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9.0 Auxiliary Systems

This chapter provides information concerning the auxiliary systems included in the station. Those systems which are essential for safe shutdown of the units or the protection of the health and safety of the public are identified. The description of the system, the design bases for each system and for critical components, a safety evaluation demonstrating how the system satisfies the design bases, and the testing and inspection performed to verify system capacity and reliability are provided.

The capability of the system to function without compromising the safe operation of the units under both normal operating and transient situations is shown by the information provided. Seismic design classifications are given with reference to detailed information provided in Chapter 3. Radiological considerations associated with the operation of each system under normal and accident conditions are summarized in this chapter and referenced to detailed information in Chapter 11 and Chapter 12.

Refer to Figure 6-1 and Figure 6-2 for symbols and system abbreviations for the flow diagrams in this chapter.

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9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

9.1.1.1 Design Bases

The design of the new fuel storage facility is based on the following criteria:

1. General Design Criterion 2 - Design for protection against natural phenomena.
2. General Design Criterion 3 - Fire Protection
3. General Design Criterion 4 - Environmental and missile design bases.
4. General Design Criterion 5 - Sharing of structures systems and components.
5. General Design Criterion 61 - Fuel storage, handling and radioactive control.
   a. Capability of periodic inspection.
   b. Shielding for radiation protection.
   c. Provisions for containment and confinement.
6. General Design Criterion 62 - Prevention of criticality in fuel storage and handling.
7. ANSI N18.2-1973, Section 5.7.4.1, which states:

   "The design of spent fuel storage racks and transfer equipment shall be such that the effective multiplication factor will not exceed 0.95 with new fuel of the highest anticipated enrichment in place assuming flooding with pure water. The design of normally dry new fuel storage racks shall be such that the effective multiplication factor will not exceed 0.98 with fuel of the highest anticipated enrichment in place assuming optimum moderation (e.g., aqueous foam envelope as the result of fire fighting). Credit may be taken for the inherent neutron absorbing effect of materials of construction or, if the requirements of Criterion 5.7.5.10 are met, for added nuclear poisons."
8. Regulatory Guide 1.29

Each unit at McGuire Nuclear Station has New Fuel Storage Racks located within a New Fuel Storage Vault shown in Figure 9-1 and Figure 9-2.

9.1.1.2 Facility Description

The new fuel storage racks are arranged to provide dry storage for 96 new fuel assemblies. The racks consist of vertical cells grouped in parallel rows, six rows wide and 16 cells long, which provide support for the fuel assemblies and maintain a minimum center-to-center distance of 21 inches between assemblies. The 96 new fuel storage locations are subdivided into four 3 x 8 arrays arranged in a 2 x 2 array and separated by concrete. See Figure 9-1. The new fuel storage racks for Unit 1 are an open cell design constructed of angle iron with support plates and the Unit 2 new fuel storage racks are a closed cell design with a stainless steel tube. The top of the new fuel storage racks have guides which provide for easy entry of the assemblies into the racks.

New fuel is received in the fuel receiving area and stored temporarily prior to being removed from the shipping container. When removed from the shipping container, the assemblies are handled one at a time by the auxiliary hoist of the fuel handling bridge crane. Upon removal from the shipping container the assembly is placed in a new fuel storage location for inspection and is transported by the auxiliary hoist to the New Fuel Storage Vault or the spent fuel pool for storage. Fuel is transferred from the vault to the spent fuel pool in a similar manner.

9.1.1.3 Safety Evaluation

9.1.1.3.1 Structural Evaluation

The New Fuel Storage Vault, located in the fuel handling area, is designed as a Category I structure. Refer to Section 3.8.4.2 for design and analysis procedures used in the design of the New Fuel Storage Vault. Loads and load combinations as defined in Section 3.8.4.2 include loads for the safe shutdown earthquake and tornado missiles. The methods, codes, and specifications used in the design of the New Fuel Storage Vault are also given in Section 3.8.4.2.

The new fuel storage racks are designed as a Category I structure to withstand the load conditions as defined in Table 3-42 and loading combinations as defined in Table 3-44.

9.1.1.3.2 Criticality Evaluation

Calculated values of K_{eff} for the storage arrays, including the effects of calculational and geometrical uncertainties, are less than those required by ANSI N18.2-1973, Section 5.7.4.1 when a full loading of the assemblies listed in assumption #5 below is considered. The computer codes and techniques utilized in the analysis have been validated against experimental data for water moderated UO_2 lattices with characteristics similar to the fuel analyzed.

The following assumptions are made in evaluating criticality safety:

1. Parameters are chosen to maximize K_{eff}.
2. No credit was taken for the inherent neutron-absorbing effect of the new fuel storage rack materials.
3. No burnable poisons, control rods, or supplemental neutron poisons are assumed to be present.
4. Effects of reflectors other than water are included if their neglect would have been nonconservative. This includes the storage vault's concrete walls, ceiling, and floor.
5. All assemblies are unirradiated Westinghouse or B&W 17x17 designs that have been or are currently being used in the McGuire reactors. Of the fuel designs analyzed, the W-STD, MkBW, and W-RFA are enriched to a nominal 5.0 wt % U-235. The W-OFA fuel assembly design is limited to a maximum nominal 4.76 wt % U-235. A reactivity uncertainty is included in the new fuel vault calculations to account for an as-built enrichment tolerance of ±0.05 wt % U-235.
6. The new fuel storage vault is modeled as 2 rooms which are separated by a 2-foot thick concrete wall. Each room contains 3 infinite rows of 12-foot high fuel assemblies. (See Figure 9-1 and Figure 9-2).
7. Each fuel assembly is treated as a heterogeneous system with the fuel pins, control rod guide tubes, and instrumentation thimble guide tube modeled explicitly.
8. Mechanical uncertainties and biases due to construction tolerances are considered by using worst-case conditions. Uncertainties considered include cell I.D. and center-to-center spacing.

The following accidents are considered in the criticality design of the new fuel storage area:

1. Flooding: complete immersion of the entire array in pure, unborated, full density water.
2. Envelopment of the entire array in a uniform density aqueous foam of optimum density (that density which maximizes the reactivity of the array), for example as a result of fire fighting.

Accidents resulting in an increase in K_{eff} because of geometrical changes of the racks or fuel handling accidents are not considered credible due to the following design bases:

1. The facility is designed in accordance with GDC 2 and 4.
2. The racks are designed to Seismic Category I requirements.
3. The only-Category I structure that could disrupt the array should it fail during a seismic event is the crane trolley. Administrative procedure prohibits the trolley from being parked over the New Fuel Storage Vault.
4. The runway conductors for the trolley are divided and power to each section is provided through separate circuit breakers. Power to the conductors in the area of the new fuel storage vault is provided only during handling operations. The conductors are divided at a point which will prohibit the trolley being positioned over the vault when power to that end is interrupted.
5. The racks and anchorages can withstand the maximum uplift force available without a significant change in geometry.
6. The design of the Fuel Handling System and administrative procedures insure subcritical spacing of fuel assemblies.

The introduction and retention of moderators into the facility is prevented by the following:

1. The facility is located above the probable Maximum Flood Level (See Section 9.1.1.3.7).
2. A drainage system sized to handle all probable means of inadvertent flooding exists.
3. Administrative policy has been established to preclude the use of hydrogenous fire fighting material from the vault. All extinguishers are of the dry chemical of CO_2 type.

Two gamma radiation monitors, which alarm both locally and in the control room area upon the detection of high radiation levels, are located in the vault in accordance with the criteria set forth in 10CFR 50.68(b)(6).

9.1.1.3.3 Fire Protection

There are normally no combustible materials present in the storage vault. A manual fire fighting system of appropriate capacity and capability is present should combustible material be inadvertently introduced. Fire and smoke detectors have been installed in the new fuel storage area pursuant to the requirements of Regulating Guide 1.120.

9.1.1.3.4 Ventilation

Ventilation for this facility is provided by the fuel handling area ventilation supply and the fuel handling area ventilation exhaust subsystems (Refer to Section 9.4.2.2).

9.1.1.3.5 Sharing of Systems

Each unit has its own completely independent new fuel storage facility.

9.1.1.3.6 Shielding for Radiation Protection

Only fresh uranium assemblies are stored in the vault, which provides more than sufficient radiation shielding.

9.1.1.3.7 Prevention of Flooding

The storage of the new fuel within the New Fuel Storage Vault precludes flooding of the new fuel assemblies by the probable maximum flood (PMF) and the probable maximum precipitation (PMP). The dike at McGuire Nuclear Station is at elevation 780+0, which protects the site area. The bottom of the new fuel assemblies are at approximately elevation 762+6. This elevation (762+6) is higher than the maximum elevation of either the PMF or the PMP. The PMF and PMP are described in Section 2.4.10.

There is no piping, the rupture of which could introduce moderator into the vault, routed through this area.

9.1.1.3.8 Tests and Inspections

The gamma radiation monitors are self testing. Should either unit become inoperable, indication of the failed state is provided in the Control Room.

No degradation of the storage racks is expected; thus no routine inspections are made. Any unusual deterioration would be noted during routine fuel handling.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases

The design bases of the spent fuel storage system are the following:

1. The prevention of criticality during storage.
2. The prevention of damage to the fuel.
3. Adequate radiation shielding.
4. Protection against radioactivity release.
5. Adequate monitoring of the fuel storage.
6. Ability to withstand design seismic loads.

Conformance with Regulatory Guide 1.13, “Fuel Storage Facility Design Basis” is as follows:

1. Regulatory Position 1

   The spent fuel storage as part of the Auxiliary Building is analyzed and designed as a Category I structure (see Table 3-1). For details of the loading conditions and loading combinations of the spent fuel pool, refer to Table 3-42 and Table 3-43.

2. Regulatory Position 2

   The spent fuel pool is approximately 47 feet four inches deep. Fuel is stored in the pool approximately 33 feet four inches below the fuel pool operating floor with approximately 25 feet of water above the top of the fuel.

   The spent fuel pool is housed in a concrete and steel superstructure. The concrete superstructure encloses the spent fuel pool except for the North end of the structure which is enclosed by a steel structure with siding. The concrete structure provides protection from turbine generator, tornado winds and tornado missiles. The North end of the spent fuel building does not provide tornado missile protection.

An evaluation has been performed to determine the maximum damage to spent fuel elements stored in the spent fuel storage pool in the event a tornado missile enters the pool through the North end of the building. The spent fuel storage pool is described in Section 3.8.4.1.1. The structural grid supporting the fuel racks is described in Section 9.1.2.2.

Figure 9-5 defines the critical missile trajectory entering the pool at the operating deck level, elevation 778 + 10, and maintaining an unaltered path through the pool. Damage to the supporting grid systems and fuel cells is not considered for missile energy dissipation. The most critical missile is considered to be the utility pole, 13.5 inches in diameter. To be conservative, it is assumed that the missile trajectory is unaltered until it contacts the pool floor or wall and that all fuel pins contained within the missile cross section are ruptured. With this assumption, a maximum of thirty-eight equivalent fuel cells in Region 2 could be ruptured. A dose analysis has been performed based upon this damage and is discussed in Section 15.10.

3. Regulatory Position 3

   See Section 9.1.4.2.2, under Fuel Handling Bridge Crane, and Section 9.1.4.3.1, for a discussion of Regulatory Position 3. Note that, “the spent fuel crane has identical safety features to those on the manipulation crane with the exception that the spent fuel crane does not have an RCC hoist system.”

4. Regulatory Position 4

   Refer to the answer to Regulatory Position No. 1 for a discussion of the type of building enclosing the fuel pool. For a discussion of the ventilation and filtration system, see Sections 9.4.2.2 (paragraphs 1 & 2), 9.4.2.3 (paragraph 2), and 5.4.5.

5. Regulatory Position 5

   a. Refer to the above response to Regulatory Position No. 3.

   b. Refer to the above response to Regulatory Position No. 3.

6. Regulatory Position 6

   See Section 9.1.3.2, paragraph 2, for a discussion of Regulatory Position No. 6.

7. Regulatory Position 7

   For a discussion of instrumentation which monitors fuel storage pool water level, see Section 9.1.3.2.6. See Sections 11.4.2.2 and 11.4.2.2.6 for a discussion of instrumentation which monitors radiation level in the spent fuel pool area.

8. Regulatory Position 8

   See Section 9.1.3.3 for a discussion of Regulatory Position No. 8. Consideration of the fuel pool leakage rate is discussed in Section 3.8.4.1.1 and Section 9.1.3.1.4 in the answer to Regulatory Position No. 8.

9.1.2.2 Facility Description

Each unit of the McGuire Station has an independent fuel storage system. The Fuel Handling System associated with the pool is discussed in Section 9.1.4, and Spent Fuel Cooling System is presented in Section 9.1.3. Radiation shielding and monitoring are presented in Sections 12.1 and 11.4, respectively. Originally the fuel pool contained sufficient fuel storage racks to accommodate the number of fuel assemblies discharged from four normal McGuire refueling cycles plus one complete McGuire core. Due to projected shortage of spent fuel storage in 1990's two region poison fuel storage racks have been installed at both McGuire Units 1 and 2. This modification increases the total storage capacity of each spent fuel pool from about 500 locations to 1463 locations. Provisions are also made to store control rods and burnable poison rods in empty locations in the fuel racks. In response to an industry wide concern over Boraflex degradation, the Boraflex Region 1 racks at Units 1 and 2 have since been replaced with Boral racks of equal size and capacity. Also due to the shortage of spent fuel storage, McGuire has installed an Independent Spent Fuel Storage Installation (ISFSI). The dimensions and location of the fuel pool are included on Figure 9-1 and Figure 9-2. Major components, piping, valves and instrumentation in contact with the fuel pool water are stainless steel.

9.1.2.2.1 Rack Design

The function of the spent fuel storage racks is to provide for safe storage of spent fuel assemblies in a flooded pool, while maintaining a coolable geometry, preventing criticality, and protecting the fuel assemblies from excess mechanical or thermal loadings.

A list of design criteria is given below:

1. The racks are designed to meet the nuclear requirements of the NRC guidance, “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Plants” contained in an NRC letter to all power reactor licensees from Laurence Kopp dated August 19, 1998. The effective multiplication factor, K_{eff}, in the spent fuel pool, including all uncertainties, is less than 1.00 under normal conditions with no credit for soluble boron, and less than or equal to 0.95 under normal conditions with credit for 800 ppm boron (2475 ppm boron under accident conditions).
2. The racks are designed to allow coolant flow such that boiling in the water channels between the fuel assemblies in the rack does not occur.
3. The racks are designed to Seismic Category I requirements, and are classified as ANS Safety Class 3 and ASME Code Class 3 Component Support structures. The structural evaluation and seismic analyses are performed using the specified loads and load combinations in Section 9.1.1.
4. The racks are designed to withstand loads which may result from fuel handling accidents and from the maximum uplift force of the fuel handling crane without violating the criticality acceptance criterion.
5. Each storage position in the racks is designed to support and guide the fuel assembly in a manner that will minimize the possibility of application of excessive lateral, axial and bending loads to fuel assemblies during fuel assembly handling and storage.
6. The racks are designed to preclude the insertion of a fuel assembly in other than design locations.
7. The materials used in construction of the racks are compatible with the storage pool environment and do not contaminate the fuel assemblies.

9.1.2.2.2 Specified Loads and Definitions

The following are load combinations specified for the Region 1 racks:

Elastic Analysis	Acceptance Limits[(1)]

(1) D + L	Normal limits of NF 3231.1a
(2) D + L + T_o	Normal limits of NF 3231.1a
(3) D + L + T_o + E	Normal limits of NF 3231.1a
(4) D + L + T_a + E	Normal limits of NF 3231.1a
(5) D + L + T_o + P_f	Normal limits of NF 3231.1a
(6) D + L + T_a + E'	Faulted Condition limits of NF 3231.1c
(7) D + L + T_o +F_d	The functional capability of the fuel racks should be demonstrated.

Note:

1. The Region 1 racks are freestanding; thus, there is minimal or no resistance against free thermal expansion of the rack. Moreover, thermal stresses are secondary, which strictly speaking, have no stipulated stress limit in Class 3 structures or components. Thermal loads applied to the rack are, therefore, not included in the stress combinations.

The following are load combinations specified for the Region 2 racks:

Elastic Analysis	Acceptance Limits
(1) D + L	Normal limits of NF 3231.1a
(2) D + L + E	Normal limits of NF 3231.1a
(3) D + L + T_o	Lesser of $2S_y$ or S_u stress range$^{(1)}$
(4) D + L + T_o + E	Lesser of $2S_y$ or S_u stress range$^{(1)}$
(5) D + L + T_a + E	Lesser of $2S_y$ or S_u stress range$^{(1)}$
(6) D + L + T_a + E'	Faulted Condition limits of NF 3231.1c$^{(2)}$
(7) D + L + F_d	The functional capability of the fuel racks should be demonstrated.
(8) D + L + P_f	Normal limits of NF 3231.1a
(9) D + L + P_f + T_o	Lesser of $2S_y$ or S_u stress range$^{(1)}$

Note:

1. Application of this acceptance limit for the combination of primary and thermal stresses will typically limit the stress to S_y. However, when proper justification is provided to show that the thermal stresses are self-limiting, the combined stresses may exceed S_y provided the lesser of $2S_y$ or S_u stress range limit is met.
2. For the faulted load combination, thermal loads will be neglected when they are secondary and self-limiting in nature and the material is ductile.

Definitions:

D	- Dead loads or their related internal moments and forces including any permanent equipment and hydrostatic loads.

L	-	Live loads or their related internal moments and forces including any movable equipment loads.
T_o	-	Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.
T_a	-	Thermal effects and loads resulting from the highest temperature associated with the postulated abnormal design condition.
E	-	Loads generated by the operating basis earthquake (OBE).
E'	-	Loads generated by the safe shutdown earthquake (SSE).
P_f	-	Upward force on racks caused by postulated stuck fuel assembly.
F_d	-	Force caused by accidental drop of the heaviest load from the maximum possible height.

Analyses were performed to verify the acceptability of the critical load components and paths under the load combinations given above.

9.1.2.2.3 Applicable Codes and Standards

The two region poison rack design meets the “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Power Plants” dated August 19, 1998, and 10CFR50.68(b).

The design of the racks complies with the following:

1. NRC Regulatory Guides

R.G. 1.13 Spent Fuel Storage Facility Design Basis (Proposed Revision 2)

R.G. 1.29 Seismic Design Classifications

R.G. 1.60 Design Response Spectra for Seismic Design of Nuclear Power Plants

R.G. 1.61 Damping Values for Seismic Design of Nuclear Power Plants

R.G. 1.92 Combining Modal Responses and Spatial Components in Seismic Response Analysis

R.G. 1.124 Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports

2. NRC Standard Review Plans

SRP 3.7 Seismic Design

SRP 9.1.2 Spent Fuel Storage

SRP 9.1.3 Spent Fuel Pool Cooling and Cleanup System

3. Industry Codes and Standards

   American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Division 1.

   American National Standards Institute, ANSI/ANS-57.2-1983, “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants.”

American National Standards Institute, N16.1-1975, “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.”

American National Standards Institute, N16.9 - 1975, “American National Standard for Validation of Calculational Methods for Criticality Safety.”

American National Standards Institute, N18.2 - 1973, “Nuclear Safety Criteria for the Design of Stationary PWR Plants.”

The spent fuel pool storage rack arrangement is shown in Figure 9-6 for Unit 1 and Figure 9-7 for Unit 2.

Each pool contains a two-region rack design. Region 1 (286 locations) has high density fuel assembly spacing (10.4 inches) obtained by utilizing a neutron absorbing material (Boral) and is reserved for temporary core off loading and storage of non-irradiated fully enriched fuel. Region 2 (1177 locations) has high density fuel assembly spacing (9.125 inches) and provides normal long term storage for irradiated fuel assemblies. The high-density Region 2 storage cells also contain a neutron absorber (Boraflex). However, the Boraflex in these cells has experienced significant degradation, and is no longer credited in the Region 2 criticality analysis.

There are no restrictions on placement of fuel assemblies within the Region 1 storage cells. For Region 2 storage, fuel assemblies must meet certain minimum burnup requirements – based on fuel assembly design, initial enrichment, and post-irradiation cooling time – in order to be stored without restriction in these racks. Fuel assemblies that have not achieved the minimum “unrestricted” burnup may still qualify for storage in “restricted” or “checkerboard” storage configurations, which have progressively lower minimum burnup requirements. Fuel assemblies failing to meet the burnup requirements for any of these storage configurations are not qualified for storage in Region 2, and must instead be stored in the Region 1 racks. The requirements for these different storage configurations in the spent fuel pool are defined in Section 3.7.15 of the McGuire Nuclear Station Technical Specifications.

The Region 1 storage racks are composed of individual storage cells made of stainless steel. These racks utilize a neutron absorbing material, Boral, which is attached to each cell. The cells within a module are interconnected at six locations along the length of the cell using spacer plates to form an integral structure as shown in Figure 9-9. Each rack module is provided with 4 pedestal supports which contact the spent fuel pool floor and are remotely adjustable from above through the cells at installation. The modules are freestanding and are not anchored to the floor nor braced to the pool walls.

The pedestal supports transmit the loads to the pool floor and provide a sliding contact. The pedestal screw permits the leveling adjustment of the rack.

The major components of the cell module are the fuel assembly cell, the Boral (neutron absorbing) material, and the wrapper. Spot welding is used to attach the wrapper to the outside of the cell. The wrapper covers the Boral material and also provides for venting of the Boral to the pool environment. Depending on the criticality requirements, some cells have a Boral wrapper on all four sides, some on three sides, and some on two sides.

The Region 2 storage racks consist of stainless steel cells assembled in a checkerboard pattern, producing a honeycomb type structure. The Region 2 cells are similar in design to the Region 1 cells; i.e., the major components are the cell, the neutron absorbing material, and the wrapper. The one key difference is the type of neutron absorbing material; the Region 2 cells contain Boraflex instead of Boral, although the Boraflex that remains in the Region 2 cells is no longer credited in the fuel storage criticality analyses. The cells are welded to a base support assembly and to one another (Figure 9-10) to form an integral structure without use of grids as used in Region 1 racks. This design is also provided with leveling pads which contact the spent fuel pool floor and are remotely adjustable from above through the cells at installation.

9.1.2.3 Safety Evaluation

The spent fuel pool modules have been evaluated using the techniques of seismic and stress analysis under various conditions of full, partially filled and empty fuel assembly loadings. The spent fuel pool racks have been evaluated for both OBE and SSE Seismic events and meet Seismic Category I requirements. The racks are designed to preclude insertion of fuel assemblies in other than permitted locations, thereby assuring the necessary spacing between assemblies. To further assure subcritical arrays in the fuel handling facilities, only one assembly can be manipulated at a time. It is permissible, however, to have one assembly within a test fixture while manipulating a second assembly as long as sufficient distance exists between the two assemblies (>10 inches).

Since each unit has its own fuel pool, there are no safety considerations related to sharing of components.

The fuel pool is designed to withstand the following:

1. normal dead and equipment loads plus design seismic loads,
2. all normal dead, equipment and live loads,
3. normal dead and equipment loads plus tornado wind load,
4. thermal stresses, and
5. cask drop accident.

Additionally, Section 9.1.3.3 presents a safety evaluation of the Spent Fuel Cooling System explaining in detail the provisions for continuous spent fuel cooling as required by Regulatory Guide 1.13. These provisions include:

1. redundant active components,
2. capacity to cool the maximum amount of spent fuel,
3. slow heatup rate which provides time to perform maintenance in the case of multiple component failures,
4. makeup water from the Nuclear Service Water System, and
5. system connections such that the pool cannot be inadvertently drained below the level required for shielding.

An alternative methodology for the safety evaluation of the spent fuel pool cooling system (described in Section 9.1.3.3) has been performed specifically for the new Region 1 Boral storage racks, Reference 9.1.5-12 and –13. This methodology has been previously approved by the NRC and the results are bounded by those presented in Section 9.1.3.3. Therefore, both methodologies (Section 9.1.3.3 and Reference 9.1.5-12 and –13) are acceptable for the Spent Fuel Cooling system safety evaluation. Section 9.1.3.3 and its supporting analyses continue to describe the licensing basis of the Spent Fuel Cooling System.

Section 11.4 discusses the continuous monitoring provided to detect airborne activity.

The spent fuel crane has two trip-off switches. The first switch trips at approximately 2300 lbs. In the event the first trip should fail to operate, the second switch trips at approximately 3200 lbs.

The fuel storage racks are designed to preclude structural damage to the racks due to uplift forces.

The highest level above the fuel storage racks from which a fuel assembly could be dropped during normal operation is three feet six inches through water. The Region 2 racks are designed to withstand an impact load from a dropped fuel assembly, assuming a six feet drop. The drop height assumed for the design analysis of the Region 1 racks is four feet.

The heaviest objects moved over the racks are the two spent fuel pool weir gates. Analysis of the dropping of either one of the two weir gates is presented in Section 15.7.4.3.

The fuel pool and fuel racks are located in the Auxiliary Building which is a Category I structure designed to protect the fuel pool and fuel racks from missile hazards as discussed in Section 3.5. Physical obstructions are provided to preclude insertion of fuel assemblies in other than permitted locations. Fuel handling devices are designed with a minimum safety factor of five based upon the ultimate stress of the material. This safety factor is consistent with overhead crane and wire rope industrial standards.

Details of the seismic and testing are presented in Sections 3.7 and 3.8.

9.1.2.3.1 Criticality Analysis

The design methodology which ensures the criticality safety of the fuel assemblies in the spent fuel storage rack is discussed in Section 4.3.2.6 and in Reference 17.

9.1.2.3.1.1 Neutron Multiplication Factor

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies, inserting neutron poisons between assemblies and introducing soluble boron in the spent fuel pool water.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than 0.95 with 800 ppm boron for most conditions (2475 ppm Boron for accident conditions) as recommended in the NRC guidance. “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Power Plants” contained in an NRC letter to all power reactor licensees from Laurence Kopp dated August 19, 1998. The guidance also states that there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than 1.0 with no credit for soluble boron. The acceptance criteria for criticality is further discussed in Section 9.1.2.3.1.4.

9.1.2.3.1.2 Normal Storage

Under normal storage conditions, the following assumptions were used in the criticality analysis.

1. For the criticality analysis of the Region 1 spent fuel pool racks, fresh fuel of the maximum enrichment (5.0 wt % U-235) is conservatively assumed. For the Region 2 storage rack evaluation, credit is taken for decreases in reactivity associated with fuel assembly burnup and post-irradiation cooling time.
2. Seven different “types” of fuel assemblies stored in the McGuire spent fuel pools, based on common design features, are considered: MkBI (old 15x15 Oconee transhipped fuel stored); W-STD (original standard 17x17 Westinghouse design); W-OFA (17x17 Westinghouse

"Optimized Fuel Assembly"); MkBW (standard 17x17 Framatome {B&W} assembly); MkBWb1 (MkBW design with solid, 6-inch, 2.0 wt % U-235 axial blankets); MkBWb2 (MkBW design with solid, 6-inch, 2.6 wt % U-235 axial blankets); and W-RFA (advanced 17x17 Westinghouse design with annular 6-inch, 2.6 wt % U-235 axial blankets). For the Region 2 criticality analysis, minimum burnup requirements for fuel storage are determined for each of these seven different fuel designs.

3. The storage cell nominal geometry is shown in Figure 9-11 for Region 1 and Figure 9-12 for Region 2.
4. The moderator is pure water at the temperature within the design limits of the pool which yields the largest reactivity. Credit is taken for soluble boron under normal and postulated accident conditions and during fuel movement. For accident conditions the double contingency principle of ANSI N16.1-1975 is applied. This principle states that it shall require at least two unlikely, independent, and concurrent events to produce a criticality accident.
5. McGuire Region 1 calculations are performed in 2-D, with perfect axial reflection. This is acceptable, because only fresh fuel is considered in the criticality evaluation for the Region 1 racks. It is also conservative, because it ignores axial leakage. All McGuire Region 2 criticality calculations are performed in three dimensions, with 24 axial fuel segments analyzed. The 3-D model includes top and bottom axial reflectors containing a mix of water, steel, and Zircaloy. Extensive historic and projected 3-D burnup, temperature, boron, and burnable poison data are employed to appropriately quantify the isotopic content of the fuel assembly designs considered for the Region 2 analysis.
6. Mechanical uncertainties and biases due to mechanical tolerances during construction are treated by either using "worst case" conditions or by performing sensitivity studies and obtaining appropriate values. The items included in the analysis are:
   a. Deleted Per 2006 Update
   b. Fuel pellet outside diameter
   c. Can and wrapper plenum thickness
   d. Can wall thickness
   e. Can ID
   f. Center-to-center spacing
   g. Fuel assembly position in can
   h. Fuel enrichment
   i. Fuel pellet dishing volume
   j. Fuel pellet density
   k. Fuel cladding outer diameter
   l. Boral width

   Other applicable uncertainties and biases are discussed in Section 4.3.2.6.
7. In the McGuire SFP Region 2 criticality analysis, credit is taken for the spacer grids in each fuel assembly design considered. A standard grid model that homogenizes the grid material into the coolant surrounding the fuel assembly is used to account for the effects of the grids.

8. No credit is taken for fuel assembly control components which can be removed (e.g. burnable poisons and control rods).
9. Credit is taken for the inherent neutron absorbing effect of some of the rack structure materials and in solid materials added specifically for neutron absorption (Boral in the Region 1 Racks) in accordance with Section 6.4.2.2.8 of ANSI/ANS-57.2-1983.
10. A bias is included in the Region 1 rack reactivity calculation to account for B_4C particle self shielding.

9.1.2.3.1.3 Postulated Accidents

Most accident conditions will not result in an increase in k_{eff} of the rack. For an assembly dropped on top of the storage rack, the rack structure pertinent for criticality is not excessively deformed and the dropped assembly has more than eight inches of water separating it from the active fuel height of stored assemblies which precludes interaction. Although the dropped assembly is more reactive outside rather than inside the poisoned storage cell, the assembly is no more reactive dropped on top of the storage rack than located anywhere else in the pool outside the storage rack.

However, accidents can be postulated which would increase reactivity. For accident conditions, two techniques are employed to ensure that sufficient criticality margin exists, the double contingency principle and increasing the soluble boron requirement to 2475 ppm. The acceptance criteria for criticality is further discussed in Section 9.1.2.3.1.4.

The double contingency principle of the NRC guidance, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Power Plants" contained in an NRC letter to all power reactor licensees from Laurence Kopp dated August 19, 1998 states that it is not required to assume two unlikely, independent concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool up to 2475 ppm can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

The "optimum moderation" accident is not a problem in the spent fuel storage racks because possible water densities are too low ($\leq$0.01 gm/cm^3) to yield k_{eff} values higher than for full density water and the rack design prevents the preferential reduction of water density between the cells of a rack (e.g. boiling between cells).

9.1.2.3.1.4 Acceptance Criteria for Criticality

The neutron multiplication factor in the spent fuel pools shall be less than or equal to 0.95 with 800 ppm Boron (2475 ppm under accident conditions) and less than 1.0 with no credit for soluble boron, as specified in 10CFR50.68(b) and the NRC guidance, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Power Plants" contained in an NRC letter to all power reactor licensees from Laurence Kopp dated August 19, 1998.

9.1.2.3.1.5 Cask Drop Accident

Cask drop accidents are not analyzed for criticality consequences since an evaluation has been performed to determine that a cask dropped in the fuel storage areas at McGuire will not enter the spent fuel pool. This analysis is discussed in Section 9.1.2.3.2.

9.1.2.3.2 Cask Drop Evaluation

An evaluation has been performed to assess the possibility of a spent fuel transfer cask entering the spent fuel pool. This evaluation included analysis of the following six casks which are currently regarded as potential candidates for spent fuel transfer intrastation or between McGuire Nuclear Station and Oconee Nuclear Station or transfer to the McGuire Independent Spent Fuel Storage Installation (ISFSI):

1. Model No. NFS-4, Nuclear Fuel Services, Inc.,
2. Model No. NLI-1/2, National Lead Company,
3. Model No. TN-8/TN-8L, Transnuclear, Inc.,
4. Model No. TN-32A, Transnuclear, Inc. (ISFSI Storage Cask).
5. Model No. NAC-UMS Universal Storage System, NAC International, Inc. (ISFSI Storage Cask)
6. Magnastor Storage System (ISFSI)

No special equipment is required to handle the Oconee spent fuel cask since cask hoisting equipment at McGuire is similar to Oconee.

The area in which the casks are handled is designed for a 30 foot drop of a proposed 125 ton cask and the structure is reinforced concrete with a rock foundation. The cask handling crane stops are located in a position to prevent the cask from being moved into the fuel pool area. The cask area is separated from the spent fuel pool by a three foot reinforced concrete wall (Figure 9-1 and Figure 9-2). Local damage to the concrete will be negligible and no safety related equipment is located in the cask travel path.

A description of the analysis and the results of the analysis for the NFS-4 truck cask is given below and in Figure 9-3. The analyses of the NLI-1/2, TN-8/TN-8L, NAC-UMS, Magnastor and TN-32A casks are similar to that of the NFS-4 cask and all show that the respective casks will not enter the spent fuel pool. In order to provide assurance that a cask will not fall into the spent fuel pool, the path of the cask will be controlled by application of administrative control as shown in Figure 9-4. By requiring the cask to follow either of these paths, any potential for the cask falling into the spent fuel pool is negated. The acceptable transfer paths have been incorporated as Selected Licensee Commitment 16.9.20.

Case #1- The cask handling crane is assumed to be traveling along the path(s) defined in McGuire Selected Licensee Commitment Figure 16.9.20-1 at its maximum speed of 50 fpm and hits the crane stops nearest the spent fuel pool. The crane stops, and the cask is assumed to continue traveling toward the spent fuel pool rotating about a line through the centerline of the crane drum. The cask continues to swing until the kinetic energy is completely converted to potential energy (i.e. the cask raises up as it rotates about the crane drum). At the instant the cask swings as close to the spent fuel pool as possible, the cable breaks and the cask falls. The conclusion is that the cask falls on the NE or NW edge of the cask pit and spent fuel pool walls and falls away from the spent fuel due to its center of gravity being within the cask pit envelope. See Figure 9.3 (1 of 3) for an illustration of this case.

Case #2- The cask handling crane is assumed to be traveling along the path(s) defined in McGuire Selected Licensee Commitment Figure 16.9.20-1 at its maximum speed of 50 fpm and hits the crane stops nearest the spent fuel pool. The crane stops and the cable breaks at the same instant, therefore the cask does not swing but is moving at 50 fpm. The cask is assumed to be at its highest position. The distance the cask travels at 50 fpm in the time it takes for the cask to fall, is determined. The cask will hit the NE and NW edge on the wall(s). It is determined that the amount of kinetic energy is not sufficient to cause the cask to fall into the spent fuel pool. See Figure 9-3(2 of 3) for an illustration of this case.

Case #3- The cask is assumed to be over the corner edge of the north and east or west cask pit walls and the cable breaks (Other locations will result in the cask overturning and falling away or perpendicular to the spent fuel pool wall). Energy losses at impact with the cask pit and spent fuel pool wall are conservatively considered and the results of the analysis show that the remaining energy is not sufficient to cause the cask to fall into the spent fuel pool. See Figure 9-3 (3 of 3) for an illustration of this case.

At such time that another cask is contemplated for use, a similar analysis will be performed prior to using the cask to assure that it will not enter the spent fuel pool.

9.1.2.4 Storage of Oconee Spent Fuel

The interim spent fuel storage plans for Duke Energy nuclear facilities call for storage of Oconee spent fuel assemblies in the McGuire Spent Fuel Pools. A detailed description of Oconee fuel assemblies is given in Final Safety Analysis Report, Oconee Units 1, 2, and 3. Oconee fuel storage will proceed within the system design bases listed in Section 9.1.2.1. The safety evaluation presented in Section 9.1.2.3 for the two region poison racks also applies to the storage of Oconee spent fuel assemblies.

9.1.2.5 Degassing Bent Irradiated Fuel Pin In Unit 2 Spent Fuel Pool

The licensing basis for the Unit 2 Spent Fuel Pool permits a one time evolution to degas and straighten a bent irradiated fuel pin, which has been stored in the spent fuel pool since 1993. The evolution involves the intentional puncture of an irradiated fuel rod in order to transfer the fuel rod gap gasses to a collection chamber to be released later in a controlled manner in accordance with approved procedures. Following the gas collection, the bent fuel rod is then straightened and stored in a broken rod capsule. Straightening of the bent fuel rod is necessary for eventual offsite transfer and disposal.

A safety evaluation determined that offsite dose consequences are insignificant for either an uncontrolled release from the spent fuel pool due to a ruptured fuel rod or as a controlled release from the waste gas system. The highest potential dose occurs to a worker in the fuel building, with whole body doses of less than 3 mRem and a thyroid dose of less than $3E^{-11}$ mRem. The NRC reviewed the licensing submittal dated January 31, 2003, as supplemented by letter dated May 1, 2003, and the pertinent Safety Evaluation Report was approved August 4, 2003.

9.1.3 Spent Fuel Cooling and Purification

The Spent Fuel Pool Cooling System (KF) is designed to remove heat from the spent fuel pool and maintain the purity and optical clarity of the pool water during fuel handling operations. The purification loop provides an alternate means for removing impurities from either the refueling cavity/transfer canal water during refueling or the refueling water storage tank water following refueling.

As part of the FLEX mitigation strategy in response to NRC Order EA-12-049, the ability to provide makeup from the UHS to the Spent Fuel Pool is required following a postulated beyond design basis event. A spent fuel pool cooling mechanical process connection is provided for this capability on the 767' Elevation of the Auxiliary Building.

9.1.3.1 Design Bases

KF System design parameters are given in Table 9-1.

9.1.3.1.1 Spent Fuel Pool Cooling

The existing Spent Fuel Cooling System is designed to maintain the spent fuel pool water temperature within acceptable limits under normal and maximum heat load conditions (Table 9-5).

Deleted Per 2008 Update.

Various operational evolutions may utilize decay heat predictions based on the ORIGEN methodology (i.e. ORIGEN-ARP or SAS2H/ORIGEN-S) presented in References 15 and 16.

For the normal heat load case, the spent fuel cooling system is designed for a single failure for an indefinite period of time. Table 9-5 shows that temperatures remain within acceptable limits established by Reference 18.

For the maximum heat load case, the spent fuel pool cooling system is not designed for a single failure for an indefinite period of time. For single failure conditions, Table 9-5 shows that temperatures remain below boiling values established by Reference 18.

Prior to a Full Core discharge, the spent fuel heat load is determined by calculation and offload requirements are procedurally established to assure that the decay heat load in the pool is less than the maximum allowable decay heat load as shown in Table 9-5. These offload requirements include meeting the constraints established in SLC 16.9.17.

McGuire has a design requirement for the spent fuel pool to be qualified as an assured source for the Standby Shutdown Facility. The design basis of the Standby Shutdown Facility includes a total loss of spent fuel pool cooling for a period of 72 hours. Analysis of this scenario has shown that all spent pool structures, systems and components are qualified for the 72 hour period. This provides adequate time for recovery of spent fuel pool cooling. This Standby Shutdown Facility scenario bounds the loss of one train of spent fuel pool cooling under maximum heat loads for 72 hours.

In summary, the spent fuel pool is not indefinitely qualified for loss of a single train under maximum heat loads. Spent fuel pool structures, systems and components have been qualified for a period of 72 hours given a complete failure of the pool cooling system. This provides adequate time for restoration of pool cooling should a train be lost during maximum heat load conditions.

9.1.3.1.2 Water Purification

The system demineralizer and filters are designed to maintain adequate purification to permit unrestricted access to the spent fuel storage area for plant personnel, provide means for purifying transfer canal and refueling pool water during refueling, and provide purification capability for the refueling water storage tank. The KF System also maintains the optical clarity of the spent fuel pool water surface by use of the skimmer trough, strainers, and skimmer filters.

9.1.3.1.3 Spent Fuel Pool Dewatering Protection

System piping is arranged so that failure of any pipeline cannot drain the spent fuel pool below the water level required for radiation shielding. A water level of ten feet or more above the top of the stored spent fuel assemblies is maintained to limit direct gamma dose.

9.1.3.1.4 Spent Fuel Pool Makeup

In order to provide specified shielding and water volumes in the fuel pool during plant operation, system piping provides makeup capabilities. Borated makeup water can be supplied to the spent fuel pool from the refueling water storage tank. Demineralized water can be supplied to the pool by the Reactor Makeup Water Pumps, and emergency makeup water can be supplied to the pool from the Nuclear Service Water System. All means of makeup are manually initiated and manually terminated.

9.1.3.2 System Description

An identical Spent Fuel Cooling System, as shown in Figure 9-13, is provided for each unit. The system consists of two cooling loops, one purification loop, and one skimmer loop.

The fuel pool cooling pumps take suction from the spent fuel pool. These pumps circulate the water through the cooling loops and the purification loop in various combinations prior to returning the water to the spent fuel pool. The spent fuel pool heat load is transferred to the Component Cooling System by the fuel pool cooling heat exchangers. The fuel pool cooling pre-filter, demineralizer, and post-filter will adequately remove corrosion and fission products from the spent fuel pool water during fuel handling operations.

The fuel pool skimmer pump takes suction from the skimmer trough, that collects water from the spent fuel pool surface. Floating debris is removed by the fuel pool skimmer strainer and filter. Optically clear water is then discharged below the pool surface at various locations. Discharge throttling valves are provided for optimizing the spent fuel pool skimmer loop operation.

The pool cooling and purification system is manually controlled from a local control panel. High temperature and low liquid level in both the fuel pool and the refueling canal, plus high radiation in the fuel pool area alarms are provided in the Control Room as per Regulatory Guide 1.13. Also alarmed in the Control Room are high liquid level in both the fuel pool and the refueling canal. Local gages are provided for high differential pressure across each strainer and filter.

9.1.3.2.1 Pool Cooling Subsystem

The cooling subsystem of the Spent Fuel Pool Cooling System is a closed loop system consisting of two full-capacity pumps and two full-capacity heat exchangers. Each pump-heat exchanger loop is designed to accommodate the decay heat from the one-third core case.

The fuel pool contains water with approximately 2000 to 4000 ppm of boron and has the capacity to store 1463 spent fuel assemblies including one and one-third cores of freshly discharged fuel. A capacity of one and one-third cores enables handling the removal of one full core during that period of time when one-third core is stored in the fuel pool following a refueling. When in storage racks the fuel elements are spatially distributed so as to preclude criticality in the event of an accidental dilution of the fuel pool water. The fuel pool water is maintained at refueling water storage tank concentration to assure that mixing of the fuel pool water and the refueling canal water cannot dilute the refueling canal water concentration.

The pool cooling subsystem removes the decay heat from the spent fuel stored in the fuel pool. The System is designed to limit the fuel pool temperatures as given in Section 9.1.3.1.1.

9.1.3.2.1.1 Component Description

Component design parameters are given in Table 9-1.

<u>Fuel Pool Cooling Pumps</u>

The two fuel pool cooling pumps are designed for 2200 to 2900 gpm per pump and installed for parallel operation. Under normal operating conditions one pump is operating. The pump has mechanical seals and leakoff, vent and drain connections. The internal wetted surfaces of the pumps are stainless steel. The units are constructed in accordance with Section III, Class 3 of the ASME Code.

Fuel Pool Cooling Heat Exchangers

The two fuel pool cooling heat exchangers are shell and tube design. The shell side component cooling water flow is 2500 gpm per heat exchanger, which is the flowrate required to remove the design heat load. Both the shell and tube side of the unit are constructed in accordance with Section VIII, Div. 1, Class C of the ASME Code. The tube side is stainless steel, and the shell side is carbon steel.

Fuel Pool Cooling Pump Strainers

Submerged debris and trash that might be harmful to the fuel pool cooling pump is collected by this strainer.

