CHAPTER 9

AUXILIARY AND EMERGENCY SYSTEMS

9.1 GENERAL DESIGN CRITERIA

The General Design Criteria which apply to specific auxiliary and emergency systems are discussed in the appropriate system design section presented in this Chapter. The criteria which apply primarily to systems described in other Chapters of the FSAR are only stated, and cross-references are provided to identify the specific Chapter where the system is described and the general design criteria discussed.

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Part 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

9.1.1 Related Criteria

Reactivity Control Systems Malfunction

Criterion: The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits. (GDC 31 of 7/11/67)

As described in Chapter 7, and justified in Chapter 14, the Reactor Protection Systems are designed to limit reactivity transients to the applicable DNBR limit due to any single malfunction in the deboration controls.

Engineered Safety Features Performance Capability

Criterion: Engineered Safety Features such as the emergency core cooling system and the containment heat removal system shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public. (GDC 41 of 7/11/67)

Each of the auxiliary cooling systems which serve an emergency function provide sufficient capability in the emergency operational mode to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

Containment Heat Removal Systems

Criterion: Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure this system shall perform its required function, assuming failure of any single active component. (GDC 52 of 7/11/67)

Each of the auxiliary cooling systems which serves an emergency function to prevent exceeding containment design pressure, provides sufficient capability in the emergency operational mode to accommodate any single failure of an active component and still perform its required safety function.

9.2 CHEMICAL AND VOLUME CONTROL SYSTEM

The Chemical and Volume Control System performs the following functions: 1) adjusts the concentration of the chemical neutron absorber for chemical reactivity control, 2) maintains the proper water inventory in the Reactor Coolant System, 3) provides the required seal water flow for the reactor coolant pump shaft seals, 4) maintains the proper concentration of corrosion inhibiting chemicals in the reactor coolant and 5) maintains the reactor coolant and corrosion product activities to within design levels. The system is also used to fill and hydrostatically test the Reactor Coolant System.

During normal operation, this system also has provisions for supplying the following chemicals:

a) Regenerant chemicals to the deborating demineralizers
b) Hydrogen to the volume control tank
c) Nitrogen as required for purging the volume control tank
d) Hydrazine and Lithium Hydroxide, as required, via the chemical mixing tank to the charging pumps suction.

9.2.1 Design Bases

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has since completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

Redundancy of Reactivity Control

Criterion: Two independent reactivity control systems, preferably of different principles, shall be provided. (GDC 27 of 7/11/67)

In addition to the reactivity control achieved by the Rod Cluster Control (RCC), as detailed in Chapter 7, reactivity control 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 that might cause system parameters to exceed design limits.

Reactivity Hold-Down 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 to the public. (GDC 30 of 7/11/67)

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 boric acid tanks and ready for injection will always exceed that quantity required for the normal cold shutdown. This quantity will always exceed the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

The boric acid solution is transferred from the boric acid tanks by boric acid 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 offsite AC power. Using either one of the two boric acid transfer pumps, in conjunction with any of the three charging pumps, the RCS can be borated to hot shutdown even with the control rods fully withdrawn. Additional boration would be used to compensate for xenon decay. At a minimum CVCS design boration rate of 132 ppm/hr, the boron concentration required for cold shutdown can be reached well before xenon decays below its pre-shutdown level.

The RWST is a suitable backup source for emergency boration. When two charging pumps are used to transfer borated water from the RWST to the reactor coolant, the boron concentration required for cold shutdown can be reached before xenon decays below its full-power pre-shutdown level.

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.

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. (GDC 28 of 7/11/67)

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating 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 life of the initial core. The full length Rod Cluster Control (RCC) assemblies are divided into two categories comprising a control group and shutdown groups.

The control group, used in combination with chemical shim provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of RCC assemblies 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.

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. (GDC 29 of 7/11/67)

The reactor core, together with the reactor control and protection systems, is designed so that the minimum allowable DNBR is above the applicable limit located in the Core Operating Limits Report (COLR) and there is no fuel melting during normal operation including anticipated transients.

The shutdown groups of RCC assemblies are provided to supplement the control group of RCC assemblies to make the reactor at least 1.3% subcritical ($k_{eff}<0.99$) following trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCC assembly remains in the fully withdrawn position

Sufficient shutdown capability is also provided to maintain the core subcritical for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a steam bypass or relief valve. This is achieved with a combination of control rods and automatic injection of borated water from the Refueling Water Storage Tank (RWST) by the Safety Injection System with the most reactive rod assumed to be fully withdrawn. Manually controlled boric acid addition is used to maintain the shutdown margin for the long term conditions of xenon decay and plant cooldown.

Codes and Classifications

All pressure retaining components (or compartments of components) which are exposed to reactor coolant, 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, fittings and piping – USAS B31.1, including nuclear code cases.

System integrity was ensured by conformance to applicable code listed in Table 9.2-1, and by the use of austenitic stainless steel or other corrosion resistant materials in contact with both reactor coolant and boric acid solutions.

The regenerative heat exchanger and the tube side of the excess letdown heat exchanger were designed as per ASME III, Class C. This designation is based on the following considerations:

a) Two fail-closed air operated valves are installed in the line between the Reactor Coolant System and the regenerative heat exchanger shell side. Each of these

valves are provided with an independent signal to trip closed on pressurizer low level.

b) Two fail-closed air operated valves are installed in the line between the Reactor Coolant System and the excess letdown let exchanger.

9.2.2 System Design and Operation

The Chemical and Volume Control System, shown in Plant Drawings 9321-F-27363 and -27373 [Formerly Figures 9.2-1 and 9.2-2], provides a means for injection of control poison in the form of boric acid solution, chemical additions for corrosion control, and reactor coolant cleanup degasification. This system also adds makeup water to the Reactor Coolant System, reprocesses water letdown from the Reactor Coolant System, and provides seal water injection to the reactor coolant pumps.

Overpressure protective devices are provided for system components whose design pressure and temperature are less than the Reactor Coolant System design limits.

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 post-operational hydrostatic testing to applicable code test pressures. The relief valves are gagged during hydrostatic testing. The relief valves in systems that are hydrostatically tested after refueling operations are set at the system design pressure.

During plant operation, reactor coolant flows through the letdown line from the reactor coolant loop cold leg on the suction side of the pump and is returned to the same cold leg on the discharge side of the pump via a charging line. An alternate charging connection is provided to the hot leg of another loop. An excess letdown line is also provided.

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 each 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 that reduces coolant pressure. The cooled, low pressure water leaves the Reactor Containment and enters the Primary Auxiliary Building where it undergoes a second temperature reduction in the tube side of the non-regenerative 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 filters and enters the volume control tank through a spray nozzle. This would be the preferred normal in-service line-up. However, temporary isolation of the inservice demineralizers not greater than 8 hours to support plant operation is acceptable provided that chemistry is maintained within acceptable parameters.

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 to the Waste Disposal System prior to a cold or refueling shutdown.

During normal operation, the volume control tank gas space will contain approximately 85-96 volume percent H_2 (remainder is water vapor) and the holdup tank gas space from 0 to a maximum of 100 volume percent H_2 the remainder is N_2 and small amount of water vapor). The CVCS volume control tank, CVCS holdup tanks, and associated piping were all designed to accommodate up to 100% hydrogen in the vapor space. Flammable mixtures are precluded by excluding oxygen. The gas analyzer samples these vapor spaces automatically and alarms any sample point where an oxygen concentration of 2% is detected. Exclusion of O_2 is accomplished by leak tight construction of tanks and piping systems and by always maintaining a positive pressure inside these tanks and piping systems.

From the volume control tank, 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.

Demineralizer(s) loaded with Cation 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 B_{10} (n, α) Li_7 reaction.

Boric acid is dissolved in hot water in the batching tank to a concentration of approximately 12 percent by weight. The lower portion of the batching tank is jacketed to permit heating of the batching tank solution with low pressure steam. A transfer pump is used to transfer the batch to the boric acid tanks. 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 normal operation. Electric immersion heaters maintain the temperature of the boric acid tank solution high enough to prevent precipitation.

During plant startup, normal operation, 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 nitrogen 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 essentially zero at the end of the core cycle. A recirculating pump is provided to transfer liquid from one holdup tank to another.

There are three identical CVCS holdup tanks. The liquid contents of one tank are normally being process by the Waste Disposal System while another tank is being filled. The third tank is normally kept empty to provide additional storage capacity when needed. Liquid effluent in the holdup tanks is processed as a batch operation. This liquid is pumped by the gas stripper feed pumps to the waste disposal system where it is processed and transferred to the monitor tanks for sampling.

Valves on all tanks leading to a common header are normally locked open to insure continuous venting. No provision is made to control the percentage of gases evolving from the liquid solution. However, an automatic gas analyzer is provided to monitor the concentrations of oxygen and hydrogen in the cover gas of tanks discharging to the radiogas vent header. Upon indication of a high oxygen level, an alarm sound to alert the operator.

Subsequent handling of the monitor tanks is dependent on the results of sample analysis. Discharge from the monitor tanks may be pumped to the primary water storage tank, returned to the holdup tanks for reprocessing or discharged to the environment with the condenser circulating water when within the allowable activity concentration as discussed in Chapter 11. 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. As the effluent leaves, it is continuously monitored by the Waste Disposal System liquid effluent monitor. If an unexpected increase in radioactivity is sensed, one of the valves in the discharge line to the condenser circulating water closes automatically and an alarm sounds in the Control Room.

The deborating demineralizer(s) can be used intermittently to control Cesium and Lithium during normal plant operation or Boron toward end of core life depending on the type of Resin resident within the demineralizer(s). For Cesium and Lithium control, the affected demineralizer will be loaded with Cation Resin. For Boron control, the affected demineralizer will be loaded with Anion resin. When the deborating demineralizers are in operation, the letdown stream passes from the mixed bed demineralizers and then through the deborating demineralizers and into the volume control tank after passing through the reactor coolant filter.

During plant cooldown, when the residual heat removal loop is operating and the letdown orifices are not in service, a flow path is provided to remove corrosion impurities and fission products. A portion of the flow leaving the residual heat exchangers passes through the non-regenerative heat exchanger, mixed bed demineralizers, reactor coolant filter and volume control tank. The fluid is then pumped, via the charging pump, through the tube side of the regenerative heat exchanger into the Reactor Coolant System.

<u>Expected Operating Conditions</u>

Tables 9.2-2, 9.2-3, and 9.2-5 list the system performance requirements data for individual system components and reactor coolant equilibrium activity concentration. Table 9.2-4 supplements Table 9.2-5.

<u>Reactor Coolant Activity Concentration</u>

The parameters which were used in the calculation of the reactor coolant fission product inventory, including pertinent information concerning the expected coolant cleanup flow rate and demineralizer effectiveness, are presented in Table 9.2-4. The results of the calculations are presented in Table 9.2-5. In these calculations defective fuel rods are assumed to be present at initial core loading and uniformly distributed throughout the core through the use of fission product escape rate coefficients.

The fission product activity in the reactor coolant during operation with small cladding defects* in 1% of the fuel rods was computed using the following differential equations:

For parent nuclides in the coolant,

$$\frac{dN_{wi}}{dt} = D\gamma_i \, Nc_i - \left(\lambda_i + R\eta_i + \frac{B'}{B_o - tB'} \right) N_{wi}$$

for daughter nuclides in the coolant,

$$\frac{dN_{wj}}{dt} = D\gamma_j \, Nc_j - \left(\lambda_j + R\eta_j + \frac{B'}{B_o - tB'} \right) N_{wj} + \lambda_i N_{wi}$$

where:

- N = population of nuclide
- D = fraction of fuel rods having defective cladding
- R = purification flow, coolant system volumes per second
- B_O = initial boron concentration, ppm
- B' = boron concentration reduction rate by feed and bleed, ppm per second
- η = removal efficiency of purification cycle for nuclide
- λ = radioactive decay constant
- ν = escape rate coefficient for diffusion into coolant

*NOTE: fuel rods containing pinholes or fine cracks

Subscript C refers to core
Subscript w refers to coolant
Subscript i refers to parent nuclide
Subscript j refers to daughter nuclide

Tritium is produced in the reactor from ternary fission in the fuel, irradiation of boron in burnable absorbers and irradiation of boron, lithium and deuterium in the coolant. The parameters used in the calculation of tritium production rate are presented in Table 9.2-6.

Reactor Makeup Control

The reactor makeup control consists of a group of instruments arranged to provide a manually pre-selected makeup composition to the charging pump suction header or the volume control tank. The makeup control functions are to maintain desired operating fluid inventory in the volume control tank and to adjust reactor coolant boron concentration for reactivity and shim control.

The boric acid batch integrator is one part of this instrument loop that consists of the flow transmitter, a flow-signal-to-pulse converter and the integrator itself. The integrator "counts" the flow pulses on a non-resettable digital register for all 12% boric acid solution handled by the reactor makeup control system ("automatic makeup" and "borate" modes of operation). Each one-tenth of a gallon is counted and registered. During the "borate" mode of operation, an additional register is active. This additional register counts the same flow pulses, compares them to a preset quantity, and stops the boration when the preset quantity is reached.

The accuracy, in terms of total boric acid addition, is somewhat variable beyond the instrument accuracies because of the tolerance allowed in mixing the nominal 12% solution (i.e., 11-2/3% to 13%). However, the absolute accuracy is unimportant since the first operating check on the reactor coolant boron concentration is the control rod position with final verification achieved through chemical analysis. The operation of the integrator is not intended to keep an accurate inventory of boron in the Reactor Coolant System. It merely provides relative indication as a guide to changes which are made between periodic chemical analysis of samples.

The boric acid blend system is furnished as a convenience for the operator and has no safety functions. The system is not required to operate during or following an accident. In the event of boric acid integrator malfunction, there are no safety related consequences.

Makeup for normal plant leakage is regulated by the reactor makeup control, which is set by the operator to blend water from the primary water storage tank with concentrated boric acid to match the reactor coolant boron concentration.

The makeup system also provides concentrated boric acid or primary water to increase or decrease the boric acid concentration in the Reactor Coolant System. To maintain the reactor coolant volume constant, an equal amount of reactor coolant at existing reactor coolant boric acid concentration is letdown to the holdup tanks. Should the letdown line be out of service during operation, sufficient volume exists in the pressurizer to accept the amount of boric acid necessary for cold shutdown. Additionally, the Reactor Coolant System volume “shrinks” by approximately 25% upon cooling down from the hot operating conditions to cold shutdown.

Makeup water to the Reactor Coolant System is provided by the Chemical and Volume Control System from the following sources:

a) The primary water storage tank, which provides water for dilution when the reactor coolant boron concentration is to be reduced

b) The boric acid tanks, which supply concentrated boric acid solution when reactor coolant boron concentration is to be increased

c) The refueling water storage tank, 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 chemical are necessary.

The reactor makeup control is operated form the Control Room by manually preselecting 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 Reactor Coolant System. The operator can stop the makeup operation at any time in any operating mode by remotely closing the makeup stop valves.

One primary water makeup and one boric acid transfer pump are normally aligned for operation on demand from the reactor makeup control system.

A portion of the high pressure charging flow is injected into the reactor coolant pumps between the thermal barrier 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. Parts of the shaft seal injection flow cools the lower radial bearing, and part passes through the seals and is cooled in the seal water heat exchanger, filtered, and returned to the volume control tank.

Seal water injection to the Reactor Coolant System requires a continuous letdown of reactor coolant to maintain the desired inventory. In addition, bleed and feed or reactor coolant are required for removal of impurities and adjustment of boric acid in the reactor coolant.

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, open the concentrated boric acid control valve and the primary water makeup control valve. The flow controllers then blend the makeup stream according to the present 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 primary water makeup control valve closes, the concentrated boric acid control valve closes and the makeup stop valve to charging pump suction closes.

Dilution

The “dilute” mode of operation permits the addition of a preselected quantity of primary water makeup 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 pumps suction in the closed position, the mode selector switch to “dilute”, the primary water makeup flow controller set point to the desired flow rate, and the primary water makeup batch integrator to the desired quantity. If the dilution flow deviates +5 gpm from the preset flow rate, an alarm indicates the deviation. Makeup water is added to the volume control tank by opening a makeup stop valve. Water in the volume control tank 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 primary water makeup has been added, the batch integrator causes the primary water makeup control valve to close.

For a discussion of the level of borated water in the Safety Injection System accumulators, sampling capabilities, and instrumentation, see Section 6.2.

Boration

The “borate” mode of operation permits the addition of a pre-selected 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 set point to the desired flow rate, and the concentrated boric acid batch integrator to the desired quantity. Opening the makeup stop valve to the charging pumps suction shifts the selected boric acid transfer pump to fast pump speed, and the concentrated boric acid is 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 boric acid transfer pump to return to the slow speed and the concentrated boric acid control valve to close.

The operation of the boric acid transfer pumps and the transfer of concentrated boric acid from the boric acid tanks to the suction of the charging pumps is checked by flow meter FT-110 and/or the boric acid tank levels LT-106, LIT-106 for tanks No. 31 and LT-102, LIT-102 for tank No. 32.

The delivery of concentrated boric acid to the reactor coolant by the charging pumps is checked by flow meter FT-128 in the main charging line and the local flow indicators FI-115, 116, 143, 144 in the seal water supply line.

FT-110 is a magnetic flow meter operating on the “Hall Effect” principle. Instrument power is necessary for its operation. Its signal is both indicated and recorded in the Control Room.

The Boric Acid Storage tanks have been provided with the following instrument systems:

1. LT-102 and LT-106 ∆P transmitters using Nitrogen Bubbler for providing differential pressure that is proportional to Level in the tanks.

2. LE-102 / LIT-102 and LE-106 / LIT-106 Radar level measuring instrument. Signal from these devices are the same as from the ∆P transmitters.

Both systems produce the same level signal for local and CCR indication and therefore the indication and control / alarm function remains unchanged.

Although the preferred instrument will be the Radar level instrument system, either of the above two level instrument systems (i.e., Nitrogen ∆P instrument of the Radar level instrument) can be used for day-to-day operation. The ∆P instrument could also be used at the operator’s discretion.

FT-128 is an electronic transmitter sensing flow by means of the differential pressure generated across an orifice. Instrument power is required for its operation. Two indicators are provided, one in the Control Room and one near the charging pumps.

F1-115, 116, 143 and 144 are the local differential pressure gauges sensing flow by means of the differential pressure generated across an orifice. These indicators are located outside the Containment and are self-actuated (i.e., no power required).

The 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 9.2-2. One set of values is given for the addition of boric acid from a boric acid tank with one transfer 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 at the end of the core life when the moderator temperature coefficient is most negative.

Administrative controls require that if one boric acid tank is out of service, the other boric acid tank must contain sufficient boric acid to bring the plant to a cold shutdown condition. In the event that one boric acid tank would have to be taken out of service, the operator would, prior to taking the one tank out of service, make certain that the remaining tank contained sufficient boric acid to meet cold shutdown requirements.

If unable to comply with administrative controls, the unit could then be placed in cold shutdown condition following normal cooldown procedures as described in the plant operating instructions.

Alarm Functions

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

a) Deviation of primary water makeup flow rate from the control set point
b) Deviation of concentrated boric acid flow rate from the control set point
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.

Charging Pump Control

Three positive displacement, variable speed drive charging pumps are used to supply charging flow to the Reactor Coolant System.

The speed of each pump can be controlled manually or automatically. During normal operation, only one of the three pumps is automatically controlled. During normal operation, only one charging pump is operating and the speed is modulated in accordance with pressurizer level. During load changes, the pressurizer level set point is varied automatically to compensate partially for the expansion or contraction of the reactor coolant associated with the T_{avg} changes. T_{avg} compensates for power changes by varying the pressurizer level set points in conjunction with pressurizer level for charging pump control.

The level set points are varied between 20 and 60 percent of the adjustable range depending on the power level. 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 and a second charging pump is manually started. 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.

Components

A summary of principal component design data is given in Table 9.2-3.

Regenerative Heat Exchanger

The regenerative heat exchanger was designed to recover the heat from the letdown stream by reheating the charging stream during normal operation. This exchanger also limits the temperature 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 (instantaneous changes from initial to the final condition) in shell side fluid temperature from 130 F to 552.2 F during the design life of the unit.

Temperature gradients that exist as a result of the step change in fluid conditions were determined, and the design and stresses that result from this condition were considered. Fatigue analysis results showed a usage factor much less than the 1.0 allowed by the ASME Code. The design considerations that minimize the effects of this service condition were proper design analysis and elimination of unnecessary excess metal thickness in various locations through the heat exchanger.

The in-service inspection program for verifying the equipment condition is discussed in Section 4.5.

Letdown Orifices

One of the three letdown orifices controls the flow of the letdown stream during normal operation and reduces its pressure to a value compatible with the non-regenerative heat exchanger design. Two of the letdown orifices were designed to pass normal letdown flow.

The other orifice was designed to be used in conjunction with one normal letdown flow orifice for maximum purification flow at normal Reactor Coolant System operating pressure. The orifices are placed in and taken out of service by 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 is an austenitic pipe containing a bored corrosion and erosion resistant insert.

Non-Regenerative (Letdown) Heat Exchanger

The non-regenerative 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.

The letdown heat exchanger was designed in accordance with the ASME code requirements given in Table 9.2-1. The design parameters given in Table 9.2-3 and 9.2-4 were based on the following operating parameters:

Operating Parameter	Normal Letdown	Plant Heatup
Shell Side		
Flow (lbs/hr)	203,000	492,000
T_{in} (F)	95	95
T_{out} (F)	125	125
Tube Side		
Flow (lbs/hr)	37,050	59,280
T_{in} (F)	295	371
T_{out} (F)	130	130

The consequences of a tube side rupture in this heat exchanger cannot have greater nuclear safety significance than the release of the contents of the volume control tank. The volume control tank safety significance is addressed in Chapter 14.

A shell break, resulting in loss of component cooling water, will require operator action to shut off the letdown flow after the volume control tank high temperature alarm is activated. An alternate letdown path from the Reactor Coolant System is provided in the event that the normal letdown path is inoperable. When the normal letdown line is not available, the normal purification path is also not in operation.

Therefore, this alternate condition would allow continued power operation for limited periods of time dependent on Reactor Coolant System chemistry and activity.

Monitors R-17a and R-17b* continuously monitor the component cooling loop for radiation indicative of a leak of reactor coolant from the components being cooled by component cooling water.

Mixed Bed Demineralizers

Two flushable mixed bed demineralizers maintain reactor coolant purity. A Lithium-7 or hydrogen form cation resin and a hydroxyl form anion resin are initially charged to the demineralizers. Both forms of resin remove fission and corrosion products, and in addition, the reactor coolant causes the anion resin to be converted to the borate form. The resin bed is designed to reduce the concentration of ionic isotopes in the purification stream, except for cesium, yttrium, and molybdenum, by a minimum factor of 10.

Each demineralizer was 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. Local sample points are provided at the effluent pipe of each demineralizer. The vessels are equipped with a resin retention screen. The resin retention screens were designed to withstand a differential pressure of 25 psi. Each demineralizer has sufficient capacity to enable refueling after operation for one core cycle with one percent defective fuel rods.

Failure of a mixed-bed demineralizer shell is not considered credible due to the following design features:

1) A pressure relief valve is located on the letdown line at a point upstream of the mixed-bed demineralizers. This relief valve has a relief set point of 200 psig

2) The mixed-bed demineralizers have a design pressure of 200 psig

3) The mixed-bed demineralizers were designed to ASME B&PV Code, Section III, Class C, which results in a significant margin of safety in the design.

However, should the very unlikely rupture of the shell occur as postulated, the contents of the demineralizer would be released to the shielded demineralizer room. Any radioactive gases thus introduced into this compartment are directed to the plant ventilation system and plant vent radiation monitors, thus providing indication to the control room personnel of the level of activity released from the plant. For conditions of high activity signaled from the stack gas monitor, additional air flow is provided for dilution purposes to reduce the concentration in the plant vent discharge (Section 9.8). Any liquids or solids released from the ruptured shell would be routed to the building sump tank through the installed floor drains. For this condition, the activity released from the ruptured shell is retained within the Waste Disposal System. Refer to Chapter 11 for a description of the Waste Disposal System and the installed shielding. Recovery procedures would be dependent on the extent of shell rupture but, in all cases, releases from the plant are first sampled and analyzed prior to discharge.

*NOTE: The measurement range of these monitors are given in Section 11.2

Cation Bed Demineralizer

A flushable cation resin bed in the hydrogen form is located downstream of the mixed bed demineralizers and is used intermittently to control the concentration of Lithium-7 that builds up in the coolant from the B^{10} (n, α) Li^7 reaction. The demineralizer also has sufficient capacity to maintain the Cesium-137 concentration in the coolant below 1.0 μc/cc with one percent 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. A local sample point is provided at the demineralizer effluent pipe. The vessel is equipped with a resin retention screen, designed to withstand a 25 psi differential pressure.

For the mixed-bed, cation-bed and deborating demineralizers, an upper limit for allowable pressure drop is, based on manufacturer recommendations, approximately 35 psi. This limit is required to preclude resin bed compaction and bead fracture. A maximum operating differential pressure of approximately 17 psi is specified to allow sufficient margin below this upper limit. A design differential pressure of 25 psi for the screens is considered adequate margin above the maximum operating pressure.

At the beginning of resin life, the pressure drop across the demineralizer at design flow is approximately 13 psi. The design basis was that resin bed fouling, and increased pressure drop occurs very slowly and that resin replacement, due to depletion (low DF) or high radioactivity, is required well before the maximum operating pressure is reached.

The resin retention screen is not a part of any system pressure boundary, its failure would not result in any radioactive release to the environment. The only result of such a failure would be the loss of resin from the demineralizer vessel. It is standard design practice to provide a filter downstream of any demineralizer to collect any resin that might be flushed out of the demineralizer. Therefore, even in the unlikely event of a failure of the resin retention screen, there would be no safety hazard.

Deborating Demineralizers

If desired, one [or both] of the two deborating demineralizer(s) can be loaded with Cation Resin to support control of Cesium and Lithium within the Reactor Coolant System (RCS). As plant operation reaches end of core life condition, the deborating demineralizers may then be loaded with Anion Resin to support removal of boric acid from the RCS fluid.

The demineralizers are provided for use near the end of a core cycle, but can be used at any time. Hydroxyl form ion-exchange resin is used to reduce Reactor Coolant System boron concentration by releasing a hydroxyl ion when a borate ion is absorbed. Facilities are provided for regeneration. When regeneration is no longer feasible, the resin is flushed to the spent resin storage tank.

Each demineralizer is sized to remove the quantity of boric acid that must be removed from the Reactor Coolant System to maintian full power operation near the end of core life should the holdup tanks be full. A local sample point is provided at the demineralizer effluent pipe.

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 demineralized water and resin slurry can be sluiced into the demineralizer by opening the dump valve. The tank, designed to hold approximately one-third the resin volume of one mixed bed demineralizer, is made of austenitic stainless steel.

Reactor Coolant Filter

The reactor coolant filter is located downstream of the deborating demineralizers. This filter is located inside a shielded compartment, and a shield wall is also provided for maintenance personnel during filter cartridge change operations. Filter disassembly and cartridge handling tools were designed to limit personnel exposures to within the limits of 10 CFR 20.

The filter collects resin fines and particulates larger than 25 microns from the letdown stream. The vessel is made of austenitic stainless steel and is provided with connections for draining and venting. Design flow capacity of the filter is equal to the maximum purification flow rate. Disposable synthetic filter elements are used. Indications that determine when the reactor coolant filter should be replaced are: (1) a high pressure differential across the filter, (2) a set time limit after which the filter will be replaced, and (3) a portable radiation monitor reading that shows radiation in excess of established limits.

Volume Control Tank

The volume control tank collects the reactor coolant surge water volume resulting from a change from zero power 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 25 to 35 cc per kg of water (standard conditions).

A spray nozzle is located inside the tank on the inlet line from the reactor coolant filter. This spray nozzle provides intimate contact to equilibrate the gas and liquid phases. A remotely operated vent valve discharging to the Waste Disposal System permits removal of gaseous fission products that are stripped from the reactor coolant and collected 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. The tank is constructed of austenitic stainless steel.

Hydrogen is supplied to the volume control tank for the purpose of maintaining the reactor coolant hydrogen concentration, 25 to 35 cc/kg @ STP. Normal consumption was estimated to be less than 100 scf/day; startup from cold conditions will require 600 to 800 scf.

The source of hydrogen for the volume control tank is the hydrogen supply manifold (discussed in Chapter 11). A pressure reducing valve at the manifold reduces the hydrogen pressure to 100 psig (note that the hydrogen supply header has a relief valve sized to pass full flow from the manifold if the pressure reducing valve fails open) in the supply header. The supply header to the volume control tank is also equipped with a pressure regulator to control downstream pressure to 15 psig.

Leakage of hydrogen from the hydrogen supply manifold and rupture of the manifold piping is prevented by the following design features:

a) Use of class 152 piping (Schedule 40) which has a design pressure of 150 psig at 500 F in a system that operates at 15 to 100 psig and a maximum temperature of 127 F

b) All-welded construction. Pneumatic or hydrostatic tests were performed on the completed system with a thorough examination for leaks. No other special tests were considered necessary

c) All manual valves are bellows sealed and all pressure regulators are self-contained thereby eliminating any packing leaks.

In the unlikely event of any very small leakage of hydrogen into the volume control tank cubicle, buildup of hydrogen gas is prevented by the ventilation system in the building.

Leakage into the hydrogen piping is prevented by the fact that the pressure inside the piping is always greater than atmospheric.

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. These pumps have mechanical packing followed by a leakoff to collect reactor coolant before it can leak to the outside atmosphere. Pump leakage is piped to the drain header for disposal. The pump design prevents lubricating oil from contaminating the charging flow, and the integral discharge valves act as check valves. A recirculation line from the discharge of the charging pumps to the volume control tank is provided. This recirculation line enables warmup running of the pumps against low discharge pressure prior to full load operation.

Warm-up running allows for any air that has accumulated in the pumps to be bled out and all internal gearing and bearings to be fully lubricated.

Each pump was designed to provide the normal charging 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 is operating) and the piping, valve and equipment pressure losses at the design charging flows. The capacity of the three charging pumps permits operation at normal charging line flow with one reactor coolant pump shaft seal operating normally while other reactor coolant pumps are operating with floating ring seal flow.

To reduce the hydraulic pulsations created by the positive displacement pumps, a pulsation stabilizer/separator is installed in the pump suction lines. The discharge line of the No. 33 charging pump is provided with a pulsation dampener to further reduce hydraulic pulsations.

Any one of the three charging pumps can be used to hydrotest the Reactor Coolant System.

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 was determined by the quantity of 35 percent 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.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools reactor coolant letdown flow until the flow rate is equal to the nominal injection rate through the reactor coolant pump labyrinth seal, if letdown through the normal letdown path 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 was designed to withstand 2000 step changes in the tube fluid temperature from 80 F to the cold leg temperature.

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 tube sheet 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 was 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 pump seals are leaking at the maximum design leakage rate.

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 floating ring seals. The vessel is constructed of austenitic stainless steel and is provided with connections for draining and venting. Disposable synthetic filter elements are used.

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.

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 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.

Boric Acid Tanks

The boric acid tank capacities are sized to store sufficient boric acid solution for a cold shutdown shortly after full power operation is achieved following a refueling shutdown. The most reactive RCC is assumed completely withdrawn. One tank supplies boric acid for reactor coolant makeup while recycled solutions form the concentrates holding tank is accumulated in the other tank.

The concentration of boric acid solution in storage is maintained between 11.5 and 13% by weight. Periodic manual sampling and corrective action are provided, if necessary, to ensure that these limits are maintained. Therefore, measured quantities 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.

For cold shutdown purposes, there must be a minimum of 6100 gallons of boric acid solution available in the boric acid tank. The operator is alerted to an approach to the cold shutdown level in either tank by a low level alarm in each tank corresponding to 45% level (about 3800 gallons). It is, however, optional whether the operator chooses to operate normally above the low level alarm in both tanks.

The Boric Acid Storage tanks have been provided with the following instrument systems:

1. LT-102 and LT-106 ∆P transmitters using Nitrogen Bubbler for providing differential pressure that is proportional to Level in the tanks.
2. LE-102/LIT-102 and LE-106/LIT-106 Radar level measuring instrument. Signal from these devices are the same as from the ∆P transmitters.

Both systems produce the same level signal for local and CCR indication and therefore the indication and control / alarm function remains unchanged.

Although the preferred instrument will be the Radar level instrument system, either of the above two level instrument systems (i.e., Nitrogen ∆P instrument of the Radar level instrument) can be used for day-to-day operation. The ∆P instrument could also be used at the operator's discretion.

This indication is provided on the Chemical and Volume Control System supervisory panel in the Control Room.

The low level condition is audibly annunciated in the Control Room with the annunciator drop located on the same panel.

<u>Boric Acid Tank Heaters</u>

Each boric acid tank has two 100% capacity electric heaters that are connected in parallel and controlled from a single controller, a single temperature sensing controller and a single temperature sensing device (TIC-107 in tank No. 31 and TIC-103 in tank No. 32). They are powered by a single source. The heaters maintain the boric acid solution at 170°F (temperature range of 165°F to 175°F), thus ensuring a temperature in excess of the solubility limit (for 20,000 ppm boron this is 130°F). The heaters are shielded in austenitic stainless steel.

TIC-107 (and TIC-103) are "filled system" temperature devices. The instrument mechanism is connected to the thermal bulb in the tank by a capillary. Thermal expansion of the full fluid is converted into a motion that:

1) Controls the local indicating pointer directly
2) Controls an electronic transmitter
3) Controls the contacts used for controlling the tank heaters.

The local indicating pointer operates independently of any power source.

The electronic transmitter provides a signal to a control board indicator and to an alarm unit that provides audible and visual low alarm in the Control Room.

The contacts which control the heaters operate through an internal relay. Loss of instrument power will cause a low alarm and turn the heaters on and will cause the remote indicator to give minimum temperature readings. Since the meter is calibrated from 50 to 200°F, an erroneous reading is obvious to the operator.

Batching Tank

The batching tank is used to prepare solutions of boric acid for filling the boric acid tanks. The tank is provided with a steam jacket for heating the tank contents and an agitator to improve mixing during operations. The steam jacket was designed to heat a batch (approximately 300 gallons) of 12 weight percent boric acid from 32°F to 165°F in 3 hours. Steam is supplied at 250°F and 15 psig. Although no design code applies, the jacket and tank were both fabricated by code-qualified welders (ASME Section IX).

The batching tank was sized to hold one week's makeup supply of boric acid solution for the boric acid tank. The basis for makeup was a reactor coolant leakage of ½ gpm at beginning of core life. The tank may also be used for solution storage, and a transfer system for accumulator makeup is provided from this tank. Refer to Section 6.2 for details of this makeup system. A local sampling point is provided for verifying the solution concentration prior to transferring it to the boric acid tank or for draining the tank. The tank manway is provided with a removable screen to prevent entry of foreign particles.

Boric Acid Transfer Pumps

Two 100% capacity, 2-speed centrifugal pumps are used to circulate or transfer chemical solutions. Redundancy is thus provided for the pumps to permit maintenance during operation of the plant. The pumps circulate boric acid solution through the boric acid tanks at the slow pump speed and inject boric acid into the charging pump suction header at the fast pump speed.

Although one pump is normally used for boric acid batching and transfer and the other for boric acid injection, either pump may function as standby for the other. At fast speed, each pump is capable of delivering boric acid to the charging pump suction header at flow rates that exceed the minimum required boration rate of 132 ppm/hr. All parts in contact with the solutions are austenitic stainless steel or other adequate corrosion-resistant material. When on slow speed, the pump continuously circulates boric acid within the transfer system to provide mixing of the boric acid.

The RWST is a suitable backup source for emergency boration. When two charging pumps are used to transfer borated water from the RWST to the reactor coolant, the boron concentration required for cold shutdown can be reached before xenon decays below its full-power pre-shutdown level.

The transfer pumps are operated either automatically or manually from the 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. Current indicators in the Control Room monitor each pump fast speed motor operation.

The pumps are covered by a well-insulated easily removable heated enclosure. Thermostatically controlled electric strip heaters maintain a temperature well above the solubility limit of the 12% boric acid solution and prevent pump rotor biding form boric acid crystals.

The Boric Acid Transfer are tested quarterly to ensure their ability to function in the emergency boration mode. Flow indicator FI-916 was used prior to implementation of Reference 3 to provide indication of flow during quarterly operability testing. However, Reference 3 valved off the recirculation line (between the Boron Injection Tank and the Boric Acid Tanks), which

contains FI-916, retiring it in place. Instead, Reference 3 installed ultrasonic flow transducer pair FE-197A and FE-197B in the common discharge header of the Boric Acid Transfer Pumps, to permit operability testing to continue. Testing entails recirculating concentrated boric acid from/to the Boric Acid Storage Tanks (BASTs). The flow transducers interface with microprocessor-based portable electronic equipment to provide flow indication during testing.

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. The inner pipe carries the boric acid solution and the outer pipe transports the primary water. These pipes are manufactured and assembled in accordance with ANSI B16.11-1966. The design pressure and temperature are 150 psig and 250 F and the normal operating pressure and temperature are approximately 75 psig and 175 F. All material is austenitic stainless steel. The blender decreases the pipe length required to homogenize the mixture for taking a representative local sample.

Recycle Process

Holdup Tanks

Three holdup tanks contain the radioactive liquid that enters the tanks from the letdown line. The liquid is released from the Reactor Coolant System during startup, shutdowns, load changes and from boron dilution to compensate for burnup. The contents of one tank are normally being processed by the Waste Disposal System while another tank is being filled. The third tank is normally kept empty to provide additional storage capacity when needed. Adequate protection against an internal vacuum condition in the tanks has been verified by the installation of pressure switches and actuation circuitry.

A level indicating system is provided for each CVCS holdup tank. Pneumatic differential pressure transmitters are mounted in a pipe loop external to the tanks.

The total liquid storage sizing basis for the holdup tanks is given in Table 9.2-3. The tanks are constructed of austenitic stainless steel.

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.

Gas Stripper Feed Pumps

The two gas stripper feed pumps transfer water from the holdup tanks to the waste disposal system for processing. These canned centrifugal pumps are constructed of austenitic stainless steel.

