9 AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

9.1.1 NEW FUEL STORAGE

New fuel is delivered by truck to the site in NRC-Department of Transportation-approved containers. The assemblies are removed, inspected, and transferred to the new fuel storage racks using the auxiliary building crane (see Figure 9.1-1). The storage location on the operating level of the auxiliary building facilitates the unloading of trucks and the transfer of the fuel assemblies. The Seismic Category I storage vault contains specially constructed racks which ensure a minimum 20-in. center-to-center spacing of the new fuel assemblies. This spacing ensures a K_{EFF} less than 0.95 for the accidental full water density flooding scenario and less than 0.98 for the accidental low water density (optimum moderation) flooding scenario. The use of Westinghouse 422+ Vantage Fuel Assemblies satisfies these K_{EFF} requirements (*Reference 67* and *Reference 68*). The storage area is located above grade to help prevent this from occurring.

The new fuel storage area is configured to store 12 fuel assemblies. The fuel storage area is isolated from potential contamination resulting from work performed in the auxiliary building maintenance shop, or from normal activities on the auxiliary building operating floor.

The design of the new fuel storage racks is in compliance with 10 CFR 50.68, Criticality accident requirements.

9.1.2 SPENT FUEL STORAGE

The original spent fuel storage racks provided capacity for the storage of 210 fuel assemblies. In 1976, the NRC approved the replacement of the original racks with higher density flux trap type racks (*References 1* and *2*). This expanded the storage capability from 210 to 595 fuel assemblies.

In 1984, the NRC approved the conversion of six flux trap type racks to high-density fixed poison type racks (*References 3* and *4*). This further expanded the storage capacity from 595 to 1016 fuel assemblies. At this point, the spent fuel pool (SFP) was divided into two regions. Region 1 comprised three flux trap type racks to accommodate a full core off-load. Region 2 consisted of six high-density fixed poison (Boraflex) type racks for the storage of 840 fuel assemblies that satisfied minimum burnup criteria and had cooled for a minimum of 60 days.

In 1998, the NRC approved re-racking the spent fuel pool (*Reference 38*). This re-rack effort, to be done in two phases, reconfigures the pool to accommodate a net increase of 353 locations. This is accomplished by retaining the six existing high-density region 2 racks (840 minus 12 for attachment of new racks = 828 locations) and installing new borated stainless steel (BSS) racks with up to 541 additional storage locations for a total of 1369 storage locations after completion of both phases.

After completion of phase 1 of the re-rack in November 1998, the pool has three types of racks in two regions. Region 1 contains new high-density flux-trap design BSS racks designated as type 3 for fresh and spent fuel. Region 2 contains the existing Boraflex racks designated as type 1 and new high-density BSS racks designated as type 2. With the completion of phase 1, the pool contains 1321 storage locations. Figure 9.1-3 shows the phase 1 re-racked configuration.

In addition to intact fuel assemblies, consolidated fuel canisters can also be stored in region 1 and region 2 of the pool. In 1985, the NRC approved the storage of consolidated fuel in the spent fuel pool (*Reference 5*). This process involves placing spent fuel containing, at most, all the rods from two standard spent fuel assemblies, which have decayed at least 5 years, into one canister. The canisters are designed to hold 358 fuel rods and can be placed in either region 1 or region 2 rack locations. The canisters are fabricated from stainless steel.

Prior to Plant Uprate the number of fuel rods contained in the intact fuel assemblies and/or consolidated rod storage canisters was limited to no more than the number of rods contained in 1879 fuel assemblies (179 fuel rods per assembly x 1879 assemblies = 336,341 fuel rods). The Technical Specifications limited storage at that time to 1879 fuel assemblies. As part of plant uprate in 2006, the maximum number of fuel assemblies that could be stored in the spent fuel pool was limited to 1321, which is consistent with the regulatory requirements imposed by Reference 68.

The fuel assembly grids, guide tubes, upper tie plates, and lower tie plates that remain after removal of the fuel rods are crushed and stored in a waste canister. The waste canister will maintain the fuel assembly non-fuel-bearing components in a physically stable configuration such that under all postulated conditions there will be no damage to the stored spent fuel in the spent fuel pool (SFP). The waste canister will be stored in the spent fuel storage pool until ultimate disposal.

The spent fuel pool inventory consists of intact fuel assemblies and other components (both fuel bearing and non-fuel bearing, including consolidated fuel canisters, consolidated hardware canisters, a failed fuel rod storage basket, dummy fuel assemblies, a dummy canister, trash baskets, an irradiated sample basket, and a coupon tree). In addition, one cell in the type 1 racks is capped.

The design of the spent fuel storage racks is in compliance with 10 CFR 50.68, Criticality accident requirements.

9.1.2.1 Design Criteria

<u>9.1.2.1.1 General</u>

The original design was based on the General Design Criteria (GDC) included in the Atomic Industrial Forum (AIF) version of proposed criteria issued by the AEC for comment on July 10, 1967. These criteria (AIF-GDC 66, 67, 68, and 69) are discussed in Section 3.1.1. Criteria for the design and performance of the current spent fuel storage system are defined by AIF-GDC 62, ANSI/ANS Standard 57.2-1983 and Regulatory Guide 1.13. The spent fuel racks satisfy these criteria as described below. In addition, the spent fuel rack design complies with the "Staff Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, as modified January 18, 1979; and section 9.1.2 of the Standard Review Plan dated July 1981 (*Reference 52*).

9.1.2.1.2 Effective Multiplication Factor

The design of spent fuel storage racks and transfer equipment shall be such that the effective multiplication factor will not exceed 0.95 with new fuel of the highest anticipated enrichment in place assuming limited credit for soluble boron. Credit may be taken for the inherent neutron absorbing effect of materials of construction or, if the requirements are met, for added nuclear poisons or for fuel assembly burnup.

The effective multiplication factor is less than or equal to 0.95, including biases and uncertainties, provided the combination of assembly average burnup and initial U-235 enrichment satisfies the requirements of Technical Specification LCO 3.7.13. Assemblies are stored in regions 1 and 2 locations based on their initial enrichments and minimum burnups as specified in LCO 3.7.13. Credit is taken in the LCO calculations for the presence of borated stainless steel in the type 2 and type 3 racks.

9.1.2.1.3 Protection Against Damage

Fuel handling system facilities shall be designed to prevent damage to fuel assemblies while in storage or during transport from one location to another.

Each fuel assembly or consolidated fuel canister is stored in a stainless steel box, which physically separates that fuel assembly from all other fuel assemblies. The stainless steel box is strong enough to prevent damage to the contained fuel assembly in the unlikely event that another fuel assembly should be dropped anywhere on top of the spent fuel racks.

The rack design contains no protuberances that could cause damage to a fuel assembly being lowered into or being lifted out of a storage position. Lead-ins are provided at the top of some boxes in region 1. No lead-ins are provided in region 2.

9.1.2.1.4 Storage Capacity

The fuel storage pool capacity shall accommodate at least one shipping cask and one complete core, in addition to the maximum number of fuel assemblies normally stored in the pool. Consideration should be given to potential for highly radioactive components, which may require storage in the pool.

The rack design provides the capability to store projected spent fuel discharges resulting from operation through the fall of 2009, while still retaining the capability to accommodate one shipping cask and removal of the complete core from the reactor vessel. The current Ginna operating license expires in 2029. The racks will have the capability to accommodate removal of the core from the reactor during the operating cycle that ends in the spring of 2011. Following receipt of new fuel in preparation for the 2011 spring outage, the racks will not have the capability to store a complete core removal from the reactor vessel. Ginna plans to implement on-site dry cask storage after the 2009 refueling outage to accommodate spent fuel storage requirements through 2029, the end of the current license period.

Operational in 2010, the Independent Spent Fuel Storage Installation (ISFSI) will serve as a facility for the interim storage of sealed, leak proof, and self-contained dry shielded canisters.

Storage capacity is intended to satisfy spent fuel storage requirements through the end of extended plant life with the reactor defueled and the SFP full.

During MODE 6 (Refueling) periods, and whenever the shipping cask is not in the pool, the cask area will be available to store radioactive components or to perform underwater inspection or mechanical operations on radioactive components.

9.1.2.1.5 Fuel Pool Cooling System Instrumentation

Suitable provisions shall be made in the design of the fuel storage pool cooling system to permit installation of instrumentation to monitor system performance.

The pressure and flow of service water (SW) and the temperature, pressure, and flow of spent fuel pool (SFP) water circulating through SFP heat exchanger A are measured and indicated locally. Service water (SW) pressure and flow through SFP heat exchanger B has local indication, as well as installed resistance temperature detectors (RTD) to allow local measurement of service water (SW) inlet/outlet temperatures. SFP heat exchanger B also has indication for the pressure and flow of spent fuel pool (SFP) water as well as installed resistance temperature detectors (RTD) to allow local measurement of spent fuel pool (SFP) water inlet/outlet temperatures. The spent fuel pool (SFP) water temperature is measured and a high temperature alarm is actuated in the control room if the spent fuel pool (SFP) water temperature exceeds 115°F. To provide increased monitoring capabilities during a full core off-load, and depending upon lake temperature this high temperature alarm setpoint may be lowered to a level as determined by the Reactor Engineer. The spent fuel pool (SFP) water level is also measured and a high/low alarm is actuated in the control room if the water level exceeds preset values.

9.1.2.1.6 Seismic Design

The fuel storage pool and storage racks shall be designed to accommodate, within applicable code stress limits, normally imposed loads due to half the design-basis earthquake.

The fuel storage pool and storage racks shall be designed so that normally imposed loads plus loads imposed by the design-basis earthquake will not cause failure. Plastic deformation may take place but with a substantial margin to that which might result in failure.

These criteria are satisfied as described in Section 3.7.

The spent fuel (SFP) pool is founded on sound rock. The spent fuel storage racks are capable of withstanding loads imposed by the safe shutdown earthquake without plastic deformation of the racks and without damage to spent fuel assemblies. The bearing loads are sufficiently low to prevent damage to the stainless steel liner of the spent fuel pool (SFP) and supporting concrete. The reinforced-concrete structure of the pool is capable of transmitting these loads to the rock without plastic deformation of the pool structure.

9.1.2.1.7 Fuel Handling System

Lifting and transport equipment of the fuel handling system shall be designed to prevent dropping of fuel assemblies. Heavy loads shall not be carried over stored fuel assemblies. The design shall prevent lifting a fuel shipping cask over fuel storage racks.

The design includes these provisions except for the full consolidated fuel canisters. The canisters containing consolidated fuel are considered a heavy load per NUREG-0612 criteria. They may be carried over stored fuel assemblies provided that the spent fuel racks beneath the transported canister contain only spent fuel that has decayed at least 60 days since reactor shutdown. See Section 9.1.5 for the discussion of control of heavy loads at Ginna Station.

9.1.2.1.8 Minimum Center-to-Center Spacing

Fuel storage racks shall physically prevent placing more than one fuel assembly in a single storage location; specified minimum center-to-center distances between individual fuel assemblies shall be maintained to meet criticality requirements.

The rack design permits only one fuel assembly or consolidated fuel canister to be inserted into a storage box. Minimum center-to-center spacings between fuel assemblies are maintained by the rack structure.

In the fuel consolidation process, fuel rods are removed from the fuel assembly and stored in a canister. The canister, containing no more than 358 fuel rods (two fuel assemblies), is then placed in a rack storage location. The design of the storage racks has been verified to be able to withstand the loads associated with a maximum dead weight of a canister with 358 fuel rods in each storage location.

Criticality analyses show that even with the consolidated fuel storage, the K_{EFF} criterion of less than or equal to 0.95 is satisfied.

9.1.2.1.9 Stability of Fuel Storage Racks

Fuel storage rack design shall prevent geometric changes due to environmental conditions characteristic of this site. The design shall be stable against tipping with provisions to prevent unplanned movement of the fuel or the racks.

The region 2 type 1 racks, which were not replaced in phase 1 of the re-rack in 1998, are free standing racks which are supported on the pool floor only. The gaps between racks and those between the racks and the pool walls are designed such that the new racks installed in phase 1 of the re-rack do not impose any additional loading on the resident racks or on the pool walls. The new region 1 type 3 racks and region 2 type 2 racks installed in phase 1 are also free standing and self-supporting.

The new type 2 and 3 racks stainless steel square tubes are fillet welded to the base plate. The stainless steel cells are joined together along their length by connecting tabs welded to the square tube faces. This forms the cells in each rack into a continuous structure.

The rack pedestals are adequate in size and number to ensure that the rack structure is stable, thus minimizing tilting, and to equally distribute and minimize the resulting bearing loads onto the pool liner and floor. The pedestals also provide threaded connections to ensure the overall rack module levelness during installation, thus minimizing any load eccentricities and imbalances. The rack design is stable against tipping. Each stored fuel assembly is completely surrounded by a relatively close-fitting box.

9.1.2.1.10 Fuel Pool Leakage Prevention

The spent fuel storage pool (SFP) and refueling canal shall have provisions, such as a water-tight liner, to prevent leakage of pool water.

A stainless steel liner is provided. The spent fuel racks are designed to limit local mechanical loadings on the pool liner to prevent damage to the liner.

9.1.2.1.11 Depth of Water Over Fuel

The fuel storage pool minimum depth shall be determined by dose considerations at the top of the pool considering irradiated fuel or components stored in the pool or in transit and radioactive contaminants in the pool water.

The top of the fuel assemblies stored in the spent fuel storage racks are approximately 26 ft below the surface of the water. A radiological evaluation of the rack design is presented in Section 9.1.2.6.

9.1.2.1.12 Fixed Neutron Poisons

Fuel storage racks using nuclear poisons additional to those inherent in the structural materials shall be designed and fabricated in a manner to prevent inadvertent removal of the additional poisons by mechanical or chemical action. Prior to installation of the additional nuclear poisons, the quantity and effectiveness of the additional poisons shall be verified. Effectiveness of the additional poisons may be checked by isotopic analysis. Provisions shall be made to permit periodic inspection or verification or both, thereafter.

Both borated stainless steel (BSS) and Boraflex are used as neutron absorber materials in the storage racks. BSS fixed absorber plates are provided in the region 2 type 2 and region 1 type 3 racks of the spent fuel pool (SFP). The BSS was specified as ASTM A887-89, grade B, type B6/B7, with a minimum boron content of 1.70%. BSS has an exceptional resistance to corrosion by electrolytic hydridation, oxidation, or other chemical reactions in borated and pure water. There are no significant changes to the mechanical properties of the BSS upon exposure to the levels of irradiation encountered over the design life of the fuel storage racks.

The BSS plate is a free-standing member in the type 2 and type 3 rack designs. The BSS is neither bent nor welded in these racks which precludes any cracking or thermal alteration of the metal.

Boraflex fixed absorber material is provided in the region 2 type 1 racks of the spent fuel pool (SFP). The absorber assemblies are welded in place in each storage cell, thus precluding inadvertent mechanical removal.

To address concerns with Boraflex degradation as presented in Generic Letter 96-04 (*Reference 15*), Ginna Station performed tests in February 1998 of the B-10 areal density of 24 representative Boraflex panels in region 2 of the spent fuel pool (SFP) using the Boron Areal Density Gauge for Evaluating Racks (BADGER). During the testing, degradation beyond the four-inch gap assumption of the criticality analysis was noted on selected Boraflex panels. This data indicated that some panels had undergone dissolution beyond expected levels and placed the spent fuel pool in an unanalyzed condition.

This event and the results of the associated assessment that was performed were reported to the NRC in *Reference 10*. In addition, the Technical Specifications were changed to ensure that controls were in place to verify at least 2300 ppm of soluble boron was maintained in the spent fuel pool (SFP).

A subsequent criticality analysis (Section 9.1.2.4) was performed without crediting any Boraflex in the Region 2, Type 1 racks (*Reference 60*). This analysis demonstrated that a soluble boron concentration of 975 ppm, based on a B-10 isotropic fraction of 0.197 for recycled boron, was necessary to maintain $k_{EFF} \leq 0.95$ under normal and accident conditions. The Technical Specifications retained the control that at least 2300 ppm of soluble boron be maintained in the spent fuel pool. The 2300 ppm ensures that there is sufficient time to detect and mitigate any postulated dilution event before the concentration of soluble boron is diluted to 975 ppm. A boron dilution analysis (*Reference 61*) demonstrated that the volume necessary to dilute from 2300 ppm to 975 ppm was 183,000 gallons. Based on this analysis, any postulated dilution event is not credible. Plant Uprate performed in 2006 has no impact on this boron dilution analysis.

<u>9.1.2.1.13 Bearing Loads on Pool Liner</u>

Provisions shall be made to accommodate the necessary heavy equipment loads in the fuel storage pool without subjecting the pool liner to mechanical damage.

The bearing loads on the pool liner are low and will not cause mechanical damage to the liner.

The spent fuel storage facility should be designed to Seismic Category I requirements. The spent fuel pool (SFP) and spent fuel racks are designed to Seismic Category I requirements.

9.1.2.2 Description

<u>9.1.2.2.1 Spent Fuel Pool (SFP)</u>

The spent fuel pool (SFP) is a Seismic Category I design, reinforced-concrete structure located in the west end of the auxiliary building. The spent fuel pool (SFP) is totally clad with stainless steel. The spent fuel pool (SFP) contains approximately 255,000 gallons of water (*Reference 27*), which is maintained borated to at least a 2300-ppm concentration. A leak chase system under the floor liner plate minimizes the chances of accidental drainage, and the weir gate access to the refueling canal has a high sill to prevent inadvertent drainage through the canal from uncovering the stored fuel assemblies.

The normal makeup water sources to the spent fuel pool (SFP) are from the refueling water storage tank (RWST) or one of the chemical and volume control system holdup tanks. The minimum required refueling water storage tank (RWST) water volume is 300,000 gal. The refueling water storage tank (RWST) capacity is approximately 338,000 gal. Water is supplied from the refueling water storage tank (RWST) by the refueling water purification pump to the spent fuel pool (SFP) purification system to the spent fuel pool (SFP). Alternative sources of makeup water are available from the reactor makeup water tank or the monitor tanks.

The spent fuel pool (SFP) water leak collection system consists of channels in the concrete pool floor which are designed to collect any water that may leak through the stainless steel liner. Leakage is directed to a collection tank, leading to the liquid waste processing system.

9.1.2.2.2 Spent Fuel Storage Racks

The spent fuel pool (SFP) capacity is discussed in Section 9.1.2. Vertical storage racks are seated on the pool floor and cover the entire area except for a section in the southeast corner reserved for fuel shipping cask loading operations. Control rods are stored in the fuel assemblies. The new fuel elevator is located at the northeast corner and is used to lower new fuel elements into the spent fuel pool (SFP) for transfer to the reactor.

The storage pool is divided into two regions that may contain either intact fuel assemblies or consolidated fuel canisters. The inherent strength of the rack designs results from their honeycomb box structure arrangement. The rack assemblies are made up of a repeating array of square stainless steel boxes in a checkerboard arrangement. Region 1 allows fresh fuel storage with a combination of a fixed neutron absorber, flux traps, and a checkerboard arrangement of fresh and burned fuel assembles. Lead-in funnels are provided on some cells that are allowed to contain fresh fuel. Region 2 rack types ensure criticality safety with a combination of fixed neutron absorbers, burnup credit, and a checkerboard arrangement of burned fuel assemblies. The checkerboard arrangements are set forth in Technical Specification LCO
3.7.13. No lead-in funnels are provided in any cell in region 2. In both regions, the lower end of each box contains a horizontal plate, with a circular hole in the center, both to position the spent fuel assembly or consolidated fuel canister and to allow cooling water flow.

Region 1 contains five modules of the rack design designated as type 3. The boxes forming the rack cells are joined at the corners in a checkerboard arrangement to create an array with a nominal 9.23 in. center-to-center pitch. Each cell formed by the box array contains a borated stainless steel insert with an approximate 8.14 in. square inner dimension. The insert is positioned vertically to span the active fuel region of the fuel assemblies. Eight horizontal belts on alternate borated stainless steel (BSS) cells maintain a water gap between the BSS cells to provide a flux trap for slowing neutrons between the BSS absorber plates. Region 1 provides storage locations for 144 fresh, 145 burned, and 5 damaged (bowed) assemblies (see Figure 9.1-3). Either fresh or burned assemblies may be stored in these locations, as long as a fresh/ burned checkerboard configuration is maintained for the fresh fuel assemblies.

Upon completion of phase 1 of the re-rack in 1998, region 2 contains two different rack designs designated as types 1 and 2 for the storage of burned fuel. All rack types are formed with stainless steel boxes. The type 1 racks are the high-density fixed absorber racks which were converted in 1984. This type of rack has an approximate 8.11 in. square inner dimension with two Boraflex panels per cell. Six type 1 modules (140 cells/module) provide

839 storage locations (840 cells less 1 capped cell) for storage of burned fuel assemblies or canisters with burned fuel rods. The type 2 racks are high-density free-standing racks with borated stainless steel (BSS) plates as the fixed absorbers. This type of rack is fabricated with stainless steel cells joined at the corners, with BSS inserts in every other cell. Two type 2 modules provide 187 storage locations for burned fuel assemblies or canisters with burned fuel rods.

The phase 1 re-rack in 1998 installed the type 2 and 3 racks to augment the storage remaining in the type 1 racks. Thus, as of 1998 the pool contains six type 1 modules, two type 2 modules, and five type 3 modules for a total of 1321 storage cells.

9.1.2.3 Design Evaluation

The original spent fuel storage racks provided capacity for the storage of 210 fuel assemblies. In 1976, the NRC approved the replacement of the original racks with higher density flux trap type racks (*References 1* and *2*). This expanded the storage capability from 210 to 595 fuel assemblies. In the submittal to the NRC, a nuclear criticality analysis was made assuming a fuel assembly design enriched to 3.5 wt % of Uranium-235. This criticality analysis was applicable to the previously delivered Westinghouse fuel.

With the fuel reload of 1984, fuel assemblies of a Westinghouse design incorporating axial natural uranium blankets were used. This change in design, along with the adoption of low radial leakage fuel management, requires central region enrichments in excess of the 3.5% used in the analysis of *Reference 1*.

In 1984, the NRC approved the conversion of six flux trap type racks to high-density fixed poison type racks (*References 3* and *4*). This further expanded the storage capacity from 595 to 1016 fuel assemblies and resulted in a two-region spent fuel pool (SFP).

In 1983, the NRC approved a new analysis which assumes an unirradiated fuel assembly enrichment of 4.25 wt % Uranium-235 (*References 6* and *28*). The analysis did not include any changes to the storage rack or pool design.

In 1985, NRC approval was received for the use of consolidated fuel canisters (*Reference 29*). In 1996, NRC approval was received to store unirradiated fuel assemblies with integral burnable poisons with up to a nominal 5.0 w/o U-235, provided the K_∞ is ≤ 1.458 (*References 26* and *30*).

In 1998, NRC approval was received for re-racking portions of the spent fuel pool (*Reference 38*). This included replacement of the three region 1 flux trap racks with two types of high density fixed neutron absorber type racks. In addition, the approval allowed attachment of similar high-density fixed neutron absorber type racks to the north and south faces of the Boraflex neutron absorber racks that constituted region 2 prior to the approval. The installation of the new high-density racks was planned in two phases. In 1998, phase 1 added five rack modules to create region 1 for storage of fresh fuel and two additional rack modules to augment region 2. This modification increased the number of usable cells from 1015 to 1320 (one additional cell is capped). The phase 2 attachment of the remaining six high-density rack modules to the Boraflex racks in region 2 will be done in the future as needed.

9.1.2.4 Nuclear Analysis

9.1.2.4.1 Methods of Analysis

The criticality calculation method and cross-section values are verified by comparison with critical experiment data for fuel assemblies similar to those for which the racks are designed. This benchmarking data is sufficiently diverse to establish that the method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps and low moderator densities.

New Fuel Storage Rack

The design method that insures the criticality safety of fuel assemblies in the new fuel storage rack was described in the application for the 1996 enrichment upgrade (*Reference 31*). This design method uses the AMPX (*References 7* and *8*) system of codes for cross-section generation and KENO Va (*Reference 9*) for reactivity determination.

The 227 energy group cross-section library that is the common starting point for all cross-sections used for the benchmarks of KENO Va and the KENO Va storage rack calculations is generated from ENDF/B-V (*Reference 7*) data. The NITAWL (*Reference 8*) program includes, in this library, the self-shielded resonance cross-sections that are appropriate for each particular geometry. The Nordheim Integral Treatment is used. Energy and spatial weighting of cross-sections is performed by the XSDRNPM (*Reference 8*) program which is a one-dimensional S_n transport theory code. These multigroup cross-section sets are then used as input to KENO Va (*Reference 9*) which is a three-dimensional Monte Carlo theory program designed for reactivity calculations.

A set of 44 critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and uncertainty. The benchmark experiments cover a wide range of geometries, materials, and enrichments, ranging from relatively low enriched (2.35, 2.46, and 4.31 w/o), water moderated, oxide fuel arrays separated by various materials (B_4C, aluminum, steel, water, etc.) that simulate LWR fuel shipping and storage conditions to dry, harder spectrum, uranium metal cylinder arrays at high enrichments (93.2 w/o) with various interspersed materials (plexiglass and air). Comparison with these experiments demonstrates the wide range of applicability of the method.

The highly enriched benchmarks show that the criticality code sequence can correctly predict the reactivity of a hard spectrum environment, such as the optimum moderation condition often considered in fresh rack and shipping cask analyses. However, the results of the 12 highly enriched benchmarks are not incorporated into the criticality method bias because the enrichments are well above any encountered in commercial nuclear power applications. Basing the method bias solely on the 32 low enriched benchmarks results in a more appropriate and more conservative bias.

The 32 low-enriched, water moderated experiments result in an average KENO Va K_{EFF} of 0.9930. Comparison with the average measured experimental K_{EFF} of 1.0007 results in a method bias of 0.0077. The standard deviation of the bias value is 0.0014 ΔK. The 95/95 one-sided tolerance limit factor for 32 values is 2.20. Thus, there is a 95 percent probability with a 95 percent confidence level that the uncertainty in reactivity, due to the method, is not greater than 0.0030 ΔK.

Material and construction tolerance reactivity effects and reactivity sensitivities are determined using the transport theory computer code, PHOENIX (*Reference 11*). PHOENIX is a depletable, two-dimensional, multigroup, discrete ordinates, transport theory code which utilizes a 42 energy group nuclear data library.

Spent Fuel Storage Racks

The analysis methods employ: (1) the SCALE 4.4 code system, as documented in *Reference 53*, with the 44 group ENDF/B-V cross-section library, and (2) the two-dimensional integral transport code DIT, *Reference 54*, with an ENDF/B-VI neutron cross section library.

SCALE 4.4 is used for calculations involving infinite arrays of storage cells and checker-boarded storage cells depending on the storage features of individual rack types. In addition, it is employed in a full pool representation of the storage racks to evaluate soluble boron worths and postulated accidents.

SCALE 4.4 modules employed in both the benchmarking analyses and the spent fuel storage rack analyses include the control module CSAS and the following functional modules: BONAMI, NITAWL-II, and KENO V.a. All references to KENO in the text to follow should be interpreted as referring to the KENO V.a module.

The DIT code is used for simulation of in-reactor fuel assembly depletion. The following sections describe the application of these codes in more detail.

Validation of SCALE 4.4

Validation of the SCALE 4.4 code system for the purposes of fuel storage rack analyses is based on 123 criticality experiments documented in NUREG/CR-6361 (*Reference 82*) with geometries, materials, and neutron interaction characteristics representative of storage array. The 123 critical experiments include 58 in core-type category, 45 in separator plate category, 12 in separator plate-soluble boron category, and 8 in flux trap-void category. The mean calculational bias, the mean calculational variance, and the 95/95 confidence level multiplier are deduced as 0.0008, $(0.0046)^2$, and 1.896, respectively (*Reference 60* and *Reference 83*).

Application to Fuel Storage Pool Calculations

As noted above, the CSAS control module was employed to execute the functional modules within SCALE 4.4. The CSAS25 control module was used in the majority of the cases to analyze either infinite arrays of single or multiple storage cells or the full spent storage pool.

Standard material compositions were employed in the SCALE 4.4 analyses consistent with those of *Reference 39*. For fresh fuel conditions, the fuel nuclide number densities were derived within the CSAS module. For burned fuel representations, the fuel isotopics were derived from the DIT code as described below.

The DIT Code

The DIT (Discrete Integral Transport) code performs a heterogeneous multigroup transport calculation for an explicit representation of a fuel assembly. The neutron transport equations are solved in integral form within each pin cell. The cells retain full heterogeneity throughout the discrete integral transport calculations. The multigroup spectra are coupled between cells though the use of multigroup interface currents. The angular dependence of the neutron flux is approximated at cell boundaries by a pair of second order Legendre polynomials. Anisotropic scattering within the cells, together with the anisotropic current coupling between cells, provide an accurate representation of the flux gradients between dissimilar cells.

The multigroup cross sections are based on the Evaluated Nuclear Data File Version 6 (ENDF/B-VI). Cross sections have been collapsed into an 89 group structure which is used in the assembly spectrum calculation. Following the multigroup spectrum calculation, the region-wise cross sections within each heterogeneous cell are collapsed to a few groups (usually 4 broad groups), for use in the assembly flux calculation. A B1 assembly leakage correction is performed to modify the spectrum according to the assembly in-or-out-leakage.

Following the flux calculation, a depletion step is performed to generate a set of region-wise isotopic concentrations at the end of a burnup interval. An extensive set of depletion chains are available, containing 33 actinide nuclides in the thorium, uranium and plutonium chains, 171 fission products, the gadolinium, erbium and boron depletable absorbers, and all structural nuclides. The spectrum-depletion sequence of calculations is repeated over the life of the fuel assembly. Several restart capabilities provide the temperature, density and boron concentration dependencies needed for three dimensional calculations with full thermal-hydraulic feedback effects.

The DIT code and its cross-section library are employed in the design of initial and reload cores and have been extensively benchmarked against operating reactor history and test data.

For the purpose of spent fuel pool criticality analysis calculations, the DIT code is used to generate the detailed fuel isotopic concentrations as a function of fuel burnup and initial feed enrichment. Each selected set of fuel isotopics is equivalenced to a reduced set of burned fuel isotopics at specified time points after discharge. The latter burned fuel representation includes the following nuclides: ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{149}Sm, ^{16}O, and ^{10}B. The DIT code lists the Samarium-149 isotopics for ^{149}Sm and ^{149D}Sm (a metastable isomer). Since ^{149}Sm is a stable isotope, the concentration of this Samarium isotope is the sum of the individual concentration of these two isomers.

The isotopic number densities from the DIT calculation are based upon Cell average values. The input to KENO calculations require that the number densities be specified for the fuel pellet. Therefore, the number densities from the DIT calculations are scaled by the ratio of area of the cell to the area of the fuel pellet for use in the KENO calculations. The concentration of Boron - 10 is determined by reactivity equivalencing a given DIT cell calculation with a corresponding KENO cell calculation to within the KENO one sigma uncertainty level.

9.1.2.4.1.1 Criticality Methodology

A summary of the methodology follows.

1. Determine the fresh and spent fuel storage configuration of the spent fuel pool using no soluble boron conditions such that the 95/95 upper tolerance limit value of K_{EFF} for the storage pool, including applicable biases and uncertainties, is less than unity.
2. Next, using the resulting storage configuration from the previous step, calculate the spent fuel rack effective multiplication factor with the chosen concentration of spent fuel pool

soluble boron present. Then calculate the sum of: (a) the latter multiplication factor, (b) the reactivity uncertainty associated with fuel assembly and storage rack tolerances, and (c) the biases and other uncertainties required to determine the final 95/95 confidence level effective multiplication factor and show that at the chosen concentration of soluble boron, the system maintains the overall effective multiplication factor less than or equal to 0.95.

3. Use reactivity equivalencing methodologies to determine the minimum fuel assembly burnup for fuel assembly enrichments higher than allowed in Step 1, above. As a function of time after discharge and burnup, calculate the reactivity credit due to actinides for each fuel assembly.
4. Determine the increase in reactivity caused by postulated accidents and the corresponding additional amount of soluble boron needed to offset these reactivity increases.

An alternative form of expressing the soluble boron requirements is given in *Reference 50*. The final soluble boron requirement is determined from the following summation:

$$SBC_{TOTAL} = SBC_{95/95} + SBC_{RE} + SBC_{PA}$$

where:

SBC_{TOTAL} = total soluble boron credit requirement (ppm),

$SBC_{95/95}$ = soluble boron requirement for 95/95 $K_{EFF} \leq 0.95$ (ppm),

SBC_{RE} = soluble boron required for reactivity equivalencing methodologies (ppm),

SBC_{PA} = soluble boron required for $K_{EFF} \leq 0.95$ under accident conditions (ppm)

For purposes of the analyses contained herein, minimum burnup limits established for fuel assemblies to be stored in the different types of storage racks do include burnup credit established in a manner which takes into account conservative approximations to the operating history of the fuel assemblies. Variables such as the axial burnup profile as well as the axial profile of moderator and fuel temperatures have been factored into the analyses.

The methodology employed in this analysis for soluble boron credit is analogous to that of *Reference 49* and employs analysis criteria consistent with those cited in the Safety Evaluation by the Office of Nuclear Reactor Regulation, *Reference 50*.

The design input employed in this analysis is basically the same as that employed in the Ginna SFP Reracking Licensing Report, *Reference 39*. However, the current analyses employ a broader scope by implementing the Soluble Boron Credit Methodology, taking credit for the decay of ^{241}Pu, and quantifying the spent fuel storage limits for Region 2, Type 1 and 2 Racks independently. The Soluble Boron Credit Methodology provides additional reactivity margin in the spent fuel storage analyses which may then be used to implement added flexibility in storage criteria and, for example, eliminate the need to implement the degraded boraflex modeling as well as eliminate credit for IFBA in fresh fuel assemblies with enrichments above 4 wt% ^{235}U.

Please note that for the Region 1, Type 3 storage racks, reactivity control is achieved by means of a checkerboard of burned and fresh fuel assemblies having initial enrichments of up to 5.0 wt% ^{235}U(nominal); no Integrated Fuel Burnable Absorber (IFBA) credit for fresh fuel assemblies with nominal enrichments above 4.0 wt% ^{235}U is required. Region 2 accommodates burned fuel assemblies having initial enrichments up to 5 wt% ^{235}U(nominal) at prescribed minimum burnups.

The selection of design basis fuel assembly types was based on an evaluation of the variety of fuel assemblies employed in the reactor to date and selecting the most reactive type for a given evaluation. The candidate fuel assembly types include the Westinghouse Standard, Westinghouse OFA, Westinghouse VANTAGE+, Mixed Oxide fuel assemblies and other Lead Test Assemblies, as well as the Consolidated Fuel Assembly Canisters and the damaged fuel rod basket.

The selection of the Westinghouse OFA as the design basis fresh fuel assembly is predicated on the fact that this assembly is an optimized design and is more reactive than the Westinghouse Standard in the fresh fuel condition. This result is consistent with the analyses of *Reference 39*; the latter analyses also concluded that the Westinghouse Standard assembly becomes more reactive than the OFA assembly beyond burnups greater than about 12,000 MWD/MTU burnup. Thus, the design basis burned fuel assembly employed for these analyses is taken to be a variant of the Westinghouse Standard fuel assembly because of its burnup characteristics and the fact that it is, in general, more representative of fuel assemblies employed in the past operation of the plant. The design basis burned fuel assembly is taken to be a Westinghouse Standard fuel assembly with the instrument tube replaced by a fuel rod and the RCC guide tubes made of zircaloy. These changes were simply added conservatisms to assure enveloping of the variety of fuel assemblies that had passed through the core and presently reside in the spent fuel pool.

The reactivity characteristics of the different rack Types 1, 2, and 3 were evaluated using infinite lattice analyses; this environment was employed in the evaluation of the burnup limits versus initial enrichment for each rack type as well as the evaluation of physical tolerances and uncertainties. The full spent fuel pool model was also employed to evaluate soluble boron worths and the reactivity worth of postulated accidents.

9.1.2.4.1.2 Criticality Analysis of Consolidated Rod Storage Canisters in Spent Fuel Racks

The fuel rod consolidation canister is employed to store burned fuel rods removed from multiple, typically two or less, fuel assemblies. The purpose of these canisters is to increase the storage capacity of the pool consistent with the load bearing capability of the pool structure by removing fuel rods from the burned fuel assembly cage structure and storing the rods in the consolidated fuel rod canister at a reduced water to fuel ratio. The base canister model assumed only a square stainless steel can with an outer square dimension of 8.02 in. and a wall thickness of 0.089 in. The upper tolerance value of the outer dimension of the canister and the lower tolerance value of the thickness of the steel enclosure were used to maximize the capacity of the canister. The divider plate was not modeled for conservatism.

The KENO calculations were performed with both the Westinghouse Standard and Westinghouse OFA fuel rods. The pitch of the fuel rods inside the canister was varied to obtain the near-optimum pitch for the canister. The fuel rods were enriched to 1.30 wt% ^{235}U, a conservative value for Region 2 Type 1 cells. The KENO results show the most reactive case occurs when 225 fuel rods from the Westinghouse Standard fuel assembly are optimally spaced in the canister. This value of K_{EFF} is, however, lower than for the case of the design basis spent fuel assembly in a Type 1 cell. It can therefore be concluded that results based on loading design basis spent fuel assemblies in Type 1 cells is bounding and the canisters can be excluded from further treatment. This argument is also extended to other type storage cells in the spent fuel pool so as to permit use of the consolidated canisters in those locations.

These evaluations indicate that the criticality condition is satisfied for storage of consolidation containers in locations for intact spent assemblies. The criticality criterion of 95/95 $K_{EFF} \leq 0.95$ is also met for the fuel rods that satisfy the burnup vs. enrichment curves for either region 1 or 2.

9.1.2.4.1.3 Summary of Criticality Results

Fresh Fuel Racks

The acceptance criteria for criticality requires the effective neutron multiplication factor, K_{EFF}, in the fresh fuel storage rack to be less than or equal to 0.95, including uncertainties, under flooded conditions and less than or equal to 0.98, including uncertainties, under optimum moderation conditions.

The acceptance criteria for criticality is met for the Ginna Fresh Fuel Storage Racks for the storage of both Westinghouse 14x14 OFA and 422 VANTAGE+ fuel assemblies with nominal enrichments up to 5.0 w/o Uranium-235.

Spent Fuel Racks

A summary of the results is as follows.

1. In *Reference 60*, the SFP criticality analysis was performed for the Extended Power Uprate (EPU) conditions and pre-EPU conditions with the present Pu-241 half-life values of 14.4 years. A summary of the results is as follows:

   a) Pre-EPU conditions: Soluble boron credit methodology was employed to establish a target K_{EFF} value of 0.98225 for the spent fuel pool at zero soluble boron. The allowance for applicable biases and uncertainties was deduced to be 0.01640; thus, the 95/95 upper tolerance limit value of K_{EFF} was deduced to be 0.99865. The total soluble boron requirement for achieving a 95/95 value of $K_{EFF} \leq 0.95$ was deduced to be the summation of the following three terms: $SBC_{95/95}$ = 389 ppm, SBC_{RE} = 212 ppm, and SBC_{PA} = 363 ppm for a total of 964 ppm. The soluble boron concentration was increased due to the difference in the B-10 atom percent used in the analysis (19.9 a/o) and a lower bounding operating value (19.0 a/o). This results in a soluble boron concentration equal to 1010 ppm.

   b) EPU conditions: Soluble boron credit methodology was employed to establish a target K_{EFF} value of 0.98318 for the spent fuel pool at zero soluble boron for the Pre-EPU conditions. The allowance for applicable biases and uncertainties was deduced to be 0.01640; thus, the 95/95 upper tolerance limit value of K_{EFF} was deduced to be 0.99958. The total soluble boron requirement for achieving a 95/95 value of $K_{EFF} \leq 0.95$ was deduced to be the summation of the following three terms: $SBC_{95/95}$ = 407 ppm, SBC_{RE} = 219 ppm, and SBC_{PA} = 374 ppm for a total of 999 ppm. The soluble boron concentration was increased due to the difference in the B-10 atom percent used in the analysis (19.9 a/o) and a lower bounding operating value (19.0 a/o). This results in a soluble boron concentration equal to 1047 ppm.

   c) Note that the minimum spent fuel pool boron concentration value of 2300 ppm required by Technical Specifications is more than sufficient to maintain K_{EFF} less than or equal to 0.95 under accident conditions. By virtue of the double contingency principle, which has been endorsed by the NRC staff, two unlikely independent and concurrent events are beyond the scope of the required analysis. Therefore, credit for the presence of the entire 2300 ppm of soluble boron may be assumed in evaluating other accident conditions such as fuel misplacement. Without including the additional 363 and 374 ppm of soluble boron required for K_{EFF} less than or equal to 0.95 under accident conditions for pre-EPU and EPU conditions respectively, the total soluble boron requirement is 629 ppm and 655 ppm at 19.0 a/o B-10, for pre-EPU and EPU conditions respectively. However, for conservatism, the boron concentration required by Technical Specifications of 965 ppm, which was previously determined in *Reference 60* for pre-EPU conditions with the former Pu-241 half-life values of 13.2 years, is retained as total required boron concentration for K_{EFF} less than or equal to 0.95. In effect, the minimum requirement for the criticality analysis is 120 B-10 atoms per million, which is based on B-10 abundance of 19.0 a/o and soluble boron of 629 ppm. At a soluble boron of 965 ppm, 120 B-10 atoms per million corresponds to a B-10 abundance of 12.1 a/o. These parameters define the limit of the analyzed boron supported by the analysis and account for changes in abundance and surveillance measurement uncertainty.

d) Also note this soluble boron concentration includes an allowance for 5% burnup uncertainty. In addition, all of the burnup versus enrichment storage curves have been increased by 5%. Therefore, a 5% burnup uncertainty has been double counted.

2. The design basis fuel assembly for the fresh fuel storage cells in the Region 1, Type 3 racks was taken to be a conservative representation of the Westinghouse OFA 14 x 14 fuel assembly having a nominal enrichment of 5 wt% ^{235}U, no IFBA loadings, and the instrument tube location replaced by a fuel rod. The design basis fuel assembly for the burned fuel storage cells in both Region 1 and 2 racks was taken to be a conservative approximation to the Westinghouse Standard 14 x 14 fuel assembly wherein the RCC guide tubes were represented as zircalloy-4 and the instrument tube was replaced by a fuel rod. This conservative approximation to the burned fuel assembly envelops the characteristics of all burned fuel assemblies, including lead test assemblies, currently stored in the spent fuel pool. This design basis burned fuel assembly was represented by an 8-node axial representation of the assembly burnup and applicable fuel and moderator temperatures.

3. All representations of the Region 2, Type 1 spent fuel storage racks, originally containing boraflex inserts between the L-shaped insert and the storage cell tube wall, were represented in both the infinite cell array and full storage pool analyses as having nominal pool water in place of the boraflex.

4. Minimum fuel assembly burnup limits versus fuel assembly initial average enrichment were established for Region 2, Type 2 spent fuel storage cells. These limits were established on both a nominal basis and an equivalent dual tier approach for 0, 5, 10, 15, and 20 years of ^{241}Pu decay so as to provide more efficient utilization of the available spent fuel storage capacity of the storage racks.

5. It was demonstrated that the existing fuel assembly burnup versus initial enrichment criteria established in *Reference 39* are applicable for Region 1, Type 3 cells. These analyses also demonstrated this objective is easily achieved with fresh fuel enrichments of 5 wt% ^{235}U and no requirements for IFBA credit in the fresh fuel assemblies.

It was further established that either a fuel rod consolidation canister or a damaged rod storage basket is less reactive than a fuel assembly of equivalent burnup when placed in a spent fuel storage cell. Consequently, there are no restrictions as to placement of these storage devices in the spent fuel storage cells.

Other items may be stored in the spent fuel pool in addition to fresh or discharged fuel assemblies. These items, in general, fall into the category of Non-Special Nuclear Material (SNM). These items are non-multiplying and, in general, are parasitic to the spent fuel rack local reactivity. Some of the items which fall under this category that can be safely stored in the spent fuel pool are: Dummy Canisters containing Non-SNM, Consolidation Hardware, Dummy Fuel Assemblies, Trash Basket containing full length control rods, etc. The general rule for safely storing these types of items is very simple: any non-multiplying and non-fissile item can be safely stored in any cell location. Note that neutron sources are considered to be non-multiplying and non-fissile.

Technical Specification defines the limits on storage of spent fuel assemblies versus assembly burnup, initial enrichment, and years of ^{241}Pu decay.

The analytical methods employed in the criticality analysis conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," Section 5.7 Fuel Handling System; ANSI 57.2-1983, "Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations," Section 6.4.2; ANSI 57.3-1983, "Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants"; ANSI N16.9-1975, "Validation of Calculational Methods for Nuclear Criticality Safety"; and the NRC Standard Review Plan, Section 9.1.2, "Spent Fuel Storage".

As discussed in *Reference 67*, the critical design parameters (fuel pellet diameter and fuel stack height) of the 422 Vantage+ fuel assemblies used for the plant uprate are bounded by the critical parameters used for the Westinghouse Standard 14 x 14 fuel assembly in the design basis criticality analyses. Additionally, the burn-up characteristics of the 422 Vantage+ fuel assembly at 1775 MWt nominal power level are bounded by the burn-up characteristics used in the analysis for the Westinghouse Standard fuel assembly. Therefore, the pre-uprate criticality analyses for spent fuel remaining bounding.

9.1.2.4.2 Accident Analysis

9.1.2.4.2.1 Fresh Fuel Storage Racks

Under normal conditions, the fresh fuel racks are maintained in a dry environment. The introduction of water into the fresh fuel rack area is the worst-case accident scenario. The water flooding cases analyzed in this report are bounding accident situations which result in the most conservative fuel rack K_{EFF}.

Other accidents can be postulated which would cause some reactivity increase (i.e., dropping a fuel assembly between the rack and wall, or dropping an assembly on top of the rack). For these other accident conditions, the double contingency principle is applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for these other accident conditions, the absence of a moderator in the fresh fuel storage racks can be assumed as a realistic initial condition since assuming its presence would be a second unlikely event.

Experience has shown that the maximum reactivity increase associated with postulated accident (dropping a fuel assembly between the rack and wall, or dropping an assembly on top of the rack) is less than 10 percent ΔK.

Therefore, since the normal, dry fresh fuel rack reactivity is less than 0.55, and the maximum reactivity increase for the postulated accidents is less than 10 percent ΔK, the maximum rack K_{EFF} under these other postulated accidents conditions will be less than 0.95.

9.1.2.4.2.2 Spent Fuel Storage Racks

The soluble boron concentration (ppm) required to maintain K_{EFF} less than or equal to 0.95 under accident conditions is determined by first surveying all possible events which increase the K_{EFF} value of the spent fuel pool. The accident event which produced the largest increase in spent fuel pool K_{EFF} value is employed to determine the required soluble boron concentration necessary to mitigate this and all less severe accident events.

Several fuel mishandling events and one seismic event were simulated to assess the possible increase in the K_{EFF} value of the spent fuel pool. The fuel mishandling events all assumed that a fresh 5.0 wt % ^{235}U assembly with no IFBAs was mislocated into any cell of the spent fuel pool intended for less reactive fuel assemblies. The seismic event results in the reduction of the gap between all the modules to 0.1 cm. The inter-module gap was originally simulated based on the minimum fabricated values.

A survey of the various fuel mishandling events considered in an earlier analysis (Section 4.3.6, *Reference 39*) indicated that the most disruptive accident would be the misplacement of a fresh fuel assembly with 5.0 wt % ^{235}U in a Type 1 location. Therefore, all the other mishandling events were simulated with equivalent fresh fuel assemblies in all the cells; this event was simulated with burnt fuel assemblies occupying the Region 2 Type 1 and Type 2 locations. The various mishandling accidents (including the location) were simulated and the calculated K_{EFF} values were compared.

The calculations determined that the highest worth of a fuel mishandling event is based on a misloading in the Type 1 racks with a worth of 5.809% delta K_{EFF} for pre-EPU conditions and 5.910 % for EPU conditions. The soluble boron concentration (ppm) necessary to compensate for this reactivity insertion is conservatively calculated to be 363 ppm for pre-EPU conditions and 374 ppm for EPU conditions.

As noted above, the misplaced fuel assembly was cited in *Reference 39* as being the most adverse postulated mishandling event. Other dropped assembly events such as, for example, the postulated Rack Type 2 T-bone fuel assembly accident configuration and the postulated deep drop type accidents, have little effect on the local K_{EFF} of the spent fuel storage rack as indicated by the analyses cited in Table 4.3-14 of *Reference 39*. In the case of the T-bone accident, the delta K_{EFF} resulting from a fuel assembly lying on top of the storage rack is quite minimal due to the relatively large separation distance between the top of the fuel columns for the assemblies standing vertical in the storage rack and the fuel assembly lying across the top of the rack. In the case of the postulated deep drop accident, the deflection of the base plate is limited by the height of the pedestals supporting the rack above the concrete floor and the structural design of the rack. A significant fraction of the base plate deflection distance would be taken up by the fuel assembly structure below the active fuel columns. Thus, one would again conclude the misplaced fuel assembly accident overshadows the reactivity insertion resulting from the postulated deep drop event.

9.1.2.5 Thermal-Hydraulic Analysis

The evaluation of the heat removal criteria of the spent fuel pool (SFP) cooling system is presented in Section 9.1.3.4.

9.1.2.6 Radiological Evaluation

The principal source of radiation levels observed at the surface of the spent fuel pool (SFP) water is due to the concentration of radionuclides within the pool water. This has been verified by calculations. The observed dose rate has been typically less than 20 mrem/hr. The radionuclides are removed from the water by the spent fuel pool (SFP) demineralizer with the need for changing the demineralizer resin determined by the demineralizer Decontamination Factor (DF), radiation levels, or the pressure drop across the demineralizer.

Increased fuel storage may result in an increased frequency of changing the demineralizer resin, but is not expected to result in any increase in the radionuclide concentrations or in subsequent radiation levels at the surface of the water. As dose rates show a very weak relationship to the amount of fuel stored in the pool, the increase in fuel storage capacity due to the use of consolidated fuel canisters will not affect the station's ability to maintain individual occupational doses within the limits of 10 CFR 20.

The top of the fuel assemblies stored in the spent fuel storage racks are approximately 26 ft below the surface of the water. The 26-ft water shield reduces the direct radiation from the stored fuel assemblies to values that are negligible when compared to background.

The extent of damage that might result to a fuel assembly during fuel handling and the radiological consequences of such an event are discussed in Section 15.7.

9.1.2.7 Radiological Consequences of Tornado Missile Accident (TMA)

Previous analyses (*References 12*, *44* and *66*) and evaluations (*Reference 65*) performed for the original 1998 re-rack modification identified that a tornado missile impact of spent fuel assemblies in Region 1 of the spent fuel pool resulted in the limiting control room and off-site radiological consequences. Subsequent to the spent fuel pool re-rack in 1998, the tornado missile analysis (*Reference 71*) was revised (per *References 72* and *73*) using Alternate Source Term methodology as part of the installation of the control room emergency air treatment system (CREATS) in 2004. The control room dose for the TMA was re-analyzed as part of the control room emergency air treatment system (CREATS) upgrade modification to reflect the new configuration. For consistency the atmospheric dispersion factors (X/Q) and off-site doses were also re-analyzed. The resulting radiological consequences were analyzed (*Reference 69*) and evaluated (*Reference 70*). As part of the plant uprate the limiting tornado missile accident radiological consequences due to uprate were re-analyzed using Alternate Source Term methodology (*Reference 67*) and evaluated by the NRC (*Reference 68*).

The tornado missile accident assumed that a hypothetical tornado missile representing a 1490-pound wooden pole, 35 feet in length and 13.5 inches in diameter, propelled by the wind, penetrates the auxiliary building roof. Since the pool is divided into two regions, it is possible that the hypothetical tornado missile can impact and damage up to nine fuel assemblies in either region. For uprate, the missile is assumed to impact nine fuel assemblies in Region 1 of the spent fuel pool (five fuel assemblies decayed for 100 hours and four fuel assemblies decayed for 60 days) which represents the limiting condition for radiological releases. Neither control room isolation nor re-circulating filtration are conservatively assumed. The major assumptions and parameters used in this analysis are summarized in Table 9.1-2.

The resulting offsite and control room doses due to the tornado missile accident occurring in region 1 are shown in Table 9.1-5.

9.1.2.8 Radiological Consequences of a Dropped Consolidated Canister

A consolidated canister can contain all of the fuel rods from two assemblies and is considered a heavy load per NUREG-0612 criteria. There are established controls which govern the movement of the canisters in the spent fuel pool (SFP). The canisters will be lifted using a single-failure proof crane and a single-failure proof lifting system and will be handled in accordance with the guidelines in NUREG-0612 with regard to limiting the chance of an unacceptable heavy load drop. This action will prevent potential fuel damage and the subsequent release of fission products. Thus, the offsite radiological consequences need not be determined for this accident.

9.1.3 SPENT FUEL POOL COOLING

9.1.3.1 Design Bases

The spent fuel pool (SFP) cooling system is designed to remove from the spent fuel pool (SFP) the heat generated by stored spent fuel elements. Piping is so arranged that failure of any pipeline does not drain the spent fuel pool (SFP).

The heat removal criteria of the spent fuel pool (SFP) cooling system are that the system should be capable of maintaining the spent fuel pool (SFP) temperature less than or equal to 120°F during normal refueling operations and less than or equal to 150°F during full core discharge situations. The 120°F is not a safety requirement but is a limit set for operator comfort during refueling operations. As discussed in UFSAR Section 9.1.3.4.1.7, it is possible that the 120°F administrative limit may be exceeded for a short period of time at the beginning of a normal refueling off-load. For structural integrity reasons, the pool water temperature is not to exceed 180°F *(Reference 76)*. In order to provide sufficient time to take corrective action in the event of spent fuel pool (SFP) cooling system failure, the pool temperature limit is not to exceed 150° for all modes of operation including a full core discharge.

Normal refueling operations are conducted approximately every 18 months and are defined for the purpose of these criteria as having approximately 45 fuel assemblies (one-third of the core) being removed from the core and placed in the spent fuel pool (SFP).

Full core discharge occurs when all the fuel in the reactor (121 fuel assemblies) is placed in the spent fuel pool (SFP). The full core will be discharged once every 10 years for inservice inspection. Full core discharge may also occur on other occasions when it is deemed necessary, i.e., full core discharges may occur several times during a 10-year inservice inspection interval.