9.1.3.2.2 Pool Purification Subsystem

The purification subsystem removes particulates, dissolved fission products and surface dust from the fuel pool and the canal, thereby maintaining clarity of the fuel pool water and the refueling canal water and permitting visual observation of underwater operations. The purification subsystem can also remove dissolved fission products from the refueling water storage tank.

One spent fuel pool water volume can be circulated through the purification loop every 24 hours, which should be sufficient to maintain the water chemistry specified for the spent fuel pool in WCAP-7452, Rev. 1, “Chemistry Criteria and Specifications for Westinghouse Pressurized Water Reactors”, dated July 30, 1973.

The fuel pool water or refueling water is circulated by its respective pump through a fuel pool cooling pre-filter, which removes particulates, and then through the fuel pool cooling demineralizer which removes ionic material. The purified water then goes through the fuel pool cooling post filter before it is returned to the fuel pool or the refueling water storage tank.

The pool purification subsystem also consists of a spent fuel pool skimmer loop which removes floating debris from the spent fuel pool surface by use of an adjustable skimmer trough, strainer, skimmer pump, and filter. The suction and return lines of the skimmer loop are arranged so that the maximum area of surface water is circulated through the skimmer loop.

Deleted paragraph(s) per 2002 revision.

9.1.3.2.2.1 Component Description

Component design parameters are given in Table 9-1.

Fuel Pool Cooling Pre-Filter

Suspended particles are collected on the pre-filter instead of on the fuel pool cooling demineralizer. The disposable filter cartridge is an absolute filter that can be varied in filtration porosity, depending on cartridge size, up to a 70 micron size.

Fuel Pool Cooling Demineralizer

The demineralizer is of the mixed bed type with H+ and OH- type resin which removes corrosion and fission product ionic contaminants from the spent fuel pool water or the Refueling Water System water.

Fuel Pool Cooling Post-Filter

Resin fines are collected on the post-filter. The disposable filter cartridge is an absolute filter that can be varied in filtration porosity, depending on pool clarity conditions, up to a 20 micron size.

Skimmer Trough

The skimmer collects water from the spent fuel pool surface. Surface skimming is optimized by skimmer trough adjustment, or by raising the pool surface by filling.

Fuel Pool Skimmer Strainer

Floating debris and trash that might be harmful to the fuel pool skimmer pump is collected by this strainer.

Fuel Pool Skimmer Pump

This pump is sized to circulate 150 gpm through the skimmer loop and return the water to the spent fuel pool at four discharge points. Throttling valve (1KF100) can be used along with the four discharge valves to optimize skimmer loop operation.

Fuel Pool Skimmer Filter

Suspended particles are collected on this filter. The disposable filter cartridge is an absolute filter that can be varied in filtration porosity, depending on cartridge size, up to a 40 micron size.

9.1.3.2.3 Piping

All piping used in the Pool Cooling and Purification System is stainless steel with welded connections throughout, except for flanged connections to support routine maintenance.

9.1.3.2.4 Electrical Power Supply

Each fuel pool cooling pump is supplied power by its corresponding emergency diesel generator within one hour after LOCA or blackout but must be restarted manually. These pumps are supplied from a reliable station bus during all normal operating periods.

The fuel pool skimmer pump is supplied power from the normal station bus. This pump is not required to operate during an emergency.

9.1.3.2.5 Water Chemistry

The boron concentration of the Spent Fuel Pool will be verified at a frequency which meets or exceeds the requirements of Station Technical Specifications. Minimum boron concentration is set by the Core Operating Limits Report (COLR) for each unit as required by Technical Specifications; Maximum boron concentration is 4000 ppm boron.

Additional water chemistry limits and sample frequencies are listed in the Station Chemistry Manual.

9.1.3.2.6 System Instrumentation and Control

9.1.3.2.6.1 Spent Fuel Pool Cooling Loop Instrumentation

Temperature

Spent Fuel Pool Temperature - This instrument provides Control Room indication of the water temperature in the Spent Fuel Pool. High water temperature is alarmed to the computer.

Fuel Pool Cooling Heat Exchanger Inlet and Outlet Temperature - These instrument test points are used to check the temperature at each heat exchanger inlet and outlet.

Level

Spent Fuel Pool Level - This instrument provides control room indication of water level in the spent fuel pool. High and low pool water levels are alarmed to the computer.

Two wide range Spent Fuel Pool level instrument channels were provided in response to NRC Order EA-12-051, “ORDER MODIFYING LICENSES WITH REGARD TO RELIABLE SPENT FUEL POOL INSTRUMENTATION.” These instrument channels were required for mitigation of Beyond Design Bases External Events (BDBEE). The instrument channels were designed to be electrically independent and spatially separated to provide reasonable assurance they were not both vulnerable to failure due to a postulated BDBEE.

Pressure

Fuel Pool Cooling Pump Discharge Pressure - These instruments provide local indication of each pump's discharge pressure.

Fuel Pool Cooling Heat Exchanger Inlet and Outlet Pressure - These instrument test points are used to check the pressure at each heat exchanger inlet and outlet to monitor tube cleanness.

Flow

Fuel Pool Cooling Heat Exchanger Flow - These instruments provide local flow indication. High or low flow through the tubeside of each heat exchanger is alarmed through the computer.

9.1.3.2.6.2 Spent Fuel Pool Purification Loop Instrumentation

Pressure

Fuel Pool Cooling Pre-filter and Post-filter ΔP - These instruments provide local indication of the differential pressure across each filter. Filters are changed on either high pressure drop, or high radiation as measured by a probe inserted into each filter cubicle.

Flow

Fuel Pool cooling demineralizer Flow - This instrument provides local flow indication. Low flow is alarmed through the computer.

9.1.3.2.6.3 Spent Fuel Pool Skimmer Loop Instrumentation

Pressure

Fuel Pool Skimmer Filter ΔP - This instrument provides local indication of the differential pressure across the filter.

Fuel Pool Skimmer Pump Discharge Pressure - This instrument provides local indication of the pump discharge pressure.

Flow

Fuel Pool Skimmer Loop Flow - This instrument provides local flow indication. Low flow is annunciated.

9.1.3.3 Safety Evaluation

9.1.3.3.1 Availability and Reliability

The KF System is located in a Seismic Category I structure. The redundant active components of the cooling portion of the system are located above the design basis flood level in the Auxiliary Building. The KF System heat removal equipment is designed to remain functional for the Safe Shutdown Earthquake and within the required stress limits for the Operational Basis Earthquake.

Electrical power is supplied from emergency power buses to each of the spent fuel pool pumps. Each pump is connected to these emergency power buses so that it receives power from a separate diesel generator set from the other pump, should offsite power be lost. The use of emergency power buses assures the operation of these pumps for open reactor cooling during plant refueling canal flooding conditions.

The KF System provides adequate capacity to maintain an acceptable temperature range for the normal and maximum heat loads. The large heat capacity of the fuel pool allows enough time for maintenance to assure necessary cooling in the event of component failures including a complete loss of forced cooling.

The system is designed to maintain the pool temperature below 140°F for the maximum case and 120°F for the normal case when both cooling trains are in operation. If only one train is in operation, the pool temperature remains below 140°F for the normal heat load and saturation for the maximum heat load. Table 9-5 shows the peak pool temperatures for each heat load case. Complete loss of cooling capability is considered highly unlikely since the cooling system is Seismic Category I, and redundant in active components such that a single failure would not prevent the system from performing its intended safety function.

In the event of a loss of forced cooling, the large volume of water in the pool would take several hours to heat up to boiling. The amount of time before the pool begins to boil depends on the heat load and the initial pool temperature. With two pump-cooler configurations in operation prior to loss of forced cooling, the time to adiabatically heat up to boiling under normal and maximum heat loads is given in Table 9-6. For the case where either of the two cooling trains is in operation prior to the loss of forced cooling, Table 9-6 gives the time to adiabatically heat up to boiling under normal maximum heat load. There is ample time to effect repairs to the cooling system or arrange alternate cooling should adequate cooling capacity be lost.

Loss of forced cooling calculations have been performed to verify the adequacy of the spent fuel pool as an assured emergency source of primary coolant pump seal water. These Safe Shutdown Facility calculations show that the boil-off time following loss of cooling and makeup capability with 26 GPM being withdrawn for Reactor Coolant Pump seals is in excess of 3 days. Ample time is provided to effect repairs, restore makeup capability, and maintain the spent fuel pool water level above the spent fuel assemblies.

9.1.3.3.2 Spent Fuel Pool Dewatering

The most serious failure of this system would be complete loss of water in the storage pool. To protect against this possibility, the spent fuel pool cooling suction connections enter near the normal water level such that it cannot be lowered appreciably by siphoning. The discharge lines to the pool have vacuum breakers to prevent siphoning.

9.1.3.3.3 Water Quality

Except for operation of this system during refueling only a very small amount of water is interchanged between the refueling canal and the spent fuel pool as fuel assemblies are transferred in the refueling process. Whenever a fuel assembly with defective cladding is transferred to the spent fuel pool, a small quantity of fission products may enter the spent fuel cooling water. The purification loop provided removes fission products and other contaminants from the water. Radioactivity concentrations in the spent fuel pool water are maintained at a level such that the dose rate at the surface of the pool is low enough to allow unrestricted access for plant personnel.

9.1.3.4 Tests and Inspections

Active components of the Spent Fuel Cooling System are either in continuous or intermittent use during normal plant operation. Periodic visual inspections are made during system operation and preventive maintenance is conducted per established Preventive Maintenance Program, using normal industry practice.

9.1.4 Fuel Handling System

9.1.4.1 Design Bases

The Fuel Handling System consists of equipment and structures utilized for the refueling operation in a safe manner.

The following design bases apply to the Fuel Handling System:

1. Fuel handling devices have provisions to avoid dropping or jamming of fuel assemblies during transfer operation.
2. Fuel lifting and handling devices are capable of supporting maximum loads under safe shutdown earthquake conditions.
3. The fuel transfer system, where it penetrates the Containment, has provisions to preserve the integrity of the Containment pressure boundary.
4. Cranes and hoists used to lift spent fuel have a limited maximum lift height so that the minimum required depth of water shielding is maintained.

9.1.4.2 System Description

The Fuel Handling system consists of the equipment needed for the refueling operation on the reactor core. Basically, this equipment is comprised of cranes, handling equipment and a fuel transfer system. The structures associated with the fuel handling equipment are the refueling cavity, the refueling canal, the spent fuel pool, and the new fuel storage area. The arrangement and location of fuel handling equipment in the Auxiliary Building is shown on Figure 9-1 and Figure 9-2. The arrangement and location of fuel handling equipment in the Containment is shown on Figure 9-24 and Figure 1-13.

New fuel assemblies received are removed one at a time from the shipping cask and stored in the fuel storage racks located in the fuel storage area, or temporarily stored in the new fuel vault.

New fuel is delivered to the reactor by placing a fuel assembly into the new fuel elevator, lowering it into the spent fuel pool and taking it through the fuel transfer system. New fuel may also be delivered to the reactor via the spent fuel storage racks.

The fuel handling equipment is designed to handle the spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Under-water transfer of spent fuel provides an effective, economic and transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. The boric acid concentration in the water further reduces the value of K_{eff} below that indicated in Section 9.1.2.3.1.

The associated fuel handling structures may be generally divided into three areas: the refueling cavity and refueling canal which are flooded only during unit shutdown for refueling, the spent fuel pool which is kept full of water and is always accessible to operating personnel, and the new fuel storage area which is separate and protected for dry storage. The refueling canal and the spent fuel pool are connected by a fuel transfer tube. This tube is fitted with a blind flange on the canal end and a gate valve on the spent fuel pool end.

The blind flange is in place except during refueling to ensure Containment integrity. Fuel is carried through the tube on an underwater transfer car.

Fuel is moved between core positions or between the reactor vessel and the refueling canal by the reactor manipulator crane. A rod cluster control changing fixture, located on the refueling canal wall, can be used for transferring control elements from one fuel assembly to another.

The upender at either end of the fuel transfer tube is used to pivot a fuel assembly. Before entering the transfer tube the upender pivots a fuel assembly to the horizontal position for passage through the transfer tube. After the transfer car transports the fuel assembly through the transfer tube, the upender at that end of the tube pivots the assembly to a vertical position so that it can be lifted out of the fuel container.

In the spent fuel pool, fuel assemblies are moved about by a fuel pool manipulator crane similar in the fuel handling features to the reactor cavity manipulator crane. A short tool is used to handle new fuel, and place it in the new fuel elevator which is used to lower the assembly to a depth at which the fuel pool manipulator crane can engage the new assembly and move it into the fuel transfer container in the upending device, or into the storage racks. Fuel components can be shuffled in the spent fuel pool by using specific components and manipulator crane hoists.

Decay heat, generated by the spent fuel assemblies in the spent fuel pool, is removed by the Spent Fuel Cooling System. After a sufficient decay period, the fuel can be removed from the racks and loaded into shipping casks for removal from the site or can remain stored in the spent fuel pool.

9.1.4.2.1 Refueling Procedure

The refueling operation follows a detailed procedure which provides a safe, efficient refueling operation. The following significant points are assured by the refueling procedure:

1. The boron concentration of the refueling water and the reactor coolant is maintained at a concentration that will, together with the negative reactivity of control rods, keep the core 5 percent or more $\Delta k/k$ subcritical during the refueling operations. It is also sufficient to maintain the core subcritical in the unlikely event that all of the rod cluster control assemblies were removed from the core.

2. The water level in the refueling cavity is high enough to keep the radiation levels within acceptable limits when the fuel assemblies are being removed from the core.

The refueling operation is divided into four major phases: 1) preparation; 2) reactor disassembly; 3) fuel handling; and 4) reactor assembly. During phases 2 and 4, the reactor's accessible components are inspected. A general description of a typical operation through the four phases is given below:

1. Phase I - Preparation

   The reactor is shutdown and cooled to cold shutdown conditions with a final K_{eff} <0.95 (all rods in). Following a radiation survey, the Containment Vessel is entered. After Reactor Coolant System cleanup, the coolant level in the reactor vessel is lowered to a point slightly below the vessel flange.

2. Phase II - Reactor Disassembly

   Electrical connections and seismic supports are removed. The head lift rig is then inspected and installed. The CRDM cable bridges are removed. Insulation is then removed from appropriate areas. The control rod drive mechanism (CRDM) cooling duct and supports are removed. The head vent line flanges are disassembled. The conoseals are then disassembled and protective sleeves are installed over thermocouple (T/C) connectors. The vessel head coolant level indication tubing is removed. The vessel head studs are detensioned and studs, nuts, and washers are removed, cleaned, inspected, and stored. Guide pins and stud hole plugs are installed. The vessel canal seal is installed. The NIS detectors cover O-rings are removed and replaced periodically. The permanent vessel nozzle inspection hatch covers are installed, sealed, and tested. The head is lifted about four inches and stopped. This position is held for at least ten minutes during which time the sling bolt lugs to the lifting block welds, and the spreader lugs to the spreader arm welds are visually inspected. The head is then lifted to the top of the guide pins where it remains suspended less than 20 feet above the vessel until the reactor cavity water level is raised to a depth of at least 8 feet above the vessel in order to meet the bounding conditions of the Westinghouse head drop analysis (Reference 37). The analysis qualifies a 20 foot head drop through air and a 40 foot head drop through air and water when there is at least 8 feet of water covering the reactor vessel. The polar crane has been classified as single failure proof equivalent in accordance with requirements and acceptance criteria in NEI 08-05 Revision 0. NEI 08-05 has been formally endorsed by the NRC. Therefore, as an alternative to controlling the lift height with medium present under the load as described in the preceding paragraph, the reactor vessel head may be lifted without the presence of the medium (water) under the load (vessel head). The head is then raised off of the guide pins to an appropriate height and moved to the vessel head storage stand. The refueling canal is flooded to full pond elevation. The CRDM drive rods are then disconnected and, with the upper internals, are removed from the vessel. The manipulator cranes are checked for proper operation. The fuel assemblies and rod cluster control assemblies are then free from obstructions and the core is ready for refueling.

3. Phase III - Fuel Handling

   The refueling sequence is started with the reactor manipulator crane. Spent fuel assemblies are removed from the core in the sequence presented in the refueling procedure which is prepared before each refueling.

   The general fuel handling sequence is:

   a. The reactor manipulator crane is positioned over a fuel assembly.

b. The fuel assembly is lifted by the manipulator crane to a pre-determined height sufficient to clear the reactor vessel and still leave sufficient water covering to eliminate any radiation hazard to the operating personnel.

c. The fuel transfer car is moved into the refueling canal from the spent fuel pool.

d. The fuel assembly container is pivoted to the vertical position by the upender.

e. The manipulator crane is moved to line up the fuel assembly with the fuel transfer system.

f. The manipulator crane loads a fuel assembly into the fuel assembly container of the transfer car.

g. The container is pivoted to the horizontal position by the reactor side upender.

h. The fuel container is moved through the fuel transfer tube to the spent fuel pool by the transfer car.

i. The container is pivoted to the vertical position by the pit side upender.

j. The fuel assembly is placed in the spent fuel storage rack by the fuel pool manipulator crane.

k. The new fuel assembly is brought from dry storage, lowered into the spent fuel pool with the new fuel elevator, and loaded into the spent fuel pool. Alternatively, the new fuel assembly may be already stored in the spent fuel racks.

l. Components are shuffled as necessary for next cycle.

m. The fuel assembly is loaded into the fuel assembly container by the fuel pool manipulator crane.

n. The fuel assembly container is pivoted to the horizontal position and the transfer car is moved back into the refueling canal.

o. The container is pivoted to the vertical position by the reactor side upender.

p. Fuel assemblies are located in the reactor core by the reactor manipulator crane.

4. Phase IV - Reactor Assembly

Reactor assembly, following refueling, is essentially achieved by reversing the operations given in Phase II - Reactor Disassembly.

9.1.4.2.2 Component Description

Reactor Manipulator Crane

The reactor manipulator crane, Figure 9-14, is a rectilinear bridge and trolley crane spanning the refueling canal with a vertical mast extending down into the refueling water. The mast supports and guides the gripping and hoisting devices for handling fuel assemblies. The bridge and trolley motions are used to position the mast over the fuel assembly positions in the core.

Fuel assemblies are lifted using a long tube with a pneumatic gripper on the end which is lowered down out of the mast to grip the fuel assembly. The gripper tube is long enough so that the upper end is still contained in the mast when the gripper end contacts the fuel. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. The fuel is transported while inside the mast tube to its new position.

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All controls for the reactor manipulator crane are mounted on a console on the trolley. The console contains the Programmable Logic Controller (PLC), Human-Machine Interface (HMI) computer/touch screen display, motor drive controllers, and manual inputs. The Bridge and trolley are positioned on a coordinate system defined by geared racks paralleling one bridge rail and one trolley rail. Dual redundant rotary encoders mounted to the bridge and trolley engage these gears and transmit position data to the PLC. Hoist (vertical) positioning utilizes dual redundant encoders mounted on the mast and connected to the movable inner mast. The inputs are compared for reliability and the position information is displayed on the console. The drives for the bridge, trolley, and winch are variable speed. Scalable controllers provide for inching capability on all three axes. Electrical interlocks and limit switches, plus a mechanical stop, prevent a fuel assembly from being raised above a safe shielding depth. The bridge, trolley, and winch can be operated manually using a handwheel on the motor shaft.

Fuel Pool Manipulator Crane

The fuel pool manipulator crane, Figure 9-16, is similar to the reactor manipulator crane but has two auxiliary hoists compared to one on the reactor manipulator crane.

New Fuel Elevator

The new fuel elevator, Figure 9-17, consists of a box-shaped elevator assembly with its top end open and sized to house one fuel assembly.

The new fuel elevator is used to lower a new fuel assembly to the bottom of the spent fuel pool where it is transported to the fuel transfer system or the fuel storage racks by the fuel pool manipulator crane.

Fuel Transfer System

The fuel transfer system, Figure 9-18, includes an above-water electric-motor driven transfer car that runs on tracks extending from the refueling canal through the transfer tube and into the spent fuel pool and an upender lifting frame at each end of the transfer tube. The upender in the refueling canal receives a fuel assembly in the vertical position from the reactor manipulator crane. The fuel assembly is then lowered to a horizontal position for passage through the transfer tube and raised to a vertical position after passage through the transfer tube by the upender in the spent fuel pool. The fuel pool manipulator crane takes the fuel assembly to a position in the spent fuel storage racks.

During reactor operation, the transfer car is stored in the spent fuel pool. A blind flange is bolted on the refueling canal end of the transfer tube to seal the reactor Containment.

Rod Cluster Control Changing Fixture

Although RCC elements are normally moved from one fuel assembly to another using an auxiliary hoist on the fuel pool manipulator crane, the RCC elements can also be shuffled by means of the rod cluster control changing fixture, Figure 9-19, located in the refueling canal inside of the reactor building. Five major subassemblies comprise the changing fixture including: (1) frame and track structure, (2) carriage, (3) guide tube, (4) gripper, and (5) drive mechanism. The carriage is a moveable container supported by the frame and track structure. The tracks provide a guide for the four flanged carriage wheels and allow horizontal movement of the carriage during changing operations. Positioning stops on both the carriage and frame locate each of the three carriage compartments directly below the guide tube. Two of these compartments are designed to hold individual fuel assemblies while the third is made to support a single rod cluster control element. Situated above the carriage and mounted on the refueling canal wall is the guide tube. This assembly provides for the guidance and proper orientation of the gripper and rod cluster control element as they are being raised and lowered. The gripper is a pneumatically actuated mechanism responsible for engaging the rod cluster control element. It has two flexure fingers which can be inserted into the top of the rod cluster control element when air pressure is applied to the gripper position. Mounted on the operating deck is the drive mechanism assembly. Its components include: (1) manual carriage drive mechanism, (2) revolving stop operating handle, (3) pneumatic selector valve for actuating the gripper piston, and (4) electric hoist for elevation control of the gripper.

New Fuel Assembly Handling Fixture

This short-handled tool is used to handle new fuel on the operating deck of the fuel storage building; to remove the new fuel from the shipping container; and to facilitate inspection and storage of the new fuel and loading of fuel into the new fuel elevator.

Reactor Vessel Head Lifting Device

The reactor vessel head lifting device, Figure 9-21, consists of a welded and bolted structural steel frame with suitable rigging to enable the crane operator to lift the head and store it during refueling operations. The lifting device is normally attached to the reactor vessel head. Attached to the head lifting device are the monorail and hoists for the reactor vessel stud tensioners.

Reactor Internals Lifting Device

The reactor internals lifting device, Figure 9-22, is a structural frame suspended from the overhead polar crane. The frame is lowered onto the upper core support plate of the internals, and is manually engaged to the support plate by three roto-lock studs. Bushings on the frame engage guide studs in the vessel flange to provide guidance during removal and replacement of the internals package.

Reactor Vessel Stud Tensioner

Stud tensioners, Figure 9-23, are employed to secure the head closure joint at every refueling. The stud tensioner is a hydraulically operated device that uses oil as the working fluid. The device permits preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. Stud tensioners minimize the time required for the tensioning or unloading operation. Three tensioners are provided and are applied simultaneously to three studs located 120 degrees apart. A single hydraulic pumping unit operates the tensioners which are hydraulically connected in parallel/series. Relief valves on each tensioner prevent over-tensioning of the studs due to excessive pressure.

Fuel Handling Bridge Crane

The fuel handling bridge crane is a CMAA specification No. 70 Class 1A electric overhead traveling bridge crane. The main hoist is rated at 125 tons and the auxiliary hoist is rated at 4 tons. The crane and accessories are used to handle both new and spent fuel assemblies contained in protective containers. Handling of fuel by the crane main hoist is limited to the area between the new fuel storage vault and the spent fuel shipping cask area. Handling of fuel by the auxiliary hoists is limited to the area between the new fuel storage vault and the new fuel elevator. Neither of the hoists are capable of being immersed in the spent fuel pool. Mechanical stops are installed to prevent the main hoist from being positioned over the spent fuel pool.

Design criteria for the 125-ton crane are as follows:

1. As a minimum the crane complies in all respects with the specification for electric overhead traveling cranes, C.M.A.A. Specification No. 70 and Service Class 1A.
2. Mechanical parts are designated to have a minimum safety factor of five when under rated load and based on the ultimate strength of the material used.

3. The main and auxiliary hoists are not required to handle rated capacity loads at the same time.
4. Mechanical parts are also proportioned to withstand loads produced by the rated pull out torque of the motors with the unit stress not to exceed 70 percent of the elastic limit of the material involved.
5. The maximum allowable stresses used in structural design of the crane are those established in the specifications for electric overhead traveling cranes, C.M.A.A. Specification No. 70.
6. The main and auxiliary hooks are load tested to 200 percent of the rated load. The hooks are annealed and design stresses of the hooks are limited to 10,000 psi tension in the stem and 20,000 psi combined tension and bending stress.
7. All welding design and procedures conform to "Welding in Building Construction", AWS D1.0-69. Component parts of built up members are fabricated with fillet welds. Butt welds, where used, are made from both sides and are full penetration welds. The structural welds for the bridge, trucks, drums, and trolleys are 100 percent inspected by magnetic particle testing, liquid penetrant testing, or equivalent methods on NDE examination as defined by applicable welding code per C.M.A.A Specification No. 70 (Ref. Design Criteria #1 above).
8. Materials used conform to the following:

a.	Structural Steel	A36
b.	Rivet Steel	A242
c.	High Strength Bolts and Nuts	A325
d.	Cold Finished Steel, GDS 1010 to 1040, Inclusive	A108

9. Wire rope is used with a minimum safety factor of five (5) based on the static rated load without allowance for impact, frictional resistance or for bending stresses over sheaves.
10. Gearing conforms to AGMA Standards.

The crane is seismically designed. The seismic loads are not considered acting simultaneously with the crane loaded. Hold down devices are provided.

9.1.4.3 Safety Evaluation

9.1.4.3.1 Safe Handling

The reactor manipulator crane design includes the following provisions to ensure safe handling of fuel assemblies:

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1. Bridge, trolley, and hoist drives are mutually interlocked to prevent simultaneous operation of any two drives when operating in the manual mode. In either semi-automatic or automatic modes of operation, simultaneous operation of the bridge and trolley drives is permitted but the bridge/trolley drives and hoist drive are mutually interlocked to prevent simultaneous horizontal and vertical motion.

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2. Bridge and trolley drive operation is prevented except when a fuel assembly is completely withdrawn into the mast (loaded gripper) or when the gripper alone is completely withdrawn into the mast (unloaded gripper). Redundant inputs from dual hoist encoders determine the required hoist position, and loaded gripper or unloaded gripper conditions are determined by combined inputs from the load cell and the gripper engaged position switch.
3. An interlock is supplied which prevents the opening of a solenoid valve in the air line to the gripper except when the load cell indicates that a fuel assembly is not suspended. As backup protection for this interlock, the mechanical weight actuated lock in the gripper prevents operation of the gripper under load even if air pressure is applied to the operating cylinder.

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4. The fuel hoist is equipped with overload and underload protection with primary setpoints based on the fuel manufacturer's recommendations to protect fuel assemblies from excessive drag forces. These overload and underload limit values are determined by the position of a fuel selector switch mounted on the control console. Secondary overload protection set at approximately 3200 lbs is provided by redundant switches exernal to the PLC (one in the load cell amplifier and another mechanical overload switch). Additionally a geared limit switch provides overtravel up and overtravel down protection by stopping the hoist when travel exceeds normal full up or full down positions.

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5. An interlock in PLC controls allows bridge, trolley, and hoist motion only when either the engaged or the disengaged indicating switch on the gripper is actuated (one switch must be open and one switch closed in order for the permissive to be active). If one or both switches fail, or remain open for more than 1 second, audible and visual alarms on the control console are actuated and all drives are disabled.

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6. Bridge and trolley movement at normal speeds is restricted to defined secure zones where the crane can operate without risk of collision with walls or other fixed structures or obstacles. At some locations mechanical stops also prevent collision of the mast with obstructions. Operation outside of the secure zones is only permitted at reduced speed using the travel override bypass switch.

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7. Interlocks are provided on the hoist drive to limit hoist speed to slow speeds in predetermined "slow zones" where potential interferences exist. These "slow zones" include:
   a. A zone starting ≥ 10" above a storage location (core, upender, etc.) and extending ≥ 10" into a storage location where hoist speed is limited to slow speed in the downward direction.
   b. A zone beginning ≥ 10" above full down and extending to full down in a storage location where hoist speed is limited to slow speed in both directions.
   c. A zone starting ≥ 6" before the gripper enters the mast and extending ≥ 6" into the mast where hoist speed is limited to slow speed in the upward direction.
   d. A zone starting ≥ 6" before the gripper reaches the full up position extending to full up where hoist speed ramps down in the upward direction.

8. An interlock prevents hoist movement when the mast is indexed over the upender and the upender is not at its full up position.
9. Suitable restraints are provided between the bridge and trolley structures and their respective rails to prevent derailing due to the safe shutdown earthquake. The manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper under the safe shutdown earthquake.
10. The fuel, and auxiliary hoists are equipped with two independent braking systems. A solenoid release - spring set electric brake is mounted on the motor shaft. This brake operates in the normal manner to release upon application of current to the motor and set when current is interrupted. The second brake is a mechanically actuated load brake internal to the hoist gear box that sets if the load starts to overload the hoist. It is necessary to apply torque from the motor to raise or lower the load. In raising, the motor cams the brake open; in lowering, the motor slips the brake allowing the load to lower. This brake actuates upon loss of torque from the motor for any reason and is not dependent on any electrical circuits. On the fuel hoist, the motor brake and the mechanical brake are capable of holding their rated loads.

The fuel hoist system is supplied with redundant paths of load support such that failure of any one component does not result in free fall of the fuel assembly. Two wire ropes are anchored to the winch drum and carried over independent sheaves to a load equalizing mechanism attached to the swivel on the top of the gripper tube. In addition, supports for the sheaves and equalizing mechanism are backed up by passive restraints to pick up the load in the event of failure of this primary support. Each cable system is designed to support 13,750 pounds or 27,500 pounds acting together.

The working load of fuel assembly plus gripper is approximately 2500 pounds.

The gripper itself has four fingers gripping the fuel, any two of which will support the fuel assembly weight.

The gripper and hoist system are routinely load tested to 4000 pounds.

The following safety features are provided for in the fuel transfer system control circuit:

1. Transfer car operation is possible only when both upenders are in the down position as indicated by the limit switches.
2. The remote control panels have a permissive switch in the transfer car control circuit that prevents operation of the transfer car in either direction when either switch is open, i.e., with two remote control panels, one in the refueling canal and one in the spent fuel pool, the transfer car can not be moved until both “go” switches on the panels are closed.
3. Redundant switches allow upender operation only when the transfer car is at either end of its travel.
4. Transfer car operation is possible only when the transfer tube valve position switch indicates the valve is fully open.
5. The refueling canal upender is interlocked such that it cannot be operated unless the reactor manipulator crane fuel gripper tube is fully retracted with a fuel assembly engaged, the fuel gripper tube is completely in the mast with no fuel assembly engaged, or the crane is located over the core. Similarly, the fuel pool upender is interlocked such that it cannot be operated unless the fuel pool manipulator crane fuel gripper tube is fully retracted with a fuel assembly engaged, the fuel gripper tube is completely in the mast with no fuel assembly

engaged, or the crane is located outside of the upender basket area. The interlocks associated with gripper position are controlled by redundant interlocks.

6. All fuel handling tools and equipment handled over an open reactor vessel are designed to prevent inadvertent decoupling from crane hooks (i.e., lifting rigs are pinned to the crane hook and safety latches are provided on hooks supporting tools).

   Tools required for handling internal reactor components are designed with fail safe features that prevent disengagement of the component in the event of operating mechanism malfunction. These safety features apply to the following tool:

   a. Control Rod Drive Shaft Unlatching Tool: The air cylinders actuating the gripper mechanism are equipped with back up springs which close the gripper in the event of loss of air to the cylinder. Air valves are equipped with safety locking rings to prevent inadvertent actuation.

The fuel pool manipulator crane design includes the following provisions to ensure safe handling of fuel assemblies:

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1. All of the features listed for the reactor manipulator crane also apply to the fuel pool manipulator crane except that the travel limits, hoist overtravel limit, and "slow zones" differ as follows:

   a. An interlock of the bridge and trolley drives prevents the bridge from entering the transfer canal area from either the upender basket area or through the transfer canal gate unless the trolley is positioned with the fuel mast centered on an unobstructed travel path. Similarly, an interlock prevents the bridge from entering or exiting the cask area gate unless the trolley is positioned with the fuel mast centered on the cask area gate. In both cases the trolley drive is locked out when the crane enters the interlocked area. In addition, travel zone boundaries and mechanical stops prevent collision of the mast with walls and other obstructions throughout the spent fuel pool, transfer canal, and cask areas.

   b. Instead of the core slow zones, the fuel pool manipulator crane has slow zones entering the fuel storage racks (beginning approximately 10" above the rack and extending 10" into the rack) and at the bottom of the fuel storage racks (approximately 10" above the rack bottom extending to the rack bottom) that prevent further downward hoist movement except in slow speed. Similar slow zones are provided for the new fuel elevator and upender entrance (beginning approximately 10" above the elevator/upender and extending 10" into the elevator/upender) and bottom (approximately 10" above the elevator bottom extending to the elevator bottom).

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2. An interlock between the fuel pool manipulator crane and the new fuel elevator prevents the crane from bridging over the new fuel elevator unless the elevator is in the full down position. Also, the new fuel elevator is prevented from raising when the manipulator crane is indexed over the elevator.

3. An interlock prevents operation of the manipulator crane hoist, bridge, and trolley drives unless the north monorail is in the full up position.

9.1.4.3.2 Seismic Considerations

The safety classifications for all fuel handling and storage equipment are listed in Table 3-4. These safety classes provide criteria for the seismic design of the various components. Class 1 and Class 2 equipment are designed to withstand the forces of the OBE and SSE. For normal conditions plus OBE loadings, the resulting stresses are limited to allowable working stresses as defined in the American Institute of Steel Construction (AISC) Manual with the allowed increase for seismic stresses. For normal conditions plus SSE loadings, the stresses are limited to within the material yield strength for critical parts of the equipment which are required to maintain the capability of the equipment to perform its safety function. Permanent deformation is allowed for the loading combination which includes the SSE to the extent that there is no loss of safety function.

The Class 3 fuel handling and storage equipment satisfies the Class 1 and Class 2 criteria given above for the SSE. Consideration is given to the OBE only insofar as failure of the Class 3 equipment might adversely affect Class 1 or 2 equipment.

For Non-Nuclear Safety equipment, design for the SSE is considered if failure might adversely affect a Safety Class 1, 2 or 3 component. Design for the OBE is considered if failure of the Non-Nuclear Safety component might adversely affect Safety Class 1 or 2 component.

9.1.4.3.3 Containment Pressure Boundary Integrity

The fuel transfer tube which connects the refueling canal (inside the reactor Containment) and the spent fuel pool (outside the Containment) is closed on the refueling canal side by a blind flange at all times except during refueling operation. Two seals are located around the periphery of the blind flange with leak-check provisions between them.

9.1.4.3.4 Radiation Shielding

During all phases of spent fuel transfer, the gamma dose rate at the surface of the water will be maintained ALARA by maintaining sufficient depth of water above the fuel assembly. At normal water levels the dose rate will be 3 millirem/hour or less. However, it should be noted that during some fuel handling operations in the Reactor Building the gamma dose rate is higher at the surface of the water due to other contributing radiation sources, (the reactor head assembly on the head stand, CRDMs, etc.).

The two cranes used to lift spent fuel assemblies are the reactor manipulator crane and the fuel pool manipulator crane. Both cranes contain positive stops which prevent the top of the fuel pellets in a fuel assembly from being raised to above a minimum safe shielding depth in the refueling cavity and the spent fuel pool.

9.1.4.4 Tests and Inspections

As part of normal unit operations, the fuel-handling equipment is inspected for operating conditions prior to each refueling operation. During the operational testing of this equipment, procedures are followed that affirm the correct performance of the fuel handling system interlocks.

In response to NRC 1E Bulletin 84-03, "Reactor Cavity Water Seal," which cited a failure of the reactor cavity water seal at the Haddam Neck Plant, evaluations and tests of the McGuire Nuclear Station Units 1 and 2 refueling cavity water seals were performed to determine the potential for and consequences of a refueling cavity water seal failure. Of particular concern in this bulletin was the potential for water inventory loss should the refueling cavity water seal between the reactor vessel and the refueling cavity fail. As concluded in the DPC final response to the NRC for 1E Bulletin 84-03 (letter from H.B. Tucker to the NRC, dated November 12, 1985), the evaluations of both unit's refueling cavity water seals demonstrated that such failure was precluded due to inherent design features. Nonetheless, emergency operating procedures for both McGuire units were established to address operator actions in the unlikely event of unanticipated refueling cavity inventory reduction.

9.1.4.5 Oconee Fuel Handling Equipment

Since the Oconee fuel assemblies are not compatible with the gripper used for the McGuire fuel assemblies, handling cannot be performed with the spent fuel crane. Oconee fuel handling is performed with a special handling tool which is suspended from an auxiliary hoist mounted on the north end of the fuel pool manipulator crane bridge. The handling tool has mechanically actuated grippers which are manually operated to engage or disengage a fuel assembly. The auxiliary hoist is provided an excessive suspended weight switch, and bridge and trolley drive interlocks to prevent simultaneous operation.

9.1.5 Overhead Heavy - Load Handling Systems

9.1.5.1 Introduction and Licensing Background

As a result of Generic Task A-36, "Control of Heavy Loads Near Spent Fuel," NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," was developed. Following the issuance of NUREG-0612, a Generic Letter dated December 22, 1980 (as supplemented on February 3, 1981 by Generic Letter 81-07), was sent to all operating plants, applicants for operating licenses and holders of construction permits requesting that responses be prepared to indicate the degree of compliance with the guidelines of NUREG-0612. Phase 1 responded to Section 5.1.1 of NUREG-0612 and addressed applicable codes and standards for the subject cranes and special lifting devices, crane operator training and qualification and procedures for heavy load handling. Phase 2 responded to Sections 5.1.2, 5.1.3, 5.1.5, and 5.1.6 of NUREG-0612 and addressed the need for mechanical stops or electrical interlocks, the need for single-failure-proof handling systems and load drop consequence analyses. By letters dated June 2, August 5, October 8 and November 23, 1981; January 15, March 3, June 4 and July 26, 1982; July 15, 1983; August 17, 1984; and January 31 and February 22, 1985 Duke provided responses to this NRC request.

On March 12, 1985, the NRC issued its Safety Evaluation Report (SER) for McGuire Nuclear Station (Reference 28), concluding that "... the guidelines in NUREG-0612, Sections 5.1.1 (Phase 1) and 5.3 have been satisfied." The SER further states that "... Phase 1 for McGuire Nuclear Station, Units 1 and 2 is acceptable."

On June 28, 1985, the NRC issued Generic Letter 85-11. This generic letter concluded that Phase 1 had provided improvements in heavy load handling and that Phase 2 was no longer required. On April 11, 1996, the NRC issued Bulletin 96-02, "Movement of Heavy Loads over Spent Fuel in the Reactor Core or Over Safety Related Equipment." Duke's response concluded that existing regulatory guidelines associated with the control and handling of heavy loads while the plant is operating, were being met. In its summary of the NRC staff's review of licensee responses for Bulletin 96-02, the NRC concluded that Duke Power's response to the Bulletin was acceptable." On October 31, 2005 the NRC issued Regulatory Issue Summary (RIS) 2005-25, "Clarification of NRC Guidelines for Control of Heavy Loads", as a result of recommendations developed through Generic Issue 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants." The RIS reemphasized the guidelines of NUREG-0612 and identified relevant operating experience and inspection information related to the movement of heavy loads. On May 29, 2007, the NRC issued Supplement 1 of RIS 2005-25 addressing remaining recommendations associated with GI 186 and communicated regulatory expectations related to safe load handling.

On September 14, 2007, and "Industry Initiative on Heavy Lifts" was initiated by NEI to specify those actions to be taken by each plant to ensure that heavy lifts continue to be conducted safely and that each plant's licensing bases accurately reflected those plant practices.

9.1.5.2 Design Basis

The design bases of the overhead heavy load systems are to:

1. Assure that the potential for a load drop is extremely small,
2. That in the event of a postulated reactor vessel head drop, the core remains covered and cooled, and
3. Assure the consequences of a load drop in the spent fuel pool meet the acceptance criteria of NUREG-0612.

9.1.5.3 Scope of Heavy Load Handling Systems

All cranes and hoists lifting heavy loads over spent fuel or safe shutdown equipment comply with the guidelines of NUREG-0612 and are consistent with McGuire's responses and commitments related to the handling of heavy loads. Additional details regarding each load handling device in the Reactor Building and Auxiliary Building are provided in Table 2.1 of the NRC SER dated March 12, 1985 (Reference 28).

9.1.5.4 Control of Heavy Lifts Program

The control of heavy lifts consists of the following:

1. McGuire's commitments in response to NUREG-0612, Phase 1 elements
2. McGuire's response to NRC Bulletin 96-02 dated May 13, 1996.
3. McGuire's response to the NEI Initiative on Heavy Load Lifts
4. Reactor vessel head lift load drop analysis assumptions (lift height and medium present) are incorporated into plant procedures
5. The polar crane is single failure proof equivalent for lifting the reactor vessel head as defined in NEI 08-05 Revision 0, Industry Initiative on Control of Heavy Loads. This is an acceptable alternative method to the load drop analysis.
6. Load drop analyses have been performed for loads over the spent fuel pool.

McGuire maintains a Lifting Program to minimize the potential for adverse interaction between overhead load handling operations and: 1) nuclear fuel assemblies to ensure a subcritical configuration and preclude radiological consequences and; 2) structures, systems and components (SSCs) selected to ensure safe, cold shutdown of the plant following a postulated heavy load drop event. A "heavy load" has been defined by NUREG-0612 - Section 1.2 as “any load, carried in a given area after a plant becomes operational, that weights more than the combined weight of a single spent fuel assembly and its associated handling tool for the specific plant in question”. For McGuire this definition of a “heavy load” applies to a load weighing 1500 lbs. or more. However, NUREG-0612 also applies to potentially lighter loads as stated in Section 5.1.5: “In other plant areas, loads may be handled which, if dropped in a certain location, may damage safe shutdown equipment....Some of these loads may be less than the weight of a fuel assembly with its handling tool, but may be sufficient to damage safe shutdown equipment." The bases of the NRC acceptance of McGuire's program is summarized in the March 12, 1985 SER as revised by McGuire's response to NRC Bulletin 96-02 dated May 13, 1996. The objective of the program is to ensure that all load handling systems are designed, operated, and maintained such that their probability of failure is uniformly small and their use appropriate for the critical tasks in which they are employed.

9.1.5.4.1 McGuire Commitments in Response to NUREG-0612, Phase 1 Elements

The McGuire Lifting Program is based on the NEI "Industry Initiative on Heavy Lifts" and the following general guideline areas of NUREG-0612, Section 5.1.1:

Guideline 1 - Safe Load Paths

Guideline 2 - Load Handling Procedures

Guideline 3 - Crane Operator Training

Guideline 4 - Special Lifting Devices

Guideline 5 - Lifting Devices (not specifically designed)

Guideline 6 - Cranes (inspection, testing and maintenance)

Guideline 7 - Crane Design

The following Sections summarize the commitments made by McGuire in compliance with Section 5.1.1 of NUREG-0612:

<u>Safe Load Paths</u>

NUREG-0612, Section 5.1.1 defines a "Safe Load Path" as one which minimizes the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment.

Reactor and Auxiliary Building Monorails

Safe load paths for monorails located in the reactor and auxiliary buildings are not required because the monorail safe load path can only be a vertical projection of the monorail on the underlying floor.

Crane Movement in the Reactor Buildings

Safe load paths have been established in load handling procedures for major equipment handled in the reactor building. Deviations from safe load paths require formal review and approval, and

McGuire procedures prohibit the movement of heavy loads over the reactor vessel, with head removed and irradiated fuel exposed.

