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10.1 Summary Description

10.1.1 Introduction

This section describes the reactor and plant auxiliary systems that provide operational convenience but which are not integral portions of the reactor, power conversion equipment or their safety systems. Included are reactor auxiliaries, plant services and cooling systems.

10.2 Reactor Auxiliary Systems

10.2.1 Fuel Storage and Fuel Handling Systems

10.2.1.1 Design Basis

The design objectives of the fuel storage and handling equipment are: (1) to provide a fuel storage pool for the underwater storage of fuel assemblies; (2) to provide a storage pool for the underwater storage of reactor vessel internals; (3) to provide adequate protection against the loss of water from the fuel pool storage; (4) to provide equipment for handling both new and irradiated fuel; and (5) to provide sufficient water depth for decontamination of releases from damaged fuel in the case of a refueling accident, and prevent release of contamination or exposure of personnel to radiation in excess of 10CFR20 limits during normal operations. To achieve these objectives, the fuel storage and handling equipment is designed using the following design bases: 01445693

a. New fuel is received and stored in a manner which precludes inadvertent nuclear criticality.

b. Reactor refueling generally involves replacement of about one-third of the fuel assemblies in the core every twenty-four months. 01457570

c. The fuel assemblies and other reactor components to be handled are of the size and weight given in Section 3.4.

d. Undamaged, used flow channels are reused on new fuel bundles (subject to NRC review) or new unirradiated flow channels are used on new fuel bundles.

e. There is no release of contamination or exposure of personnel to radiation in excess of 10CFR20 limits during normal operations, or in excess of 10CFR50.67 limits during a refueling accident.

f. It is possible, at any time, to perform limited work on irradiated components.

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

h. The fuel storage pool is designed to withstand earthquake loadings of a Class I structure as described in Section 12.2.

i. The new fuel storage vault is designed to withstand earthquake loadings of a Class I structure as described in Section 12.2.

10.2.1.2 Description

The major components of these systems are the new fuel storage vault, spent fuel storage pool, dryer-separator assemblies pool, the refueling platform and fuel handling equipment and tools. A pictorial representation of the refueling floor is shown in Figure 10.2-1. A plan view of the refueling floor is shown in Drawing NF-36058, Section 15.

The new fuel storage vault is a reinforced-concrete Class I structure, accessible only through top hatches. Racks in the vault have 150 locations that cannot be used to store fuel. The center-to-center spacing of bundles in the racks is 6.5 inches by 10 inches minimum. There is an open drain in the floor of the vault. 01445693

The spent fuel storage pool has been designed to withstand earthquake loadings as a Class I structure. It is a reinforced concrete structure, completely lined with seam-welded, stainless steel plates welded to reinforcing members (channels, I-beams, etc.) embedded in concrete. The stainless steel liner prevents leakage even in the event the concrete develops cracks. To avoid unintentional draining of the pool, there are no penetrations that would permit the pool to be drained below a safe storage level and all lines extending below this level are equipped with valving to prevent syphon backflow. The passage between the spent fuel storage pool and the refueling cavity above the reactor vessel is provided with two double-sealed gates with a monitored drain between the gates. This arrangement permits monitoring of leaks from the passage and facilitates repair of a gate or seal, if necessary. Given a nominal fuel pool water level of 1026 feet 8 inches, the depth of water in the pool is 37 feet 9 inches and the depth of the water in the transfer canal during refueling is 22 feet. The water in the pool is continuously filtered and cooled by the spent fuel pool cooling and demineralizer system described in Section 10.2.2.

The spent fuel pool has been modified to increase the original capacity from 740 fuel assemblies to 2217 by installing 13 new High Density Fuel Storage System (HDFSS) modules, one original low density rack, and two control blade racks. Of this capacity, only 2209 spent fuel storage slots are usable pending resolution of a bulging/binding concern with eight of the storage slots. The new modules are composed of rectangular fuel storage tubes, arranged in a 13x13 array, and fabricated by forming an inner and outer sheet of 304 stainless steel sandwiching a core of Boral (clad by aluminum). Each fuel tube is vented top and bottom to provide a positive flow path for any gases formed by the interaction of aluminum with stainless steel. The module design, materials, and fabrication are in accordance with the requirements of ASME Code, Section III, Subsection NF (Reference 37). The modules meet the Seismic requirements for Class I equipment. The modules are free standing. Analysis has confirmed that the frictional forces between the module support and the pool floor liner, and the low seismic overturning moment of the modules, make them stable under all conditions of storage. Calculated horizontal displacement of the modules during an earthquake is less than the nominal spacing, assuring no interaction between modules resulting from an SSE.

A re-evaluation of the fuel pool structural capacity for the HDFSS installation has shown that the existing structure is capable of supporting the increased loadings. This analysis justified the use of one control blade rack and two low-density fuel racks, for a total of 2237 fuel assemblies. Although the spent

fuel storage pool has been analyzed for 2237 fuel assemblies, the current installed fuel storage rack configuration provides for 2217 storage slots of which only 2209 are usable. The current configuration is bounded by this analysis.

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The basic design criterion associated with the storage of both irradiated (spent) and new fuel is that the effective multiplication factor of fuel stored under normal conditions and abnormal storage conditions are limited to an effective multiplication factor of 0.95 or less for both high and regular density fuel storage designs.

A Criticality Safety Analysis (CSA) for the Monticello spent fuel pool storage pool that verifies conformance with the above criterion is documented in ANP-3113(P) (Reference 80). The CSA shows that the criterion is met if the fuel assemblies exhibit a maximum lattice k-infinity of 0.8825 in the spent fuel storage pool rack geometry. The analysis supports storage in both the high density storage racks and the remaining original low density storage rack as long as a minimum of 12 inch spacing is maintained between the original and adjacent high density rack.

As detailed in Table 2.1 of ANP-3113(P), adherence to the following enrichment and gadolinia loading requirements will ensure that the in-rack reactivity requirement is met.

Maximum Lattice Average Enrichment, wt% U-235	4.70	
Minimum Number of Rods containing Gd_2O_3	8	Note 1
Minimum wt% Gd_2O_3 in these Gd Rod: Top Lattices	3.5	Note 2
Bottom Lattices	3.919	Note 3

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Notes:

1. These eight gadolinia rods cannot be loaded on the perimeter of the lattice or adjacent to the water channel. An equivalent of 2 gadolinia rods must be loaded along each side, however; two face adjacent gadolinia rods count as one in fulfilling this requirement. Gadolinia is not required in natural Uranium blankets and there are no restrictions on the number, concentration, or placement of any additional gadolinia rods.

2. The 3.5 wt% minimum Gd_2O_3 concentration applies to the top lattices, those axially located above the top of the active fuel region in the part length fuel rods.

3. The 3.919 wt% minimum Gd_2O_3 concentration applies to the bottom lattices, those axially located in the bottom of the assembly up to the top of the active fuel region in the part length fuel rods.

Assemblies containing lattices that do not meet the above enrichment and gadolinia loading requirements may still be stored in the storage pool if an analysis is performed in accordance with the requirements of Table 2.1 and Appendix A of ANP-3113(P) that explicitly confirms that the in-rack k-infinity requirement has been met.

The Monticello CSA provides a result that bounds the reactivity of the fuel to be loaded into the spent fuel storage pool on a 95 percent probability with a 95 percent confidence level. This 95/95 result includes allowances for uncertainties in parameters that have the potential to affect the final result as well as adders for known biases. These uncertainties include manufacturing uncertainties for the fuel assembly (fuel rod pitch, enrichment, UO2 pellet density, channel growth, channel thickness, pellet diameter, clad diameter, pellet void volume, gadolinia concentration, and gadolinia pellet density), manufacturing uncertainties for the storage rack (Areal B-10 density, boral sheet width, boral thickness, stainless steel wall thickness, and storage cell pitch), and uncertainties related to the KENO V.a computer code used in the analysis. The biases include those related to validation of the methodology, limiting accident, and potential abnormal conditions such as a conservative treatment of hypothetical boral blistering. 01445693

The CSA is performed using Reactivity Equivalent at Beginning Of Life (REBOL) lattices that bound the peak reactivity of the corresponding reference bounding lattices including a conservative bias to account for depletion and other uncertainties. The underlying reference bounding lattices are described by the enrichment and gadolinia requirements specified above. Evaluation at a variety of conditions including fuel temperature, moderator void fraction, power density, controlled versus uncontrolled depletions, and channeled versus unchanneled assemblies were used to ensure that the limiting reactivity condition was selected for these reference lattices.

10CFR50.68 requires that the effective multiplication factor of the new fuel vault not exceed 0.98 under optimum moderator conditions. No evaluation exists showing that the effective multiplication constant of the new fuel vault will remain below 0.98 under optimum moderator conditions; therefore the new fuel vault cannot be used to store fuel (Reference 82). 01445693

The existing cooling capacity of the SFP is sufficient to remove the heat load during normal end of cycle discharges, assuming a 24-month refueling cycle. For higher heat loads, such as a full core offload, the RHR cooling system can be used if necessary.

The spent fuel storage pool may hold, in addition to the fuel assemblies, control rod blades on temporary storage hangers, temporary control curtains, defective fuel containers, used flow channels, and small reactor vessel components to provide for a complete core unloading. Currently, storage is provided for 2209 fuel bundles, 141 control blades, and 20 flow channels. A 7 ft x 7 ft area of the pool is reserved for loading a spent-fuel cask. Additional storage for large components, such as the steam dryer and separator assemblies is provided in a separate storage pool adjacent to the reactor head cavity.

A refueling platform, equipped with a refueling grapple and two 1/2-ton auxiliary hoists is provided. Either of these hoists can be positioned for servicing the reactor head cavity or the fuel storage pool. The operating floor is serviced by a single failure proof (redundant) reactor building crane, which is equipped with a 105-ton main hoist and a 5-ton conventional auxiliary hoist. These hoists can reach any major equipment storage area on the refueling (operating) floor. The reactor building crane may either be cab or remotely operated.

The single failure proof reactor building crane was provided by installing a new redundant crane trolley and modifying the existing controls to minimize single failure vulnerability. These modifications provide a crane which complies with Reg. Guide 1.104, “Overhead Crane Handling Systems for Nuclear Power Plants”, Feb. 1976 (Reference 39), as applicable to an operating plant.

Underwater vacuum-cleaning equipment is available for removal of dirt and small particles. A variety of tools are provided for remote handling of the fuel, the reactor internals and the flow channels.

There are several options for the sequence of events involved in inspecting and readying new fuel bundles for insertion into the reactor core. The basic steps involve: (1) movement of new fuel bundles in shipping containers to the refueling floor and criticality safe temporary storage configurations there, (2) removal of the bundles from these shipping containers and placement in the new fuel inspection stand, (3) inspection of the new fuel followed by installation of a new unirradiated channel and placement in the spent fuel pool, or placement in the spent fuel pool for immediate or delayed installation of a “reused” irradiated fuel channel or a new unirradiated channel. 01445693

In preparation for refueling, the following sequence of events occurs:

a. Concrete shield plugs are removed from the reactor head cavity and dryer-separator storage canal.

b. The Drywell Head is unbolted and removed.

c. The reactor vessel head insulation is removed.

d. The reactor vessel head is unbolted and removed.

e. The Steam Dryer is transferred to the dryer-separator storage pool.

f. The Steam Separator is unbolted from the core shroud.

g. The water level is raised in the reactor cavity and dryer-separator storage pool and the steam separator is transferred to the storage location in the pool (the steam separator is sometimes transferred prior to raising water level).

h. The Fuel Storage pool gates are removed.

Spent fuel is removed from the reactor vessel, using the main fuel grapple attached to the refueling platform, and placed in racks in the spent fuel storage pool. The same equipment is used to transfer the new fuel from the spent fuel storage pool to the reactor vessel. The sequence of fuel movements is determined for each refueling based on the specific conditions associated with that refueling.

After refueling is completed the reactor is prepared for operation by essentially reversing the preparation procedure above.

The flow channels and spent-fuel bundles are temporarily stored in spent fuel storage pool racks. When the decay heat of the spent fuel declines sufficiently, a spent-fuel cask is set into the pool using the reactor building crane. The spent fuel is loaded into the cask under water and the cask is closed and removed from the pool. After decontamination, the spent-fuel cask is lowered by the reactor building crane to a truck or railway car in the equipment access area.

Spent Fuel casks may be transferred offsite or to the site's Independent Spent Fuel Storage Installation (ISFSI). The ISFSI utilizes the NUHOMS® Standardized Storage System consisting of Dry Shielded Canisters (DSCs) stored in concrete Horizontal Storage Modules (HSMs). The ISFSI is designed and licensed in accordance with 10CFR72.

Spent fuel storage pool level monitoring can be achieved by two independent, permanent, wide span level measurement channels. The level measurement is accomplished by a MOHR Test and Measurement LLC supplied guided wave type measurement probe and signal processor. Each signal processor features an integral display that will indicate the SFP level. The main signal processor and battery enclosure is located in the Control Room and the backup signal processor and battery enclosure is located in the EFT Building near the alternate shutdown panel. 01477432

10.2.1.3 Performance Analysis

The new fuel vault is not used for fuel storage. The vault floor drain prevents flooding. 01445693

The spacing of fuel bundles in the spent-fuel storage pool and Boral in the storage racks maintains k_{eff} <0.95.

A radiation monitor near the new fuel storage vault provides warning of any radiation level increase above normal operating conditions.

Fuel stored in the spent fuel storage pool is covered with sufficient water for radiation shielding and irradiated fuel being moved is normally covered by approximately eight feet of water over the actual fuel pellets. There is sufficient water depth above the fuel to provide decontamination of releases from damaged fuel resulting from a fuel-handling accident.

Special handling activities such as fuel assembly reconstitution can bring fuel pins to within 5 feet of the pool surface. Radiation levels above the fuel pool are monitored constantly by Area Radiation Monitors and Airborne Activity Monitors, equipped with audible alarms, that could activate and cause evacuation of the refueling floor.

A set of interlocks prevents handling of fuel over the reactor when a control rod is withdrawn, except when bypassed as allowed by Technical Specifications, and another set of interlocks prevents control rod withdrawal when fuel is being handled over the reactor. Load limit switches are provided on the hoist to prevent damaging fuel in the event a fuel element should become snagged. Limit switches on the fueling platform hoists interrupt power to the hoists when fuel is raised to a point approximately eight feet below the surface of the pool

water and the brakes on all equipment lock upon loss of power. Mechanical interlocks are provided on the hoist to ensure that spent fuel is not handled with an inadequate depth of water for shielding.

A liquid level switch monitoring pool water level is provided to detect loss of water and warn the reactor operator in time to permit refilling of the pool from the condensate storage and transfer system or other available systems. In addition, a second level switch in the skimmer surge tank is provided to permit almost instantaneous water loss detection.

Protection of the spent fuel pool from loss of water through the liner has been analyzed. Any unexpected leakage is primarily drained off from the liner seam leak detection system. Other seepage to the floor below is picked up in the reactor building floor drain radwaste subsystem. Available refill systems would far exceed makeup requirements for any expected leakage. Normal fuel storage pool makeup capacity is 100 gpm. As a backup system, a 450 gpm condensate backwash subsystem is also available. A supply of water from the Fire System is always available as a source for emergency makeup.

A redundant reactor building overhead crane system has been supplied for handling heavy loads both during refueling operations and during operations involving the transfer of spent fuel offsite or to the site’s Independent Spent Fuel Storage Installation (ISFSI). Such spent fuel transfers can take place either when the plant is operating or shut down. The redundant crane has been installed to reduce the probability of a heavy load drop to the category of an incredible event.

The heaviest load, other than the spent fuel cask, that has to be handled during refueling operations, when the plant is shut down, is comparable in weight to the IF-300 shipping cask or the loaded NUHOMS® spent fuel transfer cask.

Such “refueling” loads would only be handled when the plant is in a cold shutdown condition. Sufficient diversity in equipment exists to maintain the reactor in a cold shutdown condition should any one of the “refueling” loads be dropped. Therefore, this operation does not pose a significant safety hazard.