The spent fuel pool (SFP) cooling system consists of three SFP pumps, two installed SFP heat exchangers, associated piping, valves, and hoses. Six (6) spent fuel pool (SFP) cooling loop options, as listed in the Technical Requirements Manual (TRM), are available, provide 100% cooling capability (under normal and refueling operation). These SFP cooling loop options are:

1. SFP Loop A (normal)
2. SFP Loop A (cross-connect)
3. SFP Loop B (normal)

4. SFP Loop B (cross-connect)
5. SFP Standby Loop (cross-connect "A")
6. SFP Standby Loop (cross-connect "B")

The flow path starts at the common suction from the spent fuel pool to the three SFP pumps, through one or more SFP pumps and one or more SFP heat exchangers, and to the spent fuel pool via the common return line. The primary loop (Loop Option #3) is made up of the SFP pump B, SFP heat exchanger B, and piping. The SFP pump B can also supply SFP heat exchanger A (Loop Option #4). Additional backup loops include:

a. the SFP pump A, supplying SFP heat exchanger A (Loop Option #1) or SFP heat exchanger B (Loop Option #2), and piping, and
b. the SFP standby pump, SFP standby heat exchanger supplying SFP heat exchanger A (Loop Option #5) or SFP heat exchanger B (Loop Option #6), and hoses.

There is also the ability to align the SFP Pump A and SFP standby Pump individually to the SFP Heat Exchanger B. Service water (SW) circulates through the shell while SFP water circulates through the tubes of the SFP heat exchangers. There is also the ability to provide fire water for cooling of the SFP heat exchanger A. It should be noted that the standby SFP heat exchanger is not credited within the Technical Requirements Manual (TRM) as evaluated by *Reference 80*. The heat exchanger remains available for use with associated procedures to allow for installation if needed.

SFP Loop B is designed to maintain the spent fuel pool (SFP) water below 150°F with a heat load of 32 x 10^6 Btu/hr. It is also designed to maintain the spent fuel pool (SFP) water below 120°F with a heat load of 16 x 10^6 Btu/hr. Adequacy of the cooling capability for normal and full core off-loads is discussed in Section 9.1.3.4.1. SFP heat exchanger B is sized to remove the safety basis and normal basis heat loads.

SFP heat exchanger A is designed to remove a normal basis heat load of 5.3 x 10^6 Btu/hr with a pool temperature of 120°F and service water (SW) at 80°F. At a 150°F pool temperature, SFP heat exchanger A is capable of removing 7.93 x 10^6 Btu/hr with an 80°F service water temperature. It is capable of removing the normal basis heat load. When SFP Pump A and SFP Standby Pump are operated in parallel with SFP heat exchanger B, they are capable of removing the safety basis heat load.

The ratings of each cooling loop are shown in Table 9.1-3.

Impact of 85°F Lake Temperature

Increasing the maximum lake temperature from 80°F to 85°F results in decreasing the heat removal capability the spent fuel pool (SFP) loops. As discussed in *Reference 45*, the corresponding decrease in the SFP Heat Exchanger A heat removal capability would be approximately 0.65 x 10^6 Btu/hr for each loop. This change results in decreasing SFP heat removal capability by approximately 8% for a 150°F spent fuel pool (SFP) temperature. SFP heat exchanger B was designed for 85°F lake water temperature and has sufficient capacity to maintain SFP temperature below 150°F.

During normal plant operation the decrease in SFP heat removal capability associated with an 85°F maximum lake temperature would result in either an increase in service water (SW) flow to the SFP heat exchangers or a slight increase in SFP temperature. Since SFP temperatures typically are maintained well below 120°F during normal plant operation, any increase in SFP temperature would be acceptable. During a full core off-load refueling outage, the decrease in SFP heat removal capability associated with an 85°F lake temperature would have no impact on SFP temperature. As discussed in Section 9.1.3.4.1.8, the minimum required reactor shutdown time before performing a full core off-load is cycle-specific based upon SFP total heat load and a bounding assessment of expected lake temperature. Consequently, performing a full core off-load coincident with lake temperatures approaching 85°F would result in increasing the required shutdown time prior to initiating the full core off-load.

9.1.3.2 System Design and Operation

9.1.3.2.1 System Design

The spent fuel pool (SFP) cooling system is shown in Drawing 33013-1248.

The spent fuel pool (SFP) cooling system removes residual heat from fuel stored in the spent fuel pool (SFP). The system is normally required to handle the heat load from one-third of the core freshly discharged from the reactor, plus that stored from previous refuelings, and it can safely accommodate the heat load from a full core discharge plus that stored from previous refuelings.

The spent fuel pool (SFP) is located outside the reactor containment and is not affected by any loss-of-coolant accident in the containment. The water in the pool is separated from that in the refueling canal by a removable weir gate. Only a very small amount of interchange of water occurs as fuel assemblies are transferred between containment and the SFP.

The spent fuel pool (SFP) cooling system is described in Section 9.1.3.1. Components are described in Section 9.1.3.3. SFP heat exchanger B is supplied service water (SW) from one section of the service water (SW) loop header. SFP heat exchanger A is supplied service water (SW) from another section of the service water (SW) loop header. Motor-operated valves provide automatic and remote manual isolation of the service water (SW) supply to the SFP heat exchangers and the component cooling water heat (CCW) exchangers. These motor-operated valves are automatically isolated on a safety injection (SI) signal concurrent with an associated 480-V safeguards bus undervoltage condition. Handwheels are provided for manual operation. Radiation detectors, alarms, and recorders are provided to detect radioactivity in the service water (SW) in the event that tube leaks occur in SFP heat exchanger A or SFP heat exchanger B. Normally locked closed manual valves can be opened to align the service water (SW) flow from one section of the service water (SW) loop header to a desired SFP heat exchanger when performing maintenance on another section of the SW loop header.

9.1.3.2.2 System Operation

Operation of the spent fuel pool (SFP) cooling system is manual. Control for all three spent fuel pool (SFP) cooling pumps is from local control stations near the pumps. An electrical interlock prevents simultaneous operation of the spent fuel pool (SFP) pumps A and B, which are supplied from 480-V safeguards buses. SFP pump A is operated from MCC 1C. MCC 1C is supplied from 480-V safeguards bus 14. SFP pump B is operated from 480-V safeguards bus 16. Normally either SFP pump A or SFP pump B is operated alone to maintain the desired pool temperature. The spent fuel pool (SFP) standby pump has a dual train power supply with the ability to be connected to MCC 1C or MCC 1D. The design of the dual train power supply prevents simultaneous connection to both trains, therefore ensuring train separation requirements are met while providing increased redundancy.

Following a safety injection signal, spent fuel pool (SFP) pump B, if operating, will be shed from 480-V safeguards bus 16. After reset of safety injection, spent fuel pool (SFP) pump B can be manually started.

Following a safety injection signal, spent fuel pool (SFP) standby pump, if operating, will be shed from its current power supply, MCC 1C or MCC 1D. After reset of safety injection, the spent fuel pool (SFP) standby pump can be manually started.

Following a safety injection signal with loss of offsite power (1A diesel generator output breaker closed), spent fuel pool (SFP) pump A, if operating, will be shed from MCC 1C. After reset of safety injection and reset of the MCC 1C load shed device, spent fuel pool (SFP) pump A can be manually started.

In the event of spent fuel pump shedding, the cooling of the spent fuel pool (SFP) water will be interrupted and the water temperature will increase as indicated in Section 9.1.3.4.3 until cooling is restored.

The clarity and purity of the spent fuel pool (SFP) water is maintained by passing approximately 60 gpm of the flow through a filter and demineralizer.

In the event of low water level in the spent fuel pool (SFP), a level switch will trip spent fuel pool (SFP) pump B. The switch actuates at elevation 275 ft-11.5 in., which is approximately 2 ft below the top of the spent fuel pool (SFP) and approximately 2 ft above the pump upper suction line. To protect against the possibility of complete loss of water in the spent fuel pool (SFP), the upper suction line penetrates the spent fuel pool (SFP) near the top of the pool. The lower suction line penetrates the spent fuel pool (SFP) approximately 5 ft-4 in. above the top of the fuel racks to preclude the possibility of draining the pool and to ensure a minimum water level of 5 ft-4 in. above the top of the fuel. See Figure 9.1-7

The spent fuel pool (SFP) cooling water return line, which terminates at the bottom of the spent fuel pool (SFP), contains a passive siphon breaker device near the normal spent fuel pool (SFP) water level so that the pool water cannot be siphoned.

9.1.3.2.3 Suction Lineup Using Spent Fuel Pool (SFP) Pump A

When spent fuel pool (SFP) pump A is in operation by itself, it is normally lined up to the upper spent fuel pool suction with the lower suction line isolated. For off normal conditions where the pool level is temporarily lowered (such as maintenance, transfer slot filling evolutions or emergency conditions), operation using the lower suction has been evaluated (*Reference 34*). This configuration has been shown to be acceptable as long as the spent fuel pool is maintained at an elevation greater than 261 ft and the pool temperature is less than 150°F. This evaluation involved a test which showed that spent fuel pool stratification is negligible when SFP pump A is operating on the lower suction.

9.1.3.3 Spent Fuel Pool (SFP) Cooling System Components

Table 9.1-4 lists the spent fuel pool (SFP) cooling system component data.

9.1.3.3.1 Spent Fuel Pool (SFP) Heat Exchangers

The spent fuel pool (SFP) heat exchangers are of the shell and U-tube type with the tubes welded to the tubesheet. Service water (SW) circulates through the shell, and spent fuel pool (SFP) water circulates through the tubes. The tubes are austenitic stainless steel, the shell of SFP Heat Exchanger B is stainless steel, and the shell of SFP Heat Exchanger A is carbon steel.

9.1.3.3.2 Spent Fuel Pool (SFP) Pumps

The spent fuel pool (SFP) pumps circulate water in the spent fuel pool (SFP) cooling system to accomplish the heat removal function. All wetted surfaces of the pumps are austenitic stainless steel or equivalent corrosion resistant material. The pumps are operated manually from local stations. Spent fuel pool (SFP) pump A, if operating, is automatically shed from MCC 1C following a safety injection signal, with a loss of offsite power (1A diesel generator output breaker closed). Spent fuel pool (SFP) pump B, if operating, is automatically shed from 480-V safeguards bus 16 following a safety injection signal. Spent fuel pool (SFP) standby pump, if operating, is automatically shed from its current power supply, MCC 1C or MCC 1D following a safety injection signal.

9.1.3.3.3 Spent Fuel Pool (SFP) Filter

The spent fuel pool (SFP) filter removes particulate matter larger than 5 microns from the spent fuel pool (SFP) water. The filter cartridge is of synthetic fiber and the vessel shell is austenitic stainless steel.

9.1.3.3.4 Spent Fuel Pool (SFP) Strainer

A stainless steel strainer is located at the upper inlet of the spent fuel pool (SFP) suction line for removal of relatively large particles, which might otherwise clog the spent fuel pool (SFP) demineralizer.

9.1.3.3.5 Spent Fuel Pool (SFP) Demineralizer

The demineralizer is sized to pass approximately 60 gpm to provide adequate purification of the fuel pool water for unrestricted area access to the working area, and to maintain optical clarity.

9.1.3.3.6 Spent Fuel Pool (SFP) Skimmer

A skimmer pump and filter are provided for surface skimming of the spent fuel pool (SFP) water.

9.1.3.3.7 Spent Fuel Pool (SFP) Valves

Manual stop valves are used to isolate equipment and lines, and manual throttle valves provide flow control. Valves in contact with spent fuel pool (SFP) water are austenitic stainless steel or equivalent corrosion-resistant material. Motor-operated valves are used to isolate service water (SW) flow to the spent fuel pool (SFP) heat exchangers. They have remote manual control from the control room and close automatically upon coincidence of safety injection and loss of offsite power (normal feed breaker to bus 14/16 open coincident with safety injection).

9.1.3.3.8 Spent Fuel Pool (SFP) Piping

All piping in contact with spent fuel pool (SFP) water is austenitic stainless steel. The piping is welded except where flanged connections are used at the pump, heat exchanger, and filter to facilitate maintenance. The hoses from the SFP Standby Loop to the spent fuel pool (SFP) piping and the service water (SW) piping are styrene butadiene rubber or equivalent.

9.1.3.4 System Evaluation

9.1.3.4.1 Thermal-Hydraulic Analysis

9.1.3.4.1.1 Heat Removal Requirements

The heat removal criteria of the spent fuel pool (SFP) cooling system are given in Section 9.1.3.1 and are that the system should be capable of maintaining the spent fuel pool (SFP) temperature less than or equal to 120°F during typical normal refueling operations and less than or equal to 150°F during full core discharge situations. Depending on the lake temperature and off-load time for a normal refueling outage, spent fuel pool (SFP) temperature may exceed 120°F for a short time period.

9.1.3.4.1.2 Service Water Temperature

The spent fuel pool (SFP) heat exchanger transfers heat from the spent fuel pool (SFP) water to the service water (SW). The service water (SW) system is discussed in Section 9.2.1.

The temperature of the service water (SW) going into the spent fuel pool (SFP) heat exchanger is a controlling factor in determining the heat transfer capability of the spent fuel pool (SFP) cooling system. The service water (SW) temperature is approximately the same as the intake (lake) water temperature except during the winter months when recirculation is used as necessary to moderate the service water temperature.

The intake water temperature has been recorded since December 1969. The data through the end of 1975 showed the following:

1. The instantaneous daily maximum temperature exceeded 80°F three times and then only by a maximum of 2°F.
2. The monthly average of the daily maximum temperatures had not exceeded 75°F.
3. The monthly average of the daily average temperatures had not exceeded 73°F.

The data from 1978 through 1988 confirmed items 1, 2, and 3 above with no occurrence of the daily maximum temperature exceeding 80°F. The service water (SW) temperature to the inlet of the spent fuel pool (SFP) cooling system heat exchanger can therefore be assumed to be 80°F or less. See Section 9.1.3.1 for a discussion of the current maximum service water (SW) temperature.

9.1.3.4.1.3 Analysis of Heat Removal System

The spent fuel pool (SFP) cooling system is described in Section 9.1.3.1. Water is drawn from the spent fuel pool (SFP) by the spent fuel pool (SFP) pump, forced through the heat exchanger, and returned to the spent fuel pool (SFP). The heat exchanger is cooled by the service water (SW). Approximately 60 gpm of the water from the spent fuel pool (SFP) bypasses the heat exchanger and is passed through a demineralizer and filter.

The design capabilities of the spent fuel pool (SFP) cooling system were calculated for 120°F and 150°F (maximum normal pool temperature). For SFP heat exchanger A, a service water (SW) flow of 700 gpm at 80°F was assumed with a spent fuel pool (SFP) outlet flow of 610 gpm and with only 550 gpm flowing through the spent fuel pool (SFP) cooling system heat exchanger. Under these conditions, the heat exchanger, with design fouling, will transfer 5.3 x 10^6 Btu/hr with a spent fuel pool (SFP) outlet temperature of 120°F and 7.93 x 10^6 Btu/hr with a spent fuel pool (SFP) outlet temperature of 150°F. For SFP heat exchanger B, a 1600-gpm service water (SW) flow at 85°F was assumed with a spent fuel pool (SFP) outlet flow of 1200 gpm passing through spent fuel pool (SFP) heat exchanger B. Under these conditions the heat exchanger will transfer 32 x 10^6 Btu/hr with a spent fuel pool (SFP) outlet temperature of 150°F. For SFP heat exchanger B, with a service water (SW) flow of 1000 gpm, pool water temperature of 120°F, pool water flow of 1200 gpm through the heat exchanger, the heat exchanger will transfer 16 x 10^6 Btu/hr. The ratings of each cooling loop are shown in Table 9.1-3.

9.1.3.4.1.4 Cooling Water Flow in Fuel Pool

Water is returned to the spent fuel pool (SFP) from the spent fuel pool (SFP) cooling system heat exchanger through a discharge pipe entering the pool at the west wall. Water enters the spent fuel pool (SFP) cooling system through another pipe located on the south wall. To ensure proper cooling of the fuel assemblies, the discharge pipe is routed along the west wall of the spent fuel pool (SFP). The pedestal and rack baseplate designs provide sufficient cutouts for fluid cooling while ensuring adequate structural strength. Details of the rack baseplate designs are available in *Reference 39*.

9.1.3.4.1.5 Cooling Analysis of Individual Fuel Assemblies

Pre-uprate Analysis

The major cooling mechanism for fuel assemblies stored in the spent fuel pool is natural circulation cooling that is induced by the decay heat generated in the fuel assemblies. The density difference between the hot water in the spent fuel pool racks and the bulk pool water above the racks results in a thermal driving head for establishing water flow through each canister that contains a spent fuel assembly. The water flow through an individual spent fuel canister is determined by balancing the thermal head or buoyancy term for the individual canister with its associated unrecoverable pressure losses due to water flow. The canister thermal head term is determined by the decay heat generation for the stored fuel assembly.

The natural circulation pressure losses occur due to frictional losses in the downcomer flow region, turning losses, canister entrance and exit losses, frictional losses for flow past the fuel assembly, and contraction and expansion losses at a number of locations associated with flow at the base of a spent fuel rack, underneath the spent fuel rack and through the cooling holes contained in the rack pedestals.

Due to the different spent fuel rack designs used in the spent fuel pool (e.g. types 1-4) and differences in decay heat limitations between region 1 and region 2 of the spent fuel pool, a number of individual natural circulation cooling analyses were performed to identify the limiting spent fuel canister location and verify adequacy of cooling water flow. Adequate cooling water flow exists for all spent fuel assemblies by demonstrating that the peak cladding temperature and the exiting cooling water temperature for the limiting fuel assembly are both below the boiling water temperature at the discharge of the spent fuel canister. The boiling water temperature limit is 238.9°F which corresponds to the saturation temperature due to the static head associated with 23 feet of water at a bulk temperature of 150°F.

To verify adequate natural circulation cooling water flow to individual spent fuel assemblies the following configurations were analyzed in *Reference 39*:

1. Type 2 rack in region 2
2. Type 3 rack in region 1
3. Type 4 rack in region 2

For each region and rack type, the cooling water flow for the hottest spent fuel assembly was calculated. The maximum fuel clad and water exit temperature were calculated with the following conservative assumptions:

a. Maximize spent fuel assembly decay heat based on bounding fuel enrichment and burn-up
b. Maximize decay heat based on minimum shutdown time of 100 hours for region 1 fuel assemblies and 60 days for region 2 fuel assemblies (region 2 was also analyzed for a decay heat based on minimum shutdown time of 100 hours in *Reference 74*).
c. Maximize canister water exit temperature by assuming a maximum pool bulk temperature of 150°F.
d. Maximize hot assembly decay heat based upon hot channel peaking factor of $F^N_{\Delta H} = 1.75$

The results of the *Reference 39* local fuel assembly natural circulation cooling analyses identified that the limiting fuel assembly was located in a type 3 canister in region 1. The maximum cladding and exiting water temperature for the limiting type 3 canister are 232°F and 222°F respectively. These results are below the 238.9°F boiling temperature for the water at the top of the spent fuel racks. The type 3 canister location was more limiting than the region 2 racks analyzed due to the significantly higher decay heat used for its hot assembly based upon the 100-hour shutdown assumption. Therefore, the natural circulation cooling results for the type 3 racks bound the results for the other three rack types.

Uprate Analysis

Based on the pre-uprate analysis, the limiting hot assembly cooling analysis is for a Type 3 fuel assembly discharged to spent fuel pool region 1 after a 100-hour shutdown time. At 1520 MWt the maximum temperature leaving the hot assembly is 222°F which is below the 238.9°F boiling temperature for water at the top of the spent fuel pool rack. At uprate with a core power level of 1775 MWt, the heat load in the hot assembly at 100 hours after a shut-down would increase proportionally with the change in power level or ~17% (1775 MWt / 1520 MWt). Conservatively ignoring the increase in mass flow through the hot assembly due to the increased heat load, a 17 % increase in assembly decay heat would result in a corresponding 17 % increase in temperature rise across the fuel assembly. The water inlet temperature for the existing hot assembly analysis is 150°F which results in a temperature rise of 72°F for the pre-uprate analysis. Therefore, at uprate the 17 % increase in core power would conservatively result in a hot assembly temperature rise of ~ 85°F (1.17 x 72°F). The resulting water temperature leaving the hot assembly spent fuel pool rack would be ~235°F. Since this temperature is still below the boiling temperature of the water at the spent fuel pool rack location, adequate cooling water is provided to the hot assembly at uprate. In reality, the actual temperature rise across the hot assembly would be less than 85°F. The increase in water outlet temperature results in increased thermal buoyancy in the hot assembly. This increased buoyancy result in a higher mass flow through the hot assembly at uprate and correspondingly a lower water temperature rise.

The increase in water exit temperature leaving the spent fuel rack would also result in an increase in the maximum cladding temperature. The pre-uprate analysis determined that the peak cladding temperature would be approximately 232°F or 10°F higher than the water exit temperature. With the ~13°F increase in water exit temperature at uprate, the maximum clad temperature would also be expected to increase a comparable amount. This would result in the peak clad temperature exceeding the water saturation temperature which would result in the onset of boiling heat transfer. The transition to pool boiling heat transfer from single phase heat transfer would result in an increase in the heat transfer coefficient between the cladding and the bulk water. This would result in a decrease in the required temperature differential between the water and cladding to remove the cladding decay heat. Therefore, the increase in cladding peak temperature would be smaller than the increase in water temperature. The transition to boiling heat transfer could cause the peak cladding temperature to be slightly higher than the water saturation temperature. Since the peak temperature is only slightly higher than the water saturation temperature, adequate cooling of the cladding for the hot assembly at uprate is still being maintained.

9.1.3.4.1.6 Cooling Analysis of Consolidated Fuel Canisters

In addition to the local fuel assembly cooling analysis discussed in Section 9.1.3.4.1.5, *Reference 39* also re-evaluated natural circulation cooling of fuel canisters that contained consolidated fuel rods. The re-analysis assumed that the limiting consolidated canister contained fuel rods from two fuel assemblies (358 fuel rods) with a minimum decay time of 5 years. To maximize decay heat generation, the fuel rods were assumed to have a burn-up of 60 GWD/ MTU. The consolidated canister was located in the most limiting position in region 1 of the spent fuel pool (SFP). The resulting peak cladding and water outlet temperatures are 231°F and 222°F, respectively, which are both below the 238.9°F boiling temperature limit at the top of the spent fuel racks. This analysis was based on a fuel assembly decay heat load of 4335

BTU/hr (*Reference 74*) which exceeds the Ginna Technical Specifications Section 4.3.1.1 limit of 2150 BTU/hr for a fuel assembly to be stored in a consolidated canister.

9.1.3.4.1.7 Normal 1/3 Core Off-Load Cooling Capability

For a normal refueling outage, approximately 1/3 of the core is off-loaded to the spent fuel pool (SFP). SFP cooling is typically provided by operating one SFP cooling loop with 100% back-up capability being available by having another SFP cooling loop operable, as defined in the Technical Requirements Manual (TRM). The available SFP cooling system is required to maintain the bulk SFP temperature below 150°F. The actual heat load from irradiated fuel assemblies stored within the SFP is a variable based on the total number of assemblies stored, the power history of the individual assemblies, and the time since the assemblies were last irradiated. The actual heat removal capabilities of each SFP cooling loop are also variables, and the heat removal capabilities are determined by Nuclear Engineering Services based on plant conditions. For a 150°F SFP bulk temperature, SFP Loop B is capable of removing 32 x 10^6 Btu/hr with a service water temperature of 85°F. Although SFP heat exchanger A is required to remove 7.93 x 10^6 Btu/hr at 150°F SFP temperature, actual heat removal capability for SFP heat exchanger A at this temperature is significantly higher than the design requirement. As discussed in *Reference 43*, the heat exchanger is capable of removing a minimum of 9.3 x 10^6 Btu/hr for a 150°F SFP temperature and an 80°F service water temperature.

A normal refueling outage occurs approximately every 18 months and is based upon removal of approximately one-third of the 121 reactor core fuel assemblies. For the uprate operating conditions, the typical number of fuel assemblies to be discharged is expected to alternate between 44 and 45 fuel assemblies per outage. For normal core refueling in 2029 at the expiration of present operating license, the decay heat was calculated based upon 45 fuel assemblies irradiated for 18 months at a power level of 1811 MWt and off-loaded 100 hours after a reactor shutdown. This decay heat load was calculated by the ORIGEN2 computer program (*Reference 40*) based upon bounding assessments of fuel burn-up and a reactor power of 102% of rated full EPU power and corresponds to approximately 9.0 x 10^6 Btu/hr. ORIGEN2 has previously been used for the Ginna Re-racking Licensing Amendment Request in 1998. In addition, the background heat load in the SFP is based upon the assumption that 1200 previously discharged fuel assemblies are resident in the spent fuel pool with a combined decay heat of 3.9 x 10^6 Btu/hr. The combined decay heat for previously discharged assemblies is based upon the actual fuel assembly discharge history for all refueling outages prior to 1997 and the offloading sequence described above. Presently, the actual number of spent fuel rack storage positions installed in the SFP is 1321. Due to the present inventory of stored fuel assemblies, Ginna plans to implement on-site dry cask storage around 2009 to accommodate the off-loads out through 2029. Therefore, to maximize the pool residual heat load from the existing 1321 storage positions a bounding full core off-load in 2029 was assumed. The off-load analysis assumed completely filling all 1321 available storage locations. The previously discharged fuel was assumed to be the most recent spent fuel available to completely fill the pool. This assumes that fuel assemblies being stored in dry casks are the oldest fuel assemblies with the lowest decay heat rates.

Therefore, for a normal 45 fuel assembly core refueling outage, a bounding estimate of the total spent fuel pool heat load at the time of core off-load is 12.9 x 10^6 Btu/hr.

Since the total heat load for a normal 45 fuel assembly off-load 100 hours after the reactor shutdown is well below the available heat removal capability of SFP heat exchanger B and the available heat removal capability of either SFP pump A or the standby SFP pump cross connected to SFP heat exchanger B, the maximum expected spent fuel pool temperature would be below the 150°F limit.

Although a 1/3 core discharge to the spent fuel pool after a shutdown time of 100 hours would not approach the 150°F pool operational limit, the bounding off-load at 100 hours could cause the SFP temperature to exceed the 120°F value identified in Section 9.1.3.4.1 as the nominal upper bound for a normal refueling outage. The actual pool temperature for a 1/3 core off-load scenario would be dependent upon actual service water temperature as well as actual shutdown time to off-load and total decay heat inventory of the SFP. For the bounding 1/3 core off-load initiated at 100 hours time, the time period that the pool temperature could exceed 120°F would be small due to the continued reduction with time of the off-loaded fuel decay heat.

9.1.3.4.1.8 Full Core Off-Load Cooling Capability

For a full core off-load scenario spent fuel pool (SFP) cooling is provided by six (6) SFP cooling system options, as listed in the Technical Requirements Manual (TRM). These SFP cooling system options are:

1. SFP Loop A (normal)
2. SFP Loop A (cross-connect)
3. SFP Loop B (normal)
4. SFP Loop B (cross-connect)
5. SFP Standby Loop (cross-connect A)
6. SFP Standby Loop (cross-connect B)

The Technical Requirements Manual (TRM) specifies the minimum required time after shut-down for each of the six (6) options listed above. The available SFP cooling system is capable of maintaining the bulk SFP temperature below 150°F as required by the Technical Requirements Manual (TRM). Use of the 150°F pool temperature limit provides 30°F of margin to the spent fuel pool structural integrity design limit of 180°F.

Consideration of a single passive failure for SFP cooling is not required by the NRC Standard Review Plan (SRP) (*Reference 46*) and is not a part of the licensing basis for the Ginna Station SFP cooling system. Per Section 9.1.3 of the SRP, for the “maximum normal heat load” (of the spent fuel pool) there should be suitable redundancy of components so that safety functions can be performed assuming a single active failure of a component coincident with the loss of offsite power. For the “abnormal maximum heat load” (defined in the SRP as full core unload), a single active failure need not be considered. The NRC

Systematic Evaluation Program (SEP) evaluated Ginna Station's conformance to the SRP. Ginna Station's conformance to SRP Section 9.1.3 is addressed in *Reference 47* and *Reference 48*.

The minimum amount of time required after plant shutdown before the full core can be off-loaded is a function of the available SFP heat removal capability. Since cooling is provided by service water from Lake Ontario, the SFP heat removal capability is a function of lake temperature. The allowed combinations of lake temperature (as measured by screen house bay water temperature), time after shutdown, and SFP cooling system options, are cycle/outage specific and these combinations need to be re-evaluated for each full core off-load. The current requirements are listed in the Technical Requirements Manual (TRM).

The Technical Requirements Manual (TRM) indicates that the required time after shutdown prior to initiating a full core off-load is strongly dependent upon the available lake temperature. Use of an 85°F lake temperature for assessing full core off-loads during a typical Ginna fall or spring outage associated with an eighteen-month fuel cycle may require delaying core off-load for an extended period. Consequently, for performing full core off-loads, each off-load scenario is conservatively evaluated on a case by case basis to identify the minimum required time after shutdown.

The minimum time after shutdown before initiating a full core off-load is determined by a conservative assessment of an upper bound lake temperature for the actual time of year and a review of historic lake temperature data. The total pool background decay heat is also conservatively assessed for the actual loading of spent fuel in the pool and operating characteristics of the off-loaded fuel. The performance of the available spent fuel pool cooling system is conservatively assessed based upon the latest heat exchanger thermal performance test results along with the actual available heat exchanger surface area (e.g. current plugging level of the SFP heat exchangers). These parameters are used to determine both the available heat removal cooling capability at a 150°F pool temperature as well as the total spent fuel pool heat load as a function of shutdown time. Full core off-loads are not initiated until the spent fuel pool decay heat with the off-loaded core is less than the heat removal capability of the available cooling systems.

Additionally, as required by *Reference 38* for any full core off-load, the cycle specific limits for lake temperature and time after shutdown will be specified by the Technical Requirements Manual (TRM). The Technical Requirements Manual (TRM) also requires the availability of a 100% back-up SFP cooling heat removal system.

9.1.3.4.2 Leakage Provisions

Whenever a leaking fuel assembly is transferred from the fuel transfer canal to the spent fuel storage pool, a small quantity of fission products may enter the spent fuel cooling water. A small purification loop is provided for removing these fission products and other contaminants from the water. Radiation monitors detect leakage of spent fuel pool (SFP) water from the spent fuel pool (SFP) heat exchangers A or B into the service water (SW) system (see Section 11.5.2.2.13).

The probability of inadvertently draining the water from the spent fuel pool (SFP) cooling system is exceedingly low. The only means is through an action such as opening a valve on the cooling line and leaving it open when the pump is operating. In the unlikely event of water being drained from a portion of the SFP cooling system, the spent fuel storage pool itself cannot be drained, and no spent fuel is uncovered since the spent fuel pool (SFP) cooling connections enter near the top of the pool. Although the spent fuel pool (SFP) cooling pump discharge piping discharges near the bottom of the storage racks, a 1/4-in.-diameter drilled hole in the discharge piping, located approximately 18 in. below the normal spent fuel pool (SFP) water level, anti-siphoning protection for events such as a mispositioned valve or a cracked pipe. At the same depth below the surface, a 1-1/2 in. diameter vent pipe has been added to allow air to enter at the highest point of the discharge pipe and is sized to accommodate the worst-case scenario of a double-guillotine pipe break, thereby precluding lowering of the water level and uncovering the spent fuel assemblies. The temperature and level indicators in the spent fuel pool (SFP) would warn the operator of the loss of cooling. The slow heatup rate of the spent fuel pool (SFP) would allow sufficient time to take any necessary action to provide adequate cooling while the cooling capability of the spent fuel pool (SFP) cooling system is being restored.

9.1.3.4.3 Interruption of Spent Fuel Pool (SFP) Cooling

For plant uprate to 1775 MWt, the impact of a complete loss of spent fuel pool (SFP) cooling capability was evaluated. The loss of SFP cooling was assumed to occur 100 hours after reactor shutdown immediately following a full core off-load with an initial SFP temperature of 150°F and SFP Loop B in operation. Since the shutdown time required for a full core off-load is a function of lake temperature as discussed in Section 9.1.3.4.1.8, SFP heat-up was analyzed for lake temperatures of 40°F, 60°F and 80°F.

The SFP heat-up analysis conservatively neglected any cooling associated with heat transfer to the SFP concrete walls, convective cooling to the ambient air and any evaporative cooling from the pool surface. The analysis took credit for the thermal inertia of the SFP water as well as the thermal inertia associated with the SFP racks, stored fuel assemblies and the SFP steel liner. The SFP heat load was assumed to be constant over the duration of the SFP heat-up. The SFP heat load assumed for the heat-up analysis is listed in Table 9.1-6 for the three lake temperatures analyzed. For the 40°F, 60°F, and 80°F lake temperature, the heat load assumed is equal to the heat removal capability of the SFP cooling system at a 150°F pool temperature.

Since the SFP structural design basis temperature is 180°F, the heat-up time from 150°F to the 180°F limit was analyzed for heat-up rates based on the operation of SFP heat exchanger B. Heat-up rates were calculated for lake temperatures of 40°F, 60°F and 80°F. The results of these analyses assuming a complete loss of cooling are presented in Table 9.1-6. Heat up times to 212°F have also been calculated, though the 180°F heat-up times are controlling for Ginna since they correspond to the SFP design temperature of 180°F. These heat-up rates to 212°F and corresponding boil-off rates are also listed in Table 9.1-6. The heat-up rates to 212°F are conservative because they do not include any operator action to restore cooling from an alternate source.

A makeup water flow rate of 60 gpm can be made available from the refueling water storage tank in less than 15 minutes. This make-up rate exceeds all of the boil-off rates listed in Table 9.1-6. As a back-up, 50 gpm of water from the chemical and volume control system hold up tanks can also be made available in approximately 15 minutes. For cases where the initial boil-off rate exceeds the back-up make-up source of 50 gpm, the mass imbalance would result in a maximum inventory loss from the SFP of less than 2 inches (*Reference 75*). Ginna Technical Specifications (T.S.) 3.7.11 requires that the water level be greater than or equal to 23 feet above the top of the fuel assemblies in the SFP. Additionally, T.S. 3.7.11 requires stopping any fuel movement activities if the SFP level falls below 23 ft. Since the potential decrease in SFP inventory following a loss of cooling event is small, the resulting loss in inventory for the bounding case was determined to be acceptable (*Reference 68*).

The use of redundant components in the SFP cooling system provides adequate protection against primary component failures. Valves are also provided for isolating individual branch connections, pumps and heat exchangers should they fail. The existence of the installed SFP cooling systems and the availability of the SFP Standby Loop and other options as listed in the Technical Requirements Manual (TRM) provide assurance that SFP cooling can be restored should SFP cooling be rendered inoperable during a safety basis heat load scenario.

Following a complete loss of SFP cooling, operator actions would be taken to restore SFP cooling system capability, using one or more of the options as listed in the Technical Requirements Manual (TRM). The SFP cooling system can be aligned to establish cooling prior to the SFP heat-up to 180°F.

Following a complete loss of SFP cooling due to a loss of SFP heat exchanger B during a full core off-load, previous evaluations credited the availability of SFP heat exchanger A and SFP standby heat exchanger to be aligned prior to reaching the structural temperature of the pool. *Reference 80* and *Reference 81* re-evaluated the loss of SFP heat exchanger B and determined the most limiting single active failures that cause a loss of the heat exchanger are a failure of a service water (SW) supply isolation Motor Operated Valve (MOV) in the closed position and a loss of a SFP pump. In contrast to the previous evaluation that credited manual action to align other heat exchangers, manual action is credited for opening the failed closed valve to restore cooling to SFP heat exchanger B and aligning another SFP pump to SFP heat exchanger B, respectively. In the event of either failure, sufficient time (2.4 hours per *Reference 81*) would be available to perform the procedurally directed actions to ensure the structural temperature limit of the pool would not be exceeded. An analysis was performed to demonstrate the time to boil if there is a complete loss of cooling to the SFP. A makeup water flow rate of 60 gpm can be made available from the refueling water storage tank in less than 15 minutes. As an alternative, 50 gpm of water from the chemical and volume control system hold up tanks can also be made available in approximately 15 minutes such that cooling the SFP by adding makeup water during an unlikely event of a complete loss of SFP cooling conforms with the guidance described in the NRC Standard Review Plan (SRP), (*Reference 46*); therefore, it is acceptable. The results of this analysis are documented in *Reference 38*, where the NRC concludes that cooling the SFP by adding makeup water during an unlikely event of a complete loss of SFP cooling conforms with the guidance described in the NRC Standard Review Plan (SRP) (*Reference 46*); therefore, it is acceptable. To conform to this conclusion, the maximum SFP heat load is limited to 25.67 x 10^6 Btu/hr (*Reference 81*). The SFP cooling capability provided is capable of maintaining SFP temperature below the 180°F structural design limit for the SFP.

It should be noted that a Standby SFP heat exchanger is available for use for beyond design basis events where additional heat removal capacity is required. The Standby SFP heat exchanger was originally designed to have a heat removal capacity equivalent to SFP heat exchanger A. The Standby SFP heat exchanger is not permanently installed and requires hose installation to allow use. The Standby SFP heat exchanger is no longer credited within the Technical Requirements Manual (TRM) and was removed from the Technical Requirements Manual (TRM) due to the evaluation within Reference 80.

9.1.3.5 Minimum Operating Conditions

The spent fuel storage pool is normally limited to 120°F except in unusual circumstances as previously described. Boric acid concentration in the pool fluid must be maintained at a minimum of 2300 ppm.

9.1.4 FUEL HANDLING SYSTEMS

The fuel handling systems provide a safe and effective means for transporting and handling reactor fuel from the time the fuel reaches the plant in an unirradiated condition until it leaves the plant after post-irradiation cooling. The fuel handling systems can be divided into the two categories of fuel storage and fuel handling. Fuel storage is discussed in Sections 9.1.1 through 9.1.3. The Independent Spent Fuel Storage Installation (ISFSI) will serve as a facility for the interim storage of sealed, leak proof, and self-contained dry shielded canisters holding spent fuel. The fuel handling category covers the facilities other than storage, equipment, and tools used to refuel the reactor and are discussed below.

The existing fuel handling equipment is too long and incapable of transferring spent fuel assemblies from storage locations into the ISFSI Dry Shielded Canister. A new shorter Spent Fuel Cask Loading Tool (SFCLT) (see Figure 9.1-11) was designed and fabricated. A SFCLT storage bracket was installed in the Spent Fuel Pool.

As a result of using Westinghouse 422Vantage+ fuel assemblies, the following changes were made to existing fuel handling tools:

- Replace refueling machine gripper with any equivalent gripper design that is compatible with both the existing Ginna fuel assembly types and the new 422V+ design.
- Obtain new spent fuel and new fuel handling tools that are compatible with both the existing Ginna fuel assembly types and the new 422V+ design.
- Adjust RCCS stop on fuel handling car to prevent interference when transferring new 422V+ fuel from containment and the SFP.
- Modify portable RCCS tool to be compatible with both the existing Ginna fuel assembly types and the new 422V+ design.

9.1.4.1 Reactor Cavity

The reactor cavity is a reinforced-concrete structure that forms a pool above the reactor when it is filled with borated water for MODE 6 (Refueling). The cavity is filled to a depth that limits the radiation at the surface of the water to 50 mR/hr during those brief periods when a fuel assembly is transferred over the reactor vessel flange. The reactor vessel flange is sealed to the bottom of the reactor cavity by a cam-lock seal ring, which prevents leakage of refueling water from the cavity. This seal is installed after reactor cooldown but prior to flooding the cavity for MODE 6 (Refueling) operations. The previously used inflatable seal is available in the event it is needed. The cavity is large enough to provide storage space for the reactor upper and lower internals, the rod cluster control drive shafts, and miscellaneous refueling tools.

The likelihood of reactor cavity seal failure and its consequences were analyzed in response to IE Bulletin 84-03 (*References 16* and *17*). It was concluded that seal failure, such as occurred at Haddam Neck, is improbable due to differences in both plant and seal design. Since the cam-lock seal was developed after IE Bulletin 84-03, the design is different than that used at Haddam Neck. However, should a postulated gross seal failure occur, operator action is necessary to prevent fuel uncovery only in the event that a fuel assembly is in the manipulator crane mast at the time of the event. Procedures direct the crane operator to take appropriate action to protect the fuel assembly from overheating. For all other cases, the fuel remains covered, although some water shielding will be lost. High-radiation fields, which could result, will have negligible effect on the general public.

To maintain clarity of the water in the reactor cavity during MODE 6 (Refueling), a reactor cavity filtration system has been installed, which consists of a 600-gpm centrifugal pump and four filters.

9.1.4.2 Refueling Canal

The refueling canal is a passageway extending from the reactor cavity to the inside surface of the reactor containment. The canal is formed by two concrete shielding walls which extend upward to the same elevation as the reactor cavity. The floor of the canal is at a lower elevation than the reactor cavity to provide the greater depth required for the fuel transfer system upender and the rod cluster control changing fixture located in the canal. The transfer tube enters the reactor containment and protrudes through the end of the canal. Canal wall and floor linings are similar to those for the reactor cavity.

9.1.4.3 Fuel Handling Equipment

9.1.4.3.1 Auxiliary Building Crane

The auxiliary building crane is used in moving the new fuel assemblies into and out of their storage area and in the movement of the spent fuel shipping cask. The crane is electrically interlocked to prevent movement over the spent fuel storage racks. These interlocks may be defeated by keys, and when defeated, indicate the condition by rotating flashing yellow or red lights. When in this condition, the crane operator is responsible to observe the following restrictions: (1) A load in excess of one fuel assembly and its handling tool may not be carried over storage racks containing spent fuel unless control of the heavy load adheres to the guidance provided in NUREG-0554 and NUREG-0612 for assuring that the potential for a load drop is "extremely small," through the use of a single-failure-proof crane and redundant rigging and attachment points; and (2) the restriction in (1) shall not apply to the movement of canisters containing consolidated fuel rods if the spent fuel racks beneath the transported canister contain only spent fuel that has decayed at least 60 days since reactor shutdown. The weight of one standard fuel assembly and its handling tool is 2000 lb. The weight of a fully loaded canister is approximately 2300 lb.

The auxiliary building crane main hoist meets the single-failure criteria of NUREG 0554 and NUREG 0612. The main hoist is rated at 32.5 tons; however, the maximum critical load is 30 tons, which is characterized as a fully loaded single-element spent fuel cask with a redundant yoke. No single failure of any lifting component of the main hoist will result in the drop of any load up to 32.5 tons. The main hoist is used for handling heavy loads such as the spent fuel cask. The main hoist is capable of stopping and holding the load under all conditions, including the safe shutdown earthquake. The 5-ton auxiliary hoist, which does not meet the single-failure criteria of NUREG 0554 and NUREG 0612, is used to handle new fuel assemblies and canisters. Administrative procedural controls and the Technical Requirements Manual restrict the use of the auxiliary building crane when handling heavy loads.

9.1.4.3.2 New Fuel Elevator

The new fuel elevator is a box-shaped carriage (sized to contain a single fuel assembly), which rides on a track mounted to the spent fuel pool (SFP) wall. It is used to lower new fuel assemblies into the spent fuel pool (SFP). It is also utilized under special administrative controls to perform fuel reconstitution activities on spent fuel assemblies. The elevator is not interlocked, but the button must be continuously depressed for up or down movement. An electric winch is used as the motive power for the elevator and is controlled from the spent fuel pool bridge control panel located on the operating floor level.

9.1.4.3.3 Spent Fuel Pool (SFP) Bridge

The spent fuel pool (SFP) bridge is a wheel-mounted walkway which spans the spent fuel pool (SFP) in the north-south direction. It carries two electric motor driven monorail hoists on overhead structures which may be manually positioned along the walkway. Fuel assemblies are moved within the pool by means of a long-handled tool (spent fuel handling tool) suspended from the hoist in service. The hoist travel and tool length limit the maximum lift of a fuel assembly, thus maintaining a safe shielding depth of water above the fuel. A hoist load cell used between the hoist and motorized trolley allows for a constant check on fuel assembly load conditions. The monorail hoists and trolleys are rated at 1 ton. Rod Cluster Control Assemblies (RCCAs) are moved within the pool by means of another long-handled tool (Portable RCCA Tool).

9.1.4.3.4 Fuel Transfer System

The fuel transfer system is an underwater, variable speed cable/winch driven conveyor car that runs on tracks extending from the spent fuel pool (SFP) through the transfer tube and into the containment refueling canal. The conveyor car basket receives a fuel assembly in the vertical position, is lowered to the horizontal for passage through the fuel transfer tube, and is raised to the vertical for removal of the fuel assembly. Conveyor car motion and the upending and lowering functions are controlled from panels on the operating floor.

Two electric winches are mounted above the water in the Auxiliary Building at the South end of the transfer slot. The winches are electrically interlocked such that when one winch is energized, the counter-torque of the other hoist is energized, maintaining cable tension at all times. Reeved and attached to the conveyor car, the hoist moves the transfer conveyor back and forth between the Spent Fuel Pool (SFP) upender and the containment refueling canal upender. The conveyor is capable of speeds up to 80 fpm.

The conveyor car winches are provided with load sensitive bases, which are used to de-energize the winches if the cable tension builds up to a preset limit, thereby providing overload protection during operation. The load sensitive bases also function to protect from a preset underload limit or slack cable condition.

The conveyor car winch, which pulls the conveyor car into the Auxiliary Building, is equipped with a resolver, which provides both the Auxiliary Building and Containment Refueling Canal stops and slow zones. As a redundancy feature, the Programmable Logic Controller (PLC) monitors a resolver mounted on the reactor conveyor winch and compares the positions reported by the two resolvers to ensure that they are within a reasonable delta.

The fuel assembly basket is pin-hinged to the conveyor car to permit tipping it to a vertical position for fuel assembly loading and unloading. The basket engages with an upending frame at either end of its travel, and when the frame is raised with an electric winch on the operating floor, the basket and fuel assembly are raised with it. The lifting frames are interlocked such that the conveyor car must be at the end of its travel before the frame will operate. An interlock is also provided to prevent car motion unless the fuel assembly basket is in the DOWN position. Note that the gate valve interlock is not currently functional and is permanently jumpered out. Conveyor car motion is administratively controlled by refueling procedures. The conveyor car is also provided with an emergency cable which allows the car to be retrieved from the transfer tube should a cable break or the motor fail.

The fuel transfer tube through which the conveyor car runs may be closed on the spent fuel pool (SFP) side with a gate valve, manually operated by means of a reach rod from the operating floor. A blind flange is provided for the containment side. During normal plant operations the conveyor car is stored in the spent fuel pool (SFP), the gate valve seals off the reactor containment, and the blind flange seals the containment side of the tube.

9.1.4.3.5 Manipulator Crane

The manipulator crane transfers fuel assemblies within the core and between the core and the fuel transfer system conveyor car. It is a rectilinear bridge and trolley crane with a vertical mast, which extends down into the refueling water. The bridge spans the reactor cavity and runs north-south on rails set along the cavity edge. The trolley traverses east-west along the bridge. This allows exact positioning of the mast above any fuel assembly.

A long gripper tube with a pneumatic gripper assembly on the end is lowered down from the mast to grip the fuel assembly. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. While inside the mast tube, the fuel is transported to its new position. The outer mast is mounted on a support bearing that allows rotation of the mast about its centerline to allow proper alignment of the fuel.

All controls for the manipulator crane are located on a console on the trolley. The drives for the bridge, trolley, and winch are all variable speed. In an emergency, the drives may be operated manually using a handwheel on the motor shaft.

The suspended weight on the gripper tool is monitored by electric load cells with an indicator mounted on the console. A load in excess of 150lbs greater than a standard fuel assembly with RCCA insert, stops the winch drive in the up direction. A load in excess of 150lbs less than a standard fuel assembly, stops the winch drive in the down direction. The load cells also sense a slack cable condition. Without slack cable indication an interlock prevents the control system from signaling the gripper to disengage fuel. This interlock is backed up by a mechanical weight actuated lock in the gripper which prevents gripper operation even if air pressure is applied, thereby preventing opening when a fuel assembly is being supported.

The Manipulator Crane has an advanced control system which monitors mast position via a series of encoders. A set of redundant encoders is provided for all axis of motion (Bridge, Trolley and Winch). Should two encoders in a set differ by a given amount all crane motion is locked out. The redundant encoders feed into the manipulator crane PLC based control logic. The crane position is utilized by the PLC to provide three modes of control. Manual mode during which all control is via operator input thru joystick, Semi-Auto during which the bridge and trolley are simultaneously controlled by the PLC based control logic system after the operator inputs the desired destination and Automatic during which the bridge and trolley are simultaneously controlled by the PLC based control logic system after the destination is provided by the fuel shuffle program uploaded into the control system and verified by the operator. During all modes of operation the manipulator crane winch is controlled in manual.

There are several other safety features or interlocks incorporated in the manipulator crane:

1. Travel of the manipulator crane is limited to “safety to travel” zones.

2. The winch is interlocked with the bridge and trolley to prevent motion in any other axis while the winch is in motion.

3. An interlock within the PLC based control system permits the hoist to be operated only when either the Open or Closed indicating switch on the gripper is actuated. An alarm will occur if the gripper remains in a transition zone between Open or Closed in excess of a set duration of time.

4. Several zones of winch operation exist which prevent operation of the bridge and trolley once the fuel has entered the core on index, restrict the winch to slow motion during fuel transitions, etc.

5. The manipulator crane is designed so as not to drop a fuel assembly or to fall into the cavity during a safe shutdown earthquake.

The manipulator crane was modified in 1997 to accept a skid mounted sipping system for in-mast fuel assembly leakage examinations. The permanent aspects of this modification include suction connections at the fuel gripper and associated hoses routed through the mast to a dedicated location on the mast support structure.

9.1.4.3.6 Reactor Vessel Head Lifting Device

The reactor head lifting device shown in Figure 9.1-10 consists of a welded and bolted structural steel frame with suitable rigging for removal and storage of the reactor head for servicing and replacing the reactor internal components.

The device, an annular ring girder, is permanently attached to three lifting lugs which are an integral part of the reactor head through three lug assemblies. The legs are pinned and held fastened by a jam nut. They extend vertically upward and pass through the girder and a circular platform assembly to which they are based and welded. The upper portion of the lift rig was modified by PCR 2001-0042 during the Fall 2003 refueling outage, to serve as a portion of the control rod drive cooling system ductwork.

The platform elevation is such that it encloses the top of the control rod drive housing shroud and thus provides access to and allows maintenance on the rod drives and position indication equipment. A removable handrail is placed around the outer periphery of the platform. Located around the outside bottom portion of the ring girder is an I-beam. The beam acts as a support and monorail for the stud tensioners used during the refueling procedure. An adjustable tripod lifting sling is pinned to the hook of the crane and lowered over the platform assembly of the lifting rig. Each leg of the sling is positioned for alignment by its turnbuckle and attached to the top of the three legs of the lifting rig in the same manner as to the head.

The lifting sling must be removed and stored on the operating deck when not in use. Permanent installation would impede placement of the missile shield located above the control rod drive housing.

9.1.4.3.7 Reactor Internals Lifting Device

The internals lifting rig is a structural frame device used to handle the upper and lower reactor vessel internal packages. The rig consists of a sling assembly, spreader assembly, leg assembly, and support ring.

The rig is suspended from the main crane hook. The alignment of the rig, with respect to the internals lift points, is obtained through bushings attached to the support ring which fit over the reactor vessel guide studs. When the rig is not in use, it is normally stored on the upper internals storage stand.

The rig is placed on the reactor vessel after the vessel head is removed and is bolted to the internals. The rig normally remains on the internals until replacement in the reactor vessel after the refueling operations are complete and the vessel head is to be reinstalled.

A tension or load cell is provided with the rig as part of the sling assembly. Its purpose is to monitor the load during all handling operations. Any deviations from the established normal load readings will indicate interference or binding and allow corrective action to prevent damage.

9.1.4.3.8 Rod Cluster Control Assembly Changing Fixture

The rod cluster control assembly fixture is used to remove, install, and temporarily store rod cluster control assemblies. It is located in the refueling canal near the containment terminus of the fuel transfer system. It is made up of a guide tube section, a carriage section, a frame and track section, and a gripper and drive mechanism.

The wheel-mounted carriage section rides on a track anchored to the refueling canal floor. It is made up of three compartments. The center and one end compartment will accept fuel assemblies while the other section will accept only a rod cluster control assembly.

The guide tube is mounted on the refueling pool wall above the carriage and acts to maintain alignment of the rod cluster control assembly rodlets when the pneumatic gripper raises and lowers them as required. All positioning of the carriage and operation of the gripper is accomplished from the operating floor. The fuel assemblies are moved into and out of the fixtures with the manipulator crane.

9.1.4.3.9 Upper Internals Storage Stand

The upper internals storage stand is a structural stainless steel fixture installed in the refueling cavity and is used to support the upper internals package when removed from the reactor vessel. The construction of the upper internals does not permit them to be supported from the bottom. During MODE 6 (Refueling) the stand is underwater.

9.1.4.3.10 125 Ton Cask Handling Crane

The new 125 ton single failure proof cantilevered gantry crane has a stationary runway mounted to an embedded steel support system. A rolling bridge is mounted on top of the stationary runway. A main trolley is mounted on the rolling bridge with a flying trolley mounted to the main trolley. The crane is designed to safely transport a Transfer Cask with a fully loaded Dry Shielded Canister from the Spent Fuel Pool to the Self-Propelled Modular Transporter located in the Canister Preparation Building.

9.1.4.4 Fuel Handling/Refueling Tools

The fuel handling tools are used to prevent close operator exposure to the fuel. Several other specialized tools are also available to aid the operators in performing specific MODE 6 (Refueling) functions.

9.1.4.4.1 New Fuel Assembly Handling Tool

The new fuel assembly handling tool, shown in Figure 9.1-11, is used to lift and transfer fuel assemblies from the new fuel shipping containers to the new fuel storage racks. The tool is also used to transfer fuel assemblies from the new fuel storage racks to the new fuel elevator. The tool employs four cam-actuated latching fingers which grip the underside of the fuel assembly top nozzle. The operating handles to actuate the fingers are located on the side of the tool. With the fingers latched, the operating handles are in the DOWN position; with the fingers unlatched, the operating handles are in the UP position. When the fingers are latched, the safety locking device on the side of the tool is turned in to prevent the accidental unlatching of the fingers.

9.1.4.4.2 Spent Fuel Handling Tool

The spent fuel handling tool shown in Figure 9.1-11, also called the long-handled tool, is used to manually handle and inspect new and spent fuel in the spent fuel pool (SFP) and to move fuel to and from the fuel transfer system conveyor car. Its operation is similar to that of the new fuel handling tool. The spent fuel assembly handling tool employs four cam-actuated latching fingers which grip the underside of the fuel assembly top nozzle. The operating handle to actuate the fingers is located at the top of the tool. With the operating handle in the DOWN position, the fingers are latched; with the handle in the UP position, the fingers are unlatched. Once the fingers are latched, insertion of a pin in the operating handle prevents the fingers from being accidentally unlatched during fuel handling operations.

