Chapter 9: AUXILIARY SYSTEMS

TABLE OF CONTENTS

TABLE OF CONTENTS

Chapter 9: AUXILIARY SYSTEMS

TABLE OF CONTENTS

Chapter 9: AUXILIARY SYSTEMS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Chapter 9: AUXILIARY SYSTEMS

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

Chapter 9

AUXILIARY SYSTEMS

This chapter describes the objectives, design bases, system design safety considerations, and the inspection and testing requirements of the auxiliary systems listed below:

- New fuel storage.
- Spent fuel storage.
- Fuel pool cooling and cleanup system.
- Fuel handling system.
- Tools and servicing equipment.
- Well water system.
- River water supply system.
- Residual heat removal (RHR) and emergency equipment service water systems.
- General service water system.
- Reactor building cooling water system.
- Condensate storage and transfer system.
- Makeup water treatment system.
- Potable and sanitary water systems.
- Engineered safeguards, heating and ventilating system.
- Instrument and service air systems.
- Process sampling systems.
- Equipment and floor drainage systems.

- Standby liquid control system.
- Heating, ventilation, and air conditioning (HVAC) systems.
- Fire protection system.
- Communications systems.
- Plant lighting systems.
- Diesel generator auxiliary systems.
- Plant heating boiler system.

9.1 FUEL STORAGE AND HANDLING

These facilities include the dry storage vault for new fuel, the water-filled pool for spent fuel, storage racks, and cranes for handling fuel and shipping casks. See Section 1.1 concerning onsite interim storage at an Independent Spent Fuel Storage Installation.

9.1.1 NEW FUEL STORAGE

9.1.1.1 Design Bases

9.1.1.1.1 Power Generation Objective

The power generation objective of the new fuel storage racks arrangement is to provide a specially designed dry storage place for the new fuel assemblies or bundles (see Figure 9.1-1).

9.1.1.1.2 Power Generation Design Bases

1. New fuel storage racks are provided for 30% of the full core load of fuel assemblies to accommodate fuel storage space.

2. New fuel storage racks are designed and arranged so that the fuel assemblies can be efficiently handled during refueling or pre-refueling operations.

9.1.1.1.3 Safety Design Bases

1. The new fuel storage racks are designed and maintained with sufficient spacing between the new fuel assemblies to ensure that the array, when racks are fully loaded, shall be substantially subcritical under all conditions.

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

9.1.1.2 Facilities Description

The location of the new fuel storage vault is shown in Figure 1.2-7. [redacted] Construction details are shown in Figures 9.1-1 and 9.1-2. [redacted]

The new fuel racks are anchored to the reinforced-concrete walls as shown in Figure 9.1-3. The vault, racks, and rack supports are designed to withstand the design basis earthquake (DBE) seismic forces.

Each new fuel storage rack (shown in Figure 9.1-4) holds as many as 10 channeled or unchanneled fuel assemblies in a row, spaced 6.625 in. apart center to center. The racks are designed so that arrangement in rows on an 11-in. center to center spacing will limit the effective multiplication factor of the array (k_{eff}) to not more than 0.90 in air. New fuel storage racks are provided for 30% of the reactor core load. Space is provided in the new fuel storage vault to store up to 43% of a full core load by installing additional fuel racks.

The fuel assemblies are loaded into the rack through the top with the auxiliary hoist on the overhead crane. Each hole for a fuel assembly has enough clearance for inserting or withdrawing the assembly while enclosed in a protective wrapping. Sufficient guidance is provided to preclude damage to the fuel assemblies. Guides are provided to guide the spacers of the fuel elements for the full length of their insertion into the rack. The design of the racks prevents accidental insertion of the fuel assembly in a position not intended for the fuel. The weight of the fuel assembly is supported at the bottom and the rack provides a full longitudinal support of the new fuel assembly.

[redacted] These gratings can withstand loads of 100 lb/ft^2. The vault is provided with enough drainage to prevent water collection and flooding.

Each new fuel storage rack loaded with fuel is designed as a Seismic Category I structure.

9.1.1.3 Safety Evaluation

The calculations of k_{eff} are based on the geometrical arrangements of the fuel array. Subcriticality does not depend on the presence of neutron-absorbing materials. The most reactive arrangement of the fuel assemblies in the fuel storage racks results in k_{eff} below 0.90 in a dry condition, or in the absence of a moderator. In an abnormal condition when the fuel is flooded with water, k_{eff} will not exceed 0.95. These k_{eff} values are satisfied if the maximum infinite lattice multiplication factor (k_∞) of the individual fuel bundles is ≤ 1.31.

Stresses in a fully loaded rack are designed not to exceed applicable specification requirements of the American Institute of Steel Construction or the American Society of Civil Engineering when subjected to a horizontal earthquake load of 0.75g applied in any direction. A safety factor of two, based on the material yield or local critical buckling, is used where these specifications are not applicable.

The storage rack structure is designed to absorb an impact energy of at least 7000 ft.-lb. on an impact surface no larger than 3 in. in diameter. Under this impact force, those members whose function it is to physically maintain the subcritical spacing to ensure that k_{eff} will not exceed 0.95 will remain intact. Those members whose local and general strain exceed 25% of the material's ultimate strain are assumed to be nonexistent for further energy absorption or for spacing purposes. Those members and their connections whose continued presence is required

to maintain subcriticality are designed using a minimum safety factor of 1.33 based on the lower of the material's yield or buckling stresses.

The storage racks are designed to withstand a pull-up force equal to the load rating of the overhead crane's auxiliary hoist. (This is necessary in the event that the fuel assembly or grappling device binds during removal.) The stress in those members required to maintain the abnormal storage subcriticality conditions will not exceed 75% of the material's yield strength or 75% of that stress at which local buckling occurs.

The new fuel racks are designed to be restrained by hold-down lugs to ensure that rack spacing does not vary under specified earthquake loads. Hold-down bolts will restrain the rack in case a stuck fuel assembly is inadvertently hoisted. [redacted] All material used in the construction of the new fuel storage racks is specified in accordance with the applicable ASTM specifications, and all welds are in accordance with the AWS standards for materials used. Materials selected are corrosion resistant or treated to provide the necessary corrosion resistance.

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

9.1.2 SPENT FUEL STORAGE

9.1.2.1 Design Bases

9.1.2.1.1 Power Generation Objective

The power generation objective of the spent fuel storage racks is to provide specially designed underwater storage space for the spent fuel assemblies or bundles that require shielding, cooling, and criticality control during storage and handling.

9.1.2.1.2 Power Generation Design Bases

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

2. High density spent fuel storage racks are designed to provide maximum storage space in the spent fuel pool.

9.1.2.1.3 Safety Design Bases

1. The fuel array in the fully loaded spent fuel racks will be substantially subcritical and prevent fuel barrier damage caused by overheating. For any operating or accident condition which is a design basis for DAEC, the subcritical multiplication factor (k_{eff}) is

maintained below 0.95. This includes the worst-case postulation of a dropped fuel assembly.

2. Each spent fuel storage rack, empty or loaded with fuel, is designed to withstand earthquake loading to minimize distortion of the spent fuel storage arrangement.

9.1.2.2 Facilities Description

The spent fuel storage racks provide a storage place [redacted] for the spent fuel received from the reactor vessel as shown in Figures 9.1-5 and 9.1-29. The racks are full length top entry and designed to maintain the spent fuel in a space geometry that precludes the possibility of criticality under normal and abnormal conditions. Normal conditions exist when the spent fuel is stored at the bottom of the fuel pool in the design storage position. Abnormal conditions may result from an earthquake or mishandling of spent fuel.

Each spent fuel storage rack and fixture, empty or loaded with fuel, is designed as a Seismic Category I structure to resist sufficiently the response motion at the installed location within the supporting structures for the DBE.

The following sections describe and evaluate the design of the high density spent fuel storage racks.

9.1.2.2.1 General Description and Arrangement

Currently, the DAEC spent fuel pool contains two types of spent fuel storage racks:

1) Twelve (12) PaR (Programmed and Remote Systems Corporation) racks, and

2) Nine (9) Holtec racks.

PaR racks were approved for installation via license amendment number 45 in 1978, and Holtec racks were approved for installation via license amendment number 195 in February of 1994.

2017-006 The rerack project of 1994 removed 7 PaR racks from the east side of the spent fuel pool along with the Control Rod Blade Storage Rack and the Channel Storage Rack and installed nine (9) Holtec racks to increase the spent fuel pool capacity to 2411 assemblies (Figure 9.1-7).

In addition, a Cask Pit is also licensed to contain a rack with storage capacity of 152 assemblies. The Cask Pit rack is used as a means to retain full-core-offload capability after such capacity is exhausted in the spent fuel pool. The DAEC may or may not exercise this option in the future.

Following is the general description and arrangement of the spent fuel storage racks:

PaR Racks

PaR spent fuel racks are a bolted anodized aluminum construction having a neutron absorber medium of natural B_4C in an aluminum matrix core clad with 1100 series aluminum. The neutron absorber, marketed under the trade name of Boral, is sealed within two concentric square aluminum tubes forming the "poison can." The minimum weight of total boron per unit area of poison material is 0.129 g/cm^2 (0.0232 g/cm^2 Boron-10). 2017-006

Holtec Racks

Holtec racks are also, like PaR racks, free standing and self supporting. The principal construction materials for the Holtec racks are ASME 240-type 304 stainless steel sheet and plate stock, and SA564 (precipitation hardened stainless steel) for the adjustable support spindle. The only non-stainless steel material utilized in the rack is the neutron absorber material which is a boron carbide aluminum cermet manufactured under a US patent and sold under the brand name Boral by AAR Advanced Structures, Livonia, Michigan. Boral panels are placed in the pockets formed between the (box) cell and outer sheathing plate (Figure 9.1-31).

2017-006 Both PaR and Holtec Racks

Figure 9.1-6 shows the general location of the fuel pool with respect to other plant structures and Figure 9.1-7 shows the existing arrangement of the spent fuel racks in the spent fuel pool. There are a total of 21 racks for a total of 2411 cavities. The following table shows the different PaR rack sizes, the size indicating the number of fuel positions along each horizontal axis of the rack.

Quantity	Size	Approximate Rack Deadweight (lb)
2	8 x 10	10,880
5	8 x 11	11,975
4	10 x 11	14,960
1	11 x 11	16,456

Holtec rack details are shown below with the rack number, number of cells, and weight of each rack.

Rack No.	Size	No. of Cells	Approximate Rack Dead Weight (lb)
A1	12×12	144	10300
A2	12×12	144	10300
A3	12×12	144	10300
B1	12×10	120	8600
B2	12×10	120	8600
C	12×12 (-1×10)	134	9600

D	14×12	168	12000
E	14×10 (-4×7)	112	8000
F	14×12 (-4×1)	164	11700

Rack A1 [redacted] is also known as a "Dual Purpose Rack." 96 cells of this rack are convertible into 24 square prismatic cells of
2017-006 approximately 12" × 12" opening which can be used to store miscellaneous object of larger cross-section such as the defective fuel containers, control rods, etc. Modules A2 and A3 are also designed to support a specially engineered, overhead platform which permits storage of miscellaneous objects up to five tons total weight without interfering with the normal function of the module as an assemblage of spent fuel storage cavities. (That platform was not
2017-006 purchased and would require further review by NRC staff prior to use per Reference 6.) The structural and thermal-hydraulic qualification of these racks includes appropriate consideration
2017-006 of the overhead platform.

9.1.2.2.2 Spent Fuel Rack Construction

PaR Racks

[redacted] Figures 9.1-5, 9.1-8, and 9.1-9 show the basic structural design. The racks consist of the following six basic components:

1. Top grid castings.

2. Bottom grid casting.

3. Poison can assembly.

4. Side plates.

5. Corner angle clips.

6. Adjustable foot assembly.

Each component is anodized separately. The top and bottom grids are machined to accurately maintain nominal fuel element spacing of 6.625 in. center-to-center within the rack. The spacing between the outermost fuel elements in adjacent racks is 9.375 in. center-to-center. The grid structures are bolted and riveted together by four corner angles and four side shear panels. Large leveling screws are located at the rack corners to adjust for variations in pool floor level of up to ± 0.75 in. The bearing pad at the bottom of the screws pivots to allow for maintaining a flat uniform contact area. The close-spaced arrangement of the storage racks is such that a fuel assembly cannot be inserted between racks or anywhere within the rack other than in a designed location.

Pockets are cast in alternate cavity openings of the grids into which the poison cans rest. This arrangement provides enough separation to ensure that no structural loads will be imposed on the poison cans. The Boral in the poison cans is positioned so that it extends at least 1 in. beyond the top and bottom of a fuel assembly of maximum active length. The outer can is formed into the inner can at the ends and totally seal welded to isolate the Boral from the pool water. Each can is pressure and vacuum leak tested.

The racks have good corrosion resistance because of the use of anodized aluminum and are not expected to incur corrosion problems during the life of the plant. This is supported by experience at other installations using similar materials in unborated spent fuel pools.

Corrosion of the Boral will not be a problem because of the protection provided by the canning and the corrosion resistance of the Boral itself. A study by the manufacturer, Brooks and Perkins, shows that a 40-year life would be expected for the Boral with no reduction in neutron-absorbing capability following a rupture in a poison can. For license renewal extension, an aging
2017-006 management program was established to manage aging effects of Boral. See UFSAR Section 18.1.41.

Holtec Racks

Holtec racks are all welded stainless steel construction, free standing and self supporting (Figure 9.1-29). The cells are fabricated from two precision formed channels (Figure 9.1-30) by seam welding them together. Each cell has two lateral holes (Figure 9.1-30) punched near its bottom edge to provide auxiliary flow. In the next step, a picture frame sheathing is pressed formed in a precision die. The "picture frame sheathing" is attached to each side of the cell (Figure 9.1-31) with the poison material (Boral) installed in the sheathing
2017-006 cavity. The top of the sheathing is connected using a smooth continuous fillet weld near the top of the cell. The edges of the sheathing and the cell are welded together to form a smooth lead-in edge. The cell with integrally connected sheathing is referred to as the "composite box."

The composite boxes are arranged in a checkerboard array (Figure 9.1-33) to form an assemblage of storage locations. The inter-box welding and pitch adjustment is accomplished by a small longitudinal austenetic stainless connector.

2017-006 Figures 9.1-29 through 9.1-42 show the basic structural design of the maximum-density poisoned BWR spent fuel rack of all stainless steel construction. These racks consist of the following five basic components:

1) The Storage Box Assembly
2) The Base Plate
3) The Neutron Absorber Material
4) Picture Frame Sheathing, and
5) Support Legs

9.1.2.2.3 Rack Interfaces with the Spent Fuel Pool

PaR Racks

The racks are a freestanding design. Their only interface with the floor is the four stainless steel bearing pads attached to the corner leveling screws. A 0.25-in. ABS plastic sheet separates this pad and the aluminum leveling screw to prevent galvanic corrosion. The ABS plastic sheet is held in place by the geometric configuration of the adjustable foot.

[redacted] The installation tolerance between racks is -0.0, +0.125 in.

[redacted] The periphery of the racks clears the walls, sparger pipes, and any other wall attachments by at least 2.1 in. This arrangement provides ample clearance for thermal downflow and seismic displacement. Provisions have been made for cooling flow in the corner cavities between the foot assembly and the bottom of the casting.

Holtec Racks

[redacted]
These bearing pads are specially designed to assure clearance of the pool liner seam welds and
2017-006 other obstacles such as swing bolt, and abandoned structures, etc.

[redacted]

The rack consists of individual cells with 5.9 inch (nominal) inside square dimension. Each cell accommodates a single boiling water reactor (BWR) fuel assembly. The fuel assembly can be stored in a storage location in channeled or unchanneled configuration. Each cell has two lateral holes punched near its bottom edge to provide auxiliary cooling flow.

[redacted] This arrangement provides sufficient margin for the seismic, nuclear criticality, and thermal-hydraulic considerations.

9.1.2.2.4 Quality Assurance Program

The rack design control, design verification, material control, and rack fabrication were accomplished by procedures that satisfy the requirements of ANSI N45.2 "Quality Assurance Program for Nuclear Power Plants."

Special quality control programs were in effect to ensure that the Boral had the required minimum and uniform B_4C density in the sheet. Included was a nondestructive chemical analysis sampling program that maintained a high level of confidence of uniform B_4C density. Traceability of all components to a heat lot was maintained during the rack fabrication. The Boral has complete traceability along with a map of its final position in the rack. The Boral traceability is as follows: The stock sheets are etched with a serial number by the manufacturer (Brooks and Perkins), who maintains traceability to original aluminum and B_4C lots and test samples. This serial number along with a dash number is etched into each part cut from the stock sheet. The cavities are also serialized. At assembly, a log is maintained including the cavity assembly weight and record of all dimensional, seal, and liquid-penetrant tests. Finally, dimensional, visual, and functional inspections of the rack were performed by the manufacturer at the site before rack installation. Sealed Boral coupons are provided for inservice surveillance.

The following documents comprise the final documentation package:

Design Documents

1. As-built module assembly/detail drawings.
2. Installation drawing.
3. Design report.
4. Installation procedures.

Quality Control Documents

1. Inspection status form, modules.
2. Inspection status form, cavities.
3. Map location of cavities and Boral.
4. Nominal material test reports.
5. Weld rod certifications.
6. Weld identification and welder qualifications.
7. Inspection identification and qualifications.
    a. Liquid penetrant.
    b. Seal test.
8. Certification of conformance for anodizing.

9.1.2.3 Safety Evaluation

9.1.2.3.1 Criticality Considerations

9.1.2.3.1.1 Design Criteria

The design of the revised fuel storage rack complies with the criteria established for the spent fuel storage racks as described in the Section 9.1.2.1.3. For any operating or accident condition that is a design basis for the DAEC, the subcritical multiplication factor (k_{eff}) is
2017-006 maintained below 0.95. This k_{eff} value is satisfied if the maximum exposure dependent k_{∞} and enrichment of each individual bundle are within limits as discussed later.

9.1.2.3.1.2 Analysis Methods

PaR Racks

2017-006

The analysis is performed using CASMO-4 for the depletion analysis for the determination of SCCG infinite multiplication factor (k_{inf}) and MCNP6 for the rack analysis and calculation of the in-rack effective multiplication factor (k_{eff}).

CASMO-4 is a two-dimensional multi-group transport theory code for burnup calculations of BWR and PWR fuel assemblies developed by Studsvik. The code handles a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array including fuel rods loaded with Gadolinia. The code handles the BWR channels as well as water gaps and cruciform control rods in the regions separating fuel assemblies.

The CASMO-4 neutron data library is based on data from ENDF/B-IV although some data comes from other sources. It contains cross sections for 108 materials, most of which are individual nuclides. Nuclear data are collected in a library containing microscopic cross sections in 70 energy groups. The code models multiple individual fission products, and uses two lumped fission products for those that are not specifically modeled.

MCNP6 is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The calculations use continuous energy cross section data based on ENDF/B-VII.1. MCNP6 was selected because it has a history of successful use in fuel storage criticality analyses and has most of the necessary features for the analyses performed. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. Pointwise cross-section data is used. All reactions given in a particular cross-section evaluation (such as ENDF/B-VII.1) are accounted for. Thermal neutrons are described by both the free gas and S(alpha, beta) models.

The convergence of a Monte Carlo criticality case is sensitive to the number of histories (neutrons) per generation, the total number of generations, the number of generations skipped before averaging, and the initial source distribution. Sufficient neutrons and generations were used to ensure the Monte Carlo uncertainty was in the range of 0.0001 to 0.0003. The initial source is placed in the highest reactive area of the model. All cases were verified to have source convergence.

Holtec Racks

Criticality analyses for the Holtec maximum density racks were performed with the CASMO-3 code, a two-dimensional multi-group transport theory code. Independent verification calculations were made with the KENO-5a computer package, using the 27-group SCALE (Standardized Computer Analysis for Licensing Evaluation, a standard cross-section set developed by the Oak Ridge National laboratory for the USNRC).

Benchmark calculations indicate a bias of 0.0000 +/- 0.0024 for CASMO-3 and 0.0101 +/- 0.0018 (95% probability at the 95% confidence level) for NITAWL-KENO5a. In the geometric model used in the calculations, each fuel rod and its cladding were explicitly described and reflecting boundary conditions (zero neutron current) were used in the axial direction and at the centerline of the Boral and steel plate between storage cells. These boundary conditions have the effect of creating an infinite array of storage cells in all directions.

The CASMO-3 computer code was used as the primary method of analysis as well as a means of evaluating small reactivity increments associated with manufacturing tolerances. Burnup calculations were also performed with CASMO-3, using the restart option to describe spent fuel in the storage cell. KENO-5a was used to assess the reactivity consequences of eccentric fuel positioning and abnormal locations of fuel assemblies.

9.1.2.3.1.3 Bases and Assumptions

The following conservative assumptions were used for both normal and abnormal configuration analyses:

a) Both PaR and Holtec Racks:

Conservative assumptions used for both PaR and Holtec racks are as follows:

- The racks are assumed to contain the most reactive fuel for the case being analyzed, without any control rods or burnable poison, except gadolinium, as appropriate.
- The moderator is assumed to be pure, unborated water at a temperature corresponding to the highest reactivity (4°C) over the expected range of water temperatures.
- Neutron absorption in minor structural members is neglected, i.e., spacer grids are assumed to be replaced by water.
- 2017-006 Criticality safety analyses are based upon the infinite multiplication factor (k_{∞}.), i.e., lattice of storage racks is assumed infinite in all directions. No credit is taken for axial or radial neutron leakage, except in the assessment of certain abnormal/accident conditions where neutron leakage is inherent.
- Fuel assemblies are assumed to contain the high reactivity limiting lattice for the entire length of the assembly (i.e., natural uranium blankets are not modeled).

b) PaR Racks:

The nominal spent fuel storage cell used for the criticality analyses of the PaR racks is as follows:

- [redacted]
- The fuel assemblies are assumed to be centrally located in each storage cell on a lattice spacing of 6.625 ± 0.050 inches.
- 2017-006 The Boral absorbers have a core thickness of 0.080 inches with a loading of 0.0150 gm B-10/cm^2, well below the nominal areal density of 0.0250 gm B-10/cm^2 (Reference 16). The Boral absorbers are clad on both sides with 0.0175-inch thick aluminum.

c) Holtec Racks:

The following conservative assumptions were made for the Holtec racks to assure that the true reactivity will always be less than the calculated reactivity:

- 2017-006 [redacted]
- 2017-006 The 0.070 inch thick boral absorber has a nominal loading of 0.0162 g B-10/cm^2.

2017-006 d) Fuel Assemblies (PaR Racks):

Nine different fuel designs have been used in the Duane Arnold reactor core from its initial cycle in 1974: GE 7×7, GE 8×8, GE5, GE6B, GE8B, GE10, GE12, GE14, and GNF2 2019-001 (including GNF2.02). Note that in Cycle 8 lead test assemblies were used, which were similar to the design used in later cycles. Also note that GNF2.02 is identical to GNF2 for criticality purposes.

The objective of the depletion analysis is to find the most limiting composition at any time of life for any fuel assembly under any operating condition. Since BWRs generally use Gadolinia bearing rods in all the assemblies, the most limiting composition is when the reactivity peaks at around 12-18 GWd/MTU. The depletion analysis is done using CASMO-4.

In order to find the most limiting composition, all of the lattices from all the fuel assemblies that have been used at DAEC were analyzed with CASMO-4. For each case, the depletion analysis was performed by depleting without the control rod at voids of 0%, 40%, 70%, and 100%, and also with the control rod inserted at voids of 0%, 40%, and 70%. Using the depleted atom densities from each burnup step and each depletion condition, a k_{inf} calculation was performed where the channeled fuel assembly is centered in a 6 inch cell of water at 20 °C. This condition is called the Standard Cold Core Geometry (SCCG). For each lattice, the highest k_{inf} is determined and this is the maximum SCCG k_{inf} for that lattice.

2017-006 The maximum SCCG k_{inf} for the limiting GE14 and GNF2 fuel designs is the same, 1.267. Since the maximum SCCG k_{inf} of these fuel designs is more than 2% lower than the selected SCCG k_{inf} limit of 1.29, a new GNF2 lattice was created that matches the SCCG k_{inf} of 1.29. The peak reactivity is 1.28995 which occurred when depleted unrodded at 0% void. The pin isotopics at peak reactivity were imported into the PaR rack model. This lattice was used for the tolerance and uncertainty calculations as well as the eccentricity, interface, and accident calculations.

The maximum SCCG k_{inf} for the 7×7 fuel is 1.190, whereas for the 8×8 fuel it is 1.246. Since there is a large margin to the selected SCCG k_{inf} limit of 1.29, no additional calculations are performed for those designs. Nevertheless, to assure that the margin is maintained, the analysis is only applicable to the 7×7 and 8×8 legacy fuel assemblies used at DAEC during
2017-006 Cycles 1 through 16.

2017-006 e) Fuel Assemblies (Holtec Racks):

Two different fuel assembly configurations were assumed in the analysis, as follows:

- GE-10 8×8 fuel assemblies, and
- 2017-006 GE-12, 10×10 fuel assemblies.

Each of the fuel assembly configurations has a maximum enrichment of 4.95 wt%
2017-006 U-235, 7 gadolinia rods, and a zircalloy channel. The GE-12 fuel assemblies contain part-length rods, creating an array of higher water-to-fuel ratio near the top. Furthermore, the physical differences between GE-12 and GE-14/GNF2 fuel (10×10 arrays) are compensated by the conservative assumptions made for part length rods, UO_2 blankets, and spacers.
2017-006

Of the various fuel assembly types investigated, the 10×10 rod assembly was found to
2017-006 be the most reactive. All of the fuel assemblies have natural UO_2 blankets at each end. The calculations reported here do not include blankets, and are based on the lattice average enrichments. The blankets, therefore, do not contribute to the calculated reactivities. Blankets of natural UO_2 would result in lower and more conservative reactivities (k_{eff}) for the entire assembly. Fuel enrichments, as used in this report, refer to the enriched lattices in the assembly
2017-006 without consideration of any axial blankets that might be present.

Calculations were also made to determine the effect of removing the zircalloy channel. Results showed a decrease in reactivity with removal of the channel (-0.0098 Δk). Therefore, the reactivity with the flow channel installed on the assembly yields the higher and controlling reactivity.

9.1.2.3.1.4 Results

2017-006 PaR Racks

The criticality analysis documented in Reference 16 confirms that the PaR storage racks can safely accommodate, within the regulatory guidelines, all fuel assemblies which were at the

2017-006 DAEC through March 2016. The analysis uses a Boral areal density of 0.0150 gm B-10/cm^2 which provides a large margin to the as-built Boral areal density for the PaR spent fuel pool racks and matches the Boral areal density of the Holtec spent fuel pool racks. This analysis reduces the SCCG limit to 1.29, same as for the Holtec spent fuel pool racks. This new criticality analysis has a calculated $k_{95/95}$ of 0.9300, which provides considerable margin to the regulatory limit of 0.95.

The calculated $k_{95/95}$ includes statistically combined uncertainties on fuel tolerances, rack tolerances, depletion uncertainty, and code validation uncertainty. It also includes biases on assembly orientation, assembly eccentricity, de-channeled fuel assemblies, fission product worth, zirconium cladding alloys, rack bulging, dry cavity, rack interface effects, and code validation. The analysis also addresses potential accidents, and includes the limiting accident in the $k_{95/95}$ calculation.

The following are the limits to this criticality safety analysis of the PaR spent fuel pool racks:

1. Fuel assemblies must have a SCCG k_{inf} of 1.29 or less.

2. Only 7×7 and 8×8 fuel assemblies allowed for storage are legacy fuel assemblies from Cycles 1 through 16.

3. Future 10×10 fuel assemblies must be bounded by the fuel pellet diameter, clad outer diameter, pin pitch and pellet stack density of the GNF2 fuel lattices analyzed.

4. Fuel cladding material must be a zirconium alloy.

5. Integral burnable poison must be Gadolinia (Gd_2O_3).

6. Reconstituted fuel must contain a stainless steel or natural uranium fuel rod in any location where fuel was removed. (The reconstitution process is safe if done in an isolated area.)

7. The Boral areal density in the panels is greater than or equal to 0.0150 gm B-10/cm^2.

8. Rack center to center distance greater or equal to 6.625 inches.

9. Spent Fuel Pool water temperature greater or equal to 4 °C.

Each future fuel assembly must be analyzed to determine its SCCG k_{inf} and demonstrate compliance with this criticality safety analysis. Those future calculations must be performed with the same process used in this analysis to ensure its validity. The analyst is expected to utilize the same CASMO-4 input decks used for this analysis and only change the fuel rod enrichments, the number and position of the Gadolinia rods, and job titles. A brief summary of the process is provided below.

2017-006 First, the SCCG k_{inf} must be calculated using CASMO-4. Use of any other computer code or cross section library may introduce a bias not included in this analysis. Second, the analysis must include depletion with bounding reactor parameters with unrodded conditions at 0, 40, 70 and 100% void, and also with rodded conditions at 0, 40 and 70% void. The bounding reactor parameters are documented in Section 6.1 of the criticality report (Reference 16). Third, the SCCG k_{inf} is at 20 °C and unrodded. The maximum value from all depletions evaluated for each fuel assembly must meet the limit established by this analysis.

Holtec Racks

The analysis of Reference 5 proves that all fuel assemblies at the DAEC by August, 1997 may be stored in the Holtec storage racks while acceptably meeting the regulatory guidelines. A similar analysis was performed for advanced fuel designs initially enriched to
2017-006 4.95 wt % U-235. These calculations verified that the GE-12, 10×10 array fuel assemblies may be acceptably stored in the Holtec racks providing that the core k_∞ is 1.29 or less. This same limit applies to the GE-14, 10 × 10 array, due to the bounding analysis assumptions.
2012-007 Supplement 1 of Reference 13 notes that the GNF2 fuel design is bounded by this requirement
2019-001 and can be stored in the Holtec racks similar to other 10 × 10 fuel designs. Note that GNF2.02 fuel design is identical to GNF2 for criticality purposes.

9.1.2.3.1.5 Temperature and Boiling Effects

2017-006 PaR Racks

Criticality calculations are performed at normal ambient temperature (about 300K, which is the standard temperature for MCNP6 and its libraries). A water density of 1.0 gm/cm^3 is used for all cases, which is the maximum value of water density at all temperatures. This bounding density corresponds to a water temperature of 4 °C.

The accident analysis includes the loss of spent fuel pool cooling resulting in an over-temperature accident. The analyses presented in Section 9.1.2.3.2 are applicable for the PaR racks.

Holtec Racks

Using the normal geometry, the temperatures of the pool water and the fuel were allowed to range from 68°F to 200°F. The conditions necessary to cause such a temperature
2017-006 excursion are discussed below in Section 9.1.2.3.2. The reactivity change was calculated at 95°F, 120°F, 160°F, and 200°F. The result was that reactivity decreases as temperature increases and k_{eff} remained less than 0.92.

9.1.2.3.1.6 Accident and Abnormal Conditions

2017-006

PaR Racks

The following accidents were evaluated for this criticality analysis:

1. Dropped fuel assembly into a rack location. A postulated accident is that a fuel assembly is dropped into a cell, the grids fail, and all pins are separated to an optimum pitch.

2. Missing absorber panel. It is postulated that there is one missing Boral panel in the rack.

3. Rack movement into the worst alignment.

4. Misplaced fuel assembly outside yet adjacent to the racks. There is a corner between modules E15 and E12 where, if an assembly were to be misplaced, that assembly would not be exposed to any face-adjacent Boral panels. The limiting assembly was modeled in this corner next to three other eccentric-in assemblies with no Boral panel between assemblies.

5. Loss of spent fuel pool cooling resulting in an over-temperature accident. It is assumed that the most reactive condition in the pool is full density water. To test this assumption, the base case was run with water at a lowered density, and the k_{eff} decreased. As expected, increasing temperature in the pool reduces k_{eff}.

The limiting accident was found to be the missing Boral panel, with a delta k of 0.0030. After accounting for accidents, the criticality analysis has considerable margin to the regulatory limit of 0.95.

Holtec Racks

Although the storage rack is designed to prohibit the insertion of a fuel assembly anywhere except at a design location, the dropping of a fuel assembly could result in an unintended fuel assembly location adjacent to the rack. Two locations are credible, as follows:

1. On top of the storage rack.

2. Outside the rack assembly between the outermost rack and spent fuel pool wall.

The consequences of a dropped fuel assembly on top of the other fuel assemblies result in k_{eff} less than 0.95.

The evaluation of a fuel assembly dropped alongside the rack was performed by conservatively assuming that the dropped assembly lodges parallel to an off-centered assembly in the outermost cavity. The analysis indicates that k_{eff} is less than 0.95.

9.1.2.3.1.7 Conclusion

The analyses performed for both the PaR and Holtec spent fuel storage racks shows that the k_{eff} of the spent fuel pool remains substantially below the limit of 0.95. This is assured if individual fuel assemblies have the following limits for maximum k_∞ in the normal reactor core configuration at cold conditions and maximum lattice-averaged U-235 enrichment weight percents:

			k_∞	wt %
2017-006	i)	7×7 and 8×8 pin arrays (Legacy Fuel Assemblies only)	≤ 1.29	≤ 4.6
	ii)	10×10 pin arrays (Holtec and PaR racks)	≤ 1.29	≤ 4.95

2011-015 In addition to meeting the reactivity and enrichment requirements noted, the following fuel design requirements must also be met for the criticality analysis to remain valid:

2017-006 Allowable Fuel Designs (7×7, 8×8):	All previously used at DAEC through GE8; GE10
2012-007 Allowable Fuel Designs (10×10):	GE12; GE14; GNF2 (including GNF2.02)
2019-001 Cladding Material:	Zircaloy-2
Burnable Poison Type:	Gd_2O_3

2011-015 Furthermore, the following spent fuel pool requirements must also be met for the criticality analysis to remain valid:

Pool Water Temperature:	≥ 4°C (39.2°F)
Absorber Panel Material:	Boral
2014-012 Absorber Panel Areal Density (PaR):	≥ 0.0150 g/cm^2 B-l0
2017-006 Absorber Panel Areal Density (Holtec):	≥ 0.0150 g/cm^2 B-l0
Center to Center Distance (PaR):	≥ 6.625 in
Center to Center Distance (Holtec):	≥ 6.060 in

2017-006 For both nominal and accident cases, analyses indicate that a fully loaded fuel pool would remain substantially subcritical. This is based on conservative analyses that take no credit for poisons in the fuel, soluble poisons in the water, or in-core fuel depletion.

9.1.2.3.2 Cooling Considerations

9.1.2.3.2.1 Design Bases

The racks and the pool structure are designed for an accident thermal excursion to 212˚F. For the freestanding rack design, thermal load resulting from confined expansion of the racks is negligible. Therefore, the only effect of this thermal excursion on the rack design was its associated reduction of material yield strengths, which was considered. The thermal excursion to 212˚F was considered in the design of the pool structure. No additional thermal analysis of the pool structure was required due to changes in fuel rack design.

The Spent Fuel Pool Cooling System design was analyzed as part of the rerack performed in 1994. Four heat load analyses (Cases 1, 2, 3 and 4) were performed to demonstrate the adequacy of the DAEC Spent Fuel Pool Cooling System to cool the discharged fuel under four separate scenarios.

Case 1, the Maximum Normal Heat Load scenario and Case 2, the Abnormal Maximum Heat Load scenario were performed to illustrate compliance of the FPCCU System to the provisions of NUREG-0800, SRP 9.1.3. These scenarios were not found to be the most limiting and are therefore not discussed here. Case 1 and Case 2 scenarios and associated analyses can be found in Reference 5.

Case 3, the - Normal Heat Load scenario describes the bounding discharge practices at the DAEC involving a total off-load of the core 120 hours after shutdown. This scenario is the most limiting case for the 120 hour incore decay scenarios in that the analyzed peak temperature is the highest value for the four cases (1 though 4) of Reference 5.

The spent fuel pool is assumed to have 3152 spent fuel storage locations (actual number of storage locations is 2411) and that, when 128 spent fuel assemblies are discharged into the pool, insufficient space remains to accept another normal batch, while maintaining full core off-load capability. The transfer of the fuel to the pool begins after 120 hours of in-core decay and at a rate of 144 fuel assemblies per 24 hours. Both loops of FPCCU are assumed to be in service. The peak fuel pool water temperature is sought to be no greater than 180° F, which is below the regulatory limit of 212° F. Analysis shows that the maximum bulk pool temperature peaks at 164.61° F, 190 hours after reactor shutdown.

This case was used to calculate maximums for local water and fuel clad temperatures. Calculations showed that the maximum local water temperature would reach 216.3° F and the maximum local fuel clad temperature would reach 264.4° F. No nucleate boiling is indicated at any location in the spent fuel pool. It was, therefore, concluded that the reracked spent fuel pool complies with all thermal hydraulic regulatory criteria (reference Figure 9.1-45).

Case 4, the Abnormal Heat Load scenario is the same as Case 3 except the scenario occurs when the reactor has been operating for 36 days after a 45-day refueling outage. This

scenario was not found to be the most limiting and is therefore not discussed here. Case 4 scenario and associated analysis can be found in Reference 5.

Early Core Discharge Scenario

Three additional analyses (Cases A, B, and C) were performed to demonstrate the adequacy of the FPCCU system and/or RHR Supplemental Fuel Pool Cooling (RHR-SFPC) to support an early core discharge scenario (Reference 10). The initial conditions for the scenarios analyzed are similar to Cases 3 and 4 discussed previously, however, core alteration begin as soon as 60 hours after the reactor shutdown (i.e., reactor scram or all rods in). The three scenarios analyzed included: a planned (i.e., follows 24 months of power operation) full core offload with one FPCCU loop operating before RHR-SFPC is initiated (Case A), a planned full core offload with two loops of FPCCU operating before RHR-SFPC is initiated (Case B) and an unplanned full core offload with both loops of FPCCU in operation prior to initiating RHR-SFPC (Case C). Case C assumes that the core has been operated for 45 days after a 36 day refueling outage, during which a discharge of 152 fuel assemblies to the spent fuel pool took place.

The scenarios assume the core is offloaded at a rate of 6 fuel assemblies per hour. The scenarios assume that RHR Shutdown Cooling (SDC) is in operation (one loop) prior to the start of the core alterations with 85°F cooling water to the RHR Heat Exchanger. One (Case A) or both (Cases B and C) loops of FPCCU are in service at rated flow with 95°F RBCCW cooling water to the FPCCU heat exchangers.

The spent fuel pool temperature begins to rise and, for analysis purposes, at 120˚F, the FPCCU system is removed from service and RHR supplemental fuel pool cooling is initiated. RHR-SFPC takes a suction on the reactor through the SDC line. The RHR heat exchanger removes the decay heat and then the flow is split between the reactor cavity and the spent fuel pool. Fuel pool water flows through the fuel transfer canal into the reactor cavity and back through the SDC suction to complete the flow path. The fuel pool bulk water temperature is analyzed to peak at 152.50°F, 126 hours after shut down for Cases A and B and 152.46˚F, 125 hours after shut down for Case C (reference Figures 9.1-47 through 9.1-49).

Plant procedures exist that allow the FPCCU system to be in operation with RHR-SFPC in service while the fuel pool gates are removed (as assumed in Cases A through C). However, the FPCCU system must be shut down if RHR is to be used to cool the Fuel Pool with the fuel pool gates installed. The scenarios of Cases A through C describe worse case situations and take no credit for the operation of the FPCCU system, even though it is allowed by procedures.