MNGP has upgraded the overhead crane system by making certain modifications. These modifications consist of replacing the existing trolley and hoisting system. Within the constraints of available space and requirements relative to performance, movement and weight, the new trolley and redesigned hoisting system, where practical, provides a dual load path, single-failure-proof hoisting system which complies with the provisions of draft NRC Regulatory Guide 1.104, “Overhead Crane Handling Systems for Nuclear Power Plants” (Reference 39). The non-redundant components, e.g., main hook, load block, shafts and structures have increased load safety factors to reduce the likelihood of their failure.

It can be concluded that modifications to the reactor building crane have incorporated all the provisions of draft Regulatory Guide 1.104, as applicable to an operating plant, that are practical for the Monticello design. In addition to the modifications to the reactor building crane, adequate measures exist to preclude the occurrence of a cask drop accident and to mitigate its effect in the very unlikely event that it should occur.

To minimize the potential of a cask dropping into the spent fuel storage pool, a detailed inspection of the reactor building crane is performed at the time of a spent fuel transfer or at a minimum of once per year. In addition, the spent fuel cask travel path is never over the reactor vessel or the fuel racks in the spent fuel storage pool. Travel over the spent fuel storage pool with the spent fuel cask is limited to that small area located for cask storage.

Prior to cask lifting operations a detailed visual inspection is made of all mechanical and electrical components of the crane. Following the visual inspection an operational test is conducted with no load on the hook. This test verifies that all controls are operating correctly. Following these tests a load test is conducted by raising the spent fuel cask approximately one foot off its transfer vehicle. The prime purpose of this test is to verify that there is no load movement after a fixed period of time. Hoisting and lowering rates are then determined to see that they comply with the vendor’s recommendations.

After confirmation of the operational acceptability of the crane, the fuel cask is hoisted to the refueling floor and moved over a controlled path to its position in the fuel storage pool. Specific load paths are approved by the Plant Operating Review Committee to ensure that the requirements of NUREG-0612 (Reference 40) are met (References 15 and 16).

10.2.1.4 Inspection and Testing

Prior to all fuel handling, all hoists, cranes and tools that are to be used are inspected and tested to assure safe operation.

Leak detection channels are provided on the concrete side of the spent fuel storage pool liner. Visual surveillance of these leak channels permit early determination and localization of any leakage and permit repair of leaks during construction. The surveillance lines are piped to a radwaste system sump to provide control of potentially contaminated water in the event of future leaks.

Per TS program 5.5.14 (Spent Fuel Pool Boral Monitoring Program), the Boral in the spent fuel pool is monitored to ensure that the neutron attenuation capability described in USAR section 10.2.1 is maintained (Reference 82). This monitoring includes periodic physical examination to ensure that the average thickness of the Boral meets the criticality analysis assumptions and neutron attenuation testing to ensure that the boron areal density meets the criticality analysis assumptions. 01445693

10.2.2 Spent Fuel Pool Cooling and Demineralizer System

10.2.2.1 Design Basis

The design objectives of the spent fuel pool cooling and demineralizer system are to handle the spent fuel cooling load and to maintain pool water purity and clarity. The system has been designed to provide sufficient filtering capacity to filter the entire spent fuel pool water volume every 12 hours. The fuel pool temperature is normally maintained at 125°F or less in order to maintain a reasonable working environment in the pool area, to keep the demineralizer system at an operable temperature, and to maintain visual clarity of the air above the pool. However, operation at temperatures up to 140°F is acceptable in order to remove decay heat from the spent fuel. The filtration flow may be varied to maintain required pool water clarity and purity levels.

10.2.2.2 Description

The spent fuel pool cooling and demineralizer system consists of circulating pumps, heat exchangers, filter/demineralizers, piping, valves and instrumentation. Table 10.2-1 lists the design parameters of the principal equipment of this system. The pumps take suction from the skimmer surge tank, located at the top of the spent fuel storage pool water level, which continuously skims the water from the surface, and circulates the water to the heat exchangers, and filter/demineralizers before discharging the water through the diffusers at the bottom of the spent fuel pool. This arrangement of taking suction from the top and discharging to the bottom of the pool provides a cross flow which tends to sweep the pool and to carry off dirt and small particles. Foreign material entering the pool either sinks to the bottom to be removed by a portable vacuum cleaner or floats in the pool and eventually enters the skimmer surge tanks and filtering loop.

This system may also be used to drain the steam separator-dryer assemblies storage pit and the reactor head cavity after refueling. The lines permit draining the water to either the reactor building equipment drain tank or to the spent fuel pool cooling and demineralizer system for processing, depending upon water condition. The fuel pool filters and the skimmer surge tank are shielded with concrete. Provision is made for connecting to the Residual Heat Removal System to provide for additional backup heat removal capacity. The liquid radwaste system has a tie-in connection to the fuel pool filter-demineralizers. The fuel pool filter-demineralizers can be used as needed as a back-up for processing liquid waste.

Section 15 Drawings NH-36256 and NH-36257 show the piping and instrumentation diagrams for the spent fuel pool cooling and demineralizer system.

10.2.2.3 Performance Analysis

The system is designed to maintain a maximum spent fuel storage pool temperature of 140°F. System capacity is sufficient to allow for normal reload variations of number of assemblies, cycle length, burn-up, and cooling times. 01101248

Normal and emergency heat loads are explicitly calculated prior to each fuel movement that could increase the spent fuel pool heat load. Primary considerations include fuel bundle exposure, cooldown and discharge intervals, uncertainty, and cooling system source water temperature.

The decay heat used in the evaluations is based on the ANSI/ANS 5.1-1994 standard (Reference 74) with a one sided 95% confidence interval. The normal spent fuel pool heat load is evaluated using conditions such that would be expected during routine refuel shuffles. The Normal Heat Load for this condition is 5.55 x 10^6 Btu/hr for a discharge completed 216 hours after shutdown. The Emergency Heat Load is nominally 24.7 x 10^6 Btu/hr for conditions which are as defined as: 01445693 / 01101248

> Emergency Heat Load - assumes full core discharge required 30 days following last refueling discharge and fills last 484 spaces; full core discharge is complete approximately 192 hours after shutdown.

The removal of heat for the emergency heat load, is accommodated by the use of either the spent fuel pool cooling and demineralizer system or by the residual heat removal system. During refuel outages, full core offloads are allowed because heat loads are explicitly calculated and compared to cooling capabilities prior to any fuel movement that would increase the spent fuel pool heat load. Removal of the heat from the full core offloads is also accommodated by the use of either the spent fuel pool cooling and demineralizer system or by the residual heat removal system. The RHR system has the capability of removing 26.43 x 10^6 Btu/hr from the fuel pool (Reference 68). Analysis has shown that the use of ATRIUM 10XM fuel does not increase the limiting spent fuel pool decay heat loads (Reference 81). 01445693

With an emergency heat load of 24.7 MBTU/hr, if fuel pool cooling capability is lost, the minimum possible time to achieve bulk pool boiling is 6.5 hours (assuming a maximum initial fuel pool temperature of 140°F). The maximum evaporation rate after bulk boiling commences is 53 gpm. 6.5 hours would be sufficient time to establish 53 gpm makeup rate from the Residual Heat Removal Service Water System (emergency makeup source). Under the bulk boiling conditions the temperature of the fuel does not exceed 382°F (Reference 61). 01101248 01101248

The fuel pool cooling heat exchangers are cooled by the reactor building closed cooling water system, thus preventing contamination outside the reactor building in the event of fuel pool heat exchanger tube failure.

The spent fuel pool cooling system is controlled from either a local panel in the reactor building, or a remote panel located in the radwaste building. The plant operators are provided with indications and/or alarm of system flow, pool water level, water temperature, skimmer surge tank level, and valve positions. Initial filling and level maintenance in the spent fuel storage pool and surge tanks is from the condensate storage and transfer system. When objects are placed in or removed from the pool, the water level varies in the surge tank.

10.2.2.4 Inspection and Testing

The spent fuel pool cooling and demineralizer system is in continuous operation except for preventive and corrective maintenance; thus, periodic testing of the system is not required. Initial testing was performed prior to plant startup operation.

10.2.3 Reactor Cleanup Demineralizer System

10.2.3.1 Design Basis

The design objectives of the reactor cleanup demineralizer system are: 1) to maintain high reactor water purity to minimize disposition on fuel surfaces by reducing the amount of water-borne impurities in the reactor primary system and 2) to reduce the secondary sources of beta and gamma radiation resulting from the deposition of corrosion products, fission products and impurities in the reactor primary system. To achieve these objectives the system is designed to meet the following design requirements:

Design Flow Rate	90,000 lbm/hr
Design Pressure (major components)	1450 psig
Design Code (major components)	ASME Boiler and Pressure Vessel Code, Section III, Class C

The reactor cleanup demineralizer system provides a continuous purification of a portion of the reactor coolant recirculation system flow with a minimum of heat loss and water loss from the cycle. It can be operated during startup, shutdown, cooldown, and refueling operations, as well as during normal operations. Water is normally removed at reactor pressure from one of the reactor coolant recirculation system loops and from the reactor vessel bottom head drain, and then cooled in regenerative and nonregenerative heat exchangers to about 120°F, filtered, demineralized, and pumped through the shell side of the regenerative heat exchanger to raise the temperature back up to about 440°F before returning it to the reactor through both reactor feedwater system lines. Return water may be isolated to one reactor feedwater system line as required with no detrimental effects with respect to feedwater nozzle fatigue. Section 15 Drawings NH-36254 and NH-36255 show the reactor cleanup demineralizer system P&IDs.

10.2.3.2 Description

The major components of the reactor cleanup demineralizer system are regenerative heat exchangers, nonregenerative heat exchangers, filter-demineralizers, circulating pumps and the necessary control and support equipment. Table 10.2-2 lists the principal design parameters of the system components.

The regenerative heat exchanger transfers heat from the water leaving the reactor to the water which returns to the reactor. The nonregenerative heat exchanger cools the water further to 120°F normal (140°F maximum) by transferring heat to the reactor building closed cooling water system. The nonregenerative heat exchanger is capable of maintaining this low temperature and not exceeding a maximum of 140°F even during a blowdown of the cleanup flow, when no cooling is provided by the regenerative heat exchanger.

Blowdown may be used during hot standby, startup and normal operation to remove excess reactor coolant inventory. The blowdown water may be directed to the main condenser or to the liquid radwaste system. Maximum blowdown flow is limited by the capability of the non-regenerative heat exchanger.

Two filter/demineralizers are provided for continuous operation. Spent cleanup resins are not regenerated because of the radioactivity of the impurities removed from the reactor coolant. They are sluiced from the demineralizer vessels directly to the Solid Radwaste System for processing, storing and eventual off-site disposal. Y-type post-strainers on the outlet of each of the filter/demineralizers prevent resins from entering the reactor primary system in the event of a resin support failure.

The operating reactor cleanup demineralizer system pump normally delivers 180 gpm to the filter demineralizers. The pumps are horizontal electric motor-driven centrifugal pumps.

10.2.3.3 Performance Analysis

Operation of the reactor cleanup demineralizer system is controlled from the main control room. Resin-sluicing operations are controlled from a local panel in the radwaste building. Radioactive equipment is shielded with concrete and access is controlled. The bases for shielding design were determined by the estimated frequency of operating, inspecting and maintaining the various devices.

The two 50% capacity filter-demineralizer units permit maintenance and recharging operations at reduced cleanup rates during system operation.

The system is provided with instrumentation to protect against overheating of the resins. The reactor cleanup demineralizer system is isolated by automatically closing the isolation valves on a reactor low-low water level, high drywell pressure, RWCU high flow or RWCU high room temperature signal from the plant protection system, described in Section 7.6. The reactor cleanup demineralizer system is also automatically shut down on a high-temperature signal on water entering the filter/ demineralizers. A bypass switch is provided

to allow operators to override the high water temperature shutdown and restore flow to the system. An interlocking system isolates the cleanup system upon initiation of the reactor standby liquid control system.

10.2.3.4 Inspection and Testing

Many of the components of the system are unavailable for inspection during operation due to the high radioactivity levels associated with the cleanup equipment. Many sample points are provided on the system. Use of the sample points and the flow and temperature instruments on the loop permit effective determination of the operation of the loops without access for inspection. After flushing the spent resins from the facility, accessibility to the filter vessel is improved and inspection accomplished if other tests indicate a desirability of inspection.

10.2.4 Reactor Shutdown Cooling System

10.2.4.1 Design Basis

The design objective of the reactor shutdown cooling system is to take over the cooling of the reactor water when the temperature and pressure in the reactor fall below the point at which the main condenser can no longer be used as a heat sink following reactor shutdown.

Once the reactor dome pressure has been reduced to 75 psig, the reactor shutdown cooling system is capable of cooling reactor water down to 125°F within 26.5 hours after reactor shutdown and maintaining it at this temperature 01101248
by removing fission-product decay heat absorbed by the reactor water (Reference 73, 75 and 76). To achieve this objective, the RHR heat exchanger was designed as presented in Table 6.2-2.

10.2.4.2 Description

The reactor shutdown cooling system is an integral part of the RHR System and is placed in operation during a normal shutdown and cooldown. The initial phase of nuclear system cooldown is accomplished by dumping steam from the reactor vessel to the main condenser with the main condenser acting as the heat sink. When nuclear system temperature has decreased to the value where the steam supply pressure is not sufficient to maintain the turbine shaft gland seals, vacuum in the main condenser cannot be maintained and the RHR system is placed in the shutdown cooling mode of operation. An evaluation was performed (Reference 76) which demonstrates that the RHR shutdown cooling system is capable of completing cool down to 125°F in 26.5 hours and maintaining the nuclear system at 125°F or lower so that the reactor can be refueled and serviced. The evaluation assumes RHR system heat removal is initially accomplished by two RHR heat exchangers each served by one RHR 01101248
pump and one RHRSW pump. After reactor pressure is reduced to a level to ensure RHRSW pressure can be maintained higher than RHR pressure, then all four RHRSW pumps and all four RHR pumps are assumed to be placed in service and remain in service until a reactor coolant temperature of 125°F is achieved. This is based on a river temperature of 90°F. Figure 10.2-6 shows a simplified P&ID of the system. The complete RHR P&IDs are shown on Section 15 Drawings NH-36246 and NH-36247.

Reactor coolant is pumped by the RHR pumps from one of the reactor coolant recirculation system loops and discharged through the RHR heat exchangers where cooling takes place by transferring heat to the RHR-service water system. Reactor coolant is returned to the reactor vessel via either one of the recirculation loops.

During a reactor primary system shutdown and cooldown, when the shutdown cooling subsystem is initially placed in operation, decay heat levels can be high and operation of both RHR heat exchangers may be required to remove the heat rapidly to the 125°F value assumed above for refueling and service to the reactor. When the decay heat level has decreased sufficiently, the entire shutdown cooling load can be shifted to one RHR heat exchanger.

10.2.4.3 Performance Analysis

The reactor shutdown cooling system is placed into operation during a normal plant cooldown when reactor dome pressure is below 81.8 psig. Operation of this portion of the RHR system for shutdown cooling does not compromise the ability of the RHR system to operate in the low pressure coolant injection system (LPCI) mode. During shutdown, the probability of requiring LPCI operation is very low. However, if LPCI operation is required, the operator can manually terminate shutdown cooling and start LPCI operation from the main control room.

Interlocks are provided to prevent over pressurizing the shutdown cooling system by inadvertent operation of the valves in the suction line from the recirculation loop (MO-2029 and MO-2030 Section 15 Drawing NH-36247, RHR System P&ID). The interlocks prevent the valves from opening if the pressure is greater than 81.8 psig reactor dome pressure as determined by either of two pressure switches.

The design pressure and temperature of the shutdown cooling suction line is 185 psig and 330°F. The design pressure and temperature of the shutdown cooling discharge lines is 510 psig and 330°F. Thus, an interlock setting of 81.8 psig is very conservative (Reference 14). If these valves were opened at low pressure and the pressure subsequently increased to 81.8 psig or more, the same circuitry causes MO-2029 and MO-2030 to close and isolate the shutdown cooling system from the reactor primary system. Additionally, the shutdown cooling discharge lines incorporate check valves (AO-10-46A and AO-10-46B) to prevent reverse flow and subsequent over-pressurization. Relief valves are also incorporated to provide protection: RV-2031, on the shutdown cooling suction line, is set at 150 psig; relief valves RV-2004 and RV-2005 on the discharge piping, are set at 500 psig.