As part of McGuire's response to NRC Bulletin 96-02, the polar crane may be used in Modes 1 through 4 to lift the pressurizer enclosure hatch plugs, make lifts in support of polar crane maintenance activities, and make lifts that are of the same order of magnitude as the pressurizer hatch plug. The load drop analysis for the pressurizer enclosure hatch plug was reviewed and approved by the NRC as documented in the SER dated April 5, 2005 related to the Technical Specification 3.6.14 revision (Amendment Nos. 228/210).

The polar crane may also be used in Modes 1 through 4 to perform inspections and preventative maintenance associated with the reactor vessel upper internals lift rig. This requires the lift rig to be lifted approximately 35 feet 7 inches above its storage stand located in the refueling canal floor to approximately four feet above the upper containment operating floor. The load drop analysis determined that the consequences were within the guidelines of NUREG-0612, Section 5.1. At no time during the lift, inspection or maintenance will the lift rig be moved in any horizontal plane. Based on a conservative weight of 21,500 pounds local damage to the refueling canal floor will be limited to rupture of the stainless steel liner in the refueling canal floor. No safety-related equipment is located in the drop zone.

Crane Movement in the Spent Fuel Pool Buildings

Loads in excess of 3000 pounds are prohibited from travel over fuel assemblies in the storage pool. Spent fuel pool weir gates may be moved over the stored fuel in accordance with the requirements of the Selected Licensee Commitments (SLC) Manual,

Designated travel paths have been identified in fuel handling procedures, and

A safe load path has been established for movement of spent fuel casks in the fuel pit and fuel pool area.

<u>Load Handling Procedures</u>

The rigger is responsible for the selection, inspection and safe use of rigging hardware. The crane/equipment operator is responsible for the safe operation of the crane/equipment during lift, transport and set of a load and shall be the final authority on decisions related to performing safe handling of the lift.

The removal or movement of equipment associated with special lifting devices listed below is controlled by written procedures.

All lifting and lowering of the reactor vessel head and internals is required to be continuously monitored with the load cell.

<u>Crane Operator Training</u>

Crane operators are qualified, trained and conduct themselves in accordance with ANSI B30.2-1976.

<u>Special Lifting Devices</u>

The following special lifting devices were designed, fabricated and load tested in compliance with the requirements or intent of ANSI N14.6-1978, as applicable:

- Reactor vessel head lifting rig and load cell
- Reactor internals lifting rig
- Reactor coolant pump motor lifting rig
- Control rod drive mechanism missile shield lifting rig

Six additional special lifting devices have been identified which were not included in the March 12, 1985 SER. Each of these devices were designed and fabricated in accordance with ANSI N14-6 requirements:

- TN-32 Cask Lift Beam
- ISFSI Extension
- NAC-UMS TFR

- NAC-UMS Lifting Yoke
- Magnastor Transfer Canistor (MTC)
- Magnastor Lifting Yoke

Lifting Devices (not specifically designed)

Lifting devices consist of the appropriate size and number of chain-falls, chokers and slings as determined by the rigger. In making that selection, the rigger draws on his experience and training. Chocker and sling sizing are determined by the estimated weight of the load. All lifts are made by qualified people who, by experience and/or training, are cognizant in the movement of loads.

No hoists at McGuire are capable of speeds greater than 30 feet per minute (fpm) while loaded. In addition, lifts over the reactor vessel and spent fuel are performed at much slower speeds. Therefore, the dynamic loads are small relative to the static loads. Since all slings are manufactured in accordance with the requirements of ANSI B30.9 with a safety factor of 5, the dynamic loads are of insignificant concern relative to sling integrity.

Inspection, Testing and Maintenance

The McGuire Crane Inspection Program is discussed in Section 18.2.7 of the McGuire UFSAR.

All slings are required to be inspected prior to each use and receive a documented periodic inspection.

Crane inspection, testing and maintenance programs at McGuire comply with Chapter 2-2 of ANSI B30.2-1976. All other lifting devices are inspected in accordance with the applicable ANSI standard.

The following requirements have been imposed on the reactor coolant pump motor and control rod drive mechanism missile shield lifting rigs by Duke's letter of August 17, 1984:

- Major modifications or repairs are subject to a 125% load test,
- All load-carrying components are fitted with cotter pins or lock pins and/or lockwire,
- Dimensional testing and visual inspection of major load-carrying welds and critical areas is required to be performed in accordance with ANSI N14.6-1978, Section 5.5, as permitted by Section 5.3.1, and
- Inspection of each device is performed in accordance with preventive maintenance procedures during each outage (if they are intended to be used).

The following requirements have been imposed on the reactor vessel head and internals lifting rigs by Duke's letter dated January 31, 1985:

- Weld repairs are required to be performed in accordance with the requirements identified in ASME III, NF-4000 and NF-5000,
- Major modifications or repairs are subject to a 125% load test,
- An inspection procedure meeting the requirements of ANSI N14.6-1978 has been written for these lifting devices. All load bearing parts and welds are inspected and any part or weld that appears suspicious or for which a defect is

detected is subjected to an appropriate non-destructive examination by QC personnel. Inspection of each device is performed in accordance with preventive maintenance procedures during each outage (if they are intended to be used),

- Critical welds and/or parts are inspected by nondestructive examination once every ten (10) years,
- Replacement parts, should they be required, are to be made of identical (or equivalent) material and inspections as originally required. Only pins, bolts and nuts are considered replacement parts for the reactor vessel head and internal lift rigs. Pins, bolts or other fasteners would be replaced in lieu of repair,
- A visual inspection of all lifting rig components and critical welds for cracks or deformation is required prior to use, and
- After connection of the lifting device to the reactor vessel head or upper/lower reactor internals, the load is to be lifted slightly off its support and held for 10 minutes. During this time, the sling block to lifting block welds, and spreader lug to spreader arm weld (for reactor head lifting device) and sling block lugs to lifting block welds (internals lifting device) are to be inspected. It is considered that this 100% load test, performed on each device, followed by a visual check of critical welds is sufficient to demonstrate compliance with ANSI N14.6-1978, Section 5.3.1.

Station procedures require that the TN-32 Cask Lift Beam, ISFSI Extension, NAC-UMS Lifting Yoke, NAC-UMS TFR, MTC and Magnastor Lifting Yoke be visually examined prior to use and inspected, tested and maintained (load test, visual inspection of load bearing welds and critical areas, and nondestructive examination, as appropriate) in accordance with ANSI N14.6 requirements.

Nondestructive examination of major load-carrying welds and critical areas of special lifting devices will be performed by personnel qualified in accordance with the rules in the edition of ANSI/ASNT CP-189 adopted by the NRC approved McGuire Inservice Inspection Plan. Acceptance standards shall be as indicated in paragraphs NF-5350 and NF-5340 of the latest NRC approved edition of ASME Section III, Division 1.

Crane Design

The reactor building polar crane and fuel handling bridges were designed in accordance with CMAA-70 and Chapter 2-1 of ANSI B30.2-1976. Other load-handling devices (e.g., monorails and jib hoists) were designed in accordance with appropriate industrial standards.

9.1.5.4.2 McGuire Response to NEI Initiative on Heavy Load Lifts

9.1.5.4.2.1 Reactor Vessel Head Lifting Procedures

In response to the September 14, 2007 NEI "Industry Initiative on Heavy Loads Lifts," McGuire Procedures used to control the lift and replacement of the reactor vessel head establish limits of load height and the medium present under the load. These procedures are based on (1) analyses performed using the guidance and acceptance criteria developed by NEI as a part of its initiative, and (2) provide additional assurance that the core will remain covered and cooled in the event of a postulated reactor vessel head drop. The motion restrictions, such as load height and medium present under the load to cushion postulated drops, and load weight are identified in the plant procedures governing load handling.

The polar crane has been classified as single failure proof equivalent in accordance with requirements and acceptance criteria in NEI 08-05 Revision 0. NEI 08-05 has been formally endorsed by the NRC. Therefore, as an alternative to controlling the lift height with medium present under the load as described in the preceding paragraph, the reactor vessel head may be lifted without the presence of the medium (water) under the load (vessel head).

9.1.5.4.2.2 Load Drops in the Spent Fuel Pool Building

The area in which the spent fuel casks are handled is designed for a 30 foot drop of a proposed 125 ton cask and the structure is reinforced concrete with a rock foundation. The cask handling crane stops are located in a position to prevent the cask from being moved into the fuel pool area. The cask area is separated from the spent fuel pool by a three foot reinforced concrete wall. Local damage to the concrete will be negligible and no safety related equipment is located in the cask travel path.

The evaluation and consequence of spent fuel cask drops in the Spent Fuel Pool are discussed in Section 9.1.2.3.2 of the UFSAR.

The evaluation and consequence of a weir gate drop, while being maneuvered in the spent fuel pool, are discussed in Section 15.7.4.3 of the UFSAR.

9.1.5.5 Safety Evaluation

The McGuire Lifting Program provides a defense-in-depth approach which ensures that all load handling systems are designed, operated, and maintained such that the probability of their failure is very small and the use of said handling systems appropriate for the tasks in which they are employed. In addition, procedures to lift and replace the reactor vessel head ensure the core remains covered and cooled when a reactor vessel head drop is postulated.

The consequences of a load drop in the spent fuel pool meet the acceptance criteria of NUREG-0612.

9.1.6 References

1. Deleted Per 2006 Update.
2. Deleted Per 1996 Update.
3. Deleted Per 1996 Update.
4. Deleted Per 1996 Update.
5. Deleted Per 1996 Update.
6. Deleted Per 1996 Update.
7. Deleted Per 1996 Update.
8. Deleted Per 1996 Update.
9. Deleted Per 2002 Update.
10. Deleted Per 2002 Update.
11. Deleted Per 2006 Update.
12. HI-2022937, “Local Temperature Evaluation of the McGuire Units 1 and 2 Spent Fuel Pools,” Revision 0, approved September 19, 2002.

13. HI-2022932, “Bulk Thermal-Hydraulic Analysis for the McGuire Units 1 and 2 Rerack,” Revision 0, approved August 30, 2002.
14. Deleted Per 2006 Update.
15. NUREG/CR-0200, Section S2, “SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module.”
16. NUREG/CR-0200, Section D1, “ORIGEN-ARP: Automatic Rapid Process for Spent Fuel Depletion, Decay, and Source Term Analysis.”
17. Letter from G. Peterson (Duke) to U.S. NRC, “McGuire Nuclear Station Units 1 and 2, Proposed Technical Specification (TS) Amendments, TS 3.7.15 – Spent Fuel Assembly Storage, and TS 4.3 – Fuel Storage” – September 29, 2003.
18. MCC-1201.30-00-0009, "Two Region Storage Rack Expanded Heat Load".
19. Letter from D.G. Eisenhut (NRC) to all licensees dated December 22, 1980. Subject: Control of Heavy Loads.
20. Letter from W.O. Parker, Jr. (Duke) to H.R. Denton (NRC) dated March 3, 1982. Subject: Response to Section 2.1 of Enclosure 3 of the Dec. 22, 1980 Generic Letter.
21. Letter from W.O. Parker, Jr. (Duke) to H.R. Denton (NRC) dated June 4, 1982. Subject: Control of Heavy Loads, NUREG-0612.
22. Letter from W.O. Parker, Jr. (Duke) to H.R. Denton (NRC) dated July 26, 1982. Subject: Control of Heavy Loads, NIREG-0612.
23. Letter from H.B. Tucker (Duke) to H.R. Denton dated November 1, 1982. Subject: Response to Concerns Related to NUREG-0612.
24. Letter from E.G. Adensam (NRC) to H.B. Tucker (Duke) dated June 10, 1983. Subject: Control of Heavy Loads, Phase 1.
25. Letter from H.B. Tucker (Duke) to H.R. Denton dated August 17, 1984. Subject: NUREG-0612, Control of Heavy Loads at Nuclear Power Plants.
26. Letter from H.B. Tucker (Duke) to H.R. Denton dated January 31, 1985. Subject: NUREG-0612, Control of Heavy Loads at Nuclear Power Plants.
27. Letter from H.B. Tucker (Duke) to H.R. Denton dated February 22, 1985. Subject: NUREG-0612, Control of Heavy Loads at Nuclear Power Plants.
28. Letter from T.M. Novak (NRC) to H.B. Tucker (Duke) dated March 12, 1985. Subject: Control of Heavy Loads [Safety Evaluation Report].
29. Letter from H.L. Thompson, Jr. (NRC) to All Licensees for Operating Reactors dated June 28, 1985. Subject: Completion of Phase 2 or "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612 (Generic Letter 85-11).
30. Letter from M.S. Tuckman (Duke) to Document Control Desk (NRC) dated May 13, 1996. Subject: Response to NRC Bulletin 96-02: Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment.
31. Letter from Frank Rinaldi (NRC) to M.S. Tuckman (Duke) dated May 1, 1998. Subject: Completion of Licensing Action For NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, over fuel in the reactor core, or over safety-related equipment".
32. Selected Licensee Commitment (SLC) 16.9.20, Crane Travel - Spent Fuel Pool Storage Building.

33. MP/1/A/7650/060, "Operation of Polar Crane in Unit 1 Upper Containment;" MP/2/A/7650/116, "Operation of Polar Crane in Unit 2 Upper Containment."
34. Duke Energy Nuclear Lifting Program.
35. NAC-UMS FSAR, Sections 1.2.1.4 and 1.2.1.5.9
36. TN-32 FSAR, Section 8.1.5
37. McGuire UFSAR Section 9.1.2.3.2
38. MCC-1134.02-00-0043, Westinghouse CN-MRCDA-07-94, "Evaluation of a Closure Head Assembly Drop for McGuire Units 1 and 2, and Catawba Units 1 and 2"
39. MCC-1134-02-00-0034, "Reactor Vessel Head Drop Operating Floor"
40. NRC Enforcement Guidance Memorandum 07-006, "Enforcement Discretion for Heavy Load Handling Activities," dated September 28, 2007.
41. McGuire UFSAR Section 18.2.7
42. NEI "Industry Initiatives on Heavy Load Lifts," dated September 14, 2007
43. NEI 08-05 Revision 0, "Industry Initiative on Control of Heavy Loads" dated July 2008.
44. NRC letter and attached SER (Safety Evaluation Report) by the Office of Nuclear Reactor Regulation Related to Nuclear Energy Institute (NEI) 08-05, Revision 0 Industry Initiative on Control of Heavy Loads. Thomas C. Houghton from William H. Ruland, September 5, 2008.
45. MP/2/A/7150/057A, Rx Vessel Head Removal
46. MP/2/A/7150/057B, Rx Vessel Head Installation
47. MP/0/A/7700/096, Quarterly/Annual Inspection and Servicing of Overhead and Gantry Cranes

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9.2 Water Systems

9.2.1 Nuclear Service Water System

Note:

This section of the UFSAR contains information on the design bases and design criteria of this system/structure. Additional information that may assist the reader in understanding the system is contained in the design basis document for this system/structure.

9.2.1.1 Design Bases

The Nuclear Service Water (NSW) System provides assured cooling water for various Auxiliary Building and Reactor Building heat exchangers during all phases of station operation. Each unit has two redundant "essential headers" serving two trains of equipment necessary for safe station shutdown, and a "non-essential header" serving equipment not required for safe shutdown. In conjunction with the Ultimate Heat Sink, comprised of the Standby Nuclear Service Water Pond (SNSWP), the NSW System is designed to meet design flow rates and heads for normal station operation and also those flow rates and heads required for safe station shutdown under the following conditions while sustaining a single failure:

Normal shutdown or shutdown as the result of a postulated (single unit) Loss of Coolant Accident (LOCA), or

A LOCA on one unit with a controlled shutdown on the alternate unit concurrent with a loss-of-offsite power on both units and a seismic event (SSE), or

A seismic event (greater than OBE) causing loss of Lake Norman resulting in controlled shutdown on both units concurrent with a loss-of-offsite power on both units.

According to Regulatory Guide 1.27, the ultimate heat sink is required to provide cooling for a period of time to evaluate an accident scenario and to take necessary corrective actions. A period of 30 days is identified in this guidance to be adequate for these purposes; therefore, the mission time for the RN system is also considered to be 30 days.

The effects of low water level have been considered.

The NSW System is designed to be compatible with plant requirements as stated in Section 2.4.11.5 and dependability requirements as stated in Section 2.4.11.6. Environmental acceptance of effluents heated by the NSW heat exchangers is considered in Section 2.4.12.

Safety classifications and code requirements for major equipment in the NSW System are listed in Table 3-4. Table 9-13 lists water chemistry data for Lake Norman water supplied to the NSW System.

9.2.1.2 System Description

Normal Operation

The summary flow diagram for the Nuclear Service Water System with Channel A in normal operation is shown in Figure 9-31. Valves shared between units are shown in Table 9-9 and their operation is related to the safety of both units. The NSW System is made up of four sections, which when put together in series provide an assured source of water for all the station safety related water demands and some non-safety related demands. These sections are, in order of flow, the main supply section, the strainer/pump section, the heat exchanger section, and the main discharge section.

The Nuclear Service Water System is designed to meet single failure criteria, with two redundant channels per unit to serve components essential for safe station shutdown. Channel B components and piping provide 100% backup to channel A components. Engineered Safety Features provide for automatic valving and component actuation for both channels of the unit affected, while non-safety related components are isolated and shut off. Channel A and B crossover double valving is also closed as an Engineered Safety Feature, assuring channel integrity.

The four basic sections of the NSW System are discussed in the following paragraphs:

1. Main Supply Section

The main supply section of the NSW System includes the Low Level Intake Cooling Water System, the Condenser Circulating Water System (CCW), the Standby Nuclear Service Water Pond (SNSWP) and all piping and valves up to and including the channel supply isolation valves preceding the NSW Strainers.

Plans and profiles of the Low Level Nuclear Service Water pipes and the Standby Nuclear Service Water pipes are shown in Figure 9-43, Figure 9-44, and Figure 9-45.

a. As the normal source of water from Lake Norman, the single line from the Low Level Intake Cooling Water system provides feedwater to both channels of NSW pumps. Should any of the channel A redundant safety-related components malfunction, the corresponding unit's “A” NSW pump is shut down and the “B” NSW pump started, supplying the units' channel B heat exchangers. As an Engineered Safety Features, this low-level intake supply is automatically valved to provide feed to the A channels of both units following an S signal from either. Sufficient NPSH is available to meet NSW pump requirements at Lake Norman maximum drawdown of 745 ft. MSL and LOCA plus cooldown flow.

   Stainless steel mesh panels installed on inlet of the Low Level Intake (LLI) structure provide a macrofouling barrier that precludes introduction of fish and other debris into the Low Level Intake. The LLI structure supports concurrent LLI pump and nuclear service water pump operation (See Section 10.4.5). The LLI inlet average velocity with two LLI pumps and two nuclear service water pumps operating is less than 0.5 ft/sec. The LLI pumps are operated during specific periods during late summer and typically are not operated when fish are concentrated in the vicinity of the LLI. Fouling of the macrofouling barrier is precluded by LLI inlet low velocities and typical operation of the LLI pumps. Fish population behavior in the vicinity of the LLI is monitored periodically each summer.

b. A secondary source of water is available from the CCW supply cross-over. Due to the potential for introducing Asiatic clams into the system, CCW is not relied upon for normal operation or accident response functions performed by NSW. Alignment of this source requires deliberate Operator action to complete. The B channel of CCW supply has had all automatic isolation features removed. It is no longer an allowed configuration.

   On loss of instrument air, operating procedures manually align channel B of the nuclear service water system to the SNSWP. This suction source diversification provides additional protection against strainer fouling when automatic backwash is not available.

c. Two lines are provided from the SNSWP to meet single failure criteria should a seismic event cause loss of Cowans Ford Dam and resulting loss of Lake Norman. As an

Engineered Safety Feature, the Channel B SNSWP supply is automatically valved to provide feed to the B channels of both units following an S signal from either unit. The channel A SNSWP supply then acts as a 100% backup should any B component fail to function properly. Each channel is of sufficient size to provide total station flow for a unit LOCA and a unit cooldown.

See Table 9-9 for a tabulation of main supply and discharge valve positions for various modes of operation.

"The shared supply headers from the LLI and the SNSWP are designed with sufficient capacity to support both units for design basis accident flow requirements. General Design Criteria (GDC) 5 applies to this system. This design meets the requirements of GDC 5 for shared systems."

2. Strainer and Pump Section

   a. Strainers are of the automatic backwash type, and backwash discharge is automatically in service when the respective Nuclear Service Water pump is started. The normal backwash discharge pump flow path returns to the Nuclear Service Water return header, thus preventing any unnecessary loss of water from the SNSWP for all design basis events that require the use of the UHS. Backwash Supply is provided by Nuclear Service Water from the NSW pump discharge when a pressure drop across the strainer reaches a predetermined value. Safety related assured air supply is available to the strainer supply isolation valve to permit operation in the event of a Safety Injection Signal or a Loss of Instrument Air. The strainer drum motors and strainer backwash discharge pumps are powered by normal and emergency sources. The strainer function is in service during normal and accident conditions.

   b. Normally, only one pump per unit is in operation, as each pump meets system maximum flow requirements, but as an Engineered Safety Feature, all available pumps are automatically started upon S signal or loss of station and offsite power. Only one channel is necessary, so the redundant channel is maintained in standby. Each pump is supplied with power from separate normal and emergency sources. Emergency power is provided to each pump from its corresponding channel diesel generator, thereby assuring a continuous flow of water under all conditions. Each NSW pump motor receives cooling water from its corresponding NSW pump discharge at all times while that pump is in operation.

      See Table 9-8 and Table 9-10 for NSW equipment operating and design parameters.

      See Figure 1-3 for NSW pump location.

3. Heat Exchanger Section

   The heat exchanger section of the Nuclear Service Water System includes components both essential and non-essential for safe plant shutdown. Essential components are necessarily redundant and served by redundant NSW headers in each unit to meet single failure criteria. Non-essential components have no backup, and are served by the NSW pump in operation.

   a. The following components and emergency water supplies are essential for safe plant shutdown, so they are redundant for each unit and served by corresponding redundant channels of the Nuclear Service Water System. They are also designed for operation during and after seismic conditions.

      1) Coolers for:

a) Component Cooling Pump Motors

b) Centrifugal Charging Pump Motors

c) Safety Injection Pump Motors

d) Residual Heat Removal Pump Motors

e) Containment Spray Pump Motors

f) Fuel Pool Cooling Pump Motors

g) Nuclear Service Water Pump Motors

h) Auxiliary Feedwater Pump Motors

2) Containment Spray Heat Exchangers

3) Diesel Generator Heat Exchangers

4) Component Cooling Heat Exchangers

5) Centrifugal Charging Pump Bearing Oil Coolers

6) Centrifugal Charging Pump Gear Oil Coolers

7) Assured Auxiliary Feedwater Supplies

8) Assured Diesel Generator Cooling Supplies

9) Assured Fuel Pool Makeup Supplies

10) Assured Component Cooling Supplies

11) Safety Injection Pump Bearing Oil Coolers

12) VC/YC Chiller Condensers

b. Each channel of the NSW system provides assured auxiliary feedwater to the Auxiliary Feedwater System. Each motor driven AFW pump motor is cooled and supplied with suction from its corresponding channel of the NSW System. The steam turbine driven auxiliary feedwater pump is supplied from whichever channel of the NSW System is in operation. Nuclear Service Water is used for feedwater only when the normal condensate supplies for the Auxiliary Feedwater System, are unavailable. Automatic start signals are provided from each motor driven AFS pump to the respective NSW pump. See Figure 10-47 for a flow diagram of this system.

c. Water returning from the Containment Spray heat exchanger is monitored for radioactivity to detect tube leakage. Any radioactivity present sets off an alarm in the Control room, and immediately the changeover is made to the redundant channel.

d. The following components are not redundant since they are not essential for safe shutdown. Water is supplied to these components during normal operation, but they are automatically isolated from Nuclear Service Water supply and return after a Safety Injection Signal.

1) Reciprocating Charging Pump Bearing Oil Cooler

2) Reciprocating Charging Pump Fluid Drive Oil Cooler

3) The Auxiliary Building Ventilation Units

Water is supplied to the following components during normal operation, but they are automatically isolated from the Nuclear Service Water supply after a Safety Injection

Signal. The return lines are automatically isolated from the Nuclear Service Water After a Hi-Hi Containment Isolation Signal.

4) The Upper Containment Ventilation Units

5) The Lower Containment Ventilation Units

See Table 9-8 for flow rates required by each component during various modes of station operation.

e. The reactor coolant pump motor air coolers are not essential for safe shutdown, but are set up to receive cooling flow until the Containment, high-high pressure of 3 psig is received. Provisions are made for testing the Containment penetration valves for out-leakage. The Containment isolation valves are diaphragm valves which are capable of slight leakage under high differential pressure to accommodate thermal expansion during an accident.

4. Main Discharge Section

a. During normal operation with supply from the Low Level Intake Cooling Water System, water returns to Lake Norman via the NSW system CCW return line to the CCW crossover line in the Turbine Building.

b. When NSW requirements are being supplied from the channel B supply line from the SNSWP, return to the pond is accomplished by the channel B return line. Complete, 100% redundancy of safety related components and piping is provided by channel A heat exchangers, supply, and return piping to the SNSWP.

c. The shared discharge headers to Lake Norman and the SNSWP are designed with sufficient capacity to support both units for design basis accident flow requirements. General Design Criteria (GDC) 5 applies to this system. This design meets the requirements of GDC 5 for shared systems.

5. Crossover valving

At the interface of each section of the Nuclear Service Water System with the next section, there are crossover lines with double isolation valves. These are identified as follows:

a. Main supply crossover valves

b. Pump discharge header crossover valves

c. Main discharge crossover valves

These valves give the system added flexibility to operate should more than one malfunction occur. As an Engineered Safety Feature, the main supply and main discharge crossover valves have electric motor actuators that close upon a Safety Injection Signal. This assures channel isolation and properly aligned supply and return to the CCW crossover and SNSWP as outlined previously. An exception to the engineered safety features described occurs in the 0RN4AC and 0RN5B valves. As part of the response to an extended loss of AC power (ELAP), 0RN4AC has been left in the open position with all Engineered Safety Feature signals removed from the valve closure scheme. This ensures a continuous source of water for the turbine-driven AFW pump for Unit 2 for ELAP events. 0RN5B is locked closed and de-energized to maintain separation of the NSW and CCW systems. This configuration satisfies ELAP and SSS requirements for AFW sources while preventing the use of the CCW source for RN without deliberate action.

The RN pump discharge crossover valves have the capability to cross connect the Essential headers. However, in accordance with GDC 5 requirements and the MNS design basis, the RN pump discharge crossover valves may not be used to share RN trains between units for a Loss of Service Water event.

A fourth crossover ties the channel A and channel B pump discharge headers together to supply the nonessential header. The crossover valves are left open normally to pressurize both essential channels with one NSW Pump. Channel B is isolated from the crossover, after a Safety Injection Signal or loss of offsite power, by double isolation valves. Channel A is isolated from the nonessential header on a safety injection signal but continues to supply the reactor coolant pump motor coolers until it is isolated from the crossover after a Phase B Containment Isolation signal. Channel A continues to supply the nonessential header following a loss of offsite power.

Startup

The Nuclear Service Water System is a water solid system and should be allowed to fill by normal supply and return head before the NSW Pump is started and circulation is initiated. Each component has local vents and drains as needed and the system high point vents are located atop the containment spray heat exchangers. Standby Pond supply and return piping has high point vents to allow filling by SNSWP head. The low-level cooling water intake pipe will also fill without vacuum connections since its highest point is still below Lake Norman maximum drawdown elevation and is vented to relieve airlock. All manual control valves are properly set to maintain proper flow with the system in LOCA operational mode.

Normal Station Cooldown

During a one unit cooldown phase following initiation of normal unit shutdown, two nuclear service water pumps per unit are required to be in operation, and full flow through both component cooling heat exchangers is required for normal cooldown capability. The station can be safely cooled down by one component cooling heat exchanger at reduced flow but this will require considerably more time.

Cold Shutdown

During periods of unit cold shutdown for maintenance and refueling following the unit cooldown phase, one train of the system remains in operation. Flow through one component cooling heat exchanger is required at all times during station shutdown for removal of residual heat from the Reactor Coolant System, the spent fuel pool, and various other non-essential equipment.

Engineered Safety Features Actuation

1. A Containment high pressure signal (S signal) causes actuation of ESF equipment as a result of postulated accident conditions and causes automatic startup of both emergency diesel generators for the unit so affected. The safety injection signal causes the following to occur:

   a. Automatic startup of the redundant channel of essential equipment.

   b. Automatic separation of the essential trains and closing off of the Auxiliary Building branch of the non-essential header and Auxiliary Building ventilation cooling water supplies and returns.

   c. Initiation of water flow to the emergency diesel generator heat exchangers and all ES air handling fan coil units by automatic opening of “S” signal activated valves.

   d. Separation of seismic Category I piping from non-seismic CCW supply piping and alignment of channel A supplies from low level supply and channel B supply from the Standby Nuclear Service Water Pond.

e. Upon an ESF (S_s) signal, the NSW System Train A will be aligned to the Low Level Intake supply for higher NPSH availability, and Train B will be aligned to the SNSWP. The redundant train of the affected unit's NSW system starts automatically. Redundant isolation valves, located on the crossover of the NSW pump discharge headers, are automatically closed to assure separation between the affected unit's NSW trains and to provide isolation between the SNSWP and Lake Norman. This alignment protects the Train B NSW pump from cavitation damage due to low NPSH.

2. A Containment high-high pressure signal (P signal) causes the following to occur:

   a. Isolation of the reactor coolant pump motor cooler supply and return lines at the penetration.

   b. Isolation supply and returns for Containment upper and lower compartment ventilation system.

<u>Loss of Offsite Power (Blackout)</u>

Blackout causes startup of all emergency diesel generators and system alignment as described under <u>Engineered Safety Features Actuation</u> for a containment high pressure signal with the exception that the Auxiliary Building branch of the non-essential header is not isolated.

Deleted Per 2011 Update.

<u>Loss of Nuclear Service Water on a Single Unit</u>

McGuire Nuclear Station provided a response to Generic Letter 91-13 "Esstential Service Water System Failures at multi-Unit Sites" on February 27, 1992. In this response, several mitigation strategies were identified, but only two are currently approved and in the Current Licensing Bases. These are (i) aligning the Containment Ventilation Cooling Water (RV) System to provide cooling water backup through the RN non-essential header, or (ii) activating the SSF standby makeup pump to provide NC pump seal cooling. The NRC review of this Generic Letter was provided on July 15, 1992.

9.2.1.3 Safety Evaluation

The Nuclear Service Water System is designed to withstand a safe shutdown earthquake and to prevent any single failure from curtailing normal station operation or limiting the ability of the engineered safety features to perform their functions. The normal source of water from the low level intake is not qualified to a Safe Shutdown Earthquake (SSE) but is qualified to an Operating Basis Earthquake (OBE). Operating procedures manually align both channels of the nuclear service water system to the SNSWP if an OBE is exceeded. Sufficient pump capacity is included to provide design cooling water flow under all conditions, and the headers are arranged in such a way that loss of a header does not jeopardize unit safety. Radiation monitors are located in the system for detection of potentially radioactive leaks. The system is designed to operate at either maximum drawdown of the lake or pond and also at a maximum water elevation.

The Nuclear Service Water System is designed to tolerate a loss of offsite power during: 1) a LOCA on one unit with a controlled shutdown on the alternate unit concurrent with loss of Lake Norman and a seismic event (SSE) or; 2) a controlled cooldown on both units concurrent with loss of Lake Norman and a seismic event (SSE). By adhering to channel A and B separation with a double valved main supply crossover, both units are assured of having a source of water, two 100 percent capacity pumps, and two redundant trains of heat exchangers essential for safe shutdown. Channels A and B are connected together only at 4 places by crossover piping, and in these cases double isolation valves, actuated automatically or normally locked closed, protect channel integrity, and meet single failure criteria.

Shared valves beginning with "0RN" listed in Table 9-9 are provided with normal and emergency diesel power from Unit 1 and Unit 2. The A channel shared valves are normally aligned to Unit 1 A channel diesel and the B channel shared valves are normally aligned to the Unit 2 B channel diesel. If a diesel on one unit is out of service, the shared valves normally powered from that channel are provided with manual switchover to the other unit diesel of corresponding channel. In this manner, any one diesel generator can be out of service and the RN System can still perform its safety functions. Shared valve 1RN1 is normally open with the associated breaker also open to prevent inadvertent actuation.

Deleted Per 2015 Update.

For a detailed failure analysis, see Table 9-11.

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A flood analysis was performed as required by 10CFR50. Flood analysis criteria were also established in Regulatory Guide 1.59. Exceptions were taken to Regulatory Guide 1.59 and described in Section 3.4.

The RN system piping that penetrates the Auxiliary Building below ground level is encased in concrete and does not require water seals. The Probable Maximum Flood (PMF) does not have any detrimental effect on the SNSWP. Also, the portion of the RN system located within the Auxiliary and Reactor Buildings is protected from the effects of flooding. Internal flooding due to failure of nonseismic piping is also mitigated.

Cowans Ford Dam, which impounds Lake Norman, was checked for a combination of the less severe natural phenomena of one-half DBE and the 100 year flood (discussed in Section 2H.3).

9.2.1.4 Tests and Inspections

All system components are hydrostatically tested and full operational tests are performed before station operation. During normal operation, periodic tests and inspections are performed per the requirements of the plant Technical Specifications, Inservice Inspection Program, and Inservice Testing Program to ensure operability of the Nuclear Service Water System.

Containment penetration isolation valves in the supply and return line for the Reactor Building header are tested during refueling operations. Test vents and drains along with other associated tests valves are provided so the line can be drained of water on both sides of each isolation valve and tested for leakage.

Deleted paragraph(s) per 2002 revision.

9.2.1.5 System Instrumentation and Control

See Table 9-12 for a listing of instrumentation by type.

The instrumentation listed in Table 9-12 serves to inform the plant operator as to the operation and status of the Nuclear Service Water System. Detection of leakage from the system is accomplished by using the NSW instrumentation in conjunction with the floor drain tank level alarm, and the sump level alarms in all the various locations in the Auxiliary Building. For instance, if the floor drain tank high level alarm sounds, the operator will be alerted to a leak condition in some system. The high-high level alarm would indicate a major leak. Nuclear Service Water main channel supply header low pressure would indicate a possible leak from this system. The location and extent of the leak can be determined by low flow alarms, motor high temperature alarms, and high level alarms in the appropriate sump. Upon such condition, the redundant channel of equipment will be actuated and the leaking component isolated for repair. No automatic channel changeover is provided other than automatic actuation of all safety equipment upon Engineered Safety Features signals.

9.2.1.6 Corrosion and Fouling in the RN System

The issues related to corrosion and fouling in the RN system were addressed in McGuire's initial response, and in subsequent supplemental responses, to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," issued on July 18, 1989. Generic Letter 89-13 required licensees to address biofouling or accumulations of silt, corrosion products, or other debris by instituting NRC recommendations or acceptable substitute actions.

In the original design of the RN System, no provisions were made for minimizing corrosion or corrosion product fouling. The 3/16-inch openings of the strainers on the suction side of the RN pumps helps to reduce the impact of corrosion product fouling on downstream components. However, due to problems with occlusion, high pressure drop in piping, localized pitting, and corrosion product fouling of strainers and heat exchangers, minimizing corrosion in the RN system has become an important issue. As the plant has aged, corrosion product detachment from RN system pipe walls has resulted in additional and more frequent flushing and/or cleaning of RN system supply lines to ensure operational readiness. Minimizing the effects of corrosion in the RN system is now being accomplished through material replacement and mechanical pipe cleaning.

Control of macro-biological fouling, primarily Asiatic clams and fish, is achieved by a series of design features and operating procedures. The normal operating alignment of the RN System takes suction from the bottom of the lake, at the Low Level Intake (LLI), as does the ESF alignment for A Train RN. Macro-biological fouling control, primarily for fish, is achieved at the LLI through a macrofouling barrier consisting of 3/4" X 3/4" stainless steel mesh panels installed on the inlet of the LLI-structure (Section 9.2.1.2). The water temperature at the LLI is normally less than the temperature for clam spawning and livelihood, and therefore, Asiatic clams are not likely to be introduced into the RN system via the LLI. The ESF alignment of the B Train RN System takes suction from the bottom of the Standby Nuclear Service Water Pond (SNSWP), which goes anoxic much earlier than the lake and also has temperatures normally less than the temperature for clam spawning and livelihood. These anoxic conditions, in combination with low temperatures in the SNSWP, inhibit clams from being introduced into the RN system from the SNSWP. Fish intrusion at the SNSWP is controlled through monitoring surveys and chemical treatment. A third potential alignment of the RN System, from the RC system, does have the potential to introduce clam larvae into the RN System; however, this alignment is not normally used, and is controlled by operating procedures. Finally, the water supply from any of these three potential alignments passes through a strainer with 3/16-inch openings, on the suction of each RN pump, which helps to reduce the impact of macro-biological fouling. Environmental monitoring for both clams and fish at the intake structures provides early detection of potential macro-biological fouling concerns in the RN system.

Due to their smaller size, macrofouling barriers and strainers are not effective for control of micro-biological and particulate fouling, which primarily affect heat exchangers. Control of micro-biological fouling is achieved by flushing and mechanical and chemical cleaning, as determined by performance monitoring. Control of particulate fouling, primarily mud, silt, and oxide precipitants, is achieved by coatings, wet lay-up, flushing, and mechanical cleaning, as determined by performance monitoring. It is also controlled by operating at flow rates which prevent particles from settling out on heat exchanger tubes. In addition, oxide precipitant fouling is controlled by inspection of the LLI structure seals to prevent seasonal in-leakage of oxygenated water.

As a result of increasing concerns over corrosion and fouling, and in response to NRC Generic Letter 89-13, McGuire has instituted a program of performance monitoring, consisting of flushing and flow testing, heat exchanger testing, material replacement, coatings, wet lay-up, mechanical and chemical cleaning, system operational changes, and inspection and maintenance of the RN system. This program minimizes the impact of corrosion and fouling in the RN system.

9.2.2 Component Cooling System

9.2.2.1 Design Bases

The Component Cooling System is designed to:

1. Remove residual and sensible heat from the Reactor Coolant System, via the Residual Heat Removal System, during station shutdown.
2. Cool the letdown flow to the Chemical and Volume Control System during power operation.
3. Cool the spent fuel pool water.
4. Provide cooling to dissipate waste heat from various primary station components during normal operation and under accident conditions.

Active system components considered vital to the cooling function are redundant. Such redundancy of components does not degrade the performance or reliability of any system served by the Component Cooling System. Any single passive failure in this system does not prevent the system from performing its design function.

The design provides means for the detection of radioactivity entering the system from the Reactor Coolant System and its associated auxiliary systems, and includes provision for isolation of system components.

9.2.2.2 System Description

The Component Cooling System normally functions as two independent subsystems. A summary flow diagram for the Component Cooling System is shown in Figure 9-57. The Component Cooling System consists of eight (four per unit) component cooling pumps, four (two per unit) component cooling heat exchangers, two (one per unit) split volume surge tanks, one drain tank, one drain tank pump and associated valves, piping, and instrumentation.

The equipment receiving cooling flow is arranged in parallel circuits with the component cooling pumps and component cooling heat exchangers placed at a common supply and return location for the various cooling circuits. The surge tanks are connected to the suction piping of the component cooling pumps, and are placed at the highest points in the system in order to facilitate easy filling and venting of the system.

Cooling water is normally available to all components served by the system, even though one or more of these components may be individually isolated. Valves actuated by an Engineered Safety Features signal

are used to provide the residual heat removal heat exchangers with cooling water should it become necessary to place these components in service under loss-of-coolant accident conditions.

The four component cooling heat exchangers are of the shell and straight tube type. Raw river water from the Nuclear Service Water System (Section 9.2.1) is circulated through the straight tubes.

The heat exchangers are placed at the discharge of the component cooling pumps in order to provide a higher component cooling water pressure, relative to service water pressure, for the purpose of preventing in-leakage of nuclear service water.

The eight component cooling pumps are horizontal, centrifugal units. These pumps receive electric power from normal or emergency sources. Mechanical seals are provided to minimize leakage.

Two split-volume surge tanks, one per unit, accommodate expansion, contraction, and in-leakage of water and in the event of a leak in the system, assures a continuous supply of component cooling water until the leak can be isolated. The normal supply of makeup water is provided from demineralized water storage tanks through the make up supply header connected to the surge tank. An assured supply of makeup water is available from the Nuclear Service Water System. Because the tanks are normally vented to the atmosphere, a radiation monitor is provided at the discharge of each component cooling heat exchanger. The monitor actuates an alarm if the radiation reaches a preset level above the normal background and sends a signal to close the surge tank vent isolation valve. A major portion of the component cooling system material is carbon steel. Therefore, corrosion is controlled by chemical addition to the surge tanks.

The component cooling drain tank and component cooling drain tank pump are located at the lowest point in the system. All equipment drains and low point drains are piped to the drain tank and then pumped to the appropriate component cooling surge tank, thus minimizing makeup and waste treatment problems associated with the chemically treated component cooling water. The drain tank, drain tank pump, surge tanks, and all drain piping and associated valves are stainless steel.

All other piping in the Component Cooling System is carbon steel. Welded joints and connections are used except at components which require removal for maintenance. All line valves two inches or less in size are of packless stem or diaphragm design, or have stem leak off lines to further reduce leakage. Self-actuated, spring-loaded relief valves are provided for lines and components that could be pressurized to their design pressure by improper operation or malfunction.

During normal station operation, two pumps per unit and one heat exchanger per unit provide the necessary cooling requirements. A minimum of two pumps and one heat exchanger are adequate for cooldown of one unit. This equipment is also adequate during refueling of one unit. In the event of a LOCA, two pumps per unit and one heat exchanger per unit are capable of fulfilling system requirements. The remaining two pumps and one heat exchanger per unit serve as a backup system.

Design data of system components are listed in Table 9-21. System safety class requirements are presented in Table 3-4.

Component cooling is provided for the following heat sources:

1. residual heat removal heat exchangers,
2. fuel pool cooling heat exchangers,
3. letdown heat exchanger,
4. excess letdown heat exchanger,

5. seal water heat exchanger,
6. reactor coolant pump motor bearings and thermal barriers,
7. boron recycle evaporator condenser,
8. boron recycle evaporator vent condenser,
9. boron recycle evaporator distillate cooler,
10. waste evaporator condenser,
11. waste evaporator vent condenser,
12. waste gas hydrogen recombiners,
13. sample heat exchangers,
14. residual heat removal pump mechanical seal heat exchanger,
15. reactor coolant drain tank heat exchanger,
16. recycle evaporator concentrates pump seal cooling water heat exchanger, and
17. waste evaporate concentrates pump seal cooling water heat exchanger.

Nominal flow rates during various unit operating modes are tabulated in Table 9-22. Typical valve lineups for various system operational modes are shown in Table 9-23.

9.2.2.3 Safety Evaluation

Most of the equipment, piping, and instrumentation associated with the Component Cooling System is located outside Containment and, therefore, is available for inspection and maintenance during power operation. Repair of a pump or heat exchanger can be performed while the other components are in service.

Sufficient cooling capacity is provided to fulfill all system requirements under normal and accident conditions. Adequate safety margins are included in the size and number of components to preclude the possibility of a component malfunction adversely affecting operation of safety features equipment. Active system components considered vital to the operation of the system are redundant. Also, any single passive failure in the system does not prevent the system from performing its design function.

In consideration of single failure criteria, the Component Cooling System contains separate flow paths to the two trains of Engineered Safety Features equipment. Any pipes connecting the separate flow paths contain isolation valves in series.