Monitor Tanks

Two monitor tanks are provided for storage of water processed by the Waste Disposal System. When one tank is filled, the contents are analyzed and either reprocessed, discharged to the Waste Disposal System, or pumped to the primary water storage tank. Water from the Waste Holdup Tanks can be pumped to the demineralizer system. The demineralizer system consists of a shielded pre-filter/roughing demineralizer, a shielded main demineralizer, and a pump to deliver water to the monitor tanks. These tanks are stainless steel construction and contain a bladder in the air space above the stored liquid. (For a discussion of the Waste Disposal System refer to Section 11.1).

The bladder in the monitor tanks is made of nylon reinforced Buna-N. The expected shelf life (limiting factor) is approximately 10 years.

The most probable causes of failure are the following:

1) Overpressurization of the bladder caused by improper system operation. If air is not removed from the tank before initial filling and kept out of the liquid during operation, as with improper venting, several psi gas pressure under the bladder could conceivably cause failure of the bladder and/or tank.

2) Tank internals causing mechanical failure of the bladder. Improper design of internals could cause interference, and possible tearing of the bladder.

Failure of the bladder for any reason results in air contaminated primary makeup water. Fatigue failure, that is, cracking from folding and unfolding as the water level changes, is not expected. The manufacturer stated that with greater than 10 years experience using this material for tank diaphragms, including use at several nuclear stations, there had been no reported failures from fatigue.

Although it is considered unlikely that the bladder material would fail in a manner that would generate pieces; if this were the case, clogging would occur in the tank outlet line, in a valve, or in the pump suction. However, there would be no significant impact on the plant's safety or operation since there are duplicate tanks and pumps. Clogging is detected by observation of no flow output from the monitor tank pump.

The monitor tank manways are moved periodically to allow for diaphragm inspection.

Monitor Tank Pumps

Two monitor tank pumps discharge water from the monitor tanks. The pumps are sized to empty a monitor tank in approximately 2 hours. The pumps are constructed of austenitic stainless steel.

Primary Water Storage Tank

The primary water storage tank is used to store makeup water that is supplied from the monitor tanks and the water treatment plant. Makeup water form the tank discharges to the suction of the primary water makeup pumps. The tank is stainless steel, operates at atmospheric pressure and has a volume of 165,000 gallons.

The primary water storage tank is insulated and freeze-protected by the auxiliary steam system. The primary water storage tank lines which are exposed to the environment are electrically heat traced to protect them from freezing.

A tank high and low level alarm is provided for the storage tank.

These alarms have no control over level and are used only to inform the operator of tank conditions.

Primary Water Makeup Pumps

Two primary water makeup pumps discharge from either the monitor tanks or the primary water storage 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. One pump operates continuously while the other pump is available for use on an as-needed basis. These pumps are constructed of austenitic stainless steel.

Electrical Heat Tracing

Electrical heat tracing is installed under the insulation on all piping, valves, line-mounted instrumentation, and components normally containing concentrated boric acid solution. The heat tracing was designed to prevent boric acid precipitation due to cooling, by compensating for heat loss.

Exceptions are:

1) Lines that may transport concentrated boric acid but are subsequently flushed with reactor coolant or other liquid of low boric acid concentration during normal operation

2) The boric acid tanks, which are provided with immersion heaters

3) The batching tank, which is provided with a steam jacket

4) Boron injection tank (SIS) which is provided with immersion heaters. References 2 through 5 de-energized these heaters since they are no longer required because the concentrated boric acid contained in the tank heretofore, was replaced by refueling water, with nominally 2,500 ppm boron concentration.

   *NOTE: Technical Specifications Amendment 139 eliminates the requirement to maintain a boron injection tank and related heat tracing. (For a discussion of the Safety Injection System boron injection tank, refer to Section 6.2.)

All boric acid piping is provided with primary and redundant electrical tracings, in conjunction with insulation, to maintain the concentrated solution within a temperature range above the precipitation point and below 212°F when subjected continuously to an ambient temperature of 40°F in still air.

Either tracing (primary or redundant) is capable of supplying enough heat to maintain these temperatures.

The normal source of power for the tracing on the boric acid piping totaling 60 kW, is 480 volt motor control center No. 36A. This motor control center is powered from Diesel Generator No. 33 on loss of offsite power. In addition, all circuits can be manually switched to motor control center No. 36B, which is supplied, from Diesel Generator No. 32 under the same condition. The electric tank heaters in the boric acid tanks are supplied from motor control center No. 37, which can be manually switched to Diesel Generator No. 32 on power loss. The boron injection tank heaters (6 kW each) were supplied from motor control centers No. 36A and No. 36B. Reference 3 de-energized those heat trace circuits on piping, which will no longer convey concentrated boric acid, notably those associated with the Boron Injection Tank (BIT).

Each individual pipe tracing circuit (excluding the boric acid storage tank overflow lines, boric acid storage tank sample lines and the instrumentation tubing associated with the boric acid transfer pumps and boric acid filter) has a local control cabinet containing operating and alarm devices as follows:

1) Operating Thermostat – A line thermostat with remote bulb temperature sensor. The bulb is strapped on the pipe underneath the insulation. This thermostat energizes the tracing when the pipe temperature falls below the low temperature operating set point and de-energizes the circuit on a temperature rise to the high temperature operating set point (these set points are lower than the desired temperature range because of the difference in temperature between the pipe exterior and the fluid inside the pipe).

2) Alarm Thermostat – A two-stage thermostat with remote bulb sensing device strapped on the pipe in the same area as the operating thermostat bulb. It is used to monitor the pipe temperature. A high-high temperature condition (i.e., above the high temperature operating set point but within 212°F) or a low-low temperature condition (i.e., below the low temperature operating set point but above the precipitation point) on any tracing circuit is indicated on a local annunciator panel in the Primary Auxiliary Building. In addition, this condition is alarmed on the main annunciator in the Control Room, as is loss of power to the local annunciator.

3) Test Circuit – A manually operated circuit consisting of test switch, current relay, and indicating light is used to monitor and insure the integrity of the de-energized redundant tracing, and to check the status of the operating tracing. Power for this circuit is supplied from the same source as the heat tracing circuit.

Failure of the operating tracing associated with the piping will result in a decrease in pipe temperature, and will alarm in the Control Room. Redundant tracing can be used to restore affected flow path. If temperature of the affected line decreases to less than 145°F, then this line will be deemed inoperable until either the primary or redundant system is placed in service with line temperature restored. Likewise, failure of any operating device in the local control cabinet will result in an alarm.

Spares are available so that any defective device can be replaced within one hour.

Heat tracing associated with the boric acid storage tank overflow lines are continuously energized and contain a test and alarm device as follows:

1) Alarm Thermostat – A thermostat with remote bulb sensing device strapped to the piping underneath the insulation. It is used to monitor the pipe temperature for a low temperature condition and is indicated on a local annunciator panel in the Primary Auxiliary Building. In addition, this condition is alarmed on the main annunciator in the Control Room.

2) Test Circuit - A manually operated circuit consisting of a test switch and an indication light is used to monitor and ensure the integrity of the de-energized redundant tracing, and to check the status of the operating tracing. Power for this circuit is supplied from the same source as the heat tracing circuit.

Failure of the operating tracing associated with the overflow lines will result in a decrease in pipe temperature, and will alarm in the Control Room. Redundant tracing can be used to restore affected flow path. If temperature of the affected line decreases to less than 145°F, then this line will be deemed inoperable until either the primary or redundant system is placed in service with line temperature restored.

Heat tracing associated with the boric acid storage tank sample lines and the instrumentation tubing associated with the boric acid transfer pumps and boric acid filter is continuously energized and contains an indication light to verify the operation of the heat tracing. The indication light issued to check the status of the operating tracing only.

Failure of the operating tracing associated with the boric acid storage tank sample lines and the instrumentation tubing of the boric acid transfer pumps and the boric acid filter will be detected by plant operators through observation of the heat trace indication lights and the response of the instrumentation. A failure of the heat tracing on any of these sample lines or pressure indicators does not impact any boric acid flow path necessary for the safe shutdown of the reactor. Connection of the redundant tracing can be made once the failure is detected.

<u>Valves</u>

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. For a description of the valves, their identification numbers and a malfunction analysis refer to Table 9.2-7.

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 a spring loaded check valve installed around CH-AOV-204B which is designed to crack open when pressure under the seat exceeds reactor coolant pressure by 75 psi and fully open when pressure under the seat exceeds reactor coolant pressure by 200 psi.

For the active failure analysis of the Safety Injection System, the single failure analysis of the Containment Spray System, and the malfunction analysis of the Chemical and Volume Control System, see Section 6.2 and 6.3 and Table 9.2-7, respectively.

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. Reference 3 permanently de-energized those heat trace circuits on piping and valves that will no longer convey concentrated boric acid, notably those associated with the BIT.

9.2.3 System Design Evaluation

Availability and Reliability

A high degree of functional reliability is assured in the CVCS by providing standby components where performance is vital to safety and by assuring 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 flow.

The electrical equipment of the Chemical and Volume Control System is arranged so that multiple items receive their power from various 480 volt buses (see Plant Drawing 617F644 [Formerly Figure 8.2-4]). Each of the three charging pumps are powered from separate 480 volt buses. The two boric acid transfer pumps are also powered from separate 480 volt buses. One charging pumps and one boric acid transfer pump are capable of meeting cold shutdown requirements shortly after full-power operation. In cases of loss of AC power, a charging power and a boric acid transfer pump can be placed on the emergency diesels, if necessary.

Control of Tritium

The Chemical and Volume Control System is used to control the concentration of tritium in the Reactor Coolant System. 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 dew point 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:

1) Possible plant personnel hazard during access to the Containment. Leakage of reactor coolant during operation with a closed containment causes an accumulation of tritium in the containment atmosphere. It is desirable to limit the accumulation to allow containment access for two hours per week for incore instrumentation maintenance.

2) Possible public hazard due to release of tritium to the environment.

Neither of these considerations is limiting for Indian Point 3.

Leakage Prevention

Quality control of the material and the installation of the Chemical and Volume Control System valves and piping that are designated for radioactive service is provided in order to 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 in each charging pump suction and discharge, on each boric acid 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 that are larger than 2 inches and that 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.

Incident Control

The letdown line and the reactor coolant pumps seal water line penetrate the Reactor Containment. The letdown line contains air-operated valves inside the Reactor Containment and two air-operated valves outside the Reactor Containment that are automatically closed by the containment isolation signal.

The reactor coolant pumps seal water return line contains one motor-operated isolation valve outside the Reactor Containment that is automatically closed by the containment isolation signal.

The four seal water injection lines to the reactor coolant pumps and the normal charging line are inflow lines penetrating the Reactor Containment. Each line contains double check valves inside the Reactor Containment to provide isolation if a break occurs in these lines outside the Reactor Containment.

Malfunction Analysis

To evaluate system safety, failure or malfunctions were assumed concurrent with a Loss-of-Coolant Accident, and the consequences were analyzed and are presented in Table 9.2-7. As a result of this evaluation, it was concluded that proper consideration was 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 Chapter 14.

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 isolated valve, actuation of the valve would limit the release of coolant and assure continued functioning of the normal means of heat dissipation from the core. 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 Chapter 14.

When the reactor is subcritical, i.e., during cold or hot shutdown, refueling and approach to criticality, the relative reactivity status (neutron source multiplication) is continuously monitored and indicated by FB_3 counters and count rate indicators. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate (approximately 480 ppm/hr), is slow enough to give ample time to start a corrective action (boron dilution stop and/or emergency boron injection) to prevent the core form becoming critical. The maximum dilution rate was 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 during refueling when the boron concentration is at the maximum valve and the water volume in the system is at a minimum.

At least two separate and independent flow paths are available for reactor coolant boration, i.e., either of the charging line, or the reactor coolant pumps labyrinths. The malfunction or failure of one component will not result in the inability to borate the Reactor Coolant System. An alternate supply path is always available for emergency boration of the reactor coolant. As backup to the boration system, the operator can align the refueling water storage tank outlet to the suction of the charging pumps.

On loss of seal injection water to the reactor coolant pump seals, seal water flow may be re-established by manually starting a standby charging pump. Even if the seal water injection flow is not re-established, the plant can be operated indefinitely, since the thermal barrier cooler has sufficient capacity to cool the reactor coolant flow that would pass through the thermal barrier cooler and seal leakoff from the pump volute.

Boration during normal operation, to compensate for power changes, is indicated to the operator from two sources: (a) the control rod movement and (b) the flow indicators in the boric acid transfer pump discharge line. When the emergency boration path is used, two indications to the operator are available. The charging line flow indicator indicates boric acid flow since the charging pump suction is aligned to the boric acid transfer pump suction for this mode of operation. The change in boric acid tank level is another indication of boric acid injection.

<u>Galvanic Corrosion</u>

The only types of materials that are in contact with each other in borated water are stainless steels, Inconel, Stellite valve materials and Zircaloy fuel element cladding. These materials have been shown(1) to exhibit only an insignificant degree of galvanic corrosion when coupled to each other.

For example, the galvanic corrosion of Inconel versus type 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 8 days.

Further galvanic corrosion would be trivial since the cell currents at the conclusion of the tests were approaching polarization. Zircaloy versus type 304 stainless steel was shown to polarize

at 180°F in lithiated boric acid solution in less than 8 days with a total galvanic attack of -3.0 mg/dm^2.
Stellite versus type 304 stainless steel was polarized in 7 days at 575°F in lithiated boric acid solution. The galvanic corrosion for this couple was -0.97 mg/dm^2.

As can be seen from the tests, the effects of galvanic corrosion are insignificant to systems containing borated water.

9.2.4 Minimum Operating Conditions

Minimum operating conditions are specified in plant administrative controls to assure adequate boration flow paths are available.

9.2.5 Tests and Inspections

The minimum frequencies for testing, calibrating, and/or checking instrument channels for the Chemical and Volume Control System as well as the Boric Acid Transfer Pumps operability testing are dictated by administrative controls.

References

1) Sammarone, D. G., “The Galvanic Behavior of Materials in Reactor Coolants,” WCAP 1844, August 1961.

2) Revised Feasibility Report for BIT Elimination for Indian Point Unit 3, dated July 1988 (Westinghouse).

3) Modification MOD 86-03-150 SIS, “Elimination of Boron Injection Tank, Phase I. ”

4) Nuclear Safety Evaluation No. NSE 86-03-150 SIS, “Elimination of Boron Injection Tank, Phase I.”

5) Classification CLAS 86-03-150 SIS, “Elimination of Boron Injection Tank, Phase I.”

6) DELETED

TABLE 9.2-1

CHEMICAL AND VOLUME CONTROL SYSTEM CODE REQUIREMENTS

Regenerative heat exchanger	ASME III*, Class C
Non-Regenerative heat exchanger	ASME III, Class C, tube side ASME VIII, shell side
Mixed bed demineralizers	ASME III, Class C
Reactor coolant filter	ASME III, Class C
Volume control tank	ASME III, Class C
Seal water heat exchanger	ASME III, Class C, tube side ASME VIII, shell side
Excess letdown heat exchanger	ASME III, Class C, tube side ASME VIII, shell side
Chemical mixing tank	ASME VIII
Deborating demineralizers	ASME III, Class C
Cation bed demineralizers	ASME III, Class C
Seal water injection filters	ASME III, Class C
Holdup tanks	ASME III, Class C
Boric acid filter	ASME III, Class C
Piping and valves	USAS B31.1**

NOTE:
* ASME III – American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

** USAS B31.1 – Code for Pressure Piping, and special nuclear cases, where applicable.

TABLE 9.2-2

CHEMICAL AND VOLUME CONTROL SYSTEM PERFORMANCE REQUIREMENTS [1]

Deleted	
Seal water return flow rate, gpm	12
Normal letdown flow rate, gpm	75
Maximum letdown flow rate, gpm	120
Normal charging pump flow (one pump), pgm	87
Normal seal injection flow to reactor coolant pumps, gpm	32
Normal charging line flow, gpm	55
Maximum rate of boration with one transfer and one charging pump, ppm/min	24
Equivalent cooldown rate to above rate of boration, °F/min	7.0
Maximum rate of boron dilution (maximum design letdown rate), ppm/hour	300
Two-pump rate of boration (using refueling water), ppm/min	7.4
Equivalent cooldown rate to above rate of boration, °F/min	2.1
Temperature of reactor coolant entering system at full power, °F	555.0
Temperature of coolant return to Reactor Coolant System at full power, °F	505.0
Normal coolant discharge temperature to holdup tanks, °F	127.0
Amount of 12% boric acid solution required to meet cold shutdown requirement shortly after full Power operation, gallons	6100

NOTE:
(1) Volumetric flow rates in gpm are based on 127°F and 15 psig.
Reactor coolant water quality is given in Table 4.2-2.

TABLE 9.2-3

CHEMICAL AND VOLUME CONTROL SYSTEM
PRINCIPAL COMPONENT DATA SUMMARY

	Quantity	Heat Transfer Btu/hr	Design Letdown Flow 1b/hr	Letdown T F	Design Pressure psig, shell/tube	Design Temperature °F, shell/tube
Heat Exchangers						
Regenerative	1	10.35×10^6	37050	249	2485/2735	650/650
Letdown	1	14.74×10^6	37050	253	150/500	200/400
Seal Water	1	2.88×10^6	159,000	14	150/150	200/250
Excess Letdown	1	3.8×10^6	9880	355	150/2485	200/650

	Quantity	Type	Capacity gpm	Head ft or psi	Design Pressure psig	Design Temperature °F
Pumps						
Charging	3	Pos. Displ.	98	2500 psi	3000	250
Boric acid transfer	2	Centrifugal	75	235 ft	150	250
Hold up tank recirculation	1	Centrifugal	500	195 ft	150	150
Primary water makeup	2	Centrifugal	150	235 ft	150	250
Monitor tank	2	Centrifugal	120	200 ft	150	150
Gas stripper feed	2	Canned	25	320 ft	150	150

	Quantity	Type	Volume	Design Pressure psig	Design Temperature °F
Tanks					
Volume control	1	Vertical	400 ft^3	75/15	250
Boric acid	2	Vertical	7000 gal	atmos.	250
Chemical mixing	1	Vertical	5.0 gal	150	250
Batching	1	Jacket Btm.	400 gal	atmos.	250
Holdup	3	Vertical	8,500 ft^3	15	200
Primary water storage	1	Vertical	165,000 gal	atmos.	150
Monitor	2	Diaphragm	10,000 gal	atmos.	150
Resin Fill	1	Open	8 cu. ft.	-	-

	Quantity	Type	Resin Volume ft^3	Flow	Design Pressure psig	Design Temperature °F
Demineralizers						
Mixed Bed	2	Flushable	30	120	200	250
Cation Bed	1	Flushable	12.0	40	200	250
Deborating	2	Flushable	30	120	200	250

TABLE 9.2-4

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT FISSION PRODUCT ACTIVITIES

1.	Core thermal power (maximum calculated), MWt	3280.3
2.	Fraction of fuel containing clad defects	0.01
3.	Reactor coolant liquid volume, ft^3	10,520
4.	Reactor coolant average temperature, F	572
5.	Purification flow rate (normal, minimum), gpm	45
6.	Effective cation demineralizer flow, gpm	4
7.	Volume control tank volumes	
	a. Vapor, ft^3	270
	b. Liquid, ft^3	130
8.	Fission product escape rate coefficients:	
	a. Noble gas isotopes, sec^{-1}	6.5×10^{-8}
	b. Br, I and Cs isotopes, sec^{-1}	1.3×10^{-8}
	c. Te isotopes, sec^{-1}	1.0×10^{-9}
	d. Mo, Te, and Ag isotopes, sec^{-1}	$2.0 \text{ X } 10^{-9}$
	e. Sr and Ba isotopes, sec^{-1}	1.0×10^{-11}
	f. Y, Zr, Nb, Ru, Rh, La, Ce and Pr isotopes, sec^{-1}	1.6×10^{-12}
9.	Mixed bed demineralizer decontamination factors:	
	a. Noble gases and Cs-134, 136, and 137	1.0
	b. All other isotopes	10.0
10.	Cation bed demineralizer decontamination factor for Cs-134, 137, and Rb-86	10.0
11.	Volume control tank noble gas stripping fraction (closed system):	

Isotope	Stripping Fraction
Kr-83m	8.7E-01
Kr-85	1.3E-04
Kr-85m	7.5E-01
Kr-87	9.0E-01
Kr-88	8.1E-01
Kr-89	1.0E-01
Xe-131m	2.9E-02
Xe-133	6.4E-02
Xe-133m	1.4E-01
Xe-135	4.8E-01
Xe-135m	9.7E-01
Xe-137	9.9E-01
Xe-138	9.7E-01

TABLE 9.2-5

REACTOR COOLANT SYSTEM EQUILIBRIUM ACTIVITIES (572°F)

Activation Products	uCi/g
Cr-51	5.50E-03
Mn-54	1.60E-03
Mn-56	2.00E-02
Fe-55	2.00E-03
Fe-59	5.20E-04
Co-58	1.56E-02
Co-60	1.98E-03

Non-Volatile Fission Products (Continuous Full Power Operation)

	uCi/g		uCi/g		uCi/g
Br-83	1.10E-01	Sr-90	4.90E-04	Te-127m	6.48E-03
Br-84	5.10E-02	Sr-91	7.34E-03	Te-127	2.16E-02
Br-85	5.86E-03	Sr-92	1.43E-03	Te-129m	1.96E-02
I-129	1.45E-07	Y-90	1.68E-04	Te-129	2.08E-02
I-130	9.60E-02	Y-91m	4.09E-03	Te-131m	3.80E-02
I-131	4.67E+00	Y-91	9.91E-04	Te-131	1.67E-02
I-132	3.18E+00	Y-92	1.36E-03	Te-132	4.68E-01
I-133	6.28E+00	Y-93	4.87E-04	Te-134	3.28E-02
I-134	6.82E-01	Zr-95	1.09E-03	Ba-137m	4.19E+00
I-135	3.05E+00	Nb-95	1.09E-03	Ba-140	7.14E-03
Cs-134	8.82E+00	Mo-99	1.23E+00	La-140	2.95E-03
Cs-136	5.46E+00	Tc-99m	1.15E+00	Ce-141	1.10E-03
Cs-137	4.43E+00	Ru-103	1.09E-03	Ce-143	7.48E-04
Cs-138	1.08E+00	Rh-103m	1.08E-03	Pr-143	1.07E-03
Rb-86	6.92E-02	Ru-106	5.71E-04	Ce-144	4.92E-04
Rb-88	4.48E+00	Rh-106	5.71E-04	Pr-144	4.92E-04
Rb-89	2.06E-01	Ag-110m	8.70E-03		
Sr-89	7.43E-03	Te-125m	2.01E-03		

Gaseous Fission Products	uCi/g
Kr-83m	5.04E-01
Kr-85m	2.03E+00
Kr-85	1.37E+01
Kr-87	1.30E+00
Kr-88	3.81E+00
Kr-89	1.03E-01
Xe-131m	3.23E+00
Xe-133m	3.52E+00
Xe-133	2.46E+02
Xe-135m	6.25E-01
Xe-135	9.56E+00
Xe-137	1.97E-01
Xe-138	7.14E-01

TABLE 9.2-6

CALCULATED TRITIUM PRODUCTION

BASIC ASSUMPTIONS -

Plant Parameters:

1. Core thermal power, MWt	3216
2. Coolant water volume, ft^3	10,690
3. Core water volume, ft^3	684.5
4. Core water mass (grams)	1.45E+07
5. Plant full power operating times	
a. Initial cycle	60 weeks (14 months)
b. Equilibrium	98 weeks (22.5 months)
6. Boron Concentrations (Peak hot full power equilibrium Xe)	
a. Initial cycle, ppm	890
b. Equilibrium cycle, ppm	1,240
7. Burnable poison boron content (total-all rods), lb	17.6
8. Fraction of tritium in core (ternary fission + burnable boron) diffusing thru clad	
Initial Cycle	0.30 (design value)
Equilibrium Cycle	0.10 (design value)
Equilibrium Cycle	0.02 (expected value)
9. Ternary fission yield	8 x 10-5 atoms/fission

TRITIUM PRODUCTION AND RELEASE VALUES

A. Tritium from Core (Curies)	Initial Cycle	Design Equilibrium Cycle Value	Expected Equilibrium Cycle Value
1. Ternary Fission	11,450	22,280	22,280
2. B^{10}(n, 2 a)T (in poison rods)	800	1,045	1,045
3. B^{10}(n, a) Li^{7} (n, na) T (in poison rods)	1,500	3,200	3,200
4. Release fraction	x0.30	x0.10	x0.02
5. Total release to Coolant	4,125	2,653	531
B. Tritium from Coolant (Curies)			
1. B^{10}(n, 2a)T	1,130	1,013	1,013
2. Li^{7}(n, na) T (limit 3.5 ppm Li, decreasing with Core Burnup for pH control)	8.8	36.6	36.6
3. Li^{6}(n, a) T (purity of Li^{7} = 99.9%)	8.8	286	286
4. Release Fraction (1.0)			
5. Total Release to Coolant	1147.6	1,341	1,341
C. Total Tritium in Coolant (Curies)	5273	3,994	1,872

TABLE 9.2-7

MALFUNCTION ANALYSIS OF CHEMICAL AND VOLUME CONTROL SYSTEM

Component	Failure	Comments and Consequences
1) Letdown line	Rupture in the line inside the Reactor Containment	The remote air-operated valves (LCV-459 & LCV-460) located near the main coolant loop are closed on low pressurizer level to prevent supplementary loss of coolant through the letdown line rupture. The containment isolation valves (201 & 202) in the letdown line outside the Reactor Containment and also the orifice block valves (200A, 200B, & 200C) are automatically closed by the containment isolation signal initiated by the concurrent Loss-of-Coolant Accident. The closure of these valves prevents any leakage of the reactor containment atmosphere outside the Reactor Containment.
2) Normal and alternate charging lines	See above	The check valves 210A & 201B located near the main coolant loops prevent supplementary loss of coolant through the line rupture. The check valve (374) located at the boundary of the Reactor Containment prevents any leakage of the reactor containment atmosphere outside the Reactor Containment.
3) Seal water return line	See above	The motor-operated isolation valve (222) located outside the Containment is manually closed or is automatically closed by the containment isolation signal initiated by the concurrent Loss-of-Coolant Accident. The closure of that valve prevents any leakage of the reactor containment atmosphere outside the Reactor Containment. The safety analyses allow for failure of MOV-222 to close, due to the single failure of its associated electrical Bus on a Containment

Isolation signal. In such a case, the accident analysis assumes leakage of 1.6 gph from Containment into the Primary Auxiliary Building (PAB) for the first 4 hours of the accident. This allows the operators time to manually close one of the other shutoff valves on the same line.

9.3 AUXILIARY COOLANT SYSTEM

9.3.1 Design Basis

The Auxiliary Coolant System consists of three loops, as shown in Plant Drawings 9321-F-27203, and -27513 [Formerly Figures 9.3-1, 9.3-2A and 9.3-2B]; the component cooling loop, the residual heat removal loop, and the spent fuel pit cooling loop.

Performance Objectives

Component Cooling Loop

The component cooling loop was designed to remove residual and sensible heat from the Reactor Coolant System via the residual heat removal loop during plant shutdown, to cool the letdown flow to the Chemical and Volume Control System during power operation, and to provide cooling to dissipate waste heat from various primary plant components.

Active loop components which are relied upon to perform the cooling function are redundant. Redundancy of components in the process cooling loop does not degrade the reliability of any system which the process loop serves.

In order to ensure the long-term functioning of the Component Cooling Water System following a Loss-of-Coolant Accident, the system was designed to accommodate a single failure which may be either active or passive. The Component Cooling Water System is provided with two main headers. The cooling loads are divided between the two headers in such a manner as to ensure that each header is capable of supplying the necessary service to enable continued containment sump and core recirculation following a LOCA. To meet this requirement, one residual heat exchanger, one residual heat removal pump, one recirculation pump, and at least one high head pump are supplied from each header. Isolation valves are furnished to allow each loop to be isolated and operated as an independent component cooling loop. The loop design provides for detection of radioactivity entering the loop from the Reactor Coolant System and also provides for means for isolation.

Residual Heat Removal Loop

The residual heat removal loop was designed to remove residual and sensible heat from the core and to reduce the temperature of the Reactor Coolant System during the second phase of plant cooldown. During the first phase of cooldown, the temperature of the Reactor Coolant System is reduced by transferring heat from the Reactor Coolant System to the Steam and Power Conversion system. All active loop components which are relied upon to perform their function are redundant.

The loop design provides means to detect radioactivity migration to the ultimate heat sink environment and includes provisions which permit adequate action for continued core cooling, when required, in the event that radioactivity limits are exceeded.

The loop design precludes any significant reduction in the overall design reactor shutdown margin when the loop is brought into operation for decay heat removal or for emergency core cooling by recirculation.

The loop design includes provisions which enable periodic hydrostatic testing to applicable code test pressures.

Loop components, whose design pressure and temperature are less than the Reactor Coolant System design limits, are provided with overpressure protection devices and with redundant means of isolation. It is permissible to de-energize these devices with proper administrative controls to prevent inadvertent loss of RHR.

Spent Fuel Pit Cooling Loop

The spent fuel pit cooling loop was designed to remove, from the spent fuel pit, the heat generated by stored spent fuel elements. Loop design incorporates redundant active components. Loop piping is so arranged that failure of any pipeline does not drain the fuel pit below the top of the stored fuel elements. The thermal design basis for the loop provides for a full core offload preceded by storage within the pool of 6 cores.

Design Characteristics

Component Cooling Loop

Two pumps and two component cooling heat exchangers are normally operated to provide cooling water for the components located in the Primary Auxiliary Building, the Fuel Storage Building, and the Containment Building. The water is normally supplied to all components being cooled even though one of the components may be out of service.

Primary makeup water is provided to the CCW Surge Tanks via the Primary Water System by manually opening valves AC-831A and AC-831B as required. The boundary is at the upstream side of these isolation valves.

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

a) Pressure detector on the lines between the component cooling pumps and the component cooling heat exchangers
b) Temperature and flow indicators in the outlet line from the heat exchangers
c) Radiation monitors in the outlet lines from the heat exchangers
d) Temperature indicators on the main inlet lines to the component cooling pumps.

Residual Heat Removal Loop

Two pumps and two residual heat exchangers perform the decay heat cooling functions for the reactor. After the Reactor Coolant System temperature and pressure have been reduced to between 250°F and 350°F and 400 psig, respectively, decay heat cooling is initiated by aligning the pumps to take suction from one reactor hot leg and discharge through the heat exchangers into the reactor cold legs. If only one pump and one heat exchanger are available, reduction of reactor coolant temperature is accomplished at a lower rate.

The equipment utilized for decay heat cooling is also used for emergency core cooling during Loss-of-Coolant Accident conditions as described in Section 6.2

Spent Fuel Pit Cooling Loop

The spent fuel pit pump and heat exchanger will handle the decay heat load from a partial core offload (which is defined as an offload which maintains the Spent Fuel Pit feat load below 17 x 10^6 BTU/hr) while maintaining the spent fuel pit water temperature below 150°F. With a full core discharge the water temperature is maintained below 200°F.

Codes and Classifications

Those portions of the Component Cooling Water System (including the system pumps and heat exchangers) shown on Plant Drawings 9321-F-27203 and -27513 [Formerly Figures 9.3-1, 9.3-2A and 9.3-2B] which serve the residual heat removal pumps, recirculation pumps, residual heat exchangers and the high head safety injection pumps are considered safety related.

For an emergency shutdown situation (non LOCA), component cooling water is also required for the charging pumps. The charging pumps are needed to borate the reactor coolant and shutdown the reactor.

All piping and components of the Auxiliary Coolant System were designed to the applicable codes and standards listed in Table 9.3-4. The component cooling loop water contains a corrosion inhibitor to protect the carbon steel piping. Austenitic stainless steel piping is used in the remaining piping systems which contain borated water without a corrosion inhibitor.

9.3.2 System Design and Operation

Component Cooling Loop

Component cooling is provided for the following heat sources:

a) Residual heat exchangers (Auxiliary Coolant System, ACS)
b) Reactor coolant pumps (Reactor Coolant System, RCS)
c) Non-regenerative heat exchanger (Chemical and Volume Control System CVCS)
d) Excess letdown heat exchanger (CVCS)
e) Seal water heat exchanger (CVCS)
f) Sample heat exchangers (Sampling System and Radiation Monitoring)
g) Waste gas compressors (WDS)
h) Reactor vessel support pads
i) Residual heat removal pumps (ACS)
j) Safety injection pumps (Safety Injection System, SIS)
k) Recirculation pump motors (SIS)
l) Spent fuel pit heat exchanger (ACS)
m) Charging pumps (CVCS).

At the reactor coolant pump, component cooling water removes heat from the bearing oil and the thermal barrier. Since the heat is transferred from the component cooling water to the service water, the component cooling loop serves as an intermediate system between the reactor coolant pump and service water cooling system. This double barrier arrangement reduces the probability of leakage of high pressure, potentially radioactive coolant to the Service Water System.

During normal, full power operation, two component cooling pumps and two component cooling heat exchangers accommodate the heat removal loads. One standby pump provides 50% backup and a heat exchanger provides 100% backup during normal operation provided that service water flow is increased to the operating heat exchanger. Three pumps and two 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 operable, safe shutdown of the plant is not affected; however, the time for cooldown is extended.

Two surge tanks, one for each header, accommodate expansion, contraction and inleakage of water, and ensure a continuous component cooling water supply until a leaking cooling line can be isolated. The tanks are vented to the waste holdup tanks, and a high radiation alarm actuates in the control room in the unlikely event of gross inleakage to the component cooling system.

A non-Chromate based corrosion inhibitor is added to the Component Cooling water system. This inhibitor contains a compound for corrosion control on non-ferrous materials.

The fluorides are kept below 0.15 ppm each, chlorides are kept below 150 ppm, and the makeup is of reactor coolant water quality. Experience at other operating plants has shown that sodium molybdate corrosion inhibitors as a whole are effective in controlling corrosion of carbon, alloy and stainless steel.

Assurance of proper component cooling water chemistry is provided through periodic sampling. The Component Cooling Water is sampled and analyzed at least monthly for gross activity, corrosion inhibitor and pH. The maximum time between analyses is 45 days.

<u>Residual Heat Removal Loop</u>

The residual heat removal loop, as shown in Plant Drawings 9321-F-27203 and -27513 [Formerly Figures 9.3-1 and 9.3-2A] consists of heat exchangers, pumps, piping and the necessary valves and instrumentation. During plant shutdown, coolant flows from the Reactor Coolant System to the residual heat removal pumps, through the tube side of the residual heat exchangers and back to the Reactor Coolant System. The inlet line to the residual heat removal loop starts at the hot leg of one reactor coolant loop and the return line connects to the Safety Injection System piping. The residual heat exchangers are also used to cool the water during the latter phase of Safety Injection System operation. These duties are defined in Chapter 6. The heat loads are transferred by the residual heat exchangers to the component cooling water.

During plant shutdown, the cooldown rate of the reactor coolant and the component cooling water heat exchanger outlet temperature are controlled by regulating the flow through the tube side of the residual heat exchangers. Two remotely operated control valves, downstream of the residual heat exchangers, are used to control flow. Manual throttle valves are used to control component cooling water flow to the residual heat removal heat exchangers and service water flow to the component cooling water heat exchangers.

Double, remotely operated valving is provided to isolate the residual heat removal loop from the Reactor Coolant System. Whenever the reactor coolant system pressure exceeds the design pressure of the residual heat removal loop, separate reactor coolant system pressure channel interlocks will automatically close these valves. In addition, the interlocks also prevent the

valves from opening until a predetermined set point is reached. Two remotely operated valves and one check valve isolate each line to the Reactor Coolant System cold legs from the residual heat removal loop.

Recirculation lines that branch off from the RHR pump discharge piping upstream of the discharge check valves have been installed to ensure a minimum pump recirculation flow of 300 gpm. This recirculation line configuration also eliminates the possibility of “dead heading” an RHR pump during dual pump operation by effectively separating the two pump discharge lines.

Spent Fuel Pit Cooling Loop

The spent fuel pit cooling loop removes residual heat from fuel stored in the spent fuel pit. The loop is normally required to handle the heat load from 88 freshly discharged fuel assemblies from the reactor, but it can safely accommodate the heat load from all of the 1345 assemblies for which there is storage space available.

The spent fuel is placed in the pit during refuelings for long-term storage. The spent fuel pit capacity allows for storage of 6 cores while retaining enough capacity for a full core unload.

The spent fuel pit is located outside the Reactor Containment and is not affected by any Loss-of-Coolant Accident in the Containment. During refueling, the water in the pit is connected to that in the refueling canal by the fuel transfer tube. Only a very small amount of water interchange occurs as fuel assemblies are transferred.

The spent fuel pit cooling loop consists of pumps (main and standby), heat exchanger, filters, demineralizer, piping and associated valves and instrumentation. The operating pump draws water from the pit, circulates it through the heat exchanger and returns it to the pit. Component cooling water cools the heat exchanger. A second pumping system is used to circulate refueling water through the demineralizer and filter for purification. This is permitted under administrative controls (i.e., an operator familiar with the operational restrictions of the RWST Purification System who is in contact with the control room). Redundancy of this equipment is not required because of the large heat capacity of the pit and the slow heat up rate as shown in Table 9.3-3. However, connections are provided for an additional future heat exchanger. In the event of a failure of the spent fuel pump, the standby pump can be put into operation immediately from a local startup push button station.

In addition, reactor cavity filter tie-ins have been added to the spent fuel pit cooling loop to assist in purifying refueling water as it is drained from the reactor cavity to minimize the concentration of particulates in the refueling water storage tank.

The clarity and purity of the spent fuel pit water are maintained by passing approximately 5 percent of the loop flow through a filter and demineralizer. The spent fuel pit pump suction line, which is used to drain water from the pit, penetrates the spent fuel pit wall above the fuel assemblies. The penetration location prevents loss of water as a result of a possible suction line rupture.