10.2.4.4 Inspection and Testing

Periodic inspection and maintenance testing of the RHR pumps, pump motors, and heat exchangers are carried out in accordance with the manufacturer’s instructions. See also Section 6.2.3.

10.2.5 Reactor Core Isolation Cooling System (RCIC)

10.2.5.1 Design Basis

a. The system shall operate automatically in time to maintain sufficient coolant in the reactor vessel so that the core is not uncovered as a result of loss of off-site AC power or for a Loss of Feedwater event.

b. Provision shall be made for remote-manual operation of the system by a reactor operator.

c. To provide a high degree of assurance that the system shall operate when necessary, the power supply for the system shall be from immediately available energy sources of high reliability.

d. To provide a high degree of assurance that the system shall operate when necessary, provision shall be made so that periodic testing can be performed during normal plant power operation.

10.2.5.2 Description

The RCIC System consists of a turbine-driven pump unit capable of delivering makeup water to the reactor vessel. A detailed P&ID is shown on Section 15 Drawings NH-36252 and NH-36251.

Following a reactor shutdown, steam generation continues at a reduced rate due to the core fission product decay heat. Normally at this time, the main turbine bypass system diverts the steam to the main condenser, and the reactor Feedwater System supplies the makeup water required to maintain reactor vessel inventory.

In the event the reactor vessel is isolated, and the reactor Feedwater System supply is unavailable, primary system relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel drops due to relief valve operation. Upon reaching a pre-determined low-low level, the RCIC System is initiated as the steam admission valve automatically opens. The RCIC System supplies makeup water until a high level point is reached at which time the steam admission valve automatically closes to prevent turbine water induction. The auto restart of the turbine on low-low water level following a high water level trip satisfies NUREG-0737 (Reference 41), Item II.K.3.13.

The turbine driven pump supplies demineralized makeup water from the condensate storage tank to reactor vessel; an alternate safety related source of water is available from the suppression pool. Valves from the suppression pool suction line are automatically opened when condensate storage water level drops to the setpoint specified in the Technical Specifications. This automatic switchover satisfies NUREG-0737 Item II.K.3.22 (Reference 12). The condensate storage tank suction valve is automatically closed at the same time via an interlock with the suppression pool suction valves.

The RCIC System is not credited in the design mitigation sequence for SBO. Part of the RCIC System design basis is to show that the system will operate for two hours following a LOOP. This is a requirement of NUREG-0737 Item II.K.3.24. (References 77 and 78) 01101248

The turbine is driven with a portion of the decay heat steam from the reactor vessel, and exhausts to the suppression pool. Table 10.2-3 lists the principal design parameters for the system.

The components of the RCIC System include:

One 100% capacity turbine and accessories.

One 100% capacity pump assembly.

Piping, valves, and instrumentation for:

1. Steam supply to the turbine
2. Turbine exhaust to the suppression pool
3. Makeup supply from the condensate storage tank to the pump suction
4. Makeup supply from the suppression pool to the pump suction
5. Pump discharge to the feedwater line, including a test line to the condensate storage tank, and a minimum flow bypass line to the suppression pool
6. Condensing sparger, submerged in the torus water.

The turbine-pump assembly is located below the level of the reservoir and below the minimum water level in the suppression pool to assure positive suction head to the pump. Pump NPSH requirements are met by providing adequate suction head and adequate suction line size. All components necessary for initiating operation of the RCIC system are completely independent of any auxiliary AC power, plant service air, and external cooling water systems, requiring only 125 and 250 Vdc power from the plant batteries.

The RCIC system has sufficient capacity to keep the core covered following a loss of all AC power, assuming a conservative decay heat generation model at 2004 MWt. The general sequence of events in the Loss of Feedwater (LOFW) analysis is as follows: The reactor is assumed to be at 102% of 2004 MWt when the LOFW occurs. The initial level in the model is conservatively set at the low-level scram setpoint and reactor feedwater is instantaneously isolated at event initiation. Scram is initiated at the start of the event. When the level decreases to the low-low level setpoint, the RCIC system and MSIV closure are initiated. The RCIC flow to the vessel begins at 48 seconds into the event, minimum level is reached at 72 seconds and level is recovered after that point. Only RCIC flow is credited to recover the reactor water level. There are no additional failures assumed beyond the failure of the HPCI system. The decay heat used bounds ANS 5.1-1979 + two sigma. 01101248

This LOFW analysis is performed to demonstrate acceptable RCIC system performance. The design basis criterion for the RCIC system is confirmed by demonstrating that it is capable of maintaining the water level inside the shroud above the top of active fuel during the LOFW transient. The minimum level is maintained at least 77 inches above the top of active fuel, thereby demonstrating acceptable RCIC system performance. (Reference 72) 01101248

Upon receipt of the reactor vessel low-low water level signal, the RCIC System starts automatically and delivers design flow within 30 seconds.

The RCIC turbine has two systems for controlling power, with the control valve positioned by the lower setting of the two:

a. Speed governor, limiting the turbine speed to its maximum operation speed.

b. A control governor with automatic speed set point control which is positioned by a demand signal from a flow controller to maintain constant flow over the pressure range of RCIC operation.

The reactor operator has the capability to select manual control of the turbine governor and adjust power and flow to match decay heat steam generation during the period of RCIC operation.

The makeup water is delivered into the reactor vessel and distributed through the reactor feedwater system spargers to obtain mixing with the hot water and steam within the reactor pressure vessel. Bechtel calculations have shown that the RCIC-to-feedwater connection is not jeopardized by thermal shock, and consequently, there is no thermal sleeve installed in this connection.

Vacuum breaker lines were added to the RCIC turbine exhaust lines to prevent damage to the exhaust line check valves and improve low-load operation of the turbine. The vacuum breakers are installed in a line connecting the torus air space with the turbine exhaust line. The RCIC sparger support was redesigned to take the increased load due to the installation of the SRV T-quenchers.

During RCIC operation, the suppression pool acts as the heat sink for steam generated by reactor decay heat, resulting in a rise in pool water temperature. Heat exchangers in the Residual Heat Removal System are used to maintain pool water temperature within acceptable limits.

Provisions exist for automatic shutdown of the RCIC turbine upon receipt of any one of the following signals:

a. Turbine overspeed

b. High water level in the reactor vessel

c. Low pump suction pressure

d. High turbine exhaust pressure

The steam supply to the turbine is automatically isolated upon receipt of one of the following signals:

a. High pressure drop across a flow device in the steam supply line after a ≤7.16 second time delay. This time delay reduces the probability of a spurious trip during startup (Reference 11).

b. High area temperature, utilizing temperature switches connected in one-out-of-two-twice logic. High area temperature is also alarmed in the plant main control room, using independent temperature sensing elements.

c. Low reactor pressure, utilizing pressure switches connected in one-out-of-two-twice logic.

In addition to the automatic operational features, provisions are included for remote-manual startup, operation, and shutdown of the RCIC system provided low-low and high water level signals do not exist.

10.2.5.3 Performance Analysis

10.2.5.3.1 RCIC Operation Following Loss of AC Power

Upon loss of off-site AC power while the plant is operating at design power, the reactor is automatically scrammed and the reactor vessel is automatically isolated. The electric reactor Feedwater System pumps lose power and coast to a stop. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts and pumps make-up water to the vessel. The pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other makeup water system in operation.

Space cooling for the RCIC room is handled by one air cooling unit. On loss of AC power, this unit is powered from an emergency source, but there is no provision for a supply of emergency service water. Testing of Terry steam turbines and Woodward electronic control systems by the GE BWR Owners' Group has shown that realistic RCIC equipment environmental limits are 150°F at 100% relative humidity for two hours. An analysis of RCIC room heat-up after a complete loss of power indicates that the RCIC room will not exceed 150°F in two hours (Reference 9).

10.2.5.3.2 RCIC Operation Following Normal Reactor Shutdown

Following any reactor shutdown, steam generation continues due to heat produced by the radioactive decay of fission products. Initially the rate of steam generation can be as much as approximately 6% of rated flow and is augmented during the first few seconds by delayed neutrons and some of the residual energy stored in the fuel. The steam normally flows to the main condenser through the turbine bypass or, if the main condenser is isolated, to the suppression pool.

The fluid removed from the reactor vessel is normally made up by the reactor feedwater pumps or partially made up by leakage from the control rod drive system which is supplied by the control rod drive feed pumps. If make-up water is required to supplement or substitute for these primary sources of water, the RCIC turbine-pump unit starts automatically upon receipt of a reactor vessel low-low water level signal.

10.2.5.3.3 RCIC Action with RCIC Break

Should a break occur on the RCIC steam line either during operation of the RCIC or when the RCIC is not in operation, the RCIC steam line isolation valves close in time to prevent uncovering the core. The emergency core cooling systems (ECCS) provide core cooling in this situation.

10.2.5.4 Inspection and Testing

Provisions exist for testing the operability and performance of several components of the RCIC turbine-pump unit. The following are capable of being tested periodically:

a. It is possible to carry out a complete turbine-pump test at any time during normal plant operation. The steam admission valve is opened, and the motor-operated valve in the pump discharge line is closed. The system then feeds makeup water through a bypass line to the condensate storage tank. The bypass line is throttled by a motor operated control valve (drag type) to simulate a discharge into the pressurized reactor. A fatigue evaluation for the RCIC turbine exhaust torus nozzle (Reference 27) concluded that a usage factor of 0.729 for RCIC turbine operational testing corresponds to 4835 allowable operational tests of the RCIC turbine.

b. It is possible to isolate the system completely from the reactor primary system by valve operations if maintenance, adjustments, or tests on individual components are necessary during station operation.

c. It is possible to exercise and demonstrate the operability of all motor-operated valves.

The level of water in the condensate storage tank and suppression chamber is continuously indicated in the plant main control room. Instrumentation provides full monitoring of the RCIC operation. The turbine-pump unit is accessible for inspection and maintenance during operation at normal reactor power level.

10.2.6 Refuel Floor Auxiliary Bridge

10.2.6.1 Design Basis

The design objective of the Refuel Floor Auxiliary Bridge is to provide a platform to facilitate In-Vessel work performed during refueling outages. The auxiliary bridge is designed to meet the guidelines of NUREG-0612 (Reference 40). The bridge, monorails and hoists are designed to meet ANSI N14.6 requirements.

10.2.6.2 Description

The refuel floor auxiliary bridge is a personnel work platform. It is comprised of sections consisting of beams, two end truck assemblies, a work platform, and handrails. Deck plates between the parallel girders, supported by suitable cross members welded to the girders, make up a work platform that spans the reactor cavity. The bridge may be equipped with overhead monorails and hoists to raise and lower tooling. The platform and bridge assembly is placed on to the refueling platform rails then stored over the fuel pool until after reactor vessel disassembly. Following refueling activities and reactor vessel re-assembly, the auxiliary bridge is generally removed from the refueling platform rails.

10.2.6.3 Performance Analysis

The Refueling Floor Auxiliary Bridge is a tool designed to facilitate In-Vessel work during refueling outages. The additional load placed on the refuel floor will not exceed the design load of the floor. The auxiliary bridge is designed to the same degree of safety as the existing refueling platform and is seismically qualified by analysis to assure that the bridge will not fail, slip off the tracks or overturn due to a seismic event.

10.2.6.4 Inspection and Testing

Inspection and testing consists of standard receipt inspection. The inspection includes checking welded and bolted connections for structural integrity and applicable load testing of monorails and hoist assemblies per the requirements of NUREG-0612 and ANSI N14.6, Heavy Loads Design Criteria.

Table 10.2-1 Reactor Auxiliary Systems
Spent Fuel Pool Cooling and Demineralizer System
Principal Design Parameters

(Page 1 of 2)

SYSTEM

Reactor Core Thermal Power	2004 MWt 01101248
Fission Product Buildup Level	Equilibrium
Number of Spent Fuel Bundles Removed to Storage	Average Discharge Batch
Time after Reactor Shutdown when Spent Fuel Pool Cooling System Assumes Entire Heat Load, hours	Consistent with Refueling Schedule
Temperature at or below which the Storage Pool Water is Maintained	140°F

COMPONENTS

Spent Fuel Pool Cooling and Demineralizer Pump

Type of Pump	Centrifugal, horizontal
Number	2-50% each
Service	Continuous

Design Conditions, (each pump)

Pressure	200 psig
Temperature	150°F

Normal Operating Conditions, (each pump)

Capacity	450 gpm
Total Developed Head	175 ft
Suction Pressure	0 psig
Pumping Temperature	125°F
Type of Drive	Electric Motor
Rating	30 hp

Spent Fuel Pool Demineralizer Units

Type of Unit	Filter/Demineralizer
Number	2-50% ea.
Filter Element	304 stainless steel

Table 10.2-1 Reactor Auxiliary Systems
Spent Fuel Pool Cooling and Demineralizer System
Principal Design Parameters

(Page 2 of 2)

Design Conditions (each unit)

Flow Rate	450 gpm
Flow Area	225 ft^2
Pressure	125 psig
Temperature	140°F
Pressure Drop	
Clean	5 psi
Maximum	25 psi

Spent Fuel Pool Heat Exchangers

Type of Unit	Shell and tube heat exchanger, horizontal
Number	2-50% ea.

Design Conditions, (each unit)

	Tube Side	Shell Side
Fluid	Demineralized Water	Reactor Building Closed Cooling Water System
Design Flow/Max Flow	225/450 gpm	675 gpm
Design Pressure	250 psig	150 psig
Maximum Operating Temperature	140°F	140°F

	Tube Side	Shell Side
Maximum Pressure Drop	10 psi	7 psi
Heat Transferred	2.87×10^6 Btu/hr	2.87×10^6 Btu/hr
Inlet Temperature	125°F	95°F
Outlet Temperature	99.6°F	103.5°F

Power Sources

Pumping	Normal Auxiliaries and Diesel Generators
Control	Plant Batteries

Table 10.2-2 Reactor Auxiliary Systems
Reactor Cleanup Demineralizer System
Principal Design Parameters

(Page 1 of 2)

COMPONENTS

Cleanup Recirculation Pump	
Type	Horizontal, Centrifugal
Number	2-100% ea.
Design Conditions (each pump)	
Flow	180 gpm
Head	366 ft
Design Pressure	1450 psig
Design Temperature	120°F
Material	304 stainless steel
Motor	40 hp
$NPSH_r$	10 ft
Cleanup Filter-Demineralizer Unit	
Type of Unit	Filter/Demineralizer
Number	2-50% ea.
Design Conditions (each unit)	
Design Flux	1 psid @1 gpm per sq. ft of filter
Design Filter Area	94 ft^2
Design Pressure	1450 psig
Design Temperature	150°F
Filter Cartridge Material	304 stainless steel
Pressure Drop	
Clean	5 psi
Maximum	25 psi
Cleanup Regenerative Heat Exchanger	
Type of Unit	Multi-shell and tube heat exchanger, horizontal
Number Required	One unit: 3 shells, each shell with two tube passes

Table 10.2-2 Reactor Auxiliary Systems
Reactor Cleanup Demineralizer System
Principal Design Parameters

(Page 2 of 2)

Design Conditions (each unit)

	Tube Side	Shell Side
Fluid	Reactor Coolant	Reactor Coolant
Design Flow	90×10^3 lb/hr	90×10^3 lb/hr
Design Pressure	1450 psig	1450 psig
Design Temperature	575°F	575°F
Maximum Pressure Drop	10 psi	10 psi

Cleanup Nonregenerative Heat Exchanger

Type of Unit		Multi-shell and tube heat exchanger, horizontal
Number Required		One unit: two shells, each shell with two tube passes

Design Conditions

	Tube Side	Shell Side
Fluid	Reactor Water	Reactor Building Closed Cooling Water System
Design Flow	90×10^3 lb/hr	152×10^3 lb/hr
Design Temperature	575°F	370°F
Design pressure	1450 psig	150 psig
Maximum Pressure Drop	10 psi	10 psi
Heat Transferred	8.4×10^6 Btu/hr	8.4×10^6 Btu/hr

POWER SOURCES

Pumping	Normal Auxiliaries
Control	Plant Batteries

NOTE: Cleanup Heat Exchanger Designs are Based Upon Normal Operating Conditions.