9.1.4.4.2.1 Spent Fuel Cask Loading Tool (SFCLT)

The SFCLT is used to latch onto an assembly and transfer it between spent fuel pool rack locations and to the racks within the Dry Shielded Canister. Operation of the tool is similar to the Spent Handling Tool (Section 9.1.4.4.2), although the SFCLT is 55 inches shorter in overall length (Figure 9.1-11). The SFCLT allows latching/unlatching at different set elevations to enable cask loading. This is a long-handled tool that can be used with either SFP bridge hoist and is guided and latched manually by an operator. Due to its shorter length, higher dose rates are expected when moving fuel assemblies as compared to other fuel handling systems.

9.1.4.4.3 Burnable Poison Rod Assembly Handling Tool

(Note: The information in this section is for historical purposes only. The burnable poison rod assembly (BPRA) handling tool is no longer used since plate mounted BPRAs are no longer utilized at Ginna.) The burnable poison rod assembly handling tool, shown in Figure 9.1-11, is used to transfer plate-mounted burnable poison rod assemblies between fuel assemblies or between a fuel assembly and a burnable poison rod assembly storage insert in the spent fuel racks. Transfer is accomplished by raising the burnable poison rod assembly out of one location, drawing it up inside the tool, then lowering it down out of the tool into the other location.

This tool is used in the spent fuel building, suspended from the hoist on the spent fuel bridge and operating on the bridge walkway. The tool enters into the guide holes of a fuel assembly top nozzle or rack insert. Two sleeve actuated fingers engage the holddown bar at the top of the burnable poison rod assembly. The fingers actuator assembly is spring loaded in the DOWN (holding) position and can be moved upward to release the fingers only when the burnable poison rod assembly is in the FULL DOWN position.

Four pneumatic-cylinder-driven comb assemblies at right angles form an interlocking grid to position and guide the poison rods during withdrawal or insertion. During the up or down movement of the burnable poison rod assembly inside the tool, an indicator is tripped to notify the operator when moving the combs in or out to avoid the thimble plugs.

The 6-in.-long thimble plugs are larger in diameter than the burnable poison rods and will interfere with extended combs; consequently, the combs must be fully retracted when the thimble plugs are at or below the level of the combs.

The comb cylinders are driven by an attached air supply and controlled by a valve package mounted at the top of the tool. A safety latch over the valve handles prevents the combs from being accidentally withdrawn.

Two types of burnable poison rod assembly handling tool guides are provided to assist in locating the handling tool on a fuel assembly or rack insert. One type aligns over the top funnel portion of a spent fuel rack; the other is designed to align over a spent fuel rack insert or the upender fuel assembly container. Each guide has two attached nylon ropes for lowering into position prior to landing the handling tool. The guide remains there during burnable poison rod assembly removal or insertion and is removed after the handling tool has been raised for transfer to the next specified location.

Tool guides are necessary because the large base plate near the bottom end of the handling tool completely overhangs the three locating pins which must enter close-fitting holes in the fuel assembly top nozzle or rack insert. Thus, the tool operator standing on the spent fuel bridge cannot observe the pins to guide them into their respective holes.

9.1.4.4.4 Control Rod Drive Shaft Tool

The control rod drive shaft tool, shown in Figure 9.1-11, is utilized by the operator to disconnect the rod drive split shaft from the collet coupling on the rod cluster control spider. This is done after the head is removed and before the upper internals lifting device is attached. Once disconnected, the drive shafts remain with the upper internals package as it is lifted and wet stored.

The control rod drive shafts are removed from and reconnected to the rod cluster control assembly by means of the control rod drive shaft unlatching tool. The tool employs two sets of cam-actuated latching fingers, which grip the control rod drive shaft and disconnect button, respectively. The cams are actuated by air cylinders controlled by valves equipped with a locking ring on the operating handle to prevent accidental energizing of the air cylinders. The air cylinders actuating the drive shaft latching fingers are spring loaded causing them to lock in the shaft latch position in case of loss of air, thus preventing accidental dropping of the control rod. The disconnect button finger assembly is connected directly to an indicator rod by the double ended air cylinder. The upper end of the indicator rod is calibrated to the FULL UP and FULL DOWN position of the drive shaft disconnect button.

9.1.4.4.5 Thimble Plug Handling Tool

The thimble plug handling tool, shown in Figure 9.1-11, is used to handle the Westinghouse Optimized Fuel Assemblies and the Westinghouse VANTAGE + Fuel Assemblies. It employs two sleeve-actuated latching fingers to grip the thimble plug handling bar. The fingers and the finger housing move with respect to the outside frame thereby permitting the thimble plug to be withdrawn up inside the frame. The operating handle to actuate the fingers is located at the top of the tool. With the handle in the DOWN position, the fingers are latched; with the handle in the UP position the fingers are unlatched.

9.1.4.4.6 Irradiation Sample Handling Tool

The irradiation sample handling tool, shown in Figure 9.1-11, is a long-handled tool suspended from the containment building crane used to remove the irradiation specimens from their holders located on the outer surface of the neutron shield panels along the lower core barrel. This tool, operated from the manipulator crane bridge, extends through aligned access openings in the upper core barrel flange, thereby preventing unnecessary lifting of the core barrel out of the reactor vessel.

The irradiation sample handling tool employs three cam-actuated latching fingers which grip the top plug of the irradiation sample capsule. The operating handle to actuate the fingers is located at the top of the tool. With the fingers latched, the operating handle is in the DOWN position; with the fingers unlatched, the handle is in the UP position. As a safety feature, a pin is to be inserted into the operating shaft to preclude the possibility of the fingers becoming unlatched during removal of capsules. The tool is also equipped with a secondary operating sleeve which will permit an operator to remove the irradiation specimen access plug. The pin would be removed during this operation and the operating handle would be held in either the UP or DOWN position by means of the ball detents.

9.1.4.4.7 Stud Tensioners

The stud tensioners are hydraulically operated devices which unload the reactor vessel head studs to transition from MODE 5 (Cold Shutdown) to MODE 6 (Refueling) conditions and then preload them following refueling to transition back to MODE 5. The tension is normally applied simultaneously to as many studs as there are tensioners, and studs are tensioned to their operational load according to a sequence designed to prevent high stresses in the flange region, and unequal loading in the studs.

Trolleys operate on an I-beam (welded to the ring girder) around the reactor vessel head to position the hoists, which are used to move the stud tensioners from one stud to another. A hydraulic pumping unit, control console, and pressure release/relief valve provide the means to operate the tensioners. When the hydraulic pressure release valve is operated, the hydraulic fluid pressure is decreased, thus reducing the force applied to the stud by the tensioner.
Because it contains aluminum and electronics, all removable tensioning equipment (trolleys, hoists, pumping unit, etc.) is taken out of containment before reactor operation.

9.1.4.4.8 Portable Rod Cluster Control Assemble (RCCA) Tool

The portable RCCA Tool is used by the operator to move rod control cluster assemblies in the spent fuel pool. The tool is suspended from the SFP bridge hoist when in use. The tool consists of a telescoping grapple that engages the RCCA hub and retracts the RCCA into a cage that maintains alignment of the rodlets. The portable RCCA tool is compatible with OFA and 422V+ fuel assembly top nozzles.

9.1.4.5 Fuel Handling System Operation During MODE 6 (Refueling)

9.1.4.5.1 Introduction

The MODE 6 (Refueling) sequence follows detailed procedures which ensure a safe, efficient operation. These procedures, along with Technical Specifications requirements and the previously discussed equipment interlocks and safety features, provide assurance that no threat to the public health and safety will occur. The MODE 6 (Refueling) operation may be divided into three major phases: preparation, MODE 6 (Refueling), and reactor reassembly.

9.1.4.5.2 Preparation Phase

In the preparation phase, the reactor is shut down, cooled down to less than 140°F, and borated to a value that meets the requirements of the Section 3.9.1 of the Ginna Technical Requirements Manual (TRM). The containment is surveyed and ventilated as necessary, refueling equipment is checked out, and reactor disassembly begun. The control rod drive mechanism missile shield is removed to storage, then ventilation ducting, control rod drive mechanism cables, reactor vessel head insulation, instrument leads, and core exit thermocouple nozzle assemblies (CETNAs) are removed. The reactor vessel studs are detensioned, the studs removed, and stud hole plugs and three guide studs installed. The flux mapping thimbles are retracted and low-pressure seals made up. The final preparation of underwater lights and tools is made, the reactor vessel to cavity seal ring is installed, and the fuel transfer tube blind flange is removed.

At this point the reactor vessel head lifting rig is installed and the head is unseated, checked for levelness, and lifted 108 in. above the flange. The reactor cavity is now filled with water from the refueling water storage tank (RWST) as the head is raised until at least 23 ft of water exists above the flange. The reactor vessel head is then removed to storage.

The control rod drive shafts are disconnected and, with the upper internals, are removed by the vessel internals lifting rig and containment crane. The upper internals are wet stored on a stand in the refueling cavity. The fuel assemblies are now free from obstructions, and the core is ready for MODE 6 (Refueling).

9.1.4.5.3 MODE 6 (Refueling) Phase

With the initiation of the MODE 6 (Refueling) phase, the reactor cavity water level is verified to be covering the transfer canal and the reactor cavity boron concentration is verified to be within Technical Specifications limits. The fuel transfer tube valve is then opened. This provides a fuel movement path and allows level monitoring, cooling, and cleanup of the refueling water by the spent fuel pool (SFP) cooling system.

The MODE 6 (Refueling) sequence is begun by a manipulator crane. It is positioned over a spent fuel assembly, the gripper tube is lowered, and the gripper engaged with the upper nozzle. The fuel is lifted up into the protective mast of the machine. This height is sufficient to clear the top of the reactor vessel yet still leave sufficient water depth to provide radiation shielding for personnel. The manipulator crane then transfers the fuel to the fuel transfer system for movement to the spent fuel pool (SFP).

Once the spent fuel is removed from the core, partially spent fuel is transferred to the vacated positions and new fuel assemblies are brought in via the spent fuel pool (SFP) and fuel transfer system and loaded into the appropriate locations. The fuel management plan and the resulting plant procedures specify the specific fuel moves for each fuel assembly.

The MODE 6 (Refueling) sequence is modified for fuel assemblies containing rod cluster control elements. If transfer of the rod cluster control elements between fuel assemblies is required, the assemblies are taken to the rod cluster control change fixture to exchange the rod cluster control elements from one assembly to another. Such a change is required whenever a spent fuel assembly containing a rod cluster control element is removed from the core and whenever a fuel assembly is placed in or taken out of a control position during MODE 6 (Refueling) rearrangement.

The RG&E has incorporated the Westinghouse guidelines currently in effect entitled Subcriticality and Core Coupling Guidelines for Core Loading into the Ginna fuel assembly movement sequence. Following these guidelines ensures that the shutdown margin requirements of the Ginna Technical Specifications are maintained during MODE 6 (Refueling) operations.

In the interest of saving time in the MODE 6 (Refueling) operation, the operators may elect to temporarily wet store the new fuel assemblies in designated locations in the spent fuel pool (SFP) prior to actual MODE 6 (Refueling).

Whenever new fuel is added to the reactor core, a reciprocal curve of source neutron multiplication (inverse count rate ratio plot) is recorded to verify the subcriticality of the core.

9.1.4.5.4 Reactor Reassembly

Once all the fuel has been positioned in the core, the fuel transfer conveyor car is parked in the spent fuel pool (SFP) and the fuel transfer tube isolation valve is closed and the containment crane replaces the reactor vessel upper internals package in the vessel. The control rod drive shafts are then reattached to the rod cluster control assemblies. New O-ring seals are installed on the reactor head which is positioned over the reactor vessel and slowly lowered as the cavity water level is lowered. When the reactor vessel head is about 1 ft above the flange, the reactor cavity is completely drained and the flange is cleaned. The reactor vessel head is then seated on the flange surface.

The reactor vessel cavity seal ring is removed and stored. The guide studs and plugs are removed and the stud holes cleaned. The refueling cavity is then washed and decontaminated. The blind flange on the fuel transfer tube is reinstalled. The hold-down studs are replaced and torqued with the hydraulic tensioners and all instrument port thermocouple seals, thermocouples, and electrical connections are restored. Cooling ducts and reactor vessel head insulation are replaced and the missile shield moved into place above the reactor.
Finally, the in-core detector thimbles are inserted into the core, seals are made up, and all necessary connections are made.

Any maintenance required on the refueling equipment is accomplished at this time and it is then returned to its power operation storage position. When cleanup and restorage is complete, all preoperational tests and checks are made.

9.1.4.6 Fuel Handling System Evaluation

Underwater transfer of spent fuel provides essential ease and corresponding safety in handling operations. Water is an effective, economic, and transparent radiation shield and a reliable cooling medium for removal of decay heat.

Basic provisions to ensure the safety of MODE 6 (Refueling) operations are as follows:

A. Gamma radiation levels in the containment and fuel storage areas are continuously monitored. These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. Continuous monitoring of reactor neutron flux provides immediate indication and alarm of an abnormal core flux level in the control room.

B. Containment penetrations shall be in the status specified in the Technical Specification LCO 3.9.3 during core alterations and during movement of irradiated fuel assemblies within containment.

C. Whenever new fuel is added to the reactor core, a reciprocal curve of source neutron multiplication is recorded to verify the subcriticality of the core.

D. Direct communication between the control room and the refueling cavity manipulator crane is available whenever changes in core geometry are taking place to allow the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

9.1.4.6.1 Incident Protection

Direct communication between the control room and the refueling cavity manipulator crane is available whenever changes in core geometry are taking place.

This provision allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

9.1.4.6.2 Malfunction Analysis

An analysis is presented in Section 15.7.3, concerning cladding damage to all fuel rods in one assembly, for evaluating environmental consequences of a fuel handling accident.

9.1.4.7 Minimum Operating Conditions

Limiting conditions for MODE 6 (Refueling) operations are specified in the Technical Specifications.

Whenever the core cooling or containment spray systems are specified to be operable, the refueling water storage tank (RWST) must have a minimum water volume of 300,000 gallons and have a boron concentration not less than 2750 ppm and no more than 3050 ppm. The refueling water storage tank (RWST) capacity is 338,000 gallons and the quantity of water required for MODE 6 (Refueling) is 230,000 gallons.

Analysis of loss-of-coolant incidents shows that the quantity of water in storage is sufficient for limiting core temperatures and containment pressure following any incident. These analyses are discussed in Section 15.6.

9.1.4.8 Tests and Inspections

Upon completion of core loading and installation of the reactor vessel head, certain mechanical and electrical tests were performed prior to initial criticality. The electrical wiring for the rod drive circuits, the rod position indicators, the reactor trip circuits, the in-core thermocouples, and the reactor vessel head water temperature thermocouple were tested at the time of installation. The tests were repeated on these electrical items before initial plant operation.

9.1.5 CONTROL OF HEAVY LOADS

As a result of the NRC review of load-handling operations at nuclear power plants, NUREG 0612, Control of Heavy Loads at Nuclear Power Plants, was issued. Following the issuance of NUREG 0612, a generic letter, dated December 22, 1980, was sent to all plants requesting that responses be prepared to indicate the degree of compliance with the guidelines of NUREG 0612. The responses were made in two stages. The first response (Phase I) was to identify the load-handling equipment within the scope of NUREG 0612 and to describe the associated load paths, procedures, operator training, special and general-purpose lifting devices, the maintenance, testing and repair of equipment, and the handling equipment specifications. The second response (Phase II) was intended to show that either single-failure-proof handling equipment was not needed or that single-failure-proof equipment had been provided.

Ginna Station responded with submittals to the NRC on February 1, 1982, (*Reference 18*) March 2, 1983, (*Reference 19*) and October 12, 1983 (*Reference 20*). The NRC staff and its consultant, the Franklin Research Center, have reviewed the submittals for Ginna Station and have issued a technical evaluation report, (*Reference 21*) and a safety evaluation report (*Reference 22*) concluding that Phase I of the control of heavy loads issue for Ginna Station is acceptable.

Since the issuance of these reports, an updated inspection program for the reactor head lifting rig was proposed in a letter to the NRC dated May 30, 1986 (*Reference 23*). This program of 100% visual inspection of the lifting rig welds, prior to first use at each MODE 6 (Refueling) outage, together with 10-year surface examinations on exposed portions of the welds is considered adequate testing to verify that the lifting device is in compliance with NUREG 0612. The auxiliary building crane has been upgraded to the single-failure requirements of NUREG 0554. See Section 9.1.4.3.1.

The Ginna Station responses to Phase II of the issue were submitted on March 26, 1984, (*Reference 24*) and July 31, 1984. (*Reference 25*). Based on improvements in heavy loads handling obtained from the implementation of Phase I, further action is not required to reduce the risk associated with the handling of heavy loads. Therefore, Phase II is considered complete.

In 1996, the NRC issued Bulletin 96-02 (*Reference 32*) to alert licensees to the importance of complying with existing regulatory guidelines associated with the control and handling of heavy loads at nuclear power plants while the plant is operating. In *Reference 33*, RG&E responded to Bulletin 96-02 by stating that a review was performed of planned activities into 1998 related to heavy loads and that all potential heavy load movements were determined to be within the scope of the Ginna Station licensing basis. In *Reference 44*, the NRC determined that RG&E's response to Bulletin 96-02 was acceptable and therefore considers the issue to be closed.

The "Industry Initiative on Control of Heavy Loads," NEI-05 (Revision 0) was reviewed to verify the Ginna's heavy load lifts continue to be conducted safely and that the plant procedures accurately reflect the licensing bases.

9.1.5.1 CONDUCT OF HEAVY LOADS MOVEMENTS

The movement of heavy loads at Ginna Station is controlled under plant procedures. These procedures give requirements for material handling equipment and their inspections. Any lifted load greater than 1500 pounds is treated as a heavy load at Ginna Station.

A specific procedure exists for each crane that can move loads over safety related equipment at Ginna Station. Each procedure gives safe load paths for load movements with its associated crane.

Since the movement of the reactor vessel head from the reactor to its temporary storage stand during a refueling outage is considered to be a high-risk evolution, that movement is further controlled through the use of refueling procedures. Those procedures apply the general mechanical maintenance administrative controls for heavy loads, but also restrict the height that the reactor vessel head may be lifted above the core before it is moved away from above the core. This restriction is imposed to assure compliance with the load drop analysis of the reactor vessel head that was performed as part of Ginna's response to NUREG 0612.

The Independent Spent Fuel Storage Installation (ISFSI) Project installed a 125-ton single failure proof cantilevered gantry crane in the Canister Preparation Building. The 125-ton crane has a single failure proof hoist design which includes a yoke and block for lifting a Spent Fuel Transfer Cask with a fully loaded Dry Shielded Canister. The 125-ton crane satisfies all regulatory requirements in accordance with ASME NOG-1, NUREG-0554, CMAA-70, and the Ginna UFSAR.

REFERENCES FOR SECTION 9.1

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3. Letter from R. W. Kober, RG&E, to H. R. Denton, NRC, Subject: Increase of the Spent Fuel Storage Capacity, dated April 2, 1984.
4. Letter from J. A. Zwolinski, NRC, to R. W. Kober, RG&E, Subject: Increase of the Spent Fuel Storage Capacity, dated November 14, 1984.
5. Letter from G. E. Lear, NRC, to R. W. Kober, RG&E, Subject: Storage of Consolidated Fuel, dated December 16, 1985.
6. Letter from J. E. Maier, RG&E, to H. R. Denton, NRC, Subject: Application for Amendment to Operating License, dated February 23, 1983.
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8. N. M. Greene, AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B, ORNL/TM-3706, March 1976.
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21. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: Transmittal of Technical Evaluation Report on Control of Heavy Loads, dated August 19, 1982.

22. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: Transmittal of Safety Evaluation on Control of Heavy Loads (Phase I), dated January 18, 1984.

23. Letter from R. W. Kober, RG&E, to G. E. Lear, NRC, Subject: Control of Heavy Loads, dated May 30, 1986.

24. Letter from R. W. Kober, RG&E, to D. M. Crutchfield, NRC, Subject: Control of Heavy Loads, dated March 26, 1984.

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28. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: Insertion of a Higher Enrichment Fuel Assembly Into the Spent Fuel Racks, dated February 8, 1984.

29. Letter from G. E. Lear, NRC, to R. W. Kober, RG&E, Subject: Storage of Consolidated Fuel, dated December 16, 1985.

30. Letter from A. R. Johnson, NRC, to R. C. Mecredy, RG&E, Subject: Issuance of Amendment No. 60 to Facility Operating License No. DPR-18 (TAC No. M92188), dated February 6, 1996.

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Table 9.1-1
FUEL PARAMETERS EMPLOYED IN THE CRITICALITY ANALYSIS

Parameters	Exxon 14x14	Westinghouse[a] 14x14 STD	Westinghouse 14x14 OFA/14 x 14 VANTAGE+	422V+
Number of fuel rods per assembly	179	179	179	179
Rod Zirc-4 (ZIRLO®/ Optimized ZIRLO™) clad O.D. (in.)	0.424	0.422	0.400	0.422
Clad Thickness (in.)	0.030	0.0243	0.0243	0.0243
Fuel pellet O.D. (in.)	0.3565	0.3669	0.3444	0.3659
Fuel pellet density (% of theoretical)	95	95	95	95
Fuel pellet dishing factor	1.187	1.187	1.1926	1.187
Rod pitch (in.)	0.556	0.556	0.556	0.556
Number of Zirc-4 (ZIRLO®) guide Tubes	16	16	16	16
Guide tube O.D. (in.)	0.524	0.539	0.526	0.526
Guide tube thickness (in.)	0.015	0.017	0.017	0.017
Number of instrument tubes	1	1	1	1
Instrument tube O.D. (in.)	0.424	0.422	0.399	0.422
Instrument tube thickness (in.)	0.039	0.0240	0.0235	0.0240

a. Parameters used in criticality analyses for Westinghouse standard 14 x 14 fuel bound the geometry parameters associated with the Ginna Westinghouse 422 VANTAGE+ fuel assemblies.

Table 9.1-2
TORNADO MISSILE ACCIDENT DOSE ANALYSIS ASSUMPTIONS

Parameter	Value
Reactor Power, MWt (including fuel management factor)	1811
Power Peaking Factor	1.75
Number of damaged fuel assemblies[a] Region 1 Region 2	 5 hot, 4 cold 9 cold
Time after reactor shut-down hot assemblies cold assemblies	 100 hours 60 days
Fuel rod gap fractions I-131 other halogens Kr-85 other noble gases	 0.08 0.05 0.1 0.05
Iodine specials above water elemental iodine organic iodide	 0.57 0.43
Pool DF elemental iodine organic iodide Particulate Overall Pool DF	 500 1 ∞ 200
Exhaust flow rate, cfm puff (5 second activity release)	 1.11E+08
Iodine removal efficiency for all forms to environment	0
Control Room isolation	No
Control Room filtration operation	No

a. Based on damage to 9 assemblies in a 3x3 group of storage cells. The bounding source term for the TMA is due to damage in Region 1 Type 3 SFP storage racks. The source term also bounds storage of one hot fuel assembly in Region 2 Type 2 SFP storage racks per Technical Requirements Manual (TRM) Section 3.9.1

Table 9.1-2
TORNADO MISSILE ACCIDENT DOSE ANALYSIS ASSUMPTIONS (OFFSITE X/Q)

Boundary	2 hr[a]	0-8 hr	8-24 hr	24-96 hr	96-720 hr
EAB	2.17E-04[b]	-	-	-	-
LPZ		2.51E-05	1.78E-05	8.50E-06	2.93E-06

a. Any two-hour period

b. 0 to 1 min tornado value is 1.87E-6

Table 9.1-2
TORNADO MISSILE ACCIDENT DOSE ANALYSIS ASSUMPTIONS (OFFSITE BREATHING RATES)

Boundary	2 hr	0-8 hr	8-24 hr	24-96 hr	96-720 hr
EAB	3.47E-04	-	-	-	-
LPZ		3.47E-04	1.75E-04	2.32E-04	

Table 9.1-3
SPENT FUEL POOL (SFP) COOLING SYSTEM RATING

	Safety Basis Heat Load	***Normal Basis Heat Load***
SFP HEAT EXCHANGER B		
Heat removal capacity, Btu/hr	32[a] x 10^6	16 x 10^6
Service water temperature in, °F	85	80
Service water temperature out, °F	125	113
Service water temperature differential, °F	40	33
Pool water temperature, °F	150	120
Service water flow, gpm (approximate)	1600	1000
Pool water flow, gpm (approximate)	1200	1200
SFP HEAT EXCHANGER A		
Heat removal capacity, Btu/hr	7.93 x 10^6	5.3 x 10^6
Service water temperature in, °F	80	80
Service water temperature out, °F (approximate)	103	95
Service water temperature differential, °F (approximate)	23	15
Pool water temperature, °F	150	120
Service water flow, gpm (approximate)	700	700
Pool water flow, gpm	610	610

a. Maximum SFP heat load is limited to 25.67 x 10^6 BTU/HR per section 9.1.3.4.3

Table 9.1-4
SPENT FUEL POOL (SFP) COOLING SYSTEM COMPONENT DATA

System design pressure, psig	150
System design temperature, °F	200

Spent fuel pool heat exchanger

Quantity	2
Type	Shell and U-tube
Material, shell/tube	Carbon steel/stainless steel (SFP A)
	stainless steel/stainless steel / SS (SFP B)

	SFP HEAT EXCHANGER A	**SFP HEAT EXCHANGER B**
Design, Btu/hr[a]	7.93×10^6	32×10^6
Service water flow, design gpm[b]	700	1600
Tube flow, design gpm[b]	550	1200

Spent fuel pool pump data

Quantity	3
Type	Horizontal centrifugal
Material	Stainless steel

	SFP PUMP A	**SFP PUMP B**	**SFP STANDBY PUMP**
Flow, design gpm[b]	610	1200	610
Head, ft H_2O	150	150	150
Motor horsepower	50	100	50

Spent fuel pool	
Volume, ft^3 / gallons	34,100 / 255,000
Boron concentration, ppm boron minimum	≥2300

a. Pool temperature at or below 150°F, service water temperature at 80°F, for SFP Heat Exchanger A and 85°F for SFP Heat Exchanger B.

b. Design flow rates represent rated values and do not represent flow limits. Flow limits are greater than or equal to design flowrates.

Table 9.1-5
OFFSITE AND CONTROL ROOM DOSES FOR THE SPENT FUEL POOL TORNADO MISSILE ACCIDENT

	Doses (Rem)	*Limit (Rem)*
CONTROL ROOM DOSE	0.63	5.0
TORNADO MISSILE ACCIDENT IN REGION 1 (100 HRS DECAY)		
Exclusion Area Boundary (0-2 hours)	0.03	6.3
Low Population Zone (0-2 hours)	0.01	6.3

Table 9.1-6

HEAT-UP TIMES ASSOCIATED WITH LOSS OF SPENT FUEL POOL COOLING[a]

Operating Cooling Loop	Service Water Temp (°F)	Decay Heat Load (MBtu/hr)	Time to 180°F in Fuel Pool[b] (hours)	Time to 212°F in Fuel Pool[b] (hours)	Required Make-up Rate (gpm)
SFP Loop B (normal) OR	40	25.67	2.4	4.9	53
SFP Pump A and SFP	60	25.67	2.4	4.9	53
Standby Pump cross-connected to SFP HX B	80	25.67	2.4	4.9	53

a. Structural design temperature limit for SFP is 180°F. The times to 212°F and the boil-off rate at 212°F were calculated as requested by the NRC in *Reference 42*.

b. Initial SFP temperature is 150°F.

9.2 WATER SYSTEMS

9.2.1 SERVICE WATER (SW) SYSTEM

9.2.1.1 Design Bases

The service water (SW) system takes suction from Lake Ontario via the screen house and supplies cooling water to various turbine plant loads as well as auxiliary reactor plant loads. The system supplies seal water to the circulating water pumps and the vacuum pumps, flushing water to the traveling screens and makeup water to the fire water storage tank via the fire booster pump. Service water (SW) is the normal supply to the standby auxiliary feedwater system and an alternate supply to the preferred auxiliary feedwater system. The system is designed to provide adequate cooling to critical and noncritical loads during MODES 1 and 2 and to critical loads during accident conditions. The system normally discharges back into Lake Ontario via the discharge canal. A discharge line to Deer Creek is available for selected Auxiliary Building SW loads.

The service water (SW) system consists of four service water (SW) pumps, a single loop supply header, isolation valves, and a normal and standby discharge header. The physical design of the SW system is such that four(4) pumps, two(2) from each class 1E electrical bus (Buses 17 and 18), supply the SW loop header. All portions of the service water (SW) system (pumps, piping, etc.) serving safeguards equipment are designed as Seismic Category I. All other portions of the service water (SW) system serving nonsafety loads are designated as nonseismic and are capable of being isolated from the Seismic Category I portion.

9.2.1.2 Description

9.2.1.2.1 General Description

The service water (SW) loop header supplies the cooling water to all safety related and non-safety related components. The nonsafety related and long-term safety functions (e.g., component cooling water heat exchangers) can be isolated from the loop header through use of redundant motor operated isolation valves. These valves automatically close on a coincident safety injection signal and undervoltage signal on Buses 14 and 16.

The system is sized to ensure adequate heat removal based on the highest expected temperatures of cooling water, maximum loadings, and leakage allowances. The system is monitored and operated from the control room. Isolation valves are incorporated in all service water (SW) lines penetrating the containment.

The service water (SW) system flow diagram is shown in Drawings 33013-1250, 33013-1251, and 33013-1925. Drawing 33013-1250, Sheets 1 through 3 is the safety-related service water (SW) system P&ID. Drawing 33013-1251, Sheets 1 and 2 is the non-safety-related service water (SW) system P&ID. Drawing 33013-1925 is the service water (SW) system P&ID for the instrument air compressors and after coolers, which are supplied by the non-safety-related portion of the service water (SW) system.

The four service water (SW) pumps are located in the screen house. They are two-stage, vertical turbine pumps (original specified rating of 5300 gpm each, 1760 rpm, 198 ft (total discharge head), and 308 brake horsepower). Each pump has a clip-on type basket strainer installed on its suction end bell. The service water (SW) pumps were originally supplied with 300-hp motors. Between 1995 and 1997, all four motors were replaced with 350-hp motors that have anti-reverse rotation devices. An evaluation showed that this configuration met the design and performance requirements of the system. Periodic service water (SW) pump testing ensures that the inservice testing (IST) program performance requirements are satisfied.

The service water (SW) system circulates water from the screen house on Lake Ontario to various heat exchangers and systems inside the containment and the auxiliary, intermediate, turbine, and diesel generator buildings. These buildings are Seismic Category I structures except for the turbine building. Two or three of the pumps are generally in use to carry the required normal cooling load. Table 9.2-1 lists the loads supplied by the service water (SW) system.

During MODES 1 and 2, the service water (SW) system supplies flow to all necessary loads except pump suction flow to the preferred auxiliary feedwater and standby auxiliary feedwater pumps (SAFW). During residual heat removal operation for a normal plant cooldown, almost all noncritical loads may be removed from the service water (SW) system, if necessary. Following a safeguards actuation signal (with bus undervoltage) the service water (SW) system would continue to supply all required critical loads except the supply to the preferred auxiliary feedwater and standby auxiliary feedwater pumps (SAFW), which require operator action to receive service water (SW) flow.

The fire protection system can be used as a backup for the service water (SW) system supply to spent fuel pool (SFP) heat exchanger A, the standby spent fuel pool (SFP) heat exchanger, either component cooling water (CCW) heat exchanger (under emergency, beyond design basis conditions only), preferred auxiliary feedwater pumps, standby auxiliary feedwater pumps (SAFW), and the diesel generator lube-oil coolers and jacket water heat exchangers via temporary hoses.

Originally, plant load requirements dictated two or three pumps for normal full load, one pump for accident conditions during the injection phase, and two pumps for accident conditions during the recirculation phase. Based on subsequent analyses, a single service water (SW) pump has been shown to provide adequate cooling to all loads during the recirculation phase, coincident with nonessential load isolation. In January of 1999, the NRC issued a Safety Evaluation Report (SER) accepting the results of RG&E’s evaluations (*Reference 14*). However, as a result of performing the power uprate to 1775 MWt, the required SW Pumps needed during the re-circulation phase of a design basis LOCA was modified from one operating SW Pump to two operating SW Pumps. The number of SW Pumps required during the injection phase of a design basis LOCA after uprate remains as one SW Pump.

The post-accident containment pressure and critical reactor integrity parameters are not affected since the limiting conditions for these parameters occur during the injection phase of the design basis event. Typical service water (SW) flows supplied during normal operation and reference design flows are shown in Table 9.2-2. Service water (SW) flow design limits for critical loads are evaluated based on heat removal requirements and documented by engineering design analyses. These tabulated major loads on the service water (SW) system formed the basis for the sizing of the service water (SW) pumps. Service water (SW) flow rates analyzed for the various cases evaluated are consistent with the assumed single failure for each case. See Section 6.2.2 for details of these cases. Electric power to the service water (SW) pumps is provided by 480-V safeguards buses 17 and 18. Following a safety injection (SI) signal and/or loss of offsite power, one service water (SW) pump is automatically started on each diesel generator.

9.2.1.2.2 Service Water System Design

The service water (SW) system consists of a single loop header. Four(4) pumps, two(2) from each class IE electrical bus (Buses 17 and 18) is arranged on a common piping header which then supplies the service water (SW) loop header. A service water (SW) train is based on electrical source only. Cross-tie valves are located in the loop header which could be used to split the header. Some of the cross-tie valves are operated normally open, while others are operated normally closed; this functions to balance service water (SW) system flows and pressures. The loop header is designed so that no single failure will cause a plant shutdown.

The service water (SW) system piping is arranged so that all pumps can provide flow to the critical loads identified on Table 9.2-2 (and also to the noncritical reactor compartment coolers and containment penetration coolers). During normal operation, branch headers supply various noncritical loads (see Table 9.2-1). Six pairs of motor-operated valves are provided to automatically isolate the loop header from the component cooling water (CCW) heat exchangers, spent fuel pool (SFP) heat exchangers, and the noncritical loads (excepting the noncritical reactor compartment coolers and containment penetration coolers); in addition, these valves can be controlled remotely from the control room. The redundant valves in each pair are powered from independent 480-V safeguards buses (buses 14 and 16). The motor-operated valves within each independent train will automatically receive a close signal following a safety injection (SI) signal concurrent with a trip of the normal supply breaker (i.e., undervoltage) on their associated 480-V safeguards bus. These valves will then close automatically upon reenergization of their associated 480-V safeguards electrical bus by its diesel generator.

The two component cooling water (CCW) heat exchangers, the "A" spent fuel pool (SFP) cooling heat exchanger, the safety-related pump motor coolers located in the auxiliary building, and the standby auxiliary feedwater (SAFW) pump room cooling units have redundant service water (SW) discharge lines, thus providing Seismic Category I redundant service water (SW) supply and discharge lines to these loads. The primary service water (SW) discharge line discharges to the discharge canal and then to Lake Ontario. The redundant service water (SW) discharge line discharges to a Seismic Category I discharge structure, then to Deer Creek and to Lake Ontario. The redundant service water (SW) discharge line is normally in standby; however, it is occasionally placed in service for such activities as surveillance testing or maintenance work.

9.2.1.2.3 Service Water System Initiation on Loss of Offsite Power

The service water (SW) pumps are connected to the 480-V safeguards buses that can be supplied by the emergency diesels in the event of a loss of all offsite power. One service water (SW) pump per diesel is automatically started on either an undervoltage condition or a safety injection (SI) signal. For a safety injection (SI) signal coincident with an undervoltage condition, the service water (SW) system is designed to supply cooling water only to the required critical loads by means of automatically closing redundant motor-operated non-essential load isolation valves, with the exception that the noncritical reactor compartment coolers and containment penetration coolers are not isolated. Under these conditions, any one pump using emergency power is capable of supplying the required cooling capacity to the injection phase loads shown in Table 9.2-2. On a loss of ac power to the 480-V safeguards buses, the service water (SW) pumps restart automatically upon reenergization of the buses. They do this under the following conditions:

A. Following an undervoltage condition on a safeguards bus, the selected service water (SW) pump will restart after a 40-sec time delay following reenergization of the bus by the emergency diesel.

B. On a safety injection signal, two service water (SW) pumps will start after 15-sec and 17-sec time delays for trains A and B, respectively, following reenergization of the buses by the emergency diesel (assuming an undervoltage condition existed), or from the time of the safety injection signal, if no undervoltage existed.

The diesel generators employ jacket cooling and shell and tube heat exchangers. Service water flow to these heat exchangers is provided through normally closed air operated valves. The valves open upon receipt of a diesel generator start signal. The valves will fully open prior to service water pump flow becoming available. Adequate heat absorption capacity is provided to operate until the service water (SW) system starts.

The motor driven auxiliary feedwater pumps (MDAFW) are equipped with ball bearings and require only supplementary cooling to the thrust-bearing jacket. The pumps are designed to operate satisfactorily until the service water (SW) system starts.

The auxiliary feedwater pump drive turbine has an oil reservoir which provides a supply of cool oil until the service water (SW) system starts. Tests conducted to determine the capacity of the oil cooler have demonstrated that the oil remains within acceptable temperature limits for at least 2 hours even without service water (SW) flow. Refer to Section 10.5.4.2 for details of this test.

9.2.1.2.4 Containment Cooling Coils

The containment cooling coils are provided with both automatic and manual temperature controls. A three-way selector valve (V-15532) is provided for this purpose. The automatic position of this valve allows normal automatic operation of the air-operated containment coolers service water (SW) outlet flow control valve (AOV-4561). In the manual mode, signal air to AOV-4561 can be controlled through a regulator valve (V-15531). The safety injection signal to AOV-4561 is not affected while in the manual mode. If the containment temperature rises above 100°F in the manual mode, regulator valve V-15531 can be repositioned as required to reduce containment temperature.

Automatic temperature control is provided for the containment cooling coils. An automatic bypass valve is provided around the containment cooling coil temperature control valve and both valves will trip wide open on a safety injection signal. Both the control and the automatic bypass valves are of the fail open type. Manual globe valves are provided on the outlet side of most service water (SW) system cooling services. Exceptions are the containment coolers, spent fuel pool (SFP) heat exchanger B, component cooling water (CCW) heat exchangers, and reactor compartment coolers which have butterfly valves that can be used for flow adjustment and balancing.

Indicating alarms are provided to monitor each fan cooler discharge for flow and temperature. A radiation-indicating alarm is located on the discharge line downstream of the discharge header. In addition, each fan cooler inlet is provided with a pressure indicator.

9.2.1.2.5 Radiation Monitors

A common radiation monitor is provided in the service water (SW) discharge line from the four containment coolers and the reactor compartment coolers. Individual coolers may be manually isolated to determine which unit is leaking if the monitor indicates radioactivity. Radiation monitors are also provided in the service water (SW) discharge lines from the spent fuel pool (SFP) cooling system heat exchangers.

9.2.1.2.6 Service Water Fouling

Lake Ontario has an infestation of zebra mussels, which makes Ginna Station's cooling systems potentially vulnerable to plugging. To control this problem, RG&E has installed sodium hypochlorite injection lines in the screen house inlet plenum and service water (SW) pump bays to prevent colonization of zebra mussels in the screen house bays. Chlorine monitoring stations were also installed to monitor chlorine concentrations in the service water (SW) supply headers in the screen house basement; in service water (SW) discharge headers in the turbine building, intermediate building, and auxiliary building; and in the discharge canal. The sodium hypochlorite injection system includes four injection pumps, two of which may discharge into the inlet plenum and serve the circulating water system and two of which may discharge into the service water (SW) pump bay and serve the service water (SW) system (see Drawing 33013-1885, Sheet 2). Either of the service water (SW) sodium hypochlorite injection pumps is capable of meeting the total service water (SW) demand.

Based on zebra mussel biofouling experienced in 1991, the diesel generator jacket water heat exchangers and turbine lube-oil coolers were identified as being particularly susceptible to zebra mussel fouling. Valves with fire hose connections on the service water (SW) discharge and supply sides of these heat exchangers and coolers provide backflushing capability to clear out the zebra mussels. Side stream monitoring stations (bio-boxes) on the supply and discharge sides of both the circulating water and non-safety-related service water (SW) systems allow monitoring of biological growth (particularly zebra mussels) and chlorine effectiveness.

NRC Generic Letter 89-13 requests licensees to implement a surveillance and control program for the service water (SW) system to reduce the incidence of flow blockage problems as a result of biofouling and conduct a test and retest program to verify the heat transfer capability of safety-related heat exchangers cooled by service water (SW). Ginna Station has equipped the spent fuel pool (SFP) cooling B heat exchanger, component cooling water (CCW) heat exchangers, standby auxiliary feedwater pump (SAFW) room coolers, diesel generator coolers, and containment recirculating fan coolers with pressure and temperature connections and instrumentation to support a heat transfer capability and performance testing program. This installation is part of the Ginna Station Service Water System Reliability Optimization Program, which also includes intake structure inspection, service water (SW) and lake water sampling, and service water (SW) system flushing and cleaning programs and the above chlorination program. The Service Water System Reliability Optimization Program (SWSROP) was established to define the techniques, equipment, methods, and responsibilities that are used to ensure that the service water (SW) system performs the following functions: transfer the necessary heat from safety related equipment to the ultimate heat sink under both normal and accident conditions, provide a source of water to the preferred auxiliary feedwater system and the standby auxiliary feedwater system for decay heat removal, and support reliable and economic operation of Ginna Station. The program fulfills the recommendations of Generic Letter 89-13.

9.2.1.3 Design Evaluation

The service water (SW) system is designed to prevent a single active failure from curtailing normal station operation. As will be noted from *Drawings 33013-1250* and *33013-1251*, the 20-in. service water (SW) supply loop is isolated by normally closed loop isolation valves (4610 and 4779) and provides split flow to the safety-related component cooling water (CCW) heat exchangers and spent fuel pool (SFP) heat exchangers. Certain cross-tie valves are open to provide a balanced flow to the four containment fan coolers (valves 4639 and 4756) and to the two emergency diesel generators (valves 4760 and 4669). Supply lines to the safety-related pump area coolers in the auxiliary building and to the nonsafety-related reactor compartment coolers are also cross-tied. The system in this configuration is consistent with the analysis of the service water (SW) loads during accident conditions and procedures used during MODES 1 and 2. The service water (SW) loop header supplies the cooling water to all safety related and non-safety related components. The non-safety related and long-term safety functions (e.g., component cooling heat exchangers) can be isolated from the loop header through use of redundant motor operated isolation valves. These valves automatically close on a coincident safety injection signal and undervoltage signal on buses 14 and 16.

In addition to the loop isolation valves, each component also has individual isolation valves to permit isolating any piece of equipment from the system.

The design basis for the service water (SW) system includes design against a single active failure only; therefore, discussions relative to passive failures (i.e., critical pressure boundary pipe crack, *Reference 1*) are provided for information only.

From a system reliability standpoint, service water (SW) flow required for long-term safety functions (e.g., component cooling water (CCW) heat exchangers) has the capability of being provided by means of manual operation of the normally closed loop isolation valves (4610 and 4779), adding system operation flexibility. In the normal system alignment, no single active or passive failure could result in the loss of service water (SW) flow to redundant critical loads, although the noncritical reactor compartment coolers could both be partially disabled by a single passive failure.

The service water (SW) system is a moderate energy system; therefore, a passive pipe failure would probably result in a leak rather than a complete pipe rupture. Using the method described in *Reference 1*, the estimated leakage for a service water (SW) system header is 640 gpm for a 20-in. header at 88 psig. Although this leak may cause a flooding problem, the supply function of the affected header would not be significantly impaired (*Reference 18*). A leak from the 2.5-in. supply line to the noncritical reactor compartment coolers would result in the loss of about 25 gpm. This leak rate would not completely disable the coolers, each of which normally receives about 45 gpm of service water (SW) flow. Leaks were reviewed as a comparison to the Standard Review Plan; however, the service water (SW) system design is based upon ensuring that post-loss-of-coolant accident (post-LOCA) requirements are met, assuming a single active failure (AIF-GDC 41).

Double valves in series are provided for redundancy to automatically isolate nonessential service during an incident in case of the failure of a motor-operated valve.

All control valves used in the service water (SW) system fail in the open position. The control valve shown in the containment cooler discharge line is bypassed with a spring-loaded, fail open, quick action valve which will automatically open in the event of an accident or any malfunction of the valve closing signal system.

9.2.1.4 Postaccident Conditions

Minimum postaccident injection phase operating requirements are met with one pump and one loop header system (see Section 9.2.1.2.1). The remote operated isolation valves permit isolation of all noncritical services for one pump operation with the exception that the non-critical reactor compartment coolers and containment penetration coolers are not isolated. During transfer to the sump recirculation phase post-LOCA, the component cooling water (CCW) heat exchangers are provided with service water (SW) flow; therefore, a second service water (SW) pump is utilized in the analysis to accommodate this additional load. The motor-operated valves which are opened to supply service water (SW) to the component cooling water (CCW) heat exchangers also provide flow to branch headers that lead to the spent fuel pool (SFP) heat exchangers. Since the branch headers are open during normal power operation, the spent fuel pool (SFP) heat exchangers would be provided with service water (SW) flow during the recirculation phase unless operator action is taken. Emergency operating procedures provide guidance to operators for the isolation of the spent fuel pool (SFP) heat exchangers, if necessary, to ensure adequate flow is provided to the component cooling water (CCW) heat exchangers.

9.2.1.4.1 Recirculation Phase

Pre-Uprate

During the recirculation phase post-LOCA the service water (SW) loads dictate operation of two service water (SW) pumps to accommodate the additional component cooling water (CCW) heat exchanger loads. The use of two service water (SW) pumps allows the flow for each pump to operate closer to its rated flow of 5300 gpm, as compared to the flow of only one pump operating in a similar mode. Analyses have been performed assuming only one service water (SW) pump in operation in the recirculation phase post-LOCA. This condition is assumed to result from the failure of an emergency diesel generator based on an initial condition where only two of the four service water (SW) pumps were available as allowed by the Technical Specifications. The results showed that the containment temperature and pressure response was within the limits allowed (Figures 6.1-1 and 6.1-2 respectively) (*Reference 3*). Although the service water (SW) flow resulting from a single operating pump would be beyond its rated flow, it would still be less than its runout condition.

In their Safety Evaluation Report (SER) dated January 29, 1999, (*Reference 14*), the NRC concurred that one service water pump was acceptable for post-LOCA recirculation phase cooling. Although not required, the option to utilize the bus tie breaker between 480-V safeguards ac buses 17 and 18 during the recirculation phase post-LOCA in order to provide power to two service water (SW) pumps from a single diesel generator is retained, after appropriate evaluation by Technical Support Center personnel. See also Section 8.3.1.1.6.6.

Uprate

However, as a result of performing the power uprate to 1775 MWt, the required SW Pumps needed during the re-circulation phase of a design basis LOCA was modified from one operating SW Pump to two operating SW Pumps. The bases for the Service Water Technical Specification LCO has been revised to identify that two Service Water Pumps in each SW loop are required to be operable for the SW loop to be considered operable.

9.2.1.4.2 Limiting Steam Line Break Events

The accident that produces the limiting containment integrity (peak pressure) response is not a LOCA but a steam line break. See UFSAR Section 6.2.1.2.3 for a discussion of the cases analyzed and assumptions. Previous analyses have determined that a limiting assumption for containment pressurization due to a steam line break is no loss of off-site power. This assumption causes the two Reactor Coolant Pumps to continue to operate throughout the transient. In combination with this assumption, containment air cooler heat removal capability is based on assuming only one service water pump in operation. This combination of conservative assumptions ensures that the resulting containment pressurization analysis results are bounding. Therefore, only one service water pump is needed to mitigate the effects of a steam line break inside containment. Since a steam line break does not result in loss of reactor coolant inventory, operation of the service water system in a recirculation phase is not entered.

9.2.1.4.3 Accident Considerations With Offsite Power Available

The design basis accident analysis parameters affected by the flow capability of the service water (SW) system are fuel peak clad temperature (LOCA), containment pressure integrity (LOCA and steam line break), and equipment qualification (LOCA). For fuel peak clad temperature analyses, SW flow impacts the containment back-pressure that exists during the blowdown and reflood phase of a LOCA. As discussed in UFSAR Section 15.6.4.2.3.2, a lower containment backpressure reduces the core reflooding rate due to the increased difficulty in venting steam as a result of increased steam binding within the reactor and reactor coolant system. This in turn maximizes peak clad temperature. Therefore, for calculating minimum containment back-pressure to support LOCA analyses, service water system flow is maximized so as to maximize flow to the containment air coolers. Therefore, off-site power is assumed to be available for this analysis. Additionally, the minimum back-pressure analysis assumes operation of all four containment air cooler and minimum service water temperature. This set of assumptions maximize the heat removed from containment by the containment air coolers and thereby result in minimizing the containment back-pressure that is used as input into the LOCA peak cladding analyses described in UFSAR Section 15.6.4.

For containment integrity the limiting parameters to be maintained following a break are containment pressure and containment temperature. The steam line break produces the peak containment pressure results for Ginna. As previously discussed, loss of off-site power is not assumed for steam line breaks so as to maximize energy release from operating reactor coolant pumps. Therefore, other single failures are assumed. Typically, although failure of one service water pump would result in reduced flow it is not limiting, since four containment air coolers would be still operating as well as two containment spray pumps. The additional margin gained from the two operating pumps and operation of four containment air coolers compensate for the reduction in SW flow to the containment air coolers. At uprate the limiting single failure for a steam line break is a vital bus failure as described in UFSAR Section 6.2.1.2.3.

The LOCA is the limiting accident related to the environmental qualification (EQ) temperature limits shown in UFSAR Figure6.1-1. The results shown in Figure6.1-1 are based on assuming loss of off-site power with a single failure of one emergency diesel generator. Therefore, the results are based on one SW pump operating during the injection phase of a design basis LOCA; and, two SW pumps operating during the recirculation phase of the LOCA. The results are also based on assuming a maximum service water temperature of 85°F. Although the uprate temperature at 24 hours after a design basis LOCA is greater than the EQ profile for a short period of time, the long-term accident temperature drops below the qualification profile. An aging equivalency analysis was performed for a range of activation energies which demonstrated that the existing equipment qualification profile bounds the uprate accident profile (*Reference 15*).

9.2.1.4.4 Postulated Service Water Pump Discharge Check Valve Failure

In a configuration with two service water (SW) pumps in operation, if one of the pumps trips or is stopped and its discharge check valve fails to close, a reverse flowpath would be created through this idle pump. The start of another service water (SW) pump would allow reverse flow through this idle pump and reduce the service water (SW) flow delivered to the system loads to less than normal flow for two-pump operation. Analysis has shown, though, that the minimum service water (SW) flow delivered to the system loads for this configuration will exceed that of a single service water (SW) pump. Therefore, the postulated discharge check valve failure, which is a passive failure, is less limiting than either the failure of a service water (SW) pump or the failure of a diesel generator assumed in the LOCA and steam line break analyses.

9.2.1.5 Tests and Inspections

All system components were hydrostatically tested prior to station startup and periodic inspections are performed during operation. All electrical components, switchovers, and starting controls are tested periodically.

Ginna Station has equipped service water (SW) system header A with a 20-in. removable flange for periodic underground piping inspection. This access point located in the screen house basement, in addition to a removable section of piping located in the control building air handling room, allows for periodic robotic inspection of the condition of the underground concrete liner of the service water (SW) system header A piping running from the screen house to the auxiliary building.

Ginna Station has instituted a program for periodic performance testing, inspection, or cleaning of critical safety related service water (SW) heat exchangers. Testing and/or cleaning frequencies were established based on previous testing or inspection, consistent with Generic Letter 89-13 guidance.

9.2.2 COMPONENT COOLING WATER (CCW) SYSTEM

The component cooling water (CCW) system is shown in Drawing 33013-1245 and 33013-1246, Sheets 1 and 2.

9.2.2.1 Design Bases

The component cooling water (CCW) system is designed to remove heat from plant components during plant operation, plant cooldown, and during postaccident conditions. Component cooling water circulates through parallel flow paths into various components where it picks up heat from other systems and transfers the heat to the service water (SW) system via the component cooling water (CCW) heat exchangers. The component cooling loop serves as an intermediate system between the radioactive fluid systems and the service water (SW) system. This arrangement reduces the probability of radioactive fluid leakage to the environment via the service water (SW) system. The system design provides for the detection of radioactivity entering the system from any of the components serviced and includes the ability to isolate any component. Active system components that are relied upon to perform the cooling function are redundant.

The component cooling loop is a closed system inside containment. Makeup water is taken from the reactor makeup water transfer pumps and delivered to the component cooling surge tank. A backup source of water is provided from the demineralized water system.

All piping and components of the component cooling water (CCW) system were designed to the applicable codes and standards listed in Table 3.2-1. The component cooling water system contains a corrosion inhibitor to protect the carbon steel piping.

9.2.2.2 System Design and Operation

Component cooling is provided for the following heat sources:

A. Residual heat removal heat exchangers (residual heat removal system).

B. Reactor coolant pumps and motors (reactor coolant system).

C. Nonregenerative heat exchanger (chemical and volume control system).

D. Excess letdown heat exchanger (chemical and volume control system).

E. Seal-water heat exchangers (chemical and volume control system).

F. Boric acid recycle evaporator (chemical and volume control system).

G. Sample heat exchanger (sampling system).

H. Waste evaporator condenser (waste disposal system) (system physically removed in 1999).

I. Waste gas compressors (waste disposal system).

J. Reactor support cooling pads.

K. Residual heat removal pump mechanical seal coolers and bearing water jackets (residual heat removal system).

L. Safety injection pump mechanical seal coolers (safety injection system).

M. Containment spray pump mechanical seal coolers (safety injection system).

At the reactor coolant pump, component cooling water (CCW) removes heat from the bearing oil and the thermal barrier. Since the heat is transferred from the component cooling water (CCW) to the service water (SW), the component cooling loop serves as an intermediate system between the reactor coolant and service water (SW) cooling systems and ensures that any leakage of radioactive fluid from the components being cooled is contained within the plant.

During normal full-power operation, one component water pump supplies flow to both component cooling water heat exchangers, however, cooling one component cooling water heat exchanger can accommodate the heat removal loads. Therefore, especially at lower service water (lake) temperatures, service water may be limited or isolated to one component cooling water heat exchanger. The standby pump provides a 100% backup during MODES 1 and 2. Both pumps and both heat exchangers are utilized to remove the residual and sensible heat during plant shutdown. If one of the pumps or one of the heat exchangers is not operative, safe operation of the plant during cooldown is not affected; however, the time for shutdown is extended.

Based upon the discussion provided above, increasing the component cooling water (CCW) heat exchanger service water (SW) inlet temperature to 85°F has no adverse impact on normal plant operation or plant cooldown. For both scenarios, an increase in SW flow to the component cooling water (CCW) heat exchangers would compensate for the impact of the increased inlet temperature. Additionally, for a plant cooldown scenario, safe operation of the plant during cooldown is not affected; however, the time for shutdown is extended. Normal plant cooldown at uprate with an 85°F lake temperature was evaluated by Westinghouse (*Reference 16*) and demonstrated that plant cooldown to cold shutdown conditions were obtained. Although the time required was increased over that achievable with the pre-uprate power level, it was determined that the operation of one CCW Pump and one RHR Pump were still capable of placing the RCS in Mode 5 in less than 30 hours after a reactor shutdown.

