CHAPTER 9

AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

9.1.1 New Fuel Storage

9.1.1.1 Design Bases

9.1.1.1.1 Safety Design Bases

<u>Safety Design Bases - Structural</u>

1. New fuel storage racks containing a full complement of fuel assemblies are designed to withstand all credible static and dynamic loadings, prevent damage to the structure of the racks and contained fuel, and minimize distortion of the rack arrangements. For design criteria see Table 3.9B-2n.

2. Modules are designed to protect fuel assemblies and bundles from physical damage that may cause the release of radioactive materials in excess of 10CFR20 and 10CFR50.67 requirements under normal or abnormal conditions caused by impacting from fuel assemblies, bundles, or other equipment.

3. Racks are constructed in accordance with quality assurance (QA) requirements of 10CFR50 Appendix B.

4. New fuel storage racks are categorized as Safety Class 2 and Category I.

5. The new fuel storage facility is housed within a Category I structure that is tornado, missile, and flood protected.

6. The new fuel storage facility is designed to conform to the requirements of Regulatory Guide (RG) 1.29.

7. The new fuel storage facility is designed in accordance with General Design Criteria (GDC) 2, 3, 4, 5, 61, 62, and 63.

<u>Safety Design Bases - Nuclear</u>

New fuel storage racks are designed and maintained with sufficient spacing between the new fuel assemblies to assure that the array, when racks are fully loaded, is subcritical by at least 5 percent k_{eff} including allowance for calculational biases and uncertainties. The calculations performed assure that k_{eff}

≤ 0.95. Two computational methods were used in the criticality analysis for the new fuel storage rack: a lattice design code TGBLA06 to calculate the in-core k_∞ values, and a Monte Carlo code, MCNP, to obtain fuel storage rack k_{eff} values[1].

It is assumed that the storage array is infinite in all directions. Since no credit is taken for leakage, the values reported as effective neutron multiplication factors are, in reality, infinite medium neutron multiplication factors.

The biases between the calculated results and experimental results as well as the uncertainty involved in the calculations are taken into account as part of the calculational procedure to assure that the specified k_{eff} limits are met.

All nonsafety-related systems in the vicinity of the new fuel storage facility are seismically supported and, therefore, will not fail and cause increase in k_{eff} beyond the allowable.

9.1.1.1.2 Storage Design Bases

1. New fuel storage racks are supplied to accommodate a fuel load of 270 fuel assemblies representing 35 percent of a full core.

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

9.1.1.2 Facilities Description

The location of the new fuel storage facility within the reactor building is shown on Figure 1.2-11. Details of the new fuel storage vault are shown on Figure 9.1-1a. Each new fuel storage rack (Figure 9.1-1) holds up to 10 channeled or unchanneled assemblies in a row.

Fuel spacing (7.00 in nominal center-to-center within a rack, 12.25 in nominal center-to-center between adjacent racks) within the rack and from rack-to-rack limits the effective multiplication factor of the array (k_{eff}) to not more than 0.90 in a dry condition and 0.95 in a flooded condition.

Fuel assemblies are loaded into the rack through the top. Each hole for a fuel assembly has adequate clearance for inserting or withdrawing the assembly channeled or unchanneled. Sufficient guidance is provided to preclude damage to the fuel assemblies. The upper tie-plate of the fuel assembly rests against the rack to provide lateral support.

Design of the racks prevents accidental insertion of the fuel assembly in a position not intended for the fuel. This is

achieved by abutting the sides of each casting to the adjacently installed casting. In this way, the only spaces in the assembly are those into which it is intended to insert fuel. The weight of the fuel assembly is supported by the lower tie-plate which is seated in a chamfered hole in the base casting.

The floor of the new fuel storage vault is sloped to a nonclosable drain located at the low point. This drain removes any water that may be accidentally and unknowingly introduced into the vault. The drain is part of the reactor building equipment drains.

Radiation monitoring equipment for the new fuel storage area is described in Section 12.3.

9.1.1.3 Safety Evaluation

9.1.1.3.1 Criticality Control

The subcriticality of the low-density new fuel storage racks is determined by the interaction of three variables: spacing between fuel bundles, fuel reactivity, and moderator density. The analyses were performed with a nominal center-to-center spacing between rows of 12.25 in. The infinite lattice reactivity ($k\infty$) of the design basis fuel was 1.34, calculated in the uncontrolled reactor core geometry. All of the analyses assumed that the storage racks were fully loaded with the design basis fuel. All storage rack analyses conservatively assumed an infinite array, or no neutron leakage. Subcriticality does not depend upon the presence of neutron-absorbing materials.

Channeled and non-channeled assemblies, 90° assembly rotation, and eccentric assembly positioning were considered for both GE14 and GNF2 fuel assemblies. The centered non-channeled assemblies were determined to be bounding for GE14 fuel, where as the non-channeled 4-middle eccentric loading pattern was determined to be bounding for GNF2 fuel. An average planar enrichment of 4.9 wt% U-235 was assumed. In addition to considering biases and uncertainties in the method of calculation, tolerances on the following parameters are also evaluated: enrichment, pellet stack density, Gadolinia loading, cladding thickness, in-rack pitch, and between-rack pitch.

The new fuel storage racks in the normal (dry) storage condition have an effective multiplication factor of less than 0.78 including biases and uncertainties. In the abnormal condition of vault flooding, partial or full flooding, the effective multiplication factor is 0.87697 for GE14 fuel and 0.89275[15] for GNF2 fuel. The flooded condition conservatively assumes the normal maximum allowable spent fuel moderator temperature of 100°C, which is more reactive than the 20°C condition.

In the optimum moderation condition, with a water moderator density range of approximately 0.1 to 0.6 g/cc, the effective multiplication factor for the new fuel storage rack exceeds 0.98. To prevent an optimum moderation condition, 32 noncombustible aluminum covers are placed over the new fuel vault. Any misting or aerosols would be precluded from entering the vault when the covers are in place. Only the individual cover, in the area where fuel is being handled, will be removed at any given time. During this period the major portion of the new fuel vault will be covered, in turn limiting the likelihood of a mist or aerosol entering the vault. Also to preclude optimum moderation, water and firefighting materials are avoided in the new fuel storage area. To support this approach, the racks use noncombustible materials and by procedure combustible materials are restricted from the area.

9.1.1.3.2 Structural Design (Figure 9.1-1)

1. The new fuel storage vault contains 27 sets of racks, each of which may contain up to 10 fuel assemblies. A maximum of 270 fuel assemblies may be stored.

2. The storage racks provide an individual storage compartment for each fuel assembly and are secured to the vault wall through associated hardware. Fuel assemblies are stored in a vertical position, with the lower tie-plate engaging in a captive slot in the lower fuel rack support casting. Additional restraints are provided to restrict lateral movement.

3. The weight of the fuel assembly is held by the lower support casting.

4. New fuel storage racks are made of aluminum. Materials used for construction are specified in accordance with the applicable ASTM specifications. The material choice is based on a consideration of the susceptibility of various metal combinations to electrochemical reaction. When considering the susceptibility of metals to galvanic corrosion, aluminum and stainless steel are relatively close together insofar as their coupled potential is concerned. The use of stainless steel fasteners in aluminum to avoid detrimental galvanic corrosion is a recommended practice and has been used successfully for many years by the aluminum industry.

5. The minimum center-to-center spacing for the fuel assembly between rows is 12.25 in. The minimum center-to-center spacing within the rows is 7.00 in. Fuel assembly placement between rows is not possible.

6. Lead-in and lead-out guides at the top of the racks provide guidance of the fuel assembly during insertion or withdrawal.

7. The rack is designed to withstand the impact load of 4,000 ft-lb while maintaining the safety design basis. This impact load could be generated by the vertical free-fall of a fuel assembly from the height of 6 ft.

8. The storage rack is designed to withstand the pull-up force of 4,000 lb and a horizontal force of 1,000 lb. There are no expected vertical or horizontal forces in excess of 1,000 lb. The racks are designed with lead-outs to prevent sticking. However, in the event of a stuck fuel assembly, the maximum lifting force of the fuel handling platform grapple (assuming limit switches fail) is 3,000 lb.

9. The storage rack is designed to withstand horizontal combined loads up to 222,000 lb, well in excess of expected loads.

10. The maximum stress in the fully-loaded rack in a faulted condition is in Table 3.9B-2n.

11. The fuel storage rack is designed to handle nonirradiated, low emission radioactive fuel assemblies.

12. The fuel storage racks are provided protection from adverse environmental effects by proper design of the new fuel storage facility (Section 9.1.1.3.3).

9.1.1.3.3 Protection Features of Fuel Storage Facilities

A curb approximately 4 in high is provided around the top opening of the new fuel vault. The top opening is covered with noncombustible plates which protect the fuel and preclude the possibility of optimum moderation. The coverplates are removable sections, one over each new fuel storage rack. This will minimize the total number of uncovered assemblies during operations. Section 9.1.1.1.1 discusses additional protection features for the new fuel storage facility.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases

The spent fuel storage facility which includes the reactor building provides safe storage for approximately 4,000 fuel assemblies (523 percent of the full core fuel load). The spent fuel storage facility is designed to prevent severe natural phenomena, including missiles generated by high winds, from causing damage to the spent fuel. Also in the event of a release

of radioactivity from stored fuel, the spent fuel storage facility is capable of limiting offsite exposure to a small fraction of the 10CFR50.67 criteria.

The spent fuel storage pool is a Category I structure constructed of reinforced concrete and lined with stainless steel. The flooded spent fuel pool provides a water barrier which ensures sufficient shielding to protect plant personnel from exposure to radiation in excess of 10CFR20.

The freestanding spent fuel storage racks located in the spent fuel pool are also Category I. They are designed to maintain the stored spent fuel in a subcritical geometry under normal and abnormal conditions. The design provides for adequate cooling to handle the decay heat loads under normal and abnormal conditions. The racks will withstand both operating basis earthquake (OBE) and safe shutdown earthquake (SSE) loads, maximum uplift forces generated by the fuel handling equipment, and dropped fuel accident loads. The design and general arrangement of the spent fuel storage racks will allow handling of fuel assemblies during refueling operations.

The fully-loaded spent fuel storage rack array is designed to maintain K_{eff} ≤0.95 when the pool is filled with nonborated water at 68°F. The criticality analysis for GE14 fuel[13] demonstrates compliance with this limit for a maximum average planar enrichment of 4.9 wt% U-235 and in-core k_∞ ≤1.3392. The criticality analysis for GNF2 fuel[15] demonstrates compliance with this limit for a maximum average planar enrichment of 4.9 wt% U-235 and in-core k_∞ ≤1.32266. Additional margin exists due to Technical Specification 4.3.1.1.c, which limits storage to fuel with an in-core k_∞ ≤1.32. The assumed infinite geometric array is considered to be in the most critical arrangement considering mechanical tolerances and abnormal conditions.

9.1.2.2 Facilities Description

The spent fuel storage pool is a Category I reinforced concrete structure lined with stainless steel, which will be resistant to damage during normal refueling operations. The spent fuel storage pool is shown on Figure 9.1-2. The stainless steel liner is designed to remain in place and retain its leak-tight integrity during a SSE. The general arrangement of the spent fuel racks in the pool, Figure 9.1-3, will permit the storage of approximately 4,000 fuel assemblies.

The stainless steel liner plates are seam welded together and the liner is anchored to the concrete walls with reinforcing S-beam stiffeners. Except for a floor coverplate joint which is backed by a redundant weld, each seam is backed by a leak chase channel which forms a test cavity to verify the leak-tightness of the welds. The leak chase system also provides a means for detecting

leaks in the spent fuel pool and prevents the uncontrolled loss of contaminated water into the reactor building.

Loading combinations, as identified in Section 3.8.4.4.1, are used to develop concrete strains against the SST liner. The SST liner analysis incorporates these concrete strains, concrete placement loads, thermal loads, hydrostatic loads, static loads and seismic loads. The calculated SST liner strains are within the allowables set forth in the ASME Code, Section III, Division II, 1977 Edition, Subsection CC.

The refueling canal, which connects the spent fuel pool with the reactor head cavity, is sealed by two gates in series with a monitored drain between them. This arrangement will permit the monitoring of any leaks and allow a gate seal to be repaired, if necessary.

No connections are provided to the spent fuel pool below the normal water level that may cause the pool to be drained and, therefore, the fuel would not be uncovered should these lines fail. Two skimmer surge tanks are provided to accommodate water displacement due to the installation or removal of large items into the spent fuel pool. Sparger lines are equipped with siphon breakers to prevent siphoning of the spent fuel pool water in the event of a pipe break. The elevation of the spent fuel storage facility will preclude the possibility of flooding.

The design of the reactor building polar crane (RBPC) control circuit prevents the movement of the main hoist from going over the spent fuel pool (by means of electrical interlocks).

An interlock bypass keylock switch has been provided which allows the movement of main hoist of the RBPC over the spent fuel pool. This is required in order to support refueling operations. In addition, this modification can permit, if required, the RBPC to be used to support spent fuel pool servicing equipment (e.g., spent fuel pool vacuuming equipment).

Crane operation is not permitted in the spent fuel storage pool area during the following conditions:

1. Spent fuel storage pool water level is not within the limits contained in Technical Specifications.

2. Less than the minimum required operable ac electrical power sources listed in Technical Specifications are operable.

All usages of this keylock bypass switch will be in accordance with load handling procedures (per NUREG-0612), and will be administratively controlled by the Shift Manager (SM) and will require his written approval.

The cask storage pool location is structurally separated from the spent fuel pool and allows for underwater transfer of fuel. To allow for the use of the cask storage area for maintenance activities such as control rod blade storage, the gates between the spent fuel pool and the cask storage pool area are normally left in their stored location on the east wall of the spent fuel pool. When the cask storage pool is to be used for the transfer of irradiated components out of the pool, the gates can be installed before heavy loads are handled over the cask storage pool. However, during dry cask storage operations the gates remain out during the movement of the transfer cask and placing the shielf plug into the DSC in the transfer cask. The cask storage pool area is lined with stainless steel independent of the spent fuel pool. The spent fuel pool is connected to the cask storage pool area by a canal that can be sealed by two gates in series when the shipping cask is to be removed. Travel restrictions on the RBPC ensure that the shipping or storage cask will not travel over the spent fuel pool.

The spent fuel storage racks have been installed in phases. Figure 9.1-3 depicts the two-phased installation (Phase II and Final Phase) of the storage racks. The 10 storage racks installed in Phase II accommodate approximately 1,500 fuel assemblies. The Final Phase consists of 16 storage racks and provides locations for approximately 2,500 fuel assemblies.

The spent fuel racks are designed to maintain the stored spent fuel in a geometry that precludes the possibility of criticality. The racks will maintain this subcritical array when subjected to maximum earthquake conditions, dropped fuel assembly accident conditions, and any uplift forces generated by the fuel handling equipment.

The control blade storage frame which securely supports three control blade racks is normally stored in the cask storage pool. The freestanding frame is safety related (Category I) and is designed to maintain the stored racks and control blades under normal and abnormal conditions. The frame will withstand both OBE and SSE earthquake loads, maximum uplift forces generated by the control blade handling equipment, and demonstrate that no overturning of the storage frame occurs with and without the control blades in place. The frame, as designed and analyzed, will not alter the function or operation of the cask storage pool and its safety-related system. The existing racks, appropriated from the Nine Mile Point Nuclear Station - Unit 1 (Unit 1) fuel pool, are aluminum material and nonsafety related. The design of the frame is such that the aluminum will not come in contact with the stainless steel cask storage pool liner.

The spent fuel storage racks are designed and fabricated in accordance with ASME III Subsection NF, but are not code stamped because they are not a pressure-retaining component, and Subsections NA/NCA do not apply. The spent fuel rack material is 304SS in accordance with ASTM specifications. The racks comply

with Category I requirements in accordance with RG 1.29. Loads and load combinations are in accordance with Standard Review Plan (SRP) Section 3.8.4 for steel structures. The racks are analyzed using time-history inputs into a nonlinear seismic finite analysis. The results show an adequate safety margin for all earthquake conditions. The acceptance criteria are in accordance with Appendix D of SRP Section 3.8.4.

Phase II & Final Phase Racks

Figure 9.1-3b is a schematic of the dynamic model of a single rack used as an element in the development of the overall model of both the new and existing rack modules in the pool. It depicts many of the characteristics of the model including all of the degrees-of-freedom and some of the spring restraint elements. The model includes all fluid coupling interactions and mechanical coupling appropriate to performing an accurate nonlinear simulation. This 3-D simulation is referred to as a whole pool multi-rack (WPMR) model. The WPMR model is implemented by the computer program DYNARACK (also known as MR216) that performs dynamic simulations on systems and structures[5]. It is used to simulate rack structure response to seismic excitation. DYNARACK is benchmarked and verified in accordance with Holtec International quality procedures. Both Phase II and Final Phase rack modules are included. The existing racks are included in the analytical model to the extent necessary to properly compute fluid coupling coefficients for the entire pool.

The spent fuel racks are of welded stainless steel construction and follow the proven design concept of nonflux trap honeycomb prismatic construction. The basic building block for the modules of new racks is a square cross-section box. Each box is equipped with Boral neutron absorber panels on its sides to form a composite box. The Boral panels are held in place by stainless steel sheathing. The welding of the sheathing to the box is accomplished in a manner to prevent any heat-induced damage to the encased Boral.

The composite box assemblies of the new racks are arrayed in a vertical fixture over a solid monolithic baseplate, which is machined, with an array of equispaced cylindrical holes containing tapered crowns. These tapered holes serve as the seating surface for the nose of the fuel assembly. The centerlines of these machined holes define the geometrical axis of symmetry for each storage location. The composite boxes, arranged in a vertical checkerboard pattern, are joined to each other at their contiguous corners through connector bars. These connector bars are shaped to provide an effective groove weld (made using a TIG process) between the adjoining boxes without protruding into the square prismatic space formed by the flat surfaces of the exterior of the composite boxes. The length and axial spacing of the connecting bars is set to minimize weld-induced distortion of the boxes. The bottom edges of the honeycomb cell assemblage formed by their edge-to-edge fastening,

described in the foregoing, are attached to the baseplate by fillet welds.

The cellular array equipped with a lateral baseplate, completed through the preceding steps, is next equipped with vertical pedestals. The pedestals consist of an internally-threaded austenitic stainless steel slab with an externally-threaded spindle made from a high-strength alloy (SA564-630 precipitation hardened stainless steel). The support pedestal is connected to the underside of the baseplate through suitably sized continuous fillet welds. The high-density fuel storage rack assembly is supported at four corners. The supports elevate the rack baseplate above the pool floor level, creating a water plenum for coolant inventory. The upright fuel rack, fabricated in the manner described above, stands on the pool slab, with bearing pads serving as intervening elements sized to diffuse the concentrated load from the pedestals into the body of the reinforced concrete without local overstress.

In addition to the structural and nonstructural stainless material, the racks employ Boral, a patented product of AAR Manufacturing, as the neutron absorber material. A brief description of Boral and its pool experience list follows.

Boral is a thermal neutron poison material composed of boron carbide and 1100 alloy aluminum. Boron carbide is a compound having a high boron content in a physically stable and chemically inert form. The 1100 alloy aluminum is a lightweight metal with high tensile strength, which is protected from corrosion by a highly resistant oxide film. The two materials, boron carbide and aluminum, are chemically compatible and ideally suited for long-term use in the radiative, thermal, and chemical environment of a nuclear reactor or a spent fuel pool.
Boral's use in spent fuel pools as the neutron absorbing material can be attributed to its proven performance and the following unique characteristics:

1. The content and placement of boron carbide provides a very high removal cross-section for thermal neutrons.

2. Boron carbide, in the form of fine particles, is homogeneously dispersed throughout the central layer of the Boral panels.

3. The boron carbide and aluminum materials in Boral do not degrade because of long-term exposure to radiation.

4. The neutron absorbing central layer of Boral is clad with permanently bonded surfaces of aluminum.

5. Boral is stable, strong, durable and corrosion resistant.

The cell geometry of the storage racks is shown on Figure 9.1-4a. The rack construction materials have an established history of in-pool use and are physically, chemically and radiologically compatible with the pool environment. Boral has been successfully used in a number of fuel pools in the U.S. and abroad and is suitable for use in this application. Austenitic stainless steel is perhaps the most widely used stainless alloy in nuclear power plants. The quality assurance requirements for fabrication of the spent fuel racks are in accordance with 10CFR50 Appendix B.

Criticality analyses[13, 15] were performed to ensure the acceptability of the NMP2 fuel to be stored in the spent fuel storage racks with the pool flooded with unborated water at a normal temperature of 20°C. The analytical methodology used consists primarily of two computer codes. TGBLA06 was used to calculate burned fuel compositions and the in-core k_∞ values. The burned fuel compositions were then used in MCNP to obtain fuel storage rack k_{eff} values. The fuel assemblies used as the principal design basis for the storage racks are the GE14[13] and GNF2[15] (10x10) assemblies containing UO_2 fuel rods clad in Zircaloy, and using a maximum average planar enrichment of 4.9 wt% U-235. Prior analyses[14] performed with CASM04 and MCNP[7,8,9,10, 16] have demonstrated that GE14 fuel bounds the earlier GE6/6B, GE9B, GE11, and GE13 fuel types in the NMP2 boral high-density racks.

The effects of calculation method uncertainties and manufacturing tolerances, such as B_{10} loading of the boral neutron absorber, boral width and thickness, fuel enrichment, stainless steel box wall thickness, lattice spacing, moderator temperature, and eccentric positioning, were evaluated and statistically combined in determining the maximum k_{eff} in the storage rack. The effect of uncertainty in depletion calculations, an MCNP calculational bias from comparison to benchmark critical experiments, and a benchmark gap uncertainty to account for the absence of credited fission product absorbers in critical experiments used for benchmarking MCNP, were additionally calculated[13, 15] and added to the maximum k_{eff} in the storage rack. The final result, excluding accident or abnormal conditions, shows a maximum k_{eff} of 0.9492 for GE14 fuel and 0.9415[15] for GNF2 fuel.

Abnormal and accident conditions, such as a dropped fuel assembly, fuel rack lateral movement, and abnormal location of a fuel assembly adjacent to the fuel rack, were also evaluated. Since the analysis of maximum k_{eff} is based on an infinite array of storage cells, loaded with the most reactive lattice analyzed, the above-mentioned abnormal and accident conditions are bounded. Thus, none of the abnormal or accident conditions that have been identified as credible will result in exceeding the limiting reactivity (K_{eff} of 0.95). The double contingency principle of ANSI 16.1-1975 specifies that at least two unlikely independent

and concurrent events must occur to produce a criticality accident. This principle precludes consideration of the simultaneous occurrence of multiple accident conditions. Other hypothetical events were considered and no credible occurrences or configurations have been identified that might have any adverse effect on the storage rack criticality safety.

The spent fuel pool is cooled by the spent fuel pool cooling and cleanup (SFC) system (Section 9.1.3).

9.1.2.3 Safety Evaluation

The design of the spent fuel storage racks provides for a subcritical effective multiplication factor K_{eff} <0.95 for both normal and abnormal storage conditions.

Normal conditions exist when the fuel storage racks are located in the pool and are covered with a normal depth of water (approximately 23 ft above the top of stored fuel), and with the fuel assemblies in their design storage positions. The spent fuel is sufficiently covered with water to meet the radiation zone requirements in Section 12.3 under this condition. The spent fuel pool is designed to prevent any inadvertent drainage and/or siphoning of water. There are no permanent drains in the spent fuel pool.

The reactor cavity and internals storage pools contain piping which, were it to leak, could result in loss of water from the spent fuel pool during refueling. However, the leakage rate from a crack in these low-pressure (moderate energy) lines would be slow. Sufficient time would be available for the plant Operators to initiate makeup water and stop the leak prior to any significant decrease in the spent fuel pool level.

An abnormal condition may result from accidental dropping of equipment or damage caused by the horizontal movement of fuel handling equipment without first disengaging the fuel from the hoisting equipment. To ensure that the design criteria are met, the following normal and abnormal spent fuel storage conditions are analyzed:

1. Normal positioning in the spent fuel storage array.

2. Eccentric positioning in the spent fuel storage array.

3. Pool water temperature increases to 212°F.

4. Normal storage array of ruptured fuel.

5. Moving fuel bundle between work area rack and storage area.

The spent fuel storage racks are designed to meet Category I requirements (Section 9.1.2.2). The maximum uplift forces that

can be developed by the fuel grapple hoist of the refueling platform, the auxiliary hoist of the refueling platform, or the refueling jib cranes, do not exceed the weight of an empty spent fuel storage rack.

The storage rack structures are assessed to withstand the impact resulting from a falling weight possessing 12,742 ft-lb of kinetic energy. When subjected to this impact load, those members that maintain spacing to ensure $k_{eff} \leq 0.95$ remain intact. The fuel handling accident evaluation (Section 15.1) is based on the dropping of a raised fuel assembly onto the top of the core. Since the fuel assembly density in the core exceeds that in the spent fuel pool, the accident evaluation in Section 15.7.4 bounds the conditions that could result from a fuel assembly dropping in the spent fuel pool.

The spent fuel storage racks are made of Type 304 stainless steel. Materials used for construction are specified in accordance with the latest issue of applicable ASTM specifications at the time of fabrication. Material selection is made considering the following:

1. The inherent high corrosion resistance of Type 304 stainless steel makes it well suited for use in demineralized water.

2. The high strength and high modulus of elasticity of Type 304 stainless steel make it well suited to withstand the design seismic vibratory loads.

3. The stainless steel acts as a partial neutron poison.

The spent fuel storage arrangement is designed considering the following:

1. The spent fuel storage facility is designed to protect stored fuel from cyclonic winds and postulated missiles generated by these winds. The effects of missile impact upon tornado- and missile-protected structures are presented in Section 3.5. Reactor building design is discussed in Section 3.8.4.

2. Any high levels of radioactivity released at the refueling level of the reactor building are detected in the refueling level ventilation exhaust duct by redundant radiation monitors and also by spent fuel pool area radiation monitors.

From the foregoing analysis, it is concluded that the spent fuel storage arrangement meets its safety design bases and satisfies the requirements of RG 1.13.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

9.1.3.1 Design Bases

The SFC system is designed to remove the decay heat released from the spent fuel elements and maintain a specified fuel pool water temperature, water clarity, and water level by:

1. Filtering fuel pool water to minimize corrosion product buildup and control fuel pool water clarity.

2. Filtering and demineralizing fuel pool water to minimize fission product concentration. This minimizes the release of fission products from the pool to the reactor building environment.

3. Monitoring fuel pool water level and providing makeup water to maintain an adequate height of water above the fuel. This provides required shielding for fuel storage and fuel handling operations and ensures adequate cooling.

4. Maintaining the fuel pool water temperature at or below 125°F* under normal operating conditions and below a maximum fuel pool design temperature of 150°F under all other conditions.

The SFC system is also designed to provide spent fuel pool, reactor cavity pool and reactor internals pool cooling during plant refueling outages. In this mode, the SFC system is required to backup the nonsafety-related alternate decay heat (ADH) removal system in the event of failure due to equipment malfunction or natural phenomena. The SFC system will maintain the average reactor coolant exit temperature below 150°F.

9.1.3.2 Description

The SFC system is shown on Figure 9.1-5. The cooling sections can operate independently from the cleanup section. Spent fuel pool water flows over adjustable weirs into the spent fuel pool surge tanks. Each spent fuel pool circulation pump takes suction from the surge tanks and circulates the spent fuel pool water through one or both of two heat exchangers, where it is cooled by the reactor building closed loop cooling water (RBCLCW) system, and then returns the water to the spent fuel pool through spargers located at the bottom of the spent fuel pool. The service water (SWP) can also be used to remove heat from the heat exchangers.

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* Following normal refueling outages of short duration, the fuel pool water will be maintained at lower temperatures, in accordance with station procedural controls, to ensure the design temperature of 150°F is not exceeded in the event of a station blackout (SBO).

The return spargers are located on the side of the pool opposite the overflow weirs, thus ensuring spent fuel pool water flow upward and across the spent fuel pool to the overflow weirs to maintain uniform spent fuel pool water conditions. Both sparger lines are equipped with siphon breakers to prevent siphoning of the spent fuel pool water and uncovering of the spent fuel in the event of a pipe break. The spent fuel pool circulating pumps can circulate the water through one or two filter demineralizers arranged in parallel and return it to the spent fuel pool through the diffusers associated with the SFC system. The cleanup section of the system can be isolated from the cooling section by means of two automatically operated Category I isolation valves in series.

The spent fuel pool cooling section of the system is classified as Safety Class 3, Category I in accordance with RG 1.26 and 1.29, respectively. The spent fuel pool cooling section is nuclear safety related and consists of redundant 100-percent capacity circulating pumps, 100-percent capacity heat exchangers and spent fuel pool surge tanks, complete with necessary piping, valves, and instrumentation. All equipment, piping, and valves are manufactured to the applicable codes, as specified in Table 3.2-2. All SFC system equipment is located in the reactor building.

The spent fuel pool cooling portion of the system is designed to remove decay heat loads for the following cases:

Case 1: A core shuffle following a refueling outage containing a batch size of 365 bundles discharged to the pool at a rate of 10 bundles per hr beginning 48 hr after reactor shutdown from 3,988 MWt power operation. The remainder of the pool is filled with 8 similar refueling discharges cooled for multiples of 24 months. The refueling outages begin in December to mid-April.

Case 2: Off-normal operation after an emergency full core offload discharged at a rate of 10 bundles per hr beginning 48 hr after reactor shutdown from 3,988 MWt power operation, discharged 180 days after the last refueling. The remainder of the pool is filled with 9 previous discharges of 365 bundles. The last discharge is cooled 180 days while the rest are cooled 180 days plus a multiple of 24 months.

Case 3: A full core offload following a refueling outage to the pool at a rate of 5 bundles per hr beginning 80 hr after reactor shutdown from 3,988 MWt power operation. The remainder of the pool is filled with 8 refueling discharges of 365 bundles cooled for multiples of 24 months. The refueling outages begin in December to mid-April.

The analysis and assumptions for the cases discussed above are cycle independent and can be found in reference 15. Decay heat loads will be established on a cycle specific basis in order to help manage the Spent Fuel Pool heat load. The cycle specific decay heat loads are determined in accordance with NRC Branch Technical Position (BTP) ASB 9-2 with uncertainty factor K set equal to 0.1 for long term cooling, or ORIGEN ARP and will be bounded by cases discussed above.

The spent fuel cooling system heat exchanger design data is provided in Table 9.1-6.

The heat load for the normal core shuffle (Case 1) is evaluated for the condition when the spent fuel communicates with the reactor cavity (refueling operations) and the condition when the spent fuel pool is isolated from the reactor cavity. When the spent fuel pool is isolated from the reactor cavity, one SFC train is normally operating and the other train is maintained available for service. The maximum spent fuel pool temperature will be maintained at or below 125°F*.

The heat load for the maximum emergency full core offload (Case 2) is also evaluated for conditions with the spent fuel pool isolated and/or open to the reactor cavity. When the full core has been moved to the spent fuel pool and the spent fuel pool is isolated from the reactor cavity, the maximum spent fuel pool temperature can be maintained below 150°F by the two SFC loops. For this off-normal case both loops of SFC are available, single active failure is not required to be considered by NUREG-0800. The RHR system (SFC assist mode) and ADH can also be used for SFC if needed. It is permissible to use the RHR system because of the availability of both RHR loops when there is no fuel in the vessel.

The heat load for the normal full core offload during refueling operations (Case 3) is evaluated for both conditions when the spent fuel pool is open to the reactor cavity and for the condition which would require the spent fuel pool to be isolated from the reactor cavity with the core offloaded to the spent fuel pool. The maximum spent fuel pool temperature will be maintained at or below 140°F.

For an unplanned event (emergency) which would require the spent fuel pool to be isolated from the reactor cavity, single failure criteria does not apply (similar to Case 2). For the case where the gates are put in for convenience, single failure criteria apply. This mode is limited to times when one loop of SFC can handle the entire heat load.

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* Following normal refueling outages of short duration, the fuel pool water will be maintained at lower temperatures, in accordance with station procedural controls, to ensure the design temperature of 150°F is not exceeded in the event of a station blackout (SBO).

SFC maintenance outages are normally performed during plant power operating conditions when the spent fuel pool is isolated from the reactor cavity. During these planned evolutions, an alternate method of decay heat removal will be available whenever the planned maintenance would render one loop of SFC unavailable for a period longer than the time required to heat up the spent fuel pool to 150°F, assuming the single remaining SFC loop is lost due to a single failure. The alternate method of decay heat removal may be a nonsafety-related system. The alternate method may be the primary cooling system, and the other available train of SFC can be the reserve.

The spent fuel pool cleanup portion of the system is classified as Safety Class 4, non-Category I. The cleanup system is designed to remove 6-micron and larger particulate matter while maintaining the conductivity of the water below 1.0 umho/cm at 25°C. The system consists of two 100-percent capacity filter demineralizers and necessary piping, valves, and instrumentation. All equipment, piping, and valves are manufactured to the applicable codes, as specified in Table 3.2-1. The spent fuel pool filter demineralizers are located in the reactor building. Cleanup requirements of the spent fuel pool water volume are accommodated by the use of one filter demineralizer. Cleanup requirements of the refueling water volume are accommodated by the use of one filter demineralizer.
Grab samples are obtained from the fuel pool cooling and cleanup system via the reactor sample system, as listed in Table 9.3-1.

Typically, pH, gross gamma, and conductivity analyses are performed on a weekly basis, and chloride analysis is performed when conductivity is above 1 umho/cm at 25°C. During periods when spent fuel pool work is not performed, frequency of analysis will vary according to continuous conductivity monitor indications. Generally, filterable solids and metal analysis will be performed monthly when materials are introduced which could cause significant changes to these parameters. The above grab samples will be performed in accordance with appropriate procedures to ensure that the water quality parameters are within design limits as shown in Table 9.1-5. These limits are based on GE recommendations to ensure maintenance of a proper storage environment for the spent fuel. A radiation monitor located in the common inlet to the filter demineralizers alerts plant personnel to the presence of abnormal radioactivity levels in the fuel pool water.

The filter demineralizers are normally backwashed manually. The backwash of the filter demineralizers is based upon an analysis of filter demineralizer effluent and influent water quality, or high differential pressure across the filter or filter demineralizer radiation levels or outlet level strainer differential pressure. The analysis is used to establish when filter media is chemically exhausted. Spent demineralizer resins are backwashed to the LWS system for disposal.

Reactor water cleanup (RWCU) requirements during refueling are augmented by the use of the SFC system filter demineralizer units. Additional cleanup capabilities are included for use during refueling operations by providing connections from the reactor internals storage pit and the bottom of the reactor head cavity to the suction line of the spent fuel pool cleanup circulating pumps. During refueling operations, a portion of the SFC system return flow may be diverted to the refueling volume spargers located in the reactor head cavity.

Normally, makeup water for the spent fuel pool is automatically provided to the spent fuel pool surge tanks from the condensate storage tank (CST) via the condensate makeup and drawoff system. Emergency makeup water to the spent fuel pool is available from the Category I portion of the SWP system.

Provisions are made to supply service water in lieu of RBCLCW to the spent fuel pool heat exchangers for long-term cooling considerations if the RBCLCW is unavailable.

The SFC system serves as a backup to the nonsafety-related ADH removal system described in Section 9.1.6. The ADH removal system is a nonsafety-related system used to perform shutdown cooling functions during outages in lieu of the RHR system.

9.1.3.3 Safety Evaluation

As previously described, sufficient redundancy and design flexibility are provided in the SFC system to safely accommodate faulted conditions. Safety requirements of the system are long term in nature and, as such, the SFC circulating pumps do not start automatically from the diesel generators on loss of offsite power (LOOP). Power to the pumps is supplied manually from the diesel generator when required.

Refueling Outages

When the spent fuel pool is isolated from the reactor cavity, each loop of the SFC system is capable of providing cooling for the Case 1 heat load of approximately 36 10^6 btu/hr described in Section 9.1.3.2. Cycle specific analyses verify that should spent fuel cooling be lost due to a single failure in one loop, the other cooling loop can be placed into service within 1 hr. During the 1 hr, the pool temperature will not increase to exceed 140°F. Should spent fuel pool cooling be lost, it would take more than 3 hr for the temperature of the spent fuel pool to approach its maximum design temperature of 150°F.

For the full-core emergency offload condition corresponding to approximately 59 million btu/hr of heat load, with the spent fuel pool isolated from the reactor cavity (Case 2), described in Section 9.1.3.2, two trains of SFC will maintain the spent fuel pool temperature at approximately 143°F, when the cooling water

is at 84°F (maximum service water temperature). Cycle specific analyses verify that should one loop of SFC be lost under this condition, it would take more than 2 hr for the temperature of the spent fuel pool to approach its maximum design temperature of 150°F. Two hr is a sufficient amount of time for SFC to be restored under this condition. All that is required to restore spent fuel pool cooling in this situation is to start one loop of the RHR system (SFC assist mode) or one loop of ADH, whichever is available.

For the full-core offload condition, with the spent fuel pool isolated from the reactor cavity (Case 3), described in Section 9.1.3.2, one train of SFC will maintain the spent fuel pool temperature at or below 140°F. Cycle specific analyses verify that should the SFC be lost under this condition, it would take 1 hr for the temperature of the spent fuel pool to approach its maximum design temperature of 150°F. The other cooling loop can be placed into service within 1 hr.

The above evaluations (Cases 1, 2 and 3) credit the service water directly removing the decay heat from the SFC heat exchangers. The maximum expected service water temperature coincident with the maximum heat loads (Cases 1 and 2) is approximately 50°F based on the historical record for the December through April months. The heat removal by the SFC heat exchanger is based on the maximum service water temperature of 52°F.

The SFC system, in combination with natural circulation, is capable of providing an alternate method of decay heat removal and circulation of reactor coolant when the plant is in the refueling mode. This requires a portion of the SFC return flow to be directed to the reactor cavity spargers and a portion of the fuel to remain in the vessel. Other restrictions apply to this use of SFC, such as time after shutdown, cooling water temperature available to the heat exchangers, and reactor power level at time of shutdown.

During refueling outages when the spent fuel pool is open to the reactor cavity, spent fuel pool cooling is performed in conjunction with reactor cavity cooling. Operation in this mode is administratively controlled. Cavity cooling is not designed for single active failure. SFC is required to meet single failure requirements. For the worst-case heat load conditions during refueling, either one RHR shutdown cooling loop in conjunction with one SFC loop, or one ADH loop in conjunction with one SFC loop, would maintain the spent fuel pool temperature below 125°F and the average reactor coolant exit temperature below 150°F. The other SFC would be maintained available. In the event the ADH removal system is lost and RHR is out of service for maintenance, the two trains of the SFC system, with 64°F cooling water to the SFC heat exchangers, are capable of maintaining the average reactor coolant exit temperature at or below 150°F, and the average cavity temperature at or below 140°F. If one operating loop of either SFC or the shutdown

cooling loop (i.e., ADH or RHR) is lost, it would take about 4 hr before the reactor cavity and spent fuel pool temperature approaches 140°F, or the average reactor coolant exit temperature approaches 150°F. Under normal conditions, 2 hr is sufficient time to place the idle available system into service.