Table 10.2-3 Reactor Auxiliary Systems
Reactor Core Isolation Cooling System (RCIC)
Principal Components Parameters

RCIC Pump

Flow Rate	400 gpm
Water Temperature	40°F to 140°F
NPSH	20 ft
Developed Head	2800 ft @ 1135 psia Reactor Pressure 525 ft @ 165 psia Reactor Pressure

RCIC Turbine

	H.P. Condition	L.P. Condition
Reactor Pressure	1135 psia	165 psia
Steam Inlet Pressure	1120 psia	150 psia
Exhaust Pressure	25 psia	25 psia
Steam Consumption	16,500 lb/hr	6,000 lb/hr

Valve Requirements

Steam Supply Valve	Open and/or close against full pressure within 15 sec
Pump Discharge Valves	Open and/or close against full pressure within 15 sec
Pump Minimum Flow Bypass Valve	Open and/or close against full pressure within 5 sec

Condensate Storage Requirement

Inventory for RCIC (with HPCI as backup to RCIC) 75,000 gallons

The condensate storage requirement exceeds that required for RCIC operation at maximum capacity during the two hour RCIC mission time for a loss of feedwater event. (Reference 78) 01101248

10.3 Plant Service Systems

10.3.1 Fire Protection System

10.3.1.1 Design Basis

The Monticello Nuclear Generating Plant (MNGP) Fire Protection Program has been established to minimize the likelihood of fires, ensure the capability to shut down the reactor and maintain it in a safe shutdown condition, and minimize radioactive releases to the environment in the event of a fire. The Fire Protection Program implements the philosophy of defense-in-depth protection against the hazards of fire and its associated effects on equipment important to safety by:

a. Preventing fires from starting

b. Rapidly detecting, controlling and promptly extinguishing fires that do occur

c. Providing protection for structures, systems, and components important to safety so that a fire not promptly extinguished by fire suppression activities will not prevent safe shutdown of the plant

Plant design features reflecting the above are described in Appendix J, “Fire Protection Program”.

Section 1 - Introduction - Background and objectives.

Section 2 - Licensing Summary - Summary of past submittals and evaluations related to fire protection and post-fire safe shutdown evaluations at MNGP.

Section 3 - Comparison to Regulatory Guidance - Comparison of the MNGP Fire Protection Program against the guidance contained in Standard Review Plan 9.5-1 (1976).

Section 4 - Post-Fire Safe Shutdown Analysis (SSDA) - Description of the post fire safe shutdown capability and evaluation of the adequacy of the separation of the redundant systems to ensure post fire safe shutdown.

Section 5 - Updated Fire Hazards Analysis (UFHA) - Evaluation of the adequacy of the plant fire protection features to control and contain the fire hazards within each plant area.

Significant MNGP and NRC correspondence, commitments, and approvals are documented in References 1-8, 10, 13, 18-26, and 55.

10.3.1.2 Description

10.3.1.2.1 Fire Control Systems

The Fire Protection System consists of a 1500 gpm diesel-driven vertical centrifugal pump, two 1500 gpm electrical motor-driven vertical centrifugal pumps, and a 50 gpm electric motor-driven horizontal centrifugal jockey pump, plus associated piping, valves, strainers, instrumentation and controls. The system P&ID is shown in Section 15 Drawings NH-36516 and NH-36666. One of the electric motor driven pumps supplies the fire system and is known as the fire pump. The second electric motor driven pump supplies the needs of the screen wash system in addition to being a fire pump and is known as the screen wash/fire pump. The 1500 gpm pumps each have a duplex basket strainer located in their discharge. The jockey pump takes its suction from the service water system header after the auto-strainer.

The jockey pump and/or the screen wash/fire pump are operated to maintain fire protection system header pressure greater than the electric motor-driven fire pump automatic start setpoint. Upon a drop of header pressure to 70 psig, the electric motor-driven fire pump and the diesel engine-driven fire pump start. After a time delay, if the pressure in the system header remains at 70 psig or less, the screen wash/fire pump starts as a third fire system pump. Since the diesel engine-driven fire pump is battery started, it will start when header pressure decreases to 70 psig, even if a loss of plant AC power were to occur. Provisions are made for the diesel fire pump to supply an alternate source of cooling water to the core or drywell spray. The connection is shown on Section 15 Drawing NH-36664.

The three fire pumps and the jockey pump are located in the intake structure with the diesel engine-driven fire pump in a separate room. The fire pumps take suction from the service water suction bay, and the jockey pump takes suction from the service water system header. The screen wash/fire pump connects to the electric fire pump discharge piping just before the basket strainer through a check valve.

Both the electric and the diesel fire pump discharge lines tie into the fire line loop. The fire line loop extends to the turbine, reactor and the administration buildings and to the cooling towers. The fire main and extensions have post indicator valves for isolation of any portion of the loop or the extensions. Fire hydrants are located at each cooling tower (see Section 15 Drawing NH-36516). The reactor and turbine building fire suppression systems are provided with an alternate supply line in case the normal supply header breaks.

Automatic wet pipe sprinkler and deluge systems are provided in the turbine building to protect the lube oil systems including the feedwater pumps, the turbine lube oil tanks, pumps and conditioner area, clean and dirty lube oil storage tank room, hydrogen seal oil unit, the warehouse area, and throughout the condenser level which contains the turbine lube oil piping and some feedwater heaters. Deluge systems are provided in the reactor recirculation pump motor-generator set room in the reactor building, for oil filled transformers located outside the turbine building, the turbine building siding adjacent to these transformers and the cooling towers. Pre-action sprinkler systems are provided in the radwaste shipping building, No. 13 diesel

generator enclosure, the intake structure, turbine and exciter bearings and the diesel generator and day tank rooms. The pre-action sprinkler system in the diesel generator and day tank rooms has a design density of 0.30 gpm/sq ft over the entire diesel generator area. The sprinkler and deluge systems are designed and maintained in compliance with the provisions of NFPA 13 and NFPA 15 with exceptions taken when justified. Fire protection systems are not backfit to changes in later NFPA codes.

Hose stations equipped with 1-1/2 inch synthetic jacket-lined hose are provided in various fire areas of the plant. The nozzles associated with the hose stations are the all-fog type incorporating a wide angle to narrow angle fog pattern adjustment.

A Halon fire extinguishing system is provided in the cable spreading room. The Halon system consists of cylinder storage units pressurized with Halon 1301. Halon is discharged into the room through wide angle nozzles.

Portable fire extinguishers are provided in the reactor building, turbine building, and administration building. NFPA 10 guidance is used for extinguisher placement and selection.

The HEPA/charcoal filter units in the emergency filtration train (EFT) building are provided with a deluge system which can be connected to the fire water line located near the unit by means of a hose.

10.3.1.2.2 Fire Detection and Signaling Systems

Ionization and multi-sensor (photo-electric and thermal) type fire detectors are provided throughout the plant. Fire detectors of these types provide prompt response and have adequate sensitivity to the products of combustion for the types of combustibles typically found in nuclear power plants. The detectors in each fire area initiate an alarm locally and in the control room upon detecting either combustion or failure in the detector system.

Battery operated ionization type fire detectors are provided inside closed control room cabinets and in the ceiling above other cabinets not visible from the general area of the control room. These also detect the airborne products of combustion always present before a significant temperature rise has occurred. They do not require the presence of smoke or flame for immediate detection.

The fire detection and signaling system also includes heat activated devices that utilize the heat from a fire to operate pneumatic systems to either sound alarms or automatically initiate a deluge or sprinkler system.

10.3.1.2.3 Emergency Communication and Lighting Systems

The plant is equipped with emergency communication and lighting systems that are effective and useful for fire fighting operation (see Sections 10.3.8 and 10.3.9).

10.3.1.2.4 Administrative Controls

Administrative controls and procedures are provided which satisfy the requirements of Sections III.H, I and K of Appendix R to 10CFR Part 50 (Reference 6). The administrative controls for fire protection consist of the fire protection organization, fire brigade training, the controls over combustibles and ignition sources, and quality assurance provisions for fire protection.

Administrative controls for the Fire Protection Program are comprised of the following key documents which are, therefore, part of the USAR by reference:

a. Administrative Work Instruction 4 AWI-08.01.00 (FIRE PROTECTION PROGRAM PLAN) establishes the basis for the Fire Protection Program. It documents the licensing background of the MNGP Fire Protection Program. For each Program Element, it identifies the applicable principal regulatory documents, major MNGP commitments, NRC approval of commitments, implementing documents and organizational responsibilities. Fire Protection Program organizational and implementing document relationship charts are also provided.

b. Fire impairments and surveillance requirements for fire protection systems and equipment per Table A.2-1 and Table A.2-2 in Operations Manual B.08.05-05.

c. Fire brigade requirements per Table A.2-4 in Ops Manual B.08.05-05.

10.3.1.2.5 Fire Barriers

The plant is comprised of numerous separate fire areas, some of which may contain multiple fire zones. 10CFR50, Appendix R fire barriers are installed between different fire areas to separate redundant trains of equipment required to achieve and maintain safe shutdown.

When necessary, structural steel forming a part of or supporting such barriers are coated with a qualified fireproofing material.

10.3.1.3 Performance Analysis

10.3.1.3.1 General

The fire suppression water system is supplied by three identical vertical centrifugal pumps rated at 1500 gpm at 100 psig each. Two of these pumps are motor driven and one diesel driven. One of the motor driven pumps normally supplies the needs of the screen wash system and is designated the screen wash/fire pump. Transfer from screen wash duty to fire duty occurs automatically. The screen wash/fire pump and motor is equivalent in every respect to a UL Listed fire pump. All pumps are started automatically by instrumentation sensing header pressure. Any two fire pumps are capable of supplying fire fighting water requirements in any fire area of the plant. Fire zones 23A, 23B and 16 as defined in Reference 1 contain power supply cables for both electric fire pumps. A fire in these zones could disable both electric fire pumps and leave only the diesel fire pump to extinguish the fire. Due to the low combustible loading in these zones, the diesel fire pump would be adequate to suppress the fire.

If any of the automatic suppression systems included in Operations Manual B.08.05-05 Table A.2-1 are inoperable, compensatory measures are implemented in accordance with Operations Manual B.08.05-05.

Hose stations and yard hydrant hose houses are provided in all safety related fire areas of the plant and surrounding all principal plant buildings. These stations are supplied from the fire suppression water system. If the water supply to these areas is interrupted, a hose supplied from an operable source is provided to protect the affected fire area having the inoperable station.

Fire detection instrumentation is installed throughout the plant to protect safety related structures, systems, and components. The detectors in each fire zone initiate a local alarm and an alarm in the control room. All circuits are supervised. Installation meets the applicable circuitry technical requirements of NFPA-72D (Reference 59) and NFPA-72 (Reference 60). Operations Manual B.08.05-05, Table A.2-3 requires a minimum number of detectors to be operable in each fire zone. If more detectors are inoperable, a fire watch patrol is established in the affected fire zone until the required number of detectors are restored to operable status. The loss of one detector will not significantly degrade the ability to detect fires in zones of the plant having multiple detectors.

Piping and electrical penetrations are provided with seals where required by the combustible loading. If a seal is made or found to be inoperable for any reason, the penetration area is continuously attended until an effective fire seal is restored or the operability of the fire detectors on at least one side of the inoperable seal is verified and an hourly fire watch patrol is established. Seals were qualified for the maximum fire severity present on either side of the barrier.

10.3.1.3.2 Smoke Removal

The turbine building is equipped with turbine floor fans which discharge to the main exhaust plenum for smoke removal. The fans and dampers are actuated by high temperature sensors provided that the main exhaust plenum differential plenum interlock is satisfied. No other smoke removal mode of ventilation is provided for the other plant buildings. In the buildings without a smoke exhaustingmode, the normal ventilation systems could be used for smoke removal for some types of fires although not specifically designed for this purpose. The fans and other equipment in the normal air handling systems are not designed to withstand high temperatures, and could be incapacitated by the heat from a significant fire. However, in plant fire areas other than the turbine building the fire loading is not high, and the installed and proposed fire protection features will limit the amount of smoke and hot gases generated. When normal air handling cannot be used for smoke removal, the plant personnel will use the portable and semi-portable fans available at the plant.

10.3.1.3.3 Effects of Water Spray

Existing water suppression systems are not located in fire areas with equipment that would be highly sensitive to the effects of water impingement. The water spray systems protecting the outside transformers are manually initiated. Transformers are provided with linear heat detectors (main, 1R, & 2R) or heat activated devices (turbine building siding) which initiate an alarm in the control room enabling operations to determine the need to activate the local deluge system. Water spray systems do not have a detrimental effect on these units as this type of equipment is designed for outside environments. The suppression systems protecting the seal oil areas in the turbine building will not discharge in the area of safety-related equipment such as switchgear which is sensitive to water. Pumps and equipment are mounted on pedestals which will prevent the water accumulations on the floors from affecting them. The deluge systems in the reactor recirculation pump motor generator set room utilize directed spray nozzles aligned in such a way that the water sensitive portions of the equipment are not wetted. The sprinkler systems protecting the warehouse area and cooling towers will have no effect on safety related equipment because of distance. The safe shutdown equipment in the intake structure is protected by water deflection shrouds.

10.3.1.4 Inspection and Testing

Equipment and systems are inspected and tested upon installation in compliance with requirements of the authorities and fire underwriters having jurisdiction and in general conformance with the guidelines of NFPA standards. The applicable fire protection system surveillance requirements are discussed in Ops Manual B.08.05-05, Table A.2-2.

10.3.1.5 Safe Shutdown Analysis

10.3.1.5.1 General

To comply with 10CFR 50, Appendix R, Section G, a safe shutdown analysis (SSDA) was performed (Reference 10). Two trains of safe shutdown systems were identified to achieve safe shutdown in a fire event (Figure 10.3-7). Based on the fire areas established (References 1 and 10), an analysis was conducted to establish that one train of safe shutdown systems would be free of fire damage for any fire event. The analysis revealed four fire areas that contain components from both trains of safe shutdown systems. Below is a list of the fire areas and actions needed. All other fire areas contain components for one train of shutdown equipment only. 01483863

Fire Area	Description	Resolution
IV	Suppression Pool Area	NRC Exemption approved based on adequate separation and low fire loading (Reference 13).
VI	Cable Spreading Room	Alternate Shutdown System installed.
VIII	Control Room	Alternate Shutdown System installed and approved NRC exemption from suppression system in the Control Room (Reference 25).
IX	Intake Structure	NRC Exemption approved based on a horizontal separation greater than 20 feet without intervening combustibles. (Reference 69 and 70) 01483863

The minimum safe shutdown system consists of one of the two trains of Safety/Relief Valves (SRVs), core spray, torus cooling mode of the Residual Heat Removal System (RHR), and emergency diesel generators. The minimum systems provide for a means to depressurize (SRVs), make-up coolant inventory (core spray), cool the torus (RHR), and supply power (EDG). Cold shutdown is achieved with alternate shutdown cooling by using core spray to pump water from the torus to the reactor vessel. Water is then discharged through an SRV while RHR pumps and heat exchangers are used for cooling the torus water.

In the event of a fire of reasonable size, there would normally be additional equipment available beyond the minimum equipment needed to achieve a safe shutdown condition. Because of this, vessel depressurization and floodup would most likely be unnecessary.

Two areas of the plant, the Control Room and Cable Spreading Room, contain many cables from both trains of safe shutdown equipment. An alternate shutdown system was installed during the 1986 refueling outage to provide the required protection for a fire in either of these areas. The alternate shutdown system provides control of the minimum necessary Division II systems, once the transfer switches are activated, to achieve safe shutdown.