For design basis accident scenarios, the reduction in CCW heat exchanger heat removal capability, in combination with the reduced containment recirculation fan cooler (CRFC) heat removal capability, would slightly decrease the long-term cooling capability of containment following the transfer to the post-LOCA recirculation phase. As discussed in Section 9.2.1.4.3, the resulting impact on the containment cooldown transient has been determined to be bounded by the design basis post-LOCA environmental qualification (EQ) temperature envelope shown in Figure 6.1-1.

The surge tank accommodates expansion, contraction, and inleakage of water, and ensures a continuous component cooling water (CCW) supply until a leaking cooling line can be isolated. Because the tank is normally vented to the atmosphere, a radiation monitor in the component cooling system annunciates in the control room and closes a valve in the vent line in the unlikely event that the radiation level reaches a preset level above the normal background.

9.2.2.3 Component Description

Component Cooling Water Heat Exchangers

The two component cooling water (CCW) heat exchangers located on the upper level of the auxiliary building are of the shell and straight tube type. Service water (SW) circulates through the tubes while component cooling water (CCW) circulates through the shell side. Parameters are presented in Table 9.2-3.

Component Cooling Water Pumps

The two component cooling water (CCW) pumps which circulate component cooling water (CCW) through the component cooling loop are horizontal, centrifugal units. The pump casings are made from cast iron (ASTM 48) based on the corrosion-erosion resistance and the ability to obtain sound castings. The material thickness is indicated by high-quality casting practice and ability to withstand mechanical damage, and, as such, is substantially overdesigned from a stress-level standpoint. The design parameters are listed in Table 9.2-3.

Component Cooling Water Surge Tank

The component cooling water (CCW) surge tank, which accommodates changes in component cooling water (CCW) volume, is constructed of carbon steel. Parameters are presented in Table 9.2-3. Piping is provided for the addition of the chemical corrosion inhibitor to the component cooling loop.

Component Cooling Valves

The valves originally installed in the component cooling water (CCW) system are constructed of carbon steel with bronze or stainless steel trim. Stainless steel bodied valves have been installed on a case-by-case basis as an equivalent replacement. Since the component cooling water (CCW) is not normally radioactive, special features to prevent leakage to the atmosphere are not provided.

Self-actuated spring-loaded relief valves are provided for lines and components that could be pressurized to their design pressure by improper operation or malfunction.

Component Cooling Piping

All component cooling loop piping is carbon steel with welded joints and connections, except at components which might need to be removed for maintenance.

9.2.2.4 System Evaluation

9.2.2.4.1 Availability and Reliability

9.2.2.4.1.1 Accessibility

For component cooling of the reactor coolant pump and the excess letdown heat exchanger inside the containment, most of the piping, valves, and instrumentation are located outside the concrete shields for the reactor vessel, steam generators, and reactor coolant pumps at an elevation above the water level in the bottom of the containment at postaccident conditions. The exceptions are the cooling lines for the reactor coolant pumps and reactor supports, which are not required to be operable following an accident. This location provides a measure of protection from postaccident dynamic conditions and flooding and also provides shielding which allows for maintenance and inspections to be performed during power operation.

Outside the containment, the component cooling pumps and heat exchangers, and associated valves, piping, and instrumentation can be maintained and inspected during power operation. Replacement of one pump or one heat exchanger may be performed while the second units are in service.

9.2.2.4.1.2 Seismic Design

The component cooling loop components are Seismic Category I and were designed to the codes given in Table 3.2-1. In addition, the components of the component cooling loop are not subjected to any high pressures (see Table 9.2-3) or stresses. Hence, a rupture or failure of the system is very unlikely.

9.2.2.4.1.3 Loss of Component Cooling Water System

Valves are provided for isolation of individual leaking components. Also, although the component cooling water (CCW) pumps and heat exchangers are redundant, they are connected by single pipe headers whose failure could disable the system. However, at the operating pressure and temperature of the system (100 psig, 200°F) a passive failure could probably result in a leak rate estimated to be no greater than 210 gpm. The normal volume of water in the surge tank (1000 gal) would provide the operators with about 5 min at a leak rate of 210 gpm to stop a leak from the system. It is improbable that the operator could act within this time period, and it is possible that the leak may be in an unisolable portion of the system. If a loss of the component cooling water (CCW) systems occurs during MODES 1 and 2, an operating procedure directs the operator to shut down the reactor and commence decay heat removal using the steam generators with natural circulation of the reactor coolant system. If component cooling water (CCW) cannot be readily restored, a plant cooldown would be commenced. For a cooldown with no component cooling water (CCW), the cooldown method and system described in *Reference 12* (with the exception of the component cooling water (CCW) and residual heat removal systems) would be available, and a method is available to achieve MODE 5 (Cold Shutdown) conditions independent of the component cooling water (CCW) and residual heat removal systems using the steam generators as described in *Reference 13*.

Loss of the component cooling water (CCW) system during postaccident recirculation operation was considered in the Provisional Operating License review of Ginna Station. At that time it was concluded that the residual heat removal (RHR) pumps could continue to operate to recirculate containment sump water with decay heat being removed by the containment fan coolers. It has since been determined that the mechanical seals for the RHR pumps would require cooling during the recirculation phase. Therefore, during Recirculation the component cooling water (CCW) system is not required for the RHR bearings, but it is required for the RHR Seal Water Heat Exchangers (*Reference 17*). Current criteria for piping system passive failures do not require the assumed passive failures of moderate energy systems (like the component cooling water) under postaccident conditions, although system leaks are assumed (*Reference 4*). Therefore, the component cooling water (CCW) system makeup capability should be capable to cope with normal system leakage in postaccident operation.

The effects of a loss of component cooling water (CCW) during a cooldown of the plant with the residual heat removal system operating have been considered. In this case, with the reactor vessel head installed, the reactor coolant system temperature would rise to greater than 200°F and decay heat could continue to be removed via the steam-generator atmospheric steam dump valves using natural circulation. Steam-generator feed would be accomplished by the preferred auxiliary feedwater system. The plant could remain in this condition while component cooling water (CCW) repairs were made. For normal decay heat removal when the reactor vessel head is removed, adequate cooling can be provided by keeping the core flooded (using various systems such as the residual heat removal and chemical and volume control systems) while repairs are made to the component cooling water (CCW) piping. The component cooling water (CCW) system is accessible for repairs and can be filled with water in less than 2 hours after the repairs are completed starting with a completely drained system.

9.2.2.4.1.4 Component Cooling Water Surge Tank

During normal and postaccident operation, thermal expansion and contraction of the component cooling water (CCW) system liquid is accommodated by the component cooling water (CCW) surge tank, and leakage into or out of the system can be detected by surge tank level changes. High and low surge tank levels are alarmed in the control room, and a radiation monitor and alarm alerts the control room operator to the leakage of radioactive fluid into the component cooling water (CCW) system from components which contain reactor coolant.

The surge tank also maintains a positive suction head on the component cooling water (CCW) pumps during normal and postaccident operation. Makeup water to the component cooling water (CCW) system is normally supplied by the reactor makeup water system via a remotely operated valve in the auxiliary building. The makeup rate is sufficient to accommodate system leakage. The demineralized water system is also a makeup source utilizing manual valves. Installation of redundant water level alarms on the component cooling water (CCW) surge tank ensures early warning and detection of leaks in the component cooling water (CCW) system so that operator action can be taken to prevent damage to the reactor coolant pumps.

9.2.2.4.1.5 Safety-Related Functions

The safety-related functions of the component cooling water (CCW) system are to provide cooling for the residual heat removal heat exchangers and Emergency Core Cooling System

(ECCS) pumps. Other functions of the component cooling water (CCW) system include cooling to the reactor coolant pumps, reactor support cooling pads, excess letdown heat exchanger, and the nonregenerative heat exchanger.

On loss of component cooling water (CCW) flow, plant procedures require the operator to trip the reactor and then trip the reactor coolant pumps. Loss of component cooling water (CCW) flow to the excess letdown heat exchanger or reactor support cooling pads could cause a reactor shutdown, but does not require immediate operator action, and adequate protection is provided by plant procedures.

Regulatory Guide 1.97 recommends instrumentation for the component cooling water (CCW) flow to the engineered safety features system with a range of 0 to 110% of design flow. Although such instrumentation is not provided at Ginna Station, other instruments provide adequate information. Ginna Station has redundant component cooling water (CCW) pumps with pump status indication, as well as component cooling water (CCW) surge tank level indication in the control room. Also, alarms are provided for the following: low surge tank level, low system flow, low system pressure, and low component cooling water (CCW) flow from the residual heat removal, core spray, and safety injection pumps. Thus, substantial information exists to verify operability of the component cooling water (CCW) system.

The component cooling water (CCW) system is normally aligned with each supply line to the residual heat exchangers closed by a motor-operated valve that can be remote manually operated from the control room. This normal alignment is consistent with the cooling requirements during the injection phase of emergency core cooling. One of these valves is required to open upon transfer to the long-term recirculation phase following a loss-of-coolant accident.

9.2.2.4.1.6 Flow-Induced Vibration

To minimize the potential for flow-induced vibration in the component cooling water (CCW) heat exchangers and residual heat removal heat exchangers during normal cooldown and postaccident recirculation modes, analyses were performed (*References 5 through 8*) to determine the effects of reducing flow through the heat exchangers. The analyses supported a reduction in flow and as a result, since 1994, component cooling water (CCW) flow has been limited to approximately 2500 gpm through the shell side of each component cooling water (CCW) heat exchanger. To accomplish this, the component cooling water (CCW) system outlet valves from the residual heat removal heat exchangers (780A and 780B) were throttled and remain in a position of approximately 30 degrees. Analysis showed that throttling these valves would cause the least impact on system valves and would also reduce flow through the residual heat removal heat exchangers to approximately 1800 gpm thereby minimizing the potential for flow-induced vibration in these exchangers.

During normal power operation and the postaccident injection phase, the component cooling water (CCW) system inlet valves to the residual heat removal heat exchangers (MOV-738A and 738B) are in the closed position. Therefore, during these evolutions the component cooling water (CCW) system flow is much less than its rated capacity and flow-induced vibration is not a concern to system reliability.

These flow rate reductions were determined to result in a 6% reduction in heat removal capability of the residual heat removal and component cooling water (CCW) systems (as compared to the design flow rates listed in Tables 5.4-6, 6.3-5, and 9.2-3). The reduction in heat removal capability would not significantly affect cooldown operations. A normal plant cooldown with an 85°F lake temperature and the reduced component cooling water heat exchanger flow rates was evaluated as part of the plant uprate to 1775 MWt (*Reference 16*). The evaluation demonstrated that plant cooldown to Mode 5 conditions in less than 30 hours after a reactor shutdown was achievable with operation of only one CCW Pump and one RHR Pump.

9.2.2.4.2 Leakage Provisions

9.2.2.4.2.1 Introduction

Water leakage from piping, valves, and equipment in the system inside the containment is not considered to be generally detrimental unless the leakage exceeds the makeup capability. With respect to water leakage from piping, valves, and equipment outside the containment, welded construction is used where possible to minimize the possibility of leakage. The component cooling water (CCW) could become contaminated with radioactive water due to one of the following:

a. A leak in any heat exchanger tube in the chemical and volume control, the sampling, or residual heat removal systems, or a leak in the cooling coil for the mechanical seal on a reactor coolant pump.

b. Absorption of radioactive products from the containment air during actual postaccident operations.

9.2.2.4.2.2 Leakage Detection

Reactor coolant leakage into the component cooling loop from components being cooled are detected by the leak detection system described in Section 5.2.5 for components within the containment. Such leaks are detected by a radiation monitor located in the component cooling system and also by an increase in level in the component cooling surge tank.

Leakage from the component cooling loop can be detected by a falling level in the component cooling surge tank. The leaking component can be ascertained by sequential isolation or inspection of equipment in the loop. If the leak is in the on-line component cooling water (CCW) heat exchanger, the standby exchanger would be put on stream and the leaking exchanger isolated and repaired. During MODES 1 and 2, the leaking exchanger could be left in service with leakage up to the capacity of the makeup line to the auxiliary building from the demineralized water system. Should a large tube-side to shell-side leak develop in a residual heat exchanger, the water level in the component cooling surge tank would rise, and the operator would be alerted by a high-water alarm. The atmospheric vent on the tank is automatically closed in the event of high radiation level in the component cooling water (CCW) system. If the leaking residual heat exchanger is not isolated from the component cooling loop before the inflow completely fills the surge tank, the relief valve on the surge tank lifts. The discharge of this relief valve is routed to the auxiliary building waste holdup tank.

Engineering analysis shows that the automatic closure of the component cooling water (CCW) surge tank vent upon detection of reactor coolant system inleakage by the radiation monitor will not overpressurize the system. In this case, the maximum system pressurization was calculated to be approximately 229 psig, which is well below the 500-psig hydrostatic test pressure limit of the component cooling water (CCW) pump seals, the most limiting system component. The pressurization is limited by maximum component cooling water (CCW) pump discharge pressure, maximum system deviation head, and highest-possible relief valve pressure setting.

The severance of a cooling line serving an individual reactor coolant pump cooler would result in substantial leakage of component cooling water (CCW). Several indications and alarms are available to alert the operator of this loss of component cooling water (CCW). The water storage in the surge tank after a low-level alarm, together with makeup flow, provides the operator with time to close the valves external to the containment to isolate the leak. Operator actions are dictated by the Ginna Emergency Procedures to prevent damage to the reactor coolant pumps.

9.2.2.4.2.3 Relief Valves

The relief valves on the component cooling water (CCW) lines downstream from each reactor coolant pump are designed with a capacity equal to the maximum rate at which reactor coolant can enter the component cooling loop from a severance type break of the reactor coolant pump thermal barrier cooling coil. Flow indication is available and isolation valves can be closed to prevent the continued inflow of reactor coolant into the component cooling water (CCW) system. The isolated portion of piping is designed to withstand full reactor coolant system pressure (2500 psig).

The relief valve on the component cooling surge tank is sized to relieve the maximum flow rate of water which enters the surge tank following a rupture of a reactor coolant pump thermal barrier cooling coil prior to the time it is isolated. However, a full break is not required to be considered with respect to causing a LOCA, since component cooling water (CCW) is a moderate energy piping system which only requires consideration of cracks and conservatism in the tube design will prevent tube collapse. Therefore, only cracks in accordance with *Reference 11* need to be postulated. The set pressure of the relief valve is such that none of the components in the component cooling water (CCW) system would be damaged due to the inflow of reactor coolant.

The relief valves on the cooling water lines for the sample, excess letdown, seal-water, nonregenerative, and residual heat exchangers are sized to relieve the volumetric expansion occurring if the exchanger shell side is isolated when cool, and high-temperature coolant flows through the tube side. The set pressure equals the design pressure of the shell side of the heat exchangers.

9.2.2.4.3 **Incident Control**

Since the component cooling water (CCW) system loop is used as an engineered safety feature, containment isolation valves are not automatically closed. That portion of the loop located outside the containment is not required to be a closed system. Each of the cooling water supply lines to the reactor coolant pumps contains a check valve inside and a remote operated valve outside the containment wall. Each return line has a remote operated valve outside the containment wall. The cooling water supply line to the excess letdown heat exchanger contains a check valve (inside the containment wall), normally open supply and return manual isolation valves (located outside the containment wall) and a return line air operated globe valve (outside the containment wall) which are closed during MODES 1 and 2. Except for the normally closed makeup line and equipment vent and drain lines, there are no direct connections between the cooling water and other systems. The equipment vent and drain lines outside the containment have manual valves which are normally closed unless the equipment is being vented or drained for maintenance or repair operations.

Following a loss-of-coolant accident, one component cooling pump and one component cooling heat exchanger accommodate the heat removal loads. If either a component cooling pump or component cooling heat exchanger fails, the standby pump and heat exchanger provide 100% backup. Valves on the component cooling return lines from the safety injection, containment spray, and residual heat removal pumps are locked open. Each of the component cooling supply lines to the residual heat exchangers has a normally closed, remotely operated valve. If one of the valves fails to open at initiation of long-term recirculation, the valve which does open supplies a heat exchanger with sufficient cooling to remove the heat load.

If a break of a cooling line occurs inside the containment, adequate valving is available outside the containment on the component cooling supply and return lines to isolate the leak (see Drawing 33013-1246, Sheet 1). None of the components inside the containment require component cooling water (CCW) during recirculation. If the break occurs outside the containment, the leak could either be isolated by valving or the broken line could be repaired, depending on the position in the loop at which the break occurred.

Once the leak is isolated or the break has been repaired, makeup water is supplied from the reactor makeup water tank by either one of the reactor makeup water pumps or the monitor tank pump. If the loop drains completely before the leakage is stopped, it can be refilled by either a reactor makeup water pump or the monitor tank pump in less than 2 hours.

To comply with Appendix R requirements related to ensuring the capability to achieve cold shutdown within 72 hours and to relieve pump casing brittle fracture concerns, in 1983 RG&E purchased a spare component cooling water (CCW) pump to be stored on site which could be manually placed in service, if needed. Modifications to the Appendix R program as accepted by the NRC (*Reference 9*) and an evaluation by RG&E addressing the brittle fracture concerns (*Reference 10*), later eliminated the commitment to maintain a spare pump.

In the review of SEP Topic III-4A, Tornado Missiles, it was concluded that a loss of the component cooling water (CCW) system due to tornado effects will not compromise safe shutdown capability, because alternative safe shutdown means are available which do not rely on the component cooling water (CCW) system.

9.2.2.4.4 **Malfunction Analysis**

A failure analysis of pumps, heat exchangers, and valves is presented in Table 9.2-4.

9.2.2.5 Instrumentation Requirements

The operation of the component cooling water (CCW) system is monitored with the following instrumentation:

A. Temperature detectors in the main inlet and outlet lines for the component cooling heat exchangers.

B. A pressure detector on the line between the component cooling pumps and the component cooling heat exchangers.

C. A temperature and flow indicator in the outlet line from the heat exchangers.

D. A radiation monitor on the main inlet line to the component cooling pumps.

E. Wide range water level indication with redundant alarm instrumentation at the component cooling water (CCW) surge tank.

The following is a list of alarms that are monitored in the control room:

AA. Component cooling surge tank high level.

BB. Containment spray pump cooling water outlet low flow.

CC. Reactor coolant pumps component cooling water (CCW) return high temperature or low flow.

DD. Residual heat removal pump cooling water outlet low flow.

EE. Component cooling heat exchanger outlet high temperature.

FF. Component cooling pump discharge low pressure.

GG. Component cooling water from reactor support high temperature.

HH. Component cooling pump inlet header high temperature.

II. Component cooling loop low flow.

JJ. Component cooling service water low flow.

9.2.2.6 Minimum Operating Conditions

Minimum operating conditions for the component cooling water (CCW) system are shown in Table 9.2-5 and are part of the Technical Specifications.

9.2.2.7 Tests and Inspections

The active components of the component cooling water (CCW) system are in either continuous or intermittent use during MODES 1 and 2. System motor-operated valves are exercised per surveillance program requirements. Periodic visual inspections and preventative maintenance are conducted following normal industrial practice.

9.2.3 DEMINERALIZED WATER MAKEUP SYSTEM

The condensate demineralizer system which maintains the purity of the feedwater is described in Section 10.7.7.

The GE Betz Water Treatment System provides demineralized water to the reactor makeup water tank, the component cooling water (CCW) surge tank, and the condensate storage tanks. In addition, the GE Betz Water Treatment System provides demineralized water for use throughout the plant. Drawing 33013-1908, Sheet 1, shows the GE Betz Water Treatment System. The reactor makeup water tank provides demineralized water to the chemical and volume control system for primary system makeup as discussed in Section 9.3.4 and to the component cooling water (CCW) system. The condensate storage system is described in Section 9.2.4. The component cooling water (CCW) system is described in Section 9.2.2.

9.2.4 CONDENSATE STORAGE FACILITIES

The condensate and feedwater systems are described in Section 10.4. The condensate storage facilities are described in Section 10.7.4.

REFERENCES FOR SECTION 9.2

1. Nuclear Regulatory Commission, Branch Technical Position ASB 3-1, Appendix A to Appendix C, Criteria for Determination of Postulated Break and Leakage Locations in High and Moderate Energy Fluid Piping Systems Outside of Containment Structures, July 12, 1972.

2. Letter from D. M. Crutchfield, NRC, to L. D. White, RG&E, Subject: SEP Topic III-5.B -Pipe Break Outside Containment, dated June 24, 1985.

3. Letter from R. C. Mecredy, RG&E, to A. R. Johnson, NRC, Subject: Service Water System Operational Performance Inspection, dated September 1, 1992, with attached Summary Report, Long-Term Containment Response to LBLOCA With One Service Water Pump Operating.

4. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: SEP Topic IX-3, Station Service and Cooling Water Systems, dated November 3, 1981.

5. Rochester Gas and Electric Corporation Safety Evaluation SEV-1011, Revision 1, Throttling of CCW to RHR Flow Control Valves 780A and 780B, dated March 2, 1994.

6. Rochester Gas and Electric Corporation Safety Evaluation NSL-0000-SE023, Shutdown Throttling of CCW to RHR Flow Control Valves 780A, 780B, dated March 12, 1993.

7. Rochester Gas and Electric Corporation Design Analysis DA-ME-93-157, Impact of CCW Flow Reduction on CCW and RHR Heat Exchanger Performance, dated December 3, 1993.

8. Rochester Gas and Electric Corporation Design Analysis DA-ME-93-0052, Component Cooling Water Heat Exchanger Flow Analysis for Potential Flow Induced Vibration (FIV), dated August 30, 1993.

9. Letter from J. A. Zwolinski, NRC, to R. W. Kober, RG&E, Subject: Exemptions to Section III.G of Appendix R, dated March 21, 1985.

10. SEV-1010, UFSAR Change on CCW Pump Material, dated December 16, 1993.

11. Rochester Gas and Electric Corporation Design Analysis ME-92-0008, NRC IEN 89-54 Evaluation, dated March 17, 1992.

12. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: Ginna - SEP Topics V-10.B, RHR System Reliability, V-11.B, RHR Interlock Requirements, and VII-3, Systems Required for Safe Shutdown (Safe Shutdown Systems Report), dated May 13, 1981.

13. Letter from L. D. White, RG&E, to D. L. Ziemann, NRC, Subject: Fire Protection - Shutdown Analysis, dated December 28, 1979.

14. Letter from G. S. Vissing, NRC, to R. C. Mecredy, RG&E, Subject: Service Water System at R. E. Ginna Nuclear Power Plant (TAC No. M84947), dated January 29, 1999.

15. Letter from M. Korsnick, Ginna, to Document Control Desk, NRC, "License Amendment Request Regarding Extended Power Uprate," (Letter # 1001353) dated July 7, 2005.

16. Westinghouse Calculation CN-SEE-04-84, "Ginna Uprate Cooldown Analysis," Revision 0.

17. Letter from PWR Systems Division, to J. L. Gallagher, Subject: "Safeguards Pump Operating Without Component Cooling Water", dated February 6, 1969.

18. Ginna Calculation DA-ME-10-026, "Service water Leakage Rate Analysis," Revision 0.

Table 9.2-1
LOADS SUPPLIED BY SERVICE WATER (SW) SYSTEM

Diesel-generator coolers (4) and expansion tank makeup (2)

Condensate pump motor coolers (3)

Heater drain pump motor coolers (2)

Instrument air compressors (3)

Generator exciter cooler (1)

Generator bus duct coolers (2)

Generator seal-oil coolers (2)

Main feed pump lube-oil coolers (2)

Electrohydraulic control oil coolers (2)

Turbine lube-oil coolers (2)

Vacuum priming pumps (2)

Fire service water booster pump supply (1)

Traveling screen flushing valves supply (4)

Seal-water to circulating water pumps (2)

Relay room air conditioning units (2)

Battery room air conditioning unit (1)

Containment air test aftercooler (1)

Air conditioning water chillers (2)

Containment recirculation fan coolers (4 units, 3 coils per unit) and fan motor coolers (4)

Reactor compartment coolers (2)

Component cooling water heat exchangers (2)

Spent fuel pool heat exchangers (2 normal, 1 standby)

Safety injection pump outboard thrust bearing housing oil coolers (3)

Residual heat removal pump room coolers (2)

Charging pump room coolers (2)

Containment penetration cooling

Administrative computer room air conditioner unit (1)

Telephone equipment room air conditioning unit (1)

Degasifier and instrumentation and control shop

Alternative supply to Preferred auxiliary feedwater system (AFW) pumps (3)

Normal supply to standby auxiliary feedwater system (SAFW) pumps (2)

Safety injection and containment spray pump area coolers (3)
(Service water (SW) lines to these were blanked closed in 1992. See Section 9.4.9.1)

Standby auxiliary feedwater pump (SAFW) area coolers (2)

Motor-driven auxiliary feedwater pump (MDAFW) oil coolers (2)

Turbine-driven auxiliary feedwater pump (TDAFW) oil cooler

Turbine-driven auxiliary feedwater pump (TDAFW) pump (outboard) thrust bearing

Motor-driven auxiliary feedwater pumps (MDAFW) outboard thrust bearing (2 pumps)

House heating boiler sample cooler

Component cooling water (CCW) area emergency shower and eyewash

Sample coolers (6)

Secondary cooling temperature control unit

GE Betz water treatment system source water

Table 9.2-2
MAJOR SERVICE WATER SYSTEM FLOWS

Service (Number)	***Design Flow Each (gpm)***[a]	***Typical Flow at Power Two Pumps Total (gpm)***[b]	***Typical Flow at Power Three Pumps Total (gpm)***[b]	***Number of Components Assumed to Receive Service Water Flow During Injection Phase Post LOCA***	***Number of Components Assumed to Receive Service Water Flow During Recirculation Phase Post LOCA***
Containment fan cooler units (4)[c] consisting of:					
Cooling coils (3 per unit)	915[d]	4,769	5,460	4 cooler units	4 cooler units
Fan motor cooler (1 per unit)	31	235	270	4 fan motor coolers	4 fan motor coolers
Subtotal	946	5,004[e]	5,730[e]	NA	NA
Component cooling water (2)[c]	5,070[f]	2,642[g]	4,200[g]	None	2
Reactor compartment coolers (2)	45	98	115	2	2
Diesel generators (2)[c]	320	751	865	2	2
Motor-driven auxiliary feedwater pumps (MDAFW) oil coolers (2)[c]	7	14	14	2	2
Turbine driven auxiliary feedwater pump (TDAFW) oil cooler (1)[c]	25	25	25	1	1
Main turbine lube oil coolers (2)	600	651	735	None	None
Penetration cooler (1)	20	34	40	1	1
Electrohydraulic control oil coolers (2)	20	39	45	None	None
Seal oil coolers (2) (air side/H^2 side)	100/70	259	290	None	None
Exciter (1)	90	308	350	None	None
Pump area coolers					
Safety injection and containment spray (3)	NA[h]	NA[h]	NA[h]	None	None
Residual heat removal (2)	12.5	33	40	2	2
Charging (2)	9	32	40	2	2

Service (Number)	***Design Flow Each (gpm)***[a]	***Typical Flow at Power Two Pumps Total (gpm)***[b]	***Typical Flow at Power Three Pumps Total (gpm)***[b]	***Number of Components Assumed to Receive Service Water Flow During Injection Phase Post LOCA***	***Number of Components Assumed to Receive Service Water Flow During Recirculation Phase Post LOCA***
Safety injection pump bearing housing oil cooling (3)[c]	3	9	9	3	3
Standby auxiliary feedwater(SAFW):					
Pump supplies (2)[c]	215[i]	NA	NA	None	None
Area Coolers (2)[c]	25[i]	NA	NA	None	None
Air compressors (3)	12 [j]	36	36	None	None
Air conditioning	525	270	310	None	None
Sample coolers and chillers (4)	15	41	50	None	None
Bus duct coolers (2)	70	162	180	None	None
Main feedwater pump lube oil coolers (2)	35	82	95	None	None
GE Betz water treatment system source water	600	200	200	None	None
Spent fuel pool heat exchanger A	700	474[k]	530[k]	None	None
Spent fuel pool heat exchanger B[c]	1,600	661[k]	840[k]	None	1
Screen wash (4)	320[l]	505[m]	570[m]	None	None
TOTAL	NA	12,328	15,305	NA	NA
Number of pumps required (4)	NA	2[n]	3[n]	1	2
Service water pump flow (gpm) per pump	5,300	6,164[o]	5,102[op] 4,872[q]		

a. These values represent flows utilized for design purposes and during normal operation (MODES 1 and 2) and testing are applied to the critical loads as alert values.

b. Flows represent typical values determined by hydraulic analysis of the service water (SW) system using a computerized model, which represents the system configuration and which has been baselined against system testing and operations data.

c. These loads have been classified as critical loads as they have either a postaccident function or a function important to safety.

d. Minimum required flowrate for 80°F service water (SW) system temperature and 33,000 cfm air flow is presented. Requirements to ensure containment integrity are based on heat removal rate in Btu/hr in the accident analysis and are dependent on service water flow and temperature and fan cooler air flow rate. See Section 6.2.2.1 and 9.2.1.4.

e. During normal operation (MODES 1 and 2) the actual flowrate to the containment recirculating fan coolers may be throttled based on the setting of the common outlet control valve.

f. Value is based on sizing of the heat exchanger with system valves full open for the purpose of satisfying the maximum required decay heat removal for the normal plant cooldown evolution.

g. During normal operation (MODES 1 and 2), flow to the component cooling water heat exchanger is throttled based on service water temperature and number of service water pumps operating.

h. The service water (SW) piping to these pump area coolers is blanked closed. These coolers are not required for operation of the pumps. See Section 9.4.9.1.

i. The service water (SW) system provides flow (manually initiated) to the standby auxiliary feedwater pumps (SAFW) as a backup to the Preferred auxiliary feedwater system (AFW) only for postulated special events when all auxiliary feedwater pumps are unavailable.

j. Instrument air compressor C was replaced in 1995. The service water (SW) design flow to this new unit is 20 gpm.

k. During normal operation (MODES 1 and 2), flow to the spent fuel pool heat exchanger is throttled based on lake temperature and spent fuel pool load.

l. The screen wash design flow was based on 60 psig at the nozzles. The actual pressure is approximately 50 psig.

m. Two traveling water screen sprays are normally in operation as the screens operate by a timer cycle.

n. The number of service water (SW) pumps in operation while the plant is at power is dependent on lake temperature and pump header pressure.

o. This flow exceeds “rated” service water (SW) pump capacity but is within the maximum runout flow of 7600 gpm which is a net positive suction head (NPSH) limit.

p. Reflects the flow for the service water (SW) pump that is operating on the single pump service water (SW) header during three-pump normal service water (SW) operation.

q. Reflects the flow for each of the two service water (SW) pumps that are operating on the two-pump service water (SW) header during three-pump normal service operation.

Table 9.2-3
COMPONENT COOLING LOOP COMPONENT DATA

Component cooling water pumps	
Quantity	Two
Type	Horizontal centrifugal
Rated capacity, gpm	2980
Rated head, ft H_20	165
Normal cooldown capacity (throttled), gpm[a]	2400
Head at normal cooldown capacity, ft H_2O	180
Capacity during normal power operation, gpm[b]	1300-1400
Motor horsepower, hp	150
Casing material	Cast iron
Design pressure, psig	150
Design temperature, °F	200
Component cooling water heat exchangers	
Quantity	Two
Type	Shell and straight tube
Heat transferred, Btu/hr	[c]25.15×10^6
Shell side (component cooling water)	
Inlet temperature, °F	117
Outlet temperature, °F	100
Flow rate, lb/hr	[c]1.475×10^6(approx 2970gpm)
Design temperature, °F	200
Design pressure, psig	150

Tube side (service water)	
Inlet temperature, °F	80[d]
Outlet temperature, °F	90
Flow rate, lb/hr	2.53×10^6
Design pressure, psig	150
Design temperature, °F	200
Tube material	Admiralty
Shell material[e]	Carbon steel

Component cooling water surge tank

Volume, gal	2000
Volume above normal operating water level, gal	1000
Design pressure, psig	100
Design temperature, °F	200
Construction material	Carbon steel
Relief valve setpoint, psig	100

a. During plant cooldown or postaccident recirculation with residual heat removal heat exchangers in service.
b. Residual heat removal heat exchangers not in service.
c. To minimize the potential for flow-induced vibration in the component cooling water heat exchangers, as of 1994 component cooling water flow has been limited to approximately 2500 gpm through the shell side of each exchanger. See Section 9.2.2.4.1.6.
d. Maximum possible inlet temperature is 85°F. Impact of 85°F inlet temperature on CCW Heat Exchanger performance is discussed in Section 9.2.2.2.
e. In an effort to minimize corrosion on the service water side of the "B" component cooling water heat exchanger, the outlet channel and inlet/outlet tubesheets, which are carbon steel, have been coated with an epoxy.

Table 9.2-4
FAILURE ANALYSIS OF PUMPS, HEAT EXCHANGERS, AND VALVES

Components	Malfunction	Comments and Consequences
Component cooling water pumps	Rupture of a pump casing	The casing and shell are designed for 150 psi and 200°F, which exceed maximum operating conditions. Furthermore, the system can withstand even higher conditions without failure. Rupture due to missiles is not considered credible; however, each unit is isolable and second unit can carry total pumping load.
	Pump fails to start	Additional pump is available. Only one is required to perform the required cooling functions.
	Manual valve on a pump suction line closed	This is prevented by prestartup and operational checks. Further, during normal operation (MODES 1 and 2), each pump is operated on a periodic basis, which would show if a valve is closed.
Component cooling water	Valve on discharge line sticks closed	The valve is shown to be open during periodic operation of the pumps during normal operation (MODES 1 and 2).
Check valve at inlet penetrations	Sticks closed	For flow loops required for normal operation (MODES 1 and 2), there is flow through this line at all times. Hence, the valve is normally open and that it sticks closed at the time of accident is considered incredible.
Component cooling heat exchanger	Tube or shell rupture	Rupture is considered improbable because of the low operating pressure. Each unit is isolable. Second unit can carry total heat load for normal operation (MODES 1 and 2).
Demineralized water makeup line check valve	Sticks open or manual valve is open	The check valve is backed up by the manually operated valve. Manual valve is normally closed.
Component cooling heat exchanger vent or drain valve	Left open	This is prevented by prestartup and operational checks. On the operating unit such a situation is readily assessed by makeup requirements to system. On the second unit such a situation is ascertained during periodic testing.

Table 9.2-5
MINIMUM ALLOWED COMPONENTS FOR THE COMPONENT COOLING WATER (CCW) SYSTEM

Component	Number Installed	Minimum to[a] ~~be Operative~~
Component cooling pumps	2	2
Component cooling heat exchangers	2	2
Residual heat removal heat exchangers	2	2
Service water pumps	4	2

a. As defined in the Ginna Station Technical Specifications, certain components may be out of service for specified time durations without requiring plant shutdown.

9.3 PROCESS AUXILIARIES

9.3.1 INSTRUMENT AND SERVICE AIR SYSTEMS

9.3.1.1 System Description

The instrument air system supplies clean, dry air for valve operators, and piping penetration pressurization. The service air system supplies air for maintenance and service use and the backup eductor for vapor extraction of the turbine-generator bearing drains. A backup source of air supply to the instrument air header is from the service air system. The flow diagram of the service air systems is shown in Drawing 33013-1886, Sheets 1 and 2.

The instrument air system produces 120 to 125 psig dry, filtered air used chiefly as the motive power for valve actuation. The system consists of three air compressors with an associated aftercooler and air reservoir for each compressor. Air from the receivers is supplied to the instrument air header through filters and an air dryer. The instrument air header delivers air to the various valve actuators, piping penetration pressurization system, and containment air and proof test system. The instrument air compressors, receivers, filters, and dryers are shown in Drawing 33013-1900, Sheets 1 and 2.

The service air system produces 115 to 125 psig dry, filtered air used in the maintenance air connections throughout the station, for fire water storage tank pressurization, and the turbine lube-oil system. The system consists of one air compressor with an integral aftercooler and associated air receivers. A cross-tie between service air and instrument air allows the service air system to supply the instrument air header if instrument air pressure drops below 90 psig. The cross-tie occurs prior to the instrument air filters. Therefore, air being supplied to the instrument air header will always pass through the filters and dryer. A cross-connect between the service air system and the instrument air system allows both systems to be supplied by a single rotary screw air compressor. A pressure regulator valve will stop air flow to the service air system if pressure on the service air side drops below 100 psig. Administratively, the instrument air system and service air system are cross-connected only when one of the rotary screw air compressors is in operation.

The instrument and service air compressors, aftercoolers, air receivers, and filters are located in the turbine building basement (253-ft level).

All controls and instrumentation that are required for safe operation and shutdown of the plant are electrical. Instrument air is used chiefly as the motive power for valve actuation. Air supply failure does not affect the safe operation of the plant; it affects only the means of positioning air controlled equipment. All air-operated containment isolation valves are listed in Tables 6.2-29 and 6.2-32. Table 6.2-32 lists the effects of loss of air supply on air-operated valves containment isolation valves. The effects of loss of air to these valves were considered in the safety analysis of all systems in the plant.

Instrument air distribution is shown in Drawings 33013-1887 through 33013-1899. Drawings 33013-1887 and 33013-1888 shows instrument air in the containment building. Drawings 33013-1889 through 33013-1892 shows instrument air in the auxiliary building. Drawing 33013-1893 shows instrument air in the intermediate building. Drawing 33013-1894 and

33013-1895 shows instrument air in the turbine building. Drawing 33013-1896 shows instrument air in the turbine building and screen house. Drawing 33013-1897, Sheets 1 and 2 shows instrument air in the all-volatile-treatment (condensate demineralizer) building. Drawings 33013-1898 and 33013-1899 shows instrument air in the service building.

The instrument air system, although supplying valves in safety-related systems, is not designed as a safety-related system. All safety-related systems using instrument air are designed such that upon loss of air pressure each component will fail in a position of greater safety.

9.3.1.2 Component Description

9.3.1.2.1 Compressors

The instrument air and service air compressors are comprised of a combination of two types of equipment; two two-stage rotary screw type compressors and two vertical, canned, piston type compressors. All air compressors utilize oil-free construction to minimize the possibility of introducing oil into the instrument air system. The piston type instrument air compressors include an attached aftercooler. The rotary screw air compressors include integral intercooler and aftercoolers. The compressor air receivers are located adjacent to the compressors.

The two rotary screw air compressors are controlled locally by microprocessors. There is also a start-stop control switch located on the back of the main control board. The rotary screw compressors load or unload based on system pressure.

The two piston type compressors have local control capabilities for constant, off, and automatic operation and a START-STOP control switch located on the back of the main control board. Normally, compressor operation is controlled by the local selector switches. When a compressor is placed in the constant operation configuration, the associated compressor will start and run continuously. In this mode of operation, the associated compressor is loaded or unloaded based upon system pressure.

The two piston type instrument air compressors, when in the automatic configuration, will start and run continuously when the instrument air header pressure drops to 110 psig.

9.3.1.2.2 Aftercoolers

The instrument air aftercoolers used with the piston type compressors provide cooling of the compressed air. The aftercoolers are counterflow shell and tube type heat exchangers. A solenoid-operated service water inlet isolation valve opens whenever the compressor is running. Service water flow rate is throttled by the temperature control valve in the flow path between the aftercooler and compressor. A moisture separator is located at the air outlet of the aftercooler. Any condensation resulting during cooling is removed by the moisture separator and drained to the waste system.

The two stage rotary screw instrument air compressor utilizes an intercooler following the first stage of compression and an aftercooler following the second stage of compression.

Both heat exchangers utilize counter flow shell and tube heat exchangers. A moisture trap is located on the outlet of both heat exchangers to remove condensation to the waste system.

The two stage rotary screw service air compressor is air cooled. The integral intercooler and aftercooler are designed to achieve a 20°F aftercooler approach temperature. A moisture trap is located on the outlet of both coolers to remove condensation to the waste system.

9.3.1.2.3 Air Receivers

The air receivers provide a storage volume of compressed air. The service and instrument air receivers are located adjacent to the compressor. The air receiver is provided with a safety valve, moisture drain trap, and pressure indications. The safety valves on the service and instrument air receivers are set for 135 psig.

The three instrument air receivers supply a common air header to the filters and air dryers. Also connected to this common header ahead of the filters and dryers is the service air crosstie. A pressure regulating valve in this line will automatically supply the instrument air header from service air if the instrument air pressure drops below 90 psig.

9.3.1.2.4 Filters and Dryers

Two heaterless air dryers in the instrument air header reduce the dewpoint of the air to -70°F at atmospheric pressure. A prefilter before each drying unit removes entrained moisture and oil to prevent fouling of the dehydration towers. An automatic drain trap directs any moisture or oil collected to the waste system. Each dryer unit contains two desiccant-filled absorption towers. An automatic timer controls the air flow such that one tower per unit is in the drying stage while the other is being regenerated. Regeneration is accomplished by passing dry air through the regenerating tower and venting the moisture-laden air. Dry air from each dryer is passed through an afterfilter to remove any dessicant dust which may be present in the air. To eliminate problems due to corrosion particles in the instrument air system, all piping and valves downstream of the air dryers are brass or copper alloy. The two stage rotary screw service air compressor utilizes a similar heaterless air dryer.

The two stage rotary screw instrument air compressor utilizes absorption/heat of compression type dryer to remove any moisture from the compressed air. A rotor, made of material designed to absorb water, rotates slowly and is passed by two flows of air. A flow of wet compressed air to be dried passes through three quarters of the rotor. A flow of hot compressed air is passed through one quarter of the rotor to regenerate the drying material after moisture is absorbed from the wet compressed air. The dewpoint of air discharged to the remainder of the system will be -22°F at a temperature and pressure of 68°F and 100 psig, respectively. This drying process neither utilizes nor produces any type of particulate, therefore no downstream filtration is required.

9.3.2 SAMPLING SYSTEMS

9.3.2.1 Nuclear Sampling System

<u>9.3.2.1.1 Design Bases</u>

9.3.2.1.1.1 Functional Requirements

The nuclear sampling system provides representative primary coolant samples for laboratory analysis during MODES 1 and 2. Typical information obtained from the analyses includes reactor coolant boron and chloride concentrations; fission product radioactivity level; corrosion product concentration and chemical additive concentrations; and oxygen, hydrogen, and fission gas content. The system has no active emergency function, but it interfaces with the postaccident sampling system (see Section 9.3.2.3).

The system is capable of obtaining reactor coolant samples during reactor operation and during cooldown when the system pressure is low and the residual heat removal loop is in operation. Access is not required to the containment for the collection of samples.

Equipment for sampling secondary and nonradioactive fluids is separated from the equipment provided for reactor coolant samples (Section 9.3.2.2). Leakage and drainage resulting from the sampling operations are collected and drained to tanks located in the waste disposal system.

Two types of samples are obtained by the system: high temperature-high pressure reactor coolant system and steam generator blowdown samples which originate inside the reactor containment, and low temperature-low pressure samples from the chemical and volume control and auxiliary coolant systems.

<u>High Pressure-High Temperature Samples</u>

A sample connection is provided from each of the following:

a. The pressurizer steam space.
b. The pressurizer liquid space.
c. Reactor coolant system hot legs A and B.
d. The steam generator blowdown from each steam generator.

<u>Low Pressure-Low Temperature Samples</u>

A sample connection is provided from each of the following:

a. The chemical and volume control system mixed-bed demineralizer inlet header.
b. The chemical and volume control system mixed-bed demineralizer outlet header.
c. The residual heat removal loop, just downstream of the heat exchangers (Sample line is cut and capped, and no longer in service).
d. The volume control tank gas space (Sample line is cut and capped, and no longer in service).

The high pressure-high temperature samples leaving the sample heat exchangers are held to a maximum temperature of 127°F to minimize the generation of radioactive aerosols.

All components, piping, and valves of the nuclear sampling system are designed to the applicable codes listed in Table 9.3-1 and in Section 3.2.

9.3.2.1.1.2 Operational Requirements

The nuclear sampling system is designed to be operated manually, and on an intermittent basis under conditions ranging from full power operation to MODE 5 (Cold Shutdown). In the design of sampling system piping five design requirements were imposed:

a. The piping is routed such that representative samples of the primary system can be obtained.
b. The piping internal diameter is sized such that the quantity of liquid that must be purged in order to obtain a representative sample is minimized.
c. The piping internal diameter is sized such that sample fluid velocities are high enough to maintain suspended solids in solution.
d. Flow through the system is limited during normal and accident conditions to prevent the release of fission products beyond the limits of 10 CFR 20. During postaccident operation, the postaccident sampling system may be used for highly radioactive material sampling and analysis (see Section 9.3.2.3).
e. Piping runs contain sufficient delay in order to minimize radiation exposure of the sample system operator (N-16 Gamma).

9.3.2.1.2 System Design and Operation

9.3.2.1.2.1 Sampling System

The nuclear sampling system, shown in Drawing 33013-1278, Sheets 1 and 2, provides the representative samples for laboratory analysis. Analysis results provide guidance in the operation of the reactor coolant and chemical and volume control systems. Analyses show both chemical and radiochemical conditions.

Typical information obtained includes reactor coolant boron, chloride, and fluoride concentrations, fission product radioactivity level, hydrogen, oxygen, and fission gas content, corrosion product concentration, and chemical additive concentration.

The information is used in regulating boron concentration adjustments, evaluating fuel element integrity and mixed-bed demineralizer performance, and regulating additions of corrosion controlling chemicals to the systems. The sampling system is designed to be operated manually, on an intermittent basis. Samples can be withdrawn under conditions ranging from full power to MODE 5 (Cold Shutdown).

Reactor coolant liquid and steam lines, which are normally inaccessible or which require frequent sampling, are sampled by means of permanently installed tubing leading to the sampling room.

Sampling system equipment is located inside the intermediate building with most of it in the sampling room. The delay coil and sample lines with remotely operated valves are located inside the reactor containment.

9.3.2.1.2.2 Reactor Coolant Samples

Reactor coolant liquid from both hot legs, pressurizer liquid, and pressurizer steam samples originating inside the reactor containment flow through separate sample lines to the sampling room. Each of these connections to the reactor coolant system has a remote operated isolation valve located close to the sample source. The samples pass through the reactor containment, to the intermediate building, and into the sampling room, where they are cooled (pressurizer steam samples condensed and cooled) in the sample heat exchangers. The sample stream pressure is reduced by a manual throttling valve located downstream of each sample pressure vessel. The sample stream is purged to the volume control tank in the chemical and volume control system until sufficient purge volume has passed to permit collection of a representative sample. After sufficient purging, the sample pressure vessel is isolated and then disconnected for laboratory analysis of the contents.

Alternately, liquid samples may be collected by bypassing the sample pressure vessels. After sufficient purge volume has passed to permit collection of a representative sample, a portion of the sample flow is diverted to the sample sink where the sample is collected.

The reactor coolant sample originating from the residual heat removal loop of the auxiliary coolant system has a remote operated, normally closed isolation valve located close to the sample source. This sample line is cut and capped, and no longer in service.

9.3.2.1.2.3 Chemical and Volume Control System Samples

Liquid samples originating at the chemical and volume control system letdown line at demineralizer inlet and outlet pass directly through the purge line to the volume control tank. Samples are obtained by diverting a portion of the flow to the sample sink. If the pressure is low in the letdown line, the purge flow is directed to the chemical drain tank. The sample line from the gas space of the volume control tank is cut and capped, and no longer in service.

9.3.2.1.2.4 Steam Generator Liquid Samples

Samples of the steam generator liquid are obtained from the blowdown lines (see Drawing 33013-1278, Sheets 1 and 2). These sample lines are routed separately from each steam generator into the sample room where the liquid is cooled and the pressure reduced. Each individual sample is then split into two routes: one goes to the sample sink to provide samples for chemical analysis; in case of a primary to secondary steam generator leak, the second goes to a radiation monitor and then to drain. This second line handles a continuous flow for a constant reading of conductivity and a constant monitoring for radiation. These lines are missile protected within the containment and are equipped with an automatic isolation valve and manual isolation valve in each line immediately outside the containment. The automatic isolation valve is closed upon receipt of a signal from the blowdown sample radiation monitor or the containment isolation system.

9.3.2.1.2.5 Sample Sink

The sample sink, which is contained in the laboratory bench as a part of the sampling hood, contains a drain line to the waste disposal system. A sample hood is provided around the valves at the containment penetration area, which directs airborne activity that may result from valve leakage to the intermediate building ventilation system.

9.3.2.1.2.6 Instrumentation

Local instrumentation is provided to permit manual control of sampling operations and to ensure that the samples are at suitable temperatures and pressures before diverting flow to the sample sink.

9.3.2.1.2.7 Steam Generator Blowdown

See Section 9.3.2.2.1 for a discussion of steam generator blowdown sampling for secondary side chemistry control.

9.3.2.1.3 Component Description

Design parameters of the nuclear sampling system components are listed in Table 9.3-2.

9.3.2.1.3.1 Sample Heat Exchangers

Five sample heat exchangers reduce the temperature of samples from pressurizer steam space, pressurizer liquid space, the hot legs, and each steam generator to 127°F or less before samples reach the sample vessels and sample sink. The tube side of the heat exchangers is austenitic stainless steel, the shell side is carbon steel.

The inlet and outlet tube sides have socket-weld joints for connections to the high-pressure sample lines. Connections to the component cooling water lines are socket-weld joints. The samples flow at 0.42 gpm through the tube side and component cooling water from the auxiliary coolant system circulates through the shell side.

9.3.2.1.3.2 Delay Coil

The hot-leg sample lines contain a delay coil, consisting of coiled tubing, which has sufficient length to provide at least a 40-sec sample transit time within the containment and an additional 20-sec transit time from the containment to the sampling hood. This allows for decay of short-lived isotopes to a level that permits normal access to the sampling room.

9.3.2.1.3.3 Sample Pressure Vessels

The pressurizer sample trains, the residual heat removal loop sample train, and the volume control tank gas space sample train each contain sample pressure vessels which are used to obtain liquid or gas samples. The RHR loop and VCT gas space sample lines are cut and capped, and no longer in service. The hot-leg sample lines have a single sample pressure vessel in common. Integral isolation valves are furnished with the vessel and quick-disconnect coupling valves containing poppet-type check valves, are connected to nipples extending from the valves on each end. The vessels, valves, and couplings are austenitic stainless steel.

9.3.2.1.3.4 Sample Sink

The sample sink is located in a hooded enclosure which is equipped with an exhaust ventilator. The work area around the sink and the enclosure is large enough for sample collection and storage for radiation monitoring equipment. The sink perimeter has a raised edge to contain any spilled liquid.

The enclosure is penetrated by sample lines from the reactor plant, a demineralized water line, and possibly steam system lines, all of which discharge into the sink. The sink and work area are stainless steel.

9.3.2.1.3.5 Piping and Fittings

All liquid and gas sample lines are austenitic stainless steel tubing and are designed for high pressure service. With the exception of the sample vessel quick-disconnect couplings and compression fittings at the sample sink, socket-welded joints are used throughout the sampling system. Lines are so located as to protect them from accidental damage during routine operation and maintenance.

9.3.2.1.3.6 Instrumentation

A temperature detector is located in the sample line downstream of each of the sample heat exchangers to provide the sample system operator with local temperature information. A pressure indicator is located in the sample line downstream of each sample pressure vessel. A flow meter is provided in the purge line to the volume control tank for use when purging sample lines.

9.3.2.1.3.7 Valves

Remotely operated stop valves are used to isolate all sample points and to route sample fluid flow inside the reactor containment. Manual stop valves are provided for component isolation and flow path control at all normally accessible sampling system locations. Manual throttle valves are provided to adjust the sample flow rate as indicated in Drawing 33013-1278, Sheets 1 and 2.

Check valves prevent gross reverse flow of gas from the volume control tank into the sample sink. There is a check valve in the sample line from the residual heat removal loop to prevent overpressurization of the residual heat removal system due to backflow from the reactor coolant system. The RHR loop and VCT gas space sample lines are cut and capped, and no longer in service.

All valves in the system are constructed of austenitic stainless steel or equivalent corrosion resistant material.

Each sample line coming from the containment and the sample line from the residual heat removal loop contain an air-operated valve operated from outside the sampling room. The valves fail closed on loss of air.

Each sample line penetrating containment contains an air-operated globe valve used for containment isolation. These valves are operated from the main control room or local operating station outside the sample room and are closed automatically on a containment isolation signal. They fail closed on loss of air to the valve. To resolve concerns identified in Generic Letter (GL) 96-06 (Section 6.5.1.3), bypass lines with check valves were installed in 1997 inside containment around the air-operated sample line valves at penetrations 205, 206a and 207a.

9.3.2.1.4 System Evaluation

9.3.2.1.4.1 Availability and Reliability

Neither automatic nor operator action is required of the sampling system during an emergency or to prevent an emergency condition. The system is therefore designed in accordance with standard practices of the chemical processing industry.

9.3.2.1.4.2 Leakage Provisions

Leakage of radioactive reactor coolant from the system within the containment evaporates to the containment atmosphere and is removed by the cooling coils of the recirculation air heating and cooling system. Leakage of radioactive material from the most likely places outside the containment is collected by placing the entire sampling station under a hood provided with an offgas vent to waste gas processing. Liquid leakage from the valves in the hood is drained to the chemical drain tank.

9.3.2.1.4.3 Malfunction Analysis

To evaluate system safety, the failures or malfunctions were assumed concurrent with a loss-of-coolant accident, and the consequences analyzed. The results are presented in Table 9.3-3. From this evaluation it is concluded that proper consideration has been given to station safety in the design of the system.

9.3.2.1.5 Minimum Operating Conditions

All radioactive wastes are sampled and evaluated prior to release.

9.3.2.1.6 Tests and Inspections

The frequency and description of sample analyses are included in the Technical Specifications and the Technical Requirements Manual (TRM).

9.3.2.2 Nonnuclear Sampling System

The secondary sampling system is provided with a number of sampling points. All sample points are provided with manual sampling capability. Inline analyzers are provided for selected parameters to allow continuous information useful in evaluating secondary conditions and in developing corrective actions when required. Drawing 33013-2711, Sheets 1 through 4 shows the nonnuclear sampling systems.

9.3.2.2.1 Steam Generator Blowdown Sampling

Steam Generator blowdown analysis provides the closest approximation of the chemistry which exists inside the steam generator. Steam generator blowdown water samples provide early indication of impurity ingress to the secondary system because of the concentrating effect of boiling. Parameters monitored continuously are:

1. pH.
2. Conductivity.
3. Cation conductivity.
4. Chloride.
5. Sulfate.
6. Sodium.

The sampling system conditions samples from each steam generator blowdown header and provides the conditioned samples to inline analyzers and to a manual sample point.