Since the SWP system is always available and provisions are included to use service water as makeup water for the spent fuel pool, adequate water coverage of spent fuel in the spent fuel pool is assured at all times. Design features of the SFC system also include siphon breakers in the SFC return lines to preclude the potential for pool drainage from a broken line.

Normal Plant Operation

During the normal plant operation, one of the SFC loop pipes is postulated to be damaged due to a break in an adjacent high-energy RWCU pipe with one additional single active failure. This situation would require the use of one of the crossties between the two redundant SFC pumps and may require valve realignments. The remotely-actuated cross-connect valves and the pumps are located in a room not affected by the postulated high-energy line break (HELB). Valves which could require manual/local isolation action depending on the leak location could be reached after 1 hr even if located in the secondary containment break area. Plant operating procedures describe how to utilize the cross-connect capability of the SFC system. These crossties permit each SFC pump to take suction from either skimmer surge tank and to discharge back to the fuel pool through either heat exchanger. Figure 9.1-5 shows the SFC system and the cross-connect capability. During this event, it will take longer than 3 hr for the spent fuel pool temperature to approach its design temperature of 150°F. Three hr is sufficient time to restore SFC cooling for the above-described scenario.

Due to the time available for required Operator actions following a faulted (i.e., line break) condition and the redundancy of the SFC system, SFC is assured for any single active or passive failure. Therefore, the integrity of the spent fuel pool can be maintained.

Originally, the failure modes and effects analysis (FMEA) of the SFC system was contained in the Unit 2 FMEA document, which is historical. FMEAs for plant systems are now performed and controlled by the design process.

9.1.3.4 Inspection and Testing

No special equipment tests are required because at least one pump of each system, one heat exchanger, and one filter/demineralizer are normally in continuous operation while fuel is stored in the pool. Operating and standby components are alternated periodically to verify operability of all equipment. Routine visual inspection of the system components and instrumentation is

adequate to verify system operability. The spent fuel storage pool level alarms are periodically tested. Isolation valves are also tested on a periodic basis, when the respective system is not operating, to ensure their operability.

9.1.3.5 Instrumentation Requirements

Description

Redundant safety-related instruments and controls are provided for automatic and manual control of the SFC system. The controls and monitors described in the following paragraphs are located in the main control room. The control logic is shown on Figure 9.1-6.

Operation

The SFC system valves are opened and closed manually. Interlocks prevent the filter subsystem header inlet and outlet valves from opening and automatically trip closed when there is sustained SFC system circulation pump operation with low suction pressure.

The SFC system water circulation pumps are started and stopped manually. Interlocks prevent the starting of a pump on a diesel generator if an emergency load sequencing signal has been initiated, and prevent pump operation on low suction pressure or sustained low pump discharge pressure or flow.

The SFC system surge tank makeup valves open automatically when the surge tank water level is low and close automatically on high water level. The valves can be opened, unless tank water level is high, and closed manually.

When required, water may be discharged from the SFC system to the condenser and radwaste systems by the spent fuel pool cooling water dump valve, which is controlled manually.

Monitoring

Indicators are provided for:

1. SFC system surge tank water level.

2. SFC system outlet water temperature.

3. SFC system water heat exchanger outlet temperature.

4. Spent fuel pool water level.

5. SFC system water circulation pumps suction and discharge pressures.

6. SFC system division total flow.

7. SFC system discharge total flow.

8. SFC system flow to condenser/radwaste systems.

1. Spent fuel pool level high/low.

2. Spent fuel pool level low.

3. Spent fuel pool leakage level high.

4. Spent fuel surge tank outlet temperature high.

5. Spent fuel surge tank level high/low.

6. Spent fuel surge tank level high/high.

7. Spent fuel circulation pumps discharge flow low.

8. Spent fuel circulation pump suction pressure low.

9. Spent fuel circulation pump discharge pressure low.

10. Spent fuel circulation pump auto trip.

11. Spent fuel pool cleanup trouble.

12. Fuel pool cooling systems inoperable.

13. Spent fuel pool water temperature high.

14. Spent fuel pool cooling systems flow low.

15. Spent fuel heat exchangers outlet temperature high.

16. Spent fuel cask-handling system trouble.

17. Reactor plant sample system trouble.

18. Process liquid radiation monitor activated.

19. SF WTR circulating pump motor overload.

9.1.4 Fuel Handling System

9.1.4.1 Design Bases

The fuel handling system is designed to provide a safe, effective means for transporting and handling fuel from the time it reaches the plant until it leaves the plant after post-irradiation cooling. Safe handling of fuel includes design considerations for maintaining occupational radiation exposures as low as reasonably achievable (ALARA) during transportation and handling.

Design criteria for major fuel handling system equipment are provided in Tables 9.1-1 through 9.1-3, which list the safety class, quality group, seismic category, and QA category. Additional design criteria are discussed below and expanded in Section 9.1.4.2. Where applicable, the appropriate ASME, ANSI, Industrial, and Electrical Codes are identified.

9.1.4.1.1 Fuel Handling Equipment

The transfer of new fuel assemblies between the refueling floor uncrating area and the new fuel inspection stand and/or the new fuel storage vault is accomplished using the RBPC equipped with a general purpose grapple.

The RBPC auxiliary hoist is used with a general purpose grapple to transfer new fuel from the new fuel vault to the fuel storage pool. From this point, the fuel is handled by the telescoping grapple on the refueling platform.

The refueling platform (Figure 9.1-7) is constructed in accordance with the applicable provisions of 10CFR50 Appendixes A and B and categorized Category II. Allowable stress due to SSE loading is 100 percent of yield or 70 percent of ultimate, whichever is less. A dynamic analysis is performed on the refueling platform structures using the response spectrum method with load contributions resulting from each of three orthogonal directions being combined by the root mean square (RMS) procedure.

Working loads of the refueling platform structures are in accordance with the AISC Manual of Steel Construction. All parts of the hoist systems are designed to have a minimum safety factor of 5 based on the ultimate strength of the material. A redundant load path is incorporated in the fuel hoists so that no single component failure could result in a fuel bundle drop. Maximum deflection limitations are imposed on the main structures to maintain relative stiffness of the platform. Welding of the platforms is in accordance with AWS D14.1 or ASME Boiler and Pressure Vessel Code Section IX. Gears and bearings meet AGMA Gear Classification Manual and ANSI B3.5 requirements. Materials used in construction of load bearing members meet ASTM specifications. For personnel safety, OSHA Part 1910.179 is applied. Electrical equipment and controls meet ANSI CI, National Electric Code, and NEMA Publication No. IC1, MG1 specifications.

The fuel grapple (Figure 9.1-8) is used for lifting and transporting fuel bundles. It is designed as a telescoping grapple that can extend to the proper work level and in its fully retracted state still maintains adequate water shielding over the fuel. The main telescoping fuel grapple has redundant hooks and an indicator that confirms positive grapple engagement.

In addition to redundant electrical interlocks to preclude the possibility of raising radioactive material out of the water, the cables on the auxiliary hoists on the refueling platform incorporate an adjustable, removal stop that prevents hoisting when the free end of the cable is at a preset distance below water level.

Provision of a separate cask loading pool, isolated from the fuel storage pool, eliminates the potential consequences of dropping the cask and rupturing the fuel storage pool. Furthermore, the positive interlocks on the RBPC prevent the transfer path of the cask from ever being over the spent fuel pool. Refer to Chapter 15 for accident considerations.

9.1.4.1.2 Reactor Building Polar Crane

The RBPC is designed in accordance with the following criteria:

1. Category I requirements.

2. Applicable design requirements of AWS D1.1, NEMA MG1, NFPA 70, and CMAA Specification No. 70 as a Service Class A1 crane.

3. Requirements of ANSI B30.2.0.

9.1.4.2 System Description

Table 9.1-4 is a listing of typical tools and servicing equipment supplied with the nuclear system. The following sections describe the use of some of the major tools and servicing equipment and address safety aspects of the design where applicable.

9.1.4.2.1 Spent Fuel Shipping Cask

The design of the spent fuel shipping cask and handling facilities is based on a cask that is approximately 8 ft in diameter and 21 ft in length, with a maximum critical loaded weight of approximately 120 tons. These large casks are shipped to and from the site on specially-designed rail cars. Smaller casks are shipped on flatbed trucks equipped with suitable cask hold-down devices and heat exchangers as required. The design of the casks and transport vehicles conforms to the rules and regulations for packaging and transportation of radioactive materials (10CFR71 and 49CFR171-178).

9.1.4.2.2 Reactor Building Polar Crane

The RBPC is designed as a Service Class A1 crane in accordance with CMAA Specification No. 70. The design features described in this section are consistent with the guidelines of NUREG-0554.

The RBPC consists of two girders and a trolley. The crane is of welded-box girder construction with the girders connected by end ties at the left and right ends of the girders. This bridge assembly rests on circular runway rails (el 387 ft 4 in) which provide for 360-deg rotation of the crane girders. The runway rails are mounted on a circular support structure mounted on the secondary containment wall. The trolley travels laterally on its rails mounted on the crane girders and serves as the platform for the main (132-ton) and auxiliary (25-ton and 1/2-ton) hoisting mechanisms. The main hoist has a single-failure proof design based on a maximum critical load of 132 tons, so that the single failure of any component in the hoist train does not result in loss of the lifted load. This also applies to the electrical components in the main hoist path. The crane can be controlled from the bridge-mounted cab or by using a radio transmitter.

The RBPC is used to move the fuel transfer shielding bridge, all shield plugs, the drywell head, the reactor vessel head, the steam dryer, the steam separator, and the reactor vessel head insulation and support frame. It will also be used to handle the spent fuel shipping or storage casks. During construction, the RBPC is used for the lifting and placement of major machinery and equipment. Crane design features have been specified for construction service; however, the RBPC will not be used to lift loads greater than the maximum critical load. The auxiliary hoist of the crane will not handle individual spent fuel assemblies.

All materials for structural members essential to structural integrity, as well as the main hook, have been impact tested in accordance with the requirements of ASTM A370 for Charpy V-notch tests.

The crane has been seismically designed to withstand the combined effects of the maximum critical load and the SSE loads without loss of load or structural integrity. The bridge and trolley remain on their rails and no other part of the crane will be dislodged and fall during the SSE.

Welding, welding procedures, and welders used for the crane construction have been qualified to the requirements of AWS D1.1 except that undercutting is in accordance with AWS D14.1.

Two independent reeving systems are provided for the main hoist. With all parts intact, the rope static and dynamic design factors for the maximum critical load are 12.3 to 1 and 10.9 to 1, respectively. In the event of a failure of one of the reeving systems, this design is adequate to effect a load transfer and load support by the remaining reeving system. All fleet angles within the main hoist reeving are within 3.5 deg. The main load block assembly includes the main hook and a backup eyebolt for holding the maximum critical load. This assembly is designed so that stresses in each main hoist load holding assembly are less than one-third yield under static and dynamic loading conditions. The main hook and eyebolt have undergone a static load test of

264 tons (200 percent of design rated loads [reference NUREG-0554], 132 tons). Each hoist will be initially load tested in the field to 1.25 times its rated capacity. Load bearing shafts and gears have been designed to ensure proper alignment of components. Redundant gear trains are provided for the main hoist. Main hoist full rated load speed is 7.5 fpm and the rope line speed is 52 fpm at the drum.

The Main Hoist, 25-ton auxiliary hoist, ½-ton auxiliary hoist, bridge and trolley motors are inverter duty AC motors that are VFD and PLC controlled. The control system includes protection that prevents the main hoist and 25-ton auxiliary hoist brakes from being released until the motor field is energized and armature current is flowing. Load control for the main hoist and 25-ton auxiliary hoist are provided by the VFDs. Each hoist has an emergency dynamic lowering feature that automatically lowers the load at a safe rate in the event of simultaneous loss of ac power and holding brakes. Both the main and auxiliary hoists have two independent shoe-type brakes that are automatically applied to the motor shaft when the motor is de-energized. Each hoist holding brake has a rated braking torque capacity of at least 125 percent of the developed torque at the point of brake application. The bridge movement is equipped with an electric released shoe-type braking system. The brake is applied automatically upon loss of power or violation of travel limits.

The trolley traverse motor is equipped with an electric brake that is applied automatically to the motor shaft upon loss of power, the opening of the main line contactor, violation of travel limits, or the return of the trolley controllers to the off position. The bridge and trolley VFDs provide "inch" speed control for precision positioning. Buffers are provided at the end of the trolley rails to physically prevent overtravel in the event of limit switch failure.

The following additional control features prevent uncontrolled motion or loss of load in the event of a component malfunction or inadvertent Operator action. Each hoist has redundant limit devices that stop the hook at its maximum safe travel to prevent "two blocking." Interlocks on the main hoist prevent the movement of heavy loads over the spent fuel pool. Positive Operator action to bypass travel restriction interlocks is required to operate the main hoist in the cask loading area. Additional restrictions placed on bypassing the travel interlocks are discussed in Section 9.1.2.2. Hoist limit switches apply the hoist holding brakes in the event of an overspeed condition. Hoist load limiting devices prevent lifting in the event of an overload condition. Radio controls are such that loss of radio signal causes a fail-safe, no-motion condition. All crane motions or the motion involved stop when the operating range of the transmitter is exceeded. The Radio has switches that can stop all motions. Design features permit lowering the load manually upon loss of power.

Specific requirements regarding heavy load handling over fuel assemblies in the spent fuel storage pool are described in Technical Requirements Manual (TRM) Section 3.9.5.

9.1.4.2.3 Fuel Servicing Equipment

The fuel servicing equipment described in the following sections has been designed in accordance with the criteria listed in Table 9.1-1.

Fuel Preparation Machine

The fuel preparation machine is mounted on the wall of the spent fuel storage pool and is used for removing and installing reusable channels or fuel bundles. The machine is also used with the fuel inspection fixture to provide an underwater inspection capability, and with the defective storage container to contain a defective fuel assembly for stripping the channel.

The fuel preparation machine consists of a work platform, a frame, and a movable carriage. The frame and movable carriage are located below the normal water level in the fuel storage pool, providing a water shield for the irradiated fuel assemblies being handled. Fuel preparation machine 2FNR*TL1B carriage has an installed up-travel-stop to prevent raising irradiated fuel above the safe water shield level. The movable carriage is operated by a foot pedal-controlled air hoist.

Fuel preparation machine 2FNR*TL1A is used to assist in new/nonirradiated fuel transfer operations with its up-travel-stops removed. This is to prevent submerging the polar crane's hook into the spent fuel pool water. When this machine is placed in service, administrative controls will be in place to limit its use to new/non-irradiated fuel only.

New Fuel Inspection Stand

The new fuel inspection stand (Figure 9.1-9) serves as a support for the new fuel bundles undergoing receiving inspection and provides a working platform for technicians engaged in performing the inspection. The new fuel inspection stand consists of a vertical guide column, a lift unit to position the work platform at any desired level, bearing seats, and upper clamps to hold up to two fuel bundles in position.

Channel Bolt Wrench

The channel bolt wrench (Figure 9.1-10) is a manually-operated device approximately 12 ft in overall length. The wrench is used for removing and installing the channel fastener assembly while the fuel assembly is held in the fuel preparation machine. The channel bolt wrench has a socket that mates and captures the channel fastener capscrew.

Channel Handling Tool

The channel handling tool (Figure 9.1-11) is used in conjunction with the fuel preparation machine to remove, install, and transport fuel channels in the fuel storage pool. The tool is composed of a handling bail, a lock/release knob, an extension shaft, angle guides, and clamp arms that engage the fuel channel. The clamps are actuated (extended or retracted) by a manually rotating lock/release knob. The channel handling tool is suspended by its bail from a spring balancer on the channel handling boom located on the fuel pool periphery.

Fuel Pool Sipper

The fuel pool sipper, which is used to detect leaking fuel bundles, provides a means of isolating a fuel assembly in demineralized water to concentrate fission products in relation to a controlled background. The fuel pool sipper consists of a control panel assembly and a sipping container.

Refueling Mast Sipper

This equipment may be used by a sipping vendor and is not stored at the site when not in use. The refueling mast sipper is used to detect leaking fuel while the bundle is being transferred from the reactor vessel to the spent fuel pool, or while the fuel bundle is being raised and then lowered in the reactor vessel or spent fuel pool. This system allows fission product gases to concentrate in a collection chamber. The fission product gases are counted by radiation detection equipment which will show a significant rise in count rate when a leaking fuel bundle is being sipped. The refueling mast sipper consists of a suction tube assembly which attaches to the refueling mast, a sipping skid, a beta radiation detector and associated electronics and computer for interpreting and presenting the data collected. The refueling mast sipper also provides connections for obtaining gas and water samples for independent analysis.

Channel Gauging Fixture

The channel gauging fixture (Figure 9.1-12) is a go/no-go gauge recommended for evaluating the condition of an irradiated fuel channel only if difficulty is encountered in channeling with irradiated channels. This fixture tests channels for outside and inside dimensions and squareness on the ends only. The fixture consists of a frame, a gauging plate, and a gauging block. The gauging plate is shimmed to correspond to the outside dimension of a reusable fuel channel. The gauging block conforms to the inside dimension of the lower end of a usable fuel channel. The fixture is installed in the vertical position between the two fuel preparation machines and hangs from the fuel storage pool curb.

General Purpose Grapple

The general purpose grapple (Figure 9.1-13) is a fuel handling tool. The grapple can be attached to the reactor building auxiliary hoist, the jib crane, and the auxiliary hoists on the refueling platforms. The grapple is used to remove new fuel from the vault, place it in the inspection stand, and transfer it to the fuel pool. It can be used to handle new fuel during channeling.

Fuel Inspection Fixture

The fuel inspection fixture is used in conjunction with the fuel preparation machine to permit remote inspection of fuel elements. The fixture consists of two parts: 1) a lower bearing assembly, and 2) a guide assembly at the upper end of the carriage. The fuel inspection fixture permits the rotation of the fuel assembly in the carriage, and in conjunction with the vertical movement of the carriage provides complete access for inspection.

Jib Cranes

Two 1,000-lb capacity fuel-handling jib cranes with hoist and trolley are used over the open reactor vessel and around the fuel pool for handling fuel and reactor vessel servicing tools. One 200-lb capacity channel handling jib crane is used to lift channels from spent fuel assemblies and lower channels onto new fuel assemblies.

9.1.4.2.4 Servicing Aids

General area underwater (Figure 9.1-14) lights are provided with suitable reflectors for illumination. Suitable light support brackets are furnished to support the lights in the reactor vessel to allow the light to be positioned over the area being serviced independent of the platform. Local area underwater lights (Figure 9.1-15) are small diameter lights for additional illumination. Drop lights are used for illumination where needed.

A radiation-hardened, portable, underwater, closed-circuit television camera is provided, which can be lowered into the reactor vessel and/or fuel storage pool to assist in the inspection and/or maintenance of these areas.

A general purpose, plastic viewing aid (Figure 9.1-16) is provided to float on the water surface to provide better visibility. The sides of the viewing aid are brightly colored to allow the Operator to observe it in the event that it fills with water and sinks.

A portable, submersible, underwater vacuum cleaner is provided to assist in removing crud and miscellaneous particulate matter from the pool floors or the reactor vessel. The pump and filter unit are completely submersible for extended periods. The filter

package can be remotely changed, and the filters will fit into a standard shipping container for offsite burial.

Fuel pool tool accessories are also provided to meet servicing requirements. A fuel sampler is provided to detect defective fuel assemblies during open vessel periods while the fuel is in the core. The fuel sampler head isolates individual fuel assemblies by sealing the top of the fuel channel. Water is then pumped from the bottom of the fuel assembly, through the fuel channel, to a sampling station, and returned to the primary coolant system. After a soaking period, a water sample is obtained and radiochemically analyzed to determine possible fuel bundle leakage.

As an alternative to using the fuel sampler, the fuel bundles may be sipped using the refueling mast sipper during transfer from the reactor vessel to the spent fuel pool or by raising and then lowering the bundle while in the reactor vessel or spent fuel pool. Water is recirculated from the top of the fuel bundle to the sipping system and back to the spent fuel pool or reactor, depending on where the sipping is being performed. The fission product gases are separated from the water and are collected for sampling and analysis. A leaking fuel bundle will be identified by the elevated count rates recorded by the radiation detection equipment. The fuel bundles are not raised above limits established by the Technical Specifications during this sipping method.

9.1.4.2.5 Reactor Vessel Servicing Equipment

The safety classifications, quality group, and seismic category and QA category for this equipment are listed in Table 9.1-2. The following sections describe the equipment designs in reference to that table.

Reactor Vessel Service Tools

These tools are used when the reactor is shut down and the reactor vessel head is being removed or reinstalled. Tools in this group are:

1. Stud handling tool.
2. Stud wrench.
3. Nut runner.
4. Stud thread protector.
5. Thread protector mandrel.
6. Bushing wrench.
7. Seal surface protector.

8. Stud elongation measuring rod.

9. Dial indicator elongation measuring device.

10. Head guide cap.

These tools are designed for a 40-yr life in the specified environment. Lifting tools are designed for a safety factor of 5 or better with respect to the ultimate strength of the material used. When carbon steel is used, it is either hard chrome plated, parkerized, or coated with an acceptable paint.

Steam Line Plugs

The steam line plugs are used during reactor refueling or servicing. They are inserted in the steam outlet nozzles from inside the reactor vessel to prevent a flow of water from the reactor well into the main steam lines during servicing of safety relief valves (SRVs), main steam isolation valves (MSIVs), or other components of the main steam lines, while the reactor water level is maintained at the refueling level. The steam line plug design provides two seals of different types. Each one is independently capable of holding full head pressure. The equipment is constructed of noncorrosive materials and all calculated safety factors are 5 or greater. The plug body is designed in accordance with the Aluminum Construction Manual by the Aluminum Association.

Shroud Head Bolt Wrench

This is a hand-held tool for operation of the shroud head bolts. Designed for a 40-yr life, it is made of aluminum to be easy to handle and to resist corrosion. Testing has been performed to confirm the design. Shroud head bolts may also be operated utilizing the service pole caddy system from the refueling platform.

Head Holding Pedestal

Three pedestals are provided for mounting on the refueling floor for supporting the reactor vessel head (Figure 9.1-17). The pedestals have studs which engage three evenly-spaced stud holes in the head flange. The flange surface rests on replaceable aluminum wear pads. When resting on the pedestals, the head flange is approximately 3 ft above the floor to allow access to the seal surface for inspection and O-ring replacement.

The pedestal structure is a carbon steel weldment coated with an approved paint. It has a base with bolt holes for mounting it to the concrete floor. The structure is designed in accordance with the AISC Manual of Steel Construction.

Head Nut and Washer Rack

The RPV head nut and washer rack (Figure 9.1-18) is used for transporting and storing up to eight nuts and washers. The rack is a box-shaped aluminum structure with dividers to provide individual compartments for each nut and washer. Each corner has a lug and shackle for attaching a 4-leg lifting sling. The rack is designed in accordance with the Aluminum Construction Manual, and for a safety factor of 5.

Head Strongback Carousel

The head strongback carousel and stud tensioners are used to tighten and loosen the vessel closure studs. The use of the stud tensioners ensures a balanced loading on the reactor vessel flange without damage to the studs or nuts and without distortion of the O-rings.

The head carousel and the stud tensioner assembly consist of four hydraulically-powered stud tensioning mechanisms mounted to a circular carousel, a dolly-mounted power and control unit, a frame that is supported by the reactor vessel head and from which the stud tensioner mechanisms are suspended, and the required piping, gauges and connectors.

The stud tensioners are lowered over four studs and nuts, and engage the upper stud threads. Air pressure is applied to the pump, and hydraulic pressure elongates the studs. A T-handle mounted on the side of the tensioners is then turned clockwise to tighten the nuts. Proper preloading is assured by monitoring the pump pressure. Hydraulic pressure is then released and the stud tensioners are moved to the next operation.

Head Stud Rack

The head stud rack is used for transporting and storing eight RPV studs. It is suspended from the RBPC when studs are lifted from the reactor well to the operating floor. The rack is made of aluminum to resist corrosion.

Lifting Strongback

The lifting strongback (Figure 9.1-19) is used for lifting the pressure vessel head, the drywell head, and all other critical loads during refueling. The strongback conforms to the lifting configuration required for each lift. This strongback has dual-load attaching points providing a redundant lifting design.

The strongback is designed to support three times the rated load (static and dynamic) in accordance with NUREG-0554. All welding is in accordance with the ASME Boiler and Pressure Vessel Code Section IX and AWS D1.1. The completed assembly is proof tested at 125 percent rated load. After the load test, all structural welds are magnetic particle inspected.

The head strongback carousel (Figure 9.1-27) is used for suspending the stud tensioners and for lifting the pressure vessel head. The head strongback carousel has dual-load attaching points providing a redundant lifting design.

The strongback is designed to support three times the rated load (static and dynamic) in accordance with NUREG-0554. All welding is in accordance with the ASME Boiler and Pressure Vessel Code Section IX. The completed assembly is proof tested at 125 percent rated load. After the load test, all structural welds are magnetic particle inspected.

Lifting Slings and Strongbacks

Three types of lifting devices will be used to lift and transport critical loads during refueling:

1. Solid bar slings in conjunction with the primary and redundant strongbacks (Figure 9.1-19).

2. Wire rope slings attached directly to the polar crane main hook (Figure 9.1-19a).

3. Kevlar slings attached directly to the polar crane main hook (Figure 9.1-19a).

Each pair of slings and strongbacks are designed to withstand three times the rated load without any permanent deformation. The slings are designed to withstand the rated load without exceeding 20 percent of the material tensile stress allowables.

A complete set of four slings, either solid bar, wire rope, or Kevlar, will be used when lifting and transporting the shield plugs, drywell head, reactor vessel head insulation frame, RPV head, fuel transfer shielding bridge, SFC and reactor water cleanup (WCS) filter removal plugs, storage pool gate, service platform, and other critical loads identified in Appendix 9C, Figure 5-2. For the fuel pool gates (outer and inner), two slings can be used as an alternate.

A separator sling set and lifting rig consisting of four wire rope or solid bar slings and two spreader beams will be used when lifting the dryer and separator (Figures 9.1-20 and 9.1-20a). The dryer/separator sling assembly (Figure 9.1-20a) which utilizes Kevlar slings and a strongback may also be used to handle the dryer and separator. These slings (i.e., wire rope, solid bar, or Kevlar) and spreader beams/strongback are designed for underwater service. The spreader beams/strongbacks are designed to the same criteria as the slings.

Each completed sling assembly (solid bar, wire rope or Kevlar) and the spreader beam/strongback are proof tested at 150 percent of the rated load. All welds after load testing are subjected to nondestructive testing (NDT) inspection.

During use, slings and lifting rigs shall be maintained in accordance with ANSI 14.6-1978.

Service Platform

The service platform (Figure 9.1-21) is provided to facilitate maintenance work on reactor internals. It provides a working platform for people and hand-guided tools, and it also can support a jib crane. The service platform is supported by four wheels that run on a circular track resting on the vessel flange and confined by the vessel closure studs.

The physical size of the device is such that it cannot enter the RPV. The structure design is in accordance with the AISC Manual of Steel Construction. Materials are in accordance with ASTM Standards. Welding is in accordance with ASME Boiler and Pressure Vessel Code Section IX or AWS D1.1 structural welding. The electrical system is in accordance with ANSI/ANS C1, National Electrical Code, and NEMA Publications No. IC1 and MG1. Painting and surface preparation are in conformance with SSPC and RG 1.54.

Service Platform Support

The service platform support serves both as a sealing surface protector for the reactor vessel flange and as a track for the service platform. It has continuous vertical support on the vessel flange, and horizontally it is confined by the vessel studs by strapping to the outer edge of the flange. The service platform support is made from aluminum and all welding is done in accordance with AWS Code D1.0.

Service Pole Caddy System

The service pole caddy system is an auxiliary platform attached to the refueling platform. It is used to carry service poles to desired locations over the reactor cavity to perform underwater servicing activities on reactor equipment, such as shroud head bolts removal and installation, steam line plugs installation and removal, and underwater camera manipulation.

Steam Line Plug Installation Tool

The steam line plugs are designed to be installed using the installation tool under water from the refuel bridge. The installation tool is used for transporting and installing the steam line plugs in the steam line nozzles of the reactor vessel. This tool is made of aluminum; it is designed for a safety factor of 5, and is in accordance with Aluminum Construction Manual.

9.1.4.2.6 In-Vessel Servicing Equipment

The instrument strongback attached to the RBPC auxiliary hoist is used for servicing the local power range monitor (LPRM), source

range monitor (SRM), and intermediate range monitor (IRM) dry tubes should they require replacement. The strongback initially supports the dry tube into the vessel. The in-core dry tube is then decoupled from the strongback and is guided into place while being supported by the instrument handling tool (Figure 9.1-22).

Final in-core insertion is accomplished from below the reactor vessel. The instrument handling tool is attached to the refueling platform auxiliary hoist and is used for removing and installing LPRM fixed in-core dry tubes as well as handling the SRM and IRM dry tubes.

Each in-core instrumentation guide tube is sealed by an O-ring on the flange. In case the seal needs replacing, an in-core guide tube sealing tool is provided. The tool is inserted into an empty guide tube and sits on the beveled guide tube entry in the vessel. When the drain on the water seal cap is opened, hydrostatic pressure seats the tool. The flange can then be removed for seal replacement.

The auxiliary hoist on the refueling platform is used with appropriate grapples to handle control rods, flux monitor dry tubes, sources, and other internals of the reactor. Interlocks on both the grapple hoists and auxiliary hoist are provided for safety purposes; the refueling interlocks are described and evaluated in Section 7.7.1.13.

9.1.4.2.7 Refueling Equipment

Fuel movement and reactor servicing operations are performed from a platform that spans the refueling, servicing, and storage cavities.

Refueling Platform

The refueling platform (Figure 9.1-7) is a gantry crane that is used to transport fuel and reactor components to and from pool storage and the reactor vessel. The platform spans the fuel storage and vessel pools on rails embedded in the refueling floor. A telescoping mast and grapple suspended from a trolley system are used to lift and orient fuel bundles for core, storage rack, storage cask, or shipping cask placement. Control of the platform is from an Operator station on the main trolley with a position-indicating system provided to position the grapple over core locations. The platform control system includes interlocks to verify hook engagement and grapple load, prevent unsafe operation over the vessel during control rod movements, and limit vertical travel of the grapple. Two 1,000-lb capacity auxiliary hoists, one main trolley mounted and one auxiliary trolley mounted, are provided for in-core servicing such as LPRM replacement, fuel support piece replacement, jet pump servicing, and control rod replacement. The grapple in its fully retracted position provides approximately 7 ft 7 1/2 in minimum water shielding over the active fuel during transit. The refueling

platform has an auxiliary platform installed to accommodate the service pole caddy system.

Fuel Transfer Shielding Bridge

The portable radiation shield (Figure 9.1-23) is a temporary shielding device that is installed prior to the transfer of the spent fuel bundles from the reactor to the fuel storage pool. The fuel bundles are passed through the shield to reduce radiation levels in the upper drywell area. The shield is handled by the RBPC. When installed, one end of the shield is supported by the reactor vessel flange and the other end is supported by the refueling floor at el 329 ft 7 in. Following its use the shield is decontaminated and stored in the reactor building.

Auxiliary Service Platform

The auxiliary service platform (Figure 9.1-24) is used as a work area by personnel performing refuel outage-related activities. The auxiliary service platform’s framed bridge rides on the refueling platform rails in the reactor building.

Reactor Cavity Dewatering

The dewatering of the reactor cavity, by using the reactor internals storage pool gate, may be performed during a plant outage for the purpose of accessing the reactor cavity and vessel.

A. The following steps dewater the reactor cavity (with the spent fuel pool gates installed). Basic sequence may be altered to accommodate underwater transfer of the steam dryer.

    1. Remove steam dryer and place in the storage pit with the sprinklers on.

    2. Flood up the reactor refueling cavity.

    3. Remove the moisture separator while ensuring the separator is continuously submerged (Section 1.12, Item 40, "Generic Licensing Issues").

    4. After the moisture separator and the steam dryer are in the storage pit, the reactor internals storage pool gate can be installed.

    5. Dewater the reactor cavity by opening the reactor cavity drains.

    The following steps refill the reactor cavity and return reactor equipment (with the spent fuel pool gates installed).

1. With reactor cavity drains closed, open reactor cavity fill system.

2. On completion of reactor cavity filling, remove reactor internals storage pool gate.

3. Return moisture separator to the reactor vessel.

4. Lower water level in the storage and reactor cavity pools.

5. As water level lowers, return steam dryer to reactor vessel.

B. Restrictions on the Use of the Reactor Internals Storage Pool Gate

Operating procedures and maintenance procedures shall contain restrictions on the use of the reactor internals storage pool gate to explicitly prevent the draining of the reactor internals storage pool during two situations when the reactor cavity is filled.

1. Any time that the reactor cavity is flooded and the spent fuel pool gates are removed, the reactor internals storage pool will be flooded, thus the reactor internals storage pool gate will not perform a safety-related function. Consequently, at no time could a failure of the reactor internals storage pool gate or seal result in lowering water level to unacceptable levels in the spent fuel pool. The reactor internals storage pool gate and gate seals are classified nonsafety related as they do not perform a safety function of maintaining water level in the spent fuel storage pool.

   In addition, if the storage pit must be drained, then the reactor cavity shall be drained prior to the storage pool draining or occur in parallel to the storage pool draining but not ahead of the storage pool. These restrictions are necessary primarily due to the design of the storage pool gate.

   The gate was designed to withstand the combined loads from seismic and hydrostatic concurrently but only with the hydrostatic load being on the reactor internals storage pool side of the storage pool gate.

2. Also, any time that reactor cavity is flooded, the reactor internals storage pool shall be flooded. This eliminates the possibility of a personnel and/or radiological hazard or safety problem from occurring in

the event of a gate failure resulting from nondesigned load combinations.

9.1.4.2.8 Storage Equipment

Specially designed equipment storage racks are provided. Additional storage equipment is listed in Table 9.1-4. For a description of fuel storage racks and fuel arrangement, see Sections 9.1.1 and 9.1.2.

The fuel pool sipper may be used for out-of-core wet sipping at any time. It is used to detect a defective fuel bundle while circulating water through the fuel bundle in a closed system. The containers cannot be used for transporting a fuel bundle. The bail on the container head is not designed to fit into the fuel grapple.

9.1.4.2.9 Refueling Seal

During refueling operations, it will be required to isolate the space above the reactor vessel flange from the drywell atmosphere. To permit flooding of the reactor cavity for refueling, two separate air gaps will have to be closed: first, the inner annular air space between the reactor vessel flange and the primary containment, refueling bulkhead, and second, the outer air space between the reactor primary containment and the reactor head cavity liner. The inner space will be sealed with a steel bellows expansion joint. The outer space will be closed with an elastomer seal. The design consists of an inflatable elastomer seal used with a broad top flange which serves as a secondary seal in case of failure of the inflated section. Because the seismic adequacy of surrounding structures and qualification of the RPV response under accident loading are dependent on the stiffness of the seal, the predictable value of zero stiffness for the seal makes it possible to qualify the seal.

9.1.4.2.10 Under Reactor Vessel Servicing Equipment

The functions of the under reactor vessel servicing equipment are to remove and install control rod drives (CRD), thermal sleeves, and the SRM/IRM detectors, and to perform work associated with LPRM maintenance. Table 9.1-3 lists the equipment and tools required for servicing. Of the equipment listed, the CRD service platform is powered electrically and pneumatically.

The weight of the CRD handling tool is not to exceed 3,000 lb. When required, the CRD handling tool will be installed during refueling outages and shall be removed from the primary containment prior to startup. The CRD handling tool is equipped with adequate brakes or gearing to prevent uncontrolled movement.

The CRD service platform provides a working surface for equipment and personnel performing work in the under vessel area. It is a

polar platform capable of rotating in both forward and reverse directions for 360 deg. This equipment is designed in accordance with the applicable requirements of OSHA (Vol. 37, No. 202, Part 1910N), AISC, and ANSI-C-1 (National Electric Code).

The spring reel is used to pull the in-core guide tube seal or in-core detector into the in-core guide tube during in-core servicing.

The thermal sleeve installation tool locks, unlocks, and lowers and raises the thermal sleeve from the CRD guide tube.

The in-core flange seal test plug, used to determine the pressure integrity of the in-core flange O-ring seal, is constructed of noncorrosive material.

The key bender is designed to facilitate installation and removal of the antirotation key that is used on the thermal sleeve.

9.1.4.2.11 Description of Fuel Transfer

The fuel handling system provides a safe and effective means for transporting and handling fuel from the time it reaches the plant until it leaves the plant after post-irradiation cooling. The following sections describe the design bases of the fuel handling system. The requirements of RG 1.13 are satisfied.

Fuel handling procedures are described below. A typical refueling floor layout is shown on Figure 9.1-25.

The fuel handling process takes place primarily on the refueling floor above the reactor. The principal locations and equipment are shown on Figure 9.1-25. The reactor, fuel pool, internals pool, and shipping cask pool are connected to each other by canals. The fuel transfer canal is open during reactor refueling but otherwise is closed by two gates which make watertight barriers. The canal between the spent fuel pool and the shipping cask pool is normally open to allow use of the cask storage pool during various maintenance activities, but can still be sealed by two gates when required to support spent fuel shipping or other activities. The gate seal leak detection drain between the gates to the shipping cask pool canal has been sealed with a welded plate.

<u>Receipt and Inspection of New Fuel</u> The incoming new fuel is delivered to a receiving station. The crates are unloaded from the transport vehicle and examined for damage during shipment. The crate dimensions are approximately 32 in x 32 in x 18 ft. Each crate contains two fuel bundles supported by an inner metal container. The shipping weight of each unit is approximately 3,000 lb. The receiving station includes a separate area where the crate covers and the inner metal containers can be removed from the crates. Both inner and outer shipping containers are reusable.

The reactor building 25-ton auxiliary hoist will lift the inner container up through the equipment hatch to the refueling floor.

The fuel bundle will not be handled horizontally without support. The top and front of the metal shipping container are opened, and the bundles removed in a vertical position.

Either the 25-ton auxiliary or the 1/2-ton hoist of the RBPC equipped with general purpose grapple is used to transfer the new fuel bundles to either the new fuel storage vault or the new fuel inspection stand.

The new fuel inspection stand holds two bundles in vertical position. The inspector(s) ride up and down on a platform, and the bundles are manually rotated on their axes. Thus the inspectors can observe all visible surfaces on the bundles. After inspection, a new channel is usually installed on the bundle before transfer to the new fuel storage vault by the auxiliary hoist of the RBPC. The auxiliary hoist is also used with a general-purpose grapple to transfer new fuel from the new fuel vault or inspection stand to a storage rack position in the fuel pool. From this point on, the fuel is handled by the telescoping grapple on the refueling platform.