A time-to-boil calculation for Cases A, B and C was performed assuming a total loss of cooling to the spent fuel pool. If all forced cooling were lost, calculations show that Cases A and B are the most limiting. These scenarios show the spent fuel pool would start to boil approximately 64 minutes after the total loss of forced cooling. Given there is 16 feet of water above the fuel racks, makeup water to the pool would have to be provided within 3.8 hours of the

onset of boiling at a rate of 53.05 gpm to makeup for the steam being generated. A special test of the ESW flow to the spent fuel pool conducted on September 26, 1996, demonstrated an ESW flow greater than 59 gpm, which is in excess of the 53.05 gpm required (Reference 7). This loss of cooling was conservatively analyzed assuming the fuel pool gates were installed simultaneously with the loss of forced cooling. In other words, no credit is taken for the existence of the reactor cavity water volume or the ability to reflood or cool the fuel pool from the reactor cavity through the transfer canal (reference Figures 9.1-47 through 9.1-49).

The maximum local water and fuel cladding temperatures were calculated for a limiting (i.e. highest decay heat and bounding bulk temperatures) early discharge scenario (Reference 10). The net heat generation for this scenario bounds the calculated net spent fuel pool heat load coincident with the maximum temperature for all discharge/cooling scenarios at DAEC. The maximum spent fuel pool bulk temperature is used in this calculation. Additionally, the east and west downcomers are assumed to be 2/3 blocked thereby reducing cooling flow to the pool. Under these conditions, the maximum local water temperature was calculated as 177.53˚F and the maximum fuel cladding temperature as 211.28˚F. [redacted] and the saturation temperature of the coolant at this depth is approximately 239.29˚F. Therefore, a substantial margin against nucleate boiling exists on the fuel rod surfaces and the fuel rack cells.

Procedures exist that control the initial conditions necessary to discharge the core 60 hours after reactor shutdown. Normal system lineups are expected to be used that keep the actual initial conditions more conservative than the assumed initial conditions. The analysis provides the maximum operating parameters that would allow discharge of the core to the spent fuel pool beginning 60 hours after shutdown.

Fuel Shuffling

The bounding analysis for the DAEC is Case 3, (Reference 3), which restricts fuel movement to the spent fuel pool to begin no earlier than 120 hours after shutdown and at a rate not to exceed 144 fuel assemblies in any 24 hour period. Should the pool temperatures approach 150° F, actions will be taken to increase cooling or fuel handling rates will be adjusted to reduce the rate of heat addition to the spent fuel pool. The analyses for Cases A through C (Reference 10), which take credit for RHR-SFPC, allow fuel movement to begin at 60 hours after shutdown, but result in a lower maximum bulk pool temperature than Case 3. Fuel shuffling would be well within the bounds of Case 3 for fuel discharge beginning 120 hours after shutdown or Cases A through C for early core discharge beginning 60 hours after shutdown provided the initial plant
conditions are bounded by these cases. Total core off-loads or fuel shuffling activities will be performed such that the 150˚F limit is not exceeded.

9.1.2.3.2.2 <u>Cooling System Capability</u>

The limiting design factor of the FPCCU System is the piping for the FPCCU System is designed to accommodate 150° F water at 200 psi. Procedural restrictions prevent the

coolant exiting the FPCCU heat exchangers from exceeding 130° F to protect the demineralizers. The system design is such that a bulk pool temperature limit of 150° F is imposed. The design of the existing FPCCU System and the RHR System permit operations of the systems in parallel. For fuel discharges at the DAEC starting 60 hours after shutdown, RHR-SFPC will be utilized as the fuel pool temperatures approach 120°F, if decay heat curves warrant, to preclude spent fuel pool temperatures from exceeding the value of 150°F.

The analyses performed for the total core off-loads bound the heat load that could be anticipated during a partial off-load or core shuffle. The plant configuration and analyses parameters are adhered to assure compliance to the safety analyses.

9.1.2.3.3 Mechanical, Material, and Structural Considerations

9.1.2.3.3.1 Design Requirements

The spent fuel pool and spent fuel pool storage racks are Seismic Category I. The storage racks are designed to withstand the effects of a design-basis earthquake (DBE), postulated jammed fuel and fuel drop accidents without loss of structural integrity or functional adequacy, that is, the retention of fuel element spacing and overall geometry. The fuel pool structure is analyzed for the resulting storage rack interface loads.

9.1.2.3.3.2 Loading Combinations and Allowable Stresses

The loading combinations and factored limits for the PaR racks are included in Table 9.1-3. The Holtec and PaR Storage Racks were designed to meet applicable requirements of Subsection NF, Section III, of the ASME Code.

The allowable stresses for stainless steel are in accordance with the ASME Code, Section III, Appendix XVII. This is interpreted as being identical to the AISC Steel Construction Manual, Section 5.

The allowable stresses for aluminum members are based on the Aluminum Construction Manual, Section 1, "Specifications for Aluminum."

The following specifications from the manual were used:

Table No.	Description
3.3.3	Factors of Safety for Use with Aluminum Allowable Stress Specification
3.3.4 and 3.3.4b	Formulas for Buckling Constraints
3.3.6	General Formulas for Determining Allowable Stress
5.1.1a	Allowable Bearing Stresses for Building-Type Structures

5.1.1b Allowable Stresses for Rivets, Bolts for Building-Type Structures

9.1.2.3.3.3 Seismic Analysis

Analysis Method

Following a seismic event with accelerations in excess of the Operating Basis Earthquake, the gaps between the spent fuel racks are to be inspected and, if necessary, restored to their original dimensions. (Reference NRC letter dated February 24, 1994.)

A combination time-history/static seismic analysis was performed. A horizontal time history was developed such that the corresponding response spectra enveloped the E-W and N-S Design Basis Event spectra for 6% damping for the PaR racks and 2% damping for the Holtec racks. Both are conservative with respect to Regulatory Guides 1.60 and 1.61.

It was determined in the original seismic analysis that the building will cause no amplification of motion in the vertical direction. A vertical time history was developed such that the corresponding spectra would conservatively envelope the ground response spectra. The horizontal and vertical time histories were then input simultaneously to the dynamic model at the floor spring location. The forces computed from the time-history analysis were applied to the static model. Symmetry of the PaR storage rack about the principal axes accounts for the equivalence of this method to simultaneous excitation in three orthogonal directions.

The combination time-history/static seismic analysis was done for the PaR racks via computer solution programs ANSYS and SAP IV, respectively. The ANSYS User Manual, Swanson Analysis Systems Inc., Elizabeth, Pennsylvania, documents this program.

SAP IV (public version) for static and dynamic analysis of linear structural systems was used to analyze the mathematical model. The development and documentation of SAP IV was sponsored by grants from the National Science Foundation and was authored by Klaus-Jurgan Bathe, Edward L. Wilson and Fred Peterson of the University of California, Berkeley, California. It is available as Report EERC 73-11 revised April, 1974, from the Earthquake Engineering Research Center at the University of California. SAP IV has been installed on a Control Data Corporation Cyber 74 computer in Minneapolis, Minnesota, where the model was analyzed.

The seismic analysis for the Holtec racks used several different models to provide, as accurate as possible, the seismic response of the fuel racks. Single rack 3-D models were used and compared to the Holtec computer code DYNARACK and the Whole Pool Multi-Rack 3-D analysis. The intent of this parallel approach was to foster added confidence and to uncover any peculiarities in the dynamic response which was germane to the structural safety of the Holtec Storage System. More detailed information is available in Reference Five, Section 6.0 - Structural / Seismic Considerations.

9.1.2.3.3.3.1 Seismic Analysis of PaR Fuel Racks

The following paragraphs describe the mathematical models employed and assumptions used in the seismic analysis of the PaR fuel racks.

9.1.2.3.3.3.1.1 ANSYS Seismic Model (PaR Fuel Racks)

The rack structure consists of four side panels bolted top and bottom to a very stiff box grid. The corners of the side panels are riveted together via formed angles. The structural system may, therefore, be visualized as a large square or rectangular tube enveloped by the side panels with no structural stiffness added for either the poison cans or fuel assemblies. Dynamic analyses of a detailed SAP IV model have determined the first two natural frequencies to be orthogonal and simple cantilever modes at 8 Hz. Successive horizontal frequencies are greater than 28 Hz. A vertical damping frequency of the bottom casting exists at 14 to 18 Hz.

The rack structure for the simplified dynamic model used in the ANSYS analysis is idealized as a planar frame consisting of a cantilever beam at the base (bottom casting elevation) with leg beams connecting the ends of this member to the floor (see Figure 9.1-10). Section properties 2-4 are calculated directly from the composite of the four side panels and bottom casting legs. Section 5 is located at the same elevation as Section 3 and is pinned to it at the ends. It represents the vertical damping of the bottom casting. Fundamental frequencies of this idealized system agree closely with the detail model.

To consider the nonlinear effects of module rocking and sliding and fuel rattling, the ANSYS model is expanded and shown in Figure 9.1-11. The center pole, Section 1, representing the mass and stiffness of all the fuel assemblies extends the height of the rack. It is pinned at the bottom of the rack and is allowed to impact at the top and top quarter point, nodes 1 and 2, and 3 and 4. A 3/8-in. gap on each side occurs at these points, which represents the fuel assembly to can clearance. For worst-case analysis, it is assumed that all fuel in the rack is channeled (thus providing the stiffest section). This transmits the highest impact and overturning loads to the rack. Based upon the stiffness of this member and based on past analysis, fuel-can impact below the top quarter is unlikely, so that the 3/8-in. gap at nodes 5 and 6 will not close. This model conservatively assumed that all fuel assemblies are in phase and move together at all times.

The vertical spring under each leg is known as an "interface element." The interface element represents two plane surfaces that may maintain or break physical contact and slide relative to each other. At each time step, the program compares the horizontal force in the interface element against the coefficient of friction to see if sliding will occur and also allows for uplift and rocking by vertically releasing the element if tensile forces exist in the leg.

A single vertical degree of freedom represents the pool floor under the racks. Its mass is the total pool mass under the area of each rack. The spring rate is calculated to give the same first mode diaphragm frequency as the entire spent fuel floor, water, and racks.

The following assumptions are made relative to the rack submergence in the spent fuel pool:

1. All water entrapped within the rack envelope is added to the horizontal mass but not to the vertical mass.

2. Since the depth of water above the racks is large (greater than 20 ft), surface waves or sloshing effects are ignored.

3. Because the linear dimension of the pool is much smaller than the pressure waves generated by typical earthquakes ($\mathbf{l}/\lambda<<1$), water in the pool will move in phase with the ground, provided the walls are rigid. Therefore, external water effects between the rack and the walls are ignored, which conservatively assumes that damping forces generated in "pumping" this confined water from the wall-rack gap as a result of the relative motion of the racks are greater than any added external mass effects of this water.

Figure 9.1-12 represents a two-rack model. It includes all the effects of the single-rack model plus the maximum interaction or potential for banging with other racks in the pool. Gap springs are located at the top and bottom casting elevation and are initially closed.

The coefficients of friction values used in the analysis are based on the following test reports: Simulated Rack Minimum Coefficient of Friction by Programmed and Remote Systems Corp. (PaR) and Friction Coefficients of Water-Lubricated Stainless Steels for a Spent Fuel Rack Facility by Professor Ernest Rabinowicz of the Massachusetts Institute of Technology, performed for Boston Edison Company. In the latter report, results of the 100 tests performed show a mean value of 0.503 with a standard deviation of 0.125. The upper ($x+2\sigma$) and lower ($x-2\sigma$) limits are 0.753 and 0.253, respectively. The values used in this analysis are 0.2 as lower limit and 0.8 as upper limit. Values measured under similar conditions agree closely for both independent tests.

The following freestanding and rack conditions were analyzed:

1. 0.2 coefficient of friction, empty, single rack.

2. 0.8 coefficient of friction, two full racks.

Condition 1 was analyzed to determine maximum displacement of the racks relative to the pool floor. Condition 2 determined the maximum rack loads for the SAP IV static analysis. The coefficients of friction remained constant throughout the time history.

9.1.2.3.3.3.1.2 SAP IV Finite-Element Model

Figure 9.1-13 shows the SAP IV computer model. The PaR spent fuel rack is idealized as a three-dimensional detailed finite-element model of nodal points, consisting of over 400

flexural beam column elements and over 800 plate elements representing the side plates and formed angles.

Only two of the module feet are fixed. Reactions for the other two feet and nodal forces needed to put the rack in equilibrium are developed for worst-load cases from the ANSYS time-history analysis. These horizontal and vertical static forces were applied to the SAP IV model in the same manner as on the ANSYS model. An equal load set was applied in an orthogonal plane. Stresses were computed using the SRSS method for all members and plates for each of these two load sets and compared against their factored allowables.

9.1.2.3.3.3.1.3 Results of Seismic Analyses for the PaR Spent Fuel Racks

Displacements and loads resulting from the response of the PaR racks to seismic events were calculated for simultaneous vertical and horizontal safe-shutdown earthquake motion using conservative time histories as described above. The coefficient of friction is calculated to be greater than 0.2 and less than 0.8 under all conditions, including variations in rack loading and floor smoothness. Decreasing coefficients of friction increase sliding displacements. A conservatively low coefficient of 0.2 was used in determining these displacements. Increasing coefficients of friction increases rack and floor loads, rocking displacements, and rack-to-rack interaction forces. A conservatively high coefficient of 0.8 was used in determining these forces and displacements. The results are as follows:

9.1.2.3.3.3.1.4 Sliding displacement

The maximum sliding displacement of the PaR racks relative to the pool floor was calculated as 1.05 in. This displacement would occur during a condition of minimum friction and would be accompanied by no significant rocking displacement, that is, only pure rigid body sliding occurred.

9.1.2.3.3.3.1.5 Rocking displacement

The maximum rocking displacement of the PaR racks relative to the pool floor was calculated to result in one side of a rack lifting approximately 1 in. off the floor. This displacement would occur during a condition of maximum friction. The feet on the other side of the rack would remain in contact with the floor and very little sliding displacement would occur. Rocking displacements of this magnitude would only be on the outside rows of racks. Rocking displacement of racks on inner rows would be limited by interactions with other racks.

9.1.2.3.3.3.1.6 Rack-to-rack impact loads

The maximum PaR rack-to-PaR rack impact load was calculated as 120,000 lb. This impact load would result from the impact of racks having undergone rocking displacement.

9.1.2.3.3.3.1.7 Foot impact loads

The maximum foot impact load for a PaR was calculated as 276,084 lb. This impact load would occur at each foot of an 11 by 11 rack having undergone rocking displacement. This load would exert a bearing stress of 4838 psi on the pool floor, along with a punching shear stress of 84.5 psi. Allowable DBE stresses are 8806 psi and 344 psi, respectively. The uniform floor loading resulting from foot impact loads would be 2535 psf. This compares to the allowable DBE uniform floor load of 3200 psf.

9.1.2.3.3.3.1.8 Rack member stresses

The stresses in the various members of the PaR rack side plates, castings, and legs were computed and compared to the factored allowable stresses given in Standard Review Plan 3.8.4 (see Table 9.1-3). The most limiting stresses are listed below in terms of the appropriate factored allowable stresses. The symbols used are taken from the Standard Review Plan and are identified on Table 9.1-3. Because the rack poison cans and alignment lugs are not structural members of the racks, stresses for these members have not been computed.

Equation Number	Loading Combination	Factored Allowable Stress Limit	Side Plates	Largest Calculated Interaction Stress: Casting	Legs
1	D + L	1.0	0.219	0.484	0.206
2	D + L + E	1.0	<1.0	<1.0	<1.0
3	D + L + To	1.5	0.226	0.509	0.217
4	D + L + To + E	1.5	<1.5	<1.5	<1.5
5	D + L + Ta + E	1.5	<1.5	<1.5	<1.5
6	D + L + DF				
	Condition 1	1.6	0.299	0.633	0.297
	Condition 2	1.6	0.726	0.513	0.701
	Condition 3	1.6	0.381	1.2	0.278
	Condition 4	1.6	0.024	0.052	0.024
7	D + L + Ta + E	2.0	0.708	1.618	1.16

Note: E = DBE; E = OBE

The analysis discussed above in the ANSYS seismic model is a worst-case analysis for PaR fuel rack loading, but is not the worst case with respect to possible fuel damage due to loss of cladding integrity.

The compressive strength of concrete and the yield strength of reinforcing steel were determined by laboratory analyses of actual samples drawn from each pour of concrete and each heat of reinforcing steel. The most limiting of the results obtained were used as the bases for performing the structural analysis.

As discussed above, a conservative value is assumed for the coefficient of friction in each computation of rack response. Actual rack responses will therefore be bounded by the calculated response regardless of variations in K_f across the floor.

The overall floor load was calculated using the double-rack model shown in Figure 9.1-12. The seismic portion of the floor load was first determined separately from the dead load, using the SRSS method. To accomplish this, the seismic load for each rack pair was determined as a fraction of the dead load (F_S/F_D) using the following relation:

$$F_S/F_D = (F_{max}/F_D) - 1$$

where

F_S = maximum force in floor as a result of seismic load only

F_D = force in floor as a result of dead load only
F_{max} = maximum load calculated

This seismic portion (F_S/F_D) was summed for the total number of pairs of racks in the pool by the SRSS method to obtain the average seismic load as a fraction of dead load $(F_S/F_D)_T$. The total dead plus seismic floor load was then determined by the following:

$$P = (N)(D)[1 + (F_S/F_D)_T]$$

where

N = total number of cavities
D = deadweight of rack plus fuel per cavity
P = total dead plus seismic floor load

9.1.2.3.3.3.1.9 <u>Dropped Fuel Bundle Analysis</u>

Analyses were done to define the equivalent static load for the following drop conditions:

1. 18-in. fuel drop on the corner of the top grid castings and fuel rollover.

2. 18-in. drop in the middle of the top castings.

3. A fuel drop full length through the cavity impacting on the bottom grid.

The following methods were used in defining the impact loads.

For condition 1, the impact energy losses of the inertia of the rack module and the collapsing of the bottom tripod on the fuel bundle fitting were quantified for the 18-in. vertical

drop to determine the net impact energy. Using the SAP IV model, spring rates were determined at various impact locations on the module. A static impact load was then determined for each of these locations by equating the elastic structural strain energy balance with the net impact energy. These impact loads have been verified by full-size tests on an actual top grid casting.

For condition 3, an unimpeded fuel drop through an empty cavity, a static load was determined to shear out the bottom fuel support. After shear out, the fuel bundle impacts the pool liner plate. The resulting load is applied to the pool as an interface load.

Equivalent static loads for different dropped fuel bundle cases were then applied at proper locations to the SAP IV finite-element model of the module and combined with the deadweight vertical load (rack full of fuel). Stresses for each member and plate were then tabulated and compared against the factored allowables.

A structural analysis was made to establish the maximum load-carrying capacity of the existing spent fuel pool. This analysis was based on the actual material strength and latest ACI Code requirements (ACI 318-71). A compressive concrete strength of 7400 psi and a yield strength of reinforcing steel of 65,700 psi, as determined from laboratory test reports, were used. The results of the analysis indicated that the maximum live load (including the associated earthquake loading from fuel rack and fuel elements) should not exceed 2.56×10^6 lb.

9.1.2.3.3.3.1.10 Pool Interface Loads

Rack leg vertical gap forces are computed for each time step of the analysis. These loads are used to determine the bearing and punching shear stress in the reinforced-concrete floor. The allowable stresses are defined by Section 1.10, "Alternative Design Method," of American Concrete Institute Building Code Requirements for Reinforced Concrete (ACI 318-71). As described in the Commentary to the Code, this section carries forward the working stress design method of ACI 318-63. Under dynamic impact loads, a factor of 1.25 is applied to allowable compressive stress. Information supporting the use of this factor is from a publication entitled Structural Analysis and Design of Nuclear Plant Facilities, prepared by the Committee on Nuclear Structures and Materials of the Structural Division of the American Society of Civil Engineers.

The overall floor load was checked taking the force in the floor spring "K_f" in Figure 9.1-11 and calculating a total for all the racks by an SRSS technique. This load, 2.04×10^6 lb, was compared against floor slab capacity of 2.56×10^6 lb.

9.1.2.3.3.3.1.11 Analysis of Rack and Pool Interaction

The maximum dry weight of the rack is 136 lb/cavity. For an 11 by 11 rack, this amounts to 16,456 lb.

Figure 9.1-8 presents a detail of the leveling foot assembly. A flat ABS plastic sheet separates the steel from the aluminum and is mechanically confined between these parts. The steel and the plastic are fastened to the aluminum with stainless steel bolts. ABS plastic washers on the bolts and oversize holes through the aluminum prevent contact between the aluminum and the bolts.

Calculations show that the plastic will withstand all design loadings while remaining within its elastic limits. The plastic will also withstand temperatures far in excess of the maximum expected without significant changes to the mechanical properties. The plastic will not affect the pool water chemistry and will not be significantly affected by irradiation. In the book Radiation Effects on Organic Materials by R. O. Bolt and J. G. Carrol, 1963 edition, test data of a styrene-acrylonitrile copolymer similar to ABS demonstrated that this material retains up to 80% of its initial strength at a total radiation dose of 10^8 rads. It should also be noted that because of mechanical confinement of the plastic, the integrity of the assembly would be maintained even if the plastic suffered some deterioration or failure such as cracking.

Figure 9.1-14 shows a section view of the underside of the corner of the bottom casting that indicates the water path through the casting into the corner cavity.

9.1.2.3.3.3.2 Holtec Rack Modeling for Dynamic Simulations, 3D-Single Rack Analysis

Spent fuel storage racks are Seismic Class I equipment. They are required to remain functional during and after a Design Basis Event (DBE). The racks are free-standing; they are neither anchored to the pool floor nor attached to the sidewalls. Individual rack modules are not interconnected. Figure 9.1-35 shows a typical module. The baseplate extends beyond the cellular region envelope ensuring that inter-rack impacts, if any, occur first at the baseplate elevation; this area is structurally qualifiable to withstand any large in-plane impact loads.

A rack may be completely loaded with fuel assemblies (which corresponds to greatest total mass), or it may be completely empty. The coefficient of friction, μ, between pedestal supports and pool floor is indeterminate. Analyses are, therefore, performed for coefficient of friction values of 0.2 (lower limit) and for 0.8 (upper limit), and for random friction values clustered about a mean of 0.5. The bounding values of $\mu = 0.2$ and 0.8 have been found to bracket the upper limit of module response in rerack projects at other facilities.

Since free-standing racks are not anchored to the pool slab, not attached to the pool walls, and not interconnected, they can execute a wide variety of motions. Racks may slide on the pool floor, one or more rack support pedestals may momentarily tip and lose contact with the floor slab liner, or racks may exhibit a combination of sliding and tipping. The structural models developed permit simulation of these kinematic events with inherent built-in conservatisms. The rack models also include components for simulation of potential inter-rack and rack-to-wall impact phenomena. Lift-off of support pedestals and subsequent liner impacts are modeled using impact (gap) elements, and Coulomb friction between rack and pool liner is simulated by piece-wise linear (friction) elements. Rack elasticity, relative to the rack base, is included in the model with linear springs representing a beam-like action. These special

attributes of rack dynamics require strong emphasis on modeling of linear and non-linear springs, dampers, and compression only gap elements. The term "non-linear spring" is a generic term to denote the mathematical element representing the case where restoring force is not linearly proportional to displacement. In the fuel rack simulations, the Coulomb friction interface between rack support pedestal and liner is typical of a non-linear spring.

Three dimensional dynamic analyses of single rack modules require a key modeling assumption. This relates to location and relative motion of neighboring racks. The gap between a peripheral rack and adjacent pool wall is known, with motion of the wall prescribed. However, another rack, adjacent to the rack being analyzed, is also free-standing and subject to motion during a seismic event. To conduct the seismic analysis of a given rack, its physical interface with neighboring modules must be specified. The standard procedure in analysis of a single rack module is that neighboring racks move 180° F out-of-phase in relation to the subject rack. Thus, the available gap before inter-rack impact occurs is 50% of the physical gap. This "opposed phase motion" assumption increases the likelihood of intra-rack impacts and is, thus, conservative. However, it also increases the relative contribution of fluid coupling, which depends on fluid gaps and relative movements of bodies, making overall conservatism a less certain assertion. Three dimensional Whole Pool Multi-Rack (WPMR) analyses performed indicate that single rack simulations predict smaller rack displacement during seismic responses. Nevertheless, 3-D analyses of single rack modules permit detailed evaluation of stress fields, and serve as a benchmark check for the much more involved WPMR analysis.

Particulars of modeling details and assumptions for 3-D Single Rack analysis and for WPMR analysis are given in the following subsections.

9.1.2.3.3.3.2.1 <u>The 3-D 22 DOF Model for Single Rack Module (Assumptions)</u>

1. The fuel rack structure is very rigid; motion is captured by modeling the rack as a twelve degree-of-freedom structure. Movement of the rack cross-section at any height is described by six degrees-of-freedom of the rack base and six degrees-of-freedom at the rack top. Rattling fuel assemblies within the rack are modeled by five lumped masses. Each lumped fuel mass has two horizontal displacement degrees-of-freedom. Vertical motion of the fuel assembly mass is assumed equal to rack vertical motion at the baseplate level. The centroid of each fuel assembly mass can be located off center, relative to the rack structure centroid at that level, to simulate a partially loaded rack.

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

3. Fluid coupling between rack and fuel assemblies, and between rack and wall, is simulated by appropriate inertial coupling in the system kinetic energy. Fluid coupling terms for rack-to-rack coupling are based on opposed phase motion of adjacent modules..

4. Fluid damping and form drag is conservatively neglected.

5. Sloshing is negligible at the top of the rack and is neglected in the analysis of the rack.

6. Potential impacts between rack and fuel assemblies are accounted for by appropriate "compression only" gap elements between masses involved. The possible incidence of rack-to-wall or rack-to-rack impact is simulated by gap elements at top and bottom of the rack in two horizontal directions. Bottom elements are located at the baseplate elevation.

7. Pedestals are modeled by gap elements in the vertical direction and as "rigid links" for transferring horizontal stress. Each pedestal support is linked to the pool liner by two friction springs. Local pedestal spring stiffness accounts for floor elasticity and for local rack elasticity just above the pedestal.

8. Rattling of fuel assemblies inside the storage locations causes the gap between fuel assemblies and cell wall to change from a maximum of twice the nominal gap to a theoretical zero gap. Fluid coupling coefficients are based on the nominal gap.

9.1.2.3.3.3.2.2 Whole Pool Multi-Rack (WPMR) Model

The single rack 3-D model, outlined in the preceding subsection, is used to evaluate structural integrity, physical stability, and to initially assess kinematic compliance (no rack-to-rack impact in the cellular region) of the rack modules. Prescribing the motion of the racks adjacent to the module being analyzed is an assumption in the single rack simulations. For closely spaced racks, demonstration of kinematic compliance is further confirmed by modeling all modules in one comprehensive simulation using a WPMR model. In WPMR analysis, all racks are modeled, and their correct fluid interaction is included in the model.

9.1.2.3.3.3.2.3 Whole Pool Fluid Coupling

The presence of fluid moving in the narrow gaps between racks and between racks and pool walls causes both near and far field fluid coupling effects. A single rack simulation can effectively include only hydrodynamic effects due to contiguous racks when a certain set of assumptions is used for the motion of contiguous racks. In a WPMR analysis, far field fluid coupling effects of all racks are accounted for using the correct model of pool fluid mechanics. The external hydrodynamic mass due to the presence of walls or adjacent racks is computed in a manner consistent with fundamental fluid mechanics principles using conservative nominal fluid gaps in the pool at the beginning of the seismic event. Verification of the computed

hydrodynamic effect by comparison with experiments is also provided. The fluid flow model used to obtain the whole pool hydrodynamic effect reflects actual gaps and rack locations.

9.1.2.3.3.3.2.4 Coefficients of Friction

To eliminate the last significant element of uncertainty in rack dynamic analyses, the friction coefficient is ascribed to the support pedestal / pool bearing pad interface consistent with data at other facilities. Friction coefficients, developed by a random number generator with Gaussian normal distribution characteristics, are imposed on each pedestal of each rack in the pool. The assigned values are then held constant during the entire simulation in order to obtain reproducible results. Thus, the WPMR analysis can simulate the effect of different coefficients of friction at adjacent rack pedestals. The friction coefficients at the interface between rack support pedestals and pool liner is assumed distributed randomly with a mean of 0.5 and permitted to vary between the limits of 0.2 - 0.8.

9.1.2.3.3.3.2.5 Material Properties

Physical properties of the rack and support materials were obtained from the ASME Boiler & Pressure Vessel Code, Section III and appendices. Maximum pool bulk temperature is less than 200° F; this is used as the reference design temperature for evaluation of material properties.

9.1.2.3.3.3.2.6 Results of 3-D Non-linear Analyses of Single Racks

This section focuses on results from all 3-D single rack analyses. The following section presents results from the whole pool multi-rack analysis and discuss the similarities and differences between single and multi-rack analysis.

The racks chosen to be analyzed are Rack G (the rack with maximum aspect ratio), Rack J (the largest rack in the pool), and Rack R (the rack in the cask pit). Altogether, 18 runs are carried out for governing cases using Holtec proprietary computer program DYNARACK. Results are abstracted from output files and presented here for the governing cases. Analyses have been carried out for regular fuel (680 lb. dry weight) and for opposed-phase motion assumption. The chosen racks would be installed in Campaigns II and III.

9.1.2.3.3.3.2.7 Racks in the Fuel Pool

A summary of results of all analyses performed for racks in the pool and in the cask pit as well, using a single rack model, is presented in Reference 5. The tabular results for each run give maximax (maximum in time and in space) values of stress factors at important locations in the rack. Results are given for maximum rack displacements, maximum impact forces at pedestal-liner interface, and rack cell-to-fuel, rack-to-rack, and rack-to-wall impact forces. It is shown that no rack-to-rack or rack-to-wall impacts occur in the cellular region of the racks.

In the single rack analysis, kinematic criteria are checked by confirming that no inter-rack gap elements at the top of the rack close. By virtue of the symmetry assumption discussed

in Reference Five, impact is assumed to occur if the local horizontal displacement exceeds 50% of the actual rack-to-rack gap.

Structural integrity at various rack sections is considered by computing the appropriate stress factors. Results corresponding to the SSE event yield the highest stress factors. Limiting stress factors for pedestals are at the upper section of the support and are to be compared with the bounding value of 1.0 (OBE) or 2.0 (DBE). Stress factors for the lower portion of the support are not limiting and are not reported. The analysis shows all stress factors are below the allowable limits.

9.1.2.3.3.3.2.8 Impact Analyses

1. Impact Loading Between Fuel Assembly and Cell Wall

   Local cell wall integrity is conservatively estimated from peak impact loads. Plastic analysis is used to obtain the limiting impact load. Reference 5 compares limiting impact loads with the highest value obtained from any of the single rack analysis. The limiting load is much greater than the load obtained from any of the simulations calculated.

2. Impacts Between Adjacent Racks

   No non-zero impact loads are found for the rack-to-rack gap elements (in the cellular region), or for the rack-to-wall elements; it is concluded that no impacts between racks or between racks and walls are likely to occur during a seismic event. This is confirmed by the WPMR results in Reference 5.

9.1.2.3.3.3.2.9 Weld Stresses

Weld locations subjected to significant seismic loading are at the bottom of the rack at the baseplate-to-cell connection, at the top of the pedestal support at the baseplate connection, and at the cell-to-cell connections. Results from dynamic analyses of single racks are surveyed and maximum loading used to qualify the welds.

9.1.2.3.3.3.2.10 Rack in the Cask Pit Area

The cask area of the fuel pool is a separate pit area with a 108” x 120” horizontal envelope. Analyses have been carried out for a 17 x 19 free standing rack (Rack R) installed in the cask pit area. To evaluate the rack in the cask pit, analysis is performed using fluid gaps between rack and cask pit wall that reflects the actual dimensions of the cask pit area and the rack envelope. Runs were carried out for coefficient of friction of 0.2 and 0.8 and for different rack fuel loading scenarios. From all analyses performed for a spent fuel rack in the cask pit area, the bounding structural and kinematic results are given in Table 6.7.2 of Reference 5.

9.1.2.3.3.3.2.11 Results from Whole Pool Multi-Rack (WPMR) Analyses

Figure 9.1-7B shows the DAEC spent fuel pool with 18 new Holtec spent fuel racks. In the WPMR Analysis, a reduced degree-of-freedom (8-DOF) model for each rack and its contained fuel is employed. The WPRM dynamic model for DAEC contains 144 degrees-of-freedom and requires a non-linear analysis. All racks are assumed to be fully loaded with 680-pound fuel assemblies. Thirty-percent of the fuel load is assumed to be rattling and impacting the rack top.

Table 6.8.1 of Reference 5 shows maximum corner absolute displacements at both the top and bottom of each rack in global x and y direction from the multi-rack runs. As noted previously, a random set of friction coefficients in the range of 0.2 - 0.8 with mean value being 0.5 is used. The seismic loadings are the DBE earthquake time-histories which are the corresponding OBE time-histories multiplied by a factor of 2.0. No non-zero values found for impact indicate that there is no impact between racks and between rack and pool wall during a DBE seismic event. The absolute displacement values are higher than those obtained from single rack analysis. Thus, it appears essential to perform WPMR analyses to verify that racks to not impact or hit the wall. A survey of all of the rack-to-rack and rack-to-wall impact elements confirms that there are no rack-to-rack or rack-to-wall impacts in the cellular region of any rack in the spent fuel pool. The inter-rack gap elements in the whole pool analysis have an initial gap equal to the actual gap.

The WPMR analysis confirms that no new concerns are identified; overall structural integrity conclusions are confirmed by both single and multi-rack analyses. Because the values of all the stress factors obtained for DBE are less than 1.0 and no rack-to-rack / wall impacts are found, it is not necessary to perform the WPMR Analysis for OBE seismic.

9.1.2.3.3.3.2.12 Bearing Pad Analysis

To protect the slab from high localized dynamic loadings, bearing pads are placed between the pedestal base and the slab. Fuel rack pedestals impact on those bearing pads during a seismic event and pedestal loading is transferred to the liner. Bearing pad dimensions are set to ensure that the average pressure on the slab surface due to a static load plus a dynamic impact load does not exceed the American Concrete Institute (ACI-349-85) limit on bearing pressures. Pedestal locations are set to avoid overloading of leak chase regions under the slab. Time-history results from dynamic simulations for each pedestal are used to generate appropriate static and dynamic pedestal loads which are then used to develop the bearing pad size.

The limiting bearing pad size with the maximum liner stress from bearing pad pressure was found to be a 12” x 12” pad. The maximum load was found to be 103500 lbs. The calculated stress to the liner was calculated to be 719 lbs. which is well below the 2380 lbs. allowed. (See Reference Five, Table 6.9.1).

9.1.2.3.3.3.2.13 Refueling Accidents

1. Dropped Fuel Assembly

   The consequences of dropping a fuel assembly as it is being moved over stored fuel is discussed below. It is assumed that the lowest part of the fuel assembly being carried is 18" above the top of the new spent fuel racks. The fuel assembly weighs 680 lbs. and associated handling equipment is assumed to weight 120 lbs.

   a. Dropped Fuel Assembly Accident (Deep Drop Scenario)

      An 800 lb. fuel assembly plus handling equipment is dropped from 18" above the top of the storage location and impacts the base of the module. Local failure of the baseplate is acceptable; however, the rack design should ensure that gross structural failure does not occur and the subcriticality of the adjacent fuel assemblies is not violated. Calculated results show that there will be no change in the spacing between cells. Local deformation of the baseplate in the neighborhood of the impact will occur, but the dropped assembly will be contained and not impact the liner. Calculations also show that even if there is local cell-to-baseplate weld overstress in individual cells, the maximum movement of the baseplate toward the liner after the impact is at most between .94" and 1.52". The load transmitted to the liner through the support by such an accident is well below the loads caused by seismic events.

   b. Dropped Fuel Assembly Accident (Shallow Drop Scenario)

      One fuel assembly plus the channel is dropped from 18" above the top of the rack and impacts the top of the rack. Permanent deformation of the rack is acceptable, but is required to be limited to the top region such that the rack cross-sectional geometry at the level of the top of the active fuel (and below) is not altered. Assuming a minimal area of impact, it is shown that damage, if it occurs, will be restricted to a depth of less than or equal to 1.09" below the top of the rack. this is above the active fuel region.

9.1.2.3.3.3.3 Conclusions of Seismic Analysis

The analyses performed show that the PaR and Holtec spent fuel storage racks are capable of withstanding the loads associated with all the design loading conditions without exceeding allowable stresses. The analysis also indicates that the racks can withstand overturning moments and horizontal forces without structural attachment to the pool.

Interface loads transmitted to the fuel pool are within the load-carrying capability of the pool structure, including dropped fuel element loading.

9.1.2.3.4 Summary of Safety Evaluation

The safety evaluation of the spent fuel storage modifications was performed to consider the consequences of modifying the storage racks to accommodate 3152 fuel elements for the purpose of allowing continued operation of the DAEC at its licensed power level without dependence on offsite facilities. The spent fuel pool storage capacity includes storing no more than 323 fuel assemblies in the cask pit if the following requirements are met: 1) The transfer of spent fuel that has decayed less than 5 years to the cask pit is prevented, 2) The cask pit floor drain is sealed, 3) The installation of the gate between the cask pit and the spent fuel pool is prevented, and 4) The transfer of heavy loads over the cask pit if it is utilized to store spent fuel is prevented (see Reference 4). (The actual installation consisted of 2411 storage spaces.)

The evaluation considered all plant features that would be affected by the modification. It was concluded that the changes necessary were limited to storage rack replacement. Supporting systems were determined to be adequate to satisfy the design requirements for the modified conditions. The evaluation confirmed the adequacy of the spent fuel pool cooling and cleanup system, HVAC systems, and structural interfaces, which were included in the mechanical, structural, and criticality considerations. Acceptance criteria for those features that were not modified were based on FSAR commitments. The storage rack itself was analyzed using updated methods and evaluated in accordance with current criteria contained in applicable regulatory guides and NRC positions stated in the Standard Review Plan. This included requirements established for seismic and structural analysis.

The criticality evaluation confirmed that the stored fuel would remain substantially subcritical ($k_{eff} < 0.95$) with a fully-loaded fuel pool conservatively assuming loading of nondepleted fuel. This condition is met for nominal configuration, worst-case clustering due to gaps and fabrication tolerances, and postulated fuel drop locations.

Mechanical evaluation confirmed the acceptability of supporting cooling systems, and structural evaluation verified that the rack could withstand the design bases loading combinations. Interface loads transmitted to the fuel pool are within the load-carrying capability of the structure. The structural evaluation included a seismic analysis equivalent to a three-dimensional excitation using methods that conform to Regulatory Guides 1.60 and 1.61.

9.1.2.4 Inspection and Testing Requirements

The spent fuel storage racks require no special inspection and testing for nuclear safety purposes.

9.1.2.5 Instrumentation

Fuel Pool Water Level Indication is provided in the Control Room and at a local instrument panel. Level is sensed by an ultrasonic sensing element mounted along the north side of the Fuel Pool in a stainless steel 2-inch diameter seven (7) foot long stilling well. The

level signal also provides High and Low alarms to the Control Room and local instrument panels.