9.3.2.2.2 Hotwell Sampling

Two hotwell sample pumps allow sampling of the individual hotwell sections. Each hotwell sample can be analyzed for sodium and cation conductivity. Condenser leaks of approximately 0.05 gpm can be detected and isolated to a specific waterbox with the online sodium analyzers.

9.3.2.2.3 Condensate Sampling

A sample connection on the condensate pump outlet header feeds inline pH, cation conductivity, ion chromatograph, and dissolved oxygen analyzers at the secondary sample sink.

9.3.2.2.4 Feedwater Sampling

A feedwater sample taken from a point between the high pressure feedwater heaters outlet and the main feedwater regulating valve (MFRV) feeds inline pH, conductivity, cation conductivity, dissolved oxygen, corrosion products, ion chromatograph, and hydrazine analyzers at the secondary sample sink.

9.3.2.2.5 Main Steam Sampling

A sample connection on each main steam line feeds an inline cation conductivity analyzer at the secondary sample sink.

9.3.2.2.6 Heater Drain Tank Sampling

A sample connection on the heater drain tank discharge line feeds an inline cation conductivity, and corrosion products analyzer at the secondary sample sink.

9.3.2.2.7 Sampling Cooling

Cooling for the samples is accomplished by a service water primary cooling header with primary heat exchangers, along with secondary cooling by a closed cooling water system containing a temperature control unit with secondary heat exchangers. The samples that require both primary and secondary cooling are:

1. Feedwater to steam generator.
2. Steam generator blowdown.
3. Main steam samples.
4. Heater drain pump discharge.

The samples that require only secondary cooling are:

1. Condenser hotwell sample pump discharge.
2. Condensate pump discharge.

9.3.2.3 Postaccident Sampling System

9.3.2.3.1 Design Bases

The postaccident sampling system is designed to meet the postaccident sampling requirements of NUREG 0578 and NUREG 0737, Item II.B.3, and to meet the containment sump sampling and pH, and oxygen analysis requirements of Regulatory Guide 1.97, Revision 2.

In January 2002, the NRC staff issued Amendment No. 81 to the Ginna Station Technical Specifications (*Reference 4*). This amendment eliminated the Technical Specifications requirement for having and maintaining a postaccident sampling system. However, as a condition of this amendment to the Technical Specifications, three regulatory commitments were made. Initially, it is not intended to physically eliminate or modify the postaccident sampling system. The commitments will continue to be satisfied by maintaining and using the appropriate portions of the PASS. For future reference, the regulatory commitments are effective with the implementation of the license amendment and are as follows:

1. Maintain contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. The contingency plans are contained in plant procedures.
2. Maintain a capability for classifying fuel damage events for core damage levels approximating radioactivity levels of 300 μCi/gm dose equivalent iodine. This capability is described in plant procedures.
3. Maintain the capability to monitor radioactive iodines that have been released to offsite environs. This capability is described in plant procedures.

The postaccident sampling system is designed to allow the station to obtain and analyze reactor coolant, containment air, and containment sump samples within 3 hours after the decision is made to sample. The postaccident sampling system also permits routine sampling of these process streams during MODES 1 and 2. In-line chemical instrumentation is provided in a liquid and gas sample panel which remotely determines important chemical parameters of the reactor coolant, containment air, and containment sump A. In addition, the liquid and gas sample panel enables acquisition of both diluted and undiluted grab samples of the reactor coolant and containment air for isotopic analysis in the counting lab. Radiation exposure during postaccident sampling is minimized by operating the postaccident sampling system liquid and gas sample panel remotely from an electric control panel and instrument panel located in the hot shop.

The postaccident sampling system is nonseismic except for the component cooling water and volume control tank purge line tie-ins, which are Seismic Category I. The supports of the system components are designed to ensure their structural integrity and the integrity of the interfacing Seismic Category I structures in the event of a seismic occurrence.

9.3.2.3.2 System Description

The postaccident sampling system shown in Drawing 33013-1279 consists of the following components:

A. One liquid and gas sample panel.

B. One electrical control panel and instrumentation panel.

C. Four postaccident sampling system coolers.

D. One postaccident sampling system waste tank.

E. One postaccident sampling system tank transfer pump.

F. One postaccident sampling system waste tank evacuating compressor.

G. One containment sump A sample pump.

H. Seven gas bottles.

I. Piping, valves, and other instrumentation.

The shielded liquid and gas sample panel located at elevation 235 ft in the south section of the intermediate building provides reactor coolant, containment sump, and containment atmosphere sampling capability. Control and monitoring of the liquid and gas sample panel is accomplished remotely by the electrical control panel and instrument panel located in the hot shop at elevation 253 ft approximately 30 ft away from the liquid and gas sample panel. Analytical laboratories are available onsite in the service building at elevation 271 ft and in the ground floor of the Training building. Grab samples of the containment sump fluid and reactor coolant can be collected at the liquid and gas sample panel, transported in a lead-shielded container to the next elevation and passed through a "passbox" to the radiochem lab for analysis. Containment air samples are also collected at the liquid and gas sample panel and transported manually to the "passbox" for isotopic analysis in the radiochem lab. Containment hydrogen monitoring and radiation monitoring of containment air are described in Sections 6.2.5.1 and 6.2.1.5.3, respectively.

9.3.2.3.3 Component Description and Operation

9.3.2.3.3.1 Liquid and Gas Sample Panel

The liquid and gas sample panel provides reactor coolant, containment sump, and containment atmosphere sampling capability. All postaccident sample system analysis and sample components which contain postaccident liquids and gases are mounted in the liquid and gas sample panel behind a shield structure. The panel incorporates integral lead shot and steel shielding to limit operator radiation exposure levels from the panel components. The rear of the panel is enclosed and includes provisions for exhausting to the plant ventilation system to prevent airborne contamination of the sample area. An integral spray system is provided in the panel for washdown and decontamination prior to maintenance.

The panel can be used for routine in-line chemical analysis. Reactor coolant gas stripping and sampling during MODES 1 and 2 is provided. Under accident conditions, the panel provides reactor coolant gas stripping, liquid and gas dilution, in-line chemical analysis, and liquid and gas grab sample capabilities. All high-level samples are confined to the heavily shielded areas.

The postaccident sampling system functional requirements are listed in Table 9.3-4. The inline instrumentation required to achieve those requirements are contained in the liquid and gas sample panel. The panel contains the means to:

a. Strip reactor coolant of dissolved gases for subsequent analysis.

b. Obtain an undiluted containment air sample.

c. Dilute reactor coolant, containment air, and containment sump samples by a factor of approximately 1000. The diluted samples can be withdrawn by a shielded syringe via port connections located on the front of the liquid and gas sample panel. The liquid and gas sample panel also contains a means to obtain an undiluted reactor coolant gas and liquid grab sample during MODES 1 and 2.

Liquid purges during MODES 1 and 2 can go to the volume control tank or to the postaccident sampling system waste tank. Liquid samples normally gravitate down to the postaccident sampling system waste tank which is located underneath the liquid and gas sample panel. In addition, the liquid and gas sample panel has an overpressure relief valve that relieves to the postaccident sampling system waste tank.

Flushing flows are set at a rate (about 1900 cm^3/minute) so that representative samples are available in about 10 minutes.

To minimize personnel exposure during accident conditions, the liquid and gas sample panel integral lead shielding is of sufficient thickness to limit the direct radiation dose at 1 m in front of the panel to less than 100 mR/hr at 1 hr after the accident from sources within the panel, excluding backscatter and other background sources. This ensures that the total dose that a single operator can receive while obtaining and analyzing a single sample is 5 rem whole body and 75 rem to the extremities.

The specific analyses that the liquid and gas sample panel is required for, as well as the required types of instrumentation, range, and accuracy, are included in Table 9.3-5.

9.3.2.3.3.2 Gas Sampling

The gas sampling section of the liquid and gas sample panel provides the capability to sample the containment atmosphere. Samples are obtained with an internal vacuum pump, thereby providing assurance that samples can be obtained under positive or negative containment pressures. To ensure representative sampling, the entire system is purged prior to sample acquisition and analysis. Postaccident gas samples are routed to the containment after exiting the panel.

After adequate purging has been completed, gas samples may be routed directly to the liquid and gas sample panel mounted gas chromatograph for in-line hydrogen and oxygen analysis. The analysis portion of the instrument is located in the liquid and gas sample panel and the control and readout portion is located on the instrument panel.

Gas samples may also be routed through a dual range dilution loop to reduce the specific activity to a level that is acceptable for grab sampling. Adequate dilution capability is provided by this loop to allow isotopic analysis of postaccident gas samples with existing hot-lab equipment.

The basic concept of the dilution loops involves capturing an undiluted gas sample "bite" in either of two dilution loops of preset fixed volume. The captured gas is then purged with argon into a preevacuated vessel of fixed volume. Finally, additional argon is added to achieve a preselected vessel pressure. The use of two dilution loops of different volume provides a dual-range dilution capability. The final vessel pressure setpoint can also be changed to provide variation in the ultimate dilution ratio.

The panel provides the capability of obtaining grab samples of primary coolant dissolved gases and containment atmosphere through septum ports located on the front of the liquid and gas sample panel. A shielded syringe is used for sample acquisition. Normal operation undiluted gas samples can also be obtained. Valve interlocks are provided to prevent undiluted gas sampling under accident conditions.

The gas sampling system can be evacuated and purged with nitrogen after each sample operation to reduce radiation levels.

9.3.2.3.3.3 Liquid Sampling

System pressure or the containment sump sample pump provides the motive force to obtain liquid samples from all sources. The liquid system can be purged prior to each operation to ensure that representative sampling and analysis is achieved. During MODES 1 and 2, the purged liquid will be routed to the waste disposal system. Under accident conditions, valve interlocks are provided to ensure that the purge can only be routed to the postaccident sampling system waste tank and, ultimately, returned to the containment.

After the system is purged, liquid samples can be routed to various in-line instruments for analysis. Each device has been qualified for use during MODES 1 and 2 as well as under postaccident conditions. Conductivity, pH, and dissolved oxygen are measured in-line using sample probes. Boron analysis is provided by an automatic titration device. Chloride analysis is provided through analysis of the diluted liquid sample in the chemistry laboratory. Analysis parameters are displayed on the instrument panel.

Liquid samples can be processed through a gas stripping loop to remove dissolved gases. The stripped gas can then be routed to the gas sampling system for hydrogen analysis, dilution, and/or grab sampling. Normal operation stripped gas may be collected undiluted for routine isotopic analysis. Interlocks are provided to allow grab sampling of only diluted stripped gas in a postaccident situation.

Postaccident liquid samples from the stripping loop are processed through a dilution loop prior to grab sampling. The loop provides a nominal 1:1000 diluted sample for isotopic analysis, chemical analysis, or offsite shipment.

The liquid section of the liquid and gas sample panel incorporates features to flush components with demineralized water after each operation to reduce radiation levels. Liquid wastes from the analysis and dilution sections of the panel are routed to the waste holdup tank during MODES 1 and 2 and to the postaccident sampling system waste tank during postaccident use.

9.3.2.3.3.4 Instrument Panel

The instrument panel provides remote indication of analytical parameters associated with operation of the liquid and gas sample panel. The panel also houses the controls for the gas chromatograph. Chemicals required for operation and calibration of the in-line instruments are located in the instrument panel. The necessary support equipment for the in-line boron analysis is housed in the instrument panel.

The instrument panel contains its own controls, instruments, and mechanical components necessary for its operation. Since no sample fluids enter the panel, shielding is not required. The instrument panel is located in an area away from the liquid and gas sample panel in a low dose rate area to reduce operator exposure associated with monitoring and control functions. The following display devices are provided on the instrument panel:

a. Digital pressure and temperature indicators.

b. pH indicator.

c. Conductivity monitor.

d. Oxygen monitor.

e. Recorders for hydrogen and oxygen concentration.

f. Boron concentration meter (integral with analyzer controls).

9.3.2.3.3.5 Electrical Control Panel

The electrical control panel houses all electrical support equipment necessary for operation of the liquid and gas sample panel. All remotely operated valves in the liquid and gas sample panel are operated from the electrical control panel. A lighted mimic display showing the status of valves and equipment is provided as part of the electrical control panel. The accident isolation switch which locks out all normal sampling functions under accident conditions is also located on this panel.

9.3.2.3.3.6 Postaccident Sampling System Coolers

In order to cool down influent samples, four tube and shell type coolers are provided using component cooling water as their cooling medium. These coolers serve the three reactor coolant sample lines and the containment sump A sample line. A separate cooler for each liquid sample line ensures representative samples and minimizes the likelihood of sample cross-contamination. The coolers are mounted behind the liquid and gas sample panel to take advantage of the liquid and gas sample panel integral shielding to protect adjacent areas. Design parameters are as follows: tube temperature of 700°F, shell temperature of 350°F, tube pressure of 2485 psig, shell pressure of 150 psig, tube flow of 0.1 gpm, and shell flow of 10 gpm.

9.3.2.3.3.7 Postaccident Sampling System Waste Tank

The postaccident sampling system waste tank is provided to collect sample waste from calibration operations, purges, flushes, and analyses. It is sized to contain the fluid generated during any single sample operation, including a purge of at least three line volumes. It is located within the liquid and gas sample panel support pad and below the liquid and gas sample panel so that it receives the sample drainage by gravity. It is an 18-gallon tank having a design pressure range from full vacuum to 150 psig and a design temperature of 150°F. It is normally vented to the intermediate building heating, ventilation, and air conditioning exhaust via the liquid and gas sample panel exhaust plenum through a normally open valve. During postaccident conditions, an evacuation compressor will cycle on tank pressure to vent the tank to the containment atmosphere. The tank level is maintained by a level controlled transfer pump cycling on and off. The tank contents can be emptied by either the transfer pump (primary) or by nitrogen blowdown (backup) to either the waste holdup tank or sump A in containment. The postaccident sampling system waste tank will collect potentially low quality water from the containment sump sample and chemicals from instrument calibration. Waste from these operations should not normally be returned to the volume control tank.

For overpressure protection a rupture disk is used for leaktightness and to ensure adequate response to rapid pressure excursions as would occur with a hydrogen detonation. During operation the rupture disk discharge would be routed directly to the intermediate building heating, ventilation, and air conditioning system.

9.3.2.3.3.8 Postaccident Sampling System Waste Transfer Pump

A postaccident sampling system waste transfer pump is provided to empty the postaccident sampling system waste tank of its contents during typical operation. For postaccident operation the waste transfer pump discharges to the containment A sump. This is accomplished by pumping in the reverse-to-normal flow direction through the containment A sump pump discharge line. During MODES 1 and 2, the pump discharges to the waste holdup tank. It has an auto/manual switch where auto causes it to cycle to maintain the postaccident sampling system waste tank in a controlled level band. Its design parameters are 1 gpm flow, 150°F temperature, and 150 psig pressure, with a discharge pressure of 85 psig.

9.3.2.3.3.9 Postaccident Sampling System Waste Tank Evacuating Compressor

A postaccident sampling system waste tank evacuating compressor is provided to maintain the postaccident sampling system waste tank at atmospheric pressure during accident conditions to ensure adequate gravity drainage from the liquid and gas sample panel. The evacuating compressor will discharge to the containment during postaccident operation since the process stream could contain postaccident gases which evolve from the reactor coolant.

During MODES 1 and 2, the tank is vented via the liquid and gas sample panel plenum to the intermediate building heating, ventilation, and air conditioning system.

The compressor has an auto/manual switch where auto causes it to cycle off of a pressure transmitter in the waste tank. Its design parameters are 150 psig, 120°F, and 0.5 scfm minimum. It has a discharge pressure of 75 psia and a suction pressure of 14.0 to 14.7 psia.

9.3.2.3.3.10 Containment Sump A Sample Pump

The containment sample pump is an air-operated, 1-gpm pump. Its design parameters are 150 psig and 250°F, and it has a discharge head of 50 psig. It can discharge either to the waste holdup tank or to the liquid gas and sample panel via a heat exchanger.

9.3.3 EQUIPMENT AND FLOOR DRAINS SYSTEMS

The equipment and floor drain systems serve to route leakage from equipment and compartments in order to provide proper control of leakage, prevent uncontrolled communication between areas as necessary, and to allow monitoring of leakage prior to disposition. Pedestals and curbs are provided to prevent safety-related equipment from being flooded with standing water. The equipment and floor drains are included in the liquid waste disposal system and are included in Drawings 33013-1259 and 33013-1270 through 33013-1272.

The floor drain systems for the diesel generator rooms, battery rooms, and the control building air handling room, are equipped with backflow devices to prevent both internal and external flooding from affecting the rooms.

9.3.4 CHEMICAL AND VOLUME CONTROL SYSTEM

9.3.4.1 Design Bases

The following design criteria were used during the licensing of Ginna Station. They represent the Atomic Industrial Forum version of proposed criteria issued by the AEC for comment on July 10, 1967. Conformance with 1972 General Design Criteria of 10 CFR 50, Appendix A (i.e., General Design Criteria 1, 2, 5, 14, 29, 33, 35, 60, and 61), as they relate to the chemical and volume control system and components is discussed in Section 3.1.2.

9.3.4.1.1 Redundancy of Reactivity Control

CRITERION: Two independent reactivity control systems, preferably of different principles, shall be provided (AIF-GDC 27).

In addition to the reactivity control achieved by the control rods, reactivity control is provided by the chemical and volume control system which regulates the concentration of boric acid solution neutron absorber in the reactor coolant system. The system is designed to prevent, under anticipated system malfunction, uncontrolled or inadvertent reactivity changes which might stress the system beyond allowable limits.

9.3.4.1.2 Reactivity Holddown Capability

CRITERION: The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public (AIF-GDC 30).

Normal reactivity shutdown capability is provided by control rods with boric acid injection used to compensate for the long-term xenon decay transient and for plant cooldown. Any time that the plant is at power the quantity of boric acid retained in the sources of boric acid and ready for injection will always exceed that quantity required for normal MODE 5 (Cold Shutdown) and will also exceed the quantity of boric acid required to bring the reactor to MODE 3 (Hot Shutdown) and to compensate for subsequent xenon decay.

The boric acid solution is transferred from the boric acid storage tanks by boric acid transfer pumps to the suction of the charging pumps which inject boric acid into the reactor coolant. Any charging pump and boric acid transfer pump can be operated from diesel-generator power on loss of primary electric power. Boric acid injection from the Boric Acid Storage Tanks (BAST) to the RCS by one charging pump operating at its nominal charging flow rate of 46 gpm is capable of shutting down the reactor with no rods inserted in approximately 81 minutes.

Sufficient boric acid from the BAST or the RWST can also be injected to compensate for xenon decay beyond the equilibrium level, with one charging pump operating at its minimum speed, and thereby delivering in excess of the required minimum flow of approximately 9 gpm into the reactor coolant system. If three charging pumps are available, these time periods are reduced. Additional boric acid is employed if it is desired to bring the reactor to a cold shutdown condition.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability, independent of control rod clusters which normally serve this function in the short-term situation. Shutdown for long-term and reduced temperature conditions can be accomplished with boric acid injection using redundant components.

9.3.4.1.3 Reactivity Hot Shutdown Capability

CRITERION: The reactivity control system provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition (AIF-GDC 28).

The reactivity control systems provided are capable of making and holding the core subcritical for any hot operating (MODES 1 and 2) condition, including those resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning of each cycle. The control rods are divided into two categories comprising a control group and shutdown groups.

The control group, used in combination with chemical shim (soluble boron) provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of control rods is used to compensate for short-term reactivity changes at power that might be produced due to variations in reactor power requirements or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup and decay.

9.3.4.1.4 Reactivity Shutdown Capability

CRITERION: One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn (AIF-GDC 29).

The shutdown groups are provided to supplement the control group of control rods to make the reactor subcritical with the required shutdown margin following trip from any credible operating condition to the hot zero-power condition assuming the most reactive rod cluster control assembly remains in the fully withdrawn position. Manually controlled boric acid addition is used to supplement the control rods in maintaining the shutdown margin for the long-term conditions of xenon decay or plant cooldown.

9.3.4.1.5 Codes and Classifications

All pressure-retaining components (or compartments of components) which are exposed to reactor operating pressure and temperatures at rated power and pressure-retaining components (or compartments of components) through which reactor coolant circulates at reduced pressures and temperatures generally comply with the following codes:

A. System pressure vessels - ASME Boiler and Pressure Vessel Code, Section III, Class C, including paragraph N-2113.

B. System valves and fittings, USAS B16.5; piping, USAS B31.1, including nuclear code cases.

The regenerative heat exchanger and excess letdown heat exchanger (tube side) are specified as Class A vessels.

The components of the chemical and volume control system comply with the codes and standards as discussed in Section 3.2.

9.3.4.2 System Design and Operation

9.3.4.2.1 General

The chemical and volume control system is designed to perform the following functions:

A. To control the reactor coolant inventory, chemistry conditions, activity level, and boron concentration.

B. To provide seal-water injection flow to the reactor coolant pumps.

C. To process reactor coolant effluent for reuse of boric acid and makeup water.

Reactor coolant water chemistry specifications are listed in plant procedures. Contaminant limits are included in the Technical Specifications and the Technical Requirements Manual (TRM).

In order to perform the above functions of the chemical and volume control system, a continuous feed-and-bleed is maintained between the reactor coolant system and the chemical and volume control system.

Water is letdown from the reactor coolant system, through a regenerative heat exchanger to minimize thermal loss to the reactor coolant system. The pressure is reduced through orifices and further cooling occurs in a nonregenerative heat exchanger followed by a second pressure reduction. Water is returned to the reactor coolant system by the charging system, which also provides seal injection flow to the reactor coolant pumps.

The chemistry of the letdown flow may be altered by passing the flow through demineralizers that remove ionic impurities. A filter removes solids, and the gases dissolved in the coolant are removed in the volume control tank. The boric acid concentration in the coolant is changed by the reactor makeup portion of the chemical and volume control system as required for reactivity control. Excess coolant may be diverted into the boron recycle portion of the chemical and volume control system for reprocessing into pure water and concentrated boric acid. System components that have a design pressure and temperature less than the reactor coolant system design limits are provided with overpressure protective devices.

System discharges from overpressure protective devices (safety valves) and system leakages are directed to closed systems. Effluents removed from such closed systems are monitored and discharged under controlled conditions. The system design enables postoperational hydrostatic testing to applicable code test pressures. The relief valves will be gagged during hydrostatic testing. The relief valves in systems that are hydrostatically tested after MODE 6 (Refueling) operations will be set at the system design pressure.

9.3.4.2.2 Letdown and Charging Systems

9.3.4.2.2.1 General

During plant operation, reactor coolant flows through the letdown line from the reactor coolant loop B crossover leg and is returned to the loop B cold leg or (alternatively) the loop B hot leg via a charging line. An alternate charging line is provided to the cold leg of loop A. The charging and letdown systems flow diagrams are provided in Drawings 33013-1264 and 33013-1265.

Each of the connections to the reactor coolant system has an isolation valve located close to the loop piping. In addition, a check valve is located downstream of the charging line isolation valve. Reactor coolant entering the chemical and volume control system flows through the shell side of the regenerative heat exchanger where its temperature is reduced. The coolant then flows through a letdown orifice which reduces the coolant pressure. The cooled, low-pressure water leaves the reactor containment and enters the auxiliary building where it undergoes a second temperature reduction in the tube side of the nonregenerative heat exchanger followed by a second pressure reduction by the low-pressure letdown valve. After passing through one of the mixed-bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filter and enters the volume control tank through a spray nozzle.

Hydrogen is automatically supplied, as determined by pressure control, to the vapor space in the volume control tank, which is predominantly hydrogen and water vapor. The hydrogen within this tank is, in turn, the supply source to the reactor coolant. Fission gases are periodically removed from the system by venting the volume control tank prior to a cold or MODE 6 (Refueling) shutdown.

Next, the coolant flows to the charging pumps, which raise the pressure above that in the reactor coolant system. The coolant then enters the containment, passes through the tube side of the regenerative heat exchanger, and is returned to the reactor coolant system.

The cation bed demineralizer, located downstream of the mixed-bed demineralizers, is used intermittently to control cesium activity in the coolant and also to remove excess lithium which is formed from Boron-10 (n, α) Lithium-7 reaction.

Letdown flow from the residual heat removal system allows purification of the reactor coolant during shutdown conditions when the temperature of the reactor coolant is maintained by the residual heat removal system.

Excess letdown is used to maintain the flow balance between the letdown and charging systems if the normal letdown path is inoperable, or for additional letdown when necessary. Excess letdown is taken from the reactor coolant loop A crossover leg and flows to the excess letdown heat exchanger. The individual components of the letdown and charging systems are described in Section 9.3.4.3.

9.3.4.2.2.2 Charging Pump Control

The speed of each charging pump can be controlled manually or automatically. During MODES 1 and 2, two of the three pumps are running with one in automatic and one in manual control. The automatically operated charging pump speed is modulated in accordance with pressurizer level. During load changes, the pressurizer level setpoint is varied automatically to compensate partially for the expansion or contraction of the reactor coolant associated with the T_{AVG} changes. Charging pump speed does not change rapidly with pressurizer level variations due to the reset action of the pressurizer level controller.

If the pressurizer level increases, the speed of the pump decreases; likewise, if the level decreases, the speed increases. If the charging pump on automatic control reaches the high-speed limit, an alarm is actuated. The speed of the second pump is manually regulated. If the speed of the charging pump on automatic control does not decrease and the second charging pump is operating at maximum speed, the third charging pump can be started and its speed manually regulated. If the speed of the charging pump on automatic control decreases to its minimum value, an alarm is actuated and the speed of the pumps on manual control is reduced.

To ensure that the charging pump flow is always sufficient to meet both the seal-water and minimum charging flow requirements, the pump has a variable control stop that does not permit pump flow lower than the specified minimum. This control stop is adjustable to permit higher flow limits to be set if mechanical reactor coolant pump seal leakage increases during plant life.

Charging flow is indicated on duplicate indicators in the control room on the left and middle sections of the main control board. Seal injection flow is indicated on a dual indicator in the control room on the middle section of the main control board beside the charging flow indicator and in proximity of the charging flow controller. The charging flow indicators are scaled from 0 to 75 gpm and the seal injection flow indicators are scaled from 0 to 15 gpm. Total charging pump flow is determined by adding the two indicators. Also, the two flows are input to the plant process computer system where they are combined to produce total flow.

9.3.4.2.3 Seal-Water Injection System

A portion of the high-pressure charging flow is injected into the reactor coolant pumps between the pump impeller and the shaft seal so that the seals are not exposed to high temperature reactor coolant. Part of the flow is the shaft seal leakage flow and the remainder enters the reactor coolant system through a labyrinth seal on the pump shaft. The shaft seal leakage flow passes through the seals, is filtered, cooled in the seal-water heat exchanger, and returned to the volume control tank. The remaining flow is diverted through the labyrinth seal, cools the lower radial bearing, and enters the reactor coolant system. Seal water inleakage to the reactor coolant system requires a continuous letdown of reactor coolant to maintain the desired inventory.

The seal-water injection system is shown in Drawing 33013-1265, Sheets 1 and 2.

9.3.4.2.4 Reactor Makeup Control System

9.3.4.2.4.1 System Description

The reactor makeup control system is shown in Drawings 33013-1266 and 33013-1269. The reactor makeup control, operated from the control room, manually preselects makeup composition to the charging pump suction header or the volume control tank in order to adjust the reactor coolant boron concentration for reactivity control. Makeup is provided to maintain the desired operating fluid inventory in the volume control tank. The operator can stop the makeup operation at any time in any operating mode by remotely closing the makeup stop valves.

One reactor makeup water pump and one boric acid transfer pump are normally selected for auto standby. The other two pumps are placed in the pull stop position. Tripping of either pump during its operation would cause either a reactor makeup water flow deviation alarm or boric acid flow deviation alarm on the main control board.

Makeup water to the reactor coolant system is provided by the chemical and volume control system from the following sources:

a. The reactor makeup water tank, which provides water for dilution when the reactor coolant boron concentration is to be reduced.
b. The boric acid storage tanks, which supply concentrated boric acid solution when reactor coolant boron concentration is to be increased.
c. The refueling water storage tank (RWST), which supplies borated water for emergency makeup.
d. The chemical mixing tank, which is used to inject small quantities of solution when additions of hydrazine or pH control chemicals are necessary.
e. The monitor tanks, which provide water for dilution when the reactor makeup water tank is out of service.

Makeup for normal plant leakage is regulated by the reactor makeup control system, which is set by the operator to blend water from the reactor makeup water tank with concentrated boric acid to match the reactor coolant boron concentration. Makeup is added automatically if the volume control tank level falls below a preset point.

Boric acid is dissolved in hot water in the batching tank to the desired concentration. A transfer pump is used to transfer the batch to the boric acid storage tanks, which are maintained at a concentration, minimum volume, and minimum solution temperature in accordance with the Technical Requirements Manual (TRM). Small quantities of boric acid solution are metered from the discharge of an operating transfer pump for blending with makeup water as makeup for normal leakage or for increasing the reactor coolant boron concentration during MODES 1 and 2. Electric immersion heaters maintain the temperature of the boric acid storage tank solution high enough to prevent precipitation. The lower portion of the batching tank is jacketed to permit heating of the batching tank solution with low-pressure steam.

The boric acid flow control valve in the line from the boric acid storage tanks to the boric acid blender provides accurate fluid flow regulation throughout the range from 1 to 10 gpm.

The original Westinghouse reactor water makeup and blender system design did not account for the volumetric expansion of water in the isolated heat traced piping of the blender subsystem. During dilution evolutions, reactor makeup water is introduced into the blender.

The reactor water makeup system water is not heated and therefore, during normal operation, this water can be as cold as 60° F. After the dilution evolution, the heat traced blender piping is isolated by a combination of AOVs and check valves. This isolated blender piping is then heated by three different heat trace systems. Since there is no room for the expansion of the heated water, a bellows style accumulator has been added to the reactor makeup water system to allow for the expansion of the heated water from the isolated heat traced piping connected to the blender piping assembly. An enlarged section of pipe has also been added to the system to prevent any nitrogen intrusion into the charging system in the event of accumulator failure.

The various modes of operation of the reactor makeup control system are described below.

9.3.4.2.4.2 Automatic Makeup

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

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

9.3.4.2.4.3 Dilution

The dilute mode of operation permits the addition of a preselected quantity of reactor makeup water at a preselected flow rate to the reactor coolant system. The operator places the makeup system control switch in the stopped position and verifies the makeup stop valves to the volume control tank (AOV-110C) and to the charging pump suction (AOV-110B) are closed.
The reactor makeup water controller setpoint is adjusted to the proper flowrate, and the reactor makeup water batch integrator is set to the proper quantity. Then the makeup system mode selector switch is placed to the dilute position. When the makeup system control switch is placed in the armed position, the selected reactor makeup pump starts, the reactor makeup water control valve (AOV-111) opens to the preselected position and the makeup stop valve to the volume control tank inlet (AOV-110C) is opened. If the dilution flow deviates +/-5 gpm from the preselected flow rate, an alarm will annunciate on the main control board. The reactor makeup is added in the volume control tank and then goes to the charging pump suction header. Excessive rise of the volume control tank water level is prevented by automatic actuation (by the tank level controller) of a three-way diversion valve, which routes the reactor coolant letdown flow to the holdup tanks. When the preset quantity of reactor makeup water has been added, the batch integrator causes the reactor makeup water control valve and the makeup water stop valve to the volume control tank inlet header to close, and the makeup water pump stops. The operator then realigns the system as desired.

9.3.4.2.4.4 Boration

The borate mode of operation permits the addition of a preselected quantity of concentrated boric acid solution at a preselected flow rate to the reactor coolant system. The operator sets the makeup stop valves to the volume control tank and to the charging pump suction in the closed position, the mode selector switch to borate, the concentrated boric acid flow controller setpoint to the desired flow rate, and the concentrated boric acid batch integrator to the desired quantity. Opening the makeup stop valve to the charging pump suction permits the concentrated boric acid to be added to the charging pump suction header. The total quantity added in most cases is so small that it has only a minor effect on the volume control tank level. When the preset quantity of concentrated boric acid solution has been added, the batch integrator causes the concentrated boric acid control valve to close.

The normal capability to add boron to the reactor coolant is sufficient so that no limitation is imposed on the rate of cooldown of the reactor upon shutdown. The maximum rates of boration and the equivalent coolant cooldown rates are given in Table 9.3-6. One set of values is given for the addition of boric acid from a boric acid storage tank with one boric acid transfer pump and one charging pump operating. The other set assumes the use of refueling water but with two of the three charging pumps operating. The rates are based on full operating temperature and on the end of the core life when the moderator temperature coefficient is most negative.

9.3.4.2.5 Boron Recycle System

9.3.4.2.5.1 System Description

The boron recycle system is designed to reduce the amount of liquid waste produced by plant operations by recycling the discharge from the reactor coolant system. The boron recycle system is shown in Drawings 33013-1266 and 33013-1268.

During plant startup, MODES 1 and 2, load reductions, and shutdowns, liquid effluents containing boric acid flow from the reactor coolant system through the letdown line and are collected in the holdup tanks. As liquid enters the holdup tanks, the cover gas is displaced to the gas decay tanks in the waste disposal system through the waste vent header. The concentration of boric acid in the holdup tanks varies throughout core life from the refueling concentration to the boron concentration at which the deborating demineralizers are used. A recirculation pump is provided to transfer liquid from one holdup tank to another.

Liquid effluent in the holdup tanks is processed as a batch operation. This liquid is pumped through the base removal ion exchanger and cation ion exchanger which primarily remove lithium hydroxide and long-lived cesium. It then flows through the ion exchanger filter and into the gas stripper where dissolved gases are removed from the liquid. Effluent from the gas stripper enters the boric acid evaporator where dilute boric acid solution is concentrated to a selected weight percent boric acid solution.

The condensate leaves the condenser and is pumped through a distillate cooler before entering one of the two evaporator condensate demineralizers where evaporator boron carryover is removed. Condensate then flows through the condensate filter and accumulates in a monitor tank.

Subsequent handling of the condensate is dependent on the results of sample analysis. Discharge from the monitor tanks may be pumped to the reactor makeup water storage tank, aligned to the suction of the reactor makeup water pumps (when the reactor makeup water tank is out of service), recycled through the evaporator condensate demineralizers, returned to the waste holdup tanks for reprocessing or discharged to the environment with the condenser circulating water within the allowable activity concentration. If the sample analysis of the monitor tank contents indicates that it may be discharged safely to the environment, two valves must be opened to provide a discharge path. There is only one discharge path from the plant. As the effluent leaves, it is continuously monitored. If an unexpected increase in radioactivity is sensed, one of the discharge valves closes automatically and an alarm sounds in the control room.

Boric acid evaporator bottoms are discharged through a concentrates filter to the concentrates holding tank. Solution collected in the concentrates holding tank is sampled and then transferred to the boric acid storage tanks if analysis indicates that it meets specifications. Otherwise the solution is pumped to the holdup tanks for reprocessing by the evaporator train.

The concentrated solution can also be pumped from the evaporator to containers. These containers can then be stored at the plant site for ultimate shipment offsite for disposal.

The deborating demineralizers are used intermittently to remove boron from the reactor coolant near the end of the core life. When the deborating demineralizers are in operation, the letdown stream passes from the mixed-bed demineralizers, through the deborating demineralizers, and into the volume control tank after passing through the reactor coolant filter.

9.3.4.2.5.2 Alarm Functions

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

a. Deviation of reactor makeup water flow rate from the control setpoint.

b. Deviation of concentrated boric acid flow rate from the control setpoint.

c. Low level (makeup initiation point) in the volume control tank when the reactor makeup control selector is not set for the automatic makeup control mode.

9.3.4.2.6 Heat Tracing System

Electrical heat tracing is installed under the insulation on all piping, valves, line-mounted instrumentation, and components normally containing concentrated boric acid solution, including those lines beyond shut-off valves which would contain stagnant concentrated solution following closure. The heat tracing is designed to prevent boric acid precipitation due to cooling, by compensating for heat loss. There are two trains of heaters. Each is capable of heating components, piping, and valves above the minimum required solution temperature.
One train is normally used with the second train used as a backup in case of failure. Locations in which heat tracing is not used are as follows:

A. Lines that may transport concentrated boric acid but are subsequently flushed with reactor coolant or other liquid of low boric acid concentration during MODES 1 and 2.

B. The boric acid storage tanks, which are provided with immersion heaters.

C. The batching tank, which is provided with a steam jacket.

D. The concentrates holding tank, which is provided with an immersion heater.

Duplicate tracing on sections of the chemical and volume control system normally containing boric acid solution provides standby capacity if the operating tracing malfunctions.

The heat tracing system is capable of maintaining the piping contents above the minimum solution temperature corresponding to the boron concentration ranges specified in the Technical Requirements Manual (TRM). An internal upper limit of approximately 200°F is currently established, but authorization for temperatures up to 250°F has been given for specific analyzed piping. The heat tracing system will be supplied with power from the emergency diesel generators following a loss of offsite power. The heat trace system is required to be operational during normal power, startup, and transient conditions. Temperature detectors, alarm and control functions, and electrical power requirements for the heat tracing are not shown on the process flow diagram.

9.3.4.3 Component Description

Tables 9.3-6 and 9.3-7 list the system performance requirements and data for individual system components.

9.3.4.3.1 Letdown and Charging Systems

9.3.4.3.1.1 Regenerative Heat Exchanger

The regenerative heat exchanger is designed to recover the heat from the letdown stream by reheating the charging stream during MODES 1 and 2. This exchanger also limits the temperature rise which occurs at the letdown orifices during periods when letdown flow exceeds charging flow by a greater margin than at normal letdown conditions.

The letdown stream flows through the shell of the regenerative heat exchanger and the charging stream flows through the tubes. The unit is made of austenitic stainless steel, and is of all-welded construction. The exchanger is designed to withstand 2000 step changes in shell side fluid temperature from 100°F to 560°F during the design life of the unit.

9.3.4.3.1.2 Letdown Orifices

One of the three letdown orifices controls the flow of the letdown stream during MODES 1 and 2 and reduces the pressure to a value compatible with the nonregenerative heat exchanger design. Two of the letdown orifices are designed to pass normal letdown flow. The other orifice is used to attain maximum purification flow at normal reactor coolant system operating pressure. The orifices are placed in service by remote manual operation of their respective isolation valves. One or both of the standby orifices may be used in parallel with the normally operating orifice in order to increase letdown flow when the reactor coolant system pressure is below normal. This arrangement provides a full standby capacity for control of letdown flow. Each orifice consists of bored pipe made of austenitic stainless steel.

9.3.4.3.1.3 Nonregenerative Heat Exchanger

The nonregenerative heat exchanger cools the letdown stream to the operating temperature of the mixed-bed demineralizers. Reactor coolant flows through the tube side of the exchanger while component cooling water flows through the shell. The letdown stream outlet temperature is automatically controlled by a temperature control valve in the component cooling water outlet stream. The unit is a multiple-tube-pass heat exchanger. All surfaces in contact with the reactor coolant are austenitic stainless steel, and the shell is carbon steel.

9.3.4.3.1.4 Mixed-Bed Demineralizers

Two flushable mixed-bed demineralizers maintain reactor coolant purity. Each demineralizer is charged with anion and cation resin. The resin removes fission and corrosion products, along with other ionic impurities. The resin bed is designed to reduce the concentration of these impurities in the purification stream by a minimum factor of 10.

Each demineralizer is sized to accommodate the maximum letdown flow. One demineralizer serves as a standby unit for use if the operating demineralizer becomes exhausted during operation.

The demineralizer vessels are made of austenitic stainless steel, and are provided with suitable connections to facilitate resin replacement when required. The vessels are equipped with a resin retention screen. Each demineralizer has sufficient capacity to enable MODE 6 (Refueling) after operation for one core cycle with 1% of the fuel rods containing pinholes or fine cracks.

9.3.4.3.1.5 Cation Bed Demineralizer

The original cation demineralizer has a broken retention screen and is no longer used. The "A" letdown deborating demineralizer is being used as the cation demineralizer. It is a flushable resin bed in the hydrogen form and is located downstream of the mixed bed demineralizers and is used to control the concentrations of Lithium-7 which build up in the coolant from the Boron-10 (n, α) Lithium-7 reaction. The demineralizer also has the capacity to maintain the Cesium-137 concentration in the coolant below 1.0 $\mu Ci/cm^3$ with 1% defective fuel. The demineralizer would be used intermittently to control cesium.

The demineralizer is made of austenitic stainless steel and is provided with suitable connections to facilitate resin replacement when required. The vessel is equipped with a resin retention screen.

9.3.4.3.1.6 Deborating Demineralizers

When required, two anion demineralizers remove boric acid from the reactor coolant system fluid. The demineralizers normally are used only near the end of the core cycle. Hydroxyl based ion-exchange resin is used to reduce reactor coolant system boron concentration by releasing a hydroxyl ion when a borate ion is absorbed. The spent resin is flushed to the spent resin storage tank.

9.3.4.3.1.7 Resin Fill Tank

The resin fill tank is used to charge fresh resin to the demineralizers. The line from the conical bottom of the tank is fitted with a dump valve and may be connected to any one of the demineralizer fill lines. The demineralizer water and resin slurry can be sluiced into the demineralizer by opening the dump valve. The tank, designed to hold approximately two-thirds of the resin volume of one mixed-bed demineralizer, is made of austenitic stainless steel.

9.3.4.3.1.8 Reactor Coolant Filter

Two filters collect resin fines and particulates from the letdown stream. A coarse filter (approximately 20 micron) can be operated by itself or in parallel with a fine filter (submicron up to approximately 20 micron). The coarse filter serves as a relief path when the fine filter is in operation to protect against pressure and flow transients that may occur if the fine filter fouls. Each vessel is made of austenitic stainless steel, and is provided with connections for draining and venting. Design flow capacity of each filter is equal to the maximum purification flow rate. However, the fine filter is aligned in parallel with the coarse filter for fouling protection. The filters are equipped with an external bypass line which allows for continuous letdown flow while the filter cartridge is being replaced.

9.3.4.3.1.9 Volume Control Tank

The volume control tank (VCT) collects the excess water released during ascension from zero to full power that is not accommodated by the pressurizer. It also receives the excess coolant release caused by the deadband in the reactor control temperature instrumentation. Overpressure of hydrogen gas is maintained in the volume control tank to control the hydrogen concentration in the reactor coolant at the recommended levels of EPRI TR-105714, "PWR Primary Water Chemistry Guidelines."

A spray nozzle is located inside the tank on the inlet line from the filter. This spray nozzle provides intimate contact to equilibrate the gas and liquid phases. A remotely operated vent valve permits removal of gaseous fission products which collect in this tank. The volume control tank also acts as a head tank for the charging pumps and a reservoir for the leakage from the reactor coolant pump controlled leakage seal. Two level transmitters have been installed on the VCT to control the operation of the:

- Holdup diversion
- VCT isolation on low-low level
- automatic makeup valves

VCT isolation also occurs on a high-high temperature in the VCT as sensed by two temperature elements. Charging pump suction swaps over to the refueling water storage tank (RWST) on either a low-low level or high-high temperature in the VCT.

The tank is constructed of austenitic stainless steel.

9.3.4.3.1.10 Charging Pumps

Three charging pumps inject coolant into the reactor coolant system. The pumps are the variable speed positive displacement type, and all parts in contact with the reactor coolant are fabricated of austenitic stainless steel or other material of adequate corrosion resistance. Special low chloride content packing is used in the pump glands. These pumps have stuffing boxes with leakoffs to collect reactor coolant. A closed system for reclaiming the packing leakoff has been installed to reduce the release of fission gases from the chemical and volume control system and the amount of high activity water going into the liquid waste system. The pump design prevents lubricating oil from contaminating the charging flow, and the integral discharge valves act as check valves.

Each pump is designed to provide the full charging line flow and the reactor coolant pump seal-water supply during normal seal leakage. Each pump is designed to provide rated flow against a pressure equal to the sum of the reactor coolant system normal maximum pressure (existing when the Pressurizer Power Operated Relief Valve (PORV) is operating) and the piping, valve, and equipment pressure losses at the design charging pump flow with two pumps in operation.

9.3.4.3.1.11 Charging Pump Leakoff Tank

The pump packing gland leakoff goes to a charging pump leakoff tank. The leakoff tank has two pumps with local controls which pump at 3 gpm to the holdup tanks.

9.3.4.3.1.12 Charging Pump Dampener

The charging pump pulse dampener is a device meant to eliminate pulsations in discharge pressure from the pump. The dampener is 9 ft long and 2 ft in diameter with a divider plate and a pipe through the plate. Charging flows into the tank through the center pipe and out one of the two outlets.

9.3.4.3.1.13 Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools reactor coolant letdown by an amount equal to the nominal injection rate through the reactor coolant pump labyrinth seal, if letdown through the regenerative heat exchanger is blocked. The unit is designed to reduce the letdown stream temperature from the cold-leg temperature to 195°F. The letdown stream flows through the tube side and component cooling water is circulated through the shell side. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel. All tube joints are welded. The unit is designed to withstand 12,000 step changes in the tube fluid temperature from 80°F to the cold-leg temperature.

<u>9.3.4.3.2 Seal-Water Injection System</u>

9.3.4.3.2.1 Seal-Water Heat Exchanger

The seal-water heat exchanger removes heat from two sources: reactor coolant pump seal-water returning to the volume control tank and reactor coolant discharge from the excess letdown heat exchanger. Reactor coolant flows through the tubes and component cooling water is circulated through the shell side. The tubes are welded to the tubesheet because leakage could occur in either direction, resulting in undesirable contamination of the reactor coolant or component cooling water. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

The unit is designed to cool the excess letdown flow and the seal water flow to the temperature normally maintained in the volume control tank if all the reactor coolant pumps controlled leakage seals are leaking at the maximum design leakage rate.

9.3.4.3.2.2 Seal-Water Filter

The filter collects particulates larger than 25 microns from the reactor coolant pump seal-water return and from the excess letdown heat exchanger flow. The filter is designed to pass the sum of the excess letdown flow and the maximum design leakage from the reactor coolant pump seals. The vessel is constructed of austenitic stainless steel and is provided with connections for draining and venting. Disposable synthetic filter elements are used.

9.3.4.3.2.3 Seal-Water Injection Filters

Two filters are provided in parallel, each sized for the injection flow. They collect particulates larger than 5 microns from the water supplied to the reactor coolant pump seal.

9.3.4.3.3 Reactor Makeup Control System

9.3.4.3.3.1 Boric Acid Filter

The boric acid filter collects particulates larger than 25 microns from the boric acid solution being pumped to the charging pump suction line. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously. The vessel is constructed of austenitic stainless steel and the filter elements are disposable synthetic cartridges. Provisions are available for venting and draining the filter.

9.3.4.3.3.2 Boric Acid Storage Tanks

The boric acid storage tank capacity is sized to store sufficient boric acid solution for MODE 6 (Refueling) plus enough boric acid solution for a MODE 5 (Cold Shutdown) shortly after full power operation is achieved. In addition, sufficient boric acid solution is available for MODE 5 (Cold Shutdown) if the most reactive control rod is not inserted. The boric acid storage tanks are also discussed in Sections 6.3.2.2.5 and 6.3.6.4.

The concentration of boric acid solution in storage is maintained at a concentration, minimum volume, and minimum solution temperature in accordance with the Technical Requirements Manual (TRM). Periodic manual sampling and corrective action is provided, if necessary, to ensure that these limits are maintained. As a consequence, measured amounts of boric acid solution can be delivered to the reactor coolant to control the chemical poison concentration. The combination overflow and breather vent connection has a water loop seal to minimize vapor discharge during storage of the solution. The tank is constructed of austenitic stainless steel.

Redundant tank heaters and line heat tracing are provided to ensure that the solution will be stored at a temperature that is above the solubility limit for the range of concentrations allowed by the Technical Requirements Manual (TRM). The heating elements are located near the bottom of the tank.

Low concentration boric acid from either boric acid storage tank can be transferred to the boric acid batching tank where the concentration can be increased through piping connections between the discharge lines of the two boric acid transfer pumps and the inlet to the boric acid batching tank.

9.3.4.3.3.3 Batching Tank

The batching tank is sized to hold a 1-week makeup supply of boric acid solution for the boric acid storage tank. The basis for makeup is reactor coolant leakage of 0.50 gpm at beginning of core life. The tank may also be used for solution storage. A local sampling point is provided for verifying the solution concentration prior to transferring it to the boric acid storage tank or for draining the tank. The tank manway is provided with a removable screen to prevent entry of foreign particles. In addition, the tank is provided with an agitator to improve mixing during batching operations. The tank is constructed of austenitic stainless steel and is not used to handle radioactive substances. The tank is provided with a steam jacket for heating the boric acid solution to 165°F.

9.3.4.3.3.4 Boric Acid Tank Heaters

Two 100% capacity electric immersion heaters in each boric acid storage tank are capable of maintaining the temperature of the boric acid solution at 165°F with an ambient air temperature of 40°F. The concentration, minimum volume, and minimum solution temperature are maintained in accordance with the Technical Requirements Manual (TRM). The heaters are sheathed in austenitic stainless steel.

9.3.4.3.3.5 Boric Acid Transfer Pumps

Two 100% capacity canned centrifugal pumps are used to circulate or transfer chemical solutions. The pumps circulate boric acid solution through the boric acid storage tanks and inject boric acid into the charging pump suction header. The design capacity of each pump is equal to the normal letdown flow rate (40 gpm). The design head is sufficient, with one pump out of operation to permit maintenance and considering line and valve losses, to deliver rated flow to the charging pump suction header when volume control tank pressure is at the maximum operating value (relief valve setting). All parts in contact with the solutions are austenitic stainless steel or other adequately corrosion resistant material.

The transfer pumps are operated either automatically or manually from the main control room or from a local control center. The reactor makeup control operates one of the pumps automatically when boric acid solution is required for makeup or boration. Instrumentation is provided to allow boric acid transfer pump discharge pressure to be monitored in the control room.

9.3.4.3.3.6 Boric Acid Blender

The boric acid blender promotes thorough mixing of boric acid solution and reactor makeup water from the reactor coolant makeup circuit. The blender consists of a conventional pipe fitted with a perforated tube insert. All material is austenitic stainless steel. The blender decreases the pipe length required to homogenize the mixture for taking a representative local sample.

9.3.4.3.3.7 Chemical Mixing Tank

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

The capacity of the chemical mixing tank is determined by the quantity of 35% hydrazine solution necessary to increase the concentration in the reactor coolant by 10 ppm. This capacity is more than sufficient to prepare solution of pH control chemical for the reactor coolant system.

The chemical mixing tank is made of austenitic stainless steel.

9.3.4.3.3.8 Heat Tracing

The heat tracing system is described in Section 9.3.4.2.6.

9.3.4.3.3.9 Reactor Makeup Water Pumps

Two reactor makeup water pumps take suction from either the monitor tanks or the reactor makeup water tank. These pumps are used to feed dilution water to the boric acid blender and are also used to supply makeup water for intermittent flushing of equipment and piping.

Each pump is sized to match the maximum letdown flow (60 gpm). One pump serves as a standby for the other. These centrifugal pumps are constructed of austenitic stainless steel.

9.3.4.3.3.10 Reactor Makeup Water Tank

The reactor makeup water tank is used to store makeup water which is supplied from the monitor tanks and the GE Betz water treatment plant. Makeup water from the tank discharges to the suction of the reactor makeup water pumps. The tank is Plasite-lined carbon steel.

9.3.4.3.4 Boron Recycle System

9.3.4.3.4.1 Holdup Tanks

Three holdup tanks (Drawing 33013-1267) contain radioactive liquid which enters the tank from the letdown line. The liquid is released from the reactor coolant system during startups, shutdowns, and load changes, and from boron dilution to compensate for burnup. The contents of one tank are normally being processed by the gas stripper and evaporator train while another tank is being filled. The third tank is normally kept empty to provide additional storage capacity when needed.

The total liquid storage sizing basis for the holdup tanks is given in Table 9.3-7. Nitrogen cover gas is supplied to the tanks at a rate sufficient to prevent vacuum formation at the design pumping rate of 60 gpm. The three tanks hold two reactor coolant system volumes. The tanks are constructed of austenitic stainless steel.

Seismic qualification for the holdup tanks is listed in Table 3.2-1.

9.3.4.3.4.2 Holdup Tank Recirculation Pump

The recirculation pump is used to mix the contents of a holdup tank or transfer the contents of a holdup tank to another holdup tank. The wetted surface of this pump is constructed of austenitic stainless steel.

9.3.4.3.4.3 Gas Stripper Feed Pumps

The two gas stripper feed pumps supply feed to the gas stripper boric acid evaporator train from a holdup tank. The capacity of each pump is equal to the gas stripper-evaporator capacity. One pump is a standby and is available for operation in the event the operating pump malfunctions. These canned centrifugal pumps are constructed of austenitic stainless steel.

9.3.4.3.4.4 Base Removal Ion Exchanger

Two base removal flushable demineralizers remove anions from the holdup tank effluent. The resin is initially in the hydrogen form. The design flow rate is equal to the gas stripper boric acid evaporator processing rate. The demineralizer vessel is constructed of austenitic stainless steel and contains a resin retention screen.

9.3.4.3.4.5 Cation Ion Exchanger

Two cation flushable demineralizers remove cations (primarily cesium) from the holdup tank effluent. The resin is initially in the hydrogen form. The design flow rate is equal to the gas stripper boric acid evaporator processing rate. The demineralizer vessels are constructed of austenitic stainless steel and contain a resin retention screen.

9.3.4.3.4.6 Ion Exchanger Filter

The filter collects resin fines and particulates larger than 25 microns from the cation ion exchanger. The vessel is made of austenitic stainless steel, and is provided with connections for draining and venting. Disposable synthetic filter elements are used. The design flow capacity is equal to the boric acid evaporator flow rate.

9.3.4.3.4.7 Gas Stripper Equipment

The gas stripper removes nitrogen, hydrogen, and fission gases from the holdup tank feed using steam. The gas stripper equipment consists of a preheater, stripping column with a reflux condenser and associated pumps, piping, and instrumentation.

The gas stripper preheater located upstream of the gas stripper heats the liquid effluent from the holdup tanks from ambient temperature, at the evaporator train processing rate, to approximately 205°F using the gas stripper bottoms which are cooled in the preheater from approximately 220°F to 120°F. The preheater is a regenerative type shell and tube unit constructed of austenitic stainless steel.

The gas stripper consists of a hotwell to store stripped water, a stripping section packed with pall rings, a spray type liquid inlet header, and an overhead integral reflux condenser. Liquid flowing to the gas stripper is controlled to constant rate by a flow controller. The gas stripper is designed for the same flow rate as the evaporator and is designed to reduce the influent gas concentration by a factor of 10^5.

Two gas stripper bottom pumps, operated from level control, transfer effluent from the gas stripper hotwell to the boric acid evaporator via the gas stripper preheater. Each centrifugal pump is rated at the evaporator processing rate. The pumps are austenitic stainless steel and one is an installed standby for the operating pump.