The Nuclear Service Water System (Section 9.2.1) provides an assured source of cooling water to the component cooling heat exchangers. The Component Cooling System serves as an intermediate system and a second boundary between the Reactor Coolant and Auxiliary Systems and Nuclear Service Water System and assures that any leakage of radioactive fluid from the components being cooled is contained within the station. Radiation monitors are placed in the discharge lines of the component cooling heat exchangers to detect any radioactive leaks into the Component Cooling System. To minimize the possibility of leakage from piping, valves, and equipment, welded construction is used whenever practical. Further instrumentation to detect both inleakage and outleakage is presented in Section 9.2.2.5. Normal makeup to the system is provided by the Makeup Demineralized Water System as indicated on Figure 9-57. An assured supply of makeup water is available from the Nuclear Service Water System.

A relief valve on the component cooling water return header downstream of the reactor coolant pumps is designed with a sufficient capacity to envelope the maximum rate at which reactor coolant can enter the Component Cooling System from a credible break of the reactor coolant pump thermal barrier cooling coil. The relief valves on the cooling water lines downstream of other heat exchangers in the system are sized to relieve the thermal expansion occurring if the exchanger shell side is isolated and high temperature fluid continues to flow through the tube side. Actuation pressure for these relief valves equals the design pressure of the component cooling piping.

The relief valves on the component cooling surge tanks are sized to relieve the maximum flow rate of water which could possibly enter the surge tank following a rupture of a reactor coolant pump thermal barrier cooling coil. The actuation pressure equals the design pressure of the surge tanks.

The surge tank has sufficient capacity to provide a continuous supply of component cooling water for 30 minutes during a leak of 50 gpm. This is sufficient time for operating personnel to isolate the leaking component. The pumps are able to operate with surge tanks empty, but the operator manually provides makeup upon low level indication.

The Component Cooling System does not service any Engineered Safety Feature inside Containment. Upon a safety injection signal, Component Cooling flow to the reactor coolant drain tank heat exchanger and excess letdown heat exchanger is isolated along with the reactor building Component Cooling drain lines. Flow to the reactor coolant pump motor coolers is continued, however, until a Phase B Containment Isolation signal is received.

Active and passive failure analyses of pumps, heat exchangers, valves and piping are presented in Table 9-24. The Safety Classes of major system components are listed in Table 3-4.

9.2.2.4 Tests and Inspections

System components are hydrostatically tested, and full operational tests are performed before unit operation. Active components of the Component Cooling System are in either continuous or frequent use during normal station operation and are tested per the McGuire In-Service Testing Program. Containment isolation valves are tested periodically in accordance with Technical Specifications. Periodic visual inspections and preventive maintenance are conducted according to good industrial practice.

9.2.2.5 Instrumentation Application

The operation of the system is monitored with the following instrumentation:

1. temperature detectors at the discharge of each component cooling heat exchanger,
2. pressure detectors at the discharge of each component cooling pump,
3. flow detectors in the main inlet lines of the component cooling heat exchangers,
4. flow detectors in the main inlet or outlet lines of the component serviced by the system,
5. a radiation monitor in the outlet line of each component cooling heat exchanger,
6. a level indicator and alarms in each compartment of each surge tank, and
7. temperature and pressure test points at strategic points in the system.

9.2.3 Treated Water Systems

9.2.3.1 Design Bases

The treated water systems are designed to provide the following:

1. drinking and sanitary water requirements,
2. demineralized water makeup requirements.

9.2.3.2 System Description

The treated water systems of the following: (See Figure 9-66 through Figure 9-67).

1. Drinking Water Systems, and
2. Makeup Demineralized Water System.

Lake water is supplied to the Water Treatment Building from the Conventional LP Service Water System (see Figure 9-54).

As system demands dictate, water is pumped from the Water Treatment Building to two 42,500 gallon storage tanks located on the Service Building roof. Demineralized water is supplied from the storage tanks to the Makeup Demineralized Water Systems.

The Drinking Water System is supplied by Charlotte Municipal Utilities Department.

The Makeup Demineralized Water System is supplied by demineralized water from vendor supplied water treatment equipment located in the Water Treatment Building providing high purity demineralized makeup water to the station primary and secondary systems.

9.2.3.3 Safety Evaluation

The treated water systems are shared systems that perform no safety functions. The Systems supply the station with drinking, and demineralized water in addition to providing condensate makeup capability. The demineralized water supply header is the only connection between the treated water systems and other systems which may contain radioactive material. Backflow into the supply header is normally prevented by the header being at a higher pressure than the receiving source. Should header pressure decrease and the booster pumps fail to operate, check valves are strategically located to prevent backflow into the demineralized water system, respectively.

9.2.3.4 Tests and Inspections

Initial operation acceptance tests are performed to assure components meet design capacity and quality. Normal operation performance monitoring indicates deviation of any component from design operating conditions and the deviation is corrected by appropriate methods as necessary.

9.2.3.5 Instrumentation Application

Instrumentation is provided to monitor treated water systems process water flow and quality to assure proper operation and alarm any significant malfunction. Sufficient instrumentation is provided to assure that storage tank levels and/or pressure are maintained.

9.2.4 Deleted Per 2009 Update

9.2.5 Ultimate Heat Sink

Note:

This section of the UFSAR contains information on the design bases and design criteria of this system/structure. Additional information that may assist the reader in understanding the system is contained in the design basis document for this system/structure.

9.2.5.1 Design Bases

Two independent sources of nuclear service water are available to provide a normal supply of cooling water: Lake Norman, via the Low Level Intake (LLI) and the Standby Nuclear Service Water Pond (SNSWP). However, to dissipate the waste heat rejected during a unit LOCA plus a unit cooldown the Standby Nuclear Service Water Pond is the only source qualified as the ultimate heat sink. These two water sources are separated and protected such that failure of one does not induce failure of the other. Lake Norman is the normal source of nuclear service water. The emergency source is the Standby Nuclear Service Water Pond (SNSWP).

The Nuclear Service Water System is designed to operate properly within a specific temperature range. Calculations are performed to verify that the intake water is within that range, and that there is a sufficient quantity of water to supply the plant for a 30-day period.

The SNSWP provides an assured supply of makeup water to the auxiliary feedwater, component cooling water, EDG diesel engine cooling water, and spent fuel pool cooling systems when aligned as the cooling water source for the Nuclear Service Water System.

9.2.5.2 System Description

The Nuclear Service Water System is described in Section 9.2.1. During normal operation Lake Norman is the preferred source of nuclear service water and also dissipates the waste heat from the discharge. A lake surface elevation of approximately 745 ft. or greater is required for operation of the system. If Lake Norman falls below this elevation the SNSWP can be used to supply water and dissipate the waste heat during shutdown and/or LOCA. Plans and profiles of the Low Level Nuclear Service Water pipes and the Standby Nuclear Service Water pipes are shown in Figure 9-43, Figure 9-44, and Figure 9-45.

At minimum pond elevation of 738 ft. msl the SNSWP has a surface area of approximately 30.5 acres, as shown in Figure 9-42.

Effects of flood waters on the SNSWP are discussed in Section 2.4.10.

9.2.5.3 Design Evaluation

9.2.5.3.1 Analytical Model

For analytical purposes the code to calculate temperatures in the McGuire Standby Nuclear Service Water Pond (SNSWP) employs two basic assumptions. First, the heated water loses heat to the atmosphere at a rate proportional to its excess above the equilibrium temperature. Second, the pond can be treated as a stack of horizontal layers of water, each with its own temperature. However, heat loss is assumed to occur only at the surface.

The operation of the SNSWP is simulated in the computer model as follows:

1. water is withdrawn from the bottom layer of the pond, heated up, and discharged to the pond surface;
2. the warm buoyant surface water undergoes surface cooling for the time period ts = Vu/Q where Vu is the unit volume of the water layer and Q is the McGuire flow through rate;
3. each of the stacked horizontal layers of water are shifted down one layer retaining their previously defined temperatures;
4. checks for density instabilities between water layers are made and mixing is affected where necessary;
5. the complete process is repeated to simulate a 30 day period.

Since the intake is at the bottom of the pond and the discharge is on the surface, the low velocities and the density difference between the intake and discharge prevent hydraulic short circuiting. Harleman and Elder (Reference 6) analyzed the potential for pulling an upper warm buoyant layer down into a submerged intake. If a computed depth H is less than the depth from the warm interface to the bottom of the intake, then less than 5% of the intake will be water pulled down from the surface.

1. $$H = \frac{3}{2}(\left[\frac{(Q/B)^2}{g'}\right])^{\frac{1}{3}}$$

where:

H=computed depth of water above lowest point of intake opening, ft.

Q=intake flow, cfs

B=width of intake opening, ft

g'=g Δd/d, ft/sec^2

g=acceleration of gravity, 32.17 ft/sec^2

d=density of bottom water layer, slug/ft^3

Δd=density difference between surface and bottom water layers, slug/ft^3

Assuming a peak plant intake temperature of 102°F and a temperature difference between the intake and surface layers of 8°F, this yields a heated surface layer temperature of 110°F:

d=1.923 slug/ft^3, ASME Steam Tables, 1967

Δd=0.003 slug/ft^3, ASME Steam Tables, 1967

g'=0.05

Using B=28 ft and g'=0.05 with 56,000 gpm flowrate a withdrawal depth of H=11 ft can be calculated. The minimum depth of the SNSWP surface (738 ft msl) to the intake structure opening (702.5 ft msl) is 35.5 feet. Thus, significant recirculation of the heated surface layer will not occur until the depth of the heated layer exceeds 24.5 feet.

Supporting information can be found in Ref 14. A listing of the computer program used to calculate the thermal behavior of the McGuire SNSWP is presented in Table 9-17. A list of the inputs and variable for the program with explanations is provided in Table 9-18. A sample run of the program is provided in Table 9-19. These results are from the computer model with the input data as presented in the listing.

The program simulates the pond by taking water from the bottom of the pond at a constant rate, heating it, and discharging it onto the surface with a temperature of T_{dis}. Surface heat transfer cools it to temperature T_{sfc} using the following equation:

$$T_{sfc} = (T_{dis} - E)e^{\frac{-Kts}{\rho hCp}} + E$$

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where:

E=equilibrium temperature (°F)

K=heat transfer coefficient (BTU/(ft^{2}hour °F))

h=depth of cooling layer (top layer), (ft)

t_s=length of cooling time period for the surface layer, (hour)

Cp=specific heat at constant pressure (BTU/lbm°F)

ρ=density (lbm/ft^3)

If the temperature of a layer is less than the temperature of the layer below it, then the layers are mixed since a cooler, more dense surface layer would be unstable. Then the new bottom layer is withdrawn and treated in the same manner.

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The meteorological data from 1951 to 1972 was scanned in order to find the highest monthly average dewpoint (71°F in July 1961), and the lowest average wind speed (4.7 mph in August 1953). These values were then used to calculate the heat exchange coefficient K. The heat exchange coefficient is the net rate at which heat is lost or gained by a body of water, is defined by using the following equation (Ref 1):

K = 15.7 + (0.26 + B) f(w), BTU/FT^2/Day/F

where:

K = Heat Exchanger Coefficient,

B = Vapor Pressure slope,

f(w) = Wind Speed function.

The wind speed function used in this analysis is known as te Meyer equation (Ref 2), and is defined as follows:

f(w) = 73 + 7.3w, BTU/FT^2/DAY/mmHg

where:

w = Wind Speed, mph

The combination of low wind speed (4.7 mph) and high dew point temperature (71°F) results in the lowest expected monthly average exchange coefficient. Assuming a water surface temperature of 95°F and using the method and meteorological parameters presented above the heat exchange coefficient is calculated to be 146 BTU/FT^2/DAY/F. A value of 150 BTU/ FT^2/DAY/F is used in the model.

The equilibrium temperature, F, is given by Brady, Graves, and Geyer (Ref 5) as:

$E = T_{dp} = + R/K$, °F

where:

T_{dp} = Dew Point Temperature

R = Short Wave Solar Radiation

K = Heat Exchange Coefficient

The highest monthly average solar radiation recorded at Greensboro/Highpoint Airport over a twenty year record (1951-1970) is 2500 BTU FT^2/DAY/F. Using the highest summer monthly average dew point temperature of 71°F and the lowest expected summertime heat exchange coefficient of 150 BTU/FT^2/DAY/F previously computed, a worst case monthly average equilibrium temperature of 88°F is computed. This value is used in the computer model.

The same method as used for the 1951-1975 data was used to scan meteorological data for the period January 1976 to December 2012 in order to determine whether the period contained a monthly average equilibrium temperature higher than the original 88°F. The highest equilibrium temperature was found to be 84.2°F for July 1993, therefore the original equilibrium temperature of 88°F for June 1952 remains bounding.

Recognizing that overall heat exchange is a function of the heat exchange coefficient as well as the equilibrium temperature, the methodology previously used for the worst case 30-day period was replicated and used to calculate equilibrium temperature and heat exchange coefficient for 1976-2012 months for which the meteorological data fell outside the original parameters. Specifically the months where wind speed was less than 4.7 mph, dewpoint temperature was greater than 71°F, or short wave solar radiation was greater than 2500 BTU/ft^2/day. The source data was the monthly average meteorological data used for the equilibrium temperature scan, with the period of interest limited to the hottest summer months (June, July and August). There were no periods in which short wave solar radiation exceeded 678 Langleys/day, one period in which dewpoint temperature exceeded 71°F, and several periods in which wind speed was less than 4.7mph.

In all cases, equilibrium temperature was less than 88°F and heat exchange coefficient was greater than 150 BTU/ft^2/°F/day with the exception of August 2003. The heat exchange coefficient for August 2003 was 148 BTU/ft^2/°F/day, but the corresponding equilibrium temperature was 80.5°F. In terms of overall heat exchange, the relatively low equilibrium temperature more than compensates for the shortfall in the heat exchange coefficient. Therefore it is concluded that the 88°F equilibrium temperature and 150 BTU/ft^2/°F/day used in the Edinger-Geyer equation for heat transfer bound all combinations of equilibrium temperature and heat exchange coefficient for the period 1976-2012

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Heat input to the standby pond was calculated using heat rejected from one unit LOCA plus one unit cooldown with simultaneous station blackout.

The total heat rejected (see Figure 9-41 and Table 9-16 for the 30 day period) comes from three basic sources: Supporting information can be found in Ref 15.

1. Containment spray plus residual heat removal heat exchanger heat load following one unit LOCA (See Figure 9-39). Together, these heat exchangers remove the residual and sensible heat from the LOCA unit.
2. Residual and sensible heat from one unit cooldown (See Figure 9-40). Heat rejection from this source begins approximately four hours after initiation of shutdown when steam dump to atmosphere is terminated.
3. Heat rejected to cooling water from all station auxiliary equipment in operation, including:
   a. Diesel generator cooling water heat exchangers
   b. Control Room air conditioning condenser
   c. Pump and motor air and oil coolers

The computer model predicts a maximum value of less than 102°F water supplied from the pond to the station and is in agreement with the maximum allowable temperature for all safety related components. The flow rate selected for heat exchangers represents flow required for maximum service water temperature at maximum heat load on the major heat exchangers.

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The Nuclear Service Water Pond full level is at elevation 740 ft. MSL. See Figure 9-42 for the area-capacity relationship. Level is alarmed at elevation 739.75 ft. during normal station operation. Makeup is provided by aligning the operating or idle NSW train to discharge to the pond to maintain a minimum pond level of 739.5 ft. Should Lake Norman be lost during filling, the check valve on the pump discharge prevents siphoning of the SNSWP. All system components and piping are located in temperature controlled areas or below the frost line to protect against freezing.

The SNSWP is not used as a fire water source with level below 739.5 ft, and all NSW pipes and equipment are maintained full of water, so no drawdown occurs due to these considerations. Drawdown for use as auxiliary feedwater (900 GPM/Unit for six hours) is negligible. The initial pond elevation in the computer model simulation is assumed to be 738 ft msl. The removal of 1.5 ft depth of pond inventory at the beginning of the simulation is a conservative approach since the combination of the inventory losses:

1. evaporation losses – 34.16 ac-ft
2. system losses – 5.16 ac-ft
3. seepage losses – 2.65 ac-ft (through dam)
4. system makeup – 1.5 ac-ft (Aux Feedwater)

total 43.47 ac-ft which is less than the volume of 45 ac-ft assumed in the thermal analysis. Supporting information can be found in Ref 14.

The temperature calculated by the computer program show that the SNSWP is adequate to supply all heat exchangers with cooling water below 102°F, which is an acceptable temperature to meet their capabilities at all times during shutdown. Heat transfer calculations were performed at points of maximum heat loads and points of maximum service water temperature (which do not occur simultaneously) to show the large heat exchangers are capable of operating within their design conditions. All small essential heat exchangers are designed for a conservatively high service water temperature based on this SNSWP analysis.

This design meets the requirements of General Design Criteria (GDC) 5 for shared systems.

Twenty-two years (1951-1972) of daily average meteorological records from the Charlotte, North Carolina airport were scanned by computer to determine the most severe 1, 4, and 30-day periods for surface cooling and maximum evaporation conditions. The most severe period for surface cooling is defined as the period giving the highest calculated equilibrium temperature as described above. The most severe period for evaporation is defined as the period with the maximum difference between dry bulb and dew point temperature, as recommended in Regulatory Guide 1.27, Revision 1.

Tabulations of the most severe 1, 4, and 30-day periods for surface cooling and evaporation are presented in Tables 9-14 and 9-15, respectively. Note that the solar radiation values are from Greensboro, North Carolina, the nearest such recording station. Tables 9-14 and 9-15 were generated for historical purposes. This is an exception to Regulatory Guide 1.27 and these tables were not used in the thermal analysis.

The Ultimate Heat Sink meets all of the requirements set forth in Regulatory Guide 1.27 with the exception that a plant-specific computer model using worst case meteorological constants in lieu of the recommended methods in Regulatory Position C.1. The SNSWP Dam was designed for the individual occurrences of the probable maximum flood and the design basis earthquake, as discussed in Section 2G.3 of former Appendix 2G and complies with Regulatory Guide 1.27. Cowans Ford Dam, which impounds Lake Norman, was checked for a combination of the less severe natural phenomena of one-half DBE and the 100 year flood (discussed in Section 2H.3). The design of the SNSWP Dam also takes into account a possible failure of Cowans Ford Dam and subsequent loss of Lake Norman. No known site related events have occurred or are expected to occur during the plant lifetime which would affect the SNSWP or dam and were therefore not considered in the SNSWP Dam design. Since the SNSWP and dam have been designed to withstand all design basis natural phenomena, no mechanistic passive single failure resulting from such phenomena is postulated. Supporting information can be found in Ref 13. Two 100% capacity intake and discharge pipes route water to the auxiliary building and return it to the SNSWP in compliance with paragraph C-3 of Regulatory Guide 1.27. These pipes are separated and protected from missiles so that failure of one does not induce failure of the other. The SNSWP intake structure itself is designed for design basis earthquake (DBE), wind, and missiles (discussed in Appendix 2G).

The Standby Nuclear Service Water Pond Intake Structure has a frame/truss mechanism covered with a wire mesh foulant/debris barrier with a weighted polyethylene net beneath. The weighted high modulus polyethylene net extending from the bottom of the wire mesh mechanism to the bottom of the Pond assures that Design Function of the RN service water system will be met, by lifting in the unlikely event of significant blockage of the mechanism's wire mesh barrier. The designed weight of this net assures that the net will lift to provide the adequate Service Water flow in this event and there are no credible failure mechanisms or hazards which would prevent the lifting of the net. The wire mesh foulant/debris barrier provides "defense-in-depth" confidence that debris and macro-fouling will be prevented from entering the Nuclear Service Water system. The materials and fabrication of the wire mesh/foulant barrier and its frame were equivalent to the design criteria required for safety-grade equipment, and appropriate for use in the SNSWP. A hazards analysis was performed which concluded there are no credible hazards or failure modes of the wire mesh foulant/debris barrier or frame capable of defeating the design function of the RN system.

9.2.5.4 Tests and Inspections

All system components are hydrostatically tested and full operational tests are performed before station operation. During normal operation, inspections and tests of equipment are performed by plant operating personnel. Periodic inspection and testing is performed to ensure operability and structural integrity of the Standby Nuclear Service Water Pond Dam, intake structure, discharge structure, and associated piping in accordance with Technical Specifications.

Deleted paragraph(s) per 2002 revision.

9.2.5.5 System Instrumentation and Control

See Table 9-12 for a listing of instrumentation by type.

The instrumentation listed in Table 9-12 serves to inform the plant operator as to the operation and status of the ultimate heat sink. Standby Nuclear Service Water Pond temperature and level are monitored from the control room via computer points and alarms are provided to indicate an abnormal condition. No automatic channel changeover is provided other than automatic actuation of all safety equipment upon Engineered Safety Features signals as described in Section 9.2.1.

9.2.6 Condensate Storage System

The condensate storage facilities are discussed in Section 10.4.7 and 10.4.10.

9.2.7 Refueling Water System

9.2.7.1 Design Bases

The Refueling Water System is designed to provide:

1. a source of borated water at refueling water boron concentration, for use during refueling or a postulated loss-of-coolant accident,
2. recirculation of the refueling cavity and fuel transfer canal water for cleanup during refueling, as necessary,
3. recirculation of water in the refueling water storage tank for cleanup following refueling, as necessary, and
4. borated makeup water to the fuel pool.

The Refueling Water Storage Tank and associated piping are used in beyond design basis events for makeup to the other unit Refueling Water Storage Tank and to supply containment flooding through the Containment Spray system. Further information on the components which may be utilized in responding to a B.5.b related severe accident is maintained in design basis specification MCS-1465.00-00-0025, "Design Basis for the Extensive Damage Mitigation". Guidance for administration of the B.5.b accident response program is located in NSD-226, "Extensive Damage Mitigation Program."

As part of the FLEX mitigation strategy in response to NRC Order EA-12-049, the RWST and associated piping are relied upon as a source of reactor coolant system makeup during a postulated beyond design basis event. A refueling water mechanical process connection is provided for this capability on the 750' Elevation of the Auxiliary Building.

9.2.7.2 System Description

A Refueling Water System, as shown in Figure 9-65 is provided for each unit. Operation of the Refueling Water System is described as follows for various modes of system operation.

1. Safety Injection Operation

The refueling water storage tank provides a source of borated water for the injection mode during operation of the Emergency Core Cooling System.

The missile wall (a Category 1 structure) surrounding the refueling water storage tank is designed to protect the lower portion of the tank from missiles generated by environmental conditions or plant events. The limiting environmental condition has been determined to be a tornado that generates missiles as described in UFSAR Table 3-8. An example of a plant event that could generate missiles is a high speed turbine blade ejection. In the event a rupture of the refueling water storage tank occurs, the missile wall will retain a water level in the tank equal to the height of the missile wall. This assured borated water volume is sufficient to mitigate the consequences of a postulated main steam line rupture outside of containment, which is considered to be a limiting event.

2. Refueling Cavity Filling and Emptying

   The Refueling Water System provides a secondary means of filling and emptying the refueling cavity, the primary means being provided by the Residual Heat Removal System. The elevation of the refueling water storage tank provides sufficient static head to fill the refueling cavity to approximately 75 percent of its required water level. The refueling water pump is used to complete the filling process. After refueling, the refueling cavity water level is lowered to the top of the reactor vessel by using the Residual Heat Removal System to transfer approximately two-thirds of the water to the refueling water storage tank. The remaining water is transferred by the Refueling Water System. During the filling or emptying operation, water can be bypassed through the Spent Fuel Cooling System demineralizer and filters for cleanup.

3. Refueling Water Cleanup

   The refueling water in the refueling water storage tank and the refueling water cavity is recirculated through the Spent Fuel Cooling System demineralizers and filters for cleanup by using the refueling water pump. The refueling water in the refueling water cavity is recirculated through the Spent Fuel Cooling System demineralizer and filters or the Boron Recycle System demineralizers and filters for cleanup by using the reactor coolant drain tank pumps.

4. Component Description

   Component safety classifications and design codes are given in Section 3.2 and a summary of principal component design parameters is given in Table 9-25.

   a. Refueling Water Storage Tank (RWST)

      The refueling water storage tank usable capacity is based on the requirement for filling the refueling cavity to a depth that limits the radiation at the surface of the water to 2.5 mrem/hr during the period when a fuel assembly is transferred over the reactor vessel flange.

   b. Refueling Water Pump

      The refueling water pump has the capacity to pump one refueling cavity volume in approximately 24 hours. This flow rate is sufficient to handle any cleanup problem normally encountered. The total dynamic head is determined by the most stringent operating mode, which occurs during the emptying of the refueling cavity by going through the filters and demineralizer of the Spent Fuel Cooling System.

   c. Refueling Water Pump Strainer

The refueling water pump strainer is provided to protect the refueling water pump from debris that might be dropped in the refueling cavity.

d. Refueling Water Recirculation Pump

The refueling water recirculation pumps are provided to maintain the water in the pipe header feeding the charging pumps, safety injection pumps, residual heat removal pumps, and the Containment spray pump at or above 70°F. This is accomplished by recirculating the volume of water in this header at least once every 2½ hours from the heated refueling water storage tank. This is only required during the coldest winter months.

e. Refueling Water Pipe Trench Sump Pumps A&B

Two small submersible sump pumps are located in the pipe trench to remove rainwater which collects in the trench and prevent damage to heat tracing due to submersion in water.

9.2.7.3 Safety Evaluation

The refueling water storage tank provides a source of borated water for use during refueling and a loss-of-coolant accident. The tank capacity is based on the requirement for filling the reactor cavity and fuel transfer canal for refueling. A missile wall protects the lower portion of the tank to assure sufficient borated water is available for plant shutdown in the event of a steamline rupture due to tornado missiles. This capacity provides an amount of borated water to assure:

1. the volume of borated refueling water needed to increase the boron concentration of initially spilled water to a point that assures no return to criticality with the reactor at cold shutdown and all control rods, except the most reactive rod cluster control assembly, inserted in the core,
2. a volume sufficient to refill the reactor vessel above the nozzles after a loss-of-coolant accident, and
3. a sufficient volume of water in the Containment sump to permit the initiation of recirculation.

The only portion of the Refueling Water System shown in Figure 9-65 that is safety related Seismic Category I, is the refueling water storage tank and associated NRC Quality Class B piping that connects to the ECCS. A failure analysis of the lines from the refueling water storage tank to the safety injection, centrifugal charging, and residual heat removal pumps is provided in Chapter 6, Section 6.3. The failure of non-seismic Category I equipment and piping in the Refueling Water System and interfacing systems does not affect the ability of the Refueling Water System to perform its intended safety related function.

The water in the tank is borated to a concentration which assures reactor shutdown by approximately five percent Φk/k when all rod cluster control assemblies are inserted and the reactor is cooled down for refueling. Upon an Engineered Safety Features signal, the centrifugal charging pumps, the safety injection pumps, and the residual heat removal pumps take suction from the tank.

Automatic heating is provided to maintain the tank temperature above 70°F. Refueling water recirculation pumps are used to recirculate heated water from the tank through the transport line as necessary to maintain the line water temperature above 70°F.

9.2.7.4 Tests and Inspections

The Refueling Water System design is verified by pre-operational testing. Testing includes verification of system flow paths used in normal operation and a verification of electrical heater input and associated control interlocks. The instrumentation associated with the refueling water storage tank is checked preoperationally and during each refueling.

9.2.7.5 Instrumentation Application

Refer to Section 6.3.5.4 for RWST level instrumentation application during Emergency Core Cooling System operation. RWST level indications also aid the operator during the emptying and refilling of the refueling water storage tank during refueling. The tank has an overflow line to the spent fuel pool and a auxiliary vent line (not the 12 inch main vent) available to handle overflow. The temperature indication is provided to monitor the tank water temperature so that action can be taken if tank water temperature starts approaching the minimum tank water temperature.

All other instrumentation is provided to aid the operation of the system in its various modes.

9.2.8 Conventional Low Pressure Service Water and Recirculated Cooling Water Systems

9.2.8.1 Design Bases

The conventional Low Pressure Service Water System is designed to supply low pressure service water for various functions on the secondary side of the station. Specifically it supplies the following: cooling water for the Unit 1 and Unit 2 main turbine lube oil coolers, cooling water for the air compressors, makeup to the Water Treatment Building, and seal water flow to the Main Vacuum and Vacuum Priming pumps. It can also supply air handling units in the Service Building and Turbine Building.

The Recirculated Cooling Water System is a closed cooling system that delivers clean, rust-inhibited cooling water of a regulated temperature to various equipment in the Turbine Building, Auxiliary Building, and Service Building.

9.2.8.2 System Description

Conventional Low Pressure Service Water System

The system serving both units consists of three pumps, a duplex strainer and associated piping and valves (see Figure 9-54, Figure 9-55, and Figure 9-56). Normally, two of the pumps are in operation with the third in reserve. Supply for the pumps is from the crossover pipe between Unit 1 and Unit 2 Condenser Circulating Water System intake piping. Flow supplies water to the Water Treatment Building from which the water is processed into demineralized water for various uses throughout the station. Seal water flow is supplied to the Main Vacuum and Vacuum Priming pumps. Another part of the flow is used for cooling the air compressor after coolers. Flow also goes to the Unit 1 and Unit 2 main turbine lube oil coolers. There are two 100 percent capacity main turbine lube oil coolers per unit and during normal operation one of the two on each unit will be in operation with the other in reserve. The cooling water is discharged into either the Unit 1 or Unit 2 Condenser Circulating Water System discharge piping. Thus Conventional Low Pressure Service Water to both units is operational, even if one tunnel is unwatered. The system's cooling requirements are based on maximum lake temperature and minimum lake level.

An alternate flow path for the Conventional Low Pressure Service Water System is provided by the Unit 2 side condenser circulating water system via Booster Pump skid. The Booster Pump skid consists of two 100% capacity pumps and a duplex strainer. The Booster Pump skid can be used to supply the Conventional Low Pressure Service Water System with 8000 gpm at its system design pressure conditions to support Unit 2 operation during a Unit I outage. See Figure 9-54 for Booster Pump skid connection to system.

Air and other released gases are removed from the main turbine lube oil cooler inlet and discharge piping by the Vacuum Priming System and high point vents. The conventional service water strainer is cleaned by backwashing it with water from the conventional service water pump discharge header. For system component design see Table 9-20.

Recirculated Cooling Water System

Three fifty percent capacity (two required for normal operation) recirculated cooling water pumps take suction from the recirculated cooling water storage tank and deliver cooling water to the following equipment:

1. Air Compressor Water Jackets
2. Air Compressor Intercoolers
3. Air Compressor Oil Coolers
4. Boron Thermal Regeneration System Chillers (This equipment is no longer in service)
5. Ice Condenser Refrigeration Units
6. Ice Bin Condensing Units
7. Ice Annex Condensing Unit
8. Ice Condenser Air Cooler
9. Air Chiller Condensing Unit
10. Feedwater Pump Turbine Oil Coolers
11. Condensate Booster Pump Oil Coolers
12. Generator Seal Oil Cooler
13. Isolated Phase Bus air Cooler
14. Electro-hydraulic Control Unit Oil Coolers
15. C-Heater Drain Pump Seals
16. C-Heater Drain Pump Motors
17. Exciter Air Cooler
18. Conventional Sampling System Sample Coolers
19. Hotwell Pump Motor Bearings
20. G-Heater Drain Pump Motor Bearing
21. Electrical Penetration Room (733' & 750') Condensing Units
22. Deleted Per 2005 Update

Deleted Per 2015 Update.

23. S/G BB Tank Sample Hx.

24. Aux. Electric Boilers Conductivity Sample Coolers
25. Aux. Electric Boilers Feed Pump Thrust Bearing
26. Aux. Electric Boiler Recirc. Pump Bearing
27. Aux. Electric Boiler Recirc. Pump Mechanical Seal
28. Upper Surge Tank Sample Hx.
29. Air Compressor Aftercoolers

The Cooling water then goes to four one-third (1/3) capacity recirculated cooling water exchangers. Normally only three heat exchangers are required. The cooling medium for these heat exchangers is the condenser circulating water system. An alternate supply is provided by the Conventional Low Pressure Service Water System. This alternative supply is typically used during a Unit 1 outage to support Unit 2 operation. The water then flows back to the recirculated cooling water storage tank to complete the cycle (see Figure 9-26 through Figure 9-30). For system component design parameters, see Table 9-7.

9.2.8.3 Safety Evaluation

Conventional Low Pressure Service Water System

No safety functions are performed by this system. It is not designed to be operable during earthquake or tornado conditions. During normal operation, the system remains functional even if one unit is out of service and its circulating water tunnels are unwatered.

Recirculated Cooling Water System

The Recirculated Cooling Water System does not perform any safety function. The recirculated cooling water piping in the Auxiliary Building is Duke Class F where required to prevent damage to safety related equipment. A flood analysis of the Auxiliary Building has shown that postulated cracks in the recirculated cooling water piping in the auxiliary building would not flood any safety related equipment.

9.2.8.4 Tests and Inspections

Conventional Low Pressure Service Water System

System components are hydrostatically tested prior to station startup and are accessible for periodic inspections during operation by station personnel. The Conventional Low Pressure Service Water System is functionally tested prior to unit startup to verify that integrated system operation is capable of supplying cooling water to the required components. Proper pump operation, system flow paths and strainer operation are demonstrated.

Recirculated Cooling Water System

The system is fully tested and inspected before initial operation. All components are hydrostatically tested where these tests apply. Working components are tested to make sure they operate properly. During normal operation, testing is not required; adequate operating performance monitoring assures system integrity.

9.2.8.5 Instrumentation Application

Conventional Low Pressure Service Water System

Pressure gauges on the inlet and outlet piping of the strainer indicate the need to backwash the strainer. Instrumentation on the discharge of the main turbine oil tank coolers regulates the flow of the conventional low pressure service water, as required, by discharge temperature of the oil. Instrumentation on the discharge of the conventional service water pumps indicates pressures which relate to flow. Flow elements at the discharge of the air compressor aftercoolers indicate flow; this information is used to set cooling water throttling valves. Temperature and pressure test points are located on the inlet and discharge sides of the main turbine lube oil coolers and the air compressor aftercoolers.

Recirculated Cooling Water System

The flow of cooling water to the various equipment is controlled in the following ways:

1. The cooling water to an air compressor water jacket is controlled by a temperature regulating valve set to maintain the exit cooling water temperature.
2. The cooling water to an air compressor intercooler is controlled by a temperature regulating valve set to maintain the exit cooling water temperature.
3. Cooling water to a feedwater pump turbine oil cooler is controlled by a modulating control valve downstream of the equipment which acts on a signal from a temperature transmitter located on the exit oil line. There are also pressure test points located on the cooling water lines before the cooler, and pressure and temperature test points located after the cooler.
4. Cooling water to a condensate booster pump oil cooler is regulated by a self-contained flow regulating valve on the cooler cooling water discharge line.
5. Unit 1 - Generator seal oil cooler water for both the air and the hydrogen sides is controlled by control valves that are modulated according to the exit oil temperature. There are also pressure and temperature test points on the cooling water lines on either side of the air side cooler and there are temperature test points on either side of the hydrogen side cooler and pressure test points before the cooler.

   Unit 2 - Generator seal oil temperature control is provided by a Temperature Control Valve that modulates oil through the cooler. Cooling water flow is not modulated.
6. Isolated phase bus (IPB) air cooler cooling water is controlled by throttling a globe valve downstream of the IPB to get the desired flow. The flow is measured by flow orifices and is controlled at approximately 309 gpm.
7. Electro-hydraulic control unit cooling water is controlled by a control valve downstream of the cooler and is modulated by a signal from a temperature transmitter, located on the main turbine hydraulic oil reservoir, to maintain a reservoir temperature of approximately 120°F.
8. C-Heater drain pump mechanical seal and C-Heater drain pump motor cooling water is controlled by self-contained flow regulating valves set for a 5 gpm flow rate. There are flow meters upstream of the equipment.
9. Exciter air cooler cooling water is controlled to maintain an exit air temperature of approximately 112°F. The flow is measured using flow orifices and pressure gages and is regulated by throttling valves all downstream of the equipment.
10. The boron thermal regeneration (BTR) chiller condenser cooling water is modulated by a control valve acting on a signal from a pressure transmitter on the BTR chiller condenser. There are pressure test points on the cooling water lines before the equipment and pressure and temperature test points after the equipment. This equipment, however, is no longer in service.
11. Ice condenser refrigeration package cooling water is controlled using a modulating control valve which maintains condenser pressure to control freon temperature.

12. Ice bin condensing unit cooling water is controlled by a control valve upstream of the equipment using instrumentation on the Ice Condenser Refrigeration System.
13. Ice annex condensing unit cooling water is controlled by a control valve upstream of the equipment using instrumentation on the Ice Condenser Refrigeration System.
14. Ice condenser air cooler cooling water is controlled by a control valve upstream of the equipment using instrumentation on the Ice Condenser Refrigeration System.
15. Air chiller condensing unit cooling water is controlled by a control valve upstream of the equipment using instrumentation on the Ice Condenser Refrigeration System.
16. Hotwell pump motor bearing cooling water is controlled by self-contained flow regulating valves and measured by flow meters downstream of the equipment.
17. G-Heater drain pump bearing cooling water flow is controlled by self-contained flow regulating valves and measured by flow meters downstream of the equipment.

An alarm is sent to the operator in the event a low level is reached in the recirculated cooling water storage tank. The low level condition can then be corrected by opening valve 1KR97 makeup from the upper surge tank or by opening valve 1KR16, make up from the demineralized water system.

The recirculated cooling water heat exchangers have temperature test points on the cooling water lines after the equipment. Downstream of each heat exchanger there is a temperature transmitter than sends a signal to modulate the amount of condenser circulating water going through the heat exchangers. There is also a thermocouple well in the heat exchanger discharge header.

There is a pressure gage in each RCW pump discharge line that will show a flow related pressure. There is also a pressure transmitter in the RCW pumps common discharge header. This pressure transmitter opens the recirculation line valve and also alarms on high or low header pressure.

The flow alarm is provided on the shell side of each recirculated cooling water heat exchanger to alarm excessive velocity through the shell. This alarm is activated by a differential pressure switch across the shell side of each heat exchanger.

9.2.9 Conventional Waste Water Treatment System

9.2.9.1 Design Bases

The Conventional Waste Water Treatment System accepts all secondary side plant waste water (except sanitary sewage), monitors it for radioactivity, treats it through a system of basins with chemical treatment capabilities and discharges it to the Catawba River at a quality equal to or better than applicable State and Federal Water Quality Standards.

9.2.9.2 System Description

The Conventional Waste Water Treatment System receives waste water in two separate headers (Unit 1 header and Unit 2 header). The origins of the waste water flows for each header are as follows (see Figure 9-77):

1. Unit 1 Header
   a. Unit 1 condensate polishing demineralizer backwash tank discharge.

   b. Unit 1 Turbine Room sump discharge.
   c. Unit 1 unwatering pump discharge.
   d. Diesel Generator Room sumps 1A and 1B discharges.
   e. Standby Shutdown Facility Sump discharge.
   f. Unit 1 steam generator blowdown blowoff tank conventional waste water dump.

2. Unit 2 Header
   a. Unit 2 condensate polishing demineralizer backwash tank discharge.
   b. Unit 2 Turbine Room sump discharge.
   c. Unit 2 unwatering pump discharge.
   d. Diesel Generator Room sumps 2A and 2B discharges.
   e. Unit 2 steam generator blowdown blowoff tank conventional waste water dump.

The Service Building 739' treated water area sump discharge, Water Treatment Building waste discharge, and the Landfill Leachate Pond discharge feeds into the open, concrete channel which feeds into the initial holdup pond.

The only sources discussed above which are potentially radioactive are the Turbine Room sumps, the condensate polishing demineralizer backwash and the steam generator blowdown blowoff tank conventional waste water dump. If radioactivity from any source approaches 10CFR 20 limitations, it is handled as radioactive waste by the Liquid Waste Monitor and Disposal System and the Solid Waste Disposal System, respectively (see Section 9.2.9.3 and Table 9-26 for method of determining radioactivity at these sources).

This flow is transported via one of two 12" diameter pipes to the Initial Holdup Pond. The two 12" diameter pipes have a crossover valve arrangement such that the Unit 1 waste water may be transported separate from the Unit 2 waste water if it becomes necessary. The Initial Holdup Pond discharges through an open concrete channel into either Settling Pond A or Settling Pond B. The Settling Ponds discharge to either the Final Holdup Pond which in turn discharges to the Catawba River, or directly to the Catawba River.

9.2.9.3 Safety Evaluation

The Conventional Waste Water Treatment System is a shared system which performs no safety function, however, radiation monitors 1 and 2EMF31 are provided in the respective unit's waste water header to monitor the secondary side liquid waste discharges.

Under normal operating conditions, i.e., no known or only small primary to secondary coolant leakage, the potentially radioactive sources discussed in Section 9.2.9.2 are sampled and analyzed in accordance with Table 9-26, and the secondary coolant is sampled and analyzed in accordance with the Technical Specifications. This sampling and analysis would detect radioactivity on the secondary side as a result of minute primary coolant leakage prior to detection by monitors 1 and 2EMF33 (CSAE gaseous release monitor) or monitors 1 and 2EMF34 (Steam Generator Blowdown sample monitor). Once radioactivity has been detected, either by the sampling and analysis of Table 9-26 or radiation monitors, 1EMF33, 2EMF33, 1EMF34 or 2EMF34, sampling and analysis of the particular unit involved would be changed in accordance with the appropriate Radiation Protection and Chemistry procedures. In addition to this, monitors 1 and 2EMF31 discussed above, provide additional indication to operating personnel that waste water approaching 10CFR 20 limits is not entering this system.

9.2.9.4 Tests and Inspections

Initial operation and acceptance tests are performed to assure all system components meet design capacity and quality.

9.2.9.5 Instrumentation Application

Instrumentation is provided to monitor radioactivity of each unit's waste stream as described in Sections 9.2.9.2 and 9.2.9.3. Process instrumentation is provided to monitor the flow rate and quality of the waste stream within the system to assure proper treatment, and the discharge quality is continuously monitored to assure that the system discharge to the Catawba River is of a quality equal to or better than applicable State and Federal water quality standards.

9.2.10 Containment Ventilation Cooling Water System

9.2.10.1 Design Bases

The Containment Ventilation Cooling Water System is designed to provide a source of raw cooling water to various containment, fuel handling, and auxiliary building ventilation systems. The Containment Ventilation Cooling Water System is designed to support this function during normal operation and shutdown conditions.

9.2.10.2 System Description

The shared system consists of three parallel configured pumps, a duplex suction strainer and associated piping and valves (see Figure 10-29). The system cooling water supply can be provided by any combination of the system pumps from the low level intake, and/or from the Nuclear Service Water System non-essential header. The system cooling water source is normally provided from the Nuclear Service Water System non-essential header with the pumps in a standby mode. Based on thermal load requirements, the pump(s) are placed in operation on an as needed basis. The system cooling water return flow discharges into the Nuclear Service Water System discharge header (train 'A' for Unit 1, and train 'B' for Unit 2). The system pumps have the capability to automatically start, if RN non-essential header pressure on either Unit is too low. Refer to Table 9-45 for component design data.

The RV System supplies cooling water primarily to the following loads;

1. Upper Containment Ventilation Units (UCVUs)
2. Lower Containment Ventilation Units (LCVUs)
3. Incore Instrumentation Room Ventilation Units (IRVU)
4. Auxiliary Building Ventilation Units (ABSU)
5. Fuel Handling Area Ventilation Unit (FPSU)

Table 9-8 presents the nominal cooling water flow-rates supplied to the ventilation units within the Auxiliary and Containment buildings. The system duplex suction strainer is equipped with a manual backwash capability. The system pumps are equipped with an automatic mini-flow recirculation valve. Vacuum breakers are utilized within the system to prevent unacceptable pressure transients during pump starts, subsequent to a low system pressure condition. The Lower Containment Ventilation Units are equipped with an automatic tube cleaning system.

The system pumps cn also be utilized for filling the Nuclear Service Water System after intrusive maintenance. The system has the ability to deliver cooling water to the Service and Turbine

Building air conditioning units. This capability is provided by cross-connect piping to the Conventional Low Pressure Service Water System.