2015-012 In addition, two guided wave radar level probes are respectively mounted [redacted] The level probes measure level from approximately the top of the fuel rack up to the level required to support normal operation of the spent fuel pool cooling system. Each probe provides a continuous level signal via level transmitter back to their associated electronics display enclosure located inside the control room. The guided wave radar level instrumentation system is installed to meet NRC Order EA 12-051, Order Modifying Licenses With Regard to Reliable Spent Fuel Pool Instrumentation.

9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Bases

9.1.3.1.1 Power Generation Objective

The power generation objective of the fuel pool cooling and cleanup system is to remove the decay heat and radioactivity released from the spent fuel elements. The system maintains a specified fuel pool water temperature, purity, water clarity, and water level.

9.1.3.1.2 Safety Objective

The safety objective of the fuel pool cooling and cleanup system is to maintain fuel pool water temperature at a level that will prevent damage to the fuel elements.

9.1.3.1.3 Safety Design Basis

The fuel pool cooling and cleanup system is designed to remove the decay heat from the fuel assemblies and maintain fuel pool water temperature for spent fuel storage and refueling operations and to prevent damage to the fuel elements caused by overheating.

9.1.3.1.4 Power Generation Bases

1. The fuel pool cooling and cleanup system minimizes corrosion product buildup and controls water clarity, so that the fuel assemblies can be efficiently handled underwater.

2. The fuel pool cooling and cleanup system minimizes fission product concentration in the water that could be released from the pool to the reactor building environment.

3. The fuel pool cooling and cleanup system monitors fuel pool water level and maintains a water level above the fuel sufficient to provide shielding for normal building occupancy.

9.1.3.2 System Description

The fuel pool cooling and cleanup system is shown in Figure 9.1-15. The system cools the fuel storage pool by transferring the spent fuel decay heat (see Table 9.1-2) through a heat exchanger to the reactor building closed cooling water system. The plant has installed a system cross-tie to allow well water to augment the GSW cooling for the reactor building closed cooling water system. This cross-tie is only used during the GSW out-of-service windows during refuel outages. Water purity and clarity in the storage pool, reactor well, and dryer-separator storage pit are maintained by filtering and demineralizing the pool water through a filter-demineralizer, which is shown in Figure 9.1-16. The system consists of two circulating pumps, two heat exchangers, two filter-demineralizers, and two skimmer surge tanks, all connected in parallel and the required piping, valves, and instrumentation. The pumps circulate the pool water in a closed loop, taking suction from the skimmer surge tanks, circulating the water through the heat exchangers and filters, and discharging it into the fuel pool and through diffusers near the bottom of the reactor well when the well is flooded. The water flows from the pool surface through skimmer weirs and scuppers to the surge tanks. [redacted]

Fuel pool water is continuously recirculated except during the period when the reactor well and dryer-separator pit are being drained or the FPCCU System is shutdown for maintenance. The heat exchangers are operated to remove the decay heat load from spent fuel to maintain bulk pool temperature at or below 150° F. The operating temperature of the fuel pool is permitted to rise to 150°F maximum when the circulating flow is interrupted to drain the reactor well, or when the system is shutdown. The heat exchangers in the RHR system can be used in conjunction with the FPCCU system to supplement pool cooling when the reactor is shut down, reactor head removed, and fuel pool gates open, and in the event that the bulk pool temperature cannot be maintained at or below 150° F by the FPCCU System. Makeup water for the system is normally transferred from the Condensate Storage Tank to the skimmer surge tanks to make up evaporative and leakage losses. The circulation patterns within the reactor well are established by the placement of the discharge and skimmers so as to sweep particles dislodged during refueling operations away from the work area and out of the pools. The normal flow pattern may be altered by taking suction from the bottom of the dryer-separator storage pit to control particles dislodged from parts transferred to the dryer-separator storage pit. A portable, submersible-type, underwater vacuum cleaner can be used to remove crud and miscellaneous objects from the pool walls and floor.

Pool water clarity and purity is maintained by a combination of filtering and ion exchange. The filter-demineralizer maintains total heavy element content (Cu, Ni, Fe, Hg, etc.) at 0.1 ppm or less with a pH range of 5.3 to 8.5 for compatibility with aluminum fuel racks and other equipment. Particulate material is removed from the circulated water by the pressure precoat filter-demineralizer unit in which a finely divided disposable filter medium is supported on permanent filter elements. The filter medium is replaced when the pressure drop is excessive or the ion exchange resin is depleted. Backwashing and precoating operations are manually controlled from a local panel in the reactor building. The spent filter medium is

flushed from the elements and transferred to the waste sludge tanks by backwashing with air and Condensate. The new filter medium is mixed in a precoat tank and transferred as a slurry by a precoat pump to the filter where the solids deposit on the filter elements. The holding pump maintains circulation through the filter in the interval between the precoating operation and the return to normal system operation.

The filter-demineralizer units are designed to operate with water flowing at normal 2 gpm/ft^2 of the filter area. Earth cellulose or powdered ion-exchange resin is used as a filter medium. The holding element for the filter material is a stainless steel mesh, mounted vertically in a tubesheet and replaceable as a unit. Venting is possible from below the tubesheet and from the upper head of the filter vessel. The upper head has a removable manhole for installation and replacement of the holding element. The filter vessel is constructed of Type 304 stainless steel, phenolic resin-coated carbon steel, or material of equivalent structural properties and corrosion resistance. A strainer is provided in the effluent stream of the filter-demineralizers to limit the migration of the filter material. The filter-holding element is capable of withstanding a differential pressure greater than the developed pump head for the system.

The ion-exchange resin typically is a mixture of finely ground, 300 mesh or less, cation and anion resins in proportions as determined by service. The cation resin is a strongly acidic, polystyrene with a divinyl-benzene cross-linkage. The resin is supplied in the fully regenerated hydrogen form. The anion resin is strongly basic, Type I, quaternary ammonium polystyrene with a divinyl-benzene cross-linkage. The resin is supplied in fully regenerated hydroxide form.

The maximum pressure drop across the filter and associated process valves and piping at the time for filter media replacement should not exceed the value shown in Table 9.1-2. A holding pump is connected to each filter-demineralizer. This pump starts automatically to maintain sufficient flow through the filter media to retain it on the filter elements during loss of system flow. The holding flow rate is 0.1 gpm/ft^2 of the filter area. The backwash system is used to completely remove resins and accumulated sludge from the filter-demineralizers with a minimum volume of water. Backwash slurry is drained to the radwaste system waste sludge tank, located in the radwaste building. The precoat system is designed to rapidly apply a uniform precoat of filter media to the holding elements of a filter-demineralizer. The precoat tank is carbon steel coated with phenolic or epoxy materials and sized to provide adequate volume for one precoating.

An agitator is furnished with the tank for mixing. One precoat pump and associated piping and valves are provided to precoat either filter-demineralizer and to recirculate the water to the precoat tank or suction side of the precoat pump at a rate of 1.5 gpm/ft^2 of filter area. The precoat system is also capable of cleaning or decontaminating either filter-demineralizer unit with a detergent or citric acid solution. The two filter-demineralizer units are located separately in shielded cells. Sufficient clearance is provided in the cell to permit the removal of the filter elements from the vessels. Each cell contains only the filter-demineralizer and piping. All inlet, outlet, recycle, vent, drain, and other valves are located on the outside of one

shielding wall of the cell, together with necessary piping and headers, instrument elements, and controls. Penetrations through shielding walls are located so as not to compromise radiation shielding requirements.

The system instrumentation is provided for both automatic and remote manual operations. Fuel Pool level indication is provided in the control room. A Fuel Pool High/Low Level annunciator alarm is also provided in the control room. Surge tank high, low, and low-low switches are provided. The high- and low-level switches alarm in the control room and at a local control panel in the reactor building. Skimmer Surge Tank level indication is provided in the Control Room and to the Plant Process Computer. Local indication is also provided. 2012-008 The low-low-level switches trip their respective circulating pumps when surge tank reserve capacity is reduced to the volume that can be pumped in one minute with one pump at rated capacity. A level indicator is provided to monitor reactor well water level during refueling. The indicator is mounted on the fuel pool pump rack, which controls flow to or from the reactor well during refueling. A Fuel Pool high-low water level alarm relay operates a local indicator light and sounds an alarm in the control room whenever the level is either too high or too low. The trip point is adjustable over the entire range of Fuel Pool level indication.

The pumps are controlled from a local panel in the reactor building. Pump low suction pressure automatically turns off the pumps. A pump low discharge pressure alarms in the control room and at the local panel. The controls for the remote-controlled valve that discharge the fuel pool water to the condenser hotwell or Condensate storage tank are located on the local control panel. The open or closed condition of each of these valve is indicated by a light on the local panel.

A high rate of leakage through the refueling bellows assembly, drywell to reactor seal, or the fuel pool gates is indicated by lights on the operating floor instrument racks and is alarmed in the control room.

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

9.1.3.3 Safety Evaluation

Section 9.1.2.3.2.1 describes several decay heat load scenarios for the spent fuel pool (SFP) and the FPCCU System. These scenarios show the possible decay heat removal needs for the DAEC spent fuel pool required by the most recent SFP rerack (1994) and analyses of the early core discharge (2000). In all but Case One of Reference 5, the imposed limit of 150° F bulk pool temperature is exceeded according to the analysis. The RHR System may be operated alone or in parallel with the FPCCU System in the event bulk pool temperatures cannot be maintained at or below 150° F for any decay heat addition scenario. The SFP bulk pool temperature may approach the 150° F limit during cavity drain down or maintenance

activities but every effort should be made to keep the temperature at or below the limit of 150° F to mitigate approaching the SFP and storage rack design limits.

A loss of cooling event was analyzed. The event assumes that both the cask pool and reactor cavity drain, resulting in the loss of SFP level and external cooling sources. Forced circulation is assumed to be lost when the SFP level gets to 16 feet above the pool liner (starting at the minimum SFP level of 36 feet). Cases A and B (reference Section 9.1.2.3.2.1) are the most limiting cases analyzed for all of the DAEC discharge scenarios. For Cases A and B, it is demonstrated that the maximum boil-off rate is 53.05 gpm. The analysis also shows that the SFP level can be maintained above the top of active fuel if the makeup water can be initiated within 3.8 hours of the SFP level reaching 16 feet.

A hose connection is provided on the Emergency Service Water (ESW) System on the Refuel Floor as shown in Figure 9.2-5. This ensures a Seismic Class 1 water supply to replace the fuel pool water as it evaporates (boils off).

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

There are no connections to the fuel storage pool that could allow the fuel pool to be drained below the pool gate between the reactor well and the fuel pool. The return cooling water supply piping terminates just below the surface of the spent fuel pool.

Flow control valves at the operating floor enable the operator to achieve optimum recirculation patterns to control and maintain the specified water quality and operational conditions.

A suction line and discharge line connect the fuel pool cooling and cleanup system and the RHR system as shown in Figure 9.1-15. The discharge line (from RHR) contains two normally closed manually operated valves, one adjacent to the RHR system and one adjacent to the fuel pool cooling system. The suction line (to RHR) contains one normally closed manually operated valve adjacent to the fuel pool cooling system. The RHR pumps are isolated from the fuel pool cooling system suction line by the shutdown cooling RHR pump suction valves. The interconnecting piping from the RHR system through the second interconnecting valve is designed to Seismic Category I criteria. Fuel pool cooling piping is supported to ensure that it will not fall and damage the interties with the RHR system in the event of a DBE.

Figure 9.1-15 shows the fuel pool cooling and cleanup system and its connections to the RHR system. Both connections to the RHR system are Seismic Category I from the RHR system up to and including the closed isolation valve. In addition, there is a removable spool piece in the 8-in. line [redacted] downstream of the manual isolation valve in that line. The manual isolation valves in the 8-in. lines [redacted] are designated as locked closed

valves. This protection ensures that the RHR system will not be degraded by a failure of the non-seismic fuel pool cooling and cleanup system.

The interconnections are used (i.e. the spool piece installed and the isolation valves are open) only at times when the RHR system is in operation in the shutdown cooling mode with the reactor shut down and depressurized or when the LPCI mode of the RHR system is not required to be operable. The RHR system can be used for fuel pool cooling in the unlikely event of a prolonged outage of both fuel pool cooling pumps. More likely, it will be used at times when heat loads in the pool are high. The fuel pool cooling system may be capable of handling such heat loads, but by supplementing that system with the RHR system, more comfortable temperatures can be maintained for the benefit of personnel working in the vicinity of the pool.

The design of the spent fuel pool includes a separate cask pool (Figure 9.1-6). In the unlikely event that the cask were dropped inside the cask pool, there would be damage to the reactor building, but no loss of fuel pool water would occur. In addition, interlocks on the reactor building crane prevent positioning the fuel cask or any other load over the spent fuel pool.

9.1.3.4 Inspection and Testing Requirements

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

9.1.4 FUEL HANDLING SYSTEM

9.1.4.1 Fuel Servicing Equipment

Two fuel preparation machines are used to strip the channel from spent fuel assemblies and to install the used channels on new-fuel bundles (see Figure 9.1-17). These machines are designed to be removed from the pool for servicing.

A new-fuel inspection stand is used to restrain the fuel bundle in a vertical position for inspection. The inspection stand can hold two bundles. The general purpose grapple, a small, hand-actuated tool used generally with fuel, can be attached to the reactor building auxiliary hoist, jib crane, and the auxiliary hoists on the refueling platform. The general purpose grapple is used to remove new fuel from the vault, place it in the inspection stand, and transfer it to the fuel pool. It also can be used to shuffle fuel in the pool and to handle fuel during channeling.

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

9.1.4.2 Refueling Equipment

The refueling platform is used as the principal means of transporting fuel assemblies back and forth between the reactor well and the storage pool (see Figure 9.1-18). The platform travels on tracks extending along each side of the reactor well and the fuel pool. The platform supports the refueling grapple and auxiliary hoists. The grapple is suspended from a trolley system that can traverse the width of the platform. Platform operations are controlled from an operator station on the trolley.

The drawings of major refueling and reactor servicing equipment are presented in Figures 9.1-17 through 9.1-25.

The fuel grapple is designed to provide positive indication of fuel bundle engagement and grapple hook closure (see Figure 9.1-19). Proximity switches (for hook closure) and a limit switch, 400 lb. hoist-loaded, for fuel bundle bail engagement, are wired in series to indicating lights. Both switches must be closed up to allow a fuel assembly to be lifted. The design includes a lock tab washer installed as recommended in General Electric SIL No. 125 to maintain proper grapple hook adjustment. The grapple hook is modified per SIL No. 119 recommendations.

Positive indication that a fuel bundle is properly engaged in the fuel grapple is necessary to prevent dropping fuel bundles.

The fuel grapple will ensure the operator that a fuel bundle is properly engaged and that the grapple hook is fully closed, thereby minimizing the potential for a dropped fuel bundle accident. Also, the new grapple hook release bail has been modified per General Electric SIL No. 119 to prevent accidentally engaging a fuel bundle to the grapple hook release bail.

The refueling platform as well as all fuel-handling equipment is designed to Seismic Category I requirements.

9.1.4.3 Refueling Procedures

9.1.4.3.1 Pre-shutdown Preparations for Refueling (Typical)

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

9.1.4.3.2 Post-shutdown Preparations for Refueling (Typical)

NOTE: The information listed below is typical of the activities necessary to prepare the Reactor for refueling. Actual sequences are determined by outage schedules.

[redacted]

[redacted]

9.1.4.3.2.1 Post Refueling Operations in Preparation for Startup

[redacted]

[redacted]

9.1.4.3.3 Refueling Operations

When verification has been made that the reactor and all refueling equipment and appurtenances are in readiness and administrative requirements of Section 9.1.4.3.1 above have been completed, refueling operations are initiated. These operations involve the removal of spent fuel assemblies and either reshuffling these fuel assemblies back into the core or replacing them with new-fuel assemblies as described below.

9.1.4.3.3.1 Spent Fuel

Spent fuel assemblies are those assemblies in which the reactivity burnup has been too high to permit their replacement in the reactor core for further power operation. These assemblies are stored in the spent fuel storage racks in the fuel pool or may, after appropriate decay, be transferred to an Independent Spent Fuel Storage Installation for interim onsite storage.

9.1.4.3.3.2 Shuffled Fuel

Shuffled fuel is that fuel that is moved out of the reactor core, placed in the spent fuel storage racks in the fuel pool, or reloaded into the core in another location. This relocation position is based on a determination of fuel burnup and core physics calculations.

9.1.4.3.3.3 New Fuel

New-fuel assemblies are removed from the storage vault racks and placed in the fuel pool storage racks in preparation for reactor core loading. These fuel assemblies are loaded into specific locations in the core based on core physics calculations.

9.1.4.3.3.4 Fuel Assembly Orientation

Fuel assembly orientation is very important when performing core alterations. Fuel assembly identification is verified by its location in the core or spent fuel pool. These locations are used in lieu of verifying actual fuel assembly serial numbers. Administrative controls exist to account for the location of special nuclear material. Fuel assembly identification and orientation are verified each time it is moved as part of a core alteration.

9.1.4.3.4 Failed Fuel Inspection Operations

In the event that there is positive indication that significant fuel leakage has occurred during plant operations, the following inspection procedures may be implemented.

9.1.4.3.4.1 Sipping

With regard to "sipping," fuel assemblies in the core are water sampled in order to determine whether or not the assemblies contain failed fuel rods. The water samples are analyzed for iodine-131 (I-131) and iodine-132 (I-132), and concentrations of these isotopes are compared with an analyzed reactor water sample. If the I-131 and I-132 levels are higher than the level of the reactor water sample, the fuel assembly is considered as a suspected failed fuel assembly.

9.1.4.3.4.2 Suspect Fuel

When the suspect failed fuel assembly is removed from the reactor core, it will be placed in a fuel storage rack in the spent fuel pool.

9.1.4.3.4.3 Inspection in Fuel Pool

After suspect failed fuel assemblies are stored in the fuel storage racks, these assemblies may be subjected to nondestructive examination.

The channel and upper tie plate may be removed. Suspect fuel rods may be individually examined by nondestructive techniques (such as eddy current techniques) to detect holes or cracks and by ultrasonic techniques to detect moisture inside the fuel rod. Occasionally, a detected failed rod will be visually inspected by the use of underwater visual aids such as a television, camera, boroscope, or a periscope.

9.1.4.3.4.4 Reconstitution of Failed Fuel Assemblies

After nondestructive examination and visual inspections are performed on suspect fuel assemblies and all of the failed fuel rods are identified, the data are analyzed and calculations may be made for the reconstitution of fuel assemblies. This is the replacement of defective fuel rods with sound rods so as to permit continued irradiation of the assembly. Factors to be considered for the exchange of rods between fuel assemblies are as follows:

1. Reactor core burnup calculations
2. Type of pellets.
3. U-235 enrichment.
4. Rod traverse burnup modes.

Reconstituted fuel assemblies may be reloaded into the core. [redacted]

9.1.4.4 Control of Heavy Loads

9.1.4.4.1 Introduction/Licensing Background

NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants," issued as a resolution to Generic Technical Activity A-36, established guidelines to ensure that the probability and consequences of dropping a heavy load on irradiated fuel or equipment required to achieve safe shutdown and continued decay heat removal are acceptably small. This report contained several recommendations to be implemented by all licensees and applicants to ensure the safe handling of heavy loads.

NRC Generic Letter 80-113 requested a review of the controls for the handling of heavy loads to determine the extent to which the guidelines of NUREG-0612 were being satisfied and to identify the changes and modifications that would be required in order to fully satisfy those guidelines. GL 80-113 requested a response in two phases.

Phase I responses were to address Section 5.1.1 of NUREG-0612 which covers the following areas:

1. Definition of safe load paths,
2. Development of load handling procedures,
3. Periodic inspection and testing of cranes,
4. Qualifications, training and specified conduct of crane operators,
5. Special lifting devices should satisfy the guidelines of ANSI N14.6,
6. Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9,
7. Design of cranes to ANSI B30.2 or CMAA-70.

DAEC responded to Phase I with a report showing how the guidelines of NUREG-0612 will be satisfied. DAEC also committed to upgrade the reactor building crane and the special lifting devices for those loads handled by the reactor building crane whose postulated drop has been shown to have unacceptable consequences. The NRC provided a Safety Evaluation and Technical Evaluation Report and concluded the guidelines of NUREG-0612, Sections 5.1.1 and 5.3 had been satisfied and Phase I of this issue was acceptable.

Phase II addressed Section 5.1.2 through 5.1.6 of NUREG-0612 (5.1.4 through 5.1.6 for BWR's) which cover the need for electrical interlocks/mechanical stops, or alternatively, single-failure-proof cranes or load drop analyses and the specific guidelines for single-failure-proof handling systems.

In GL85-11, (Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants" NUREG-0612), the NRC recognized the completion of the requirements of Phase I and Phase II and based on improvements in heavy loads handling obtained from the implementation of NUREG-0612 (Phase I), further action to reduce the risks associated with the handling of heavy loads was not required. DAEC completed all heavy loads modifications for Phase II.

NRC Bulletin 96-02, "Movement of Heavy Loads over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety Related Equipment," requested that licensees review plans and capabilities for handling heavy loads in all modes of operation, including while the reactor is at power, in accordance with existing regulatory guidelines and licensing basis.

A review, which was attached to the response to this bulletin, confirmed that DAEC continued to meet the commitments to existing regulatory guidelines and licensing basis. Some minor inconsistencies between plant documents and the UFSAR had to be resolved and a resolution to the single failure proof status of the Reactor Building Crane by the NRC was requested. A license amendment request was later submitted and approved (Amendment 251, References 11 & 12) that clarified the status of the Reactor Building Crane. The NRC Safety Evaluation related to that amendment stated that the NRC Staff recognizes and accepts the DAEC Reactor Building Crane as being single-failure proof for handling loads up to 100 tons.

9.1.4.4.2 Safety Basis

The risk associated with load-handling failures is acceptably low based on either (1) use of a single failure proof crane or (2) avoiding carrying loads over irradiated fuel or equipment necessary to safely shut down the plant and maintain it in a safe shutdown condition.

9.1.4.4.3 Scope of Heavy Loads Handling Systems

The following overhead handling systems and equipment are those at the DAEC from which a load drop could result in damage to irradiated fuel, plant shutdown system, or decay heat removal systems:

1. Reactor building crane,
2. Turbine building crane,
3. Recirculation pump motor hoist,
4. Drywell Shield Blocks and personnel air lock hoist,
5. Fuel pool demineralizer area hoist,
6. Steam valve area monorails,
7. Drywell maintenance hoists,
8. Spent fuel pool gamma-scan collimator port hoist,
9. Torus monorail.

9.1.4.4.4 Control of Heavy Loads Program

The Control of Heavy Loads Program consists of the following:

1. DAEC commitments in response to NUREG-0612 Phase I elements,
2. Reactor Pressure Vessel Head lifts using a single failure proof crane,
3. Spent Fuel Cask movement using a single failure proof crane and avoiding spent fuel cask lifts over the spent fuel pool.

9.1.4.4.4.1 Commitments in Response to NUREG-0612, Phase I Elements

For Overhead Handling Systems that are within the scope of NUREG-0612, seven elements must be met as described in NUREG 0612, Section 5.1.1, which is referred to as Phase I. Loads that could be brought in proximity to or over safe shutdown equipment or irradiated fuel in that spent fuel pool area or in reactor building and other plant areas should satisfy the following seven elements:

- Definition of safe load paths

Use of Safe Load Paths are required such that, to the extent possible, heavy loads are not carried over irradiated fuel or equipment necessary to safety shutdown the plant and maintain it in a safe shutdown condition. Such loads are defined by procedure as Critical Loads. Moving heavy loads over the Spent Fuel Pool (irradiated fuel) is prohibited. If it is essential to carry loads over safe shut down equipment necessary to safely shutdown the plant and maintain it in a safe shutdown condition, special precautions are directed by procedures to assure the loads are in compliance with the requirements of NUREG-0612. Plant procedures refer to safe load path drawings (BECH-HLR series) that show the location of safe shutdown equipment, safe shutdown piping, and safety-related conduit and cable trays. Those drawings may be used to develop a safe load path.

Refuel procedures direct the use of single-failure-proof crane in handling loads on the refuel floor and define the Safe Load Path for all lifting operations described in the refuel procedures.

- Development of Load Handling Procedures

The DAEC complies with the guidance of NUREG-0612 for the control of heavy loads. Procedures are in effect that prohibit movement of heavy loads over the spent fuel pool.

Specific procedures are provided for the handling of loads by the reactor building crane above the reactor refueling building floor which includes the following:

1. Identification of required equipment,
2. Inspections and acceptance criteria required before movement of a heavy load,
3. The steps and proper sequence to be followed in handling the load,
4. Safe load paths for the movement of heavy loads.

General load handling procedures are provided for the handling of loads by the reactor building crane, turbine building crane, and the other overhead handling systems in the vicinity of safe shutdown equipment. The procedures contain safe load path drawings that show the location of all safe shutdown equipment, safe shutdown piping, and safety-related conduit and cable trays.

[redacted]

- Qualifications, training, and specified conduct of crane operators.

Crane operator training, qualification and conduct is controlled by procedures consistent with Chapter 2-3 of ANSI B30.2-1976, Overhead and Gantry Cranes.

- Special Lifting Devices

The design of the special lifting devices in use at the DAEC has been compared with Section 5.1.1(4) of NUREG-0612 and ANSI-N14.6-1978 criteria related to component design and load handling reliability. After modifications were made to the vessel head strongback to provide safety margins of 10 to 1, all devices were shown to comply with the ANSI criteria. Maintenance and testing are performed on these lifting devices in accordance with ANSI-N14.6-1978, Section 5 requirements.

Intermediate hoists for critical loads, such as chain falls between the main hook and special lifting device, are prohibited by plant procedures.

- <u>Lifting Devices (Not specially designed)</u>

Lifting devices that are not specially designed are inspected, tested, and tagged according to plant procedures.

Slings are designed, fabricated, and proof-tested per the requirements of ANSI B30.9. Engineered softeners are required by plant procedures to protect slings from sharp edges on the load.

- <u>Periodic inspection and testing of cranes</u>

The DAEC program for inspection, testing and maintenance of overhead and gantry cranes satisfies the criteria of Guideline 6, NUREG-0612, Section 5.1.1(6), which states that the crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI-B30.2-1976, Overhead and Gantry Cranes, except that tests and inspections should be

performed prior to use where it is not practical to meet the frequencies of ANSI-B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency.

- Design of Cranes

The Turbine Building Crane and Reactor Building Crane are specified, inspected, tested, and maintained in accordance with CMAA-70. The Reactor Building Crane meets the requirements of ANSI B30.2, NUREG-0554 and NUREG-0612.

9.1.4.4.4.2 Reactor Pressure Vessel Head (RPVH) and other critical lifts

The following critical lifts are performed with the Reactor Building single failure proof crane and special lifting devices and/or rigging that is in accordance with NUREG-0612:

- Reactor Well Plugs
- Drier Separator Canal Plugs
- Spent Fuel Pool Refueling Slot Plugs
- Drywell Head
- Reactor Vessel Insulation Head
- Reactor Vessel Head
- Steam Dryer
- Moisture Separator
- Spent Fuel Pool Refueling Slot Gates
- Equipment Hatch Plugs in-Vessel Work Platform and Shield System

The four Stud Tensioners are lifted by a strongback which is not single failure proof. The maximum lift height of 6 feet is governed by a load drop calculation and is limited by procedures.

The Fuel Pool Demineralizer Area Shield Plugs are hoisted by a manual hoist on a monorail. The maximum lift height of 1 foot is governed by a load drop calculation and is limited by procedures and a placard mounted on the wall.

The Personnel Air Lock and associated Shield Blocks are hoisted by an installed monorail. A load drop analysis indicates acceptable consequences of a postulated load drop from the maximum height as limited by the location of the monorail.

The South Torus Hatch Plugs are hoisted by an installed monorail. Procedures govern the removal and installation process to prevent a hatch plug from dropping through the opening.

The reactor Feed Pump Motors are lifted with the Turbine Building Crane. A load drop analysis indicates acceptable consequences of a postulated load drop from the maximum height allowed by the crane.

New Fuel Crates are lifted by the Reactor Building Crane. The load drop analysis of the New Fuel Crates is bounded by the four thousand pound load drop analysis. The lift height is limited by the maximum height allowed by the crane.

9.1.4.4.4.3 Single Failure Proof Crane for Spent Fuel casks

The Reactor building crane meets the requirements of NUREG-0554 and NUREG-0612, as specified in the DAEC's responses to Generic Letters 81-07, 83-42, and 85-11 (References 11 and 12). The physical design of the DAEC Refuel Floor does not require a spent fuel cask to be lifted over irradiated fuel and procedures are in effect that prohibit movement of heavy loads over the spent fuel pool.

Analysis demonstrated that the Reactor Building crane's capability to withstand a seismic event is within acceptable limits. Therefore, the Reactor Building single-failure-proof crane will safely performed the intended function of retaining a maximum critical load of 100 tons under OBE and SSE conditions.

9.1.4.4.4.4 Movement of Heavy Load During Refueling

Table 9.1-4 contains a list of all objects that are required to be moved over the reactor core during refueling. Table 9.1-5 is a list of all objects that are required to be moved over the spent fuel pool. Currently, procedures prohibit movement of heavy loads over the spent fuel pool.

[redacted]
The cranes are equipped with interlocks to prevent any other load from passing over the spent fuel pool.

The reactor building with its entire lifting system is designed to Seismic Category I criteria as described in Section 3.8. Consequently, a postulated drop of the reactor vessel head onto the opened reactor vessel or the dryer-separator assembly into an opened reactor vessel due to hardware failure or procedural error is considered incredible. The consequences of dropping the reactor vessel head while it is in a position over the vessel would be damaging to the reactor vessel closure studs, and in some cases, cause damage to the sealing surfaces on both the vessel and vessel head. In all cases, no direct or indirect contact with the fuel would be possible, as the top of the fuel bundles are 27 ft below the vessel flange and the size of the vessel head (with respect to the vessel flange area) with all the possible orientations in the drop would not permit it to impact the fuel.

The consequences of dropping the dryer assembly onto an opened reactor vessel would be damaging to the reactor closure studs if the dryer impacted against them. If the dryer were directly over the vessel, the falling assembly would pass by the closure studs and impact upon the guide rods which control the azimuth position of the dryer and finally upon the dryer support blocks on the vessel wall. Again, no direct or indirect contact with the fuel bundles would be possible.

9.1.4.4.5 Safety Evaluation

Controls implemented by NUREG-0612 Phase I elements make the risk of a load drop very unlikely.

The use of a single failure proof crane makes the risk of a load drop extremely unlikely and acceptably low.

The risks associated with the movement of heavy loads is evaluated and controlled by station procedures.

9.1.5 Spent Fuel Cask Movement

Figures 9.1-26 and 9.1-27 show the physical relationship between the reactor, the fuel transfer canal, the steam dryer and separator storage pool, the spent fuel storage pool, and the cask pool. The reactor building crane will be used to move any spent fuel cask used to transport spent fuel or irradiated components from the cask pool to the reactor building equipment hatch for subsequent shipment or transfer to an Independent Spent Fuel Storage Installation. The reactor building crane has been upgraded in accordance with NUREG-0554. Additionally, limit switches are installed which prevent the crane from inadvertently being moved over the spent fuel pool or the reactor cavity. Safe load paths are also employed which identify the path of load travel which in the unlikely event of a load drop would have the least impact on safety related equipment. The physical design of the DAEC Refuel Floor does not require a spent fuel cask to be lifted over irradiated fuel. A separate cask pool has been provided which is used for cask loading operations. The cask is typically staged in the Reactor Head wash down area. It is then moved to the cask pool for loading and then moved back to the Reactor Head wash down area for decontamination and preparation for shipment. At no time is the cask lifted over irradiated fuel.

Secondary Containment shall be operable if the fuel cask is being moved in the reactor building. Fuel cask movement shall be suspended in the reactor building if Secondary Containment becomes inoperable.

Secondary Containment isolation valves/dampers shall be operable if the fuel cask is being moved in the reactor building. Fuel cask movement shall be suspended if a secondary containment isolation valve/damper inoperable and the associated penetration is open.

The SBGT system shall be operable if the fuel cask is being moved in the reactor building. If one train of SBGT inoperable and is not restored within the completion time, reactor building fuel cask movement shall be suspended.

For certain cask designs, rigging which meets the single failure proof criteria of ANSI-N14.6 cannot be installed on the cask until the cask has been upended and removed from the transporter and lowered to the reactor building [redacted] floor. During this evolution the cask rigging does meet a safety factor of 5 to 1; however, it will not meet the single failure proof requirements of ANSI-N14.6. For this evolution the cask shall only be lifted in the area of the floor directly supported by the corner room wall below. Lift height of the cask while in this configuration shall be limited to that required to clear the cask transporter to support loading and unloading of the cask. For all other cask movements, from the [redacted] elevation up to the Refueling Floor and back, and to and from the cask pool, the consequences of a load drop have been determined to be unacceptable. A single failure proof load handling system shall therefore be employed. This shall consist of the Reactor Building Crane and rigging which conforms to the single failure proof criteria of ANSI-N14.6 by either providing redundant load paths or by providing a safety factor of 10 to 1 when comparing the actual load to the ultimate breaking strength of the rigging. By employing these criteria, the probability of a load drop is sufficiently small that it is not considered to be a credible event, and as such, DAEC's commitments to NUREG-0612 are satisfied.

9.1.6 TOOLS AND SERVICING EQUIPMENT

9.1.6.1 Introduction

Tools and servicing equipment required for boiling water reactor (BWR) general servicing provide for efficient, safe serviceability in a minimum of time. Table 9.1-6 is a listing of tools and servicing equipment supplied with the nuclear system. The paragraphs below describe some of the major tools and servicing equipment for the following:

1. Fuel servicing equipment.

2. Servicing aids.

3. Reactor vessel servicing equipment.

4. In-vessel servicing equipment.

5. Refueling equipment.

6. Storage equipment.

7. Under reactor vessel servicing equipment.

The fuel servicing equipment and refueling equipment are described in Section 9.1.4.

9.1.6.2 Servicing Aids

General area underwater lights are provided with a suitable reflector for downward illumination. Lights can be supported by suitable support brackets in the reactor vessel to allow the light to be positioned over the area being serviced independent of the platform. Local area underwater lights are small diameter lights for additional downward illumination. Drop lights are used for intense radial illumination where needed. These lights are small enough in diameter to fit into fuel channels or control blade guide tubes.

A portable underwater television camera and monitor are part of the plant optical aids. The transmitted image can be viewed on the refueling platform. This assists in the inspection of the vessel internals and general underwater surveillance in the reactor vessel and fuel storage pool. A general purpose, clear plastic viewing aid that will float is used to break the water surface for better visibility.

Two portable, submersible-type underwater filter/vacuum units are provided to assist in removing crud and miscellaneous objects from the pool floor or the reactor vessel. The pumps and the filter units are completely submersible for extended periods. Fuel pool tool accessories are also provided to meet servicing requirements.

9.1.6.3 Reactor Vessel Servicing Equipment

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

The head strongback is used for lifting the drywell head and the vessel head. The strongback is designed to keep the head level during lifting and transport (see Figure 9.1-25). It is cruciform in shape with four equally spaced lifting points. The strongback is designed such that no single component failure would cause the load to drop or to swing uncontrollably. It has also been modified to meet higher safety margin criteria than originally designed to comply with the guidelines of NUREG-0612. The strongback has been proof-tested to 150% rated capacity.

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

The head holding pedestals are designed to properly support the reactor vessel head and permit reactor o-ring removal and replacement and seal surface cleaning and inspection (see Figure 9.1-22). The mating surface between vessel and pedestal is selected to minimize the possibility of damaging the vessel head.

9.1.6.4 In-Vessel Servicing Equipment

The instrument strongback is attached to the reactor building crane auxiliary hoist and is used to lift replacement in-core detectors. The instrument handling tool is attached to the in-core detector by the personnel on the refueling floor. The strongback initially supports the in-core detector into the vessel. Final incore insertion is accomplished with the instrument handling tool. The instrument handling tool is attached to the refueling platform auxiliary hoist and is used for removing and installing fixed incore detectors as well as handling neutron sources and the source range monitor/intermediate range monitor dry tubes.

In the unlikely event that incore guide tube flange O-rings need replacing, an incore guide tube seal and a test plug are provided. The guide tube seal seats on the beveled guide tube entry in the vessel. When the drain on the spring reel is opened, water drains from the incore housing and guide tube; hydrostatic pressure seats the guide tube seal and allows the flange to be removed. The incore guide tube seal contains a bail, similar to the control rod and fuel bail. A fuel bail cleaner is provided to brush the bails and improve bundle number legibility.

The one-half ton auxiliary hoist can be used with appropriate grapples to handle control rods, flux monitors, sources, and other internals of the reactor. Interlocks on both the grapple hoist and auxiliary hoists are provided for safety purposes; the refueling interlocks are described and evaluated in Section 7.6.2.

A Lightweight Auxiliary Work Platform has been provided as a temporary work platform on the refueling floor [redacted] The Lightweight Work Platform is a fully engineered portable work platform that provides personnel access over the reactor cavity, spent fuel pool and equipment storage pool. It provides personnel the ability to perform inspections, servicing, or modification activities in parallel with fuel moves from the existing Refuel Platform. It is equipped with a 950-lb jib hoist. The Lightweight Work Platform will not be used to move fuel. Administrative controls are in place to prohibit use of the Lightweight Work Platform hoist for fuel movement.

The 360 Degree Work Platform is a single-unit, multiple configuration structure that can accommodate a wide variety of outage refueling work scopes. The platform is located on the refuel floor in the northwest corner and is used only during refueling outages. The platform is capable of supporting in-vessel visual inspection, in-vessel servicing, and modification work scopes. It is a steel structure, temporarily installed over the reactor vessel cavity, and is supported by structural feet sitting on the reactor cavity ledge. It has a centralized access opening, which allows multiple work teams to simultaneously have access to the reactor vessel internal components.

The 360 Degree Work Platform is equipped with two 1000 lb. capacity hoists for transportation of tools, cameras, calibration, etc. in the cavity area. The hoists are not intended for use in moving fuel and are not equipped with grapple hooks. Procedural and administrative controls are also used to prevent these hoists from handling fuel.

9.1.6.5 Storage Equipment

Specially designed fuel storage racks are provided. Additional storage equipment is listed in Table 9.1-6. For fuel storage racks' description and fuel arrangement, see Sections 9.1.1 and 9.1.2.

Defective fuel assemblies may be placed in damaged fuel containers, which in turn are normally stored in the defective fuel storage racks. Each can is adaptable for individual sipping. For channel removal, the can may be removed from the rack and placed in the fuel preparation machine where the can cover is removed. Before shipment, the sipping head is removed and the shipping lid is installed. Provisions for dry sipping are provided. This system allows for the detection of leaking fuel pins in the fuel pool during refueling.

Fuel sipping techniques and equipment have been improved over the years. Several nuclear services providers are available to perform the "sipping" of nuclear fuel. The use of their equipment and procedures will be reviewed prior to implementation.

9.1.6.6 Under Reactor Vessel Servicing Equipment

The necessary equipment to remove several control rod drives during a refueling outage is provided. An equipment handling platform with a rectangular open center is provided. This platform can rotate to provide space under the vessel so the control rod drive can be lowered and removed. A thermal sleeve installation tool is used to rotate the thermal sleeve within the CRD housing. Sleeve rotation permits disengagement of the guide tube. A rope and pulley integral with the tool permits complete sleeve removal. Special tools and instruments to service and test individual control rod hydraulic units are also provided.