9.3.4.3.4.8 Boric Acid Evaporator Equipment

The boric acid evaporator concentrates boric acid for reuse in the reactor coolant system. Borated water enters the evaporator and the liquid is concentrated to the selected weight percent boric acid. Vapors leave the evaporator and are condensed. The solids decontamination factor between the condensate and the bottoms is approximately 10^6. All evaporator equipment is constructed of austenitic stainless steel and is supplied as a unit. The boric acid evaporator equipment consists of the boric acid evaporator concentrates pumps, boric acid evaporator, boric acid evaporator condenser, boric acid evaporator condensate pumps, boric acid evaporator condensate cooler, air ejector system, and associated piping and instrumentation.

The boric acid evaporator feed tank has sufficient capacity to hold a 1-day production of boric acid solution produced from MODE 6 (Refueling) concentration feed. The evaporator and condenser heat transfer area is sufficient to maintain the required feed rate. The evaporator is steam heated. Component cooling water flows through the tube of the condenser.

The boric acid distillate cooler reduces the temperature of the condensate to approximately 100°F. The condensate flows through the shell and component cooling water flows through the tubes.

9.3.4.3.4.9 Evaporator Condensate Demineralizers

Two anion demineralizers remove any boric acid contained in the evaporator condensate. Hydroxyl based ion-exchange resin is used to produce evaporator condensate of high purity by releasing a hydroxyl ion when a borate ion is adsorbed. When resin is exhausted, the spent resin is flushed to the spent resin storage tank. The resin volume in each demineralizer is selected to keep resin replacements to an average of once per core cycle. The demineralizers are sized for a flow rate equal to the evaporator flow rate.

9.3.4.3.4.10 Condensate Filter

The filter collects resin fines and particulates larger than 25 microns from the boric acid evaporator condensate stream. The vessel is made of austenitic stainless steel, and is provided with a connection for draining and venting. Disposable synthetic filter elements are used.
The design flow capacity of the filter is equal to the boric acid evaporator flow rate.

9.3.4.3.4.11 Concentrates Filter

A disposable synthetic cartridge type filter removes particulates larger than 25 microns from the evaporator concentrates. Design flow capacity of the filter is equal to the boric acid evaporator concentrates transfer pump capacity. The vessel is made of austenitic stainless steel.

9.3.4.3.4.12 Concentrates Holding Tank

The concentrates holding tank is sized to hold approximately the concentrates produced during 1 day of evaporator operation. The tank is supplied with an electrical heater which prevents boric acid precipitation. It is constructed of austenitic stainless steel.

9.3.4.3.4.13 Concentrates Holding Tank Transfer Pumps

Two holding tank transfer pumps discharge boric acid solution from the concentrates holding tank to the boric acid storage tanks. The canned centrifugal pumps are sized to empty the concentrates holding tank in 30 minutes. The wetted surfaces are constructed of austenitic stainless steel.

9.3.4.3.4.14 Monitor Tanks

The original design was that two monitor tanks permit continuous operation of the evaporator train. When one tank is filled, the contents are analyzed and either reprocessed, discharged to the circulating water discharge, or recycled to the reactor makeup water tank. Each tank is sized to hold the condensate produced during 10 hr of operation from all evaporators at full output to ensure a maximum of two lab analyses per day. These tanks contain a diaphragm membrane and are stainless steel lined.

When the waste evaporator was decommissioned, a new demineralizer waste treatment system was installed. Since installation of this system, the A monitor tank is used for the waste demineralizer effluent. This effluent would be recirculated, sampled and released. To prevent cross contamination, only the B monitor tank is used for the evaporator output.

9.3.4.3.4.15 Monitor Tank Pump

One monitor tank pump discharges water from the monitor tanks. The pump is sized to empty a monitor tank in 2.0 hours. To protect the monitor tank pump, a low-level monitor tank cutout control is provided on the pump. The pump is constructed of austenitic stainless steel.

9.3.4.3.5 Valves

Some valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the waste disposal system. All other valves have stem leakage control. Globe valves are installed with flow over the seats when such an arrangement reduces the possibility of leakage. Basic material of construction is stainless steel for all valves except the batching tank steam jacket valves which are carbon steel.

Isolation valves are provided at all connections to the reactor coolant system. Lines entering the reactor containment also have check valves inside the containment to prevent reverse flow from the containment.

Relief valves are provided for lines and components that might be pressurized above design pressure by improper operation or component malfunction. Pressure relief for the tube side of the regenerative heat exchanger is provided by the charging line isolation valve which is designed to open when pressure under the seat exceeds reactor coolant pressure by 250 psi.

All relief valves used in systems handling radioactive fluids are of the closed bonnet design and are constructed of stainless steel.

9.3.4.3.6 Piping

All chemical and volume control system piping handling radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanged connections are required to facilitate equipment removal for maintenance and hydrostatic testing. Piping, valves, equipment, and line-mounted instrumentation, which normally contain concentrated boric acid solution, are heated by electrical tracing to ensure solubility of the boric acid.

9.3.4.4 System Evaluation

9.3.4.4.1 Availability and Reliability

A high degree of functional reliability is ensured in this system by providing standby components where performance is vital to safety and by ensuring fail-safe response to the most probable mode of failure. Special provisions include duplicate heat tracing with alarm protection of lines, valves, and components normally containing concentrated boric acid.

The system has three high-pressure charging pumps, each capable of supplying the normal reactor coolant pump seal and makeup flows.

The electrical equipment of the chemical and volume control system is arranged so that multiple items receive power from two 480-V buses (see Drawing 33013-0652). Two of the charging pumps and one of the boric acid transfer pumps are powered from one 480-V safeguards bus and the remaining charging pump and boric acid transfer pump, are powered from the other 480-V safeguards bus. One charging pump and one boric acid transfer pump are capable of meeting MODE 5 (Cold Shutdown) requirements shortly after full power operation. One charging pump taking suction from the refueling water storage tank (RWST) is capable of meeting MODE 5 (Cold Shutdown) requirements following cooldown from MODE 3 (Hot Shutdown) conditions. In case of a loss of offsite power, a charging pump and a boric acid transfer pump can be placed on the emergency diesels, if necessary.

The dc feed for the charging pump 1A control circuits comes from a common dc bus in bus 14 switchgear. A fire in the control complex, cable tunnel, or auxiliary building basement and mezzanine could damage the control circuits for charging pump 1A. Therefore, as part of the originally-designed alternative shutdown system, a transfer switch is available to transfer the dc feed to an alternative source of dc power. See Section 7.4.4. The transfer is a manual operation.

The power feed for charging pump 1B from bus 16 to the charging pump motor could fail due to direct impingement from a high-energy heating or process line break in the auxiliary building basement. A spare cable is stored in an area outside the auxiliary building.

It is estimated that the auxiliary building could be restored to ambient conditions and the spare cable installed in less than 8 hr, which would precede the need for the charging pump. (See Section 3.6.2.5.1.8.)

9.3.4.4.2 Seismic Analysis

The majority of the chemical and volume control system piping, as shown in Drawings 33013-1264 through 33013-1269 as well as certain components of the auxiliary and emergency systems such as pumps, heat exchangers, tanks, and valves are designated as Seismic Category I. The seismic analysis methods and criteria are presented in Section 3.9.2.

9.3.4.4.3 Leakage Prevention

Quality control of the material and the installation of the chemical and volume control valves and piping, which are designated for radioactive service, is provided in order to essentially eliminate leakage to the atmosphere. The components designated for radioactive service are provided with welded connections to prevent leakage to the atmosphere. However, flanged connections are provided on each charging pump suction and discharge, on each boric acid transfer pump suction and discharge, on the relief valves inlet and outlet, on three-way valves, and on the flow meters to permit removal for maintenance.

The positive displacement charging pumps stuffing boxes are provided with leakoffs to collect reactor coolant before it can leak to the atmosphere. All valves, which are larger than 2 in. and which are designated for radioactive service at an operating fluid temperature above 212°F, are provided with a stuffing box and lantern leakoff connections. Leakage to the atmosphere is essentially zero for these valves. All control valves are either provided with stuffing box and leakoff connections or are totally enclosed. Leakage to the atmosphere is essentially zero for these valves.

Diaphragm valves are provided where the operating pressure and the operating temperature permit the use of these valves. Leakage to the atmosphere is essentially zero for these valves.

9.3.4.4.4 Incident Control

The letdown line and the reactor coolant pump seal-water return line penetrate the reactor containment. The letdown line contains three air-operated orifice valves inside the reactor containment (AOV 200A, 200B and 202) and one air-operated valve outside the reactor containment (AOV-371), which are automatically closed by the containment isolation signal.

The reactor coolant pumps seal-water return line contains no containment isolation valves inside containment and one motor-operated isolation valve outside the reactor containment, which is automatically closed by the containment isolation signal. The line is a 3-in. line and terminates in the volume control tank, which has design pressure higher than containment accident pressure (see Section 6.2.4.4).

The two seal-water injection lines to the reactor coolant pumps, the normal charging line, and the alternate charging line are inflow lines penetrating the reactor containment. Each line contains multiple check valves inside the reactor containment to provide containment isolation in the event of a pipe break.

9.3.4.4.5 Malfunction Analysis

9.3.4.4.5.1 System Failures

To evaluate system safety, failures or malfunctions were assumed concurrent with a loss-of-coolant accident and the consequences analyzed and presented in Table 9.3-8. As a result of this evaluation, it is concluded that proper consideration has been given to station safety in the design of the system.

If a rupture were to take place between the reactor coolant loop and the first isolation valve or check valve, this incident would lead to an uncontrolled loss of reactor coolant. The analysis of loss-of-coolant accidents is discussed in Section 15.6.

Should a rupture occur in the chemical and volume control system outside the containment, or at any point beyond the first check valve or remotely operated isolation valve, actuation of the valve would limit the release of coolant and ensure continued functioning of the normal means of heat dissipation from the core. Even in the event of a failure of the check valve, the high-energy line break evaluation discussed in Section 3.6 demonstrated no loss of safety function. For the general case of rupture outside the containment, the largest source of radioactive fluid subject to release is the contents of the volume control tank. The consequences of such a release are considered in Section 15.7.

9.3.4.4.5.2 Inadvertent Dilution

When the reactor is subcritical, i.e. during MODE 5 (Cold Shutdown) or MODE 3 (Hot Shutdown), MODE 6 (Refueling), and approach to criticality, the relative reactivity status (neutron source multiplication) is continuously monitored and indicated by BF_3 counters and rate indicators. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate (see Table 9.3-6), is slow enough to give ample time to start a corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical. The maximum dilution rate is based on the abnormal condition of two charging pumps operating at full speed delivering unborated makeup water to the reactor coolant system at a particular time when the boric acid concentration is at the maximum value and the water volume in the system is at a minimum. The worst-case conditions for maximum boric acid concentration and minimum system water volume are chosen for the accident analysis for each reactor condition, i.e., MODE 6 (Refueling), cold or MODE 3 (Hot Shutdown), startup, and at power (see Section 15.4.4).

9.3.4.4.5.3 Alternative Methods of Boration

Normally, two of the three charging pumps are used in series with one of the two boric acid transfer pumps. An alternative method of boration would be to use the charging pumps directly from the refueling water storage tank (RWST). A third method would be to depressurize and use the safety injection pumps. There are two sources of borated water available for injection through three diverse methods:

a. The boric acid transfer pumps can deliver the boric acid storage tank contents to the charging pumps.

b. The charging pumps can take suction from the refueling water storage tank (RWST).

c. The safety injection pumps can take suction from the refueling water storage tank (RWST).

The malfunction or failure of one component will not result in the inability to borate the reactor coolant system. An alternate flow path for each of the methods above is available for boration of the reactor coolant.

9.3.4.4.5.4 Inadvertent Dilution of Boric Acid Storage Tanks

To guard against inadvertent dilution of the boric acid solution in the boric acid storage tanks, there is an interlock between the boric acid transfer pumps and the flow control valve FCV-110A that will close the valve when both pumps are not operating. The line between the blender and flow control valve includes a check valve V-355. These features prevent reactor makeup water from flowing back through the blender to the boric acid storage tanks. See Drawing 33013-1266.

9.3.4.4.5.5 Loss of Seal Injection Water

On loss of seal injection water to the reactor coolant pump seals, seal-water flow may be reestablished by manually starting a standby charging pump. Even if the seal-water injection flow is not reestablished, the plant can be operated for a relatively long period of time, such as 24 hours, without evaluation, because the thermal barrier cooler has sufficient capability to cool the reactor coolant flow, which would pass through the thermal barrier cooler and seal leakoff from the pump volute (*Reference 3*).

9.3.4.4.6 Overpressurization Protection

9.3.4.4.6.1 Suction Lines

Overpressurization protection for the chemical and volume control system refers to the system isolation capabilities from the high-pressure reactor coolant system. The chemical and volume control system suction line pressure reduction is provided by the three parallel letdown orifices, each of which is in series with a solenoid-operated valve. Each of these valves is operated from the control room where the valve position is indicated. The letdown orifices reduce the reactor coolant pressure below that of the chemical and volume control system. In addition, a relief valve, downstream of the letdown orifice valves, which has a capacity greater than the combined capacity of the three orifices, is located inside the containment and relieves to the pressurizer relief tank inside the containment.

Under SEP Topic XV-16, RG&E reviewed the radiological consequences of failure of small lines carrying primary coolant outside containment. The worst-case break was taken to be the chemical and volume control system letdown line break, with a break flow, rate of 60 gpm, and the assumption that the flash fraction of fission products contained in the leaked coolant was released. Assuming a previous iodine spike, the primary coolant activity was set at 60 μCi/g Iodine-131 dose equivalent. After a 20-min delay, operator action to isolate the break was assumed, based on available information such as volume control tank level, letdown line flow and pressure, and radiation monitors in the auxiliary building. The conclusion of this review, confirmed by an independent review by the NRC, was that the offsite radiological consequences are 1 rem whole body and 12 rem thyroid, a small fraction of the 10 CFR 100 guidelines.

9.3.4.4.6.2 Discharge Lines

The isolation of the chemical and volume control system discharge line is provided by a common discharge line check valve and a branch check valve in each of the three branches downstream of the common check valve. Drain fittings on the discharge line upstream of each check valve allow the valves to be tested. The discharge line of the chemical and volume control system is not classified as a low-pressure system connected to the reactor coolant system because the piping is 2500-psi-rated piping throughout its length to the positive displacement charging pumps. Ginna Station has not experienced failure of the positive displacement pumps to hold primary system pressure nor is any failure anticipated.

The charging and alternative charging lines were evaluated relative to GDC 55 and 56 requirements under SEP Topic VI-4, Containment Isolation Systems. Each line has a check valve inside containment that is leak tested (CV-370B in the charging line and CV-383B in the alternate charging line). These lines do not have a postaccident function. Acceptance of these lines was based on the following:

a. The piping system is designed to operate at 2250 psi.
b. The piping is Seismic Category I.
c. The charging pumps are positive displacement pumps and, therefore, leakage back through the pumps is expected to be minimal (see also Section 6.2.4.4, Class 3B).

9.3.4.4.7 Galvanic Corrosion

The only types of materials that are in contact with each other in borated water are stainless steels, Inconel, Stellite valve materials, and zircaloy/ZIRLO®/Optimized ZIRLO™ fuel element cladding. These materials have been shown (*References 1, 2, and 7*) to exhibit only an insignificant degree of galvanic corrosion when coupled to each other.

For example, the galvanic corrosion of Inconel versus 304 stainless steel resulting from high temperature tests (575°F) in lithiated boric acid solution was found to be less than -20.9 mg/dm^2 for the test period of 9 days. Further galvanic corrosion would be trivial since the cell currents at the conclusion of the tests were approaching polarization. Zircaloy versus 304 stainless steel was shown to polarize at 180°F lithiated boric acid solution in less than 8 days with a total galvanic attack of -3.0 mg/dm^2. Stellite versus 304 stainless steel was polarized in 7 days at 575°F in lithiated boric acid solution. The total galvanic corrosion for this couple was -0.97 mg/dm^2. These tests show that the effects of galvanic corrosion are insignificant to systems containing borated water.

9.3.4.4.8 Control of Tritium

The chemical and volume control system is used to control the release of tritium introduced into the reactor coolant system from the sources shown in Figure 9.3-1. Essentially all of the tritium is in chemical combination with oxygen as a form of water. Therefore, any leakage of coolant to the containment atmosphere carries tritium in the same proportion as it exists in the coolant. Thus, the level of tritium in the containment atmosphere, when it is sealed from outside air ventilation, is a function of tritium level in the reactor coolant, the cooling water temperature at the cooling coils, which determines the dewpoint temperature of the air, and the presence of leakage other than reactor coolant as a source of moisture in the containment air.

There are two major considerations with regard to the presence of tritium:

A. Possible plant personnel hazard during access to the containment.

B. Potential release of tritium to the environment.

Neither of these considerations is limiting at Ginna Station.

The concentration of tritium in the reactor coolant is maintained at a level which precludes personnel hazard during access to the containment. This is achieved by discharging part of the condensate from the boric acid recovery process to the lake via the plant circulating cooling water. The tritium released to the environment in this manner is between 10^{-2} and 10^{-3} of 10 CFR 20 limits.

9.3.4.4.9 Reactor Coolant Activity Concentration Calculations

9.3.4.4.9.1 Computation Method

The reactor coolant activity calculation assumes that the defective fuel rods are uniformly distributed throughout the core and the fission product escape rate coefficients are therefore based upon an average fuel temperature. The fission product activity in the reactor coolant during operation with small cladding defects in 1% of the fuel rods is computed using the following differential equations:

For parent nuclides in the coolant,

$$\frac{dN_{w_i}}{dt} = Dv_j \cdot N_{C_i} - \left(\lambda_i + R_{\eta_i} + \frac{B'}{B_o - tB'}\right) \cdot N_{w_i}$$

(Equation 9.3-1)

for daughter nuclides in the coolant,

$$\frac{dN_{w_j}}{dt} = Dv_j \cdot N_{C_j} - \left(\lambda_j + R_{\eta_i} + \frac{B'}{B_o - tB'}\right) \cdot N_{w_j} + \lambda_i N_{w_i}$$

(Equation 9.3-2)

where: N = population of nuclide

D =	fraction of fuel rods having defective cladding
R =	purification flow, coolant system volumes per sec
B_0 =	initial boron concentration, ppm
B'=	boron concentration reduction rate by feed-and-bleed, ppm per sec
η =	removal efficiency of purification cycle for nuclide
λ =	radioactive decay constant
ν=	escape rate coefficient for diffusion into coolant

Subscript C refers to core

Subscript w refers to coolant

Subscript i refers to parent nuclide

Subscript j refers to daughter nuclide

Table 9.3-9 lists the reactor coolant system equilibrium activities for fission products for use in dose and shielding calculations. Table 9.3-10 shows the parameters used in the calculation of the Table 9.3-9 Equilibrium Activities.

9.3.4.4.9.2 Tritium Production

Tritium is produced in the reactor from ternary fission in the fuel, irradiation of boron in the burnable poison rods and irradiation of boron, lithium, and deuterium in the coolant. The parameters used in the calculation of tritium production rate are presented in Table 9.3-11a.

9.3.4.4.9.3 Radioactivity Monitoring

During plant operation, continuous monitoring of the reactor coolant is accomplished by means of a detector assembly mounted at the letdown line. The detector is a Geiger-Mueller tube with a range of 0.1 mR/hr to 1E7 mR/hr. Indication and alarming in the control room of high radiation level requires that the operator immediately determine if the source of additional activity has resulted from failed fuel. The alarm setpoint is 200 mR/hr. A reading of 200 mR/hr corresponds to approximately 0.1% fuel rod cladding defects. Additional activity monitoring instrumentation is provided in the charging pump room.

The charging pump room instrumentation is used primarily for personnel warning. The normal reading is about 20 mR/hr and the alarm setpoint is 100 mR/hr.

9.3.4.4.9.4 Technical Specifications Limits

The Technical Specifications limits on reactor coolant leakage and activity are:

a. A known leakage source of 10 gpm.

b. An unidentified leakage source of 1 gpm.

c. Primary to secondary leakage of 150 gpd through any one steam generator.

d. Activity:

1. The Xenon-133 equivalent of the xenon activity in the reactor coolant shall not exceed 650 μCi/gm.
2. The Iodine-131 equivalent of the iodine activity in the reactor coolant shall not exceed 1.0 μCi/gm.
3. The Iodine-131 equivalent of the iodine activity on the secondary side of a steam generator shall not exceed 0.1 μCi/gm. This limit is required whenever the plant is above MODE 5 (Cold Shutdown).

Basis

The total activity limit for the primary system corresponds to operation with the plant design basis of l% fuel defects. The limit for secondary iodine activity is conservatively established with respect to the limits on primary system iodine activity and primary-to-secondary leakage.

The specified activity limits provide protection to the public against the potential release of reactor coolant activity to the atmosphere, as demonstrated by the analysis of a steam generator tube rupture accident (see Section 15.6.3).

9.3.4.4.9.5 Tritium Limit

A tritium limit is established to meet the allowable concentration in the circulating water discharge. The production rate of tritium in the fuel is calculated to be 10,240 Ci/cycle and 10% (design) or 2% (expected) is assumed to be released to the coolant by recoil through the cladding (see Table 9.3-11a and Table 9.3-11b). To this is added tritium from other sources for a total of approximately 2500 Ci/cycle (design) or 1609 Ci/cycle (expected) of total tritium activity added to the reactor coolant. With a turnover of 4.5 reactor coolant volumes per year in addition to the 3.7 reactor coolant volumes discharged per cycle through boron dilution, it is anticipated that tritium activity will remain below 3.5 μCi/g for the design release throughout the cycle. The methodology to determine fuel damage and clad damage based on coolant area radiation activity is provided in a procedure.

9.3.4.4.9.6 R.E. Ginna Normal Operation RCS and Secondary Coolant Sources

Normal Operation Sources

The normal operation source terms are based on the American National Standard (ANS) Source Specification, ANSI/ANS-18.1-1999 (*Reference 6*) entitled, "Radioactive Source Term for Normal Operation of Light Water Reactors." This standard establishes typical long-term concentrations of principal radionucides in fluid streams of light-water-cooled nuclear power plants for use in estimating the expected release of radioactive materials from various effluent streams. These fluid streams are the reactor coolant and the steam generator water and steam.

The purpose of this standard is to provide a uniform approach, applicable to light-water-cooled nuclear power plants, for the determination of expected concentrations in fluid streams. Through application of this standard, a common basis for the determination of radioactive source terms is established with the goal of providing a consistent approach for those involved in the design, licensing, and operation of nuclear power plants.

The numerical values given in the standard are based on available data from operating plants that use Zircaloy-clad, uranium-dioxide fuel. However, the standard stipulates that the values given will be revised periodically as additional plant operating data becomes available.

If the parameters such as power level, flow rates, and fluid quantities are those given in ANSI/ANS-18.1-1999, the source-term values given in the standard are to be used without modification. In cases where any parameter differs from the values given in ANSI/ANS-18.1-1999, one must account for these differences by using adjustment factors.

The pertinent plant parameters and assumptions are listed in Table 9.3-12 along with normal values specified in ANSI/ANS-18.1-1999. Since several of these quantities differ from the normal values specified in ANSI/ANS-18.1-1999, these values are considered in the determination of the adjustment factors, which are then applied to the standard source term values listed in the standard.

Table 9.3-13 gives the normal source based on ANSI/ANS-18.1-1999 for the R.E. Ginna Plant based on the 1775 MWt uprate power level.

9.3.4.5 Minimum Operating Conditions

The chemical and volume control system provides complete control of the reactor coolant system boron inventory. All three charging pumps are capable of injecting concentrated boric acid directly into the reactor coolant system. The volume of boric acid solution required to meet MODE 5 (Cold Shutdown) requirement shortly after full power operation is specified in the Technical Requirements Manual (TRM). (See also Table 9.3-6.)

The minimum volume from the boric acid storage tanks for the various concentrations allowed to meet MODE 5 (Cold Shutdown) conditions is tabulated in the Technical Requirements Manual (TRM). A range of concentrations from 6000 ppm to 23,000 ppm are allowed. Alternatively, 32,000 gallons of 2750 ppm borated water from the refueling water storage tank (RWST) will meet MODE 5 (Cold Shutdown) conditions. The amount of boric acid will vary from cycle to cycle. The required volume is associated with boration from just critical, hot zero power, peak xenon with control rods at the insertion limit, with single reactor coolant loop operation.

REFERENCES FOR SECTION 9.3

1. D. G. Sammarone, The Galvanic Behavior of Materials in Reactor Coolants, WCAP 1844, August 1961.

2. S. L. Davidson, VANTAGE + Fuel Assembly Reference Core Report, SCAP-12610-P-A, April 1995.

3. Westinghouse Electric Company Nuclear Safety Advisory Letter Number NSAL-99-005, Reactor Coolant Pump Operation During Loss of Seal Injection, dated June 1, 1999.

4. Letter from Robert L. Clark, NRC, to Robert C. Mecredy, RG&E, Subject: Amendment Re: Elimination of Post Accident Sampling System (TAC No. MB3387), dated January 17, 2002.

5. Deleted

6. American National Standard ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors," approved by the American Nuclear Society Standards Institute, Inc., LaGrange Park, Illinois, September 21, 1999.

7. Shah, H. H., “Optimized ZIRLO™,” WCAP-12610-P-A & CENPD-404-P-A Addendum 1-A, July 2006, Westinghouse Proprietary.

Table 9.3-1
NUCLEAR PROCESS SAMPLING SYSTEM CODE REQUIREMENTS

Sample heat exchanger	ASME III, Class C, tube side ASME VIII, shell side
Sample pressure vessels	ASME III, Class C
Piping and valves	USAS B31.1

Table 9.3-2
NUCLEAR PROCESS SAMPLING SYSTEM COMPONENTS

Sample Heat Exchanger(HX)

General	
Number	5
Type	Coil-in-shell
Design heat transfer rate (duty for 652.7°F saturated steam to 127°F liquid), each	2.14 x 10^5 Btu/hr
Shell	
Design pressure	150 psig
Design temperature	350°F
Component cooling water flow (maximum flow per HX)	40 gpm
Pressure loss at 40 gpm	15 psi
Nominal Component Cooling Water Flow (per HX)	15 gpm
Operating cooling water temperature, in (maximum)	105°F
Operating cooling water temperature, out (maximum)	130°F
Material	Carbon steel
Tubes	
Tube diameter	3/8 in., O.D.
Design pressure	2485 psig
Design temperature	680°F
Sample flow, normal, each	209 lb/hr
Maximum allowable pressure loss, each 209 lb/hr	10 psi
Operating sample temperature, in (maximum)	652.7°F
Operating sample temperature, out (maximum)	127°F
Material	Austenitic stainless steel

Sample Pressure Vessels

Number, total	8
Volume, pressurizer steam sample, two supplied	75 ml
Volume, pressurizer liquid sample, two supplied	75 ml
Volume, reactor coolant hot-leg sample, two supplied	75 ml
Volume, volume control tank sample, two supplied	75 ml

Design pressure	2485 psig
Design temperature	680°F

Table 9.3-3
MALFUNCTION ANALYSIS OF NUCLEAR PROCESS SAMPLING SYSTEM

Sample Chains	***Malfunction***	***Comments and Consequences***
Pressurizer steam space sample, pressurizer liquid space sample, or hot-leg sample	Remote operated sampling valve inside reactor containment fails to close	Diaphragm-operated valve outside the reactor containment closes on containment isolation signal
Any sample chain	Sample line break inside containment	Same as above

Table 9.3-4
POSTACCIDENT SAMPLING SYSTEM FUNCTIONAL REQUIREMENTS

Sample Source[a]	***In-Line Analysis Objective***	***Grab Sample***[b]
Reactor Coolant		
Pressurizer vapor space Pressurizer liquid space Hot leg	Boron, pH, dissolved oxygen, dissolved hydrogen, conductivity[c]	Diluted (1000:1)[d] grab samples (postaccident) Undiluted dissolved gas sample (normal operation (MODES 1 and 2)) Undiluted reactor liquid sample (normal operation (MODES 1 and 2))
Containment		
Sump A	Boron, pH, conductivity[bc]	Diluted (1000:1)[d] sample (postaccident) Undiluted (normal operation (MODES 1 and 2))
Atmosphere	Hydrogen, oxygen	Diluted (1000:1)[d] sample postaccident)

a. Sample must be obtained and analyzed within 3 hrs of the decision to sample.

b. Grab samples to be used for isotopic analysis in the Ginna Station counting room or for offsite analysis.

c. Connections are provided for chloride analysis by a portable instrument. These connections also used for obtaining undiluted samples during routine operation.

d. (1000:1) is a nominal design value. Gas samples diluted at approximately 200:1 or 1500:1.

Table 9.3-5
LIQUID AND GAS SAMPLE PANEL ANALYTICAL EQUIPMENT REQUIREMENTS

Parameter	***Measurement Technique***	***Range***	***Required Accuracy***
LIQUID SAMPLES			
pH	Probe	1-13	±0.1 pH Unit
Conductivity	Probe	0.1-500 μmho/cm	±3%
Dissolved oxygen[a]	Probe	0.01-20 ppm	±10%
Dissolved hydrogen[a]	Gas chromatograph	10-2000 cm^3/kg	±15%
Boron	Automatic titrimeter	50-6000 ppm	±1%
GAS SAMPLES			
Hydrogen	Gas chromatograph	0-10%	±5%
Oxygen	Gas chromatograph	0-30%	±5%

a. Not required for sump samples.

Table 9.3-6
CHEMICAL AND VOLUME CONTROL SYSTEM PERFORMANCE PARAMETERS[a]

Plant design life, years	40
Seal-water supply flow rate, gpm	16
Seal-water return flow rate, gpm	6
Normal letdown flow rate, gpm	40
Maximum letdown flow rate, gpm	60
Normal charging pump flow (one pump), gpm	46
Normal flow to reactor coolant pumps, gpm	16
Normal charging line flow, gpm	30
Maximum rate of boration with one transfer and one charging pump, ppm/min	31
Equivalent cooldown rate to above rate of boration, °F/min	9.4[b]
Maximum rate of boron dilution (two charging pumps), ppm/hr	707[b]
Two-pump rate of boration, using refueling water, ppm/min	6.2[b]
Equivalent cooldown rate to above rate of boration, °F/hr	1.9[b]
Temperature of reactor coolant entering system at full power, °F	544.8/540.2[c]
Temperature of coolant return to reactor coolant system at full power, °F	499.2/497.6[c]
Normal coolant discharge temperature to holdup tanks, °F	127.0
Volume of 2750 ppm borated water from the refueling water storage tank (RWST) required to meet MODE 5 (Cold Shutdown) requirements (gallons)	32,000[d] 37,000[e]
Volume of boric acid solution from the boric acid storage tanks required to meet MODE 5 (Cold Shutdown) requirements (gallons)	As required by the Technical Requirements Manual (TRM)

NOTE:—Volumetric flow rates in gpm are based on 127°F and 15 psig. Reactor coolant water quality is summarized in plant procedures.

a. Values listed in the table represent those from the original plant design and may differ from the current plant values due to power uprate.

b. Historic information. Boration capability evaluated on a cycle specific basis as parts of reload report review.

c. Original design/Uprate at T_{AVG} = 576°F.

d. Calculated for Power Uprate. Minor cycle to cycle variations may occur, which are reviewed by cycle specific reload reports.

e. Cycle 39 Value.

Table 9.3-7
PRINCIPAL COMPONENT DATA SUMMARY

HEAT EXCHANGERS	Quantity	Heat Transfer (Btu/hr)	Letdown Flow (lb/hr)	Letdown Δ T (°F)	Design Pressure (psig)	Design Temperature (°F)
					Shell/tube	Shell/tube
Regenerative	1	5.65×10^6	19,760	262	2485/2735	650/650
Nonregenerative	1	7.4×10^6	29,640	244	150/600	250/400
Seal-water	1	1.19×10^6	79,940	16	150/150	250/200
Excess letdown	1	1.88×10^6	4,940	357	150/2485	250/650

PUMPS	Quantity	Type	Capacity (gpm)	Head (ft)	Design Pressure (psig)	Design Temperature (°F)
Charging	3	Positive displacement	60	[a]	3000	250
Boric acid	2	Canned	40	235	150	250
Recirculation	1	Centrifugal	500	100	75	200
Reactor makeup water	2	Centrifugal	60	235	150	250
Monitor	1	Centrifugal	60	235	150	250
Concentrates holding tank transfer	2	Canned	20	150	150	250
Gas stripper feed	2	Canned	12.5	200	150	200
Gas stripper bottom	2	Canned	12.5	125	75	300

TANKS	Quantity	Type	Volume (Gal)	Design Pressure (psig)	Design Temperature (°F)
Volume control	1	Vertical	1500	75/15 Int/Ext	250
Charging pump pulse dampener	1	Horizontal	NA	3000	250

Boric acid	2	Vertical	4348[b]	Atmosphere	250
Chemical mixing	1	Vertical	3.0	150	200
Batching	1	Jacket bottom	400	Atmosphere	250
Holdup	3	Vertical	4165 ft^3 [c]	15	200
Reactor makeup water	1	Vertical	75,000	Atmosphere	100
Concentrates holding	1		700	Atmosphere	250
Monitor	2	Diaphragm	7500	Atmosphere	125
Hydropad Accumulator	1	Vertical	Gas: 500in^3 Water: 300in^3	500	-325 to 1200
Nitrogen Gas Trap	1	Horizontal	9.75(1.303ft^3)	20	250

DEMINERALIZERS	**Quantity**	**Type**	**Resin Volume (ft^3)**	**Flow (gpm)**	**Design Pressure (psig)**	**Design Temperature (°F)**
Mixed-bed	2	Flushable	12.0	40	200	250
Cation bed	1	Flushable	12.0	40	200	250
Base removal and cation	4	Flushable	12.0	12.5	200	250
Evaporator condensate	2	Fixed/Flushable	12.0	12.5	200	250
Deborating	2	Fixed/Flushable	30.0	40	200	250

a. Head limited by discharge relief valve.

b. Per EWR 3881 calculations.

c. The storage capacity of three of these tanks equals two reactor coolant system volumes.

Table 9.3-8
MALFUNCTION ANALYSIS OF CHEMICAL AND VOLUME CONTROL SYSTEM

Component	***Failure***	***Comments and Consequences***
Letdown line	Rupture in the line inside the reactor containment	The remote air-operated valve (AOV 427) located near the main coolant loop is closed on low pressurizer level or a containment isolation signal. The containment isolation valve in the letdown line (AOV 371) outside the reactor containment is automatically closed by the containment isolation signal initiated by the concurrent loss-of-coolant accident. The closure of AOV 371 and the letdown system orifice valves (AOVs 200A, 200B, and 202, which close on a containment isolation signal fed from AOV 427) limits the leakage of the reactor containment atmosphere outside the reactor containment.
Normal and alternate charging line	Rupture in the line inside the reactor containment (upstream of the check valves)	The check valves located near the main coolant loops (CV 295 and CV 9314, normal charging, and CV 383A, alternate charging) prevent loss of coolant through the line rupture. The air-operated valve located upstream of the check valve (AOV 294, normal charging, AOV 392B, alternate charging) in the defective line can also be closed to isolate the reactor coolant system from the rupture. The check valves located at the boundary of the reactor containment (CV 370B, normal charging at penetration 100; CV 383B, alternate charging at penetration 102) limit the leakage of the reactor containment atmosphere outside the reactor containment.
Seal-water return line	Rupture in the line inside the reactor containment	The motor-operated isolation valve MOV 313 located outside the containment is manually closed or is automatically closed by a containment isolation signal. The closure of that valve limits the leakage of the reactor containment atmosphere outside the reactor containment.

Table 9.3-9
REACTOR COOLANT SYSTEM EQUILIBRIUM ACTIVITIES

Activation Products

Nuclide	µCi/g
Cr-51	5.40E-03
Mn-54	1.60E-03
Mn-56	2.20E-02
Fe-55	2.10E-03
Fe-59	5.10E-04
Co-58	1.40E-02
Co-60	1.30E-03

Non-Volatile Fission Products (Continuous Full Power Operation)

Nuclide	µCi/g	Nuclide	µCi/g	Nuclide	µCi/g	Nuclide	µCi/g
Br-83	1.00E-01	Rb-86	3.76E-02	Mo-99	8.38E-01	Te-132	3.15E-01
Br-84	4.90E-02	Rb-88	4.40E+00	Te-99m	7.78E-01	Te-134	3.15E-02
Br-85	5.70E-03	Rb-89	2.00E-01	Ru-103	6.11E-04	Ba-137m	2.15E+00
I-129	6.86E-08	Sr-89	4.56E-03	Rh-103m	6.14E-04	Ba-140	4.43E-03
I-130	4.41E-02	Sr-90	2.33E-04	Ru-106	2.12E-04	La-140	1.52E-03
I-131	3.05E+00	Sr-91	6.00E-03	Rh-106	2.12E-04	Ce-141	6.80E-04
I-132	2.97E+00	Sr-92	1.32E-03	Ag-110m	1.99E-03	Ce-143	5.41E-04
I-133	4.72E+00	Y-90	6.68E-05	Te-125m	7.75E-04	Pr-143	6.55E-04
I-134	6.49E-01	Y-91m	3.26E-03	Te-127m	3.46E-03	Ce-144	5.14E-04
I-135	2.59E+0	Y-91	6.00E-04	Te-127	1.43E-02	Pr-144	5.14E-04
Cs-134	3.22E+00	Y-92>	1.16E-03	Te-129m	1.17E-02		
Cs-136	3.90E+00	Y-93	3.96E-04	Te-129	1.46E-02		
Cs-137	2.27E+00	Zr-9	6.99E-04	Te-131m	2.68E-02		
Cs-138	1/06E+00	Nb-95	7.03E-04	Te-131	1.40E-02		

Gaseous Fission Products

Nuclide	µCi/g	>Nuclide	µCi/g
Kr-83m	4.74E-01	Xe-133m	3.84E+00
Kr-85m	1.93E+00	Xe-133	2.71E+02
Kr-85	8.21E+00	Xe-135m	5.58E-01

Kr-87	1.24E+00	Xe-135	9.49E+00
Kr-88	3.60E+00	Xe-137	1.91E-01
Kr-89	1.00E-01	Xe-138	6.92E-01
Xe-131m	3.54E+00		

Table 9.3-10
PARAMETERS USED IN THE 1811 MWT UPRATE CALCULATION OF REACTOR COOLANT FISSION PRODUCT ACTIVITIES

1.	Core thermal power, MWt	1811
2.	Cycle Length (days)	575.5
3.	Initial Boron Concentration (ppm)	1967
4.	Fuel Defect Level (%)	1
5.	Reactor Coolant Mass (g)	1.123×10^{8}
6.	Purification System Flow Rate (gpm)	40
7.	Purification System Flow Temperature (°F)	127
8.	Purification System Flow Pressure (psig)	15
9.	Purification System Demineralizer Resin Volume (ft^3	12
10.	Volume Control Tank Volumes	
	Vapor, ft^3	100
	Liquid, ft^3	100
11.	Volume Control Tank Temperature (°F)	127
12.	Volume Control Tank Vapor Purge Rate (cfm)	0
13.	Fission Product Escape Rate Coefficients	
	Noble gas isotopes, sec^{-1}	6.5×10^{-8}
	Br, Rb, I and Cs isotopes (sec^{-1})	1.3×10^{-8}
	Te isotopes, sec^{-1}	1.0×10^{-9}
	Mo, Tc and Ag isotopes, sec^{-1}	2.0×10^{-9}
	Sr and Ba isotopes, sec^{-1}	1.0×10^{-11}
	Y, Zr, Nb, Ru, Rh, La, Ce and Pr (sec^{-1})	1.6×10^{-12}
14.	Mixed-bed demineralizer decontamination factors	
	Noble gases and Cs-134, Cs-136 and Cs-137	1
	All other isotopes	10
15.	Cation Bed Demineralizer Decontamination Factor for Cs-134, Cs-137 and Rb-86	10

Table 9.3-11a
PARAMETERS USED IN THE CALCULATION OF TRITIUM PRODUCTION IN THE REACTOR COOLANT - BASIC ASSUMPTIONS

Plant Parameters:

1.	Core thermal power, MWt	1811
2.	Coolant water volume, ft^3	5441[a]
3.	Core water volume, ft^3	354
4.	Core water mass (grams)	7.324 E+06
5.	Plant full power operating time	
	• Equilibrium cycle	79.71 weeks (18.33 months)
6.	Boron concentrations (Peak hot full power equilibrium Xenon)	
	• Equilibrium cycle, ppm	1801
7.	Burnable poison boron content (total-all rods), kg	2.93[b]
8.	Fraction of tritium in core (ternary fission + burnable boron) diffusing through clad	
	• Equilibrium (design)>	0.10
	• Equilibrium (expected)	0.02
9.	>Ternary fission yield	8×10^{-5} atoms/fission

a. Minimum value based on PCWG parameters for Cases 3 and 4 with 10% tube plugging and 35% pressurizer water level.

b. Effective derived quantity based on 3 regions loaded in subsequent cycles with each having an initial mass of 1123 grams.

Table 9.3-11b
CALCULATION OF TRITIUM PRODUCTION IN THE REACTOR COOLANT

Calculations	***Equilibrium Cycle Design Value (Ci/cycle)***	***Equilibrium Cycle Expected Value (Ci/cycle)***
Tritium from core		
1. Ternary fission	10240	10240
2. ^{10}B (n, 2α) T (in poison rods)	152	152
3. ^{10}B (n, α) ^{7}Li (n, nα) T (in poison rods)	744	744
4. Release fraction	0.10	0.02
5. Total released to coolant	1114	223
Tritium from coolant		
1. ^{10}B (n, 2α) T	1033	1033
2. Li (n, nα) T (limit 2.2 ppm Li)	10.19	10.19
3. ^{6}Li (n, α) (purity of ^{7}Li = 99.9%)	337.80	337.80
4. D2 (n,y)	5.38	5.38
5. Release fraction	1.0	1.0
6. Total release to coolant	1386	1386
Total tritium in coolant	2500	1609

Table 9.3-12
ANSI/ANS 18.1-1999 NORMAL SOURCE INPUT PARAMETERS

Parameter	Symbol	Value	Units	Nominal Value	
Core Thermal Power	P	1.811E+03[a]	MWt	3.4E+03	MWt
Weight of water in reactor coolant system	WP	4.07E+04	gal	2.5E+05	kg
Reactor coolant letdown flow rate (purification)	FD*	4.00E+04	gal	4.7E+00	kg/sec
Reactor coolant letdown flow rate (yearly average for boron control)	FB	1.66E-01	gpm	6.3E-02	kg/sec
Flow through the purification system cation demineralizer	FA	4.00E+00	gpm	4.7E-01	kg/sec
Steam flow rate	FS	7.41E+06	lb/hr	1.9E+03	kg/sec
Weight of secondary side water in all steam generators<	WS	1.67E+05	lb	2.0E+05	kg
Steam generator blow-down flow rate (total)	FBD	1.00E+02	gpm	9.5E+00	kg/sec
Density of RCS Water	Drcs	4.48E+01	lb/ft^3		
VCT Liquid Volume	VOL-L*	1.00E+02	ft^3		
VCT Vapor Space Volume	VOL-V*	1.00E+02	ft^3		
VCT Purge Rate	PR*	0.00E+00	scfm		
Density of VCT Water	Dvct*	6.16E+01	lb/ft^3		
VCT Temperature	TEMP*	1.27E+02	deg F		
VCT Vapor Pressure	PRESS	1.50E+01	psig		

Notes: Values for NB, NA, NBD, NC, NS and NX are N-18.1 values. Symbols marked with (*) are used in noble gas stripping factor calculations (Y).

a. 102% of 1775 MWt

Table 9.3-13
R. E. GINNA NORMAL PLANT OPERATIONAL SOURCES BASED ON ANSI/ANS 18.1 – 1999 (uCi/g)

		Secondary Side				Secondary Side	
Nuclide	**RCS**	**Water**	**Steam**	**Nuclide**	**RCS**	**Water**	**Steam**
Class 1				**Class 6**			
Kr-85m	1.5E-02	>nil	6.5E-09	Na-24	5.0E-02	2.6E-06	1.3E-08
Kr-85	1.3E+00	>nil	5.6E-07	Cr-51	3.0E-03	1.9E-07	9.2E-10
Kr-87	1.8E-02	>nil	2.2E-08	Mn-54	1.5E-03	9.5E-08	4.8E-10
Kr-88	1.8E-02	nil	7.7E-09	Fe-55	1.2E-03	7.2E-08	3.6E-10
Xe-131m	7.7E-01	nil	3.2E-07	Fe-59	2.9E-04	1.8E-08	8.9E-11
Xe-133m	6.3E-02	nil	2.8E-08>	Co-58	4.5E-03	2.8E-07	1.4E-09
Xe-133	2.7E-02	nil	1.2E-08	Co-60	5.1E-04	3.2E-08	1.6E-10
Xe-135m	1.5E-01	nil	6.4E-08	Zn-65	4.9E-04	3.1E-08>	1.5E-10
Xe-135	6.2E-02	nil	2.6E-08	Sr-89	1.4E-04	8.3E-09	4.2E-11
Xe-137	4.1E-02	nil	1.7E-07	Sr-90	1.2E-10	7.2E-10	3.6E-12
Xe-138	7.2E-02	nil	3.1E-08	Sr-91	1.0E-03	5.2E-08	2.6E-10
				Y-91m	5.4E-04	8.9E-09	4.4E-11
				Y-91	5.0E-06	3.1E-10	1.6E-12
Class 2							
Br-84	1.9E-02	2.2E-07	2.2E-09	Y-93	4.5E-03	2.2E-08	1.1E-09
I-131	2.0E-03	1.3E-07	1.3E-09	Zr-95	3.8E-04	2.3E-08	1.2E-10
I-132	7.0E-02	2.2E-06	2.2E-08	Nb-95	2.7E-04	1.6E-08	8.4E-11
I-133	2.8E-02	1.6E-06	1.6E-08	Mo-99	6.4E-03	3.8E-07	1.8E-09
I-134	1.2E-01	2.0E-06	2.0E-08	e-99m	5.2E-03	2.2E-07	1.2E-09
I-135	6.2E-02	2.8E-06	2.8E-08	Ru-103	7.3E-03	4.5E-07	2.3E-09

		Secondary Side	
Nuclide	**RCS**	**Water**	**Steam**
Class 1			
Class 3			
Rb-88	2.3E-01	1.6E-06	7.8E-09
Cs-134	3.6E-05	2.2E-09	1.2E-11
Cs-136	8.5E-08	5.3E-08	2.7E-10
Cs-137	5.1E-05	3.3E-09	1.6E-11
Class 4			
N-16	4.0E+01	2.6E-06	2.6E-07
Class 5		<	
H-3	1.0E+00	1.0E-03	1.0E-03>

		Secondary Side	
Nuclide	**RCS**	**Water**	**Steam**
Class 6			
Ru-106	8.7E-02	5.4E-06	2.6E-08
Ag-110m	1.3E-03	7.7E-08	3.9E-10
Te-129m	1.8E-04	1.1E-08	5.7E-11
Te-129	2.8E-02	5.9E-07	2.9E-09
Te-131m	1.5E-03	8.7E-08	4.4E-10
Te-131	9.2E-03	8.5E-08	4.4E-10
Te-132>	1.7E-03	1.0E-07	5.0E-10
Ba-140	1.3E-02	7.7E-07	3.8E-09
La-140	2.5E-02	1.5E-06	7.3E-09
Ce-141	1.5E-04	8.9E-09	4.5E-11
Ce-143	2.8E-03	1.6E-07	8.2E-10
Ce-144	3.9E-03	2.3E-07	1.2E-09
W-187	2.6E-03	1.4E-07	7.3E-10
Np-239	2.2E-03	1.3E-07	6.5E-10

9.4 AIR CONDITIONING, HEATING, COOLING, AND VENTILATION SYSTEMS

9.4.1 CONTAINMENT VENTILATION SYSTEM

9.4.1.1 Design Bases

9.4.1.1.1 Design Objectives

The containment ventilation systems are designed to accomplish the following:

A. Remove the normal heat loss from the equipment and piping in the reactor containment during plant operation and maintain a normal ambient temperature of less than 125°F, 50% relative humidity.

B. Provide sufficient air circulation and filtering throughout all containment areas to permit safe and continuous access to the reactor containment within 2 hours after reactor shutdown, assuming defects exist in 1% of the fuel rods.

C. Provide for positive circulation of air across the refueling water surface to ensure personnel access and safety during shutdown.

D. Provide a minimum containment ambient temperature of 50°F during reactor shutdown.

E. Provide for purging of the containment to the plant vent for dispersion to the environment as allowed by applicable regulations.

F. Provide for backup purging of the containment following an accident. The design for post-accident conditions and operating criteria are described in Section 6.2.2.

In order to accomplish these objectives, the following systems are provided:

AA. Containment recirculation cooling and filtration system.

BB. Control rod drive mechanism cooling system.

CC. Reactor compartment cooling system.

DD. Refueling water surface and purge system.

EE. Containment auxiliary charcoal filter system.

FF. Containment post-accident charcoal filter system.

GG. Containment shutdown purge system.

HH. Containment mini-purge system.

II. Penetration cooling system.

The design characteristics of the equipment required in the containment for cooling, filtration, and heating to handle the normal thermal and air cleaning loads during MODES 1 and 2 are presented in Table 9.4-1. In certain cases where engineered safety features functions also are served by the equipment, component sizing is determined from the required operating specifications associated with the design-basis accident, described in Sections 6.3 and 15.6.

9.4.1.1.2 Design Criteria

The design criteria below, associated with inspection and testing of the air cleanup system, were used during the licensing of Ginna Station. They represent the Atomic Industrial Forum version of proposed criteria (AIF-GDC) issued by the AEC for comment on July 10, 1967. Conformance of the ventilation systems to the General Design Criteria of 10 CFR 50, Appendix A (i.e., GDC 2, 4, 5, 17, 19, 60, and 61) is discussed in Section 3.1.2. Conformance with Regulatory Guides and IEEE Standards is discussed in Section 1.8.

Inspection of Air Cleanup Systems

CRITERION: Design provisions shall be made, to the extent practical, to facilitate physical inspection of all critical parts of containment air cleanup systems, such as ducts, filters, fans, and dampers (AIF-GDC 62).

Access is available for visual inspection of the fan cooler and recirculation filtration systems and components.

Testing of Air Cleanup System Components

CRITERION: Design provisions shall be made, to the extent practical, so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance (AIF-GDC 63).

Periodic tests of the dampers associated with the charcoal filter units of the containment air cleanup system are conducted. Each damper is started and operation (including stroke time) is checked by personnel in containment. An indicating light in the control room provides indication of damper movement. Periodic tests also verify that the dampers fail in a safe position on loss of air and that air flow and orientation for accident operation is acceptable.

Testing Air Cleanup Systems

CRITERION: A capability shall be provided, to the extent practical, for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits (AIF-GDC 64).

Each containment recirculation fan unit is checked periodically for water in the filtration area. Also, charcoal filters are tested for bypass flow and pressure drop and are visually inspected for damage and loss of charcoal.

Further, a representative sample cell is removed during shutdown and tested periodically to verify its continued efficiency. After reinstallation, the filter units will be tested in-place by aerosol injection to determine integrity of the flow path.

Initial Testing of Operational Sequence of Air Cleanup Systems

CRITERION: A capability shall be provided to test initially under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability (AIF-GDC 65).

Means were provided to test initially under conditions as close to design and as near as was practical the full operational sequence that would bring the fan cooler and recirculation filtration systems into action, including transfer to the emergency diesel-generator power source.

9.4.1.2 System Design

9.4.1.2.1 Introduction

The containment ventilation, purging, and recirculation cooling and filtration systems flow diagram is shown in Drawings 33013-1863 through 33013-1866. The containment recirculation cooling and filtration system and the purge and depressurization system are designed as Seismic Category I.

The containment recirculation fans and reactor compartment fans are direct-driven units, each with standby units for redundancy. Control rod drive fans were sized to provide adequate cooling with one fan operation. However, standard practice is two fan operation. The fans and motors of these units are provided with vibration detecting devices to detect abnormal operating conditions in the early stages of the disturbance. Each of the associated systems is provided with flow switches to verify existence of air flow in the associated duct system. Dampers in the following systems and ducts are provided with air by dual supply air mains: primary compartment ducts, dome ducts, containment auxiliary charcoal filter systems, butterfly valves which isolate the post-accident charcoal filters, and containment purge supply and exhaust ducts.

9.4.1.2.2 Containment Recirculation Cooling and Filtration System

The containment recirculation fan cooler (CRFC) function during MODES 1 and 2 is accomplished using the air handling units with common, headered discharge ducting to ensure adequate distribution of filtered and cooled air throughout the containment (see Drawing 33013-1863). The cooling capacity and air flow rates of the units are discussed in Section 6.2.2.1. Condensate collected from the units is discussed in Section 5.2.5.4.3.

Each air handling unit consists of the following equipment arranged so that during MODES 1 and 2, air flows through the assembly in the following sequence: entering louvers, cooling coils, moisture separators (demisters), high efficiency particulate air filters, direct-drive motor and centrifugal fan, and supply header.

In the event of a loss-of-coolant accident, the sequence of flow would be as above but in addition, for two of the units, after passing through the fan the flow would be directed through an alternative post-accident path containing charcoal filters and then through gratings discharging onto the operating floor area of containment.

During MODES 1 and 2, the charcoal filters are isolated from air flow on both the upstream and downstream sides by butterfly valves. These valves are automatically opened by the accident signal, which also closes a similar valve to block the normal discharge path from the fan. The isolation of the charcoal under normal conditions maintains a high degree of charcoal activation.

The containment recirculation fan cooler (CRFC) units are located in the space between the reactor coolant loop shield wall and the containment wall. The shielded location makes inspection of the equipment possible at power under restricted area access conditions and immediately after entry into MODE 3 (Hot Shutdown).

The fans, motors, electrical connections, and all other equipment in the containment necessary for operation of the system are capable of operating under the environmental conditions following a loss-of-coolant accident.

Two of the four fans and coolers plus one containment spray pump (i.e. one train of each system) are required to provide sufficient capacity to maintain the containment pressure within design limits after a loss-of-coolant accident or steam line break accident. See Section 6.2.2 for a discussion of the post-accident performance of these systems.

During power operation, containment integrity is maintained with no release from the containment ventilation system to the atmosphere except as described in Section 12.3.2.2.7 and as required to maintain containment pressure within the requirements of Technical Specifications. Prior to purging the containment air using the containment mini-purge system, particulate and radiogas monitor indications of the closed containment activity levels are used to guide releases from the containment. During power operation, the containment particulate and radiogas monitor (R-11 and R-12) indications help determine the desirability of using either one or both of two auxiliary particulate and charcoal filter units installed in the containment primarily for pre-access cleanup.

When containment purging for access is in progress, releases from the plant vent are continuously monitored with a radiogas monitor.