10.3.1.5.2 <u>Associated Circuits</u>

Safety related and non-safety related cables that are associated with the equipment and cables of the selected shutdown systems are those that have a physical separation from the fire area less than that required by Section III.G.2 of Appendix R to 10CFR50 and (1) a common power source with the safe shutdown equipment and the power source is not electrically protected from the shutdown circuit by coordinated circuit breakers, fuses or similar devices, or (2) a connection to circuits of equipment whose spurious operation will adversely affect the shutdown capability, or (3) a common enclosure (e.g., raceway, panel, junction box) with shutdown cables and not electrically protected from the shutdown circuits of concern by circuit breakers, fuses or similar devices, or will allow propagation of the fire into the common enclosure.

An ongoing fuse and circuit breaker coordination study is used to show that associated circuit types 1 and 3 do not exist at the Monticello Plant. The study includes all protective devices throughout the plant auxiliary power distribution system that serve identified safe shutdown equipment. This study, together with review of existing documentation on the 4.16 KV switch gear and the 480V load centers, demonstrate that hot shorts, open circuits, or shorts to ground will not adversely affect nor prevent operation of the selected shutdown equipment.

Associated circuit type 2 as defined above has been addressed in the “Fire Protection and Safe Shutdown Systems Analysis Report”, (Reference 10). Three lines have been identified that contain motor operated valves which, if spuriously operated, could drain reactor coolant. To address this concern, motor operated valves MO-2029 (RHR Shutdown Cooling Suction Inboard Isolation), MO-2032 (RHR Discharge to Waste Surge Tank Inboard Isolation) and MO-2401 (RWCU Discharge Orifice Bypass) are closed and their control and power circuits deenergized during reactor power operation.

10.3.1.5.3 <u>Fire Barriers</u>

Fire barrier upgrades have been made in accordance with 10CFR50 Appendix R, Section III.G, and in response to “Fire Protection and Safe Shutdown Systems Analysis Report”, (Reference 10). This report defines the fire area boundaries throughout the plant and describes the upgrades made to maintain separation.

Where required, new fire walls were constructed to an approved design to give a 3-hour fire-resistive rating. All replacement and new doors in fire barriers shall be certified to UL, or equivalent, standard(s), and to a specified duration rating (e.g., 3-hour rating). New fire dampers installed in ducts penetrating fire barriers are UL listed, or equivalent, 3-hour rated thermally actuated dampers. All other fire barrier penetrations (i.e., pipe, cable tray, bus ducts, etc.) are sealed with a qualified sealant material in accordance with an approved penetration seal design.

Existing fire area boundaries that have a fire rating of 150% or more of the loading on either side of the boundary were judged acceptable. Therefore, all fire area boundaries are not necessarily 3-hour rated if they existed prior to the issuance of Appendix R and meet the above criterion.

10.3.1.5.4 Alternate Shutdown System (ASDS)

The alternate safe shutdown system is required for a 10CFR50, Appendix R event in the control room and/or cable spreading room. The system utilizes existing Division II systems and equipment and a remote ASDS control panel. The ASDS control panel is located on the third floor of the emergency filtration train (EFT) building adjacent to the turbine building and the control room. This ASDS panel is safety grade (Class 1E) and provides the controls, AC circuitry, and instrument readouts for the operator to safely shutdown the plant at a centralized location assuming a fire in the control room or the cable spreading room. The design of the ASDS panel includes a master transfer switch which, when activated, enables ASDS operation, initiates an annunciator in the control room, and initiates an indication light and activates other transfer switches at the ASDS control panel. It also includes an MSIV isolation switch and four system transfer switches which, when activated, will ensure closure of MSIVs and enable the manual control and operation of the four safety/relief valves, core spray system, RHR system and other auxiliary systems from the ASDS control panel independent of the control and cable spreading rooms. The instrumentation provided at the ASDS panel is also independent of the control and cable spreading rooms. A fire at the ASDS control panel will result in loss of control for only the equipment controlled from the panel. One train of systems needed for safe shutdown will be available and controllable from the main control room.

For post-fire shutdown, the performance goals of the alternative safe shutdown capability will be met utilizing the reactor pressure relief system, the core spray system, and the RHR System. Reactivity control will be provided by a manual scram of the control rods from the control room. A minimum of two safety/relief valves will be manually controlled at the ASDS panel to reduce reactor coolant system pressure. After depressurization, the Core Spray System will provide reactor inventory make-up. The Core Spray System will be used to establish a cooling path by allowing the reactor vessel water level to rise until water flows through the safety/relief valve lines into the suppression pool. Decay heat removal is provided by manual operation of the RHR system in the suppression pool cooling mode. The suppression pool is capable of providing sufficient cooling until the RHR System is initiated. Direct indication of process variables including reactor coolant level, reactor coolant pressure, suppression pool temperature and suppression pool level is provided at the ASDS control panel. Diagnostic monitoring at the ASDS panel includes AC/DC power supply availability, pump breaker positions, valve positions, flow indications, and diesel fuel oil day tank level. The Division II support systems available include the No. 12 Emergency Diesel Generator and the Essential Electrical Distribution System, RHR Service Water System, Emergency Service Water System, ECCS room cooler, RHR Auxiliary Air System, communication, lighting, and ventilation for the ASDS room. 01483863

An associated circuits review was performed for the control room and cable spreading room. Where concerns were identified methods consistent with the guidelines contained in Generic Letter 81-12 (Reference 50) were used to protect the safe shutdown capability from fire-induced failures of these circuits.

Information in support of the ASDS design was submitted to the NRC for review and approval in References 19, 20 and 21. Reference 22 contained the NRC Staff's Safety Evaluation which concluded that, "...the Monticello Nuclear Generating Plant design provides one train of systems necessary to achieve

and maintain safe shutdown conditions by utilizing either the control room or the Alternate Shutdown System control panel, and thus meets the requirements of Appendix R to 10CFR50, Items III.G.3 and III.L with respect to alternate Safe Shutdown in the event of a fire".

10.3.2 Plant Heating Ventilating and Air Conditioning Systems

10.3.2.1 Design Basis

The design objectives of the plant heating, ventilating and air conditioning systems are:

a. To provide appropriate ambient temperature and humidity environmental conditions for plant operating personnel and equipment.

b. To provide means for purging the primary containment.

c. To provide sufficient fresh air for personnel and to provide volume removal of equipment odors and fumes.

10.3.2.2 Description

10.3.2.2.1 General

The plant heating, ventilating and air conditioning systems provide individual air supplies to main areas of the plant. Normal air flow is routed from areas of lesser to areas of progressively greater contamination potential prior to final exhaust.

The general ventilation air exhaust from areas that have the potential to contain radioactive airborne activity are monitored and released above reactor building roof level in such a manner as to minimize the possibility of the same air being drawn into a fresh air intake.

10.3.2.2.2 Reactor Building

Refer to Section 5.3.4.

10.3.2.2.3 Standby Gas Treatment System

Refer to Section 5.3.4.1.

10.3.2.2.4 Turbine Building

The turbine building ventilating system is designed to provide filtered and heated air at an approximate rate of one change per hour. It is ventilated by supply and exhaust fans through distribution ductwork. The supply system is equipped with filters for cleanliness. Clean supply air is directed to those areas with the lowest contamination rating and exhausted from those areas with higher contamination ratings so as to ensure movement of air from clean areas to areas with progressively greater contamination potential. Exhaust air from the main condenser area is discharged through the Main Exhaust Plenum Room and Reactor Building Stack. The turbine building air flow is illustrated in Section 15 Drawing NH-36263.

Due to concerns over the operability of Safety Related equipment located in areas of the turbine building served by Non-Safety Related ventilation systems, special testing was performed to confirm that compensatory measures were effective in controlling temperatures in the event that the ventilation system failed. (Reference 32).

10.3.2.2.5 Radwaste Building

The radwaste building heating and ventilating system is designed to provide conditioned air at an approximate rate of one change per hour. This supply air may be filtered or unfiltered depending on seasonal conditions. It has a supply and exhaust ventilating system. Air is removed and controlled so as to insure movement of air from the clean areas to areas with progressively greater contamination potential. All exhaust air is treated as potentially contaminated and is filtered. Provision is made to sample the exhaust air prior to discharge into the Reactor Building Stack. Flow of air in the radwaste building is shown on Section 15 Drawing NH-36266.

10.3.2.2.6 Primary Containment

Refer to Section 5.2.2.5.2.

10.3.2.3 Performance Analysis

The primary purpose for all ventilation systems is to remove heat produced by equipment, piping, and motors. The quantity of air required is determined as that required to remove the heat loads in the equipment areas. Where spaces are occupied by personnel and are without high heat producing equipment, the ventilation quantity requirements are defined by establishing air change rates required for personnel comfort and safety in accordance with latest heating, ventilating and air conditioning practices.

Spare or standby equipment is installed in heating, ventilation, and air conditioning systems which are essential to maintain proper air pressures or temperatures for the safe and continuous operation of the plant. Controls for the ventilating system are designed to be as simple as possible and centrally located when practical. Equipment for temperature indication, high temperature alarm, air flow failure indication and differential pressure measurement is installed.

Air flow is controlled so that air moves from nonradioactive areas to potentially radioactive areas by supplying filtered air to clean areas and exhausting air from the potentially contaminated areas. This arrangement ensures a positive air pressure in clean areas and a negative air pressure in potentially contaminated areas.

Filters are arranged to permit easy removal and replacement to minimize exposure for personnel. Actual filter life cannot be anticipated, since it depends on the amount of contamination in the air handled and the size and concentration of particles entering the filter.

10.3.2.4 Inspection and Testing

Ventilating and air conditioning equipment and systems are monitored and maintained during normal operation. Standby equipment is tested and inspected periodically to assure their availability when required.

10.3.3 Plant Makeup Water Treatment System

10.3.3.1 Design Basis

The Plant Makeup Water Treatment System is designed to provide a supply of treated water suitable as makeup for the plant and reactor cycles and other demineralized water requirements.

10.3.3.2 Description

The Plant Makeup Water Treatment System is sized to continuously produce demineralized water suitable for use in the plant and reactor cycles. Raw water is supplied from the plant domestic well water. The water is then demineralized to make it suitable for use in the reactor and turbine systems. A storage tank for the demineralized water is provided and water is distributed to the various services as required.

The Plant Makeup Water Treatment System supplies demineralized water for the following plant equipment and systems:

Condensate Storage System
Reactor and Turbine Building hose stations
Spent Fuel Storage Pool and Spent Fuel Pool Cooling and Demineralizer System
Reactor Building Closed Cooling Water System
Reactor Water Cleanup System
Heating Boiler Deaerator
Standby Liquid Control System
Radioactive Waste Control Systems
Pressure Suppression Chamber (Pool)
Chemistry Laboratory
Condensate Storage and Transfer System
Makeup Demineralizer
Hot Water Heating System
Stator Winding Liquid Cooling Units
Reactor Core Isolation Cooling System
High Pressure Coolant Injection System
General Electric Zinc Injection Passivation (GEZIP) System
On-Line Noble Chemistry System

10.3.3.3 Performance Analysis

The piping, tanks and other equipment of the demineralized water treatment systems are of corrosion resistant metals or with coatings which prevent contamination of the makeup water with foreign material.

Transfer of water from the demineralizer storage tank to the condensate storage tank is accomplished. The discharge from the demineralizer is continuously monitored for quality by conductivity measuring devices which initiate an alarm.

10.3.3.4 Inspection and Testing

All systems and equipment were inspected and tested before operation. Equipment is monitored and checked for performance on a continuous basis during normal operation, where no special testing or inspection is considered necessary.

10.3.4 Plant Air Systems and Nitrogen Systems

10.3.4.1 Design Basis

The plant pneumatic systems consist of a plant Instrument and Service Air System, an outboard main steam isolation valve (MSIV) air supply, an Alternate Nitrogen System, and an instrument nitrogen supply to containment. The plant Instrument and Service Air System provides the plant with a continuous supply of oil-free compressed air. Dry air is supplied to plant instruments and controls as required and is provided at hose stations throughout the plant. The outboard MSIV air supply provides a source of high pressure air (nominal 280 psig) to open and maintain open the outboard MSIVs. A safety related Alternate Nitrogen System provides an alternate pressure source to equipment required during or following an accident. An instrument nitrogen supply is provided to instruments, controls, and equipment located in the drywell in lieu of instrument air. This is to eliminate a source of oxygen build-up in containment through in-leakage.

10.3.4.2 Description

10.3.4.2.1 Instrument and Service Air Systems

The compressed air systems are shown on Section 15 Drawing NH-36049, Sheets 1, 2, 3, 4, 11, 12, 14 and 15. The Instrument and Service Air System includes three non-lubricated, variable speed drive air compressors, each with an intake filter, air cooler, moisture separator/trap and associated dryer and receiver. These compressors are rated at a nominal 717 scfm at a discharge pressure of 125 psig.

The compressors are controlled from a common control panel and associated system header pressure transmitters to allow for any combination of lead or standby function of each compressor as desired. In addition to this common control panel, each compressor can be controlled from its integral controller and discharge pressure transducer. Normally, one air compressor is in the lead position and the others are in standby as controlled by the common control panel.

The three compressors discharge air to their respective air dryers through aftercoolers with moisture separator/traps. Each compressor is cooled by its dedicated ethylene glycol closed cooling system. The air dryers supply the air receivers and the Instrument and Service Air System. The receivers act as a surge volume to minimize the effects of pressure surges caused by usage, and the loading and unloading of the compressors. Flow measuring instrumentation allows monitoring of compressor performance and instrument and service air usage.

The air for the Instrument and Service Air System is dried by means of 900 scfm twin bed desiccant type air dryers, each installed between a pre-filter and after-filter. When one bed of a dryer is in service, the alternate bed is in standby. When a dryer senses too high of a moisture content in the standby bed, it will automatically put that bed in the regenerating cycle. The beds are alternated in service to provide a continuous flow of dry air to the system. An after-filter is located downstream of each dryer to trap any particulates which pass through or are released by the compressors or air dryers.

The instrument air portion of the Instrument and Service Air System supplies the compressed air requirements for most pneumatic instruments and controls located throughout the plant.

The service air portion of the Instrument and Service Air System consists mainly of a piping system which supplies hose stations throughout the plant and is intended for various miscellaneous usages by maintenance and operations personnel.

10.3.4.2.2 Outboard MSIV Main Air Supply

The normal main air supply for the outboard MSIVs is provided by bottled air pallets stored in an air pallets storage building located west of the condensate storage tanks. (The outboard MSIVs' pilot air supply is provided by the plant Instrument and Service Air System.) Two main air supply headers are located in the air pallets storage building. One air pallet is coupled to each air header but only one pallet is required to be valved in service at a time. Bottled air at a pressure range of 600 psig to 2500 psig is fed from the pallets to the air supply headers where it is reduced to a nominal pressure of 280 psig. The air headers connect to a single common 2-inch main air supply through which the air is transported to an accumulator located in the Turbine Building. Four lines, one for each outboard MSIV, are routed from the accumulator to the MSIVs located in the steam chase. The accumulator prevents open outboard MSIVs from drifting closed during momentary periods of high air demand and mitigates the effects of thermal expansion of air in the MSIV actuators.

Two air driven air compressors (intensifiers), located in the reactor building, provide a backup source of main air for the outboard MSIVs upon loss of the normal air supply. Each intensifier is capable of boosting its supply air pressure by a factor of five. At a flow rate of about 9 scfm the output pressure would be 300 psig. Approximately 50 scfm of supply air (total of drive air and main air) will be consumed by each intensifier at this output rate. Drive and main air supplies for the intensifiers are provided by the Instrument and Service Air System. The intensifiers are manually valved in service when needed.