Miscellaneous wrenches, a tapering tool, and a flaring tool are provided to install and remove the neutron detectors. The spring reel pulls the fixed incore detector string into the incore guide tube and also seals the opening in the incore flange during incore servicing. A drain can be opened after incore insertion to drain any residual water. Correct seating of the incore string is indicated when drainage ceases.

Undervessel servicing equipment can be provided by several nuclear services vendors. Their equipment is of the same form and function as the equipment at the DAEC but it has been modified to perform the tasks more efficiently. The equipment and procedures used by these vendors is reviewed and approved by DAEC personnel prior to their use.

Additional nuclear system tools and servicing equipment not covered in these paragraphs are listed in Table 9.1-6.

9.1.6.7 Dryer-Separator Pool Seal

During a refueling outage, the dryer and separator assemblies are placed in the dryer-separator pool. The dryer-separator storage pool may be filled with water to reduce radiation exposure for people on the refueling floor. The dryer-separator pool seal prevents leakage of water between the dryer-separator storage pool and the reactor cavity when the storage pool is flooded and the reactor cavity is drained for servicing.

REFERENCES FOR SECTION 9.1

1. Letter from Thomas A. Ippolito, NRC, to Duane Arnold, Iowa Electric, Subject: Amendment No. 45 to Facility License No. DPR-49 for the Duane Arnold Energy Center, dated July 7, 1978.

2. U. S. Nuclear Regulatory Commission, Single-Failure Proof Cranes for Nuclear Power Plants, NUREG-0554, May 1979.

3. U. S. Nuclear Regulatory Commission, Control of Heavy Loads at Nuclear Power Plants, NUREG-0612, July 1980.

4. Letter from Robert M. Pulsifer, NRC, to Lee Liu, IES Utilities, Subject: Amendment No. 195 to Facility License No. DPR-49 for the Duane Arnold Energy Center, Dated February 2, 1994.

5. Holtec Report, HI-92889, Licensing Report for Spent Fuel Storage Capacity Expansion DAEC, transmitted to NRC along with RTS-252, NG-93-0566, dated March 26, 1993.

6. Letter from K. Young, IES Utilities, to W. Russell, NRC, dated November 15, 1994, NG-94-3874.

7. Holtec Report, HI-971746, Thermal Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie, transmitted to NRC along with RTS-296, NG-97-1578, dated October 3, 1997.

8. Holtec Report, HI-971708, Criticality Safety Evaluation of the Spent Fuel Storage Racks in the Duane Arnold Energy Center for Maximum Enrichment Capability, dated August, 1997.

9. Deleted

10. Holtec Report, HI-971746, Thermal-Hydraulic Evaluation of the DAEC Spent Fuel Pool with RHR Intertie, transmitted to NRC along with TSCR-040, NG-00-1904, dated November 17, 2000.

11. Letter from D. Hood, NRC, to M. Peifer, NMC, Subject: Issuance of Amendment Regarding Reactor Building Crane (TAC No. MB 8003), dated May 16, 2003.

12. Letter from D. Hood, NRC, to M. Peifer, NMC, Subject: Supplement to Safety Evaluation for Amendment No. 251 Regarding Reactor Building Crane (TAC No. MB8003), dated January 30, 2004.

2012-007 13. Holtec Report, HI-2115034 Revision 1, Criticality Evaluation of the Spent Fuel Racks at Duane Arnold with Boral Degradation, dated June 20, 2012.

2014-012 14. Deleted

2017-006 15. Deleted

16. NextEra Energy Report, Criticality Safety Evaluation of the PaR Racks in the Duane Arnold Spent Fuel Pool, Attachment 6 to NG-16-0052, March 2016.

Table 9.1-1

DAEC EXISTING AND PROJECTED FUEL DISCHARGE SCHEDULE

Cycle Number	Batch Size	Number of Bundles	Date Discharged
1A	4	4	June, 1975
1B	84	88	February, 1976
2	100	188	March, 1977
3	88	276	March, 1978
4	88	364	February, 1980
5	84	448	March, 1981
6	128	576	February, 1983
7	120	696	February, 1985
8	128	824	March, 1987
9	120	944	September, 1988
10	104	1048	June, 1990
11	104	1152	February, 1992
12	128	1280	July, 1993
13	128	1408	February, 1995
14	120	1528	September, 1996
15	120	1648	May, 1998
16	128	1776	December, 1999
17	136	1912	June, 2001
18	152	2064	June, 2003
19	152	2216	June, 2005
20	152	2368	June, 2007
21	152	2520	June, 2009
2017-002 22	152	2672	June, 2011
23	152	2824	June, 2013
24	368	3192	June, 2015

Note: This table provides the basis for decay heat evaluations. The assumptions made at the end of the plant licensing life are solely to provide a bounding analysis condition.

Table 9.1-2

FUEL POOL COOLING AND CLEANUP SYSTEM DESIGN SPECIFICATIONS

System Function	System Specification
Total pool, well, and pit volume	[redacted]
Fuel storage pool volume	[redacted]
System design flow	[redacted] (each of 2 units)
Maximum flow	[redacted]
Design heat load	[redacted]
Maximum heat load	[redacted]
Pump characteristics	[redacted] (each of 2 units)
Heat Exchanger - capacity (30°F delta T)	[redacted]
Filter-demineralizer	190 sq. ft, 400 gpm
	20 psi max dp (dirty)
Holding pump flow	19 gpm
Precoat flow	285 gpm
Flow control valve	100 psi max dp
	10 psi min dp

Note 1: See Section 5.4 for RHR heat exchanger capacity.
Note 2: Outage Management Guidelines contain typical decay heat and system capacity curves.

Table 9.1-3

LOADING COMBINATIONS AND FACTORED ALLOWABLES FOR PaR RACKS

Load Combinations	Factored Allowable
D+L	S
D+L+E	S
D+L+To	1.5S
D+L+To+E	1.5S
D+L+Ta+E	1.6S
D+L+DF	1.6S
D+L+Ta+E[1]	2.0S

S = normal allowable stresses according to section 9.1.2.3.3.2
D = dead load, buoyant rack weight
L = live load, buoyant fuel weight
To = operating thermal loads
Ta = accident thermal loads
E = OBE seismic loads including impact of fuel and modules
E[1] = SSE seismic loads including impact of fuel and modules
DF = Dropped fuel bundles loads

LOADING COMBINATIONS AND STRESS LIMITS FOR HOLTEC RACKS

Load Combinations	Stress Limit[a]
D+L	Level A service limits
D+L+To	Level A service limits
D+L+To+E	Level A service limits
D+L+Ta+E	Level B service limits
D+L+To+Pf	Level B service limits
D+L+Ta+E[1]	Level D service limits[b]
D+L+Fd	Level D servic limits[b]

D = dead weight-induced stresses (including fuel assembly weight)
L = live load (0 for the structure, since there are no moving objects in the rack load path
To = differential temperature induced loads (normal or upset condition)
Ta = differential temperature induced loads (abnormal design conditions)
E = Operating Basis Earthquake (OBE)
E[1] = Safe Shutdown Earthquake (DBE)
Fd = force caused by the accidental drop of the heaviest load from the maximum possible height
Pf = Upward force on the racks caused by postulated stuck fuel assembly

[a] ASME Boiler and Pressure Vessel Code, Section III, Subsection NF.
[b] The functional capability of the fuel racks should be demonstrated.

Table 9.1-4

OBJECTS MOVED OVER REACTOR CORE

Object	Approximate Size	Approximate Weight
[redacted]		

2011-013

2010-007

Table 9.1-5

OBJECTS MOVED OVER SPENT-FUEL POOL

Object	Approximate Size	Approximate Weight (lb)	Height Over Pool Floor	Frequency
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]

[redacted]

Table 9.1-6 Sheet 1 of 2

TOOLS AND SERVICING EQUIPMENT

Fuel servicing equipment

- Fuel preparation machines
- New-fuel inspection stand
- Channel bolt wrench
- Channel handling tool
- Channel transfer grapple
- Fuel container sipping head
- Fuel inspection fixture
- Channel gauging fixture
- General purpose grapple

Servicing aids

- Pool tool accessories
- Actuating pole
- General area underwater lights
- Local area underwater lights
- Drop lights
- Underwater TV monitoring system
- Underwater vacuum cleaner
- Viewing aids
- Light support brackets
- Incore cutting tool
- Incore manipulator

Reactor vessel servicing equipment

- Reactor vessel servicing tools
- Steam line plugs
- Shroud head-bolt wrenches
- Vessel nut handling tool
- Head holding pedestals
- Head nut plus washer racks
- Head stud rack
- Dryer-separator sling (strongback)
- Head strongback
- Service platform support (Reactor Flange Protector)
- Refueling Shield (Cattle Chute) Strongback
- Fuel Transfer Slot Plug Strongback
- HI-TORQUE Pole Handling System
- 360 Degree Work Platform

Table 9.1-6 Sheet 2 of 2

TOOLS AND SERVICING EQUIPMENT

In-vessel servicing equipment

- Instrument strongback (for single LPRM)
- Multiple LPRM strongback
- Control rod grapple
- Control rod guide tube grapple
- Fuel support grapple
- Grid guide
- Control rod latch tool
- Instrument handling tool
- Orifice holder (peripheral)
- Blade guide
- Fuel bail cleaner
- Fuel bundle sampler
- Lightweight Work Platform

Refueling equipment

- Refueling equipment servicing tools
- Refueling platform equipment

Storage equipment

- Spent-fuel storage racks
- Channel storage racks
- Storage racks (control rod/defective fuel)
- Defective fuel storage containers
- New-fuel storage racks

[redacted]

[redacted]

[redacted]

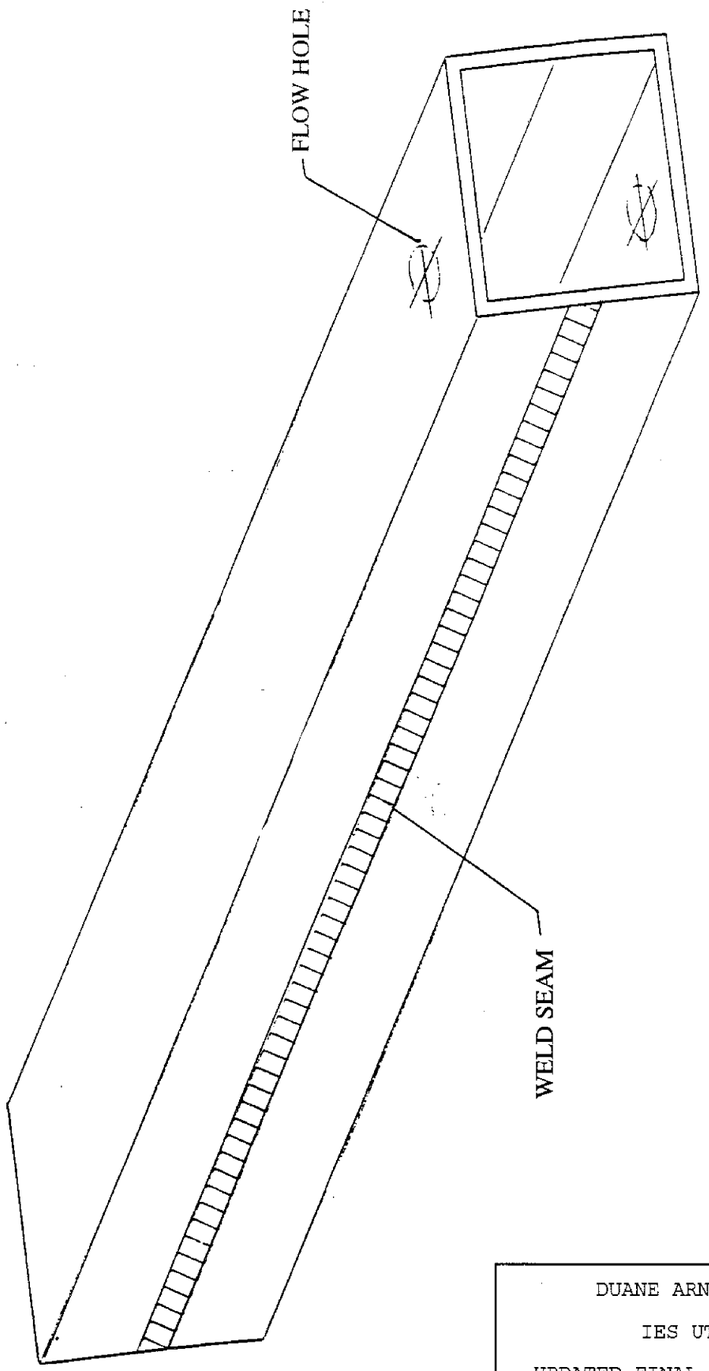
[redacted]

[redacted]

[redacted]

[redacted]

[redacted]



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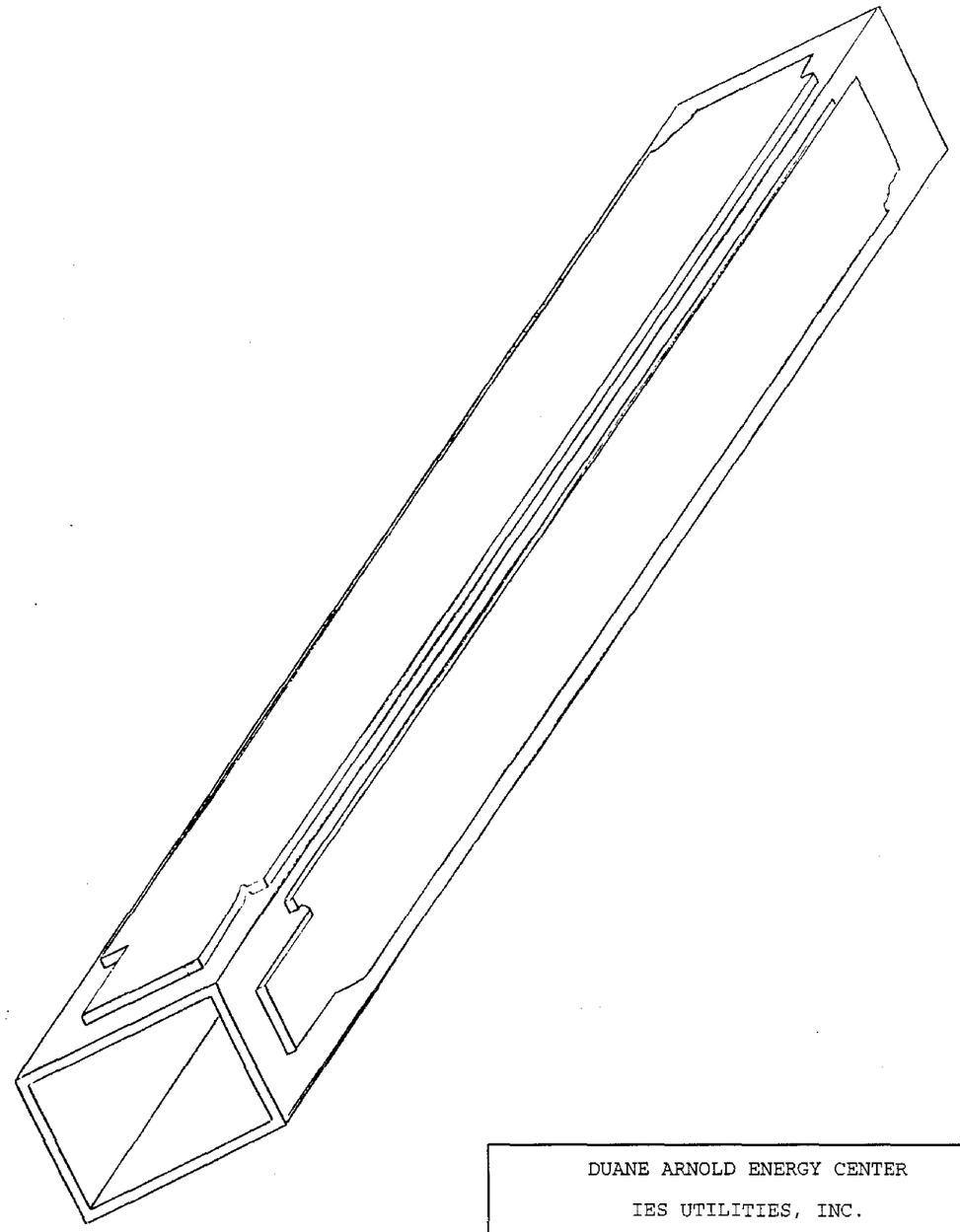
UPDATED FINAL SAFETY ANALYSIS REPORT

Holtec Stainless Steel Spent Fuel Racks
Seam Welding Precision Formed Channel

Figure 9.1-30



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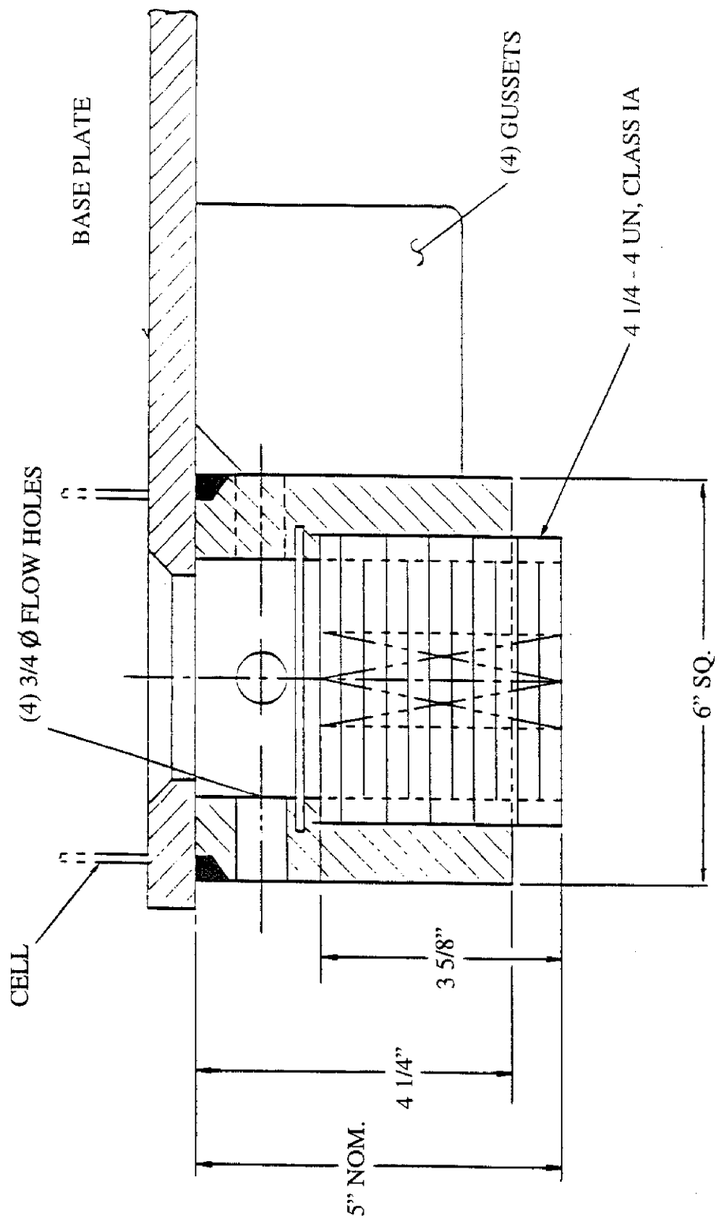
Holtec Stainless Steel Spent Fuel Racks
Sheathing Shown Installed on the Cell

FIGURE 9.1-31



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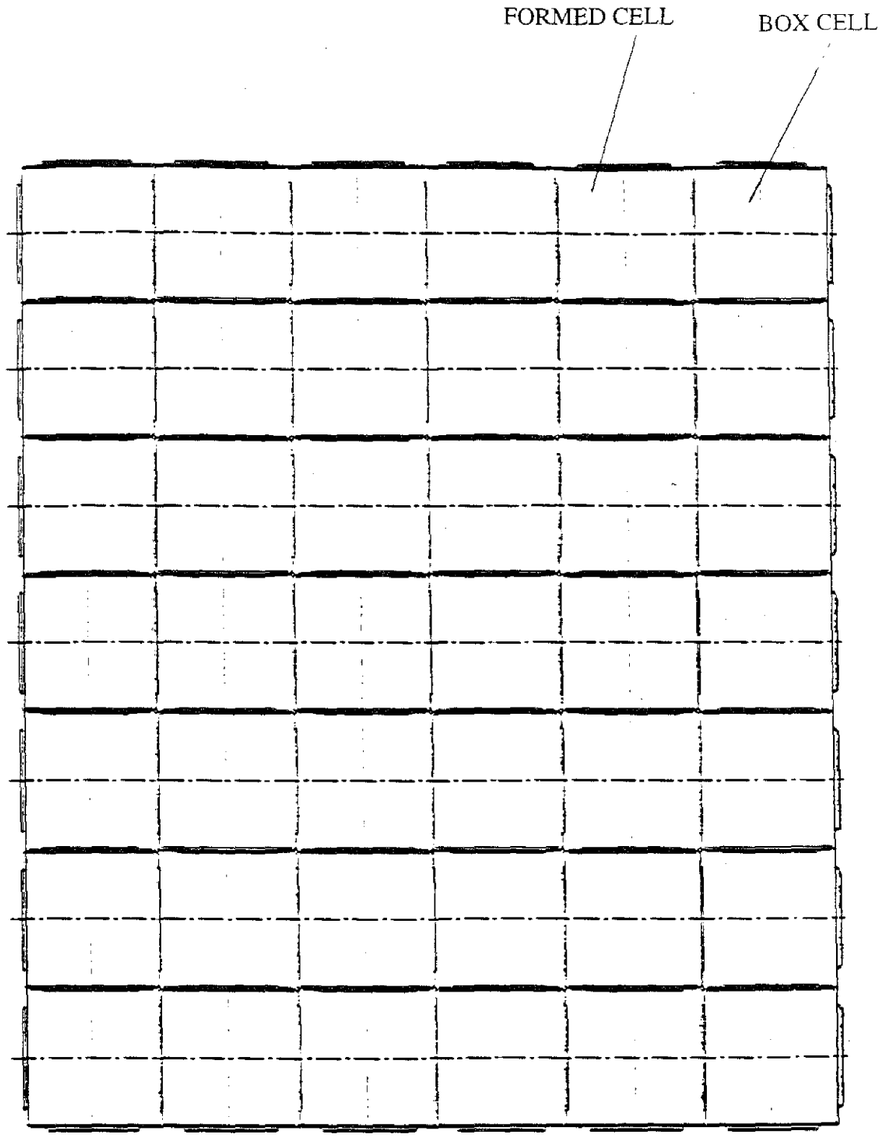
Holtec Stainless Steel Spent Fuel Racks

Adjustable Support

Fig. 9.1-32



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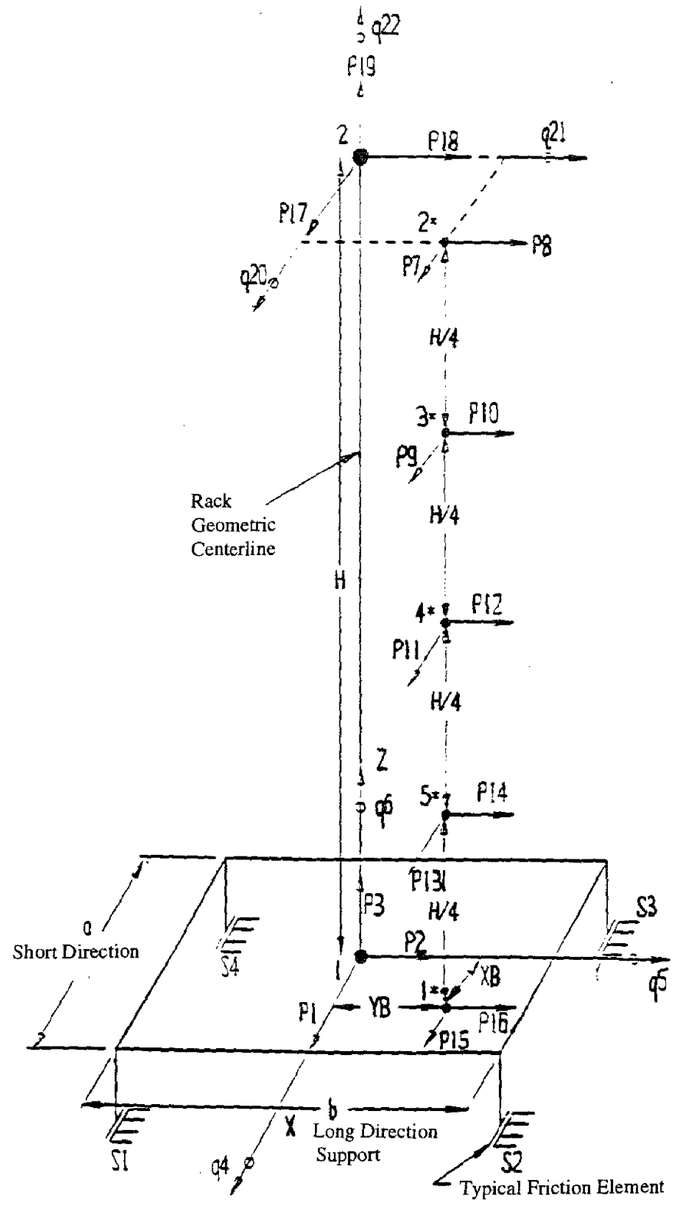
Holtec Stainless Steel Spent Fuel Racks

Cross Sectional View of An Array of Storage Location
Fig. 9.1-33

[redacted]



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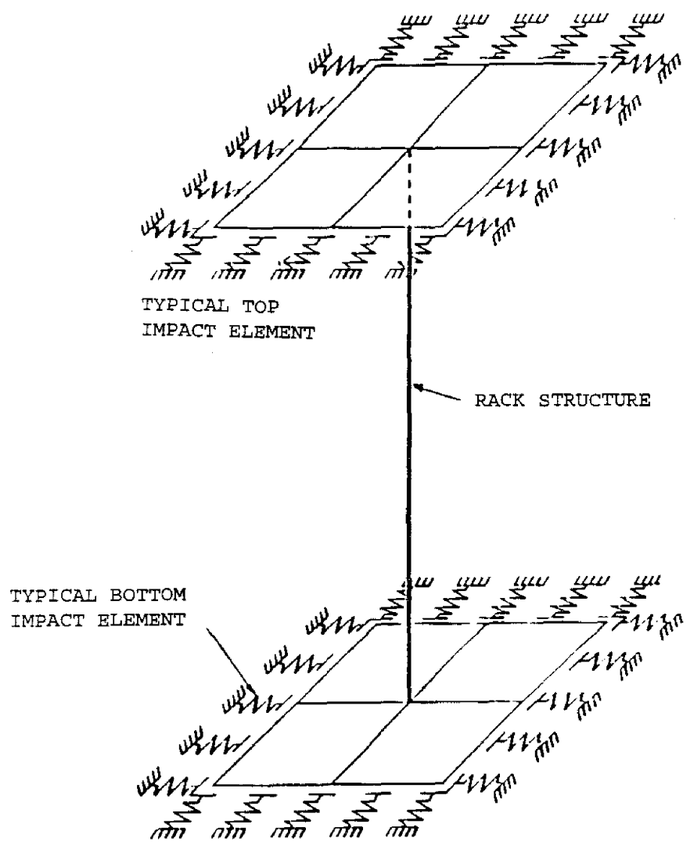
Holtec Stainless Steel Spent Fuel Racks

Schematic Model for DYNARACK

Fig. 9.1-36



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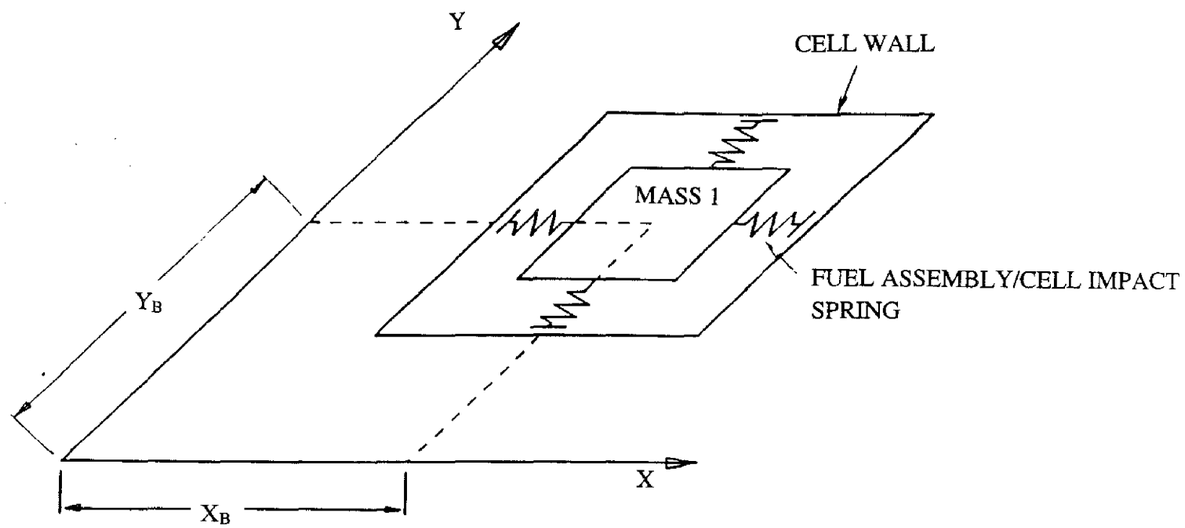
Holtec Stainless Steel Spent Fuel Racks

Rack-To-Rack Impact Spring

Fig. 9.1-37



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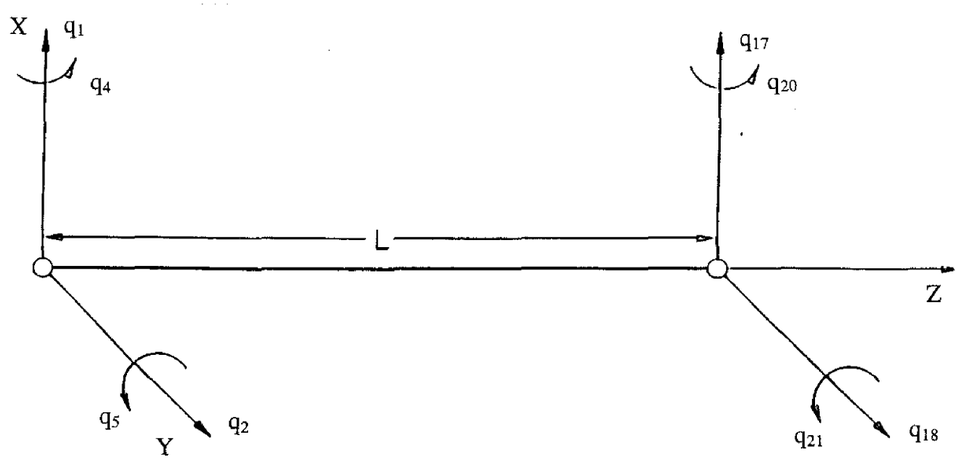
Holtec Stainless Steel Spent Fuel Racks
Fuel-To-Rack Impact Spring

FIGURE 9.1-38



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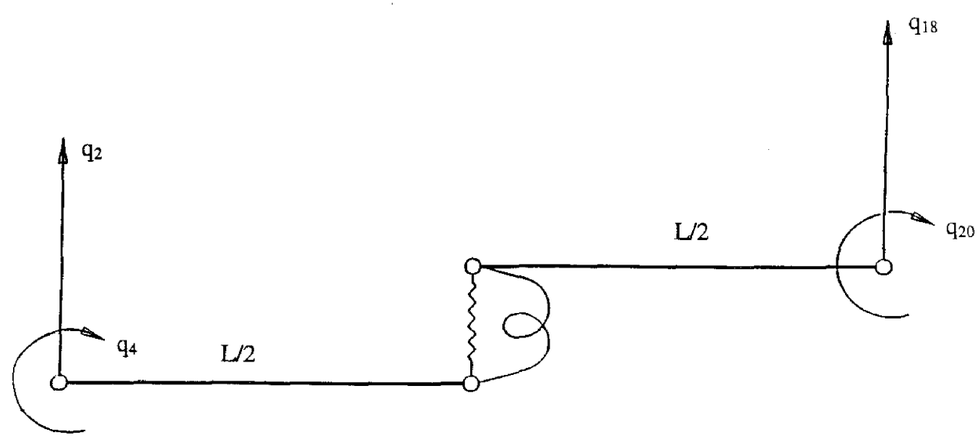
Holtec Stainless Steel Spent Fuel Racks Degree-of-Freedom Modeling Rack Motion Location

FIGURE 9.1-39



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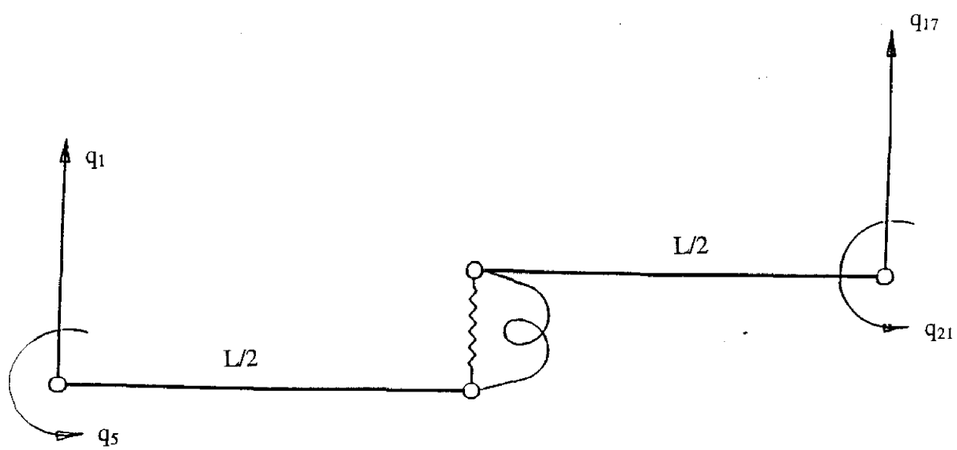
Holtec Stainless Steel Spent Fuel Racks
Rack Degree-of-Freedom for Y-Z Plane Bending

Figure 9.1-40



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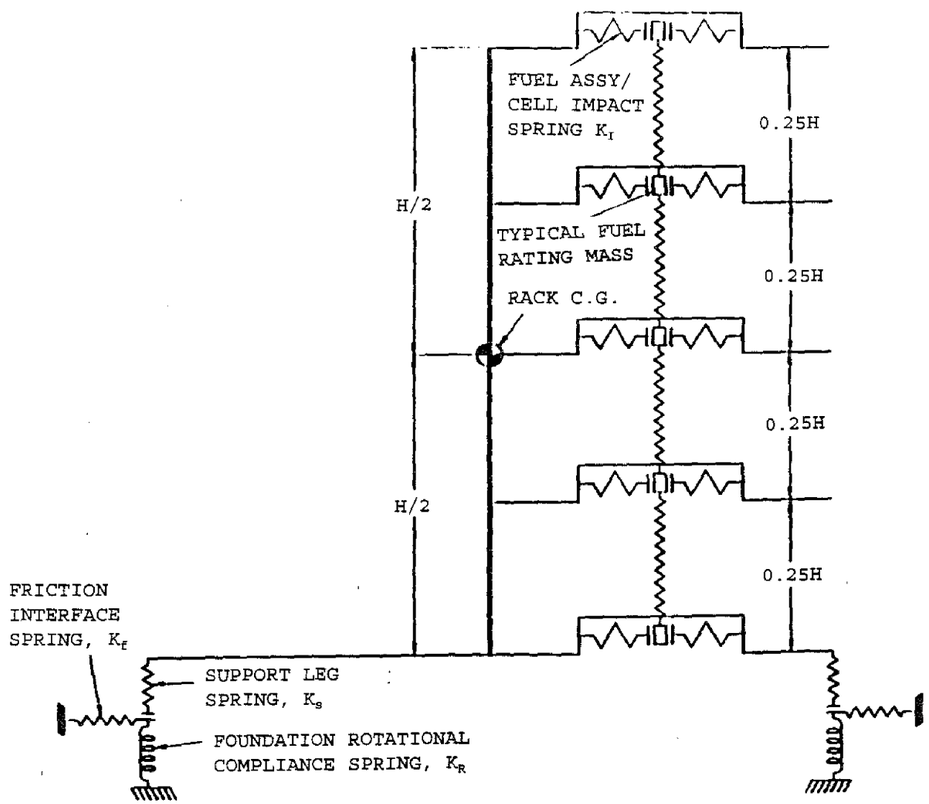
Holtec Stainless Steel Spent Fuel Racks
Rack Degree-of-Freedom for XZ Plane Bending

Figure 9.1-41



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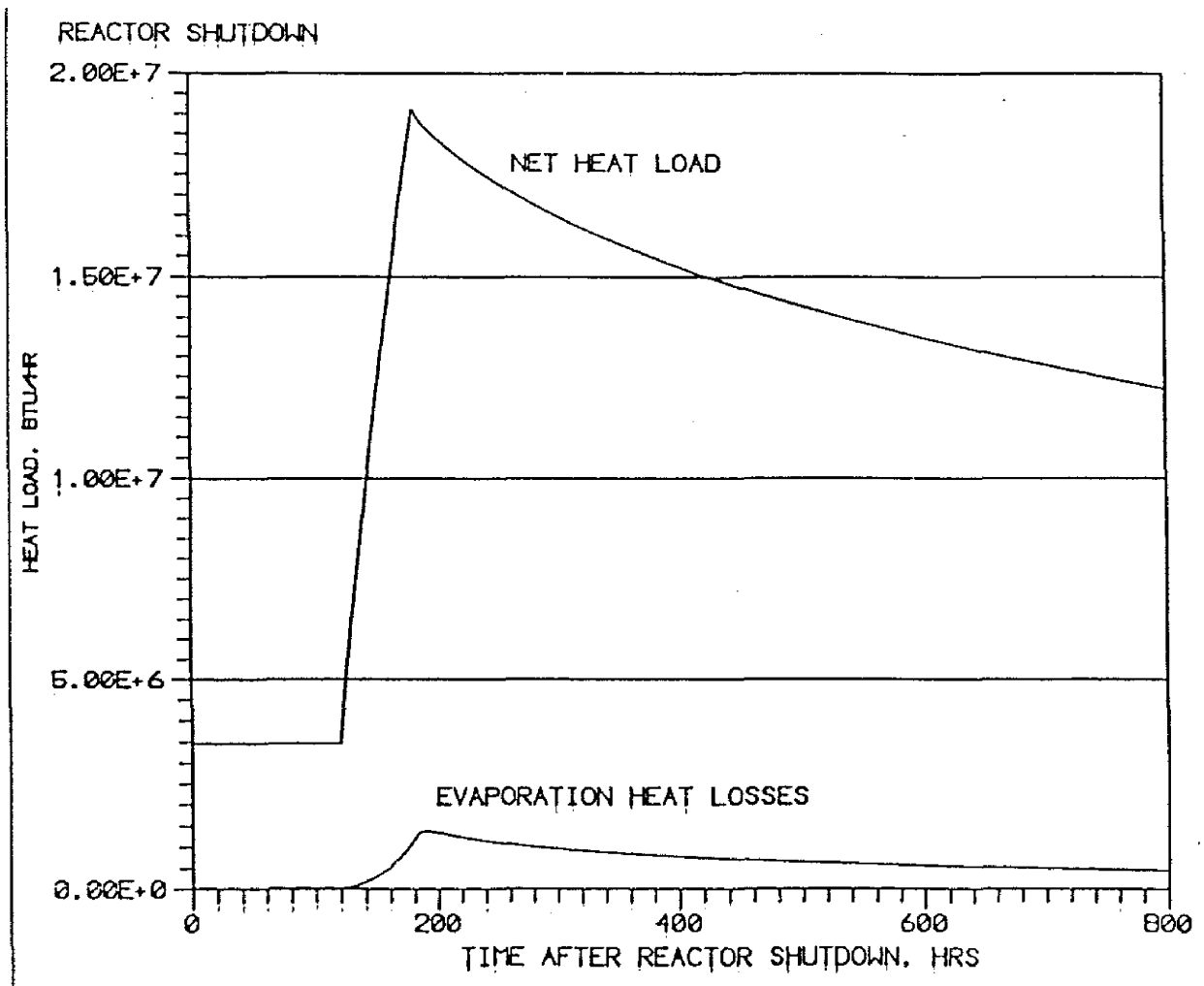
Holtec Stainless Steel Spent Fuel Racks