The refueling platform uses a grapple on a telescoping mast for lifting and transporting fuel bundles or assemblies. The telescoping mast can extend to the proper work level and, in its fully retracted state, maintains adequate water shielding over the fuel being handled.

<u>Channeling Fuel</u>

Two fuel preparation machines are located in the spent fuel pool.

When a new channel is used on new fuel that has not been previously channeled, the procedure is as follows: Either by using the refueling platform grapple, the bundle is placed in a fuel preparation machine. The fuel preparation machine carriage is lowered to its lowest position. A new fuel channel is lowered into the fuel preparation machine and attached to a channel handling tool at the top of the channel. The fuel preparation machine is raised lifting the fuel bundle up into the new channel. The channel is then bolted to the bundle using a channel fastener bolt and the channel bolt wrench. Or, the new channel can be lowered over the fuel bundle while the fuel bundle is in the fuel inspection stand.

If a new channel is to be used on irradiated fuel which will be returned to the reactor, the procedure is as follows: Using the refueling platform grapple, the irradiated bundle is transferred to a fuel preparation machine. The used channel is unbolted from the bundle using the channel bolt wrench. The channel handling tool is fastened to the top of the fuel channel and the fuel

preparation machine carriage is lowered removing the fuel bundle from the channel. Using the channel handling tool, the used channel is placed in the channel storage rack. A new fuel channel is lowered into the fuel preparation machine and attached to the channel handling tool at the top of the channel. The fuel preparation machine carriage is raised lifting the fuel bundle up into the new channel. The channel is then bolted to the bundle using a channel fastener bolt and the channel bolt wrench.

New channels should be placed on new fuel bundles and will remain on the same fuel bundle until it is finally discharged. Irradiated channels normally are not used on different fuel bundles.

<u>Equipment Preparation</u> Prior to use for refueling, all equipment is placed in readiness. All tools, grapples, slings, strongbacks, and stud tensioners, etc., are given a thorough check and any defective or worn parts replaced. Air hoses on grapples are leak tested and crane cables are inspected. All necessary maintenance and interlock checks are performed to ensure no extended outage due to equipment failure.

The in-core flux monitors, in their shipping container, are on the refueling floor. The channeled new fuel and the replacement control rods are ready in the storage pool.

<u>Reactor Shutdown</u> The reactor power is lowered according to procedures. At 36-percent power level, the top four reactor cavity shield plugs may be removed. Once the reactor is shut down, the four remaining reactor cavity shield plugs can be removed. During cooldown, the RPV is vented and filled to above the flange level to equalize cooling. The plug removal is accomplished with the RBPC and the supplied slings. Following the removal of these four plugs, the three fuel canal and three internal canal plugs can be removed by use of the RBPC and supplied slings. Thus, a total of 14 separate plugs must be removed and placed on the refueling floor. A refueling equipment storage and crane clearance arrangement drawing is issued to locate placement of these plugs on the refueling floor. The outer fuel pool gate may also be removed at this time or deferred until completion of reactor cavity flooding. The gate sling is attached to the gate lifting lugs and the RBPC lifts the gate and places it on the fuel pool storage lugs.

After removal of all reactor head cavity and canal shield plugs, the work to remove the drywell head can begin. The drywell head is attached by a quick-disconnect mechanism. To remove the head, the quick-disconnect pins are withdrawn and stored separately for reinsertion when the head is replaced. The drywell head is lifted by the RBPC with strongbacks and lifting slings attached in conjunction with the four hydraulic jacks located around the chimney base, and is stored on the refueling floor.

When the drywell head has been removed, an array of piping is exposed that must be serviced. Various vent piping penetrations through the reactor head cavity must be removed and the penetrations made watertight. Vessel head piping and head insulation must be removed and transported to storage on the refueling floor. Water level in the vessel is now brought to flange level in preparation for head removal.

Reactor Vessel Opening

The stud tensioners are transported by the RBPC on the head strongback carousel and positioned on the reactor vessel head. Each stud is tensioned and its nut loosened in one and one-half passes. After the stud is fully detensioned, the stud is turned out and the stud, nut and washer are raised approximately 3 ft and held in place with a spacer supported by the vessel head flange. The studs, nuts and washers will then be removed in one lift along with the reactor vessel head by the RBPC. If any studs are not removed, i.e., stuck studs or studs which will interfere with the head holding pedestals, the nut and washer will be removed and placed on adjacent studs or placed in the nut and washer rack for storage on the refuel floor.

The strongbacks and lifting slings, transported by the RBPC, are then attached to the vessel head and the head is transported to the head holding pedestals on the refueling floor. These pedestals keep the vessel head elevated to facilitate inspection and O-ring replacement.

As an alternate method of moving the RPV head, with the head strongback carousel attached to the vessel head, the head is transported to the head holding pedestals on the refueling floor by the RBPC. These pedestals keep the vessel head elevated to facilitate inspection and O-ring replacement.

The studs in line with the fuel transfer canal are removed from the vessel flange and placed in the rack provided. The loaded rack is transported to the refueling floor for storage.

Dryer Removal The dryer-separator sling and lifting cruciform are lowered by the RBPC and attached to the dryer lifting lugs. The dryer is lifted from the reactor vessel and transported to its storage location in the internals storage pool adjacent to the reactor head cavity.

Separator Removal In preparation for separator removal, the refueling platform with the service pole caddy system is positioned over the reactor cavity. The service poles are then utilized to operate the shroud head bolts.

When unbolting is accomplished, the separator sling and lifting cruciform are lowered into the vessel and attached to the separator lifting lugs. The separator is transferred underwater to its allotted storage place in the adjacent pool.

The four main steam line plugs are installed from inside the vessel using the plugs furnished for this duty. Servicing or testing of the main steam system components can thus be accomplished without adding to the critical refueling path time. As an alternate method in preparation for separator removal, the service platform and service platform support are installed on the vessel flange. From the service platform work area, the four main steam lines are plugged from inside the vessel using the plugs furnished for this duty. Servicing of the safety and relief valves can thus be accomplished without adding to the critical refueling path time. Working from the service platform, the separator is unbolted using the shroud head bolt wrenches furnished.

When unbolting is accomplished, the service platform is removed and stored on the refueling floor. The service platform support remains on the vessel flange during the remainder of the refueling outage and acts as the flange seal surface protector. The separator sling and lifting cruciform are lowered into the vessel and attached to the separator lifting lugs. The water in the reactor head cavity and in the internals storage is raised to fuel pool water level and the separator is transferred underwater to its allotted storage place in the adjacent pool.

<u>Fuel Bundle Sampling</u> During reactor operation, the core offgas radiation level is monitored. If a rise in offgas activity is noted, the reactor core may be sampled during shutdown to locate any leaking fuel assemblies. The fuel sampler rests on the channels of a four-bundle array in the core. An air bubble is pumped into the top of the four fuel bundles and allowed to stay about 10 min or more as defined in applicable procedures. This stops water circulation through the bundles and allows fission products to concentrate if a bundle is defective. After about 10 min, a water sample is taken for fission product analysis.

As an alternative to using the fuel sampler, the fuel bundles may be sipped using the refueling mast sipper during transfer from the reactor vessel to the spent fuel pool or by raising and then lowering the bundle while in the reactor vessel or spent fuel pool. Water is recirculated from the top of the fuel bundle to the sipping system and back to the spent fuel pool or reactor, depending on where the sipping is being performed. The fission product gases are separated from the water and are collected for sampling and analysis. A leaking fuel bundle will be identified by using count rates recorded by the radiation detection equipment. The fuel bundles are not raised above limits established by the Technical Specifications during this sipping method.

<u>Refueling and Reactor Servicing</u> The inner and outer spent fuel pool gates isolating the fuel pool from the reactor head cavity are now removed by the RBPC and placed on the spent fuel pool wall lugs, thereby interconnecting the fuel pool, the reactor

head cavity, and the internals storage pool. The refueling of the reactor can now begin.

Refueling

Prior to movement of irradiated fuel in the RPV, the reactor shall be subcritical for at least 24 hr as described in TRM Section 3.9.2.

During a normal equilibrium outage, a large percentage of the fuel is removed from the reactor vessel, the remaining fuel is shuffled in the core, and new fuel is installed. The actual fuel handling is done with the fuel grapple which is an integral part of the refueling platform. The platform runs on rails over the fuel pool and the reactor head cavity. In addition to the fuel grapple, the refueling platform is equipped with two auxiliary hoists that can be used with various grapples to service other reactor internals.

To move fuel, the fuel grapple is aligned over the fuel assembly, lowered and attached to the fuel bundle bail. The fuel bundle is raised out of the core, moved through the refueling slot to the fuel pool, positioned over the storage rack, and lowered into the storage rack. Fuel is shuffled and new fuel is moved from the storage pool to the reactor vessel in the same manner.

To preclude the possibility of raising radioactive material out of the water, redundant electrical limit switches are incorporated in the hoist and interlocked to prevent hoisting above the preset limit. In addition, the cables on the auxiliary hoists incorporate adjustable stops that will jam the hoist cable against the hoist structure, which prevents hoisting if the limit switch interlock system should fail.

Requirements regarding direct communication between the control room and refueling floor personnel are described in TRM Section 3.9.3.

Vessel Closure The following steps return the reactor to operating condition. The procedures are the reverse of those described in the preceding sections (many steps are performed in parallel and not as listed).

1. Core verification: The core position of each fuel assembly must be verified to assure that the desired core configuration has been attained.

2. CRD tests: The CRD timing, friction, and scram tests are performed as described in Section 4.6.

3. Install inner and outer fuel pool gates.

4. Remove main steam line plugs.

5. Replace separator and bolt.
6. Replace dryer.
7. Drain dryer-separator storage pool and reactor head cavity.
8. Install reactor head.
9. Replace vessel studs.
10. Remove drywell seal surface covering.
11. Decontaminate reactor head cavity.
12. Decontaminate dryer-separator storage pool.
13. Install vessel head piping and insulation.
14. Open drywell vents, install vent piping.
15. Hydrotest vessel, if required.
16. Install drywell head.
17. Replace storage pool canal plugs.
18. Replace fuel pool canal plugs.
19. Install reactor head cavity shield plugs.
20. Inert drywell and suppression chamber.
21. Perform startup tests. The reactor is returned to full power operation. Power is increased gradually in a series of steps until the reactor is operating at rated power. At specific steps during the approach to power, the in-core flux monitors are calibrated.

<u>Shipping Fuel from Site</u>

After the spent fuel assemblies have been stored in the spent fuel pool for a sufficient time (at least 90 days) to allow radioactive decay heat to decrease to levels suitable for transport, they are transferred from the spent fuel storage racks and inserted into the spent fuel shipping cask. This is done using the fuel grapple on the refueling platform. After completion of filling the cask with spent fuel and cover installation, the gates are replaced and the cask is removed.

The cask head, which is dry stored on the refueling floor, is transferred to the spent fuel storage pool shipping cask area by the RBPC. The cask is then washed down to remove liquid and particulate surface contaminants in preparation for shipment

offsite. The decontaminated cask is then attached to its lifting yoke, raised out of the storage pool by the RBPC, and then lowered through the equipment hatch directly onto its transport vehicle. The transfer vehicle is equipped with the necessary heat exchangers and integral tie-downs to protect and secure the spent fuel shipping cask, in accordance with applicable Nuclear Regulatory Commission (NRC) and Department of Transportation (DOT) regulations.

9.1.4.3 Safety Evaluation - Fuel Handling System

Safety aspects (evaluation) of the fuel servicing equipment are discussed in Section 9.1.4.2.3 and safety aspects of the refueling equipment are discussed in Section 9.1.4.2.7. A description of fuel transfer, including appropriate safety features, is provided in Section 9.1.4.2.11. In addition, the following summary safety evaluation of the fuel handling system is provided.

The fuel preparation machine 2FNR*TL1B removes and installs channels with all parts remaining underwater. Mechanical stops prevent the carriage from lifting the fuel bundle, channel, or assembly to a height where water shielding is less than 7 ft above the fuel bundle or channel. Irradiated channels, as well as small parts such as bolts and springs, are stored underwater. The spaces in the channel storage rack have center posts that prevent the loading of fuel bundles into this rack.

Fuel preparation machine 2FNR*TL1A is used to transfer new nonirradiated fuel into the spent fuel pool with its up-travel-stops removed so the assembly in the fuel preparation machine is not fully submerged. When this fuel preparation machine is placed in service, controls will be in place to limit its use to new/nonirradiated fuel only.

The refueling platform is designed to prevent it from toppling into the pools during a SSE. The grapple utilized for fuel movement is on the end of a telescoping mast. At full retraction of the mast, the grapple is a minimum of 6 ft 1 1/2 in (approximately) below the water surface, so there is no chance of raising a fuel assembly to the point where it is inadequately shielded by water. The grapple is hoisted by redundant cables inside the mast and is lowered by gravity. A digital readout is displayed to the Operator, showing him the exact coordinates of the grapple over the core.

The mast is suspended and gimbaled from the trolley, near its top, so that the mast can be swung about the axis of platform travel to remove the grapple from the water for servicing and for storage.

The grapple has two interlocking hooks, that are operated by an air cylinder. Engagement is indicated to the Operator.

In addition to the main hoist on the trolley, there is an auxiliary hoist on the trolley and another hoist on its own monorail. These three hoists are precluded from operating simultaneously because control power is available to only one of them at a time.

The two auxiliary hoists have electrical interlocks that will maintain a minimum of 6 ft 1 1/2 in (approximately) of water coverage above the lifted load. Adjustable mechanical jam-stops on the cables back up these interlocks.

The following is a list of light loads (weight of one fuel assembly or less) that are carried over the open reactor vessel or spent fuel pool.

Item	Approximate Weight (lb)
Blade Guide	145
Control Rod	215
Control Rod Guide Tube	257
Fuel Support Piece	60
Fuel Assembly Sampler Head	140
In-vessel Storage Rack	575

These items, with the exception of a double-blade guide, are not carried at a height greater than that of a fuel assembly. A double-blade guide will be lifted a maximum of 5 in higher than a fuel assembly. However, in all cases, the potential energy of these items, in the case of an accidental drop, is less than that of a fuel assembly.

The accidental drop of a light load into the reactor core could only occur during fuel loading. This event will not cause a change in criticality when the control rod blades are fully inserted.

The accidental drop of a light load over the spent fuel racks will not affect the geometric array of the stored fuel. Therefore, this event would not cause a change in criticality.

None of these accident events will result in a release of radioactivity in excess of the design basis fuel handling accident.

In summary, the fuel handling system complies with GDC 2, 3, 4, 5, 61, 62, and 63, and applicable portions of 10CFR50.

A system-level, qualitative-type FMEA relative to this system is discussed in Section 15A.6.5. The safety evaluation of the new and spent fuel storage is presented in Sections 9.1.1.3 and 9.1.2.3.

9.1.4.4 Inspection and Testing Requirements

9.1.4.4.1 Inspection

Refueling and servicing equipment is subject to strict QA controls, incorporating the requirements of 10CFR50 Appendix B. Components defined as essential to safety, such as the fuel storage racks and the refueling platform, have engineering-specified quality requirements that identify safety-related features which require specific QA verification of compliance to drawing requirements.

9.1.4.4.2 Testing

Qualification testing is performed on refueling and servicing equipment prior to multiunit production. Test specifications are defined by the responsible design Engineer and may include sequence of operations, load capacity, and life cycles tests. These test activities are performed by an independent test engineering group, and in many cases a full design review of the product is conducted before and after the qualification testing cycle. Any design changes affecting function that are made after the completion of qualification testing are requalified by test or calculation.

Functional tests are performed in the shop prior to the shipment of production units and generally include electrical tests, leak tests, and sequence of operations tests.

When the unit is received at the site, it is inspected to ensure that no damage has occurred during transit or storage. Prior to use and at periodic intervals, each piece of equipment is again tested to ensure that the electrical and/or mechanical functions are operational.

Passive units, such as the fuel storage racks, are visually inspected prior to use. There is an operation and maintenance instruction manual for each tool that additionally requires a series of functional checks each time the unit is operated for reactor refueling or servicing.

Fuel handling and vessel servicing equipment preoperational tests are described in Section 14.2.

Specific operability requirements for the refueling platform and fuel grapple position refueling interlock are described in TRM Section 3.9.4.

9.1.4.5 Instrumentation Requirements

9.1.4.5.1 Refueling and Servicing Equipment

The majority of the refueling and servicing equipment is manually operated and controlled locally by the Operator's visual observations. This type of operation does not require a dynamic

instrumentation system. However, there are several components, essential to prudent operation that do have instrumentation and control systems.

9.1.4.5.2 Refueling Platform

The refueling platform has a nonsafety-related X-Y-Z position indicator system that informs the Operator which core fuel cell the fuel grapple is accessing. Refer to Section 7.7.1.4 for discussion of refueling interlocks.

Additionally, there is a series of mechanically activated switches and relays that provides monitor indications on the Operator's console for grapple limits, hoist and cable load conditions, and confirmation that the grapple's hook is either engaged or released.

A series of load cells is installed to provide automatic shutdown whenever threshold limits are exceeded on either the fuel grapple or the auxiliary hoist units.

9.1.4.5.3 Fuel Support Grapple

Although the fuel support grapple is not essential to safety, it has an instrumentation system consisting of mechanical switches and indicator lights. This system provides the Operator with a positive indication that the grapple is properly aligned and oriented and that the grappling mechanism is either extended or retracted.

9.1.4.5.4 Reactor Building Polar Crane

The RBPC may be controlled from the cab or by using a radio. The radio receiver is designed so that any malfunction of operation causes all motions to fail safe. The radio receiver is designed to assure proper and unique operational signals, preventing operation from illogical or extraneous commands.

9.1.4.5.5 Radiation Monitoring

The radiation monitoring equipment for the refueling and servicing area is discussed in Sections 11.5.2.1.2 and 12.3.4.

9.1.5 This section has been deleted.

9.1.6 Alternate Decay Heat Removal System

9.1.6.1 Design Basis

The ADH system, in conjunction with natural circulation, is designed to remove the decay heat released from the spent fuel pool, reactor core, reactor internals storage pool and cavity, during refueling outages that begin in December to mid-April, to maintain reactor coolant temperatures suitable for refueling by:

1. Maintaining the spent fuel pool, reactor cavity and reactor internals storage pool bulk average temperature below 110°F.

2. Maintaining the spent fuel pool, reactor cavity and reactor internals storage pool average surface temperature below 110°F.

3. Maintaining the reactor core coolant exit temperature below 150°F.

The ADH system can also be used at other times; however, supplemental cooling may be needed to maintain the above temperature limits.

The system is designed for use when the vessel head is removed, the reactor cavity is flooded to an elevation greater than 22 ft 3 in above the vessel flange, and the spent fuel pool is interconnected to the reactor cavity pool. The purpose of the ADH system is to allow both trains of RHR to be taken out of service simultaneously during refueling operations for maintenance.

During refueling, the ADH system is intended to perform the normal spent fuel pool cooling function of the SFC system described in Section 9.1.3. However, the SFC system will be maintained available such that it can be restored to provide spent fuel pool, reactor cavity and reactor internals storage pool cooling in the event ADH is lost.

During normal plant operation and when extended maintenance of one SFC loop is planned, the ADH system can be used to supplement the spent fuel pool cooling. One train of ADH has the capability to maintain the spent fuel pool temperature below 125°F under all conditions.

9.1.6.2 Description

The ADH system is shown on Figure 9.1-28. The ADH system consists of a primary loop for removing decay heat from the spent fuel pool and the reactor core and a secondary loop to transfer the decay heat to the atmosphere. The primary loop is housed entirely in the reactor building. The primary loop draws water from the northeast corner of the spent fuel pool, utilizes one of two 100-percent capacity primary loop pumps to circulate the water through one of two 100-percent capacity flat plate heat exchangers, and returns the cooled water to the reactor cavity and the spent fuel pool via existing SFC system spargers. The secondary loop transfers heat from the plate heat exchangers to the atmosphere via one of two 100-percent capacity mechanical draft cooling towers. The secondary loop is located outdoors with the exception of the portion of the supply and return piping which connects to the plate heat exchangers.

The fundamental design provides two 100-percent capacity cooling trains, each rated at 54.9 x 10^6 Btu/hr. Electrical power is provided to all primary and secondary loop equipment from the facilities power loop, with portable diesel generator tie-in capability. Connections are provided to allow bypassing of the water exiting the cooling tower through temporary chillers during summer outages. Chillers may be required during summer outages due to higher wet-bulb temperatures.

9.1.6.3 Safety Evaluation

As previously described, the ADH system, in conjunction with natural circulation, does not perform any safety-related functions and is only intended to perform RHR shutdown cooling functions during refueling operations with the head removed from the vessel, the reactor cavity flooded to an elevation of greater than 22 ft 3 in above the vessel flange, and the spent fuel pool interconnected to the reactor cavity pool. The SFC system will be used to backup the ADH system in the event the ADH system is lost due to either equipment malfunction or natural phenomena.

As described in Section 9.1.3, two trains of SFC will maintain the cavity, the spent fuel pool, and reactor core coolant exit temperatures below design limits following a loss of ADH cooling.

The portion of the ADH system which interfaces with the safety-related SFC system is designed in accordance with design requirements consistent with the requirements of the SFC system, as described in Section 9.1.3.2.

The portion of the ADH system which penetrates the reactor building (secondary containment boundary) is designed consistent with the requirements of Section 6.2.3.1 for the reactor building structure. The design of the inboard check valves, which provide secondary containment integrity in the event of an ADH pipe break, are designed for a maximum pressure differential between the inside and outside of the building of 6.92 in of water. This incorporates the worst-case pressure loadings for the various events described in Section 6.2.3.1.

The impact of internally-generated missiles and missiles generated by natural phenomena is evaluated. Internally-generated missiles from the ADH primary loop pumps is not credible based on Section 3.5.1.1.5, which considers catastrophic failure of rotating equipment that leads to generation of missiles to be not credible. Postulated missiles generated by natural phenomena are evaluated based on an analysis of the probability of a missile strike on the reactor building penetrations for the ADH components. The analysis calculated the probability of a missile strike to these openings to be less than the RG 1.117 accepted value of 1.0 x 10^{-7} per year and, therefore, the new penetrations are not protected against missile strikes from external missiles. The probability analysis described above

is based on the methodology developed by L. A. Twisdale, as documented in EPRI Report No. NP-2005.

The nonsafety portions of the ADH system housed inside the reactor building are designed and supported, where required, for seismic forces to ensure that failure of these components does not affect the operation of any Category I equipment or cause damage to Category I structures.

The equipment located in the yard, including cooling towers, secondary pumps and motor control center (MCC), is evaluated for impact on atmosphere dispersion factors for accident and effluent releases in accordance with Sections 2.3.4 and 2.3.5. The evaluation concludes the ADH components in the yard will not impact atmosphere dispersion factors.

The ADH components located in the yard are evaluated for commitments in Section 2.4 regarding protection of safety-related facilities, systems and components against flood damage resulting from probable maximum precipitation (PMP) and historical maximum lake level. The evaluation concludes the ADH components will not impact external flood protection features.

The ADH equipment and piping heat loads are evaluated for impact on the reactor building heating, ventilating and air conditioning (HVAC) system. The reactor building HVAC system is capable of performing the design functions of Section 9.4.2 with the ADH heat loads.

The ADH piping is evaluated for flooding in the reactor building. The ADH piping is moderate-energy piping. The evaluation concludes the flood height in the reactor building resulting from a postulated crack in the moderate-energy ADH piping is below the level which results from the limiting case 18-in RHS line. Therefore, existing flood protection features are unaffected by the ADH piping.

9.1.6.4 Inspection and Testing

No special equipment tests are required for the nonsafety-related ADH system. The ADH plate heat exchangers will be leak tested prior to utilizing the ADH system during an outage, or any other time its use may be required, if the system has not been in operation for more than 30 days. This test is provided to ensure heat exchanger pressure boundary integrity and to minimize the potential for primary to secondary side leakage. The piping and check valves which form secondary containment pressure boundary will be tested for leakage prior to use of the ADH system. When the ADH system is not in use, spectacle flanges located between the check valves and secondary containment boundary may be used to provide the necessary isolation. Other than the above, routine visual inspections during operation are adequate to verify system operability.

Post-modification testing to verify that installation meets design standards includes system pressure tests, secondary containment isolation valve leak testing, and component operability testing. Startup testing included system hydraulic and thermal testing.

9.1.6.5 Instrumentation Requirements

Description

Instruments and controls are provided for manual control of the ADH system. The controls and monitors described in the following paragraphs are located local to the major pumps, heat exchangers and cooling towers.

All outdoor 480-V motor services are provided with local controls on the MCC. These controls have a start-stop switch for each secondary loop pump and each cooling tower fan. No interlocks are provided for the secondary loop pumps. The cooling tower fans are provided with vibration trip switches and reducer gear oil level protective switches.

The primary loop pumps located in the reactor building have local controls mounted near the pumps to start and stop these pumps. An interlock from a differential pressure indicating switch will trip these pumps on low differential pressure across the heat exchanger boundary to maintain any leakage directed into the primary loop. This interlock circuit is designed to electrically fail in the low differential pressure direction.

Operation

The ADH valves are opened and closed manually.

The ADH pumps and cooling towers are placed into service manually from local remote switches. Interlocks prevent operation of the primary loop pumps whenever secondary loop pressure is less than 10 psi greater than primary loop pressure.

City water makeup to the cooling towers opens automatically when the cooling tower basin water level is low and closes automatically on high water level.

Monitoring

Local indicators (no control room indication) are provided for:

1. primary pump suction pressure

2. primary pump discharge pressure

3. primary pump flow

4. secondary pump suction pressure

5. secondary pump discharge pressure
6. secondary pump flow
7. heat exchanger inlet temperature (process)
8. heat exchanger outlet temperature (process)
9. heat exchanger inlet temperature (service)
10. heat exchanger outlet temperature (service)

9.1.7 Dry Cask Storage

Nine Mile Point Nuclear Station, LLC constructed and operates a dry cask storage ISFSI facility under a 10 CFR 72 general license at the NMPNS site. The ISFSI is used for interim storage of NMPNS Unit 1 and Unit 2 spent nuclear fuel assemblies utilizing the Transnuclear, Inc., standardized NUHOMS® modular storage system for irradiated nuclear fuel with NUHOMS®-61BT and NUHOMS®-61BTH Dry Shielded Canisters (DSCs). Each DSC can contain 61 spent fuel assemblies and each Horizontal Storage Module (HSM) contains only one DSC.

The ISFSI is southwest of NMPNS Unit 1 and the security fencing was expanded to enclose the foundation pads for the HSMs. The ISFSI is sized for the storage of 200 DSCs using the Standardized NUHOMS® horizontal cask system. The initial campaign is for 40 Model 102 HSMs and 40 Model H HSMs (HSM-Hs). The type and model of the remaining 120 HSMs ad DSCs will be determined at a later date, but space for 120 HMS-Hs was reserved. Likewise, the reinforced concrete approach slabs and HSM foundation pads for the remaining 120 HSMs will be construced at a later date; however, the site was cleared and graded for the future concrete slabs.

9.1.8 References

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11. QA Validation of ANSYS 5.3, Holtec Report HI-961580, 1996 (Proprietary).

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14. Criticality Safety Evaluation for the Nine Mile Point 2 Rack Installation Project, Holtec Report HI-2012621, Revision 1, March 2001.

15. GNF2-Nine Mile Point Nuclear Station, Unit 2: Fuel Storage Criticality Safety-GNF2, S22.2-XX-G102NF, November 2015.

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TABLE 9.1-1

FUEL SERVICING EQUIPMENT

Component Identification	Safety Class	Quality Group	Seismic Category	Quality Assurance Category
Fuel prep machine	2	E	I	I
New fuel inspection stand	O	E	NA	NA
Channel bolt wrench	O	E	NA	NA
Channel handling tool	O	E	NA	NA
Fuel pool sipper	O	E	NA	NA
Fuel inspection fixture	O	E	NA	NA
Channel gauging fixture	O	E	NA	NA
General purpose grapple	2	E	NA	I
Jib cranes	O	E	I	NA

KEY: O = Other
E = Industrial code applies
NA = Not applicable
I = Category I

TABLE 9.1-2

REACTOR VESSEL SERVICING EQUIPMENT

Component Identification	Safety Class	Quality Group	Seismic Category	QA Category
Reactor vessel servicing tools	O	E	NA	NA
Steam line plug	O	E	NA	I
Shroud head bolt wrench	O	E	NA	NA
Vessel nut handling tool	O	E	NA	NA
Head holding pedestal	O	E	I	NA
Head nut and washer rack	O	E	NA	NA
Head stud rack	O	E	NA	NA
Head strongback carousel	O	E	NA	I
Stud tensioner assembly	O	E	NA	NA
Lifting strongback	O	E	NA	I
Lifting slings	O	E	NA	I
Service platform	O	E	NA	NA
Service platform support	O	E	NA	NA
Steam line plug installing tool	O	E	NA	NA
Service pole caddy system	O	E	I	NA

KEY: O = Other; E = Industrial code applies; NA = Not applicable; I = Category I

TABLE 9.1-3

UNDER-REACTOR VESSEL SERVICING EQUIPMENT AND TOOLS

Equipment/Tool	Safety Class	Seismic Category	Quality Assurance Category
CRD handling Tool	O	NA	NA
Equipment handling Platform	O	NA	NA
Spring reel	O	NA	NA
Thermal sleeve removal tool	O	NA	NA
In-core flange seal test plug	O	NA	NA
Key bender	O	NA	NA

KEY: O = Other
NA = Not applicable

TABLE 9.1-4

TOOLS AND SERVICING EQUIPMENT

<u>Fuel Servicing Equipment</u>
Fuel preparation machines
New fuel inspection stand
Channel bolt wrenches
Channel handling tool
Fuel pool sipper
Channel gauging fixture
General purpose grapples
Fuel inspection fixture
Jib cranes
<u>Servicing Aids</u>
Pool tool accessories
Actuating poles
General area underwater lights
Local area underwater lights
Drop lights
Underwater TV monitoring system
Underwater vacuum cleaner
Viewing aids
Light support brackets
In-core detector cutter
In-core manipulator
<u>Reactor Vessel Servicing Equipment</u>
Reactor vessel servicing tools
Steam line plugs
Shroud head bolt wrenches
Head holding pedestals
Head stud rack
Head strongback carousel
Lifting strongback
Lifting slings
Service platform
Service platform support
Steam line plug/installation tool
Vessel nut handling tool
Head nut and washer storage racks
Service pole caddy system
<u>In-Vessel Servicing Equipment</u>
Auxiliary service platform
Instrument strongback
Control rod grapple

TABLE 9.1-4 (Cont'd.)

Control rod guide tube grapple
Fuel support grapple
Grid guide
Control rod latch tool
Instrument handling tool
Control rod guide tube seal
In-core guide tube seals
Fuel bundle sampler
Blade guides
Peripheral orifice grapple
Orifice holder
Peripheral fuel support plug
Fuel bail cleaner

<u>Refueling Equipment</u>

Refueling equipment servicing tools
Refueling platform
Fuel transfer shielding bridge
CAVSPAN system

<u>Storage Equipment</u>

Spent fuel storage racks
Channel storage racks
In-vessel racks
New fuel storage rack
Control rod guide tube storage rack

<u>Under-Reactor Vessel Servicing Equipment</u>

Control rod drive servicing tools
CRD hydraulic system tools
Neutron monitoring system servicing tools
Spring reels
Control rod drive handling tool
Equipment handling platform
Thermal sleeve installation tool
In-core flange seal test plug
Key bender

TABLE 9.1-5

WATER QUALITY FOR SPENT FUEL POOL

Conductivity	≤3 umho/cm at 25°C
Chlorides	≤0.5 ppm
pH	5.3 to 7.5 at 25°C
Heavy elements	≤0.1 ppm
Total Insolubles	<1 ppm

TABLE 9.1-6

SPENT FUEL POOL COOLING HEAT EXCHANGERS
2SFC*E1A and *E1B

Manufacturer's Type: TEMA AEU

Total Duty (each): 15,000,000 Btu/hr

	Fluid Entering (lb/hr)	Temp in	Temp out
Tube side flow	1,125,000	125°F	111.7°F
Shell side flow	1,200,000	95°F	107.5°F

	Design Pressure	Design Temp	Materials of Construction
Shell	150 psig	150°F	Carbon steel
Tube	300 psig	150°F	SA-688 Type 304 SS

TABLE 9.1-7

ALTERNATE DECAY HEAT REMOVAL SYSTEM COMPONENTS

Primary Pumps	No.	2
2ADH-P1A, P1B	% Capacity each	100%
	Head[1]	218 ft
	Capacity[1]	3450 gpm
	Brake horsepower	228 hp
Secondary Pumps	No.	2
2ADH-P2A, P2B	% Capacity each	100%
	Head	417 ft
	Capacity	4000 gpm
	Brake horsepower	506 hp
Heat Exchangers	No.	2
2ADH-E1A, E1B	% Capacity each	100%
	Primary flow (hot)	3450
	Secondary flow (cold)	4000
	Hot inlet temperature	106°F
	Hot outlet temperature	74°F
	Cold inlet temperature	71°F
	Cold outlet temperature	99°F
	Heat Load	54.9×10^6 Btu/hr
Cooling Towers	No.	2
2ADH-TWR1A, TWR1B	% Capacity each	100%
	Flow	4000 gpm
	Inlet temperature	99°F
	Outlet temperature	71°F
	Wet-bulb temperature	64°F
	Approach	7°F
	Range	28°F

(1) Primary loop pumps have operating head capacity of 3900 gpm @ 205 ft.

9.2 WATER SYSTEMS

9.2.1 Service Water System

9.2.1.1 Design Bases

9.2.1.1.1 Safety Design Bases

The SWP system is designed with suitable redundancy to provide a reliable supply of cooling water during and following a design basis loss-of-coolant accident (LOCA) for the following essential components and systems:

1. RHR heat exchangers.
2. Emergency diesel generator coolers.
3. Control building chilled water chillers.
4. RHR pump seal coolers.
5. Design basis accident (DBA) hydrogen recombiners.
6. Reactor building ventilation recirculation cooling coils.
7. Reactor building, control building, diesel generator building, and service water pump bay unit coolers.
8. Spent fuel pool heat exchangers, if needed, as a backup to the RBCLCW system.
9. Spent fuel pool, if needed, as emergency makeup.

In addition, an intertie with the RHR system is provided to allow flooding of the containment, if required, during the post-LOCA recovery period. Essential portions of the SWP system are designed, fabricated, and installed in accordance with Category I criteria and ASME Boiler and Pressure Vessel Code, Section III, Safety Class 3 requirements.

9.2.1.1.2 Power Generation Design Bases

The SWP system is designed to provide cooling water to the secondary sides of the RBCLCW and turbine building closed loop cooling water (TBCLCW) heat exchangers during normal plant operation and planned outages. Service water is also supplied to the secondary side of the RHR heat exchangers during planned unit outages. In addition, the system is designed to provide makeup water to the circulating water system (CWS) and cooling water to miscellaneous nonessential turbine and reactor building components during normal plant operation.

Nonessential portions of the SWP system are designed, fabricated, and installed in accordance with ANSI B31.1 and nonseismic requirements.

9.2.1.2 System Description

The SWP system configuration is shown on Figure 9.2-1. The water enters the SWP system from Lake Ontario through two intake structures, passes through trash racks and traveling water screens (Section 9.2.5), and enters the SWP intake bay.

SWP pumps take their suction from the intake bay. Six motor-driven, horizontal, centrifugal pumps are provided. Each pump is rated at 600 hp and designed to deliver 10,000 gpm at 185 ft of head (TDH). Pumps are mounted in two separate bays, three pumps in each bay, at el 224 ft in the screenwell building. The SWP pump impeller centerline is located at el 227 ft 10 1/2 in. A conservative margin is provided over the required net positive suction head (NPSH) at the design minimum low water level for the intake bay (Section 9.2.5).

Service water is pumped from the intake bay through an automatically-operated strainer located in the discharge line of each pump. Strainers are designed to retain any particle greater than 1/32 in. The backwash effluent is directed to the discharge bay.

The SWP system has a design pressure and temperature of 150 psig and 130°F, respectively. All components cooled by the SWP system are designed for a maximum service water inlet temperature of 84°F. All essential and nonessential SWP piping is 150-lb carbon steel with a corrosion allowance of 0.125 in, with the exception of the piping to the service water radiation monitoring cabinets which are made up with sections of stainless steel pipe.

The SWP system is designed with three loops. Two are essential and one is nonessential. All essential components are fed by the safety-related loops. During an accident the nonessential loop is isolated as stated below. There are no essential heat loads on the nonsafety-related loop.

From the strainers, the service water is directed to a common header in the screenwell building. Two motor-operated isolation valves are provided in the header. These isolation valves separate the SWP system into two separate, redundant systems, an A loop and B loop. With the system isolated, pumps 2SWP*P1A, 2SWP*P1C, and 2SWP*P1E supply the A loop, while pumps 2SWP*P1B, 2SWP*P1D, and 2SWP*P1F supply the B loop. All essential electrical components in the A and B loops are powered from Division I and II electrical buses, respectively.

Two takeoffs from the common screenwell header, one from each side of the isolation valves, supply service water to the essential components in the reactor building. One takeoff from

each of the reactor building supply lines is routed to essential components in the control and diesel generator buildings. One takeoff from the A loop reactor building supply line supplies service water to reactor building nonessential components during normal plant operation. One takeoff from the A loop portion of the common screenwell header supplies service water to the turbine building nonessential components during normal plant operation. Refer to Tables 9.2-1 and 9.2-2 for tabulations of essential and nonessential components that are cooled by the SWP system and cooling water flow rates.

Service water discharge from all buildings is combined outside the buildings in two separate, redundant discharge headers. A tap-off from each discharge header is routed to the CWS to supply makeup water. The service water discharges at el 279'-9". This discharge and the blowdown from the CWS are returned to Lake Ontario through a discharge tunnel and diffuser system.

Each of the main supply and return lines that supply service water to the reactor and turbine building nonessential components is provided with two isolation valves located in series. The discharge headers that supply CWS makeup water each contain two pneumatically-operated isolation valves and one check valve in series.

Two cross-connections have been provided between the B loop reactor building supply line and the service water supply lines to nonessential reactor and turbine building components. These components normally receive cooling water from the A loop. The two cross-connections allow the B loop to supply these loads whenever the A loop is unavailable during unit outages or for periodic inservice pressure testing required by ASME Section XI (Section 6.6). In addition, a cross-connect exists between two nonessential discharge headers downstream of the outlet from the nonessential loads in the reactor and turbine buildings, which may be used to direct flow through the available loop discharge piping. Each of the cross-connections contains a normally closed manual isolation valve.

The high-pressure core spray (HPCS) diesel generator (Division III) and associated switchgear room unit cooler receive a redundant supply of cooling water from either A or B loop of the SWP system. Service water to these components also returns through either A or B loop. The SWP system is sized so that if one loop is lost, the remaining loop is capable of providing necessary cooling water flow through these components.

Service water to each control building chilled water condenser contains a recirculation loop with a pump (2SWP*P2A and 2SWP*P2B). These pumps work in conjunction with the temperature control valves located in the chiller return lines to maintain the service water inlet temperature to the chillers above 53°F.