10.3.4.2.3 Alternate Nitrogen System

The Alternate Nitrogen System consists of two separate safety related trains providing a safety related back-up pneumatic source from nitrogen bottle racks located on the 931’ elevation of the turbine building. The location of the bottle racks permits replacement of nitrogen bottles to maintain the nitrogen pressure during normal operation and following an accident. Train A provides backup pneumatic supply to the T-ring seals of the inboard Primary Containment Atmospheric Control System purge and vent valves, the T-ring seals and actuators of the reactor building to suppression chamber vacuum breakers, and safety relief valves RV-2-71A, RV-2-71B, and RV-2-71E. Train B provides backup pneumatic supply to the T-ring seals of the outboard Primary Containment Atmospheric Control System purge and vent valves, the inboard main steam isolation valves AO-2-80A, AO-2-80B, AO-2-80C, and AO-2-80D, and safety relief valves RV-2-71C, RV-2-71F, and RV-2-71H. Train B also provides the sole pneumatic supply to the Primary Containment Hard Pipe Vent System. Manifold and system pressures of each train are monitored by pressure switches which give control room annunciation on low pressure.

10.3.4.2.4 Instrument Nitrogen Supply to Containment

The Primary Containment must be maintained at a oxygen concentration less than 4 percent by volume. Constant leakage from the Instrument and Service Air System increases the oxygen content if compressed air is used to supply air operated equipment located in the drywell. When the concentration exceeds 4 percent by volume, the primary containment must be purged. In order to reduce the need to purge, a nitrogen supply to the instrument air header inside the drywell is provided. Nitrogen is supplied through a regulator from the plant nitrogen storage tank. In the event of loss of nitrogen pressure, the containment supply is automatically transferred to the Instrument and Service Air System.

10.3.4.3 Performance Analysis

Since vital instrumentation in the plant is primarily electrically powered, large capacity instrument air storage is not required as plant safety does not depend on this system. One compressor can be operated by essential power supplied by the Emergency Diesel Generator System or No. 1AR transformer. The second and third can be operated by the non-essential diesel generator.

The plant air systems and instrument nitrogen systems are not considered to be a safety related systems since equipment requiring compressed air or nitrogen for operation during or immediately subsequent to an accident is supplied from local accumulators, other air sources, or the Alternate Nitrogen system.

Accumulators or other pneumatic sources are provided at the following points:

RCIC Minimum Flow Valve (1).

RHR Pump Minimum Flow Valves (4) - The containment response analyses assume the RHR pump minimum flow valves fail open due to loss of accumulator pressure after 10 minutes. See USAR section 5.2.3.3. 01101248

Primary Containment purge and vent valves (7)-Local accumulators, in conjunction with the Alternate Nitrogen Supply System, provide sufficient air/nitrogen to maintain T-ring seals pressurized to maintain Primary Containment integrity.

HPCI Pump Minimum Flow Valve (1).

Safety/relief valves (SRVs) - Two SRVs, G and D, are each served by a bank of four accumulators charged by the non-safety related Instrument Air/Instrument Nitrogen Systems in the drywell. The remaining six SRVs are connected to the Alternate Nitrogen System. The accumulators and Alternate Nitrogen System provide a safety-related backup supply in the event the non-safety related pneumatic systems are lost.

Inboard Main Steam Isolation Valves (MSIVs) - The inboard MSIVs are normally supplied by the Instrument Air/Instrument Nitrogen systems with the Alternate Nitrogen System serving as a safety-related backup. There are four accumulators connected to the pneumatic supply header to the inboard MSIVs. These accumulators are located in close proximity to the MSIVs and serve to minimize pressure drop and to speed up valve closure.

Standby Gas Treatment System Valves - (1) Air operated valves and instrumentation associated with the Standby Gas Treatment System are supplied from the plant Instrument and Service Air System. The valves fail safe.

RHRSW Control Valves - (2) Are supplied by instrument air normally, and upon occurrence of low air pressure, are supplied by the RHR auxiliary air compressors. The RHRSW auxiliary air compressors supply air to the RHRSW control valves.

Reactor Building to suppression chamber vacuum breakers - (2) Alternate Nitrogen Supply System available to close valves and pressurize T-ring seals for Primary Containment integrity.

10.3.4.4 Inspection and Testing

The Instrument and Service Air Systems are in continuous operation and are observed and maintained during normal operations. In response to Generic Letter 88-14 “Instrument Air Supply Problems Affecting Safety-Related Equipment” (Reference 63), a commitment was made to add semi-annual tests for dewpoint, particulate and oil content to the Instrument Air Quality Program (Reference 64).

10.3.5 Domestic Water System

10.3.5.1 Design Basis

The Domestic Water System provides water for drinking and sanitary purposes. It also provides raw water for the Plant Makeup Demineralizer System and normal supply for the Seal Water System.

10.3.5.2 Description

The Domestic Water Supply is supplied from wells on site.

Shower and lavatory waste water and drains such as in the office building that do not contain radioactive material are directed to a sewage treatment system.

The Seal Water is supplied to various river water pumps and the overflow is discharged to the river.

See Section 10.3.3 for a discussion of the Plant Makeup Water Treatment System.

10.3.5.3 Performance Analysis

The Domestic Water System provides water for drinking, sanitation, the Seal Water System, and the Plant Makeup Water Treatment System.

10.3.6 Plant Equipment and Floor Drainage Systems

10.3.6.1 Design Basis

The design objective of the plant equipment and floor drainage is to collect and remove all waste liquids from their points of origin to a suitable treatment area in a controlled and safe manner. Water from radioactive drains is collected, treated, sampled, stored and/or analyzed prior to disposal to the environment in accordance with NRC regulations 10CFR20 and Appendix I to 10CFR50. Drain line penetrations through containment barriers are designed to maintain containment during normal operations and postulated design-basis accidents.

10.3.6.2 Description

10.3.6.2.1 General

The three basic drainage systems in the plant are:

a. Drywell Radioactive Drainage Systems.

b. Reactor and Turbine Building Radioactive Drainage Systems.

c. The Nonradioactive Equipment and Floor Drainage Systems.

The first two systems, as the names imply, handle wastes which are radioactive. The third system handles wastes originating in areas which are not radioactive. All radioactive drains are pumped to the radwaste system for treatment prior to reuse. Nonradioactive drains are drained to a normal drain sump or to the storm drain system.

10.3.6.2.2 Drywell Radioactive Drainage Systems

The drywell floor drain sump, the equipment drain sump, and the pumps associated with these sumps provide the system for removal of leakage from the drywell. Drainage into the drywell equipment drain sump is considered to be identified leakage. This leakage is piped from valve and pump seal leak-offs directly to the sump. The floor drain sump collects unidentified leakage which may come from unexpected leaks. Background flow into this sump is very low (less than one gpm). Steam leaks collect in this sump due to condensation in the drywell coolers or on the drywell walls. Measurement of the leakage rate is accomplished by measuring the interval between two level switches as the sump fills. At any time that the interval decreases over the previous minimum interval (indicating an increased leak rate) an alarm operates in the control room. A dial is provided for the operator to set the minimum interval timer to any time. After such a setting, the first refill interval becomes the standard to which subsequent intervals are compared.

Each sump is also provided with a level transmitter. Control board recorders provide continuous indication of leakage rates. Process computer alarms on high leakage rates provide rapid notification to operators of abnormal leakage.

Two pumps of 50 gpm capacity are provided in each sump. The live storage of each sump is about 180 gallons. Total sump capacity is about 250 gallons which allows sufficient working margin at low and high levels.

The operator can also determine the leakage over a period of time from an integrating flowmeter and recorder which indicate the sump flow. The operator can supplement this information by observing the drywell cooler cooling water temperature differential.

Besides the quantitative detection methods discussed above, the following provisions are made for determining the source of leakage. It should be noted that most of these methods are applicable only for localizing leakage sources and do not necessarily give a quantitative indication of the leak size.

Leak Detection Methods

a. Local air temperature monitors. (Located in each quadrant of the drywell).

b. Leak-off collection lines from reactor head flange seal, pump seals, valve stem leak-offs.

c. Air sampling.

d. Sump water sampling.

e. Loose parts monitoring system.

f. Portable radiation instruments.

g. Visual inspection during shutdown.

10.3.6.2.3 Reactor Building Radwaste Drainage System

10.3.6.2.3.1 Reactor Building Drains

Reactor building leakage is collected in two separate sumps.

One sump handles drainage from all equipment drains while the other collects floor drains.

Pumps are provided to transfer these wastes from the sumps (via their respective collection tank) to the waste collector tank in the liquid radwaste system. The reactor building equipment drain system begins with drains at all items of equipment, collects in branch lines, and discharges to the reactor building equipment drain sump for eventual transfer to the waste collector tank in the liquid radwaste system. Containment is maintained in transferring wastes from the sumps to the radwaste system by maintaining a minimum water level in the sump which seals the pump suction lines so that there is no continuous air path between the pump suction inlet and the containment. Drains from the reactor standby liquid control system are not collected in the radioactive waste system, but to a separate collecting tank, e.g., 55 gal. drum, for independent disposal. Valves on each side of a primary containment wall close upon a high pressure signal to prevent blowout of water seals.

10.3.6.2.3.2 Turbine Building Drains

The turbine building radioactive equipment drainage system drains all items of equipment that require draining, collects in branch lines, empties into main waste lines, and discharges into the equipment drain sump located below the basement level. A sump pump is provided to pump the discharge from the turbine building to the liquid radwaste system.

10.3.6.2.4 Nonradioactive Water Drainage System

The turbine building, administration building, radwaste building roof drains and some floor drains in the turbine building service areas are collected in branch lines, emptied to headers or main drain lines, and discharged to the plant storm drain system.

Nonradioactive drains located in the lower basement level of the turbine buildings are collected in branch lines which empty into a main drain line and discharge to the nonradioactive water normal drain sump. This waste is normally pumped into the plant intake bay. Sometimes, during outages, this waste is sent via temporary lines to the retention ponds located near the intake structure for treatment prior to release to the Mississippi River as allowed by the MNGP water discharge permit.

10.3.6.3 Performance Analysis

10.3.6.3.1 Drywell Radioactive Drainage System

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally determined behavior of cracks in pipes and on the ability to make up coolant system leakage in the event of loss of off-site a-c power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USNRC-sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study) (Reference 56). Analysis, utilizing the data obtained in this study, has shown that there is a high probability that a leaking crack can be detected before it grows to a dangerous or "critical" size. "Critical" size is considered to be the size that would result in self propagation at the stress level existing. Mechanically or thermally induced cyclic loading, stress corrosion cracking, earthquake and normal vibration stresses are considered in the determination of the critical crack size. The critical crack size results in water leakage of about 150 gpm.

Identified leakage (equipment drain sump) originates predominantly from pump seals and valve packing leakoffs. Background leakage is normally 1 to 3 gpm. It is estimated that a detection capability of 5 gpm is achievable. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than 10^{-5}. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin.

Measurement of the leak rate into the sump is considered to be of greater importance to the operator than the actual water level in the sump. As described above, the sump filling rate is measured by determining the time for the level to change a fixed amount. An increase in the filling rate causes an alarm in the control room. Gross level measurement is recorded in the control room and can be inferred by the condition of the various controls operated by the level switches such as pumps off, interval timer running, one pump running and two pumps running.

Final localization of leaks detected by the above system will, in all probability, require the presence of operators in the drywell; however, some localization can be accomplished by the instrumentation available as follows:

Local air temperature monitors are located in different quadrants of the drywell. Comparison of temperatures in the quadrants will infer sources of leakage depending on equipment and major pipe locations.

Leak-off collection lines from the recirculation pump seals and head seal are instrumented with flow switches which indicate in the control room when the leakage exceeds the switch set point. Leakage in the other leak-off lines cannot be remotely identified.

Water sampling can be used to determine whether leakage is coming from the reactor closed cooling water system since this system contains corrosion inhibitors which are not in the reactor coolant loop.

Air sampling can be used to determine radioactivity levels in the containment air which by inference will indicate whether the leak is from the reactor coolant system. Radiochemical analysis of either air or water samples can be used to establish whether a leak source is steam or reactor water.

Portable sonic and radiation instruments can aid visual inspection when the drywell becomes accessible. The loose parts monitor may also provide information on the source of leakage prior to entry.

10.3.6.3.2 Reactor Building Radwaste Drainage System

Reactor Building floor drain and equipment drain sumps are provided to collect and direct flow to the radwaste processing system.

When the reactor building equipment sump level increases due to accumulated leakage, the 50 gpm sump pump will start and discharge its contents to the reactor building equipment drain tank. Overfilling the equipment sump will result in a high level alarm and overflow to the adjacent floor drain sump. This latter sump has dual 100 gpm pumps which discharge to the reactor building floor drain tank. Overfilling of the floor drain sump also causes a high level alarm.

Both the reactor building equipment drain tank and floor drain tank have redundant 100 gpm capacity pumps that discharge to separate collector tanks. These collector tanks are integral components of the liquid radwaste processing system.

Engineered safeguard (RHR and Core Spray) pumps are located in the Southeast and Southwest corner rooms. Equipment and floor drains from each room are directed to the reactor building equipment and floor drain sumps, respectively. In case of an internal flooding event affecting the RHR and Core Spray corner rooms, two 100 gpm sump pumps, each powered from the same essential switchgear division as the equipment in that room, are installed in each room at floor level. The pumps discharge to the torus room at approximately the same elevation as normal suppression pool water level, 909′ - 3.5″. In addition to the pumps, each room is equipped with a flooding alarm to inform control room personnel of increasing water levels in the rooms.

10.3.7 Plant Process Sampling System

10.3.7.1 Design Basis

The design objectives of the plant process sampling are to:

a. Monitor the operation of plant equipment.

b. Provide information via local analyzers and data acquisition systems for making operational decisions.

c. Monitor plant radioactive effluent.

In all cases, the plant process sampling system will:

a. Obtain representative samples which provide sample flow to in-line instrumentation. Representative samples are available for laboratory analysis.

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

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

10.3.7.2 Description

Samples are taken from the following streams at the indicated location in Table 10.3-1.

10.3.7.3 Performance Analysis

The quality and quantity of samples taken from the process systems provides sufficient information to the reactor operators and plant staff to enable them to make operational decisions related to plant chemistry and materials effecting plant performance.

10.3.7.4 Inspection and Testing

All the plant samples are taken on frequent enough intervals that further testing of the system is not required.

10.3.8 Plant Communication System

10.3.8.1 Design Basis

Internal and external communication will be established by telephones and loudspeakers. These systems are designed to provide convenient, effective operational communications between various plant buildings and locations.

10.3.8.2 Description

The plant communication system utilizes hand sets and loudspeakers to provide for paging and communications in all important areas. The communications system can be broken down into the following three subsystems:

a. Paging and Evacuation Alarm System

This paging system is a two-channel system permitting simultaneous “page” and “party line” channel usage. The paging system includes a siren tone generator which can inject emergency signals into the system. In addition, a motor driven siren provided on the roof of the reactor building covers the outdoor areas. Both the siren tone generator and outdoor siren can be interrupted from the control room to allow announcements to be made over the Public Address system. The paging system is normally powered from the normal plant AC system with backup power supplied from the station battery system. Pagers are available for supervisory personnel to contact plant personnel when they are in sections of the plant site which are not normally covered by the paging and telephone systems.

b. Sound Powered Telephone System

Portable sound powered handsets are provided for a maintenance system. Several phone jacks are located around the plant to provide reasonably uniform coverage with emphasis on specific areas of potential usefulness.

c. Intra-Plant Communication System

The intra-plant communication system is a private automatic exchange system.

d. Emergency Communications

Due to loud background noise and the potential for fire damage the systems described in a., b., and c. are not always effective for fire fighting operation. An AC-powered radio receiver/transmitter console is provided in the control room. Handheld radio units stored in the fire brigade room can provide radio communications between the control room personnel and the fire brigade. The handheld units operate independently from the control room console unit and are not disabled by any failures of the console.

10.3.8.3 Performance Analysis

The plant communication system has adequate flexibility to keep plant personnel informed of the plant operational status at all times.

10.3.8.4 Inspection and Testing

The continual usage of the plant communications system assures the knowledge of the communication system's operability status, therefore, no further testing is necessary. Fire brigade hand sets are periodically checked to ensure their operability.

10.3.9 Plant Lighting System

10.3.9.1 Design Basis

Normal and emergency lighting is provided for the plant with power utilized from normal sources with backup from the standby diesel power system or the plant battery system. Power is available for critical areas in the building. Emergency lighting is provided to support operator actions in accordance with the Monticello compliance for 10CFR50 Appendix R and CFR50.63 Station Blackout (SBO).