Component Cooling Loop Components

Component Cooling Heat Exchangers

The two component cooling heat exchangers are of the shell and straight tube type. Service water circulates through the tubes while component cooling water circulates through the shell side. The outlet water temperature of the component cooling heat exchangers is controlled manually by throttling the service water throttle valves. Design parameters are presented in Table 9.3-1.

Component Cooling Pumps

The three component cooling pumps which circulate component cooling water through the component cooling loop are horizontal, centrifugal units. The pump casings were 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, was substantially overdesigned from a stress level standpoint. Design parameters are presented in Table 9.3-1.

Auxiliary Component Cooling Pumps

All four auxiliary component cooling pumps receive an automatic start signal at the initiation of an event producing a Safety Injection. The four pumps are configured in pairs with each pair supplying cooling water to an individual internal recirculation pump motor cooler. They are located outside Containment and are seismic Class1. The function of these pumps is not needed for the injection phase. However, at least one auxiliary component cooling water pump of each pair is credited during the recirculation phase. For further discussion of the auxiliary component cooling pumps refer to Section 6.2.

A booster pump is also connected to the motor shaft of each safety injection pump to cool the safety injection pump bearings.

Design parameters are presented in Table 9.3-1.

Component Cooling Surge Tanks

The component cooling surge tanks, which accommodate changes in component cooling water volume, were constructed of carbon steel. Design parameters are presented in Table 9.3-1. In addition to piping connections, the tanks have a flanged opening at the top for the addition of the chemical corrosion inhibitor to the component cooling loop.

The internals of the relief valves have been removed to provide a direct path to the Waste Holdup Tanks to prevent a potential overpressurization of the component cooling system.

Component Cooling Valves

The valves used in the component cooling loop are standard commercial valves constructed of carbon steel with bronze or stainless steel trim. Since the component cooling water 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.

Residual Heat Removal Loop Components

Residual Heat Removal Heat Exchangers

The two residual heat removal heat exchangers located within the Containment are of the shell and U-tube type with the tubes welded to the tube sheet. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell side. The tubes and other surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

Residual Heat Removal Pumps

The two residual heat removal pumps are vertical, centrifugal units with special seals to prevent reactor coolant leakage to the atmosphere. All pump parts in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

The residual heat removal pump seal heat exchangers and stuffing boxes are cooled from the Component Cooling Water System. A backup cooling water supply is provided to the 31 RHR pump from the city water system in the event the component cooling water loop is out of service. The 31 residual heat removal pump can be operated for an unlimited length of time, providing the supply of city water is uninterrupted. Based on evaluation of pump and pump seal performance, the residual heat removal pumps can operate without cooling for a limited period of time before they must either be shut down or supplied with cooling flow. The period of time depends on the event involved and the associated pump suction fluid temperature.

Residual Heat Removal Valves

The valves used in the residual heat removal loop are constructed of austenitic stainless steel or equivalent corrosion resistant material.

Stop valves are provided to isolate equipment for maintenance. Throttle valves are provided for remote and manual control of the residual heat exchanger tube side flow. Check valves prevent reverse flow through the residual heat removal pumps.

Two remotely operated series stop valves at the inlet with independent pressure interlocks isolate the residual heat removal loop from the Reactor Coolant System. The residual heat removal loop is isolated from the Reactor Coolant System by one check valve and two remotely operated stop valves on the outlet line. Overpressure in the residual heat removal loop is relieved through relief valves to the pressurizer relief tank. In addition, Technical Specification Section 3.4.12 restricts operation of the SI pumps when the RCS average cold leg temperature is below the OPS enable temperature. These restrictions help to preclude RHR overpressurization.

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.

Manually operated valves have backseats to facilitate repacking. Valve backseats are capable of limiting the stem leakage when used.

Residual Heat Removal Piping

All residual heat removal piping is austenitic stainless steel. The piping is welded with flanged connections at the pumps.

Spent Fuel Pit Loop Components

Spent Fuel Pit Heat Exchanger

The spent fuel pit heat exchanger is of the shell and U-tube type with the tubes welded to the tube sheet. Component cooling water circulates through the shell, and spent fuel pit water circulates through the tubes. The tubes are austenitic stainless steel and the shell is carbon steel.

Spent Fuel Pit Pumps

The spent fuel pit pumps (main and standby) circulate water in the spent fuel pit cooling loop. All wetted surfaces of the pumps are of austenitic stainless steel or equivalent corrosion resistant material. The pumps are operated manually from a local station.

Refueling Water Purification Pump

The refueling water purification pump circulates water in a loop between the Refueling Water Storage Tank and the spent fuel pit demineralizer and filter. All wetted surfaces of the pump are austenitic stainless steel. The pump is operated manually from a local station.

Spent Fuel Pit Filter

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

Spent Fuel Pit Strainer

A stainless steel strainer is located at the inlet of the spent fuel pit loop suction line for removal of relatively large particles which might otherwise clog the spent fuel pit demineralizer.

Spent Fuel Pit Demineralizer

The demineralizer was sized to pass 5% of the loop circulation flow in order to provide adequate purification of the fuel pit water for unrestricted access to the working area, and to maintain optical clarity.

Spent Fuel Pit Skimmer

A skimmer pump, strainer and filter are provided for surface skimming of the spent fuel pit water.

Spent fuel Pit Valves

Manual stop valves are used to isolate equipment and lines, and manual throttle valves provide flow control. Valves in contact with spent fuel pit water are austenitic stainless steel or equivalent corrosion resistant material.

Reactor Cavity Filter System

This system is discussed in Section 9.5.

Spent Fuel Pit Piping

All piping in contact with spent fuel pit water is austenitic stainless steel. The piping is welded except where flanged connections re used at the pump, heat exchanger and filter to facilitate maintenance.

9.3.3 System Evaluation

Availability and Reliability

Component Cooling Loop

For component cooling of the reactor coolant pumps, the excess letdown heat exchanger and the residual heat exchangers inside the Containment, most of the piping, valves, and instrumentation are located outside the primary system concrete shield at an elevation above the water level in the bottom of the Containment at post-accident conditions. (The exceptions are the cooling lines for the reactor coolant pumps and reactor supports, which can be secured following the accident.) In this location the systems in the Containment are protected against credible missiles and from being flooded during post-accident operations. Also, this location provides shielding which allows for maintenance and inspections to be performed during power operation.

Outside the Containment, the residual heat removal pumps, the spent fuel heat exchanger, the component cooling pumps and heat exchangers, and associated valves, piping and instrumentation are maintainable and inspectable during power operation. Replacement of one pump or one heat exchanger is possible while the other units are in service.

Several of the components in the component cooling loop were fabricated from carbon steel. The component cooling water contains a corrosion inhibitor to protect the carbon steel. Welded joints and connections are used except where flanged closures are employed to facilitate maintenance. At least 10% of the component cooling line welds inside the Containment are 100% radiographed. The entire system is seismic Class I and is housed in structures of the same classification, with the exception of the piping and components which serve the Spent Fuel Pit Heat Exchanger. Analysis has demonstrated that the CCW safety function will be retained following natural phenomena. See Sections 16.1 and 16.2. The components were

designed to the codes given in table 9.3-4. In addition, the components are not subjected to any high pressures (see Table 9.3-1) or stresses. Hence, a rupture or failure of the system is very unlikely.

During the recirculation phase following a Loss-of-Coolant Accident, the component cooling water pumps are required to deliver flow to the shell side of the residual heat exchangers. To ensure operability, system flow must be maintained within the maximum flow capacity of a single component cooling water pump. This has been accomplished by establishing fixed component throttle valve positions for power operation. Cooling water flow to the non-regenerative heat exchanger had originally been required to be manually isolated prior to the start of a component cooling water (CCW) pump during the switchover to the LOCA recirculation phase to prevent potential run out of the pump. Subsequent analyses demonstrated that run out of the pump would not occur even with the non-regenerative heat exchanger not isolated. However, isolation of the non-regenerative heat exchanger is still implemented in applicable Emergency Operating Procedures since it results in increased CCW flow to the residual heat exchangers. These pumps are supplied by an emergency power source when required. However the component cooling water pumps are not started during the injection phase combined with a blackout following a LOCA. During this time the only heat removal requirement is for the bearings on the safety injection pumps. Although the auxiliary component cooling water pumps start at the initiation of the injection phase and operate throughout its timeframe, their function is not credited during injection. Recirculation pump motor qualification testing shows that the motors are acceptable without specifically dedicated cooling during the injection phase. Only during the recirculation phase are the auxiliary component cooling water pumps required to operate to support recirculation pump function. See more details on this in Section 6.2.

Since the component cooling pumps are not running during this injection phase, the water volume of the Component Cooling Water System is used as a heat sink. The temperature rise of the fluid is discussed in Section 6.2.2.

Residual Heat Removal Loop

Two pumps and two heat exchangers are utilized to remove residual and sensible heat during plant cooldown. If one of the pumps and/or one of the heat exchangers is not operative, safe operation or safe cooldown of the plant is not affected; however, the time for cooldown is extended. The function of this equipment following a Loss-of-Coolant Accident is discussed in Chapter 6.

Spent Fuel Pit Cooling Loop

This manually controlled loop may be shutdown safely for reasonable time periods, as shown in Table 9.3-3, for maintenance or replacement of malfunctioning components.

The Backup Spent Fuel Pool Cooling system (BSFPCS) has been installed to allow maintenance and repair and operate in parallel with the Normal SFP Cooling System to improve pool conditions during refueling activities. When required the BSFPCS may be operated stand alone to allow maintenance and repair (see Section 9.13). The heat removal capacity of the Backup Spent Fuel Pool Cooling System is not allowed to be credited when calculating the delay time prior to spent fuel offload from the reactor.

Leakage Provisions

Component Cooling Loop

Water leakage from piping, valves, and equipment in this 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 could become contaminated with radioactive water due to a leak in any heat exchanger tube in the Chemical and Volume Control, Sampling, or Auxiliary Coolant Systems, or due to a leak in the thermal barrier cooling coil for the mechanical seal on a reactor coolant pump.

Tube or coil leaks in components being cooled would be detected during normal plant operation by the leak detection system described in Sections 4.2 and 6.7. Such leaks are also detected anytime by the radiation monitors located on the main cooling lines downstream of the component cooling heat exchangers.

Leakage from the component cooling loop can be detected by a falling level in the component cooling surge tanks. The rate of water level fall and the area of the water surface in the tanks permit determination of the leakage rate. To assure accurate determinations, the operator would check that temperatures are stable.

The component which is leaking can be located by sequential isolation or inspection of equipment in the loop. If the leak is in a component cooling water heat exchanger, it can be detected by a radiation monitor which monitors the Service Water Return line from the CCW Heat Exchangers sand the leaking heat exchanger can be isolated for repairs. System heat loads can be accommodated by one heat exchanger provided that service water flow is increased. Overall leakage within the Containment is limited to the value given in the Technical Specifications.

Should a large tube-side-to-shell-side leak develop in a residual heat exchanger, the water level in the component cooling surge tanks would rise, and the operator would be alerted by a high water alarm. The tanks are vented to the waste hold-up tanks, therefore any gross inleakage would overflow to these tanks. Additionally, a radiation alarm would actuate in the control room.

The unlikely severance of a cooling line serving an individual reactor coolant pump cooler would result in substantial leakage of component cooling water. This small bore (1 to 3") piping inside the missile shield (which is not required to be missile protected) will however leak more slowly than larger piping (up to 12"), which is missile protected. For smaller leaks, the water stored in the surge tank after a low level alarm, together with makeup flow, provides ample time for the closure of the valves external to the Containment to isolate the leak before cooling is lost to the essential components in the component cooling loop. For larger leaks, the CCW pumps would be tripped before CCW inventory is depleted and cooling lost to the essential components in the component cooling loop.

Should there be a tube rupture of the reactor coolant pump thermal barrier cooling coil, the water level in the component cooling surge tanks would rise, and the operator would be alerted by a high water alarm. The tanks are vented to the waste hold-up tanks; therefore, any gross inleakage would overflow to these tanks. A high flow signal from the RCP Thermal Barrier HX

CCW return indicating flow switch FIC-625 closes RCP Thermal Barrier HX CCW return isolation valve AC-FCV-625. Piping and components downstream of the thermal barrier, inside containment are designed for RCS pressure.

The relief valves on the cooling water lines downstream from the sample, excess letdown, seal water, non-regenerative, spent fuel pit and residual heat exchangers were 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.

The relief line to the waste hold-up tanks from the component cooling surge tanks 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.

Should a tornado missile cause the rupture of small bore piping on the CCW line to the Spent Fuel Pit Heat Exchanger, low level in the surge tank would alert the control room to the leakage from the line and an operator could be dispatched to investigate. The CCW pumps would be stopped at a low low surge tank level until corrective action allowing refilling of the system from the Primary Water Storage Tank. Isolation is not critical to system function because CCW is not immediately required for plant shutdown and cooling water to the RHE heat exchangers can be interrupted 30 days or more after an accident.

Residual Heat Removal Loop

During reactor operation all equipment of the residual heat removal loop is idle and the associated isolation valves are closed. During the Loss-of-Coolant Accident condition, water from the recirculation sump is recirculated through a loop inside the Containment using the recirculation pumps and the residual heat exchangers. The residual heat removal pumps (which are located outside of the Containment) serve as backup to the internal recirculation pumps.

Each of the two residual heat removal pumps is located in a shielded compartment with a floor drain. Piping conveys the drain water to an external sump which is capable of handling the flow which would result from the failure of a residual heat removal pumps seal. A 50 gpm sump pump discharges to the Waste Holdup Tanks.

The original design of the RHR and HHSI Pump seals incorporated a disaster bushing that would limit the flow to 50 GPM if the seal faces were severely damaged. For GL 2004-02 compliance, an analysis determined the wear of these disaster bushings if debris laden fluid passed through a failed seal. The potentially abrasive nature of the fluid can wear non-metallic disaster bushings over time, whereby the flow out past the damaged seal could eventually exceed 50 GPM. However, this effect is not immediate and as before, actions would be taken to isolate the pump before the 50 GPM flow rate is reached. The Chesterton seal, an alternate type to the original seal, was tested to demonstrate that severely damaged seal faces would result in a flow rate of less than 50 GPM past the seal. Both the original seal designs and later Chesterton model seals are acceptable and may be in use in the HHSI and RHR pumps.

Spent Fuel Pit Cooling Loop

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 the contaminants from the water.

The probability of inadvertently draining the water from the cooling loop of the spent fuel pit is exceedingly low. The only means is through actions such as opening a valve on the cooling line and leaving it open when the pump is operating. In the unlikely event of the cooling loop of the spent fuel pit being drained, the spent fuel storage pit itself cannot be drained and no spent fuel is uncovered since the spent fuel pit cooling connections enter near the top of the pit. Temperature and level indicators in the spent fuel pit warn the operator of the loss of cooling.

No movement of irradiated fuel in the reactor shall be made until the reactor has been subcritical for at least 84 hours. The 84 hour decay time and the 23 feet of water above the top of the reactor pressure vessel flange are consistent with the assumptions used in the dose calculation for the fuel handling accident.

As irradiated fuel is added to the Spent Fuel Pit, the pit bulk temperature will begin to increase with increasing decay heat load. The thermal-hydraulic analysis for the maximum density Spent Fuel Pit racks has determined that, for a decay heat load no greater than 17.6×10^6 BTU/hr, the Spent Fuel Pit bulk temperature shall not exceed 150°F with the Spent Fuel Pit Cooling System in service. During normal Spent Fuel Pit operation, the decay heat load is below 17.6×10^6 BTU/hr. The maximum allowable decay heat load is that which will ensure that the bulk temperature does not exceed 200°F with the Spent Fuel Pit Cooling System in service.

The thermal-hydraulic analysis for the maximum density Spent Fuel Pit racks has established default cooling times for core offload. A full-core offload completed no earlier than 254 hours subcritical will ensure that the pit bulk temperature will not exceed 200°F.

This time limit may be relaxed for any core offload, provided that it can be shown that the heat load in the Spent Fuel Pit, bulk temperature shall not exceed 200°F at any time. This must be proven in a formal calculation, which shall determine Spent Fuel Pit heat load in accordance with the requirements of Branch Technical Position ASB 9-2 of the USNRC Standard Review Plan. (1)

In summary, two (2) criteria must be met before spent fuel can be discharged to the SFP:

- Spent fuel cannot be discharged to the SFP until at least 84 hours after shutdown to satisfy the dose assumptions of the design basis fuel handling accident; and
- An additional delay time prior to spent fuel discharge is administratively controlled by operating procedures to ensure that the total spent fuel pool heat load is within the capacity of the spent fuel cooling loop to maintain SFP bulk temperature no greater than 200°F. this is a variable time limit primarily dependent upon SW temperature.

It should be noted that the use of auxiliary Spent Fuel Pit cooling can significantly reduce pit temperature. Although auxiliary cooling is not credited in the Spent Fuel Pit thermal-hydraulic analysis, it is a useful tool for maintaining working conditions in the vicinity of the Spent Fuel Pit and for keeping the pit temperature below 150ºF during partial and full-core offloads.

Spent Fuel Pit bulk temperature is calculated assuming one of the two Spent Fuel Pit coolant pumps is in service. No credit is taken for heat loss to the pool liner, slab, concrete walls or the air, nor is credit taken for heat loss through evaporation.

During normal Spent Fuel Pit operation, with an equilibrium bulk temperature of 150°F, a complete loss of cooling will result in a temperature increase of 7.30°F/hr. In this case, 8.5 hours are available to reestablish pool cooling before boiling occurs. For a partial offload of 88 assemblies, pool boiling occurs in 4.9 hours, presuming an initial pit temperature of 150°F. For the full-core offload case, with an equilibrium bulk temperature of 200°F with no heat removal by installed or supplemental cooling capability, the time for SFP water to rise from 200°F to 212°F is at least 33 minutes.

The results of the Spent Fuel Pit thermal-hydraulic analysis are shown on Table 9.3-3.

The primary source of makeup water to the spent fuel pool is the Primary Water Storage Tank. Additional makeup water may be provided from the Refueling Water Storage tank or the city water supply.

Incident Control

Component Cooling Loop

Should the break occur outside the Containment the leak could either be isolated by valving or the broken line could be repaired, depending on the location in the loop at which the break occurred.

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

If the break occurs inside the Containment, on a cooling water line to a reactor coolant pump, the leak can be isolated. The cooling water supply line to the reactor coolant pumps contains a check valve inside containment and remotely operated valves outside the containment wall. Each return line has remote operated valves outside the containment wall. The cooling water supply line to the excess letdown heat exchanger contains a check valve inside the containment wall and both supply and return lines have automatic isolation valves outside the containment wall which are open during normal operation.

Should the break occur inside the Containment, and the leak cannot be isolated locally, the main header feeding the break is isolated. Component cooling is still provided in the remainder of the system by the second main header. The cooling loads are divided between the two headers in such a manner as to ensure continued containment sump and core recirculation following a Loss-of-Coolant Accident.

Flow indication is provided on the component cooling return lines from the safety injection and residual heat removal pumps. Each of the component cooling supply lines to the residual heat exchangers contains a check valve; each return line has a remotely operated valve (normally closed) outside the containment wall. 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.

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.

A failure of pumps, heat exchangers, and valves for the Component Cooling Water System is presented in Table 9.3-5.

Residual Heat Removal Loop

The residual heat removal loop is connected to one reactor hot leg on the suction side and to the reaction cold legs on the discharge side. On the suction side, the connection is through two electric, motor-operated gate valves in series. Each valve is independently interlocked with reactor coolant system pressure. On the discharge side, the connection is through two electric, motor-operated gate valves and one check valve for each reactor cold leg. The motor-operated valves are open whenever the reactor is in MODES 1, 2, or 3, in accordance with Technical Specification requirements.

Spent Fuel Pit Cooling Loop

The most serious failure of this loop is complete loss of water in the storage pool. To protect against this possibility, the spent fuel storage pool cooling connections enter near the water level so that the pool cannot be either gravity-drained or inadvertently drained. For this same reason, care was exercised in the design and installation of the fuel transfer tube. The water in the spent fuel pit below the cooling loop connections could be removed by using a portable pump.

9.3.4 Minimum Operating Conditions

Minimum operating conditions for the Auxiliary Coolant System are given in the Technical Specifications.

9.3.5 Tests and Inspections

The residual heat removal pumps flow instrument channels are calibrated during each refueling operation.

The check valves on the lines from the residual heat removal loop to the cold legs of the Reactor Coolant System are leak tested every time the plant is shutdown and the reactor coolant system has been depressurized to 700 psig or less. This test is also performed following valve maintenance, repair or other work which could unseat these check valves.

The portion of the Residual Heat Removal System outside containment shall be tested for leakage at least every 24 months as follows:

1. The portion of the Residual Heat Removal System that is outside the containment shall be tested either by use in normal operation of hydrostatically tested at 350 psig.

2. The piping between the residual heat removal pumps suction and the containment isolation valves in the residual heat removal pump suction line from the containment sump shall be hydrostatically tested at no less than 100 psig.

3. Visual inspection shall be made for excessive leakage during these tests from components of the system. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

4. The maximum allowable leakage from the Residual Heat Removal System components and Safety Injection System components, located outside of the containment and used during the recirculation phase of design basis accident, shall not exceed two gallons per hour.

5. Repairs of isolation shall be made as required to maintain leakage within the acceptance criterion.

Reference

1) Branch Technical Position ASB 9-2, USNRC Standard Review Plan, “Residual Decay Energy for Light-Water Reactors for Long-Term Cooling”.

TABLE 9.3-1

COMPONENT COOLING LOOP COMPONENT DATA

Component Cooling Pumps	
Quantity	3
Type	Horizontal, Centrifugal
Rated capacity (each), gpm	3600
Rated head, ft H_20	220
Maximum flow rate, gpm	5817 analyzed; 6050 runout
Motor horsepower, hp	250
Casing Material	Cast iron
Design pressure, psig	150
Design temperature, °F	200
Component Cooling Heat Exchangers	
Quantity	2
Type	Vertical shell and straight tube
Heat exchanged, Btu/hr	31.4×10^6
Fouled transfer rate, Btu/hr-°F-ft^2	298
Clean transfer rate, Btu/hr-°F-ft^2	600
Surface area, ft^2	8569
Overall heat transfer coefficient(1), BTU/hr-°F	2.55×10^6

(1) Fouled transfer rate multiplied by the design surface area.

TABLE 9.3-1
(Cont.)

COMPONENT COOLING LOOP COMPONENT DATA

Component Cooling Heat Exchangers Design Conditions*		
Parameter	Tube Side	Shell Side
Pressure, psig	150	150
Temperature, °F	200	200
Flow, 1b/hr	4.55×10^6	2.66×10^6
Inlet Temperature, °F	95	116.9
Outlet Temperature, °F	101.9	105
Material	Admiralty	Carbon steel

Component Cooling Surge Tanks	
Quantity	2
Volume, gal.	2000
Normal water volume, gal.	1000
Design pressure, psig	100
Design temperature, °F	200
Construction material	Carbon steel

Auxiliary Component Cooling Pumps (Recirculation Pump Motor Coolers)	
Quantity	4
Type	Vertical, centrifugal
Rated capacity, gpm	80
Rated head, ft H_20	100
Motor Horsepower, hp	5
Casing material	Cast iron
Design pressure, psig	150
Design temperature, °F	200

*NOTE: Excluding the RCP thermal barrier heat exchanger.

TABLE 9.3-1
(Cont.)

COMPONENT COOLING LOOP COMPONENT DATA

Auxiliary Component Cooling Pumps (SI Pump Coolers)	
Quantity	3
Design pressure, psig	150
Design temperature, °F	200
Design flow rate, gpm	40
Design head, ft	102
Type	Centrifugal
Component Cooling Loop Valves	
Design pressure, psig	150
Design temperature, °F	200
Component Cooling Loop Piping	
Design Pressure, psig	150
Design Temperature, °F	500
Component Cooling RCP Thermal Barrier Heat Exchanger	
Design Pressure, psig	2500
Design Temperature, °F	650

TABLE 9.3-2

RESIDUAL HEAT REMOVAL LOOP COMPONENT DATA

Cooldown parameter	Normal	Appendix R
Reactor core power, MWt	3216	3216
Component cooling water heat exchanger performance		
Number	2	2
Heat transfer coefficient(1) (each), Btu/hr-°F	2.43×10^{6}	1.77×10^{6}(2)(3)
Tube side inlet temperature, °F	95	95
Tube side flow (each), lb/hr	4.55×10^{6}	1.42×10^{6}
Shell side flow (each, lb/hr	2.66×10^{6}	1.15×10^{6}
Residual heat removal heat exchanger performance		
Number	2	1
Heat transfer coefficient(1) (each, Btu./hr-°F	1.16×10^{6}	1.07×10^{6}(2)
Tube side flow (each, lb/hr	1.44×10^{6}	1.44×10^{6}
Shell side flow (each), lb/hr	2.46×10^{6}	1.74×10^{6}
Auxiliary heat loads at 4 hours after plant shutdown, Btu/hr	27.3×10^{6}	18.9×10^{6}
Auxiliary heat loads at 20 hours after plant shutdown, Btu/hr	20.2×10^{6}	18.4×10^{6}
Reactor coolant temperature at initiation of cooling loop, °F	350	350
Maximum reactor coolant cooldown rate, °F/hr	50	50
Maximum component cooling water supply temperature, °F	120	125

NOTE:
(1) Surface area reduced by 5% to allow for tube plugging.
(2) Fouled transfer rate reduced for flow less than design. (“Corrected UA”)
(3) Other Appendix R cases evaluated had different “Corrected” heat transfer coefficients.

TABLE 9.3-2
(Cont.)

RESIDUAL HEAT REMOVAL LOOP COMPONENT DATA

	Normal	Appendix R
Time after plant shutdown cooling loop initiated, hr	5	29
Time after plant shutdown to reach cold shutdown condition (200°F), hr	21	72
Time after plant shutdown to reach refueling condition (140°F), hr	105	
Refueling water storage temperature, °F	Ambient	
Decay heat generation at 20 hours after plant shutdown, Btu/hr	$71.2 \times 10^{(6)}$	
Reactor cavity fill time, hr	1	
H3BO3 concentration in refueling water storage tank, ppm boron	2000-2600	
Residual Heat Removal Pumps		
Refer to Table 6.2-5		
Residual Heat Removal Heat Exchangers		
Refer to Table 6.2-6		
Residual Heat Removal Loop Piping and Valves		
1. Isolated loop		
Design pressure, psig	600	
Design temperature, °F	400	
2. Loop Isolation		
Design pressure, psig	2485	
Design temperature, °F	650	

TABLE 9.3-3

SPENT FUEL COOLING LOOP COMPONENT DATA

Spent Fuel Pit Heat Exchanger

Number	1
Type	Shell and U-tube
Heat Exchanged, Btu/hr	7.96×10^6
Fouled transfer rate, Btu/hr-°F-ft^2	310
Clean transfer rate, Btu/hr-°F-ft^2	468
Surface area, ft^2	2000
Overall heat transfer coefficient(1), Btu/hr-°F	0.62×10^6

Design Conditions:

Parameter	Tube Side	Shell Side
Pressure, psig	150	150
Temperature, °F	200	200
Flow, lb/hr	1.1×10^6	1.4×10^6
Inlet Temperature, °F	120	100
Outlet Temperature, °F	112.8	105.7
Material	Stainless steel	Carbon steel

System Cooling Capability

Spent fuel pit heat load, Btu/hr

Normal SFP Operation	$<17.6 \times 10^6$
Partial Offload (88/193 core), 84 hr discharge	30×10^6
Full Core Offload (nominal)	35×10^6

Component cooling water heat exchanger performance

Number	2
Heat Transfer coefficient (each), Btu/hr-°F	2.43×10^6

NOTE:
(1) Fouled transfer rate multiplied by the design surface area.

TABLE 9.3-3
(Cont.)

SPENT FUEL COOLING LOOP COMPONENT DATA

Tube side inlet temperature, °F	95
Tube side flow (each), lb/hr	4.55×10^6
Shell side flow (each), lb/hr	2.66×10^6
Spent fuel pit heat exchanger performance	
Heat transfer coefficient(1), Btu/hr-°F	0.59×10^6
Tube side flow, lb/hr	1.1×10^6
Shell side flow, lb/hr	1.4×10^6
Pit water inertia, no heat removal(2) (3)	
Time to heat from 150°F to 212°F, normal SFP operation, hr	8.5
Time to heat from 150°F to 212°F, 88 spent fuel assembly discharged, hr	4.9
Time to heat from 200°F to 212°F, full core, min	>33
Spent Fuel Pit Skimmer Pump	
Quantity	1
Type	Horizontal, Centrifugal
Rated Capacity, gpm	100
Rated head, ft $H^2 0$	50
Design pressure, psig	50
Design temperature, °F	200
Casing Material	Stainless steel
Refueling Water Purification Pump	
Quantity	1
Type	Horizontal, Centrifugal
Rated capacity, gpm	100

NOTE:
(1) Overall heat transfer coefficient reduced by 5% to allow for tube plugging
(2) The initial temperatures are maximum calculated equilibrium, temperatures with cooling system operating.
(3) If SFP level is presumed to be reduced to 88' 0" due to a hypothetical loss of SFP volume, the heat up times are reduced by about 19%, to 6.9 hours, and 26.7 minutes, respectively.

TABLE 9.3-3
(Cont.)

SPENT FUEL COOLING LOOP COMPONENT DATA

Rated head, ft H_20	150
Design pressure, psig	150
Design temperature, F	200
Casing Material	Stainless steel
Spent Fuel Pit Cooling Loop Piping and Valves	
Design pressure, psig	150
Design temperature, F	200
Spent Fuel Pit Skimmer Loop Piping and Valves	
Design pressure, psig	100
Design temperature, F	200
Refueling Water Purification Loop Piping and Valves	
Design pressure, psig	150
Design temperature, F	200
Spent Fuel Pit Pump	
Quantity	2
Type	Horizontal, Centrifugal
Casing Material	Stainless steel
Rated capacity, gpm	2300
Rated head, ft H_20	125
Design pressure, psig	150

TABLE 9.3-3
(Cont.)

SPENT FUEL COOLING LOOP COMPONENT DATA

Design temperature, F	200
Motor horsepower	100
Spent Fuel Storage Pool	
Volume, ft^3	37,300
Boron concentration, ppm boron	2000 to 2500
Spent Fuel Pit Filter	
Quantity	1
Internal design pressure of housing, psig	200
Design temperature, F	250
Rated flow, gpm	100
Maximum differential pressure across filter element at rated flow (clean cartridge), psi	5
Maximum differential pressure across the filter element prior to removing, psi	20
Filtration requirement	98% retention of particles down to 5 micron
Spent Fuel Pit Strainer	
Quantity	1
Rated flow, gpm	2300
Maximum differential pressure across the strainer element at rated floor (clean), psi	1
Perforation, inches	Approximately 0.2”

TABLE 9.3-3
(Cont.)

SPENT FUEL COOLING LOOP COMPONENT DATA

Spent Fuel Pit Demineralizer	
Type	Flushable
Design pressure, psig	200
Design temperature, F	250
Flow rate, gpm	100
Resin volume, ft^3	15 to 25
Spent Fuel Pit Skimmers	
Quantity	2
Flow per unit, gpm	50
Vertical fluctuation range: Floating, inch	4
Manual adjustment, feet	2
Spent Fuel Pit Skimmer Strainer	
Quantity	1
Type	Basket
Rated flow, gpm	100
Design pressure, psig	50
Design temperature, F	200
Maximum differential pressure across strainer at rated flow, psi	1
Perforation, inch	1/8

TABLE 9.3-3
(Cont.)

SPENT FUEL COOLING LOOP COMPONENT DATA

Spent Fuel Pit Skimmer Filter

Quantity	1
Type	Replaceable
Internal design pressure, psig	200
Design temperature, F	250
Rated flow, gpm	100
Maximum differential pressure across filter at rated flow (clean), psi	5
Maximum differential pressure across filter prior to replacement, psi	20
Filtration requirement	98% retention of particles above 5 microns

TABLE 9.3-4

AUXILIARY COOLANT SYSTEM CODE REQUIREMENTS

Component	Code
Component cooling heat exchangers	ASME VIII
Component cooling surge tank	ASME VIII
Component cooling loop piping and valves	USAS B31.1
Residual heat exchangers	ASME III, Class C, tube side
	ASME VIII, shell side
Residual heat removal piping and valves	USAS B31.1
Spent fuel pit filter	ASME III, Class C
Spent fuel heat exchanger	ASME III, Class C, tube side
	ASME VIII, shell side
Spent fuel pit loop piping and valves	USAS B31.1

TABLE 9.3-5

FAILURE ANALYSIS OF PUMPS, HEAT EXCHANGERS, AND VALVES

Components	Malfunction	Comments and Consequences
1. Component cooling water pump	Rupture of a pump casing	The casing and shell are designed for 150 psi and 200°F which exceed maximum operating conditions. Pump is inspectable and protected against credible missiles. Rupture is not considered credible. However, each unit is isolatable. One of the three pumps is capable of meeting system flow requirements.
2. Component cooling water pumps	Pump fails to start	One operating pump supplies sufficient water for emergency cooling.
3. Component cooling water pumps	Manual valve on a pump suction line closed	This is prevented by pre-startup and operational checks. Further, during normal operation, each pump is checked on a periodic basis which would show if a valve is closed.
4. Component cooling water valve	Normally open valve	The valve is checked open during periodic operation of the pumps during normal operation.
5. Component cooling heat exchanger	Tube or shell rupture	Rupture is considered improbable because of the low operating pressures. Each unit is isolatable. Second unit can carry total heat load for normal operation provided service water flow is increased.
6. Demineralized water makeup line check valve	Sticks open	The check valve is backed up by the manually operated valve. Manual valve is normally closed.
7. Component cooling heat exchanger vent or drain	Left open	This is prevented by pre-startup 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.
8. Component cooling water inlet valve to residual heat exchanger	Fails to open	There is one valve on each outlet line from each heat exchanger: One heat exchanger remains in service and provides adequate heat removal during long tern recirculation. During normal operation the cooldown time is extended.

9.4 SAMPLING SYSTEM

9.4.1 Sampling During Normal Operation

9.4.1.1 Design Basis

Performance Requirements

This system provides samples for laboratory analysis to evaluate reactor coolant, feedwater, steam systems, and other reactor auxiliary systems chemistry during normal operation. This system is normally isolated at the containment boundary.

Sampling system discharge flows are limited under normal and anticipated fault conditions (malfunctions or failure) to preclude any fission product releases beyond the limits of 10 CFR 20. The Sampling System can also be used for removing fluid from any individual SI Accumulator as described in Section 6.2.

Design Characteristics

The system is capable of obtaining reactor coolant samples during reactor operation and during cooldown. Access is not required to the containment for operation for the sampling system.

Sampling of other process coolants, such as from tanks in the Waste Disposal System, is accomplished locally. Equipment for sampling secondary and non-radioactive fluids are separated from the equipment provided for reactor coolant samples. 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 that originate inside the reactor containment, and low temperature – low pressure samples from the Chemical and Volume Control and Auxiliary Coolant Systems.

High Pressure – High Temperature Samples

A sample connection is provided from each of the following:

a) The pressurizer steam space – RCS
b) The pressurizer liquid space – RCS
c) Hot legs of loops 1 and 3 – RCS
d) Secondary steam blowdown from each steam generator – SGBDS

Low Pressure – Low Temperature Samples

A sample connection is provided from each of the following:

a) The mixed bed demineralizer inlet header – CVCS
b) The mixed bed demineralizer outlet header – CVCS
c) The residual heat removal loop, just downstream of the heat exchangers – ACS
d) The volume control tank gas space – CVCS
e) The accumulators – SIS
f) The recirculation pump discharge – SIS / Accident

Operating Temperatures

The high pressure, high temperature samples and the residual heat removal loop samples leaving the sample heat exchangers are held to a temperature of 130F to minimize the generation of radioactive aerosols.

Codes and Standards

System component code requirements are given in Table 9.4-1.

9.4.1.2 System Design and Operation

The Sampling System, shown in Plant Drawings 9321-F-27453 [Formerly Figure 9.4-1], provides the representative samples for laboratory analysis. Analysis results provide guidance in the operation of the Reactor Coolant, Auxiliary Coolant, Steam and Chemical and Volume Control Systems. Analyses show both chemical and radiochemical conditions. Typical information obtained includes: reactor coolant boron and chloride 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. Samples can be withdrawn under conditions ranging from full power to cold shutdown.

Reactor coolant liquid and steam sample 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 auxiliary building in the sampling room. The delay coil and sample lines with remotely operated valves are located inside the Reactor Containment.

Reactor coolant hot leg liquid, pressurizer liquid and pressurizer steam samples originating inside the Reactor Containment flow through separate sample lines to the sampling room. The hot leg sample lines are of sufficient length to provide at least a 40-second transit time within the containment and an additional 20 seconds from the Containment to the sample hood to minimize personnel exposure. Each of these connections to the Reactor Coolant System has a remote operated isolation valve located close to the sample source. The samples are passed through the Reactor Containment to the auxiliary building, and into the sampling room, where they are cooled (pressurizer steam samples are condensed and cooled) in the sample heat exchangers. The sample stream pressure is reduced by a manual throttling valve located upstream of the quick connection for a temporary 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. Pressurized samples are collected by purging a portion of the sample stream through the temporary sample pressure vessel to the sample sink. After sufficient purging, the sample pressure vessel is isolated and analyzed.

Depressurized 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 two remotely operated, normally closed isolation valves located close to the sample source outside the Containment. The sample line from this source is connected into the sample line coming from the hot leg at a point ahead of the sample's heat exchanger. Samples from this source can be collected either in the sample vessels or at the sample sink as with hot leg 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. The sample line from the gas space of the volume control tank delivers gas samples to the volume control tank sample pressure vessel in the sampling room. Purge flow for these samples is discharged to the vent header in the Waste Disposal System.

Samples of the steam generator liquid are obtained from the blowdown lines. These sample lines are routed by separate lines from the steam generator into the sample room. These lines are missile protected within containment and are equipped with a remote operated valve.