The Upper Containment Ventilation units are equipped with a cooling water back-pressure control valve, which serves to minimize the potential for column separation and cavitation damage on upstream valves.

9.2.10.3 Safety Evalation

With the exception of the system containment isolation valves, the Containment Ventilation Cooling Water System performs no required safety related functions. The system provides a non-safety related cooling water supply to various ventilation units to control building ambient temperatures and relative humidity.

On receipt of a safety injection signal ('Ss'), the Nuclear Service Water System cooling water supply to the system is automatically isolated. The Auxiliary Building Ventilation Unit cooling water return header is isolated on receipt of a safety injection signal ('Ss').

The Containment Ventilation Cooling Water System pumps can continue to provide cooling water flow to the containment ventilation units until receipt of a high-high containment pressure signal ('Sp').

Refer to additional discussion for the Nuclear Service Water System in Section 9.2.1.

9.2.10.4 Tests and Inspections

The design of the Containment Ventilation Cooling Water System was verified by testing prior to initial plant start-up. The testing demonstrated adequate system flow capability, building temperature control, and proper operation of associated instrumentation controls.

Refer to Section 6.2.4.4 regarding testing of containment isolation valves.

9.2.10.5 Instrument Application

The primary instrumentation and controls provided for the system include the following:

- Pump suction, discharge pressure indication
- Pump differential pressure indication and mini-flow valve control
- Pump supply header temperature indication
- Strainer differential pressure indication
- Lower Containment Ventilation cooling water flow indication
- Upper Containment Ventilation cooling water header backpressure control
- Cooling water flow control valves for the Upper Containment Ventilation Units
- Cooling water flow control valves for the Incore Instrumentation Room Ventilation Units

9.2.11 References

1. “Generic Cooling Pond Analysis”, C00-2224-1, May 1972 – October 1972.
2. Ryan, P.J. and Harleman R.F., “An Analytical and Experimental Study of Transient Cooling Pond Behavior”, MIT Report 161, January 1973.

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6. Harleman, D. R. F. and R. A. Elder, “Withdrawal From Two-Layer Stratified Flows,” ASCE, Journal of the Hydraulics Division, Vol. 91, p. 43-58, 1965.
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10. Deleted Per 2002 Update.
11. Deleted Per 2002 Update.
12. Deleted Per 2002 Update.
13. Calculation MCC-1150.01-00-0004 (current revision)
14. Calculation MCC-1150.01.00.0008 (Current revision)
15. Calculation MCC-1223.24-00-0130 (Current revision)

THIS IS THE LAST PAGE OF THE TEXT SECTION 9.2.

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

9.3.1.1 Design Bases

9.3.1.1.1 Instrument Air System

The Instrument Air System is designed to provide an adequate capacity of dry oil-free air for instrumentation, testing, and control air requirements. The Instrument Air System is not a safety class system and is not seismically designed, except for the nitrogen supply line from the Cold Leg Accumulators to the Pressurizer PORVs, the containment penetration piping and containment isolation valves and the Diesel Building Instrument Air Piping. The Pressurizer PORV's nitrogen supply line and associated valves are class F, but they do have a safety significance (Steam Generator Tube Rupture, LTOP), as a result of NRC Generic Letter 90-06. Safety class equipment requiring compressed air is provided with an air reserve sufficient to perform its safety function should the compressed air system fail. Reference Figure 9-79 for general system layout and Table 9-28 and Table 9-29 for equipment design parameters.

During normal operations, instrument air is provided by three centrifugal air compressors. Two diesel powered and three reciprocating compressors provide instrument air if the centrifugal compressors cannot maintain system pressure at acceptable levels. Downstream of the compressors, air dryers dry air to a dew point of -35ºF at 100 psig.

9.3.1.1.2 Station Air System

The Station Air System is designed to provide an adequate capacity for general station service air requirements. Normally, the Instrument Air System provides the station air requirements through system crossconnect valve 1VI820 (Figure 9-83). However, if needed, one station air compressor is provided to furnish the station air requirements if the instrument air supply is not available or desired (Figure 9-83). In addition, the output of the Diesel Powered Instrument Air Compressors can be directed to supply the Station Air System if needed.

A smaller capacity high pressure service air system provides high pressure air supply to the fire protection pressurizer tank.

See Table 9-28 and Table 9-29 for equipment design parameters.

9.3.1.1.3 Breathing Air System

The Breathing Air System is designed to provide an adequate capacity of air which meets ANSI specification CGA-G-7.1-1989. Two full capacity compressors are provided to furnish the total average breathing air requirements. See Table 9-28 and Table 9-29 for equipment design parameters.

9.3.1.1.4 Piping

The compressed air piping system which furnishes air inside the Containment is equipped with Containment isolation valving in accordance with the criteria for Containment Isolation Systems as discussed in Section 6.2.4.

9.3.1.2 System Description

9.3.1.2.1 Instrument Air System

The Instrument Air System consists of three centrifugal compressors, three reciprocating compressors, and two diesel powered rotary screw compressors. After being compressed, air is passed through aftercoolers, moisture separators, air dryers, and filters. Dry, oil free air is then provided throughout the plant.

The centrifugal compressors operate in base mode supplying all plant instrument air demands. Each centrifugal compressor takes suction from the Service Building basement through a filter. The compressors are oil-free, centrifugal, two-stage, and water cooled. The compressors are designed to deliver 1500 ICFM of air at 100 psig. Cooling water is provided for the intercooler and aftercooler from the Recirculated Cooling Water System. Moisture is removed by moisture separators internal to each compressor. Outlet air from the centrifugal compressors is routed to two parallel headers between the Instrument Air Receivers and the Instrument Air Dryers.

Two diesel powered Instrument Air Compressors operate in standby mode and are capable of supplying Instrument air loads should the centrifugal compressors be unavailable. They will auto start upon decreasing air pressure, failure of the VI compressor sequencer panel, or loss of Recirculated Cooling Water System flow (KR) to VI Compressors D, E, and F. Controls are provided to align the diesel powered compressors to provide normal instrument air system flows in lieu of the centrifugal compressors if desired. The diesel powered compressors are single stage, rotary screw type compressors. Each of the compressors is designed to deliver 1200 SCFM of air at 100 psig. Outlet air from each compressor passes through an air cooled aftercooler and afterfilters and is then routed to a header between the Instrument Air Receivers and the VI Centrifugal Air Compressor “D”.

The reciprocating compressors are available to supply instrument air loads should the centrifugal and diesel powered compressors be unable to maintain adequate instrument air system pressures. They start on decreasing air pressure. Start up of a standby reciprocating compressor is annunciated in the Control Room. Controls are provided to switch the reciprocating compressors to base mode if it is desired that they provide normal instrument air system flows in lieu of the centrifugal or diesel powered compressors. The reciprocating compressors are positive displacement, oil-free, two-stage, and watercooled. Each of the compressors is designed to deliver 650 SCFM of air at 100 psig. Cooling water for the air compressor intercoolers and water jackets is from the Recirculated Cooling Water System. Inlet air passes through individual air filters and then to the respective air compressor. The three reciprocating compressors discharge air to a common header which then feeds two trains of an aftercooler and moisture separator combination. Cooling water for the aftercoolers is supplied by the Conventional Low Pressure Service Water System. Air leaving the aftercoolers then passes through moisture separators when swirling action removes water particles in the air. The outlet of the two aftercooler/moisture separator trains then joins a common header upstream of the three Instrument Air Receiver Tanks. Each of the three air receivers has a storage capacity of 312 cu ft and a relief valve setting of 115 psig. Air from the receivers is routed to two parallel headers between the receivers and the Instrument Air Dryers.

Outlet air from the Instrument Air Compressors is routed via two parallel headers to three Instrument Air Dryers. Each of the dryers is designed to deliver a dew point of -35ºF when supplied with 1600 SCFM of air at 100 psig and 100ºF. The instrument air system is a shared system in that all compressors and air dryers can supply both or either unit. Portions of the Instrument Air System are unitized within the Auxiliary Building, Turbine Buildings, Service Building, Interior & Exterior Doghouses, and Reactor Buildings.

Assured air accumulators are provided to supplement the Instrument Air supply for the RN strainer backwash supply AOVs. Refer to Section 9.2.1.1. Assured air accumulators are provided to supplement the air supply for the MSIV's. Refer to section 10.3.2. Assured air accumulators are provided to supplement the normal pilot air supply to the Unit 1 Main Feedwater Containment Isolation Valve (CFIV) control panels. Unit 2 CFIV's are supplied from assured nitrogen accumulators. Refer to section 10.4.7.2. As part of the FLEX mitigation strategy in response to NRC Order EA-12-049, assured air supplies have been installed to provide a continuous air supply to the instrument air blackout header until portable compressors can be aligned to the blackout header. This consists of two 1400 ft^3 storage tanks to provide the initial air supply in an extended loss of AC power event and adding connections to the instrument air system in the Unit 1 and Unit 2 interior and exterior doghouses for portable compressors to facilitate use of a backup air supply.

9.3.1.2.2 Station Air System

The Station Air System provides air for pneumatic tool operation and other miscellaneous uses in the plant. Normally, the Instrument Air centrifugal compressors provide the station air requirements through system crossconnect valve 1VI820. If operating, the diesel-powered or reciprocating compressors can supply station air through the same valve. In addition, the output of the diesel-powered Instrument Air Compressors can be directed to supply station air through a normally closed manual isolation valve. In addition, station air loads can be supplied by one Station Air Compressor if the instrument air supply is not desired.

Station Air equipment consists of one compressor, one aftercooler, one moisture separator, and two air receivers. The compressor aftercooler cooling water is supplied by the Conventional Low Pressure Service Water System. The moisture separator section swirls the air to remove the water particles by centrifugal force. One of the receivers is located in the discharge path of the compressor. The other receiver does not receive flow directly from the compressor. Instead, it acts as an additional storage volume for the Station Air System. After leaving the receivers, the air is then routed to the station air headers for tool operation and other miscellaneous usage. Isolation valves located outside the Containment are installed in series with check valves located inside the Containment for each service air containment penetration.

The control circuit for the air compressor has the following pressure settings for operation of the compressor.

Falling Pressure Start	Rising Pressure Stop
94 psig	100 psig

The Station Air System (high pressure) consists of two full capacity 26 SCFM compressors. Each compressor is equipped with an inlet air filter and an air receiver tank. The system provides pneumatic pressurization of the fire protection pressurizer tank. The Instrument and Breathing Air Systems were not used for this function to eliminate the possibility of water entering these systems.

9.3.1.2.3 Breathing Air System

Breathing Air is supplied by two compressors. The identical compressors are oil lubricated rotary screw type compressors. Each of the compressors is designed to deliver 450 SCFM of air at 125 psig. Each compressor takes air from intakes located outside and above the roof of the Service Building. Inlet air passes through individual air filters and then to the respective air compressor.

Downstream of each compressor, the air flows through filters which filter the air to meet ANSI CGA-G-7.1-1989 air quality. After the filters, air from each compressor discharges into a common receiver tank.

From the receiver tank, breathing air is supplied to various locations in the Auxiliary Building, Turbine Building, and inside the Containment. Isolation valves located outside the Containment are installed in series with check valves located inside the Containment for each breathing air containment penetration.

9.3.1.3 Safety Evaluation

The compressed air systems are designed to provide a dependable source of compressed and cooled air for station service, breathing, testing, and instrumentation requirements. Sufficient redundancy is provided to give a high degree of reliability of air supply at all times. Sufficient air receiver capacity is provided to meet system high air demand transients.

A loss of instrument air due to a loss of offsite power (LOOP) during normal operation causes all pneumatically operated valves in the station which are essential for safe shutdown to fail to the safe position. Therefore, air headers have been provided in the Auxiliary Building and both Containments to supply air to those valves which are used to normally proceed to cold shutdown. These valves, which have a blackout air supply, are listed in Table 9-30. These headers are protected against depressurization by check valves 1VI122, 1VI153, 2VI122 and 2VI153. Air storage for these headers are provided by the Instrument Compressed Air Tanks. Since the emergency diesel generator power is not available for the Instrument Air Compressors, the blackout air supply headers are aligned to the Diesel Generator Starting Air headers through a normally closed solenoid valve and a normally closed manual isolation valve. The solenoid valve will open upon receipt of a LOOP and a “Diesel Running” signal without a LOCA signal after a 30 second delay. The manual valves will be opened and then throttled to supply as much Diesel Generating Starting Air as possible to the blackout air supply headers without lowering the pressure of the Diesel Generating Starting Air System to levels that might adversely effect the operation of the Diesel Generators. Since the installation of diesel powered instrument air compressors, the need for relying on the Emergency Diesel Starting Air Compressors is reduced. The use of these Starting Air Compressors will be only at the direction of the Technical Support Center during a plant event.

9.3.1.4 Tests and Inspections

The system is fully tested and inspected before initial operation. Adequate operating performance monitoring assures system integrity.

The NRC issued Generic Letter 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment," on August 8, 1988 (Reference 1). The purpose of Generic Letter 88-14 was to request for McGuire Nuclear Station an Instrument Air System design review of NUREG-1275 and verification of the following:

1. Verification by test that actual instrument air quality is consistent with the manufacturer's recommendations for individual components served.
2. Verification that maintenance practices, emergency procedures, and training are adequate to ensure that safety-related equipment will function as intended on loss of instrument air.
3. Verification that the design of the entire instrument air system, including air or other pneumatic accumulators, is in accordance with its intended function, including verification by test that air operated safety-related components will perform as expected

in accordance with all design-basis events, including a loss of the normal instrument air system. This design verification should include an analysis of current air-operated component failure positions to verify that they are correct for assuring required safety functions.

Responses detailing these verifications, along with descriptions of the Instrument Air Program, and design and programmatic enhancements necessary for maintaining proper instrument air quality were submitted to the NRC, beginning with the letter from H.B. Tucker to the NRC, dated February 10, 1989 (Reference 2), and concluding with the letter from T.C. McMeekin to the NRC, dated July 9, 1996 (Reference 3), which detailed Generic Letter 88-14 action item closeout.

9.3.1.5 Instrumentation Application

9.3.1.5.1 Instrument Air System

Instrumentation is provided on each compressor to trip it on compressor trouble. Local indication of air pressure is provided by instruments at the following locations: each reciprocating instrument air compressor, each centrifugal instrument air compressor, each diesel powered instrument air compressor, each instrument compressed air tank, each instrument air dryer, and the combined discharge of the instrument air dryers. Local and remote indication of air flowrates is provided on the discharge of each centrifugal instrument air compressor, the outlet of each instrument air dryer, the Unit 1 instrument air supply header, and the Unit 1 Reactor Building/Auxiliary Building instrument air supply header. Local indication of air flow is provided for the yard instrument air supply header.

Pressure switches and transmitters on the instrument air receivers provide operating control for the instrument air compressors. However, if desired, the centrifugal compressors can be controlled by instruments in the discharge of each compressor. Instrumentation in the Control Room is provided to indicate Instrument Air system pressure. Pressure switches on the instrument compressed air tanks, containment instrument air headers, and the combined discharge piping downstream of the instrument air dryers provide low pressure alarms.

9.3.1.5.2 Station Air System

Local indication of air pressure is provided by pressure gages at each air receiver. Instrumentation is provided to indicate Station Air System pressure in the Control Room. Pressure switches on the air receiver discharge header provide operating control for the station air compressors.

Local indication of high pressure station air pressure is provided by a pressure gage on the piping immediately downstream of the high pressure station air receiver discharge header. Pressure switches on the high pressure station air receiver discharge piping provides low air pressure alarm and operating control for high pressure station air compressors.

9.3.1.5.3 Breathing Air System

Local indication of air pressure is provided by pressure gages at the compressors. Instrumentation is provided to indicate Breathing Air System pressure in the Control Room. A pressure switch on the air discharge header provides a low air pressure alarm which is annunciated in the Control Room. A flow element and differential pressure gage is provided on the receiver discharge for testing.

9.3.2 Nuclear Sampling System

9.3.2.1 Design Basis

The Nuclear Sampling System is designed to provide means to obtain samples from various systems in each of the two units for chemistry and radiochemistry analysis. Primary system chemical addition can be done via the sample purge header to the VCT. Adequate safety features are provided to protect laboratory personnel and prevent the spread of contamination from the sample room. The system has no emergency function. During a LOCA, this system is isolated at the Containment. Nuclear Sampling System discharges are designed to limit flow under normal operation and anticipated malfunctions or failures to preclude any fission product release leading to exposures exceeding 10CFR 20 limits. The system is designed to operate manually and on an intermittent basis under conditions ranging from full power operation to cold shutdown. Access to the Containment is not required for sampling.

9.3.2.2 System Description

A Nuclear Sampling System, as shown on Figure 9-90 through Figure 9-92 is provided for each unit. The system provides representative samples for laboratory analyses which are used for guidance in the operation of various primary and secondary systems. Typical information obtained includes reactor coolant boron and chloride concentrations, fission product radioactivity level, hydrogen and oxygen content, fission gas content, corrosion product concentration, and chemical additive concentration. A sample room is provided for each unit.

Samples are obtained within the sample room of each unit from the following locations:

Inside Containment

Pressurizer steam space

Pressurizer liquid space

Hot leg of two different reactor coolant system loops

Each of four steam generator's steam space

Each of four steam generator blowdown lines

Each of the four accumulators

Outside Containment

Downstream of both residual heat removal heat exchangers

Volume control tank gas space

Volume control tank discharge

Upstream of the NV mixed bed demineralizers

Downstream of the NV mixed bed demineralizers

Downstream of the NV cation bed demineralizers

Downstream of the NV boric acid blender

Downstream of the NV seal injection filters

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Upstream of the NB evaporator feed demineralizers (Unit 1)

Downstream of both NB evaporator feed demineralizers (Unit 1)

Downstream of the NB evaporator feed pumps (Unit 1)

Downstream of the NB condensate demineralizers (Unit 1)

Waste evaporator feed pump discharge (Unit 1)

Waste drain tank pump discharge (Unit 1)

Waste evaporator condensate demineralizer inlet (Unit 1)

Waste evaporator condensate demineralizer outlet (Unit 1)

Recycle monitor tank pump discharge (Unit 1)

Reactor makeup water storage tank discharge

Upstream of fuel pool cooling pre-filter

Downstream of fuel pool cooling post-filter

Refueling water storage tank discharge

Spent resin sluice filter discharge (Unit 1)

Local sample connections are provided at various other locations outside the Containment, but are not considered part of the sampling system.

Sample lines originating within the Containment are provided with remote, motor-operated Containment isolation valves, located both inside and outside the Containment, which are closed automatically by a Containment isolation signal in the event of a LOCA. In addition, a manual valve is located close to each sample source, and manually operated valves for flow control and isolation are located in the sample room.

Sample lines originating outside the Containment are provided with a manually operated valve close to the source of the sample, with the exception of the residual heat removal loops which are provided with a remote, motor-operated valve. All remotely operated valves are controlled from the sample room.

All sample lines are provided with a local sample valve in the sample room sink. The sample sink is located in a hooded enclosure which is equipped with an exhaust ventilator. The work area around the sink and enclosure is large enough for sample collection and storage of radiation monitoring equipment.

Shielding is provided where required for personnel protection (refer to Section 12.1). Local instrumentation is provided to permit manual control of sampling operations and to assure that the samples are at suitable temperature and pressures before diverting flow to the sample sink.

Sample heat exchangers are provided to cool samples from the reactor coolant loops, residual heat removal loops, pressurizer steam and liquid spaces, and the steam generator blowdown lines. In the sample room manually controlled valves are provided to reduce pressure. Purge lines are provided through which sample fluids may flow until sufficient volume has passed to permit collection of a representative sample. Sufficient flow is directed to the sample sink to purge the remainder of the sample line prior to obtaining a sample.

A gaseous sample from the volume control tank in the chemical and volume control system is collected in a sample vessel.

The steam generator blowdown sample lines are taken off the blowdown lines as close to the steam generator as practical in order to provide representative samples and satisfactory radioactivity control. The steam generator upper shell sample lines are taken directly off the steam generator. They are tied, along with the blowdown sample lines, into common steam generator sample lines to the sample room. The sample line flow from each steam generator is cooled by a sample heat exchanger. Downstream of the sample heat exchanger each sample line separates into 3 paths. One path provides for manual sampling at the primary sample sink. Another path connects all four sample lines together and directs the flow through a radiation monitor. The flow from the radiation monitor is directed to the steam generator blowdown tank of the Steam Generator Blowdown System. Individual lines are provided for each steam generator sample header, directing flow to the conventional sampling panel.

Each line contains an air operated isolation valve which will automatically close on a signal from the radiation monitor. Containment isolation valves are provided in each of the steam generator sample lines and are arranged so that the operator may select sample flow from either the upper shell or the blowdown line of each steam generator. These valves close automatically on the safety injection signal. The ability to obtain a manual sample in the primary sample sink for identifying the responsible steam generator following a radiation monitor alarm is provided.

The principal components of the system are the sample heat exchangers, sample vessel and the sample sink. Component design is as follows:

Sample Heat Exchangers

Sample heat exchangers are provided to cool samples originating from the pressurizer, the reactor coolant loops, the residual heat removal loops and the steam generator blowdown sample lines. The samples flow through the tube side and component cooling water circulates through the shell side.

Sample Vessels

A sample vessel is provided in the sample line from the volume control tank to obtain gas samples for laboratory analysis.

Sample Sink

Two sample sinks are provided in each sample room. One sink includes all the samples which require heat exchangers or sample vessels plus the accumulator sample lines. The second sink includes all the other sample lines. Both sinks drain to the waste evaporator feed tank sump A.

The sinks are located in a hooded enclosure which is supplied with a continuously running exhaust fan. The fan discharges to the unit vent. The enclosure is penetrated by the sample lines, and a demineralized water line for each sink. Manual valves, for sample line discharge isolation and flow control, are mounted on panels above each sink.

Table 3-4 presents the Sample System safety class requirements.

9.3.2.3 Safety Evaluation

The Sampling System is designed to operate manually and on an intermittent basis under conditions ranging from full power operation to cold shutdown. Access to the Containment is not required. The system serves no emergency function. All sample lines which penetrate the Containment are isolated on a Phase A containment isolation signal. The portion of the system utilized as Containment isolation is designed in accordance with applicable safety and code criteria. Adequate provision is made to protect laboratory personnel and prevent the spread of contamination from the sample room. Operating procedures specify the precautions to be observed when purging and drawing samples.

9.3.2.4 Tests and Inspections (Sampling System)

The systems are fully tested and inspected before initial operation. Adequate operating performance monitoring assures system integrity.

9.3.2.5 Instrumentation Application

Sufficient instrumentation is included in the system to assure satisfactory operation.

9.3.3 Equipment and Floor Drainage System

9.3.3.1 Design Bases

Section 11.2, contains a discussion of the different means of processing liquid wastes. The Equipment and Floor Drainage system provides the means by which the liquid wastes are appropriately segregated and transported to the liquid waste systems in order to minimize the liquid and gaseous radioactive releases. This system accomplishes this function in a manner that is consistent with normal station operating procedures.

9.3.3.2 System Description

The Equipment and Floor Drainage System piping is embedded in the floor where possible. The various areas of the Equipment and Floor Drainage System are described below.

1. Laundry and Hot Shower Tank Drains

   The drains from the laundry, showers, and sinks that may contain radioactive wastes are piped to the laundry and hot shower tank.

2. Floor Drain Tank Drains

   Most Auxiliary Building floor drains feed either directly to the floor drain tank or indirectly to the tank through the floor drain tank sumps. The Containment floor drains and sumps are piped to the floor drain tank. All equipment that has a low probability of containing tritium or other radioactive elements is drained to the floor drain tank. The residual heat removal and Containment spray pump rooms sump pumps are normally aligned to pump to the floor drain tank.

3. Waste Evaporator Feed Tank Drains

   All equipment that contains tritiated water that may also be aerated is routed either directly to the waste evaporator feed tank or indirectly to the tank through the waste evaporator feed tank sumps. The residual heat removal and containment spray pump room sump pumps can be aligned to pump to the waste evaporator feed tank.

4. Waste Drain Tank Drains

   Equipment that contains water with entrained fission product gases is flushed out with water from the reactor makeup water storage tank into the waste drain tank. After all water containing entrained fission product gases has been removed, the remaining flush water is then drained to the waste evaporator feed tank.

5. Diesel Generator Room Floor Drains

   Each diesel generator area drains to a diesel generator sump system as described in Section 9.5.10.

9.3.3.3 Safety Evaluation

Drains are sized large enough to facilitate draining of equipment in a reasonable time. Sump sizes and sump pump capacities are designed to eliminate undesirable sump pump cycling operation. Two independent means are provided to empty all sumps except the incore instrumentation room sump and the waste evaporator feed tank sumps later in radwaste facility sump. Sump pump capacities are sized large enough to handle a credible rupture or the maximum leakage rate into their respective sumps.

All piping capable of flooding components needed for safe shutdown and accident prevention is designed to seismic Category I regardless of system safety class. This minimizes the potential for flooding safety related components. Good operating practice dictates that system operation be either terminated or quickly diverted from the excessive leaking component or ruptured line. This minimizes the quantity of water available for flooding safety related equipment.

Flood design considerations are discussed in Section 2.4.2.2. Critical station areas are not flooded by a probable maximum flood.

The system accomplishes the necessary segregation of deaerated radioactive liquids, aerated radioactive liquids, cleaning (soapy) liquids that are potentially radioactive, and other potentially radioative liquids required by the liquid waste systems.

9.3.3.4 Tests and Inspections (Equipment & Floor Drainage System)

The system is tested and inspected before initial operation. Adequate operating performance monitoring assures system integrity.

9.3.3.5 Instrumentation Application

Sufficient instrumentation is included in the system to assure satisfactory operation.

9.3.4 Chemical and Volume Control System

The Chemical and Volume Control system, shown in Figure 9-96 and Figure 9-98 is designed to provide the following services to the Reactor Coolant System:

1. maintenance of predetermined water level in the pressurizer, i.e., maintain required water inventory in the Reactor Coolant System,
2. maintenance of seal-water injection flow to the reactor coolant pumps,
3. control of water chemistry conditions, activity level, soluble chemical neutron absorber concentration and makeup, and
4. emergency core cooling (part of the system is shared with the Emergency Core Cooling System).

9.3.4.1 Design Bases

Quantitative design bases are given in Table 9-31 with qualitative descriptions given below.

1. Reactivity Control

   The Chemical and Volume Control System regulates the concentration of chemical neutron absorber in the reactor coolant to control reactivity changes resulting from the change in reactor coolant temperature between cold shutdown and hot full-power operation, burnup of fuel and burnable poisons, buildup of fission products in the fuel, and xenon transients.

a. The Chemical and Volume Control System is capable of borating the Reactor Coolant System through either one of two flow paths and from either one of two boric acid sources.

b. The amount of boric acid stored in the Chemical and Volume Control System always exceeds that amount required to borate the Reactor Coolant system to cold shutdown concentration assuming that the control assembly with the highest reactivity worth is stuck in its fully withdrawn position. This amount of boric acid also exceeds the amount required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

c. The Chemical and Volume Control System is capable of counteracting inadvertent positive reactivity insertion caused by the maximum boron dilution accident (see Chapter 15).

2. Regulation of Reactor Coolant Inventory

The Chemical and Volume Control System maintains the coolant inventory in the Reactor Coolant system within the allowable pressurizer level range for all normal modes of operation including startup from cold shutdown, full power operation and unit cooldown. This system also has sufficient makeup capacity to maintain the minimum required inventory in the event of minor Reactor Coolant System leaks (see Technical Specifications for a discussion of maximum allowable Reactor Coolant System leakage).

The Chemical and Volume Control System flow rate is based on the requirement that it permits the Reactor Coolant System to be heated to or cooled from hot standby condition at the design rate and maintain pressurizer level within the limits of the operating band.

3. Reactor Coolant Purification

The Chemical and Volume Control System removes fission products and corrosion products from the reactor coolant during operation of the reactor. The Chemical and Volume Control System can also remove excess lithium from the reactor coolant, keeping the lithium concentration within the desired limits for pH control (see Table 5-14).

The Chemical and Volume Control System is capable of removing fission and activation products, in ionic form or as particulates, from the reactor coolant in order to provide access to those process lines carrying reactor coolant during operation and to reduce activity releases due to leaks.

4. Chemical Additions for Corrosion Control

The Chemical and Volume Control System provides a means for adding chemicals to the Reactor Coolant System which control the pH of the coolant during initial startup and subsequent operation, scavenge oxygen from the coolant during startup, and control the oxygen level of the reactor coolant due to radiolysis during all operations subsequent to startup.

The Chemical and Volume Control System is capable of maintaining the oxygen content and pH of the reactor coolant within limits specified in Table 5-14.

5. Seal Water Injection

The Chemical and Volume Control System is able to continuously supply filtered water to each reactor coolant pump seal, as required by the reactor coolant pump design.

6. Hydrostatic Testing of the Reactor Coolant System

The Chemical and Volume Control System is capable of supplying water at the maximum test pressure specified to verify the integrity of the Reactor Coolant System.

7. Emergency Core Cooling

The centrifugal charging pumps in the Chemical and Volume Control System also serve as the high-head safety injection pumps in the Emergency Core Cooling System. Other than the centrifugal charging pumps and associated piping and valves, the Chemical and Volume Control System is not required to function during a loss-of-coolant accident. During a loss-of-coolant accident, the Chemical and Volume Control System is isolated except for the centrifugal charging pumps seal injection flow, CCP minflow and the piping in the safety injection path.

9.3.4.2 System Description

The Chemical and Volume Control System is shown in Figure 9-96 and Figure 9-98 with system design parameters listed in Table 9-31. The Chemical and Volume Control System consists of two subsystems: the charging, letdown and seal water system and the chemical control, purification and makeup system.

1. Charging, Letdown and Seal Water System

The charging and letdown functions of the Chemical and Volume Control System are employed to maintain a predetermined water level in the Reactor Coolant System pressurizer, thus maintaining proper reactor coolant inventory during all phases of unit operation. This is achieved by means of a continuous feed and bleed process during which the feed rate is automatically controlled based on pressurizer water level. The bleed rate can be chosen to suit various unit operational requirements by selecting the proper combinations of letdown orifices in the letdown flow path.

Reactor coolant is discharged to the Chemical and Volume Control System from a reactor coolant loop cold leg; it then flows through the shell side of the regenerative heat exchanger where its temperature is reduced by heat transfer to the charging flow passing through the tubes. Letdown isolation valves are interlocked with pressurizer level and low charging flow to prevent uncovering the pressurizer heater elements. The coolant then experiences a large pressure reduction as it passes through the letdown pressure break-down device and flows through the tube side of the letdown heat exchanger where its temperature is further reduced to the operating temperature of the mixed bed demineralizers (105°F). Downstream of the letdown heat exchanger a second pressure reduction occurs. This second pressure reduction is performed by the low pressure letdown valve, the function of which is to maintain upstream pressure which prevents flashing downstream of the letdown orifices. Both the temperature and pressure control functions are performed by the Ovation PCS, using proportional-integral-derivative (PID) controllers, with associated M/A stations.

During normal operation, the coolant then flows through a mixed bed demineralizer. The flow may then pass through the cation bed demineralizer which is used intermittently when additional purification of the reactor coolant is required. The cation bed demineralizer flowrate is limited to 75 gpm.

The coolant then flows through one of the reactor coolant filters and into the volume control tank through a spray nozzle in the top of the tank. The gas space in the volume control tank is normally purged with hydrogen. The partial pressure of hydrogen in the volume control tank determines the concentration of hydrogen dissolved in the reactor coolant.

The charging pumps normally take suction from the volume control tank and return the cooled, purified reactor coolant to the Reactor Coolant System through the charging line. Normal charging flow is handled by one of the three charging pumps. The bulk of the charging flow is pumped back to the Reactor Coolant System through the tube side of the regenerative heat exchanger. The letdown flow in the shell side of the regenerative heat exchanger raises the charging flow to a temperature approaching the reactor coolant temperature. The flow is then injected into a cold leg of the Reactor Coolant System. Two charging paths are provided from a point downstream of the regenerative heat exchanger. A flow path is also provided from the regenerative heat exchanger outlet to the pressurizer spray line. An air operated valve in the spray line may be employed to provide auxiliary spray to the vapor space of the pressurizer during unit cooldown. This provides a means of cooling the pressurizer near the end of unit cooldown, when the reactor coolant pumps are not operating.

A portion of the charging flow is directed to the reactor coolant pumps (nominally 8 gpm per pump) through a seal water injection filter. It is directed down to a point between the pump shaft bearing and the thermal barrier cooling coil. Here the flow splits and a portion (nominally 5 gpm per pump) enters the Reactor Coolant System through the labyrinth seals and thermal barrier. The remainder of the flow is directed up the pump shaft, cooling the lower bearing, and to the number 1 seal leakoff. The number 1 seal leakoff flow discharges to a common manifold, exits from the Containment, and then passes through the seal water return filter and the seal water heat exchanger to the volume control tank or by alternate path to the suction side of the charging pumps. A very small portion of the seal flow leaks through to the number 2 seal. A number 3 seal provides a final barrier to leakage to Containment atmosphere. The number 2 and 3 seal leakoff flows are discharged to the reactor coolant drain tank in the Liquid Waste Recycle System.

An alternate letdown path from the Reactor Coolant System is provided in the event that the normal letdown path is inoperable. Reactor coolant can be discharged from a cold leg to flow through the tube side of the excess letdown heat exchanger where it is cooled by component cooling water. Downstream of the heat exchanger a remote-manual control valve controls the letdown flow. The flow normally joins the number 1 seal discharge manifold and passes through the seal water return filter and heat exchanger to the volume control tank or the suction side of the charging pumps. The flow can also be directed to the reactor coolant drain tank. When the normal letdown line is not available, the normal purification path is also not in operation. Therefore, this alternate condition would allow continued power operation for a limited period of time, dependent on Reactor Coolant System chemistry and activity. The excess letdown flow path is also used to provide additional letdown capability during the final stages of unit heatup. This path removes some of the excess reactor coolant due to expansion of the system as a result of the Reactor Coolant System temperature increase. In this case, the excess letdown is diverted to the reactor coolant drain tank.

Surges in Reactor Coolant System inventory due to load changes are accommodated for the most part in the pressurizer. The volume control tank provides surge capacity for reactor coolant expansion not accommodated by the pressurizer. The volume control tank level control scheme utilizes a primary level control loop and a backup control. If the water level in the volume control tank exceeds the normal operating range, a PID controller modulates a three way valve downstream of the reactor coolant filter to divert a portion of the letdown to the Boron Recycle System. If the high-level limit in the volume control tank is reached, an alarm is actuated in the Control Room and the letdown flow is completely diverted to the Boron Recycle System.

The Boron Recycle System (Section 9.3.6) receives and processes reactor coolant effluent for reuse of the boric acid and purified water. The system decontaminates the effluent by means of demineralization and gas stripping, and uses evaporation to separate and recover the boric acid and reactor makeup water.

Low level in the volume control tank initiates makeup from the reactor makeup control system. If the reactor makeup control system does not supply sufficient makeup to keep the volume control tank level from falling to a lower level, a low alarm is actuated. Manual action may correct the situation or, if the level continues to decrease, an emergency low level signal from the two level channels causes the suction of the charging pumps to be transferred to the refueling water storage tank.

The reciprocating charging pump is used to perform hydrostatic tests which verify the integrity of the Reactor Coolant System. The pump can pressurize the Reactor Coolant System to the maximum designated test pressure. The hydrostatic test was performed prior to initial operation and as part of the periodic Reactor Coolant System in-service inspection program.

2. Chemical Control, Purification and Makeup System

pH Control

The pH control chemical employed is lithium hydroxide. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/Inconel systems. In addition, lithium-7 is produced in the core region due to irradiation of the dissolved boron in the coolant.

The concentration of lithium-7 in the Reactor Coolant System is maintained in a range specified for pH control. If lithium needs to be removed from the reactor coolant system, this can be accomplished by placing a standby mixed bed or cation bed demineralizer in service for a short period of time. Diverting letdown to the recycle holdup tank and making up to the NV system is also a means of reducing lithium if the mixed bed and cation bed demineralizers are not available. If the concentration of lithium-7 is below the specified limits, lithium hydroxide can be introduced into the Reactor Coolant System via the charging flow. The solution is prepared in the laboratory and poured into the chemical mixing tank. Reactor makeup water is then used to flush the solution to the suction manifold of the charging pumps. An orifice in the flush line limits the flow rate to approximately 2.5 gpm.

Oxygen Control

During reactor startup from the cold condition, hydrazine is employed as an oxygen scavenging agent. The hydrazine solution is introduced into the Reactor Coolant system in the same manner as described above for the pH control agent. Hydrazine is not employed at any time other than startup from the cold shutdown state.

Dissolved hydrogen is employed to control and scavenge oxygen produced due to radiolysis of water in the core region. Sufficient partial pressure of hydrogen is maintained in the volume control tank such that the specified equilibrium concentration of hydrogen is maintained in the reactor coolant. A pressure control valve maintains a minimum pressure in the vapor space of the volume control tank. This valve can be adjusted to provide the correct equilibrium hydrogen concentration for NC System (Table 5-14).

Reactor Coolant Purification

Mixed bed demineralizers and a cation bed demineralizer are provided in the letdown line to provide cleanup of the letdown flow. The demineralizers remove ionic corrosion products and fission products. Normally, a mixed bed demineralizer is in continuous service and can be supplemented intermittently by the cation bed demineralizer, if necessary, for additional purification. The cation resin removes principally cesium and lithium isotopes from the purification flow. The second mixed bed demineralizer normally serves as a standby unit for use if the operating demineralizer becomes exhausted during operation.

A further cleanup feature is provided for use during cold shutdown and residual heat removal. A remote operated valve admits a bypass flow from the Residual Heat Removal System into the letdown line upstream of the letdown heat exchanger. The flow passes through the heat exchanger, mixed bed demineralizer(s) and the reactor coolant filter to the volume control tank. The fluid is then returned to the Reactor Coolant system via the normal charging route.

Filters are provided at various locations to assure filtration of particulate and resin fines and to protect the seals on the reactor coolant pumps.

Fission gases are removed from the system by routine purging of the volume control tank to the Waste Gas System.

Chemical Shim and Reactor Coolant Makeup

The soluble neutron absorber (boric acid) concentration is controlled by the reactor makeup control subsystem of the Chemical and Volume Control System. The reactor makeup control subsystem is also used to maintain proper reactor coolant inventory. In addition, for emergency boration and makeup, the capability exists to provide refueling water or 4 weight percent boric acid solution directly to the suction of the charging pumps.

One boric acid tank is normally aligned to supply each unit. Two pumps are provided for each unit. On each unit, one pump is normally aligned to provide boric acid to the boric acid blender, with the second pump in reserve.

On a demand signal by the reactor makeup control system, the pump starts and delivers boric acid to the boric acid blender. The pump can also be used to recirculate the boric acid tank fluid.

All portions of the Chemical and Volume Control System which contain concentrated boric acid solution (4 weight percent boric acid) are located within a heated area in order to maintain solution temperature at ≥65°F. If a portion of the system which normally contains concentrated boric acid solution is not located in a heated area, it is provided with some other means (e.g., heat tracing) to maintain solution temperature at ≥65°F.

The reactor makeup water pumps, taking suction from the reactor makeup water storage tank, are employed for various makeup and flushing operations throughout the systems. One of these pumps also starts on demand from the reactor makeup control system and provides flow to the boric acid blender.

The flow from the boric acid blender is directed to either the suction manifold of the charging pumps or the volume control tank through the letdown line and spray nozzle.

During reactor operation, changes are made in the reactor coolant boron concentration for the following conditions:

a. Reactor startup - boron concentration must be decreased from shutdown concentration to achieve criticality.

b. Load follow - boron concentration must be either increased or decreased to compensate for the xenon transient following a change in load.

c. Fuel burnup - boron concentration must be decreased to compensate for fuel burnup and the associated buildup of fission products in the fuel.

d. Cold shutdown - boron concentration must be increased to the cold shutdown concentration.

The Boron Thermal Regeneration System was used to control boron concentration to compensate for xenon transients during load follow operations. Boron thermal regeneration could also have been used during dilution to reduce the amount of effluent to be processed by the Boron Recycle System. The Boron Thermal Regeneration System has been functionally disabled and is no longer used to carry out these functions.

The reactor makeup control subsystem consists of a group of instruments arranged to provide a manually pre-selected makeup composition to the charging pump suction header or the volume control tank. The makeup control functions are those of maintaining desired operating fluid inventory in the volume control tank and adjusting reactor coolant boron concentration for reactivity control. The system can operate in the following modes:

a. Automatic Makeup

The “automatic makeup” mode of operation of the reactor makeup control subsystem provides boric acid solution preset to match the boron concentration in the Reactor Coolant System. The automatic makeup compensates for minor leakage of reactor coolant without causing significant changes in the coolant boron concentration.

Under normal unit operating conditions, the mode selector switch and makeup stop valves are set in the “automatic makeup” position. A preset low level signal from the volume control tank level controller in the Ovation PCS causes the automatic makeup control action to start a reactor makeup water pump, start a boric acid transfer pump, open the makeup stop valve to the charging pump suction, open the concentrated boric acid control valve and the reactor makeup water control valve. The flow controllers then blend the makeup stream according to the preset concentration. Makeup addition to the charging pump suction header causes the water level in the volume control tank to rise. At a preset high level point, the makeup is stopped, the reactor makeup water pump stops, the reactor makeup water control valve closes, the boric acid transfer pump stops, the concentrated boric acid control valve closes and the makeup stop valve to charging pump suction closes.

If the automatic makeup fails or is not aligned for operation and the tank level continues to decrease, a low level alarm is actuated. Manual action may correct the situation or, if the level continues to decrease, an emergency low level signal from both channels opens the stop valves in the refueling water supply line to the charging pumps and closes the stop valves in the volume control tank outlet line.

b. Dilution

The “dilute” mode of operation permits the addition of a pre-selected quantity of reactor makeup water at a pre-selected flow rate to the Reactor Coolant System. The operator sets the mode selector switch to “dilute,” the reactor makeup water flow controller set point to the desired flow rate, the total blender counter to the desired quantity and initiates system start. This opens the reactor makeup water control valve to the volume control tank and starts a reactor makeup water pump which delivers water to the volume control tank. From here the water goes to the charging pump suction header. Excessive rise of the volume control tank water level is prevented by automatic actuation (by the tank level controller) of a three-way diversion valve which routes the reactor coolant letdown flow to the Boron Recycle System.

When the preset quantity of water has been added, the Ovation PCS total blender counter causes the pump to stop and the control valve to close.

Dilution could also have been accomplished by operating the Boron Thermal Regeneration System in the boron storage mode. The Boron Thermal Regeneration System has been functionally disabled and is no longer used for this function.

c. Alternate Dilution

The “alternate dilute” mode of operation is similar to the dilute mode except a portion of the dilution water flows directly to the charging pump suction and a portion flows into the volume control tank via the spray nozzle and then flows to the charging pump suction.

d. Boration

The “borate” mode of operation permits the addition of a pre-selected quantity of concentrated boric acid solution at a pre-selected flow rate to the Reactor Coolant System. The operator sets the mode selection switch to “borate”, the concentrated boric acid flow controller set point to the desired flow rate, the concentrated boric acid counter to the desired quantity, and starts the “borate” mode of operation. This opens the makeup stop valve to the charging pumps suction and starts the selected boric acid transfer pumps, which delivers a 4 weight percent boric acid solution to the charging pumps suction header. The total quantity added in most cases is so small that it has only a minor effect on the volume control tank level. When the preset quantity of concentrated boric acid solution is added, the counter stops the boric acid transfer pump and closes the makeup stop valve to the suction of the charging pumps.