2-D View of the Rack Module

Fig. 9.1-42

HOLTEC INTERNATIONAL

DAEC, SFP, EOC FULL CORE OFFLOAD, TWO TRAINS - CASE 3



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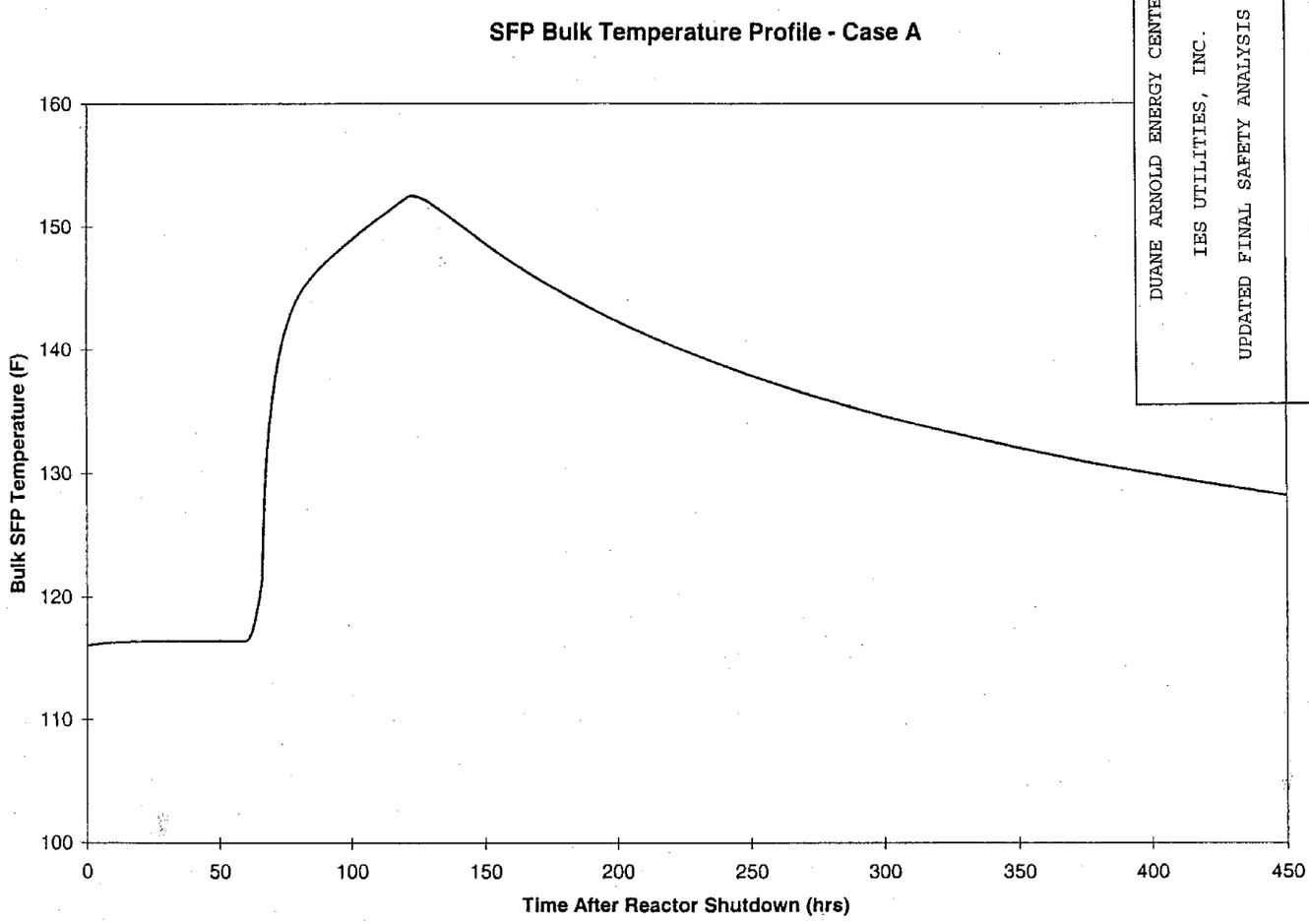
Analyzed Spent Fuel Pool Net Decay Heat Load and Heat Losses for Case 3

Figure 9.1-45



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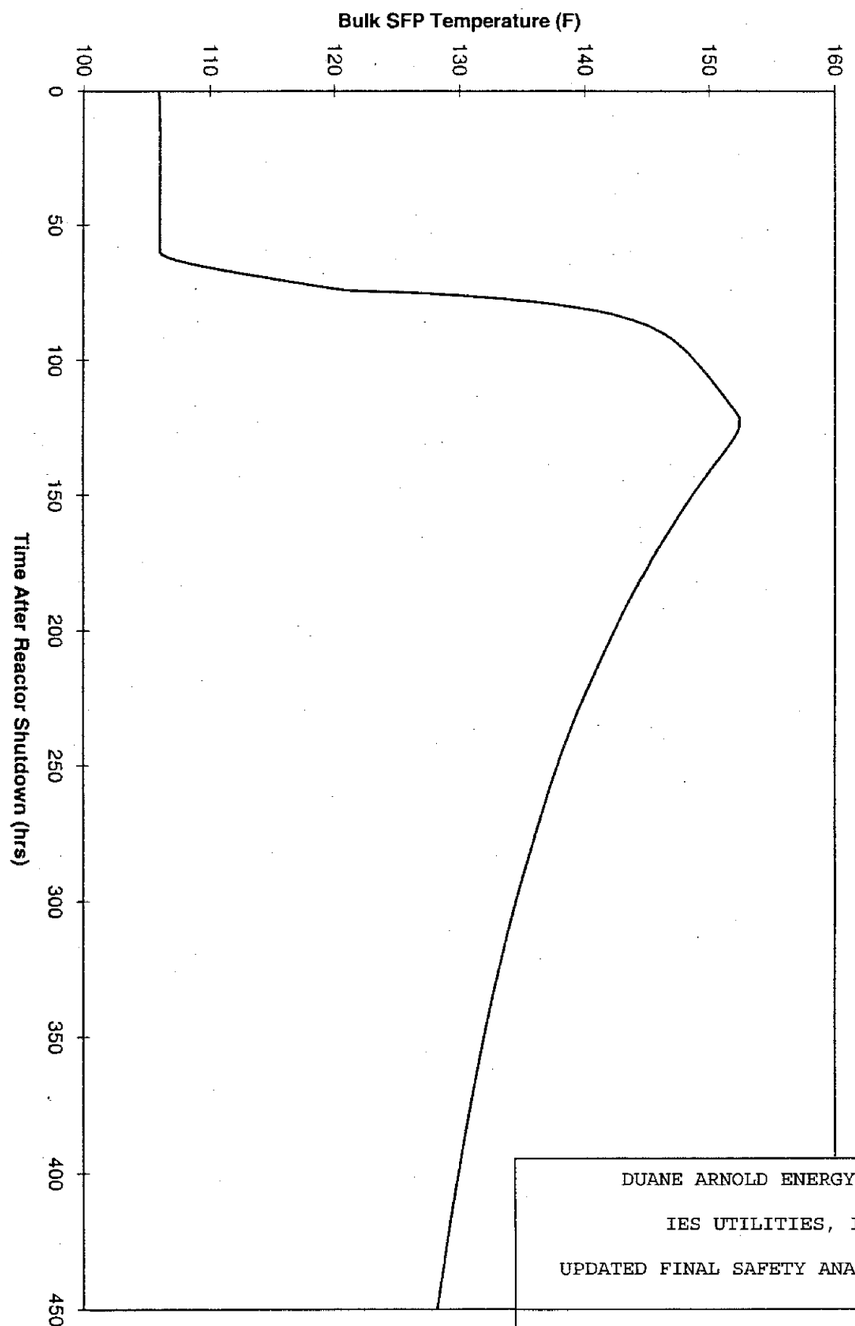
SFP Bulk Temperature Profile
Case A
Figure 9.1-47

Revision 17 - 10/03



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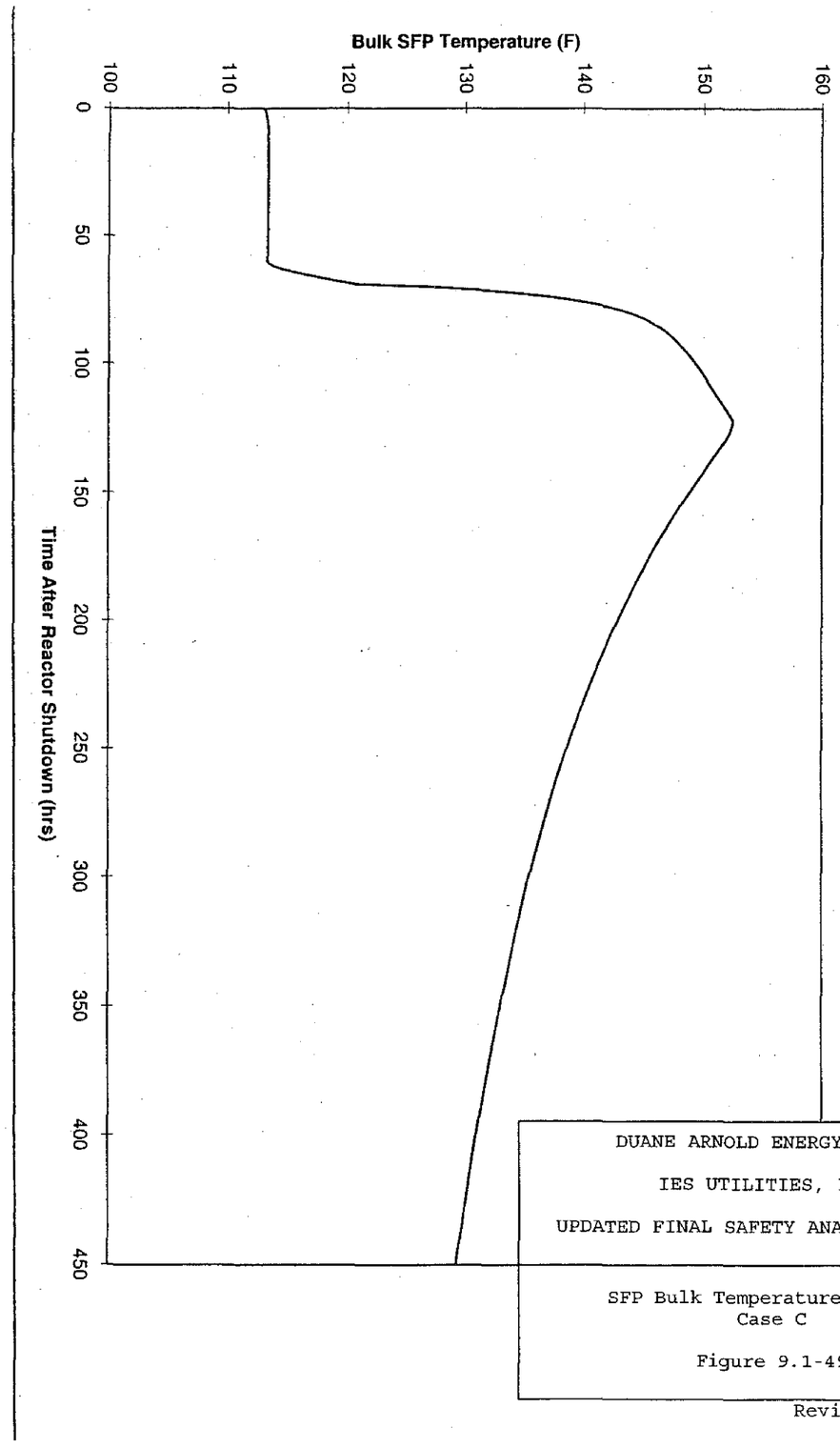
SFP Bulk Temperature Profile
Case B

Figure 9.1-48



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SFP Bulk Temperature Profile
Case C

Figure 9.1-49

9.2 WATER SUPPLY SYSTEMS

9.2.1 WELL WATER SYSTEM

9.2.1.1 Design Bases

9.2.1.1.1 Power Generation Objectives

The power generation objectives of the well water system are to provide cooling water for all the plant ventilation cooling units, supply potable water for the plant requirements, and supply the required water for demineralizer makeup. Discharge from the plant ventilation cooling units is reused for cooling water in the offgas recombiner, offgas glycol refrigeration unit, and the containment N_2 compressor.

9.2.1.1.2 Power Generation Design Basis

The design is based on using the well water to remove heat from the components during startup, normal operation, shutdown, and cooldown and to discharge the water into the circulating water system as part of the makeup for that system.

9.2.1.2 Description

9.2.1.2.1 General

The system consists of four independent wells. Two have a 750-gpm pump capacity, one has a 1200-gpm pump capacity, and one has a variable speed pump with a maximum output of 1650 gpm (see Figure 9.2-1). [redacted] All four production well locations are shown in Figure 11.2-8. All are located away from the plant. The supply headers from each well join one main supply header before entering the plant.

All of these wells are sealed to prevent the collection of the less desirable ground water from the more shallow aquifers. The wells are supplied by the deeper Devonian/Silurian formations. A discussion of the ground water in the site vicinity is presented in Section 2.4.13.

Should radwaste enter the ground water at the plant, it would flow away from the wells toward the river.

A backflow preventer is provided to ensure that contaminated water cannot flow into the wells or into the potable water system. The backflow preventer is shown in Figure 9.2-1.

During startup, normal operation, shutdown, and cooldown, combinations of one, two, or three pumps will be in service. [redacted]

During startup, normal operation, shutdown, and cooldown, the well water system with the selected pump(s) in operation will supply the following equipment:

Note: The GPM shown adjacent to the plant equipment are nominal values for reference only. These numbers vary with system demand, winter or summer and/or day or night operation.

1. Plant ventilation cooling water.

   a. Drywell cooling units (six either train and two additional), 268 gpm, or 448 GPM with all the coolers inservice.

   b. Main plant air cooling coils, 480 gpm.

      Control building chillers (two).*

   c. Air compressors (three)

      1) Control Building/SBGT air compressors (two)*

      2) Backup instrument air compressor

   d. Control-rod drive (CRD) room coolers (two), 40 gpm total.

   e. Radwaste building cooler, 44 gpm.

   f. Reactor building cooler, 25 gpm.

   g. Recombiner room cooler, 40 gpm.

   h. Condenser area coolers (two), 160 gpm.

---

* Control building chillers and air compressors [redacted] are supplied by water discharged from the main plant air cooler, 480 gpm at 66°F.

i. Switchgear area cooler, 40 gpm.

2. Potable and sanitary water system (intermittent), 20 gpm.

3. Makeup demineralizer (100 gpm maximum), normal, 25 gpm.

4. Radwaste and machine shop rotoclones (three), 7 gpm.

5. Jockey fire pump, 25 gpm.

6. Room cooling unit for KAMAN system, [redacted]

Discharge from some of the above units is reused for the cooling of the following equipment:

1. Offgas recombiner.

2. Offgas glycol refrigeration unit.

3. Containment N_2 compressor.

If flow from the pumps in operation should fall below the total requirements tabulated above, the components listed below will be supplied until additional capacity can be brought on line. The inactive pumps can be started manually from the control room. The well water cooling requirements of the plant normally exceed the capacity of one low capacity well water pump (wells A or C). Therefore, if all well water pumps become inoperable except for one of these two low capacity pumps, it will be necessary to shut down the plant until a second pump becomes available. [redacted] are capable of supplying all loads with no other Well Water pump in service.

1. Drywell cooling units (six either train and two additional), 268 gpm or 448 gpm with all coolers in service.**

2. Control building chillers, 480 gpm at 54°F.***

** The control building chillers will be switched over to the emergency service water system if well water is not available or adequate.

*** During single-pump operation, control valve [redacted] opens and the main plant air cooling coils are bypassed.

3. CRD room coolers, 40 gpm.

4. Air compressors (two), 6 gpm.

5. Makeup demineralizer, 25 gpm.

6. Potable water system (intermittent), 20 gpm.

7. Radwaste building cooler, 44 gpm.

8. Switchgear area cooler, 40 gpm.

9. Condenser area cooler, 80 gpm.

10. Radwaste and machine shop rotoclones (three), 7 gpm.

11. Reactor building cooler, 25 gpm.

12. Recombiner room cooler, 40 gpm.

Discharge from some of the above units is reused for the cooling of the following equipment:

1. Offgas recombiner.

2. Offgas glycol refrigeration unit.

3. Containment N_2 compressor.

All other components will be isolated from the well water system.

The cooling water to the control building chillers and the H & V Instrument Air compressors during the design-basis accident is supplied from the emergency service water system (see Section 9.2.3.2.2).

The well water system is the source of water for the potable and sanitary water system (Section 9.2.1.2.2), the makeup water treatment system (Section 9.2.1.2.3) and the training center. In addition, the well water system provides cooling water for the plant ventilation systems.

The well water system includes a chemical injection system to add chemicals to the well water to mitigate the formation of calcium carbonate buildup on heat-exchanger surfaces and related piping and components.

9.2.1.2.2 Potable and Sanitary Water System

9.2.1.2.2.1 Design Bases

Power Generation Objective

The power generation objective of the potable and sanitary water system is to provide water for drinking and sanitary purposes.

Power Generation Design Bases

1. Well water is supplied in sufficient quantities and will be treated, as required, to satisfy the plant potable water demand.

2. System pressure is reduced, as required, to ensure that no plumbing equipment or fixture outlet is subjected to a static pressure greater than 65 psig under normal operating conditions.

9.2.1.2.2.2 Description

Water for drinking and sanitary use is supplied from wells on the site. The water is filtered and purified as necessary to meet applicable drinking water standards. Shower and lavatory waste water that does not contain radioactive material is directed to the sewage disposal facilities.

Potable water for plant use is supplied by a takeoff from the well water system supply header upstream of a reduced pressure backflow preventer. All takeoffs to other components supplied from the well water system are located downstream of the backflow preventer.

The possibility of contaminating the potable water is therefore impossible because the pressure in the well water system will always be higher than the pressures at the components supplied.

A shutdown of the well water pumps, reducing the upstream (supply) pressure, will immediately close both check valves in the backflow preventer. This will ensure that there is no backflow from any system components to the potable supply header.

9.2.1.2.2.3 Testing and Inspection Requirements

The system was inspected for compliance with the codes having jurisdiction. Before being placed in service, the domestic water piping proved leaktight while subjected to a hydrostatic test pressure of 100 psig. The system was then disinfected, flushed, and placed in service. The potable and sanitary water system is in continual use and requires no periodic testing. Inspection and maintenance is provided during periods when individual pumps are shut down.

9.2.1.2.3 Makeup Water Treatment System

9.2.1.2.3.1 Design Bases

Power Generation Objective

The power generation objective of the makeup water treatment system is to provide a supply of treated water suitable for makeup to the plant and reactor coolant cycles and other demineralized water requirements. The water is to be of the required purity to prevent corrosion, minimize deposits on heat transfer surfaces and mechanical parts, and minimize the amount of foreign material subjected to irradiation activation.

Power Generation Design Bases

The makeup water treatment system is designed to

1. Process well water by means of a portable demineralizer. The originally installed demineralizers are not used at this time.

2. Maintain water purity by the correct choice of storage and piping material.

3. Provide makeup water of reactor coolant quality.

4. Provide an adequate supply of treated water for all plant operating requirements.

5. Provide an adequate supply of treated water to the condensate storage tank for refueling (see Section 9.2.6).

6. Provide an adequate supply of treated water for other miscellaneous requirements (e.g., startup flushing and auxiliary boiler). These demands could be required when the reactor and power generation plant is out of commission.

9.2.1.2.3.2 Description

[redacted] The water is processed through the makeup demineralizers and stored. The storage tank and piping are constructed of materials to prevent metallic contamination of the water.

The makeup water system portable demineralizers are located in the truck bay of the turbine building and receive supply water from onsite wells. An alternate location for the portable demineralizers is outside the south turbine building rollup door.

[redacted]

The original makeup system is not used at this time however portions are still used to monitor and transfer water. The mobile demineralizer is equipped with anion, cation and mixed polishing beds. The mobile unit has been specifically designed to produce water acceptable for use within the nuclear power industry. The plant typically makes water periodically to maintain adequate inventory in the storage tank(s).

The demineralizer system and piping are coated carbon steel or stainless steel to prevent the contamination of the makeup water.

The demineralized water is stored in a 50,000-gal lined carbon steel tank from which it is pumped by two 100-gpm transfer pumps (one standby) to supply the following plant requirements for demineralized water or as makeup to the condensate storage tank (see Section 9.2.6):

1. [redacted]

2. Condenser, makeup and reject.

3. Heating system fill and makeup.

4. Stator winding liquid cooler unit.

5. Reactor building cooling water system.

6. Residual heat removal (RHR) heat exchanger - flush.

7. Offgas system - components.

8. Standby liquid control tank.

9. [redacted]

10. Laboratories.

11. Decontamination sinks.

12. Drywell area.

13. Hose stations.

14. Other miscellaneous locations throughout the plant.

The effluent from the portable demineralizers is continually monitored. On high conductivity, the demineralizer is isolated from the storage tank until the conductivity returns to acceptable limits.

The quality of water from the makeup demineralizers is maintained within the following limits:

Conductivity	< 0.1 μmho/cm at 25°C
Chlorides (as C1)	< 5 ppb
Silica (as SiO_2)	< 20 ppb

9.2.1.3 Safety Evaluation

The well water system, the potable and sanitary water system, and the makeup water treatment system are not safety related.

The DAEC has conducted a review of all cooling water systems inside containment in response to IE Bulletin 80-24, Prevention of Damage Due to Water Leakage Inside Containment (10/17/80 Indian Point 2 Event). Details of the results of the review were reported in Reference 1. A summary of the results is as follows. The drywell cooling water system, a subsystem of the well water system, is the only cooling water system inside containment that fits the NRC definition of an "open" cooling water system. The drywell cooling water passes through the four control rod drive cooling coils, six drywell cooling coils, and four recirculation pump cooling coils all physically located inside the drywell and is then sent to the circulating water system via a connection with the RHR service and emergency service water header. The drywell cooling water system supply and return containment isolation valves close automatically

only upon receipt of a reactor vessel low-low-low water level isolation signal. The drywell cooling water is not open to either the reactor or the drywell. In the event of a LOCA, failure of the drywell cooling water system piping would have to occur in order for there to be any communication between the reactor or the drywell environment and the area outside the drywell.

There are two redundant drywell cooling water subsystems. There are six cooling coils in each subsystem, plus two coils which may be operated by either subsystem. It is possible to isolate either an entire drywell cooling subsystem or any individual cooling coil within a subsystem inside the drywell without entering the drywell. However, the two cooling coils which may be operated by either subsystem may be isolated as a group, but not individually.

There is a drywell floor drain sump (unidentified leakage) and a drywell equipment drain sump (identified leakage). Each sump has two sump pumps inside the drywell which are designed to handle all normal identified equipment drainage and a significant unidentified source of leakage. The sumps are arranged such that high level in one sump will overflow into the other sump. Unidentified leakage is measured every 4-hr period in order to ensure that the Technical Specifications, allowable drywell leakage, is not exceeded. Drywell unidentified leakage is determined by run times on the floor drain sump pumps utilizing flow instrumentation (integrator and timers). Also, each sump has a high level switch and alarm which annunciates in the control room and each set of sump pumps has a "pump out" timer which will activate an alarm in the control room should the pump run excessively, a condition which would be indicative of a leak from some source inside containment or of a pump failure. Also, drywell atmosphere radiation detectors are provided (air sampling system) as a backup method of leak detection. If the sump pumps fail, the DAEC drywell/suppression pool arrangement is such that a significant accumulation of water inside the drywell would drain into and be detected in the suppression pool. There is direct communication between the drywell and the suppression pool such that an accumulation of approximately 2 ft of water on the floor of the drywell would spill into the downcomers to the suppression pool. Any significant amount of water would be detected by suppression pool level instrumentation. High and low suppression pool level alarms are provided so that an alarm actuates before the Technical Specification torus water volume limits are reached.

The Technical Specification require action to be taken if the unidentified drywell leakage is observed to increase by 2 gpm in any 24-hr period.

9.2.1.4 Testing and Inspection Requirements

The site has elected to perform periodic testing of the backflow preventers in accordance with local requirements.

9.2.1.5 Instrumentation Requirements

[redacted] The indicating instruments have high-flow and low-flow alarms.

9.2.2 RIVER WATER SUPPLY SYSTEM

9.2.2.1 Design Bases

9.2.2.1.1 Safety Objective

The safety objective of the river water supply system is to provide enough flow to meet all emergency plant requirements for cooling, particularly for the RHR service water and emergency service water systems.

9.2.2.1.2 Safety Design Bases

1. Pumping equipment activates automatically following a design-basis loss-of-coolant accident (LOCA) or loss of offsite power (LOOP), including the combination of LOOP and LOCA.

2. Equipment is housed in a Seismic Category I structure that is designed so that water may be obtained from the source (Cedar River) under any condition of river flow.

3. The full capacity of the system is available to the essential safeguard systems at all times.

9.2.2.1.3 Power Generation Objective

The power generation objective of the river water supply system is to provide a dependable supply of water for all normal cooling requirements for the plant.

9.2.2.1.4 Power Generation Design Basis

Sufficient flow is provided to meet all normal plant requirements for cooling.

9.2.2.2 Description (Figure 9.2-2)

[redacted]

2015-005 [redacted] A detailed discussion of the availability of water for an ultimate heat sink under all design conditions is contained in Amendment 16 to the DAEC PSAR.

Fig 9.2-3 and 9.2.4 depict the [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

2015-005 [redacted]

[redacted]

DAEC retained the services of University of Iowa's IIHR Hydroscience and Engineering in June 2002. [redacted]

Withdrawing water from shallow sand-bed rivers such as the Cedar River will always require attention to operational issues involving river sediments and debris (deadwood) build up. Dredging of the river sediment to re-align the river flow directly towards the intake structure and to keep the [redacted] is performed periodically. River surveys are conducted to determine if debris and sediment build up requires removal.

[redacted]

2015-005 [redacted]

[redacted]

[redacted] Each pump is rated at 6000 gpm at 57-ft head. Characteristics of the pumps are such that under all river flow conditions, from extreme flood conditions when static head on the system is zero to extreme low flow when head is maximum, the pumps may be operated continuously. The integrity of the pumps, motors, and associated electrical equipment is maintained to the maximum probable flood elevation. [redacted] which in turn discharges into the wet-pit sump of the RHR and emergency service water systems.

2012-019 [redacted]

2012-019 [redacted]

[redacted]

The four river water pumps deliver water through two lines to a stilling basin supplying the RHR and emergency service water systems wet-pit pump sumps to maintain these sumps at their safe operating level at all times. An overflow weir in the stilling basin makes the excess flow available as makeup to the circulating water system and general service water system. Water for one method of radwaste dilution is made available by branch connections from each of these 24-in. pipelines located immediately upstream of the flow control valves at the entrance of the lines to the stilling basin. A valve in each branch connection and a valve in the common radwaste dilution header automatically close on drywell high pressure or low reactor water level or low wet-pit sump level to ensure an adequate supply of water for the RHR and emergency service water systems. An alternative method of radwaste dilution is provided by the return flow from the RHR and emergency service water systems.

The RHR and emergency service water systems are discussed in Section 9.2.3.

Water supply requirements for the river water system are as follows: Accident requirements are 4080 gpm for the RHR service water system and 889 gpm for the emergency service water system for a total requirement of 4939 gpm. (This is based on required minimum flows. Actual flows may be as high as 6000 gpm based on actual ESW and RHRSW pump performance.) During normal operation, the river water supply requirements are dependent on evaporative dissipation from the cooling towers and cooling tower blowdown, which are variable. The maximum requirements are expected to be 8100 gpm for evaporative dissipation, including drift, and 3100 gpm for blowdown (at 3.5 cycles of concentration) for a total of 11,200 gpm. To ensure that the available flow at the intake provides the required emergency cooling flows, a water depth of at least 12 in. at the intake (Reference 6) is maintained by the Technical Specifications.

9.2.2.3 Safety Evaluation

On a Loss-of-Offsite Power, the running river water pump will be automatically load shed as essential bus voltage drops to less than 20%. When the bus is re-energized, the pump selected for automatic start will start immediately if it was not previously running. If the pump selected for automatic start was previously running, 2 minutes must elapse between pump trip and pump restart in order to ensure that the pump column has drained. A 2 minute timer in the pump control logic provides this protection. Valves at the pump house will go to their fail-safe position and ensure that the entire output is available to the safeguard system. Alternative or standby pumping capacity is available by manually connecting the idle pumps to the essential buses.

9.2.2.4 Testing and Inspection Requirements

As part of the plant normal preventive maintenance activities, the river water system will be periodically inspected during service. Pumps and auxiliary equipment can be maintained, put into service, and tested without affecting the system operational objectives. The frequency and scope of periodic maintenance of the pumps and equipment will be in accordance with plant practices, manufacturer's recommendations and operating history.

The DAEC has conducted an evaluation effort in response to IE Bulletin 81-03 and determined that there are no Corbicula (Asiatic clam) and Mytilus (mussel) present in the vicinity of the DAEC that could cause flow blockage problems of the DAEC cooling water systems. In order to detect the possible intrusion of these organisms into the system in the
2015-010 future, the DAEC conducts a sampling program of the intake structure and cooling tower basin on a semiannual basis. See References 2 and 3 for details.

9.2.2.5 Instrumentation Requirements

Instrumentation is provided at the intake structure to measure river water level and temperature. Excessive level differential across the screen will be alarmed in the control room.

9.2.3 RHR SERVICE WATER AND EMERGENCY SERVICE WATER SYSTEMS - (FIGURES 9.2-5 AND 9.2-6)

9.2.3.1 Design Bases

9.2.3.1.1 Safety Objectives

The safety objectives of the RHR service water system are to provide a reliable supply of cooling water for heat removal from the RHR system under postaccident conditions and supply a source of water if postaccident flooding of the core or primary containment is required.

The safety objective of the emergency service water system is to provide a reliable supply of cooling water to essential safeguards equipment under a loss-of-offsite-power (LOOP) condition or a LOCA, including the combination of LOOP and LOCA.

9.2.3.1.2 Safety Design Bases

1. The emergency service water system uses Cedar River water to provide long-term cooling for the essential safeguards systems both during and following the design-basis accident. The RHR service water system uses river water to remove heat from the primary containment under post-accident conditions. Both systems have the capability to return the water either to the cooling towers or directly to the river (if necessary), via the circulating water system.

2. For each of the two systems, two completely independent cooling water loops are provided to ensure redundant service water supply for emergency mode operation.

3. A normally closed cross-connection is provided between the RHR service water system supply header and the RHR system. Flow in this cross-connection is accomplished by opening two remotely operated, key-locked valves in series with a check valve, which prevents backflow from the RHR system to the RHR service water system.

4. The two emergency service water pumps [redacted] start automatically, in combination with the emergency core cooling systems following a design-basis LOCA or loss of offsite ac power. The RHR service water pumps [redacted] can be started after adequate core cooling has been ensured as described in Section 8.3.

9.2.3.1.3 Power Generation Objectives

The power generation objective of the RHR service water system is to provide cooling water to the RHR heat exchangers during conditions of normal shutdown and cooldown.

The power generation objective of the emergency service water system is to provide cooling water to all emergency equipment except the RHR heat exchangers.

9.2.3.1.4 Power Generation Design Bases

1. During normal cooldown and shutdown, the design is based on discharging the water from both systems through the 24-in. [redacted] to the circulating water system to remove heat from the systems.

2. To ensure that radioactive fluids are not released into the Cedar River or the circulating water system, the pumps of the two systems have sufficient head to maintain design flow through the RHR heat exchangers and the emergency equipment coolers, with the cold-side pressure exceeding the component hot-side pressure.

9.2.3.2 <u>Description</u>

9.2.3.2.1 RHR Service Water System

The RHR service water system provides coolant only for the RHR heat exchangers. A cross-connection (12-in. line [redacted] in Figure 9.2-5) to the RHR system provides capability for core or containment flooding. The system consists of two independent and redundant trains each containing one full-size RHR heat exchanger supplied by two half-size RHR service water pumps. Each half-size RHR service water pump is rated at 2400 gpm at 674 ft total developed

head. Analysis has shown that this rated flow of RHR service water is more than adequate to allow the RHR system to meet its design-basis requirements of 2040 gpm in the shutdown cooling mode (References 4 and 5).

2016-008 The values of parameters and results from analyses for each RHR heat exchanger used in the accident analyses is contained in UFSAR Chapter 15 Accident Analysis. During this mode, 2015-013 two RHR service water pumps are in service to supply the required 4080 gpm river water flow.

During normal shutdown, the two RHR heat exchangers and the four RHR service water pumps are in service to achieve reactor cooldown to 125°F within ~27 hr, and the RHR service water supply temperature must be 85°F or less. For temperatures over 85°F, the cooldown time will be extended accordingly. The design maximum river water temperature is 95°F. As discussed in Section 5.4.7.2.2, this slightly exceeds the original system design specification of 20 hours. However, this is an operational issue and not a safety impact. During normal shutdown, one RHR heat exchanger and two RHR service water pumps are capable of bringing the reactor to cold shutdown (reactor coolant temperature less than 212°F) in~10 hr following reactor trip using 95°F river water.

River water temperature data taken over a 29-yr period revealed only 8 days when the river water daily mean temperature exceeded 90°F. Data covering the highest recorded river water temperature over a 31-day period indicated a maximum temperature variation of 10°F in a 24-hr period, a 10.5°F daily mean temperature variation during a 31-day period, and a maximum daily mean temperature of 93.3°F. This 31-day period contained 5 days when the daily mean temperature exceeded 90°F. This data was recorded downstream of the City of Cedar Rapids. The DAEC is located 18 miles upstream of the City of Cedar Rapids.

[redacted] This temperature is monitored in accordance with the Technical Specifications.

To ensure against radioactive releases into the Cedar River or the circulating water, when the RHR service water pumps are running, the pressure of the RHR service water on the tube-side discharge will be maintained at a minimum of 20 psi higher than the process fluid on the shell-side inlet of the heat exchanger. This is accomplished by a valve controlling the RHR service water discharge on the tube side of each heat exchanger. The heat exchanger contains a differential-pressure sensor that is set to annunciate if the required 20 psi differential is not present. The control room operator manually throttles tube side flow to achieve the desired flow rate and heat exchanger differential pressure. In addition, the RHR service water discharge is monitored by a process radiation monitor (Section 11.5.4). A keylock switch with an indicator light is provided for operating each of these valves while the RHR service water pumps are not running. This provides for operator actions which may be procedurally required during an emergency.

The RHR heat exchanger design pressure is 450 psi. The maximum design RHR service water pump shutoff head is 450 psi with actual pump shutoff head being approximately 406 psi.

Piping and valving exist to apply 35 psig pressure to the demineralized water flush lines to the RHR heat exchangers. Seismic-qualified valves are used for the isolation boundary to preclude loss of RHR service water as a result of the DBE.

Each of the RHR service water cooling loops has a self-cleaning (automatic continuous backwash) strainer in the discharge header of the system loop pumps to prevent plugging during system operation. [redacted]

9.2.3.2.2 Emergency Service Water

The emergency service water system provides coolant for all emergency equipment except the RHR heat exchangers. The system consists of two independent and redundant trains, each supplied by one full-size emergency service water pump rated at 1200 gpm. Each pump supplies the following components:

1. Emergency diesel-generator.

2. RHR pump seal coolers (two).

3. RHR and core spray pump room cooling unit.

4. High-pressure coolant injection (HPCI) room cooling unit.

5. Reactor core isolation cooling (RCIC) room cooling unit.

6. Control building chiller.

7. Core spray pump motor bearing cooler.

8. RHR service water pump motor coolers.

9. Heating and ventilation instrument air compressors.

The flow requirements of equipment cooled by the emergency service water system for critical water temperatures are shown in Table 9.2-1. The flow requirement of the control building chiller at 95°F is 199 gpm. This flow is based on dual-range operation of the equipment, which causes automatic load shedding of nonessential equipment, such as the

administration building cooler, when the emergency diesel-generator starts and emergency service water is used for cooling.

The component cooling requirements given in Table 9.2-1 indicate flow rates calculated on the basis of heat-transfer rates consistent with general engineering design and adequate velocities to prevent fouling. An additional consideration is flow velocity through the heat exchangers. An additional column is listed in table 9.2-1 to show the flows required in gpm to establish a 3 ft/sec flow through the heat exchanger tubes. System flows in this range will reduce cleaning costs for the system, but are not required for the heat exchangers to meet their safety function. The component requiring the most water flow, the emergency diesel-generator, has a series arrangement of three heat exchangers whose heat-rejection requirements and cooling flow rates are essentially determined by the unit's lubrication oil cooler. The next major component requiring substantial cooling flows is the control building chiller cooler condenser, where an increase in cooling water temperatures subsequent to the design-basis accident would result in a reduction in personnel comfort.

To prevent overheating of the diesel engines even considering a hot start, the two emergency service water pumps receive an auto-start signal from the diesel generator start logic and will start provided power is available.

Each of the emergency service water cooling loops has a self-cleaning (automatic continuous backwash) strainer in the discharge header of the system loop pumps to prevent plugging during system operation.

The control building chillers are connected to both the well water system and the emergency service water system. The system having the highest pressure will supply the chillers. During normal operation and shutdown, the control building chillers are supplied from the well water system (see Section 9.2.1.2) and discharged into the general service water system. Although the Seismic Category I emergency service water system and the Non-seismic well water system are cross connected on the supply side, flow between systems is prevented by check valves. During the design-basis accident, the two systems can be isolated from each other by a manually controlled motor-operated valve allowing supply to the chillers from the Seismic Category I emergency service water system pumps only. The discharge side of the control building chillers is also cross connected to the Nonseismic well water system discharge header and the Seismic Category I emergency service water system discharge header. Valves are located in each connection between the two systems. During the design-basis accident, the valve to the emergency service water header opens automatically allowing discharge from both of the systems. The second remote-operated valve can be manually closed isolating the two systems from each other and allowing discharge only to the Seismic Category I emergency service water system header.

Each of the two redundant and independent trains of the emergency service water system discharge into one of the two equally redundant and independent trains of the RHR service water

system downstream of the RHR heat exchanger. The discharge from the two systems normally discharges into the circulating water system (see Figure 9.2-5).

The emergency service water from the diesel-generators is discharged directly to the river through one of the storm sewers.