9.4.1.2.3 Control Rod Drive Mechanism Cooling System

The control rod drive cooling system consists of fans and ductwork to draw air through the control rod drive mechanism shroud and eject it to the main containment volume (see Drawing 33013-1864). *Reference 10* provided upgraded ductwork for the control rod drive mechanism (CRDM) cooling system.

The purpose of the control rod shroud fans is to remove the heat generated by the control rod drive mechanism coils. This heat is dissipated from the reactor vessel through the shroud surrounding the control rod drive mechanism coils. The two fans take a suction from the shroud and discharge to the containment atmosphere above their missile barrier location. The fan motors are 60 hp, and each fan is rated for 14,000 cfm air flow. Backdraft dampers prevent reverse air flow through the fans.

A new Head Assembly Upgrade Package (HAUP) was provided by PCR 2001-0042 during the 2003 refueling outage.

The HAUP provided a shroud and shield assembly with retractable doors as shown in *Reference 11.* The HAUP provided a duct system for the control rod drive mechanism cooling air flow and also provides radiation shielding in the lower portion of the assembly.

The shroud assembly attaches to the existing lift rig circular ring beam lower face in order to form part of the ductwork for the control rod drive mechanism cooling. A plenum with access doors for the control rod drive mechanism and microprocessor rod position indication (MRPI) coil connectors has been added to the top of the existing lift rig to complete the cooling air flow path. The plenum attaches to existing duct work on the missile shield via a two piece removable duct section.

9.4.1.2.4 Reactor Compartment Cooling System

The reactor compartment cooling system consists of a plenum, cooling coils, fans, and ductwork arranged to supply cool air to the annulus between the reactor vessel and the primary shield and to the nuclear instrumentation external to the reactor (see Drawing 33013-1864).

These fans take a suction from the containment atmosphere and cool the air with service water (SW) through cooling coils before discharging through supply dampers to the area near containment sump A (directly under the reactor vessel). Air exits to the containment atmosphere around the loop nozzles and vessel seal ring area.

Each fan has a 30-hp motor rated for 21,175-cfm air flow. The cooling coils are supplied by service water (SW) through two normally open butterfly isolation valves which are located outside containment. The total cooling capacity is 342,000 Btu/hr for each unit, based on 80°F service water (SW) temperature and 120°F air inlet temperature. As discussed in *Reference 9 and Reference 12*, the increase in the maximum lake temperature to 85°F and the increase in containment temperature to 125°F does not prevent the reactor compartment cooling system from providing adequate cooling since a review of past plant instrumentation records during summer operation indicates that the air temperature exiting the reactor vessel annulus is maintained well below the design basis temperature.

Counterweighted manual backdraft dampers prevent reverse air flow through the fans.

9.4.1.2.5 Refueling Water Surface and Purge System

The purpose of this subsystem is to supply air to the surface of the refueling cavity and exhaust from the area above the refueling manipulator crane to protect the operators during MODE 6 (Refueling) operations. The system consists of a supply fan (3 hp, 6900 cfm) and an exhaust fan (7.5 hp, 11,000 cfm) located near the refueling cavity in containment. The flow path requires that the supply fan blow air over the surface of the refueling cavity where an exhaust fan removes the air and delivers it to the purge exhaust system. The fans are controlled from within the containment during MODE 6 (Refueling) operations. (See Drawing 33013-1864.)

9.4.1.2.6 Containment Auxiliary Charcoal Filter System

The purpose of this subsystem is to absorb radioactive iodine vapor and radioactive particles that may occur as a result of normal primary system leakage inside the containment. These fans are used during MODES 1 and 2 prior to containment entries and when containment radioactivity levels are high. The system consists of two fans and filter plenums. Each fan is rated at 5 hp and 5100 cfm. Each filter unit has six high efficiency particulate air filter cells rated at 6000 cfm and 16 charcoal filter cells rated at 5280 cfm. The air flow path is from the fan suction through a discharge damper and plenum entry damper (both air-operated) to the filter plenum and exhausts above the refueling cavity area. The fan discharge and plenum entry dampers are interlocked to open fully when the fan is started and to close when the fan is stopped. (See Drawing 33013-1864.)

9.4.1.2.7 Containment Post-accident Charcoal Filter System

Two of the containment air handling units have their air discharge routed automatically during an accident condition through charcoal filters before being discharged onto the operating floor area of containment. These post-accident charcoal filters are designed to remove iodine and/or radioactive particulates following an accident. The inlet valves to the filter plenums are air-operated, spring-loaded butterfly valves with control solenoids. Upon a loss of control signal or power, the valves are spring-loaded to open directing air through the filters as air pressure bleeds off. Each charcoal filter bank has 120 cells originally rated at 38,000 cfm. (See Section 6.5.1.2.1 for the latest rating figures.)

Each charcoal filter plenum also includes a dousing system (part of the containment spray discussed in Sections 6.2.2.2 and 6.5.1.2).

9.4.1.2.8 Containment Shutdown Purge System

The containment shutdown purge system is independent of the main auxiliary building exhaust system and includes provisions for both supply and exhaust air. The supply system includes an outside air connection to roughing filters, heating coils, fans, duct system, and supply penetration with a butterfly valve outside containment and a blind flange inside containment. The exhaust system includes an exhaust penetration with a butterfly valve and a blind flange identical to those above, a duct system, a filter bank with high efficiency particulate air and charcoal filters, fans, and a building exhaust vent. The charcoal filters are of the same type as those used in the post-accident system (Section 9.4.1.2.7). Both supply and exhaust systems include two fans with isolating dampers so that purging can be performed using one supply and exhaust fan or both supply and exhaust fans. Minimum design flow for operation using one supply and exhaust fan is 10,000 cfm assuming clean filters. (See Drawings 33013-1865 and 33013-1866.)

The shutdown purge supply and exhaust duct blind flanges inside the containment are closed during MODES 1, 2, 3 and 4. The blind flanges are equipped with double O-ring seals. Leakage can be checked periodically by pressurizing the duct between the blind flanges and butterfly valves outside containment. The outboard valves are designed for rapid automatic closing by a containment isolation signal or upon a signal of high activity level within the containment ventilation system. The blind flanges can only be removed during cold and MODE 6 (Refueling) shutdowns. The flanges and associated double seals provide containment isolation and are a containment boundary for MODES 1, 2, 3 and 4. During cold and MODE 6 (Refueling) shutdown, when the flanges are removed, closure of the containment penetration is provided by the butterfly valves outside the containment. See also Section 6.2.4.4.9.

The steam heating system is part of the containment purge system. It heats inlet air if outside temperatures are excessively low. System preheaters and two reheaters are supplied with 100-psig steam through a flow control valve which is thermostatically controlled. Steam pressure is normally reduced to 2 psig. Each reheater is rated at 94,600 Btu/hr and drains to the intermediate building floor drains. The heater coils will automatically receive steam when outside air supply temperature falls below 50°F.

9.4.1.2.9 Containment Mini-Purge System

The containment mini-purge system is capable of purging containment during MODES 1 and 2 at a relatively low flow rate (approximately 1500 cfm). The exhaust is through a 6-in. line to the auxiliary building charcoal filters. See Drawing 33013-1870. The exhaust line has an automatic air-operated butterfly isolation valve inside and outside containment. The supply system includes a 2000-cfm rated blower for supplying ambient air from inside of the intermediate building through a 6-in. line penetrating containment with air-operated, butterfly type, automatic inboard and outboard isolation valves. (See Drawing 33013-1865.) The supply and exhaust system isolation valves are capable of closing fully against 60 psig in a maximum of 2 sec after receiving an isolation signal. Operation of the mini-purge system is remote manual from the control room. It is intermittent and may be started, operated, and secured during all normal modes of plant operation. The containment isolation valves are quality Group B, Seismic Category I. Their control circuitry is Class 1E, Seismic Category I. (See also Section 6.2.4.4.)

The mini-purge system is connected to the plant vent. The system is automatically isolated on high radiation in containment. To ensure the containment sample monitored by the radiation detectors (R-11 and R-12) is representative of the containment atmosphere, at least one recirculation fan is required to be in operation during mini-purge operation.

9.4.1.2.10 Penetration Cooling System

Two penetration cooling fans are provided to cool hot mechanical penetrations. Each fan is powered from a separate motor control center. One cooling coil serviced from the service water (SW) system provides the heat sink. (See Drawing 33013-1866.) The inlet air ductwork to the fans draws outside air from the auxiliary building roof. The ductwork includes a backflow preventer to minimize the potential of an unmonitored radioactive discharge from the auxiliary building to the outside. The ductwork also includes a steam heating coil with a temperature control system that keeps the air temperature at the discharge of the containment penetration cooling fans, upstream of the cooling coil, in the range of 60°F to 65°F.

The containment penetration cooling system is designed to prevent the bulk concrete temperature surrounding the penetrations from exceeding 150°F.

9.4.2 AUXILIARY BUILDING VENTILATION SYSTEM

9.4.2.1 Design Basis

The auxiliary building ventilation system (ABVS) is designed to meet the following principal criteria:

A. Ensure adequate heat removal from equipment rooms and open areas such that ambient temperature limits are not exceeded. Conservatively estimated design parameters, dictated by the requirements for engineered safety features operation and by good engineering practice, were used to establish ambient temperature limits. An evaluation of heat loads and temperatures in the auxiliary building following a loss of offsite power was conducted as part of the RG&E environmental qualification program. It was shown that required ambient conditions were not exceeded even with the minimum complement of ventilation equipment available (*Reference 1*).

B. Control the direction of flow of airborne radioactivity from areas of low activity toward areas of higher activity.

9.4.2.2 System Design and Operation

9.4.2.2.1 System Design Objective

The auxiliary building ventilation system (ABVS) provides clean, filtered, and tempered air to the operating floor of the auxiliary building. Air from within the Auxiliary Building sweeps the surface of the decontamination pit and spent fuel pool (SFP). The system exhausts air from the equipment rooms and open areas of the auxiliary and intermediate buildings, the decontamination pit area, and spent fuel pool (SFP) area through a closed exhaust system. The exhaust system includes a 100% capacity bank of high efficiency particulate air filters and redundant 100% capacity fans discharging to the atmosphere via the plant vent. This arrangement ensures the proper direction of air flow for removal of airborne radioactivity from the auxiliary building. The auxiliary building main exhaust fans have inlet flow control dampers that modulate automatically to maintain an acceptable negative pressure at the fan's suction header. On receipt of a high radiation alarm, the auxiliary building supply fans and all exhaust fans of the system are tripped except those exhausting to the plant vent through the charcoal filters.

The auxiliary building ventilation system (ABVS) is included in Drawings 33013-1869 through 33013-1872.

9.4.2.2.2 Charcoal Filter Circuit

A separate charcoal filter circuit is included in the auxiliary building exhaust system which exhausts from rooms where fission product activity may accumulate during MODES 1 and 2 in concentrations exceeding the average levels expected in the rest of the building. Following a loss-of-coolant accident, this filter circuit is capable of providing exhaust ventilation from the areas containing pumps and related piping and valving which are used to recirculate containment sump liquid.

A full-flow charcoal filter bank is provided in the circuit, along with two 50%-capacity exhaust fans. The air-operated suction and discharge dampers associated with each fan are interlocked with the fan such that they are fully open when the fan is operating and fully closed when the fan is stopped. These dampers fail to the open position on loss of control signal or control air. Each fan is connected to a separate emergency power bus, but both fans can be operated from the same bus after passage of the peak post-accident diesel loading.

The charcoal filter fans discharge through the 1G exhaust fan to the main auxiliary building exhaust system containing the high efficiency particulate air filter bank. A fail open damper is installed in a bypass circuit around the two main exhaust fans to ensure a path for the charcoal and high efficiency particulate air filtered exhaust to the plant vent if the main exhaust fans are not operating.

9.4.2.2.3 System Operation

The auxiliary building ventilation system (ABVS) provides a minimum of six air exchanges per hour for each of the rooms and open areas of the building; this ensures adequate heat removal from most operating equipment. However, a total of seven separate cooling units are provided for safety-related pumps. Two redundant cooling units each are provided for both the charging pump room and the residual heat removal pump pit. Three cooling units are provided for the safety injection and containment spray pumps, headered into a common set of ductwork. The coils for the safety injection and containment spray pumps have been blanked off such that the associated fans are operable but the air is no longer cooled by service water (SW). One unit is powered from bus 14, one from bus 16, and one has a swing unit design during automatic fan operation mode that is not used. This unit is preferentially loaded from a bus 16 source, with a bus 14 source (breaker) available through manual transfer. The cooling units are operated whenever any of the associated pumps are in service and are designed with separation of cooling water and electrical services, and provision for operation on emergency power. These cooling units consist basically of a fan, a water-cooled heat exchanger and ductwork for circulating cooled air to the residual heat removal and charging pump rooms, and to an area near the safety injection and containment spray pump motors.

The cooling units are designed and installed in accordance with Seismic Category I criteria. However, since at least one charging pump is run intermittently/continuously for normal plant operation, a charging pump cooling unit may be run intermittently/continuously to cool the pump motor(s) to extend equipment life. Charging pumps are not required to survive a harsh environment. Also, each cooling unit is sufficient to maintain acceptable room temperatures with the minimum number of charging pumps required for system operation in service. The remaining five room coolers are not required for the operation of their associated pump motors even with both trains of engineered safety features operating (*Reference 2*).

Ventilation for the decontamination pit area and spent fuel pool (SFP) area is provided by the main auxiliary building supply and exhaust system (see Section 9.4.4). Operation of these systems would be interrupted by a loss of normal power supplies, as the main supply and exhaust fans are not vital to operation of engineered safety features equipment and are not among the loads operable from the emergency diesel power supplies. However, a reduced quantity of air is circulated and exhausted by redundant fans in separate exhaust paths, bypassing the main exhaust fans via the auxiliary building exhaust fan bypass damper. The main auxiliary building exhaust fans can be returned to service upon restoration of the normal power supplies.

9.4.2.3 System Components

9.4.2.3.1 Auxiliary Building Air Handling Unit

The air handling unit consists of an outside air inlet, damper, roughing filters, heating coils, discharge heating coils, and dampers. Associated with the unit, one supply fan 1A supplies air to the new fuel area, general operating floor area, drumming station, and general area. Supply fan 1B supplies air to the spent fuel pool (SFP) area and the decontamination pit (see Drawing 33013-1872).

9.4.2.3.2 Auxiliary Building Exhaust Fan 1C

Exhaust fan 1C draws air from the spent fuel pool (SFP) and decontamination pit areas and discharges through a damper to the plant discharge header (see Drawing 33013-1871).

9.4.2.3.3 Auxiliary Building Exhaust Fans 1A and 1B

Auxiliary building exhaust fans 1A and 1B are the main fans and each has a 100% capacity (see Drawing 33013-1871). The fans discharge to the plant vent stack. Air flows through the high efficiency particulate air filter, pressure alarms, minimum static pressure controller, dampers, exhaust fans, and bypass damper. A fail open bypass damper is installed around the main exhaust fans to ensure a path for charcoal filtered exhaust to the plant vent if main exhaust fans 1A and 1B are not operating. This bypass damper opens if no exhaust fans are operating in order to start an auxiliary building charcoal filter fan. Dampers on the auxiliary building exhaust fans are interlocked with the fan starting circuit.

9.4.2.3.4 Auxiliary Building Exhaust Fan 1G

To assist in the removal of any airborne radioactivity, exhaust fan 1G takes suction from the following areas and discharges through high efficiency particulate air and charcoal filters to the plant discharge header (see Drawing 33013-1870):

A. General intermediate floor area.

B. Gas stripper.

C. Waste gas compressor.

D. Holdup tank rooms.

E. Spent fuel pool (SFP) filter area.

F. Evaporator areas.

G. Basement floor area.

H. Spent resin storage tank.

I. Chemical holdup tank.

J. Containment spray pump area.

K. Nonregenerative heat exchanger area.

L. Seal-water heat exchanger area.

M. Demineralizer and ion exchanger areas.

N. Boric Acid Tank Area

O. Drumming Station Area

P. RWST Vent

Q. CCW Surge Tank Vent

R. General Operating floor area

The charcoal filter associated with auxiliary building exhaust fan 1G is protected by an automatic fire suppression system.

9.4.2.3.5 Auxiliary Building Charcoal Filter Fans 1A and 1B

Charcoal filter fans 1A and 1B include the charcoal filter bank and suction and discharge dampers (interlock) (see Drawing 33013-1870). A low-flow alarm is associated with the fans. The fans are each 50% capacity and take a suction from the following areas:

A. Leakoff collection tank.

B. Residual heat removal pump pit.

C. Residual heat removal heat exchanger pit.

D. Gas decay tank rooms.

E. Waste evaporator vent and air ejector discharge (system physically removed in 1999).

F. Boric acid evaporator air ejector discharge.

G. Charging pump rooms.

H. Concentrates holdup tank and pump rooms.

I. Volume control tank and reactor coolant filter area.

J. Containment mini-purge exhaust.

9.4.2.3.6 Penetration Cooling Fans 1A and 1B

Penetration cooling fans 1A and 1B supply intermediate and auxiliary building penetrations through a cooling coil (see Drawing 33013-1866).

9.4.2.3.7 Pump Area Coolers

Coolers are provided for the residual heat removal, charging, safety injection, and containment spray pumps (see Section 9.4.9.1).

9.4.2.3.8 Intermediate Building Supply and Exhaust Fans

The intermediate building ventilation system includes a supply fan that exhausts air from the intermediate building cleanside to the intermediate building restricted area side. Two additional exhaust fans, which are located in the intermediate building restricted area side, draw ventilation air from various areas of both the clean and restricted area sides of the intermediate building and discharge to the auxiliary building discharge header plant vent duct. Ventilation air is provided to the intermediate building cleanside through louvered, pneumatically controlled outside air intake dampers, which are located in the east wall of the intermediate building. The dampers will open when the outside air temperature reaches approximately 50°F and close when the outside air temperature falls below approximately 50°F. Additional ventilation air capability is available to be drawn into the intermediate building cleanside from the turbine building through a louvered wall opening, which is installed in front of a rolling fire door installed in the fire barrier wall. Additional exhaust air capability is provided to the intermediate building cleanside from four roof ventilators of approximately 78,400 cfm total capacity (see Drawings 33013-1871 and 33013-1872).

An additional exhaust fan is mounted on the existing grating at the west end of the intermediate building cleanside, near column G3 at elevation 278 ft 4 in. The fan is intended to enhance air movement from the intermediate cleanside basement to the building roof exhaust fans.

<u>9.4.2.3.9 Steam Isolation Dampers</u>

Steam isolation dampers are installed to minimize the potential for steam to pass through the wall that divides the clean side of the intermediate building from the restricted area side in the event of the design-basis high-energy line break, although this function is not credited in the Ginna Station current licensing basis, per Section 3.6.2.5.1.2. Three dampers are located at the wall at the points where the following ventilation systems pass through the wall:

A. Auxiliary building exhaust system.

B. Intermediate building exhaust system.

C. Intermediate building supply system.

To provide redundancy against mechanical failure, two isolation dampers are installed back to back in each line. The dampers in all three systems are electrically connected to individual trip and alarm systems with redundant control achieved through the use of electro-thermal type fusible links designed to release the dampers at a maximum of 165°F. (See Drawings 33013-1871 and 33013-1872.)

9.4.2.4 System Evaluation

<u>9.4.2.4.1 Effect of Loss of Cooling on Pumps and Valves</u>

Section 9.4.2.4.1 is retained for historical purposes. The auxiliary building transient heatup results discussed in this section have been superseded by the revised heatup analysis discussed in Section 9.4.2.4.2.

An engineering evaluation of the auxiliary building was conducted to determine the effects of loss of cooling on the operability of safety-related pumps and valves (*Reference 3*). The residual heat removal pumps pit, basement level, intermediate level, and operating level were examined assuming a loss of all ventilation concurrent with a large break loss-of-coolant accident or an emergency cooldown condition (e.g., steam line break in containment). The equipment considered was the safety injection pumps, containment spray pumps, residual heat removal pumps, and their associated valves. The temperature rise in the areas of concern was determined by applying the RHU computer code (*Reference 4*) modified to represent the specific area and building configurations. The heat sources considered were pumps, piping, solar effects, and lighting in the areas. The evaluation concluded that the equipment would be capable of operating in the resultant environment for the durations required to mitigate the accident. Detailed results are provided below.

The basement level east end contains the high-head safety injection and containment spray pumps. The primary heat load to this area comes from the pump motors and piping. During a loss-of-coolant accident, ambient temperatures rise to about 108°F in the first 25 minutes of the event and stabilize at 94°F after 20 hours. Temperatures are slightly lower during an emergency cooldown condition due to less run time of the subject pumps. The pump motors and necessary valves for the containment spray and safety injection pumps are qualified for these environments (*Reference 2*).

Due to the large amount of uninsulated piping and relatively small volume in the residual heat removal pump pit, the ambient temperature rises to 149°F approximately 13 hours into the event during emergency cooldown conditions. This elevated temperature is maintained throughout the event since the residual heat removal system is realigned after the injection phase. During a loss-of-coolant accident, the ambient temperature also rises to 149°F, but not until 72 hours into the event. The contribution of the containment wall as a heat source and the temperature of the sump fluid during the recirculation phase contribute to these conditions. The residual heat removal pump motors are qualified for these environments (*Reference 2*).

The engineering evaluation also considered the effect of operating a single cooling unit with one tube plugged in the residual heat removal pump pit. For this configuration, the ambient temperature would rise to 106°F during emergency cooldown conditions and stabilize at 98°F. Since these temperatures are bounded by the above scenario, the residual heat removal pumps would remain operable during these conditions.

All other areas of the auxiliary building that were evaluated remained below 104°F throughout the accident.

9.4.2.4.2 Revised Auxiliary Building Loss of Cooling Analysis

As a result of the 1997 NRC design inspection (architect/engineer inspection) of the Ginna Nuclear Power Plant (*Reference 6*), the NRC identified three non-conservatisms with the original engineering evaluation (*Reference 3*). The three non-conservative areas identified were:

- Initial auxiliary building ambient temperature less than maximum temperature for normal plant operation.
- Initial refueling water storage tank (RWST) temperature less than the maximum allowable tank temperature.
- Failure to assume a 50 gpm residual heat removal (RHR) pump seal leak twenty-four (24) hours following the design basis loss-of-coolant accident (LOCA).

To address the NRC concerns, the transient heatup of the auxiliary building following a design basis loss-of-coolant accident (LOCA) was reanalyzed with the GOTHIC computer program (*Reference 7*). The GOTHIC program was first benchmarked against the original results (*Reference 3*). This benchmarking indicated that the GOTHIC peak room temperatures were comparable to those calculated by *Reference 3*, with GOTHIC typically calculating slightly higher peak room temperatures. Based upon this benchmark, GOTHIC was then used to investigate the impact of the non-conservatisms identified by the NRC inspection (*Reference 6*) on the auxiliary building transient temperature profiles following a design basis loss-of-coolant accident (LOCA).

The GOTHIC reanalysis (*Reference 8*) of the auxiliary building heatup included a parametric study of the individual impact of the three non-conservatisms on peak room temperatures. Increasing both the initial auxiliary building ambient temperature and the initial refueling water storage tank (RWST) temperature resulted in a 2°F to 5°F increase in peak room temperatures. Assuming a 50 gpm residual heat removal (RHR) pump seal leak 24 hours after the LOCA caused the RHR pump room temperature to increase by 12°F with no noticeable impact on any of the other areas of the auxiliary building.

As a result of the increased peak temperatures calculated by *Reference 8*, the environmental qualification of all safety-related equipment inside the auxiliary building was reviewed. The review determined that all safety-related equipment was still capable of performing their safety-related functions. Typically, the magnitude of the increase in peak temperatures combined with the short duration of the temperature increase resulted in a negligible decrease in equipment qualified life that did not affect operability of the equipment.

9.4.2.4.2.1 AUXILIARY BUILDING TEMPERATURE WITH MINIMUM SERVICE WATER FLOW

In addition to the three NRC concerns described in 9.4.2.4.2, *Reference 8* also evaluated the impact of terminating spent fuel pool (SFP) cooling immediately following a design basis loss-of coolant accident (LOCA) on the auxiliary building peak temperatures.

With SFP cooling isolated, the auxiliary building operating level **may** heat up to 131°F in the first **26** hours following the design basis loss-of-coolant accident (LOCA) due to heat loss from the spent fuel pool (SFP) to the operating level of the auxiliary building. The temperatures of the lower levels of the auxiliary building were unaffected by the termination of cooling to the spent fuel pool (SFP).

Finally, *Reference 8* evaluated the combined effects of the three NRC concerns and the termination of spent fuel pool (SFP) cooling on the auxiliary building peak temperatures following a loss-of-coolant accident (LOCA). The operating level peak temperature was calculated to be 131°F due to convective heat transfer from the spent fuel pool to the auxiliary building operating level. The residual heat removal (RHR) pump room peak temperature was determined to be 166°F due to the 50 gpm residual heat removal (RHR) pump seal leak. All of the other areas of the auxiliary building had peak temperatures of approximately 110°F or less.
The peak auxiliary building temperatures are summarized in UFSAR Table 3.11-1.

9.4.2.4.3 Effect of Loss of Offsite Power on Ventilation Flow

The above engineering evaluation also examined the ventilation flow in the auxiliary building and intermediate building to determine if on a loss of offsite power, the flow would be reversed such that the direction of flow of airborne radioactivity would not be from areas of low activity toward areas of higher activity. It was determined that in all cases analyzed for loss of offsite power and for ventilation isolation from a high radiation signal there would be no backflow through any path from areas of higher radiation to areas of lower radiation.

9.4.3 CONTROL ROOM AREA VENTILATION SYSTEM

The function of the control room area ventilation system is to provide a controlled environment for the safety and comfort of control room personnel and to ensure the operability of control room components during normal operating, anticipated operational transient, and design-basis accident conditions. The control room area ventilation system design is shown in Drawing 33013-1867 and is discussed in Section 6.4.

9.4.4 SPENT FUEL POOL AREA VENTILATION SYSTEM

The spent fuel pool (SFP) area ventilation system is a part of the auxiliary building ventilation system (ABVS) shown in Drawing 33013-1871. The system serves to control airborne radioactivity in the spent fuel pool (SFP) area during normal operating conditions. This is accomplished by directing air from within the auxiliary building across both the spent fuel pool (SFP) and the decontamination pit to exhaust air ducts which are connected to the suction of the auxiliary building exhaust fan C. Exhaust air from the spent fuel pool (SFP) water surface is drawn through roughing filters and, depending on system alignment, charcoal filters. Discharge from the auxiliary building exhaust fan C passes through HEPA filters, a main auxiliary building exhaust fan, and then out the plant vent.

The original design flow rate for the SFP charcoal filters was 20,000 cfm. However, system operability for Technical Specifications is based upon limiting degradation of air flow from the new condition plus maintaining a negative pressure in the auxiliary building. The actual analysis of a fuel handling accident is independent of system air flow and is discussed in Section 15.7.

The system is also credited for control of airborne radioactivity in the SFP area during anticipated operational transient. However, the system is not designed for consideration of loss of offsite power or other single failures.

9.4.5 TURBINE BUILDING VENTILATION SYSTEM

The turbine building, while not requiring a heating, ventilation, and air conditioning system, uses roof vent fans, wall vent fans, windows, and unit heaters for ventilation and temperature control. The system is shown in Drawings 33013-1873 and 33013-1874. The fans are not supplied by emergency (diesel generated) power, and loss of these fans would not be critical to a safe shutdown.

In the turbine building, the main feedwater pump room and feedwater pump equipment cooling systems use a mixture of outside air and room air to control the room and equipment

temperatures. No mechanical means of heating or cooling is used. A temperature control system controls the feedwater pump room return air dampers and equipment outside air dampers that admit air to the equipment air supply fan plenum mixed at a setpoint temperature. The equipment cooling air discharges into the feedwater pump room above the feedwater pump motors in the general vicinity of the motor air intakes. A mixture of outside air and room air enters the feedwater pump motor enclosures, providing the required motor cooling. The room temperature control system controls separate outside air inlet dampers to the feedwater pump room and two feedwater pump room exhaust fans to control feedwater pump room temperature. The feedwater pump room supply fan outside air dampers and room exhaust air dampers fail open to provide cooling on loss of instrument air to the temperature control system.
The room return air dampers and the feedwater pump room outside air inlet dampers fail closed on loss of instrument air to the temperature control system. Loss of ac power to the equipment supply and room exhaust fans would result in loss of cooling. High feedwater pump winding temperature is alarmed in the control room.

9.4.6 SERVICE BUILDING VENTILATION SYSTEM

The service building ventilation system, shown in Drawings 33013-1875 through 33013-1879 and 33013-1881, consists of six air handling units serving the various areas of the service building. Air from uncontaminated areas is exhausted through roof exhaust fans. Air from areas of potential contamination, such as laboratories equipped with hoods, are exhausted through the controlled intermediate building controlled access area exhaust fans.

The kitchen hood suppression system in the cafeteria will exhaust smoke and wet chemical agent upon system activation. This is supported by the service building air handling units.

Controlled Access Area Fans 1A and 1B

Controlled access area fans 1A and 1B include high efficiency particulate air and charcoal filter banks, a low-flow alarm, dampers, and fans (see Drawing 33013-1875). These fans take suction from the following areas and discharge to the Auxiliary Building HEPA filter vent which is exhausted by the main Auxiliary Building exhaust system to the main vent header:

A. Men’s and women’s decontamination general areas.

B. Radiation protection and chemistry office general area.

C. Primary sample room general area.

D. Primary sample hood.

E. Primary and secondary sample lab hoods.

F. Hot shop general areas.

9.4.7 ALL-VOLATILE-TREATMENT BUILDING VENTILATION SYSTEM

9.4.7.1 Introduction

This system provides ventilation and heating to maintain required temperatures for the all-volatile-treatment (condensate demineralizer) building and the condensate booster pump area of the turbine building. The system is designed for the following temperature conditions (see Drawing 33013-1874):

A. Outside design temperature.

   1. Summer 95°F, dry bulb, 75°F, wet bulb.
   2. Winter -5°F, dry bulb.

B. Inside design conditions.

   1. Maximum 104°F (40°C), dry bulb.
   2. Minimum 40°F.

9.4.7.2 Summary Description of the System

9.4.7.2.1 Compressor and Booster Pump Area Ventilation System

Ventilation is provided to the compressor and booster pump area by two 50% capacity fans supplying outside air. One fan is thermostatically started upon a rise in room temperature. Temperature control of the area is accomplished by modulation of the outside air intake damper and the recirculation damper. The room thermostat has a proportional output signal which is fed to a low-pressure selector. The output of the pressure selector modulates the outside and return air damper operators. When the room temperature rises above the thermostat setting, the outside air damper opens and the recirculation damper closes to a position which is proportional to the temperature deviation between the actual room temperature and the setpoint. The cooler outside air will then mix with the recirculation flow and lower the temperature in the room. During winter conditions, cold outside air could bring the duct air temperature down to undesirable low temperatures. The duct air temperature is therefore measured by a pneumatic pressure transmitter. The signal from the transmitter is fed to a controller with an internal adjustable setpoint. The output signal from the controller is the second input to the low-pressure selector. For low duct air temperatures, the signal from the room thermostat will be blocked in the low-pressure selector and the pressure signal from the controller will modulate the damper operator to maintain a duct air temperature of 60°F.

The two 50% capacity supply fans are automatically controlled by individual room thermostats. The first fan is started at a preset room temperature by the same thermostat which was utilized for temperature control of the intake air dampers. The output signal from the thermostats actuates a pressure switch which starts the fan. Should the temperature continue to rise, the second room thermostat will actuate its pressure switch, which will start the second fan.

A pneumatic switch has been provided by which each thermostat can be connected to control any of the two fans.

A temperature switch is located in the discharge duct for each fan. This switch will trip the fan on high temperature.

9.4.7.2.2 Demineralizer Area Ventilation System

Two supply fans of 50% capacity and four exhaust fans of 25% capacity ventilate this area. The function and modulation of air supply fans are as described for the compressor and booster pump area. Two of the four exhaust fans are switched into operation when the first supply fan starts. The other two exhaust fans start when the second supply fan is started.

An electric heating coil is capable of heating the released air of the high and low conductivity waste tanks above its saturation limit. The heater could be put into operation as soon as the air blower starts. The coil is deenergized when the air blower stops. The discharge from the common header is brought to one of the exhaust openings of the all-volatile-treatment exhaust system. The heater unit is classified as a non-nuclear safety system.

9.4.7.2.3 Demineralizer Area Control Room System

One system of 100% capacity ventilates and heats this room. The system operates continuously. As the outside temperature drops below a predetermined level, the outside air damper closes and the unit goes on recirculation. As the room temperature drops below the thermostat setting, a steam heating coil provides heat for the room. The roughing filter in this system prevents excessive dirt or insects from being drawn into the control room. Two temperature switches are located in the fan discharge duct. One switch will trip the fan on high temperature while the other switch trips the fan on low temperature.

9.4.7.2.4 Heating System

Steam and electric unit heaters supply heat to the demineralizer area control room and to the demineralizer area as required. The electric unit heaters in the demineralizer area and the control room are standby heaters used in the event of malfunction of the steam heaters.

9.4.8 TECHNICAL SUPPORT CENTER VENTILATION SYSTEM

9.4.8.1 System Description

The technical support center heating, ventilation, and air conditioning system consists of the following subsystems in the technical support center (Drawing 33013-1256):

A. Central heating, ventilation, air conditioning, and charcoal filter system.

B. Office air conditioning system.

C. Mechanical equipment room cooling system.

D. Diesel generator room cooling system.

E. Battery room cooling system.

F. Uninterruptible power system room cooling system.

G. Computer room air conditioning system.

H. Toilet and kitchen exhaust system.

I. Battery, diesel, and corridor heating system.

J. Mechanical equipment room heating system.

In addition to maintaining year-round occupancy comfort levels, the heating, ventilation, and air conditioning system provides personnel protection from airborne radiological contaminants, maintains a positive pressure in the emergency mode relative to the outside, and provides cooling, heating, and ventilation required by special areas.

The central heating, ventilation, air conditioning, and charcoal filter system serves the occupied areas of the building by providing three modes of operation: normal with mechanical cooling or cool outside air, normal with steam heating, and emergency with mechanical cooling, steam heating, and charcoal filtering. A flow controller is used to throttle the charcoal filter inlet dampers to ensure that the maximum design rate of 3000 cfm is not exceeded. The main air handler fan can move up to 9300 cfm of air. A relief damper opens under the control of a differential pressure controller to maintain a 1/8-inch water gauge (0.125 in. wg ± 10%) differential pressure in the conditioned area when in the emergency mode.

The administrative computer room office is air conditioned by the technical support center central system. The administrative computer room is cooled by a packaged service-water-cooled air conditioning unit. The function of the packaged unit is to provide the proper computer room environment for the administrative computer system.

Cooling systems for the mechanical equipment room, diesel generator room, battery room, and uninterruptible power system room induce outside air for cooling by means of exhaust fans.

The plant process computer system (PPCS) and safety parameter display system (SPDS) computer room is cooled by two packaged air conditioning units. These units are the air-cooled type with the condensers placed outside on the roof. Each unit is sized for 100% of the total room heat gain. The technical support center central heating, ventilation, and air conditioning system is also ducted to the computer room and provides backup in the event of failure of the packaged units. The function of the packaged units is to provide the proper computer room environment for the plant process computer system and safety parameter display system computers.

The kitchen and toilet rooms are exhausted by a fan inducing air from the occupied areas.

The battery room, diesel room, and corridor are provided with warm air from the uninterruptible power system room by separate circulating fans.

The mechanical equipment room is heated with a steam unit heater.

9.4.8.2 System Operation

<u>9.4.8.2.1 Cooling Systems</u>

A thermostat sensing outside air temperature above or below 60°F (adjustable) indexes operation to mechanical or outside air cooling.

When the system is indexed to mechanical cooling, the outside air damper and the recirculated air damper open to a pre-established setting to admit approximately 600 cfm of outside air. A thermostat sensing mixed air temperature maintains the mixed air temperature by modulating (loading/unloading and cycling) the air conditioning compressor capacity. Zone thermostats maintain local space temperature by varying air volume.

When the system is indexed to outside air cooling, the outside air damper and the recirculated air damper are modulated by a thermostat sensing mixed air temperature. Zone thermostats maintain local space temperature by varying air volume. The amount of outside air admitted does not fall below 400 cfm.

The diesel generator room exhaust fan and air intake dampers are controlled by a room thermostat when the diesel is not operating. When the diesel is operating, the room exhaust fan is off, but both air intake dampers are open and the diesel generator skid exhaust fan operates.

The mechanical equipment room exhaust fans and air intake damper are controlled by a two-step thermostat. The uninterruptible power source room exhaust fan and air intake damper are controlled by a room thermostat. The battery room exhaust fan and air intake damper are controlled by a room thermostat.

9.4.8.2.2 Heating Systems

The air conditioning and charcoal filtering system heating is provided by steam being admitted to the air handling unit heating coil. Space heating thermostats control the space temperature by modulating the air handling unit dampers and variable air volume boxes as required by the space conditions.

Additional heating capabilities are provided to several perimeter office areas by electric reheat coils that are controlled by individual space heating/cooling thermostats.

The battery room heating fan operates when the room temperature falls below the setpoint of the room heating thermostat. A receptacle has been provided in the technical support center south corridor for connection of a portable electric heater if needed to provide heating capabilities.

The diesel generator room heating fan operates when the room temperature falls below the setpoint of the room heating thermostat. The room is also provided with space electric heaters, which operate as needed.

The corridor heating fan operates when the corridor temperature falls below the setpoint of the corridor heating thermostat.

The mechanical equipment room unit heater fan operates whenever the room temperature falls below the setpoint of the room heating thermostat.

The toilet and kitchen exhaust fan operates continuously, except during an emergency operating mode.

9.4.9 ENGINEERED SAFETY FEATURES VENTILATION SYSTEMS

The engineered safety features ventilation systems include those ventilating and cooling systems that service equipment required either following an accident or to ensure safe plant shut-down. Equipment and/or areas serviced by these ventilating and cooling systems include the following:

A. Engineered safety features equipment.

B. Relay room.

C. Battery rooms.

D. Essential auxiliary systems.

E. Diesel generator rooms.

F. Standby auxiliary feedwater system.

G. Post-accident fan coolers and charcoal filter system.

9.4.9.1 Engineered Safety Features Equipment Ventilation and Cooling

Safety Injection System

The safety injection system includes the high-pressure safety injection pumps located in the basement level of the auxiliary building and the residual heat removal pumps located in the subbasement level of the auxiliary building. The safety injection pump drive motors are cooled by redundant, stand-alone air cooling units which are shared with the containment spray pumps and from which the cooling air is ducted to a point adjacent to the cooling intake vents of each drive motor. The residual heat removal pump room coolers are also redundant stand-alone air cooling units. The cooling units are comprised of a water-cooled heat exchanger and a blower. Service water (SW) is the cooling medium. The coolers are designed to Seismic Category I criteria. (See Drawing 33013-1869.) However, none of these air cooling units are required for operation of the pumps (*Reference 2*), and the coils for the safety injection pumps have been blanked off since 1992 such that the associated fans are operable but the air is no longer cooled by service water (SW).

Containment Spray System

Cooling of the containment spray pump drive motors on the basement level of the auxiliary building is by the redundant cooled air-cooling units shared with the safety injection pump motors and described above. The cooling air is ducted to a point adjacent to the cooling intake vents of the two drive motors, in a similar manner, as for the safety injection pump motors. (See Drawing 33013-1869.) These air cooling units are not required for operation of the containment spray pumps (*Reference 2*), and the coils for the containment spray pumps have been blanked off since 1992 such that the associated fans are operable but the air is no longer cooled by service water (SW).

9.4.9.2 Relay Room

The relay room contains two self-contained, water-cooled air-cooling units that maintain a low normal room temperature during MODES 1 and 2 (see Drawing 33013-1868).

9.4.9.3 Battery Rooms

Ventilation of the battery rooms is provided by two propeller exhaust fans that take suction from the battery rooms and discharge to a common exhaust duct. (See Drawing 33013-1868.) The battery room air conditioning unit includes a refrigeration unit, an electric heating coil, and a fan. The unit provides approximately 2900 cfm of supply air to the battery rooms; of which approximately 2500 cfm is return air from the battery room propeller fans and 400 cfm is makeup air taken from the air handling room. Exhaust air from the control room HVAC system is released into the control building air handling room and fresh outside air can enter the room through a separate outside air intake duct. The system provides sufficient makeup air and return air to maintain hydrogen in the battery rooms below the lower flammability limit.

Ideal battery room temperature is between 75 and 77°F to allow optimum battery life. Sizing of the batteries was based upon an assumed electrolyte temperature equal to the 55°F Technical Specifications minimum temperature for operability. Worst case battery temperature calculations that assume a 71°F room temperature at the start of a station blackout predict battery temperatures of 68°F after 4 hours and 65°F after 8 hours. The worst case hydrogen generation calculation assumes 104°F as the maximum battery temperature.

In the event of failure of the main air handling unit, air flow switches will actuate the two normally closed dampers in the battery rooms to provide a flow path from the air handling room through the battery rooms by way of the battery room propeller fans. In the event of loss of ac power to the battery room ventilation system, air flow switches installed downstream of each battery room propeller fan would annunciate in the control room on low air flow in either battery room. A manually actuated backup fan, powered by the dc batteries, supplies air from the air handling room to both battery rooms by the opening of two normally closed volume dampers installed in the battery room block walls. Automatic actuation of the dc-powered backup fan is not required since hydrogen buildup from battery charger operation would not be excessive or immediate. If the battery chargers are not in operation, such as during a station blackout, hydrogen generation would not occur.

An analysis was performed with no credit taken for the dc-powered fan and with the ac-powered propeller fans and air conditioning unit not operating to determine the maximum temperatures in the battery rooms. The results showed that the environmental service conditions for the battery rooms in Table 3.11-1 would not be exceeded after 5 hours, assuming the initial temperature had been 77°F. The unacceptable hydrogen concentration level of 2% would not be exceeded until after 73.3 hours, with both batteries being equalized, which allows sufficient time to manually start the backup fan after receipt of the battery room loss of ventilation alarm.

9.4.9.4 Essential Auxiliary Systems

The following essential auxiliary systems require ventilation and/or cooling.

Chemical and Volume Control System

In addition to general cooling by ambient air, the charging pump room is cooled by redundant fan-driven air coolers using service water (SW) as the cooling medium. To ensure that the pumps and drive motors are cooled, the cooled air is ducted directly to the room from the coolers. (See Drawing 33013-1869.)

The emergency diesel generators automatically supply the cooling fans by separate motor control centers in the case of loss of normal power. This places one cooler on diesel 1A and the other on diesel 1B for redundancy. The capacity of each cooling unit is sufficient to maintain acceptable operating temperatures.

The cooling units are designated safety-related, nonseismic by current Regulatory Guide 1.29 classification and Seismic Class II in accordance with the original design criterion for Ginna Station (see Section 3.7.1.1). The cooling units are nonseismic consistent with the post-accident safety function of the charging system. Should failure of both charging pump room coolers occur due to a fire, high energy line break, or seismic event, other means exist to maintain the ability to inject borated water into the reactor coolant system. These include use of the high-pressure safety injection pumps or by establishing supplemental cooling to the charging pump room to maintain the ambient temperature within allowable limits for operability of the charging pumps.

Component Cooling Water System (CCW)

The primary ventilation and cooling requirements of the component cooling water (CCW) system are associated with its circulation pump motors. These motors are located on the main operating floor of the auxiliary building where cooling is provided by the ambient air of the operating floor. This air is provided by the auxiliary building supply air handling unit described in Section 9.4.2.3. However, ambient temperatures would not exceed the capabilities of the component cooling water (CCW) pumps even if the auxiliary building air handling unit were inoperable.

9.4.9.5 Diesel Generators

The diesel generators (with associated electrical switchgear) are housed in adjacent but separate rooms, each serviced by a ventilation system. Each room is ventilated by two inlet fans supplying outside air. Each fan takes suction from a common header and discharges through separate ductwork, dampers, and discharge diffusers. One fan in each room discharges a supply of air directly on the instrument and control cabinets. Excess air is discharged to the outdoors through automatic, pressure-actuated room vents, backdraft dampers, and wall-mounted louvers. No refrigeration or service water (SW) air cooling is used. (See Drawing 33013-1873.)

When the diesels are not running, the damper in each line is held closed by 20-psig instrument air by a solenoid valve associated with each damper. A diesel generator start signal automatically opens the dampers and starts one of the two room supply fans with normal operating room temperatures. The second supply fan starts when the diesel is running and the room temperature reaches the thermostat setpoint. The fans in the diesel generator room

A are powered from a motor control center within the room that is powered from diesel generator 1A via bus 14. The fans in diesel generator room B are powered from a motor control center within the room that is powered from diesel generator 1B via bus 16. Power to actuate the solenoid valves is provided by the 125-V dc system through breakers in each room. Once opened, the dampers remain open until the diesel generator is deenergized.

In the event of loss of instrument air to the solenoid valves, the air pressure in the line causes the damper motor to retract and open the damper. Should a diesel generator start signal be received, the dampers would be in their safe position (open) to provide the needed ventilation. The dampers would remain in the open position until air pressure is returned to service and the diesel generator is deenergized. Adequate ventilation for equipment cooling and for removal of any hydrocarbon gases is therefore ensured when the diesel generators are operating.

The supply fan that directs air near the engine jacket water-sensing line in each room is equipped with a thermostat that delays starting of the fan until sufficient heat is rejected by the diesel engine to prevent freezing of the line in cold weather. Additional protection is provided by a second set of thermostat switches which will turn off the automatic supply fan on DG room low temperature. This system is designed so that no failure of a single active component will prevent operation of the ventilation system such that affected components would exceed their design limits or would cause the loss of both diesel generators.

Heating of each room is by unit steam heaters (two per room). Should one of the steam supply lines break, only its respective diesel-generator room would be affected. The other diesel generator room would be isolated from the break.

Humidity in the lower level (vault-area) of each diesel-generator room is kept low by a packaged dehumidifier, which is controlled by an internal humidistat. The dehumidifiers are nonseismic, are not required to maintain functional integrity during a seismic event, and are located so that they will not adversely affect safety-related systems or components. Vault humidity is monitored regularly, and equipment inside the vault areas is part of the routine maintenance program.

9.4.9.6 Standby Auxiliary Feedwater System (SAFW)

<u>9.4.9.6.1 System Operation</u>

The standby auxiliary feedwater pump (SAFW) room cooling and heating system provides cooling and heating as required to maintain the pump room temperature within the design temperature range of 60°F (minimum) to 120°F (maximum). The standby auxiliary feedwater pump (SAFW) room is part of the auxiliary building addition. This cooling and heating system is needed to provide an acceptable environment for the equipment in the pump room, which includes the two standby auxiliary feedwater pumps (SAFW) and their electric drive motors (see Drawing 33013-1869).

The standby auxiliary feedwater pump (SAFW) room cooling system is capable of operation whenever the standby auxiliary feedwater pumps (SAFW) are needed for operation. This is a result of the fact that the cooling system provides the air cooling required for continuous operation of the pump motors. A given cooling unit is automatically started whenever its corresponding standby auxiliary feedwater pump (SAFW) is started. Due to its safety-related nature, the cooling system must remain functional during all modes of plant operation including the period during and after a safe shutdown earthquake. Thus, the two fan drive motors for the standby auxiliary feedwater pump (SAFW) room cooling units are supplied from separate, redundant Class 1E electrical systems.

The standby auxiliary feedwater pump (SAFW) room heating system operates whenever the temperature in the pump room falls below the thermostat setting of 60°F to 65°F of the unit heater and the unit heater's source of non-safety-related electric power is available. Plant operating conditions have no effect on the heating system unless non-safety-related electric power is unavailable. The heating system is not safety-related or seismically classed since its function is not required for proper operation of the standby auxiliary feedwater system. In case of a heating system failure during subfreezing weather, the water in the feedwater system can be prevented from freezing by using a portable heater or by running one of the standby auxiliary feedwater pumps (SAFW) on recirculation with the DI water storage tank, thus warming the pump room with the pump motor heat.

9.4.9.6.2 Controls and Instrumentation

The start/stop controls for the standby auxiliary feedwater pump (SAFW) room cooling units are manually and/or automatically operated. Each cooling unit is arranged so that it automatically starts or stops at the same time that its corresponding standby auxiliary feedwater pump (SAFW) is manually started or stopped (from either the control room or the local station). In addition, the cooling units can be started/stopped independently of the standby auxiliary feedwater pumps (SAFW) from a local control panel in the pump room. This manual/automatic control is determined by a maintained contact three position control switch provided in the local control panel. This switch has run-auto-off positions. It must be manually returned to the auto (or pump control) position whenever local control is no longer being utilized.

Flow of cooling water (service water (SW)) through the cooling coil of each cooling unit is controlled by an open-closed two-way valve in the discharge line from the coil. When a cooling unit is started automatically via the standby auxiliary feedwater pump (SAFW) control, the cooling water control valve for that cooling unit opens fully and stays open until the unit is shut down, at which time it fully closes. However, when the cooling unit is started locally (manual), operation of the cooling water control valve is via a temperature indicating switch that is arranged to sense and indicate the temperature of the return air to the unit. The temperature switch maintains the return air temperature (pump room temperature) within a band between 70°F and 120°F by causing the cooling water control valve to open or close as required. The control valves are fail safe, since they fail open. The control valves close completely when their particular cooling unit is not running, thus stopping service water (SW) flow through the cooling coil.

The roughing filter bank of each cooling unit has installed ports for the connection of a differential pressure indicator for local pressure drop readout across each filter bed. A temperature indicator is installed to give local indication of the air temperature entering each cooling unit.

A common alarm function is provided if space temperature exceeds the high temperature setpoint or the low temperature setpoint.

A locally indicating temperature indicating switch is located in the standby auxiliary feedwater pump (SAFW) room. It alarms the control room upon detection of excessively high or low room temperatures. An air flow switch is located in the discharge duct of each cooling unit. They alarm the control room in case a loss of air flow is sensed. They are interconnected with the controls of the cooling unit fan motors so that the alarm for a loss of air flow is blocked when a cooling unit is intentionally stopped. All of these alarms are grouped to give a single control room malfunction alarm.

The controls for the standby auxiliary feedwater pump (SAFW) room heating system are completely self-contained within each of the system's two electric unit heaters. Each unit heater operates automatically via its built-in thermostat that starts and stops the heater fan motor and energizes and deenergizes the heater electric heating coil. The unit heater thermostats are set so that when the pump room temperature decreases to approximately 65°F, the first unit heater will automatically start. Should the temperature continue to decrease, the second unit heater will automatically start when the pump room temperature drops to 60°F. The only alarm associated with the heating system is low pump room temperature alarm.

9.4.9.7 Post-accident Fan Coolers and Charcoal Filters

This system is described in Sections 9.4.1.2.7 and 6.2.2.

9.4.10 STATION HEATING STEAM SYSTEM

The Ginna heating and process steam is provided from the house boiler, located in the screen house and from a connection from the main steam system. The systems provided with steam from this system include the unit heaters in the screen house, intermediate building, auxiliary building, turbine building, diesel generator rooms, auxiliary building air handling units, containment purge supply unit, boric acid batch tank, gas stripper, and the boron recycle evaporator.

REFERENCES FOR SECTION 9.4

1. Catalytic, Inc., Mild Environment Temperature Transient Response for R. E. Ginna Nuclear Power Plant Auxiliary, Control, and Diesel Generator Buildings (EWR 3304), dated July 1982.

2. RG&E Design Analysis NSL-4529-DA035, Auxiliary Building Ventilation Systems (ABVS), Revision 0, May 29, 1991.

3. Devonrue, Engineering Evaluation of R. E. Ginna Nuclear Power Plant Ventilation System, Revision 1, November 1989.

4. Devonrue Computer Verification for RHU Computer Code, dated January 1988.

5. Devonrue, R. E. Ginna Standby Auxiliary Feedwater Building Thermal Environment Study, Revision 1, dated January 1991.

6. NRC Inspection Report 50-244/97-201, R.E. Ginna Nuclear Power Plant Design Inspection, dated September 24, 1997.

7. GOTHIC Containment Analysis Package, Version 5.0, prepared for EPRI, RP 444-1, dated December 1995.

8. ALTRAN Technical Report 99-124-TR-001, Ginna Nuclear Plant - Gothic Model of Heatup Transient in the Auxiliary Building, Revision 0, dated October 1999.

9. Rochester Gas and Electric Corporation Safety Evaluation SEV-1147, “85°F Maximum Lake Temperature”, Revision 0.

10. Rochester Gas and Electric Corporation Plant Change Request (PCR) 2001-0042.

11. Westinghouse Electric Company, R. E. Ginna, Head Assembly Upgrade Package, General Assembly Drawing 6469E26.

12. 5059EVAL-2014-0001, “ECP-13-000048 – Containment Air Temperature Increase and Associated Changes,” Revision 000.

Table 9.4-1
CONTAINMENT VENTILATION SYSTEM PRINCIPAL COMPONENT DATA SUMMARY

System	***Units Installed***	***Unit Capacity***	***Units Required for Normal operation (MODES 1 and 2)***
Containment recirculating			
Cooling coils - normal	4	2.05×10^6 Btu/hr	3
Cooling coils - design-basis accident	4	54.6×10^6 Btu/hr	3
Demister	4	42,000 cfm	3
Filters, 40 high efficiency particulate air filter cells per unit	4	40,000 cfm	3
Fans	4	46,500 cfm	3
Fan pressure - normal	---	8.3 in. H_2O	---
Fan pressure - design-basis accident (286°F)	---	32.0 in. H_2O	---
Fan motors (440 V, three-phase)	4	300 hp	3
Control rod drive cooling			
Fans, standard conditions	2	14,000 cfm	1
Fan pressure	---	18.6 in. H_2O	---
Fan motors	2	60 hp	1
Reactor compartment cooling			
Plenum	1	-	1
Fans, standard conditions	2	21,175 cfm	1
Fan pressure	---	5 in. H_2O	---
Fan motors	2	30 hp	1
Cooling coils	2	342,000 Btu/hr	2
Containment Shutdown purge supply			
Fans, standard conditions	2	12,220 cfm	---
Fan pressure	---	4 in. H_2O	---
Fan motors	2	15 hp	---
Preheat coils	4	-	---

System	Units Installed	Unit Capacity	Units Required for Normal operation (MODES 1 and 2)
Reheat coils	2	-	---
Air filters, roughing	1	25,000 cfm	---
Containment Shutdown purge exhaust			
Fans, standard conditions	2	12,600 cfm	---
Fan pressure	---	6 in. H_2O	---
Fan motors	2	20 hp	---
Plenums	2	13,000 cfm	---
Filters, 12 high efficiency particulate air filter cells per unit	2	12,000 cfm	---
Charcoal filters, 40 cells per unit	2	13,200 cfm	---
Containment Mini purge supply			
Fan, standard conditions	1	2000 cfm	1
Fan pressure	---	47.5 in H_2O	1
Fan motor	1	30 hp	
Containment Mini purge exhaust[a]			
Refueling water surface, supply			
Fan, standard conditions	1	6900 cfm	---
Fan pressure	---	1 in. H_2O	---
Fan motor	1	3 hp	---
Refueling water surface, exhaust			
Fan, standard conditions	1	11,000 cfm	---
Fan pressure	---	2.2 in. H_2O	---
Fan motor	1	7.5 hp	---
Containment auxiliary charcoal filter			
Fans, standard conditions	2	5100 cfm	Optional
Fan pressure	---	3.5 in. H_2O	---
Fan motors	2	5 hp	Optional

System	Units Installed	Unit Capacity	Units Required for Normal operation (MODES 1 and 2)
Filters, six high efficiency particulate air filter cells per unit	2	6000 cfm	Optional
Charcoal filters, 16 cells per unit	2	5280 cfm	Optional
Containment postaccident charcoal filter			
Charcoal filters 120 cells per unit	2	38,000 cfm[b]	---
Steam Heating			
Heaters, 2-psig steam, 60°F	2	94,600 Btu/hr	---

a. The containment mini purge system exhausts through the auxiliary building ventilation system (ABVS).

b. Represents original rating. See Section 6.5.1.2.1 for the latest rating figures.