These blowdown lines are then routed on to the plant blowdown flash tank. The sample lines are taken off at an intermediate point inside the Containment and routed to the sample room where the liquid is cooled and the pressure reduced. The sample lines are equipped with remotely operated isolation valves. Each individual sample is then split into two routes: one goes to the sample sink to provide periodic samples for chemical analysis, the second goes to a conductivity cell, a radiation monitor and then to the blowdown flash tank. This second line handles a continuous flow for a constant reading of conductivity and a constant monitoring for radiation.

An interconnection between the 4" SGBD lines and the 3/8" SGBD Sample tubing was installed during RO9. This interconnection allows for the 4" SGBD lines to be filled from the Sample System. This modification was installed to prevent water hammer during SGBD restart above cold shutdown.

Liquid samples originating at each of the accumulators in the Safety Injection System run directly to the sample sink; no heat exchanger or sample vessel required. Samples are obtained by sampling the flow at the sample sink. These sample lines have air operated isolation valves located close to the accumulators.

The Sampling System can also be used for removing fluid from any individual SI Accumulator in a process similar to sample line purging. Refer to Section 6.2 for information on the potential reasons and frequencies of this operation.

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.

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.

Components

A summary of principal component data is given in Table 9.4-2.

Sample Heat Exchangers

Ten sample heat exchangers reduce the temperature of samples from the pressurizer steam space, the pressurizer liquid space, each steam generator and the reactor coolant system liquid to 130°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.

Delay Coil

The high pressure reactor coolant sample line is designed to be of sufficient length to provide at least a 40 seconds sample transit time within the Containment and an additional 20 seconds transit time from the Reactor Containment to the sampling hood. This allows for decay of short lived isotopes to a level that permits normal access to the sampling room.

Sample Sink

The sample sink is located in a hooded enclosure that is equipped with an exhaust ventilator. The work area around the sink and the enclosure is large enough for sample collection and for storage of 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 steam system lines, all of which discharge into the sink. The sink and work areas are stainless steel.

Piping and Fittings

All liquid and gas ample lines are austenitic stainless steel tubing and are designed for high pressure service. Compression fittings and 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.

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 samples flow rate as indicated on Plant Drawings 9321-F-27453 [Formerly Figure 9.4-1].

Appropriate valves and administrative controls prevent gross reverse flow of gas from the volume control tank into the sample sink.

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

Isolation valves are provided outside the Reactor Containment; they trip closed upon a containment isolation signal.

9.4.1.3 System Evaluation

Availability and Reliability

Neither automatic nor operator action is required for the Sampling System during an emergency or to prevent an emergency condition (with the exception of the closure of the containment of the containment isolation valves). The system is therefore designed in accordance with standard practices of the chemical processing industry.

Leakage Provisions

Leakage of radioactive reactor coolant from this system within the Containment is evaporated to the containment atmosphere and 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 the building exhaust. Liquid leakage from the valves in the hood is drained to the chemical drain tank.

Incident Control

The system operates under administrative manual control.

Malfunction Analysis

To evaluate system safety, the failures or malfunctions are assumed concurrent with a Loss-of-Coolant Accident, and the consequences analyzed. The results are presented in Table 9.4-3. From this evaluation it is concluded that proper consideration has been given to station safety in the design of the system.

9.4.1.4 Minimum Operating Conditions

Minimum operating conditions are specified in the Technical Specifications, Technical Requirements Manual, FSAR and ODCM.

9.4.1.5 Tests and Inspections

Examples for frequency of sample analyses are as follows:

a) Reactor coolant – radiochemical analysis – every seven days.
b) Reactor coolant – boron concentration – 2 days per week, Maximum 5 days between analyses.
c) Refueling water – storage tank water – boron concentration – monthly.

9.4.2 Post-Accident Sampling System (PASS)

9.4.2.1 Post-Accident Reactor Coolant Sampling System

Under post-accident conditions, the radioactivity of the primary coolant may be increased by several orders of magnitude. Access into the primary sample room is prohibited by extremely high exposure rates. The post-accident reactor coolant sampling system (shown in Plant Drawing 9321-F-27453 [Formerly Figure 9.4-2]) provides a safe and accurate method of obtaining a pressurized coolant sample and a means for analyzing the sample of dissolved gases, hydrogen, isotopic content, chloride, and boron. Samples of the recirculation pump discharge and the residual heat removal system can also be taken.

The requirement for a Post-Accident Sampling System was added to the Indian Point 3 licensing basis in response to NUREG 0737 (Technical Specification Amendment 38, dated October 7, 1981). As a result of subsequent evaluation by the Westinghouse Owners Group (WCAP-14986) the NRC approved the technical basis for eliminating the regulatory requirement for the PASS (NRC Safety Evaluation dated June 14, 2000). Therefore, the PASS was eliminated as a Technical Specification requirement by Amendment 210, issued February 2002. NRC approval of this amendment required the adoption of three (3) commitments. One commitment required that contingency plans be maintained for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. These contingency plans are reflected in plant procedures and may use the capabilities of the PASS.

To obtain a primary coolant sample in a post-accident condition, temporary diversion of a representative sample stream of primary coolant is made into a shielded compartment outside the sample room. The primary coolant is diverted downstream of the sample coolers through quick disconnect couplings to a sample cask in the lower portion of the shielded compartment. The shielded compartment consists of three connected compartments with hinged doors; walls and doors are steel-encased lead, with a minimum thickness of 1-1/2 inch. Channels of poured lead are installed internally at any seam areas of the compartment, and reach rods are provided from remote operation of valves.

The sample is directed into the shielded portable cask so that a nominal 62-mi sample can be safely collected and transported to the analysis apparatus. Streaming is minimized by offsetting the openings in the cask from the direct line of sample and by use of a shielded cap during transport. Dose is minimized procedurally by use of a special hand tool to disconnect the cask from the system.

After the sample is collected in the cask, it is transported and positioned under the analysis equipment behind a shielded door. The cask is connected to the analysis system via flexible tubing of minimum volume, and the sample is pumped into a modified closed loop gas expansion rig. The cask with connections, in fact, forms part of the closed loop for gas expansion. The gas expansion rig has been modified to fit into a shielded box 50cm x 35cm x 35cm ID. The box is steel-encased lead, with a minimum thickness of 3 in., with a lead glass viewing window. The apparatus uses all solenoid-operated valves to preclude high personnel exposure. The box is ventilated into the Primary Auxiliary Building’s ventilation system via an in-line vaneaxial fan.

The sample is recirculated through the gas expansion rig, which contains 30 mls of demineralized water for analysis of dissolved total gas and hydrogen. A gas sample is withdrawn by syringe through a septum for hydrogen analysis by gas chromatography. A

second gas sample is withdrawn and injected into a glass bulb for isotopic analysis on the gamma spectroscopy system. On completion of the gas analyses, the diluted (1:1.5) degassed liquid is pumped to a beaker within the shielded box. Samples of the diluted liquid in the breaker are withdrawn for liquid radioisotopic, boron and chloride analyses. An undiluted sample can be obtained within 24 hours and analyzed for chlorides within 30 days. The radioisotopic and boron samples may be diluted again as necessary to limit personnel dose. On completion of all analyses, the sample left in the beaker is pumped to a waste cask for disposal. The system can then be flushed with water.

The primary method of pH measurement would be the use of an inline sensor mounted in a shielded cask. The sample is obtained in the same manner as the initial pressurized reactor coolant sample. A pH measurement can be obtained by taking a second sample and performing a pH analysis with the analysis system prepared without dilution water. The undiluted, second sample would be pumped into a beaker within the shielded box containing a pH electrode. On completion of the analysis, the sample is pumped to a waste cask for disposal. The system can then be flushed with water.

9.4.2.2 Containment Atmosphere and Plant Vent Post -Accident Sampling System

The post-accident containment atmosphere and plant vent sampling system is designed to obtain representative samples of the containment air and stack for isotopic analysis. The system is designed to be utilized when the normal sampling system is inaccessible during periods of abnormally high release rates.

The containment atmosphere and plant vent sampling lines are electrically heat traced in order to prevent moisture condensation. The heat trace cable provided is of the self-regulating type selected to maintain the containment atmosphere sample line at 250°F and the plant vent stack sample line at 120°F. Two thermostats are provided per heat tracing zone: one for temperature regulation, the other for a low temperature alarm.

The basic system consists of containment air and stack sample shielded compartments located on the 41 foot elevation of the Primary Auxiliary Building. The containment air sample system consists of three lead shielded compartments with hinged doors. One compartment contains the sample collection cartridges for iodine and particulate. Another compartment houses the gas sampling cylinder and minimal tubing. The third compartment houses most associated tubing and a sample pump. The plant vent system is similar but all tubing, sample media and a pump are housed in one shielded compartment.

A noble gas sample is withdrawn by syringe through a port in the shielded compartment and analyzed for hydrogen by gas chromatography and for activity by gamma spectroscopy. A silver zeolite cartridge and a millifilter are used to collect iodine and particulate samples. Bottled gas is purged through the system after collecting the sample to reduce personal exposure during removal and transport of the sample media. The samples are then analyzed for iodine and particulate on the gamma spectrometer.

9.4.2.3 Main Steam Post-Accident Sampling System

The post-accident main steam sampling system is designed to collect a condensed liquid sample of the main steam during accident conditions to verify steam generator integrity.

Upstream isolation valves are opened on the steam generator side of the main steam isolation valves. The normal operation sample root valve is shut and the accident cross-connect valve is opened. A sample is obtained from the main steam sample sink isolation valve in the secondary laboratory on the 15 foot elevation of the turbine building.

An isotopic analysis of the main steam sample is performed using the gamma spectroscopy system.

TABLE 9.4-1

SAMPLING SYSTEM CODE REQUIREMENTS

Sample heat exchanger	ASME III*, Class NC, tube side ASME VIII, shell side
Piping and valves	USAS B31.1**

NOTE:
* ASME III – American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.
** USAS B31.1 – Code for Pressure Piping and special nuclear cases where applicable.

TABLE 9.4-2

SAMPLING SYSTEM COMPONENTS

Sample Heat Exchanger

Number	10
Type	Coiled tube in shell
Heat exchanged (each), Btu/hr	2.14×10^5
Surface area (each), ft^2	3.73

Design Conditions:

Parameter	Tube Side	Shell Side
Pressure, psig	2485	150
Temperature, °F	680	350
Flow, lb/hr	209	20,000 (1)
Inlet Temperature, °F	668 (2)	105
Outlet Temperature, °F	127	130
Material	Stainless Steel	Carbon steel

NOTE:
(1) Nominal cooling water flow is approximately 17 gpm.
(2) Saturated steam.

TABLE 9.4-3

MALFUNCTION ANALYSIS OF 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 9.4-4
[Historical Information]

POST-ACCIDENT REACTOR COOLANT SAMPLING SYSTEM
ANALYTICAL CAPABILITIES

ANALYTE	RANGE	ACCURACY
Gross Radioactivity	1 uCi/ml to 10 Ci /ml	factor of 2
Isotopic	not specified	factor of 2
Boron	130 to 6000 ppm	+ / - 15%
Chloride (Note 1)	0.10 to 20 ppm	+ / - 40%
Dissolved Hydrogen	0.3 to 200 cc/kg	+ / - 21%
Dissolved Oxygen	2 to 200 ppm	+ / - 0.05 ppm or 10% whichever is greater
pH	1 to 13	+ / - 0.3 pH units

NOTE 1: Initial chloride sample is diluted 1 to 1.5. The sampling system can obtain an undiluted sample for chloride within 24 hours for later analysis.

9.5 FUEL HANDLING SYSTEM

The Fuel Handling System provides a safe effective means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves the plant after post-irradiation cooling.

The system was designed to minimize the possibility of mishandling or mal-operations that cause fuel damage and potential fission product release.

The Fuel Handling System consists of:

a) The reactor cavity, which is flooded only during plant shut-down for refueling
b) The spent fuel pit, which is kept full of water and is always accessible to operating personnel
c) The fuel Transfer System, consisting of an underwater conveyor that carries the fuel through an opening between the areas listed in the discussion of plant containment in Chapter 5.

In lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24, Indian Point 3 has chosen to comply with the following seven (7) criteria of 10 CFR 50.68(b):

1) Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95% probability, 95% confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

3) If optimum moderation of fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95% probability, 95% confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95% probability, 95% confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95% probability, 95% confidence level, if flooded with unborated water.

5) The quantity of Special Nuclear Material (SNM), other than nuclear fuel, is less than the quantity necessary for a critical mass.

6) Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and initiate appropriate safety actions.

7) The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5) percent by weight.

9.5.1 Design Basis

The General Design Criteria presented and discussed in this section are those that were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NCR) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

Prevention of Fuel Storage Criticality

Criterion: Criticality in the new and spent fuel storage pits shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls (GDC 66 of 7/11/67)

During reactor vessel head removal and while loading and unloading fuel from the reactor, boron concentration is maintained at not less than that required to shutdown the core to a $k_{eff} =$ 0.95. This shutdown margin maintains the core at $k_{eff}<0.99$, even if all control rods are withdrawn from the core. Weekly checks of refueling water boron concentration ensure the proper shutdown margin.

The new and spent fuel storage racks were designed so that it is impossible to insert assemblies in other than the prescribed locations. The new and spent fuel storage pits have accommodations as defined in Table 9.5-1. Additionally, the spent fuel pit has the required spent fuel shipping area. Borated water is used to fill the spent fuel storage pit at a concentration to match that used in the reactor cavity and refueling canal during refueling operations. The spent fuel pit design assures a keff<0.99.

Detailed instructions have been issued for use by refueling personnel. These instructions, the minimum operating conditions, and the design of the fuel handling equipment, incorporating built-in interlocks and safety features, provide assurance that no incident could occur during refueling operations resulting in a hazard to public health and safety.

Fuel and Waste Storage Decay Heat

Criterion: Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities and to waste storage tanks that could result in radioactivity release which would result in undue risk to the health and safety to the public. (GDC 67 of 7/11/67)

The refueling water provides a reliable and adequate cooling medium for spent fuel transfer. Heat removal from the spent fuel pit is provided for by an auxiliary cooling system.

Design Codes and Criteria

The general controlling standard for the design, fabrication, installation and testing of the Fuel Storage Building fuel cask crane is the American Society of Mechanical Engineers ASME NOG-1-2004.

The crane, bridge, trolley and hoist are controlled by ASME NOG-1-2004. Specifications for the following organizations are referenced therein:

1) American Gear Manufacturers Association
2) American Institute of Steel Construction
3) Association of Iron and Steel Engineers
4) American National Standards Institute
5) American Society for Testing and Material
6) American Welding Society
7) National Electrical Code, National Fire Protection Association
8) National Electric Manufacturers Association

The crane rail and structural steel supporting structures were controlled by the American Institute of Steel Construction, "Manual of Steel Construction" 1964. All structural steel is ASTM A-36. The crane rail is in accordance with Manufacturers Standards and ASTM A-1. Design of the cables was controlled by the ASME NOG-1-2004 code. In addition, the following are applicable to cables:

1) RR-W-410C, which is a Federal Specification for wire rope representing the industry standard
2) ANSI B30.2.0-1967 for Overhead and Gantry Cranes.

The lifting hooks were purchased, fabricated and load tested to manufacturers standards. The hooks were ultrasonically tested to detect any flaws. These hooks are again tested at specified periods prior to all lifts.

The hooks are inspected, tested, and maintained in accordance with ANSI B30.2-1976. Overhead and Gantry Cranes. When the crane hooks are inactive for a period of time longer than a specified inspection, test, or maintenance frequency, the inspection, test, or maintenance activity should be performed prior to their use. [NCR letter dated February 13, 1985, Control of Heavy Loads (Phase 1)]

The manipulator crane structure was designed in accordance with Electric Overhead Crane Institute (EOCI), Inc., Specification No. 61. The design load was specified as 5626 lbs (the weight of three fuel assemblies at 1575 lbs each plus the weight of the gripper tube at 900 lbs). All loading supporting members were designed with a 5 to 1 safety factor on this design load.

The crane was pre-operational load tested with 6000 lbs. Normal operating load for the crane is 2475 lbs although emergency procedures for removing stuck fuel assemblies may require occasional loading up to 6000 lbs. Seismic loads used for design were based on 0.5 g horizontal and 1.0 g vertical accelerations which are greater than the accelerations at the installed location. To resist design basis earthquake forces, the equipment was designed to limit the stress in the load bearing parts to 0.9 times the ultimate stress for a combination of normal working load plus design basis earthquake forces.

The structural material for the spent fuel pit bridge and hoist was designed to ASTM-A373. The design load was specified as 2000 lbs on the hoist and 250 lbs per square foot on the bridge. All load supporting members were designed with a 5 to 1 safety factor at these design loads. The hoist was pre-operational load tested with 2500 lbs while the normal operating load will be 1750 lbs (fuel assembly plus fuel handling tool). Seismic loads used for design were based on 0.5 g horizontal and 1.0 g vertical accelerations, which are greater than the accelerations at the installed location. To resist design basis earthquake forces, the equipment was designed to limit the stress in the load bearing parts to 0.9 times the ultimate stress for a combination of normal working load plus design basis earthquake forces.

For seismic considerations of the Fuel Storage Building spent fuel cask crane, the fuel crane bridge was evaluated to determine the potential for the trolley to lift off the crane bridge rails or for the crane bridge to lift off its track support in the event of a seismic disturbance. The crane bridge and trolley were analyzed for the design basis earthquake both loaded and unloaded for various positions of the trolley using response spectra modal analysis with 7% damping per ASME NOG-1-2004. It was determined that the downward force due to gravity exceeds the maximum upward seismic wheel load due to combined vertical and horizontal accelerations by a factor of 1.2.

As this is the only potential for bridge or trolley lift-off or overturning, no potential hazard exists to any seismic Class 1 function, and vertical restraints were not required.

The wheels of the bridge and trolley are shaped such that sliding perpendicular to the rail would not be possible. The lateral load from an earthquake on the trolley crane rail is about 50% greater than the lateral loads from impact that the AISC Code specifies for design within working stress limits. The stresses on the crane rail are low due to the earthquake load. For this reason, no failure of the crane rail is anticipated. The design load rating of this crane is anticipated. The design load rating of this crane is 40 tons with a 5 ton auxiliary hook.

Other pre-operational test loads on components of the FHS were:

a) 125% of design rating on spent fuel cask crane
b) Functional check-out for operability using a test weight that approximates 100% of the operational load to be handled.

Test loads used throughout the plant life shall be equal to or greater than the maximum load to be assumed by the hoist crane during refueling operations.

9.5.2 System Design and Operation

The reactor is refueled with equipment designed to handle the spent fuel underwater from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Boric acid is added to the water to ensure subcritical conditions during refueling.

The Fuel Handling System may be divided into two areas: the reactor cavity, which is flooded only during plant shutdown for refueling, and the spent fuel pit, which is kept full of water and is always accessible to operating personnel. These two areas are connected by the Fuel Transfer System consisting of an underwater conveyor that carries the fuel through an opening in the plant containment. (See Figure 9.5-1)

The reactor cavity is flooded with borated water from the Refueling Water Storage Tank. In the reactor cavity, fuel is removed from the reactor vessel, transferred through the water and placed in the fuel transfer cart by a manipulator crane. In the spent fuel pit, the fuel is removed from the transfer system and placed in storage racks with a long manual tool suspended from an overhead hoist. The fuel can be removed from storage and loaded into a shipping cask for shipment from the site.

New fuel assemblies are received and stored in racks in the new fuel storage area. The new fuel storage area is sized for storage of the fuel assemblies and control rods normally associated with the replacement of 72 fuel assemblies.

Major Structures, Systems and Components Required for Fuel Handling

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 refueling. The cavity is filled to a depth that limits the radiation at the surface of the water to 2.0 milliroentgens per hour during fuel assembly transfer.

The cavity is large enough to provide storage space for the reactor upper and lower internals, the control cluster drive shafts, and miscellaneous refueling tools. The floor and sides of the reactor cavity are lined with stainless steel.

The reactor vessel flange is sealed to the cylindrical side walls of the reactor refueling cavity by either a Presray inflatable seal or a rigid, segmented seal. This inflatable seal design utilizes gas pressurization to inflate and uniformly compress an oval cross-section reinforced synthetic rubber (EPDW) envelope structure to effect a seal. The required inflation pressure is specified at 31 psig (equivalent head of water to be sealed plus 20 psig). This seal is installed and inflated after reactor cooldown but prior to flooding the cavity for refueling operations.

The wedge shape at the top of the device is designed to effect a pressure tight seal by virtue of the hydrostatic head of water even if pneumatic inflation pressure were to be lost.

The Presray seal design includes two independent gas inflation connections that are in simultaneous use during service. Each gas connection point at the seal is equipped with a fixed orifice device that limits seal deflation to a minimum of 10 psig should either one of the inflation sources malfunction. Both inflation sources are monitored by on-line “air supply to seal”

pressure gauges. The difference in indicated delivery pressure will indicate a gas source malfunction or the failure of a gas connection. Any inflation gas leak into the refueling pool volume will be revealed by gas bubbles appearing at the surface of the pool.

As an alternate to the Presray seal, a rigid, segmented seal design, may be used. This seal relies on mechanical forces to effect the seal. The segmented seal is comprised of a stepped rectangular EPDM rubber compression seal split into five (5) equal circumferential segments. The segmented joint is achieved by means of a beveled or skive overlap joint on the segment ends. The segmented design can be installed in approximately one (1) hour, compared to the Presray design which takes about approximately four (4) hours to install. This reduces outage time and radiation dose.

Reactor Cavity Filtration System

The Reactor Cavity Filtration System provides the capability of filtering the water in the reactor cavity whenever the cavity is filled. This filtration system maintains water clarity and removes suspended radioactive particles.

The system consists of a skid carrying four stainless steel filter units, associated piping, and valving mounted on the 95' floor elevation against the northwest face of the shielding around Steam Generator No. 33. It is enclosed on its three exposed sides by shielding. All surfaces in contact with refueling water are either stainless steel or synthetic hose and filter medium. At present, the Reactor Cavity Filtration system is partially disassembled and is not used. All disassembled equipment can be reinstalled at a future date if it is desired to use the system. Filtration of the reactor cavity water can be performed using the Spent Fuel Pool Cooling System or a temporary augmented system, as described below.

When the Reactor Cavity Filtration System is not operable, temporary submersible filtration units are placed in the reactor cavity when it is filled with water. These units use plant power and are secured to the walls of the reactor cavity during their operation. They are removed prior to draining down the reactor cavity. The Reactor Cavity Filtration System is not required for safety.

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 that 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 tipping device and the control cluster changing fixture located in the canal. The transfer tube enters the Reactor Containment and protrudes through the end of the canal. The canal wall and floor linings are similar to those for the reactor cavity.

Fuel Pool Enclosures

A controlled leakage building designed for a negative pressure of 0.50 inches of water minimum, permanently encloses the fuel pool. The design features of the fuel handling building that provide this leak tightness are as follows:

1) Special sealing features at joints that include:
   a) Sealing off all edges and ends of the walls with a combination of caulking and relatively soft neoprene strip,
   b) Installation of necessary additional closure flashing at the extremities and at openings,
   c) Supplying additional caulking in vertical and horizontal joints of liner panels,
   d) Furnishing liner panels in sufficient thickness to seat well on girt spacings and resist flexing in addition to withstanding the normal design wind loads, and
   e) Providing additional fastening for liner panels.

2) Personnel doors that seal by means of inflatable air seals. These seals are inflated upon a high radiation alarm from R-5, although R-5 operability does not require this function.

3) Motor operated dampers designed to fail closed are installed on the discharge side of the two supply fans.

Prior to handling operations, when irradiated fuel is within the Fuel Handling Building, tests are performed to verify the filtration leak tightness. A negative pressure greater than or equal to 0.125 inch water gauge shall be maintained with respect to atmospheric pressure during emergency refueling mode of operation. Fuel handling operations are performed in accordance with the Technical Specifications.

Refueling Water Storage Tank

The normal duty of the Refueling Water Storage Tank is to supply borated water to the refueling canal for refueling operations. In addition, the tank provides borated water for delivery to the core following either a Loss-of-Coolant or a steam line rupture accident. This is described in Chapter 6. The capacity of the tank was based upon the requirement for filling the reactor cavity and refueling canal.

The water in the tank is borated to a concentration that assures reactor shutdown by at least 5% $\Delta k/k$ when all Rod Cluster Control assemblies are inserted and when the reactor is cooled down for refueling. Heating is provided to maintain the temperature above freezing.

The tank design parameters are given in Chapter 6.

Spent Fuel Storage Pit

The spent fuel storage pit was designed for the underwater storage of spent fuel assemblies and fuel assembly inserts after their removal from the reactor. A pumping system recirculates the water in the pool at a flow rate of 230 gpm through four stainless steel filter units, to reduce the burden of radioactive crud.

The Backup Spent Fuel Pool Cooling system has been installed to operate in parallel with the Normal SFP Cooling System to improve pool conditions during refueling activities. The BSFPCS is a manual system served by an independent cooling water source (demineralized water). A primary loop handles the SFP water and consists of two 100% capacity pumps, a plate heat exchanger, associated piping, and local instrumentation. A secondary loop is the

heat sink for the system, and includes two, open-circuit evaporative cooling towers, two 100% capacity feed pumps, associated piping, and local instrumentation. Make-up and fill for the secondary loop is normally provided by demineralized water, with an alternate, emergency source available through the Fire Protection System.

Power for all equipment is supplied from 480 VAC MCCs E1 and E2. For greater availability of power, and reliability of the cooling function, the BSFPCS includes a transfer switch that allows alignment to either the normal power sources MCC E1 and E2, or a rental diesel generator unit (with Engineering discretion). This feature allows an alternate power source in the event the MCCs become inoperable.

The pit accommodations are listed in Table 9.5-1 and the layout of the Fuel Storage Building is shown on Plant Drawing 9321-F-25143 [Formerly Figure 9.5-2]. In 1990, the high density racks shown on Plant Drawing 9321-F-25143 [Formerly Figure 9.5-2] were replaced with maximum density racks containing 1345 cells (see Figure 9.5-2A). The Technical Specifications provide limitations on fuel storage in Regions 1 and 2 of the spent fuel pit to ensure k_{eff}<0.95 while taking no credit for boron in the water (Figures 9.5-2B and 9.5-2C).

Spent fuel assemblies are handled by a long-handled tool suspended from an overhead hoist and manipulated by an operator standing on the movable bridge over the pit.

The spent fuel storage pit is constructed of reinforced concrete having thick walls and is Class I seismic design. The entire interior basin face and transfer canal is lined with stainless steel plate. Hence, the probability of rupture of the pit is exceedingly low.

The structural steel and metal siding building surrounding the spent fuel pit is seismic Class III, as is the Fuel Storage Building crane.

The design tornado missiles will not penetrate the walls of the spent fuel pit. Should a missile hit the surface of the spent fuel pit water, by the time it reached the top of the spent fuel assemblies its velocity would be reduced so that it would not damage the spent fuel.

For a discussion of spent fuel pit dewatering as a result of a tornado, refer to Section 16.4.

The ventilation system in the Fuel Storage Building enclosing the spent fuel pit was designed so that there is a slight negative pressure inside the building during normal refueling operations. Whenever the ventilation system is required to be in operation, the bypass dampers around the charcoal filter must be manually closed and leak tested to assure that it is properly sealed. On a high radiation alarm, the following actions automatically take place:

1) Building ventilation supply fans are secured,
2) Dampers at ventilation supply fans close,
3) If open, rollup coiling truck bay door closes,
4) inflatable seals on main doors are actuated (R-5 operability does not require this function, however), and
5) Exhaust fan continues to run.

Under these conditions, the maximum calculated in-leakage to the building (caused by the operation of the exhaust fan) would be approximately 20,000 cfm with a one-half inch of water negative pressure inside the building. Thus, following the release of radioactivity in the Fuel

Storage Building, there will be zero air leakage from the building proper, and the entire exhaust from the building will pass through roughing, HEPA, and charcoal filters before going up the plant vent.

Technical Requirements Manual (TRM) require charcoal and HEPA filter testing to demonstrate operability any time a fire, chemical release or work done on the filters could alter integrity. Surveillance testing is based upon a maximum flow of 30,000 cfm giving a minimum safety factor of 2 for methyl iodide removal efficiency while allowing 1% bypass. NSE 98-3-017 HVAC demonstrates, for the purpose of TS implementation, that welding is not a fire, a chemical release or work that could alter filter integrity. The NSE also demonstrates that organic components from painting and similar activities could not alter filter integrity until the organic components are above 10 % by weight and concludes that filter testing shall be performed when the organic components are greater than or equal to 2.5 % by weight organics. Administrative controls are required to evaluate the percent (%) by weight of organics when activities that could generate organics are conducted.

A pushbutton switch is provided adjacent to the 95' elevation door leading to the Fan House. This switch allows the Fuel Storage Building Exhaust Fan to be momentarily shut down and air removed from the door seal thereby allowing the door to be opened. The fan will automatically restart and the door is resealed after a preset time has elapsed (approximately 30 seconds).

The handling of irradiated fuel in the Fuel Storage Building or movement of the spent fuel cask or cask crane over the Spent Fuel Pit are prohibited when the fuel storage building emergency ventilation system is inoperable.

Since the spent fuel cask must be loaded in the spent fuel pit, the crane must carry a heavy load, namely the cask, over the pit. The bases for the acceptability of this design are:

1) During normal fuel handling operations, heavy loads (above 2000 lbs) cannot be carried over spent fuel, and
2) Even in the event that the spent fuel cask is dropped over the pit, the loss of water from the resulting failed liner plate and cracked concrete base is inconsequential.

Loss of water in the spent fuel pit and the resultant uncovering of the spent fuel by way of drains and permanently connected system cannot take place for the following reasons:

1) The suction of the spent fuel pit pump is taken from a point approximately six (6) feet below the top of the pool wall; therefore this pump cannot be used to uncover the fuel, even accidentally.

2) The spent fuel pit pump discharge pipe terminates in the pool at elevation 74' – 4 ¾". This elevation is approximately five (5) feet above the top of the spent fuel assemblies; therefore this pipe could not accidentally become a siphon to uncover the fuel.

3) The skimmer pump takes suction from, and discharges to the surface of the pool; therefore it could not accidentally or otherwise uncover the spent fuel.

4) There are no drains on the bottom or side walls of the spent fuel pit. Draining would have to be done deliberately by a temporary pump.

5) The spent fuel pit cooling loop was designed to seismic Class II and the cleanup equipment loop was designed to seismic Class III criteria; however, their failure could not result in the uncovering of the spent fuel, as explained above.

A radiation monitor (R-5) is located in the Fuel Storage Building. The monitor provides continuous indication of the radiation level with high radiation alarm given both locally and in the Control Room. The air filtration system for the Fuel Storage Building is automatically actuated on high radiation signal. Radiation levels in the spent fuel storage area shall be monitored continuously whenever there is irradiated fuel stored therein. If the monitor is inoperable, a portable monitor may be used. Equipment is also provided to monitor spent fuel pit water level with low level annunciated in the Control Room. (See Section 11.2)

The primary source of makeup water to the spent fuel pit is the Primary Makeup Water Storage Tank, which is a seismic Class I component. The pumps and most of the piping associated with this tank are also seismic Class I. The makeup water loop to the spent fuel pit is seismic Class II, as is the spent fuel pit cooling loop. The cleanup equipment and skimmer loops are seismic Class III. Additional backup can be provided through a temporary connection from the plant demineralizers or from the Fire Water Tank.

In addition, there is a second spent fuel pool cooling system pump to provide standby capacity. There is also a provision for adding a portable cooling pump.

Storage racks provided to hold spent fuel assemblies were erected on the pit floor. The racks were designed so that it is impossible to insert fuel assemblies in other than the prescribed locations (there is insufficient space between the rack assembly and the SFP wall), thereby ensuring the necessary spacing between assemblies. Control rod clusters or other inserts are stored inside the spent fuel assemblies.

The spent fuel storage racks consist of twelve freestanding welded honeycomb arrays of type 304 stainless steel boxes that have no grid frame structure. The storage racks are arranged and categorized in two regions based on fuel assembly enrichment and burn-up. The nominal pitch for Region I is 10.76 inches. The nominal pitch for Region II is 9.075 inches. All storage cells are bounded on four sides by boral poison sheets, except on the periphery of the pool rack array.

Each of the twelve maximum density racks are supported and leveled on four screw pedestals which bear directly on the pool floor. These freestanding racks are free to move horizontally. However, with only a 0.2 friction factor, there is no wall impact even assuming five (5) OBE and one (1) SSE earthquake event all added up in the same direction. Additionally, there is no rack-to-rack impact since the maximum density racks were designed to be installed with essentially no gap between the racks. The strong hydrodynamic coupling between the racks causes the racks to move together even when a full and empty rack are adjacent to each other.

<u>Shielded Transfer Canister (STC) and HI-TRAC Transfer Cask</u>

The NRC has issued Amendment 246 for the inter-unit transfer of spent fuel from Unit 3 to Unit 2 (Ref. 1). The Amendment is based on evaluations conducted for each aspect of the inter-unit transfer of fuel as documented in the Licensing Report (Ref. 2). The non-proprietary version of the Licensing Report is incorporated by reference into the UFSAR.

The STC is a thick-walled vessel with a removable top lid capable of transferring up to twelve spent fuel assemblies and associated non-fuel hardware. For inter-unit spent fuel transfer operations between the Unit 3 Spent Fuel Pool (SFP) and the Unit 2 SFP, the STC is used in conjunction with the HI-TRAC transfer cask. During STC closure activities and spent fuel transfer operations, the STC shielding is supplemented with the HI-TRAC shielding (steel, lead and water) and the water contained in the annulus space located between the STC and the HI-TRAC. For inter-unit spent fuel transfer operations, the HI-TRAC uses a solid lid and a centering assembly that keeps the STC centered inside the HI-TRAC cavity. The centering assembly forms an annular region inside the HI-TRAC which remains mostly full of water during loading and transfer operations. An air space is left in the HI-TRAC above the STC top flange to allow the STC lid operations to occur unhindered by water and provide an expansion volume for the water inside the HI-TRAC cavity. During spent fuel transfer operations the STC is mostly full of borated water and is steam blanketed to remove all air from the STC. The STC includes a removable bolted lid with vent and drain ports for steam blanketing and water filling / draining purposes. The STC lid is coated on the top and sides to protect the carbon steel surfaces from corrosion. Should the coating system be damaged during wet fuel transfer operations, the damaged coating is removed and replaced with N-5000 or vacuum grease to prevent corrosion. The STC lid has lifting devices that can be remotely or manually actuated to engage trunnions on the STC body to lift the STC body when the STC lid bolting is removed. The STC lid also has threaded lid lifting points which provide a means to attach the STC and lid to overhead cranes.

The STC is moved between Units 3 and 2 vertically in the HI-TRAC. Neither the HI-TRAC nor the STC are handled in the horizontal orientation when loaded with spent fuel assemblies and associated non-fuel hardware. In addition to the water in the STC cavity and the water in the annulus space between the STC and HI-TRAC's inner shell, the HI-TRAC's water jacket is also filled with water. These three discrete zones of water provide shielding and aid in heat transfer.

Major Equipment Required for Fuel Handling

Reactor Vessel Stud Tensioner

The stud tensioner is a hydraulically operated (oil is the working fluid) device provided to permit preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. Stud tensioners were chosen in order to minimize the time required for the tensioning or unloading operations. Three tensioners are provided and they are applied simultaneously to three studs 120° apart. One hydraulic pumping unit operates the tensioners that are hydraulically connected in parallel. The studs are tensioned to their operational load in two steps to prevent high stresses in the flange region and unequal loadings in the studs. Relief valves are provided on each tensioner to prevent over- tensioning of the studs due to excessive pressure. Charts indicating the stud elongation and load for a given oil pressure are included in the tensioner operating instructions. In addition, micrometers are provided to measure the elongation of the studs after tensioning.

Reactor Vessel Head Lifting Device

The reactor vessel head lifting device consists of a welded and bolted structural steel frame with suitable rigging to enable the crane operator to lift the head and store it during refueling operations.

Three vertical legs and a platform assembly are permanently attached to the reactor vessel head lifting lugs. The sling assembly is attached to the three vertical legs and is used when installing and removing the reactor vessel head. During plant operations, the sling assembly is removed and the three vertical legs and platform assembly remain attached to the reactor vessel head. The total estimated weight of the reactor vessel head with lifting rig is 150 tons. (See Figure 9.5-8)

The maximum drop height of the reactor vessel head is discussed in Section 9.12.4.2.

<u>Reactor Internals Lifting Device</u>

The reactor internals lifting device is a structural frame providing the means to grip the top of the reactor internals package (upper or lower) and to transfer the lifting load to the crane. The device is lowered onto the guide tube support plate of the internals and is manually bolted to the support plate by three bolts. The device may be lowered onto the lower internals package and is manually bolted to the core barrel by the same three bolts. The bolts are controlled by long torque tubes extending up to an operating platform on the lifting device. Bushings on the fixture engage guide studs mounted on the vessel flange to provide close guidance during removal and replacement of the internals package.

This fixture is a three legged structural frame device that connects the main crane hook to the upper support plate or core barrel for handling operations. It connects to the internals flanges by means of screw threads (See Detail A of Figure 9.5-6). The total estimated weight of the lifting rig and upper internals is 67 tons. The estimated weight of the upper internals is 59 tons. The postulated drop of the upper internals is bounded by the reactor head drop analysis discussed in Section 9.12.4.2.

<u>Manipulator Crane</u>

The manipulator crane is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The bridge spans the reactor cavity and runs on rails set into the floor along the edge of the reactor cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly in the core. A long tube with a pneumatic gripper on the end is lowered down out of the mast to grip the fuel assembly. The gripper tube is long enough so that the upper end is still contained in the mast when the gripper end contacts the fuel. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. The fuel is transported while inside the mast tube to its new position.

All controls for the manipulator crane are mounted in a console on the trolley. The bridge, trolley, and main hoist are equipped with encoders for position readout on the console. Bridge and trolley position may also be read directly from position scales near the bridge and trolley rails. Drives for the bridge, trolley, and main hoist are variable speed.