Boration could also have been accomplished by operating the Boron Thermal Regeneration System in the boron release mode. The Boron Thermal Regeneration System has been functionally disabled and is no longer used for this function.

e. Manual

The “manual” mode of operation permits the addition of a pre-selected quantity and blend of boric acid solution to the refueling water storage tank, to the recycle holdup tanks in the Boron Recycle System, or to some other location via a temporary connection. While in the manual mode of operation, automatic makeup to the Reactor Coolant System is precluded. The discharge flow path must be prepared by opening manual valves in the desired path.

The operator then sets the mode selector switch to “manual”, the boric acid and reactor makeup water flow controllers to the desired flow rates, the boric acid and total blender flow counters to the desired quantities and actuates the makeup start switch. The start switch actuates the boric acid flow control valve and the reactor makeup water flow control valve to the boric acid blender and starts the pre-selected reactor makeup water pump and boric acid transfer pump.

When the preset quantities of boric acid and reactor makeup water have been added, the pumps stop and the boric acid and reactor makeup water flow control valves close. This operation may be stopped manually by actuating the makeup stop switch.

If either counter is satisfied before the other has recorded its required total, the pump and valve associated with the counter which has been satisfied terminates flow. The flow controlled by the other counter continues until that counter is satisfied.

The reactor makeup control is provided with alarm functions to call the operator's attention to the following conditions:

a. deviation of total reactor makeup water flow rate from the control set point,

b. deviation of concentrated boric acid flow rate from control set point,

c. high level in the volume control tank. This alarm indicates that the level in the tank is approaching high level and a resulting 100 percent diversion of the letdown stream to the Boron Recycle System, and

d. low level in the volume control tank. This alarm indicates that the level in the tank is approaching emergency low level and a resulting realignment of charging pump suction to the refueling water storage tank.

3. Layout

The volume control tank is located above the charging pumps to provide sufficient net positive suction head (NPSH). All parts of the charging and letdown system are shielded as necessary to limit dose rates during operation with one percent fuel defects assumed. The regenerative heat exchanger, excess letdown heat exchanger, letdown orifices, and seal bypass orifices are located within the reactor Containment. All other system equipment is located inside the Auxiliary Building.

4. Component Description

A summary of principal component design parameters is given in Table 9-32, and safety classifications and design codes are given in Section 3.2.

All Chemical and Volume Control System piping that handles radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanged connections are required to facilitate equipment removal for maintenance and hydrostatic testing.

Charging Pumps

Three charging pumps are provided to inject coolant into the Reactor Coolant System. Two of the pumps are of the single speed, horizontal, centrifugal type and the third is a positive displacement (reciprocating) pump equipped with variable speed drive. All parts in contact with the reactor coolant are fabricated of austenitic stainless steel or other material of adequate corrosion resistance. The centrifugal pump seals and the reciprocating pump stuffing box are provided with leakoffs to collect the leakage. The reciprocating pump design prevents lubricating oil from contaminating the charging flow. There is a minimum flow recirculation line to protect the centrifugal charging pumps from a closed discharge valve condition.

Charging flow rate is controlled by the Ovation PCS using a PID controller and a pressurizer level signal. The means of flow control for the reciprocating pump is by variation of pump speed. The reciprocating charging pump can also be used to hydrotest the Reactor Coolant System. When operating a centrifugal charging pump, the flow paths remain the same but charging flow control is accomplished by a modulating valve on the discharge side of the centrifugal pumps. The centrifugal charging pumps also serve as high head safety injection pumps in the Emergency Core Cooling System.

Boric Acid Transfer Pumps

A BAT and two BA pumps are provided for each unit. One BA pump is normally in recirc to supply boric acid to the boric acid blender, while the second serves as a standby. Manual or automatic initiation of the reactor coolant makeup system starts a pump to provide normal makeup of boric acid solution through the boric acid blender. Emergency boration, supplying 4 weight percent boric acid solution directly to the suction of the charging pumps, can be accomplished by manually starting either pump.

The pumps are located in a heated area to prevent crystallization of the boric acid solution. All parts in contact with the solution are of austenitic stainless steel.

Regenerative Heat Exchanger

The regenerative heat exchanger is designed to recover heat from the letdown flow by reheating the charging flow, which reduces thermal shock on the charging penetrations into the reactor coolant loop piping.

The letdown stream flows through the shell of the regenerative heat exchanger and the charging stream flows through the tubes. The unit is constructed of austenitic stainless steel, and is of all welded construction.

The temperatures of both outlet streams from the heat exchanger are monitored with indication given in the Control Room. A high temperature alarm is given on the main control board if the temperature of the letdown stream exceeds desired limits.

Letdown Heat Exchanger

The letdown heat exchanger cools the letdown stream to the operating temperature of the mixed bed demineralizers. Reactor coolant flows through the tube side of the exchanger while component cooling water flows through the shell side. All surfaces in contact with the reactor coolant are austenitic stainless steel, and the shell is carbon steel.

The low pressure letdown valve maintains the pressure of the letdown flow, located downstream of the heat exchanger, in a range sufficiently high to prevent two phase flow.

The letdown temperature control indicates and controls the temperature of the letdown flow exiting from the letdown heat exchanger. The temperature sensor, which is part of the Chemical and Volume Control System, provides input to the PID controller in the Ovation PCS. The exit temperature is controlled by regulating the component cooling water flow through the letdown heat exchanger by using the control valve located in the component cooling water discharge line. Temperature indication is provided on the main control board.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools reactor coolant letdown flow at a rate which is equivalent to the nominal seal injection flow which flows downward through the reactor coolant pump labyrinth seals.

The excess letdown heat exchanger can be employed either when normal letdown is temporarily out of service to maintain the reactor in operation or it can be used to supplement maximum letdown during the final stages of heatup. The letdown flows through the tube side of the unit and component cooling water is circulated through the shell. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel. All tube joints are welded.

A temperature detector measures temperature of excess letdown downstream of the excess letdown heat exchanger. Temperature indication and high temperature alarm are provided on the main control board.

A pressure sensor indicates the pressure of the excess letdown flow downstream of the excess letdown heat exchanger and excess letdown control valve. Pressure indication is provided on the main control board.

A flow sensor indicates the flow of the excess letdown heat exchanger. Flow indication is provided on the main control board.

Seal Water Heat Exchanger

The seal water heat exchanger is designed to cool fluid from three sources: reactor coolant pump seal water returning to the Chemical and Volume Control System, reactor coolant discharged from the excess letdown heat exchanger, and centrifugal charging pump recirc flow. Reactor coolant flows through the tube side of the heat exchanger and component cooling water is circulated through the shell. The design flow rate is equal to the sum of the excess letdown flow, maximum design reactor coolant pump seal leakage, and bypass flow from one centrifugal charging pump. The unit is designed to cool the above flow to the temperature normally maintained in the volume control tank. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

Volume Control Tank

The volume control tank provides surge capacity for part of the reactor coolant expansion volume not accommodated by the pressurizer. When the level in the tank reaches the high level setpoint, the remainder of the expansion volume is accommodated by diversion of the letdown stream to the Boron Recycle System. It also provides a means for introducing hydrogen into the coolant to maintain the required equilibrium hydrogen concentration of the NC system (Table 5-14) and is used for degassing the reactor coolant. It also serves as a head tank for the charging pumps.

A spray nozzle located inside the tank on the letdown line nozzle provides liquid to gas contact between the incoming fluid and the hydrogen atmosphere in the tank.

A remotely operated vent valve, discharging to the Waste Gas System permits normal removal of gaseous fission products which are stripped from the reactor coolant and collected in this tank. Relief protection, gas space sampling, and nitrogen purge connections are also provided. The tank normally accepts the seal water return flow from the reactor coolant pumps although this flow can be aligned directly to the suction of the charging pumps.

Volume control tank pressure and temperature are monitored with indication given in the Control Room. Alarm is given in the Control Room for high and low pressure conditions and for high temperature.

Two level channels govern the water inventory in the volume control tank. These channels provide local and remote level indication, level alarms, level control, makeup control, and emergency makeup control.

If the volume control tank level rises above the normal operating range, one channel provides an analog signal to the proportional controller which modulates the three-way valve downstream of the reactor coolant filter to maintain the volume control tank level within the normal operating band. The three-way valve can split letdown flow so that a portion goes to the Boron Recycle System and a portion to the volume control tank. The controller would operate in this fashion during a dilution operation when reactor makeup water is being fed to the volume control tank from the reactor control system.

If the modulating function of the channel fails and the volume control tank level continues to rise, the high level alarm alerts the operator to the malfunction and the letdown flow can be manually diverted to the holdup tanks. If no action is taken by the operator and the tank level continues to rise, the full letdown flow is automatically diverted.

During normal power operation, a low level in the volume control tank initiates automatic makeup which injects a pre-selected blend of boric acid and water into the charging pump suction header. When the volume control tank is restored to normal, auto makeup stops.

If the automatic makeup fails or is not aligned for operation and the tank level continues to decrease, a low level alarm is actuated. Manual action may correct the situation or, if the level continues to decrease, an emergency low level signal from both channels opens the stop valves in the refueling water supply line and closes the stop valves in the volume control tank outlet line.

Boric Acid Tanks

During normal operation, one tank supplies boric acid solution for each unit. Each tank is designed to store sufficient boric acid solution for a cold shutdown from full power operation immediately following refueling with the most reactive control rod not inserted, plus operating margins.

The concentration of boric acid solution in storage is maintained between 4 and 4.4 percent by weight. Periodic manual sampling and corrective action, if necessary, assures that these limits are maintained. As a consequence, measured amounts of boric acid solution can be delivered to the reactor coolant to control the concentration.

A temperature sensor provides temperature measurement of each tank's contents. Temperature indication is provided as well as high and low temperature alarms which are indicated on the main control board.

Each boric acid tank is provided with two redundant level detectors. Level indication with high, low, low-low level and empty alarms is provided on the main control board. The high level alarm is to protect against overfilling the tank. The low level alarm indicates that the tank volume is nearing the minimum cold shutdown volume. The low-low level alarm indicates the minimum level of boric acid in the tank to ensure sufficient boric acid to provide for a cold shutdown with one stuck rod. The empty level alarm is to alert the operator that one transfer pump will empty the tank in 10 minutes after which pump cavitation begins.

Batching Tank

The batching tank is used for mixing a makeup supply of boric acid solution for transfer to the boric acid tanks. The tank may also be used for solution storage.

A local sampling point is provided for verifying the solution concentration prior to transferring it out of the tank. The tank is provided with an agitator to improve mixing during batching operations and a steam jacket for heating the boric acid solution.

Chemical Mixing Tank

The primary use of the chemical mixing tank is in the preparation of caustic solutions for pH control, crudburst peroxide addition and hydrazine solution for oxygen scavenging.

Mixed Bed Demineralizers

Two flushable mixed bed demineralizers assist in maintaining reactor coolant purity. A lithium-form cation resin and hydroxyl-form anion resin are charged into the demineralizers. The anion resin is converted to the borate form during operation. Both types of resin remove fission and corrosion products. The resin bed is designed to provide a decontamination factor of ten for most fission products (exceptions are Cesium, Yttrium, and Molybdenum). A freshly loaded mixed bed can be intermittently used to remove excess lithium-7 for pH control. The standby mixed bed demineralizer with a fresh resin bed can also be used to reduce boron concentration of the letdown flow stream. This method of reactor coolant system de-boration is typically utilized near end of cycle, in-place of normal dilution means.

Each demineralizer has sufficient capacity for approximately one core cycle with one percent of the rated core thermal power being generated by defective fuel rods.

A temperature sensor measures temperature of the letdown flow downstream of the letdown heat exchanger and controls the letdown flow to the mixed bed demineralizers by means of a three-way valve. If the letdown temperature exceeds the allowable resin operating temperature, the flow is automatically bypassed around the demineralizers. Temperature indication and high alarm are provided on the main control board. The air operated three-way valve failure mode directs flow to the volume control tank.

Cation Bed Demineralizers

A flushable cation resin bed in the hydrogen form is located downstream of the mixed bed demineralizers and is used intermittently to control the concentration of excess of Li^7 which builds up in the coolant from the B^{10} $(n\gamma)$ Li^7 reaction. The demineralizer also has sufficient capacity to maintain the cesium-137 concentration in the coolant below 1.0 μCi/cc with 1 percent defective fuel. The resin bed is designed to reduce the concentration of ionic isotopes, particularly cesium, yttrium, and molybdenum by a minimum factor of 10.

The cation bed demineralizer has sufficient capacity for approximately one core cycle with one percent of the rated core thermal power being generated by defective fuel rods.

Reactor Coolant Filters

Two reactor coolant filters are located on the letdown line. One filter can be aligned as a pre-filter or a post-filter to the demineralizers and the second can function as a demineralizer post-filter. The filters collect resin fines and particulates from the letdown stream. The design flow capacity of each filter is sufficient to accommodate the maximum purification flow rate (Table 9-31).

A differential pressure indicator is provided to show the filter differential pressure and provide an alarm on high differential pressure.

Seal Water Injection Filters

Two seal water injection filters are located in parallel in a common line to the reactor coolant pump seals; they collect particulate matter that could be harmful to the seal faces. Each filter is sized to accept flow in excess of the normal seal water flow requirements.

A differential pressure indicator monitors the pressure drop across each seal water injection filter and gives local indication with high differential pressure alarm on the main control board.

Seal Water Return Filter

The filter collects particulates from the reactor coolant pump seal water return and from the excess letdown flow. The filter is designed to pass flow in excess of the sum of the excess letdown flow and the maximum design leakage from the reactor coolant pump seals.

A differential pressure indicator is provided to show the differential pressure across the filter.

Boric Acid Filter

The boric acid filter collects particulates from the boric acid solution being pumped from a boric acid tank by the boric acid transfer pumps. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously.

A differential pressure indicator is located on the boric acid filter to provide filter differential pressure.

Boric Acid Blender

The boric acid blender promotes thorough mixing of boric acid solution and reactor makeup water for the reactor coolant makeup circuit. The blender consists of a conventional pipe-tee fitted with a perforated tube insert. The blender decreases the pipe length required to homogenize the mixture for taking representative local sample. A sample point is provided in the piping just downstream of the blender.

Letdown Pressure-Breakdown Devices

Three different pressure-breakdown devices are arranged in parallel and serve to reduce the pressure of the letdown stream to a value compatible with the letdown heat exchanger design. The pressure-breakdown devices accommodate normal letdown flow, less than normal letdown flow and a variable letdown flow control. Any combination of these devices can be utilized to increase letdown flow such as during reactor heatup and maximum purification operations. The pressure-breakdown devices are placed in and taken out of service by remote manual operation of their respective isolation valves. The control valve is also controlled by remote manual operation.

A low pressure letdown controller controls the pressure downstream of the letdown heat exchanger to prevent flashing of the letdown liquid. Pressure indication is provided on the main control board and a computer alarm is provided on high pressure.

Valves

Typically, valves, other than diaphragm valves, that perform a modulating function are equipped with a stuffing box containing two sets of packing and an intermediate leakoff connection. Valves are normally installed such that, when closed, the high pressure is not on the packing. Basic material of construction is stainless steel for all valves which handle radioactive liquid or boric acid solutions.

Isolation valves are provided for all lines entering the reactor Containment. These valves are discussed in detail in Section 6.2.4.

Relief valves are provided for the following lines and components that might be pressurized above design pressure by improper operation or component malfunction:

a. Charging Line Downstream of Regenerative Heat Exchanger

   If the charging side of the regenerative heat exchanger is isolated while the hot letdown flow continues at its maximum rate, the volumetric expansion of coolant on the charging side of the heat exchanger is relieved to the Reactor Coolant System through a spring loaded check valve. The spring in the valve is designed to permit the check valve to open in the event that the differential pressure exceeds the design pressure differential.

b. Letdown Line Downstream of Letdown Pressure-Breakdown Devices

   The pressure relief valve down stream of the letdown pressure-breakdown devices protects the low pressure piping and the letdown heat exchanger from overpressure when the low pressure piping is isolated. The capacity of the relief valve is adequate to accommodate the maximum letdown flow through all three flow paths. The valve set pressure is equal to the design pressure of the letdown heat exchanger tube side.

c. Letdown Line Downstream of Low Pressure Letdown Valve

   The pressure relief valve downstream of the low pressure letdown valve protects the low pressure piping, demineralizers, and filter from overpressure when this section of the system is isolated. The overpressure may result from leakage through the low pressure letdown valve. The capacity of the relief valve exceeds the maximum flow rate through all letdown pressure-breakdown devices. The valve set pressure is less than the design pressure of the demineralizers.

d. Volume Control Tank

The relief valve on the volume control tank permits the tank to be designed for a lower pressure than the upstream equipment. This valve has a capacity greater than the summation of the following items: maximum letdown, maximum seal water return, excess letdown and nominal flow from one reactor makeup water pump. The valve set pressure equals the design pressure of the volume control tank.

e. Charging Pump Suction

A relief valve on the charging pump suction header relieves pressure that may build up if the suction line isolation valves are closed or if the system is overpressurized. The valve set pressure is equal to the design pressure of the associated piping and equipment.

f. Seal Water Return Line (Inside Containment)

This relief valve is designed to relieve overpressurization in the seal water return piping inside the Containment if the motor-operated isolation valve is closed. The valve is designed to relieve the total leakoff flow from the No. 1 seals of the reactor coolant pumps plus the design excess letdown flow. The valve is set to relieve at the design pressure of the piping.

g. Seal Water Return Line (Charging Pumps Bypass Flow)

This relief valve protects the seal water heat exchanger and its associated piping from overpressurization. If either of the isolation valves for the heat exchanger are closed and if the bypass line is closed, the piping could be overpressurized by the bypass flow from the centrifugal charging pumps. The valve is sized to handle the full bypass flow with all centrifugal pumps running. The valve is set to relieve at the design pressure of the heat exchanger.

h. Positive Displacement Charging Pump Discharge

The pressure relief valve on the positive displacement charging pump discharge line relieves the rated pumping capacity if the pump is started with the discharge isolation valve closed. The set pressure of the valve is equal to the design pressure of the pump discharge piping.

i. Steam Line to Batching Tank

The relief valve on the steam line to the batching tank protects the low pressure piping and batching tank heating jacket from overpressure when the condensate return line is isolated. The capacity of the relief valve equals the maximum expected steam inlet flow. The set pressure equals the design pressure of the heating jacket.

j. Letdown Line Control Valve

A letdown control valve in parallel with the other letdown pressure-breakdown devices is throttled to allow a controlled letdown flow restoration. This valve is only required to operate when returning the letdown line to service after the loss of the letdown and is not required for a normal startup. The control valve also provides added flexibility for operating the system.

k. Pressurizer Auxiliary Spray Control Valve

With the reactor coolant pumps shut down, cooling of the pressurizer liquid is accomplished by charging through the auxiliary spray control valve from the residual heat removal pumps. The rate of spray is manually controlled from the control room.

Piping

All Chemical and Volume Control System piping handling radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanged connections are required to facilitate equipment removal for maintenance and hydrostatic testing.

5. <u>System Operation</u>

Reactor Startup

Reactor startup is defined as the operations which bring the reactor from cold shutdown to normal operating temperature and pressure.

It is assumed that:

a. normal residual heat removal is in progress,

b. Reactor Coolant System boron concentration is at the cold shutdown concentration,

c. reactor makeup control subsystem is set to provide makeup at the cold shutdown concentration,

d. Reactor Coolant System is either water solid or drained to minimum level for the purpose of refueling or maintenance. If the Reactor Coolant System is water solid, system pressure is controlled by letdown through the Residual Heat Removal System and through the low pressure letdown valve in the letdown line, and

e. the charging and letdown lines of the Chemical and Volume Control System are filled with coolant at the cold shutdown boron concentration. The letdown orifice isolation valves are closed.

If the Reactor Coolant System requires filling and venting, the procedure is as follows:

a. One charging pump is started, which provides blended flow from the reactor makeup control system at the cold shutdown boron concentration,

b. the vents on the head of the reactor vessel and pressurizer are opened, and

c. the Reactor Coolant System is filled and the vents closed.

The system pressure is raised by using the charging pump and controlled by the low pressure letdown valve. When the system is adequate for operation of the reactor coolant pumps, seal water flow to the pumps is established and the pumps are operated and vented sequentially until all gases are cleared from the system. Final venting takes place at the pressurizer.

After the filling and venting operations are completed, charging and letdown flows are established. All pressurizer heaters are energized and the reactor coolant pumps are employed to heat the system. After the reactor coolant pumps are started, the residual heat removal pumps are stopped, but pressure control via the Residual Heat Removal System and the low pressure letdown line is continued as the pressurizer steam bubble is formed. At this point, steam formation in the pressurizer is accomplished by manual control of the charging flow and automatic pressure control of the letdown flow. When the pressurizer water level reaches the no-load programmed set point, the pressurizer level control is shifted to control the charging flow to maintain programmed level. The Residual Heat Removal System is the isolated from the Reactor Coolant System.

The reactor coolant boron concentration is now reduced by operating the reactor makeup control system in the “dilute” mode and when the resin beds are saturated, washing off the beds to the Boron Recycle System. The reactor coolant boron concentration is corrected to the point where the control rods may be withdrawn and criticality achieved. Nuclear heatup may then proceed with corresponding manual adjustment of the reactor coolant boron concentration to balance the temperature coefficient effects and maintain the control rods within their operating range. During heatup, the appropriate combination of letdown orifices and/or control valve is used to provide necessary letdown flow.

Prior to or during the heating process, the Chemical and Volume Control System is employed to obtain the correct chemical properties in the Reactor Coolant System. The reactor makeup control subsystem of the Chemical and Volume Control System is operated on a continuing basis to assure correct control rod position. Chemicals are added through the chemical mixing tank as required to control reactor coolant chemistry such as pH and dissolved oxygen content. Hydrogen overpressure is established in the volume control tank to assure the appropriate hydrogen concentration in the reactor coolant.

Power Generation and Hot Shutdown Operation

a. Base Load

   At a constant power level, the rates of charging and letdown are dictated by the requirements for seal water to the reactor coolant pumps and the normal purification of the Reactor Coolant System. One charging pump is employed and charging flow is controlled automatically from pressurizer level. The only adjustments in boron concentration necessary are those to compensate for core burnup. These adjustments are made at infrequent intervals to maintain the control groups within their allowable limits.

   Rapid variations in power demand are accommodated automatically by control rod movement. If variations in power level occur, and the new power level is sustained for long periods, some adjustment in boron concentration may be necessary to maintain the control groups within their maneuvering band.

   During normal operation, normal letdown flow is maintained and one mixed bed demineralizer is in service. Reactor coolant samples are taken periodically to check boron concentration, water quality, pH and activity level. The charging flow through the discharge header flow control valve to the Reactor Coolant System is controlled automatically by the pressurizer level control signal.

b. Load Follow

   A power reduction initially causes a xenon buildup followed by xenon decay to a new lower equilibrium value. The reverse occurs if the power level increases; initially, the xenon level decreases and then it increases to a new and higher equilibrium value associated with the amount of the power level change.

   The reactor makeup control subsystem may be used to vary the boron concentration in the reactor coolant to compensate for xenon transients occurring when reactor power level is changed.

   The most important information available to the station operator, enabling him to determine whether dilution or boration of the Reactor Coolant system is necessary, is the position of the control rods within the maneuvering band. If, for example, the control rods are moving down into the core, and are approaching the bottom of the maneuvering band, the operator must borate the reactor coolant to bring the rods outward. If not, the control rods may move into the core beyond the shutdown limit. However, if the rods are moving out of the core, the operator dilutes the reactor coolant to keep the rods from moving above the top of the maneuvering band. Keeping the control rods below or at the top of the maneuvering band assures the capability of immediate return to full power. However, violation of the upper limit of the maneuvering band is not safety related and is allowed. With the control rods above the top of the maneuvering band the reactor cannot return to full power immediately; it can return to some intermediate power level immediately and then reach full power at some rate determined by the xenon burnout transient.

During periods of unit loading, the reactor coolant expands as its temperature rises. The pressurizer absorbs most of this expansion as the level controller raises the level setpoint to the increased level associated with the new power level. The remainder of the excess coolant is letdown and stored in the volume control tank. During this period, the flow through the letdown orifice remains constant and the charging flow is reduced by the pressurizer level control signal, resulting in an increased temperature at the regenerative heat exchanger outlet. The temperature controller downstream from the letdown heat exchanger increases the component cooling water flow to maintain the desired letdown temperature.

During periods of unit unloading, the charging flow is increased to make up for the coolant contraction not accommodated by the programmed reduction in pressurizer level.

c. Hot Shutdown

If required, for periods of maintenance, or following spurious reactor trips, the reactor can be held subcritical, but with the capability to return to full power within the period of time it takes to withdraw control rods. During this hot shutdown period, temperature is maintained at no-load T_{avg} by initially dumping steam to remove core residual heat, or at later stages, by running reactor coolant pumps to maintain system temperature.

Following shutdown, xenon buildup occurs and increases the degree of shutdown; i.e., initially, with initial xenon concentration and all control rods inserted, the core is maintained at a minimum of 1 percent $\Delta k/k$ subcritical. The effect of xenon buildup is to increase this value to a maximum at about eight hours following shutdown. If hot shutdown is maintained past this point, xenon decay results in a decrease in degree of shutdown. Therefore, boration of the reactor coolant is necessary to counteract the xenon decay and maintain shutdown.

If rapid recovery is required, dilution of the system may be performed to counteract this xenon buildup. However, after the xenon concentration reaches a peak, boration must be performed to maintain the reactor subcritical.

Reactor Shutdown

Reactor shutdown is defined as the operations which bring the reactor to cold shutdown.

Before initiating a cold shutdown, the Reactor Coolant System hydrogen concentration is reduced by replacing the volume control tank hydrogen atmosphere with nitrogen and by continuous purging to the Waste Gas Processing System.

Before cooldown and depressurization of the reactor unit is initiated, the reactor coolant boron concentration is increased to the cold shutdown value. The operator sets the reactor makeup control system to “borate”, selects the volume of concentrated boric acid solution necessary to perform the boration, selects the desired flow rate and actuates makeup start. After the boration is completed and reactor coolant samples verify that the concentration is correct, the operator resets the reactor makeup control system for leakage makeup and system contraction at the shutdown reactor coolant boron concentration.

Contraction of the coolant during cooldown of the Reactor Coolant System results in actuation of the pressurizer level control to maintain normal pressurizer water level. The charging flow is increased, relative to letdown flow, and results in a decreasing volume control tank level. The volume control tank level controller automatically initiates makeup to maintain the inventory.

After the Residual Heat Removal System is placed in service and the reactor coolant pumps are shut down, further cooling of the pressurizer liquid is accomplished by charging through the auxiliary spray line. The auxiliary pressurizer spray flow is supplied from the discharge of the residual heat removal pumps and upstream of the residual heat removal heat exchanger. A control valve is provided in this line to provide added flexibility for pressurizer cooling. The above provisions also result in a less severe thermal transient of the auxiliary pressurizer spray lines. Coincident with unit cooldown a portion of the reactor coolant flow is diverted from the Residual Heat Removal System to the Chemical and Volume Control System for cleanup. Demineralization of ionic radioactive impurities and stripping of fission gases reduce the reactor coolant activity level sufficiently to permit personnel access for refueling or maintenance operations.

9.3.4.3 Safety Evaluation

1. Reactivity Control

The boration subsystem of the chemical and volume control system (CVCS) provides the means to meet one of the functional requirements of the CVCS, i.e., to control the chemical neutron absorber (boron) concentration in the reactor coolant system (RCS) and to help maintain the shutdown margin.

Any time that the unit is at elevated temperatures, the quantity of boric acid stored and ready for injection always exceeds that quantity required for the normal cold shutdown assuming that the control assembly of greatest worth is in its fully withdrawn position. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay. An adequate quantity of boric acid is also available in the refueling water storage tank to achieve cold shutdown.

The boration subsystem is not assumed to be operable to mitigate the consequences of any design basis accident (DBA) or transient. The boration subsystem is not part of a primary success path in the mitigation of a DBA or transient.

When the reactor is subcritical; i.e., during cold or hot shutdown, refueling and approach to criticality, the neutron source multiplication is continuously monitored and indicated. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to give ample time to start a corrective action (boron dilution stop) to prevent the core from becoming critical.

In the case of a malfunction of the CVCS that causes a boron dilution event, the operator must take manual action to close the appropriate valves in the reactor makeup system before the shutdown margin is lost. Operation of the boration subsystem is not assumed to mitigate this event.

The rate of boration, with a single boric acid transfer pump and charging pump operating, is sufficient to take the reactor from full power operation to 1 percent shutdown in the hot condition, with no rods inserted, in less than 90 minutes. In less than 90 additional minutes, enough boric acid can be injected to compensate for xenon decay, although xenon decay below the equilibrium operating level does not begin until approximately 25 hours after shutdown. Additional boric acid is employed if it is desired to bring the reactor to cold shutdown conditions.

Two flow paths are available for reactor coolant boration when the reactor is at power; i.e., the charging line and the reactor coolant pump seal injection.

If the normal charging line is not available, charging to the Reactor Coolant System can be continued via reactor coolant pump seal injection at the rate of approximately 5 gpm per pump. At this charging rate, approximately 20 gpm enters the Reactor Coolant System, enough boric acid solution can be added to counteract xenon decay following reactor shutdown.

As backup to the normal boric acid supply, the operator can align the refueling water storage tank outlet to the suction of the charging pumps.

Deleted Per 2008 Update.

An upper limit to the boric acid tank boron concentration, and a lower limit to the temperature for the tank and for flow paths from the tank are maintained in order to assure that solution solubility limit is met.

Deleted Per 2008 Update.

2. <u>Reactor Coolant Purification</u>

The Chemical and Volume Control System is capable of reducing the concentration of ionic isotopes in the purification stream as required in the design basis. This is accomplished by passing the letdown flow through the mixed bed demineralizers which remove ionic isotopes, except those of cesium, molybdenum and yttrium, with a minimum decontamination factor of 10. Through occasional use of the cation bed demineralizer, the concentration of cesium can be maintained below 1.0 μc/cc, assuming one percent of the rated core thermal power is being produced by fuel with defective cladding. The cation bed demineralizer is limited to a maximum 75 gpm flowrate. Each mixed bed demineralizer is capable of processing the letdown of the maximum purification flow rate (Table 9-31). If the normally operating mixed bed demineralizer's resin has become exhausted, the second demineralizer can be placed in service. Each demineralizer is designed, however, to operate for one core cycle with one percent of the rated core thermal power being produced by fuel with defective cladding.

3. <u>Seal Water Injection</u>

Flow through the reactor coolant pumps' seals is assured by the fact that there are three charging pumps, any one of which is capable of supplying the normal charging line flow plus the nominal seal water flow.

4. <u>Hydrostatic Testing of the Reactor Coolant System</u>

The reciprocating charging pump can pressurize the Reactor Coolant System to its maximum specified hydrostatic test pressure. The pump is capable of producing a hydrostatic test pressure greater than that required.

5. <u>Leakage Provisions</u>

Chemical and Volume Control System components, valves, and piping which see radioactive service are designed to limit leakage to the atmosphere. Leakage to the atmosphere is limited through:

a. welding of all piping joints and connections except where flanged connections are provided to facilitate maintenance and hydrostatic testing,

b. extensive use of leakoffs to collect leakage, and

c. use of diaphragm valves where conditions permit.

The volume control tank in the Chemical and Volume Control System provides an inferential measurement of leakage from the Chemical and Volume Control System as well as the Reactor Coolant System. Low level in the volume control tank actuates makeup at the prevailing reactor coolant boron concentration. The amount of leakage can be inferred from the amount of makeup added by the reactor makeup control system.

6. Ability to Meet the Safeguards Function

A failure analysis of the portion of the Chemical and Volume Control System which is safety related (used as part of the Emergency Core Cooling System) is included as part of the Emergency Core Cooling System failure analysis presented in Section 6.3.

9.3.4.4 Tests and Inspections

As part of station operation, periodic tests, surveillance inspections and instrument calibrations are made to monitor equipment condition and performance. Most components are in use regularly; therefore, assurance of the availability and performance of the systems and equipment is provided by Control Room and/or local indication.

Technical Specifications have been established concerning calibration, checking, and sampling of the Chemical and Volume Control System.

9.3.4.5 Instrumentation Application

Process control instrumentation is provided to acquire data concerning key parameters about the Chemical and volume Control System. The location of the instrumentation is shown on Figure 9-96 and Figure 9-98.

The instrumentation furnishes input signals for monitoring and/or alarming purposes. Indication and/or alarms are provided for the following parameters:

1. Temperature
2. Pressure
3. Flow
4. Water level
5. Deleted per 2003 update.

The instrumentation also supplies input signals for control purposes. Some specific control functions are:

1. letdown flow is diverted to the volume control tank upon high temperature indication upstream of the mixed bed demineralizers,
2. pressure downstream of the letdown heat exchangers is controlled to prevent flashing of the letdown liquid,

3. charging flow rate is controlled during charging pump operation,
4. water level is controlled in the volume control tank,
5. temperature of the boric acid solution in the batching tank is maintained, and
6. reactor makeup is controlled.

9.3.5 Boron Thermal Regeneration System

The Boron Thermal Regeneration System varies the Reactor Coolant System boron concentration to compensate for xenon transients and other reactivity changes which occur when the reactor power level is changed.

This system has been functionally disabled and will be decommissioned at a later date.

9.3.5.1 Design Basis

The Boron Therman Regeneration System has been functionally disabled and will be decommissioned at a later date. This information is provided as historical reference only.

The Boron Thermal Regeneration System is designed to accommodate the changes in boron concentration required by the design load cycle without requiring makeup for either boration or dilution.

9.3.5.2 System Description

The Boron Therman Regeneration System has been functionally disabled and will be decommissioned at a later date. This information is provided as historical reference only.

During normal operation of the Chemical and Volume Control System, the letdown flow from the Reactor Coolant System passes through the regenerative heat exchanger, the letdown heat exchanger, the first of two reactor coolant filters, a mixed bed demineralizer, the second reactor coolant filter and the volume control tank. The charging pumps then take suction from the volume control tank and return the purified reactor coolant to the Reactor Coolant System.

An alternate letdown path is provided which allows part or all of the letdown flow to pass through the Boron Thermal Regeneration System (shown in Figure 9-107 and Figure 9-108) when boron concentration changes are made to follow unit load. The letdown flow is directed to the Boron Thermal Regeneration System from a point downstream of the mixed bed demineralizers. After processing by the Boron Thermal Regeneration System, the flow is returned to the Chemical and Volume Control System at a point upstream of the second reactor coolant filter.

Storage and release of boron during load follow operation is determined by the temperature of the fluid entering the thermal regeneration demineralizers. A group of heat exchangers and chiller units are employed to provide the desired fluid temperatures at the demineralizer inlet for either storage or release operation of the system.

The flow path through the Boron Thermal Regeneration System is different for boron storage and release operations. During boron storage, the letdown stream enters the moderating heat exchanger and from there it passes through the letdown chiller heat exchanger. These two heat exchangers cool the letdown stream prior to its entering the demineralizers. The letdown reheat heat exchanger is valved out on the tube side and performs no function during boron storage operations. The temperature of the letdown stream at the point of entry to the demineralizers is controlled automatically by the temperature control valve which controls the shell side flow to the letdown chiller heat exchanger. After passing through the demineralizers, the letdown enters the moderating heat exchanger shell side, where it is heated by the incoming letdown stream before going to the volume control tank.

Therefore, for boron storage, a decrease in the boric acid concentration in the reactor coolant is accomplished by sending the letdown flow at relatively low temperatures to the thermal regeneration demineralizers. The resin, which was depleted of boron at high temperature during a prior boron release operation, is now capable of storing boric acid from the low temperature letdown stream.

Reactor coolant with a decreased concentration of boric acid leaves the demineralizers and is directed to the Chemical and Volume Control System.

During the boron release operation, the letdown stream enters the moderating heat exchanger tube side, bypasses the letdown chiller heat exchanger, and passes through the shell side of the letdown reheat heat exchanger. The moderating and letdown reheat heat exchangers heat the letdown stream prior to its entering the resin beds. The temperature of the letdown at the point of entry to the demineralizers is controlled automatically by the temperature control valve which controls the flow rate on the tube side of the letdown reheat heat exchanger. After passing through the demineralizers, the letdown stream enters the shell side of the moderating heat exchanger, passes through the tube side of the letdown chiller heat exchanger and then goes to the volume control tank. The temperature of the letdown stream entering the volume control tank is controlled automatically by adjusting the shell side flow rate on the letdown chiller heat exchanger.

Thus, for boron release, an increase in the boric acid concentration in the reactor coolant is accomplished by sending the letdown flow at relatively high temperatures to the thermal regeneration demineralizers. The water flowing through the demineralizers now releases boron which was stored by the resin at low temperature during a previous boron storage operation. The boron enriched reactor coolant is returned to the Reactor Coolant system via the Chemical and Volume Control System.

Although the Boron Thermal Regeneration System is primarily designed to compensate for xenon transients occurring during load follow, it can also be used to handle boron swings far in excess of the design capacity of the demineralizers. During startup dilution for example, the resin beds are first saturated, then washed off to the recycle holdup tanks in the Chemical and Volume Control System, then again saturated and washed off. This operation continues until the desired dilution in the Reactor Coolant System is obtained.

As an additional function, a thermal regeneration demineralizer can be used as a deborating demineralizer, to dilute the Reactor Coolant System down to very low boron concentrations towards the end of core life. To make such a bed effective, the effluent concentration from the bed is kept very low, close to zero ppm boron. This low effluent concentration is achieved by using fresh resin. Use of fresh resin is coupled with the normal replacement cycle of the resin; one resin bed is replaced during each core cycle.

1. Component Description

   Component safety classifications and design codes are given in Section 3.2 and a summary of principal component design parameters is given in Table 9-33.

   a. Chiller Pumps

      These centrifugal pumps circulate the water through the chilled water loop. One pump is supplied for each chiller.

   b. Moderating Heat Exchanger

The moderating heat exchanger operates as a regenerative heat exchanger between incoming and outgoing streams to and from the thermal regeneration demineralizers.

The incoming flow enters the tube side of the moderating heat exchanger. The shell side fluid, which comes directly from the demineralizers, enters at low temperature during boron storage and high temperature during boron release.

c. Letdown Chiller Heat Exchanger

During the boron storage operation, the process stream enters the tube side of the letdown chiller heat exchanger after leaving the moderating heat exchanger. The letdown chiller heat exchanger cools the process stream to allow the thermal regeneration demineralizers to remove boron from the coolant. The desired cooling capacity is adjusted by controlling the chilled water flow rate passed through the shell side of the heat exchanger.

The letdown chiller heat exchanger is also used during the boron release operation to cool the liquid leaving the thermal regeneration demineralizers to insure that its temperature does not exceed that of normal letdown to the volume control tank.

d. Letdown Reheat Heat Exchanger

The letdown reheat heat exchanger is used only during boron release operations and it is then used to heat the process stream. Water used for heating is diverted from the letdown line upstream of the letdown heat exchanger in the Chemical and Volume Control System, passed through the tube side of the letdown reheat heat exchanger and then returned to the letdown stream upstream of the letdown heat exchanger.

e. Chiller Surge Tank

The chiller surge tank handles the thermal expansion and contraction of the water in the chiller loop. The surge volume in the tank also acts as a thermal buffer for the chiller.

f. Thermal Regeneration Demineralizers

The function of the thermal regeneration demineralizers is to store the total amount of boron that must be removed from the Reactor Coolant System to accomplish the required dilution during a load cycle in order to compensate for xenon buildup resulting from a decreased power level. Furthermore, the demineralizers are able to release the previously stored boron to accomplish the required boration of the reactor coolant during the load cycle in order to compensate for a decrease in xenon concentration resulting from an increased power level.

The thermally reversible ion storage capacity of the resin applies only to borated ions. The capacity of the resin to store other ions is not thermally reversible. Thus, during boration, when borated ions are released by the resin, there is no corresponding release of the ionic fission and corrosion products stored on the resin.

The demineralizers are of the type that can accept flow in either direction. The flow direction during boron storage is therefore always opposite to that during release. This provides much faster response when the beds are switched from storage to release and vice versa, than would be the case if the demineralizers could accept flow in only one direction.

g. Chillers

One chiller is provided for each unit; also, one chiller is provided which can serve either unit if required. The chillers are located in a chilled water loop containing a surge tank, chiller pumps, the letdown chiller heat exchanger, piping, valves and controls.

The purpose of the chillers are twofold:

1) To cool down the process stream during storage of boron on the resin.

2) To maintain an outlet temperature from the Boron Thermal Regeneration System at or below 115°F during release of boron.

2. System Operation

A master switch is provided which places the system in either the boron release or the boron storage mode of operation.

When the switch is set for boron storage, it automatically:

a. aligns the proper flow path for the boron storage mode of operation,

b. shuts off the letdown reheat heat exchanger tube flow which puts this heat exchanger out of operation,

c. transfers control of the control valve at the letdown chiller heat exchanger shell side outlet to the thermocouple located between the letdown reheat heat exchanger and the demineralizers, and

d. starts a chiller and chiller pump.

When the switch is set for boron release, it automatically:

a. aligns the proper flow path for the boron release mode of operation,

b. energizes the control of the tube side flow rate to the letdown reheat heat exchanger by a signal from the thermocouple between this heat exchanger and the demineralizers,

c. transfers control of the control valve at the letdown chiller heat exchanger shell side outlet to the thermocouple located in the line leaking to the volume control tank, and

d. starts a chiller and chiller pump.

After the mode of operation has been selected and the system prepared for operation by actuation of the master switch, flow is admitted to the Boron Thermal Regeneration System by throttling back on the diversion valve in the letdown line. The flow rate through the Boron Thermal Regeneration System is dictated by the desired reactor coolant dilution (boration) rate.

When the boron concentration of the reactor coolant reaches the desired level, the Boron Thermal Regeneration System is shut down by placing the master switch in the off position.

9.3.5.3 Safety Evaluation

The Boron Therman Regeneration System has been functionally disabled and will be decommissioned at a later date. This information is provided as historical reference only.

Any partial or total malfunction of the Boron Thermal Regeneration System results only in loss of unit load-following capability. This system is non-safety related. The postulated full power dilution accident considered in Chapter 15 is not influenced by dilution with this system. The dilution flow depends solely upon the delivery capability of the charging pumps which remains unchanged with or without Boron Thermal Regeneration System operability.

9.3.5.4 Tests and Inspections

The Boron Thermal Regeneration System has been functionally disabled and will be decommissioned at a later date. This information is provided as historical reference only.

The Boron Thermal Regeneration System is in intermittent use throughout normal reactor operation. The system design is verified by pre-operational testing. Periodic visual inspection and preventive maintenance are conducted using normal industry practice.

9.3.5.5 Instrumentation Application

The Boron Thermal Regeneration System has been functionally disabled and will be decommissioned at a later date. This information is provided as historical reference only.

1. Temperature

   Instrumentation is provided to monitor the chiller outlet temperature and to control chiller operation. Instrumentation is also provided to monitor the chiller surge tank temperature. Readout for both sets of instrumentation is located on the main control board.

   Instrumentation is provided to control the temperature of the letdown flow passing through the demineralizers. During dilution, it controls a valve which throttles the letdown chiller heat exchanger shell side flow. During boration, it controls the valve which throttles the letdown reheat heat exchanger tube side flow. Readout and a high temperature alarm are provided on the main control board.

   The original design provided protection of the thermal regeneration demineralizer resins from letdown process high temperature. A process high temperature interlock was provided to automatically divert letdown process from the demineralizers to the Chemical Volume Control Tank. This interlock feature and associated field instrumentation were removed due to NR system abandonment.

   Instrumentation is provided which monitors the temperature of the flow leaving the demineralizers. Temperature indication is provided on the main control board.

   The temperature of the flow leaving the Boron Thermal Regeneration System during boration (boron release) operations is controlled by instrumentation which controls a valve which throttles the letdown chiller heat exchanger shell side flow. Thus the temperature of the fluid being routed to the volume control tank is prevented from becoming too high.

2. Pressure

   Instrumentation is provided which monitors and gives local indication of the pressure at each chiller pump suction and discharge and at the inlet and outlet to the bank of demineralizers.