9.2.3.3 Safety Evaluation

During the design-basis accident (LOCA) or a loss of offsite ac power, indication of either (or both) cause the emergency diesel generators to start, thus cooling water must be supplied by at least one of the two redundant trains of the emergency service water system. Consequently, each ESW pump gets its start signal from its respective emergency diesel generator's control circuitry. At least one of the two redundant trains of the RHR service water system is required to provide cooling water to the RHR heat exchangers under postaccident conditions. Two RHR service water pumps and one emergency service water pump are required to supply the flow capacity for one train in their respective systems.

One emergency service water pump and two RHR service water pumps are connected to each of the two independent and redundant diesel-generator buses.

As noted above, during the design-basis accident (LOCA) or loss of offsite power, the emergency service water pump on each of the two trains will start immediately (LOCA with offsite power available) and after the breakers on each diesel-generator bus are closed (LOCA with loss of offsite power).

The two RHR service water pumps on each bus will be manually connected approximately 10 min after the commencement of diesel-generator loading. This will require the tripping of one RHR pump as described in Section 8.3.

As described in Section 9.2.3.2.1, the design of the RHR service water system is such that a 20 psi positive differential pressure is maintained with respect to the RHR heat exchanger discharge. Figure 9.2-5 shows that hand switches are used to throttle the RHR heat exchanger outlet valves on the RHR service water side to maintain differential pressure control in each loop. For the postulated condition of LOCA concurrent with loss of offsite power, the RHR service water pumps cannot be started until loads are shed from the emergency power system. During this period, suppression pool water is being discharged into the reactor vessel through the shell side of the RHR heat exchanger with no RHRSW flow or pressure.

In the event of a heat exchanger tube leak, a radioactive release would be detected as the RHR service water discharge is monitored by a process radiation monitor which will alarm in the control room on high radiation. The operators would then take action to terminate the release.

The RHR and emergency service water systems obtain their water from the pump house, which is supplied with water from the river by the river water supply system.

These water supply systems are completely redundant and therefore meet the single-failure criterion. The delineation of Seismic Category I/Nonseismic piping interfaces is shown in Figures 9.2-2 and 9.2-5 as denoted by the symbol . The piping for the river water system runs from the river intake structure to the pump house, and that for the RHRSW and emergency service water systems runs from there to the reactor building. The intake structure and the reactor building are Seismic Category I structures and the Seismic Category I portion of the pump house is shown in Figure 9.2-7. The piping runs for this piping are shown in Figures 9.2-8 through 9.2-11.

Additionally, the service water flow in each of the two redundant discharge headers downstream of the heat exchangers is measured and transmitted for flow indication in the control room.

9.2.3.4 Testing and Inspection Requirements

As part of the plant normal preventive maintenance activities, the RHR service water and emergency service water systems will be periodically inspected during service. Pumps and equipment can be maintained, put into service, and tested without affecting the system operational objectives. The frequency and scope of periodic inspection and maintenance of equipment will be in accordance with normal plant practices, manufacturer's recommendation and operating history.

The tests and inspections of the river water, RHR service water and emergency service water systems as listed in the Technical Specifications will not affect the availability of the redundant trains of these systems, except for the short periods required for testing the motor- or air-operated valves in either the supply or discharge header of each system. The required tests will be scheduled on a one-train-at-a-time basis, ensuring that one train of either system will always be available.

In case of a loss of offsite power during testing, the fail-safe features ensure the availability of both redundant trains of all three systems.

9.2.3.5 Instrumentation Requirements

9.2.3.5.1 RHR Service Water System

[redacted]

The differential pressure required between the tube-side discharge and the shell-side inlet of the RHR heat exchangers is also indicated in the control room.

Pressure-differential switches, located adjacent to each of the two RHR heat exchangers, initiate an alarm in the control room if the differential pressure between the primary fluid (shell) side and the service water (tube) side of either of the two heat exchangers drops below 20 psi.

In addition, the service water flow in each of the two redundant discharge headers downstream of the heat exchangers is measured and transmitted for flow indication in the control room.

[redacted]

9.2.3.5.2 Emergency Service Water System

There are four valves and one flow element in each loop of the ESW system provided to balance the flow to each of the nine cooling units in each loop. These facilitate balancing the system with the different cooling requirements for each unit, by getting a dP indication for the control building chiller and RHR and core spray pump room unit, which are major users of the emergency service water system.

A pressure switch, located on each of the two pump discharge headers, initiates an alarm in the control room if the header pressure, because of system leakage, drops below a preset minimum required pressure.

2012-017 [redacted]

9.2.4 GENERAL SERVICE WATER SYSTEM

9.2.4.1 Design Bases

9.2.4.1.1 Power Generation Objective

The power generation objective of the general service water system is to provide cooling water for equipment throughout the plant.

9.2.4.1.2 Power Generation Design Bases

1. The general service water system is designed to meet plant requirements for startup, normal operation, and shutdown.

2. A spare pump is provided for reliability.

3. Chemical treatment of the cooling water is provided to minimize fouling of equipment.

9.2.4.2 Description

The general service water system provides water to meet cooling requirements of the reactor building cooling water system and equipment in the turbine building. The cooling water used is strained, chlorinated river water which is supplied from the circulating water system and is returned to this system for recycling after being cooled by passage through the cooling towers. See Figure 9.2-12.

[redacted] Two pumps are normally operating with the third on automatic standby. All three pumps are rated 5100 gpm at 173.5ft-head. The pumps discharge into a common header and the combined flow passes through a self-cleaning strainer that is provided with a manually operated bypass. The pumps obtain their water from the same wet pit as the circulating water pumps. [redacted] After distribution to the equipment, the flows are recombined and discharged into the circulating water system flow to the plant's cooling towers and passed through the towers back to the wet-pit sump.

The following equipment is served by this system:

1. Isolated-phase bus duct cooler.

2. Generator hydrogen coolers.

3. Stator winding liquid coolers.

4. Condensate pump motor coolers.

5. Exciter air cooler.

6. Turbine lube oil coolers.

7. Oil and motor coolers for reactor feed pumps.

8. Electro-hydraulic control system coolers.

9. Recirculation pump motor-generator coolers.

10. Reactor building closed cooling system heat exchangers.

11. Chlorination system.

12. Circulating water pump motor coolers.

13. Auxiliary heating system.

14. Instrument and service air compressor (standby only, normal supply is Well Water with GSW as backup).

15. Steam tunnel cooling units.

Local temperature and/or pressure indicators are provided at the points of application.

Two of the three General Service Water pumps may be manually connected to an essential bus on loss of offsite power when and if emergency loading on the diesels permit. Each of the two pumps are on separate essential buses.

2013-010 A sampling point is provided from the general service water discharge header to obtain a representative sample. This provides a method for sampling the circulating water for chemical analysis (chlorine).

9.2.4.3 Safety Evaluation

The general service water system is not safety related.

9.2.4.4 Testing and Inspection Requirements

As part of the plant normal preventive maintenance activities, the system is periodically inspected during service. Pumps and equipment can be maintained, put into service, and tested without affecting the system operational objectives. The frequency and scope of periodic maintenance and inspection of equipment is in accordance with normal plant practices, manufacturer's recommendation and operating history.

9.2.5 REACTOR BUILDING COOLING WATER SYSTEM

9.2.5.1 Design Bases

9.2.5.1.1 Power Generation Objective

The power generation objective of the reactor building cooling water system is to provide for the cooling of equipment in the reactor building, which may contain or have the potential to contain radioactive fluids.

9.2.5.1.2 Power Generation Design Bases

1. The design is based on using general service water for heat removal from this closed-loop system (Section 9.2.4).

2. The reactor building cooling water system is designed to meet flow requirements for startup, normal operation, and shutdown.

3. A spare pump and heat exchanger are provided to ensure design capacity in case of failure of the equipment in service.

4. The possibility of radioactivity being released from the plant is minimized.

5. System corrosion and fouling of heat exchangers are minimized by the use of inhibited demineralized water.

9.2.5.2 Description

The reactor building cooling water system is a closed cooling water system using inhibited demineralized water as the heat transfer medium to cool reactor auxiliaries. The system is designed to prevent reactor water contamination and is monitored to detect radioactive leakage into the system. Heat rejection is to the general service water system (Section 9.2.4).

The reactor building cooling water system consists of a forced-circulation closed loop, which contains three heat exchangers and three pumps (see Figure 9.2-13). The system provides coolant for the following equipment:

1. Drywell equipment drain sump cooler.

2. Reactor water cleanup nonregenerative heat exchangers (two).

3. Reactor building sample cooler.

4. Turbine building sample cooler.

5. Radwaste building sample cooler (two).

6. Fuel pool heat exchangers (two).

7. CRD pump coolers (two).

8. Reactor cleanup recirculating pump seal coolers (two).

9. Reactor recirculation pump heat exchangers (two).

10. Reactor building equipment drain sump heat exchanger.

11. Postaccident sampling system sample cooler.

Normally, two pumps and two heat exchangers are in service. For reactor cooldown and loss of offsite power, only one heat exchanger and one pump are required. However, following startup from a refueling outage, it may be necessary to place all 3 loops into service to maintain the spent fuel pool temperature within the design limits of the Fuel Pool Cooling and cleanup system. The three pumps are connected to the essential buses; two pumps are on one bus and one pump is on the other. The pumps are automatically disconnected from the essential buses by a signal initiated by a LOOP-LOCA, but may be manually reconnected when power is available.

Inhibited demineralized water is circulated through the closed loop at a pressure lower than the reactor coolant. Therefore, any leakage will be into the reactor building cooling water system from the listed items 2, 6, 7, 8, 9, and 10 above. The return header is monitored continuously to detect any radioactive leakage.

An expansion tank is provided to accommodate system volume expansion and contraction. [redacted] Provision for manual filling is provided and a low-level condition is alarmed.

9.2.5.3 Safety Evaluation

The reactor building cooling water system is not safety related.

9.2.5.4 Testing and Inspection Requirements

The reactor building cooling water system is periodically inspected during service. The spare pump and heat exchanger can be maintained and put into service and tested without affecting the system operational objectives. The frequency and scope of periodic maintenance and inspection of the pumps, pump motors, and heat exchangers is carried out in accordance with normal plant practices, manufacturer's recommendations and operating history.

9.2.5.5 Instrumentation Requirements

Local temperature indicators are provided in the outlet connections of all equipment heat exchangers and in the cooling water supply header.

The closed cooling water pumps have suction and discharge local pressure gauges. The pressure of the cooling water is indicated in the control room.

9.2.6 CONDENSATE STORAGE AND TRANSFER SYSTEM

9.2.6.1 Design Bases

9.2.6.1.1 Power Generation Objective

The power generation objective of the condensate storage and transfer system is to store the condensate required for the operation and servicing of the nuclear power plant and to transfer this condensate for the various services.

9.2.6.1.2 Power Generation Design Bases

1. [redacted]

2. The condensate storage tanks are designed to provide sufficient capacity for refueling, normal service, and emergency demand. These requirements are as follows:

   a. (1) Volume required to fill the reactor vessel from normal level to the vessel flange.

      (2) Volume required to fill the basin cavity.

      (3) Volume required to fill the slot.

      (4) Volume required to fill the transition from the basin cavity to the slot.

      (5) Volume required to fill the dryer separator pool.

      [redacted]

   b. Normal service requirements

Normally, the water required for refueling is in the condensate storage tanks and may be considered available for satisfying the requirements for regular plant operation. These requirements are for filters, filter-demineralizers, waste centrifuges, radwaste and nuclear system flushing, radwaste tanks, and pump seals.

However, during the refueling process, an additional allowance must be made. For each tank this comprises

(1) A volume equal to the volume of one waste sample tank (10,000 gal).

(2) A volume equal to the largest of any normal service demand (i.e., 3000 gal for condensate backwash).

(3) Freeboard of 30% of allowances (1) and (2).

[redacted]

c. Emergency demand

[redacted]

3. The condensate storage tanks are also designed to meet the requirements of the CRD hydraulic system, which rejects condensate to the storage tanks during normal operation at a rate of 9,000 lb/hr and the testing of the core spray system during shutdown at a rated flow of 3120 gpm. (During normal operation, with one CRD pump, the minimum flow line returns about 20 gpm to the condensate storage tank.)

4. The condensate storage tanks may be used during refueling outages to store cleaned water from the torus water cleanup system (see Section 9.5.10).

9.2.6.2 Description

[redacted]

The condensate transfer system has two 100% capacity pumps of 600 gpm each and one 125 gpm jockey pump designed to accommodate the requirements for condensate throughout the plant. The condensate storage tank requirements are physically isolated from the emergency volume by suction lines raised to an elevation above the approximate [redacted]

The quality of water in the condensate storage tanks is a composite of the quality of the water entering the tanks as makeup water from the demineralized water storage tank, water

rejected from the main condenser and processed through the condensate demineralizer, processed water from the radwaste disposal system, and water discharged from the CRD hydraulic system.

Water used for refueling does not re-enter the condensate storage tanks until it has been processed through the fuel pool cooling and cleanup system.

The condensate storage tanks overflow to the reactor building equipment drain sump by way of a 1000-gal overflow tank. In an emergency, this tank will overflow to the area around the tanks. The tanks are enclosed by a dike with a concrete pad preventing the entry of condensate into the ground.

The diked area has sufficient capacity to contain the volume of water stored in one condensate storage tank and has a sump to collect rainwater and permit sampling to determine disposal. The disposal, through normally locked closed valves under administrative control, will be to the discharge canal or to the radwaste disposal system depending on the concentration of radioactivity.

9.2.6.3 Safety Evaluation

The condensate storage and transfer system is not safety related.

9.2.6.4 Testing and Inspection Requirements

There are internal standpipes and related air lines to the level switches within the condensate storage tanks to allow air pressurization for the lowering of the water level within the standpipe.

These provide a means for surveillance testing the level switches without lowering the condensate storage tank level.

Procedural control ensures that the vent caps used during surveillance testing are removed after test completion.

9.2.6.5 Instrumentation Requirements

The vent piping is susceptible to freezing. If the vent lines froze solid, the water level in the standpipe could differ from the level in the tank. Since [redacted] provide inputs to the [redacted] it is essential that the level elements sense the actual tank level.

Heat tracing is provided to prevent the closure of the vent lines through freezing.

A grab sample point is provided on the condensate service jockey pump suction line to provide a means for grab sampling water in the condensate storage tanks.

REFERENCES FOR SECTION 9.2

1. Letter from Larry D. Root, Iowa Electric, to James G. Keppler, NRC, Subject: IE Bulletin No. 80-24, Prevention of Damage Due to Water Leakage Inside Containment, dated January 5, 1981, Serial No. LDR-81-18.

2. Letter from Larry D. Root, Iowa Electric, to James G. Keppler, NRC, Subject: IE Bulletin 81-03, Flow Blockage of Cooling Water to Safety System Components by Corbicula Sp. and Mytilus Sp., dated May 18, 1981, Serial No. LDR 81-182.

3. Letter from Larry D. Root, Iowa Electric, to Edward L. Jordan, NRC, Subject: Response to NRC Request for Additional Information Regarding IE Bulletin 81-03, Asiatic Clams, dated March 28, 1983.

4. General Electric Company, Analysis of Reduced RHR Service Water Flow at the Duane Arnold Energy Center, NEDE-30051, January 1983.

5. General Electric Company, Duane Arnold Energy Center Suppression Pool Temperature Response, NEDC-22082-P, March 1982.

6. License Amendment No. 272, dated December 3, 2008.

2015-005 7. Site-Visit Report Sediment Accumulation Concern at River-Water Intake for DAEC by Robert Ettema and Jacob Odgaard of University of Iowa. IIHR Report # 308. ECP-1735 Index Item # 4.0.1.

Table 9.2-1
EMERGENCY SERVICE WATER FLOW REQUIREMENTS[a]

Equipment	River Water Temperature				Flow for 3 ft/sec tube velocity
	95°F	90°F	85°F	80°F	
Diesel-generator	528	466	466	372	625
2011-019 RHR and Core Spray room cooler	83	50	50	50	75*
RCIC room cooler	17	12	12	8	20
2008-022 HPCI room cooler	16	16	16	16	35
Control Building Chiller	199	132	100	82	250
RHR pump seal coolers (two)[b]	11 (5.5 ea)	8.8 (4.4 ea)	8 (4.0 ea)	6.6 (3.3 ea)	N/A
Core spray pump motor cooler[c]	3	3	3	3	1.2[d]
Heating and ventilation instrument air compressor	1	1	1	1	1
RHR service water pump motor coolers[e]	10	10	10	10	2.8[d]
2011-019 2008-022 2014-004 Total flow	868	698.8	666	548.6	1010.0

* The required flow is greater than the 3 ft/sec flow for river temperatures over 85°F.

[a] Flow rates are given in gallons per minute for various river water temperatures.

[b] Flow rate required to maintain RHR pump seal inlet temperature ≤ 150˚F under DBA-LOCA Conditions (CAL-M10-010).

[c] Core Spray Pump – Two pumps arrangement with only one pump operating at a time. Each pump motor cooling coil requires a minimum of 3 gpm (ref. APED-E21-2723-007). Flow for 3 ft/sec is also for two pumps.

[d] The required flow is greater than 3 ft/sec for all river temperatures.

[e] RHRSW is a four pump arrangement with only two pumps operating at a time. Each motor cooling coil require a minimum flow of 5 gpm (ref. E025-019).

9.3 PROCESS AUXILIARIES

9.3.1 COMPRESSED AIR SYSTEMS

9.3.1.1 Design Bases

9.3.1.1.1 Power Generation Objective

The power generation objective of the compressed air systems is to provide a continuous supply of dry oil-free compressed air to plant instruments and general plant services as required.

9.3.1.1.2 Power Generation Design Basis

The compressed air systems are designed to supply air at the required pressure and temperature. Instrument air is typically dried to a dewpoint of -40°F or lower at 100 psig.

9.3.1.2 Instrument and Service Air System

9.3.1.2.1 Description

[redacted] Each compressor discharges through an integral aftercooler into a common discharge header and then into either or both of two air receivers. Instrument air then passes through an air dryer and a filter before entering the instrument air header that feeds the instrument components. There is a standby air dryer and filter which can be used for maintenance purposes. Service air is supplied directly from the receivers to the service air components. Instrument air is dried to a dewpoint of -40°F at 100 psig. Refer to Figure 9.3-1. The service air system is automatically isolated from the instrument air system on low header pressure. Refer to Section 9.3.1.2.4.

The systems are so arranged that any one compressor can supply both instrument and service air requirements. The compressors are sized so that any one compressor has adequate capacity to supply the normal plant instrument and service air requirements, and any two are capable of satisfying peak air demand. The third compressor is provided as a spare. A fourth oil-free compressor located in the turbine building basement is used as a standby compressor.

Provisions within the air systems are made to mitigate the effects of system piping breaks. Should loss of air system header pressure occur, successive header isolations will result. Refer to Section 9.3.1.2.4. Also air accumulators or high pressure storage bottles have been provided locally for critical components of the Condensate and Feedwater system. This backup air system will allow the feedwater system to control reactor water level for a brief period after a loss of instrument air.

All gaseous pneumatic systems in the primary containment use nitrogen as the working fluid in order to avoid oxygen (air) leakage into containment. Refer to Figure 6.2-44.. A dedicated nitrogen compressor located outside containment takes a suction from the inerted containment atmosphere and charges a nitrogen receiver located outside containment. The nitrogen receiver is then used to operate all pneumatic-operated valves within containment including, among others, the torus-drywell vacuum breakers, reactor pressure vessel head vent isolation valves, main steam safety/relief valves (SRVs) and main steam isolation valves (MSIVs). Safety-related accumulators are provided for the SRVs and MSIVs. Refer to Sections 6.3.2.2, 7.3.1.1.1.7 and Figure 5.1-1 Sheet 1.

A safety-related air system is provided as a backup to the normal instrument air system for several critical safety-related components and systems. The safety-related air system is comprised of two separate Seismic Class I trains, each of which consists of a compressor, air receiver, and associated distribution piping, valves and service connections. The distribution piping of the two safety-related air trains are not normally cross tied together and the compressor motors are supplied from different essential busses such that no single failure will render both trains inoperable. The safety-related compressors are normally cooled with well water, but can be cooled with Emergency Service Water to ensure adequate post-accident cooling. Below is a list of components and systems whose operability is supported by the safety-related air system:

Component/System	Supported Function
Control Building Chiller System	ventilation flow path and temperature control
Standby Filter Unit System	flow control and the CBC System provides the ventilation flow path
Standby Gas Treatment System	flow control and filter cooler bypass damper opening
Drywell Cooling Water Containment Isolation Valves	valve closure
Containment Purge and Vent Isolation Valves	leak tightness (by pressurizing the T-ring seal) when closed
Reactor Building to Torus Vacuum Breaker Butterfly Valves	valve closure and leak tightness (by pressurizing the T-ring seal) when closed

9.3.1.2.2 Safety Evaluation

Instrument and service air systems are not safety-related. Any one of the four non-safety air compressors can satisfy the normal plant instrument and service air requirements; thus the four air compressors ensure a backup capability even in the unlikely event of a failure of three compressors.

The service air systems does not supply any safety-related equipment and total failure of the service air system, therefore, need not be considered in the Safety Evaluation.

Although the normal instrument air system supplies some safety-related equipment, total failure of the system will not adversely affect the operation of the plant for the following reasons:

1. Testing was performed on the instrument air system as part of DAEC's response to NRC Generic Letter 88-14. See References 9 and 10. Results of the testing indicate, "The testing program developed to verify that air-operated safety-related components will perform their safety-related function upon loss of their normal air supply was successfully completed. All safety-related components performed as designed."

2. The existence of the safety-related air system which can supply air to support the operation of safety-related equipment if the instrument air system becomes unavailable. Transition and separation from the instrument air system to the safety-related air system occurs automatically upon low safety-related air receiver pressure.

3. The pneumatically actuated components inside the drywell have a reliable source of compressed nitrogen to provide the required motive force, and therefore, do not rely upon either the safety-related air system nor the normal instrument air system.

Therefore, total failure of the normal instrument air system will not adversely impact the safe operation of the plant due to: the redundancies built into the design of the safety-related air-system, the redundancies built into the systems whose components are supplied safety-related air and the design of the components that are supplied air exclusively from the normal instrument air system that revert to their conservative position on loss of air.

9.3.1.2.3 Testing and Inspection Requirements

The instrument and service air systems operate continuously and are observed and maintained during normal operations. An instrument air system blow down is performed periodically to remove any possible particulates from the system. Also, an instrument air quality test is performed periodically at various instrument air headers downstream of the air dryers. This test is performed to verify that the air quality (dew point, particulate and

oil content) is consistent with manufacturer recommendations. Revisions to test frequencies are based on test results.

Routine testing is performed on the safety-related air compressors to ensure that they automatically start on low pressure in the associated safety-related air receiver. Additionally, routine testing is performed on the safety-related air compressors to monitor the performance of the compressors and to monitor the integrity of the safety-related air distribution piping and tubing. The routine test also verifies that the check valves which isolate the safety-related air system from the normal instrument air system are working properly.

9.3.1.2.4 Instrumentation Requirements

The three air compressors that are normally in operation are controlled by a group of three pressure switches, set for "load-no load" control over three overlapping pressure ranges. The lead compressor (as determined by the baseload selector) operates in the highest range. If the system demand exceeds the capacity of one air compressor, a second compressor will load and unload as required until a timed period of unloaded operation is exceeded. The second compressor will then return to standby (motor off) until the next call to start. The compressor starting sequence is controlled by the baseload selector and can be altered manually.

System isolation for the instrument and service air systems is accomplished by pressure switches that provide the signals to close control valves. Interlocks are provided to automatically isolate the nonessential air lines from the air receivers on the detection of low pressure in the air receivers. This provides a period for orderly shutdown or repairs. On a decrease in air header pressure to a specific point or on loss of control power, the service air system will isolate. A further decrease in air header pressure to a lower setpoint will isolate the nonessential Balance-of-Plant and Turbine Building instrument air headers. The Reactor Building instrument air header does not have an automatic isolation.

The safety-related compressors that are normally in standby are controlled by pressure switches monitoring the safety-related air receivers.

9.3.1.3 Breathing Air System

9.3.1.3.1 Description

The breathing air system is crosstied to the instrument air system and supplies grade 'D' air at a dew point of -40°F or lower at a pressure of 100 psig. The system also consists of six-man stations located throughout the power block. Each station is designed to deliver 10 cfm to each breathing apparatus hookup, in accordance with NUREG-0041.

When necessary, breathing air for personnel use can be obtained from the instrument air mains or service air mains, both of which are oil free. When breathing air is supplied from the service air system, portable air filters are used to ensure the breathing air system air quality meets grade D air requirements as described in the Compressed Gas Association Air Specification G-7-1. The instrument air system meets the grade D requirements and, therefore, the use of portable filters is not necessary.

The breathing air system inside the drywell has been "abandoned in place" since it is supplied by service air and does not meet air quality requirements. This breathing air connection to the drywell has a removable spool piece inside the drywell, a blank flange which is installed on the air supply line in the drywell and an isolation valve outside the containment. Procedures require the isolation valve to be verified closed and the blank flange to be verified in place prior to plant startup.

9.3.1.3.2 Safety Evaluation

The breathing air system is not itself a safety-related system. Any interaction of the breathing air system with safety-related systems is kept to a minimum, specifically:

1. Pipe was routed such that a line break accident is inconsequential to plant safety.

2. Secondary containment was penetrated three times. The maximum possible inleakage from the atmosphere to secondary containment was analyzed and was found to be acceptable. There was no penetration of primary containment.

9.3.1.3.3 Testing and Inspection Requirements

The breathing air system operates as required and is observed and maintained during normal operations. No special inspection or testing is required.

9.3.2 PROCESS SAMPLING SYSTEM

9.3.2.1 Design Bases

9.3.2.1.1 Power Generation Objectives

The power generation objectives of the process sampling systems are the following:

1. Monitor the operation of plant equipment.

2. Provide information for making operational decisions with regard to effectiveness and proper performance.

9.3.2.1.2 Power Generation Design Bases

The process sampling systems are designed to perform the following:

1. Obtain representative samples in forms that can be used in radiochemical laboratory analysis for the indication of changes in the constituents.

2. Minimize the contamination and radiation effects at the sampling stations.

3. Reduce decay and sample line plateout as much as possible.

9.3.2.2 System Description

Samples are taken from various streams and locations as indicated in Table 9.3-1. Sample points are grouped at normally accessible locations, with drains to the contaminated waste system provided at these locations to limit the risk of contamination. Lines are sized to ensure prompt purging and sufficient velocities to obtain representative samples when radioactivity measurements are made. Bottled grab samples are taken to the laboratory for the appropriate analysis.

A postaccident sampling system has been installed to take postaccident reactor coolant and containment atmosphere samples. See Section 12.3.4 for a discussion of postaccident sampling.

The reactor recirculation system process sample line does have postaccident sample capabilities that could be used as a backup to the postaccident sampling system. The isolation valves on the sample line have been provided with key-locked bypass switches to override a containment isolation signal to enable sampling with the containment isolated.

9.3.2.3 Safety Evaluation

The process sampling systems are not safety related.

9.3.2.4 Testing and Inspection Requirements

The process sampling systems are operational systems and as such require no periodic testing to ensure operability.

9.3.2.5 Instrumentation Requirements

The process sampling systems are provided with continuous automatic, monitoring and alarm of undesirable conditions (except for the low-level radwaste processing and storage facility sample tank).

9.3.3 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

9.3.3.1 Design Bases

9.3.3.1.1 Power Generation Objective

The power generation objectives of the equipment and floor drainage systems are to collect and remove all waste liquids from their points of origin and to route them to a suitable disposal area in a controlled and safe manner. Water from radioactive drains is collected for sampling and analysis before processing and disposal as described in Section 11.2. Drain line penetrations through containment barriers are designed to maintain containment during normal operations and design-basis accidents.

9.3.3.1.2 Power Generation Design Basis

Plant equipment and floor drainage systems will operate satisfactorily during normal plant operations and will retain their integrity following postulated accidents. Nonradioactive drain systems are arranged to ensure that no infiltration of radioactive wastes will occur.

9.3.3.2 Description

Plant equipment and floor drainage systems handle both radioactive and nonradioactive drains. Radioactive drains may contain potentially radioactive materials. In general, radioactive drains are drained by gravity to a sump, the contents of which are pumped to the radwaste system for the determination of radioactivity before cleanup, reuse, or discharge. Nonradioactive drains are drained by gravity to a sump, the contents of which are pumped to the storm drain system that is entered at a point outside the building, or to the oil interceptor tank that serves the transformers.

Specific information regarding the number of drains, locations, capacities, and types is provided in Figures 9.3-2 through 9.3-26.

9.3.3.2.1 Radioactive Equipment and Floor Drainage Systems

Reactor Building Drains

Reactor building radioactive equipment and floor drains are collected in two separate systems. One handles drainage from all equipment and floor drains located in the primary containment, and the other handles drainage from equipment and floor drains located in the secondary containment. The primary containment equipment and floor

drain system begins with funnel drains at all items of equipment that require draining and floor drains located to facilitate the rapid and efficient removal of liquid waste from the surface of the floor. Drainage collects in branch lines, and drains by gravity to the drywell equipment and floor drain sumps. Sump pumps transfer wastes from the sumps to the radwaste system.

The secondary containment equipment and floor drainage system begins with funnel drains at all items of equipment and floor drains located to facilitate the rapid and efficient removal of liquid waste from the surface of the floor, collects in branch lines, and drains by gravity to the reactor building equipment and floor drain sumps. Sump pumps transfer wastes from the sumps to the radwaste system.

Any leakage from the spent-fuel pool is channeled into one or more of the eleven 1-in.-diameter liner drain pipes that are provided to monitor leaks, as shown in Figure 9.3-21. Each pipe contains a manual gate valve that is normally closed. An operator will periodically open each of these valves to check for leaks in the fuel pool liner. These drains are routed to the reactor building floor drain sump via a common trough.

Turbine Building Drains

The turbine building radioactive equipment and floor drainage system begins with funnel drains at all items of equipment that require draining and floor drains located to facilitate the rapid and efficient removal of liquid waste from the surface of the floor. Drainage collects in branch lines, and drains by gravity into the turbine building equipment and floor drain sumps. Sump pumps transfer wastes from the sumps to the radwaste system.

Radwaste Building Drains

The radwaste building radioactive equipment and floor drainage system begins with funnel drains at all items of equipment and floor drains located to facilitate the rapid and efficient removal of liquid waste from the surface of the floor. Drainage collects in branch lines, and drains by gravity to the radwaste building equipment and floor drain sumps. Sump pumps transfer wastes from the sumps to the radwaste system.

A conveyor floor drain sump receives drainage from the drum filling and storage area in the radwaste building, and sump pumps transfer wastes from this sump to the radwaste system.

Low-Level Radwaste Processing and Storage Facility (LLRPSF) Drains

The LLRPSF radioactive equipment and floor drains are collected in two separate systems. One system handles drainage from all equipment and floor drains located in the storage section of the LLRPSF, while the other system handles drainage from all equipment and floor drains located in the processing section of the LLRPSF.

The storage area equipment and floor drainage system begins with funnel drains at all items of equipment that require draining. The floor drains are located to facilitate the rapid and efficient removal of liquid waste from the surface of the floor. Drainage collects in branch lines and drains by gravity to the storage area sump. Sump pumps transfer wastes from the sump to the radwaste system in the radwaste building.

The processing area equipment and floor drainage system begins with funnel drains at all items of equipment that require drainage. The floor drains are located to facilitate the rapid and efficient removal of liquid waste from the surface of the floor. Drainage collects in branch lines and drains by gravity to either the processing area sump or the hydrolazing/decontamination sump. Sump pumps transfer wastes from the sumps to a sample tank for temporary holdup. The sample tank pump will transfer the liquid from the sample tank to the radwaste system in the radwaste building or to the environment.

Radiochemistry Laboratory Drains

The radiochemistry laboratory equipment and floor drainage system is divided into two categories that drain separately, by gravity, to two separate sumps. Each sump is provided with a duplex pump system that discharges the liquid waste from the sump to a header that drains by gravity to the radwaste system. One sump discharge, containing corrosive waste, is conveyed to the chemical waste tank; the other, containing detergents, is conveyed to the detergent drain tank.

9.3.3.2.2 Nonradioactive Water Drainage Systems

Drainage from building roofs is collected in branch lines, emptied to headers or main drain lines, and discharged to the storm drain system.

A separate drainage system is provided in the turbine building for nonradioactive equipment drains and floor drains. These are collected in branch lines that empty into a main drain line and drain, by gravity, to the nonradioactive wastewater sump. This waste is pumped into the storm drain system, the connection to which is outside the building.

There is a normally closed, manual isolation valve and alternate flow path for the diesel-generator room floor drains to the turbine building normal waste sump.

The manual isolation valves will prevent water backup into the diesel-generator rooms in case of leaking check valves during site flood conditions.

The turbine building pumps are each designed to deliver a flow rate of 50 gpm at 40-ft total discharge head, thus enabling them to remove water from the sump against the head created as a result of the maximum probable flood.

9.3.3.3 Safety Evaluation

Within the drainage system itself, the potential for inadvertent transfer of fluids from a contaminated volume to an uncontaminated volume does not exist as all areas of potential radioactive contamination are drained to the radioactive waste system sumps.

9.3.3.4 Tests and Inspection Requirements

Before being placed into service, the nonradioactive wastewater drainage system proved leaktight when subjected to a hydrostatic test pressure equal to not less than a 10-ft head of water.

9.3.3.5 Instrumentation Requirements

The drainage systems themselves do not require instrumentation for monitoring level, flow, temperature or radiation. All potentially radioactive drains are piped to the radwaste sump system which is shown in Figure 11.2-2. This figure shows the instrumentation that is used to monitor drainage into these sumps. LLRPSF drains are piped to sumps in the facility.

9.3.4 STANDBY LIQUID CONTROL SYSTEM

9.3.4.1 Design Bases

9.3.4.1.1 Safety Objective

The safety objective of the standby liquid control (SLC) system is to provide a backup method, independent of the control rods, to initiate and maintain the reactor subcritical as the nuclear system cools. Maintaining subcriticality thus ensures that the fuel barrier is not threatened by overheating in the improbable event that not enough of the control rods can be inserted to counteract the positive reactivity effects of a colder moderator. The SLC system was modified in 1987 to meet the requirements of the final NRC rule on Anticipated Transients Without Scram (ATWS), given in 10CFR50.62 and NRC Generic Letter 85-03.

9.3.4.1.2 Safety Design Bases

The SLC system meets the following safety design bases:

1. Backup capability for reactivity control is provided, independent of normal reactivity control provisions in the nuclear reactor, to be able to shut down the reactor if the normal control ever becomes inoperative.

2. The backup system has the capacity for controlling the reactivity difference between the steady-state rated operating condition of the reactor with voids and the cold shutdown condition, including shutdown margin, to ensure complete shutdown from the most reactive condition at any time in core life.

3. The time required for the actuation and effectiveness of the backup control is consistent with the nuclear reactivity rate of change predicted between rated operating and cold shutdown conditions. A fast scram of the reactor or operational control of fast reactivity transients is not specified to be accomplished by this system.

4. Means are provided by which the functional performance capability of the backup control system components can be verified periodically under conditions approaching actual use requirements. A substitute solution, rather than the actual neutron-absorber solution, can be injected into the reactor to test the operation of all components of the redundant control system.

5. The neutron absorber is dispersed within the reactor core in sufficient quantity to provide a reasonable margin for leakage or imperfect mixing.

6. The system is reliable to a degree consistent with its role as a special safety system; the possibility of unintentional or accidental shutdown of the reactor by this system shall be minimized.

9.3.4.2 <u>Description</u>

The SLC system is designed to inject a quantity of boron that produces a concentration of [redacted] Control capacity equivalent for DAEC to the requirements of the ATWS rule is achieved by running both SLC pumps simultaneously at their design minimum pumping rate of 26.2 gpm each, [redacted] (References 1 through 3). [redacted] described above will meet the ATWS shutdown requirements, as demonstrated in Section 15.3.1.

The SLC system (see Figure 9.3-27) is manually initiated from the main control room to pump a boron neutron-absorber solution into the reactor if the operator believes the reactor cannot be shut down or kept shut down with the control rods. However, the insertion of control rods is always expected to ensure prompt shutdown of the reactor when required.

The SLC system is needed only in the highly improbable event that not enough control rods can be inserted in the reactor core to accomplish shutdown and cooldown in the normal manner.

The boron solution tank, the test water tank, the two positive-displacement pumps, the two explosive valves, and associated local valves and controls [redacted] outside the primary containment. The liquid is piped into the reactor vessel and discharged near the bottom of the core shroud so that it mixes with the cooling water rising through the core (see Section 5.3 and Section 3.9.5).

The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the uranium fuel.

[redacted] A sparger, using compressed air, is provided in the tank for mixing. The storage tank piping to the suction of the SLC pumps is located on the side of the tank, rather than the bottom, to minimize the potential of line blockage from either foreign material or precipitation of the sodium pentaborate.

[redacted] The low temperature alarm is required to be maintained at least 5 degrees above the saturation temperature.

The equipment containing the solution is installed in a room in which the air temperature is to be maintained within the range of 68 to 90°F. In addition, a heater system maintains the solution temperature at 75 to 85°F to prevent precipitation of the sodium pentaborate from the solution during storage.

Each positive-displacement pump is sized to inject the solution into the reactor at a minimum rate of 26.2 gpm. [redacted] one pump is capable of injecting sufficient boron into the reactor to assure complete shutdown. The pump and system design pressure between the explosive valves and the pump discharge is 1400 psig. The two relief valves are set at 1350 to 1400 psig. To prevent bypass flow from one pump in case of relief valve failure in the line from the other pump, a check valve is installed downstream of each relief valve line in the pump discharge pipe.

The two explosive-actuated injection valves provide assurance of opening when needed and ensure that boron will not leak into the reactor even when the pumps are being tested.

Each explosive valve is closed by a plug in the inlet chamber. The plug is circumscribed with a deep groove so the end will readily shear off when pushed with the valve plunger. This opens the inlet hole through the plug. The sheared end is pushed out of the way in the chamber; it is shaped so that it will 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. 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 primary circuit opened. To service a valve after firing, a 6-in. length of pipe (spool piece) must be removed immediately upstream of the valve to gain access to the shear plug.

The SLC system is manually actuated by a two-position keylock switch which activates both injection pumps, opens both of the explosive valves, and isolates the reactor water cleanup system. A green light in the control room indicates that power is available to each pump motor contactor and that the contactor is open (pump not running). A pump will start even if the local switch at the pump is in the STOP position for test or maintenance. Pump discharge pressure and flow are indicated in the control room.

Although both pumps are started together, cross piping and check valves ensure that the boron solution will be injected even if only one pump runs and/or one explosive valve opens.

Equipment drains and tank overflow are not piped to the waste system but to separate containers (such as 55-gal drums) that can be removed and disposed of independently to prevent any trace of boron from inadvertently reaching the reactor.

A high-point vent is located between the SLC explosive valves [redacted] and check valve [redacted].

This vent line is required to allow for local leakage rate testing of valve [redacted] in accordance with the Primary Containment Leakage Rate Testing Program.

9.3.4.3 Safety Evaluation

The SLC system is a special safety system not required for unit operation or to meet the single-failure criterion. The system is expected never to be needed for unit safety because of the large number of independent control rods available to shut down the reactor.

However, system reliability and the availability are enhanced by providing two sets of the components required to actuate the system. Pumps and explosive valves, are provided in parallel in the design. Redundancy is not required for the tank heater or heating cable.