The suspended weight on the gripper tool is monitored by an electric load cell indicator mounted on the control console and by the Control System. Under normal operating modes, when loaded, hoist lower motion is permitted in the Core zone, Upender zone, or RCC change basket only. When unloaded, hoist lower motion is permitted in the same areas as hoist loaded, and also from “hoist full up” to “hoist inside mast” in any area. Hoist raise is permitted in any area, loaded or unloaded. Raising of the guide tube is not permitted if the gripper is unlatched and the load monitor indicates a load above normal gripper weight. Hoist raise is not permitted if the load is more than 150 lb. above the selected fuel and insert type, unless the hoist elevation is in

a load bypass zone where that overload is set at + 200 lb. In all zones including load bypass zones, there is a backup overload set at +250 lb. Hoist lower is not permitted if the load is more than the 150 lbs. below the setting per fuel type unless the hoist elevation is in the bypass zone. If it is in this zone, a slack cable and / or encorder value interlock would stop motion. The gripper is interlocked electrically (through an electric load cell in conjunction with hoist encoder position) and also a mechanical spring lock so that it cannot be opened when supporting a fuel assembly.

Safety features incorporated in the system are as follows:

a) Boundary zone control provided by the Control System in conjunction with encoders on each axis preclude the possibility of violating the boundaries established for safe operation of this system. In normal operation, traverse of the trolley and bridge is limited to the areas of the Core, RCC basket location, Upender location, or the Test fixture location and a clear path connecting those areas. Operation of the bridge or trolley outside the boundary system will not be permitted unless the bypass mode has been selected. Existing stops in the bridge and trolley rails will inhibit motion beyond designed limits when in the bypass mode of operation.

b) Simultaneous motion of the bridge and trolley will be permitted. A mapping program within the Control System allows the operators to specify safe operation zones. In addition, isolated obstructions can also be identified. Operation of the hoist will not be permitted when the bridge or trolley are in operation.

c) When the gripper is loaded, the Manipulator Crane will not traverse (between the Core and the Upender, Text fixture, or RCC change basket) unless the guide tube (inner mast) is at full up. The refueling bridge can traverse within the Core Zone with the gripper loaded and not at full up. The traverse speeds will be restricted during this scenario. When the gripper is unloaded, the Manipulator Crane will not traverse (between the Core and the Upender, Test fixture, or RCC change basket) unless the guide tube (inner mast) is protected in the mast. The Manipulator Crane can traverse within the Core zone with the gripper unloaded and not inside the mast. The traverse speeds will be restricted during this scenario. The Manipulator Crane can traverse a small distance when an unloaded gripper is extended outside the inner mast at a RCC basket location, Upender location, or the Test fixture location for the fine positioning to aid in withdrawing or inserting a fuel bundle. In normal operation, traverse of the trolley and bridge is limited to the areas of the Core, RCC basket location, Upender location or the Test fixture location and a clear path connecting those areas whether the hoist is loaded or unloaded.

d) An electrical interlock that prevents opening of a solenoid valve in the air line to the gripper disengage cylinder has been incorporated into this system, which takes into account hoist load, hoist position, and system air pressure. The fuel gripper must be in its down position in the Core, or in the Fuel Transfer System or RCC change basket, or Test fixture with a slack cable, and the air pressure interlock is not tripped in order to unlatch. The spring operated mechanical backup will prevent operation of a loaded gripper even if air pressure is applied to the operating cylinder.

e) Hoist raise is not permitted if the load is more than 150 lb. above the selected fuel and insert type, unless the hoist elevation is in a load bypass zone where the overload is set at + 200 lb. In all zones including load bypass zones, there is a backup overload set at + 250 lb.

f) Interlocks on all drive circuits prevent operation if there is a gripper failure where both limits for the gripper are made at the same time or both limits are not made at the same time.

g) The bridge and trolley drives are interlocked to prevent the manipulator crane from going outside the secure zone. In normal operation, traverse of the trolley and bridge is limited to the areas of the Core, RCC basket location, Upender location, or the Text fixture location and a clear path connecting those areas whether the hoist is loaded or unloaded. The bridge or trolley will not be able to challenge limits of a particular boundary. In bypass mode of operation, there will be no boundary limits except for hard stops on bridge and trolley rails; however, the bridge and trolley will be restricted to slow speed.

Suitable restraints are provided between the bridge and trolley structures and their respective rails to prevent derailing and the manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper in the event of a maximum potential earthquake.

Only core components or tools required for the placement or removal of core components are handled over an open reactor vessel.

Any time the reactor vessel is open, the following precautions are taken to assure that foreign materials do not inadvertently get into the reactor vessel:

1) All personnel tape the cuffs, pockets, buttons, etc. of their “anti-contamination clothing”
2) A barrier is established surrounding the reactor cavity to prevent unnecessary movement in the area
3) Only the minimum number of people required to safely perform the job are allowed in the area
4) All facets of the Quality Assurance Plan are in effect and enforced
5) The cranes are visually inspected before the reactor vessel is opened
6) Prior to lifting of the head, an NDT of the hook is performed.

All fuel handling operations, including core alterations, are performed under the supervision of an individual holding either a Senior Reactor Operator license or a Senior Reactor Operator license limited to fuel handling, as established in 10 CFR 50.54 (m) (2).

Discussions of the effects of the seismic Class III Fuel Storage Building crane on seismic Class I functions are found in Section 16.4

The manipulator crane bridge and trolley are restrained on the rails by the following means: (See Figure 9.5-3)

1) Horizontally – by guide rollers (cam follower) at each wheel on one truck only. The rollers are attached to the bridge truck at the wheels and contact the vertical faces of the rail to prevent horizontal movement.

2) Vertically – by anti-rotation bars, in the vicinity of each wheel at all 4 wheel locations. The anti-rotation bars are carbon steel bars bolted to the truck and extending under the rail flange, to prevent lifting of any wheel from the rail.

Polar Crane & Fuel Storage Building Crane (See Section 9.12)

Spent Fuel Pit Bridge

The spent fuel pit bridge is a wheel-mounted walkway spanning the spent fuel pit that carries an electric monorail hoist on an overhead structure. Fuel assemblies or inserts are moved within the spent fuel pit by means of a long handled tool suspended from the hoist. The hoist travel and tool length were designed to limit the maximum lift of a fuel assembly to a safe shielding depth.

Fuel Transfer System

The fuel transfer system, shown in Figure 9.5-1, is a motor winch driven conveyor car that runs on tracks extending from the refueling cavity through the transfer tube and into the spent fuel pit. The conveyor car received a fuel assembly in the vertical position from the manipulator crane. The fuel assembly is lowered to a horizontal position for passage through the tube, and then is raised to a vertical position in the spent fuel pit.

During plant operation, the conveyor car is stored in the refueling canal. A blind flange is bolted on the transfer tube to seal the Reactor Containment.

Rod Cluster Control Changing Fixture

A fixture is mounted on the reactor cavity wall for removing rod cluster control (RCC) elements from spent fuel assemblies and inserting them into new fuel assemblies. The fixture consists of two main components: a guide tube mounted to the wall for containing and guiding the RCC element, and a wheel-mounted carriage for holding the fuel assemblies and positioning fuel assemblies under the guide tube. The guide tube contains a pneumatic gripper on a winch that grips the RCC element and lifts it out of the fuel assembly.

By repositioning the carriage, a new fuel assembly is brought under the guide tube and the gripper lowers the RCC element and releases it. The manipulator crane loads and removes the fuel assemblies into and out of the carriage.

Refueling Procedure

Refueling requirements and procedures are contained in the Technical Specifications and in this FSAR section.

Preparation

a) The reactor is shut down, cooled to T_{avg}, $\leq$ 140 ^{o}F and boron concentration is checked

b) A radiation survey is made and the containment vessel is entered

c) The control rod drive mechanism (CRDM) missile shield is removed to storage

d) CRDM cables and cooling air ducts are disconnected from CRDM and removed to storage

e) Reactor vessel head insulation and instrument leads are removed

f) The reactor vessel head nuts are loosened with the hydraulic tensioners

g) [Deleted]

h) The canal drain holes are plugged and the fuel transfer tube flange is removed

i) Checkout of the fuel transfer device and manipulator crane is started

j) Guide studs are installed in either two or three holes and the remainder of the stud holes are plugged

k) The reactor vessel to cavity seal (either inflatable “Presray” seal, or rigid segmented seal) is in place

l) Final preparation of underwater lights and tools is made. Checkout of manipulator crane and fuel transfer system is completed

m) The reactor vessel head is unseated, raised, and placed on the storage pedestal

n) The reactor cavity is filled with water. The water is pumped into the reactor cavity by the residual heat removal pumps from the Refueling Water Storage Tank through the reactor vessel. The normal Residual Heat Removal System inlet valves from the Reactor Coolant System are closed. (See alternate method described below) Subsequent to cavity fill, the water may be purified via the Reactor Coolant Drain Tank (RCDT) and associated pumps

o) When the reactor cavity is filled, restore Residual Heat Removal System to normal operation

p) The control rod drive shafts are unlatched

q) The reactor vessel internals lifting rig is lowered into position by the plant crane and latched to the support plate

r) The reactor vessel internals are lifted out of the vessel, inspected to ensure no core component is hanging from the upper core plate and placed in the underwater storage rack

s) The core is now ready for refueling.

Also provided is an alternate method of filling the reactor cavity without the necessity of the water having to pass through the reactor vessel and thereby dislodge crud that could cloud the water and delay refueling operations.

At the time of reactor cavity fill, flanged spool pieces are removed from the containment spray lines and replaced with tee-spool pieces. One end of the tee-spool pieces is blanked off to prevent fluid from entering the containment spray headers. The alternate fill line is connected to one of the tees via a length of high pressure metal hose. Should the pump in the system feeding the alternate fill line experience trouble, the metal hose can be disconnected and

attached to the other tee for completion of the fill operation. Valves for pump startup are included with each spool piece. Additionally, each spool piece contains a vent valve, pressure gauge and pressure gauge isolation valve. The alternate fill line contains three orifice plates that limit pump discharge to its design flow rate of 2600 gpm. (See Plant Drawings 9321-F-20238, and -27353 [Formerly Figure 9.5-9])

Refueling

The refueling sequence is started with the manipulator crane. Steps a through d may be performed in any order as is most efficient for the core loading process. Alternatively, the entire core may be removed to the spent fuel pit. The strategy for fuel assembly shuffle is as follows:

a) Spent fuel is removed from the core and placed into the fuel transfer system for removal to the spent fuel pit.

b) The remaining spent fuel is shuffled to new positions as identified in the core loading pattern.

c) Fuel assembly inserts such as burnable assemblies and control rod assemblies are shuffled as identified in the core loading pattern.

d) New fuel assemblies are brought in and loaded into the designated locations.

e) Alternatively, the entire core may be unloaded into the spent fuel pit, the inserts shuffled as needed and the new core returned to the reactor vessel.

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

Reactor Reassembly

a) The fuel transfer car is parked and the fuel transfer tube gate valve closed
b) The reactor vessel internals package is picked up by the plant crane and replaced in the vessel. The reactor vessel internals lifting rig is removed to storage
c) The control rod drive shafts are relatched to the RCC elements
d) The manipulator crane is parked
e) The old seal rings are removed from the reactor vessel head, the grooves cleaned and new rings installed
f) The refueling cavity is drained using either the Residual Heat Removal (RHR) system or the refueling cavity drain pump. If necessary, any water remaining in the cavity after normal RHR pump drain down can be drained via the Reactor Coolant Drain Tank (RCDT) and associated pumps
g) The reactor vessel flange surface is manually cleaned
h) The reactor vessel head is picked up by the polar crane, positioned over the reactor vessel and lowered
i) The reactor vessel head is seated
j) The reactor vessel to cavity seal (“Presray” inflatable seal, or rigid, segmented seal) is removed

k) The guide studs are removed to their storage rack. The stud hole plugs are removed
l) The head studs are replaced and retorqued
m) The canal drain holes are unplugged and the fuel transfer tube flange is replaced
n) Electrical leads and cooling air ducts are reconnected to the CRDMs
o) Vessel head insulation and instrumentation leads are replaced
p) The CRDM missile shield is picked up with the plant crane and replaced
q) Equipment and personnel access doors are closed and sealed
r) A hydrostatic test is performed on the reactor vessel
s) Control rod drives are checked
t) Pre-operational tests are performed

9.5.3 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 reliable cooling medium for removal of decay heat.

Basic provisions to ensure the safety of 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 in the Control Room of an abnormal core flux level

b) Violation of containment integrity is not permitted when the reactor vessel head is removed unless the shutdown margin is maintained greater than 5% $\delta k/k$

c) Whenever new fuel is added to the reactor core, a reciprocal curve of source neutron multiplication (inverse count rate ratio) 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. This provision allows the Control Room operator to inform the manipulator crane operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Malfunction Analysis

An analysis is presented in Chapter 14 concerning damage to one complete outer row of fuel rods in an assembly. This accident is assumed as a conservative limit for evaluating environmental consequences of a fuel handling incident.

Any suspected defective fuel assembly can be placed in a can designed to contain failed fuel and sealed to provide an isolated chamber for testing for the presence of fission products.

The failed fuel cans are typically stainless steel cylinders with lids that can be bolted in place remotely. An internal gas space in the lid provides for water expansion and for collection and

sampling of fission product gases. Various remotely operable quick-disconnect fittings permit connection of the can to sampling loops for continuous circulation through the can.

If sampling shows the presence of fission products indicative of a cladding failure, the sampling lines are closed off by valves on the can and the fuel assembly is removed to the spent fuel storage racks to await shipment. Design of the failed fuel test cans complies with 10 CFR 72.

Failed fuel can also be detected through the use of the in-mast sipping system (which is essentially a version of the sipping can that is permanently attached to the fuel handling equipment) and the poolside ultrasonic failed fuel detector, which uses a probe to examine each fuel rod for entrained water.

Fuel assemblies containing suspected top nozzle spring screw failures may be inspected in the Spent Fuel Pit using a spring scale test to determine whether an imposed tension of approximately five pounds results in visible deflection of any of the nozzle's springs [Reference NSE 00-3-008 RCS].

<u>Drop of Spent Fuel Element Cask Into Spent Fuel Pit</u>

As discussed in Section 9.12.4.3, Single Failure Proof Cranes for Spent Fuel Casks, the fuel storage building crane's main hook that handles spent fuel casks has been upgraded to single-failure-proof in accordance with the applicable guidelines of NRC NUREG-0554 (Single-Failure-Proof Cranes for Nuclear Power Plants, May 1979) and the applicable requirements of American Society of Mechanical Engineers ASME NOG-1-2004, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder) to support spent fuel cask handling activities, without the necessity of having to postulate the drop of a spent fuel cask. With the crane's main hook qualified as single-failure-proof, and when the crane is used as part of a single-failure-proof handling system for critical lifts as discussed in NRC NUREG-0800, Revision 1 of Section 9.1.5, Overhead Heavy Load Handling Systems, Sub-section III.4.C, a cask drop accident is not a credible event and need not be postulated. The following cask drop accident results are being retained since the analysis bounds other drop accidents that may be postulated in the fuel storage building and spent fuel pit even though a cask drop accident is no longer credible.

As indicated in Section 9.12.4.1, administrative controls and / or the presence of electrical limit switches or removable mechanical stops on the crane rails ensure that the cask or other heavy loads are not transported above fuel assemblies, hence, under no circumstances can the fuel assemblies be in jeopardy from the cask. However, the event that the cask would drop into the pit has been analyzed; the basic assumptions for analysis were as follows:

a) The drop would be from the cask's highest position which is 5 feet above the water surface and 43 feet above the bottom of the pit
b) The cask is fully loaded and weights 40 tons.

The results of the analysis indicate that the cask would hit the bottom of the pit with a velocity of approximately 40 ft/sec, assuming a conservative drag coefficient of 0.5. In comparison, the cask would have reached a velocity of 52 ft/sec if dropped through 43 feet in the air.

Using the Ballistic Research Laboratories formula for the penetration of missiles in steel, the depth of penetration of the cask into the 1-inch wear plate covering the ½-inch pit liner plate would be 0.32-inch, assuming the cask struck the wear plate while in a perfectly vertical position. In the event that the cask falls through the water at an angle, its terminal velocity

would be somewhat less because of the increased drag. However, the cask would strike the wear plate with an initial line contact and would penetrate the wear plate and the pit liner plate, causing some cracking of the concrete below. This reinforced concrete is a minimum of 3'-7" thick and rests on solid rock.

Water would initially flow through the punctured liner plate and fill the cracks in the concrete. As the pit is founded on solid rock and much of the bottom of the pit is below the surrounding grade, very little water can be lost from the pit. The capacity of the makeup demineralized water supply to the pit is 150 gpm. In addition, the spent fuel pit cooling system piping includes a 4" blind flange connection for temporary cooling.

Since the bottom of the spent fuel pit is an average 24 feet below grade and since no equipment areas are in the vicinity, there can be no flooding of other areas outside the Fuel Storage Building and subsequent damage to equipment.

<u>Siding Panel as a Missile</u>

Analysis has been made for the drop of a 32-1/2 ft long by 19 ft wide by 2 in thick insulated siding panel missile weighing 1860 lbs through 50 ft of free fall onto the water surface of the spent fuel pit. Although such a missile would logically be expected to plane in the water and impact the side walls of the storage pool, the analysis shows that even with the highly conservative assumption that it penetrates the water in a guillotine fashion, such that drag is based on the 19 ft x 2 in minimum cross sectional area, the drag and buoyancy forces prevent fuel damage.

The missile kinetic energy required to damage the fuel assembly cladding if 6900 ft-lbs. The missile kinetic energy variation with water depth is computed from:

$$D(KE)/dy = W - (z i 2g)(2KE)(W/g)^{-1} C_D A - z A_y$$

where KE = missile kinetic energy
W = missile weight
z = missile cross sectional area = 3.2 ft^2
z = water density = 62.4 lb/ft^3
C_D = drag coefficient = 1.0
g = gravity constant – 32.2 ft/sec^2
y = depth of water penetrated

Since the fuel storage pool is 40 ft deep with an excess of 23 ft of water over the top of the fuel assemblies, the postulated missile will be buoyed up before it strikes the fuel storage racks and fuel. Should it be postulated that tornado winds reduced the water level by 6 ft, the missile would impact the storage cell, with a striking impact energy of 2875 ft-lb per cell. However, there would be substantial margin to fuel clad failure.

The equivalent analysis was made for a 12-ft x 12-in x 4-in wooden plant striking the water vertically with a velocity of 90-mph. After 23 ft of water penetration, the plank kinetic energy is 4784 ft-lbs under the minimum drag area assumption. This would be insufficient to cause fuel failure even if the plant were to miss the storage rack and impact the top end of a stored fuel assembly. In order to illustrate the effect of planing, which would result from unsymmetrical impact of this missile on the water surface, a three dimensional model was analyzed.

Figure 9.5-4 shows calculated motion when the missile axis and missile velocity are in alignment with each other but impact is 5 degrees of vertical. Figure 9.5-5 shows the motion when the missile axis and initial velocity are misaligned by 5 degrees. It is seen that quite small deviations from a perfectly symmetrical water impact results in rotation of the missile, significantly reducing its penetration depth.

The effect of an automobile weighting 4000 lbs, entering the pool at 17 mph with 25 sq. ft contact area, was also analyzed. Due to the slower speed and the much larger drag area, the automobile will have an impact energy of 3133 ft-lb per cell, far less than the energy required for clad damage (see NSE 00-3-039 SFPC for details on missile analysis).

It is concluded that the storage pool water and storage cells provide effective protection against tornado missiles and that the chance of fuel damage by such missiles is low. Considering this, the low probability of a strong tornado striking the site, the fact that radioactive iodine in a stored fuel assembly is less than 10% of the shutdown value except of 6% of the year (first 2.3 half-lives), and the unstable and dispersive meteorological conditions accompanying a tornado, further protection is not needed.

Uplift for CRS

The individual spent fuel racks were not designed to withstand uplift forces, as a force applied to one cell will be distributed to the entire array of fuel cells (racks) in the pit. This occurs because the racks are not attached to the bottom liner of the pit but rather are interconnected.

The dead weight of adjacent fuel cells (racks) and fuel elements relieves any uplift force on the rack support members.

No force of significant magnitude can be applied to the fuel racks when removing a fuel assembly as the inside face of each rack opening was fabricated free of all burrs and rough edges and is smooth and clean.

9.5.4 Minimum Operating Conditions

Minimum operating conditions are specified in the Technical Specifications, FSAR Sections 1.3.6, 9.5.2 and 9.5.5, and plant procedures.

9.5.5 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, and the incore thermocouples, were tested at the time of installation. The tests were repeated on these electrical items before initial plant operation.

9.5.6 Inter-Unit Spent Fuel Transfer Operations

The NRC has issued Amendment 246 for the inter-unit transfer of spent fuel from Unit 3 to Unit 2 (Ref. 1). The Amendment is based on evaluations conducted for each aspect of the inter-unit transfer of fuel as documented in the Licensing Report (Ref. 2). The non-proprietary version of the Licensing Report is incorporated by reference in the UFSAR.

In preparation for inter-unit spent fuel transfer operations between the spent fuel pool (SFP) in the unit 3 Fuel Storage Building (FSB) and the SFP in the Unit 2 FSB, the HI-TRAC top lid is removed and the empty shielded transfer canister (STC) is placed inside the HI-TRAC transfer cask. The HI-TRAC / STC Centering Assembly centers the STC inside of the HI-TRAC. The HI-TRAC's solid top lid is installed to prevent any spilling of the water during the transfer process. Movement of the HI-TRAC (containing the STC) is performed using the Vertical Cask Transporter (VCT), and using the Unit 2 Low Profile Transporter (LPT), or using Air Pads at Unit 3.

The VCT moves the HI-TRAC containing the empty STC outside the Unit 3 FSB truck bay door. The HI-TRAC is lowered onto Air Pads and the VCT releases the Hi-TRAC. The Unit 3 FSB truck bay door is opened and the HI-TRAC is positioned inside the Unit 3 FSB truck bay beneath the FSB cask handling crane using the Unit 3 Air Pads. The HI-TRAC top lid is removed and the annulus between the STC and HI-TRAC is filled with demineralized water to the required level. The STC lid nuts and washers are removed and the STC is filled with SFP water. For STC spent fuel loading activities the Unit 3 SFP water must be adjusted to have boron concentration of greater than 2000 ppm.

The FSB cask handling crane is positioned over the STC and the STC Lift Lock is fastened to the STC lid and attached to the FSB cask handling crane. The STC is removed from the HI-TRAC and positioned over the cask loading area of the SFP. A set of remotely (or manually) actuated STC Lifting Devices attach the STC lid to the STC lifting trunnions. The STC is lowered into the cask loading area and the lid is removed.

For each fuel transfer cycle, up to twelve IP3 spent fuel assemblies including associated non-fuel hardware are loaded into the STC. The STC lid is positioned over the STC and installed. The STC Lifting Devices attach the lid to the STC lifting trunnions. After the Lifting Device arms are properly engaged to the lifting trunnions, the STC is raised to the surface of the SFP and any standing water on the lid is removed. A small amount of water is removed from the STC to avoid spilling during handling. Under the direction of Radiation Protection personnel radiological controls are established and surveys taken as the STC is raised and removed from the SFP, sprayed with demineralized water and placed directly into the HI-TRAC in the IP3 truck bay. The STC lid, nuts and washers are installed with the nuts left loose. The STC Lift Lock is disconnected from the STC top lid and removed. Free flow verification through the STC lid vent and drain lines is performed. The STC lid nuts are torqued and the STC seals are tested in accordance with ANSI N14.5 to assure that the STC is properly assembled for transfer operations. The required STC water level is established by blowing steam into, and water out of, the STC cavity thereby creating a compressible water vapor space. The STC top lid radiation level is measured to verify compliance with the Technical Specification requirements. As required by the Technical Specifications the pressure inside the STC is monitored for a period of 24 hours to demonstrate that there is not a significant amount of air in the STC and that a fuel misload has not occurred. Following completion of the pressure test the STC lid vent and drain port cover plates are installed and the seals are tested in accordance with ANSI

N14.5. The HI-TRAC top lid is installed and the bolts are tightened and the seal is tested in accordance with ANSI N14.5. The HI-TRAC side radiation levels are measured to verify compliance with Technical Specification requirements. The IP3 FSB truck bay door is opened and the loaded HI-TRAC is moved outside the IP3 FSB to the VCT on Air Pads using the Prime Mover.

The VCT travels inside the Protected Area on the approved haul route between IP3 and IP2. Prior to each transfer of spent fuel assemblies, the haul route is visually inspected and repaired as necessary.

The HI-TRAC containing the loaded STC is lowered from the VCT onto the IP2 LPT and moved into the IP2 FSB. Inside the IP2 FSB, the HI-TRAC is positioned beneath the 110-Ton Ederer Crane. A drain line containing a pressure gauge is connected to the HI-TRAC's top lid vent port and opened relieving any internal pressure. The HI-TRAC top lid bolts are removed and the HI-TRAC top lid is removed. The drain line is then attached to the vent port connection located on the lid of the STC and opened relieving any internal STC pressure. STC lid nuts and washers are removed.

The Lift Cleats (with the Lift Cleat Adapter) are attached to the STC lid (the STC Lifting Devices already are installed on the STC lid). The 110-Ton Ederer Crane is attached to the STC through the Lift Cleat Adapter. The STC lifting device arms are engaged with the STC trunnions. Under the direction of Radiation Protection personnel the STC is raised out of the HI-TRAC and positioned directly over the SFP cask loading area and lowered into the pool. IP2 Technical Specification 3.7.12 requires that boron levels in the IP2 SFP have a concentration of greater than 2000ppm which is also required for the STC spent fuel unloading activities.

With the STC in the SFP cask loading area, the STC Lifting Devices are released from the STC lifting trunnions and the STC lid is removed. The spent fuel assemblies and associated non-fuel hardware are removed from the STC and placed into the SFP racks in accordance with the requirements of the IP2 Technical Specification 3.7.13. The STC lid is positioned over the STC and installed. The lid's STC Lifting Devices are attached to the STC lifting trunnions and the STC is raised to the surface of the SFP. Any standing water in the lid is removed. Under the direction of Radiation Protection personnel the STC is raised and removed from the SFP, sprayed with demineralized water, and the water inside the STC is lowered before the STC is placed into the HI-TRAC. The STC lid studs and nuts are installed and the lid studs and nuts are tightened. The Lift Cleats are disconnected from the STC top lid and the Lift Cleats and Lift Cleat Adapter are removed. The HI-TRAC top lid is installed, the bolts are tightened, and the HI-TRAC containing the empty STC is then ready to be returned to the IP3 FSB.

REFERENCES FOR SECTION 9.5

1. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 246 to Facility Operating License No. DPR-64, July 13, 2012.

2. Holtec Report HI-2094289, Licensing Report on the Inter-Unit Transfer of Spent Nuclear Fuel at Indian Point Energy Center, Revision 10.

TABLE 9.5-1

FUEL HANDLING DATA

New Fuel Storage Pit	
Core storage capacity, equivalent cores	(approximately) 1/3
Equivalent fuel assemblies	72
Center-to-center spacing of assemblies, inches	20 ½
Maximum k_{eff} with unborated water for any degree of interspersed moderation	0.95
Spent Fuel Storage Pit	
Core storage capacity, equivalent cores	6.96
Equivalent fuel assemblies	1345
Number of space accommodations for failed fuel cans	0
Number of space accommodations for spent fuel shipping casks	1
Center-to-center spacing of assemblies, inches	
Region I	10.76
Region II	9.075
Maximum k_{eff} with unborated water	0.95
Maximum $_{eff}$ with unborated water for any degree of interspersed moderation	0.95
Miscellaneous Details	
Width of refueling canal, ft	3
Wall thickness for spent fuel storage pit, ft	3 to 6
Weight of fuel assembly with RCC (dry), lb	(approximately) 1606
Capacity of refueling water storage tank, gal	355,200
Minimum contents of refueling water storage tank for Safety Injection System or Containment Spray System operability, gal	342,200
Quantity of water required for refueling, gal	342,000

9.6 FACILITY SERVICE SYSTEM

9.6.1 Service Water System

Design Basis

The Service Water System (SWS) was designed to supply cooling water from the Hudson River to various heat loads in both the primary and secondary portions of the plant. Provision was made to ensure a continuous flow of cooling water to those systems and components necessary for plant safety either during normal operation or under abnormal or accident conditions. Sufficient redundancy of active and passive components was provided to ensure that cooling is maintained to vital loads for short and long periods in accordance with the single failure criteria. The system also provides backup water required for cleaning the traveling screens. A backup supply to the SWS can be provided by three non-seismic class pumps, as shown in Plant Drawing 9321-F-20333 [Formerly Figure 9.6-1A].

System Design and Operation

The Service Water System flow diagram is shown in Plant Drawing 9321-F-20333, and -27223, [Formerly Figure 9.6-1A, B, and C]. Six identical, vertical, centrifugal sump-type pumps located at the intake structure, each rated at 6000 gpm and 195 ft TDH at best efficiency point (BEP), supply service water to two independent discharge headers; each header being supplied by three of the pumps. An automatic, self-cleaning, rotary-type strainer is in the discharge of each pump to remove solids. These strainers can also be operated in the non-automatic mode. Each header is connected to an independent supply line. Either of the two supply lines can be used to supply the essential loads, with the other line feeding the non-essential loads. Table 9.6-1 identifies the design flow requirements of the Service Water System and the loads supplied by each header for various operating conditions.

Water is drawn from the river and passes under a debris wall, through a coarse screen and, finally, a fine mesh traveling ristroph screen. Electric heaters are provided in the driving head of the traveling screens, which are located inside a weatherproof building. The heaters are no longer in service. The constant motion of the screens plus the spray wash help to prevent icing of the screen panels. Each main circulating water pump is installed in an individual chamber, while the service water pumps are in a common chamber with two intakes. Each intake is capable of passing 100% of SWS demand and each is provided with a dedicated traveling screen. Openings are also provided between the service water pump chambers and the main circulating pump chamber on either side. These two openings can be opened by gates, but are normally closed.

The service water pumps can therefore obtain water through four separate intakes, each equipped with means to prevent debris from entering the pumps and each capable of supplying all the water required for the service water pumps. Even if the main circulating pump intake were 90% blocked, that intake alone would be capable of supplying all water required for the service water pumps at design conditions. The extreme low level condition for the river at the intake structure is 4’5” below the mean sea level at the site.

With the service water pump suctions at 10’-11 3/8” below the mean sea level at the site, adequate submergence at the service water pump suctions is assured. In addition, the intake structure has been evaluated for water levels higher than the maximum level of 15 feet above

the mean sea level at the site (see Section 2.5 for a discussion of the maximum river level for the Indian Point site); no foreseeable structural damage could impair the flow of water to the pump suctions for these high water conditions.

The intake structure and the steel framed grating enclosure around the service water pumps were designed as seismic Class I, and are therefore not subject to collapse under earthquake loading. Should the facade or any other architectural (i.e., non-structural) member of the intake structure be damaged or fall into the river, the flow of water from the river to the pump suction would not be impaired.

During normal operation, the essential loads listed in Table 9.6-1 can be cooled by any one of the three service water pumps on the essential header. The non-essential loads in Table 9.6-1 can be supplied by any two of the three service water pumps on the non-essential header. By manual valve operation, the essential loads can be transferred to the supply line carrying the non-essential loads and vice versa. During cold shutdown conditions, it has been evaluated that the essential and non-essential headers can be cross-connected to allow any pump or pumps to cool the entire service water heat load, subject to restrictions on positions for key valves, as evaluated by an updated revision to the original safety evaluation that addressed this concern.

The essential loads are those which must be supplied with cooling water immediately in the event of a blackout and/or Loss-of-Coolant Accident. The cooling water for these loads is supplied by the nuclear service water header. The non-essential loads are those which are supplied with cooling water from the conventional service water header. A non-essential service water pump must be manually started when required following a Loss-of-Coolant Accident.

The component cooling heat exchangers are considered non-essential loads on the Service Water System in the sense that service water to the component cooling water heat exchangers is not required during the injection phase of a LOCA.

The only accident heat loads due to forced flow on the Component Cooling Water System during the injection phase with Black Out is bearing cooling for the high head safety injection pumps. This load is satisfied by using the Component Cooling System as a heat sink. The auxiliary component cooling water pumps start and operate during the injection phase. However, as detailed in Section 6.2, forced cooling of the internal recirculation pump motors is not required during injection. Since the recirculation pumps do not operate in the injection phase, there is no cooling air circulation or motor heat generated, and thus a negligible amount of heat is transferred to the Component Cooling System through the motor cooler coils. However, for conservatism the thermal analysis assumes that the heat load transferred is consistent with operating recirculation pumps. Both the safety injection pump bearing heat load and the recirculation pump motor enclosure heat load are satisfied by using the Component Cooling Water System as a heat sink.

Following a simultaneous incident and blackout, the cooling water requirements for all five fan cooling units and the other essential loads can be supplied by any two out of the three service water pumps on the header which is designated to supply the nuclear and essential secondary loads.

The three pumps can be powered by the emergency diesels as described in Chapter 8. These emergency powered pumps are those necessary and sufficient to meet blackout and emergency conditions. Either one of the two sets of three pumps can be placed on the diesel starting logic.

The containment ventilation cooling units are supplied by individual lines from the containment service water header. Each inlet line is provided with a manual shutoff valve and drain valve. Similarly, each discharge line from the cooler is provided with a manual shutoff valve. This allows each cooler to be isolated individually for leak testing of the system in accordance with Technical Specification requirements. The ventilation cooler discharge lines are monitored for radioactivity by routing a small bypass flow from each cooler through redundant radiation monitors.

Upon indication of radioactivity in the effluent, each cooler discharge line is monitored individually to locate the defective cooling coil, which, when identified, remains isolated. The identification and isolation of an FCU or FCU motor cooler leak may be performed up until the entry into external recirculation. This is sufficient time to detect and isolate the leak, since the passive failure of a cooling coil is assumed to occur concurrently with the LOCA. The cooling coils and service water lines are a missile protected closed system inside the Containment and together with the isolation valves located just outside the Containment, satisfy the isolation criteria for containment penetrations as discussed in Section 5.2.

During normal plant operation, flow through the cooling units is throttled for containment temperature control purposes by a valve on the common discharge header from the cooling units. Two additional independent, full flow valves open automatically in the event of an engineered safeguards actuation signal to bypass the control valve. Both valves fail in the open position upon loss of air pressure, and either valve is capable of passing the full flow required for all five fan cooling units.

Should there be a failure in the piping or valves at the header supply water to the containment cooling coils, one of the two series valves in the center of the header can be manually closed and service will continue on the side of the header opposite of the failure. The supply line attached to this side of the header would supply the essential loads, whether or not it did so before the failure.

Likewise, operation of at least one component cooling heat exchanger is assured despite the failure of any single active or passive component in the system from the service water pumps to the heat exchangers themselves.

Following a simultaneous incident and a blackout, the component cooling heat exchangers are not needed during the injection phase; thus they are normally fed from the non-essential supply line. During the switchover to the recirculation phase, both component cooling heat exchangers are placed in service on the non-essential header.

During the switchover scenario, valves SWN 35-1 and 35-2 are throttled as necessary to control CCW header temperatures in response to initiating recirculation flow. This action will ensure that CCW header temperatures are maintained within optimum ranges. Maximum opening is also prescribed to prevent single Service Water pump run out for plant operation with RCS temperature 350^{o}F and greater.

Below 350°F, these position limits do not apply provided that: (1) two non-essential service water pumps are operating; (2) the non-essential SW header low pressure alarm is maintained clear; and (3) the valves are restored to their 27.5 and 27 degrees open positions should a reduction to single non-essential service water pump operation result.

This is achieved by the implementation of administrative controls to ensure that a dedicated Operator with direct communication from the Control Room takes manual action to restore the valves to their prescribed position limits well within 2 hours. These administrative controls also include procedural guidance and restrictions, such as not allowing this configuration with the headers being swapped or cross-tied.

The total service water flow required by the three Diesel Generator Jacket Water and Lube Oil Cooler sets during injection is 906 gpm (302 gpm per each cooler set). For recirculation, the service water flow requirement is 639 gpm total (213 gpm per each cooler). Cooling water to the Emergency Diesel Generator coolers is normally supplied from the essential supply line. [Deleted: TM TA-04-3-047/049 and 50.59 RVAL 04-0561-TM-00-RE entered in 2005 UFSAR Update and removed after outage]. Whenever any diesel generator is started and in normal operation, flow through the coolers is regulated by a parallel set of flow control valves (FCV-1176 and 1176A), each sized for a maximum flow of 1500 gpm and located in the common discharge line. When the diesel generators are not in operation, these valves are maintained in a closed position to isolate flow through the coolers. On a safety injection signal or a high lube oil / jacket water temperature condition, these valves open fully to ensure a sufficient supply of cooling water to the diesels. Should either valve fail, the other valve is adequate to assure the required cooling water flow. Each valve is opened on an air failure or on loss of electrical power to the solenoid valve on each operator. The inlet valving is arranged so that each of the three diesels can be served by either of the supply lines. Furthermore, the failure of any single active or passive component per single failure criteria will not result in loss of cooling water to more than one diesel generator.

Since each flow control valve is sized to pass the full flow requirement to meet the single failure criterion, if both valves go to their full open positions, a flow rate greater than that required will be seen in the Emergency Diesel Generator Cooling loop. This results in reduced cooling capacity for other essential service water users during the injection phase. For this reason, a throttle valve, SWN-55, has been installed in the common discharge line from the diesel generator coolers. The valve is manually operated butterfly valve whose disc has been drilled with four (4) 1 ¾ inch diameter holes. The incorporation of these holes ensures that in the event of a mechanical failure of the valve or a valve component, the resulting flow is limited to that required by the Diesel Generator Cooling loop (approximately 302 gpm to each diesel, for a total service water flow of 906 gpm, when the diesel generators are operating). This configuration ensures a balanced service water flow in the event of an instrument air loss, concurrent with safety injection actuation.

Three backup service water pumps provide cooling water from the discharge canal for the containment ventilation cooling coils, the Control Room Air Conditioners, the containment ventilation fan motor coolers, the instrument air compressors, and the diesel generator coolers in the unlikely event that a storm driven vessel damages the service water intake structure. The backup service water pumps are provided with automatic, continuous, rotary-type strainers and are manually valved to discharge to the header designated to supply the essential loads. Two of the three pumps can be powered by the emergency diesels.