3. Flow

   Instrumentation on the return line to the chiller surge tank maintains chiller loop flow at a constant value by controlling the valve which adjusts the amount of flow bypassing the letdown chiller heat exchanger. Thus, if the shell side flow in the heat exchanger is restricted by the temperature controlled valve, the bypass valve is automatically adjusted to maintain full flow in the chiller loop.

   Instrumentation is provided to monitor the flow rate through the Boron Thermal Regeneration System. Indication is on the main control board.

4. Level

Instrumentation is provided to measure the fluid level in the chiller surge tank. Level readout and high and low level alarm are provided on the main control board.

9.3.6 Boron Recycle System

The Boron Recycle System receives and recycles reactor coolant effluent for reuse of the boric acid and makeup water. The system decontaminates the effluent by means of demineralization and gas stripping, and uses evaporation to separate and recover the boric acid and makeup water. The Boron Recycle System is shared by both units.

9.3.6.1 Design Bases

1. Collection Requirements

   The Boron Recycle System is designed to collect the excess reactor coolant that results from the following station operations during one core cycle for each of the two units:

   a. Two dilutions for core burnup from approximately 2000 ppm boron at the beginning of a core cycle to approximately 100 ppm near the end of a core cycle.

   b. Hot shutdowns and startups. Four hot shutdowns are assumed to take place during a core cycle.

   c. Cold shutdowns and startups. Three cold shutdowns are assumed to take place during a core cycle.

   d. Refueling shutdown and startup.

   e. Reactor coolant pump seal leak off.

   The Boron Recycle System also collects water from the following sources:

   a. Volume control tank pressure relief.

   b. Boric acid blender (Chemical and Volume Control System) - provides storage if boric acid tank must be emptied for maintenance. The boric acid solution is stored in a recycle holdup tank after first being diluted with reactor makeup water by the blender. The boric acid concentration is reduced to insure against precipitation of the boric acid in the unheated recycle holdup tank.

   c. Liquid Waste Recycle System - provides capability for using the recycle evaporator as a waste evaporator and vice versa.

   d. Fuel pool cooling pumps (Spent Fuel Cooling System) - provides a means of storing the fuel transfer canal water to allow for maintenance on the transfer equipment.

   e. Valve leakoffs and equipment drains.

   f. Gas decay tank drain pump (Waste Gas System) - provides a collection point for condensate from the gas system and for reactor makeup water used for gas decay tank maintenance.

2. Capacity Requirement

   The Boron Recycle System is designed to process the total volume of water collected during a core cycle as well as short term surges. The design surge is that produced by a cold shutdown and subsequent startup for both units during the latter part of a core cycle.

The Boron Recycle piping which interconnects the units is used in beyond design basis events to transfer Refueling Water between the units. These are committed response strategies in response to NRC Order EA-02-026 (B.5.b order).

The Boron Recycle piping which interconnects the units is used in beyond design basis events to transfer Refueling Water between the units. These are committed response strategies in response to NRC Order EA-02-026 (B.5.b order). Further information on the components which may be utilized in responding to a B.5.b related severe accident is maintained in design basis specification. MCS-1465.00-00-0025, "Design Basis for the Extensive Damage Mitigation". Guidance for administration of the B.5.b accident response program is located in NSD-226, "Extensive Damage Mitigation Program."

3. Purification Requirement

The water collected by the Boron Recycle System contains dissolved gases, boric acid, and suspended solids. Based on reactor operations with one percent of the rated core thermal power being generated by fuel elements with defective cladding, the Boron Recycle System, in conjunction with Chemical and Volume Control System purification, is designed to provide sufficient cleanup of the water to satisfy the chemistry requirements of the recycled reactor makeup water and 4 weight percent boric acid solution.

The maximum radioactivity concentration buildup in the Boron Recycle System components is based on operation of the reactor at its engineered safeguards design rating with defective fuel rods generating one percent of the rated core thermal power. For each component, the shielding design considers the maximum buildup on an isotopic basis including only those isotopes which are present in significant amounts. Filtration, demineralization, and gas stripping are the means by which the activity concentrations are controlled.

9.3.6.2 System Description

The Boron Recycle System is shown on Figure 9-110. When water is directed to the Boron Recycle System, the flow passes first through the recycle evaporator feed demineralizers and filters and then into the recycle holdup tanks. The recycle evaporator feed pumps can be used to transfer liquid from one recycle holdup tank to the other if desired. When sufficient water is accumulated to warrant evaporator operation, the recycle evaporator feed pumps take suction from the selected recycle holdup tank. The fluid then flows through the recycle evaporator package. Here, hydrogen, nitrogen, and residual fission gases are removed in the stripping column before the liquid enters the evaporator shell. These gases are directed to the Waste Gas System.

During evaporator operation, distillate from the evaporator flows continuously to the reactor makeup water storage tanks. Also located in this flow path are the recycle evaporator condensate demineralizer and the recycle evaporator condensate filter. A radiation monitor continuously checks the evaporator distillate and on detection of high activity, a three-way diversion valve is tripped in order to return the distillate to the recycle holdup tanks.

The evaporator concentrates the boric acid solution until a 4 weight percent solution is obtained. The accumulated batch is normally transferred directly to the boric acid tanks in the Chemical and Volume Control System through the recycle evaporator concentrates filter. If, for some reason, this batch cannot be discharged to the boric acid tanks, it can be diverted back to the recycle holdup tanks, the Liquid Waste Recycle System, or the Nuclear Solid Waste Disposal System.

Connections are provided so that, if necessary, the recycle evaporator can be used as a waste evaporator (and vice versa).

Thermal insulation is provided on system valves, piping, and equipment for personnel protection and to prevent heat losses. It is designed to limit insulation surface temperatures to 60°F above ambient temperature. Where necessary, antisweat insulation is provided. Materials are compatible with use on stainless steel (low chloride).

Electrical heat tracing is required on the recycle evaporator even though redundant electrical heating units are provided in the room in which it is located. Electrical heat tracing is provided on the concentrates piping which leaves the room to insure that the boric acid does not precipitate.

Heat tracing is also provided on the outdoor piping associated with the reactor makeup water storage tanks to prevent freezing of the water in these lines under cold, static conditions.

1. Component Descriptions

   A summary of principal component data is given in Table 9-34 and the code requirements are given in Section 3.2.

   a. Recycle Evaporator Feed Pumps

      Two centrifugal, canned pumps supply feed to the recycle evaporator package from the recycle holdup tanks. The pumps can be used to transfer liquid from one holdup tank to the other, to the spent fuel pool of either unit, or to the charging pumps of either unit for transfer into the respective unit's Reactor Coolant System. The pumps can be used to recirculate water from the recycle holdup tanks through the recycle evaporator feed demineralizers for additional cleanup if desired.

   b. Recycle Holdup Tanks

      Two recycle holdup tanks provide storage of reactor effluents for processing by the evaporator package.

      Each tank has a diaphragm which prevents air from dissolving in the tank liquid and prevents the hydrogen and fission gases under the diaphragm from mixing with the air. The air space in the tank above the diaphragm is swept with air to the station vent.

   c. Recycle Evaporator Reagent Tank

      This tank provides a means of adding chemicals to the evaporator; e.g., for cleanup.

   d. Recycle Evaporator Feed Demineralizers

      Two flushable, mixed bed demineralizers remove fission products from the fluid directed to the recycle holdup tanks. The demineralizers also provide a means of cleaning the recycle holdup tank contents via recirculation.

   e. Recycle Evaporator Condensate Demineralizer

      A flushable, anion demineralizer is provided as a polishing demineralizer for distillate from the recycle evaporator. Although the bed may become saturated with boron at the normally low concentration (<10 ppm) leaving the evaporator, it still removes boron if the concentration increases because of an evaporator upset. The demineralizer also provides a means for cleanup of the reactor makeup water storage tank contents.

   f. Recycle Evaporator Feed Filters

These two filters collect resin fines and particles from the fluid prior to its entry to the recycle holdup tanks.

g. Recycle Evaporator Condensate Filter

This filter collects particulate and resin fines from the boric acid evaporator condensate stream.

h. Recycle Evaporator Concentrates Filter

This filter removes particulates from the evaporator concentrate as it leaves the evaporator.

i. Recycle Evaporator Package

The recycle evaporator package processes dilute boric acid and produces distillate and approximately 4 weight percent boric acid stripped of hydrogen, nitrogen, and radioactive gases.

A boric acid solution is fed from the recycle holdup tanks to the evaporator by the recycle evaporator feed pumps. The feed first passes through a heat exchanger where condensing steam raises its temperature. The feed then passes into the top of the stripping column. Gases are stripped off as the feed passes over the packing in the tower in contra flow to stripping steam from the evaporator. After stripping, the feed is introduced into the evaporator as makeup. The vapors leaving the boiling pool are stripped of entrained liquid and volatile boron carryover. Pure vapors are then condensed in the condenser section and pumped from the system. When the desired concentration is reached in the boiling pool, the concentrates are pumped from the system.

Radioactive gases and other non-condensables are discharged from the system into the waste gas vent header.

The recycle and waste evaporators are similar units and are interconnected so that they can serve as standbys for each other under abnormal conditions.

j. Reactor Makeup Water Storage Tanks

Two reactor makeup water storage tanks (one per unit) supply reactor grade makeup water for the station. Each tank is sized to supply the water requirements for one unit during a cold shutdown followed by a startup from cold conditions assuming these events occur late in cycle life when Reactor Coolant System boron concentration is below 200 ppm. A minimum quantity of water is available at all times to effect a cold shutdown. If, after a cold shutdown and startup have been accomplished, it becomes necessary to perform an additional cold shutdown, the quantity of water required can be drawn again from the reactor makeup water storage tank which was replenished by the recycle evaporator during the time of the first cold shutdown. In addition, water may be transferred between the Unit 1 and Unit 2 reactor make up water storage tanks by utilizing connections at the inlet to the recycle evaporator condensate demineralizer and at the discharge of the recycle evaporator condensate filter.

In addition to the makeup requirements outlined above, the reactor makeup water storage tanks provide flush water to various radioactive equipment and piping throughout the station.

Each tank is fitted with a flexible diaphragm in order to maintain the specified reactor makeup water requirements for oxygen (see Table 9-35 for makeup water specifications). Construction is of lined carbon steel.

k. Recycle Holdup Tank Vent Ejector

The ejector is designed to pull gases from under the diaphragm in the recycle holdup tank. The waste gas compressor provides the motive force.

l. Recycle Evaporator Concentrate Pump

This is a double mechanical seal type pump capable of delivering 35 gpm at 125 feet of head with very low available NPSH, <2.0 feet. This pump recirculates the evaporator concentrates during concentration and pumps concentrates out when concentration is complete. Pump seal cooling water is supplied by a small seal cooling water loop.

m. Mechanical Seal Cooling Water Pump

This small 60 psig at 2 gpm pump is used to supply cooling water to the recycle evaporator concentrates pump.

n. Mechanical Seal Cooling Water Heat Exchanger

This heat exchanger cools seal water to the recycle evaporator concentrates pump. It cools 2 gpm from 140°F to less than 110°F. This heat exchanger is cooled by the shell side of the Component Cooling System.

o. Mechanical Seal Cooling Water Filter

This filter cleans the concentrates pump seal cooling water prior to the seals. A 5 micron, 98% retention disposable cartridge is used.

p. Valves

The material of construction for all valves exposed to boric acid solution is stainless steel or other corrosion resistant material.

q. Piping

All Boron Recycle System piping which handles radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanged connections are required to facilitate equipment removal for maintenance and testing.

2. System Operation

The Boron Recycle System is manually operated with the exception of a few automatic protective functions. These automatic functions protect the recycle evaporator feed demineralizers from a high inlet temperature and a high differential pressure, prevent a high vacuum from being drawn on the recycle holdup tank, protect the recycle evaporator feed pumps from low NPSH, and prevent high activity recycle evaporator condensate from being sent to the recycle monitor tank. The Boron Recycle System has sufficient instrumentation readouts and alarms to provide the operator with information to assure proper system operation.

3. Evaporation

Water is accumulated in the recycle holdup tank until sufficient quantity exists to warrant an evaporator startup. Prior to startup of the evaporator, the contents of the recycle holdup tank are analyzed and, if necessary, the contents recirculated through the recycle evaporator feed demineralizers and filters. The flow can be discharged back to the recycle holdup tank or to the evaporator. The evaporator is then operated to produce a batch of 4 weight percent boric acid.

During the normal operation of the evaporator, condensate is sent to the reactor makeup water storage tanks via the recycle evaporator condensate demineralizer. The condensate is monitored for high activity and, on a high radiation alarm, the flow is automatically diverted to the recycle holdup tank for reprocessing.

After a batch of boric acid is concentrated to 4 weight percent, it is analyzed to assure that it is within specifications for reuse. If it meets the specifications, it is pumped to the boric acid tanks. If it does not, it can be returned to the recycle holdup tank via the recycle evaporator feed demineralizers for re-evaporation or, if desired, the concentrated boric acid can be sent to the Liquid Waste Recycle System, or the Nuclear Solid Waste Disposal System.

4. Recycle Holdup Tank Venting

   Because hydrogen is dissolved in the reactor coolant at approximately one atmosphere over-pressure, a portion of the hydrogen along with fission gases comes out of solution in the recycle holdup tank under the diaphragm. The hydrogen and fission gases are vented to the Waste Gas System as required.

5. Maintenance Drains

   When large amounts of water must be drained from the Reactor Coolant System or the spent fuel pool (or fuel transfer canal) to the Boron Recycle System, a recycle holdup tank is drained of water and vented to the Waste Gas System. Large amounts of water can then be stored in this holdup tank until maintenance is completed and, after checking the chemistry, returned. After returning the water, the recycle holdup tank is again vented to the Waste Gas System where it may be directed to a shutdown gas decay tank to prevent accumulation of air or nitrogen in the high activity gas decay tanks.

6. Reactor Makeup Water Cleanup

   If the reactor makeup water requires purification, it can be recirculated through the recycle evaporator condensate demineralizer until its chemistry is within specifications. If further processing is necessary, water from the reactor makeup water storage tank is directed through the recycle evaporator condensate demineralizer and into the recycle holdup tank for re-evaporation.

7. Waste Processing With the Recycle Evaporator

   The recycle evaporator can be used to perform the function of the waste evaporator. Heat tracing is provided for the recycle evaporator, and the boric acid can be concentrated to 12 weight percent.

   After using the recycle evaporator to process water from the Liquid Waste Recycle System, it is thoroughly rinsed out. During initial recycle processing, the condensate is directed to the recycle monitor tanks for analysis prior to transfer to the reactor makeup water storage tank. Depending upon the purity of the evaporator bottoms, the concentrated boric acid is transferred to the boric acid tanks or it is drummed.

9.3.6.3 Safety Evaluation

Malfunctions in the Boron Recycle System do not affect the safety of station operations. The Boron Recycle System is designed to tolerate equipment faults with critical functions being met by the use of two pieces of equipment so that the failure of one does, at most, reduce the capacity of the Boron Recycle System but not completely shut it down. Because of the large surge capacity of the Boron Recycle System, the non-availability of the recycle evaporator can be tolerated for periods of time. Also, backup is provided by the waste evaporator.

All piping and components handling radioactive fluids are designed, fabricated, and inspected according to applicable code requirements. The use of diaphragm valves and welded piping joints minimizes leaks.

9.3.6.4 Tests and Inspections

The Boron Recycle System is functionally tested to verify the ability of the system to receive and recycle excess reactor coolant. A functional verification of the operability of system pumps, filters, demineralizers, heat tracing, and the evaporator package is performed. The Boron Recycle System is in intermittent use throughout normal reactor operation. Periodic visual inspection and preventive maintenance are conducted using normal industry practice.

9.3.6.5 Instrumentation Application

The instrumentation available for the Boron Recycle System is discussed below. Alarms are provided as noted. There is also a common alarm on the Main Control Board which indicates any alarms on the Boron Recycle System panel.

1. Temperature

   Instrumentation is provided to measure the temperature of the inlet flow to the recycle evaporator feed demineralizers and to control bypass valves. If the inlet temperature becomes too high, the instrumentation aligns the valves to bypass the demineralizers. Local temperature indication and a high temperature alarm on the Boron Recycle System panel are provided by this instrumentation.

2. Pressure

   Instrumentation is provided to measure the pressure differential across the recycle evaporator feed demineralizers and to control the same valves as discussed previously (but independently of the temperature control). If the pressure drop through the demineralizers is too high, this instrumentation aligns the valves to divert flow directly to the recycle evaporator feed filters. Local pressure indication and a high alarm on the Boron Recycle System panel are provided by this instrumentation.

   Instrumentation is provided to measure the pressure differential across each recycle evaporator feed filter, the recycle evaporator concentrates filter, and the recycle evaporator condensate filter. Local indication of the differential pressure between the inlet and outlet line is provided.

   Instrumentation is provided to measure and give local indication of the discharge pressures of each recycle evaporator feed pump.

   Instrumentation is provided to measure the pressure in the recycle holdup tank vent line and to control the shutoff valve in the vent line. This instrumentation is used during holdup tank venting operations. When the pressure in this line becomes too low, the valve is closed to protect the holdup tank diaphragm from an excessive differential pressure across it. Local pressure indication and a low pressure alarm on the Boron Recycle System panel are provided.

3. Flow

   Instrumentation is provided which gives local indication of the recycle holdup tank vent purge flow and of the feed flow to the recycle evaporator package.

4. Level

Instrumentation is provided to give an indication of the water level of each recycle holdup tank. Both high level and low level alarms are provided by this instrumentation at the Boron Recycle System panel. If, after reaching the low level alarm setpoint, the recycle evaporator feed pumps are not stopped, the holdup tank level continues to decrease until a second low level point is reached and a level actuated control circuit stops the pumps.

5. Radiation

   Instrumentation is provided to give an indication in the main Control Room of the radioactivity level in the recycle evaporator condensate. Upon a high level signal, this system causes a three-way valve to divert flow back to the recycle evaporator feed demineralizers. This instrumentation also has a high radiation level alarm on both the Boron Recycle System panel and in the Control Room.

9.3.7 Post Accident Sampling System

Deleted from Licensing Basis by Tech Spec Amendment 199/180 dated 9/17/01

9.3.8 Failed Fuel Detection System

During on-line operation, periodic radioisotopic sampling is performed using the Nuclear Sampling System as described in Section 9.3.2 to identify failed fuel. Additionally, radiation monitoring via the Process Radiation Monitoring System described in Section 11.4 is monitored to detect failed fuel.

9.3.9 References

1. Nuclear Regulatory Commission, Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Reactors, from Frank J. Miraglia, Jr., August 8, 1988, "Instrument Air Supply System Problems Affecting Safety-Related Equipment."

2. Duke Power Company, Letter from H.B. Tucker to NRC, February 10, 1989, re:"Instrument Air Supply System Problems Affecting Safety-Related Equipment (NRC Generic Letter 88-14)."

3. Duke Power Company, Letter from T.C. McMeekin to NRC, July 9, 1986, re: Generic Letter 88-14, "Instrument Air Supply System Problems Affecting Safety-Related Equipment."

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9.4 Air Conditioning, Heating, Cooling and Ventilation Systems

9.4.1 Control Room

The Control Area Ventilation System is included in Section 6.4,

9.4.2 Auxiliary Building

9.4.2.1 Design Bases

The design bases for the Auxiliary Building Ventilation System are to:

1. Provide a suitable environment for the operation of equipment and personnel access as required for inspection, testing and maintenance.
2. Maintain the building at a slightly negative pressure to minimize outleakage,
3. Provide purging of the building to the unit vent. The air exhausted to the environment from potentially contaminated areas is monitored and filtered, as required, so that the limits of 10CFR 20 and the Technical Specifications are not exceeded, and
4. Provide a suitable environment for the operation of vital equipment during an accident.

The Auxiliary Building Filtered Exhaust System (ABFVES) was not initially designed as a safety related system and was not credited to mitigate a design basis accident. However, during initial plant licensing the system was re-classified as an engineered safety feature (ESF) atmosphere cleanup system and was included in Technical Specifications. By letter dated August 28, 1975 from D. B. Vassallo to W. O. Parker, the NRC imposed the requirement that the system needed to mitigate the dose consequences of a postulated ECCS pump seal failure during a LOCA and meet the recommendation of Regulatory Guide 1.52. To meet this new requirement, Duke and the NRC agreed upon a minimum level of upgrades to the system, such as, upgrade the filter units to safety grade, provide 1E power to the filtered exhaust fan motors and controls, and place the fan motors on the emergency diesel sequencers. In McGuire's SER dated March 1, 1978, the NRC credited the Auxiliary Building Filtered Exhaust System with mitigating the dose consequences of a ECCS pump seal failure. By letter dated April 6, 1979 from W. O. Parker to H. R. Denton, Duke advised the NRC regarding McGuire's conformance to Regulatory Guide 1.52 including exceptions. The existing system meets most of the recommendations of Regulatory Guide 1.52 and the exceptions were documented in UFSAR Table 9-38.

The ABFVE systems are designed to be shared between units. Each unit's system is constructed with two 50% capacity fans providing flow to a 100% capacity filter package. General Design Criteria (GDC) 5 applies to this system. This design meets the requirements of GDC 5 for shared systems.

The Annulus Ventilation System (Section 6.2.3), the Control Area Ventilation System (Section 6.4) and the Diesel Building Ventilation System (Section 9.4.6) serve the engineering safety equipment areas.

9.4.2.2 System Description

The Auxiliary Building Ventilation System (Figure 9-117, Figure 9-118 and Figure 9-119) is composed of the following subsystems:

1. Fuel Handling Ventilation Supply

Makeup air from outdoors for the fuel handling area is provided for each station unit by a separate supply subsystem (Figure 9-119) consisting of one 100 percent capacity fan with heating and cooling coils and a medium efficiency (50 percent) air filter. This subsystem does not have a standby capacity. The air is distributed throughout the fuel handling area by the suction of the fuel handling ventilation exhaust subsystem.

2. Fuel Handling Ventilation Exhaust

The fuel handling ventilation exhaust subsystem consists of two 50 percent capacity fans, duct work, bypass and filters (prefilters, absolute and carbon filters). Exhaust air is directed either through the prefilter, absolute filter, and carbon filter, or through the filter train bypass, and then to the unit vent. See comment C-2-i, Table 9-37 for complete bypass description. This operation continuously purges the fuel pool area of any heat, humidity, gaseous and/or particulate matter. The maximum and minimum design temperatures expected in the fuel handling area are 90°F and 50°F, respectively.

3. Auxiliary Building General Ventilation Supply

Makeup air from the atmosphere for the general ventilation supply subsystem in the Auxiliary Building is provided for each station unit by a separate system consisting of two supply fans with heating and cooling coils and a medium efficiency (50 percent) air filter. These subsystems have neither standby capacity nor carbon filters. Normally, both fans operate at their design speed with the air distributed throughout the building by the suction of the various exhaust ventilation systems.

4. Contaminated Material Handling and Waste Handling Areas

Five supply units with prefilters, heating and cooling coils, and duct work provide conditioned air to the various compartments within the Contaminated Material Handling and Waste Handling areas which deal with the processing and handling of radioactive liquids, solids and gases. In addition, these supply units serve the Chemistry and RP office areas. A single exhaust unit consisting of dual fans, a vent stack, a filter package and duct work exhausts these areas. Radiation monitor EMF-53 samples air exhausting from the vent stack and isolates all supply and exhaust units upon receipt of a high radiation signal. Ventilation is based on limiting temperatures of 95°F maximum and 60°F minimum. A supply unit with filter, electric heat, chilled water coil and fan maintains the Hot Machine Shop to 75°F.

5. Auxiliary Building Ventilation Exhaust

Each station unit is served by two independent exhaust subsystems, Auxiliary Building filtered ventilation exhaust and Auxiliary Building general ventilation exhaust subsystems. Each system consists of two 50 percent exhaust fans. Auxiliary Building filtered ventilation exhaust system incorporates prefilters, absolute filters, carbon filters, bypass and associated duct work extending to areas subject to contamination. These contaminated areas will be maintained under a negative pressure.

The Auxiliary Building general ventilation exhaust subsystem serves areas that are not subject to contamination. These areas have ventilation rates based upon heat loads only.

All the exhaust systems in the Auxiliary Building are of greater capacity than the supply systems, thus maintaining the Auxiliary Building at a negative pressure. During operation of either unit, all associated Auxiliary Building ventilation systems are activated to “normal” operation. During shutdown of either unit, associated Auxiliary Building ventilation systems may operate in part or in total to suit maintenance, inspection, testing or refueling conditions.

Continuous monitoring is provided at appropriate areas throughout the Auxiliary Building to assure safe conditions of temperature and radioactivity and to alert operating personnel of any abnormality (Sections 12.1.4 and 12.2.4).

9.4.2.3 Safety Evaluation

The Auxiliary Building Ventilation System provides adequate capacity to assure that proper temperatures are maintained in the various portions of the building under operating and shutdown conditions in all types of weather. Originally, both fuel handling ventilation exhaust fans were 100% capacity each. The exhaust fans are now 50% capacity each, both fans must be running to assure proper operation of the System.

In the event that saturated air should pass through the fuel handling ventilation exhaust subsystem the activated charcoal would lose very little of its ability to adsorb iodine. However, as stated in Section 9.1.3, there is redundancy within the Spent Fuel Cooling System, and therefore total loss of fuel pool cooling is not a credible event.

Should a fuel handling accident occur, the fuel handling ventilation exhaust subsystem would be available to reduce dose consequences, since fuel handling is only permitted when this system is in operation. Should the exhaust ventilation subsystem become inoperable, fuel handling would be terminated. Refer to Section 15.7.4 for a discussion of the dose consequences of a fuel handling accident.

Filter train design comparison, for the fuel handling exhaust subsystem, to Regulatory Guide 1.52 is presented in Table 9-37.

Under normal operating conditions, the supply and exhaust fans are loaded so that all areas of the building are at a negative pressure with respect to atmosphere, thus minimizing outleakage. All vents from the building are directed to the vent of the respective unit and monitored before release to the atmosphere. All supply is prefiltered. All systems are located within the Auxiliary Building and arranged for ease of access, control and monitoring.

The filtered exhaust portion of the Auxiliary Building Ventilation System provides filtration for the ECCS pump rooms under post-LOCA conditions. The filtered exhaust fans receive an automatic start signal to assure the availability of these fans under post-LOCA conditions. A comparison of the Auxiliary Building filtered ventilation exhaust subsystem design to Regulatory Guide 1.52 is presented in Table 9-38.

9.4.2.4 Tests and Inspections

Tests and inspections were performed to assure and demonstrate the capability of components and the system to perform the assigned function.

Manufacturer Shop Testing

The manufacturer was required to verify by appropriate tests the following:

1. Carbon Filter:

   Carbon filter capabilities for removal of molecular iodine - 131 and methyl iodide.

   Carbon filter iodine collection capability

   Carbon filter cell leak - tightness integrity

   Carbon filter flow resistance

2. Fan:

Proper head and flow characteristics

3. Heating and Cooling Coils:

   Proper heating and cooling capabilities

System Testing and Inspection

Operational testing was performed prior to initial startup to demonstrate proper functioning of the system. Testing includes the following:

1. Leak-tightness tests of components and system.
2. Functional verification of system controls and interlocks.
3. Demonstration that the appropriate negative pressures can be established and maintained.
4. In place testing of filters.

Active components of the Auxiliary Building Ventilation System are in either continuous or frequent use during normal station operation and require no additional periodic tests. Periodic visual inspections and preventive maintenance are conducted according to good industrial practice.

The Auxiliary Building Ventilation System is in continuous operation and is accessible for periodic inspection. Essential system components are tested periodically to demonstrate system readiness and operability as required by the Technical Specifications.

9.4.2.5 Instrumentation Application

A description of the instrumentation provided to alert fuel handling personnel that the fuel handling ventilation exhaust subsystem has failed is presented in Section 7.7.1.14.

9.4.3 Radwaste Area

Note: The Radwaste Area Ventilation System (Section 9.4.3) in the NRC Standard Format (Reg. Guide 1.70) is included in Section 9.4.2.

9.4.4 Turbine Building

9.4.4.1 Design Bases

The Turbine Building Ventilation System is designed to provide a suitable environment for the operation of equipment and personnel access as required for inspection, testing and maintenance.

Ambient temperature limits within the Turbine Building are maximum inside summer design temperature of 110°F and minimum inside winter design temperature of 55°F.

Treatment and monitoring of exhaust air is not provided since the Turbine Building environment is designed not to have any areas of measurable radioactivity. Secondary side radioactivity level is monitored in the air ejector off-gas vent and at the steam generator sampling connections. For a description of these monitoring locations, refer to Sections 10.4.2 and 9.3.2, respectively. See Tables 11-27 and 11-28 for radiation monitors, (EMF-34 and EMF-33) design parameters.

Instrumentation for the Turbine Building Ventilation System is provided to control temperature in conditioned areas.

9.4.4.2 System Description

Each Turbine Building Ventilation System (Figure 9-120) is composed of nine (9) propeller type supply fans and associated wall mounted louvers, ten (10) roof mounted propeller type exhaust fans (A thru J), and three (3) platform mounted exhauster fans (K thru M). Each platform mounted exhauster fan is connected to an air shaft which extends to each Turbine building level. The three (3) platform exhauster fans are each interlocked to operate in conjunction with three (3) of the nine (9) supply fans. The nine (9) supply fans (N thru V) are furnished with wall mounted operable louvers which automatically open whenever the associated fan starts. Manually operated air intake louvers are located in the East wall of the Unit 2 Turbine Building. The exhaust fans discharge directly to the environment with no recirculation.

The Turbine Building Ventilation System also consists of sub-systems which provide air conditioning to the 250 V DC auxilliary battery rooms, the EMC room, transducer rooms, switchgear rooms, security central alarm station, and voltage regulator enclosures. Ventilation is provided to the diesel lube oil holding tank rooms, the "C" heater drain pump pits, and main turbine oil tank rooms.

The isolated phase bus cooling and standby shutdown facility (SSF) HVAC are part of this system. The SSF control room and battery room are air-conditioned while the SSF electrical equipment room and SSF diesel room are provided with ventilation fans and electric heaters. The isolated phase bus cooling is a recirculation system with water coils. KR, Recirculated Cooling, is the coil cooling supply.

9.4.4.3 Safety Evaluation

Exhaust fans A thru J operate independently of each other and all other Turbine building fans. The number of fans required varies with conditions. Operation of these fans will affect building temperature and pressure.

Supply fans N thru P operate in conjunction with exhaust fan K; as do supply fans Q thru S with exhaust fan L and supply fans T thru V with exhaust fan M. The number of fans in operation varies to maintain building temperature and pressure. The system is not safety related.

9.4.4.4 Tests and Inspections

Periodic inspections are performed to assure the capability of components to perform their assigned functions.

Manufacturer Shop Test

The manufacturer verified by appropriate tests the fan head and flow characteristics.

System Testing and Inspection

System functional tests were performed prior to initial startup to demonstrate functioning of the system. Thereafter, normal operating system performance monitoring detects any deterioration in the functional capability of the system.

9.4.4.5 Instrumentation Application

Sufficient instrumentation is included in the system to assure satisfactory operation.

9.4.5 Containment

9.4.5.1 Design Bases

The Containment Purge and Ventilation System is designed to maintain temperature in the various portions of the Containment within acceptable limits for operation of equipment and for personnel access for inspection, maintenance and testing as required. It also has capability for purging the Containment atmosphere to the environment via the unit vent. The system is not an Engineered Safety Feature.

Provisions are made to:

1. purge the Containment atmosphere to the unit vent. System capacity is sufficient to provide 1.5 changes of the Containment air volume in one hour,
2. maintain the Containment upper compartment average air temperature between 75° and 100°F during Modes 1, 2, 3 & 4. The lower limit may be reduced to 60°F during Modes 2, 3 & 4,
3. maintain the Containment lower compartment average air temperature between 100° and 120°F during Modes 1, 2, 3 & 4. The lower limit may be reduced to 60°F during Modes 2, 3 & 4. The Containment lower compartment temperature may be between 120° and 125° for up to 90 cumulative days per calendar year provided the lower compartment temperature average over the previous 365 days is less than 120°F,
4. purge the in-core instrumentation room atmosphere to the unit vent during periods of personnel access to this room,
5. assure that a reliable supply of cooling air is provided to the control rod drives,
6. reduce the concentration of airborne fission products which may be introduced into the Containment atmosphere via leakage from the Reactor Coolant System, and
7. provide cooling for the reactor vessel support structure and the concrete to prevent the concrete core temperature from exceeding 150°F.

9.4.5.2 System Description

The Containment Purge and Ventilation System depicted in Figure 9-121 consists of the independent subsystems described below.

1. Containment Purge Supply and Exhaust

   Purge air is supplied to the Containment through two 50 percent capacity fans and their associated filters and heating coils. Purged air is exhausted through two 50 percent capacity fan and filter networks (see Figure 9-121) to the unit vent where it is monitored during release to the atmosphere. The purge air supply and exhaust fans and filters are located in the Auxiliary Building.

   There are five purge air supply penetrations and four purge air exhaust penetrations in the Containment. These penetrations are in the upper compartment, lower compartment and incore instrumentation area. Two normally closed isolation valves in each penetration provide Containment isolation.

   The system has the capacity to assure approximately 1.5 complete changes of air per hour. Venting capacity is controlled by dampers.

The upper compartment purge exhaust ductwork is so arranged to draw exhaust air into a plenum around the periphery of the refueling canal, effecting a ventilation sweep of the canal, during the refueling process. The lower compartment purge exhaust ductwork is arranged as to sweep the reactor well during the refueling process.

During normal Plant operation, Modes 1-4, the containment purge and exhaust isolation valves are sealed closed. Generally, purging of Upper Containment and Lower Containment is performed during Mode 5, Mode 6 and may be performed during No Mode. The option to purge the Incore Instrumentation Area is not typically exercised. During fuel movement evolutions, purges normally last about 60 hours for unloading and 60 hours for loading.

2. Containment Upper Compartment Ventilation

   The Containment upper compartment ventilation subsystem consists of four freestanding, recirculating ventilation units and their associated cooling coils, filters and ductwork. The ventilation units located in the upper Containment compartment are designed for a total cooling capacity sufficient to maintain an average air temperature between 75° and 100°F during Modes 1, 2, 3 & 4. A minimum Temperature of 60°F may be maintained during Modes 2, 3 & 4.

3. Containment Lower Compartment Ventilation

   The lower Containment compartment ventilation subsystem is the largest of the Containment ventilation systems consisting of four recirculating ventilation units and their associated cooling coils and ductwork. This equipment is located in the annular concrete chambers around the periphery of the lower Containment compartment. The system is capable of maintaining an average air temperature between 100° and 120° F during Modes 1, 2, 3 & 4. A minimum Temperature of 60°F may be maintained during Modes 2, 3 & 4. The Containment lower compartment temperature may be between 120° and 125° for up to 90 cumulative days per calendar year provided the lower compartment temperature average over the previous 365 days is less than 120°F. The temperature in the annulus between the reactor vessel and the primary concrete shield, however, may be allowed to reach 135°F without detrimental effects to the installed instrumentation. A temperature monitoring system is used to evaluate any effect of increased lower containment temperature on qualified life of electrical equipment in lower containment.

   The steam generator, pressurizer, and pipe tunnel booster fans provide air flow distribution to these respective areas of lower containment. The pipe tunnel booster fans also provide air flow distribution to the Digital Rod Position Indication (DRPI) cabinets.

4. Control Rod Drive Ventilation System

   The control rod drive ventilation subsystem consists of four recirculating fans and associated ductwork. The fans are located in the lower compartment outside the primary shield and the supply ducts are arranged to maintain the required flow of cooling air through the control rod drive mechanism shroud.

   Air is drawn from the lower compartment and is returned to the lower compartment after passing through the mechanism shroud. The heat removal capability provided by this air flow limits the air temperature exiting the mechanism shroud during normal unit operation. The ducted air is routed in a manner to prevent causing the temperature of nearby concrete to exceed 150°F.

5. In-core Instrument Room Ventilation

The in-core instrumentation area is a dead-ended part of the lower Containment compartment. The in-core instrument area ventilation subsystem consists of two freestanding ventilation units and associated cooling coils, filters, and ductwork. A maximum temperature of 100°F during unit operation and minimum of 60°F during shutdown is maintained.

Purge air is supplied to the in-core instrumentation room through one 100 percent capacity fan and its associated filters, heating coil, and ductwork. Purged air is exhausted through one 100 percent capacity fan and filter train to the unit vent where it is monitored during release to the atmosphere. Two normally closed isolation valves in each of the two penetrations provide in-core instrumentation room isolation.

6. Containment Auxiliary Carbon Filter

   This system consists of one fan-filter unit located in the lower Containment compartment and arranged to assure uniform mixing of the lower compartment atmosphere. Prefilters, absolute and carbon filters are provided for reduction of fission product activity which is airborne in the lower compartment as the result of fuel cladding defects and reactor coolant leakage.

   The lower compartment atmosphere is continuously monitored for radioactivity during reactor power operation for display in the Control Room.

9.4.5.3 Safety Evaluation

The Containment Purge and Ventilation System provides adequate capacity to assure that proper temperatures are maintained in the various portions of the Containment under operating and shutdown conditions in all types of weather. Sufficient redundancy is included to assure proper operation of the system with one active component out of service. The system also incorporates provision for purge of the in-core instrumentation room atmosphere so that entry maybe achieved, when necessary.

All ventilation systems are of the recirculation type with the exception of the purge systems and are completely contained within the Containment structure. The purge systems isolation valves are normally sealed closed during unit operation, but the valves would close automatically on Containment isolation. The recirculation ventilation systems are so arranged that all components of each system are located wholly in the upper or lower Containment compartment. This eliminates the need for ductwork penetrating the divider barrier and enhances barrier integrity.

Safety injection and high radiation signals are provided to initiate the isolation of the Containment Purge Ventilation System. The safety injection signal includes high containment pressure signal actuation and is designed and qualified to Class 1E criteria. High radiation and safety injection signals have independant reset circuits and require deliberate manual operator action to reset following an automatic actuation.

In an accident, the containment purge and exhaust isolation valves are assumed to be closed (Modes 1-4).

Analysis has shown that the maximum torque which would be required to close the valves in a postulated accident is 2600 in-lbs. The output torque of the valve operator is 6150 in-lbs. The potential for valves to be clogged and rendered inoperable by the debris generated in a postulated accident is minimized because the isolation valves are sealed closed. In addition, duct work upstream of purge isolation valves is constructed of 10-gauge welded stainless steel;

this duct work will withstand 28.6 psig external pressure, or about twice the peak accident pressure.

Design specifications for the equipment located in the incore instrumentation area require that no loss of function should result when operating in temperatures up to 120°F and humidity up to 95 percent which may occur upon the loss of the ventilation system. Thus, there is a wide margin between the design limit and the normal operating environment for this equipment.

Section 7.7 notes that incore instrumentation is not used for unit operation. It is used for surveillance only in providing the data gathering functions as noted in Technical Specifications and effects of deviation therefore have no potentially adverse effects on unit operation.

A fuel handling accident inside containment has been analyzed assuming the purge system is in operation during refueling operations. This analysis is described in Section 15.7.4.

Design specifications for the purge system isolation valves are presented in Table 9-39.

Deleted paragraph(s) per 2002 revision.

9.4.5.4 Tests and Inspections

Tests and Inspections as described below were performed to assure and demonstrate the capability of the system and its components to perform the assigned function.

1. Manufacturer Shop Testing

   The manufacturer is required to verify by appropriate tests the following:

   a. Carbon Filter:

      Carbon filter capabilities for removal of molecular iodine-131 and methyl iodide-131.

      Carbon filter iodine collection capability

      Carbon filter cell leak-tightness integrity

      Carbon filter flow resistance

   b. Fan:

      Proper head and flow characteristics

   c. Heating and Cooling Coils:

      Proper heating and cooling capabilities

2. System Testing and Inspection

   Operational testing was performed prior to initial startup to demonstrate proper functioning of the system. Testing includes the following:

   a. Leak-tightness tests of components and system

   b. Functional verification of system controls and interlocks.

   c. During hot functional testing, a temperature survey is made in the Containment to verify acceptable air temperatures are maintained.

   d. In place testing of exhaust filters.

   Thereafter, periodic tests and inspections are performed to demonstrate system readiness and operability.

   Test criteria for the purge system isolation valves are presented in Table 9-39.

9.4.5.5 Instrumentation Application

A pressure transmitter is provided to measure the differential pressure across the Containment auxiliary carbon filter train. Temperature sensors, embedded in the Containment auxiliary carbon filter, are provided to indicate excessive filter temperature. Indication is provided in the Control Room. Computer point RTDs are located in the Lower Compartment Ventilation System discharge duct to enable discharge temperature readings from the Operator Aid Computer.

9.4.6 Diesel Building

9.4.6.1 Design Bases

The Diesel Building Ventilation System is designed to provide a suitable environment for the operation of equipment and personnel access as required for inspection, testing and maintenance.

Heating is provided by electricity.

Instrumentation for the Diesel Building Ventilation System is provided to control and indicate temperature.

9.4.6.2 System Description

The Diesel Building Ventilation System (Figure 9-123) is composed of the following components:

1. During normal ventilation operation one set of roughing filters and one set of filters with an efficiency of approximately 90% are used to filter the outside supply air. Filters are sized to accommodate the combustion air flow requirements for each diesel engine.
2. During normal ventilation operation one ventilating fan and one heating coil for maintenance of building temperature (125°F maximum and 55°F minimum) and building ventilation. Dampers are provided at the fan suction and arranged to close when the diesel engine starts to preclude the possibility of room air short cycling to the combustion air intake in the event of fan failure.
3. During general ventilation operation two large 50 percent capacity ventilation fans, ducts, and diffusers are arranged to maintain minimum ventilation requirements. The maximum temperature expected in the Diesel Building is 125°F.

9.4.6.3 Safety Evaluation

The Diesel Building Ventilation System automatically maintains a suitable environment in each diesel enclosure under all conditions. All safety related mechanical equipment for the Diesel Building Ventilation System is ANS Safety Class 3. Since the Diesel Building Ventilation System is duplicated for each diesel, a single failure would not impair the safety function of the system.

9.4.6.4 Tests and Inspections

Tests and inspections are performed to assure and demonstrate the capability of components and the system to perform the assigned function.

Manufacturer Shop Testing

The manufacturer was required to verify by appropriate tests the following:

1. Fan:

   Proper head and flow characteristics

2. Heating Coils:

   Proper heating capabilities.

<u>System Testing and Inspection</u>

Operational testing was performed prior to initial startup to demonstrate proper functioning of the system. Testing includes the following:

1. Leak tightness test of components and system.
2. Functional verification of system controls and interlocks.
3. Verification that an environment suitable for personnel access is maintained within the diesel enclosure while the engine is operated at rated load for a time sufficient to reach engine temperature equilibrium plus one hour.

Thereafter, periodic tests and inspections are performed to demonstrate system readiness and operability.

9.4.6.5 Instrumentation Application

The diesel building normal ventilation fan and the damper at the fan suction are energized when the diesel is not running. A self-contained thermostatic control is incorporated in the diesel building normal heating coil unit. The diesel building normal ventilation fan and heater perform no safety function and are not supplied with emergency power.

The diesel building normal ventilation fan and the damper at the fan suction are de-energized when the diesel starts. The diesel building general ventilation fans are automatically energized on diesel start.

Design provisions for fire containment provide for stopping fans and closing associated dampers in-the-event of system actuation of the room fire suppression.