Each of the SWP return lines from the RHR heat exchangers has radiation monitors to detect any radioactivity resulting from a postulated tube leak. Radiation monitors are also provided on each of the service water discharge headers to the discharge bay (Section 11.5).

During normal plant operation and the initial phase of normal unit shutdown, four of the six SWP pumps are required to satisfy the plant heat load. The remaining pumps are spares to accommodate periodic maintenance.

The SWP system at Unit 2 is a once-through system which utilizes raw lake water from Lake Ontario. Since fall of 1989, zebra mussels were found in Lake Ontario. Most recently the mollusks have been confirmed at Unit 1 and Unit 2. Zebra mussels are capable of rapid reproduction causing plugging and serious damage to plant equipment if not treated; therefore, a service water raw water treatment program has been implemented. This is consistent with NRC Information Notice 89-76 and Generic Letter 89-13.

The biocide is stored in storage containers located onsite. The laydown area for the storage hoppers and pumps is enclosed in a portable environmental berm designed to retain the entire volume of the chemical. The biocide is pumped from the storage hoppers and added into the lake water inlet piping in the screenwell just north of the trash rakes, or directly into the intake structure. The intake structure injection may be done from shore via temporary piping or from boat or barge using temporary piping attached to the heater bars. Both methods ensure adequate mixing of the biocide.

During the treatment, the treated water is pumped through the systems taking feed from Lake Ontario water using the system pumps (e.g., service water, fire water, screenwash, fish jet pump system), and by the SWP system makeup to the CWS and the chilled water (HVK) system. The chemical is fed through the service water side of the following components/systems provided they are in service during the treatment:

1. Standby diesel generator cooling system.

2. RHR system heat exchangers.

3. Various unit coolers.

4. Control building HVK chillers.

5. Condensate air removal system (ARC) heat exchangers.

6. Auxiliary boiler system heat exchangers.

7. Lithium bromide cooling system heat exchangers.

8. TBCLC system heat exchangers.

9. RBCLC system heat exchangers.

The addition of the chemically-treated water through all these systems is controlled through the use of approved procedures. The length of time of the treatment varies inversely with the temperature of the water being treated. Typical treatment durations are 8 to 48 hr depending on lake water temperature.

The service water chemical treatment (SCT) system is used for the storage and injection of a biocide (sodium hypochlorite and sodium bromide) and a detoxicant (sodium bisulfite). The biocide is injected into the SWP intake bay, just downstream of the traveling screens. The detoxicant, which is used to neutralize the biocide, is injected into the SWP discharge lines just upstream of the discharge bay.

The SCT system consists of three pump skids which draw off of three storage tanks, and a local control panel located in the acid storage area (northwest corner) of the screenwell building.

Carrier water is used to provide prompt delivery of the biocide and detoxicant to the appropriate points in the SWP system. Automatic control provisions allow for a preset treatment schedule.

In addition, a SWP corrosion monitoring station is installed near the trash rakes in the screenwell building. The Station allows for sample coupon analysis of the intake and discharge streams to monitor the performance of the biocide treatment. The monitoring station includes a pump which draws pretreated service water from the intake channel. Treated service water is provided off a tie-in into the nonsafety-related portion of the SWP system. The sample water is then run through a closed monitoring loop which includes sample coupons of representative system materials and sessile beads.

The SWP system is designed with a margin for the following allowances:

1. Fouled pipe.

2. Heat transfer fouling factors.

3. Pressure drop across fouled equipment.

4. Piping and component corrosion allowance.

5. Margin for minor component leakage.

To detect leakage the system has the following provisions:

1. Low flow alarms.

2. Area leakage alarms.

These alarms alert the Control Room Operator of leakage. The Operator will then take appropriate action to investigate the leakage.

9.2.1.3 Safety Evaluation

The SWP system provides cooling water to essential components through two separate redundant supply and return headers. Each supply header provides sufficient cooling water for the following minimum components essential for the safe shutdown of the reactor and mitigates the consequences of a design basis LOCA:

1. One RHR heat exchanger and its associated RHR pump seal cooler.

2. One division of the reactor building, control building, diesel generator building, and SWP pump bay unit space coolers.

3. One standby diesel generator and the HPCS diesel generator, and their associated unit space coolers.

4. One control building chilled water condenser.

5. One reactor building ventilation recirculation cooling coil.

6. One DBA hydrogen recombiner.

7. One spent fuel pool heat exchanger.

8. Emergency makeup to the spent fuel pool.

A LOCA with offsite power available and a trip (single failure) of one of the four operating SWP pumps is the limiting design basis for the SWP system. This event is limiting because the SWP supply header cross-tie valves do not close automatically and the nonsafety-related components/systems are not isolated from the SWP system. During the initial phase of recovery from a LOCA, three SWP pumps satisfy cooling requirements of the above components except the RHR heat exchanger. When this heat exchanger is required, either the SWP flow to the turbine building nonessential loads is isolated or an additional SWP pump, if available, is used.

Following a LOOP or a LOCA coincident with a LOOP and a single failure of the Division 1 or Division 2 emergency diesel generator, the operating SWP pumps trip, the SWP cross-tie isolation valves close automatically, and the nonsafety-related components/systems required for normal operation are isolated from the SWP system. In this case, during the initial phase of recovery from a LOCA, one SWP pump satisfies cooling requirements

of the above components except the RHR heat exchanger. When this heat exchanger is required, two SWP pumps are used.

Essential piping and components are located in Category I structures. These structures protect the SWP system from adverse environmental occurrences including SSE (Section 3.7) and tornadoes (Section 3.3). In the Unit 2 SWP system, there is no Category I pipe buried in the soil. Therefore, there is not a concern of failure of safety-related buried pipe resulting from soil erosion caused by failure of nonsafety-related piping. Two redundant loops of the SWP system have sufficient protection and physical and electrical separation to ensure that one loop is always available to permit safe shutdown of the plant in the event of:

1. Flooding (Section 3.4).

2. Pipe whip and jet impingement resulting from line breaks (Section 3.6).

3. Missiles that may result from equipment failures or tornadoes (Section 3.5).

4. Fire (Section 9.5.1 and Appendix 9A).

Originally, the FMEA of the SWP system was contained in the Unit 2 FMEA document, which is historical. FMEAs for plant systems are now performed and controlled by the design process.

No single-active failure coincident with a LOOP can prevent the SWP system from achieving its intended safety function.

The use of the system manual supply cross-connect valves during unit outages bypasses the automatic isolation of nonessential loads. This is considered to be an acceptable level of safety provided: a) the associated flow path provides for the stoppage of flow at either the inlet or outlet of the nonessential loads (leaving only an open pressure boundary), and b) manual valves 2SWP*V17 and 2SWP*V32 (if open) are closed by an Operator stationed at the valves following any indication of a LOOP or a compromise in the nonessential header piping.

Section 3.7.1 of the TRM describes specific requirements for plant shutdown conditions when handling irradiated fuel assemblies in the secondary containment.

9.2.1.4 Testing and Inspection Requirements

Essential portions of the SWP system are designed to permit periodic inspection and pressure testing to ensure system integrity. In-service inspection (ISI) design criteria are provided in Section 6.6. Operability of the SWP pumps can be demonstrated during normal plant operation. The spare SWP pumps (four are required during normal plant operation) will be cycled

periodically to ensure their availability. Also, all essential power-operated valves will be cycled periodically to ensure operability.

The SWP system is designed to permit periodic functional testing to verify system capability and operability. Sufficient flow and pressure instrumentation provides verification that each SWP pump operates within design limits. Flow elements and/or test taps are provided at the inlet and outlet of cooled components to verify that the cooling water passages remain unrestricted and components receive their design flow rate. Throttle valves in the return line from each component provide for system balancing.

9.2.1.5 Instrumentation Requirements

Description

Safety-related instruments and controls are provided for automatic and manual control of the SWP system. Except where noted, the controls and monitors described below are located in the main control room. The control logic is shown on Figure 9.2-2. See TRM Section 3.3.9 for requirements regarding specific instrumentation.

Operation

The six service water pumps are controlled automatically or manually by the required combination and sequence of signals generated by LOOP, automatic load sequencing, and prior pump status. On LOOP, all running pumps are stopped and one pump per division is restarted automatically in timed sequence. If a running pump fails to restart in time, a standby pump is started automatically to assure one pump is running in each loop. Interlocks prevent a pump from starting unless the associated discharge valve is fully closed.

The service water pump discharge valves open or close automatically when the associated pump is started or stopped, respectively. The valves can also be opened and closed manually.

The service water pump discharge header isolation valves close automatically when there is a full LOOP. The valves remain open on a partial LOOP (i.e., two pumps trip). The valves can also be opened and closed manually.

The service water strainers are backwashed automatically when the differential pressure is high or when a preset time has elapsed. A backwash can also be manually initiated from a local panel. Backwash will be actuated when the control room fire disconnect switch is actuated.

The supply and return isolation valves for service water to the spent fuel pool, RHR system pump seal coolers, HPCS pump room unit coolers, and RHR system heat exchangers are opened and

closed manually. The SFC system makeup valves are controlled by key-operated switches.

The isolation valves that separate the safety-related portions of the SWP system from the nonsafety-related portions close automatically upon both a full or partial LOOP. The valves can also be opened and closed from the control room, and the Division I valves can be closed from the remote transfer panel in the event of a control room fire. The flow control valves to the CWS may be stopped at any intermediate point between full open and full closed by selecting the control point on their respective controllers.

Valves for service water to the Division I and II emergency diesel generator coolers open automatically when the associated diesel generator is started and normal water header pressure is established within a preset time. The valves close automatically when the diesel generator is stopped or water header pressure is less than normal. The valves can also be opened and closed manually.

The Division I and II service water supplies to the HPCS (Division III) diesel generator each have two valves in series; the supply valves are normally open, and the return valves are normally closed. With the control switch in the AUTO position, the return valve from the selected division will open automatically when the diesel generator is started. The return valve will close automatically when the diesel generator is stopped. In addition, the supply valve will close automatically if normal water header pressure is not established after a preset time. Supply and return valves can also be opened or closed manually.

The SWP isolation valves to the control building chilled water condensers open and close automatically whenever their associated control building HVK system pump starts or stops, respectively. These valves can also be opened and closed manually.

The control building chilled water condenser service water outlet valves are controlled automatically by the associated condenser service water outlet temperatures. The temperatures are set by manual/automatic control stations. The valves can also be controlled manually. The condensing water pumps are controlled automatically by the condenser inlet water temperature. The pumps can also be controlled manually.

The service water isolation valves for the reactor recirculation system (RCS) pumps (seal, bearing, and motor-winding coolers) supply and return are locked closed.

The vacuum pump seal water cooler valves are opened and closed manually.

In those cases where air-operated isolation valves are provided in the return line from safety-related unit space coolers, the valves are normally closed. These valves automatically open whenever their associated local space temperature rises above a preset level.

Service water to the reactor building ventilation recirculation cooling coils is normally isolated. The isolation valves for these components open automatically whenever the associated cooling coil fan is energized and the space air temperature exceeds a preset level.

Monitoring

Indicators are provided for:

1. Each service water pump suction pressure.
2. Each service water pump discharge pressure.
3. Each diesel generator jacket water cooler flow.
4. Service water flow to each RHR heat exchanger.
5. Each service water supply header temperature.
6. Each service water pump discharge flow.
7. Each service water header flow to lake.
8. Each service water header flow to CWS.
9. Each service water supply header pressure.
10. Each RHR heat exchanger service water return radiation level.
11. Each service water discharge loop radiation level.
12. Service water return temperature from each component (LOCAL).

Recorders are provided for:

1. Each RHR heat exchanger service water return radiation level (Section 11.5.1.1.1, Item 5).
2. Each service water discharge loop radiation level (Section 11.5.1.1.1, Item 6).

Alarms are provided for:

1. Service water system inoperable.

2. Service water pumps suction pressure low.

3. Service water pumps autostart.

4. Service water pumps motor/pump bearing temperature high.

5. Service water pumps autotrip/fail to start.

6. Service water pumps discharge flow low.

7. Service water supply header pressure low.

8. Spent fuel pool makeup valve not closed.

9. Service water to TBCLCW heat exchangers pressure low.

10. Service water to RBCLCW heat exchangers pressure low.

11. Emergency diesel generators service water discharge pressure low.

12. Emergency diesel generators service water flow low.

13. Control building chilled water chiller condensing water flow low.

14. Chiller condensing water pump suction pressure low.

15. Service water strainers differential pressure high/high.

16. Effluent liquid (SWP discharge) radiation monitor activated.

17. Process liquid (service water return from RHR heat exchangers) radiation monitor activated.

18. Service water pumps motor/feeder electrical fault.

19. Service water pumps motor overload.

20. Service water valves motor overload.

21. Service water strainer motor overload.

22. Chiller condensing water pumps motor overload.

23. Radiation monitoring trouble/manual out of service.

9.2.2 Reactor Building Closed Loop Cooling Water System

The RBCLCW system provides cooling water to reactor auxiliary system equipment and accessories during normal plant operating

conditions. The RBCLCW system is not required to operate during emergency or faulted plant conditions. However, during this condition, portions of the system provide a Category I pressure boundary for backup cooling from the SWP system to cool the SFC heat exchangers and RHR pump seal coolers. The RBCLCW system is shown on Figure 9.2-3. Auxiliary equipment cooled by this system is listed in Table 9.2-3. Major RBCLCW system components are described in Table 9.2-4.

9.2.2.1 Design Bases

The RBCLCW system is designed to comply with the following design bases:

1. The RBCLCW system is designed to remove heat from various auxiliary equipment housed in the reactor building and turbine building during normal plant operation. The RBCLCW system is cooled by the SWP system, and makeup water is supplied from the MWS system.

2. In the event of a mechanical failure to the RBCLCW system or upon loss of offsite electrical power, the essential components that are normally cooled by the RBCLCW system will be automatically and/or remote manually isolated from the RBCLCW system, and will be connected to the safety-related SWP system.

3. The major components of the RBCLCW system, including pumps, piping, heat exchangers, expansion tank, and valves, do not perform a safety function. These components are classified as Safety Class 4. These portions of the RBCLCW system are designed to ANSI, TEMA, HIS, NEMA, and ASME codes and standards as applicable.

4. Safety-related portions of the RBCLCW system are designed to Category I and Safety Class 2 or 3 criteria to ensure compliance with 10CFR50 Appendix A, GDC 2, 4, 5, 44, 45, 46, 54, and 56, and RG 1.26, 1.29, 1.46, and 1.53 as applicable. Figure 9.2-3 shows the safety-related portions of the RBCLCW.

5. Isolation between nonnuclear safety and safety-related portions of the RBCLCW system is provided by remote manually-actuated motor or air-operated valves (AOVs).

6. During normal plant operation, the RBCLCW system provides an intermediate barrier between systems containing radioactive products and the SWP system, which transfers RBCLCW heat load to the ultimate heat sink (UHS). This design precludes a direct release of radioactive products into the environment.

7. Surveillance of system radioactivity is provided by continuous monitoring of the return line from the system heat exchangers.

9.2.2.2 System Description

The RBCLCW system is a closed loop system that provides cooling to auxiliary equipment located in the primary containment, reactor building, and turbine building.

The RBCLCW system consists of a primary loop with three 50-percent capacity main cooling water pumps, three 50-percent capacity booster pumps, three 50-percent capacity heat exchangers, one expansion tank, piping, valves, and instrumentation. A secondary loop dedicated to cooling the instrument air compressors is provided with two 100-percent capacity pumps, two 100-percent capacity heat exchangers, one expansion tank, piping, valves, and instrumentation. The secondary loop rejects heat to the primary loop. RBCLCW water is circulated through the shell side of the RBCLCW heat exchangers, where the system heat load is transferred to the SWP system, which flows through the heat exchanger tubes. The system capacity is based on the maximum heat loads that can occur during normal plant operations. The system is designed to remove 74×10^6 Btu/hr from the components listed in Table 9.2-3. Normally, a combination of any two main pumps, two booster pumps, and two heat exchangers is capable of providing this maximum heat removal capacity while maintaining a cooling water temperature of about 90°F with service water temperature up to about 72°F. With the service water temperature above 72°F, it may be necessary to shed/reduce system heat loads or place the standby heat exchanger in service.

The RBCLCW system is not a nuclear safety-related system and is not required to operate to assure the safe shutdown of the plant. However, some components and portions of the RBCLCW system pressure boundary are needed for post-accident operation. Those components that are needed to operate during this plant condition are identified in Table 9.2-3. The SFC heat exchangers and the RHR system pump seal coolers are capable of remote manual alignment to the SWP system from the control room. The need to remote manually isolate the RBCLCW and connect the SWP system to the SFC heat exchangers is determined by fuel pool temperature. The RHR pump seals receive cooling water from the SWP system prior to starting the RHR pumps. The components, piping, valving, and instrumentation needed to ensure such isolation capability are nuclear safety related.

Equipment that is cooled by the RBCLCW system during normal operation includes:

1. SFC heat exchangers.

2. RWCU nonregenerative heat exchanger.

3. RWCU pump bearings, coolers, seal jackets, and pedestals.

4. Reactor building equipment drain coolers.

5. Drywell equipment drain cooler.

6. Drywell unit space coolers.

7. Reactor recirculation pump seal coolers, motor winding coolers, and motor bearing coolers.

8. Instrument air compressors.

9. RHR pump seal coolers.

10. CRD pump bearing coolers and speed increasers.

11. Reactor plant sample panel.

12. Reactor recirculation sample cooler.

13. RHR sample coolers.

14. SFC and RWCU off-line radiation monitors.

15. Post-accident piping station panel.

During normal plant operations, two main pumps, two booster pumps, and two heat exchangers are in operation, and the third main pump, booster pump, and heat exchanger are in standby.

Makeup water for the RBCLCW system is supplied from the demineralized water storage tank by the makeup water transfer pumps to the expansion tanks. The expansion tanks serve as surge and makeup tanks for the system. The RBCLCW system expansion tanks' levels are automatically controlled. The expansion tanks accommodate system volume changes resulting from coolant expansion and contraction and are physically located above the RBCLCW pumps to provide the pumps with sufficient NPSH.

In the event of a loss of cooling, the SFC heat exchangers and the RHR pump seal heat exchangers, normally supplied by the RBCLCW, can be supplied with cooling water by means of parallel supply and return connections from the SWP system headers.

The RBCLCW supply and return headers are connected to the SWP system through normally closed isolation valves. The safety-related RBCLCW supply headers are isolated from the nonsafety portion of the RBCLCW system by means of check valves (one in each header) in series with normally open motor- or air-operated isolation valves. The safety-related return headers

are isolated from the nonsafety-related portion of the RBCLCW system by a motor- or air-operated isolation valve.

RBCLCW system temperature is controlled by bypassing part of the component cooling water flow around the RBCLCW heat exchangers. A temperature control valve is provided to regulate flow through the heat exchangers to maintain the desired temperature. The temperature control valve nominal setpoint is about 85°F. The RBCLCW temperature range, with plant in operation, is between 75°F to 90°F. A normally closed, manually-operated butterfly valve is provided to bypass the temperature control valve in the event of control valve malfunction.

The standby RBCLCW system pump starts automatically on low pressure in the pump discharge header.

System components and piping include corrosion-resistant materials or sufficient wall thickness to preclude degradation of system performance by long-term corrosion. No chemical additives are used to inhibit corrosion, as sufficient corrosion allowance is provided on all equipment and piping.

Thermal relief valves are provided on all equipment that might be overpressurized due to thermal expansion of fluid when equipment is isolated. The shell side of the SFC heat exchangers is protected from overpressure by relief valves based on tube failure criteria.

Radioactivity in the RBCLCW system is continuously monitored in the return line from the RWCU nonregenerative heat exchanger and in the branch return header that serves the SFC heat exchangers.

In addition to the normal modes of operation of the RBCLCW system, connections have been added to the inlet and outlet piping inside of the drywell, and piping has been added through the reactor building wall, such that a temporary source of chilled water can be supplied utilizing hoses connected to this piping for control of the drywell ambient air temperature during plant outages. The source of the chilled water will be a skid-mounted chiller brought on site each outage and located outside of the reactor building. In this case, the RBCLCW temperature is allowed to go below 75°F. The operation of the chilled water unit is administratively controlled as to maintain the drywell area temperature above the minimum requirements. Secondary containment will be assured in the event of a hose break by use of redundant spring-activated check valves inside of the reactor building at each penetration.

During normal plant operation, secondary containment integrity will be maintained by safety-related blind flanges on the outboard side of the reactor building penetrations.

9.2.2.3 Safety Evaluation

Those essential portions of the RBCLCW system that are required to function during accident conditions are designed to provide cooling for safety-related components in combination with a single active or passive failure.

The power supply for the safety-related components in the RBCLCW system is provided from the redundant emergency standby power source during LOOP. Essential equipment receives cooling from the safety-related redundant SWP system. The SWP system is completely redundant and powered from separate emergency buses (Section 8.3).

During normal plant operating conditions, the RBCLCW system is capable of removing the design basis heat load with the use of any two main pumps, booster pumps, and heat exchangers. During transient or accident conditions, the safety-related SWP system will remove the design basis heat load from those components discussed in Section 9.2.2.2, with the exception of the reactor recirculation pump seal coolers, motor winding coolers, and motor bearing coolers. The standby pump and heat exchanger provide redundancy in the event of a pump or heat exchanger failure.

A radiation monitor RE115 is provided in the return header of the system which serves both SFC heat exchangers. The monitor is placed downstream of the heat exchanger return branch lines that connect into the return header. An additional radiation monitor is provided in the return header of the RWCU nonregenerative heat exchanger.

Leakage into the RBCLCW system is detected by the following:

1. Expansion tank level increase.

2. Flow measurement in the RWCU nonregenerative heat exchangers return line.

3. Temperature detection in the return lines of components.

4. Radiation monitors (Section 11.5).

5. Conductivity analysis.

Leakage out of the RBCLCW system is detected by the following:

1. Excessive makeup cycles to the expansion tank.

2. Expansion tank level alarms.

3. Flow indication from the recirculation pumps and motors.

4. Differential flow measurement across the drywell unit space coolers.

The RBCLCW system serves as intermediate pressure boundary between components that could contain radioactive fluids and the SWP system. The design feature mitigates the possibility of a direct release to the plant environment during normal plant operation. Should leakage occur from the RBCLCW system to the SWP system during normal operation, the additional feature of radiation monitoring in the service water effluent is provided to detect any possible radiation leaks.

The portions of the RBCLCW system that penetrate the primary containment have motor-operated isolation valves that are capable of remote manual operation from the control room and will close automatically during an accident condition. The location and number of isolation valves comply with the requirements of GDC 56.

Major components of the RBCLCW system are housed within a Category I structure that provides protection from earthquake (Section 3.7), tornado (Section 3.3), and flooding (Section 3.4). Those portions of the RBCLCW system that are required to operate during accident conditions are additionally protected from the effects of internally-generated missiles (Section 3.5), pipe whip, and jet impingement (Section 3.6).

The transfer of cooling water from the RBCLCW system to the SWP system can occur remote manually for the SFC heat exchangers and RHR pump coolers. The presence of this service water crossover capability and the instrumentation used to actuate and/or monitor the need for crossover, in conjunction with the system redundancy, protection from adverse internal and external environmental hazards, and the high reliability of the SWP system assure that the safety-related requirements are satisfied.

Originally, a FMEA for the RBCLCW system was contained in the Unit 2 FMEA document, which is historical. FMEAs for plant systems are now performed and controlled by the design process.

9.2.2.4 Testing and Inspection Requirements

Initial construction tests such as hydrostatic leak tests are conducted in accordance with the applicable code requirements. Tests for initial system flow distribution, valve operability, alarm setpoints, and instrumentation and control loop checks will be done in accordance with the preoperational test program (Chapter 14).

Heat exchanger performance is observed during normal plant operations. Availability of the RBCLCW standby pump will be tested periodically. Pump performance in the RBCLCW system is assessed during normal plant operations.

Service water intertie valves to the SFC heat exchangers and RHR pump seal coolers will be tested periodically.

9.2.2.5 Instrumentation Requirements

Description

Instruments and controls are provided for manual and automatic control of the RBCLCW system. The controls and monitors described in the following paragraphs are located in the main control room. The control logic is shown on Figure 9.2-4.

Operation

The RBCLCW (standby) main pump starts automatically when the pump discharge header pressure is low or an operating pump has a motor electrical fault, provided standby pump suction pressure is not low. An operating pump trips automatically when the pump suction pressure is low. The pumps can also be started and stopped remote manually.

The operation of the booster pumps is similar.

Temperature controls maintain system heat exchanger outlet water temperature automatically by a water temperature control valve, so as to vary the proportion of water that bypasses the heat exchangers.

The RBCLCW expansion tank control valves permit makeup water to fill the tanks. The control valves open and close based on the water level monitored in the tanks.

The RHR pump seal cooler supply and return valves, drywell cooler block valves, and SFC heat exchanger supply and return valves are opened and closed remote manually.

RCS pump/motor cooler containment isolation valves close automatically when there is a LOCA signal present. These valves can also be opened (in the absence of a LOCA signal) and closed remote manually.

Drywell unit cooler containment isolation valves close automatically when both a LOCA signal and LOCA override are not on or a remote manual isolation signal is present. These valves can also be opened (in the absence of a LOCA or manual isolation signal) and closed remote manually.

Monitoring

Indicators are provided for:

1. Closed loop cooling water system heat exchanger discharge temperature.

2. RBCLCW pumps discharge header pressure.

3. RBCLCW booster pumps discharge header pressure.

A recorder is provided for RBCLCW flow to drywell unit coolers.

Alarms are provided for:

1. RBCLCW system trouble.
2. RBCLCW pump P1A, P1B, P1C motor overload.
3. RBCLCW pump P1A, P1B, P1C discharge pressure low.
4. RBCLCW pump P1A, P1B, P1C fail to start/autotrip.
5. RBCLCW pump P1A, P1B, P1C autostart.
6. RBCLCW booster pump P3A, P3B, P3C motor overload.
7. RBCLCW booster pump P3A, P3B, P3C discharge header pressure low.
8. RBCLCW booster pump P3A, P3B, P3C fail to start/autotrip.
9. RBCLCW booster pump P3A, P3B, P3C autostart.
10. RBCLCW to RWCU flow high.
11. RBCLCW to reactor recirculation pump coolers pressure low.
12. RBCLCW containment/recirculation pump cooler valves motor overload.
13. Drywell unit coolers leakage high.
14. RBCLCW from drywell unit coolers temperature high.
15. RBCLCW to drywell unit coolers isolation valve motor overload.
16. RBCLCW isolation valves inoperable.
17. RBCLCW to RHR pump seal cooler pressure low.
18. RBCLCW to spent fuel pool cooling heat exchanger supply/return valves motor overload.
19. Process liquid radiation monitor activated.
20. RBCLCW to drywell unit cooler Division I and Division II containment isolation valves LOCA override.

9.2.3 Makeup Water Treatment System

9.2.3.1 Design Bases

9.2.3.1.1 Safety Design Bases

The makeup water treatment (WTS) system is not required to effect or support the safe shutdown of the reactor or to perform in the operation of reactor safety features.

9.2.3.1.2 Power Generation Design Bases

The WTS system is designed to remove dissolved and suspended solids from raw lake water to produce the demineralized water quality described in Table 9.2-5. The WTS system has the capability of producing 220,000 gpd of demineralized water based on mean values of inlet impurity and 190,000 gpd based on maximum values of inlet water impurity. These mean and maximum inlet impurities are listed in Table 9.2-6.

Pressurized tanks, including water-treating filters and ion exchangers, meet the design requirements of ASME Boiler and Pressure Vessel Code, Section VIII, Division 1. All piping and valves in the system are designed in accordance with ANSI B31.1.

The acid and caustic day tanks and caustic storage tank are removed from service in accordance with New York State regulations for tank closures. The demineralizer water treatment portion of the WTS system is not functional. A mobile self-contained demineralizer unit is on site to process plant makeup water from the domestic water supply.

9.2.3.2 System Description

The WTS system is shown on Figure 9.2-5. Raw lake water, supplied to the WTS system by the SWP system, is passed through a single train of filters and ion exchangers where both organic and inorganic materials are removed to produce the required quality of demineralized water. The WTS system supplies all the water requirements initially used in the power plant, with the exception of service water and circulating water. Demineralized water is used during plant operation as makeup or washdown by various systems in the turbine, reactor, and radwaste buildings.

Service water is supplied to two fiberglass waste water recovery tanks. Each tank has a 30,000-gal capacity. Two filter pumps, one on standby, take suction from the waste water recovery tanks and deliver 310 gpm to the filtered water storage tank, a 30,000-gal fiberglass tank, through a water-treating filter. This filter is made up of graded anthracite which removes suspended solids from the water. Provisions have been made for backwashing the filter.

Water is then pumped by the filtered water transfer pumps at 210 gpm from the filtered water storage tank through the activated

carbon filter. Filtered water is also pumped by the circulating water seal pumps at 40 gpm to the circulating water pump seals and various users throughout the plant. Water from the city of Oswego may also be used as a source of water to the filtered water storage tanks.

The activated carbon filter, which removes suspended solids, is made up of a graded anthracite supporting bed topped with a carbon filter material. The filter also can be backwashed. The backwash flow from both the anthracite and activated carbon filters is directed to the sewage treatment plant via underground piping.

Water passing through the activated carbon filter enters the ion exchanger train consisting of weak and strong acid cation exchanger demineralizers, forced draft degasifier, degasified water pumps, weak and strong base anion exchanger, and mixed bed ion exchanger demineralizer. Table 9.2-7 lists the physical characteristics and capacity of each ion exchanger.

The filters and weak and strong acid cation vessels are designed for an internal pressure of 100 psig. The weak and strong base anion vessels and mixed bed exchangers are designed for an internal pressure of 125 psig. All the pressurized vessels are hydrostatically tested at 1 1/2 times their design pressure. The vessel internals are made of corrosion-resistant materials with adequate distribution of flow and collection without channeling and bypassing.

The system includes a forced-draft degasifier to remove carbon dioxide (CO_2) from the system. The degasifier is located between the cation and anion exchangers. The CO_2 is vented to the atmosphere through a roof vent. The decationized water is repressurized by the degasified water pumps located downstream of the degasifier before passing through the weak and strong base anion vessels and the mixed bed exchanger. The demineralized water is then supplied directly to the two demineralized water storage tanks located in the screenwell building. These tanks constitute part of the MWS system.

Upon depletion of exchanger capacity, the resins are chemically regenerated. The regeneration process is automatic following manual initiation. The anion and mixed bed exchangers are regenerated with sodium hydroxide (NaOH). NaOH is pumped to the exchangers from the caustic day tank. Dilution of the NaOH is accomplished by mixing makeup water supplied from the caustic dilution water tank with the caustic. Separate pumps are used to pump caustic to the mixed bed exchanger and the strong base anion exchanger. The caustic solution is directed to the weak base anion exchanger following regeneration of the strong base anion exchanger. The cation and mixed bed ion exchangers are regenerated with sulfuric acid (H_2SO_4). H_2SO_4 is pumped to each ion exchanger from the acid day tank by pumps associated with the

specific exchanger. The acid is diluted with makeup water prior to regeneration.

Regeneration frequency during plant startup is anticipated to be once per day or after producing approximately 200,000 gal of makeup water. Regeneration frequency during plant operation is expected to be infrequent (approximately twice per month).

Following regeneration, the acid and caustic solutions are discharged to the waste neutralizing tank. Acid or caustic is added as required to neutralize the solution prior to discharge to the lake through the SWP system. Drains and overflows from the various system equipment are collected in the 8,500-gal waste sump. The contents of the waste sump are also pumped to the waste neutralizing tank. The waste recirculation and transfer pumps take suction from the waste neutralizing tank, and recirculate the flow until the waste is neutralized and then sampled and, if within allowable limits, pumped to the service water discharge.

The caustic day tanks of both the makeup water treatment system and the condensate demineralizer (CND) system (Section 10.4.6) are supplied from a tank truck-filled storage tank located in the screenwell building. The acid day tanks of both systems are supplied from the acid storage tank within the acid feed system.

The demineralized water storage tanks, which are part of the MWS system (Figure 9.2-6), include a fill and supply connection for demineralized water with Unit 1. This assures demineralized water availability should the Unit 2 water treatment system need to be secured for maintenance or other purposes. The two demineralized water storage tanks each have a 30,000-gal capacity and are fabricated of fiberglass.

The demineralized water storage tanks supply makeup quality water to the power cycle and various plant closed systems. Demineralized water transfer pumps are rated at 200 gpm, take suction from the storage tanks, and supply water to the following:

1. Condensate storage tanks, makeup.
2. Reactor and turbine building closed loop cooling water systems, makeup.
3. Auxiliary boiler makeup, chemical feed, and contact condenser.
4. Chemical mixing.
5. Stator cooling water makeup.
6. Refueling floor service boxes, supply.

7. RHR heat exchanger makeup and layup.

8. Decontamination cask washdown.

9. CRD hose stations, maintenance area.

10. Reactor, turbine, and radwaste building sampling systems, dilution.

11. Standby liquid control system storage tank, makeup.

12. Instrument flushing water.

13. Water treatment system backflushing.

14. Post-accident sampling, dilution, flushing, cooling.

15. Radiation monitor purge water.

16. Hydrogen recombiner cooling, backup for functional testing.

17. Decontamination rooms, laboratory sinks, repair shops.

18. Condensate filtration system (CFS) backwash air compressor makeup supply.

9.2.3.3 Safety Evaluation

The acid and caustic tanks and the waste sump are constructed from corrosion-resistant materials. In the event of an acid or caustic tank rupture, the discharge is directed to a chemical-resistant sump. In this event, the necessary valving exists to isolate the system. All waste from the WTS system is neutralized and sampled prior to discharge to the lake.

9.2.3.4 Test and Inspection

Field tests are performed after equipment installation to verify satisfactory operation and functioning of control equipment, as well as to demonstrate proper system performance. The major components in the system are hydrostatically tested by the vendor prior to installation. Grab samples are periodically tested in the laboratory to verify filter and demineralizer performance and to ascertain stored water quality. System redundancy and routine inspection and maintenance provide for reliability.

9.2.3.5 Instrumentation Requirements

<u>Description</u>

Instruments and controls are provided for automatic and manual control of the demineralized MWS system. The controls are located on local panels. Except where noted, the monitors are

located on local panels. The control logic is shown on Figure 9.2-7.

Operation

The waste recovery tank inlet valve opens or closes automatically when the tank level is low or high, respectively. The valve can also be controlled manually.

The filter pumps start or stop automatically when the filtered water storage tank level is low or high, respectively. The pumps trip when the waste recovery tank level is low-low. The pumps can also be controlled manually.

One circulating water seal water pump starts automatically when the redundant pump is not running.

The pumps stop automatically when the filtered water storage tank level is low-low or the discharge header pressure is sustained low. The pumps can also be controlled manually. Logic controls prevent repeated pump auto start after it is stopped for low discharge pressure.

The filtered water transfer pumps start automatically when the demineralized water storage tank level is low. The pumps stop automatically when the filtered water storage tank level is low. The pumps can also be controlled manually, but low level in filtered water storage tank prevents starting the pumps, or stops them if they were running.

After manual initiation, regeneration of the demineralizers is controlled automatically.

The demineralized water transfer pump starts automatically when the discharge header pressure is low. The pumps stop automatically when the suction pressure is sustained low. The pumps can also be controlled manually. The post-accident flush valve is controlled manually.

The waste recirculation and transfer pump starts automatically when the neutralizing tank water level is high. The pumps stop automatically when the neutralizing tank water level is low. The pumps trip when the neutralizing tank water level is high and the recirculation flow is sustained low. The pumps can also be controlled manually. Logic controls prevent repeated pump autostart after it is stopped for low flow.

The waste discharge valve is opened manually. Administrative controls are in place to ensure that pH is within the required range prior to discharging the waste neutralizing tank effluent to either the service water discharge tunnel or to the site sewage treatment facility. Interlocks prevent opening the valve when neither waste recirculation and transfer pump is running.

The valve closes automatically when neither waste recirculation and transfer pump is running. The valve can also be closed manually. In addition, 2WTS-V540 will be normally closed. Procedural controls will prevent having 2WTS-V540 open when pH levels are not acceptable.

The neutralizing tank recirculation valve opens automatically when the discharge valve is closed and when any one of the waste recirculation and transfer pumps is running. The valve closes automatically when the discharge valve is open or when neither waste recirculation transfer pump is running. The valve can also be closed manually.

The sump pumps are started manually if sump water level is high. Interlocks prevent starting a pump when the waste neutralizing tank level is high-high and discharge valve is open. The pumps stop automatically when the sump level is low, the waste neutralizing tank level is high-high, or the waste discharge valve is open.

Makeup and condensate demineralizer caustic transfer pumps can be started manually. Either of the two pumps can be selected for autostart operation. The pump starts automatically when the caustic day tank level becomes low. It stops when the tank level becomes high or when the caustic storage tank level becomes low. The pump also stops when there is sustained low discharge pressure of the pump when it is running. Control logic is provided to prevent repeated autostart of the pumps after stop because of low discharge pressure. The caustic storage tank electric heater is controlled by a thermostat.

Monitoring

Indicators are provided in the main control room for:

1. Demineralized water storage tank level.

2. Demineralized water transfer pump flow.

Alarms are provided in the main control room for:

1. Demineralized water storage and transfer system trouble.

2. Makeup demineralizer pretreatment filter system trouble.

3. Makeup demineralizer treating system trouble.

4. Acid chemical feed system trouble.

9.2.4 Domestic Water and Sanitary Drains and Disposal Systems

The domestic water system and sanitary drains and disposal systems are schematically shown on Figures 9.2-8 and 9.2-9, respectively.

9.2.4.1 Design Bases

Domestic Water System

The system is designed to provide sufficient domestic water from an existing city main to satisfy the quantity and pressure requirements of all installed plumbing fixtures. The quality of water supplied to each fixture is in accordance with Chapter 4, State Building Construction Code Applicable to Plumbing, Bulletin No. 23, New York State.

Sanitary Drains and Disposal System (Sanitary Waste Treatment Facility)

The sanitary drain and disposal system is designed to treat and dispose of the waste from all plumbing fixtures, except lavatories, sinks, and drains containing waste that is contaminated or potentially contaminated with chemicals or radioactivity. Such contaminated or potentially contaminated waste is physically segregated from the sanitary drains and disposal system and is connected to the floor and equipment drainage systems (Section 9.3.3).

Sanitary waste from Unit 1 and Unit 2 is treated at a combined sanitary waste treatment plant located approximately 300 ft northwest of the Unit 1 turbine building (see Figure 2.4-1). This facility provides secondary treatment and disinfection for minimum, average, and maximum design flows of 10,000 gpd, 35,000 gpd, and 120,000 gpd, respectively.

9.2.4.2 System Description

Domestic Water System

The Town of Scriba Water District is the normal source of domestic water. The source meets the standards of quality for domestic water set by the state of New York. City water is supplied through an 8-in underground line to Unit 1. A 6-in main serves the Unit 2 area, with branch mains extended to buildings requiring domestic water (see Figure 9.2-8).