In order to satisfy Appendix R licensing commitments, one of the backup service water pumps is designated as an Appendix R service water pump to supply cooling water to the component cooling water heat exchangers in the event of a fire. This pump is powered from the Appendix R Diesel Generator. The backup service water valve pit is protected from tornado-originated missiles by a tornado proof structure.

Plant Drawings 9321-F-20333, and -27223 [Formerly Figures 9.6-2A & B] present the flow distribution on the SWS for normal operation. Plant Drawings 9321-F-20333, and -27223 [Formerly Figure 9.6-3] shows at the flow distribution on the SWS for the injection phase after LOCA assuming maximum safeguards equipment operating. Plant Drawings 9321-F-20333, and -27223 [Formerly Figures 9.6-4A & B] indicate the flow distribution on the SWS for the recirculation phase after a LOCA, assuming maximum safeguards equipment operating.

Ultimate Heat Sink

The ultimate heat sink is the Hudson River, which is capable of providing sufficient cooling for at least the required thirty days:

(a) to permit simultaneous safe shutdown and cooldown of both operating nuclear units at the Indian Point site and maintain them in a safe condition, and
(b) in the event of an accident in one unit, to permit control of that accident safely and permit simultaneous safe shutdown and cooldown of the remaining unit and maintain it in a safe shutdown condition.

The ultimate heat sink is capable of withstanding the effects of the most severe natural phenomena associated with the Indian Point site, other site related events and a single failure of man-made structural features. The ultimate heat sink consists of a single source.

For the normal service water supply system, there are no connecting canals or conduits between the heat sink and the intake structure. The intake structure is physically located directly on the bank of the Hudson River, which is tidal at the site location. The heat sink, therefore, consists entirely of natural features.

The SWS design met the requirements of AEC Safety Guide No. 27 with respect to all natural phenomena associated with the site (earthquake, tornado, hurricane, flood, drought, and low tide).

Site related accidents such as ship collisions, airplane crashes, oil spills, and fires are not expected to affect the availability of the heat sink. Reasonable combinations of less severe natural phenomena and accidental phenomena are not expected to have significant consequences.

Failure of man-made features would have to be multiple in order to block both the fresh water flow from upstream and tidal flow from downstream. These could be postulated as bridge failures both upstream and downstream of the site. Dam failures, on tributaries to the Hudson, would not have an adverse effect of the heat sink since this occurrence would raise the water level of the heat sink.

The SWS design is not capable of accepting the highly unlikely occurrence of river diversion.

The SWS was designed to accept the rather severe combination of natural and accidental phenomena of a storm driven ocean vessel crashing into the intake structure, rendering it inoperable. In this event, the ultimate heat sink includes the Hudson River and discharge canal to supply the backup service water pumps located in the discharge canal. The backup service water pump design is not capable of withstanding the additional simultaneous occurrence of a Design Basis Earthquake, as these components are designated as seismic Class III components.

Design Evaluation

The Service Water System was designed to fulfill required safety functions while sustaining: (a) the single failure of any active component used during the injection phase of a postulated Loss-of-Coolant Accident, or (b) the single failure of any active or passive component used during the long-term recirculation phase.

The operating modes of the IP3 SWS have been identified as normal, injection post-LOCA and recirculation post-LOCA. The postulated failure conditions of the SWS must include consideration of the limiting case for each operating mode of the system. The limiting failures are:

1. Loss of the 10 inch turbine building service water supply header during normal (plus seismic) conditions.
2. Loss of instrument air (LOIA), during the post-LOCA injection phase concurrent with single active component failure.
3. Loss of a SW pump on both the essential and non-essential headers (resulting from a diesel generator failure) during the post-LOCA recirculation phase.

The flows to individual essential components for each of these cases, as calculated by using a hydraulic analysis computer program, are presented in Table 9.6-2.

As shown by these results and as discussed in the following section, the SWS will perform its required safety function for the limiting postulated failure of each of the operating modes identified above.

In addition, as a result of the elevated river water temperature experienced during the summer of 1988, the Authority undertook an effort to permanently increase the design basis ultimate heat sink temperature from 85° F to 95° F. This effort included an evaluation of certain plant equipment ultimately cooled by service water to perform all normal and safety functions at river water temperatures up to 95° F. The evaluation concludes that all equipment required for safe plant operations serviced by 95° F service water will operate acceptably and the current safety limits affected by the SWS temperature will be met.

Postulated Breaks

During the recirculation phase of a Loss-of-Coolant-Accident, the following components are supplied by the essential service water header: the instrument air closed cooling water system, the control building air conditioners, the containment fan coolers, and the diesel generators. The recirculation phase flow requirements for these components are identified in Table 9.6-1.

Using system alignment shown in Plant Drawing 9321-F-20333 [Formerly Figure 9-6-1A], where service water pumps Nos. 31, 32, and 33 are on the non-essential (conventional) header and service water pumps Nos. 34, 35 and 36 are on the essential (nuclear) header, the valve

positions during the normal injection and recirculation modes of operation are as indicated in Table 9.6-3. During the switchover recirculation phase, valves FCV-1111, FCV-1112, SWN-6 and SWN-7 are closed to isolate service water to the circulating water pump seals and backup screen wash.

In addition to the system alignment required to be performed at the initiation of the recirculation phase, without a loss of instrument air, the operator must override the automatic reset of flow and temperature control valves FCV-1176, FCV-1176A, TCV-1104 and TCV-1105 in order to assure the flow conditions assumed in the service water system hydraulic analysis. All of the above are to be accomplished during the changeover from injection to recirculation, as part of the emergency operating procedure for that changeover.

Subsequent to the issuance of the Safety Evaluation Report for Indian Point 3, a pipe break analysis was performed on the Service Water System. As part of an effort to demonstrate the adequacy of replacement pumps for the Service Water System, the analysis was reviewed and the following observations were made with regard to failure criteria that should be applied to the SWS piping:

1) Failure data contained in the Indian Point Probabilistic Safety Study indicate that the probability of total failure of SWS piping during the critical 24-hour period following a LOCA does not constitute a credible event.

2) The SWS is a moderate energy fluid system. Pipe failures within such systems are postulated to be limited to small through-wall leakage cracks. The break criteria of the original analysis included guillotine and slot breaks which are exceedingly conservative for the SWS.

As a result of these observations, the SWS was reanalyzed in 1989. On the basis of the results from this reanalysis, it was concluded that the Indian Point 3 SWS, as then configured, will perform its safety functions under accident conditions and with the previously described postulated component failures. The NRC approved the use of limited size breaks in the analysis of IP3 service water piping failures as documented in reference 7. Table 9.6-2A provides the service water system flow distribution for the passive breaks considered during the recirculation phase of a LOCA, consistent with the current licensing basis of limited size through-wall leakage cracks. Furthermore, it has been demonstrated in Reference 8 that postulated pipe failures of SWS Seismic Class I piping will not flood the PAB or impact the functionality of safety related equipment such as the RHR pumps.

The validity of the hydraulic analysis computer program used in the analysis of the postulated failure conditions has been verified by comparison of results to another hydraulic computer code. The verification problem considered sufficient node points and flow branches to demonstrate the adequacy of the computer program with respect to similar nodes and branches used in the other computer program. The results of the verification showed excellent correlation of flows between the two computer programs.

As a part of the Indian Point 3 Pre-Operational Test Program, functional tests of the Service Water System were run to verify equipment performance. The results of pre-operational, functional and periodic tests for the SWS are available for inspection at Indian Point 3.

Tests and Inspections

Each service water pump was subjected to a hydrostatic test in the shop, in which all pressurized parts were subjected to a hydrostatic pressure of the greater of 1.25 times the shutoff head or 1.50 times the rated head of the pump. In addition, normal capacity vs. head tests were made on each pump.

All valves in the Service Water System underwent a shop hydrostatic test in accordance with the applicable manufacturing code or standard. Gate, globe and check valves were tested to 225 psi across the seat. Butterfly valves were tested to 200 psi across the seat. Butterfly control valves were tested to 125 psi and diaphragm valves were tested to 150 psi in 3” size and 175 psi in 2” size across the seat.

All service water piping was hydrostatically or leak tested in the field in accordance with USAS B31.1. As per the construction code record, the 1967 edition of USAS B31.1, this was accomplished using one of the two methods:

1. The piping was hydrostatically tested to 1.5 times the design pressure or the maximum test pressure of the limiting vessel or component in the section of piping to be tested, or

2. If it was not feasible to isolate a section of pipe, the piping would be slowly brought up to system operating pressure and held at pressure for a period of time to demonstrate leak-tightness

Upon commissioning, the ISI portions of the service water system were governed by ASME Section XI for ISI Class 3 piping. Per the original edition of ASME XI used at the site (1974, w. summer 1975 addenda), piping was required to be tested to 1.10 times the system design pressure. This was upheld by the 1983 edition, with summer 1983 addenda of ASME XI used in the second ISI cycle. Retests using the original construction code were also acceptable. Future tests and inspections will be performed in accordance with ASME Section XI code of record requirements as described in the IP3 Inservice Inspection Program in effect.

Electrical components of the Service Water System are tested periodically.

An Erosion/Corrosion program using both visual and volumetric inspection methods was implemented during the 8/9 refueling outage. The visual method is accomplished by using a robotic crawler with a high resolution camera. The crawler is remotely controlled and can advance through the piping while making a video record of the internal pipe surface.

9.6.2 Fire Protection

9.6.2.1 Design Bases

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

Fire Protection Criteria

> Criterion: The facility is designed so that the probability of fires and explosions and the potential consequences of such events does not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility whenever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features. (GDC 3 of 7/11/67)

Fire Prevention in all areas of the plant was provided by structure and component design which optimizes the containment of combustible materials and which maintains exposed combustible materials below their ignition temperature in the design atmosphere. Fire control requires the capability to isolate or remove fuel from an igniting source, or to reduce the combustibles' temperature below the ignition point, or to exclude the oxidant, or preferably, to provide a combination of the three basic control means. The latter two means were fulfilled by providing fixed or portable fire fighting equipment of capacities proportional to energy that might credibly be released by fire.

All areas subject to radioactive contamination or toxic combustion products were designed to rely on manual fire protection. Access to these areas is controlled by plant health physics personnel. These areas are found in the Containment Building, Fuel Storage Building, Primary Auxiliary Building, and Waste Holdup Tank Area.

Indian Point 3 was designed on the basis of limiting the use of combustible materials in construction and of using fire-resistant materials to the greatest extent possible.

The fire protection system was designed to achieve the following objectives:

1) Provide automatic fire detection in those areas where the fire danger is greatest.
2) Provide fire extinguishment by fixed systems of water, CO2 and foam, actuated automatically or manually in those areas where the fire danger is greatest.
3) Provide manually operated fire extinguishing equipment, including hose reels, and CO2, dry chemical, water, halon, foam and MET-L-X types of hand portable extinguishers

The Fire Protection System was designed in accordance with standards of the National Fire Protection Association (NFPA) and where not, deviations are identified and justified.

The Indian Point 3 Fire Protection System was designed and installed to seismic Class III standards, except for the Fire Protection System piping supports in the Diesel Generator Building, Control Building, Primary Auxiliary Building, Fan House, Auxiliary Feed Pump Room, Electrical Tunnel, Containment Building and Fuel Handling Building which was designed and/or upgraded to seismic Class I criteria. This design was used to ensure that the Fire Protection piping in Safety Related areas was seismically analyzed and will not impact seismic Class I components in a seismic event. Although these piping supports were designed and/or upgraded to seismic Class I criteria, they are not classified as Safety Related as implied in paragraph 16.1.7 of the UFSAR.

Applicable Codes and Standards During Design Phase of the Plant

From NEPIA - MAERP: Basic Fire Protection for Nuclear Power Plants (Revised and dated March 1970).

Water Supplies

Water supply systems satisfied conditions outlined in above noted NEPIA guide under Section II, PROTECTION; A, except that the wiring arrangement was evaluated with regard to National Fire Protection Association Pamphlet No. 20 indicated in Section II,A,4.

Yard Mains and Hydrants

Yard mains and hydrants satisfied the condition outlined in NEPIA guide under Section II, PROTECTION; B.

Sprinkler and Water Spray Systems

The Lube Oil Storage Room and Diesel Generator Room wet pipe sprinkler system satisfied the conditions outlined in NEPIA guide under Section II, PROTECTION; 1, 2 D and E, and 3. The water spray systems satisfied conditions outlined in NEPIA guide under 4A and 5. The Hydrogen Seal Oil Unit listed in the NEPIA guide under 4D was protected with a system that also satisfied these conditions except that it was a foam-water instead of a water spray system. This type of system also protected the Lube Oil Reservoir, Lube Oil Storage Tank and Boiler Feed Pump Oil Console and Oil Accumulators.

Portable Fire Extinguishers and Inside Hose Connections

These two items satisfied the conditions outlined in the NEPIA guide under Section II, PROTECTION; Items D2 and D1, respectively.

Special Protection

The electrical tunnel fire protection systems satisfied the conditions outlined under Section III, A of the NEPIA Guide. This was a closed head preaction system.

Materials

The materials used in the design of the Fire Protection System are in accordance with the standards of the National Fire Protection Association (NFPA). This includes pipe, fittings, valves and hydrants. Changes to these materials are controlled under the Fire Protection Program.

On February 17, 1981, 10 CFR 50.48 and Appendix R became effective. Appendix R to 10 CFR 50 established fire protection features required to satisfy Criterion 3 of Appendix A to 10 CFR 50 with respect to certain generic issues related to nuclear power plants licensed to operate prior to January 1, 1979. As a minimum, 10 CFR 50.48 required all licensees to conform to the requirements of Section III.G., III.J, and III.O, of Appendix R which addresses fire protection of safe shutdown capability, emergency lighting, and reactor coolant pump oil collection systems, respectively. Other sections of Appendix R apply to those licensees who had open items remaining from the BTP-9.5-1, Appendix A review. The review of Indian Point 3 to BTP 9.5-1, Appendix A was completed, as documented in the NRC Safety Evaluation Reports dated March 6, 1979 and May 2, 1980.

A reevaluation of Indian Point 3 against the requirements of Section III.G of Appendix R to 10 CFR 50 was completed in August 1984. The report submitted to the NRC on August 16, 1984 described the bases on which Indian Point 3 conformed to Section III.G of Appendix R. The report provided a historical chronology of correspondence between the NRC and the Authority on Appendix R compliance by summarizing all pertinent documentation submitted to the NRC in response to 10 CFR 50.48 and Appendix R through August 1984.

The Appendix R Reevaluation was supplemented September 19, 1985 and included new exemptions to Section III.G. By letter dated June 14, 1985, an exemption from the requirements of Section III.J was requested. Additional information was provided by letters dated March 15, 1985 and September 10, 1986. By Safety Evaluation dated January 7, 1987, the NRC completed their review of the Appendix R Reevaluation and granted certain exemptions.(1) A new report was issued in May 1995 which supersedes the August 1984 report. The new report will be maintained by periodic updates.

NRC approval of the Indian Point Unit 3 fire protection program including safe shutdown capability is provided in fire protection safety evaluations (SEs) dated:

- September 21, 1973
- March 6, 1979
- May 2, 1980
- November 18, 1982
- December 30, 1982
- February 2, 1984
- April 16, 1984
- January 7, 1987
- September 9, 1988
- October 21, 1991
- April 20, 1994
- January 5, 1995
- March 29, 1995

Incorporation of the NRC approved fire protection program into the UFSAR represents one of the elements of Generic Letter 88-12 required to remove the fire protection requirements from the Technical Specifications (TS). The fire protection program requirements were removed from the TS under Amendment No. 157. This information needs to remain in the UFSAR. Entergy may make changes to the NRC-approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

The Indian Point 3 Fire Protection Program description is provided separately in the following documents:

- IPEC Fire Protection Program Plan
- IP3 Fire Hazards Analysis report
- IP3 Safe Shutdown Analysis report

The Fire Protection Program Plan as required by 10 CFR 50.48 is included in IPEC Administrative Procedure, "IPEC Fire Protection Program Plan." The Administrative Procedure discusses the program purpose, design, implementation and maintenance thereof. It states the fire protection objectives and defines the program bases and key elements.

The IPEC Fire Protection Program Plan also identifies the fundamental fire protection documents and describes the method of compliance, as well as provides an explanation of the organization, responsibilities, and administrative controls which comprise the Fire Protection Program for the Indian Point 3 Nuclear Power Plant.

The IPEC Fire Protection Program Plan has been prepared to assist in accomplishing the following objectives:

- Adhere to the requirements of Appendix R to 10 CFR 50.
- Identify those documents that provide the basis for the Fire Protection Program
- Identify the location of all commitments made by the New York Power Authority relative to Appendix A to BTP (APCSB) 9.5-1, and 10 CFR50 Appendix R.
- Identify the documents which describe plant systems and procedures required to safely shutdown and cool down the plant, in the event of a fire in any plant area.
- Facilitate identification of the documents that identify Fire Protection equipment and safe shutdown components.

9.6.2.2 Fire Areas and Fire Area Boundaries

For the purposes of establishing compliance with 10 CFR 50.48 and Appendix R, Indian Point 3 has also been divided into six distinct fire areas with physical boundaries. An additional fire area, the yard area, has also been defined and includes the areas exterior to the plant structures. The six defined fire areas are:

1) Containment
2) Primary Auxiliary Building
3) Electrical Tunnels
4) Control Building
5) Turbine Building
6) Auxiliary Feedwater Pump Room

These fire areas have been subdivided into fire zones for the purposes of fire hazards analysis.

Relief from the requirements of Appendix R for the above listed fire areas is described in detail in References 1, 5 and 6.

Fire Barriers

Substantial fire barriers have been provided throughout the plant. An evaluation including a fire hazards analysis concluded that the basic wall, floor and ceiling structures bounding each fire area have adequate fire resistance to prevent the spread of unsuppressed fire through the barriers. The required rating of each barrier has been established based on the combustible loading and fire severity that is present on either side of the barrier as well as the function of the barrier; i.e., on exterior wall or a barrier separating defined fire areas. Generally, the rating of a fire barrier does not consider the presence of any fire detection or suppression systems on either side of the barrier. Walls specifically designed as fire barriers include the following:

1) Reinforced concrete fire barrier walls between main transformers and in some areas between main transformers and the Turbine Building. In addition, the main transformer area has reinforced concrete oil barriers below grade with broken stone fill to catch oil from transformers in the event of a spill or rupture.

2) 16” reinforcing concrete fire walls from floor to ceiling between diesel generator cubicles. In addition, the pipes trench between cubicles between cubicles is filled with compacted sand fill after pipes are installed and special fire guards are installed at floor drains connected to a connected to a common drain pipe to prevent passage of burning oil.

Common walls between adjacent plant structures are concrete or concrete block except for the Electrical Penetration Tunnel/Electrical Tunnel and Primary Auxiliary Building/Waste Holdup Tank Pit interfaces, which are separated by metal partitions.

The fire barriers separating the fire areas listed above are required for compliance with Section III.G and III.L of Appendix R to 10 CFR 50. Each fire barrier along with its construction and fire resistive rating is identified in the Fire Hazards Analysis (FHA). The FHA will be maintained by periodic updates.

Fire Barriers Penetration Protection

Fire Doors

Doors in fire area barriers throughout the plant which separate redundant safe shutdown systems or protect alternate shutdown systems from significant fire hazards, are fire rated or evaluated as being adequate.

The fire doors, frames and construction are generally three-hour fire rated, and are either: kept closed, provided with door closers and periodically inspected to ensure that the door is in the closed position; held open, provided with fire actuated release devices and periodically inspected to ensure that the doorways are free of obstructions; locked closed and periodically inspected to ensure that the door is in the closed position; or kept closed and electronically supervised to alert control room operators if the door is left open.

The doorway between the Control Room and Turbine Building operating floor is a three-hour equivalent fire door which consists of a metal plate which falls over a window on the door in the event of a fire. A windowless metal door to the outside at Elevation 55 feet of the Primary Auxiliary Building has been installed.

The north, east, and west walls surround #31 and #32 Battery Rooms on 33' elevation of the Control Building were found to be non-rated barriers. Evaluation of this configuration found that the loss of the 31 and 32 Batteries would not preclude the ability to achieve safe shutdown.

In lieu of three-hour fire-rated doors in the barriers separating the Primary Auxiliary Building from the transformer yard, a manually actuated water curtain, in conformance with NFPA 15, "Water Spray Fixed Systems," protects this opening.

The door separating the Auxiliary Feedwater Pump Room from the Turbine Building at the 18' elevation is not rated. An evaluation of this door in accordance with the provisions of Generic Letter 85-01, has demonstrated the capability of the door for fire barrier penetration protection.

Within the Control Building, fire doors are provided between diesel generator cubicles and between the cubicles and the Control Building. The fire doors are 3-hour rated, Class A fire doors which have been listed or approved by a nationally recognized testing laboratory.

Where penetrations have been created in fire doors, fire door frames or transoms, appropriately related penetration seals have been installed which maintain the rating of the fire door assembly.

Fire Dampers

Dampers which are rated for three hours of fire resistance have been installed in HVAC openings and duct work to maintain the integrity of the fire rated barriers. These barriers separate fire zones in the Control Building, Diesel Generator Building, Primary Auxiliary Building, and the Fan House. Fuse links on the fire dampers will melt at a predetermined temperature which will cause automatic closure of these dampers.

Electrical Cable and Mechanical Penetration Seals

Fire barrier penetration seals are installed with the intent that they remain in place and retain their integrity when subject to an exposure fire and subsequently, a fire suppression agent.

Electrical and mechanical penetrations in fire barriers are sealed with several types of construction materials. Silicone foam and silicone elastomer comprise the two principal types of penetration fire seals used at IP3. Fire seals in fire barriers providing area separation have been qualified to a typical design that has passed a fire endurance and hose stream test.

Results of several separate fire tests have been used to evaluate the silicone foam and silicone elastomer fire seal designs. The review methodology used to evaluate the these qualifying tests required that the tests configuration be subjected to a 3-hour fire endurance which corresponds to the standard-time-temperature curve as specified in ASTM E-119, and an acceptable hose stream test.

The tested configuration has withstood the fire endurance test without the passage of flame or gases hot enough to ignite cable, other penetrating items or seal material on the unexposed side.

The temperature levels recorded for the unexposed side were analyzed and shown to be sufficiently below the self-ignition temperatures of the cables, other penetrating items or seal material used; and the fire seal remained intact without the projection of water beyond the unexposed surface during the hose stream test. The limiting component for unexposed side temperatures was found to be cable. Seven hundred degrees Fahrenheit (700^{o} F) has been established as the limiting temperature which is sufficiently below the self-ignition temperatures of the cables used at IP3.

Fire Wraps and Radiant Energy Shields

Fire wraps and radiant energy shields have been installed on various cables, cable trays, and conduits in the Containment, Electrical Tunnels, and PAB. The wraps consist of HEMYC blankets and 3M INTERAM system enclosures. The radiant energy shields are comprised of marinite or transite fire board.

These protective features were added to Safe Shutdown related instrumentation in the Containment to establish compliance with Section III.G.2.f of Appendix R. Fire wraps have been installed to protect:

1) Wide Range RCS pressure transmitter PT-402 conduit from the transmitter to the electrical penetration inside containment.
2) Source Range neutron temperature flux N-31 conduit from its preamp box to the electrical penetration inside containment.
3) Wide Range RCS temperature elements and cabling for TE 413 A & B at the electrical penetrations.
4) Steam Generator wide range level instrument LT-417D at the penetrations.
5) Steam Generator wide range level instrument LT-447D at Rack 21.

Radiant energy shields have been installed at instrument racks 19 and 21 to protect steam generator wide range level transmitter LT-417D and pressurizer level transmitter LT-459, respectively. These shields protect the instruments from the effects of a floor based fire. In addition, radiant energy shields are installed on instrument trays where twenty feet of horizontal separation does not exist between redundant safe shutdown cables.

In the PAB, the power cabling for one component cooling water pump is wrapped where it is located within 20 feet of the redundant pump's cabling. A partial height, noncombustible barrier constructed of marinite board, partially surrounds CCW pump 33. The barrier includes a fire door rated for 3 hours to facilitate access to the pump and motor. The barrier protects the pump and motor from the radiant energy generated in a floor based fire, which could impact the redundant CCW pumps.

In the Electrical Tunnels, one channel of safe shutdown instrumentation is at both ends of the Tunnels where the redundant safe shutdown channels are not separated by 20 feet or by concrete floor/ceiling assembly which separates the upper and lower tunnels.

An exemption to the requirements of 10CFR50III.G.2 has been granted to the extent that fire wrap need not be rated for one hour in the Electrical Tunnels and penetrations area. This

protection, combined with the detection and suppression systems located within the tunnels, constitutes compliance with Section.III.G.2.c of Appendix R.

Specifically, channel IV safe shutdown instrument cables exiting the containment in the upper electrical penetration area are protected in HEMYC [Deleted]barrier until the cables pass through the floor into the lower electrical tunnel. Pressurizer pressure and level, steam generator level, RCS loop 31 hot and cold temperature instrumentation cables routed in the JD cable trays are HEMYC wrapped. Source range neutron detector conduit (N-31) is also HEMYC wrapped.

Within the Turbine building, the fire protection features installed at each of the cubicles of Manhole 33 provide separation between the power feeds to normal service water pumps and those of Back-up Service Water Pump 38. These features have been evaluated in accordance with the guidance of Generic Letter 86-10 and found acceptable.

An exemption to the requirements of Appendix R, Section III.G.2 has been granted to the extent that the redundant wide-range steam generator water level sensing lines and the redundant pressurizer level sensing lines, located inside containment, need not be separated by non-combustible radiant energy shields.

In response to Regulatory Guide 1.189, Rev. 2, and the effects of “Multiple Spurious Operations” (MSO) for the applicable scenarios given by the PWR Generic MSO List in NEI 00-01, Rev. 2, on the credited post-fire safe-shutdown capability, 3M INTERAM fire wrap is installed on selected cables and conduits in the ABFP room. The applicable MSO scenarios involve excessive/uncontrolled delivery of auxiliary feedwater to the credited steam generators (or non-credited steam generators) during a post-fire shutdown scenario, caused by spurious start of the AFW pump(s) and simultaneous spurious opening of associated feed regulating valves.

9.6.2.3 Fire Suppression Systems

The Fire Protection System for Indian Point 3 was originally designed as an extension of the Fire Protection System for Indian Point 1, owned by Consolidated Edison. After incorporation of a series of modifications the Indian Point 3 Fire Protection System was made independent from the Indian Point 1 Fire Protection System and met the criteria (GDC 3) specified in Section 1.3 (see Plant Drawings 9321-F-40903, and -40913 [Formerly Figures 9.6-9A and 9.6-9B]).

Water Supply and Distribution System

A separate fire water supply system was installed at Indian Point 3 and was connected to the Fire Protection System. The supply system consists of two 350,000 gallon storage tanks and their associated piping, electrical and instrumentation systems, which serve as the source of fire protection system water and as the supply for the Indian Point 3 makeup water treatment facility. The supply to the tanks is from the City Water System and is automatically controlled to maintain a minimum of 300,000 gallons of water in each tank dedicated for fire protection.

The Indian Point 3 Fire Protection System is provided with one 2500 gpm motor driven fire pump (110 psig) and with one 2500 gpm diesel driven fire pump (110 psig). The motor driven fire pump is powered through a transfer switch from two sources: 480v Switchgear 312 BKR 3B (Normal) and 480v Switchgear 5A, BKR 24D (Emergency). System pressure is maintained by two motor driven jockey pumps. Control valves at the fire pumps are electrically supervised.

The fire pumps are located in the fire pumphouse and are separated by a concrete wall; the jockey pumps are also located within this building.

The fire pumps are sized based on the largest water demand. The largest water demand is comprised of the largest assumed sprinkler system demand or water spray system demand plus an assumed hose stream demand of 500 gpm. The largest water demand is the assumed demand for the wet pipe sprinkler system provided for the Turbine Building, elevation 15'0", north half of the building plus an assumed hose stream demand of 500 gpm, or approximately 2870 gpm. The design considerations used in evaluating the acceptability of the water supply and distribution system included the ability to supply the largest water demand assuming a failure of either fire pump.

The fire water distribution system is designed as a loop system to permit water flow in either direction. Sectionalizing valves are located throughout the system to permit isolating portion of the system for repairs or maintenance activities without impairing the entire system or isolating a break without affecting both the standpipe and fire suppression systems protecting a safety related area. Sectionalizing valves are shown in Plant Drawings 9321-F-40903, and -40913 [Formerly Figures 9.6-9A and 9.6-9B]. Sectionalizing valves are either post-indicator type, key operated type (i.e., a buried valve with a roadway box) or OS&Y type. The position of all valves in the fire water distribution system, except key operated valves, whose closure may cause loss of fire water supply, are supervised by either electrical tamper switches, locks and chains, or tamper proof seals, and periodically visually inspected to ensure that the valve is in its correct position. The position of key operated valves are periodically manipulated to ensure that the valve is in its correct position. Marking signs denote the location of key operated valves in the yard.

Fire protection is provided to the exterior plant areas by yard fire hydrants. Hydrants can be removed from service for repair without shutting off a portion of the fire loop, by means of auxiliary gate valves which are provided on each hydrant lateral. Hydrants exposed to traffic are provided with vehicle barricades and post-indicator valves. Hydrants are of the dry barrel type which self drain to prevent freezing of the hydrant. Hose houses are provided at each of the fire hydrants and at the test header at the pump house. Fire hose and other required equipment is provided at these hose houses. National Standard (NH) fire hose threads provided on all hoses, nozzles and fittings at hydrant hose houses are compatible with offsite fire department.

Valved branches from the underground fire loop system supply interior fire protection lines in the enclosed sections of the plant.

<u>Fire Hose Stations</u>

The plant fire water protection loop supplies standpipes in the following buildings:

- Administrative Services Building
- Radioactive Machine Shop
- Primary Auxiliary Building
- Fuel Storage Building
- Fan House
- Control Building
- Auxiliary Feedwater Pump Building
- Outage Support Building
- Condensate Polisher Building
- Auxiliary Boiler Building

Intake Structure
Security Building
Turbine Building

The standpipe are generally located in protected stairways and each contains drain valves, hose racks/ reels on each landing, and an air vent valve or venting capability.

Standpipes and hose racks/reels are located such that all portions of each elevation of the building are within 30 feet of a nozzle, attached to not more than 100 feet of 1-1/2 inch fire hose except as described below.

The Electrical Tunnels including electrical penetration areas are protected by the fire hose station on the 54 foot elevation of the Fan House and the station located on the 33 foot elevation of the Control Building, east stairway. The fire hose station in the Control Building can be augmented with 300 feet of 2 inch fire hose and valved wye to ensure coverage of all areas of the Electrical Tunnel. The pump and motor area on the waste holdup tank pit area is protected by the fire hose station located on the 41 foot elevation of the Primary Auxiliary Building. Fire hose stations inside the Containment Building are provided to protect areas containing a significant amount of electrical cable and the areas around the reactor coolant pumps (RCPs). In addition to the fire hose stations located in the east stairwell of the Control Building, the fire hose stations in the Turbine Building are located so that they can also be used to fight a fire in portions of the 480V switchgear room, cable spreading room, battery rooms, control room and Diesel Generator Building.

Control valves are provided for standpipes protecting safety related areas at the distribution header connection to allow isolation of individual standpipes without affecting other standpipes. Control valves for these standpipes are shown on Plant Drawings 9321-F-40903, and -40913 [Formerly Figures 9.6-9A and 9.6-9B].

Fire hose with national standard (NH) hose threads, nozzles and fittings, and suitable spanner and valve wrenches are provided at each hose station. Yard post-indicator valves and interior control valves are tagged to indicate the standpipe system or area served.

Water Fire Suppression Systems

The cable trays in the electrical tunnels and penetration areas are protected by preaction water spray systems which use automatic (i.e., closed head) directional water spray nozzles. Heat detectors installed in the cable trays operate the deluge valves associated with these systems. The heat detectors have a nominal setting of 165°F. The automatic directional water spray nozzles actuate at 175°F. The systems are provided with separate feeds from the yard fire water header such that failure or isolation of any section of yard piping would not incapacitate the systems.

The turbine governor and turbine generator bearings 1-9 are protected by a preaction water spray which uses automatic (i.e., closed head) directional water spray nozzles. The system deluge valve is operated by heat detectors which have a nominal setting of 190°F. The automatic directional water spray nozzles operate at 212°F.

A wet pipe sprinkler system has been provided to serve the following areas:
- a) Diesel Generator Building (diesel generator sumps and day tanks)
- b) Turbine Building at Elevation 15’0” and 36’9” (Turbine oil piping & general area)
- c) Auxiliary Feedwater Pump Room
- d) Outage Support Building
- e) Auxiliary Boiler Annex Building (Boiler room at Elevations 15’0” and 35’0”)
- f) Administration Services Building at Elevation 15’0”
- g) Receiving Warehouse (warehouse, paint vault and office area)

A preaction sprinkler system serves the Containment Access Facility Annex. Heat detectors are provided that actuate the system.

The following hazards / areas are protected by automatic water spray systems which use open spray nozzles and are designed to actuate by heat detectors:
- a) Unit auxiliary transformer
- b) Main transformers, No. 31 and No. 32
- c) Station auxiliary transformer
- d) Wall between the Turbine Building and main transformer No. 31 and the unit auxiliary transformer including the area between the main transformer No. 31 and the unit auxiliary transformer
- e) Wall between the Turbine Building and the pipe bridge to the Auxiliary Feedwater Pump Building.

The following hazards/areas are protected by a manually actuated water spray system which uses open spray nozzles:
- a) Main boiler feedwater pumps.
- b) Charcoal filters associated with the Containment Building fan cooler units.
- c) PAB door adjacent to the transformer yard

The Demineralized Water System (see Section 9.11) provides the supply of water for fire protection inside the Containment Building during normal plant operation. The system connects either one of two sources to nine fire hose racks located inside the Containment. The two sources are as follows:
1) The plant Makeup Demineralizer
2) The Fire Protection Header in the pipe tunnel area.

Portable CO_2 extinguishers are provided during maintenance periods.

The Reactor Containment has little combustible equipment. The greatest fire hazard is from the reactor coolant pump oil collection system. This system provides the capability to collect oil from potential leakage points including oil fill points, lower central units, upper bearing cooler, lift pump and piping, and provides drainage to a container.

The Indian Point 1 Fire Protection System can be utilized as a backup to the Indian Point 3 Fire Protection System, and is tied to the Indian Point 3 system but normally valved out. Control valves on water spray, sprinklers and foam systems are electrically supervised.

Gas Fire Suppression Systems

The CO_2 Fire Protection System is provided with two ten-ton capacity low pressure tanks, a distribution header and associated piping and valves. Storage is maintained at minimum pressure of 275 psi and $0^{\circ}F$. An electric vaporizer is located downstream of the storage tanks for generator purging; the vaporizer has no fire suppression system function (see Section 10.2).

An automatic total flooding carbon dioxide (CO_2) fire suppression system is provided to protect the following areas:

a) 480V switchgear room
b) Cable spreading room
c) Each of the three diesel generator rooms
d) Turbine generator exciter enclosure

The total flooding CO_2 fire suppression systems are actuated by heat detectors; they can also be manually actuated by a local manual station.

A local application CO_2 fire suppression system is provided to protect the following hazards in the Turbine Building:

a) Main boiler feedwater pumps 31 and 32
b) Turbine governor, main steam and re-heat valves and generator bearings, 1, 2 and 3
c) Turbine generator bearings 4, 5, 6 and 7
d) Turbine generator bearings 8 and 9

The CO_2 fire suppression systems protecting the main boiler feed pumps are actuated by heat detectors; they can also be manually actuated by a local manual station. The CO_2 fire suppression systems for the turbine governor and bearings are individually operated by manual stations located on the turbine deck.

Each CO_2 fire suppression system incorporates two types of discharge control timers, a pre-discharge timer and a discharge timer. The pre-discharge timer is adjusted to insure sufficient time for personal evacuation following a manual or automatic system actuation. The discharge timer regulates discharge duration to ensure sufficient agent delivery. The capacity of the CO_2 tanks is sufficient for a second discharge. The systems annunciate alarm and trouble, including a loss of power, in the control room on the fire display control panel (FDCP).

Foam Fire Suppression Systems

Automatic, fixed mechanical, heat detector actuated, foam-water deluge systems are provided in the Turbine Building protecting the turbine oil storage tank, reservoir and conditioner; the boiler feed pump lube oil reservoir, oil accumulators, and oil console; and the hydrogen seal oil reservoir. The foam systems also supply the interior hose stations located near each protected area.

Portable Fire Extinguishers

Portable fire extinguishers have been provided throughout the plant in accordance with the requirements of the NFPA standards. These extinguishers consist of dry chemical, CO_2, halon, foam, MET-L-X, and water types. The Control Room is provided with a “Class A” rated portable extinguisher.

Fire Protection System Leak Detection

Detection of leaks or breaks in the Fire Protection System is provided either by visual observation of the break or by information relayed to the Control Room.

Constant surveillance of the Turbine Building and Control Building is provided by personnel assigned to these areas. Locations outside these buildings are patrolled.

The flow of water in any part of the plant fire protection piping resulting from the use of water in any portion of the Fire Protection System or a pipe break causes a pressure drop in the piping system which automatically starts the fire pumps.

Startup of the main fire pumps is alarmed in the Control Room. Upon acknowledging the startup of these pumps, the responding operator checks for indication of water flow in any one of the water based fire suppression systems. If no apparent cause is identified, an investigation into the cause is initiated, which if necessary, will include an inspection for evidence of a pipe break.

If the break is so located that it cannot be seen or be readily isolated, sections of the Fire Protection System can be isolated under the direction of the Shift Manager. Isolation of the leak is indicated by a return to normal operation of the pressure maintenance and booster pumps or by visual indication that water flow has stopped.

The preaction water spray systems in the electrical tunnels and penetration areas are provided with a supervisory air system. The system piping is maintained under air pressure and monitored. Upon loss of air pressure due to system actuation, a leak or a pipe break, an alarm is sounded in the control room on the Fire Display Control Panel (FDCP).