10.3.9.2 Description

Normal lighting equipment is energized from the primary station auxiliary transformer, 2R. Portions of the normal lighting are arranged for backup supply from the standby diesel generator system. In addition to the normal plant lighting, two emergency lighting systems powered by batteries are provided. The first system consists of lights powered by individual batteries, located in various areas throughout the site, that provide illumination for stairways and walkways. The second system, powered from a station battery, provides emergency lighting in the control room, battery rooms, cable spreading room, diesel generator building, and the walkways and stairs between the diesel generator building and the control room.

Two portable sealed beam battery powered hand lights are provided in the control room emergency locker with an additional one stored in the shift supervisor's locker.

Fixed and portable lighting units are provided as necessary to support fire fighting and post-fire safe shutdown activities. Self-contained batteries power these lighting units providing a minimum 8-hour capacity.

10.3.9.3 Performance Analysis

Essentially all plant areas are lighted to an adequate degree from that portion of the lighting system which has diesel power backup. The automatically energized DC system provides immediate emergency lighting for the control room, areas required for maintenance of standby AC equipment and access routes to and from main control room.

10.3.9.4 Inspection and Testing

All critical lighting systems are inspected and tested periodically to assure the proper operation of all automatic transfer switches.

10.3.10 Post-Accident Sampling System

10.3.10.1 Design Basis

The Post LOCA Accident Sampling System (PASS) can obtain highly radioactive liquid and gas samples for radiological analysis. The liquid samples are representative of liquids within the reactor coolant and Residual Heat Removal (RHR) System. The gas samples are representative of the atmosphere within the torus and Primary Containment. (Reference 30)

Requirements for the PAS system were removed from Technical Specifications per License Amendment 136 (Reference 65). As part of the License Amendment approval, Monticello made the following commitments. (Reference 66):

a. Develop contingency plans for obtaining and analyzing highly radioactive samples from the Reactor Coolant System (RCS), suppression pool and containment atmosphere. The contingency plans will be contained in the plant technical procedures.

b. The capability for classifying fuel damage events at the Alert level threshold will be established for the Monticello Nuclear Generating Plant at radioactivity levels of 300 μCi/gm dose equivalent iodine.

c. Develop an I-131 site survey detection capability for the Monticello Nuclear Generating Plant, including an ability to assess radioactive iodines released to off-site environs using effluent monitoring systems or portable sampling equipment. The capability for monitoring iodines is maintained in the emergency plan implementing procedures.

The samples for the first commitment can be obtained via local grab samples or the PAS system depending on dose rates.

10.3.10.2 Description

The PASS sampling facilities consist of a liquid and gas sample station, a piping station, and a control panel. The system is designed to minimize exposure by minimizing the required sample sizes, to optimize the weight of shielded sample containers in order to facilitate movement through potentially high level radiation areas and to provide adequate shielding at the sample station and in the laboratory.

10.3.10.2.1 Gas Samples

Provision has been made to obtain gas samples from both the drywell and suppression pool atmospheres.

10.3.10.2.2 Liquid Samples

A sample line is provided to obtain reactor coolant samples from two points in the jet pump pressure instrument system when the reactor is at pressure. A single sample line is also connected to both loops in the RHR system.

10.3.10.3 Performance Evaluation

Tie-ins to the RHR system have been seismically evaluated up to the first isolation valve. The effect of thermal movement of RHR piping on the sample piping has been evaluated and all stresses and loads are within code allowables. The effect on the original dynamic seismic analysis of the addition of the sample taps to the jet pump sample lines has been evaluated. Loads and stresses are within code allowables. The piping system is designed and constructed in accordance with ANSI B31.1, 1977 Edition (Reference 57) with Winter 1978 Addenda.

10.3.11 Control Room Breathing Air System

10.3.11.1 Design Basis

In the event of a toxic gas spill, a Control Room Breathing Air System can be used to supply uncontaminated air to Control Room personnel. This system is sized to provide breathing air to eight individuals for three hours. Cylinders of compressed air can be changed out to extend the time that personnel can rely on this system. Low pressure alarms provide notification that cylinder replacement should be initiated.

10.3.11.2 Description

The Control Room Breathing Air Supply utilizes high pressure air cylinders located in a seismically designed rack in the Heating and Ventilating equipment room adjacent to the Control Room to supply breathing air to two distribution trains located within the Control Room. Each train contains an isolation valve, a pressure regulator, a pressure gauge, low pressure alarm, and a relief valve. Breathing air is delivered through each train to a hose quick disconnect and a quick disconnect manifold for use with the dual purpose Self Contained Breathing Apparatus (SCBA) worn by Control Room personnel. The dual purpose SCBAs enable personnel to connect to this low pressure air supply system to extend the time a SCBA may be worn.

10.3.11.3 Inspection and Testing

Periodic surveillance is performed to maintain the system in standby readiness.

Table 10.3-1 Plant Process Sampling System
Locations of Sampling Points
(Page 1 of 3)

Description	Locations	Purpose
Reactor Primary Coolant Water	Recirculation System Pipe	a. Monitor Reactor Water b. Provide Information for IGSCC
On-Line Noble Chem Mitigation Monitoring System	Recirculation System Pipe	a. Monitor Reactor Water b. Provide Information for IGSCC
	Reactor Cleanup Demineralizer System and Fuel Pool Cooling and Demineralizer System (each)	
RWCU Demineralizer/ Filter Influent	Filter Inlet Pipe	Monitor Reactor Water Quality
Demineralizer/ Filter Effluent	Filter Outlet Pipe	Filter Efficiency Filter
	Nuclear Steam Supply System	
Main Steam Line Sample	Main Steam Line	Sample Line Capped at Sample Station C213
Suppression Pool	Suppressing Pool Re-circulating Pipe	Monitor Corrosion and Activity
Standby Liquid Control System	Recirculation Pipe	Borate Concentration
Reactor Cooling Water System	Heat Exchanger	Check Corrosion Inhi-bitor Concentration
	Condensate System	
Condensate	Condensate Pump Discharge	Condensate Quality and Tube Leaks
Condensate Deminer-alizer Inlet	Demineralize Inlet Pipe	Condensate Quality and Tube Leaks
Condensate Deminer-alizer Effluent	Demineralize Outlet Pipe (Combined and Individual)	Treated Condensate Quality and Demin Perfor-mance

Table 10.3-1 Plant Process Sampling System
Locations of Sampling Points
(Page 2 of 3)

Description	Locations	Purpose
	Reactor Feedwater System	
Feedwater Heater	Halfway in Heater Train Feed Piping	Sample Line Capped at Sample Station C213
Reactor Feedwater	After Last Heater	Water Analyses System
	Reactor Building Closed Cooling Water System	
Cooling Water Sample	Outlet of Each Major Heat Exchanger	Determine Locations of Heat Exchanger Leaks
Cooling Water Sample	Pump Discharge	Check Corrosion Inhibitor Concentration
	Main Condenser Circulating Water System	
Influent	Closed Cooling Water Heat Exchanger inlet or Circulating Water Pump Discharge	Determine Background
Effluent	Discharge Canal	Monitor Plant Activity Release
	Liquid Radwaste System	
Waste Surge Tank	Pump Discharge	Process Data
Waste Collector Tank	Pump Discharge	Process Data
Floor Drain Collector tank	Pump Discharge	Process Data
Laundry Drain Tanks	Pump Discharge	Process Data
Waste Sample Tank	Pump Discharge	Discharge Suitability
Floor Drain Sample Tank	Pump Discharge	Discharge Suitability
Radwaste Filter/ Demineralizer	Outlet Pipe	Filter Efficiency

Table 10.3-1 Plant Process Sampling System
Locations of Sampling Points
(Page 3 of 3)

Description	Locations	Purpose
	Plant Makeup Water Treatment System	
Mixed-Bed Demineralizer Effluent	Outlet Pipe	Demineralizer Efficiency
Raw Water Sample	Inlet Pipe	Process Data
Filter Effluent	Outlet Pipe	Process Data
Demineralizer Water Storage Tank	Pump Discharge	Water Quality
Condensate Storage Tank	Outlet Pipe	Water Quality
	Special Samples	
Precoat Sample	Precoat Transfer Line	Test Resin Mixing
	Plant Off-gas System	
Air Ejector Samples	After Air Ejectors	Activity Release, H_2, 0_2 and Air Leakage
Off-Gas Filter Samples	Inlet and Outlet	Determine Filter Efficiency
Stack Sample	Main Stack	Particle and Iodine Release
Ventilation	Fan Discharge	Activity Release

10.4 Plant Cooling System

10.4.1 Plant Service Water System

10.4.1.1 Design Basis

The Plant Service Water System supplies water to the Reactor and Turbine Buildings for cooling. Interties are provided to permit the use of this water as a standby coolant supply and an additional source of cooling water for the ECCS room air coolers and the ECCS pump motors. When used as a standby coolant supply this system provides an inexhaustible supply of screened river water directly to the main condenser hotwell for injection into the reactor system by the Feedwater System.

10.4.1.2 Description

The Plant Service Water System consists of three one-half capacity vertical wet pit service water pumps located in the Intake Structure. An automatic self-cleaning strainer is provided in the discharge line to remove suspended matter from the river water. This system supplies strained water to the Reactor and Turbine Building to meet normal, startup, and shutdown requirements. Drawings NH-36041 and NH-36041-2, Section 15, shows the plant Service Water System P&ID.

The Mississippi River serves as the ultimate heat sink for the plant. The river has sufficient capacity to meet the flow requirements of the safety related service water systems at a temperature of 90°F or less (References 73 and 79). Service Water temperature for containment analysis is discussed in Section 5.2.3.2.4. 01101248

The plant Service Water System supplies cooling water for the plant main generator, Reactor and Turbine Building air conditioning units, turbine lube oil coolers, Reactor Building Closed Cooling Water System heat exchangers, various plant motor-generator sets, reactor Feedwater System pumps and the Condensate Pump Motor bearings.

Service water pumps provide normal plant startup and shutdown requirements. The pumps are controlled so that if the operating pumps cannot maintain the required system pressure, the standby pump will start automatically.

The standby coolant supply system consists of an 18-inch interconnecting pipe between the plant Service Water System and the condenser hotwell, associated valves, and instrumentation. This system provides capability of adding river water to the hotwell. Double valves are used in the interconnecting piping to provide the capability for testing to assure that river water does not leak into the condensate.

The valves for the standby coolant system are manually operated in the Turbine Building. Both valves are provided with position switches which cause an alarm in the Control Room if either valve is moved from the fully closed position.

Two branch lines from the plant Service Water (SW) System tie into the Emergency Filtration- Emergency Service Water (EFT-ESW) System to provide a backup water supply for the control room air conditioning units as well as for the ECCS system room coolers and the ECCS pump motors. Gate valves are installed in the EFT-ESW/SW interface piping to allow leak testing of the EFT-ESW/SW cross-tie check valves without removing the EFT-ESW system from service. Additional branch lines from the plant Service Water System provide backup water supply to each Emergency Diesel Generator Emergency Service Water (EDG-ESW) line. The EDG-ESW/SW interface is by a normally closed gate valve, one in each branch.

The intake bays are treated with chlorine/bromine on a daily basis to control macrobiological fouling. (References 33 and 34). Direct injection of sodium hypochlorite and sodium bromide into the SW pump discharge piping upstream of the automatic strainer is also used on a daily basis to control system fouling.

10.4.1.3 Performance Analysis

The plant Service Water System is designed to supply water to Reactor Building, EFT Building and Turbine Building equipment during normal operation. The three pumps supplied provide 50% excess capacity so that normal operation requires only two pumps. The third pump is available for standby operation and starts to maintain system pressure. All three service water pumps are typically operated to provide increased service water capacity in the summer.

The 18-inch intertie to the condenser provides a large capacity water source for injection into the reactor by the Feedwater System. This system could be used to flood the entire containment vessel to a water level above the reactor core and is an ultimate assurance that core cooling can be maintained. The flooding feature provides a means to achieve post accident recovery under conditions that prevent normal filling of the reactor vessel. This flooding feature is dependent on Feedwater and service water pumps not normally powered from diesel generator power sources. However, two of the service water pumps could be electrically cross-tied to diesel generator power sources.

The plant Service Water System is not required during or immediately subsequent to a design basis accident and is therefore not safety related.

10.4.1.4 Inspection and Testing

Since the Service Water System is operative at all times, no periodic testing is required. Leak checking of the valves in the service water to condenser coolant intertie is performed by routinely checking the normally open telltale drain valve between the two main valves.

The intake bays are visually inspected each refueling outage for excessive biofouling, sediment and corrosion. Water and substrate samples are tested on an annual basis for Asiatic Clams, Zebra Mussels, and overly prolific species. (References 33 and 34).

10.4.2 Residual Heat Removal System Service Water System

10.4.2.1 Design Basis

The design objective of the Residual Heat Removal Service Water System is to remove the heat rejected by the residual heat removal system during normal shutdown and accident operations. In addition this system provides a source of water for the RHR-RHR Service Water intertie.

An analysis has been performed to verify the safety related portions of the system are not vulnerable to a single active failure. (References 33, 34, and 35).

10.4.2.2 Description

The RHR-Service Water System consists of four pumps grouped into two sets of two pumps. Each pump is capable of providing a strained water supply at the design flow rate to its associated Residual Heat Removal System heat exchanger, and, consequently, can absorb the heat load for any design basis loss-of-coolant accident situation. Each set of pumps is individually piped to its associated heat exchanger, resulting in two segregated systems.

Both Residual Heat Removal Service Water loops contain a single inline basket strainer and an unstrained bypass line. For both loops, the bypass line may be placed in operation by opening the valve in the bypass line, allowing flow in parallel with flow through the strainer loop. This provides screened river water, supplied by the traveling screen, to the Residual Heat Removal System heat exchanger, as described in Section 10.4.2.3.

A 1″ crosstie line is installed between the two systems. This allows both systems to be pressurized by one RHRSW pump. The RHR Service Water System P&ID is shown in Section 15 Drawing NH-36664. Other components of the RHR System are shown on the P&ID of Section 15 Drawings NH-36246 and NH-36247.

The RHR-RHR Service Water intertie is shown on Drawings NH-36664 and NH-36247. It consists of two manually operated valves and a check valve in series between an RHR Service Water line and an RHR heat exchanger discharge line. The manual valves are located in the Turbine Building and are accessible during reactor operating and emergency conditions.

Since actuation of the system is by operator action the instrumentation and controls required is minimized. The required initiation time is discussed in Section 6.2.3.2.2. The instrumentation consists of flow measurements to the RHR heat exchangers, differential pressure control between the process fluid and cooling water, and a scintillation type radiation detector on the outlet water.

10.4.2.3 Performance Analysis

Either of the two physically separated portions (loops) of the Residual Heat Removal Service Water System is capable of providing a heat sink for the design basis loss-of-coolant accident situation and the reactor shutdown cooling system.

An unstrained bypass line is available to place into operation, when high differential pressure indication across the basket strainer is received. During the limited periods of time when the bypass line may be in operation, the mesh size of the traveling screens adequately prevents the entrance of particles large enough to clog the residual heat removal heat exchanger tubes (See Engineering Evaluation 24955, Reference 71). The strainer provides the added benefit of removing debris that may get past the screens, which would otherwise be transported into and through the Residual Heat Removal heat exchanger tubes. The Residual Heat Removal Service Water system safety function, to supply water to the Residual Heat Removal heat exchangers for component cooling, is maintained when operating in bypass.

In the event of loss of normal AC power, power for both sets of pumps can be supplied from the Emergency Diesel Generators.

Provisions have been made to permit injection of screened river water into the core through the RHR-RHR Service Water System intertie. It was assumed that this system would be used after the core had been reflooded by the Core Spray and/or the LPCI Systems. The 3500 gpm capacity provides excess water over that required to make up for leakage from the core shroud and boil off due to decay heat. Valves in the system are manually operated and the pump power can be supplied from an EDG. The system is therefore independent of offsite power.