Domestic hot water is provided by several individual electric water heaters. The auxiliary service building south is served by a 20-kW water heater with a storage capacity of 120 gal. The control building is provided with a 12-kW water heater. A 100-gal storage tank is provided for emergency use, providing a 7-day supply of drinking water for control room personnel.

Safety showers throughout the plant are provided with precharged pressure tanks to ensure adequate pressure and flow for minimal operation during any interruption of the city water supply.

Sanitary Drains and Disposal System

Noncontaminated sanitary waste from Unit 2 flows by gravity to an underground wetwell (11,500-gal storage capacity). The wetwell is located adjacent to a sewage lift station that is equipped with two sewage pumps (see Figure 9.2-9). Potentially contaminated drainage is routed to the floor and equipment drainage systems (Section 9.3.3). All noncontaminated waste lines are vented to the atmosphere in accordance with the plumbing code.

All raw sanitary waste is pumped by two automatically alternated sewage pumps to the new sanitary process facility.

This waste treatment plant is an extended aeration, activated sludge process facility. The layout of the treatment plant is depicted on Figure 9.2-21. The combined sanitary waste flows from Unit 1 and Unit 2 will be treated and monitored to comply with state pollutant discharge elimination system permit effluent limitations.

9.2.4.3 Safety Evaluation

The domestic water system and sanitary drains and disposal systems are not safety related. They are not connected to any potentially radioactive process systems. The domestic water system is nonseismic except in the control building, where the emergency storage tank is seismically designed and constructed and the tank and associated piping are seismically supported. To avoid a flooding potential, a seismic Category I manual isolation valve is provided to permit isolation of the control building from the domestic water system should nonsafety-related piping within the building rupture during a seismic event. The system also supplies makeup water to the following:

1. Pressure maintenance pump supply tank in the fire protection system.

2. Filtered water storage tank in the water treating system.

3. Unit 1 domestic water system.

Radiation detection monitors and fire protection systems are not provided for the domestic water system or the sanitary drains and disposal system.

9.2.4.4 Testing and Inspection Requirements

The domestic water system and sanitary drains and disposal systems are in continuous use. The sanitary drains and disposal systems do not require any testing after the initial testing. Before placing in service, the domestic water system will be inspected and tested hydrostatically. Sanitary waste is monitored by sampling on a regular basis to verify that effluent is within water quality discharge limitations imposed under New York State SPDES permit requirements. Domestic water backflow preventers will be tested annually to verify proper operation and to maintain Department of Health certification.

9.2.5 Ultimate Heat Sink

The UHS consists of Lake Ontario and the service water intake and discharge system. This heat sink is capable of providing sufficient cooling for more than 30 days and performing its safety function following a design basis natural phenomenon event as described in RG 1.27.

The service water intake and discharge system is designed to supply water from Lake Ontario to the SWP system and fire protection system and to return to the lake the discharge from the SWP system, CWS system, WTS system, and liquid radwaste system (LWS). The service water intake and discharge system is safety related and is designed to meet the intent of RG 1.27.

9.2.5.1 Design Basis

The service water intake and discharge system is designed to supply water to the service water pumps and fire protection pumps and to return the plant discharge to the lake under all modes of operation.

The service water intake and discharge system is designed for the following conditions:

Minimum postulated lake level*	236.3 ft
Minimum controlled lake level*	243.0 ft
Minimum controlled lake level* during navigation season	244.0 ft
Maximum controlled lake level*	248.0 ft
Maximum postulated lake level*	254.0 ft
Minimum lake temperature	32°F
Maximum lake temperature	84°F

* USLS 1935 datum.

Normal service water pump flow (four pumps)	40,308 gpm
Maximum service water pump flow (five pumps)	52,461 gpm
Minimum discharge flow***	3,700 gpm
Maximum discharge flow**	52,461 gpm

The intake water may be tempered during normal winter operations using tempering flow from the CWS system (Section 9.2.5.2.4). The intake structures, bar racks, electrical heating elements, intake pipes, screenwell substructure, rectangular rotary gates, trash racks, and south shaft weir stop log are Category I. The diffuser nozzles, discharge tunnel, and traveling water screens are non-Category I.

9.2.5.2 System Description

9.2.5.2.1 Intake System

The source and discharge point of all the cooling water required by Unit 2 is Lake Ontario. Six pumps supply water to the SWP system (Section 9.2.1). After passing through the system, a variable flow of generally less than 25,000 gpm of the service water is used as makeup to the CWS system and the remaining portion is conveyed to the screenwell discharge bay.

Two identical intake structures are located approximately 950 and 1,050 ft from the existing shoreline as shown on Figure 1.2-29. The structures are located at lake bottom contour, el 223.5 ft (USLS 1935 datum). A minimum water depth of approximately 10 ft over the structures, as recommended by the U.S. Corps of Engineers and the U.S. Coast Guard, is provided during the navigational season when the mean low water elevation is 244 ft (USLS 1935 datum). Details of the two intake structures are shown on Figure 1.2-30. The structures are hexagonal in shape with a 22.5-ft width between opposite faces. The structures include a 4.5-ft bottom sill to limit the amount of sediment entering the structure, six intake openings each 7.5 ft wide by 3.0 ft high, and a 1.6-ft thick roof. The total area of the 12 openings, 6 on each structure, is designed to provide a maximum approach velocity of 0.5 fps while drawing water through both structures. The 12 openings are equipped with vertical bar racks with a 10-in clear spacing between the bars, which prevents large debris from entering the intake system. The bar racks consist of nine vertical bars for each opening; seven of these bars are electrically heated to eliminate the potential for frazil ice adhesion.

** Not including fish bypass discharges.

Fourteen heaters are required to be operating on each intake structure to prevent the formation of anchor ice and thus ensure safe shutdown flows. The design of the electrical heating elements is based on a flow rate of 18.1 cfs across each face of the intake structure or 48,740 gpm total intake flow through each structure. These flow rates result in a design velocity of 0.96 fps through each bar rack. The design of the electrical heater is based on a 2-in x 3-in rectangular bar which is conservatively treated as a circular cylinder with an equivalent diameter (for the Reynolds and Mussett numbers calculations). Based on these values, a 32°F intake water temperature and a 2°F rise in the bar temperature, the required electrical heating element capacity for each bar is 346 W. The heating elements are specified for a required minimum of 400 W and are conservatively rated at 450 W each. The electrical supply facilities are designed to provide a minimum of 396 W with the lowest voltage conditions and longest circuit length. The heating elements are powered by two separate Class 1E electrical buses. Both buses power the heating elements on three faces of each intake structure with each face separately wired. Therefore, at least one-half of the area of the bars is always heated providing sufficient flow area for safe shutdown flows. In addition, the intake structure has an inherent safety feature in that less than one-half the area of the openings on both structures is required for normal operation.

Four temperature elements in the onshore screenwell shafts, one on each electrical division in each shaft, energize the heating elements on the offshore intake structures. Because of the rapid fluctuations in temperature sometimes experienced in the lake, the heating elements are activated at a lake temperature of 38°F or less. Operation of the heating elements is monitored in the control room.

Each structure is independently connected to the onshore screenwell by a 4.5-ft diameter concrete intake encasement. The encasements are located within two 13 1/2 by 13 ft shotcreted tunnels (Figure 1.2-29). In addition to the intake encasement, tunnel no. 1 (the west tunnel) contains electrical conduits and has a cross-sectional area to accommodate plant discharge flow with a minimum of 60 sq ft. Tunnel no. 2 (the east tunnel) contains a 3.5-ft diameter fiberglass fish return pipe and electrical conduits for the heating elements in the bar racks in addition to the encased intake pipe. Both tunnels and the intake pipes are sloped downward toward the shoreline at a minimum of 0.01 ft/ft.

At the onshore screenwell, each intake encasement connects to a separate vertical shaft. The bottom of each shaft extends 10 ft below the invert of the 4.5-ft diameter encasement at the point where the intake encasement intersects the shaft. This provides a sediment trap just below the point where the maximum velocity decreases from approximately 3 fps in the 4.5-ft diameter horizontal intake encasement to approximately 1.0 fps in the vertical shafts. Access to these traps is provided through the

operating deck above each shaft for periodic cleaning, when necessary.

After passing through the two vertical shafts, the water enters the onshore screenwell building, which has a floor elevation of 224 ft. The screenwell arrangement is shown on Figure 9.2-13. Two motor-operated, rectangular rotary gates, arranged in series and normally open, are located between the north shaft and the intake bay. Each gate is powered from a separate Class 1E electrical bus. When these gates are closed, no water enters the intake bay through the north shaft.

Downstream of the rotary gates, water from both vertical shafts merges into a common bay and then is divided into two 4-ft wide screenbays. An angled, flush-mounted traveling water screen and two trash racks, one upstream and one downstream of the traveling water screens, are located in each screenbay. The two traveling water screens are angled 25 deg to the upstream direction of flow with their downstream ends converging. The trash racks upstream of the traveling water screens are cleaned by a motorized rake. The traveling water screens are cleaned by a water spray wash system that is actuated by a timer or a high differential pressure across the traveling water screens. The debris washed from the screens is directed into a trash trough that empties into a perforated trash basket.

Trash racks are provided upstream of the traveling water screens. Following a SSE, these trash racks would prevent floating debris, which might otherwise enter the flow path if the screens were dislodged, from reaching the service water pumps.

Water passes through the traveling water screens to the two screenbays which merge into a common bay. The service water and fire protection pumps take suction from this bay. Two motor-operated, rectangular rotary gates, arranged in parallel and normally closed, are located upstream of the two screenbays to provide a traveling water screen bypass flow path to the service water pumps. Each rotary gate is powered from a separate Class 1E electrical bus. A trash rack is located downstream of these gates.

9.2.5.2.2 Discharge System

The discharge consists of service water bypass (service water not utilized as CWS system makeup), CWS system blowdown, WTS system discharge, and liquid radwaste.

The discharge system consists of an onshore bay, a portion of one intake tunnel, a discharge tunnel, and a two-port diffuser (Figure 1.2-29). All discharge is conveyed to the discharge bay located on the west side of the two intake shafts and separated from them by a wall extending up to el 279 ft that acts as a weir (Figure 9.2-13). Stop-log slots are provided from the top of each weir (el 279 ft) to the operating deck (el 285 ft) with a

stop-log gate normally in place between the south shaft and the discharge bay. This provides an alternate discharge path as described in Section 9.2.5.3. The discharge normally enters a 4.5-ft diameter discharge pipe located on the north wall of the discharge bay which connects the discharge bay to the discharge portion of tunnel no. 1.

After traveling through the discharge portion of tunnel no. 1, the discharge continues past the point where the 4.5-ft diameter intake pipe rises to its intake structure, and enters into the smaller Gunite-lined discharge tunnel. The discharge portion of the intake tunnel has approximately 60 sq ft clear area and the discharge tunnel has approximately 80 sq ft.

The discharge tunnel terminates at a point approximately 1,500 ft from the existing shoreline where the discharge enters a 4.5-ft diameter steel riser leading to a two-port diffuser located on the lake bottom. The 4.5-ft diameter riser bifurcates into 3-ft diameter steel pipes with 1.5-ft diameter nozzles at the end of each (Figure 1.2-29). The nozzles are oriented so that they face offshore 120 deg apart and inclined at a 5-deg angle from horizontal. The invert of the nozzle openings is 3 ft off the lake bottom, providing 35.25 ft of water above the nozzle centerlines at the minimum controlled lake level (el 244 ft). During normal plant operation, the combined plant discharge flows by gravity from the discharge bay through the tunnels below the lake bottom to the diffuser nozzles and exits the nozzle at an approximate velocity of 18 fps. The combined plant differential discharge temperature to the lake is approximately 27°F.

9.2.5.2.3 Fish Bypass System

Fish entering the two 4-ft wide screenbays pass through the trash racks and are guided by the two angled, flush-mounted traveling water screens into 6-in wide bypass slots at the downstream end of the screens. The two slots converge and, at their junction, the fish are transported through a funnel-shaped transition to two pipes that merge into a single pipe leading to a jet pump. The jet pump discharges the fish and this bypass flow into a pipe located in tunnel no. 2. The fish are then transported through this pipe to a vertical riser and discharged into the lake in an easterly direction parallel to the lake bottom.

9.2.5.2.4 Tempering System

Tempering of the intake water in the onshore screenwell shafts occurs when the lake temperature is less than 38°F. During normal plant operation, the tempering flow is obtained from the discharge side of the condenser in the CWS system (Figure 10.4-7). Tempering flow may be controlled automatically or manually using the temperature control valve on the 16-in diameter tempering pipe. Tempering operation is governed by the restrictions or limitations in the facility’s Environmental Protection Plan (Nonradiological).

During shutdown operation when the CWS system is not operating, and the heat load on the SWP system has not significantly decreased, the tempering flow can be obtained from the Station discharge flow. A 16-in tempering line diverts a portion of the plant discharge from the 4.5-ft discharge pipe to the temperature control valve. A Category I isolation valve on this line is normally closed during normal operation to prevent recirculation of the discharge in the event of a tempering system failure.

9.2.5.3 Safety Evaluation

9.2.5.3.1 Compliance with Regulatory Guide 1.27

Compliance with RG 1.27 was addressed in response to NRC Staff Request 15, dated April 22, 1977, for additional information regarding the proposed cooling system design changes. This response was sent by cover letter dated September 30, 1977, to Mr. E. Case from Mr. G. K. Rhode of Niagara Mohawk Power Corporation (NMPC). Although the capability of the Unit 2 UHS to meet the RG 1.27 criteria was discussed in the response to Request 15, a portion of the response is repeated here for clarity:

> ...the circulating water system for NMP2 employs a cooling tower and is considered a closed loop system; the service water system is a once-through system. Reg. Guide 1.27, Rev. 2 states in Section C.3, "For once-through cooling systems, there should be at least two aqueducts connecting the source(s) with the intake structures of the nuclear power units, and at least two aqueducts to discharge the cooling water well away from the nuclear power plant to ensure that there is no potential for plant flooding by the discharged cooling water, unless it can be demonstrated that there is extremely low probability that a single aqueduct can functionally fail as a result of natural or site-related phenomena." As described in more detail below, the Unit 2 intake and discharge facilities are designed to withstand the safe shutdown earthquake, and the design, fabrication, and installation meet the requirements of 10CFR50, Appendix B. In addition, the intake and discharge tunnels are located in bedrock below Lake Ontario, and the tunnels and intake structures serve a totally passive function. For these reasons, there is an extremely low probability of failure of these structures, and the failure referred to in the above Request is not postulated. However, even if there were a single, passive failure of the intake structure or tunnel, the function of the ultimate heat sink would not be compromised.
>
> The lake intake and discharge system is designed to meet the ultimate heat sink criteria of Regulatory Guide 1.27.
>
> The ultimate heat sink will be Lake Ontario....

> The intake structures, intake pipes, screenwell substructure, bypass valves, trash racks, and butterfly gates will be designed to withstand the design basis earthquake. The traveling water screens will not be designed for seismic loadings. The seismic capability of the service water system beyond the service water pump suction is documented in Appendix C of the Preliminary Safety Analysis Report [FSAR Section 9.2.1]....
>
> The maximum water surface elevation in the discharge bay will be 277.5 ft while discharging through the diffuser. To prevent discharge water from entering the intake bay, a weir with the crest elevation at 279.0 ft will be located between each vertical intake shaft and the discharge bay as shown on Figure R12-2 [FSAR Figure 9.2-13].
>
> The diffuser nozzles, riser, and discharge tunnels are Seismic Category II designed.
>
> The screenwell discharge bay substructure and stop logs will be designed to withstand the design basis earthquake.
>
> During normal plant operation, the intake flow required for the service water pumps will be conveyed through both intake structures to the onshore screenwell. The plant discharge is normally conveyed from the discharge bay through the diffuser nozzles to the lake.

To meet the requirements of RG 1.27, each Category I intake structure and intake pipe is individually designed for the safe shutdown flow requirements. Table 9.2-8 provides the water surface elevations of the lake, the intake bay, and the discharge bay for various operating conditions. This table shows that both intake and discharge requirements can always be met. The minimum intake bay water level, el 233.1 ft, provides adequate submergence to the service water pumps. The centerline of the service water pump impeller for these horizontal centrifugal pumps is at el 227 ft 10 1/2 in.

The screenwell building is designed so that the UHS complex, as a system, can function during any reasonably probable combination of the most severe natural phenomena and/or historical site-related events that could affect the safety function of the UHS complex. The UHS is designed so that the annual alewife fish run and frazil ice have no effect on the safety function of the system. Less severe natural phenomena such as the OBE have no effect on system operability.

Frazil ice, the initial stage of all ice formation, requires supercooled water. While it is forming, it is called active frazil ice and has an adhesive property. Active frazil ice, however, has a short duration. Usually after only a few minutes it becomes inactive and loses its adhesiveness. Because the bar

racks on the offshore intake structure are Class 1E electrically heated and active frazil ice cannot adhere to a heated surface, active frazil ice will not block the intake openings. Inactive frazil ice, although unable to adhere to surfaces, can build up in the flow passages. The maximum buildup of inactive frazil ice is expected to be only 2 to 3 in. Because of the 10-in clear spacing of the bars, and because only 2 percent of the total intake area is required to pass the safe shutdown flow, the offshore intake structure will remain operational to provide the necessary service water[1].

Active frazil ice, when it passes under an ice cover, loses its adhesiveness. Any active frazil ice that enters the intake system becomes inactive before it reaches the onshore screenwell. Inactive frazil ice rises to the surface in areas of low velocity, such as in the screenwell. At the minimum postulated water depth in the screenwell, the service water pumps are submerged to a depth of 7 ft. If the SSE occurs during a frazil ice condition, the service water pumps are able to draw water from beneath the floating ice, regardless of the failure mechanism of the traveling water screens. If the normal flow path through the two screenbays is blocked either by failure of the traveling water screens or by an alewife fish run following a SSE, the traveling water screen bypass flow path assures safe shutdown flows.

If the service water pump bay water level drops to el 234 ft, two level switches open the two parallel rectangular rotary gates and thus allow the required flow to bypass the traveling water screens. Seismically designed trash racks, located downstream of the rotary gates, prevent large debris from reaching the service water pumps.

Alewife onshore spawning migrations begin in the early spring and generally last for several weeks. The maximum calculated hourly rate of fish entrapped at the Unit 1 or James A. FitzPatrick Plant is 62,220 fish/hr. The Unit 2 Category I service water pump strainers are capable of removing 18 gpm of debris. If an alewife migration occurs during a SSE, the safety-related flow requirement remains available because any fish reaching the service water pumps through either the bypass flow path or traveling water screenbays is removed by the self-cleaning strainers.

Sedimentation in the Unit 2 intake system will not affect the connection to the UHS. Analysis of the water quality of Lake Ontario in the vicinity of the Unit 2 site indicates that, although there are suspended solids in the water, insignificant amounts settle out. Of the eight offshore borings taken in the two Unit 2 tunnel locations, four indicate that the lake bottom is bedrock and there is no overburden. The remainder indicate that the lake bottom has between 0.2 and 2.8 ft of sandy gravel, mostly fine sand with some silt and clay particles over the bedrock. The sill elevation of the intake openings is set

approximately 4.5 ft above the lake bottom in order to minimize the amount of bottom sediment that enters the intake system when it is entrained in the water column by wave action. Therefore, sedimentation is not considered a site-related event. Also, no significant sedimentation has occurred at Unit 1 or the James A. FitzPatrick Plant adjacent to Unit 2.

9.2.5.3.2 Recirculation of Effluents

Recirculation of heated effluents is virtually eliminated through the design and arrangement of the cooling water intake and discharge structures. Negligible recirculation may occur under extremely rare conditions. The basic diffuser concept is to produce rapid dilution of the plant discharge water by entraining large quantities of cooler ambient lake water in the discharge plume. This is accomplished by a two-port submerged diffuser (Figure 1.2-29) with a high initial jet velocity. The submerged jets, oriented in a lakeward direction, are deflected toward the water surface due to the buoyancy of the jet. Flow continues lakeward from the point of intersection with the water surface.

The mathematical model used to predict the discharge plume was developed by Koh & Fan for a row of equally spaced round jets discharging at an arbitrary angle of inclination to horizontal into stagnant water[2]. From this model, standard nomograms as published by the Environmental Protection Agency (EPA) were generated[3]. Depth corrections by Robideau were applied to the EPA nomographs to obtain more conservative results[4]. The worst-case conditions input data entered into the mathematical models listed above are shown as parameters on Figure 9.2-15. Analysis of worst-case conditions with minimum controlled lake elevation of 243 ft resulted in a predicted temperature distribution presented on Figure 9.2-15. The model predictions indicate that the maximum surface temperature increase is 2.3°F.

The movement of lake currents in the vicinity of the discharge area is an important factor when considering the possibility of recirculation of heated effluents into the intakes. Intensive field surveys concerning the current characteristics and meteorological conditions have been conducted in the lake area adjacent to the Nine Mile Point site. Figure 9.2-16 shows the natural lake current characteristics according to the data recorded during the period July 1 to October 22, 1970, at a position approximately 1,600 ft offshore and 3,000 ft east of Unit 2. The figure indicates that for 70 percent of the time, the current speed was less than 0.25 knots (approximately 0.4 ft/sec). The alongshore currents, which are the dominant currents, occurred during almost 90 percent of the recording period and were generally weak, averaging a few tenths of a foot per second. The duration of high-speed currents is short.

The process of mixing and entrainment develops an induced lake current which moves from all directions to the discharge jets. The strength and direction of this current varies depending on

the total kinetic energy of the jets and the natural lake current in the discharge area. The location of the intake structures (Figure 1.2-29) results in the induced current supplying cooler water to the intakes from the far field. Recirculation can occur only when a shoreward lake current overcomes the current generated by the diffuser discharge. Figure 9.2-16 indicates that the percentage of time an onshore current greater than 0.05 knots exists is negligible because the continuity of water mass implies that a persistent onshore current is practically impossible. Recirculation is not expected to occur with onshore currents because the intake structures draw water from the lower depths and the buoyant discharge jets are deflected to the water surface.

Transient surface currents driven by northerly winds are expected. Wind-driven surface currents carrying heated effluent can only reach the intakes by a longer path of large-scale recirculation. The cooling water is further diluted during this path. Therefore, the further diluted heated effluents can reach the intake areas only infrequently and for short periods of time, and the quantities and temperature are negligible in terms of recirculation.

Due to the extremely low volume of discharge relative to both Unit 1 and the James A. FitzPatrick Plant, and the submerged high velocity discharge, the Unit 2 discharge has insignificant thermal effect beyond its immediate discharge area. The effect of Unit 2 discharge at the locations of either of the other discharge locations is negligible. The greatest effect of plume interaction occurs in the immediate vicinity of Unit 2 discharge when the lake conditions cause the plume from either Unit 1 or the James A. FitzPatrick Plant discharge to be in the vicinity of the Unit 2 discharge. Since the predominant currents are alongshore in either an easterly or westerly direction and the Unit 2 discharge is between Unit 1 and the James A. FitzPatrick Plant discharges, it is improbable that both these plumes could interact with the Unit 2 plume simultaneously.

When either the Unit 1 or the James A. FitzPatrick Plant plume is in the vicinity of the Unit 2 discharge, it is confined by its buoyancy to the upper 50 percent of the water column. The method used to predict the surface temperature rises due to the Unit 2 discharge only includes dilution of the jet from the lower 50 percent of the water column and assumes no dilution due to mixing with the upper layers. Therefore, the presence of a surface plume in the vicinity of the Unit 2 discharge will not alter the predicted surface temperature rises for Unit 2 at the point of jet surfacing.

Any interaction between the Unit 2 plume and either the Unit 1 or the James A. FitzPatrick Plant plume will involve the mixing of the Unit 2 surface plume, after jet surfacing, with the surrounding plume. The temperature rises resulting from the

mixing of the two plumes must necessarily be between the temperature rises in the separate plumes prior to mixing.

The temperature rises resulting from each of the individual discharges are at a maximum in the immediate vicinity of their respective discharges. These temperature rises decrease with increasing distance from the point of discharge. Only negligible quantities of heated effluent from the adjacent plant discharges can reach the Unit 2 intake structures because of the distances between the other discharge points and the Unit 2 intakes.

9.2.5.3.3 Effect of Failure of the St. Lawrence Power Project on Low Lake Levels

The St. Lawrence Power Project constructed jointly by New York Power Authority (NYPA) and Hydro-Electric Power Commission of Ontario (HEPCO) consists of two dams and a hydroelectric power plant in the St. Lawrence River. Iroquois Dam is a gated, gravity-type structure located 78 mi downstream from the outlet of the lake. Long Sault Dam, a second gated, gravity-type structure, is located 102 mi downstream from the outlet of the lake and serves as the spillway for the power dam. The central element of the St. Lawrence Power Project is the Robert Moses-Robert H. Saunders Power Dam, a gravity-type structure located 106 mi downstream from the outlet of the lake. Iroquois Dam and the Moses-Saunders Power Dam ordinarily pass the full flow of the St. Lawrence River, less relatively small diversions including the quantity of water needed to operate the locks of the St. Lawrence Seaway. This flow amounts to significantly less than 1 percent of the river's total flow. Long Sault Dam is operated infrequently and customarily discharges no water at all.

In the course of normal operation, Lake Ontario is regulated at the Moses-Saunders Power Dam pursuant to a plan of regulation approved by the International Joint Commission (IJC). Actual operation has demonstrated that the lake also can be regulated by manipulating the gates at Iroquois Dam. The power project, therefore, provides a redundant safeguard against any loss of capability to regulate Lake Ontario.

The failure of any part of the St. Lawrence Power Project is beyond reasonable expectations. The three gravity dams are structures that were designed to withstand all static and dynamic forces by wide margins. For example, the design criteria applicable to these dams provided factors of safety of at least 2.0 against overturning and at least 3.0 against sliding.

Gravity structures were designed to resist seismic forces equivalent to 0.05 g in both the horizontal and vertical directions, while the seismic design of slender structures was based on dynamic analysis. All phases of construction that had any bearing on the ultimate safety of any structure were continuously inspected by qualified engineers during performance

of the work and subjected to rigorous quality control requirements.

Since completion of the project, all structures have been under continuous surveillance by NYPA and HEPCO. Both NYPA and HEPCO conduct rigorous programs of inspection and preventive maintenance with respect to all elements of the project under their jurisdiction. Once every 5 yr, or more often if necessary, a complete safety inspection of the project is performed by independent consultants who report to the Department of Energy (DOE) with respect to the conditions of dams and other structures. The integrity of the dams that make up the St. Lawrence Power Project and regulate the levels of Lake Ontario is, therefore, ensured by the combination of conservative design, rigorous quality control during construction, and continuous surveillance following completion of the project.

In November 1968, the St. Lawrence Study Office of the Canadian Department of Energy, Mines, and Resources analyzed possible upstream and downstream effects resulting from failure of the St. Lawrence Power Project structures. Under the adverse assumptions of this study, which postulated the sudden destruction of the above-mentioned dams and the lowest supply sequence on record, it was determined that the lake level would decline gradually and, approximately 1 yr following the assumed failure, would be no more than 2.1 ft below the lowest level attained during regulation (i.e., the lake level would decline from el 242.7 to el 240.6). The study concluded that once the lake level had declined to about el 240.6, natural controls, such as existed before the project, would be reestablished and the lake levels would rise and fall thereafter in accordance with natural supplies delivered to Lake Ontario from the Great Lakes watershed. Superposition of the maximum probable seiche with the minimum still water level, 240.6 ft, would produce a further lowering of 4.3 ft to el 236.3 over a short term. Consequently, the intake and discharge system is designed for a minimum postulated lake level of 236.3 ft; this corresponds to a minimum intake level of 233.1 ft. The fire pumps are designed for a minimum intake system water level of 229.7 ft, providing a 3.4 ft margin of safety for the improbable postulated event.

Lake Ontario is the source of the St. Lawrence River and the last in the chain of the five Great Lakes. These lakes drain an area of approximately 300,000 sq mi in the United States (U.S.) and Canada. Taken together, they make up the largest body of freshwater in the world. Each lake acts as an enormous natural regulating reservoir which smoothes out inflow variations and tends to equalize seasonal outflows. Lake Ontario obtains its principal supply of water from the Niagara River which drains the four upper lakes (Erie, Huron, Michigan, and Superior) and discharges 200,000 cfs on an annual average basis. Other inflows are received from precipitation and springs within the Lake Ontario watershed. The outflow from Lake Ontario, which corresponds to the flow in the St. Lawrence River, is equal to

the quantity of water supplied from the Niagara River and all other sources, less the change in the quantity of water stored in the lake. Although storage in Lake Ontario is now governed by artificial controls, the lake was self-regulating in its natural state.

Flow in the St. Lawrence River is characterized by its extreme regularity. The maximum flow (320,000 cfs) is less than twice the minimum flow (162,000 cfs). This condition existed before construction of the St. Lawrence Power Project, and would continue today if the project did not exist. In the improbable event of the simultaneous failure of the Iroquois Dam and the Moses-Saunders Power Dam or Long Sault Dam, the actual level to which Lake Ontario would fall would be governed by supplies of water to the lake during the period following such failure and the natural resistance of the lakes to sudden changes in levels and flows. These effects would guarantee that the actual minimum still water level of the lake would be well above the design minimum water level elevation of the site during any period necessary to reestablish control of the lake.

9.2.5.3.4 Evaluation of Effects of Transportation Accidents on Operation of the Ultimate Heat Sink

Evaluation of the effects of collisions with the intake and discharge structures is provided in Section 2.2.3.1.5.

The accidental beaching of any watercraft on the plant site will not present a threat to safe operation since the unit is set back approximately 300 ft from the shoreline and both the intake and discharge tunnels are safely located well below ground (Figure 1.2-29).

9.2.5.3.5 Evaluation of Effects of Accidental Releases of Oil on the Intake Structure

The Unit 2 intake structures are designed to protect unit operations from the effects of accidental releases of oil in Lake Ontario near the site. Evaluation of the effects of such spills is discussed in Section 2.2.3.1.6.

9.2.5.4 Inspection and Testing Requirements

The four Category I rectangular rotary gates and associated control and instrumentation are tested in accordance with Technical Specifications and the TRM. The bar rack heating elements on each intake structure face are monitored to determine heating element failure. If more than 7 elements on three faces of each structure powered from the same electrical bus are out of service, replacement heating elements will be installed.

9.2.5.5 Instrumentation Requirements

Description

Safety-related instrumentation and controls are provided to automatically or manually maintain service water pump suction bay water level, service water discharge bay water level, and service water intake tunnel water temperature. Instrumentation and controls automatically control service water intake channel water temperature.

The controls and monitors described below are located in the main control room. The control logic is shown on Figure 9.2-2.

Operation

Sufficient service water pump suction bay water level is assured automatically by the intake screenwell rotary bypass gate valves. These valves automatically open on low-low pump suction bay water level, allowing intake water to bypass the normal channel. In addition, they may be controlled manually by opening or closing either intake screenwell bypass gate valve switch. Failure of the normal discharge path increases discharge bay water level. The intake gate valve will close if the water level exceeds high-high, redirecting the discharge flow through intake shaft no. 1.

The service water intake channel may be tempered. Tempering flow may be controlled automatically or manually using the temperature control valve. Tempering operation is governed by the restrictions or limitations in the facility's Environmental Protection Plan (Nonradiological). Fish that enter the intake shaft are returned to Lake Ontario via the jet motive pump.

Monitoring

Indicators are provided for:

1. Service water pump suction bay and discharge bay water levels.

2. Intake shafts inlet tempering water flow.

3. Jet motive pump flow.

Alarms are provided for:

1. Service water intake bay low water level.

2. Service water discharge bay high water level.

3. Intake shaft gate valve closed.

4. Service water intake tunnel water temperature low.

5. Jet motive pump motor electrical fault.

6. Service water intake system trouble.

9.2.6 Condensate Storage Facilities

9.2.6.1 Design Bases

9.2.6.1.1 Safety Design Bases

The condensate storage facility condensate makeup and drawoff (CNS) system is not required to effect or support safe shutdown of the reactor or to support the operation of any nuclear safety system.

9.2.6.1.2 Power Generation Design Bases

The purpose of the CNS system is to provide makeup water to various systems in the plant, to serve as a source of water during refueling operations and as a reserve for the HPCS and reactor core isolation cooling (RCIC) systems, and to provide for condenser hotwell level control. For additional information concerning refueling water, refer to Section 9.1.3.

Each of two CSTs is designed to hold 450,000 gal based on the following:

50 percent of refueling water requirement	287,000 gal
50 percent of normal water usage	25,000 gal
HPCS and RCIC reserve	135,000 gal

The design objective water quality for the condensate makeup is as follows:

Conductivity	≤1.0 umho/cm @ 25°C*
Chlorides (as Cl)	≤0.05 ppm
Soluble silica	≤0.2 ppm
Sodium (as Na)	≤0.02 ppm
Insolubles	≤0.30 ppm

System piping is designed to ANSI B31.1. Condensate transfer pumps are designed to the standards of the Hydraulic Institute.

9.2.6.2 System Description

* This limit applies after correction for dissolved CO_2.

The CNS system consists of two storage tanks, each with a nominal capacity of 450,000 gal, two parallel condensate transfer pumps, and necessary piping and instrumentation. The piping and instrumentation drawing (P&ID) is given on Figure 9.2-17. The storage tanks and pumps are housed in the condensate storage building. The tanks are vented by the condensate building exhaust fans to the main stack; therefore, any discharge of airborne radiation is monitored. Minimum operating temperature of the tanks is 40°F with a design temperature of 180°F.

Condensate is supplied from the storage tank to the makeup and drawoff system for distribution to various systems throughout the plant. Pressurized distribution is accomplished by the condensate transfer pumps. Unpressurized distribution (i.e., direct suction from the storage tank) is also supplied.

The CNS system serves the following systems:

System or Use	Purpose
RHR	Initial fill and flushing
CRD	CRD pump supply as a backup to the condensate system
Decontamination area	Washdown and decontamination
Spent fuel pool	Makeup and supply
HPCS	Testing, flushing, and reserve supply to the suppression pool
RCIC	Testing, flushing, and reserve supply to the suppression pool
LPCS	Flushing
Condensate system	Makeup
Refueling volume	Makeup and fill
CND system	Regeneration and resin sluicing
RWCU	Filter demineralizer backwash precoating
Turbine building services	Makeup, supply, and sample dilution
Radwaste building services	Makeup, supply, and sample dilution

System or Use Purpose

ABM	Supply feedwater makeup to the boilers
CFS	Alternate filter fill (makeup)

Water is supplied to the CST from the following sources: the LWS system, condenser hotwell, CRD, and MWS system. The CSTs accept clean condensate water from the LWS system. This condensate water is collected in tanks in the LWS system where it is sampled for conductivity and radiation (Section 11.5) prior to transfer to the CSTs.

The CSTs also supply and receive water from the condenser hotwell during load changes due to thermal expansion and contraction of reactor water and steam during load changes. Level control in the condenser hotwell regulates the amount of water added to the condensate system and the amount of water rejected to the CSTs from the condensate system. Normal makeup to the condenser hotwell is supplied by the condensate transfer pumps. A gravity-feed line is provided as a backup means of supply. This line also receives drawoff from the hotwell and the storage tanks.

The CRD pumps recirculate flow to the CSTs to regulate flow to the CRD system.

Initial fill and makeup to the CNS system is provided by the MWS system.

9.2.6.3 Safety Evaluation

With the exception of Station Blackout (SBO), the condensate storage facility CNS is not required to effect or support safe shutdown of the reactor or to support the operation of nuclear safety systems.

There are no essential portions of the condensate storage and transfer system. As identified in Table 3.2-1, all portions of this system are nonseismic and quality Group D. The HPCS and RCIC systems CST suction and test return lines are provided with seismic Category I, quality Group B isolation valves capable of automatic isolation.

The CSTs are the preferred water supply for the HPCS and RCIC systems. The alternate (safety-related) supply to the HPCS and RCIC is from the suppression pool. A low suction pressure signal within the HPCS or RCIC system will transfer the HPCS or RCIC pump suction to the suppression pool. Each of the storage tanks has the safeguard reserve capacity of 135,000 gal which is twice the required amount. All drawoff nozzles, other than those of the HPCS and RCIC systems, are located above the water level that holds the safeguard capacity. The HPCS-RCIC suction nozzles are 2 ft 6 in above the base of the tank and can be manually isolated with normally open block valves. The HPCS and RCIC pumps are

started periodically to assure operability. Water from the CST can also be used for this purpose. Condensate storage requirements related to Station Blackout (SBO) coping capability are addressed in Section 8.3.1.5.

The CSTs are located within the condensate storage building. In the event of a tank rupture, a release of the tank contents to Lake Ontario has been postulated. The maximum dose to the population would be orders of magnitude below values for existing background radiation (Section 15.7.3).

9.2.6.4 Test and Inspection

Field tests performed after equipment installation demonstrate satisfactory operation and functioning of control equipment, as well as guaranteed performance. The CSTs are given a water fill leak test after installation. System redundancy provides for acceptable reliability, and routine inspection and maintenance are performed periodically. Automatic actuation of system components is tested periodically.

9.2.6.5 Instrumentation Requirements

<u>Description</u>

Instruments and controls are provided for automatic and manual control of the water levels in the CSTs and the condenser hotwell. The control logic is shown on Figure 9.2-18.

<u>Operation</u>

Makeup water to the CSTs is controlled manually by opening the CST normal makeup valve in the line from the MWS system. The valve will close automatically when either CST level is normal. The valve can also be closed manually. Makeup from other sources is also controlled manually.

Two condensate transfer pumps provide normal makeup water from the CSTs to the condenser hotwell. Manual and automatic controls are provided for each transfer pump. During normal operation, one pump runs continuously. The second pump starts automatically on either a low pump discharge header pressure or a high pump discharge header flow demand. The transfer pumps stop automatically when the level in either CST decreases to the low-low limit.

The condenser hotwell is maintained at the normal condensate level automatically by the condensate normal makeup control valve modulated by a level controller. At low hotwell level, the condensate emergency makeup valve will open automatically to pass more water to the hotwell. The emergency makeup valve closes automatically when the hotwell level returns to normal. The emergency makeup valve can also be controlled manually. Excess condensate in the hotwell is returned automatically by the

condensate drawoff control valve to the CST. The condensate drawoff control valve is also modulated by a level controller. The following controls and monitors are located in the main control room.

<u>Monitoring</u>

Indicators are provided for:

1. Each CST level.
2. Condensate transfer pump discharge header flow.
3. Condenser hotwell level.
4. Condensate normal makeup control valve position.
5. Condensate drawoff control valve position.

Alarms are provided for:

1. CST level low, low-low, high, and high-high.
2. Condensate transfer pumps discharge header pressure low, flow high, and demand flow low.
3. Condensate transfer pumps autostart and autotrip/fail to start.
4. Condenser hotwell level high/low-low.
5. Condensate transfer pumps motor electrical fault.