Fire Drains and Associated Monitors

Fixed fire suppression systems have not been installed where their operation or failure could cause unacceptable damage to safety related equipment.

All areas provided with automatically operated fire protection have either gravity or pump drains. These drains were designed to handle the maximum quantity of spray water and/or oil (tank or pipe rupture) spills and will prevent local flooding.

In addition, all automatic fire protection systems are monitored to alarm in the Control Room when a system has been actuated by a fire or a possible false trip. Operators, on such a signal, are dispatched to the fire area and can, in the unlikely event of a partially plugged drain, control the water flow to the system by sectionalizing valves provided. Within the Diesel Generator Building, each control panel is protected from the effects of the sprinkler system spray. An accidental operation of the sprinklers will not cause an unsafe plant condition.

At Indian Point 3, with the exception of the CAF Annex, there is no sprinkler or deluge system installed in the following radioactive or potentially radioactive areas: Containment Building, Primary Auxiliary Building, Fuel Storage Building and Waste Holdup Tank Pit. Therefore, control and/or storage of this effluent is not a requirement. In these areas, floor drains are provided which are connected to either sump or sump tank. From there, effluent is pumped to a Waste Holdup Tank.

The Containment Access Facility Annex, which is used as a handling area for contaminated material, is protected by a preaction sprinkler system. Inadvertent sprinkler discharge is not likely. Additionally, a six-inch curb is provided on the annex floor to contain potentially radioactive system effluent.

9.6.2.4 Fire Detection System

The plant has a protective signaling system that transmits fire alarm and supervisory signals to the control room where audible and visual alarms are provided. The system includes signals for actuation of fire detectors, status of most installed fire suppression systems, control and indicating lights for the fire pumps, level indicators for the fire water storage tanks, and door status indicating lights for the operator notification of critical fire doors. Electrical supervisory signals are received from tamper switches on some of the fire water system control valves.

Portions of the Fire Detection System are supplied AC power from a lighting panel which is shed on loss of offsite power. However, the lighting panel is connected to the emergency power system and, on loss of normal AC power, is immediately restored by the operator to provide illumination in critical plant areas and thus also to restore power to the detection and signaling system.

The system provides electrical supervision of circuits for detectors in areas containing equipment and electrical cables needed for safe shutdown. Other detector circuits which are not electrically supervised are tested at various frequencies between 6 and 24 months, depending on the system protected.

Smoke detectors are provided at seven locations inside the Containment Building; at each of the four reactor coolant pumps, one at the electrical penetration areas, and two in other areas containing concentrations of electrical cable. The smoke detectors are directly connected to the fire display and control panel in the Control Room.

The Control Room is provided with smoke detectors. These detectors are installed above and below the Control Room false ceiling.

Smoke detectors have also been provided in the Electrical Tunnels, electrical penetration areas, Cable Spreading Room and 480 volt Switchgear Room. Heat detectors are provided in the Diesel Generator Building, and are used to actuate water spray systems on yard transformers, foam systems on Turbine Building oil hazards, and actuate the preaction water spray systems in the electrical tunnels and penetration areas. Heat detectors are provided in charcoal filters.

Smoke detection is provided in many fire zones within the PAB and throughout the Auxiliary Feedwater Pump Room.

There are numerous areas containing significant safety-related equipment and electrical cables that are provided with fire detection. These are detailed in Section 9.6.2.9. A complete listing of fire detection systems provided in each fire area is provided in the Fire Hazards Analysis (FHA). The FHA will be maintained by periodic updates.

A discussion of detection system redundancy within some areas of interest follows:

a) <u>Switchgear Room (Control Building, El. 15'-0" and Cable Spreading Room (Control Building, El. 33'-0")</u>.

Smoke detectors installed in these areas are wired in parallel so that the failure of one unit will not affect the integrity of the alarm system. These detection devices alarm in the Control Room.

Smoke detectors are capable of sensing the products of combustion when a fire is in the incipient stage. This time delay will permit an operator to investigate the area, make an evaluation and initiate actions to control and subsequently extinguish a fire.

Heat detectors are also installed in both rooms to activate the CO_2 fire suppression system. The heat detector actuating system consists of two detectors wired in series and the pair connected in parallel with other pairs wired in the same manner. Failure of one pair will not affect the integrity of the actuating system. The heat detector actuating system for the cable spreading room CO_2 fire protection system also initiates closure of the fire door located between the cable spreading room and the Electrical Tunnels.

b) <u>Electrical Tunnels and Electrical Penetration Areas</u>

Smoke detectors installed in these areas are wired in parallel so that the failure of one unit will not affect the integrity of the alarm system. These detection devices alarm in the Control Room.

Smoke detectors are capable of sensing the products of combustion when a fire is in the incipient stage. This time delay will permit an operator to investigate the area, make an evaluation and initiate actions to control and subsequently extinguish a fire.

Heat detectors installed in the cable trays in these areas are wired to operate the deluge valves associated with the water spray systems to flood the system piping.

Heat detectors are also installed in the entryway to the Electrical Tunnels to release the fire door located between the cable spreading room and the Electrical Tunnels. The automatic release system consists of two detectors wired in series and the pair connected in parallel with another pair wired in the same manner. Failure of one pair will not affect the integrity of the release system.

c) <u>Emergency Diesel Generator Rooms</u>

There are several heat detectors in each of the Diesel Generator Rooms for initiating an alarm in the Control Room some of which also actuate the automatic total flooding CO_2 system. The heat detector actuating systems consist of two detectors wired in series and the pair wired in parallel with other pairs wired in

the same manner. Failure of one pair will not affect the integrity of the actuating system. The CO_2 heat detector actuating system also initiates closure of the EDG Room Smoke Dampers and secures the ventilation fans.

Indication of the control room alarm will summon an operator to the area in trouble. The operator will investigate and determine the reason for the alarm.

d) <u>Primary Auxiliary Building</u>

Smoke detectors are installed outside the Residual Heat Removal Pump rooms on the 15' elevation and within each pump room, in the vicinity of the Component Cooling Water Pumps and Containment Spray Pumps on the 41' elevation, and in each Charging Pump cubical on the 55' elevation. In addition, smoke detectors are installed in the floor area underneath the motor control center cubicles on the 55' elevation of the PAB. The installed fire detection instruments provide thorough coverage of all floor elevations in the PAB on which safe shutdown equipment is located.

9.6.2.5 <u>Safe Shutdown Capability in Case of Fire</u>

Several options are available to plant operators whereby safe shutdown can be achieved following a fire. The evaluation of Indian Point 3 to the requirements of Section III.G of Appendix R to 10 CFR 50 identified the necessary systems and equipment which could be utilzed to bring the plant to a safe shutdown condition given a fire in any fire area. In accordance with the rule, the availability of equipment is ensured such that the following performance goals are met:

1) Reactivity Control - insert sufficient negative reactivity into the reactor core to maintain the core subcritical with the appropriate shutdown margin.
2) Reactor Coolant Makeup - maintain the primary system water inventory to prevent unacceptable fuel failure due to cladding heatup.
3) Reactor Coolant System Pressure Control - provide overpressure protection prior to a controlled cooldown and control of pressure for adequate subcooling margin.
4) Decay Heat Removal - remove decay heat at the appropriate rate for maintaining shutdown.
5) Processing Monitoring - provide sufficient information with regard to primary and secondary system parameters necessary to ensure maintenance of safe shutdown.
6) Support Services - provide the necessary support of systems required to achieve and maintain the above performance goals.

The performance goals are met utilizing the control rods; chemical and volume control system; closed cooling water system; auxiliary feedwater system; main steam system including steam generator, code safety valves and atmospheric steam dump valves; service water system; emergency power system; and certain instrumentation.

Where feasible, redundant safe shutdown trains are located in separate fire areas. In cases where the redundant safe shutdown trains are located in the same fire area, a combination of compliance with the requirements of Section III.G.2 and III.G.3 of Appendix R to 10 CFR 50 has been ensured or exemptions have been granted.

Alternate Shutdown Capability

There are two alternative shutdown schemes credited in compliance to Section III.G.3 of Appendix R that utilize an alternate diesel generator.

One alternate shutdown scheme makes use of local control stations in the Auxiliary Feedwater Pump Room, the Primary Auxiliary Building and the Turbine Building to affect shutdown following a fire that requires safe shutdown from outside the control room. The other alternate shutdown scheme makes use of the alternative diesel generator aligned to the 480V Vital Buses to ensure safe shutdown from the control room. The alternate diesel generator is a dedicated 2500 kw diesel generator and is located in its own enclosure in the yard area north of the Auxiliary Feedwater Pump Room.

AC power generated by the alternative diesel generator can be supplied to the 6.9 kv buses 5 and 6. These buses in turn feed 6.9 kv buses 1 and 3, which supply 480v to buses 312 and 313 through stepdown transformers. The 480 V bus 312 also feeds power distribution panel PDP-TG-1, which supplies 120 V ac power via a stepdown transformer to the instrument isolation cabinets thereby providing an alternative power supply to the safe shutdown instruments. The instrument isolation cabinets are located in the upper tunnel in the electrical penetration area.

The alternative 480 V ac switchgear 312A is powered by 480 V bus 312 directly or from bus 313 through the use of a tie breaker. The switchgear 312A feeds the following selected safe shutdown components:

- CCW pump 32
- Backup Service Water pump 38
- Charging pump 31 or 32

Component Cooling Water Pump 32 Charging Pumps 31 or 32 are powered through transfer switches, which are manually operated in the vicinity of each pump, respectively. The power supply to Backup Service Water Pump 38 is direct wired.

Supporting services for the Appendix R ac power source are independent from the supporting equipment used by the three emergency diesel generators (e.g., service water cooling, 125V dc control power, starting air HVAC, and fuel oil).

The alternative power system, as previously described, is designed to be independent and sufficiently isolated from the existing emergency power system to ensure the availability of power to the safe shutdown pumps and instruments of concern in the event of fire in the Control and Diesel Generator Buildings. In the case of a fire affecting certain portions of the Primary Auxiliary Building and Electrical Tunnels which could disable emergency diesel generator auxiliaries, the alternate diesel generator can be used to power the 480V Vital Buses to ensure safe shutdown from the control room.

In addition to the alternate diesel generator and power supply system, the capability exists to isolate the control circuits of emergency Diesel Generator No. 31, feeder breakers to 480v Buses Nos. 2A and 3A, and the tie breaker between Buses Nos. 2A and 3A. This design feature permits the operation of the normal emergency power supplies (Diesel Generator No. 31) in the event of a fire in the Cable Spreading Room or Control Room.

Switchboard type rotary selector switches in wall mounted cabinets at the Diesel Generator Control Panel for the control circuits and in floor mounted cabinets at the penetrations for the instrumentation are used. The alternate power supplies are in a floor mounted cabinet at the penetration area and they are supplied by a common 120 volt single phase source powered by Diesel Generator No. 31. When the isolation switches are in the “normal” mode, all aspects of diesel generator and 480 volt breaker control and instrument loop operation are not affected by these features. When transferring from the “normal” to “isolation” mode for these circuits, an alarm for both the diesel and the instrumentation will be initiated on the Control Room annunciator. If all the isolating switches are not operated, a local audible alarm is initiated, the “normal” indicating light is extinguished and incomplete sequence indicating light comes on. An acknowledge push button is provided to silence the audible alarm on the diesel generator isolation switch cabinet. When all the isolating switches are operated, the audible alarm will be reset and the incomplete sequence light is extinguished. The Diesel Generator and the designated breakers can be controlled only from the diesel generator panel with all remote wires passing through the cable spreading area being disconnected from the control circuit and all bypasses required for local operation mode. For the instrumentation (i.e., pressurizer pressure, pressurizer level, steam generator level, hot and cold temperature), when the isolating switches are operated, the alternate local power supply is inserted in the series loop and those sections of loop in Control and Cable Spreading Rooms are disconnected and bypassed with indication available only at the local stations in the Auxiliary Feed Pump Room and the Primary Auxiliary Building.

Instrumentation for both normal and alternate shutdown has been protected in accordance with the requirements of Section III.G of Appendix R. The plant process parameters, which are credited in the Safe Shutdown Scheme included:

a) Primary system wide range hot and cold leg temperature for loop 31.
b) RCS wide range pressure.
c) Pressurizer level.
d) Steam Generator pressure and wide range level.
e) Source range neutron flux.

The cooldown strategy is based upon a stepwise cooldown. Instrument uncertainty assumptions presume at least one fan cooler unit in service (by repair) within 8 hours of commencement of cooldown. During the cooldown, T_{cold} can be determined using local steam pressure indicators PI-2531 through PI-2534 and steam tables.

In order to minimize the effects of instrument uncertainly on pressure indication, the wide range RCS pressure indicator PT-402 will be compared against local pressure indicator PI-475 or PI-476 at the time of RHR cut-in. PI-475 and PI-476 have smaller uncertainties than PT-402. Comparing these instruments will establish a measurement bias that can be used with PT-402 throughout the remainder of the cooldown process. PI-475 and PI-476 are local pressure gauges located on the 46’ level on Containment and associated with RCS Loops 34 and 31, respectively.

Instrument isolation and transfer cabinets are located in the upper electrical penetration area. In the event of a fire in the Control Building, the safe shutdown instrumentation circuits can be isolated from the Control Building the Safe Shutdown instrumentation circuits can be isolated from the Control Building and indications transferred to the local control stations in the PAB and Auxiliary Feedwater Pump Room.

The local control station in the PAB is located outside the charging pump cubicles on the 55' elevation on Panel PL-6. Indications of pressurizer level, RCS pressure and source range neutron flux are available on PL-6. Operators at this location will control RCS boration and makeup with the charging pumps.

The local control station in the Auxiliary Pump Room, Panel PT2, is at elevation 18'6". Indications of steam generator water level and pressure, pressurized level, RCS pressure, and RCS Loop 31 hot and cold leg temperature are available at this panel.

Steam generator pressure indication is also available at the atmospheric steam dump valve local stations on the 43' elevation of the Turbine Building above the Auxiliary Feedwater Pump Room.

Local operation of the Auxiliary Feedwater Pumps would be accomplished in the Auxiliary Feedwater Pump Room for the turbine driven pump and at the emergency switchgear (in the Control Building), for the motor driven pump.

As originally designed permanent backup nitrogen supply consisting of nitrogen bottles and piping was provided for the steam generator atmospheric dump valves in the auxiliary feedwater pump room. This supply is no longer credited to support operation of the atmospheric dump valves and has been isolated by manual valve 1A-1775 to ensure an adequate supply of nitrogen is available to operate AFW pump control valves located in the auxiliary feedwater pump room. For a fire in the auxiliary feedwater pump room, nitrogen bottles and piping connections are provided independent of the AFWP room, located on the 43' elevation of the Auxiliary Feedwater Building near each local atmospheric dump valve control station.

9.6.2.6 Emergency Lighting

Self-contained emergency lighting units are installed throughout the plant. These units supplement the normal emergency lighting, which is powered from the emergency diesel generators. In the event the emergency diesel generators are inoperable or the diesel generator supplied emergency lights are damaged due to the fire, the self-contained units will provide the necessary illumination for operators to perform safe shutdown functions.

Each emergency lighting unit is compressed of a minimum eight-hour rated battery with a charger for operation from a 120V ac source. The emergency lighting units are supplied from "normal" lighting branch circuits. Pursuant to Section III.J of Appendix R to CFR 50, units are located in areas of the plant where safe shutdown operator action will be performed. Access and egress routes to these areas are also provided with emergency light coverage. Coverage of the yard area for access and egress to the alternate diesel generator, the main and backup Service Water pumps, CST or RWST, is provided by the Security Lighting System. The Security Lighting System is powered by a propane fueled generator which provides the necessary 8-hour power supply to the lights.

An exemption from the requirements of Section III.J of Appendix R has been granted (by Reference 1 and Reference 6) which allows credit for the security lighting system in lieu of eight-hour self-contained emergency light units for yard area coverage.

9.6.2.7 Reactor Coolant Pump (RCP) Oil Collection System

An oil collection system is provided for each of the reactor coolant pump motors. Oil leakage is collected and drained to tanks (one for each motor), which are located at approximately 48’6” in the containment. Each tank can accommodate the entire oil capacity of one pump motor. The tank vents are provided with flame arrestors to prevent flashback.

For RCPs 31 through 34, the oil collection piping from the collection tank is directly connected to the system piping above and is connected by means of flexible hose to the oil collection enclosures attached to the pump motor.

The seismic capability of the oil collection system has been evaluated as part of the requirements of Section III.O of Appendix R to 10 CFR 50. The results of the evaluation demonstrate that the system will withstand the Safe Shutdown earthquake.

9.6.2.8 Fire Brigade (Manual Fire Fighting)

A five-person fire brigade is available on-site to perform manual fire fighting activities. The brigade members are trained in various phases of fire fighting through academic and “hands on training.” The qualifications of each member of the brigade include satisfactory completion of training and a physical examination. The physical examination is intended to identify any condition that would prevent members from participating in strenuous activities such as fire fighting.

Yearly meetings of brigade members are held during which all classroom instruction material is reviewed. Additionally, practice sessions which provide actual experience in fire extinguishment and use of emergency breathing apparatus are provided yearly for each member. Drills are held at approximately quarterly intervals including one drill conducted on a back shift yearly. A maximum extension of 25% (three (3) months) is permitted for yearly training frequencies. A one-time change to the grace period was made for Fall 2001 yearly retraining of Security Department fire brigade members. The grace period was changed from 25% to 50% (six months). All brigade members participate in at least two drills per year. Once per year, the offsite fire fighting organizations are included in drills.

Protective clothing, portable smoke removal equipment, self-contained breathing equipment, portable hand lights, and radios are provided at strategic locations throughout the plant for use by the fire brigade. There are several self-contained breathing units on the site and a manifold cylinder emergency air supply for control room operators. Additional breathing appliance, spare cylinders and recharge capability are provided so that 10 men can be supplied for 6 hours on the basis of three air cylinders per man per hour. Breathing units are located near the Containment Air Lock, and the CO_2 flooded areas, such as the Cable Spreading Room, the Switchgear Room and the Diesel Generator Rooms and fire brigade lockers. In addition, there is an on-site compression and cascade system. Three portable smoke ejectors with a combined capacity of 1500-20000 cfm are available for fire brigade use. Portable ventilation equipment is available for those fire scenarios where HVAC systems are lost due to a fire. Portable heating units are also available for those fire scenarios where HVAC system are lost due to a fire. Portable heating units are also available for those fire scenarios where heat tracing is unavailable and piping freezing is possible.

9.6.2.9 Fire Protection of Specific Plant Areas and Equipment

A fire hazards analysis of the facility has been performed to determine the fire loading of various plant areas, to identify the consequences of fires in safety related and adjoining non-safety related areas, and to evaluate the adequacy of the Fire Protection System. The principal features for protection of specific areas are discussed in the following paragraphs.

Primary Auxiliary Building

Elevation 15, 34 and 41 Feet

Safe shutdown-related equipment at these elevations of the Primary Auxiliary Building include the two residual heat removal pumps, and three component cooling pumps, with associated valves and electrical cables. Electrical cables for two of the three charging pumps also pass through this area.

Smoke detectors are installed in the residual heat removal, containment spray, and component cooling water pump room, and fire hose stations are provided to reach all portions of these elevations. Fire suppression capability is also provided by portable fire extinguishers.

Elevation 55 feet

Safe Shutdown equipment on this elevation includes the three charging pumps, component cooling water heat exchangers, and associated piping, valves and electrical systems. One of the panels used for shutdown if the Control Room is not habitable is located in this area.

The Authority has installed smoke detectors in the motor control center area, waste drum storage room, charging pump room and the corridor outside the charging pump rooms. Fire hose stations have been provided to reach all portions of the area. Additional fire suppression capability is provided by portable extinguishers.

The Authority has installed additional communication equipment to enhance communications within the Primary Auxiliary Building, and improve overall communications between the CVCS charging pump control panel and all other areas needed for safe shutdown of the plant during a fire, should the control room become inaccessible.

Control Building

Cable Spreading Room

This area contains power, instrumentation and control cables for safety-related systems, some of which are required for shutdown. System involved include charging pumps service water pumps, component cooling pumps, auxiliary feedwater pumps, residual heat removal pumps, and atmospheric relief valves. Equipment located in this area consist of RPS motor generator sets, DC inverters and battery chargers.

Fire detection is provided by smoke detectors located at ceiling level. Fire suppression capability is provided by water hose stations and portable fire extinguishers. Water hose stations are in the Turbine Building, and at the east end of the Control Building at the Cable Spreading Room elevation. An automatic total flooding CO_2 system actuates on signals from installed heat detectors.

Battery Rooms

Three of the four redundant safety batteries which supply DC power to safe shutdown systems are each housed in their own individual enclosure within the Cable Spreading Room. Within the cable spreading room, battery rooms 31 and 32 are located adjacent to each other separated by a concrete masonry block wall, and battery room 34 is located on an opposite wall detached from the other two battery rooms. Each battery room is constructed of concrete masonry units. The fourth battery (33) is located in diesel generator room 31. A fifth battery (36), which is a non-safety battery, is housed in a concrete masonry block enclosure within the Turbine Building Hall Extension.

Flame detectors have been installed in the three safety battery rooms and diesel generator room 31, which houses the fourth battery. Smoke detectors have been provided in Battery Room 36. Procedures specifically require a periodic check of instrumentation to verify battery room ventilation flow.

Switchgear Room

This area contains the 480 volt switchgear along with the power and control cables for both redundant divisions of safety-related equipment, including the following safe shutdown equipment: charging pumps, component cooling pumps, auxiliary feedwater pumps, service water pumps, and residual heat removal pumps.

The significant combustibles in the area are lightly loaded cable trays stacked, at most, three high. The Switchgear Room also contains instrument air compressors which have a lube oil system with several gallons of lube oil each.

The switchgear are in separate metal enclosures and separated such that an unmitigated fire in one switchgear would not affect the redundant switchgear.

Protection for this area includes an automatic total flooding CO_2 system, smoke detectors at the ceiling, portable fire extinguishers and hose stations that are nearby.

Control Room

The Control Room contains cabinets and consoles within which are cables and components for safety-related systems including those systems required for safe shutdown of the plant. Manual water hose stations and portable CO_2 extinguishers provide the extinguishment capability. Fire protection equipment also includes a fire hose station at the east end of the Control Building at the Control Room elevation. With this hose station, adequate coverage is provided in the Control Room. In addition, a portable type “A” rated fire extinguisher is available in the Control Room. Smoke detectors installed above and below the Control Room false ceiling provide adequate area-wide detection capability. Smoke detectors are also installed in Control Room walk-in panels.

Electrical Tunnels

The plant contains two electrical tunnels separated by a one-foot thick concrete barrier. In general, redundant cables are in separate tunnels. The electrical tunnels contain power, control and instrumentation cables for safety equipment located in the Primary Auxiliary Building, Containment Fan House, and Auxiliary Feedwater Building Pump House. The upper tunnel

contains power cables for two of three safety injection pumps, two of three component cooling water pumps, one residual heat removal pump, two of three charging pumps, one motor-driven auxiliary feedwater pump, and control cables for atmospheric relief valves. The lower tunnel contains power cables for one component cooling pump, one residual removal heat pump, one of three charging pumps, and control cables for atmospheric relief valves.

The cable trays in the electrical tunnels and penetration areas are protected by pre-action water spray systems. The electrical tunnels and penetration areas are also provided with smoke detectors which alarm in the Control Room. These systems are supplemented with portable CO_2 fire extinguishers. Fire hose stations in the Fan House and Control Building equipped with fog nozzles approved for Class C (electrical) fires, serve the electrical tunnels and penetration areas. At the Control Building end of the electrical tunnels is a common area (i.e. Electrical Tunnels entryway) where cabling from both electrical tunnels is located in two stacks of trays horizontally separated by approximately six feet. Separate pre-action water supply systems are provided for each group of trays. One train of safe shutdown instrumentation is enclosed in fire wrap.

For a fire in the upper electrical tunnel, which contains circuits required to support operation of emergency diesel generator auxiliaries, the shutdown strategy is to utilize the alternate diesel generator aligned to the 480V vital buses located in the Control Building.

At the containment end of the cable tunnels in the upper electrical penetration area, all Safe Shutdown instrumentation passes out of the containment. One train of Safe Shutdown instrumentation is protected in a fire wrap barrier from the containment wall until it drops through the floor into the lower electrical tunnel. The metal wall surrounding the stairway between the lower and upper electrical penetration areas has been evaluated for its adequacy to protect the upper electrical tunnel/penetration area from a fire in the lower electrical penetration area. An exemption from specific requirements of Section III.G.2.a, of Appendix R applies to this wall.

Two openings have been provided in the hatchway above the upper electrical penetration area. These openings accommodate two smoke ejectors. The ventilation capability will facilitate manual fire fighting and Safe Shutdown.

Turbine Building

There is no safety related cable or equipment located within the building.

Automatic foam systems are provided for the turbine lube oil storage and reservoir, the boiler feed pump lube oil reservoir, oil console, oil accumulators, and the hydrogen seal oil unit. The foam systems have actuation alarms in the Control Room to provide fire notification to plant personnel. Foam hose stations are provided near these hazards. Manual fire hose stations and portable extinguishers are also provided throughout the Turbine Building.

Three-hour rated fire doors and dampers are provided in the barrier between the Turbine Building and Control Building. An alarm is installed on the fire doors between the Turbine Building and Control Building were upgrade to meet a three-hour fire rating. The door between the Control Room and Turbine Building is a three-hour equivalent fire rated door.

To provide overall fire protection in the Turbine Building, for areas containing lube oil lines, automatic sprinklers are provided below the operating floor on Elevation 15'0" and 36'9". Manually actuated water spray systems are provided for the main boiler feed water pumps and an automatic water spray is provided to protect the turbine governor and bearings.

Intake Structure

Six service water pumps are located within a seismically designed, steel-framed grating enclosure at the intake structure, and three back-up service water pumps are located on a platform over the discharge channel. At least one of these pumps may be required to achieve cold shutdown.

Fire hose stations are located at the North and South ends of the building at elevation 15'. Fire detection is provided by photoelectric type smoke detectors. Yard hydrants and hose are available for manual fire fighting in the area. Hose houses, equipment with fire hose and other fire fighting tools have been located in the yard area, so as to afford protection to the service water pumps.

Diesel Generator Building

The three emergency diesel generators are located in this building, along with associated day tanks and control panels. Each of the diesel generator units is located in a separate room. A set of ESF batteries is located in one Diesel Generator Room. At least one diesel generator could be utilized for safe shutdown if offsite power is interrupted. The alternate diesel generator is available for Safe Shutdown in the event the normal emergency diesel generators are disabled.

Heat detectors are provided in each Diesel Generator Room, and wet pipe automatic sprinklers are installed in the sump area beneath each diesel engine and over the fuel oil day tanks. A fire hose available in the Turbine Building can reach part of the Diesel Generator Building.

The following fire protection provisions are also provided for the Diesel Generator Building:

1) A total flooding CO_2 system which can discharge in any of the three diesel generator compartments, providing area coverage in each Diesel Generator Room.
2) A hose station in the adjacent Control Building can reach those areas in the Diesel Generator Building that are beyond the range of other hose stations.
3) Doors in the fire walls between the diesel generator cubicles are 3-hour rated fire doors having automatic closers and alarms so that the control room is alerted if the doors are left open. Additionally, the heat exchanger equipment openings between the cubicles are protected with 3-hour rated barriers.

Auxiliary Feedwater Pump Room

This area contains the two electric motor-driven auxiliary feedwater pumps, the steam turbine-driven auxiliary feedwater pump, associated valves and electrical cabling. The area also contains electrical cables for the atmospheric relief valves, and the local auxiliary feedwater control panel used for shutdown if the Control Room is not habitable. At least one of the three auxiliary feedwater pumps would be required for safe shutdown.

To provide prompt fire detection and suppression for the room, smoke detectors are provided inside the Auxiliary Feedwater Pump Room, and a hose station is provided outside the Auxiliary Feedwater Pump Room, in the Main Steam and Feedwater Piping Enclosure. Fire suppression capability in the room is provided by an area-wide, automatic wet pipe sprinkler system, portable fire extinguishers, and by fire hose from nearby yard hydrants.

Containment Building

Safety-related equipment inside the Containment Building which is required for safe shutdown includes the reactor vessel, pressurizer and steam generators; primary system piping; steam and feedwater piping; residual heat removal heat exchanger and associated valves; control rod drives; and instrumentation for pressurizer pressure and level, and steam generator level.

Charcoal filters in the Containment Building fan cooler units are protected by heat detectors and manual water spray systems that can be actuated from outside containment. Portable carbon dioxide fire extinguishers are also available for manual fire suppression.

The following provisions for fire protection are provided inside containment:

a) Fire detectors in areas containing concentrations of electrical cable and on the reactor coolant pumps.
b) Barriers to prevent a fire from causing loss of redundant instrumentation required for safe shutdown.
c) Hose stations for manual fire fighting to reach the reactor coolant pumps and areas containing a significant amount of electrical cable.
d) A Reactor Coolant Pump oil collection system which collects oil leakage.

Barriers provided inside containment to separate redundant safe shutdown instrumentation cabling have the following characteristics:

a) Thermal barriers are used to insulate the lower cable tray containing instrumentation cables of one channel where the redundant instrumentation cable trays are stacked above each other.

b) Thermal barriers, as above, are used to enclose one channel of safe shutdown instrumentation both where the cabling crosses from the stack of trays over to the penetration area, and at the penetration area.

c) Barriers and radiant energy shields have been installed for safe shutdown instrumentation to achieve conformance with Section III.G.2.f of Appendix R.

Fan House

The Fan House contains the piping penetration area. Safety-related valves are located in the piping penetration area, some of which may have to be correctly aligned to achieve safe shutdown. Containment penetration cooling fans and charcoal filters are located in the Fan House, but these are not required for safe shutdown.

The area of the Fan House housing the Primary Auxiliary Building Exhaust, Containment Purge & Containment Building Pressure Relief Systems is provided with smoke detectors. Charcoal filters in the filter units are protected by heat detectors and manual water spray systems. Fire fighters have portable extinguishers available for manual fire suppression.

Smoke detectors are also installed into the valve penetration area, and fire hose stations are provided to serve the entire Fan House.

Yard Area

The yard area contains the service water pumps, the buried fuel oil for the emergency diesel generators, the condensate storage tank, and the primary and refueling water storage tanks. The oil-filled transformers adjacent to the Control Building and Primary Auxiliary Building are protected by automatic water spray systems. The doors to the yard area from the Control Building and Primary Auxiliary Building have been upgraded to fire rating. A fire damper was installed in the ventilation opening from the electrical tunnels to the yard area, and hose houses equipped with fire hose and other fire fighting tools are provided for the yard hydrants.

The design of the door from the 15' elevation of the PAB to the yard could not be upgraded to obtain a fire rating. In lieu of a fire rated door, a water spray system is installed over the door to provide the necessary fire protection.

A valve/inspection pit housing service water valves and piping is located in the crushed stone area of the main transformer yard. The pit is of substantial construction and provides adequate fire resistance to preclude the loss of safe shutdown capability in the event of a transformer fire.

9.6.2.10 Ventilation Systems and Breathing Equipment

The following systems are provided with fire-stats (thermostats) or interlocks to shut down the fan systems in the event of pre-set high temperatures or fire:

1) The control room air conditioning system is equipped with a fire-stat installed to monitor room temperature.

2) The lube oil storage room ventilation system, is equipped with a fire-stat installed to monitor room temperature.

3) The electrical tunnel ventilation fans are interlocked with the pre-action water spray systems to shutdown when the fire protection deluge valve trips.

4) The Control Building 15' elevation exhaust fans are interlocked with the CO_2 fire suppression system to shutdown when the system actuates.

5) The Control Building 33' elevation exhaust fans and battery room exhaust fans are interlocked with the CO_2 fire suppression system to shutdown when the system actuates.

6) Diesel Generator Building exhaust fans and smoke dampers are interlocked with the CO_2 fire suppression systems to shutdown / close when the associated CO_2 system actuates.

Charcoal filters are enclosed in substantial metal housings, well separated from safe shutdown systems. Those filters in the containment cooling system are equipped with automatic fire detection and remotely operated manual water deluge systems. In addition, the filters are

separated from ignition sources and the amount of contained radioactive material is insufficient to cause ignition.

Each bank of carbon filters located in the containment fan cooler units is provided with a fire protection system. This system was designed to satisfy seismic requirements to preclude failure due to seismic events. The fire lines are located in the annulus of the Containment Building. This area is free of missiles. The Fire Protection System water supply is provided with two motor operated valves manually operated from the Control Room. These valves are installed in parallel for redundancy. A fire in the carbon filters may occur during an incident when access to the Containment Building is not possible due to high radiation levels. Because of the redundancies stated above, manual fire control in the area is not a requirement nor would it be considered necessary.

The Authority evaluated the effects of fires in radwaste areas as to the potential releases to the environment and found that the releases resulting from fires in these areas are acceptable low.

Fire dampers rated for 3-hour fire resistance in accordance with NFPA requirements are installed in ventilation openings between fire areas. Additional details are provided in Section 9.6.2.2.

Self-contained breathing apparatus (SCBAs) are located in areas of the plant protected by the CO_2 suppression system. Breathing appliances, spare cylinders and recharge capabilities are provided to supply 10 men for 6 hours on the basis of three air cylinders per man hour. Control room operators are provided with a manifold cylinder emergency air supply.

9.6.2.11 Combustible Material Control in Structures

All structures on Indian Point 3 were constructed of reinforced concrete, concrete block, structural steel and metal partitions, metal wall siding sandwich panels (consisting of 20GA galvanized steel backup liner panels, 1-1/2” fiberglass insulation and protected metal face sheets) and/or built-up roofing (over 1” hard board insulation on 15 lb. felt vapor barrier on metal decking). These are all noncombustible materials.

The metal wall siding has an Underwriter’s label indicating that all materials have a flame spread of 50 or less. The built-up roofing adhesive for attaching insulation is fire retardant BAR-FIRE which qualifies the built-up roofing system to be classified as Factory Mutual Class I as specified.

In addition to the above construction materials, the FOAMGLASR containment liner insulation is approved by NEPIA.

Electrical cables used in the plant were required to pass the ASTM-D-470-59T vertical flame test, as well as certain other tests developed by Consolidated Edison. The data indicated that the cables used will not burn vigorously under the test conditions used. Additional details of the electrical cable combustibility testing is provided in Section 8.2.2 of the FSAR. An exemption from the Appendix R requirement for 20 feet of separation and no intervening combustibles between redundant Safe Shutdown cables has been granted, based in part on the superior flame retardant capability of the cable installed in the Electrical Tunnel.

9.6.3 Compressed Air System

Instrument Air System

The Instrument Air System (Plant Drawing 9321-F-20363 [Formerly Figure 9.6-13A and 9.6-13B]) was designed such that the instrument air shall be available under all operating conditions, all essential systems requiring air during or after an accident shall be self-supporting, and after an accident, the air system shall be re-established.

To meet the design criteria the following design features have been incorporated. Duplicate compressors are installed with duplicate dryers and filters throughout. In addition, a backup supply is taken from the station air system. Those items essential for safe operation and safe cooldown are provided with air reserves or gas bottles. These supplies will enable the equipment to function in a safe manner until the air supply is reestablished. The controls are specified to fail to a safe position on loss of air or electrical power. The compressors and essential sections of the air supply system have been designed to operate after seismic shock. The non-essential header has a flow restrictor in it to limit flow in the event of a break to the capacity of one compressor.

The system is served by two 225 scfm Chicago Pneumatic non-lubricated compressors. The compressors, filters, and air dryers are located on the ground floor of the Control Building, a Class I seismic structure. Each compressor discharges into a common air receiver. The Instrument Air System is backed up from the Indian Point 3 Station Air System.

The instrument air compressors may be operated in two modes. One mode provides for the standby compressor to come on automatically in the event of low pressure in the common air receiver. In manual mode, the compressors will load and unload at predetermined pressure settings, but the motors run continuously.

To meet current and future instrument air loads, a third Non-Safety Related compressor/dryer package is available on the 15' elevation of the Turbine Building to supply the conventional plant. The third unit has a capacity of 350 scfm and is manufactured by Joy Manufacturing Co. The compressor is a two cylinder, two stage, reciprocating, water cooled unit constructed for continuous heavy duty service. The cylinders have teflon rings which do not require lubrication, therefore the air is delivered completely free of oil contamination. This compressor can also supply the Station Air System with backup air, if necessary.

The station air backup to the Instrument Air System is filtered through oil vapor and droplet removal equipment before entering the Instrument Air System. The instrument air receiver outlet enters one of two full capacity desiccant type dryers which reduce the dewpoint from saturation to approximately -40°F. This system is used for all indoor services where it is anticipated that the ambient temperature will not go below 50°F. Those services which are used for outdoor instrumentation and for lines which leave the Control Building and/or Turbine Building and enter the yard area to serve the Primary Auxiliary Building and Containment Building are also served through desiccant type dryers. These dryers reduce the dewpoint to minus 40°F, in order to compatible with the lowest expected outdoor temperatures.

The two 750 scfm capacity heatless air dryers are installed to ensure the ability of the existing Instrument Air system to consistently provide acceptable instrument air quality. Normally, one desiccant dryer is in operation at a time while the second dryer is in standby. These dryers are

installed in the 15' elevation of the Control Building. The dryers are equipped with duplex prefilters (and automatic drain valves) and afterfilters. The dryer monitors the processed air to maintain approximately -40°F dewpoint and provides an alarm if the dryer malfunctions. The afterfilters are rated for a .9 micron (absolute) particle removal. Each desiccant dryer is rated at 750 scfm and is a dual tower type dryer. An air afterfilter set is provided on the discharge of these dryers in order to filter out any desiccant which may be carried over by possible flotation of the bed. A moisture detector is provided on the common discharge header to notify the Control Room operator in the event of high dewpoint.

A power failure to the dryer which is in service causes the dryer to go to its fail safe position (fail opened). This allows for uninterrupted instrument air flow.

In order to provide continuity of service to Class I areas in the event of an outage of the conventional plant instrument air header, a restriction orifice is provided so as to limit the flow to the capacity of one instrument air compressor into a possible line break in the secondary plant air header. Upon notification of this break, a valve is operated to isolate the secondary plant and prevent pressure decay in the primary plant header.