The instrumentation described in subsection 10.4.2.2 falls into two categories - that required for system evaluation or test operation and that required when the RHR System is operated in the Shutdown Cooling mode. The flow measurement and temperature measurements are primarily used to evaluate system and equipment performance and consequently do not perform any safety related function.

When an RHRSW pump is operating, differential pressure control is used to maintain a higher pressure on the service water side of the RHR heat exchanger than the process fluid pressure. Thus, if there were a leak in the RHR heat exchanger, the reactor water would not be able to flow into the RHR Service Water Systems. When the RHR System is used in the shutdown cooling mode, radiation release limits of 10CFR20 would apply. The RHR Service Water flow to the discharge canal is monitored by a scintillation type monitor which would alarm prior to reaching 10CFR20 limits.

The differential pressure control valve solenoid is interlocked with the RHR service water pumps such that the solenoid is energized only when a pump is in service. These valves are closed during normal plant operation (RHR Service Water System not in service). In the event of a DBA (with loss of off-site power) valves can be opened when the operator initiates the containment cooling mode of RHR operation after the accident is contained, since the system is supplied with a source of air via the RHR auxiliary air compressors. Consequently, power failure does not prevent the operator from establishing coolant flow to the RHR heat exchangers.

The radiation monitor described above receives its power supply from the plant’s 24Vdc batteries and consequently is in service following the DBA.

10.4.2.4 Inspection and Testing

The RHR Service Water System pumps are tested periodically on a scheduled basis and in addition are used during refueling and normal shutdown. The leak tightness of the valves in the intertie is verified by routinely checking the normally open telltale valve between the two main valves.

Full flow testing of the RHR Service Water System is performed on a quarterly basis to verify no degradation due to flow blockage.

System heat exchangers are periodically tested for overall heat transfer capability. 01453510
A periodic non-destructive examination program was established to inspect safety-related piping and heat exchangers at known or suspected high corrosion, biofouling, or silt buildup areas. This program is supplemented by visual inspections of opened piping and heat exchangers whenever possible. (References 33 and 34).

10.4.3 Reactor Building Closed Cooling Water System

10.4.3.1 Design Basis

The Reactor Building Closed Cooling Water System is designed to remove heat from the reactor auxiliary systems equipment and their accessories.

10.4.3.2 Description

The Reactor Building Closed Cooling Water System P&ID is shown in Section 15 Drawings NH-36042 and NH-36042-2. The Reactor Building Closed Cooling Water System consists of a cooling loop containing two full capacity pumps and three heat exchangers. Any possible leakage from the reactor auxiliary systems equipment listed below is to the reactor building closed cooling water system loop where it is confined or isolated (use of manually controlled isolation valves acceptable per Reference 31). The Reactor Building Closed Cooling Water System loops are monitored continuously for radioactivity by a process radiation monitoring subsystem. The Reactor Building Closed Cooling Water System cools the following equipment:

Nonregenerative Heat Exchanger.

Reactor Coolant Recirculation System Pump Coolers and Pump Seal.

Reactor Building and Turbine Building Process System Sample Coolers.

Fuel Pool Cooling and Demineralizer System Coolers.

Primary Containment Drywell Coolers.

Residual Heat Removal System Pump Seal Coolers.

Control Rod Drive Feed Pump Coolers.

PAS System Coolers.

Drywell Equipment Drain Sump Heat Exchanger.

The Reactor Building Closed Cooling Water System is maintained at less than 95°F by heat rejection to loop heat exchangers which are cooled by the plant service water system.

10.4.3.3 Performance Analysis

The Reactor Building Closed Cooling Water System is designed for the heat load of the equipment under normal operation startup and shutdown conditions. The plant service water system is capable of supplying the required water to the Reactor Building Closed Cooling Water System to maintain the reactor auxiliary systems equipment and their accessories at a design operating temperature.

The Reactor Building Closed Cooling Water System is not required during, or immediately subsequent to, a design basis accident and is therefore not safety related.

10.4.3.4 Inspection and Testing

Since the system is operative at all times, no periodic testing is required.

10.4.4 Emergency Service Water System

10.4.4.1 Design Basis

The performance objective of the Emergency Service Water System is to remove the heat rejected by the equipment which must operate under accident conditions.

An analysis has been performed to verify the safety-related portions of the system are not vulnerable to a single active failure (References 33, 34 and 35).

10.4.4.2 Description

The following two completely segregated systems make up the Emergency Service Water System:

a. The Emergency Filtration-Emergency Service Water System (EFT-ESW, plant designator FSW) consists of two pumps which are capable of providing a strained source of cooling water to several Emergency Core Cooling System (ECCS) electric motors and compartment coolers which require cooling during accident situations. The Emergency Service Water System also supplies cooling water to the condensers of the air conditioning units for the main Control Room air conditioning system in the event of a loss of offsite power. This system is shown in RHR Service Water & Emergency Service Water Systems P&ID, Section 15 Drawing NH-36664, Service Water Systems and Make-Up-Intake Structure P&ID, Section 15 Drawing NH-36665, and Service Water System P&ID, Section 15 Drawings NH-36041 and NH-36041-2.

b. The Emergency Diesel Generator-Emergency Service Water System (EDG-ESW, plant designator ESW) consists of an additional two pumps that are capable of providing a strained source of cooling water to the Emergency Diesel Generators. This system is shown on Section 15 Drawings NH-36664 and NH-36665.

10.4.4.3 Performance Analysis

Each pump is capable of supplying sufficient water for cooling to the equipment required to handle a design basis loss-of-coolant accident including cooling water to the Main Control Room (MCR) HVAC units. See Section 15 Drawings NH-36664 and NH-36665 for pump piping arrangements and capacities. Each pump is individually piped to its associated heat exchange equipment, resulting in completely segregated systems. The EFT-ESW pumps start automatically upon an Emergency Diesel Generator (EDG) breaker closure or an essential bus transfer to the emergency offsite power source (1AR). The EDG-ESW pumps start automatically upon an automatic or manual EDG start and sufficient diesel speed. Starters are located in MCC B34, B43 and B44. Control switches in the ASDS panel allow for remote operation of the Division II EFT-ESW pump and the Division II EDG-ESW pump in the event of a control room fire.

Alarms are provided to warn operators upon loss of emergency service water flow to the Emergency Diesel Generators and to ECCS equipment. Low pressure alarms provide operator warning that line pressure is not being maintained.

In the event of loss of normal AC power, power for all pumps is supplied by the Emergency Diesel Generators. Design of the diesel generation system is discussed in Section 8.4.

10.4.4.4 Inspection and Testing

Pumps are tested periodically on a scheduled basis. The Emergency Diesel Generator - Emergency Service Water System cooling water heat exchangers are tested to verify heat transfer capability on a periodic basis. A non-destructive examination program has been established to inspect safety-related piping and heat exchangers at suspected high corrosion, biofouling, or silt buildup areas. This program is supplemented by visual inspections of opened piping and heat exchangers whenever possible (References 33 and 34). Emergency Filtration-Emergency Service Water lines to the motor coolers in the RHR rooms are flushed quarterly and backwashed, if necessary, to meet the requirements of Generic Letter 89-13 (Reference 58).

10.5 References

1. NSP (L O Mayer) letter to the NRC (V Stello), "Fire Hazard Analysis", dated March 11, 1977.

2. NSP (L O Mayer) letter to the NRC (V Stello), "Completion of Fire Protection Review", dated July 5, 1977.

3. NSP (L O Mayer) letter to the NRC, "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance", dated May 18, 1978."

4. NSP (L O Mayer) letter to the NRC, "NRC Staff Evaluation of Fire Protection Program", dated November 7, 1978.

5. NSP (L O Mayer) letter to the NRC, "NRC Staff Evaluation of Fire Protection Program", dated January 4, 1979.

6. NRC (T A Ippolito) letter to NSP (L O Mayer), "Amendment No. 41 to Provisional Operating License No. DRP-22", dated August 29, 1979.

7. NRC (D G Eisenhut) letter to all Power Reactor Licensees with Plants Licensed Prior to January 1, 1979, dated November 24, 1980.

8. NRC (T A Ippolito) letter to NSP (L O Mayer) "Supplement 1, Fire Protection Safety Evaluation Report", dated February 12, 1981.

9. Bechtel (C B Hogg) letter to NSP (D Antony) "RCIC Room Heatup", dated February 23, 1979.

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FIGURES

Figure 10.2-1 Reactor Refueling System - Pictorial

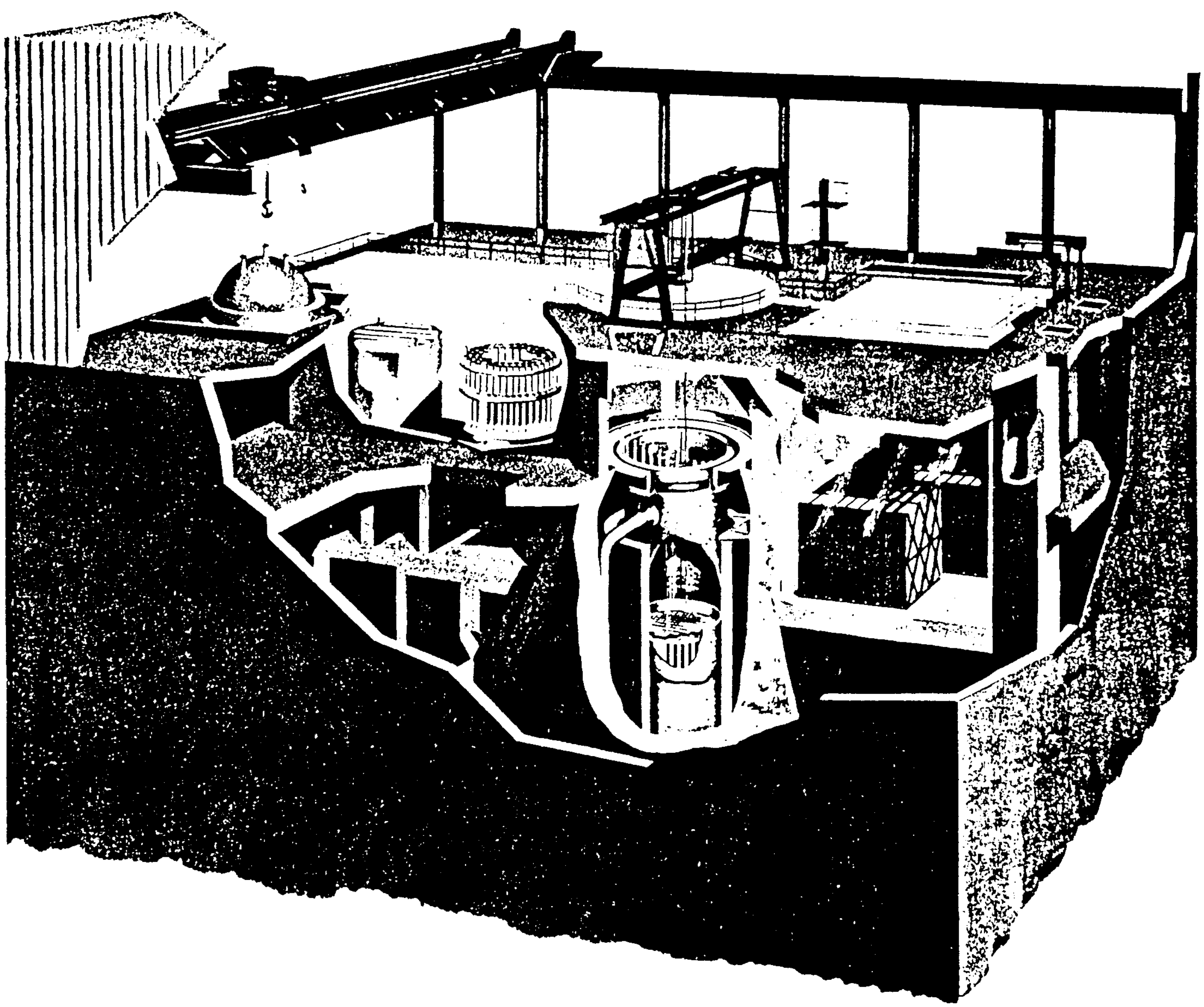

Figure 10.2-6 Shutdown Cooling Mode - RHR System Simplified P&ID

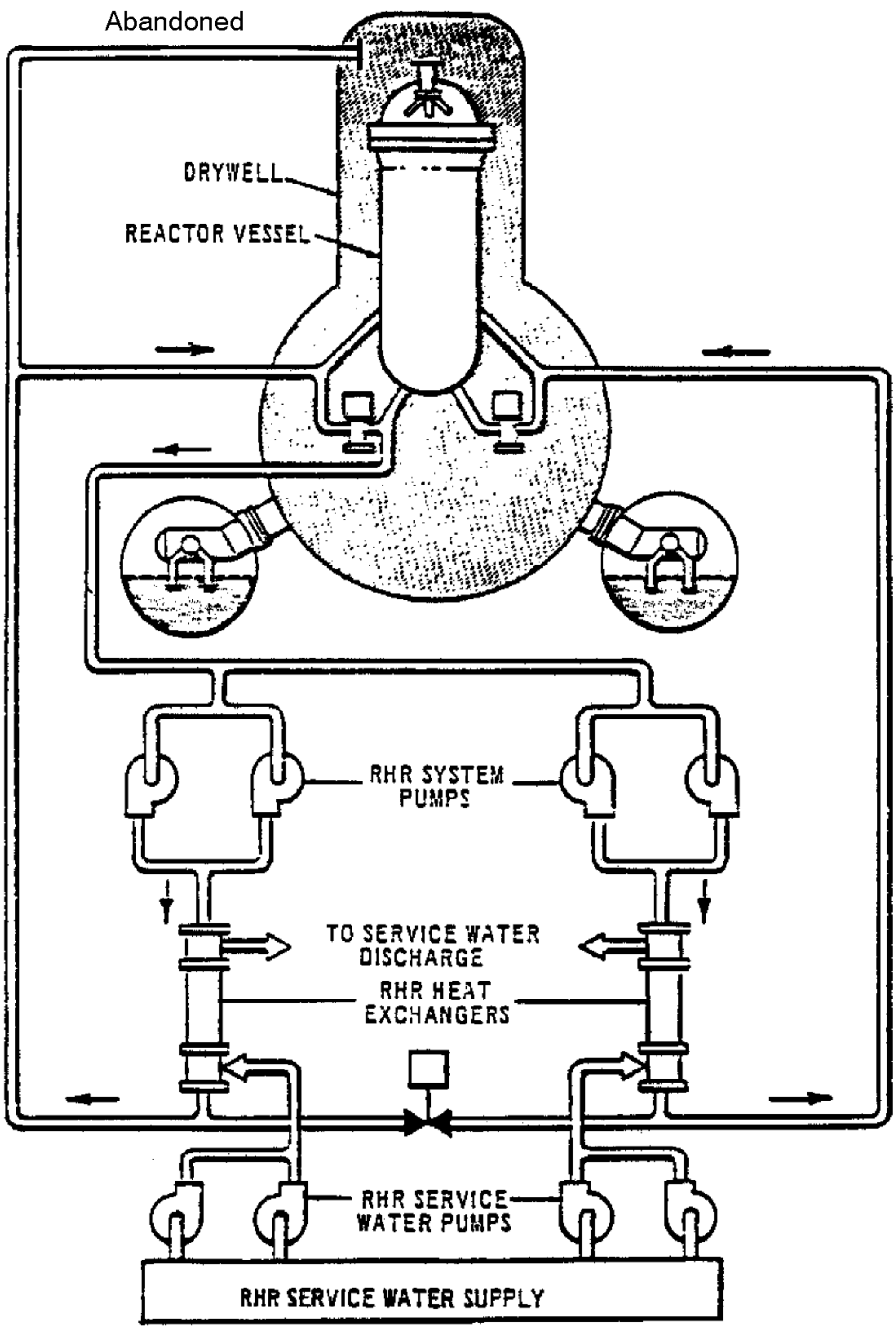

Figure 10.3-7 Principal Safe Shutdown Systems