9.2.7 Turbine Building Closed Loop Cooling Water System

The TBCLCW system is designed to remove heat from the designated heat exchangers in the turbine building and the radwaste building. The system is an intermediate cooling distribution loop that transfers heat from designated equipment to the SWP system (Section 9.2.1). The TBCLCW system is shown on Figure 9.2-19.

9.2.7.1 Design Bases

9.2.7.1.1 Safety Design Bases

The TBCLCW system is not required to effect or support safe shutdown of the reactor or to support the operation of any nuclear safety system.

9.2.7.1.2 Power Generation Design Bases

The TBCLCW system is designed in accordance with the following criteria:

1. All pumps are designed in accordance with the Hydraulic Institute standards.

2. The heat exchangers are designed in accordance with ASME Code for Unfired Pressure Vessels, Section VIII, Division 1, and also comply with the requirements of Tubular Exchanger Manufacturers Association Standards. The heat exchangers are designed to remove the full heat load generated by equipment listed in Table 9.2-9.

3. Piping is designed in accordance with ANSI B31.1, and includes allowances for corrosion.

4. The surge and makeup tank is designed in accordance with ASME Section VIII, Division 1.

9.2.7.2 System Description

The TBCLCW system consists of a single loop with three 450-hp, 50-percent system capacity, motor-driven centrifugal pumps in parallel (one on standby) feeding three half-capacity component cooling water heat exchangers also arranged in parallel (one on standby), all arranged to deliver demineralized cooling water at a variable temperature of 80 to 95°F to nonsafety-related turbine and radwaste plant equipment listed in Table 9.2-9. System capacity is based on an entering TBCLCW flow of 16,000 gpm at a temperature of 110°F and an outlet TBCLCW temperature of 80 to 95°F. The TBCLCW flowing in the shell of the heat exchangers is cooled by service water at the maximum expected temperature of 84°F entering and approximately 92°F leaving the tubes (Section 9.2.1).

A surge and makeup tank accommodates system volume changes due to temperature variations, maintains static head on the pumps, and allows detection of gross leaks in the system. It also provides for normal leakage in the system. Makeup water to the surge tank is provided by the MWS system (Section 9.2.3).

9.2.7.3 Safety Evaluation

The TBCLCW system is not a nuclear safety-related system. This system is not necessary for a safe shutdown of the plant or required during or after a design basis LOCA.

The heat exchangers associated with the radwaste system handle potentially radioactive material at an operating pressure lower than the component cooling water. Any tube leakage results in a flow from the component cooling water system to the radwaste system.

9.2.7.4 Testing and Inspection Requirements

Pumps in the TBCLCW system are proven operable by their use during normal plant operations. The standby heat exchanger and pump are placed in service periodically to ensure their operability and to allow all pumps and heat exchangers to wear evenly.

9.2.7.5 Instrumentation Requirements

Description

Instruments and controls are provided for automatic and manual control of the supply of cooling water for the auxiliary equipment in the turbine building and radioactive waste building. The controls and monitors described below are located in the main control room. The control logic is shown on Figure 9.2-20.

Operation

The TBCLCW pumps start automatically when the pump discharge header pressure is sustained low. The pumps stop automatically for low pump suction pressure, motor electrical fault, or sustained bus undervoltage. The pumps can also be controlled manually from the main control room.

The TBCLCW heat exchanger cooling water outlet temperature is controlled automatically by the CLCW temperature control valve.

The surge and makeup tank water level is controlled automatically by the tank water level control valve.

An offgas condenser outlet valve automatically opens fully from the fully closed position if the corresponding offgas isolation valve is open. Interlocks prevent it from fully opening if the other valve is partially open and both offgas isolation valves are open. The valves can also be controlled manually from local panel.

Monitoring

Indicators are provided for:

1. TBCLCW pump discharge header pressure.

2. TBCLCW system heat exchanger cooling water outlet temperature.

Alarms are provided for:

1. TBCLCW system trouble.

2. Radiation monitor trouble/manually out of service/radiation status.

3. Process liquid radiation monitor activated.

9.2.8 References

1. Journal AWWA. Frazil - The Invisible Strangler, Foulds and Wigle, April 1977.

2. Koh, R. C. Y. and Loh-Nien Fan. Mathematical Models for the Prediction of Temperature Distribution Resulting from the Discharge of Heated Water into Large Bodies of Water, Water Pollution Control Res. Ser. EPA-16130 DWO 10/70, October 1976.

3. Shirazi, M. A. and Davis, L. A. Workbook of Thermal Plume Prediction, Chapter I, Submerged Discharge, U.S. EPA, National Environmental Res. Center, EPA-R2-72-005a, August 1972.

4. Robideau, R. F. The Discharge of Submerged Buoyant Jets into Water of Finite Depth. General Dynamics, Electric Boat Division (Groton, CT), U440-72-121 (PB 214-475), 1972.

TABLE 9.2-1A
DIVISION I COMPONENTS REQUIRED SERVICE WATER FLOWS (GPM) - ASSUMING COMPLETE FAILURE OF DIVISION II
The flow rates listed below correspond to the originally specified flows (in the corresponding equipment specifications) but are not the flow rates required for the equipment to achieve the required heat removal capacity.

Equipment Identification Number	Division	LOCA(1)	LOCA & LOOP	Shutdown(1)	LOOP	Notes
Reactor Building						
2HVR*UC401A	I	29	29	29	29	
2HVR*UC401B	II	0	0	0	0	
2HVR*UC401C	II	0	0	0	0	
2HVR*UC401D	I	29	29	29	29	
2HVR*UC401E	II	0	0	0	0	
2HVR*UC401F	II	0	0	0	0	
2HVR*UC402A	I	54	54	54	54	
2HVR*UC402B	I	54	54	54	54	
2HVR*UC403A	I	77	77	77	77	(5)
2HVR*UC403B	II	0	0	0	0	(5)
2HVR*UC404A	I	21	21	21	21	
2HVR*UC404B	I	21	21	21	21	
2HVR*UC404C	II	0	0	0	0	
2HVR*UC404D	II	0	0	0	0	
2HVR*UC405	I	13	13	13	13	
2HVR*UC406	II	0	0	0	0	
2HVR*UC407A	I	16	16	16	16	
2HVR*UC407B	I	16	16	16	16	
2HVR*UC407C	I	16	16	16	16	
2HVR*UC407D	II	0	0	0	0	
2HVR*UC407E	II	0	0	0	0	
2HVR*UC408A	I	29	29	29	29	(5)
2HVR*UC408B	I	0	0	0	0	(5)
2HVR*UC409A	II	0	0	0	0	
2HVR*UC409B	II	0	0	0	0	
2HVR*UC410A	I	16	16	16	16	
2HVR*UC410B	II	0	0	0	0	
2HVR*UC410C	II	0	0	0	0	
2HVR*UC411A	I	21	21	21	21	
2HVR*UC411B	II	0	0	0	0	
2HVR*UC411C	II	0	0	0	0	
2HVR*UC412A	I	21	21	21	21	
2HVR*UC412B	II	0	0	0	0	
2HVR*UC413A	I	350	350	0	0	(5)
2HVR*UC413B	II	0	0	0	0	(5)
2HVR*UC414A	I	26	26	26	26	(5)
2HVR*UC414B	II	0	0	0	0	(5)
2HVR*UC415A	I	20	20	20	20	(5)
2HVR*UC415B	II	0	0	0	0	(5)
2RHS*E1A	I	7,400(2)	7,400(2)	7,400(2)	7,400(2)	

TABLE 9.2-1A (Cont'd.)

Equipment Identification Number	Division	LOCA[1]	LOCA & LOOP	Shutdown[1]	LOOP	Notes
Reactor Building (Cont'd.)						
2RHS*E1B	II	0	0	0	0	
2RHS*P1A	I	20[3]	20[3]	20[3]	20[3]	
2RHS*P1B	II	0	0	0	0	
2RHS*P1C	II	0	0	0	0	
2SFC*E1A	I	2,410[3]	2,410[3]	2,410[3]	2,410[3]	(5)
2SFC*E1B	II	0	0	0	0	(5)
2HCS*RBNR1A	I	10	10	0	0	
2HCS*RBNR1B	II	0	0	0	0	
Control Building						
2HVK*CHL1A	I	340	340	340	340	(5)
2HVK*CHL1B	II	0	0	0	0	(5)
2HVC*UC101A	I	29	29	29	29	
2HVC*UC101B	II	0	0	0	0	
2HVC*UC102	II	19	19	19	19	(7)
2HVC*UC103A	I	11	11	11	11	
2HVC*UC103B	II	0	0	0	0	
2HVC*104	I	29	29	29	29	
2HVC*105	II	0	0	0	0	
2HVC*106	I	52	52	52	52	
2HVC*107	II	0	0	0	0	
2HVC*108A	I	50	50	50	50	
2HVC*108B	II	0	0	0	0	
Diesel Generator Building						
2EGS*EG1	I	800	800	0	800	
2EGS*EG2	III	600	600	0	600	
2EGS*EG3	II	0	0	0	0	
2HVP*UC1A	I	11	11	0	11	
2HVP*UC1B	II	0	0	0	0	
2HVP*UC2	III	11	11	0	11	
Screenwell Building						
2HVY*UC2A	I	85	85	85	85	(6)
2HVY*UC2B	II	0	0	0	0	(6)
2HVY*UC2C	I	0	0	0	0	(6)
2HVY*UC2D	II	0	0	0	0	(6)
Total Flow (GPM)		12,687	12,687	10,905	12,327	

TABLE 9.2-1A (Cont'd.)

NOTES:

(1) Manual isolation of divisions and nonsafety-related equipment is assumed.
(2) Flow required after manual initiation.
(3) Flow required after manual initiation when cooling water from RBCLCW is unavailable; alternate cooling will be supplied within 3 hr.
(4) Deleted.
(5) Only one of two components in operation.
(6) Only two of four components in operation.
(7) The specified value represents the flow rate at original plant design conditions. Changes in the required flow rate due to variations in plant conditions are addressed in the associated system design calculations.

KEY:

HVR = Reactor building unit coolers
RHS = Residual heat removal heat exchangers and pumps
SFC = Spent fuel pool cooling heat exchangers
HCS = DBA hydrogen recombiners
HVK = Control building chilled water chillers
HVC = Control building unit coolers
EGS = Diesel generator coolers
HVP = Diesel generator control room unit coolers
HVY = Service water pump room unit coolers

TABLE 9.2-1B
DIVISION II COMPONENTS REQUIRED SERVICE WATER FLOWS (GPM) - ASSUMING COMPLETE FAILURE OF DIVISION I
The flow rates listed below correspond to the originally specified flows (in the corresponding equipment specifications) but are not the flow rates required for the equipment to achieve the required heat removal capacity.

Equipment Identification Number	Division	LOCA(1)	LOCA & LOOP	Shutdown(1)	LOOP	Notes
Reactor Building						
2HVR*UC401A	I	0	0	0	0	
2HVR*UC401B	II	29	29	29	29	
2HVR*UC401C	II	29	29	29	29	
2HVR*UC401D	I	0	0	0	0	
2HVR*UC401E	II	29	29	29	29	
2HVR*UC401F	II	29	29	29	29	
2HVR*UC402A	I	0	0	0	0	
2HVR*UC402B	I	0	0	0	0	
2HVR*UC403A	I	0	0	0	0	5
2HVR*UC403B	II	77	77	77	77	5
2HVR*UC404A	I	0	0	0	0	
2HVR*UC404B	I	0	0	0	0	
2HVR*UC404C	II	21	21	21	21	
2HVR*UC404D	II	21	21	21	21	
2HVR*UC405	I	0	0	0	0	
2HVR*UC406	II	13	13	13	13	
2HVR*UC407A	I	0	0	0	0	
2HVR*UC407B	I	0	0	0	0	
2HVR*UC407C	I	0	0	0	0	
2HVR*UC407D	II	16	16	16	16	
2HVR*UC407E	II	16	16	16	16	
2HVR*UC408A	I	0	0	0	0	5
2HVR*UC408B	I	0	0	0	0	5
2HVR*UC409A	II	29	29	29	29	
2HVR*UC409B	II	0	0	0	0	
2HVR*UC410A	I	0	0	0	0	
2HVR*UC410B	II	16	16	16	16	
2HVR*UC410C	II	16	16	16	16	
2HVR*UC411A	I	0	0	0	0	
2HVR*UC411B	II	21	21	21	21	
2HVR*UC411C	II	21	21	21	21	
2HVR*UC412A	I	0	0	0	0	
2HVR*UC412B	II	21	21	21	21	
2HVR*UC413A	I	0	0	0	0	5
2HVR*UC413B	II	350	350	0	0	5
2HVR*UC414A	I	0	0	0	0	5
2HVR*UC414B	II	26	26	26	26	5
2HVR*UC415A	I	0	0	0	0	5
2HVR*UC415B	II	20	20	20	20	5
2RHS*E1A	I	0	0	0	0	
2RHS*E1B	II	7,400(2)	7,400(2)	7,400(2)	7,400(2)	

TABLE 9.2-1B (Cont'd.)

Equipment Identification Number	Division	LOCA[1]	LOCA & LOOP	Shutdown[1]	LOOP	Notes
Reactor Building (Cont'd.)						
2RHS*P1A	I	0	0	0	0	
2RHS*P1B	II	20[3]	20[3]	20[3]	20[3]	
2RHS*P1C	II	20[3]	20[3]	20[3]	20[3]	
2SFC*E1A	I	0	0	0	0	5
2SFC*E1B	II	2,410[3]	2,410[3]	2,410[3]	2,410[3]	5
2HCS*RBNR1A	I	0	0	0	0	
2HCS*RBNR1B	II	10	10	0	0	
Control Building						
2HVK*CHL1A	I	0	0	0	0	5
2HVK*CHL1B	II	340	340	340	340	5
2HVC*UC101A	I	0	0	0	0	
2HVC*UC101B	II	29	29	29	29	
2HVC*UC102	II	19	19	19	19	7
2HVC*UC103A	I	0	0	0	0	
2HVC*UC103B	II	11	11	11	11	
2HVC*104	I	0	0	0	0	
2HVC*105	II	13	13	13	13	
2HVC*106	I	0	0	0	0	
2HVC*107	II	52	52	52	52	
2HVC*108A	I	0	0	0	0	
2HVC*108B	II	50	50	50	50	
Diesel Generator Building						
2EGS*EG1	I	0	0	0	0	
2EGS*EG2	III	600	600	0	600	
2EGS*EG3	II	800	800	0	800	
2HVP*UC1A	I	0	0	0	0	
2HVP*UC1B	II	11	11	0	11	
2HVP*UC2	III	11	11	0	11	
Screenwell Building						
2HVY*UC2A	I	0	0	0	0	6
2HVY*UC2B	II	85	85	85	85	6
2HVY*UC2C	I	0	0	0	0	6
2HVY*UC2D	II	0	0	0	0	6
Total Flow (gpm)		12,681	12,681	10,899	12,321	

TABLE 9.2-1B (Cont'd.)

NOTES:

(1) Manual isolation of divisions and nonsafety-related equipment is assumed.
(2) Flow required after manual initiation.
(3) Flow required after manual initiation when cooling water from RBCLCW is unavailable; alternate cooling will be supplied within 3 hr.
(4) Deleted.
(5) Only one of two components in operation.
(6) Only two of four components in operation.
(7) The specified value represents the flow rate at original plant design conditions. Changes in the required flow rate due to variations in plant conditions are addressed in the associated system design calculations.

KEY:

HVR = Reactor building unit coolers
RHS = Residual heat removal heat exchangers and pumps
SFC = Spent fuel pool cooling heat exchangers
HCS = DBA hydrogen recombiners
HVK = Control building chilled water chillers
HVC = Control building unit coolers
EGS = Diesel generator coolers
HVP = Diesel generator control room unit coolers
HVY = Service water pump room unit coolers

TABLE 9.2-2

SERVICE WATER FLOW REQUIREMENTS FOR
NORMAL POWER GENERATION

Equipment Identification Number	Division	Flow (gpm)
	Essential Component Flows	
Reactor Building		
2HVR*UC401A	I	29
2HVR*UC401B	II	29
2HVR*UC401C	II	29
2HVR*UC401D	I	29
2HVR*UC401E	II	29
2HVR*UC401F	II	29
2HVR*UC402A	I	54
2HVR*UC402B	I	54
2HVR*UC403A	I	77
2HVR*UC403B	II	77
2HVR*UC404A	I	21
2HVR*UC404B	I	21
2HVR*UC404C	II	21
2HVR*UC404D	II	21
2HVR*UC405	I	13
2HVR*UC406	II	13
2HVR*UC407A	I	16
2HVR*UC407B	I	16
2HVR*UC407C	I	16
2HVR*UC407D	II	16
2HVR*UC407E	II	16
2HVR*UC408A	I	29
2HVR*UC408B	I	0
2HVR*UC409A	II	29
2HVR*UC409B	II	0
2HVR*UC410A	I	16
2HVR*UC410B	II	16
2HVR*UC410C	II	16
2HVR*UC411A	I	21
2HVR*UC411B	II	21
2HVR*UC411C	II	21
2HVR*UC412A	I	21
2HVR*UC412B	II	21
2HVR*UC413A	I	0
2HVR*UC413B	II	0
2HVR*UC414A	I	26
2HVR*UC414B	II	0
2HVR*UC415A	I	20
2HVR*UC415B	II	0

TABLE 9.2-2 (Cont'd.)

Equipment Identification Number	Division	Flow (gpm)
Reactor Building (cont'd.)		
2RHS*E1A	I	0
2RHS*E1B	II	0
2RHS*P1A	I	0
2RHS*P1B	II	0
2RHS*P1C	II	0
2SFC*E1A	I	0
2SFC*E1B	I	0
2HCS*RBNR1A	I	0
2HCS*RBNR1B	II	0
Control Building		
2HVK*CHL1A	I	340
2HVK*CHL1B	II	340
2HVC*UC101A	I	29
2HVC*UC101B	II	29
2HVC*UC102	II	19
2HVC*UC103A	I	11
2HVC*UC103B	II	11
2HVC*104	I	29
2HVC*105	II	13
2HVC*106	I	52
2HVC*107	II	52
2HVC*108A	I	50
2HVC*108B	II	50
Diesel Generator Building		
2EGS*EG1	I	0
2EGS*EG2	III	0
2EGS*EG3	II	0
2HVP*UC1A	I	0
2HVP*UC1B	II	0
2HVP*UC2	III	0
Screenwell Building		
2HVY*UC2A	I	85
2HVY*UC2B	II	85
2HVY*UC2C	I	0
2HVY*UC2D	II	0
Total Essential Flow		1,738

TABLE 9.2-2 (Cont'd.)

Equipment Identification Number	Division	Flow (gpm)
Nonessential Component Flows		
Reactor Building		
2CCP-E1A(1)	-	5,834
2CCP-E1B(1)	-	5,834
2CCP-E1C(1)	-	0
2HVR-CLC2	-	400
Turbine Building		
2CCS-E1A(1)	-	8,760
2CCS-E1B(1)	-	8,760
2CCS-E1C(1)	-	0
2ARC-E1A(2)	-	300
2ARC-E1B(2)	-	0
2ARC-E2A(2)	-	1,900
2ARC-E2B(2)	-	0
2HVT-UC201A	-	58
2HVT-UC201B	-	58
2HVT-UC202A	-	58
2HVT-UC202B	-	58
2HVT-UC203A	-	58
2HVT-UC203B	-	58
2HVT-UC204	-	32
2HVT-UC205	-	32
2HVT-UC206A	-	64
2HVT-UC206B	-	64
2HVT-UC206C	-	64
2HVT-UC206D	-	64
2HVT-UC206E	-	64
2HVT-UC206F	-	64
2HVT-UC207A	-	58
2HVT-UC207B	-	58
2HVT-UC208A	-	53
2HVT-UC208B	-	53
2HVT-UC209A	-	49
2HVT-UC209B	-	49
2HVT-UC210A	-	49
2HVT-UC210B	-	49
2HVT-UC211	-	32
2HVT-UC212A	-	53
2HVT-UC212B	-	53
2HVT-UC213A	-	53
2HVT-UC213B	-	53
2HVT-UC214A	-	53

TABLE 9.2-2 (Cont'd.)

Equipment Identification Number	Division	Flow (gpm)
Turbine Building (cont'd.)		
2HVT-UC214B	-	53
2HVT-UC215A	-	53
2HVT-UC215B	-	53
2HVT-UC216A	-	65
2HVT-UC216B	-	65
2HVT-UC216C	-	65
2HVT-UC216D	-	65
2HVT-UC216E	-	65
2HVT-UC217A	-	43
2HVT-UC217B	-	43
2HVT-UC218A	-	65
2HVT-UC218B	-	65
2HVT-UC218C	-	65
2HVT-UC218D	-	65
2HVT-UC218E	-	65
2HVT-UC219	-	32
2HVT-UC220	-	25
2HVT-UC221A	-	53
2HVT-UC221B	-	53
2HVT-UC222A	-	47
2HVT-UC222B	-	47
2HVT-UC222C	-	47
2HVT-UC222D	-	49
2HVT-UC222E	-	49
2HVT-UC222F	-	49
2HVT-UC223A	-	53
2HVT-UC223B	-	53
2HVT-UC224	-	42
2HVT-UC225	-	42
2HVT-UC226	-	53
Total Nonessential Flow	-	34,888
Total Essential & Nonessential Flow	-	36,626

NOTES:

(1) Only two of three components in operation.
(2) Only one of two components in operation.

TABLE 9.2-2 (Cont'd.)

KEY:

HVR	=	Reactor building unit coolers
RHS	=	Residual heat removal heat exchangers and pumps
SFC	=	Spent fuel pool cooling heat exchangers
HCS	=	DBA hydrogen recombiners
HVK	=	Control building chilled water chillers
HVC	=	Control building unit coolers
EGS	=	Diesel generator coolers
HVP	=	Diesel generator control room unit coolers
HVY	=	Service water pump room unit coolers
CCP	=	Reactor building component cooling water heat exchangers
CCS	=	Turbine building component cooling water heat exchangers
ARC	=	Vacuum pump seal water coolers and steam jet air ejector precoolers
HVT	=	Turbine building unit coolers

TABLE 9.2-3

DESIGN DATA OF COMPONENTS SUPPLIED WITH
REACTOR BUILDING CLOSED LOOP COOLING WATER

Description	Quantity	Mode of Operation	Total Flow Requirements (gpm)	Process Heat Load (10^6 Btu/hr)
Spent fuel pool heat exchangers	2	Normal/accident[1]	2,400	16
RWCU nonregenerative heat exchanger	1	Normal	1,404	39
RWCU pump bearings, pedestals, and seal jackets coolers	2	Normal	108	0.8
Reactor building equipment drain coolers	3	Normal	317	3.3
Drywell equipment drain cooler	1	Normal	104	1.1
Drywell unit space coolers[2]	10	Normal	1,295	7
Reactor recirculation motor winding coolers, motor bearing coolers, and pump seal coolers	2	Normal	750	3.7
CCP heat exchangers for instrument air compressors	2	Normal	135	1.4
RHR pump seal coolers	3	Normal/accident[1]	60	*
Control rod drive pump bearing and seal coolers	2	Normal	50	*
Reactor plant sampling panel	1	Normal	33	*
Reactor recirculation sample cooler	1	Normal	15	*
RHR sample cooler	2	Normal	30	*
SFC and RWCU heat exchanger off-line radiation monitors	2	Normal	17	*
Post-accident piping station panel[3]	1	Accident	10	*

* The combined heat load for these components is ~1.3 x 10^6 Btu/hr.

(1) Supplied with service water for safety-related function.
(2) Only 9 out of 10 drywell unit space coolers are normally in operation.
(3) Post-accident piping station panel's heat load is normally isolated.

TABLE 9.2-4

REACTOR BUILDING CLOSED LOOP
COOLING WATER SYSTEM COMPONENT DESCRIPTION

RBCLCW Main Pumps (2-CCP-P1A, P1B, P1C)	
Quantity	3
Type	Horizontal centrifugal
Fluid	Demineralized water
Capacity, each gpm (%)	(50) 3,370
Total head, ft	110
Driver, hp	150
RBCLCW Booster Pumps (2CCP-P3A, P3B, P3C)	
Quantity	3
Type	Horizontal centrifugal
Fluid	Demineralized water
Capacity, each, gpm (%)	(50) 3,370
Total head, ft	110
Driver, hp	150
RBCLCW Instrument Air Compressor Cooling Water Pumps (2CCP-P2A, P2B)	
Quantity	2
Type	Horizontal centrifugal
Fluid	Demineralized water
Capacity, each, gpm (%)	(100) 125
Total head, ft	135
Driver, hp	10
RBCLCW Heat Exchangers (2CCP-E1A, E1B, E1C)	
Quantity	3
Type	Shell and tube, single pass
Heat transfer duty, Btu/hr, ea	37.85×10^6
Design code	ASME Section VIII, TEMA Class R
Shell side:	
Fluid	Demineralized water
Design pressure, psig	200
Design temperature, °F	150
Flow, gpm	3,255
Inlet temperature, °F	118
Outlet temperature, °F	95
Material	Carbon steel ASTM A285-C

TABLE 9.2-4 (Cont'd.)

Tube side:	
Fluid	Raw lake water
Design pressure, psig	200
Design temperature, °F	150
Flow, gpm	5,834
Inlet temperature, °F	77
Outlet temperature, °F	90
Material	Admiralty SB-111 Alloy 443
RBCLCW Instrument Air Compressor Cooling Water Heat Exchangers (2CCP-E2A, E2B)	
Quantity	2
Type	Shell and tube, single pass
Heat transfer duty, Btu/hr, ea	1,350,000
Design code	ASME Section VIII, TEMA R
Shell side:	
Fluid	Demineralized water
Design pressure, psig	150
Design temperature, °F	300
Flow, gpm	110
Inlet temperature, °F	130
Outlet temperature, °F	105
Material	Carbon steel SA-53B SMLS
Tube side:	
Fluid	Demineralized water
Design pressure, psig	150
Design temperature, °F	300
Flow, gpm	137
Inlet temperature, °F	95
Outlet temperature, °F	115
Material	90-10 Cu-Ni
RBCLCW Surge Tank (2CCP-TK-1)	
Quantity	1
Type	Horizontal, elliptical dished heads
Design pressure, psig	15
Design temperature, °F	150°F
Material	Carbon steel SA-285C

TABLE 9.2-4 (Cont'd.)

<u>RBCLCW Surge Tank (2CCP-TK-2)</u>

Quantity	1
Type	Vertical, elliptical dished heads
Design pressure, psi	15
Design temperature, °F	110
Material	SA-285, Gr. C

TABLE 9.2-5

EFFLUENT WATER QUALITY*

Constituent	Concentration (ppm)
Chlorides (as Cl)	Trace
Total dissolved solids (as $CaCO_3$)	0.04
Total iron (as Fe)	Trace
Hardness (as $CaCo_3$)	Trace
Total silica (as SiO_2)	0.02
Carbon dioxide (as CO_2)	Trace
Sodium (as Na)	0.02
Conductivity umho/cm @ 25°C	0.2
pH	6.6-7.5

* Effluent expected or calculated at the demineralizer water transfer pump discharge grab sample station.

TABLE 9.2-6

INFLUENT WATER QUALITY[1]

Constituent	Concentration[2][3] Max mg/l	Mean mg/l
Alkalinity (as $CaCO_3$)	106	88.5
Calcium (as Ca)	54	45.64
Magnesium (as Mg)	9.6	6.67
Chloride (as Cl)	70	37.78
Iron (as Fe)	0.36	0.134
Ortho-P (as P)	0.05	0.0106
(as PO_4)	0.15	0.0325
Sulfate (as SO_4)	39	29.4
Suspended solids	3	<3
Silica (as Si)	5	0.25
(as SiO_2)	10.71	0.56
Total dissolved solids	370	175
Copper (as Cu)	0.41	0.0667
Zinc (as Zn)	0.638	0.0628
pH	8.8	8.0

[1] Influent expected at outlet of the activated carbon filter.
[2] These values are based on filtered lake water.
[3] Except pH.

TABLE 9.2-7

DEMINERALIZER DATA

Equipment	Diameter		Bed Depth		Height		Resin Volume		Exchanger Capacity	
	in	cm	in	cm	in	cm	ft^3	in^3	kg/ft^3	lg/l
Weak acid cation	54	137.16	39	99.06	75	190.50	50	1.415	28.8	1.3
Strong acid cation	84	213.36	43	109.22	87	220.98	141	3.99	12.1	0.55
Weak base anion	72	182.88	39	99.06	81	205.74	88	2.49	19.0	0.86
Strong base anion	48	121.92	40	101.60	87	220.98	40	1.132	9.15	0.42
Mixed bed exchanger	48	121.92			99	251.46				
Cation resin			24	60.96			24	0.679	16.1	0.73
Anion resin			24	60.96			24	0.679	7.1	0.32

TABLE 9.2-8

INTAKE AND DISCHARGE BAY WATER SURFACE ELEVATIONS

Condition	Service Water Requirement (gpm)	Intake Flow[1] (gpm)	Discharge Flow (gpm)	Lake W.S. Elevation[2]	Intake Elevation (at Service Water Pumps)	Discharge Bay Elevation
Normal summer[3] (4 SW pumps)	39,746	54,671	32,174	236.3	233.9	243.4
	39,746	54,671	32,174	243.0	240.6	250.1
	39,746	54,671	32,174	248.0	245.6	255.1
	39,746	54,671	32,174	254.0	251.6	261.1
Extreme summer[3] (5 SW pumps)	43,315	59,585	34,216	236.3	233.6	244.4
	43,315	59,585	34,216	243.0	240.3	251.1
	43,315	59,585	34,216	248.0	245.3	256.1
	43,315	59,585	34,216	254.0	251.3	262.1
Normal winter[3] (no tempering) (4 SW pumps)	39,746	54,671	35,040	236.3	233.9	244.8
	39,746	54,671	35,040	243.0	240.6	251.5
	39,746	54,671	35,040	248.0	245.6	256.5
	39,746	54,671	35,040	254.0	251.6	262.5
Normal shutdown[3] (5 SW pumps)	51,923	66,848	51,923	236.3	233.1	254.9
	51,923	66,848	51,923	243.0	239.8	261.6
	51,923	66,848	51,923	248.0	244.8	266.6
	51,923	66,848	51,923	254.0	250.8	272.6
Normal shutdown[4] with/loss of diffuser (5 SW pumps)	51,923	66,848	51,923	243.0	233.7	249.1[5]
	51,923	66,848	51,923	248.0	238.7	254.1
	51,923	66,848	51,923	254.0	244.7	260.1
LOCA, LOOP, and loss of diffuser[4] (4 SW pumps)	30,518	30,518	30,518	236.3	233.8	238.4[6]
	30,518	30,518	30,518	243.0	240.5	245.1
	30,518	30,518	30,518	248.0	245.5	250.1
	30,518	30,518	30,518	254.0	251.2	256.1
LOCA, no LOOP, and no diffuser loss[3] (5 SW pumps)	55,381	55,381	55,381	236.3	233.9	257.6
	55,381	55,381	55,381	243.0	240.6	264.2
	55,381	55,381	55,381	245.0	245.6	269.2
	55,381	55,381	55,381	254.0	251.6	275.2

(1) Includes service water flow requirement and fish bypass flow requirement.
(2) Lake elevations: Minimum postulated el 236.3' Maximum controlled el 248.0'
Minimum controlled el 243.0' Maximum postulated el 254.0'
(3) Intake through both intake structures and discharge through diffuser.
(4) Intake through south shaft, discharge through north shaft.
(5) These elevations are for the north intake shaft which is used for discharge. The discharge bay elevation is 282.1'.
(6) See Note 2. The discharge bay elevation is 281.2'.

KEY: LOCA = Loss-of-coolant accident
LOOP = Loss of offsite power

TABLE 9.2-9

MAJOR EQUIPMENT COOLED BY THE TURBINE
BUILDING CLOSED LOOP COOLING WATER SYSTEM

Equipment	Quantity
Generator hydrogen coolers	4
Generator stator water coolers	2
Exciter alternator coolers	2
Generator leads coolers	2
Electrohydraulic control unit coolers	2
Turbine lube oil coolers	2
Reactor feed pump lubricating oil coolers and motor coolers	3
Condensate booster pump motor lubricating oil coolers	3
Condensate pump motor lube oil coolers	3
Offgas condensers	2
Heater drain pump motor bearing oil coolers and seal heat exchangers	3
Turbine building equipment drain sump coolers	5
Turbine building equipment drain cooler	1
Offgas refrigeration unit condensers	6
Circulating water pump motor oil coolers	6
Charcoal decay bed room air conditioner	1
Sample room air conditioner	1
Radwaste water chemistry sampling cooler	1
Turbine plant water chemistry sample coolers	1
Extruder evaporator, distillate cooler, and lube oil cooler	1
CFS system backwash air compressor (cooling)	1
Reactor feed pump seal heat exchangers	6

9.3 PROCESS AUXILIARIES

9.3.1 Compressed Air Systems

The compressed air systems consist of the instrument air system (IAS), the service air system (SAS), and the breathing air system (AAS). The SAS and AAS are used only for nonsafety-related equipment and components during normal plant operation. The IAS system supplies air to both safety-related and nonsafety-related equipment and components.

The IAS is described in Section 9.3.1.1, followed by descriptions of the SAS and the AAS in Sections 9.3.1.2 and 9.3.1.3, respectively.

All instrumentation and control systems located inside the reactor primary containment, including the safety-related equipment and components of the automatic depressurization system (ADS) and the four inboard MSIV actuator accumulators, are independently supplied with nitrogen gas from the instrument nitrogen system (GSN). The ADS and the GSN systems are described in Sections 9.3.1.4 and 9.3.1.5, respectively.

The four outboard MSIV actuator accumulators are supplied with air from the reactor building instrument air receiving tank.

9.3.1.1 Instrument Air System

9.3.1.1.1 Design Bases

<u>Safety Design Basis</u>

The IAS provides air to both safety-related and nonsafety-related components. However, the system is not considered a safety-related control air system since the safety-related components that it supplies all perform their safety functions without or upon loss of air, or are provided with safety-related accumulators capable of supplying the required quantities of air.

Although retaining the IAS designation, the pneumatic supply to the ADS is provided from the GSN system. This is described in Section 9.3.1.4.

All IAS piping, valves, and fittings located in Category I areas are seismically analyzed and supported in accordance with SSE design requirements so that their failure will not damage safety-related equipment, piping, and components.

The IAS component design bases are given in Section 3.2.

<u>Power Generation Bases</u>

The IAS is designed to supply clean, dry, and oil-free air at 80 to 100 psig to all plant instrumentation and control systems that

require an air supply. However, all instrumentation and control systems located inside the reactor primary containment are supplied with clean, moisture-free nitrogen gas at 80 to 100 psig from the GSN system.

The instrument air prefilter, air dryer, and afterfilter assembly installed in the IAS provides dehydrated, oil-free, and filtered air that meets or exceeds the following design basis parameters: a dew point measured at the system dryer outlet not exceeding 39°F, and with 90 percent and 98 percent of all solid contaminants larger in size than 1 micron and 10 microns, respectively, removed from the airstream.

All piping and fittings associated with the IAS are either stainless steel, carbon steel, red brass, or a combination of materials. Also, the IAS will be given a complete preoperational cleaning until the applicable acceptance cleanliness levels are established and verified. In addition, since the piping system materials are primarily corrosion-resistant, the cleanliness levels achieved during preoperational cleaning are expected to be maintained and controlled to within acceptable limits.

The pressure-retaining components of the IAS are designed, constructed, and inspected in accordance with ASME Section VIII, Division 1 requirements. The nonsafety-related portions of the instrument air piping system are fabricated and installed in accordance with the requirements of ANSI B31.1.

The safety-related portions of the instrumentation and control systems located inside the reactor primary containment, which are independently supplied with nitrogen gas (despite the retention of the IAS system designation) from the GSN system, are designed, constructed, and inspected in accordance with the applicable requirements of ASME Section III, Division 1, Subsection ND for Class 3 components, and Subsection NC for Class 2 components.

The loss of instrumentation and control air causes AOVs to fail to appropriate safety positions.

9.3.1.1.2 System Description

The compressed air supply for plant instrumentation and controls is provided by three instrument air compressors and three air receivers arranged in parallel trains with a common discharge header. This common compressed air supply header also supplies the plant SAS and AAS requirements (Figures 9.3-1 and 9.3-3).

Each two-stage instrument air compressor assembly includes an intercooler, aftercooler, and air receiver tank. The instrument air compressors are of the nonlubricated type. The intake air filters and silencers are included in the air compressor assembly. Cooling water is supplied to the air compressor cooling water system and heat exchangers from the RBCLCW system.

Two heated desiccant-type air dryers, each with a companion prefilter and afterfilter, are provided on the IAS supply header to filter and dry the air to a nominal dew point on the order of -40°F at 100 psig.

The IAS distribution piping network is supplied from a separate instrument air receiver tank located downstream of the dryers and air filters. Instrument air used only for nonsafety-related instrumentation and control systems is distributed throughout the plant from this air receiver tank.

The SAS supply header is branched off the common compressed air supply header upstream and downstream of the instrument air dryers. This allows service air to be supplied with dry air from the dryers or with air that has not been through the dryers to lessen the air dryer load. An isolation valve on the SAS main supply branch header will automatically close and shut off the service air supply when the common compressed air supply header pressure decreases to less than 85 psig. The automatic shutoff and isolation of the SAS is designed to prevent decreased compressed air supply header pressure, as during high service air demand flows, which may adversely affect the operability of the IAS.

The AAS supply header is branched off the common compressed air supply header upstream of the instrument air dryers and afterfilters. A pressure control valve on the AAS main supply branch header will regulate the supply from the IAS to a pressure level which is acceptable for use in the AAS system.

Each instrument air compressor has automatic loading/unloading controls and a local operate/off selector switch for maintenance start-stop operation. In the selected lead position the compressor starts, loads, and automatically unloads to maintain its air discharge pressure. In the lag position the compressor will automatically start when the common air supply header pressure drops below the low pressure setpoint of 100 psig. In the backup position, the compressor will automatically start when the air supply header pressure decays below the low-low pressure setpoint of 85 psig. The air compressors are operated from the normal plant power supply. Cooling water to the air compressor cooling system heat exchangers is supplied by the RBCLC system (Section 9.2.2.1).

9.3.1.1.3 Safety Evaluation

The three instrument air compressors, i.e., the lead, lag, and backup units, operate to maintain the required pressure at the air receivers which provide the Station's IAS with an air supply pressure of 100 to 125 psig.

Actuation of the selected lag unit is automatic when the compressed air supply header pressure decreases below 100 psig,

and the selected backup unit automatically starts when the header pressure further decays below 85 psig.

The instrument air compressors and air receivers have sufficient capacities to supply the requirements of plant instrumentation and control systems. The loss of instrument and control air causes AOVs to fail to appropriate safe positions.

9.3.1.1.4 Inspection and Testing Requirements

The IAS operates on a continuous basis. It is maintained and monitored, and abnormal conditions are alarmed during normal plant operation.