9.5 OTHER AUXILIARY SYSTEMS

9.5.1 FIRE PROTECTION

The fire protection program is based on the NRC requirements and guidelines, Nuclear Electric Insurance Limited (NEIL) Property Loss Prevention Standards and related industry standards. With regard to NRC criteria, the fire protection program meets the requirements of 10 CFR 50.48(c), which endorses, with exceptions, the National Fire Protection Association's (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants – 2001 Edition." R.E. Ginna has further used the guidance of NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)," as endorsed by Regulatory Guide 1.205, "Risk-Informed, Performance Fire Protection for Existing Light-Water Nuclear Power Plants."

Adoption of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", 2001 Edition in accordance with 10 CFR 50.48(c) serves as the method of satisfying 10 CFR 50.48(a) and General Design Criterion 3. Prior to adoption of NFPA 805, General Design Criterion 3, "Fire Protection" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," was followed in the design of safety and non-safety related structures, systems, and components, as required by 10 CFR 50.48(a).

NFPA 805 does not supersede the requirements of GDC 3, 10 CFR 50.48(a), or 10 CFR 50.48(f). Those regulatory requirements continue to apply. However, under NFPA 805, the means by which GDC 3 or 10 CFR 50.48(a) requirements are met may be different than under 10 CFR 50.48(b). Specifically, whereas GDC 3 refers to SSCs important to safety, NFPA 805 identifies fire protection systems and features required to meet the Chapter 1 performance criteria through the methodology in Chapter 4 of NFPA 805. Also, under NFPA 805, the 10 CFR 50.48(a)(2)(iii) requirement to limit fire damage to SSCs important to safety so that the capability to safely shut down the plant is satisfied by meeting the performance criteria in Section 1.5.1 of NFPA 805.

A Safety Evaluation was issued on November 23, 2015 by the NRC, that transitioned the existing fire protection program to a risk-informed, performance-based program based on NFPA 805, in accordance with 10 CFR 50.48(c).

EPM-FPPR, R. E. Ginna Nuclear Power Plant Fire Protection Program, (*Reference 17*) describes the Station's fire protection design/features and how the Station meets NRC fire protection program requirements. It is incorporated by reference in its entirety into the UFSAR.

9.5.1.1 Design Basis Summary

9.5.1.1.1 Defense-in-Depth

The fire protection program is focused on protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations. The fire protection program is based on the concept of defense-in-depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:

(1) Preventing fires from starting,

(2) Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage,

(3) Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

9.5.1.1.2 NFPA 805 Performance Criteria

The design basis for the fire protection program is based on the following nuclear safety and radiological release performance criteria contained in Section 1.5 of NFPA 805:

- Nuclear Safety Performance Criteria. Fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria shall be met.
  - (a) Reactivity Control. Reactivity control shall be capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity inserting shall occur rapidly enough such that fuel design limits are not exceeded.
  - (b) Inventory and Pressure Control. With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of controlling coolant level such that subcooling is maintained such that fuel clad damage as a result of a fire is prevented for a PWR.
  - (c) Decay Heat Removal. Decay heat removal shall be capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition.
  - (d) Vital Auxiliaries. Vital auxiliaries shall be capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.
  - (e) Process Monitoring. Process monitoring shall be capable of providing the necessary indication to assure the criteria addressed in (a) through (d) have been achieved and are being maintained.

- Radioactive Release Performance Criteria. Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, Limits.

Chapter 2 of NFPA 805 establishes the process for demonstrating compliance with NFPA 805.

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features.

Chapter 4 of NFPA 805 establishes the methodology to determine the fire protection systems and features required to achieve the nuclear safety performance criteria outlined above. The methodology shall be permitted to be either deterministic or performance-based. Deterministic requirements shall be “deemed to satisfy” the performance criteria, defense-in-depth, and safety margin and require no further engineering analysis. Once a determination has been made that a fire protection system or feature is required to achieve

the nuclear safety performance criteria of Section 1.5, its design and qualification shall meet the applicable requirement of Chapter 3.

9.5.1.1.3 Codes of Record

The codes, standards and guidelines used for the design and installation of plant fire protection systems are as follows: (for specific applications and evaluations of codes, refer to the Fire Protection Program Report (*Reference 17*)):

a) American National Standards Institute (ANSI)
    - ANSI B31.1 – Power Piping
b) American Society for Testing Materials (ASTM)
    - ASTM E-119
c) Factory Mutual (FM) Research Fire Protection Equipment Approval Guide
d) National Fire Protection Association (NFPA)
    - NFPA 12A – 1985 Ed., Standard on Halon 1301 Fire Extinguishing Systems
    - NFPA 13 – 1985 Ed., Standard for the Installation of Sprinkler Systems
    - NFPA 14 – 1983 Ed., Standpipe and Hose Systems
    - NFPA 15 – 1985 Ed., Standard for Water Spray Fixed Systems for Fire Protection
    - NFPA 20 – 1983 Ed., Standard for the Installation of Stationary Pumps for Fire Protection
    - NFPA 24 – 1984 Ed., Standard for the Installation of Private Fire Service Mains and Their Appurtenances
    - NFPA 26 – Supervision of Valves Controlling Water Supplies for Fire Protection, 1983 Ed.
    - NFPA 27 – 1981 Ed., Private Fire Brigades
    - NFPA 30 – 2000 Ed., Flammable and Combustible Liquids Code
    - NFPA 50A – 1973/1978 Ed., Standard for Gaseous Hydrogen at Consumer Sites
    - NFPA 72D – 1986 Ed., Standard for Proprietary Protective Signaling Systems
    - NFPA 72E – 1984 Ed., Standard for Automatic Fire Detectors
    - NFPA 600 – 2000 Ed., Standard on Industrial Fire Brigades
    - NFPA 805 – 2001 Ed., Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants

9.5.1.1.4 Required Systems

Nuclear Safety Capability Systems, Equipment, and Cables
Section 2.4.2 of NFPA 805 defines the methodology for performing the nuclear safety capability assessment. The systems equipment and cables required for the nuclear safety capability assessment are contained in the Fire Protection Program Report (*Reference 17*).

Fire Protection Systems and Features
Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features. Compliance with Chapter 3 is documented in the Fire Protection Program Report (*Reference 17*).

Chapter 4 of NFPA 805 establishes the methodology and criteria to determine the fire protection systems and features required to achieve the nuclear safety performance criteria of Section 1.5 of NFPA 805. These fire protection systems and features shall meet the applicable requirements of NFPA 805 Chapter 3. These fire protection systems and features are documented in the Fire Protection Program Report (*Reference 17*).

Radioactive Release

Structures, systems, and components relied upon to meet the radioactive release criteria are documented in the Fire Protection Program Report (*Reference 17*).

9.5.1.1.5 Definition of "Power Block" Structures

Where used in NFPA 805 Chapter 3 the terms "Power Block" and "Plant" refer to structures that have equipment required for nuclear plant operations. For the purposes of establishing the structures included in the fire protection program in accordance with 10 CFR 50.48(c) and NFPA 805, the plant structures listed in Table 9.5-2 are considered to be part of the 'power block'.

9.5.1.2 System Design

9.5.1.2.1 General

Fire detection instrumentation is located in all areas of the plant containing safety-related equipment and in areas containing large amounts of combustible or flammable materials. Actuation of fixed suppression systems and early warning alarms are provided by these detectors.

Normal fire protection is provided by fixed water deluge spray systems, fixed sprinkler systems, fixed Halon 1301 systems, hose lines, and portable and wheeled extinguishers suitably located in the required areas.

Water to the fire suppression system is supplied, via a header, by two fire pumps. The source of water is Lake Ontario. The yard loop and fire hydrants are supplied by the town of Ontario water supply.

The fire protection system can be used as a backup for the service water (SW) system supply to spent fuel pool (SFP) heat exchanger A, the standby spent fuel pool (SFP) heat exchanger, motor driven auxiliary feedwater pumps (MDAFW), standby auxiliary feedwater pumps (SAFW), and the diesel generator lube-oil coolers and jacket water heat exchangers via temporary hoses.

Fire barriers are located throughout the plant to separate established fire areas from each other and also to separate certain safety areas from the remainder of the plant. These barriers are designed to stop a fire from propagating from one area to the other. All penetrations in these barriers are sealed with appropriate materials to match the requirements of the barrier. Fire areas have been defined based upon separation of equipment and cables to ensure that at least one path of safe shutdown systems is always available.

Routing and separation standards applicable to existing cables are those that were invoked at the time of cable installation. For more information, see Section 8.3.1.4.

Fire prevention and mitigation considerations have been included in the design of ventilation systems, drain systems, lighting systems, communication systems, electrical and instrument cables, layout and materials, and oil collection systems.

Fire prevention is controlled by administrative methods to prevent accumulation of combustible materials and to practice good safety methods.

9.5.1.2.2 Fire Detection and Signaling Systems

The plant has a protective signaling system which alarms locally in selected parts of the plant and transmits fire alarm, supervisory, and trouble signals to the control room. In addition to signals from fire detection devices in various rooms or ventilating systems, the system transmits signals indicating water flow in water spray or sprinkler systems, fire pump operation, fire pump trouble, and low fire water tank level or pressure. Fire alarms are initiated by the smoke and heat detectors and by water flow or pressure switches in the water fire suppression systems. Additional protection is available by the installation of tamper switches on all major valves, unless they are locked in position.

The signaling system is powered by the emergency power supply system and automatically transfers to a 4-hour battery backup supply if its normal power source is interrupted.

Fire detection and signaling systems are generally designed and installed in accordance with NFPA 72D.

Smoke detectors and/or heat detectors have been provided in every area that contains safety-related equipment. Some detectors provide early warning fire detection and notification only. Others provide fire suppression system actuation in addition to detection and notification. These detectors are supervised to detect and annunciate circuit breaks, ground faults, and power supply failures. Remote test panels, which allow remote testing of the sensitivity of the detectors, are installed in the vicinity of smoke detectors that are difficult or hazardous to reach.

Periodic inspection and surveillance tests of the fire detection devices are in accordance with NFPA 72E and Technical Requirements Manual (TRM) section 3.3.4. The list of instruments required to be operable for fire detection and suppression systems actuation and their locations are shown in the TRM.

Fire alarm signals are provided as an integral part of the fire suppression systems to indicate an alarm in case of equipment malfunction, tampering, or in case of fire in any protected area. Additional audio and visual alarms and operating switches are provided in the control room, together with such pressure gauges, test, and reset switches as are required to completely monitor the fire protection system. The fire alarm signals in the control room are distinctive from other equipment alarms.

9.5.1.2.3 Fire Suppression Systems

Fire suppression is provided by fixed water spray and sprinkler systems, fixed gas systems, hose lines, and portable and wheeled extinguishers suitably located in the required areas. The water systems associated with the fire protection system are shown in Drawings 33013-1989 through 33013-1993. The fixed gas systems for the relay and multiplexer (MUX) rooms are shown in Drawing 33013-1242.

9.5.1.2.3.1 Water Supply

The fire protection water supply for the automatic and manual water suppression systems and hose stations inside the plant is pumped from Lake Ontario.

A fire header of sufficient size is provided to deliver an adequate quantity of water throughout the plant at a pressure of no less than 75 psi at the highest nozzle. The water supply for the fire hydrants on the yard fire main are supplied with water from the Town of Ontario. The yard hydrant system can be used as a backup to some of the fixed protection systems and all of the inside hose stations through wall hydrants in four locations (*Reference 18*). Drawing 33013-1607 shows the fire protection system yard loop (yard fire main). See also Section 9.5.1.2.3.5.

9.5.1.2.3.2 Fire Pumps

The water supply is delivered by a combination of two vertical shaft centrifugal fire pumps located in the screen house. Both pumps take suction from the circulating water intake. One pump is diesel-engine driven and the other is electric-motor driven. Each pump has a design rated minimum output of 2000 gpm at 125 psig, which is adequate to meet the largest anticipated water demand (*Reference 18*).

An automatic sprinkler system supplied by the yard fire main is provided in the area of the screen house that contains the two fire pumps. A curb has been installed around the diesel fire pump and the diesel oil storage tank to control any diesel oil leaks. The curbed area is equipped with a floor drain which drains to a holding tank buried outside the screen house.

A 15,000-gal pressure tank (10,000 gal of water) and a 120-gpm centrifugal jockey pump maintain system pressure at a minimum of 100 psig. When the system pressure drops below the nominal 95 psig setpoint (plus or minus 5 psig) the electric-motor-driven fire pump starts. If the pressure drops below the nominal 85 psig setpoint (plus or minus 5 psig), the diesel-driven fire pump starts.

An automatic controller is located with each fire pump. Each pump can be manually started from the control room or at the individual controller. Each pump can be manually stopped at the controller. In addition, the electric-driven fire pump can also be manually stopped by opening a circuit breaker located in the screen house near the fire pumps. Pump running, water flow, and pump power loss or engine trouble signals are annunciated in the control room as well as at the individual pump controllers. The fire relay panel provides these signals for the control room alarms and indications and start signals to the pumps on system pressure drop and on water flow in the fixed fire suppression systems served by the pumps.

The diesel fire pump is provided with an engine coolant heater and filter. The function of the heater is to raise the engine block temperature to a level at which the engine can start more easily and which would result in reduced wear on engine components. The coolant filter removes corrosion particles from the engine coolant.

The diesel fire pump engine is started by redundant 24-V batteries. The batteries are maintained fully charged and ready for service by an automatic dual battery charger operating in the float mode. The charger can charge one battery on high rate while maintaining the other battery on float or charge both batteries on high rate simultaneously. The charger automatically switches from float to high rate and back in order to recharge the batteries when necessary. The 120-V ac power is supplied to the battery charger from a screen house lighting panel.

An analysis was performed, which determined that the fuel consumption rate for the original diesel fire pump at full load is approximately 14.9 gal/hr. This analysis also cited that the capacity of the diesel fire pump fuel storage tank is 275 gal. The original fire pump was replaced in 1995. Fuel consumption for the installed diesel fire pump is approximately 13.1 gal/hr (*Reference 19*). Testing performed with the completion of the fire pump upgrades in 2003 determined that actual fuel consumption was approximately 12.0 gal/hr.

Specific procedures covering the diesel fire pump testing and maintenance include battery surveillance, testing of diesel oil in the day tank, and the requirement that the diesel engine be operated for a minimum of 15 minutes each month.

9.5.1.2.3.3 Piping and Valves

A separate 10-in. discharge line from each fire pump supplies the 8-in. and 10-in. interior loop main. All automatic and manual fixed water suppression systems and interior hose stations are supplied by this loop main. Outside screw and yoke gate valves subdivide the loop into a number of sections so that a single section can be isolated without impairing the entire loop. The design is such that isolation of a section of fire water piping system does not cause a loss of both the fixed suppression system protection and the manual hose coverage for the same area, with the exception of the service building. For the service building, use of manual hose lines from the exterior yard main provides backup suppression capability. Sectional valves, which are locked in the open position, are provided on the exterior yard main to allow the loop to be subdivided into a number of sections so that a single section can be isolated without impairing the entire system. Valves controlling water flow into sprinkler or deluge systems are locked open and/or are provided with electrical supervision (valve tamper switches). Sectional valves on the interior loop and valves controlling fire pump discharge are locked open and/or are provided with electrical supervision (valve tamper switches).

The discharge piping inside the Screen House is qualified as Seismic Category I. Qualification of the piping was performed during its original design in 1969 to meet USAS B31.1, Code for Pressure Piping.

When not required for fire suppression, the electric-driven fire pump can be aligned to supply water to the traveling screen wash header at a pressure higher than service water pressure. The purpose of this spray wash system is to remove debris from the traveling screens when high debris conditions exist. The high-pressure spray wash (HPSW) system is manually operated, and encompasses piping, two manual gate valves, an automatic isolation valve and a check valve which cross-connects between the fire suppression system test line and the main traveling screen wash header supply line downstream of the safety-related service water non-essential load isolation motor-operated valves. Since the electric-driven fire pump secondary function is to supply the high-pressure spray wash (HPSW) system only when plant conditions warrant its use, upon any automatic actuation of the electric-driven fire pump via a fire suppression system demand, the high pressure spray wash (HPSW) will automatically be isolated such that the primary function of supplying the fire suppression system can be satisfied.

9.5.1.2.3.4 Fire Hydrants

Yard fire hydrants are provided at approximately 250-ft intervals around the exterior of the plant. The lateral to each hydrant is controlled by a key-operated (curb) valve. Threads on hydrant outlets and hose couplings are compatible with those of fire departments which serve the plant. Impact barriers protect those fire hydrants and post-indicator valves which are located within 25 ft of roadways.

The ground area surrounding each exterior hydrant is graded to provide a clearance of at least 12 in. between the ground and the center of the lowest hydrant port.

Fire-fighting equipment is housed within hose houses. Administrative procedures cover snow removal operations and inspection of all outdoor fire hydrants for drainage immediately prior to freezing winter weather and for proper function immediately after the winter season. A yard hydrant on the southeast corner of the yard loop provides backup fire suppression capability for the transformers and primary fire suppression capability for the standby auxiliary feedwater building.

A dry hydrant assembly, located west of the screen house, provides capability to draft Lake Ontario water via the use of an offsite fire department pumper truck if needed.

9.5.1.2.3.5 Yard Loop

The fire protection system yard loop (yard fire main) is shown in Drawing 33013-1607. The yard loop supplies water to the yard fire hydrants and as a partial backup to the water suppression systems inside the plant (*Reference 18*). The yard loop provides a backup source of cooling water if service water (SW) is lost. It provides a backup to the condensate storage tanks for feedwater to the motor-driven (MDAFW) or turbine-driven (TDAFW) auxiliary feedwater pumps. It provides a backup to the DI water storage tank for feedwater to the standby auxiliary feedwater pumps (SAFW). It can be used to provide cooling water to the emergency diesel generators. It provides an alternative source of cooling to the component cooling water (CCW) heat exchangers (under emergency, beyond design basis conditions only). The yard loop is equipped with manual isolation gate valves, as shown in Drawing 33013-1607, to provide segment isolation in the case of line failures. The yard loop is supplied water from the town of Ontario water system.

9.5.1.2.3.6 Interior Hose Stations

A total of 43 interior hose stations, each equipped with 100 ft of 1.5-in. diameter UL or ULC approved municipal fire hose, are provided to protect various areas of the plant. The nozzles are 1.5-in. fog nozzles designed specifically for use in electrical fires and have a fog pattern range from 30° to 90° with no straight stream capability. A list of all available hose station locations is provided in Table 9.5-1. Operability and surveillance requirements for those hose stations required to be operable are specified in the Technical Requirements Manual (TRM) Section 3.7.4.

9.5.1.2.3.7 Water Suppression Systems

Water suppression systems include water spray systems and water sprinkler systems with open or closed-head nozzles and sprinkler heads. The water suppression systems meet the design installation requirements of NFPA 13 and/or NFPA 15. Each sprinkler system has an outside screw and yoke shutoff valve or other suitable fire protection service listed isolation valve. All control valves for spray or sprinkler systems are electrically supervised with alarms in the control room or are locked in the proper position. Other important valves on the water supply are either electrically supervised or locked in the proper position. The water flow rates follow NFPA 15 guidelines. The water suppression systems and the areas covered are tabulated below.

Automatic water spray systems (which can also be manually actuated) provide protection for

a. The turbine lube-oil system in the turbine building.

b. The hydrogen seal-oil system in the turbine building.

c. The oil storage room in the turbine building.

d. The oil-filled transformers outside the turbine building.

e. The cable trays in the screen house.

f. The cable tunnel.

g. The charcoal filter unit in the auxiliary building ventilation system (ABVS).

h. Control room-turbine building wall.

i. The cable trays in the air handling room.

Manually actuated water spray systems provide protection for

a. The turbine-driven auxiliary feedwater pump (TDAFW) and feedwater pump oil tank area.

b. The condenser pit area.

c. The relay room.

The relay room water suppression systems serve as backup to the automatic Halon system that provides primary protection.

Automatic preaction sprinkler systems provide protection for

a. The cable entrance area at the auxiliary building.

b. The cable tray area in the basement of the auxiliary building (elevation 235 ft 8 in.).

c. The cable tray area at elevation 253 ft 6 in. of the auxiliary building.

d. The cable tray area in the intermediate building.

e. The diesel generator rooms.

f. The standby auxiliary feedwater building annex diesel generator room.

The sprinkler systems for these areas have closed-head sprinklers with preaction trim on the deluge valves in accordance with NFPA 13.

Automatic sprinkler systems provide protection for the following.

a. The service water (SW) pumps in the screen house.

b. The fire pumps in the screen house.

c. The turbine island.

d. The service building.

e. The technical support center diesel room.

f. The turbine building mezzanine office and shop areas.

g. GE Betz water treatment trailers and vestibule.

An automatic and manual water curtain is provided for protection of the wall between the turbine building and the control room (superwall). The design of the water curtain is in accordance with NFPA 15. The system is actuated by heat detection.

Closed-head, close-spaced sprinklers are installed around the perimeters of the east and west stairwells and the equipment hatch, at the ceiling level of the auxiliary building mezzanine, as water curtain fire barriers to prevent the spread of fire between the mezzanine and operating levels. The sprinkler systems are wet pipe systems with automatic sprinkler heads rated at 165°F, which conform to the NFPA 13 standard temperature rating of 135°F to 170°F.

The containment postaccident charcoal filter units are protected with a water dousing system from the containment spray header, as described in Section 6.5.1.2.2.5.

The service building interior, including storage areas, shop areas, offices, locker rooms, and all rooms having combustible material, is protected by automatically operating wet-pipe sprinkler systems, with exterior water gong, which provides a local alarm annunciation at the north exterior of the building and indication in the control room.

The turbine-driven auxiliary feedwater pump (TDAFW) is protected with a water spray system consisting of six spray nozzles (Grinnell Mulsifyre Projector Type S-1 40-15) located in an array above the equipment. The north center and northwest corner nozzles are partially blocked by existing auxiliary feedwater piping in the area. An analysis was performed which demonstrated that the original design had sufficient margin such that the existing configuration met RG&E’s original criteria of 0.5 gpm/ft^2 water spray density and exceeds the NFPA 15 general range of 0.2-0.5 gpm/ft^2.

9.5.1.2.3.8 Gas Suppression Systems

Total flooding automatic Halon 1301 extinguishing systems are provided in the relay room, multiplexer (MUX) room, and technical support center PPCS/SPDS computer room. The systems are designed in accordance with NFPA 12A-1980, Section 1.5.4, to maintain a Halon concentration of 5% for at least 5 minutes following delivery (sufficient time to allow effective emergency action by trained personnel). Ionization type smoke detectors in each area alarm and annunciate in the control room. Where appropriate, a reserve supply of Halon 1301 permits prompt restoration of automatic protection following a system discharge. The Halon fire extinguishing systems are controlled by electronic control systems that are interfaced with the station fire detection system. The control system coordinates the fire detection system with local alarm actuation, air conditioning and ventilation shutdown as appropriate, electrical power disconnection as appropriate, and Halon discharge. In addition to automatic activation, Halon can be released using local manual pull stations or by operation of manual key switches on the fire control panels in the main control room. The relay room and MUX room systems are shown in Drawing 33013-1242.

9.5.1.2.3.9 Portable Fire Extinguishers

Pressurized water, dry chemical, carbon dioxide portable extinguishers are distributed throughout the plant in accordance with the provisions of NFPA 10.

9.5.1.2.3.10 Wet Chemical Suppression System

An Ansul wet chemical suppression system has been installed in the cafeteria at the south end of the service building. This system has both automatic and manual actuation capabilities. In addition, actuation of the kitchen hood system will alarm in the guard house on the Gamewell Fire Panel, activate local horn and strobes, and de-energize the fryer and grill in the kitchen area of the cafeteria. This system is UL-300 rated and the cylinder containing the wet chemical agent is of a 3-gallon capacity. The system is installed to NFPA 96, "Standard for Ventilation Control and Fire Protection of Commercial Cooking Operations."

9.5.1.2.4 Other Design Considerations

9.5.1.2.4.1 Smoke Removal

The air handling systems of the ventilation systems are capable of exhausting volumes of smoke directly to the outside.

In addition, three portable smoke ejectors, each with 5000-cfm capacity, are provided for smoke removal. Flexible hose sections are provided to channel smoke and hot gases through the buildings.

9.5.1.2.4.2 Breathing Equipment

At least 10 self-contained breathing units dedicated to emergency use are provided. Each breathing unit has one spare bottle. The plant has the capability to supply breathing air to 10 people for 6 hours at the rate of two (1.0 hour) bottles per person per hour. A compressor and cascade system are provided onsite to supply the breathing air.

9.5.1.2.4.3 Control Building Ventilation

The control building ventilation system is designed to provide a safe, controlled environment for the control room, relay room, multiplexer (MUX) room, and battery rooms under all required conditions, including high- and moderate-energy line breaks outside containment, as well as small fires in the relay room or control room. Control building ventilation systems are described in Sections 3.11.3.5, 6.4, 8.3, and 9.4.9.2.

9.5.1.2.4.4 Reactor Coolant Pump Motor Oil Collection System

The reactor coolant pump motor oil collection system consists of a package of splash guards, drip pans, and enclosures assembled as attachments to the reactor coolant pump motor at strategic locations to preclude the possibility of oil making contact with hot reactor coolant system components and piping. Any leaking oil is drained from each individual pump to its own collection tank, which is capable of handling the entire oil inventory of the motor. Strainers are placed at the drain of each drip pan or enclosure. The oil collection components are designed and attached to preclude dislodging during a seismic event.

9.5.1.2.4.5 Floor Drains and Curbs

Safety-related equipment is mounted on pedestals and floor drains provided in these areas are generally adequate to carry off fire water and prevent safety-related equipment from being flooded with standing water. In areas such as the control room, where floor drains are not provided, fire water will be drained out through door openings.

Curbs are provided in the screen house to prevent water or flammable liquid from flowing into the basement where both divisions of safety-related cables are routed. Additional curbs are provided around the diesel-driven fire pump area in the screen house.

A barrier has been installed around the turbine lube-oil reservoir area to contain possible oil spillage. The capacity of the enclosed area is large enough to retain the entire contents of the lube-oil system plus 10% margin for fire water.

Where drains from safety-related areas are tied into drains from areas which contain a large quantity of flammable liquid, backflow protection is provided to prevent possible spread of a liquid fire via the drain system *Reference 8*.

9.5.1.2.4.6 Lighting Systems

See Section 9.5.3.

9.5.1.2.4.7 Communications

There are three communication systems within the plant. The primary system is the combination paging and party system; in addition, there is a sound powered phone system and a radio paging system.

The sound powered system is hard wired with separate wires from the combination paging and party system. The radio paging system provides communication with areas inside the containment with the help of a radio antenna mounted in the containment. Additionally, a repeater located in the Nuclear Assessment Building allows for greater flexibility with radio communications. There is adequate redundancy with these three systems to ensure good communications throughout the plant during any fire emergency (see Section 9.5.2).

9.5.1.2.4.8 Electrical Cable Insulation

The cable insulation used at Ginna Station includes Kerite, oil-based rubber, neoprene, and polyvinyl chloride. The cables have, as a minimum, passed the ASTM and UL horizontal and vertical flame tests. Power cables and polyvinyl chloride control cables have passed the Consolidated Edison Bonfire Test. The majority of the electrical cables were purchased and installed prior to the publication of the IEEE 383 standard for flame testing of electrical cables; however, the potential combustion products for the materials used at the station have been evaluated from generic test reports and do not exhibit an unusual or significantly hazardous nature. All cables used for modifications meet IEEE 383 criteria unless specifically excepted. The spent fuel pool (SFP) bridge crane motive and control power cable is payed in and out from a spring-loaded storage reel assembly. Cable meeting IEEE 383-1974 flame retardant requirements and meeting the flexing duty requirements of the bridge crane cable was not available at the time replacement was required. The replacement cable was reviewed and it was determined that the proposed replacement would not adversely impact 10 CFR 50 Appendix R compliance methods or options used to maintain compliance (*Reference 7*). This determination will be made whenever it is impracticable to meet IEEE-383 criteria for cables used in modifications.

9.5.1.2.4.9 Fire Barriers

The fire hazards analysis submitted to the NRC in February 1977 (*Reference 2*) identifies the fire barriers in the plant and the requirements for maintaining their integrity. These barrier requirements were determined by the fire loadings calculated for each area subject to a potential fire hazard. As a result of this analysis, several design modifications were implemented at the plant including upgrading of the rating of original barriers and installing new barriers. The updated Ginna Station Fire Protection Program Report (*Reference 17*) was completed and provides a regularly updated fire hazards analysis of the station.

Additional definition of fire areas and barriers and analysis of fire zones were conducted as part of the 10 CFR 50, Appendix R, review effort. The addition of the water curtain around the perimeters of the stairwells and equipment hatch at the ceiling level of the auxiliary building mezzanine floor is a part of this effort. See Section 9.5.1.2.3.7. Also, 3-hour-rated dampers were installed in ducts penetrating these fire areas. New fire barrier cabling wraps were installed in battery room 1B, several cables for the equipment in the charging pump room that is located in the auxiliary building, the intermediate building, and the containment. The cable wraps inside containment are for radiant energy shield purposes. The fire barrier inside the B Diesel generator cable vault was upgraded with materials capable of three-hour rating. It is constructed of clay masonry bricks and mortar. A three-hour rated fire damper, DGVB-85, was also installed in the north wall of the brick enclosure (*Reference 21*). Penetration seals in this barrier were also sealed with approved materials. The charging pump room barriers were upgraded to provide three-hour rated resistance. Fire barriers associated with power and instrument cable systems are discussed in Section 8.3.3.

Fire protective coatings have been applied to the structural steel members forming or supporting a designated fire barrier. In this regard, the structural steel roof beams and a column that supports the roof of the A and B battery room and the floor of the relay room are provided with a fire protective coating, which will ensure that adequate margins of safety will be maintained for at least 1 hour during a fire emergency.

9.5.1.2.4.10 Electrical Cable Penetrations

The fire seals installed at Ginna Station fall into two major categories:

a. Seals installed in 1975 using BISCO SF-20 silicone room temperature vulcanizing foam rubber.

b. Seals installed since September 1979 using Dow Corning 3-6548 silicone room temperature vulcanizing foam rubber.

The adequacy of several fire endurance tests and their applicability to cable tray and conduit penetration fire seals has been demonstrated in a submittal to the NRC in June 1980, (*Reference 8*) with the conclusion that the seal designs at the station are either similar to or more conservative than the seal designs tested by the ASTM E-119 fire test method for a 3-hour rating. The NRC concurred with the evaluation (*Reference 4*). NRC Information Notice 88-04 alerted licensees that some fire barrier penetration seal designs may not be adequately qualified for the design rating of the penetrated fire barriers. As part of RG&E's review in response to Information Notice 88-04, a program was established to evaluate fire barrier penetrations against a tested configuration and examine the qualification test documentation. Branch Technical Position (BTP)-APCSB 9.5-1 requires that cable and cable tray penetrations of fire barriers (vertical and horizontal) be sealed to give protection at least equivalent to that of the fire barrier. Although not specifically stated in APCSB 9.5-1 that penetration designs must be qualified by tests, RG&E proceeded with this program in order that the penetrations would continue to meet a tested configuration, when being maintained or involved in a plant modification, thereby ensuring the barrier would not be degraded.

For fire barrier penetration seals for which it is not possible to achieve a duplication of a specific tested configuration, appropriate compensatory measures are taken, such as posting fire watch patrols when required by the Technical Requirements Manual (TRM) section 3.7.5, temporarily repairing and qualifying the penetration until it can be reworked, and performing technical evaluations to demonstrate that the penetration meets an equivalent level of protection. Guidance from Generic Letter 86-10 is employed in these cases.

9.5.1.2.4.11 Piping and Duct Penetrations

Piping penetration of fire barriers are either poured in place or sealed by one of the following methods: grout, silicone RTV foam seals, or flexible reinforced silicone-rubber boots. The piping and duct penetrations have fire resistance ratings commensurate with the fire hazards on either side of the penetration determined by the fire hazards analysis (*Reference 2*). The fire rating adequacy of the seals was demonstrated in a submittal of fire test reports to the NRC in June 1980 (*Reference 8*). Based on the data of these reports, the NRC concurred that the piping and duct penetration seals provide adequate resistance to prevent a fire from propagating through the rated fire barriers (*Reference 4*).

9.5.1.2.4.12 Cable Separation

The design and construction of Ginna Station predates current industry standards of physical separation. The criteria and design features related to cable separation at the plant are discussed in Section 8.3.1.4. Cable separation as it relates to the safe shutdown capability of the plant under a fire emergency is discussed in the references cited in Section 9.5.1.4.

9.5.1.2.4.13 Spray Shields

Water spray shields are provided in the intermediate building over the control rod drive motor control center and switch-gear, and in the auxiliary building over switchgear, motor control centers, and other electrical equipment to help protect this equipment from damage or undesirable effects from the application of fire water.

9.5.1.2.4.14 Construction Joints

The construction joints between containment and the surrounding buildings provide fire resistance commensurate with the hazards in the area.

9.5.1.3 Safety Evaluation

The Fire Protection Program Report (*Reference 17*) documents the achievement of the nuclear safety and radioactive release performance criteria of NFPA 805 as required by 10 CFR 50.48(c). This document fulfills the requirements of Section 2.7.1.2 “Fire Protection Program Design Basis Document” of NFPA 805. The document contains the following:

- Identification of significant fire hazards in the fire area. This is based on NFPA 805 approach to analyze the plant from an ignition source and fuel package perspective.
- Summary of the Nuclear Safety Capability Assessment (at power and non-power) compliance strategies.
  - Deterministic compliance strategies
  - Performance-based compliance strategies (including defense-in-depth and safety margin)
- Summary of the Non-Power Operations Modes compliance strategies.
- Summary of the Radioactive Release compliance strategies.
- Summary of the Fire Probabilistic Risk Assessments.
- Key analysis assumptions to be included in the NFPA 805 monitoring program.

9.5.1.4 <u>Fire Protection Program Documentation, Configuration Control and Quality Assurance</u>

In accordance with Chapter 3 of NFPA 805 a fire protection plan documented in the Fire Protection Program Report (*Reference 17*) defines the management policy and program direction and defines the responsibilities of those individuals responsible for the plan’s implementation. The Fire Protection Program Report (*Reference 17*) documents either by reference or directly:

- The senior management position with immediate authority and responsibility for the fire protection program.
- A position responsible for the daily administration and coordination of the fire protection program and its implementation.
- Defines the fire protection interfaces with other organizations and assigns responsibilities for the coordination of activities. In addition, the Fire Protection Program Report (*Reference 17*) identifies the various plant positions having the authority for implementing the various areas of the fire protection program.

- Identifies the appropriate authority having jurisdiction for the various areas of the fire protection program.
- Identifies the procedures established for the implementation of the fire protection program, including the post-transition change process and the fire protection monitoring program.
- Identifies the qualifications required for various fire protection program personnel.
- Identifies the quality requirements of Chapter 2 of NFPA 805.

Detailed compliance with the programmatic requirements of Chapters 2 and 3 of NFPA 805 is contained in the Fire Protection Program Report (*Reference 17*)**.**

9.5.2 COMMUNICATIONS SYSTEMS

A broad range of communications equipment is available at Ginna Station. Several systems are installed for normal and emergency communications as well as for communications with outside agencies. Equipment is periodically verified operable by plant procedures. The use of particular types is specified in the appropriate implementing procedures as first choice and backup systems. Communications systems are tested periodically.

9.5.2.1 Public Address System

A special warbling tone on the GAI-Tronics page system is sounded from the control room to warn personnel of a site evacuation. Warning is immediate to all persons onsite. High noise areas have, in addition to the public address system, red warning lights with signs to direct personnel to evacuate. Special announcements on the page and special tones are used for other emergencies. The plant evacuation alarm, plant fire alarm, and plant attention signal are each distinct tones over the GAI-Tronics page system and are actuated from the control room by pushbutton or switch.

9.5.2.2 Telephone Systems

Communications between the control room, technical support center, emergency survey center, and other operations centers can be established using either telephone, two-way intercom, radio, or the plant public address system.

The telephone system at Ginna affords a great deal of flexibility and capacity. Calls can be received or made to either the Rochester telephone system or the Ontario telephone system. The telephone system has its own power supply located onsite which could maintain house phones independent of offsite lines. There are also Rochester direct lines and Ontario direct lines.

A sound powered phone system consisting of headsets, amplifiers, power supply, and wall-mounted jacks provides party-line two-way communications throughout the plant for system tests, etc.

9.5.2.3 Radio Systems

There are several radio frequencies available for use at Ginna Station. These frequencies are assigned to the fire brigade, security, operations, Appendix R usage, and radiation survey teams. The base stations and antennae are located for maximum transmission coverage of the areas of use. The security channel is monitored at central security and at the guardhouse. The radiation survey teams have operator capability at the emergency survey center, the technical support center, and the emergency operations facility recovery center. Fire brigade communications will be monitored in the control room. Mobile radio sets are available for the use of survey teams in the field. A channel is available for indirect communication to the State Police, and Monroe and Wayne County Sheriffs.

Portable low power hand radio sets are located in the technical support center to be distributed in the event of an emergency for backup or for mobile communications. Portable hand radio sets are also located in the emergency survey center for the use of survey teams. Offsite survey teams can communicate through these portable radio sets to a base station which may be set up at either the emergency survey center, technical support center, or emergency operations facility/recovery center. The base station is capable of operating with a backup generator alternate power supply. Additionally, a sufficient number of portable radio sets are available for Operation’s use following a safe shutdown fire scenario.

9.5.2.4 Offsite Communications

Notification to state and county emergency response organizations is available 24 hours per day. The State Warning Point is staffed by the New York State Emergency Management Office. Monroe County Office of Emergency Preparedness and Wayne County Office of Emergency Management answer the New York State Radiological Emergency Communications System (RECS) line during the work day. During nonbusiness hours, weekends, and holidays, the RECS line is covered for Monroe County and Wayne County at their "911" centers.

At Ginna Station there are always control room personnel to originate calls. New York State has responsibility for communications to other counties which may fall within the ingestion exposure zone. Any contacts with Canada or Ontario Province would also be through the state agencies.

To contact appropriate offsite agencies the telephones would normally be used as discussed in Section 9.5.2.2, with direct lines or the onsite telephone system. If necessary, radio communications can be used to notify emergency agencies and relay messages through their radio systems.

Communications with federal emergency response organizations consist of telephone contact to the Department of Energy, Brookhaven Radiological Assistance Program. This call would be made by a member of the emergency response organization. Their assistance may also be requested by the state or counties.

The NRC Emergency Telecommunications System (ETS) provides for essential emergency communications with the NRC and is described in Section 9.5.2.5.

Ginna Station uses the simulated control room in the training center instead of the actual plant control room for annual emergency plan drills. To support the use of the simulated control room for these drills, the GAI-Tronics page system, sound powered phone, onsite telephone system, New York State Radiological Emergency Communication system (RECS), and the NRC Emergency Telecommunications System (ETS) emergency notification link are installed in the simulated control room.

9.5.2.5 Emergency Communications With the NRC

Essential communications with the NRC during an emergency is by use of the Emergency Telecommunications System (ETS). The functions provided by the system include the emergency notification system, health physics network, reactor safety counterpart link, protective measures counterpart link, emergency response data system, management counterpart link, and the NRC Operations Center's local area network. Telephones for the ETS are located in the technical support center, emergency operations facility, control room, simulator, and the Senior Resident NRC Inspector's office. The available ETS functions vary by telephone location.

Additional information regarding emergency communications with the NRC is discussed in the Nuclear Emergency Response Plan.

9.5.3 LIGHTING SYSTEMS

Fixed emergency lighting units are provided in safety-related areas and other areas which contain fire hazards to facilitate emergency operations, manual fire fighting, and access to and egress from each designated fire area. The lighting units are 8-hour rated. In addition to the fixed lighting systems, portable battery-powered handlights are provided.

Ginna safe shutdown panels are located in several areas of the plant. The lighting at the safe shutdown areas has been determined to be sufficient to perform all required safe shutdown tasks.

The control room normal and emergency lighting systems provide adequate illumination in accordance with the guidelines of NUREG 0700, Section 6. The control room normal lighting system is capable of functioning at all times, excluding loss of ac power, at which time the 125-V dc emergency lighting system is automatically turned on. The control room emergency lighting fixtures are fed from either the A or B station batteries. In the event of loss of either battery there is a transfer switch in the control room by which the operators can manually switch the emergency lighting feed from one train to the other. Should loss of either battery occur in the emergency lighting mode, an 8-hour-rated emergency light fixture located near the transfer switch shall remain functional to provide sufficient lighting to perform the transfer. The 125-V dc power supply up to the point of termination at the emergency lighting fixtures is Class 1E and Seismic Category I. The emergency lighting fixtures are standard. A prototype fixture has been seismically tested in accordance with IEEE 344-1975 to ensure continued operation of the fixtures in the event of an earthquake. In addition, an analysis of the seismically reinforced suspended ceiling has been performed to ensure that the ceiling, including the normal and emergency lighting fixtures, does not create a hazard to control room personnel or safety-related equipment during a seismic event.

A security lighting system along the fence at Ginna Station has been provided. The system has been designed to meet the requirements of ANSI 18.17, Industrial Security Plans for Nuclear Power Plants (see Section 13.6).

9.5.4 DIESEL GENERATOR FUEL OIL STORAGE AND TRANSFER SYSTEM

The diesel generator fuel oil storage and transfer system is shown in Drawing 33013-1239, Sheets 1 and 2.

The minimum permissible onsite fuel inventory is 10,000 gal (5000 gal for each diesel generator). This minimum diesel fuel oil inventory is maintained to ensure that both diesel generators can operate at their design ratings for 24 hours. This ensures that both diesel generators can carry the design loads of required engineered safeguards equipment for any loss-of-coolant accident conditions for at least 40 hours, or for one engineered safety feature train for 80 hours. Additional diesel fuel is stored at Ginna outside the protected area, near the warehouse, in two buried tanks that have a combined working capacity of 35,000 gallons. Trucking facilities exist to ensure deliveries of additional fuel oil within 8 hours.

Fuel oil is provided to each diesel engine by a 350-gal day tank located at the engine. When the engine starts, the engine-driven fuel pump provides fuel from the day tank. Each diesel generator day tank is normally supplied from its 6000-gal underground storage tank. A 480-V fuel oil transfer pump for each diesel engine pumps fuel oil at approximately 23 gpm at a discharge pressure of 15 psig from either storage tank to either day tank. A cross-connection allows each transfer pump to supply either day tank. The suction line to each fuel oil transfer pump includes a duplex strainer that can be serviced without interrupting flow. One fuel oil transfer pump has the capacity to supply both diesel generators at 110% load. Analysis shows the fuel oil consumption of one diesel generator is between 2.85 and 2.90 gpm at 110% load, including all uncertainties. The plant process computer system provides alarms on high and high-high differential pressure across each duplex strainer and on high and low-low fuel oil level in each day tank.

A local control switch in the diesel room for the transfer pump has two positions. In RUN, the pump will run until the day tank is full, then a fill line valve closes and a bypass valve opens to recirculate back to the storage tank. When the level in the day tank decreases, the valves will reposition to supply the day tank. In AUTO, the transfer pump starts when its diesel is running and the level in the respective day tank falls to a low level. The pump continues to run until its diesel is stopped. Again when the diesel day tank is full, the fill line valve closes and the bypass valve opens. Low level in either day tank is alarmed in the control room. Heat tracing is provided to maintain the fuel oil temperature in exposed pump suction piping in the event of a loss of heat in the diesel generator rooms. The heat tracing is thermostatically controlled to maintain the fuel oil in the pipe above 40°F. This provides sufficient margin above the point at which this portion of the suction piping could be considered inoperable based on the cloud point of the fuel oil, 23°F.

Watertight doors have been installed on the concrete manways of the underground diesel-oil storage tanks. These doors prevent the accumulation of water in the manways that might seep into the oil through the flanged manhole on the top of each storage tank.

The diesel generator fuel oil storage and transfer system surveillance tests and conditions for operation are provided in the Technical Specifications.

9.5.5 DIESEL GENERATOR COOLING SYSTEM

The diesel generator cooling system is shown in Drawing 33013-1239, Sheets 1 and 2.

The diesel generators are supplied with cooling water from the service water (SW) system. Service water (SW) is directed to the lube-oil cooler and jacket water coolers for each diesel generator, through two normally closed AOVs in parallel. The AOVs open on a diesel generator start signal is less than 10 seconds. The service water (SW) lineup is made reliable by ensuring that the service water (SW) crossover valves remain open at all times. This ensures that no matter which service water (SW) pump is selected to automatically start during an emergency, and no matter which diesel starts, the diesel that is running will receive cooling water. The 1A and 1C service water (SW) pumps are powered from bus 18 which can be supplied by diesel generator 1A. A selector switch in the screen house allows selection of either pump to automatically start. The 1B and 1D service water (SW) pumps receive their power from bus 17 which can be supplied from diesel generator 1B. Another selector switch in the screen house is provided for this set of pumps' automatic start feature.

An alternative means for diesel generator cooling is provided via a valve installed in the service water (SW) cooling to each diesel generator. The valve allows the connection of fire hoses from a fire protection valve in the diesel generator room in case of failure of the service water (SW) pump during diesel generator operation.

The cooling water is heated by jacket water heaters. Below each diesel is a subbasement which contains the buswork for that diesel. To prevent flooding of this area, a vault pump is provided for each diesel. This runs automatically as required to remove any accumulation of water.

9.5.6 DIESEL GENERATOR STARTING SYSTEM

The diesel generator starting system is shown in Drawing 33013-1239, Sheets 1 and 2.

Two 20-ft^3 air receiver tanks are provided to start each diesel. Each diesel generator air start system has a 480-V air compressor. Diesel generator 1A air compressor receives its power from the motor control center which can be supplied from diesel generator 1B. Diesel generator 1B air compressor receives its power from the motor control center which can be supplied from diesel generator 1A.

Each air system is utilized to crank the diesel with air to start the diesel within 10 sec. The compressors will automatically start at a nominal pressure of 230 psig to charge the receivers, and will automatically stop at a nominal receiver pressure of 250 psig. A relief valve set at 275 psig provides overpressure protection. Starting air from the air receiver tanks is initially supplied to two air regulators. The air regulators reduce the pressure to a nominal pressure of 130 psig to supply the air distributors. Air pressure required for starting the engine is 80 to 150 psig.

Additional testing performed during the 1992 MODE 6 (Refueling) outage demonstrated that with an initial pressure of 225 psig the air start system receiver tanks are sufficient to crank the diesels for a minimum of five starts without recharging the receivers and that the air start systems are capable of starting the diesels with a minimum air supply pressure of 80 psig.

Parallel dc-powered solenoid valves will open to admit air to the air start motor for each diesel. One solenoid valve is supplied from battery 1A and the other from battery 1B. The diesel air start systems can be cross-connected by two valves in series.

Periodic testing requirements of the diesel generator starting systems are provided in the Technical Specifications.

9.5.7 DIESEL GENERATOR LUBRICATION SYSTEM

A simplified diagram of the diesel generator engine lubrication system is shown in Figure 9.5-6. Lubrication of the engine is accomplished by a pre-lube pump and an engine-driven lube-oil pump. The engine must be kept prelubed and preheated to provide immediate starting. When the engine starts, the prelube pump and heater are deenergized and the engine-driven lube-oil pump provides oil flow.

9.5.8 DIESEL GENERATOR COMBUSTION AIR INTAKE AND EXHAUST

Fresh air for combustion is drawn into the engine through a filter and distributed to the cylinders by an air intake manifold. Turbocharging is used to increase flow or volume of air. This consists of a turbocharger driven by exhaust gases of the engine. Compressing the air with the turbocharger increases the temperature of the air. Since air intake temperature should be as low as possible for maximum operating efficiency, the air must be cooled before entering the cylinders. The system is shown in Drawing 33013-1239, Sheets 1 and 2.

REFERENCES FOR SECTION 9.5

1. Rochester Gas and Electric Corporation, Technical Supplement Accompanying Application for Full-Term Operating License, August 1972.

2. Letter from L. D. White, Jr., RG&E, to A. Schwencer, NRC, Subject: Fire Protection at R. E. Ginna Nuclear Power Plant, dated February 24, 1977.

3. Letter from D. L. Ziemann, NRC, to L. D. White, Jr., RG&E, Subject: Transmittal of Fire Protection Safety Evaluation Report (enclosure), dated February 14, 1979.

4. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: Supplement 1 to Fire Protection Safety Evaluation Report, dated December 17, 1980.

5. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: Supplement 2 to Fire Protection Safety Evaluation Report, dated February 6, 1981.

6. Letter from D. M. Crutchfield, NRC, to J. E. Maier, RG&E, Subject: Fire Protection, Ginna, dated June 22, 1981.

7. RG&E Appendix R Conformance Review, Ginna Station, Spent Fuel Pit Bridge Crane Cable, Number WR/TR 9020864, Revision 0, dated March 2, 1990.

8. Letter from L. D. White, Jr., RG&E, to D. M. Crutchfield, NRC, Subject: Fire Protection at R. E. Ginna Nuclear Power Plant, dated June 4, 1980.

9. Letter from L. D. White, Jr., RG&E, to D. L. Ziemann, NRC, Subject: Fire Protection at R. E. Ginna Nuclear Power Plant, dated May 15, 1978.

10. R. E. Ginna Nuclear Power Plant Fire Protection Plan, dated April 29, 1986.

11. Letter from J. E. Maier, RG&E, to D. M. Crutchfield, NRC, Subject: Transmittal of Appendix R, Alternative Shutdown System, dated January 16, 1984.

12. Letter from R. W. Kober, RG&E, to W. Paulson, NRC, Subject: 10 CFR 50, Appendix R, Alternative Shutdown System, Revision 1, dated October 4, 1984.

13. Letter from R. W. Kober, RG&E, to J. A. Zwolinski, NRC, Subject: Appendix R, Alternative Shutdown System, dated January 16, 1985.

14. Letter from J. A. Zwolinski, NRC, to R. W. Kober, RG&E, Subject: Safety Evaluation for Appendix R to 10 CFR Part 50, Items III.G.3 and III.L, dated February 27, 1985.

15. Letter from J. A. Zwolinski, NRC, to R. W. Kober, RG&E, Subject: Exemptions to Section III.G of Appendix R, dated March 21, 1985.

16. Letter from R. C. Mecredy, RG&E, to G. S. Vissing, NRC, Subject: Exemption to Section III.G of Appendix R, dated January 13, 1998.

17. EPM-FPPR, Ginna Station Fire Protection Program Report Volumes 1, 2, and 3.

18. Ginna Station, Design Analysis, DA-ME-2000-040, Revision 1, City Water Yard Loop Cross-Tie to Fire Yard Loop Hydraulic Calculation, dated September 8, 2003.

19. Ginna Station Design Analysis DA-ME-2001-031, Evaluation of Suppression System Flow Requirements, dated May 23, 2002.

20. Ginna Station Design Analysis DA-ME-93-108, Diesel Fire Pump Fuel Consumption Calculations, dated March 30, 2004.

21. Constellation Energy Design Analysis DA-ME-2003-018, Revision 1, Replacement of Appendix R Fire Barrier in the "B" Diesel Generator Vault, dated October 11, 2005.

22. ECP-10-000834 Rev. 0, Perform a comparison between UL and ULC standard for fire hose.

Table 9.5-1
FIRE SERVICE WATER HOSE REEL LOCATIONS

Hose Reel Number	Building	Floor	Location[a]
1	Turbine	Basement	Elevator
2	Turbine	Basement	Battery room
3	Turbine	Basement	Diesel generator rooms
4	Turbine	Basement	Steam generator feedwater pumps
5	Turbine	Intermediate	Elevator
6	Turbine	Intermediate	4160 bus
7	Turbine	Intermediate	Air ejector
8	Turbine	Intermediate	5A heater
9	Turbine	Turbine	Elevator
10	Turbine	Turbine	Control room
11	Turbine	Turbine	North wall
12	Intermediate	Level four	West
13	Intermediate	Level four	East
14	Intermediate	Level three	East
15	Intermediate	Level three	West
16	Intermediate	Level two	West
17	Intermediate	Level two	East
18	Intermediate	Level one	East
19	Intermediate	Level one	West
20	Intermediate	Level one	South
21	Intermediate	Level two	Nuclear sample room
22	Auxiliary	Operating	West
23	Auxiliary	Operating	Center
24	Auxiliary	Operating	East
25	Auxiliary	Intermediate	East
26	Auxiliary	Intermediate	Center
27	Auxiliary	Intermediate	West
28	Auxiliary	Basement	West

Hose Reel Number	Building	Floor	Location[a]
29	Auxiliary	Basement	Center
30	Auxiliary	Basement	East
31	Screen house	Main	Fire pumps
32	All-volatile-treatment building	Resin tank area	Northwest center
33	Containment	Operating	East
34	Containment	Operating	West
35	Containment	Mezzanine	East
36	Containment	Mezzanine	West
37	Containment	Basement	East
38	Containment	Basement	West
39	Service	Ground	North hall
40	Service	Main	North hall
41	All-volatile-treatment building	Technical support center	North
42	All-volatile-treatment building	Technical support center	South
43	Canister Preparation Building (CPB)	Operating	North West

a. Location indicates area served by the water hose, not necessarily the location of the water hose reel.

Table 9.5-2
Power Block Buildings

Reactor Containment Building
Auxiliary Building
Canister Preparation Building
Turbine Building
Control Building Complex
Diesel Generator Building
Screen House
Intermediate Building
Intake Structure
Standby Auxiliary Feedwater Pump Building
Cable Tunnel
Primary Hydrogen Storage Building
Secondary Hydrogen Storage Building
Nitrogen Storage Building
Service Building
Condensate Demineralizer Building
Transformer Yard