9.4.7 References

1. Letter from W. O. Parker, Jr. (Duke) to H. R. Denton (NRC) dated October 14, 1980. Subject: Override, Bypass and Reset of Safety Actuation Signal.
2. Letter from D. B. Vassallo (NRC) to W. O. Parker, Jr. (Duke) dated August 28, 1975. Subject: Additional Information Required during FSAR Review.
3. Letter from W. O. Parker, Jr. (Duke) to H. R. Denton (NRC) dated April 6, 1979. Subject: Degree of Conformance to Regulatory Guides.
4. Letter from S. K. Blackley, Jr. (Duke) to C. J. Wylie (Duke) dated June 15, 1976. Subject: Auxiliary Building Filtered Exhaust System Modifications.
5. MCC-1211.00-14-0006: Design Input Calculation to the Dose Consequence Calculation MCC-1227.00-00-0048.
6. McGuire PIP M-01-01677. Subject: Audit team questioned taking credit in the dose calculations for a non-ESF grade ventilation system.

THIS IS THE LAST PAGE OF THE TEXT SECTION 9.4.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

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

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

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

A Safety Evaluation was issued on December 6, 2016 by the NRC, that transitioned the existing fire protection program to a risk-informed, performance-based program based on NFPA 805, in accordance with 10 CFR 50.48(c).

(Clarifying Note: Throughout this UFSAR section on Fire Protection, general reference is made to the NFPA 805 Fire Protection Program Design Basis Document (DBD) in accordance with FAQ 12-0062, Revision 1, UFSAR Content (ADAMS ML121430035) and NEI 98-03, Revision 1, Guidelines for Updating Final Safety Analysis Reports. General reference of this DBD is only intended to reduce unnecessary detail in the UFSAR and direct the reader for additional information and is in no way to be construed as "incorporation by reference" as defined in NEI 98-03, Revision 1.)

9.5.1.1 Design Basis Summary

9.5.1.1.1 Defense-in-Depth

The fire protection program is focused on protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations. The fire protection program is based on the concept of defense-in-depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:

1) Preventing fires from starting,

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

9.5.1.1.2 NFPA 805 Performance Criteria

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

- Nuclear Safety Performance Criteria. Fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria shall be met.
  a) Reactivity Control. Reactivity control shall be capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity inserting shall occur rapidly enough such that fuel design limits are not exceeded.
  b) Inventory and Pressure Control. With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of controlling coolant level such that subcooling is maintained such that fuel clad damage as a result of a fire is prevented for a PWR.
  c) Decay Heat Removal. Decay heat removal shall be capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition.
  d) Vital Auxiliaries. Vital auxiliaries shall be capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.
  e) Process Monitoring. Process monitoring shall be capable of providing the necessary indication to assure the criteria addressed in (a) through (d) have been achieved and are being maintained.
- Radioactive Release Performance Criteria. Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, Limits.

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

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

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

9.5.1.1.3 Codes of Record

The codes, standards and guidelines used for the design and installation of plant fire protection systems are listed in the NFPA 805 Fire Protection Program Design Basis Document (MCS-1465.00-00-0008).

9.5.1.2 System Description

9.5.1.2.1 Required Systems

Nuclear Safety Capability Systems, Equipment, and Cables

Section 2.4.2 of NFPA 805 defines the methodology for performing the nuclear safety capability assessment. The systems, equipment, and cables required for the nuclear safety capability assessment are referenced in MCS-1465.00-00-0008, NFPA 805 Fire Protection Program Design Basis Document.

Fire Protection Systems and Features

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features. Compliance with Chapter 3 is documented in the NFPA 805 Fire Protection Program Design Basis Document (MCS-1465.00-00-0008).

Chapter 4 of NFPA 805 establishes the methodology and criteria to determine the fire protection systems and features required to achieve the nuclear safety performance criteria of Section 1.5 of NFPA 805. These fire protection systems and features shall meet the applicable requirements of NFPA 805 Chapter 3. These fire protection systems and features are documented in the NFPA 805 Fire Protection Program Design Basis Document (MCS-1465.00-00-0008).

Radioactive Release

Structures, systems, and components relied upon to meet the radioactive release criteria are documented in the NFPA 805 Fire Protection Program Design Basis Document (MCS-1465.00-00-0008).

9.5.1.2.2 Definition of “Power Block” Structures

Where used in NFPA 805 Chapter 3 the terms “Power Block” and “Plant” refer to structures that have equipment required for nuclear plant operations. For the purposes of establishing the structures included in the fire protection program in accordance with 10 CFR 50.48(c) and NFPA 805, the plant structures listed in the NFPA 805 Fire Protection Program Design Basis Document (MCS-1465.00-00-0008) are considered to be part of the ‘power block’.

9.5.1.3 Safety Evaluation

The NFPA 805 Fire Protection Program Design Basis Document (MCS-1465.00-00-0008) documents the achievement of the nuclear safety and radioactive release performance criteria of NFPA 805 as required by 10 CFR 50.48(c). This document fulfills the requirements of Section 2.7.1.2 “Fire Protection Program Design Basis Document” of NFPA 805. The document contains the following:

- Identification of significant fire hazards in the fire area. This is based on NFPA 805 approach to analyze the plant from an ignition source and fuel package perspective.

- Summary of the Nuclear Safety Capability Assessment (at power and non-power) compliance strategies.
- Deterministic compliance strategies
- Performance-based compliance strategies (including defense-in-depth and safety margin)
- Summary of the Non-Power Operations Modes compliance strategies.
- Summary of the Radioactive Release compliance strategies.
- Summary of the Fire Probabilistic Risk Assessments.
- Summary of the NFPA 805 monitoring program.

9.5.1.4 Fire Protection Program Documentaion, Configuration Control and Quality Assurance

In accordance with Chapter 3 of NFPA 805 a fire protection plan documented in the Fleet Fire Protection Program Manual (PD-EG-ALL-1500) and the NFPA 805 Fire Protection Program Design Basis Document defines the management policy and program direction and defines the responsibilities of those individuals responsible for the plan's implementation. The Fleet Fire Protection Program Manual and/or the NFPA 805 Fire Protection Program Design Basis Document:

- Designates the senior management position with immediate authority and responsibility for the fire protection program.
- Designates a position responsible for the daily administration and coordination of the fire protection program and its implementation.
- Defines the fire protection interfaces with other organizations and assigns responsibilities for the coordination of activities. In addition, the Fleet Fire Protection Program Manual identifies the various plant positions having the authority for implementing the various areas of the fire protection program.
- Identifies the appropriate authority having jurisdiction for the various areas of the fire protection program.
- Identifies the procedures established for the implementation of the fire protection program, including the post-transition change process and the fire protection monitoring program.
- Identifies the qualifications required for various fire protection program personnel.
- Identifies the quality requirements of Chapter 2 of NFPA 805.

9.5.2 Communication Systems

9.5.2.1 Design Bases

There are two primary communication systems at the McGuire Station: the station Telephone System and the Public Address System. These systems are designed in such a manner as to satisfy the single failure requirement. In addition, there are a limited number of telephones on a direct line from the commercial telephone system and a limited number of telephones on a direct line from the Duke Power microwave system.

There is an onsite emergency radio system. This system with portable radios is normally used by security but has been upgraded to provide assured communications between the Standby Shutdown Facility (SSF) and locations where local control actions are required for SSF operation and is for onsite communications only.

Duke committed to additional Communication equipment in response to NRC Order EA-02-026 (B.5.b requirements). This equipment is controlled by Emergency Management response procedures.

9.5.2.2 Use of Communication Systems

Direct voice communication within the station is handled by the Station Telephone System by use of extension telephones with direct dialing between extensions. Station personnel may be paged by use of the station telephones and a special interface between the telephone switch (i.e., PAX, PBX, and PAPBX which are used interchangeably) equipment and the P.A. amplifiers. The process consists of dialing “719” on selected station telephones and then voice paging the desired party, who in turn can direct dial the original caller at his extension. Other buildings on the McGuire site are part of the paging system. Personnel located in these buildings may be paged by dialing the appropriate 3 digit P.A. System access prefix for that building. A 3 digit access prefix is also available which will access the P.A. System in all buildings on the site. In the event of a failure of the telephone switch, voice paging and direct conversation can be accomplished by the use of P.A. handsets. Commercial telephone lines or Duke microwave lines may be accessed for outside calls by any extension telephone of the switch. These calls may be placed either by direct dialing or through the station console operator, depending upon which extension telephone is being used to place the call. Incoming outside calls may be received from the commercial telephone lines or the Duke microwave lines by the station console operator, who can then transfer the calls to any extension telephone of the switch. In addition to these interfaced lines, a commercial line and a Duke microwave line are extended directly to specific telephones in several vital locations in the station.

The onsite radio system consists of five UHF repeaters. Three of these repeaters are used exclusively by security and for medical emergency responders. Two repeaters are used to communicate in the event of a fire and by Operations in the routine performance of their duties. These two repeaters are configured to allow up to four simultaneous and independent speech paths or talk groups. The talk groups used by Operations and Fire Brigade could exist on either repeater at any time. Upon loss of one repeater, the remaining repeater automatically allows two simultaneous and independent speech paths or talk groups for use by Operations in the routine performance of their duties and the Fire Brigade in the event of a fire.

9.5.2.3 System Description

Station Telephone System

This system provides the primary means of communication for both direct conversation and for voice paging. For paging from telephones, a special interface is provided between the telephone switch and the P.A. system amplifiers. The telephone equipment is comparable to that of the local telephone company in operation and in equipment quality. Telephones are located so as to be accessible from any occupied area of the station. Therefore, the telephone system should be considered adequate for communication purposes should the P.A. system be destroyed or disabled. The telephone switch is housed in the communications facility. It is powered via an AC-DC-AC rectifier/battery inverter combination which is normally fed from retail service. A dedicated diesel generator is provided to automatically start and accept load if normal power is lost. The addition and deletion of telephones for specific applications is provided on a continuing basis. Drawings covering the Station Telephone System are shown in Figures 9-132 through 9-138.

Public Address System

In the event of a failure of the telephone switch, the P.A. system remains with an adequate number of P.A. handsets for general coverage of the station site. The necessary power for speaker amplifiers and handset preamplifiers is from a supply which is separate from the supply feeding the Station Telephone System.

Duke Power Microwave and Commercial Telephone Service

Normal outside calls are placed or received by any of the station telephones, some directly and some through the station console operator, by the use of an interface between the station switch and commercial telephone lines and another interface between the switch and Duke Power microwave lines. In addition to these interfaced lines, a commercial line and a microwave line are extended directly to specific telephones in several vital locations in the station.

The security radio communications system was expanded to provide a portable radio system that will provide communications capability between all required locations. Specifically, this system was upgraded to ensure communications between the SSF and locations where local control actions are required for operation of the SSF. This system is independent of the other communications systems.

Single Failure Analysis of the Total Communication System

As shown in Table 9-41, the total communication system is designed such that a single fault within the system does not result in a lack of adequate communication coverage of the station.

9.5.2.4 Tests and Inspections

Since the continued usage of all communication systems provides constant checks as to the operational status of the systems, periodic testing is not necessary.

Pre-operational testing is performed to verify operation of all functions available. This includes ability to communicate from all areas.

9.5.3 Lighting Systems

9.5.3.1 Normal Lighting Systems

Normal lighting is provided by the use of High Intensity Discharge (HID), fluorescent, incandescent and LED lighting units. These units provide adequate levels of light with good distribution.

9.5.3.2 Containment Lighting

The Containment has incandescent lighting units powered with 208Y/120 VAC dry type transformers. The 600 volt sources are located outside the Containment and are connected to separate switchboards to reduce the chance of failure. The units, transformers, and panelboards are separated physically in the Containment at different elevations to provide local area illumination.

9.5.3.3 Control Room Lighting

Control Room lighting is provided using a luminous ceiling and fluorescent lamps. Four panelboards furnish power at 208Y/120 VAC and the lighting units are connected so that the failure of one panelboard does not reduce the lighting level of either unit below that required for operation.

9.5.3.4 Emergency Lighting System

For each of the units, there is a separate emergency 250 volt dc lighting system and a separate emergency 208Y/120 volt ac lighting system, and a 12V DC, self contained battery lighting system provided for hot standby.

Emergency 250 Volt DC Lighting System

The 250 volt dc lighting system, which is normally de-energized, provides operating level lighting in the Control Room and lighting at selected stairs and corridors in the Containment, Auxiliary, and Turbine Buildings. The emergency lighting is energized automatically by an undervoltage sensing relay mounted on the individual panelboards located in their associated areas. Control power for the undervoltage transfer circuit is provided from the normal ac lighting panelboards. A test button is provided at each panelboard to test the operability of the system without affecting normal lighting. All associated lighting units are incandescent.

Emergency AC Lighting System

The emergency ac lighting system, which is normally de-energized, provides lighting in the following parts of the Auxiliary Building: Control Room, Cable Room and Equipment Room stairs, exits, corridors, Hot Machine Shop, Fuel Pool, Fuel Unloading Area, Decontamination Rooms, Pump and Tank Room areas, Fan and Ventilation Rooms, Penetration Rooms, Purge Rooms and Diesel Rooms. The stairs and platforms in the Containment are also provided lighting to enable personnel to leave or enter the structure. Power is provided from four essential 600 volt ac control centers through four panelboards. Each unit has two panelboards of which one is located in the Auxiliary Building and one is located in the Service Building. The emergency ac lighting is energized automatically by undervoltage sensing relays monitoring a selected 208Y/120VAC normal lighting panelboard.

Emergency 12 VDC Lighting System

The 12V DC battery lights are located throughout plant buildings to provide emergency illuminating for hot standby in accordance with the requirements of 10 CFR 50.48(c). These lights are used during fire initiated station blackouts. Battery lights with 8 hour capacity are provided in the Turbine and Auxiliary Building. Battery lights with 1.5 hour capacity are provided in the SSF.

9.5.3.5 Failure Analysis for Lighting Systems

As shown in Table 9-42, the lighting system is designed such that a power failure does not result in a lack of adequate lighting for the station.

9.5.3.6 Tests and Inspections

Preoperational testing is performed on the Emergency AC and DC Lighting Systems to verify the operability of the system and to assure sufficient lighting is provided by the system at selected stairs and corridors in the Reactor, and Auxiliary Buildings. Periodic testing of the emergency DC lighting is conducted in selected areas of the stations.

9.5.4 Diesel Generator Fuel Oil System

9.5.4.1 Design Bases

The design bases of the Diesel Generator Fuel Oil System are to supply fuel oil to the diesel engines that drive the emergency generators, provide a return path to the day tank for excess fuel oil, and remove fuel oil that leaks or drips from the fuel injector nozzles and pumps. The system allows the diesel generators to operate at accident loads for a minimum period of five days without refueling.

9.5.4.2 System Description

A complete Emergency Diesel Generator Fuel Oil Supply System is provided for each of the four emergency diesel generators as shown in Figure 9-139. All fuel oil delivered to an engine by tanker truck is first stored in a 50,000 gallon underground storage tank. Oil is then transferred by the fuel oil transfer pump to the 275 gallon day tank, which allows approximately a 1/2 hour supply of fuel to be stored near the engine.

The fuel oil is then pumped to the engine by either the fuel oil booster pump or the main engine-driven pump. The booster pump is used during start-up operations when the main pump has not yet come up to full speed. A check valve is installed in parallel with each pump to prevent recirculation of flow from the pump discharge to the pump suction. This parallel flow path allows fuel to bypass one pump when the other is operating.

The pump discharge travels through a duplex filter before entering the fuel oil headers mounted on each bank of the engine. From the headers, the fuel injection pumps supply diesel oil under pressure to the fuel injectors mounted in the cylinder heads. The amount of fuel injected is regulated by the governor through the fuel control shafts to the control rack, which alters the position of the fuel injector pump plunger and the effective stroke of the pump. The fuel injection system is designed to prevent dribble of fuel oil.

A diesel fuel oil drain header is located on each side of the engine. These headers are connected by individual pipes to cavities in the cylinder heads and in the injection pump deck of the frame. Oil leaking past the plunger and barrel of the injector pump and past the fuel injector spring seat returns through these lines to the drip tank. From here the oil is routed to the used oil storage tank.

A pressure regulating valve is provided on the engine that maintains fuel oil header pressure and returns excess fuel to the day tank.

A pressure regulating valve is provided downstream of the fuel oil transfer pump to assure adequate back pressure is provided to prevent gravity transfer of fuel oil from the storage tank to the day tank.

It is possible to align either underground fuel oil storage tank to the suction of either fuel oil transfer pump through the permanently installed cross connection line. This enables the underground storage tank for one engine to supply fuel oil to the other engine. By making this alignment, the amount of diesel fuel available to any one diesel generator is considerably in excess of 7 days supply. Additional fuel oil from diverse sources close to the McGuire plant is available on short notice. McGuire has provisions for procuring additional fuel oil under various conditions.

A permanent pump, filter and sample connection are installed to recirculate, clean-up, and sample the oil in each diesel fuel oil storage tank. At intervals not to exceed once per month, collected water is removed, and tanks are recirculated and sampled. The fuel oil is recirculated through the inline filter. This equipment is shared by fuel oil storage tanks A and B (per reactor unit).

The various modes of operation of the diesel generator fuel oil system are as follows:

1. manual start
2. automatic start
3. normal shutdown
4. emergency shutdown
5. recirculation

A manual start for testing purposes may be initiated from either the Control Room or the diesel control panel located in the Diesel Building.

The diesel starts automatically for a blackout or a LOCA. The fuel oil transfer pump is activated by the electronic current module on the fuel oil day tank to fill the tank to its proper level.

During normal operation, the fuel oil booster pump is off. Should the engine driven fuel oil pump discharge pressure fall below the setpoint value for some reason, the fuel oil booster pump will turn on to supply fuel oil to the engine.

A manual emergency stop is the only condition that does not involve a gradual reduction in load or the operation of auxiliary equipment to dissipate the buildup of residual heat. The conditions that cause an automatic emergency shutdown are diesel generator overspeed, low lube oil pressure, generator time overcurrent, and generator differential. Any of these conditions can result in the diesel generator being damaged in a short time. An automatic emergency shutdown will result from these conditions during testing or a blackout mode.

During the recirculation mode, fuel oil is sampled and filtered if necessary to prevent stratification and deterioration of fuel caused by standing for a long period of time.

The fuel oil day tank does not have an overflow return line to the storage tank, since the day tank is at a much lower elevation than the storage tank, but an overflow line is piped to the floor drains for collection in the D.G. room sump. There is a potential for oil spillage or overflow if the transfer pump does not cutoff at the desired fuel level in the day tank, however, a high alarm will be sounded for operator action to cut off the pump. Any oil overflow collected in the room sump is pumped to the Conventional Waste Water Treatment System where provisions have been made to trap the oil and prevent its release to the environment.

9.5.4.3 Safety Evaluation

Because the main storage tanks are underground, they do not require any additional protective measures from adverse environmental conditions. The storage tanks are coated for protection from corrosion. A single failure analysis of the system is presented in Table 9-43. See Table 3-4 for the safety classification of the diesel generator fuel oil system components.

Corrosion protection for the fuel oil storage tanks and underground piping complies with the MNS Coating Program.

9.5.4.4 Tests and Inspections

The system is fully tested and inspected before initial operation. Adequate operating performance is assured by periodic tests of the diesel generators as required by Technical Specifications.

9.5.4.5 Instrumentation Application

UPDATING THE INFORMATION CONTAINED IN THIS SECTION IS NOT REQUIRED.

The instrumentation information contained in this section is "historical" in nature and is intended to be used for reference only. For a complete and accurate record refer to the applicable flow diagrams and/or equipment database.

Refer to Section 7.6.12 for additional discussion of instrumentation.

9.5.4.5.1 Pressure

1. Engine Intake Fuel Oil Pressure

   This indication is performed by the following:

   a. Diesel Unit 1A - pressure gage 1FDPG5000

   b. Diesel Unit 1B - pressure gage 1FDPG5010

   c. Diesel Unit 2A - pressure gage 2FDPG5000

   d. Diesel Unit 2B - pressure gage 2FDPG5010

   This gage is a dual scale gage and indicates the oil pressure upstream and downstream of the engine intake duplex oil filter. Indications are given at the diesel control panel.

2. Engine Intake Fuel Oil Pressure

   A low fuel oil pressure alarm is located on the diesel control panel and is activated by the following:

   a. Diesel Unit 1A - pressure switch 1FDPS5000

   b. Diesel Unit 1B - pressure switch 1FDPS5010

   c. Diesel Unit 2A - pressure switch 2FDPS5000

   d. Diesel Unit 2B - pressure switch 2FDPS5010

   The alarm is activated if the fuel oil pressure falls to 22 psig while the engine is running. The pressure switch turns off the alarm and resets itself when the fuel oil pressure rises to 25 psig. The pressure switch is located at the engine fuel oil intake.

3. Engine Intake Fuel Oil Pressure

   The instrumentation provides for control of the fuel oil booster pump by the following:

   a. Diesel Unit 1A - pressure switch 1FDPS5001

   b. Diesel Unit 1B - pressure switch 1FDPS5011

   c. Diesel Unit 2A - pressure switch 2FDPS5001

   d. Diesel Unit 2B - pressure switch 2FDPS5011

The pressure switch is connected to the diesel engine fuel oil intake. The fuel oil booster pump starts when the fuel oil pressure falls to 20 psig. The pressure switch stops the pump and resets itself when the fuel oil pressure rises to 33 psig. The pressure switch is connected to the diesel control panel.

4. Fuel Oil Booster Pump Discharge Pressure The indication is performed by the following:

   a. Diesel Unit 1A - pressure gage 1FDPG5020

b. Diesel Unit 1B - pressure gage 1FDPG5030
c. Diesel Unit 2A - pressure gage 2FDPG5020
d. Diesel Unit 2B - pressure gage 2FDPG5030

This indication is given near the pump. It has a process range of 0-50 psig.

5. Fuel Oil Drip Tank Pump Discharge Pressure

This indication is performed by the following:

a. Diesel Unit 1A - pressure gage 1FDPG5080
b. Diesel Unit 1B - pressure gage 1FDPG5090
c. Diesel Unit 2A - pressure gage 2FDPG5080
d. Diesel Unit 2B - pressure gage 2FDPG5090

This indication is given near the pump. It has a process range of 0-20 psig.

6. Fuel Oil Transfer Pump Discharge Pressure

This indication is performed by the following:

a. Diesel Unit 1A - pressure gage 1FDPG5100
b. Diesel Unit 1B - pressure gage 1FDPG5110
c. Diesel Unit 2A - pressure gage 2FDPG5100
d. Diesel Unit 2B - pressure gage 2FDPG5110

This indication is given near the pump. It has a process range of 0-50 psig.

7. Fuel Oil Recirc. Filter Pressure

Inlet pressure indication is performed by the following:

a. Diesel Unit 1A - pressure gage 1FDPG5160
b. Diesel Unit 1B - pressure gage 1FDPG5160
c. Diesel Unit 2A - pressure gage 2FDPG5160
d. Diesel Unit 2B - pressure gage 2FDPG5160

This indication is given locally.

8. Fuel Oil Transfer Filter Pressure

Differential pressure indication is performed by the following:

a. Diesel Unit 1A - pressure gage 1FDPG5170
b. Diesel Unit 1B - pressure gage 1FDPG5180
c. Diesel Unit 2A - pressure gage 2FDPG5170
d. Diesel Unit 2B - pressure gage 2FDPG5180

This indication is given locally.

9.5.4.5.2 Deleted Per 2011 Update

9.5.4.5.3 Level

1. Fuel Oil Day Tank Level

   This function is performed by the following:

   a. Diesel Unit 1A - level electronic current module 1FDEM5040

   b. Diesel Unit 1B - level electronic current module 1FDEM5050

   c. Diesel Unit 2A - level electronic current module 2FDEM5040

   d. Diesel Unit 2B - level electronic current module 2FDEM5050

   This controls the fuel oil transfer pump, starting it at low set point and stopping it at high set point.

2. Fuel Oil Day Tank Level

   This instrumentation provides a high/low level alarm at the diesel control panel by the following:

   a. Diesel Unit 1A - level electronic current module 1FDEM5041

   b. Diesel Unit 1B - level electronic current module 1FDEM5051

   c. Diesel Unit 2A - level electronic current module 2FDEM5041

   d. Diesel Unit 2B - level electronic current module 2FDEM5051

3. Fuel Oil Drip Tank Level

   This instrumentation controls the drip tank pump by the following:

   a. Diesel Unit 1A - level switch 1FDLS5060

   b. Diesel Unit 1B - level switch 1FDLS5070

   c. Diesel Unit 2A - level switch 2FDLS5060

   d. Diesel Unit 2B - level switch 2FDLS5070

4. Fuel Oil Storage Tank Level

   Local indication is given by the following:

   a. Diesel Unit 1A - level transmitter 1FDMT5140 (Relay 1) [From 1FDME5140]

   b. Diesel Unit 1B - level transmitter 1FDMT5150 (Relay 3) [From 1FDME5150]

   c. Diesel Unit 2A - level transmitter 2FDMT5140 (Relay 1) [From 2FDME5140]

   d. Diesel Unit 2B - level transmitter 2FDMT5150 (Relay 3) [From 2FDME5150]

   Each storage tank has an electronic level sensing element which feeds a common transmitter. The common transmitter provides low level annunciator alarms to each diesel local alarm. It has a process range of 0-48,500 useable gallons of fuel oil. A low level alarm is also provided at the diesel control panel.

9.5.5 Diesel Generator Cooling Water System

9.5.5.1 Design Bases

The Diesel Generator Cooling Water System is designed to maintain the temperature of the diesel generator engine within a safe operating range. Auxiliary purposes are supplying cooling water for the lube oil cooler and the intercooler which cools the air leaving the turbo-charger.

9.5.5.2 System Description

The Diesel Generator Cooling Water System is a closed cooling system that circulates water through the diesel generator engine. After picking up heat in the cylinder liners and heads, the cooling water goes to a three way temperature control valve which maintains the engine outlet water temperature within a certain range by regulating the flow to the diesel generator cooling water heat exchanger. After being cooled by Nuclear Service Water in the heat exchanger the water goes to another three way temperature control valve which maintains a certain manifold inlet air temperature by regulating the cooling water flow to the intercooler. The cooling water then flows through the lube oil cooler where it cools the engine lubricating oil. The jacket water pump and the intercooler water pump circulate water through the system during normal operation. The jacket water heater and the jacket water heater circulating pump keep the engine temperature at a desired value when the engine is not operating (see Figure 9-141). Makeup water is normally supplied from the Makeup Demineralized Water (YM) system. The Nuclear Service Water (RN) system provides the assured source of makeup water to the KD system.

9.5.5.3 Safety Evaluation

The Diesel Generator Cooling Water System for each diesel unit is a Duke Class C System. Each diesel unit is housed separately in Category I structures which are part of the Auxiliary Building. Internal missiles, if generated, could only affect one diesel, since each is contained in a separate room. Since the diesel units themselves are fully independent and redundant for each nuclear unit, they meet the single failure criterion.

9.5.5.4 Tests and Inspections

System components and piping are tested to pressures designated by appropriate codes. Functional tests are performed before initial operation to demonstrate proper operation of the jacket water heater, jacket water heater circulating pumps, jacket water and intercooler water pumps. The capability of the Diesel Generator Cooling Water System to maintain engine temperatures, lube oil temperatures, and turbo charger exit air temperature is demonstrated.

Regularly scheduled diesel operation demonstrates the operational readiness of the Diesel Generator Cooling Water System.

9.5.5.5 Instrumentation Application

Refer to Section 7.6.13 for discussion of Diesel Generator Cooling Water System Instrumentation.

9.5.6 Diesel Generator Starting Air System

9.5.6.1 Design Bases

The Diesel Generator Starting Air System is designed to quickly start the diesel generator engine. Each diesel generator unit has two independent starting air subsystems, each with storage to provide at least two consecutive starts. The basis for the two start capability is given in Section 8.3. Analysis based on results obtained during testing finds the McGuire diesels are capable of starting 5 times consecutively from the initial conditions of one of the two starting air receivers isolated, the other receiver at the lowest pressure allowed by Technical Specifications and diesel room temperature at the highest allowed by Selected License Commitments. At least the first of these 5 consecutive starts will be a fast start. An auxiliary purpose is to supply control air for shutdown of each diesel generator unit. The control air shutdown function is not safety-related. Each diesel generator unit can start and continue to run without control air. On loss of control air, the run/shutdown cylinders are designed to fail to the fuel-supply position. Control air is supplied to perform the engine shutdown function. Shutdown is not a QA 1 function, but the control air must be vented to reposition the run/shutdown cylinders to the fuel supply position.

9.5.6.2 System Description

The Diesel Generator Starting Air System for each diesel unit consists of two redundant air compressors, two 100 cubic feet starting air tanks, two aftercoolers, and two air dryers, arranged as parallel, independent subsystems. Each air compressor is designed to deliver 40 SCFM at 250 psig. The air compressor discharges through an isolation valve to the aftercoolers, driers, and starting air tanks. The driers are designed to reduce air moisture to a pressure dew point ≤45°F. There is a normally-closed cross-tie connection between the redundant air compressors. The starting air tanks supply air through purifiers or filters to supply air to the following:

1. Control Air Header
2. Air distributors on both engine banks

The Diesel Generator Starting Air System normal alignment allows air from both air tanks to enter both the left and right starting air headers. When one air tank is isolated, both left and right starting air headers are supplied from the remaining air tank.

The control air passes through a pressure reducing valve and an air filter before being used for engine shutdown.

The air distributors on both banks of the diesel engine admit starting air to the starting air valves of the individual cylinder heads in firing order sequence. The starting air valves admit air to the individual cylinders to turn the engine over and start it.

Relief valves mounted on the compressor discharge lines, the starting air tanks, and the control air lines protect the system from excessive pressure (see Figure 9-143).

9.5.6.3 Safety Evaluation

The Diesel Generator Starting Air System, including its after-cooling support system, is not required for continued operation of the Diesel Generator once it has been started. The critical components of the Diesel Generator Starting Air System, i.e. portions downstream of the compressor check valve for each diesel unit are Duke Class C. Each diesel unit is housed separately in Category I structures which are part of the Auxiliary Building. Internal missiles, if generated, could only affect one diesel, since each is contained in a separate room. The diesel units themselves are fully independent and redundant for each nuclear unit. Therefore, they meet the single failure criterion.

9.5.6.4 Tests and Inspections

System components and piping are tested to pressures designated by ASME codes. Functional tests are performed before initial operation. Functional tests are performed before unit operation to verify proper operation of the air compressors, starting air tanks, and associated valves and controls. The Starting Air System for each diesel is demonstrated to have the capability of starting its respective diesel-generator within the required time independently of the starting air system for the redundant diesel. Adequate storage capacity for two successive starts is also verified. Analysis based on results obtained during testing finds the McGuire diesels are capable of starting 5 times consecutively from the initial conditions of one of the two starting air receivers isolated, the other receiver at the lowest pressure allowed by Technical Specifications and diesel room temperature at the highest allowed by Selected License Commitments. At least the first of these 5 consecutive starts will be a fast start.

Regularly scheduled diesel starting demonstrates the operational readiness of the Starting Air System.

9.5.6.5 Instrumentation Application

Refer to Section 7.6.14 for information on starting air instrumentation and control.

9.5.7 Diesel Generator Lubricating Oil System

9.5.7.1 Design Bases

The Diesel Generator Lubricating Oil System is designed to supply lubricating oil to all friction surfaces of diesel engine.

9.5.7.2 System Description

During normal operation, the lubricating oil leaving the diesel engine goes through an engine driven pump to the lube oil cooler. A before and after lube oil pump automatically circulates oil through the lube oil cooler until the engine driven pump is up to speed and after the engine is shut down. A three-way temperature regulating valve ahead of the lube oil cooler maintains a certain oil temperature from the engine by regulating the flow to the lube oil cooler. After the lube oil cooler, the oil passes through a 6-10 micron filter, and a duplex strainer and then reenters the engine. Engine lube oil is also pumped from the engine to an oil heater by the heater pump (see Figure 9-144).

9.5.7.3 Safety Evaluation

The Diesel Generator Lubricating Oil System for each diesel unit is a Duke Class C System. Each diesel unit is housed separately in Category I structures which are part of the Auxiliary Building. Internal missiles, if generated, could only affect one diesel, since each is contained in its own room. Since the diesel units themselves are fully independent and redundant for each nuclear unit, they meet the single failure criterion.

9.5.7.4 Tests and Inspections

System components and piping are tested to pressures designated by appropriate codes. Functional tests are performed before initial operation. Regularly scheduled diesel operation demonstrates the operational readiness of the Diesel Generator Lubricating Oil System.

9.5.7.5 Instrumentation Application

Refer to Section 7.6.15 for a discussion of the Diesel Generator Lubricating Oil System Instrumentation.

9.5.8 Groundwater Drainage System

9.5.8.1 Design Basis

The Groundwater Drainage System is designed to relieve hydrostatic pressure from the Reactor and Auxiliary Buildings by discharging groundwater collected in sumps to either the yard drains or the Turbine Building sumps. The Groundwater Drainage System is a shared system with three sumps in the Auxiliary Building.

9.5.8.2 System Description

The Groundwater Drainage System is shown on Figure 9-146.

Six groundwater sump pumps draw collected groundwater seepage through their pump strainers from three sump locations in the Auxiliary Building. Each sump contains two 100 percent capacity pumps (250 GPM each) one aligned to train A and the other to train B. The groundwater for sump C is pumped to the plant yard drains. Sumps A and B also collect small quantities of non-radioactive drains is collected in each sump as shown on Figure 9-146. The water from these sources is in such low quantities that the safety function of this system is not jeopardized. The sump pumps for sump A and B discharge to the Turbine Building sumps to enable monitoring of the liquid and conventional processing.

The RN pipe break is the Auxiliary Building flooding design basis event for the groundwater drainage system at 696 gpm total, 666 gpm for the pipe break and 30 gpm groundwater inleakage. The pump capacities and power supplies are such that both units are adequately served in the event of a RN pipe rupture. General Design Criteria (GDC) 5 applies to this system. This design meets the requirements of GDC 5 for shared systems.

The underdrainage grid drains to the groundwater drainage sumps as described in Section 2.4.13.5.

Each sump is isolated by its respective check valve which is provided to prevent reverse flow into the sumps.

As a part of the FLEX strategy in response to NRC Order EA-12-049, a connection has been added to the groundwater discharge piping (train A pumps) in sumps A and B to provide a flow path for connection of a submersible sump pump powered by a portable diesel powered generator. This pump would be used during an extended los of AC power (ELAP) event.

9.5.8.3 Safety Evaluation

This system is safety related because it protects a Category I seismic structure. Therefore, the groundwater sump pumps and their associated discharge valves are classified Safety Class 3 denoted in Table 3-4.

9.5.8.4 Tests and Inspections

System components and piping are tested to pressures designated by appropriate codes. Functional tests are performed before initial operation. The sump pumps for sump A and B discharge to the Turbine Building sumps which are routinely monitored for radioactivity. Groundwater Sump C is sampled periodically to ensure only groundwater is collected and not radioactive water or contaminants.

9.5.8.5 Instrumentation Application

A level switch in each sump starts the respective groundwater drainage sump pump A on high level. A computer alarm also is initiated upon high level. The level switch stops the respective groundwater sump pump A on low level.

Another level switch in each sump starts the respective groundwater drainage sump pump B on high-high level. An annunciator alarm is initiated upon high-high level. The level switch stops the respective groundwater sump pump B on low level.

Refer to Section 7.6.11 for additional discussion of the Groundwater Drainage System instrumentation.

9.5.9 Diesel Generator Crankcase Vacuum System

9.5.9.1 Design Bases

The Diesel Generator Crankcase Vacuum System is designed to reduce the concentration of combustible gases in the crankcase. It also reduces oil leakage around inspection doors and explosion relief valves.

9.5.9.2 System Description

The Diesel Generator Crankcase Vacuum System for each diesel unit consists of a vacuum blower and an oil separator. The crankcase vacuum blower is connected to the crankcase by means of a vent line. A variable orifice in the vent line controls the amount of vacuum and an oil separator in the vent line prevents drawing an excessive amount of lubricating oil from the crankcase. A drain line from the separator returns oil to the engine (see Figure 9-147).

9.5.9.3 Safety Evaluation

The Diesel Generator Crankcase Vacuum Systems' components, and piping are seismically qualified and built to applicable codes where possible. Each diesel unit is housed separately in a Category I structure which is part of the Auxiliary Building. Internal missiles, if generated, could only affect one diesel, since each is contained in a separate room. The diesel units themselves are fully independent and redundant for each nuclear unit; thus, they meet the single failure criterion.

9.5.9.4 Tests and Inspections

System components and piping are tested to pressure designated by appropriate codes. Functional tests are performed before initial operation. Equipment not in continuous use is subject to periodic tests. Visual inspections by station personnel are performed on a scheduled basis.

9.5.9.5 Instrumentation Application

The following interlock is provided in the Diesel Generator Crankcase Vacuum System:

1. Crankcase pressure

The design basis for this and other interlocks are given in Section 8.3.

Local and remote indicating and alarm devices are provided, as required for monitoring the system.

9.5.10 Diesel Generator Room Sump Pump System

9.5.10.1 Design Bases

The Diesel Generator Room Sump Pump System is designed to remove leakage and equipment drains in the Diesel Building and to protect the diesel generators from flooding due to a nuclear service water pipe rupture in the adjacent diesel room acting simultaneously with a turbine building flood.

9.5.10.2 System Description

Four sump pumps are provided for the room of each diesel generator. One is located in the pit under the diesel generator and the other three are located in the main Diesel Generator Room sump. All waste is pumped to the Conventional Waste Water Treatment System where provisions have been made to trap any oil and prevent its release to the environment (see Figure 9-148).

9.5.10.3 Safety Evaluation

The Diesel Generator Room Sump Pump System is a Duke Class C System beginning at the room sump. Each diesel unit is housed separately in Category I structures which are part of the Auxiliary Building. Internal missiles, if generated, could only affect one diesel, since each is contained in a separate room. The diesel units themselves are fully independent and redundant for each nuclear unit; thus, they meet the single failure criterion.

9.5.10.4 Tests and Inspections

System components and piping are tested to pressures designated by appropriate codes. Functional tests are performed before initial operation. Equipment not in continuous use is subject to periodic tests. Visual inspections by station personnel are performed on a scheduled basis.

9.5.10.5 Instrumentation Application

The sump pumps are controlled by level switches which turn the pumps on and off. Local and remote indicating and alarm devices are provided, as required for monitoring the system.

9.5.11 Diesel Generator Intake and Exhaust System

9.5.11.1 Design Bases

The Diesel Generator Intake and Exhaust System is designed to supply fresh air for combustion and exhaust gas cooling to the engine and dispose of the engine's exhaust.

9.5.11.2 System Description

A separate source of intake air for each diesel engine is taken from the Diesel Building Ventilation System, as shown in Figure 9-149, through an intake silencer. The turbocharger compresses the air and forces it through the intercooler which uses diesel engine cooling water to remove the heat of compression. The intercooler discharge air passes through the inlet manifold into the cylinders for combustion. Excess air flow provides exhaust gas cooling.

Exhaust gases discharge from the cylinders into the exhaust manifold and into the turbine compartment of the turbocharger. Engine exhaust passes through the exhaust silencer and through the outside wall before entering the atmosphere (see Figure 9-149).

9.5.11.3 Safety Evaluation

The Diesel Generator Intake and Exhaust Systems' components, valves and piping are seismically qualified and built to applicable codes where possible. Each diesel unit is housed separately in Category I structures which are part of the Auxiliary Building. Internal missiles, if generated, could only affect one diesel, since each is contained in a separate room. The diesel units themselves are fully independent for each nuclear unit. Therefore, they meet the single failure criterion.

The volume of air supplied for combustion and exhaust gas cooling in each Duke diesel (at rated load) is 38,800 cfm. This high flow rate means that a very large quantity of extraneous gas would be needed to cause significant dilution of combustion air. The following onsite sources of possible diluents are analyzed:

1. Engine exhaust gases - Due to the location of the exhaust system with respect to the air intake the possibility of continuous recirculation of exhaust gases is substantially reduced. Other factors which tend to assist in the dispersion of exhaust gases are local air turbulence and the high velocity of the exhaust gases. It is therefore inconceivable, even under the worst meteorological conditions, that dilution from exhaust gases could occur.
2. Gases stored onsite - The bulk storage of hydrogen and oxygen are the major sources of gas located at the McGuire site. Neither of these sources has sufficient volume to affect the diesel from developing full rated power or cause engine shutdown. Also, these gases are stored in individual bottles and it is very unlikely that accidental release of all the bottles would occur simultaneously.
3. Interaction with other plant related exhaust concurrent with fires - Because of the relative elevation level between the diesel air intake and the other plant related exhaust (diesel intake being at lower level) and since the air intake source is outside the Diesel Building very little, if any, measurable interaction would be seen during a fire.

9.5.11.4 Tests and Inspections

System components and piping are tested to pressures designated by appropriate codes. Functional tests are performed before initial operation. Equipment not in continuous use is subject to periodic tests. Visual inspections by station operating personnel are performed on a scheduled basis.

9.5.11.5 Instrumentation Application

The following local indicators are provided for monitoring the system.

1. Exhaust temperature

2. Inlet manifold pressure and temperature

9.5.12 Containment Air Release and Addition System

9.5.12.1 Design Basis

The Containment Air Release and Addition System is utilized to maintain the Containment pressure between the Technical Specification limits of -.3 and +.3 psig. The system is capable of maintaining the correct pressure during all operating modes including startup and shutdown.

9.5.12.2 System Description

Pressure increases in the Containment can be the result of several sources. Unit startup increases the Containment pressure due to the heat-up of the Containment atmosphere. Control valves and other air operated valves continuously relieve instrument air into the Containment. Nitrogen and other air systems leak a certain amount of gas to the Containment thus increasing the pressure. When the Containment pressure increases to a preset limit an alarm is initiated in the Control Room. Air samples of containment atmosphere are obtained and analyzed periodically. The results of the sample are used to prepare a Gaseous Waste Release permit (GWR) that establishes the maximum allowable release rate for VQ. Air releases are continuously monitored by process radiation monitors to ensure radiation release limits are not exceeded. When a valid GWR is in place, the operator will open valves VQ-1A, VQ-2B and VQ-4 and check to insure that the release rate is acceptable. Containment air is removed from the lower compartment, sent through a particulate, absolute and charcoal filter and released to the unit vent.

Pressure decrease in the Containment can be the result of unit shutdown. When the Containment pressure decreases to a preset limit, the system is manually started. Air is taken from the Auxiliary Building and drawn into the upper compartment of the Containment. As the Containment pressure increases due to the addition of air, the system is manually stopped at approximately 0 psig. The present limits are set to insure containment pressure will be maintained within the Technical Specification limits.

The system is shown on Figure 9-150.

9.5.12.3 Safety Evaluation

The Containment Air Release and Addition System provides adequate capacity to assure that proper pressures are maintained in the various portions of the Containment under all operating conditions. Sufficient redundancy is included to assure proper operation of the system with one filter train out of service. All the components of the system are designed for at least 15 psig. This avoids overpressurization for any postulated event. The systems isolation valves close automatically on Containment isolation. The valves have a closing time of 3 seconds to insure that the offsite dose limit is not exceeded for a postulated LOCA. Refer to Chapter 15 for the results this system has on off-site dose values. The isolation valves are capable of closing against a differential pressure of 20 psi. This insures the closure of these valves under any possible accident. The system is sized sufficiently small to prevent unacceptable offsite radiological consequences following a postulated LOCA.

9.5.12.4 Tests and Inspections

System components and piping are tested to pressures designated by appropriate codes. Functional tests are performed before initial operation. The manufacturer is required to verify by appropriate test the following items: carbon filter capabilities for removal of molecular iodine - 131 and methyl iodide carbon filter iodine collection capability, carbon filter cell leaktightness integrity, carbon filter flow resistance.

9.5.12.5 Instrumentation Application

Containment Pressure Loop NS5550 is utilized to initiate alarms to maintain the correct Containment pressure. Two pressure gages are also supplied to indicate the pressure drop across the particulate and absolute filters. A flow indicator is provided to prevent the maximum allowable release rate from being exceeded. A flow totalizer is also provided for recording the quantity of air released.

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