The IAS will be tested in accordance with the applicable requirements of RG 1.68.

Preoperational testing of the IAS is addressed in Table 14.2-43.

The actual system utilizes nonlubricated compressors, followed by desiccant dryers, each with a companion afterfilter which removes 98 percent of all particles greater than 0.07 microns in size. Therefore, no foreign material is expected to be injected into the system. However, to ensure clean air, a sample that meets the requirements of ANSI MC11.1-1976 for particle size and oil content will be performed on an annual basis.

NOTE: Since all safety-related components that use air or nitrogen have a particle size limit of no greater than 40 microns, Unit 2 will take exception to the ANSI MC11.1-1976 particle size limit. The new Unit 2 limit for particle size is 40 microns.

9.3.1.1.5 Instrumentation Requirements

Description

Manual and automatic controls are provided for maintaining an adequate supply of instrument air for air-operated instruments, equipment, and components. The controls and monitors described below are located in the main control room. The control system logic is shown on Figure 9.3-2.

Operation

Each instrument air compressor control system has a full load/no load electropneumatic regulator designed to provide manual and automatic running. The control system provides circuits for a 15-sec loading delay after starting to ensure that full compressor speed is reached before the load is applied.

The instrument air compressor cooling water supply valve automatically opens when the compressor is started and closes

when the compressor is stopped or secured. The low lubricating oil pressure trip is bypassed for 15 sec after startup.

The instrument air compressor selected as the lead unit is started manually. Selection of the lead unit places the other two units in the automatic mode as the lag and backup units. The lag compressor will automatically start when the compressed air supply header pressure decays to the low-pressure setpoint. Similarly, the backup compressor will automatically start when the air supply header pressure decreases to the low-low setpoint.

The instrument air compressors can be stopped manually. An automatic compressor stop signal is initiated by any of the following: high HP inlet air temperature, high oil temperature, high LP outlet air temperature, high HP outlet air temperature, low lubrication oil pressure after a 15-sec time delay, drive motor overload, vent fan motor overload, sustained power bus undervoltage, instantaneous/time overcurrent electrical fault.

Monitoring

Control room indicators and alarms are provided as follows:

1. Running (red) and stopped (green) indicating lights for instrument air compressors.

2. Pressure indicator for the IAS instrument air receiving tank.

3. Annunciators for the following:

    a. Instrument air compressor autotrip/fail to start.

    b. Instrument air compressor autostart.

    c. Instrument air trouble.

    d. Instrument air TK2 pressure low.

    e. Instrument air TK3 pressure low.

9.3.1.2 Service Air System

9.3.1.2.1 Design Bases

Safety Design Basis

The SAS is not required to effect or support the safe shutdown of the reactor or to perform in the operation of reactor safety features. However, all items contained within Category I areas are seismically analyzed and supported for SSE conditions so that failures will not damage safety-related equipment. For the containment penetrations, see Section 3.2.

Power Generation Design Basis

The SAS is designed to provide service air to plant services at 80 to 100 psig.

9.3.1.2.2 System Description

The compressed air supply to the SAS is provided by the same three air compressors that supply the IAS (Section 9.3.1.1). Service air is supplied from the common compressed air supply header upstream of or downstream from the instrument air dryers. This allows service air to be supplied with dry air from the dryers or with air that has not been through the dryers to lessen the air dryer load.

Service air is distributed to various plant areas by supply branch headers, each fitted with normal root valves. The service air piping at each building is designed in a loop arrangement. Typically piping terminals at the service air stations are provided with a ball valve and a 1/2-in or larger hose coupling. There are swing-type check valves at all interfaces between the IAS and the SAS to prevent reverse flow. Primary containment penetrations have one isolation valve inside the primary containment and one outside, in accordance with 10CFR50 Appendix A. Flexible hoses with quick disconnect couplings are part of the design arrangement. When the unit is on-line, the SAS is disconnected from the primary containment by way of these flexible hoses. This ensures that air cannot enter the containment.

9.3.1.2.3 Safety Evaluation

During normal power plant operation, the primary containment isolation valves are closed and the flexible hoses are disconnected from the service air supply. As a result, the piping within the containment will be isolated. In an accident condition the containment temperature may rise, increasing the piping internal pressures. Therefore, piping inside the primary containment is protected by thermal relief valves. The piping outside the primary containment is protected by the air compressor and receiver relief valves. Piping in Category I areas has been designed so that its failure will prevent damage to safety-related equipment. With the exception of the containment penetrations and isolation valves, the SAS is nonsafety related.

9.3.1.2.4 Inspection and Testing Requirements

No special inspection and testing are required following preoperational testing except for Appendix J testing (Section 6.2) and ISI of the containment penetrations (Section 6.6).

9.3.1.2.5 Instrumentation Requirements

Description

The air supply for the SAS is provided by the IAS. The only controls and monitors for the SAS are an instrument air/service air-operated isolation (globe) valve and its associated alarm which is located in the main control room. The control logic is shown on Figure 9.3-2.

Operation

In the normal mode, the SAS isolation valve can be opened locally at LCS738 only if a low-low pressure condition does not exist. The valve will close automatically on low-low pressure. The valve can be opened and closed manually.

Monitoring

An alarm is provided for SAS isolation valve closure.

9.3.1.3 Breathing Air System

The AAS provides clean, dry, oil-free air to various areas throughout the plant for breathing.

9.3.1.3.1 Design Bases

Safety Design Bases

The AAS is not required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features. However, all items contained in Category I areas are seismically analyzed and supported for SSE conditions so that their failure will not damage safety-related equipment. For containment penetrations, see Section 3.2.

Operational Design Basis

The AAS has been designed to provide air suitable for breathing at selected breathing air stations for use by unit personnel during potential or actual airborne contamination situations.

9.3.1.3.2 System Description

Compressed air is supplied to the AAS by the same three air compressors that supply the IAS (Section 9.3.1.1). Breathing air is supplied from the common compressed air supply header upstream of the instrument air dryers.

Air quality is maintained by an in-line, four-stage filtration unit that removes oil, water, particulates, and carbon monoxide, and delivers clean air that meets OSHA requirements for breathing air.

During normal unit power operation, breathing air piping within the primary containment will be physically disconnected by a flexible hose connection and isolated by valves inside and outside the containment.

Each breathing air station consists of an isolation valve, a pressure regulator with a pressure gauge, a particulate filter with a relief valve, a manifold assembly, and one-way shutoff couplings.

9.3.1.3.3 Safety Evaluation

The AAS is not required to effect or support safe shutdown of the reactor. The piping within the containment is protected from overpressure by thermal relief valves as are the piping and equipment outside the containment. All items contained within Category I areas are supported so that their failure during a SSE will not damage any safety-related equipment.

Originally, the FMEA for the containment penetrations and isolation valves of the AAS was contained in the Unit 2 FSAR FMEA document, which is historical. FMEAs for plant systems are now performed and controlled by the design process.

9.3.1.3.4 Testing and Inspection

Equipment is inspected and system performance is verified periodically and after maintenance. In addition, the containment penetrations are subject to Appendix J testing (Section 6.2) and ISI (Section 6.6).

9.3.1.3.5 Instrumentation Requirements

Description

Instruments are provided for the AAS. The monitors are located in the main control room. The logic is shown on Figure 9.3-4.

Operation

The AAS is supplied by the IAS through a pressure-reducing station. The pressure-reducing station includes manual isolation valves, a pressure-reducing valve, and a manual bypass valve. When the AAS is required, the normally closed manual isolation valves will be opened. The pressure-reducing valve is manually set to deliver 85 psig air to the AAS. The air will be filtered and routed through dryers to ensure proper air quality is maintained. In addition, a receiver tank is provided near the tie-in from the IAS to ensure that the pressure of the breathing air is maintained as designed.

Monitoring

An indicator is provided for breathing air header pressure.

Alarms are provided for:

1. Breathing air and header pressure high or low (common alarm).

2. Reactor building accumulator tank pressure low (common alarm).

9.3.1.4 Automatic Depressurization System

9.3.1.4.1 Design Bases

Safety Design Bases

The ADS is required to effect or support the safe shutdown of the reactor or to perform operational functions associated with the reactor safety system.

Main steam blowdown from a selected group of seven designated ADS SRVs, consisting of three or four SRVs in each of two subgroups. The operation of ADS, with the low-pressure coolant injection (LPCI) mode of the RHR system and/or LPCS system, functions as an alternate to the operation of the HPCS system for protection against fuel cladding damage upon LOCA. The blowdown of the main steam SRVs provides depressurization of the RPV, permitting the operation of the LPCI mode and/or the LPCS. Blowdown is activated automatically upon a signal of low reactor vessel water level (trip level-1) by the ADS. The ADS can be manually initiated or automatic initiation disabled from the Station main control room at any time.

The ADS is independently supplied with nitrogen gas (although the IAS system designation has been retained) from the high-pressure GSN system. All pressure-retaining components, piping, valves, and fittings in the ADS are seismically analyzed and supported in accordance with SSE design requirements. The system is designed to be functional in the event of a SSE.

The ADS is safety related, and all pressure-retaining components of the system are designed, constructed, and inspected in accordance with the applicable requirements of ASME Section III, Division 1, Subsection ND for Class 3 components, and Subsection NC for Class 2 components. Not included in this safety-related classification are the nitrogen gas storage tanks, equipment, and components located in the yard outside the reactor building.

Piping segments that penetrate the primary containment and serve as a containment boundary are designed to Safety Class 2, Category I requirements.

The loss of nitrogen gas for instrumentation and controls causes gas-operated valves to fail to appropriate safe positions. In the event that the nitrogen gas supply from the nitrogen gas

storage tanks is lost, a 5-day supply is available to the accumulators from ADS nitrogen receiver tanks 2IAS*TK4(Z-) and 2IAS*TK5(Z-). In addition, there are provisions for recharging the ADS nitrogen receiver tanks through its individual supply lines located in a missile-protected area outside the standby gas treatment building from special emergency tube trailer supply connections. These special, emergency recharging lines are part of the GSN system and are classified seismic Category I, Safety Class 3.

Power Generation Bases

The ADS requires clean, dry, oil-free nitrogen gas at approximately 170 psig to be supplied to the selected group of seven main steam SRVs and their respective accumulators located inside the reactor primary containment. This designated group of ADS SRVs and accumulators is divided into two subgroups with three or four valves and accumulators in each subgroup. Each subgroup is supplied with nitrogen gas from one of two separate ADS receiver tanks. Each ADS receiver tank is supplied with nitrogen gas at 355 psig from a bank of six horizontal, high-pressure nitrogen gas storage tanks located outside the reactor building. Nitrogen gas supplied for instrumentation and controls meets or exceeds the equivalent air quality requirements established for safety-related control air systems (SRCAS) by ANSI MC11.1-1975 (approved January 15, 1976) (ISA-S7.3), Quality Standard for Instrument Air.

NOTE: Since all safety-related components that use air or nitrogen have a particle size limit of no greater than 40 microns, Unit 2 will take exception to the ANSI MC11.1-1976 particle size limit. The new Unit 2 limit for particle size is 40 microns.

All piping, valves, and fittings associated with the ADS are of stainless steel materials. Also, the system will be given a complete preoperational cleaning until the applicable acceptance cleanliness levels are established and verified. In addition, since the piping system materials are corrosion-resistance, the cleanliness levels achieved during preoperational cleaning are expected to be maintained and controlled to within acceptable limits.

Each of the six high-pressure nitrogen gas storage vessels is designed, fabricated, tested, and stamped in accordance with the ASME Unfired Pressure Vessel Code, Section VIII, and conforms to Code Case No. 1205 for seamless integrally forged vessels. The six-vessel modular assembly is provided with manifold isolation valves which separate three active vessels and three reserve vessels. A fill stanchion is provided for refueling from a high-pressure tube trailer.

9.3.1.4.2 System Description

The ADS is supplied with nitrogen gas from a bank of six horizontal, high-pressure nitrogen gas storage tanks located outside the reactor building. Nitrogen gas is supplied to two ADS nitrogen receiver tanks at 355 psig. Each ADS nitrogen receiver tank supplies nitrogen gas to its corresponding subgroup of either three or four ADS valves and accumulators through a 355/170 psig pressure-reducing station. These two ADS nitrogen receiver tanks provide makeup nitrogen to compensate for valve leakage losses and to maintain the required pressure at the accumulators.

A diaphragm-type ADS air compressor, originally provided to supply instrument quality air for testing purposes, has been abandoned in place.

9.3.1.4.3 Safety Evaluation

The two ADS nitrogen receiver tanks supplied by the bank of six horizontal nitrogen gas storage tanks will operate to maintain the ADS valve accumulator pressure at approximately 170 psig. The maximum/minimum analytical values for the pneumatic system are 186/160 psig, respectively (Three Mile Island [TMI] Issue II.K.3.28).

Each nitrogen gas accumulator provides a passive safeguard which automatically supplies a motive source for the operation of each SRV in the ADS. The failure of a pilot control valve in the valve actuators can only affect a single SRV. This is due to the independence of the other, and a postulated single failure does not prevent the operation of the remaining units. The design bases covering the SRVs include additional allowances for the malfunction of any one valve.

For inventory threatening events requiring ADS, once the reactor vessel has been depressurized by the ADS and the core has been reflooded, three operable ADS valves are sufficient to maintain the system pressure below the shutoff pressure of the emergency core cooling system (ECCS) pumps. Therefore, breaks outside the primary containment that result in reactor isolation and small line breaks require the full ADS for the longest period of time. Based on a review of the Unit 2 LOCA analysis, it was determined that the full ADS is required for 30 min following a LOCA.

The ADS is designed to withstand an accident environment and still perform its safety-related functions for 100 days following an accident. An emergency nitrogen tube trailer supply connection is provided in the system to ensure that this long-term ADS operational requirement is met.

Long-term post-accident operation requires only three of the seven valves to be operable. A single failure in one of the nitrogen gas supply lines will not prevent the ADS from performing its safety-related function in either the immediate or long-term modes.

Originally, the FMEA of the ADS was contained in the Unit 2 FMEA document, which is historical. FMEAs for plant systems are now performed and controlled by the design process.

9.3.1.4.4 Inspection and Testing Requirements

The ADS is normally available for operation. The required system operating pressures are maintained and monitored, and abnormal conditions are alarmed during normal plant operation.

The ADS will be tested in accordance with Table 14.2-52. To verify that the loss of nitrogen gas will not prevent the ADS from performing its safety-related function, a loss-of-nitrogen gas-supply test will be performed.

9.3.1.4.5 Instrumentation Requirements

<u>Description</u>

Instrumentation and controls are provided for the manual and automatic operation of the ADS system. The controls and monitors described below are located in the main control room.

<u>Operation</u>

The ADS primary containment isolation valves will close when any of the following exists: the control switch is placed in closed position, a LOCA isolation signal occurs and the LOCA keylock manual override control switch is in the reset position, or the manual isolation switch is activated.

The ADS primary containment isolation valves can be opened manually when both of the following exist: a LOCA isolation signal is absent or the LOCA keylock manual override is activated, and the primary containment manual isolation switch is in the reset position. Additionally, two ADS primary containment isolation valves which supply ADS accumulator tanks are controlled from the remote shutdown panel (RSP).

The ADS low flow solenoid-operated nitrogen gas supply valve, installed in parallel with the high flow solenoid-operated valve (SOV) at each of the two supply headers to the SRV accumulators, can be manually opened and closed. These low flow valves compensate for nitrogen gas losses from small, normal leakages in the ADS system and will automatically open when its header pressure decreases below the low-pressure setpoint and close when the header pressure is restored above the low-pressure setpoint.

Operation of the low flow control valve provides the assurance that a full inventory of nitrogen gas is available at the accumulators at all times to cycle the ADS SRVs, if required.

The ADS high flow SOV, installed in parallel with the low flow SOV at each of the two air supply headers to the SRV accumulators, can be manually opened and closed. The high flow air supply valve will automatically open when its header pressure decays below the high flow setpoint and close when the pressure is increased above the high flow setpoint.

Monitoring

Control room indicators and alarms are provided as follows:

1. Running (red) and stopped (green) indicating lights for the ADS air compressor (not used during normal plant operation).

2. Open (red) and closed (green) indicating lights for the following:

    a. ADS primary containment isolation valves.

    b. ADS header high flow valves.

    c. ADS header low flow valves.

3. Inoperable (amber) indicating lights for the following:

    a. ADS isolation valves bypass/inoperable.

    b. ADS control valve power failure.

    c. ADS systems manually out of service.

4. Pressure indicators for the following:

    a. ADS nitrogen supply headers.

    b. ADS nitrogen receiver tanks.

5. Annunciators for the following:

    a. ADS air compressor autotrip/fail to start (not used during normal plant operation).

    b. ADS air compressor autostart (not used during normal plant operation).

    c. Primary containment isolation valve power failure.

    d. ADS supply nitrogen systems bypassed or inoperable.

    e. Keylock LOCA override.

    f. ADS trouble.

g. ADS nitrogen supply header pressure low.

h. ADS primary containment manual isolation.

i. Nitrogen to inboard MSIV/SRV pressure low.

9.3.1.5 Instrument Nitrogen System

9.3.1.5.1 Design Bases

Safety Design Basis

Instrumentation and control systems located inside the reactor primary containment are supplied with nitrogen gas at 120 psig from the GSN system. The IAS designation is retained for these systems which are nitrogen gas exclusively during normal plant operation.

Instrumentation and control systems located inside the reactor primary containment, except as described in Section 9.3.1.4.5, are not safety related. However, all piping, valves, and fittings located in Category I areas are seismically analyzed and supported in accordance with SSE design requirements so that their failure will not damage safety-related equipment. For containment penetrations and items within the containment areas, see Section 3.2.

Power Generation Design Bases

Nitrogen gas for instrumentation and control systems located inside the reactor primary containment areas is supplied from the vapor spaces of two 11,000-gal liquid nitrogen vertical storage tanks maintained under a constant pressure of approximately 200 psig. The liquid nitrogen tanks are located in the yard area, north-northeast of the reactor building, alongside the railroad access lock. From the liquid nitrogen tanks, nitrogen flows through an active bank of finned ambient vaporizers, a trim heater for heating to 100°F, and a pressure-reducing station that reduces the nitrogen gas pressure to 120 psig. An instrument nitrogen receiver is provided inside the reactor building for additional storage capacity. Nitrogen gas for instrumentation and controls inside the primary containment is distributed from this nitrogen receiver.

A nitrogen gas backup supply connection is provided from the high-pressure nitrogen gas storage cylinders to the instrument nitrogen receiver through a 355/100 psig pressure-reducing station.

Although instrumentation and control systems within the reactor primary containment are nonsafety related, the nitrogen gas supplied for these systems meets or exceeds the quality

requirements of ANSI MC11.1-1975 (approved January 15, 1976) (ISA-S7.3), Quality Standard Instrument Air.

NOTE: Since all safety-related components that use air or nitrogen have a particle size limit of no greater than 40 microns, Unit 2 will take exception to the ANSI MC11.1-1976 particle size limit. The new Unit 2 limit for particle size is 40 microns.

Additionally, these piping systems will receive a preoperational cleaning for the removal of contaminants and to provide the required cleanliness level required. Also, all piping associated with these systems is of stainless steel, which eliminates the potential for particulate contamination coming from the piping material.

The pressure-retaining components of the instrumentation and control systems located inside the reactor primary containment are designed, constructed, and inspected in accordance with the applicable requirements of ASME Section III, Division 1, Subsection ND for Class 3 components, and Subsection NC for Class 2 components. Piping segments that penetrate the primary containment boundary are designed to Safety Class 2, seismic Category I requirements.

Each 11,000-gal liquid nitrogen storage tank contains the equivalent gas capacity of 1,024,000 SCF of nitrogen. Each storage vessel consists of a Type 304 stainless steel inner tank fabricated to Section VIII of the ASME Code requirements, an outer carbon steel jacket, and an annular space under vacuum filled with superinsulation for maintaining a low normal evaporation rate of approximately 0.15 percent per day. The normal operating pressure of the liquid nitrogen storage tank is 190 psig.

9.3.1.5.2 System Description

The nitrogen gas supply for instrumentation and controls within the reactor primary containment areas is provided by the two 11,000-gal liquid nitrogen vertical storage tanks located in the yard area (see Figure 9.3-20 for system P&IDs). These two liquid nitrogen storage tanks also supply nitrogen gas for inerting the primary containment when required. Additionally, a low-pressure slipstream from the nitrogen gas system maintains the reactor primary containment atmosphere inerted during normal plant operations. A nitrogen gas backup connection to the GSN system is provided from the high-pressure nitrogen gas storage cylinders through a 355/100 psig pressure-reducing station during off-normal conditions for instrument nitrogen supply.

The nitrogen gas supply for instrumentation and controls is normally drawn off the vapor spaces of the liquid nitrogen storage tanks, absorbs heat energy from the surrounding environment across an active bank of finned ambient vaporizers,

heated to 100°F through one of two electric trim heaters, and its pressure reduced from 190 psig to 120 psig.

The GSN system is provided with all necessary pressure and temperature indicators, alarms, and safety relief devices for safe and reliable operation. A solenoid-operated temperature control valve installed in the main supply header is set to close if the nitrogen gas temperature drops to the low temperature setpoint. This is a safeguard feature that prevents the flow of low temperature nitrogen gas into the piping distribution system in the event of a trim heater failure.

An instrument nitrogen receiver is provided inside the reactor building for additional storage capacity. Nitrogen gas is distributed throughout the instrumentation and control systems piping network within the reactor primary containment areas from this receiver.

9.3.1.5.3 Safety Evaluation

The Station nitrogen systems operate to supply the instrumentation and control systems inside the reactor primary containment with instrument quality nitrogen gas at 120 psig.

The liquid nitrogen storage tanks and the six modular nitrogen gas cylinders, nitrogen gas receivers, and accumulators have sufficient capacities to supply the requirements of the instrumentation and control systems within the reactor primary containment.

The principal gas-operated valves supplied with instrument nitrogen gas inside the primary containment areas are the inboard MSIVs, the main steam SRVs, and the drywell vacuum breakers. The selected main steam SRVs in the ADS are independently supplied with nitrogen gas from the high-pressure nitrogen gas storage cylinders described in the previous section, Section 9.3.1.4.

Each accumulator in the GSN system provides a passive safeguard which automatically supplies a motive power source for the operation of each SRV. The failure of a pilot control valve in the valve actuators is limited to that particular single SRV, as each SRV is independent of the others; and a postulated single failure does not interfere with the operation of the remaining units. The design bases covering the total number of SRVs include additional allowances for the malfunction of any one valve. Nitrogen gas availability to the SRVs following a SBO is addressed in Section 8.3.1.5.

Originally, the FMEA of the GSN systems, as part of the overall IAS systems, was contained in the Unit 2 FMEA document, which is historical. FMEAs for plant systems are now performed and controlled by the design process.

To prevent introducing cold (less than 40°F) nitrogen into the primary containment, the nitrogen temperature for normal inerting is controlled to 100°F and monitored upstream of the normal vent and purge lines. The inerting piping located in the yard area is provided with electric heat tracing to maintain the nitrogen inerting piping temperature at 100°F as shown on Figure 9.3-20. Low nitrogen temperature (70°F) is alarmed in the control room. Should the temperature continue to fall to 40°F at the outlet of the vaporizer, an independent temperature device will trip the outlet control valve closed. The nitrogen supply to the GSN system is fed from nitrogen storage bottles and the ambient vaporizer is followed by trim heaters to hold the temperature at 100°F. The supply is fed to an accumulator prior to any containment penetration, thus essentially precluding any cold nitrogen from entering the containment. In addition, a temperature-sensing device just downstream of the trim heater will trip the downstream valve closed if the temperature drops below 40°F. In addition, there is no equipment or piping in the direct path of the injected nitrogen in either the drywell or wetwell, and the nitrogen system is normally isolated from the primary containment. Inerting is controlled administratively, and the valves are returned to a closed position after inerting.

9.3.1.5.4 Inspection and Testing Requirements

The GSN system is operated on a continuous basis. It is maintained and monitored, with off-normal conditions alarmed during normal plant operation.

The instrumentation and control systems within the reactor primary containment will be tested in accordance with the requirements of the applicable regulatory positions of RG 1.68.3, as discussed in Table 1.8-1.

9.3.1.5.5 Instrumentation Requirements

Description

Instrumentation and controls are provided for the manual and automatic operation of the GSN systems within the reactor primary containment areas. These controls and monitors are described below.

The nitrogen gas system is placed in operation manually, including the trim heaters. In normal operation, only one of two trim heaters is used to control the nitrogen gas temperature. The other trim heater is held on standby.

The GSN systems primary containment isolation valve closes when any of the following conditions exist: the control switch is placed in the close position, a LOCA isolation signal is present, or the manual isolation switch is activated. The isolation valve can be manually opened when a LOCA isolation signal is not present and the manual isolation switch is not activated.

Monitoring

Control room indications are provided for the following functions:

1. Open (red) and closed (green) indicating lights for the GSN system primary containment isolation valve.

2. Power failure (amber) indicating light for the GSN system primary containment isolation valve.

3. Annunciators for the following:

    a. GSN system trouble.

    b. Primary containment nitrogen gas purge temperature low.

9.3.2 Process and Post-accident Sampling Systems

The plant process sampling system consists of three subsystems: the reactor plant sample system, the turbine plant sample system, and the radwaste sample system. Miscellaneous sample points are provided on individual process systems where needed. These sample points and their associated equipment are considered as part of the parent process system, not the subsystems. The post-accident sampling system (PASS) is discussed in Section 1.10, Section II.B.3. This system will not be fully tested and operational prior to fuel load. The system will be appropriately tested prior to exceeding 5 percent of full reactor power. Since relatively low amounts of radioactivity and decay heat are generated from testing up to 5 percent power, the system is not needed in advance of this.

9.3.2.1 Design Bases

The plant process sampling systems are designed:

1. To continuously monitor selected plant process streams.

2. To provide grab sample points to back up the continuous analyzers and allow laboratory analysis of other process streams.

3. To provide representative samples.

4. To minimize coolant loss in case of a leak by using small diameter tubing.

5. To ensure tank samples are representative by sampling tank bulk volumes and not low points or sediment traps.

6. To ensure that piping and valves between the sample source and the outer isolation valve conform to ASME Section III, Safety Classes 1, 2, and 3 as appropriate, and that tubing and valves downstream of the outer containment isolation valve are designed in accordance with ANSI B31.1 requirements.

7. To ensure proper temperature compensation of conductivity measurements.

8. To ensure adequate sampling provisions for gaseous process systems including heat tracing where needed to preclude isotopic plateout.

9.3.2.2 System Description

The process sampling subsystems are shown on Figure 9.3-5. Table 9.3-1 shows which points are sampled, sample parameters provided for each sample point, and the purpose of each sample point. Sample equipment is arranged in the following manner:

1. Reactor plant sample system: five panels for all sample points on systems in the reactor building.

2. Turbine plant sample system: four main panels for all sample points on systems in the turbine building.

3. Radwaste sample system: four panels for all liquid radioactive waste and two local sample stations for solid radioactive waste sample points in the radwaste building.

4. Miscellaneous samples: Several other samples are provided besides those that are part of the reactor, turbine, or radwaste sample subsystems. These samples are as follows:

    a. Offgas system (Section 11.5).

    b. Drywell atmosphere (Sections 5.2.5 and 6.2.5).

    c. Drywell floor drain system: Grab samples may be drawn from manually-operated drain valves on the discharge of the drywell floor drain pumps or from the pump casing drain.

    d. Drywell and reactor building equipment drain system: A conductivity element (insertion type) is installed downstream of the drywell and reactor building equipment drain pumps on the common line to the radwaste building to monitor the impurities present in the drywell and reactor building equipment drainage. Grab samples may

also be drawn on the suction of each drain pump or from the drain pump casing.

e. Suppression pool: Grab samples may be drawn by sampling the RHR system while testing the RHR pumps from manually-operated valves in the main suction lines for the LPCS and LPCI (RHR) systems, or by direct sample from the suppression pool when the primary containment is open for access.

f. Standby liquid control system: Grab samples may be drawn by opening a manhole on top of the SLC tank and sampling directly.

g. Spent fuel pool cooling and cleanup system: Conductivity elements (insertion type) are installed in the discharge of each SFC filter and also in the SFC common pump discharge to the filters. These conductivity elements measure SFC filter efficiency and monitor spent fuel pool water quality.

h. Gland seal and exhaust steam (TME) system (Section 11.5).

i. Condenser air removal hogging system (Section 11.5).

j. Waste solidification system (WSS) (Section 11.4).

k. Other sample points identified in Table 9.3-1.

High temperature samples (steam samples or liquid samples at temperatures greater than 180°F) are condensed and/or subcooled using local (close to the sample source) coolers. These coolers are Type 316 stainless steel and are rated at 5,000 psig at 1,000°F. Their maximum working conditions exceed the design conditions of all sample sources.

All samples entering the sample panels are cooled sufficiently to ensure Operator safety. Each sample line has an AOV close to the sample source. These AOVs are manually activated by pneumatic valves from the sample panel. In the reactor and turbine sample systems, the high temperature samples have temperature switches that will automatically close the AOV if the sample temperature in the panel exceeds 180°F. In the radwaste sample system, flow switches are installed on the outlet cooling water lines for the sample coolers. The flow switches will close the sample line isolation AOVs if low cooling water flow is sensed. All AOVs fail closed upon loss of control power or air supply pressure. Manually-operated, rod-in-tube pressure-reducing valves and automatic pressure-reducing valves reduce any residual high sample pressure in the sample panel. Manual needle or globe valves regulate final grab sample flow. To ensure proper

temperature compensation of conductivity measurements, conductivity samples are conditioned to 77 ±5°F by a constant temperature bath prior to in-line analysis. Suitable panel instrumentation is provided to allow proper sample system operation and to ensure the safety of the Operator. Grab sample sinks have ventilated fume hoods to collect any airborne contamination. All sample panels are located in low radiation areas to reduce Operator exposure. All liquid sample drainage is directed to the respective building equipment drain system or is collected and returned to the plant.

To provide representative samples, all sample lines are sized to maintain Reynolds numbers in excess of 4,000 (fully turbulent flow). Sample tubing runs are as short as possible and are sized to allow the highest practicable velocities. All tubing enters the top of the sample panel, thereby allowing the final leg of tubing to be downward in direction. To minimize coolant loss in case of a leak, tubing with an internal diameter of 0.18 in (1/4 in OD reactor and turbine sample systems) and 0.245 in (3/8 in OD radwaste sample system) is used. The tubing is at least ASTM A213 Grade Type 316 stainless steel rated at a minimum of 4,812 psig at 700°F. These maximum working conditions exceed the design condition of all sample sources. Incoloy 825 tubing is used on the radwaste evaporator sample lines. This material was selected due to its corrosion resistance to sodium sulfate solution being sampled.

Parent process piping is fitted with sample probes or wall taps in turbulent flow zones to ensure representative samples. All continuous samples have bypass purge lines around the analyzers. Grab samples are purged to hooded sinks.

9.3.2.3 Safety Evaluation

The process sampling systems are not required to function during or following an accident, nor are they required to safely shut down the reactor.

9.3.2.4 Inspection and Testing Requirements

Nearly all process sampling system components are used regularly during power operation or during shutdown, thereby providing continuous assurance of system availability and performance. Routine calibration checks are performed on the continuous analyzers to ensure accurate indications and alarm functions.

9.3.2.5 Instrumentation Requirements

9.3.2.5.1 Reactor Plant Sample System

<u>Description</u>

Instruments and controls are provided to monitor the quality of reactor coolant and various reactor plant fluid systems. The controls described below are situated locally. Except where noted, the monitors described below are located in the main control room. The control logic is shown on Figure 9.3-6.

Operation

Temperature and flow rate indicators are provided on the sample panel in the reactor building to indicate that samples are properly conditioned for sampling purposes.

Sample valves for sampling lines are opened and closed manually. The sample valves for the RHR heat exchangers and the reactor recirculation inlet samples automatically trip closed on high sample temperature to prevent excessive downstream temperature in the sample line.

Monitoring

Recorders are provided for:

1. RWCU filter demineralizer inlet and outlet conductivity.

2. RWCU filter demineralizer inlet oxygen.

3. Reactor recirculation inlet Loop A conductivity.

4. Reactor recirculation inlet Loop A oxygen.

5. CRD common filter discharge conductivity.

6. CRD common filter discharge oxygen.

Alarms are provided for:

1. RWCU filter demineralizer inlet conductivity high and low.

2. RWCU filter demineralizer outlet conductivity high and low.

3. Reactor plant sample system trouble.

9.3.2.5.2 Turbine Plant Sample System

Description

Instruments are provided for monitoring the quality of reactor grade water flowing in the turbine building. The monitors described below are located in the main control room. The control logic is shown on Figure 9.3-7.

<u>Monitoring</u>

Indicators are provided for feedwater heater drain system turbidity samples.

Recorders are provided for:

1. CND system common influent sample conductivity, effluent sample oxygen, and effluent sample conductivity.

2. Feedwater system sixth-point heater discharge sample conductivity, oxygen, and pH.

Alarms are provided for:

1. Turbine plant sample system trouble.

2. CND system effluent sample conductivity high and high/high.

9.3.2.5.3 Radwaste Sample System

<u>Description</u>

The radwaste sampling system provides a means in the radwaste building for obtaining grab samples for monitoring the radioactive LWS system and the radwaste auxiliary steam system drain coolers. Manual controls are provided for the radwaste sampling system. The controls described below are located on local sample panels. The control logic is shown on Figure 9.3-8.

<u>Operation</u>

Manual controls are provided at local radwaste sample panels for the following sample valves:

1. Waste collector tank effluent pumps discharge.

2. Effluent from demineralizers.

3. Common effluent from filters.

4. Floor drain collector tanks at pump discharge.

5. Floor drain filter effluent pump discharge.

6. Recovery sample pumps discharge.

7. Regenerative waste pumps discharge.

8. Waste discharge sample pumps discharge.

9. Waste discharge sample pumps common discharge.

10. Floor drain collector surge pumps common discharge.

11. Waste collector surge pumps common discharge.

12. Waste evaporator recirculation pump suction.

13. Waste evaporator recirculation pump discharge.

14. Regenerative evaporator recirculation pump suction.

15. Regenerative evaporator recirculation pump discharge.

16. Waste evaporator distillate transfer pump discharge.

17. Regenerative evaporator distillate transfer pump discharge.

18. Radwaste auxiliary steam system distillate coolers.

An interlock prevents opening or automatically trips closed the waste evaporator distillate transfer pump discharge sample valve, the regenerative evaporator distillate transfer pump discharge sample valve, and the radwaste auxiliary steam system distillate cooler sample valves, when the associated sample cooler water flow is low.

9.3.3 Equipment and Floor Drain Systems

The equipment and floor drain system for the reactor building is schematically shown on Figure 9.3-9. The equipment and floor drain system for the turbine building is schematically shown on Figure 9.3-10. The equipment and floor drain system for the radwaste building is shown schematically on Figure 9.3-11. The equipment and floor drain system for miscellaneous buildings (CST building, main stack area, service building, screenwell building, control room building, diesel generator building, and auxiliary boiler building) is schematically shown on Figure 9.3-12.

The reactor building mat drain system is described in Section 3.4.1.2. The drains from the control building supply air special filter train system (Section 9.4.1) are routed to the radwaste system.

9.3.3.1 Design Basis

Low conductivity equipment drains are collected only in segregated equipment drain systems located in the reactor building and turbine building. The following are design bases of the equipment drain system:

1. Collect in sumps or tanks influent from non-oily waste from radioactive, potentially radioactive, and nonradioactive sources, such as:

a. Unit cooler condensate drains (outside drywell).

b. Valve stem leakoffs.

c. Pump base plates other than those terminating in segregated floor drainage sumps.

d. Relief valves.

e. Other similar equipment.

2. Prevent contamination of low conductivity equipment drains from high conductivity floor drains by terminating equipment drains in raised-rim drain hubs. Reactor building equipment drains flow by gravity to closed tanks vented to the ventilation system.

3. Discharge effluent of equipment drain pumps to the waste collector tanks of the radioactive liquid waste (radwaste) system for processing (Section 11.2.2).

4. Prevent the potential for inadvertent transfer of segregated low conductivity equipment drainage to floor drainage systems or to the storm drainage system.

5. Handle the calculated discharge from the equipment served by each sump/tank.

The equipment drain systems are not safety related and are designated nonnuclear safety, except for the following:

1. Drainage piping and valves inside the primary containment, which are Safety Class 2.

2. Primary containment wall penetrations with inboard and outboard isolation valves, which are Safety Class 2.

High conductivity floor drains are collected and segregated in the floor drain systems located in the reactor building and turbine building, and in floor and equipment drainage systems located in the radwaste building, CST building, and main stack area. The following are design bases of the floor drain systems:

1. Collect in sumps or tanks influent from oily or non-oily waste from radioactive and potentially radioactive, high (or potentially high) conductivity sources, such as:

a. Maintenance washdown water.

b. Miscellaneous surface spillage.

c. Equipment drainage from all sources other than those collected in the low conductivity equipment drainage system.

d. Accumulated fire protection water after actuation of any fire protection sprinklers.

2. Discharge effluent to the floor drain collector tanks of the radwaste system for processing (Section 11.2.2). There is no potential for inadvertent transfer of the effluent to the segregated low conductivity equipment drainage systems or to the storm drainage system.

3. Handle the calculated flow from the areas and equipment served by each sump.

The equipment and floor drain systems are not safety related and are designated nonnuclear safety, except for the following:

1. Drainage piping and valves inside the primary containment, which are Safety Class 2.

2. Primary containment wall penetrations with inboard and outboard isolation valves, which are Safety Class 2.

Equipment and floor drain systems are located in the control room building, screenwell building, service building, diesel generator building, and auxiliary boiler building. The following are design bases of the nonradioactive equipment and floor drain systems:

1. Collect in sumps oily or non-oily waste from nonradioactive sources, such as:

   a. Maintenance washdown water.

   b. Miscellaneous surface spillage.

   c. Equipment drainage from nonradioactive sources, other than those terminating in segregated equipment drainage sumps.

   d. Accumulated fire protection water after actuation of any fire protection sprinklers.

2. Discharge effluent to the storm drainage system except for the following:

   a. Sump pumps in the circulating water pump bays of the screenwell building, which discharge effluent to the discharge tunnel.

b. A sump pump in the foam room of the service building, which discharges effluent to a removable container for offsite disposal.

c. Sump pumps in the auxiliary boiler building and waste neutralizing area of the screenwell building, which discharge effluent to the radwaste system (Section 11.2.2) because of the possible presence of chemicals in the effluent.

d. Equipment and floor drain systems potentially contaminated with ethylene glycol from glycol heating/cooling equipment, which is collected in separate storage tanks for offsite disposal.

e. A sump pump in sodium hypochlorite solution cubicle of the screenwell building, which discharges effluent to the WTS system (Section 9.2.3) because of the possible presence of chemicals in the effluent.