The system is designed to bring the reactor from rated power to a cold shutdown at any time in core life. The reactivity compensation provided will reduce reactor power from rated to zero level and allow the cooling of the nuclear system to normal room temperature, with the control rods remaining withdrawn in the rated power pattern. It includes the reactivity gains that result from complete decay of the rated power xenon inventory. It also includes the positive reactivity effects from eliminating steam voids, changing water density from hot to cold, reduced Doppler effect in uranium, reducing neutron leakage from boiling to cold, and decreasing control rod worth as the moderator cools. The specified minimum final concentration of boron in the reactor core provides the capability of bringing the reactor, at any time in the cycle, to a subcritical condition.

[redacted] to provide the specified shutdown margin, after operation of the SLC system, is 660 ppm (parts per million) (see Figure 9.3-30). The calculation of the minimum quantity of sodium pentaborate to be injected into the reactor is based on [redacted] in the reactor coolant and the quantity of reactor coolant in the reactor vessel, recirculation loops, and the RHR system in the shutdown cooling mode at 70°F and reactor normal water level. The result is increased by 25% to allow for imperfect mixing, leakage, and volume in other small piping connected to the reactor. [redacted] will be achieved if the solution is prepared as defined in Section 9.3.4.2 and maintained above saturation temperature.

The cooldown of the nuclear system will require a minimum of several hours to remove the thermal energy stored in the reactor, cooling water, and associated equipment and to remove most of the radioactive decay heat. The controlled limit for the reactor vessel is 100°F/hr, and normal operating temperature is approximately 550°F. Usually, using the main condenser and various shutdown cooling systems to shut down the plant will require 10 to 24 hr before the reactor vessel is opened and much longer to reach room temperature (70°F); this is the condition of maximum reactivity and, therefore, the condition that requires the maximum concentration of boron.

The SLC system equipment essential for the injection of the neutron-absorber solution into the reactor is designed as Seismic Category I for withstanding the specified earthquake loadings (see Section 3.2). Nonprocess equipment such as the test tank is designed as Nonseismic. The system piping and equipment are designed, installed, and tested in accordance with requirements stated in Sections 3.9 and 3.10.

The SLC system is required to be operable in the event of a loss of offsite Power (LOOP); therefore, the pumps, heaters, valves, and controls are powered from the standby ac power supply or dc power in the absence of normal power. The pumps and valves are powered and controlled from separate buses and circuits so that a single failure in the power supplies will not prevent system operation.

The SLC system and pumps have sufficient pressure margin, up to the system relief valve setting of approximately 1400 psig, to ensure solution injection into the reactor above the normal pressure of approximately 1030 psig in the bottom of the reactor. The nuclear system relief and safety valves begin to relieve pressure at a nominal pressure of 1110 psig. Therefore, the SLC system positive-displacement pumps cannot overpressurize the nuclear system.

In addition, the SLC pumps need to be able to inject sufficient boron into the reactor to mitigate postulated ATWS events (Reference Section 15.3.1). A concern was raised in Information Notice 2001-13 (Reference 10) that insufficient margin was available between the required pump discharge pressure needed to ensure injection into the vessel during high pressure ATWS event and the SLCS relief valves opening settings to prevent the relief valves from opening and thereby divert sufficient boron injection from the reactor to cause unacceptable consequences. An evaluation was performed for the DAEC (Reference 11) that concluded that the potential for lifting the SLCS relief valves was minimal and that uninterrupted boron injection was assured during postulated ATWS events.

The following values show the bases for solution temperatures. The sodium pentaborate solution in the tank and pump suction line is required to be maintained 5°F above its saturation temperature.

Concentration	Solution Concentration (%)	Saturation Temperature (°F)	Minimum Required Solution Temperature (°F)
Minimum	[redacted]	53.5	58.5
Maximum		65	70

The heating system for the area of the building where the SLC system pumps and tanks B are located is designed to maintain the air temperature within the range of 68° to 90°F. The electric heaters in the tank and the heat tracing in the pump suction line act as a backup to the area heating system. The tank heaters are designed to maintain the solution within the range of 75° to 85°F and the heat tracing is designed to maintain the suction piping within a range of 75° to 95°F.

9.3.4.4 Tests and Inspection Requirements

Operational testing of the SLC system is performed in at least two parts to avoid inadvertently injecting boron into the reactor. With the valves to and from the solution tank closed and the three valves to and from the test tank opened, demineralized water in the test tank can be recirculated by locally starting either pump.

The injection portion of the system can be functionally tested by valving the injection lines to the test tank and actuating the system from the control room. Both injection valves open on actuation. System operation is indicated in the control room. After functional tests, the injection valves and explosive charges must be replaced and all the valves returned to their normal positions as indicated in Figure 9.3-27.

By closing a local locked-open valve to the reactor, leakage through the injection valves can be detected at a test connection in the line between the containment isolation check valves. Position indicator lights in the control room indicate that the local valve is closed for tests or open and ready for operation. Leakage from the reactor through the first check valve can be detected by opening the same test connection when the reactor is pressurized.

The test tank contains demineralized water for approximately 8 minutes of pump operation. Demineralized water from the makeup is available for refilling or flushing the system.

Should the boron solution ever be injected into the reactor, either intentionally or inadvertently, then after making certain that the normal reactivity controls will keep the reactor subcritical, the boron is removed from the reactor coolant system by flushing for gross dilution followed by operating the reactor cleanup system. There is practically no effect on reactor operations when the boron concentration has been reduced below approximately 50 ppm.

The concentration of the sodium pentaborate in the solution tank is determined periodically by chemical analysis.

9.3.4.5 Instrumentation Requirements

Instrumentation consisting of solution temperature indication and control, tank level, and heater system status is provided locally at the SLC tank.

A temperature indicator in the pump inlet at the pressure point permit daily recording of the temperature of the liquid control solution in the piping between the SLC tank and the pump inlet. A temperature indicator monitors line temperature to ensure that the sodium pentaborate solution does not fall below about 75°F. This could happen only if both the room air heating units and the line trace heaters failed in very cold weather.

High or low temperature, or high or low liquid level, causes an alarm in the control room. Tank level is also monitored in the control room. SLC pump discharge header pressure and flow indication are provided locally and in the control room.

9.3.5 HYDROGEN WATER CHEMISTRY SYSTEM

9.3.5.1 Design Bases

9.3.5.1.1 Power Generation Objectives

The power generation objectives of the hydrogen water chemistry system (HWCS) are to inject hydrogen into the feedwater to control intergranular stress corrosion
cracking of austenitic stainless steel piping and components, and to provide the hydrogen supply for main generation cooling.

9.3.5.1.2 Power Generation Design Bases

The HWCS is designed to meet the following design bases:

1. It will supply hydrogen for feedwater injection at a rate of 0 to 5 SCFM to each feed pump suction.

2. It will supply hydrogen for main generator purge and makeup (continuous fill) requirements.

3. It will supply the offgas system with air or oxygen to ensure a stoichiometric mixture for recombination of hydrogen and oxygen.

4. It will inject oxygen into the suction lines of running condensate pumps in order to keep the oxygen level in the condensate and feedwater systems high enough to minimize general corrosion.

5. It will automatically isolate the hydrogen and oxygen injection systems in the event of system failures.

9.3.5.2 Description

Intergranular stress corrosion cracking in the reactor coolant system is controlled by injecting hydrogen into the feedwater. The hydrogen in the feedwater enters the reactor where it removes (by combining with) dissolved oxygen, which is radiolytically produced. The composition of non-condensible gas which leaves the reactor via the condenser and the offgas is changed. Flows of both oxygen and hydrogen are decreased, and the ratio of oxygen to hydrogen is also decreased. In order to prevent combustible mixtures of hydrogen in the offgas stream, air or oxygen is injected.

2011-016 Reduction of oxygen concentration to less than approximately 30 ppb could result in an increase in the corrosion rate for carbon steel piping in the feedwater and condensate systems. Oxygen is injected into the condensate pump suction lines to preclude this problem.

Hydrogen is stored in vendor-supplied hydrogen tube trailers, and in six high pressure hydrogen storage vessels which serve as backup when the tube trailer supply is not available. A tube trailer has a capacity of approximately 125,000 cubic feet, which is sufficient for at least a ten-day injection supply. Hydrogen from the tube trailer can be used for feedwater injection, main generator fill and makeup, and recharging the hydrogen storage tanks.

2011-016 [redacted] The hydrogen supply line is 1/2-inch stainless steel pipe, coated and wrapped for underground installation. [redacted]

Hydrogen for injection is supplied to the suction of each feedwater pump through one of two redundant flow control valves. The hydrogen flow rate is programmed to increase in proportion to reactor power when in "Cascade mode". Hydrogen flow can also be adjusted manually if necessary using the "Automatic" or "Manual" controller modes of operation for the system. An isolation valve is provided to terminate the hydrogen injection if the feedwater pump is tripped or on a HWCS isolation signal.

[redacted]

9.3.5.3 Instrumentation Requirements

The following instrumentation and controls are associated with the HWCS:

- Concentrations of dissolved oxygen and hydrogen in the recirculation water are indicated and recorded.
- Residual offgas oxygen concentration is indicated and recorded.

- Local atmospheric hydrogen monitors at six locations around the plant alarm on high concentration, and initiate HWCS isolation on a high-high concentration.
- [redacted]
- Storage tank temperatures are indicated locally.
- [redacted]
- [redacted]
- Emergency shutdown buttons are provided to immediately isolate the hydrogen system. The oxygen system will be shut down after a 12 minute time delay. Air addition, if in service, will continue to operate to ensure that all excess hydrogen has had time to reach the offgas system.

The effect of hydrogen injection is monitored by instruments which measure electrochemical corrosion potential and crack growth rate. Sensors located outside the drywell are exposed to reactor coolant from a recirculation loop sample line. To verify that they were exposed to conditions which are representative of those in the primary system, additional sensors were placed in the recirculation piping and in-core via LPRM assemblies. These additional sensors are no longer in use. See Section 7.6.1.6.3.

9.3.6 ZINC INJECTION (GEZIP) SYSTEM

9.3.6.1 Description

GE Nuclear Energy has developed a system to inject zinc into the BWR primary system called GEZIP (General Electric Zinc Injection Passivation). The GEZIP process maintains trace quantities of ionic zinc in the reactor water for the purpose of reducing radiation buildup by maintaining/reducing CO-60 buildup on primary system surfaces.

The GEZIP system, SUS 63.01, consists of a zinc addition skid that is designed to inject trace amounts of Depleted Zinc Oxide (DZO) into the feedwater during normal plant operation. The system consists of a simple recirculation loop off of the feedwater system. The zinc solution is obtained by passing a stream of feedwater from the feedwater pumps' discharge header tap [redacted] by the feedwater regulating valves. This feedwater then goes through a dissolution vessel containing pelletized DZO [redacted] next to the turbine lube oil conditioner. The feedwater dissolves the pellets as it passes through the zinc vessel carrying the dissolved DZO into the feedwater pumps' suction header located on the condenser bay mezzanine. Manual valves are used

to control feedwater flow to the reactor. Instrumentation associated with the skid includes a calibrated flow meter, a differential pressure indicator, and a temperature indicator.

9.3.7 NOBLE METAL CHEMICAL ADDITION

9.3.7.1 Description

Noble Metal Chemical Addition (NMCA) is a process used to inject noble metals into the reactor coolant to enhance the effectiveness and efficiency of Hydrogen Water Chemistry (HWC) in mitigating Intergranular Stress Corrosion Cracking (IGSCC) in Boiling Water Reactor (BWR) vessel internals. In addition, use of NMCA allows lowering injection rates of HWC which in turn reduces radiation exposure to plant personnel. NMCA may be performed using Classic NobleChem™ (injection during hot standby conditions) or On-Line NobleChem (injection during power operation conditions) application specifications and procedures.

NMCA treatments have been applied into the DAEC's reactor coolant in an effort to mitigate IGSCC in the reactor vessel internals. The reactor water limits in the Technical Requirements Manual were changed to support classic application of the NMCA (References 4 through 9).

REFERENCES FOR SECTION 9.3

1. General Electric Company, Anticipated Transient Without Scram (ATWS) Response to NRC Rule 10CFR50.62, GE/NEDE-31096-P, December 1985.

2. General Electric Company, Assessment of ATWS Compliance Alternatives, GE/NEDC-30921, March 1985.

3. General Electric Company, Duane Arnold ATWS Assessment, GE/NEDC-30859-1, March 1985.

4. IES Utilities Inc., RTS-290, Reactor Water Conductivity Limit Change for Noble Metal Chemical Addition, NG-96-1297, July 5, 1996.

5. IES Utilities Inc., Noble Metal Chemical Addition 10CFR50.59 Safety Evaluation, SE 96-07, August 1996.

6. IES Utilities Inc., 10CFR50.59 Safety Evaluation for Noble Metal Pretreated Fuel Pins, SE 96-09, August 1996.

7. IES Utilities Inc., Noble Metal Chemical Addition Monitoring Equipment, Engineering Change Package 1573, October 3, 1996.

8. IES Utilities Inc., TRMCR-004, October 20, 1999.

9. IES Utilities Inc., 10CFR50.59 Safety Evaluation for 2nd Noble Metal Chemical Addition, SE 99-046, August 1999.

10. NRC Information Notice 2001-13, "Inadequate Standby Liquid Control System Relief Valve Margin," August 10, 2001.

11. NG-01-0909, "Response to Request for Additional Information (RAI) to Technical Specification Change Request TSCR-042, Extended Power Uprate," August 16, 2001.

Table 9.3-1

LOCATION OF SAMPLING POINTS

Description	Locations	Purpose
	Reactor Reciruclation System	
Reactor primary coolant water	Reactor vessel outlet	Monitor reactor when cleanup is isolated[a]
	Reactor Water Cleanup System and Fuel Pool Cooling and Cleanup system	
Filter-demineralizer influent	Filter inlet pipe	Demineralizer efficiency[a]
Filter-demineralizer effluent	Filter outlet pipe	Filter efficiency[a]
	Nuclear Steam Supply System	
Main steam	Main steam line	Carryover steam quality, H_2 and 0_2, Monitor corrosion and radioactivity
Suppression pool	Suppression pool reciruclation pipe	Monitor corrosion and radioactivity
Standby liquid control system	Reciruclation pipe	Borate concentration
Reactor shutdown cooling system	RHR system header	Check corrosion inhibitor concentration
	Condensate System	
Condensate	Condensate pump discharge	Condensate quality and tube leaks[a]
Condensate demineralizer effluent	Demineralizer system outlet	Treated condensate quality[b]

[a] Conductivity
[b] Conductivity and dissolved oxygen.

Table 9.3-1 Sheet 2 of 4

LOCATION OF SAMPLING POINTS

Description	Locations	Purpose
	Reactor Feedwater System	
Reactor feedwater system	After last heater of each train	Water analyses[c]
Reactor feedwater system	After first heater of each train	Water analyses[c]
	Reactor Building Cooling Water System	
Cooling water sample	Outlet of each major heat exchanger	Determine location of heat exchanger leaks
Cooling water sample	Pump discharge	Check corrosion inhibitor concentration
	Main Condenser Circulating Water System	
Influent	Inlet to cooling water heat exchanger	Determine radioactivity[d]
Effluent	Cooling water blowdown canal	Radioactivity[d]
	Liquid Radwaste System	
Radwaste surge tanks	Radwaste surge pumps discharge	Process data
Waste collector tank	Pump discharge	Process data
Floor drain collector tank	Pump discharge	Process data

[c] Conductivity and total iron.
[d] Conductivity and pH.

Table 9.3-1 Sheet 3 of 4

LOCATION OF SAMPLING POINTS

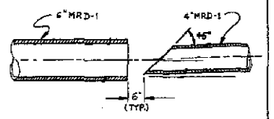
Description	Locations	Purpose
Liquid Radwaste System (Continued)		
Detergent drain tanks	Pump discharge	Process data
Waste sample tanks	Pump discharge	Process data
Fuel pool filter-demineralizer influent	Filter inlet	Fuel pool quality[a]
Floor drain sample tank	Pump discharge	Process data
Chemical waste tank	Pump discharge	Process data
Chemical waste sample tank	Pump discharge	Process data
Waste collector filter	Filter outlet	Filter efficiency
Waste collector demineralizer	Dimneralizer outlet	Demineralizer efficiency[a]
Floor drain filter	Filter outlet	Filter efficiency
Floor drain demineralizer	Demineralizer outlet	Demineralizer efficiency[a]
Low-Level Radwaste Processing and Storage Facility sample tank	Sample tank inlet	Process Data
Makeup Demineralizer System		
Cation effluent	Outlet pipe	Demineralizer efficiency
Anion effluent	Outlet of each unit	Demineralizer efficiency [a, d]
Condensate storage tank	Demineralizer water transfer pump discharge	Water quality[a]

[a] Conductivity.
[d] Conductivity and pH.

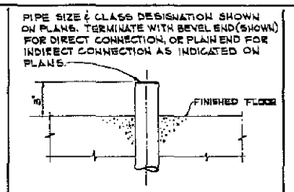
Table 9.3-1 Sheet 4 of 4

LOCATION OF SAMPLING POINTS

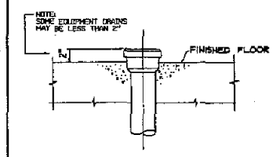
Description	Locations	Purpose
	Offgas System	
Air ejector sample	After air ejectors	Radioactivity, H_2, 0_2, and air leakage
Offgas filter samples	Inlet and outlet	Process data
Stack sample	Main stack	Radioactivity



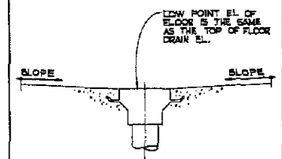
22 M-500 CONVEYOR DRAIN
RADWASTE SYSTEM.



5 M-500 EQUIPMENT DRAIN
RADWASTE SYSTEMS



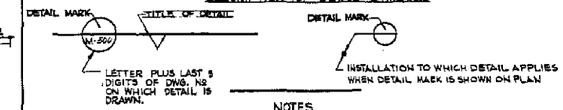
10 M-500 EQUIPMENT DRAIN
CAST IRON PIPE



15 M-500 FLOOR DRAIN WITH
INSIDE CAULK OUTLET

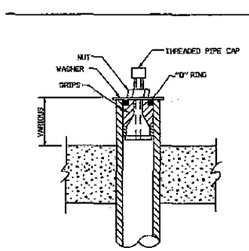
LEGEND
PLUMBING SYSTEMS OR RADIOACTIVE WASTE SYSTEMS

DESCRIPTION OF DETAIL SYMBOLS

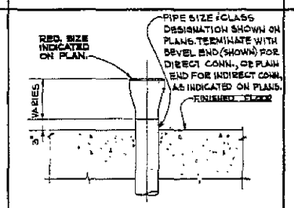


NOTES

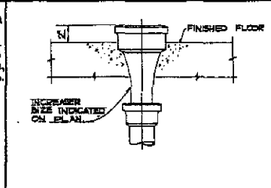
1. FOR PIPING CLASS SEE SPEC N-7854-M-150; SEC. 5T FOR ADMIN. BLDG.
2. SIZE AND CLASS OF PIPE WILL BE NOTED AT RANDOM, ADJACENT TO PIPE LINE.
3. ALL DRAINAGE LINES 4" AND LARGER SHALL HAVE A MIN. SLOPE OF (1/8) PER FOOT, UNLESS NOTED OTHERWISE.
4. ALL DRAINAGE LINES 3" AND SMALLER SHALL HAVE A MIN. SLOPE OF (1/4) PER FOOT, UNLESS NOTED OTHERWISE.
5. ELEVATIONS SHOWN TO FLOOR DRAINS AND CLEANOUTS ARE TOP OF FUTURE ELEVATIONS.
6. FLOOR DRAINS SHALL BE OF 4" NOMINAL SIZE UNLESS OTHERWISE NOTED.
7. CLEANOUTS SHALL BE OF 4" NOMINAL SIZE UNLESS OTHERWISE NOTED.
8. EQUIPMENT DRAINS SHALL BE OF 2" NOMINAL SIZE UNLESS OTHERWISE NOTED.
9. ALL CHANGES OF DIRECTION FOR RADIOACTIVE WASTE PIPING SHOWN ON PLANS SHALL BE BY MEANS OF 90° OR 45° LONG RADIUS FITTINGS, OR COMBINATIONS THEREOF UNLESS OTHERWISE NOTED.
10. REDUCERS ON RADWASTE DRAINAGE PIPING SHALL BE OF THE ECCENTRIC TYPE & INSTALLED FLAT SIDE DOWN, UNLESS OTHERWISE NOTED.



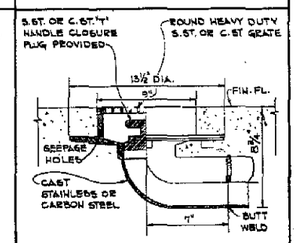
23 M-500 EQUIPMENT DRAIN
RADWASTE SYSTEMS



6 M-500 EQUIPMENT DRAIN
RADWASTE SYSTEMS



11 M-500 EQUIPMENT DRAIN
CAST IRON PIPE



16 M-500 FLOOR DRAIN
RADWASTE SYSTEMS

SYMBOLS

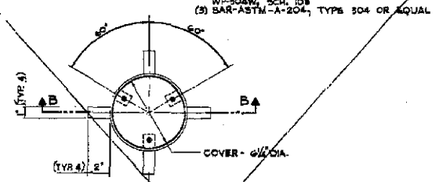
SYMBOLS	DESCRIPTIONS
	FLOOR DRAIN (NORMALLY OPEN)
	FLOOR DRAIN (NORMALLY PLUGGED)
RD AW	ROOF DRAIN ABOVE
	REDUCER, CONCENTRIC (EXCEPT - SEE NOTE 10)
	FLOOR CLEANOUT
	EQUIPMENT DRAIN (DIRECT CONNECTION)
	EQUIPMENT DRAIN (INDIRECT CONNECTION)
	HOSE BIBB
DS	DOWNSPOUT (SUSPENDED PIPING)
S	STORM DRAIN (UNDERGROUND OR EMBEDDED)
A	AIR PIPING
G	GAS PIPING
VAC	VACUUM PIPING
TP	TRAP PRIMER PIPING
V	VENT
AV	ACID VENT
DI	DISTILLED WATER
	CLEANOUT IN RUN OF PIPE

SYMBOLS	DESCRIPTION
	COLD WATER (DOMESTIC)
	HOT WATER (DOMESTIC)
	HOT WATER CIRCULATION (DOMESTIC)
4"-MRD-1	LINE DESIGNATION SYSTEM PIPE CLASS PIPE SIZE
	FIELD WELD
SAN	SANITARY WASTE
DS	ROOF DRAINAGE PIPING
AW	ACID WASTE
	NOT TO SCALE
	WALL HYDRANT
	BALANCING VALVE
	GATE VALVE
	CHECK VALVE
	PLUG VALVE

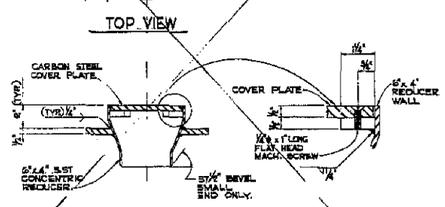
ABBREVIATIONS

B.F.P.	BACKFLOW PREVENTER	CONN.	CONNECTIONS
EL.	ELEVATION	SK.	SINK
I.E.	INVERT ELEVATION	S.S.K.	SERVICE SINK
B.L.E.	BOTTOM OF LINE ELEVATION	C.W.	CUP SINK
H.PT.	HIGH POINT	K.SK.	KITCHEN SINK
L.P.T.	LOW POINT	L.SK.	LABORATORY SINK
℄	CENTER LINE	S.D.	SHOWER DRAIN
TYP.	TYPICAL	SH.	SHOWER HEAD
FL.	FLOOR	R.D.	ROOF DRAIN
F.D.	FLOOR DRAIN	D.S.	DOWNSPOUT
E.D.	EQUIPMENT DRAIN	W.C.	WATER CLOSET
C.O.	CLEANOUT	U.R.	URINAL
F.C.O.	FLOOR CLEANOUT	LAV.	LAVATORY
W.C.O.	WALL CLEANOUT	F.H.	FUME HOOD
B.W.V.	BACK WATER VALVE	FIN. FL.	FINISHED FLOOR
H.B.	HOSE BIBB	B.V.	BALANCING VALVE
C.W.	COLD WATER	CONT.	CONTINUATION
H.W.	HOT WATER	S.SH.	SAFETY SHOWER
H.W.C.	HOT WATER CIRCULATION	E.W.	EYE WASH
E.W.C.	ELECTRIC WATER COOLER	S.ST.	STAINLESS STEEL
H.W.H.	ELECTRIC WATER HEATER	C.ST.	CARBON STEEL
W.F.	WASH FOUNTAIN	C.I.	CAST IRON
U.N.O.	UNLESS NOTED OTHERWISE	DWG.	DRAWING
E.S.E.W.	EMERGENCY SHOWER AND EYE WASH	W.H.	WALL HYDRANT
B.B.	BEVELED END	G.V.	GATE VALVE
P.E.	PLAIN END	C.V.	CHECK VALVE

NOTE
(1) ALL MATERIAL STAINLESS STEEL, EXCEPT COVER PLATE.
(2) PIPE REDUCER - ASTM-A-403 GRWP 304 OR WP-304W, SCH. 10S
(3) BAR-ASTM-A-276, TYPE 304 OR EQUAL

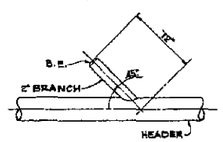


TOP VIEW

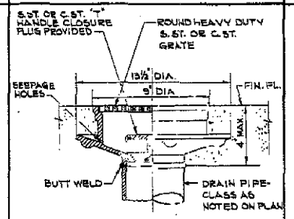


SECTION B-B

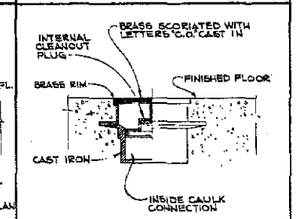
2 M-500 FLOOR CLEANOUT



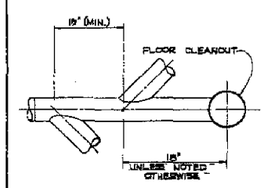
1 M-500 2 BRANCH CONNECTION
RADWASTE SYSTEMS



12 M-500 FLOOR DRAIN
RADWASTE SYSTEMS

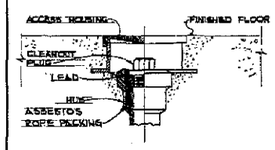


17 M-500 FLOOR CLEANOUT
CAST IRON PIPE

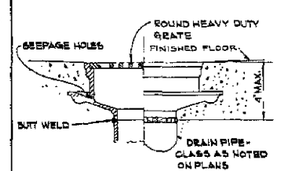


PLAN

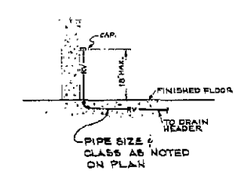
8 M-500 MAKE-UP CONNECTIONS
RADWASTE SYSTEMS



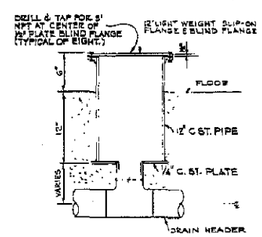
13 M-500 FLOOR CLEANOUT
ACID WASTE SYSTEM



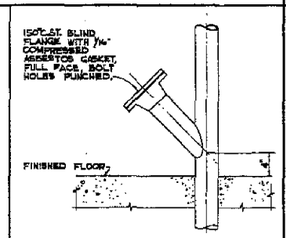
18 M-500 FLOOR DRAIN
RADWASTE SYSTEMS



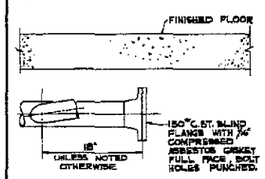
20 M-500 TERMINATION OF
RADWASTE VENT



3 M-500 INSTRUMENT SUMP
RADWASTE SYSTEMS

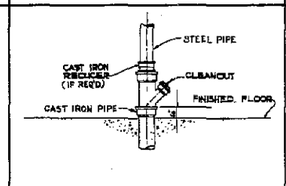


4 M-500 C.O. IN VERTICAL PIPE
RADWASTE SYSTEMS

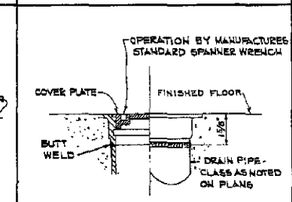


ELEVATION

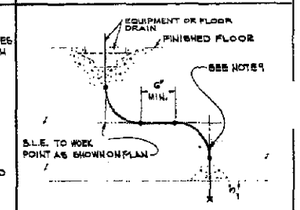
9 M-500 CLEANOUT IN HORIZ.
RUN OF PIPE



14 M-500 CLEANOUT AT BASE
OF RISER



19 M-500 FLOOR CLEANOUT
RADWASTE SYSTEMS



ELEVATION

21 M-500 PIPING OFFSET REQ'D
FOR SHIELDING

DUANE ARNOLD ENERGY CENTER
NEXTERA ENERGY DUANE ARNOLD, LLC
UPDATED FINAL SAFETY ANALYSIS REPORT

DRAINAGE DETAILS AND LEGEND

FIGURE 9.3-2

9.4 AIR-CONDITIONING, HEATING, COOLING, AND VENTILATION SYSTEMS

9.4.1 GENERAL

9.4.1.1 Design Bases

9.4.1.1.1 Power Generation Objective

The power generation objective of the heating, ventilating, and air-conditioning (HVAC) systems is to control the plant air temperatures and the flow of airborne radioactive contaminants to ensure the operability of plant equipment and the accessibility and habitability of plant buildings and compartments.

9.4.1.1.2 Power Generation Design Bases

The HVAC systems are designed

1. To provide temperature control and air movement control for personnel comfort.

2. To optimize equipment performance by the removal of the heat dissipated from the plant equipment.

3. To provide a sufficient filtered fresh air supply for personnel.

4. To provide for air movement from lesser to progressively greater areas of radioactive contamination potential before final exhaust.

5. To minimize the possibility of exhaust air recirculation into the air intake.

9.4.1.2 General System Description

Each building, such as the reactor building, turbine building, radwaste building, and the low-level radwaste processing and storage facility (LLRPSF) has a separate ventilation and/or air-conditioning system or fresh air makeup system supplying filtered outside air at a minimum flow rate of one air change per hour (Figures 9.4-1 and 9.4-2). Design flow rates are sufficient to control cross flow between areas of differing contamination levels and to control ambient temperatures. Exhaust air from the reactor building, radwaste building, and areas below the operating floor of the turbine building is discharged directly to the building exhaust vent. The exhaust air from the LLRPSF is discharged through a separate exhaust vent.

Removal of air from the space above the operating floor of the turbine building is accomplished by means of three vaneaxial exhaust fans. The air is exhausted through eight evenly distributed openings at the roof into a common duct system. The common exhaust duct is installed above the roof.

The main control room, radwaste control rooms, office areas in the LLRPSF, the office building, and other areas are air-conditioned by separate air-conditioning systems and are maintained at a positive pressure to prevent the infiltration of air and accompanying dust. Humidification or dehumidification is provided where required.

The drywell air-conditioning system controls temperatures within the values shown in Section 6.1. Temperature control ensures that the service lives of motor insulation, gaskets, cable coverings, sealants, etc., are not shortened because of high operating temperatures. The suppression chamber is not air-conditioned during normal operation. Means are provided for purging the drywell and suppression chamber before personnel entry.

Air-conditioning systems or fresh air makeup ventilation systems supply filtered air to all main areas of the plant. Sufficient fresh air makeup is provided to control contamination. The drywell air-conditioning system uses 100% recirculation during normal operation. The office building and control rooms use varying amounts of fresh air depending on the season.

Normal air flow in the reactor building, turbine building, radwaste building, and LLRSPF is controlled by the building ventilation system, consisting of the fresh air makeup systems and the exhaust systems. The air flow pattern is from potentially low-contamination areas toward potentially greater-contamination areas before exhaust.

9.4.1.3 Inspection and Testing Requirements

The HVAC systems are in continued use and no special testing or inspection is considered necessary during operation.

9.4.2 REACTOR BUILDING VENTILATION SYSTEM

9.4.2.1 Design Bases

See Section 9.4.1.1.

9.4.2.2 System Description

The ventilation system supplying filtered air to the reactor building is divided into two subsystems. One of these supplies air at the refueling floor level and the other supplies air below the refueling level. Filters for all fresh air intakes are rated at a minimum of 80% to 85% average efficiency by the ASHRAE test. [redacted] See Figures 9.4-3 and 9.4-4.

The exhaust ventilating system serving areas below the refueling level maintains negative pressures (with respect to reactor building interior ambient pressure) in the following areas:

1. Spaces between the drywell head and shield plug.

2. Cleanup filter and demineralizer area.

3. Control rod storage and repair location.

4. Reactor water cleanup system heat exchanger area.

5. Cleanup phase separator tank area.

6. Reactor water cleanup pump location.

7. Neutron monitor area.

8. Steam and feedwater piping location.

Should a release of radioactivity to the secondary containment occur of such magnitude as to activate the ventilation system radiation monitors, the reactor building is isolated by closing all intake and exhaust openings with the exception of the opening for ducts leading to the standby gas treatment system. In addition, all supply and exhaust fans are shut down. The standby gas treatment system filters and discharges the air to the elevated release point.

A separate fan supplies air to the drywell during purge. The main exhaust fan or standby gas treatment system removes purge air from the drywell during this operation.

Heating is supplied by unit heaters at the perimeter of the reactor building. Outside air is tempered to approximately 54 to 72°F during the winter by coils in the main supply air plenum. The temperature is adjusted for internal heat load.

Cooling for the reactor building is provided by cooling coils located in the main supply air plenum of the ventilation system.

When required to isolate the secondary containment, the supply and exhaust systems are shut down and all openings to the environs are closed. Potentially contaminated air within the building is exhausted by the standby gas treatment system.

Air to heat and ventilate the recombiner building is drawn from the reactor building. Supply air is cooled and passed through a medium-efficiency filter bank. Exhaust air is passed through a HEPA filter bank, after which it is discharged into the main plant exhaust plenum.

9.4.2.3 Inspection and Testing Requirements

See Section 9.4.1.3.

9.4.3 TURBINE BUILDING VENTILATION SYSTEM

9.4.3.1 Design Bases

See Section 9.4.1.1.

9.4.3.2 System Description

The turbine building is ventilated by a once-through system consisting of one supply subsystem and three distinct exhaust subsystems. Air is generally supplied to areas of less potential contamination and exhausted from areas of greater potential contamination.

Supply air is drawn through the main plant intake coils (where it is heated or cooled depending on the season) by three supply fans located in the reactor building equipment room. Each of the three fans supplies air to a common mixing header from which the air is distributed to various areas of the turbine building. See Figures 9.4-5 and 9.4-6.

Exhaust is taken from three distinct areas. Air is exhausted from the operating floor by way of eight roof exhaust ducts which are connected to three exhaust fans via a common header. The three exhaust fans then discharge to the environs by way of a monitored release point. This system is primarily provided for temperature control. Air is exhausted from the general area of

the condenser and heater bays by way of a duct to the main plant exhaust plenum. [redacted] The system consists of two redundant 100% capacity fans. Should a Group 3 isolation occur, the standby gas treatment system makes use of common ductwork to the offgas stack and exhaust flow from this system may be interrupted.

In the event of a Group 3 isolation concurrent with a main plant ventilation stack high radiation level alarm, [redacted] In accordance with annunciator response procedures, [redacted] from the reactor building via the main plant ventilation stack. In accordance with annunciator response procedures, the turbine building supply fans [redacted] to keep the turbine building at a negative pressure with respect to the environs. [redacted]

Fresh air makeup to this building is filtered by units rated at a minimum of 80% to 85% average efficiency by the ASHRAE test. Heating coils in the [redacted] temper the air during cold weather. Air is supplied to different areas of the building by supply duct systems and differential- pressure flow.

9.4.3.3 Inspection and Test Requirements

The air distribution system for the turbine building was tested in accordance with Associated Air Balance Council procedures and balanced to provide design air quantities at each outlet to a tolerance of +10% -0%. The hydronic system was also tested and balanced.

9.4.4 CONTROL ROOM VENTILATION SYSTEM

9.4.4.1 Design Bases

Refer to Sections 9.4.1.1 and 9.4.6.1.

9.4.4.2 System Description

The control room air-conditioning system has two normal modes of operation controlled from [redacted] The system can operate in a recirculation mode which will provide 1.2 air changes per hour. The system also has a fresh air (purge) mode which will provide six air changes per hour. [redacted] See Figures 9.4-7 and 9.4-8.

[redacted]

Two 1000 cfm single-pass high-efficiency filter trains are provided in parallel with the normal outside air inlet duct. The filter trains each consist of inlet and outlet isolation dampers, a heating coil, high-efficiency particulate absorber (HEPA), charcoal filter (2-in. bed, tray-type), and final HEPA filter.

Control room air is recirculated through dust filters and heated or cooled as necessary to maintain comfortable working conditions. Power for the filtration-recirculation system may be supplied from the emergency bus. The filtration-recirculation system is Seismic Category I and is located in a Seismic Category I structure.

Two types of ductwork systems distribute air from the filter trains. One supply system is connected to the cable spreading room below the control room floor and supplies cooling air directly to the space. The other supply system is for general space cooling and consists of ductwork supplying ceiling diffusers and air flows upward through the central panels, out to the

return ductwork system, and back to the filter train. Space air returns to the filter train through a return air system.

The control room is served by an air-conditioning system that also serves the cable spreading room, battery rooms, and essential switchgear rooms. [redacted]

The control room recirculation air-conditioning system consists of two completely redundant units that include roughing filters, medium-efficiency filters, heating-cooling coils, and fans.

This system is designed to maintain the room air at 75°F dry bulb and not more than 50% relative humidity when the outdoor air is at either summer design conditions, 90°F dry bulb and 76°F wet bulb, or winter design conditions, -6°F.

When normal plant and offsite power is unavailable, the emergency diesel-generators will power system fans and will allow the water chillers to operate so they can maintain the control room at its design temperature described above.

[redacted]

The control room ventilation system has the following operational capabilities:

1. [redacted]

2. [redacted] Most of the exhaust is recirculated. The remaining exhaust goes to the cable spreading room.

3. [redacted] Control room air is totally recirculated through control room air-conditioning units. Recirculated air also ventilates the HVAC equipment room and switchgear room.

4. In cases 1 and 2 above, the cable spreading room can be isolated from the control room air supply when desired, such as when carbon dioxide has been discharged inside the cable spreading room to extinguish a fire. [redacted] This venting action serves to minimize the potential for CO_2 intrusion into the Control Room upon Cardox actuation.

5. The ventilation air returning from each essential switchgear room can be isolated by bubble tight smoke dampers installed in each room's return duct. The smoke dampers automatically close upon detection of smoke by the duct mounted smoke detector or on a loss of power event. This isolation function will prevent the propagation of smoke from an essential switchgear room to the control room.

   The control room ventilation system is an engineered safety feature ventilation system and is further discussed in Section 9.4.6.