3. Handle the calculated flow from the area and equipment served by each sump.

4. Prevent the discharge of oily equipment and floor drains to the storm drainage system.

9.3.3.2 System Description

Each equipment and floor drain sump receiving radioactive influent is lined with either stainless steel, fiberglass or ceramic composite coating to prevent migration of its contents. Sumps receiving nonradioactive influent are of concrete or steel, and some are lined with acid brick. Each sump is sized to contain the influent from the equipment or area it serves.

Each drain tank of the reactor building equipment and floor drain system is served by duplex horizontal centrifugal pumps. Sumps that receive intermittent drain flow have a simplex sump pump and level switch. Sumps that receive continuous drain flow have duplex sump pumps and level switches. To minimize wear, all duplex sump pumps have one manual selector switch to provide alternation of which pump is to be selected as the lead pump on successive pump starts.

Each sump has either an open grating or a vented solid cover. Sump covers on radioactive drain sumps are vented to the building ventilation systems. Sump covers on nonradioactive drain sumps have gooseneck vents for venting to the surrounding areas.

Sump 2DFT-SUMP2H receives condensate effluent intermittently from the main steam line drain to the condenser. This sump is not vented to the building ventilation system because there is no flashing concern which could result in airborne radioactivity due

to the temperature being less than 212°F. Also, noble gases, which could become airborne even in the absence of flashing, are expected to be negligible.

Because of the high temperature of segregated low conductivity equipment drains, drain precoolers are installed to lower the temperature of the influent prior to entering the drain tanks located in the reactor building. Drain coolers are installed in sumps, and an effluent aftercooler is installed in the main discharge header of the turbine building equipment drain system.

For ease of decontamination and to minimize the potential for leakage, all piping handling radioactive flow uses corrosion-resistant materials with butt-welded construction. Piping that conveys radioactive flow is connected with the sumps below the expected minimum water level to minimize flashing conditions and gaseous radioactivity around the sumps.

Flow from equipment and floor drains that has no potential for radioactive contamination is discharged to the storm drainage system. Prior to discharge into the storm drainage system, all potentially oily drainage (except for the diesel fire pump room) is routed through a corrugated plate-type oil separator. The diesel fire pump room is provided with a depressed floor and sump sized to contain the volumetric capacity of the fuel oil tank in addition to an amount of fluid from the fire protection system. Accumulated oil in the separator or depressed floor and sump in the diesel fire pump room will be trucked from the site on an as-required basis.

9.3.3.3 Safety Evaluation

Leakage from low conductivity primary containment equipment drains and high conductivity primary containment floor drains is considered identified and unidentified leakage, respectively. This drainage is continuously monitored on a rate-of-rise basis in accordance with leak detection criteria in Section 5.2.5.

Portions of the leak detection system (LDS) are safety related, as delineated in Table 3.2-1.

The primary containment penetrations for the drywell floor and equipment drains have safety-related inside and outside isolation valves and piping. Upon a LOCA isolation signal these valves will close. After the isolation signal is reset the valves can be opened manually.

Floor and equipment drain systems are designed to prevent contamination of the storm drain system with effluent from sumps containing radioactive or potentially radioactive drainage. The effluent from all sumps/tanks in a given building is discharged to one of the following disposal points:

1. Radwaste system for radioactive or potentially radioactive drains.

2. Storm drain system or discharge tunnel for nonradioactive drains.

The equipment and floor drain systems serving buildings that house safety-related equipment have sufficient capacity to prevent excessive drain buildup that could affect the operability of the equipment. The discharge piping from each sump pump contains a check valve to prevent backflow from one pump to another.

The LPCS, RHR (RHR-A,B,C), RCIC, and HPCS pump cubicles in the reactor building each have a sump and duplex pumps which discharge to a common header located outside the cubicles whose walls are waterproofed. In the event that one of the cubicles should flood, transfer of water to other cubicles is prevented by additional check valves installed in each sump pump discharge line, prior to connection with the common header at a point outside the cubicle. The reactor building general area at el 175 ft and each cubicle have a safety-related high water level switch, wall-mounted near the floor, to detect flooding condition and to annunciate an alarm in the main control room.

All equipment drain and floor drain piping in the reactor building and control building are either seismically supported or evaluated as not requiring seismic support in order to preserve the operability of adjacent safety-related components.

Originally, the FMEA of the reactor building equipment and floor drain systems was contained in the Unit 2 FMEA document, which is historical. FMEAs for plant systems are now performed and controlled by the design process.

9.3.3.4 Testing and Inspection Requirements

All drain piping systems are tested for defects and leaks after installation and all leaks in pipes and joints are repaired. All sump pumps and sump pump discharge piping are analyzed for stress and tested to ensure that their performance meets the required design flow rates.

9.3.3.5 Instrumentation Requirements

Description

Instrumentation and controls are provided to control the sump levels by manual or automatic operation of the sump pumps when sump level is high. Safety-related instrumentation and controls are provided for automatic or manual control of the drywell equipment and floor drain system containment isolation valves. The controls and monitors described in the following sections are

located in the main control room. The control logic is shown on the following figures:

9.3-13 Reactor Building Equipment and Floor Drains

9.3-14 Turbine Building Equipment and Floor Drains

9.3-15 Radwaste Building Equipment and Floor Drains

9.3-16 Miscellaneous Buildings Equipment and Floor Drains

Operation

The operation of the reactor building and turbine building equipment and floor drain systems is similar. Operation of the radwaste building drain sump pumps and miscellaneous buildings (auxiliary boiler building, CST building, main stack, control building, and screenwell building) sump pumps is also similar.

A manual selector switch is provided for each pair of floor drain sump pumps and each pair of equipment or floor drain pumps to select one pump as lead pump each time the corresponding tank high level switch is actuated. The lead pump starts automatically when the corresponding sump or drain tank level is high. The backup pump starts automatically when the sump or drain tank level is high-high. Both pumps stop automatically when the sump or drain tank level is low.

The drywell floor drain and the drywell equipment drain containment isolation valves close automatically on a LOCA isolation signal or manual containment isolation signal. Interlocks prevent the isolation valves from opening unless all of the following conditions exist: the LOCA isolation signal is removed, the isolation signal is reset, and the valves are opened manually.

Controls are provided for manual operation for each of the following:

1. Reactor building sump pump (local control).

2. Drywell floor drain pump.

3. Drywell floor drain containment isolation valve.

4. Reactor water drain isolation valve.

5. Reactor building equipment drain pump (local control).

6. Drywell equipment drain pump.

7. Drywell equipment drain containment isolation valve.

8. Reactor water drain control valve.

9. Turbine building floor drain sump pump (local control).
10. Condenser pit sump pump (local control).
11. Turbine building equipment drain pump.
12. Radwaste building drain sump pumps (local control).
13. Auxiliary boiler building sump pumps (local control).
14. CST building sump pumps (local control).
15. Main stack sump pumps (local control).
16. Control building sump pumps (local control).
17. Screenwell building sump pumps (local control).
18. Reactor building mat drainage sump pumps (local control).
19. Service building foam room floor drain sump pump (local control).
20. Screenwell building sodium hypochlorite solution cubicle drain sump pump.
21. Diesel generator building floor and equipment drain sump pumps (local).

The diesel generator building drain sump pumps discharge to the storm sewer through the diesel generator yard area oil separator by a hand-operated valve.

<u>Monitoring</u>

<u>Reactor Building Equipment and Floor Drains</u>

Recorders are provided for:

1. Drywell floor drain tank level.
2. Drywell floor drain pump flow.
3. Drywell floor drain leak rate.
4. Drywell equipment drain tank level.
5. Drywell equipment drain pump flow.
6. Drywell equipment drain leak rate.

Alarms are provided for:

1. Reactor building floor drain leakage high.
2. Reactor building floor drain temperature high.
3. Drywell floor drain tank level high-high.
4. Drywell floor drain containment isolation valves inoperable.
5. Drywell floor drain leakage rate high.
6. General area, HPCS, LPCS, RHR-A, RHR-B, RHR-C, and RCIC pump rooms flood water level high.
7. Reactor building floor drain system trouble.
8. Reactor building equipment drain tank leakage high.
9. Drywell equipment drain containment isolation valves inoperable.
10. Drywell equipment drain tank temperature high.
11. Drywell equipment drain daily leakage rate high.
12. Drywell equipment drain tank level high-high.
13. Reactor building equipment drains system trouble.
14. Reactor water drain valves not closed.
15. Drywell floor drain containment isolation valve motor overload.
16. Drywell equipment drain containment isolation valve motor overload.
17. Drywell floor drain pump motor overload.
18. Drywell equipment drain pump motor overload.
19. Cubicles 2RHS*E1A and *E1B flooded.

<u>Turbine Building Equipment and Floor Drains</u>

Alarms are provided for:

1. Turbine building floor drains leakage high.
2. Turbine building floor drains system trouble.
3. Turbine building equipment drains leakage high.

4. Turbine building equipment drain system trouble.

Radwaste Building Equipment and Floor Drains

Alarms are provided for:

1. Radwaste building equipment and floor drains leakage high.

2. Radwaste building equipment and floor drain system trouble.

Miscellaneous Buildings Equipment and Floor Drains

Alarms are provided for:

1. Screenwell building floor and equipment drains leakage high.

2. Screenwell building floor drain sump 5 level high.

3. Screenwell building floor drain sump 9 level high.

4. Screenwell building system trouble.

5. Auxiliary boiler building floor drain system trouble.

6. Auxiliary boiler building equipment and floor drains leakage high.

7. Emergency diesel generator sump level high.

8. CST building floor drain leakage high.

9. CST building system trouble.

10. Main stack floor drain sump tank 2 leakage high.

11. Main stack floor drain tank 2 system trouble.

12. Control building floor drain system trouble.

13. Reactor building mat drainage system trouble.

14. Service water pump bay flooded.

15. Main stack floor drain tank 3 system trouble.

16. Diesel generator yard/transformer area oil separator oil level high.

17. Main stack floor drain sump tank 3 leakage high.

9.3.4 Not Applicable

9.3.5 Standby Liquid Control System

9.3.5.1 Design Bases

The SLCS is a system with capability for a special event. The process equipment, control, and instrumentation of the SLCS that are essential for injecting boron into the reactor vessel are designed as seismic Category I. The SLCS is designed with certain safety features to be highly reliable. However, the SLCS reactivity control function is not required to meet safety design basis requirements of safety systems. It meets the following design bases:

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

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

3. The time required for actuation and effectiveness of the backup control is consistent with the nuclear reactivity rate of change predicted between rated operating and cold shutdown conditions.

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

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

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

In the suppression pool pH control mode, the SLCS functions post-LOCA and meets the following additional design bases:

1. Suitable quality, reliability, and redundancy is provided to assure that for onsite or offsite power operation, the safety function of injecting sodium pentaborate solution into the reactor for suppression pool pH control can be accomplished.

2. Injection of sodium pentaborate for suppression pool pH control using the SLCS system is specified in plant procedures.

3. A sufficient concentration and quantity of sodium pentaborate is available for injection into the reactor vessel to control pH in the suppression pool for 30 days following a DBA LOCA.

9.3.5.2 System Description

The SLCS (Figure 9.3-17) can be manually initiated by the Operator or automatically by the redundant reactivity control system (RRCS).

The SLCS is not used to control reactivity for power generation. It is required only to shut down the reactor and keep the reactor from going critical as it cools. SLC is needed only in the improbable event that sufficient control rods cannot be inserted in the reactor core to accomplish shutdown and cooldown in the normal manner. The SLCS also provides suppression pool buffering following a LOCA accompanied by significant fuel damage, preventing re-evolution of iodine from the suppression pool by maintaining the pool pH above 7.0, in support of the alternative source term (AST) methodology.

The boron solution tank, the test water tank, the two positive displacement pumps, the two explosive valves, the two motor-operated pump suction valves, and associated local valves and controls are located in the reactor building. The liquid is pumped into the HPCS line downstream of the inboard containment isolation check valve. The sodium pentaborate solution is discharged radially over the top of the core through the HPCS sparger. The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the uranium fuel. The sodium pentaborate, when mixed in the suppression pool following a LOCA, also acts as a buffer to maintain the pool pH at or above 7.0 to prevent the re-evolution of iodine. The specified neutron absorber solution is sodium pentaborate ($Na_2B_{10}O_{16} \cdot 10H_2O$). It is prepared by dissolving granularly-enriched sodium pentaborate in demineralized water. An air sparger is provided in the tank for mixing. To prevent system plugging, the tank outlet is raised above the bottom of the tank.

The SLCS can deliver enough sodium pentaborate solution into the reactor (Figure 9.3-18) to assure reactor shutdown or maintain the suppression pool pH above 7.0 for 30 days following a DBA LOCA. This is accomplished by filling the SLC storage tank with

demineralized water to the low level alarm point, and then adding sodium pentaborate. The solution can be diluted with water to within 6 in of the overflow level volume to allow for evaporation losses or to lower the saturation temperature. The tank may contain boron solution from a minimum volume of 1,600 gal at 13.6% (net low level alarm volume at 1,800 gal) to a maximum of 4,815 gal (net high level alarm volume) based on a zero level of 5.1 in above the centerline of the outlet.

The minimum temperature of the fluid in the tank and piping is consistent with that obtained from Figure 9.3-19 for the solution temperature. The saturation temperature of the recommended solution is 60°F at the nominal concentration of 14.0%. Equipment containing the solution is installed in an area in which the air temperature is maintained within the range of 70°F to 104°F. An electrical resistance heater system provides a backup heat source that maintains the solution temperature between 75°F (automatic operation) and 85°F (automatic shutoff) to prevent precipitation of the sodium pentaborate from the solution during storage. High or low temperature, or high or low liquid level, causes an alarm in the main control room. The entire system is located within the reactor building, so it is unaffected by cold weather.

The positive displacement pumps are sized to inject the boron solution (minimum 41.2 gpm per pump) into the reactor within a specified time period, independent of the amount of solution in the tank.

The pump and system design pressure between the explosive valves and the pump discharge is 1,600 psig. The two relief valves are set to open at 1,600 psig with no backpressure. To prevent bypass flow in the event that a pressure relief valve fails and opens, a check valve is provided downstream of each relief valve in each pump discharge line.

The two explosive-actuated injection valves provide assurance of opening when needed and ensure that boron does not leak into the reactor even when the pumps are being tested. Each explosive valve is closed by a shear plug in the inlet chamber. The plug is circumscribed with a deep groove so the end readily shears off when pushed with the valve plunger. This opens the inlet hole through the plug. The sheared end is pushed out of the way in the chamber and is shaped so it does not block the ports after release.

The shearing plunger is actuated by an explosive charge with dual ignition primers inserted in the side chamber of the valve. Ignition circuit continuity is monitored by a trickle current, and an alarm occurs in the control room if either circuit opens. Indicator lights show which primary circuit opened.

Signals from the RRCS can automatically initiate the SLCS by actuating both loops. The SLCS can also be actuated manually by

two keylocked spring-return switches which ensure that switching from the NORMAL position to RUN position is a deliberate act. Operation of either switch starts an injection pump and simultaneously opens its respective explosive valve and storage tank outlet valve. The initiation generates a signal to close the RWCU system isolation valve to prevent loss or dilution of the boron. This isolation signal is sealed in during SLC operation and remains sealed in until reset by Operator action.

A light in the control room indicates that power is available to the pump motor contactor and that the contactor is de-energized (pump not running). Another light indicates that the contactor is energized (pump running).

Storage tank liquid level, tank outlet valve position, pump discharge pressure injection flow, and loss of continuity of the explosive valves indicate that the system is functioning. Pump discharge pressure, valve status and injection flow are indicated in the main control room.

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

Table 9.3-2 contains the process data for the various modes of operation of the SLC system. Seismic category and safety class are included in Table 3.2-1. Principles of system testing are discussed in Section 9.3.5.4. Table 9.3-3 contains the SLCS component material and specifications.

9.3.5.3 Safety Evaluation

The SLCS is a reactivity control system and is a backup method of shutting down the reactor. The probability of this system being required for reactor shutdown is very small because of the large number of independent control rods available to shut down the reactor. Since the SLCS is functionally redundant to the CRD system, it is not required to meet the single-failure criterion for this function.

To assure the availability of the SLCS, two divisional sets of the components required to actuate the system, including pumps and explosive valves, are provided in parallel so that no single failure in these components will prevent initiation of the SLCS.

The system is designed to bring the reactor from rated power to cold shutdown at any time in core life. The reactivity compensation provided will reduce reactor power from rated to zero level and allow cooling the nuclear system to room temperature, with the control rods remaining withdrawn in the rated power pattern. It includes the reactivity gains that result from complete decay of the rated power xenon inventory. It also includes the positive reactivity effects from eliminating

steam voids, changing water density from hot to cold, reduced Doppler effect in uranium, reducing neutron leakage from boiling to cold, and decreasing control rod worth as the moderator cools. The shutdown capability of the SLCS is evaluated for each fuel cycle and is reported in the cycle-specific Supplemental Reload Licensing Report (SRLR).

The SLCS also provides suppression pool buffering following a LOCA accompanied by significant fuel damage, preventing re-evolution of iodine from the suppression pool by maintaining the pool pH above 7.0, in support of the alternative source term (AST) methodology. For this function, the SLCS design relative to single failures was reviewed and determined to be acceptable by the NRC with the issuance of License Amendment 125.

The minimum average concentration of natural boron required in the reactor core to provide adequate shutdown margin after operation of the SLC is 780 ppm. Calculation of the minimum quantity of sodium pentaborate to be injected into the reactor is based on the required 780 ppm average concentration in the reactor coolant, including recirculation loops, at 68°F and reactor water level at level 8. The result is increased by 25 percent to allow for imperfect mixing and leakage. Additional sodium pentaborate is provided to accommodate dilution by the RHR system in the shutdown cooling mode. The same resulting tank concentration and level have been determined to adequately satisfy the AST support function requirements. Although procedures require injection of the entire SLCS tank contents, injection of slightly less than 1,150 gal has been determined to be sufficient to maintain the suppression pool pH at or above 7.0 for 30 days. Included in the 1,150 gal are any amounts that could remain or be trapped in the SLCS suction and discharge piping and the HPCS piping and sparger.

Cooldown of the nuclear system will require a minimum of several hours to remove the thermal energy stored in the reactor, cooling water, and associated equipment. The controlled limit for the reactor vessel cooldown is 100°F/hr, and normal operating temperature is approximately 547°F. Use of the main condenser and various shutdown cooling systems requires about 6 hr to lower the reactor vessel to a subcooled 70°F condition. This condition requires the minimum 780 ppm boron concentration. The minimum required boron injection rate is 80.0 gpm with two pumps in operation. The minimum realized boron injection rate addresses GEH safety communication 10-13, Standby Liquid Control Dilution Flow. With both pumps operating, it will take approximately 20 minutes to inject sufficient boron to achieve the minimum required boron concentration for reactivity control. The minimum 1,150 gal required for suppresion pool pH control will take less than 30 min to inject, with only one pump operating. Following a large LOCA, injection of the SLCS within 22 hr after the potential for significant fuel failure has been identified will ensure that the suppression pool pH is controlled for at least 30 days.

The SLC equipment required for injection of neutron absorber solution into the reactor is designed as Category I for withstanding the specified earthquake loadings (Chapter 3). The SLCS is protected from the effects of tornadic forces (including tornado-generated missiles) by housing the system in a tornado-protected structure as listed in Table 3.2-1. The system piping and equipment are designed, installed, and tested in accordance with requirements stated in Section 3.9B.

The SLCS is powered normally from offsite power sources. In the event of a plant offsite power failure, the pumps, valves, and controls necessary to assure boron injection are powered from the standby diesel generators. Heater electrical supply can be manually transferred to the Division I standby diesel generator in the event of loss of the normal ac source. The pumps and valves are powered and controlled from separate divisional buses and circuits.

The SLC pumps have sufficient pressure margin, up to a maximum relief valve setting of 1,600 psig with no backpressure to assure solution injection into the reactor above the normal pressure in the bottom of the reactor. The reactor SRVs begin to relieve pressure above approximately 1,100 psig. Therefore, the SLC positive displacement pumps cannot overpressurize the nuclear system.

Only one of the two SLC loops is needed for backup shutdown system or suppression pool pH control operation. If a redundant component (e.g., pump) in one of the two parallel loops is found to be inoperable, there is no immediate threat to shutdown or LOCA mitigation capability, and reactor operation can continue during repairs. The time during which one of the two parallel loops may be out of operation is given in the Technical Specifications.

9.3.5.4 Testing and Inspection Requirements

Testability of one pump at a time is possible while the reactor is in service. While one pump is being tested during reactor operation, the other pump is capable of injecting the borated solution if it receives an initiation signal. For integrated system testing during reactor shutdown, the SLCS is designed so that the pumps and all automatic valves of both loops can be tested for system performance. The system will function as if for an anticipated transient without scram (ATWS) or LOCA event except that the process fluid is demineralized water from the system test tank.

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

pump utilizing keylock switches in the main control room. This test can be accomplished with the reactor operating without affecting the operability of the other pump. During a refueling or maintenance outage, the injection portion of the system can be functionally tested by valving the suction line to the test tank and actuating the system from the main control room. System operation is indicated in the main control room.

After functional tests, the shear plugs and explosive charges for the injection valves that were tested are replaced and all the valves returned to their normal positions as indicated in Figure 9.3-17.

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

The test tank contains demineralized water for approximately 3 min of single pump operation. Demineralized water from the MWS system is available for refilling or flushing the system. Should the boron solution ever be injected into the reactor, either intentionally or inadvertently, the Operator verifies that the normal reactivity controls are adequate to keep the reactor subcritical, and then removes the boron from the RCS by initially flushing for gross dilution followed by operating the RWCU system. There is practically no effect on reactor operations when the boron concentration has been reduced below approximately 50 ppm. The concentration of the sodium pentaborate in the solution tank is determined periodically by chemical analysis. Electrical supplies and relief valves are also subjected to periodic testing. The SLCS preoperational test is described in Table 14.2-54.

9.3.5.5 Instrumentation Requirements

Instrumentation and controls are provided to maintain the temperature of the sodium pentaborate solution above the setpoint temperature (75°F) to prevent the sodium pentaborate from precipitating out of solution. Controls are provided to manually initiate injection of the sodium pentaborate solution into the reactor. Instrumentation consisting of solution temperature indication and control, solution level, and heater system status is provided locally at the storage tank. For a more detailed description of instrumentation and controls, refer to Section 7.4.

TABLE 9.3-1

PROCESS SAMPLE SYSTEMS

Sample Point	Parameter(s)	Purpose
Reactor Sample System		
1. Common influent to RWCU filter demineralizers	Conductivity, oxygen, grab, hydrogen	Reactor water quality*, RWCU F/D efficiency Hydrogen concentration (backup to RCS for H_2 concentration)
2. Effluent RWCU F/D A	Conductivity, grab	Reactor water quality, RWCU F/D efficiency
3. Effluent RWCU F/D B	Conductivity, grab	Reactor water quality, RWCU F/D efficiency
4. Effluent RWCU F/D C	Conductivity, grab	Reactor water quality, RWCU F/D efficiency
5. Effluent RWCU F/D D	Conductivity, grab	Reactor water quality, RWCU F/D efficiency
6. Reactor recirculation loop A pump discharge	Conductivity, oxygen, grab, hydrogen	Reactor water quality (backup to RWCU samples) Hydrogen concentration (primary source of sampling for H_2 concentration)
7. HPCS system test return line to condensate storage tanks	Grab	HPCS/condensate storage tank water quality
8. Common effluent control rod drive filters	Conductivity, oxygen, grab	CRD water quality
9. RHR heat exchanger A effluent	Grab	Reactor water quality while shut down and depressurized
10. RHR heat exchanger B effluent	Grab	Reactor water quality while shut down and depressurized
11. Common influent SFC filters	Grab	Spent fuel pool water quality
12. Effluent SFC filter 1A	Grab	Spent fuel pool/SFC system water quality, filter efficiency
13. Effluent SFC filter 1B	Grab	Spent fuel pool/SFC system water quality, filter efficiency
14. Effluent SFC heat exchanger A	Grab	SFC system/spent fuel pool water quality
15. Effluent SFC heat exchanger B	Grab	SFC system/spent fuel pool water quality

TABLE 9.3-1 (Cont'd.)

Sample Point	Parameter(s)	Purpose
Turbine Sample System		
1. Common influent CND system	Conductivity, turbidity[1], grab	Condenser tube leaks, condensate water quality, CND efficiency
2. Common effluent CND system	Conductivity, oxygen, grab	Condensate water quality, CND efficiency
3. Effluent CND 1A	Conductivity, grab	Condensate water quality, CND efficiency
4. Effluent CND 1B	Conductivity, grab	Condensate water quality, CND efficiency
5. Effluent CND 1C	Conductivity, grab	Condensate water quality, CND efficiency
6. Effluent CND 1D	Conductivity, grab	Condensate water quality, CND efficiency
7. Effluent CND 1E	Conductivity, grab	Condensate water quality, CND efficiency
8. Effluent CND 1F	Conductivity, grab	Condensate water quality, CND efficiency
9. Effluent CND 1G	Conductivity, grab	Condensate water quality, CND efficiency
10. Effluent CND 1H	Conductivity, grab	Condensate water quality, CND efficiency
11. Effluent CND 1J	Conductivity, grab	Condensate water quality, CND efficiency
12. Effluent fourth-point feedwater heaters (common sample from 3 fourth-point heaters)	Grab	Heater erosion, corrosion
13. Effluent low-pressure feedwater heater string	Grab	Heater erosion, corrosion
14. Inlet main steam line D to HP turbine	Grab	Moisture carryover
15. Turbine gland seal clean steam reboiler A	Grab	Clean steam reboiler water quality
16. Turbine gland seal clean steam reboiler B	Grab	Clean steam reboiler water quality
17. Auxiliary boiler steam supply header	Grab	Moisture carryover
18. Auxiliary boiler feedwater pumps discharge	Grab	Auxiliary boiler feedwater water quality
19. Condensate makeup and drawoff system	Grab	Condensate makeup water quality
20. CND cation regeneration tank effluent	Conductivity, grab	CND system performance
21. CND system anion regeneration tank effluent	Conductivity, grab	CND system performance

TABLE 9.3-1 (Cont'd.)

Sample Point	Parameter(s)	Purpose
Turbine Sample System (cont'd.)		
22. CND system resin mix tank effluent	Conductivity, grab	CND system performance
23. CND regeneration system effluent	Conductivity, grab	CND system performance
24. Deleted		
25. Deleted		
26. Deleted		
27. CND system low conductivity waste tank effluent	Grab	CND system performance
28. CND system demineralizer waste neutralizing tank effluent	Grab	CND system performance
29. CND system dilute acid influent to cation regeneration tank	Grab	CND system performance
30. CND system dilute caustic influent to anion regeneration tank	Grab	CND system performance
31. Discharge low-pressure feedwater heater drain pump 1A	Conductivity, turbidity(1), millipore filter, grab	LP feedwater heater erosion/corrosion
32. Discharge low-pressure feedwater heater drain pump 1B	Conductivity, turbidity(1), millipore filter, grab	LP feedwater heater erosion/corrosion
33. Discharge low-pressure feedwater heater drain pump 1C	Conductivity, turbidity(1), millipore filter, grab	LP feedwater heater erosion/corrosion
34. Deleted		
35. Deleted		
36. Deleted		
37. Deleted		

TABLE 9.3-1 (Cont'd.)

Sample Point	Parameter(s)	Purpose
Turbine Sample System (cont'd.)		
38. Deleted		
39. Deleted		
40. Effluent sixth-point feedwater heaters	Conductivity, oxygen, pH, grab	Reactor feedwater quality
41. Feedwater corrosion (sixth-point feedwater heater effluent)	Grab	Feedwater suspended solids, dissolved corrosion products
42. Turbine generator gland seal and exhaust steam system	Grab	TME system activity
43. Effluent fifth-point heater drains (common sample from three heaters)	Grab	Heater erosion, corrosion
44. Effluent sixth-point heater drains (common sample from three heaters)	Grab	Heater erosion, corrosion
45. Effluent moisture separator drain receiver (common sample from two drain receivers)	Grab	Heater erosion, corrosion
46. Effluent reheater drain receiver (common sample from two drain receivers)	Grab	Heater erosion, corrosion
Radwaste Sample System		
1. Effluent LWS system waste collector tank pump 1A	Grab	LWS system performance
2. Effluent LWS system waste collector tank pump 1B	Grab	LWS system performance
3. Effluent LWS system waste collector tank pump 1C	Grab	LWS system performance
4. Effluent LWS filter 1A	Grab	LWS system performance
5. Effluent LWS filter 1B	Grab	LWS system performance
6. Effluent LWS demineralizer 4A	Grab	LWS system performance
7. Effluent LWS demineralizer 4B	Grab	LWS system performance

TABLE 9.3-1 (Cont'd.)

Sample Point	Parameter(s)	Purpose
Radwaste Sample System (cont'd.)		
8. Discharge LWS floor drain collector pump 2A	Grab	LWS system performance
9. Discharge LWS floor drain collector pump 2B	Grab	LWS system performance
10. Discharge LWS floor drain filter effluent pump	Grab	LWS system performance
11. Discharge LWS recovery sample pump 4A	Grab	LWS system performance
12. Discharge LWS recovery sample pump 4B	Grab	LWS system performance
13. Discharge LWS regeneration waste pump 3A	Grab	LWS system performance
14. Discharge LWS regeneration waste pump 3B	Grab	LWS system performance
15. Discharge LWS waste sample pump 5A	Grab	LWS system performance
16. Discharge LWS waste sample pump 5B	Grab	LWS system performance
17. Common discharge LWS waste sample pumps 5A/B	Grab	LWS system performance
18. Common discharge LWS floor drain collector surge pumps 17A/B	Grab	LWS system performance
19. Common discharge LWS waste collector surge pumps 18A/B	Grab	LWS system performance
20. Discharge LWS waste evaporator distillate transfer pump 8	Grab	LWS system performance
21. Discharge regeneration evaporator distillate transfer pump	Grab	LWS system performance
22. Effluent LWS radwaste auxiliary steam drain cooler 1	Grab	LWS system performance
23. Effluent LWS radwaste auxiliary steam drain cooler 2	Grab	LWS system performance

TABLE 9.3-1 (Cont'd.)

Sample Point	Parameter(s)	Purpose
Radwaste Sample System (cont'd.)		
24. LWS waste evaporator recirculation pump 13 suction	Grab	LWS system performance
25. LWS waste evaporator recirculation pump 13 discharge	Grab	LWS system performance
26. LWS regeneration evaporator recirculation pump 11 suction	Grab	LWS system performance
27. LWS regeneration evaporator recirculation pump 11 discharge	Grab	LWS system performance
28. WSS sludge transfer pump 11 discharge	Grab	WSS system performance
29. WSS waste concentrate transfer pump 6 discharge	Grab	WSS system performance
Miscellaneous System Samples		
1. Effluent RHR heat exchanger A	Conductivity	RHR/reactor water quality
2. Effluent RHR heat exchanger B	Conductivity	RHR/reactor water quality
3. RHR to LWS flush line	Conductivity	Water quality of RHR water to LWS prior to initiation of shutdown cooling
4. Effluent SFC filter 1A	Conductivity	SFC system/spent fuel pool water quality
5. Effluent SFC filter 1B	Conductivity	SFC system/spent fuel pool water quality
6. Common effluent SFC pumps	Conductivity	SFC system/spent fuel pool water quality
7. Standby liquid control tank	Grab	Standby liquid control tank borate concentration
8. Suppression pool	Grab	Suppression pool water quality/clarity
9. Drywell floor drain pumps	Grab	Drywell floor drain water quality/clarity
10. Drywell and reactor building equipment drain system discharge to LWS	Conductivity, grab	Equipment drainage water quality
11. Service water system A and B loop discharges to Lake Ontario	Conductivity, pH, grab	Service water discharge environmental water quality
12. RBCLCW pump suction	pH, grab	CCP system water quality
13. TBCLCW pump suction	Conductivity, pH	CCS system water quality

TABLE 9.3-1 (Cont'd.)

Sample Point	Parameter(s)	Purpose
Miscellaneous System Samples (cont'd.)		
14. CWS blowdown line to service water system	pH	Environmental water quality in SWP discharge
15. CWS discharge flume	pH, conductivity	CWS system water quality
16. LWS waste collector tank pump 1A discharge/suction	Conductivity, pH	LWS system performance
17. LWS waste collector tank pump 1B discharge/suction	Conductivity, pH	LWS system performance
18. LWS waste collector tank pump 1C discharge/suction	Conductivity, pH	LWS system performance
19. LWS demineralizer 4A effluent	Conductivity	LWS system performance
20. LWS demineralizer 4B effluent	Conductivity	LWS system performance
21. LWS recovery sample tank pump 4A discharge/suction	Conductivity, pH	LWS system performance
22. LWS recovery sample tank pump 4B discharge/suction	Conductivity, pH	LWS system performance
23. LWS floor drain collector tank pump 2A discharge/suction	Conductivity, pH	LWS system performance
24. LWS floor drain collector tank pump 2B discharge/suction	Conductivity, pH	LWS system performance
25. LWS regenerative waste tank pump 3A discharge/suction	Conductivity, pH	LWS system performance
26. LWS regenerative waste tank pump 3B discharge/suction	Conductivity, pH	LWS system performance
27. LWS waste tank pump 5A discharge/suction	Conductivity, pH	LWS system performance
28. LWS waste tank pump 5B discharge/suction	Conductivity, pH	LWS system performance

TABLE 9.3-1 (Cont'd.)

Sample Point	Parameter(s)	Purpose
Miscellaneous System Samples (cont'd.)		
29. Turbine building drain systems discharge to LWS	Conductivity	Turbine building drain systems water quality
30. LWS regenerative evaporator reboiler drain	Conductivity	Radwaste steam drain water quality
31. LWS waste evaporator reboiler drain	Conductivity	Radwaste steam drain water quality
32. LWS waste evaporator EV1 distillate	Conductivity	LWS system performance
33. LWS regenerative evaporator EV2 distillate	Conductivity	LWS system performance
34. LWS filter 101 effluent	Conductivity, pH	LWS system performance
35. LWS filter 1A/B common effluent	Conductivity	LWS system performance
36. LWS waste collector surge tank pump P18A discharge/suction	Conductivity, pH	LWS system performance
37. LWS waste collector surge tank pump P18B discharge/suction	Conductivity, pH	LWS system performance
38. LWS floor drain collector surge tank pumps P17A discharge/suction	Conductivity, pH	LWS system performance
39. LWS floor drain collector surge tank pump P17B discharge/suction	Conductivity, pH	LWS system performance
40. CND individual CND effluent	Conductivity	CND system performance
41. CND system inlet header	Conductivity	CND system performance
42. CND inlet header	Conductivity	CND system performance
43. CND dilute caustic tank	Conductivity	CND system performance
44. CND dilute acid tank	Conductivity	CND system performance
45. CND low conductivity waste tank	Conductivity	CND system performance
46. CND system outlet header	Conductivity	CND system performance
47. ARC system hogging pump discharge	Grab	ARC system discharge activity
48. OFG system charcoal adsorber outlet	Grab	OFG system discharge activity
49. HVT system discharge	Grab	HVT system discharge activity

TABLE 9.3-1 (Cont'd.)

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* Water quality pertains not only to chemistry parameters such as conductivity and oxygen, but also to radiochemistry levels, both gross level and isotopic content. The radiochemistry analysis is performed on the grab samples in the laboratory.

(1) Suspended solids analysis may be performed in lieu of turbidity for backup analysis to out-of-service turbidimeters.

TABLE 9.3-2

STANDBY LIQUID CONTROL SYSTEM OPERATING PRESSURE/TEMPERATURE CONDITIONS

Test Modes[1]

Piping	Standby Mode[1]		Circulation Test		Injection Test[2]		Operating Mode[1]	
	Pressure (psig)[3]	Temperature (°F)	Pressure (psig)[3]	Temperature (°F)	Pressure (psig)[3]	Temperature (°F)	Pressure (psig)[3]	Temperature (°F)
Pump suction	Makeup water pressure[4]	70/110[5]	Test tank static head[4]	70/110[5]	Test tank static head[5]	70/110[5]	Storage tank static head	70/110[5]
Pump discharge to explosive valve inlet	Makeup water pressure	70/110	0/1325.4	70/110	105 plus reactor static head	70/110	(105 plus reactor static head) to 1325.4	70/110
Explosive valve outlet to but not including outboard isolation valves	Reactor static head to 1,150	70/100	N/A	N/A	<105 plus reactor static head	70/110	(<105 plus reactor static head) to <1325.4	70/110
Outboard Isolation Valves to Reactor								
Via HPCS sparger	Reactor static head to 1,150	70/560[6]	N/A	N/A	<105 plus reactor static head[2]	125	<105 plus reactor static head to 1220.1	70/560[6]

(1) Pump flow rate will be zero (pump not operating) during standby mode and at rated capacity during test and operating modes.
(2) Reactor to be at 0 psig and 125°F before changing from the standby mode to the injection test mode.
(3) Pressures tabulated represent pressure at the points identified below. To obtain pressure at intermediate points in the system, the pressures tabulated must be adjusted for elevation difference and pressure drop between the intermediate points and the pressure points identified as follows:

Piping	Pressure Point
Pump suction	Pump suction flange inlet
Pump discharge to explosive valve inlet	Pump discharge flange outlet
Explosive valve outlet to but not including outboard isolation valves	Explosive valve outlet
Outboard isolation valves to the reactor	Reactor HPCS nozzle or SLCS sparger nozzle

(4) Pump suction piping will be subject to demineralized water supply pressure during standby mode, flushing and filling of the piping, and during any testing where suction is taken directly from the demineralized water supply line rather than a test tank.
(5) During chemical mixing, the liquid in the storage tank will be at a temperature of 150°F maximum.
(6) 560°F represents maximum sustained operating temperature.

TABLE 9.3-3

SLCS COMPONENT MATERIAL AND SPECIFICATIONS

Component	Form	Material	Specification (ASTM/ASME)
Standby liquid control pump			
Fluid cylinder	Forging	Stainless steel	SA-182, Type F304
Cylinder head, valve cover	Plate	Stainless steel	SA-240, Type 304
Stuffing box flange plate	Plate	Stainless steel	SA-564, Type 630, Cond. H-1100
Cylinder head extension, valve stop	Bar	Stainless steel	SA-479, Type 304
Stuffing box	Bar	Stainless steel	SA-479, Type S21800
Stuffing box gland	Bar	Stainless steel	SA-564, Type 630, Cond. H-1100
Plungers	Bar	Stainless steel	A-564, Type 630, Cond. H-1100
Studs	Bar	Alloy steel	SA-540, Gr. B22, Cl. 1
Nuts	Forging	Alloy steel	SA-194, Gr. 7
Discharge and suction flange head capscrews	Forgings	Alloy steel	SA-193, Gr. B7
Standby liquid control storage tank			
Tank	Plate	Stainless steel	SA-240, Type 304
Fittings	Forgings	Stainless steel	SA-182, Type F304
Pipe	Pipe	Stainless steel	SA-312, Type 304
Welds	Electrodes	Stainless steel	FA 5.4 & 5.9, Types 308, 308L, 16, 316L