9.4.4.3 Safety Evaluation

The exposure to control room personnel occurs during postulated design basis accidents. The results of analyses performed to estimate the TEDE dose to control room personnel and associated calculation methodology are presented in Section 15.2. All calculated control room personnel doses are less than the guidelines of 10 CFR 50, Appendix A, General Design Criterion 19, and 10 CFR 50.67

9.4.4.4 Inspection and Testing Requirements

The control room ventilation system HEPA filters were tested before installation in accordance with the contract specifications as follows:

1. Thermally generated DOP test of each cell, in accordance with Edgewood Arsenal Manual, to indicate efficiency at 100% and 20% of rated capacity.

2. Retesting by AEC quality assurance stations to check on the manufacturer. These tests determine the efficiency of individual filters for a specific particle size, 0.3 μ. The test is made with monodisperse, thermally generated DOP.

3. Two units selected at random were given the rough handling test in accordance with MIL-STD-282.

Each HEPA filter bank was tested in place by a system integrity test with cold generated DOP to detect leaks in filter cells or in seals.

Carbon lot tests were made, in accordance with ORNL-65-A-2, as follows:

1. Iodine collection efficiency.
2. Particle size analysis.
3. Ignition temperature test.
4. Apparent density.
5. Moisture test.
6. CCl_4 activity.

Performance tests were made, as required by the contract specification, to verify ability to remove 99.95% of elemental iodine and 85% of methyl iodide at 95% relative humidity.

The carbon filter banks were tested in place, in accordance with the requirements of the contract specification by a system integrity test with vaporized F-112 to detect leaks in filter cells or in seals.

The air distribution system has been tested in accordance with Associated Air Balance Council procedures and balanced to provide design air quantities at each outlet.

A test of the pressure drop across the combined HEPA filters and charcoal adsorbers is performed in accordance with the Technical Specifications to indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter.

The control room ventilation system HEPA filters and charcoal adsorber banks are tested in place and charcoal samples are removed periodically from the adsorber banks for laboratory tests of adsorber efficiency.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the HEPA filters and charcoal adsorbers are performed in accordance with the Technical Specifications. Test cartridges are provided to allow removal of a representative charcoal sample without affecting the operation of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 9.4-3. Any HEPA filters found defective shall be replaced. The replacement HEPA filters should be steel cased and designed to military specifications MIL-F-51068 and MIL-F-51079. The HEPA filters should satisfy the

requirements of UL-586. The Filter Test Facility should test each filter at 100% and 20% of rated flow, with the filter encapsulated to disclose frame and gasket leaks.

Operating each Standby Filter Unit (SFU) for ≥ 15 minutes every 31 days ensures that both subsystems are operable and that all associated controls are functioning properly.

If significant painting, fire and chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign materials, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

Test frequency and limiting conditions for operation are contained in the Technical Specifications.

See Section 9.4.6.4 for additional inspection and testing requirements during operations.

9.4.5 RADWASTE BUILDING VENTILATION SYSTEM

9.4.5.1 Design Bases

See Section 9.4.1.1.

9.4.5.2 System Description

The radwaste building is served by two ventilating systems, one for the radwaste control room and one for the radwaste area and equipment room. The radwaste control room unit supplies the room with a mixture of outdoor air and recirculated air. Air passes through medium-efficiency air filters and can be treated or cooled as required. Refer to Figure 9.4-9.

[redacted] can supply 100% of design capacity. The radwaste area is exhausted by means of two redundant exhaust fan units, each of which contains a fan, prefilters, and HEPA filters to the space surrounding the torus, which is used as the reactor building exhaust plenum. Return air from the radwaste control room is recycled through both the radwaste area system and the radwaste control room system. The radwaste building can be isolated from connections to other buildings through the ventilating system by closing the dampers at the discharges of the exhaust units serving the radwaste area.

[redacted] Fresh air makeup units supply the areas of the building with filtered air through a supply ductwork system. The fresh air makeup units are equipped with tempering coils for winter usage to increase the temperature to 56 to 68°F, depending on internal heat load. Air is removed from the spaces through exhaust systems. The interaction of the supply and exhaust systems is such that air will flow from areas of lesser to areas of progressively greater contamination potential before entry into exhaust filter units. All exhaust air is treated as potentially contaminated and filtered. Heating is provided by duct and unit heaters.

9.4.5.3 Inspection and Testing Requirements

The radwaste building ventilation system HEPA filters will be tested before installation as follows:

1. Thermally generated DOP test of each cell, in accordance with Edgewood Arsenal Manual, to indicate efficiency at 100% and 20% of rated capacity.

2. Retesting by AEC quality assurance stations to check on the manufacturer. These tests determine the efficiency of individual filters for a specific particle size, 0.3 μ. The test is made with monodisperse, thermally generated DOP.

3. Two units selected at random were given the rough handling test in accordance with MIL-STD-282.

Each HEPA filter bank was tested in place by a system integrity test with cold generated DOP to detect leaks in filter cells or in seals.

The air distribution system has been tested in accordance with Associated Air Balance Council procedures and balanced to provide design air quantities at each outlet to a tolerance of +10% -0%.

9.4.6 ENGINEERED SAFETY FEATURE VENTILATION SYSTEM

9.4.6.1 Design Bases

9.4.6.1.1 Safety Objective

The safety objective of the engineered safeguards heating and ventilating systems is to maintain suitable temperatures for equipment protection in the plant emergency equipment rooms, as follows:

9.4.6.1.2 Safety Design Bases

1. The system is designed to protect the safeguards equipment against overheating.

2. The system is provided with redundant components for reliable operation.

3. Power supply to appropriate cooling and ventilating equipment is provided from the standby power supply system during loss of offsite power supplies.

4. All equipment is designed to withstand the DBE motions without impairing system function.

9.4.6.2 System Description

Engineered safeguards heating and ventilation systems are provided for the following areas:

1. Control, emergency switchgear, and battery rooms.
2. Standby diesel-generator rooms.
3. Pump structure emergency cooling water pump rooms.
4. Reactor building RHR, RCIC, HPCI, and core spray rooms.

9.4.6.2.1 Control, Emergency Switchgear, and Battery Rooms

The system consists of an air supply, a return system, and an exhaust system. Supply air to the switchgear rooms is recirculated while that to the battery room is exhausted to the atmosphere. Supply air is filtered and tempered with heating coils as required. The equipment is installed in a Seismic Category I structure. [redacted] System controls are located on a local panel, and in the control room back panel area. Redundant fans are provided for reliable system operation. See Figure 9.4-7.

In the event of loss of ventilation in the emergency switchgear rooms during shutdown of the control building ventilation system due to fire, an alternative ventilation path is established to cool the Division II switchgear room. This ventilation path is established by opening security doors and manually energizing two permanently mounted fans. These fans are provided with power from the emergency bus and provide air flow from the control building through the Division I and II switchgear rooms into the turbine building.

The ventilation air returning from each essential switchgear room can be isolated by bubble tight smoke dampers installed in each room's return duct. The smoke dampers automatically close upon detection of smoke by the duct mounted smoke detector or on a loss of

power event. This isolation function will prevent the propagation of smoke from an essential switchgear room to the control room.

9.4.6.2.2 Standby Diesel-Generator Rooms

[redacted] See Figure 9.4-5.

9.4.6.2.3 Emergency Cooling Water Pump Rooms

[redacted] Heating is provided for equipment and piping freeze protection. The ventilation system is supplied with standby power during loss of offsite power.

Supply fans introduce filtered air through roughing and medium-efficiency filters into the pump house to remove excess heat generated by equipment. The air is mostly recirculated and is tempered by mixing return air with outdoor air to maintain design temperature.

Two physically separated Seismic Category I supply fans supply cooling air to the area where the RHR service water pumps and the emergency service water pumps are located. One supply fan is used to provide cooling air to each division of RHR service water and emergency service water pumps. These fans are connected to the emergency bus. When a fan operates, the exhaust louvers automatically open to permit exhaust.

The heating of the pump house for freeze protection is by electric unit heaters.

9.4.6.2.4 Reactor Building RHR, RCIC, HPCI, and Core Spray Pump Rooms

[redacted] Fan coil units using Emergency Service Water are used to limit pump room temperatures during accident conditions. Heating is provided for equipment and piping freeze protection. The fan coil units are supplied with emergency power during loss-of-offsite-power events. See Figures 9.4-3 and 9.4-4.

A non-essential cooling system was added in the HPCI and RCIC rooms to maintain the normal operation temperatures as defined in Table 9.4-1. This system supplements the normal building ventilation when the emergency coolers are not in service. [redacted] The chiller provides glycol to the non-essential cooling units and exhausts heat directly to the atmosphere. The fan coil units and associated piping and ductwork for this system, within the HPCI and RCIC rooms, were designed to seismic

category I criteria to prevent damage to safety-related equipment within these rooms during seismic events.

9.4.6.2.5 Primary Containment

Refer to Section 6.2 for a discussion of the containment ventilation system. (Figures 6.2-59 and 6.2-60).

9.4.6.3 Safety Evaluation

The engineered safeguards equipment rooms are provided with heating, cooling, and ventilation systems required to ensure the protection of equipment during normal and accident conditions.

This is accomplished by having sufficient redundancy, where necessary, so that no single component failure will prevent this system from achieving its safety objective. In addition, equipment is installed in Seismic Category I structures and is supplied with normal and standby power.

The criteria governing the design of the air conditioning and ventilation systems for the control room, HPCI room, and RCIC room require that maximum temperatures for normal operating conditions not be exceeded assuming the failure of any one active component.

To achieve the design objectives, these areas are cooled and/or ventilated by systems that provide for 100% redundancy. The calculated temperatures are based on only one of these units operating. Furthermore, the units are housed in Seismic Category I structures. Where cooling water is required for chillers or cooling units, it is supplied by the Emergency Service Water System, which is also designed as a safety system. The cooling and/or ventilating units are powered from redundant portions of the standby power system.

Air conditioning and ventilation systems for other safety-related equipment rooms, such as the RHR/core spray pump rooms, pump house, intake structure, and diesel-generator rooms, [redacted] For example, if an RHR/core spray pump room cooler fails, the other division of RHR/core spray may have to be used.

Table 9.4-1 compares the maximum temperatures for normal equipment operation and the calculated maximum room temperature. The calculated maximum room temperatures are based on outside ambient temperatures of 90°F dry bulb and 76°F wet bulb. The local outside ambient temperature is expected to exceed these values approximately 2.5% of the time.

[redacted]

Based on this design, there is no postulated instance wherein a loss of an air conditioning or ventilation system can cause complete loss of the system or structure for which cooling is provided.

[redacted]Loss of heat exchanger and chiller output would result in the loss of air conditioning capability; however, the ventilation system could function. If the temperature rises to the 104°F maximum, the alternative ventilation flow path would maintain temperature below 104°F.

In safety-related equipment rooms other than in the control building, a more severe environment can occur in the event of a steam leak. An analysis has been conducted to determine the maximum steam leak that can be tolerated without exceeding the capacity of the air-conditioning and/or ventilation system in that particular compartment. The results of this analysis are summarized below:

Compartment	Maximum Tolerable Steam Leak
HPCI compartment	1,300 lb/hr
RCIC compartment	200 lb/hr
RHR compartments	840 lb/hr
Torus area	15,210 lb/hr

2012-016 Sources for the design inputs to determine the "Maximum Tolerable Steam Leak" have been lost, but since they were design verified at the time, the results are still valid until the piping configuration is changed or the analysis is redone.

In each compartment listed above, normal steam leakage will not exceed the value of the maximum tolerable steam leak.

In the event of excessive leakage, the temperature and relative humidity in the affected compartment could rise to 212°F and 100% respectively. However, when the temperature reaches the high-temperature setpoint (175°F in the RCIC and HPCI compartments), this condition is alarmed in the control room and isolation occurs as discussed in Section 7.3.

2012-016 The intake structure and the portion of the pump house containing the RHR service water and emergency service water pumps rooms have heating and/or ventilating systems designed to operate between 104°F and 38°F. The high-temperature has been established to prevent damage to motor control centers and the low-temperature has been established to prevent freezing. In the event of extreme outside temperatures coupled with the failure of the heating and/or ventilating systems, intolerable environmental conditions may exist in these rooms. Alarms in the control room alert the operators to an abnormal condition and allows them to take corrective action before any equipment failure.

9.4.6.4 Inspection and Testing Requirements

The engineered safeguards heating and ventilating system is proved operable by its use during normal plant operations. Portions of the system normally closed to flow can be tested to ensure operability and integrity of this system.

Equipment specifications require operability of the equipment under the indicated maximum temperatures, but no special tests are required on control and electrical equipment to ensure operability under extreme environmental conditions. Electrical cable, however, has extensive testing requirements as indicated below.

All tests are conducted in accordance with IPCEA Standard S-19-81, Part 6 (testing and test methods) or with specially furnished instructions.

1. Physical

   a. Strength.
   b. Ozone resistance.
   c. Moisture absorption.

2. Electrical

   a. Water tank.
   b. Radiation-steam.
   c. Flame resistance.

Heating and ventilation equipment and controls were operated and tested for a period of at least five consecutive 8-hr days to demonstrate operability. The tests included the operation of the heating and air-conditioning equipment and system for a period of not less than two 8-hr days at not less than 90% of full specified heating and cooling capacities.

Table 9.4-2 describes the periodic testing of the safety-related ventilation systems and the secondary containment isolation system.

9.4.7 COMPUTER ROOM VENTILATION SYSTEM

9.4.7.1 Design Bases

See Section 9.4.1.1.

9.4.7.2 System Description

[redacted] The system consists of a centrifugal fan, cooling coil, filter train unit, isolation and modulation dampers, balancing damper, smoke/fire detection devices, and the necessary air distribution system and control appurtenances. See Figures 9.4-7 and 9.4-8. The dampers are opposed-blade and are provided with manual and pneumatic actuators for isolation and modulation control, respectively. The locking type hand quadrant is furnished with the balancing damper. The air distribution system and its accessories are located inside the control building.

[redacted] The supporting structure for the air handling unit is designed as Seismic Category I in order to protect the equipment in the computer and control rooms from impact during a seismic event. [redacted]

9.4.7.3 System Operation

The computer room HVAC system can operate continuously during all periods of plant operation. Air is supplied and recirculated into the computer room.

9.4.8 LOW-LEVEL RADWASTE PROCESSING AND STORAGE FACILITY (LLRPSF) VENTILATION SYSTEM

9.4.8.1 Design Bases

See Section 9.4.1.1.

9.4.8.2 System Description

The LLRPSF is served by five ventilating systems: one for the storage area, one for the office area, one for the processing area, one for the future expansion area, and one for the equipment room.

[redacted] Any signal from a smoke detector will align dampers to purge the facility through particulate filters.

9.4.8.2.1 Storage Portion HVAC System

The HVAC system for supplying air to the storage portion of the LLRPSF consists of a filter unit, an electric heating coil, a 100% capacity supply fan, and two exhaust fans. [redacted] Under normal conditions, this exhaust air would be unfiltered, however it can be directed through a particulate filter unit. Although the HVAC system can be operated as a once-through system, during normal operation the dampers are aligned to provide recirculation flow.

9.4.8.2.2 LLRPSF [redacted]

The [redacted] consists of a 100% capacity supply fan with an electric heating coil, a direct expansion cooling coil, a humidifier, filter units, [redacted] a toilet exhaust fan and a return fan. The return fan recirculates approximately 75% of the conditioned air which mixes with outside air to make-up the design flowrate.

9.4.8.2.3 Processing Area HVAC System

The processing area HVAC consists of a 100% capacity supply fan with an electric heating coil, a filter unit, a return fan, an exhaust fan with a particulate filter unit, and five local duct heaters that provide temperature control for various rooms. Although this system can provide 100% tempered outside air to the processing areas to maintain design conditions, 77% of the air can be recirculated. The remaining 23% of the air is exhausted from the facility through the equipment room exhaust fan. Under normal conditions, the air is not filtered prior to exhaust during the recirculation mode. However, the system also has the capability to exhaust up to 100% of the air through the particulate filter unit.

9.4.8.2.4 Future Expansion Area HVAC System

The future expansion HVAC system consists of a 100% capacity supply fan with an electric heating coil and particulate filter unit and an exhaust fan. [redacted] Air exhausted from this area is normally unfiltered but can be diverted through a particulate filter unit prior to exhaust. [redacted] This system can recirculate 77% of the supply air. During this condition, recirculated air from the mechanical equipment room is mixed with the return air to partially temper the future expansion return air.

9.4.8.2.5 Equipment Room HVAC System

The equipment room HVAC system consists of a 100% capacity supply fan with a filter unit and an exhaust fan. [redacted] A recirculation mode is also provided for this system. Exhaust air from the equipment room is not passed through a filter unit.

9.4.8.3 Inspection and Testing Requirements

All filter housings were shop tested by the manufacturers to ensure negligible leakage.

After installation of the HVAC systems, the systems were tested in accordance with the requirements of the Associated Air Balance Council or the National Environmental Balancing Bureau. The systems were balanced to ensure that air flow is within ±10% of design conditions.

9.4.9 TECHNICAL SUPPORT CENTER VENTILATION SYSTEM

The Technical Support Center ventilation system recirculates the TSC atmosphere and mixes in a quantity of outside air. [redacted]

The charcoal filter units are tray-type, with 2 inch deep beds, sized to give an air residence time of 0.25 seconds. They are effective for both elemental iodine and methyl iodide.

[redacted]

Dose to TSC personnel during design basis accidents is evaluated in UFSAR Section 15.3.

Table 9.4-1

AREA TEMPERATURE REQUIREMENTS

Area	Maximum Temperature for Normal Equipment Operation[a] (°F)	Calculated Maximum Room Temperature (°F)
Control Room	104	76
Electrical and switch-gear rooms	104	102
Battery rooms	104	102
RHR and core spray rooms	104	104
HPCI room	104	104
RCIC room	104	104
Diesel-generator room		
2015-003 Generator	140	104
2015-003 Diesel	140	138

[a] Operation at temperatures higher than those listed will not affect safe system operation but may reduce the expected operating life of the equipment.

Table 9.4-2 Sheet 1 of 3

PERIODIC TESTING - SAFETY-RELATED VENTILATION SYSTEMS

Test Item/System	Test Description
Standby filter units - control building	1. Isolation signal will be simulated to start the selected train. Performance of the fan will be checked against its required flow characteristics for flow through the HEPA filters and charcoal adsorbers. Differential-pressure reading across each filter will be recorded at the set air flow rate through the train. 2. It will be demonstrated that the electric coil functions properly and its output meets the required value. 3. The second train will be tested by similar actions. 4. Automatic initiation shall be demonstrated by verifying that upon receipt of a high radiation test signal at the air intake radiation monitors, the system automatically switches to the isolation mode.
Ventilation systems - emergency diesel-generator rooms	1. Supply fan will turned on and it will be demonstrated that all the dampers are operating properly to provide proper ventilation. 2. Performance of the fan will be checked against its preoperational balancing characteristics. 3. The second system will be tested by similar actions.
Ventilation system - battery room	1. Exhaust fan will be turned on and its performance checked against its preoperational balancing characteristics . 2. It will be demonstrated that the control system functions properly to meet the accident operating modes as designed. 3. Each fan will be tested by similar actions.

Table 9.4-2 Sheet 2 of 3

PERIODIC TESTING - SAFETY-RELATED VENTILATION SYSTEMS

Test Item/System	Test Description
Room cooling units - RHR and core spray rooms, RCIC room, HPIC room	1. Room cooling unit will be turned on and performance of the fan checked against its preoperational balancing characteristics. 2. It will be demonstrated that the cooling effect of the cooling coil meets the cooling requirement. 3. It will be demonstrated that the control system functions properly; if the room temperature rises to the setpoint then the ventilation system will turn on. 4. The second equipment will be tested by similar actions.
HVAC systems - control room	1. The operating system will be checked to ensure that all dampers are operating properly to meet the set control room pressure. 2. Performance of the fans will be checked against the preoperational balancing characteristics. 3. Differential-pressure reading across the filter will be recorded at the set air flow rate. 4. It will be demonstrated that the cooling coil functions properly to meet the cooling requirements of the accident modes. 5. It will be demonstrated that the control system functions properly. The room temperature setting shall govern the system operation. 6. The second system will be tested by similar actions.

Table 9.4-2 Sheet 3 of 3

PERIODIC TESTING - SAFETY-RELATED VENTILATION SYSTEMS

Test Item/System	Test Description
Isolation dampers - reactor building	1. Isolation signal to the selected damper will be cut off and the damper operation observed to ensure that it is closed. 2. The closure time of the damper will be observed not to exceed 10 sec.[a] 3. It will be demonstrated that the pilot lights are operating properly during the operation of the damper. 4. After test, the isolation signal will be replaced to the normal connection. 5. Similar tests will be done to each damper.
Isolation valves	Refer to Technical Specifications.

[a] The dampers in the exhaust line from the refueling floor will be observed to close within 5 sec, the time required for isolation in this instance as described in Section 6.2.3.

Table 9.4-3
Summary Table of New Activated Carbon Physical Properties
(Derived from ANSI/ASME N509-1980 Table 5-1)

Test	Test Method	Acceptance Value
Performance Requirements		
Methyl Iodide, 30°C, 95% RH (1)	ASTM D3803-1989	<5.0% penetration
2013-009 Methyl Iodide, 30°C, 95% RH (2)	ASTM D3803-1989	<0.5% penetration for a 6″ bed
Physical Properties		
Particle Size Distribution	ASTM D2862 using 8 x 16 U.S. Mesh	Retained on #6 Sieve: 0.1% maximum Retained on #8 Sieve: 5.0% maximum Through #8, on #12 Sieve: 60% maximum Through #12, on #16 Sieve: 40% minimum Through #16 Sieve: 5.0% maximum Through #18 Sieve: 1.0% maximum
Ball Pan Hardness	ASTM D3802	92 minimum
C C_4, Activity (on base)	ASTM D3467	60 minimum
Apparent Density	ASTM D2854	0.38 g/cm3 minimum
Ash Content (on base)	ASTM D2866	state value
Ignition Temperature	ASTM D3466	330°C minimum
Moisture Content	ASTM D2867	state value
pH of Water Extract	ASTM D3838	state value

Notes: (1) SFU Conditions
(2) SBGT Conditions

9.5 OTHER AUXILIARY SYSTEMS

9.5.1 FIRE PROTECTION SYSTEM

2013-013 Refer to UFSAR 13.7 for bases of DAEC Fire Protection Program and associated systems and features.

9.5.2 COMMUNICATIONS SYSTEMS

9.5.2.1 Design Bases

9.5.2.1.1 Power Generation Objective

The power generation objective of the communications systems is to provide convenient, effective operational communications between various plant buildings and locations and between the plant and remote locations.

9.5.2.1.2 Power Generation Design Bases

Communications systems are provided in the plant to ensure reliable communications for startup, operation, shutdown, and maintenance under all normal and emergency conditions.

1. A public address (PA) system is equipped with a page and with a single party channel. The page channel is used to issue plant-wide instructions, for intercommunications between two or more stations, or to call personnel who may continue their conversation on the party channel or other communications system. [redacted]

2. [redacted]

3. Sound-powered telephone jacks located throughout the plant provide communications for maintenance purposes.

4. [redacted]

5. [redacted]

6. [redacted]

2016-001 7. [redacted]

8. A radio/telephone system has been installed on the refueling floor so that the refueling bridge operators can communicate "hands-free" with the control room.

[redacted]

9.5.2.2 Description

The PA system is of industrial quality using local transistorized amplifiers. All system speakers carry the conversation during the page mode of operation. Switching to the party mode makes the page channel available to others since simultaneous conversations can take place on both the page and party channels without interference. Speakers are oriented and volume levels adjusted to cover all areas inside the plant and selected outdoor areas. Outdoor stations use weatherproof speakers and amplifier enclosures and may be turned on or off from the control room.

Switches available in several locations in the plant allow the initiation of a plant evacuation or fire alarm. Actuating the switch energizes a special oscillator whose output is fed into the PA system.

The telephone systems used at the DAEC are owned by NextEra Energy Duane Arnold, LLC and were installed and are maintained by the local telephone utility. The sound powered phone system provides communications between approximately 36 locations in the plant.

[redacted]

9.5.2.3 Safety Evaluation

Communications systems are provided in the plant to ensure reliable communications for startup, operation, shutdown, and maintenance under all normal and emergency conditions. The communication system at the DAEC consists of many diverse and redundant systems. These systems are designed to be operable during a loss of offsite power.

9.5.2.4 Inspection and Testing

The design of the communications systems permits routine surveillance and testing at any time.

Audibility problems encountered with the evacuation of personnel from high-noise areas have been evaluated as follows:

1. An evaluation of the evacuation alarm system in accessible high and low noise level areas was performed. The evaluation included a special test involving the stationing of personnel in the areas during the sounding of the evacuation alarm and an interview with operating personnel to determine if they had encountered any audibility problems with the plant paging system (the evacuation alarm is relayed over the plant paging system). The evaluation identified five plant areas where the evacuation alarm was not audible and two high noise level areas having reduced audibility.

2. Additional plant paging system speakers were installed in the five plant areas where the evacuation alarm was not audible. For plant areas where the audibility of the evacuation alarm was reduced, current preparedness plan implementing procedures provide adequate mechanisms for identifying any personnel who may not have heard the evacuation alarm. The procedures require that an audit be performed to verify accountability of all personnel. Any personnel not hearing the evacuation alarm would be identified at that time and appropriate actions to locate the personnel would be initiated. [redacted]

   In the event plant circumstances necessitate a local evacuation of a certain plant area, current procedures involve only a verbal evacuation announcement over the plant paging system. In order to ensure that all affected personnel have heard the verbal announcement of a local evacuation, the announcement will be followed by an audit of the reactor building access control log.

3. The evaluation of inaccessible plant areas for evacuation alarm audibility was completed on April 1, 1980. A schedule for the implementation of any applicable corrective actions was developed at that time.

   In addition to the 1980 evaluation, a sound pressure level analysis in seven high ambient noise areas of the plant was conducted in 1986. Sound pressure ambient noise level readings were taken at different locations in the seven areas and compared with pink noise injected into the system at the same locations.

9.5.3 LIGHTING SYSTEMS

9.5.3.1 Design Bases

9.5.3.1.1 Power Generation Objective

The power generation objective of the plant lighting system is to provide adequate normal and essential plant lighting using reliable system components, with power supplied from normal or standby ac sources and from the plant or self-contained battery systems.

9.5.3.1.2 Power Generation Design Bases

1. Lighting intensities are maintained at levels recommended by the Illuminating Engineering Society.

2. Lighting fixtures of metallic vapor, fluorescent or incandescent type, were selected with due consideration for environmental conditions and ease of maintenance. Mercury fixtures and switches are not used inside the primary containment.

3. The control room is provided with a fluorescent-lighted glare-free luminous ceiling with special attention given to the reduction of glare and shadows at the main control boards.

4. [redacted]

9.5.3.2 Description

[redacted]

[redacted]

[redacted] Within the control room emergency lighting system there are 40 fluorescent type fixtures capable of maintaining illumination at the 10 foot candle level.

2013-013 [redacted]

One set of the DAEC control room light fixtures illuminating the main control boards is powered from Division I lighting panels located in the turbine building. A second set of control room light fixtures is powered from lighting panels located in the Division II switchgear room. [redacted]

[redacted]

2013-013 [redacted]

2013-013 [redacted]

2013-013 [redacted]

2013-013 [redacted]

2013-013

9.5.3.3 Safety Evaluation

2013-013

[redacted]

9.5.3.4 Inspection and Testing

The design permits routine surveillance and testing of all critical lighting systems.

Control room panel illumination tests have been conducted to determine the lighting levels in the control room with one division of lighting inoperable. Illumination levels were measured at the panels which contain instruments and controls required to achieve Safe and Stable conditions. With lighting supplied only by the Division II lights (Both Division I - powered lights and emergency fluorescent lights off), the illumination levels on some areas of the control room benchboards were found to be slightly less than the guidance of NUREG-0700. With lighting supplied only by Division I lights, illumination levels were found to exceed the 10 foot candles acceptance criteria on the same panels.

2013-013

9.5.4 DIESEL-GENERATOR FUEL OIL STORAGE AND TRANSFER SYSTEM

9.5.4.1 Design Bases

9.5.4.1.1 Safety Objectives

The safety objective of the diesel-generator fuel supply system is to provide a reliable source of fuel that is automatically initiated on demand by the diesels. The fuel storage capacity is sufficient to safely shut down the plant after a design-basis accident coincident with loss of offsite power.

9.5.4.1.2 Safety Design Bases

1. The fuel supply system consists of two physically separate and independent flow trains; one for each diesel. Additional equipment and control redundancies are provided, including the capability to manually cross connect the two fuel supply trains. Components of the fuel supply system are Seismic Category I and those components located indoors are protected by a Seismic Category I structure.

2. The quantity of fuel in the [redacted] is sufficient to provide standby ac power continuously for 7 days. This is compatible with the diesel-generator operation requirements following a LOCA with no offsite power available. Each diesel is also provided with sufficient indoor fuel storage for approximately 4 hr of full-load operation.

2016-005 The specific Emergency Diesel Generator (EDG) fuel oil volume contained in the diesel fuel oil storage tank necessary to ensure that EDG run-duration requirements are calculated using Section 5.4 of American National Standards Institute (ANSI) N195-1976, "Fuel Oil Systems for Standby Diesel-Generators," and is based on applying the conservative assumption that the EDG is operated continuously at rated capacity. This fuel oil calculation methodology is one of the two approved methods specified in Regulatory Guide (RG) 1.137, Revision 1, "Fuel Oil Systems for Standby Diesel Generators," Regulatory Position C.1.c.

[redacted] fuel sources are located 8 miles from the plant in Cedar Rapids.

9.5.4.2 Description

The two diesel-generators are Fairbanks, Morse Model 38TD8-1/8 models, each with a 2850 KW continuous capacity. Each engine uses from 1.2 gal of fuel per minute near idle speed to about 3.4 gpm at full load (2850 KW), based upon startup test data. Fuel oil to the engine is supplied from two 1000-gal day tanks located in separate rooms. The fuel oil is pumped from the day tank to the diesel-generator by an engine-driven fuel pump that supplies 11.2 gpm on a continuous basis. Fuel oil not required for combustion in the diesel engine is bypassed through a return line to the day tank.

[redacted] A 200% capacity diesel-oil transfer pump in each train supplies fuel to the 1000-gal day tank. A diesel-driven fuel supply pump transfers fuel from the day tank to the injection pump header of each engine where two injection pumps force the fuel into each cylinder (Figure 9.5-2).

The diesel-generator fuel-oil day tanks and storage tank are provided with instrumentation to accomplish the following functions:

1. Day tanks

   a. High-level alarm.

   b. High-level diesel-oil transfer pump stop (below alarm setpoint).

   c. Low-level diesel-oil transfer pump start (above alarm setpoint).

   d. Low-level alarm.

   e. Low-low-level alarm (independent of low-level alarm).

2. Storage tanks

   Redundant level indicating switches to annunciate on minimum required diesel-generator fuel reserve.

9.5.4.3 Safety Evaluation

The diesel-driven fuel supply pump and injection pumps provide fuel for combustion from the day tank as soon as the diesel begins to crank.

The day tank contains a set of level switches to automatically operate the diesel-oil transfer pump and actuate high- and low-level alarms. In addition, a separate level switch is

provided to alarm on low-low level. Manual operation of the diesel-oil transfer pump is also possible.

Diesel fuel is supplied to each unit through a separate flow train from a common [redacted]

The flow trains can be manually connected in two places. The manual cross-connections ensure the availability of fuel for both diesel-generators following a single failure in one flow train. If no manual action can be taken, a single failure may shut down the corresponding diesel-generator, but in no case will a single failure in one train shut down both units

[redacted] each day tank will supply the corresponding diesel-generator with enough fuel for about 4 hr of operation. By temporarily cross-connecting the two tanks with a hose, one diesel-generator can be operated for about 8 hr from the fuel in both day tanks.

To ensure an open flow path to the diesel, all normally open valves in the system are locked open. Through selective manual valve operation, it is possible to use either diesel-oil transfer pump to fill both day tanks.

9.5.4.4 Inspection and Testing

Functional tests of this system are conducted as part of the testing of the diesel-generators described in Section 8.3.1.

A sample line connection to the storage tank facilitates sampling diesel fuel without dismantling equipment. Sampling as required per the Technical Specifications is done at least monthly.

9.5.5 DIESEL-GENERATOR COOLING WATER SYSTEM

A completely separate service water system is used to supply cooling water to each diesel-generator. Therefore, a single failure in this system will shut down only one unit. See Section 9.2.3.

9.5.6 DIESEL-GENERATOR AIR STARTING SYSTEM

As shown in figure 9.5-2, each diesel generator has two independent starting air supply systems. One consists of an electric motor driven air compressor which automatically recharges two air receivers and the other consists of a diesel driven air compressor which is manually operated to recharge a third air receiver. Therefore, a single starting system failure will not prevent either diesel generator from starting.

The air starting compressed-air tanks are designed and fabricated in accordance with Section VIII of the ASME B&PV Code, and are Code stamped. The air starting piping external to the diesel-generator skid is designed and fabricated in accordance with the Power Piping Code, ANSI B31.1.0.

The air tanks and piping are designed for 250 psig. Relief protection is provided in accordance with the above codes. The system operating pressure may fluctuate between 240 psig and 150 psig, which is sufficient to provide adequate starting capacity. A low air pressure alarm is provided for each starting system.

In the event of a single air tank rupture or other accidental loss of one air tank, the affected diesel generator may still be started from the other starting air supply system, provided that any missiles generated by the ruptured tank have not seriously damaged critical diesel-generator components. However, of primary importance is the fact that a postulated loss of one or both of the air tanks associated with one diesel generator will not prevent the other diesel generator from starting.

9.5.7 DIESEL-GENERATOR LUBRICATION SYSTEM

Each diesel engine lube oil system contains a continuously operating standby lube oil pump with heater, a prelube pump, an engine driven lube oil pump, strainers and filters. In the standby mode of operation, the standby lube oil pump operates to circulate oil from the engine sump through the heater, strainer and filters to maintain oil temperature at acceptable level to avoid wiping bearings because of improper lubrication at startup. The prelube pump is used prior to normal engine start to ensure the upper crankshaft bearings are properly lubricated. Each diesel engine also includes a lube oil booster system which injects oil by air pressure into the upper crankshaft bearings from a 2-gal accumulator upon a start signal for the engine. After engine start, the standby pump automatically stops and engine lubrication is accomplished by the engine driven pump.

Each diesel is equipped with its own lube oil makeup system. A single system failure would limit the lube oil for one diesel-generator to only what is in the crankcase; however, the other unit would still be capable of operating for 7 days without the manual addition of lube oil.

9.5.8 DIESEL-GENERATOR COMBUSTION AIR INTAKE AND EXHAUST SYSTEM

9.5.8.1 Description

Combustion and ventilating air is supplied by axial fans that provide 38,000 cfm to each diesel room. During diesel operation, the turbo-charged engines use room air at rates that vary depending on engine load, but that reach 11,000 cfm at full load for each engine. The diesel-generators run at full load in almost every instance of operation. The remaining 27,000 cfm of ventilating air is discharged through motor-operated discharge louvers.

[redacted]

9.5.8.2 Safety Evaluation

Each diesel-generator is equipped with its own intake and exhaust piping and components. A single system failure would shut down or reduce the performance of one unit only.

Each diesel-generator room is cooled by a separate ventilation system composed of dampers, a fan, and ductwork. A single system failure may shut down or reduce the performance of only one unit.

[redacted] See Section 9.4.6.2.2.

9.5.9 PLANT HEATING BOILER SYSTEM

9.5.9.1 Design Bases

9.5.9.1.1 Power Generation Objective

The plant heating boiler system provides steam for the plant hot water heating system and preoperational steam for the HPCI and RCIC and main turbines.

9.5.9.2 Power Generation Design Bases

1. The plant heating boiler system operates as a standby for the plant heating system when the plant is operating and the feedwater heater drains are used as the primary heat source.

2. The system is also used to provide heat whenever the plant is shut down during cold weather.

9.5.9.3 Description

The plant heating boiler system consists of a water-tube boiler, a combination deaerator-direct contact heat exchanger, boiler feedwater pumps, a fuel-oil storage tank, and associated piping, valves, combustion controls, and instrumentation (Figure 9.5-3). [redacted]

The boiler is designed in accordance with the ASME B&PV Code, Section I, and the laws and regulations of the State of Iowa.

The steam from the boiler is mixed with the hot water returns from the heating system in the combination deaerator-heat exchanger. The hot water is then returned to the hot water distribution system. Demineralized water is used for boiler makeup. An atmospheric blowdown tank is provided to control the water quality in the boilers.

The boiler control system is basically designed for unattended operation, except for cold startup operation of the boiler.

The boiler also provides steam for the following functions:

1. HPCI and RCIC turbine preoperational testing.
2. Initial sealing of the main turbine.
3. Steam blanketing for the turbine reheater tubing on shutdown.

4. Liquid nitrogen vaporization for nitrogen purge of the primary containment.

Removable spool pieces are provided for temporary connection of the plant heating steam to the HPCI and RCIC systems. These connections are shown on Figure 6.3-7, Sheet 1 (for HPCI system) and Figure 5.4-9, Sheets 1 and 2 (for RCIC system). These connections permit the use of clean (non-radioactive) steam for preoperational testing of the HPCI and RCIC turbines during initial startup and after extensive maintenance on these units. The HPCI spool piece is 20.5 in. long including flanges of 8-in. pipe; the RCIC spool piece is 9 ft-0 in. of 3-in. pipe with flanges. Blind flanges are provided to isolate the systems when the spool pieces are not in use.

There is no permanent connection from the plant heating boiler system to any safety-related equipment.

9.5.9.4 Inspection and Testing

Subsequent to each period of operation, the plant heating boiler system is placed in a standby condition and is not tested in such condition. Immediate startup of this system is not critical because the plant residual heat is sufficient for a 12-hr period. The boiler is maintained in a standby condition that allows for quick restoration to service. Therefore, only routine inspections for this type of equipment are required to return the heating boiler to operation. Plant procedures dictate what inspections and testing are required before the heating boiler is left in automatic operation. The system is inspected and tested in accordance with the requirements of the State of Iowa and ASME Boiler codes and requirements. When the boiler is in automatic operation it is inspected on a shiftly basis.

9.5.10 PRESSURE SUPPRESSION POOL WATER CLEANUP SYSTEM

9.5.10.1 Design Bases

The pressure suppression pool (torus) water cleanup system provides a means of draining the torus and of cleaning and storing the water for reuse in the torus.

9.5.10.2 Description

The torus water cleanup system utilizes the condensate demineralizer system for cleaning the torus water and the condensate storage tanks and/or the condenser hotwell for storing the cleaned water. The torus water cleanup system pump [redacted] is located in the torus room (Figure 1.2-11). The torus water cleanup system is connected to the torus drain line by a removable spool piece and to the condensate demineralizer influent line by another spool piece. (See Figure 9.5-4.) These spool pieces are removed and blind flanges are installed on the torus drain line connection and on the condensate demineralizer influent line connection during normal and hot shutdown modes of operation of the plant.

9.5.10.3 Safety Evaluation

The torus water cleanup system is only connected (spool pieces in place) and operable when the fuel is removed from the reactor or when the plant is down for a refueling outage, and conditions specified in the Technical Specifications are met. The torus water cleanup system is not operable when torus water is required or when the condensate demineralizer is required. The torus water cleanup system piping passes from the torus room through the HPCI room to the air ejector room in the turbine building. A seismic analysis of the piping that runs over Seismic Category I piping and equipment in these rooms has been performed and the line has been seismically supported.

REFERENCES FOR SECTION 9.5

2013-013 | 1. None.