Valve position lights in the Control Room advise the operator as to the status of all emergency bypass or makeup control valves. A manual local reset solenoid valve is provided at each emergency valve so as to require the attention of an operator at the equipment. All air and oil filters are dual type to provide maintenance during operation.

The components or systems essential to plant safety and serviced by the Instrument Air System are as follows:

1) Containment Isolation Valves
2) Cooling Water Valves for Containment Building Fan Coolers
3) Condensate Storage Tank Shut-off Valves
4) Auxiliary Boiler Feed Pump Control Valves
5) Steam Dump Valves to Atmosphere
6) Low Pressure Steam Dump Valves to Main Condenser
7) Containment Building Penetration and Weld Channel Pressurization System
8) Emergency Diesel Cooling Water Valves
9) Deleted
10) Boron Injection Tank Recirculation Valves
11) Control Room Air Conditioning Actuators

In the event of low pressure in the instrument Air System, air is automatically supplied to the Instrument Air system from the Station Air System. In the event of low pressure at any or all of the components listed as items 4,7, or 11 above, dry nitrogen cylinders automatically supply gas pressure to those components required for safe shutdown or continued operation of the plant. Additionally, a manual supply of dry nitrogen is available to operate the steam dump valves to atmosphere in the event of low instrument air system pressure.

In the event of an instrument air line rupture in the conventional plant, a restriction orifice limits the flow to 225 scfm. One compressor supplies air to compensate for the break while the spare compressor supplies the primary plant.

Station Air System

The Station Air System (Plant Drawing 9321-F-20353 [Formerly Figure 9.6-15]) is supplied by a two-stage compressor located in the Turbine Building. The air is discharged through an aftercooler and moisture separator at 100 psig and 110°F. The maximum discharge pressure will be 125 psig. The cooling water for the intercooler, after cooler and compressor jacket is supplied from a closed cooling water system which contains treated city water. Station air can also be supplied by the third instrument air compressor located in the Turbine Building.

The Station Air Compressor is controlled by the solenoid unloader valves which are energized through a pressure switch arrangement in automatic or hand (manual) modes. In the automatic mode, the compressor will run in single or two-stage operation and unload at a predetermined pressure setting with motor and compressor stopped. In manual mode, the compressor will start and stop at predetermined pressure settings; but the motor continues to run. High water and high air temperature switches are connected to the control annunciator.

The Station Air System furnishes compressed air for pneumatic tools, circulating water pump priming, and miscellaneous cleaning and maintenance purposes throughout the secondary primary plants. A 900 cfm diesel driven air compressor and a desiccant air dryer were added to the Station Air System that will automatically start on low station air pressure. The compressor is located outside the Turbine Building and it is permanently piped to the station air receiver discharge piping. A check valve isolates the diesel compressor from the station air system when it is not running. The compressor can be isolated or disconnected when maintenance is required. The diesel driven air compressor can provide air to the Station Air and the Instrument Air systems upon loss of offsite power. All work is designated as Non-Safety Related, which is consistent with the Station Air system except for the control room alarm.

This system is backed up by the Indian Point 1 Station Air System through manually operated valve interconnection to the Indian Point 3 air receiver. The size of the connection is equal to the Indian Point 3 supply pipe. This system may also be supplemented by temporary portable air compressors connected to an outside flanged connection near the northwest corner of the Turbine Building and piped to the inlet of the station air receiver.

The system also provides for an automatic emergency supply to the Indian Point 3 Instrument Air System through an oil vapor filtering arrangement. In addition, an automatic emergency supply is supplied to the Containment Building Weld Channel and Penetration Pressurization System. The air is first filtered and then dried to –40 °F dewpoint.

Component Design and Operation Parameters

1) Station Air Compressor

2-stage, vertical angle, duplex compressor.

Modes of Operation:

a) Manual - Loads and unloads at predetermined pressure and motor continues to run.

b) Automatic - Motor stops on unloading.
   - Design Pressure - 135 psig
   - Operating Pressure - 100-125 psig
   - Design Capacity - 625 scfm @ 100 psig

2) Station Air Aftercooler

Single pass, tube and shell horizontal heat exchanger to cool air to within 15°F of inlet water temperature.

3) Station Air Receiver

10' long by 42" diameter receiver.
Design Pressure - 135 psig
Operating Pressure - 125 psig (max)

4) Instrument Air Compressors Nos. 31 and 32

Two (2) horizontal, single stage, double acting compressors, non-lube construction.

Modes of Operation:

a) Automatic - Motor starts when receiver pressure drops to 95 psig. Approximately ten seconds later unloader valve is energized and supplies air to systems. Motor stops at 105 psig.

b) Manual - Compressor runs continuously and is loaded and unloaded as receiver pressure varies between 100 and 110 psig.
Design Pressure - 135 psig
Operating Pressure - 100 - 110 psig
Design Capacity - 225 scfm @ 100 psig

5) Instrument Air Aftercoolers

Two (2) tube and shell horizontal heat exchangers to cool air to within 15°F of inlet water temperature, complete with cyclone moisture separators.

6) Instrument Air Receiver

10' long by 42" diameter receiver.

Design Pressure - 135 psig
Operating Pressure - 125 psig (max)

7) Desiccant Dryers

a) Two redundant 750 scfm heatless desiccant dryers Nos. 31 and 32 on the Instrument Air System are used to obtain a final dewpoint of approximately -40°F at 100 psig.

Design Pressure - 150 psig
Operating Pressure - 100-110 psig
Design Capacity - 750 scfm @ 100 psig
Duplex Coalescer Prefilter - 1200 scfm @ 100 psig
Duplex Afterfilter - 1200 scfm @ 100 psig, .9 micron absolute particle rating.

b) A regenerative air dryer in the Station Air System is used to obtain a final dewpoint of approximately -40°F @ 100 psig. A nonregenerative dryer is installed in parallel with each regenerative dryer for standby operation. The nonregenerative dryer is suitable for intermittent operation of 4 to 6 hours duration.

Design Pressure - 150 psig
Operating Pressure - 100-110 psig
Design Capacity - 45 scfm @ 100 psig

8) <u>Filters</u>

Two (nominal 225 scfm) prefilters and two (nominal 150 scfm) after-filters are provided for the Station Air System.

Two (nominal 45 scfm) prefilters and two (nominal 45 scfm) after-filters are provided for the Station Air System.

Emergency air makeup from the Station Air System to Instrument Air System passes through two (nominal 225 scfm) liquid oil prefilters and two (nominal 225 scfm) oil vapor prefilters.

Design Pressure - 150 psig
Operating Pressure - 110 psig (max)

9) <u>Instrument Air Compressor No. 33</u>

Two stage, two cylinder reciprocating compressor, non lube construction for continous heavy duty service.

Design capacity - 350 scfm at 110 psig
Rated discharge pressure - 110 psig

9.6.4 Heating System

An eight inch main steam header connects the two 50,000 lb steam per hour boilers at Indian Point 2 to the Indian Point 3 steam distribution system. The Indian Point 3 steam distribution system consists of three separate circuits. The first circuit starts upstream of the pressure reducing valve and serves the Circulating Water Priming Ejectors and the retired De-icing Steam Jet Vacuum Pumps. The second circuit starts from a connection located downstream from a pressure reducing station and upstream from a desuperheater section. This circuit serves the east and west side of the Turbine Hall, the Heater Bay and the Service Building. The third circuit starts from a connection located downstream from both the pressure reducing and desuperheater stations. It serves the clean and dirty turbine oil storage tanks, the Fan Room, the Fuel Storage Building, the Primary Auxiliary Building, the tank pit, the Primary Water Storage Tank and the Refueling Water Storage Tank.

Provision is made for the following heating services:

1) Primary Auxiliary Building
   a) Electric strip heaters
   b) Steam unit heaters
   c) Air makeup steam tempering units
   d) Batching mixing tank

2) Purge System Containment Building
   a) Air makeup steam tempering unit

3) Fuel Storage Building
   a) Steam unit heaters for standby heating
   b) Air makeup steam tempering units

4) Waste Tank Storage Pit
   a) Air makeup steam tempering unit

5) Fan Room
   a) Steam unit heater

6) Turbine Hall
   a) Steam unit heaters
   b) Clean and dirty oil tank
   c) Circulating water primary ejectors
   d) De-icing steam jet vacuum pumps

7) Refueling water storage tank

8) Primary water storage tank

9) Service Building
   a) Process and heating

10) Diesel Generator Building

11) Administration Building

12) Auxiliary Building

13) Radioactive Machine Shop Building

14) Liquid Radwaste Storage Building

The temperature in the Containment Building is maintained without the use of steam unit heaters. Therefore, the containment unit heaters were removed.

There is no steam heating in the Diesel Generator Building. This building is provided with thermostatically controlled electric unit heaters for standby heating service designed to maintain a minimum room temperature of 60°F when the outside temperature is approximately -5°F. Without building heat, the building temperature will be approximately 35°F.

This building except for the roof and supply and exhaust dampered openings, is built below grade. With an inside temperature of 35°F the building is subject to a heat gain from the ground and adjacent heated areas and a loss through the roof and ventilation openings. At this room temperature (35°F) and an outside temperature of -5°F, the heat gains to the building are slightly higher than the heat losses.

Each diesel engine is provided with an independently controlled 12kW electric lube oil heater, a 9 kW electric jack water heater, and a prelube circulating oil pump that circulates warm oil when the diesel unit is shutdown. Each heater and pump is controlled by an independent, local contactor and circuit breaker.

Service water supply to the engine heat exchangers is through parallel flow control valves, which are closed during normal plant operation, to prevent heat losses when the diesel unit is shutdown. The valves are provided with local control switches, and position indicating lights both local and in the control room, and are interlocked to open automatically when any unit is started (either manually or automatically).

The jacket water and lube oil heaters are sufficient to assure the engine starting capabilities at the ambient expected with loss of building heaters.

Plant Drawings 9321-F-27273, and -40573 [Formerly Figures 9.6-16 and 9.6-17] are flow diagrams and heating plans for Indian Point 3.

9.6.5 Plant Communications System

The Indian Point 3 communications system was designed to ensure the reliable, timely flow of information and action directives necessary during normal operation, and particularly for the mitigation of emergencies. Reliability of the system is provided by extensive redundancy and by alternative communications equipment; routine use of many of the systems lowers the probability of undetected system failures.

Dedicated communication links have been established for use during emergencies to preclude delays due to system swamping. A redundant power supply is provided for the communications system in the Control Room.

Public Address System

The Public Address (PA) System has two subsystems: the Plant Party Paging and the Site PA System. The system consists of three channels. Two of these channels are common to both the primary (nuclear) and secondary (conventional) portions of the plant. The third line provides an additional channel in the primary portion of the Indian Point 3 plant. Speakers for monitoring each of these lines are located in the Control Room. The Public Address System receives its power from MCC 36B through a single 3PH, 480/120V, 9KVA Sola transformer.

The Primary Auxiliary Building, Waste Holdup Tank Pit, Liquid Radwaste Storage Facility, Fuel Storage Buildings, Fan Room area, Containment Building, Auxiliary Boiler Feed Pump Building, Diesel Generator Building and tunnels are treated as the primary (nuclear) portion of the plant. The secondary (conventional) port of the plant comprises the rest of the facility.

Three handsets are located on the plant operator's desk in the Control Room. A "Page" handset is used for page purposes only and calls originating from this handset can be heard on all loudspeakers in the primary and secondary portions of the facility. The remaining two "Page-Party" handsets are used for loudspeakers paging and party-line conversations, as selected by the control room operator. A switch is provided on the control room desk, which will allow all outdoor speakers to be turned off at night.

All calls initiated from the handsets in the primary and secondary plant are heard over the party-line speaker in the Control Room. A "page" call from plant area handset will only go to the Control Room. It is possible to carry on two independent party-line conversations simultaneously between the Control Room and the primary (nuclear) areas.

A handset station is located on the Indian Point 3 plant flight panel. This station may be used on either of the three channels by selecting the desired circuit with a three-position switch mounted beside the handset.

Within the primary (nuclear) area, the handset stations are equipped with a selector switch, which allows usage of either of the two party-line channels for conversing with the Control Room operator and vice versa. At the secondary (conventional) area stations, removal of any handset from its carriage will activate a speaker through which the operator can summon the Control Room by voice communication, if so desired.

The Control Room desk is equipped with three decibel meters. One meter monitors the Page circuit; the other two monitor the Party-Line circuit. The meters will swing into the -6db area, when the circuits are in use.

A monitor console is provided for the Shift Manager. Each console is equipped with "Page" and "Party" decibel meters, which indicate when channels are in use. A speaker and speaker amplifier is also supplied. A four-position selector switch is used to select which channel shall be monitored orally, if desired. An "L" pad volume control is included so that speaker volume can be set.

<u>Sound Powered Communication System</u>

The Sound Powered Communication System (SPCS) consists of communication stations located throughout the plant and interconnected by cable run in suitable raceways. Stations are located throughout the plant.

Three independent communication channels are provided with each communication station having access to all three channels. A station consists of three telephone jacks, one connected to each channel. The sound powered phone is equipped with a cord and a plug. In use, the phone is plugged into one of the three jacks and can then communicate with one or more other phones plugged into the same channel anywhere in the plant. Energy to operate the system is generated by the user's voice; there is no external power source involved.

Communication stations at supervisory and control panel locations consist of three telephone jacks flush mounted on the front panel. Stations located in the field consist of a weatherproof junction box with three or more telephone jacks mounted on the cover.

Regular Telephone System

Office locations at Indian Point 3 have switchboard extensions and various locations have direct incoming lines without utilizing the switchboard. Along with normal telephone features, this system provides features such as automatic callback, call forwarding, three-way consultation, transfers and pickup of another individual phone from a remote location.

Incorporated into the regular telephone system is the Control Room Touch-O-Matic direct dial capability. Programmed into this system are organizations and parties for the Emergency Response Network, such as key emergency plant personnel, hospital, ambulance, police and fire numbers.

Con Edison Extensions

Placed in various locations throughout the facility are Consolidated Edison Extension telephones to aid in the communications between the Indian Point 3 and Indian Point 2 sites. The Control Room is equipped with an extension.

Emergency Communications Equipment

The emergency facilities for Indian Point 3 equipped with emergency communications equipment are: The Control Room (CR), the Emergency Operations Facility (EOF), the alternate EOF (AEOF), the Technical Support Center (TSC) and the Operations Support Center (OSC).

Direct Telephone Lines (5-party and 4-party lines)

A dedicated line from the Control Room, TSC, OSC and EOF has been installed at Indian Point 3 known as the 4-party line. In the event of relocating the EOF to the AEOF, another dedicated line from the CR, TSC, OSC, EOF and AEOF has been installed at Indian Point 3 known as the five-party line. This allows the expeditious transfer of plant operation and design information during an emergency.

Direct Telephone Lines (White Plains Office to Indian Point 3)

A dedicated line connecting Indian Point 3 to the Entergy Nuclear Northeast Corporate Office is available for information transfer during an emergency condition. The telephones are located (onsite) in the TSC and (in White Plains) in the Alternate Emergency Operations Facility (AEOF).

Direct Line with Bell Annunciator

A direct line telephone links the Indian Point 2 Control Room, the Indian Point 3 Control Room, and the Emergency Operation Facility. One another can reach each location by the use of a manual ring down circuit.

Alarms

Audible alarms are a quick and effective means of communicating emergency warnings on the Indian Point 3 site. Alarms currently installed include the:

1) Containment Evacuation
2) Site Assembly
3) Fire
4) Air Raid

Each alarm provides a distinctive sound that station personnel and contractors have been trained to recognize.

Radio Communications

Radio Communication equipment used in normal plant operations will be used in an emergency to communicate with mobile units and to provide backup to the telephone system if necessary. Radio capabilities include the following:

1) IP2 Plan Radio Frequency
2) IP3 Security Radio Frequency
3) Entergy Radio System*

*NOTE: Installed by Temporary Modifications TM 92-03820-00

The Indian Point Energy Center Emergency Plan Radio Frequency permits radio communication between the EOF, AEOF, Indian Point 2 and 3 Control Rooms, Offsite Monitoring Vehicles, TSC, OSC and Security.

The Security Radio Frequency provides, during an emergency, for monitoring site security and to have direct communication with the N.Y. State Police. During normal operation, the Security force uses portable walkie-talkies for station-to-station communication and site security. There is a backup gas-driven generator at the EOF which will automatically supply AC power for the radio system if normal power is interrupted.

The Radio System consists of repeaters and portable radios, which provide communication between the OSC, dispatched in-plant teams and the Indian Point 3 Control Room.*

Paging System ("Beeper")

"Beeper" paging is used by Entergy to call in key Emergency Plan Personnel in the event of an emergency at Indian Point Energy Center.

NRC Emergency Telecommunications System (ETS)

The NRC Emergency Telecommunications System is a dedicated telephone system that connects Indian Point 3 with the NRC Operations Center (Headquarters) in Bethesda, Maryland. It is to be used for reporting off-normal incidents affecting the facility, and for providing information concerning the operation and status of the plant. Commercial telephone lines should be used as backup communications. The purpose of these lines is to provide reliable communications with the NRC. The ETS consists of the following lines:

Line	Function
1.	Emergency Notification System (ENS)
2.	Health Physics Network (HPN)
3.	Reactor Safety Counterpart Link (RSCL)
4.	Protective Measures Counterpart Link (PMCL)
5.	Management Counterpart Link (MCL)
6.	Local Area Network Access (LAN)

These lines are located in the following areas:

Location	Line(s)
Control Room (CR)	1
Technical Support Center (TSC)	1
NRC Office	1,2,3,4,5,6
Emergency Operations Facility (EOF)	1,2,3,4,5,6
Operations Support Center (OSC)	2

*NOTE: Installed by Temp Mod 92-03820-00

Health Physics Network (HPN) Line

This line is part of a network that includes all nuclear power plants, the NRC Regional Office and the NRC Operations Headquarters in Bethesda, Maryland. In the event of an emergency at the site, either the NRC Regional Office or Headquarters may decide to establish a direct telephone link to the licensee's dose assessment team. At such time, the HPN line will be the primary means of communicating health physics and dose assessment information from the licensee to the NRC. The HPN is a restricted network and should not be used by non-government employees at any time unless needed to report a significant event when both the line and the commercial telephone lines are out of service. HPN lines are located in the NRC Office, OSC, and EOF. These lines are all tied into the same loop and there can be used as party lines.

RECS Line Telephone Network

The Radiological Emergency Communication System (RECS) is a dedicated line which connects the Control Rooms of Indian Point 2 and Indian Point 3 with the EOF, AEOF, the County Emergency Operation Centers and warning points within the 10 mile Emergency Planning Radius, the City of Peekskill, and the New York State Emergency Center in Albany [Deleted].

The RECS Line is a multipoint conferencing circuit with one drop at each of the above mentioned locations and is available 24 hours a day, 7 days a week.

References:

1. Safety Evaluation dated January 7, 1987 from S.A. Varga, Director - Project Directorate #3, Division of PWR Licensing - A, USNRC, to J.C. Brons, New York Power Authority.

2. Fire Protection Reference Manual, Volumes 1 to 4.

3. Fire Protection Plan for Indian Point 3 Nuclear Power Plant.

4. Operational Specification Manual.

5. Exemption from the Requirements of 10 CFR Part 50, Appendix R, Section III.G.2 - Indian Point Nuclear Generating Unit No. 3 (Tac No. M88323), transmitted by NRC to NYPA letter dated January 5, 1995.

6. Safety Evaluation dated March 29, 1995 from L.B. March, Director - Project Directorate I-1, Division of Reactor Projects - I/II, Office of Nuclear Reactor Regulation to W.J. Cahill, Chief Nuclear Officer, New York Power Authority.

7. Safety Evaluation dated June 12, 1989 from J. B. Neighbors, Senior Project Manager – Project Directorate I-1, Division of Reactor Projects I/II, USNRC, to J. C. Brons, New York Power Authority.

8. EC 71892, Evaluation of Service Water System Passive Failures During Long Term Recirculation

TABLE 9.6-1

(Sheet 1 of 2)

SERVICE WATER FLOW REQUIREMENTS AT 95°F*
ESSENTIAL HEADER

ESSENTIAL HEADER	FLOW EACH (GPM)	NORMAL (GPM)	INJECTION (GPM)	RECIRCULATION WITH ACTIVE FAILURES (GPM)	RECIRCULATION WITH OR WITHOUT PASSIVE FAILURES (GPM)
Containment F.C.U.	570/1400(5)	2850(4)	7000(4)	7000(3)	7000(3)
F.C.U. Motor Cooler	12	60(4)	60(4)	60(3)	60(3)
Diesel Generator	302/ 213 (9)	(6)	906(3A)	639(3A)	639(3A)
Instrument Air Compressor Cooling	65	65(1)	65(1)	65(1)	65(1)
CR Air Conditioner Unit (2 Condensers/Unit)	52.5	105(2)	105(2)	105(2)	105(2)
Turbine Oil Cooler	2350	2350(1)	2350(1)	(6)	(6)
Seal Oil Cooler	100	100(1)	100(1)	(6)	(6)
BFP Oil Cooler	120	120(1)	120(1)	(6)	(6)
[Deleted]					
Strainer Backwash	100(1)(7)	100(1)(7)	100(1)(7)	100(1)(7)	100(1)(7)
Total Flow Required	-	5,750	10,806	7,969	7,969

(1) 1 Unit
(2) 2 Units
(3) 3 Units, but all 5 not isolated
(3A) 3 Units
(4) 5 Units
(5) 570 gpm at 120°F Cont. Temp/ 1400 gpm Accident
(6) Not Required
(10) 4000 gpm (1 CCW Hx out of service)
(7) One at a time
(8) [Deleted]
(9) 302 gpm injection/ 213 gpm recirculation
(11) 5000 gpm (1 CCW Hx out of service)

(12) The Appendix R cooldown can be performed on the non-essential header or the essential header, depending on which service water pumps are available

* The service water flows shown in this table for the Containment FCU, FCU Motor Cooler, Diesel Generator, Instrument Air Compressor, CCWHX 31, CCWHX 32 and CCRAC Condenser represent the flow requirements for a 95°F river water temperature. All other flow requirements are based on 85°F river water temperature.

TABLE 9.6-1

(Sheet 2 of 2)

SERVICE WATER FLOW REQUIREMENTS AT 95°F*
NON-ESSENTIAL HEADER

NON-ESSENTIAL HEADER	FLOW EACH (GPM)	NORMAL (GPM)	INJECTION (GPM)	RECIRCULATION WITH ACTIVE FAILURES (GPM)	RECIRCULATION WITH OR W/O PASSIVE FAILURES (GPM)	APPENDIX R COOLDOWN (12) (GPM)
[Deleted]						
Component Cooling Heat Exchanger 31	Variable	3460	(6)	2750(10)	4000(11)	2500
Component Cooling Heat Exchanger 32	Variable	3469	(6)	2750(10)	4000(11)	2500
Screen Wash and Circ. Water Pump Seals and Bearings	800	800(1)	(6)	(6)	(6)	(6)
Turb. Build. Closed Cooling Water	700	700(1)	(6)	(6)	(6)	(6)
ISO-Phase, Exciter and Hydrogen Coolers	2940	2940(1)	(6)	(6)	(6)	(6)
[Deleted]						
SGBDHX-4	250	250(1)	(6)	(6)	(6)	(6)
Strainer Backwash	100(1)(7)	100(1)(7)	(6)	100(1)(7)	100(1)(7)	100
Total Flow Required	-	11,719	(6)	5,600	8,100	5,100

See sheet 1 of 2 for Notes
* The service water flows shown in this table for the Containment FCU, FCU Motor Cooler, Diesel Generator, Instrument Air Compressor, CCWHX 31, CCWHX 32 and CCRAC Condenser represent the flow requirements for a 95°F river water temperature. All other flow requirements are based on 85°F river water temperature.

TABLE 9.6-2

SERVICE WATER FLOW (gpm)
UNDER MOST LIMITING FAILURE CONDITIONS

	Normal + Seismic (Case 8)	Post-LOCA Injection + LOIA (Case 3)	Post-LOCA Recirculation +Active Failure (Case 21)
Containment FCU's & Motor Coolers:			
31	697	1529	1696
32	700	1532	1699
33	690	1510	1675
34	695	1528	1695
35	684	1500	1665
Diesel Generator Coolers:			
31	429	451	479
32	441	463	492
33	388	408	433
Component Cooling Water Heat Exchangers:			
31	*	*	3287
32	*	*	3182
CR Air Conditioning Condensers:	118	109	118
Non-Essential Pumps:			
31	*	*	6596
32	*	*	*
33	*	*	*
Essential Pumps:			
34	7092	6071	5200
35	*	*	*
36	7061	6045	5172

*NOTE: Not used for this condition
LOIA - Loss of Instrument Air

TABLE 9.6-2A
(Sheet 1 of 4)

SERVICE WATER FLOW DISTRIBUTION
ESSENTIAL HEADER
POST LOCA RECIRCULATION MODE OF OPERATION WITH PASSIVE FAILURES

ESSENTIAL HEADER	REQUIRED FLOW (GPM)	24" ESSENTIAL HEADER CRACK LWL (GPM) (Case 23)	[Deleted]	20" NON-ESSENTIAL HEADER CRACK LWL (GPM) [Deleted] (Case 25)
Containment FCU	7000	9162	[Deleted]	9290
FCU Motor Cooler	60	169	[Deleted]	171
Diesel Generator 31	213	525	[Deleted]	531
Diesel Generator 32	213	540	[Deleted]	546
Diesel Generator 33	213	475	[Deleted]	480
Instrument Air Compressor Cooling	65	135*	[Deleted]	135
CR AC Condenser	52.5	128**	[Deleted]	129**
[Deleted]				
Strainer Backwash	300	471	[Deleted]	474
Break Flow	-	729	-	-
Pump Flow Total	-	12,336	[Deleted]	11,758
Average Pump Flow	-	4,112	[Deleted]	3,919

*Two Instrument Air Compressor Coolers in Service

** Two CR AC Condensers in service

TABLE 9.6-2A
(Sheet 2 of 4)

SERVICE WATER FLOW DISTRIBUTION
NON-ESSENTIAL HEADER
POST LOCA RECIRCULATION MODE OF OPERATION WITH PASSIVE FAILURES

NON-ESSENTIAL HEADER	REQUIRED FLOW (GPM)	24" ESSENTIAL HEADER CRACK LWL (GPM) (Case 23)	[Deleted]	20" NON-ESSENTIAL HEADER CRACK LWL (GPM) [Deleted] (Case 25)
Component Cooling Heat Exchanger 31	4000	4891	[Deleted]	4746
Component Cooling Heat Exchanger 32	4000	4734	[Deleted]	4595
[Deleted]				
[Deleted]				
Strainer Backwash	200	277*	[Deleted]	275*
Break Flow	-	-	[Deleted]	509
Pump Flow Total	-	10,027	[Deleted]	10,249
Average Pump Flow	-	5.013	[Deleted]	5,124

* Flow is for two strainers but third strainer is not isolated

TABLE 9.6-2A
(Sheet 3 of 4)

SERVICE WATER FLOW DISTRIBUTION
ESSENTIAL HEADER
POST LOCA RECIRCULATION MODE OF OPERATION WITH PASSIVE FAILURES

ESSENTIAL HEADER	REQUIRED FLOW (GPM)	18" ESSENTIAL HEADER CRACK LWL (GPM) (Case 26)	10" ESSENTIAL HEADER CRACK LWL (GPM) (Case 27)	[Deleted]
Containment FCU	7000	9150	9204	[Deleted]
FCU Motor Cooler	60	169	169	[Deleted]
Diesel Generator 31	213	525	520	[Deleted]
Diesel Generator 32	213	540	535	[Deleted]
Diesel Generator 33	213	475	472	[Deleted]
Instrument Air Compressor Cooling	65	135*	135*	[Deleted]
CR AC Condenser	52.5	127**	127**	[Deleted]
[Deleted]				
Strainer Backwash	300	472	472	[Deleted]
Break Flow	-	495	331	-
Pump Flow Total	-	12,090	11,970	[Deleted]
Average Pump Flow	-	4,030	3,990	[Deleted]

* Two Instrument Air Compressor Coolers in service

** Two CR AC Condensers in service

TABLE 9.6-2A
(Sheet 4 of 4)

SERVICE WATER FLOW DISTRIBUTION
NON-ESSENTIAL HEADER
POST LOCA RECIRCULATION MODE OF OPERATION WITH PASSIVE FAILURES

NON-ESSENTIAL HEADER	REQUIRED FLOW (GPM)	18" ESSENTIAL HEADER CRACK LWL (GPM) (Case 26)	10" ESSENTIAL HEADER CRACK LWL (GPM) (Case 27)	[Deleted]
Component Cooling Heat Exchanger 31	4000	4891	4891	[Deleted]
Component Cooling Heat Exchanger 32	4000	4734	4734	[Deleted]
[Deleted]				
[Deleted]				
Strainer Backwash	200	277*	277*	[Deleted]
Break Flow	-	-	-	[Deleted]
Pump Flow Total	-	10,027	10,027	[Deleted]
Average Pump Flow	-	5,013	5,013	[Deleted]

* Flow is for two strainers but third strainer is not isolated

TABLE 9.6-3

VALVE POSITIONS

Valve No.	Normal/ Injection	Recirculation
FCV-1111	closed	closed
FCV-1112	open	closed
SWN-6	open	closed
SWN-7	closed	closed
SWN-31	open	open
SWN-32	closed	closed
SWN-33 (2)	open	open
SWN-34 (2)	open	open
TCV-1103	modulating	modulating
TCV-1104	closed/ open	open
TCV-1105	closed/ open	open
FCV-1176	closed / open	open
FCV-1176A	closed / open	open
SWN-4	open	closed
SWN-5	closed	closed
SWN-27 (2)	closed	closed
SWN-29	open	open
SWN-30	closed	closed
SWN-38	open	open
SWN-39	closed	closed
SWN-55	closed*	closed*
SWN-70 (2)	open	open
SWN-94-1	closed	closed
SWN-94-2	open	open
SWN-95	open	open
SWN-108-3	open	open
SWN-108-6	open	open

* Valve disc is modified with fixed orifice to ensure minimum flow requirements to diesel generators when SI signal is received.

9.7 EQUIPMENT AND SYSTEM DECONTAIMINATION

9.7.1 Design Basis

Activity outside the core could result from fission products from defective fuel elements, fission products from tramp uranium left on the cladding in small quantities during fabrication, products of n - Y or n - p reactions on the water or impurities in the water, and activated corrosion products. Fission products in the reactor coolant associated with normal plant operation and tramp uranium are generally removed with the coolant or in subsequent flushing of the system to be decontaminated. The products of water activation are not long lived and may be removed by natural decay during reactor cooldown and subsequent flushing procedures. Activated corrosion products are the primary source of the remaining activity.

The corrosion products contain radioisotopes from the reactor coolant that have been absorbed on or have diffused into the oxide film. The oxide film, essentially magnetite (Fe_3O_4) with oxides of other metals including Cr and Ni, can be removed by chemical means presently used in industry.

Water from the primary coolant system and the spent fuel pit is the primary potential source of contamination outside of the corrosion film of the primary coolant system. The contamination could be spread by various means when access is required. Contact while working on primary system components could result in contamination of the equipment, tools and clothing of the personnel involved in the maintenance. Also, leakage from the system during operation or spillage during maintenance could contaminate the immediate areas and could contribute to the contamination of the equipment, tools and clothing.

9.7.2 Methods of Decontamination

Surface contaminants that are found on equipment in the primary system and the spent fuel pit that are in contact with the water are removed by conventional techniques of flushing and scrubbing as required. Tools are decontaminated by flushing and scrubbing since the contaminates are generally on the surface only of non-porous materials. Personnel and their clothing are decontaminated according to the standard health physics procedures.

Those areas of the plant that are susceptible to spillage of radioactive fluids are painted with a sealant to facilitate decontamination that may be required. Generally, washing and flushing of the surfaces are sufficient to remove any radioactivity present.

The corrosion films generally are tightly adhering surface contaminates, and must be removed by chemical processes. The removal of these films is generally done with the aid of commercial vendors who provide both services and formulations. Since decontamination experience with reactors is continually being gained, specific procedures may change for each decontamination case.

Portable components may be cleaned with a combination of chemical reverse electroplating and ultrasonic method if required.

9.7.3 Decontamination Facilities

Decontamination facilities on site consist of an equipment pit and a cask pit located adjacent to the spent fuel storage pit. In the stainless steel lined equipment pit, fuel handling tools and other tools can be cleaned and decontaminated.

In the cask decontamination pit, the outside surfaces of the shipping casks are decontaminated, if required, by using steam, water detergent solutions, and manual scrubbing to the extent required. When outside of the casks are decontaminated, the casks are removed by the auxiliary building crane and hauled away.

For the personnel, a decontamination shower and washroom is located adjacent to the Radiation Control Area (RCA) locker room. Personnel decontamination kits with instructions for their use are in or adjacent to the personnel decontamination shower area.

9.8 PRIMARY AUXILIARY BUILDING VENTILATION SYSTEM

9.8.1 Design Basis

The Primary Auxiliary Building Ventilation System is designed to accomplish the following:

1) Provide sufficient circulation of filtered air through the various rooms and compartments of the Primary Auxiliary Building and Containment Access Facility (CAF) to remove equipment heat and maintain safe ambient operating temperatures.

2) Control flow direction of airborne radioactivity from low activity areas toward higher activity areas.

3) Provide purging of the Primary Auxiliary Building, CAF, and CAF Annex through roughing, HEPA, and charcoal filters to the plant vent for dispersion to the environment.

4) Provide separate exhaust ventilation for the hydrogen crib in the CAF Annex through the Annex roof.

The air exhausted by the system is filtered and monitored so that off-site dose during normal operation will not exceed 10 CFR 20 limits. These limits are specified and controlled in accordance with site ODCM.

9.8.2 System Design

The Primary Auxiliary Building is heated with thermostatically controlled steam unit heaters and electric strip heaters. Each unit has its own thermostat and the failure of one unit will not affect the operation of the others.

The Primary Auxiliary Building Ventilation System is composed of the following systems:

1) Make-up air handling system complete with fan, bypass dampers, filters, heating coils and supply ductwork.

2) Exhaust system complete with fans, ductwork, bypass dampers, roughing, HEPA, and charcoal filters.

3) Make-up air tempering unit for the waste storage tank pit.

Design parameters for the system components are given in Table 9.8-1. Plant Drawing 9321-F-40223 [Formerly Figure 6.4-2] shows the flow diagram for this system. The fire protection provisions for these systems are covered in Section 9.6.

PAB Exhaust Fan 32 is normally fed from 480V switchgear 32, Bus 6A. However, an alternate power source is provided so that the fan can be fed from 480V MCC-312A when offsite power is available during a peak accident loading condition following a design basis accident.

Branch supply ducts direct make-up air to the various floors at the east end of the Primary Auxiliary Building. This supplied make-up air is drawn into the rooms and compartments of the Primary Building and the CAF by an exhaust system, which also draws outside air through the CAF Annex.

The air is exhausted from each of the compartments through ductwork designed to make the supply air sweep across the room as it travels to the room exhaust register. The air then flows to the exhaust fan inlet plenum, and is drawn by the operating exhaust fan through roughing, HEPA and charcoal filters before discharge to the plant vent. Flow from the PAB ventilation system normally bypasses the charcoal filters, these filters are placed in service by a high radiation signal. The exhaust system has been designed to insure that air flows from the “clean” end of regions of low radioactivity level of the building through the “hot” or regions of higher radioactivity levels.

The Primary Auxiliary Building Ventilation System is divided into two separate parts: The hold-up tank pit (waste hold-up tanks and three CVCS hold-up tanks); and the remaining portions of the Primary Auxiliary Building (CAF, chemical drain tank, spent resin storage tank, gas decay tank) and engineered safety feature equipment (safety injection pumps, residual heat removal pumps, component cooling pumps). The air exhausted from each of the two separate parts of the Primary Auxiliary Building Ventilation System normally flows directly into the roughing, and HEPA filters in the exhaust fan inlet plenum. The charcoal filters are placed in service by a high radiation signal. Make-up air to this area is tempered with a steam heating coil. The roughing, and HEPA filters are installed in such a way that all gaseous flows from the PAB will pass through them. The analysis of releases from this system is contained in Chapter 11.

There are two 70,000 cfm exhaust fans (No. 31 and 32), which are common to both containment building purge system (see Section 5.3) and the Primary Auxiliary Building Ventilation System, and serve as back-up to each other. Each system has its own supply fan that operates only in its individual ventilation system. One exhaust fan is required for each supply fan operating.

The selection of the desired pair or pairs of fans is manual, using a selector switch located on the fan room control panel. All four fans can be started and stopped by the single control switch located on the fan room control panel. Each fan has indicating lights on the fan room control panel and in the main control room. An autotrip alarm is also provided. In addition, each of the fans have a “jog” push button located on the fan room control panel for testing.

The Radiation Monitoring System (RMS) room on Elevation 55’-0” of the PAB contains the RM-80 microprocessors. These microprocessors are rated at 86°F ambient for continuous operation. This room is cooled by two air conditioning units located within the RMS room with air cooled condensers located outdoors in the 138kV switchyard.

9.8.3 Component Design and Operation Data

The HEPA filters used in this system are designed to remove submicron particles 0.3 microns and larger with an efficiency of not less than 99.97%. The roughing filters are provided ahead of the HEPA filters to screen out large size particles and thereby prolong the life of the HEPA filter. Filters are designed and fabricated to conform to the following:

HEPA Filters
1) Nominal size - 24” x 24” x 11” deep
2) Filter Frame - 304L or 409 stainless steel all welded construction
3) Filter media - a continuous strip of fire resistant waterproof glass fibers, folded back and forth over corrugated separators
4) Separators - corrugated aluminum
5) Flow - approximately 1000 cfm per filter
6) Tests - factory and inplace tested
7) Leakage - sealed with pressure sealing tape to prevent air bypass
8) Pressure - designed for 6 inch H_2O pressure differential across filter
9) Seismic - designed to satisfy seismic design criteria

Roughing Filters
1) Nominal Size - 24” x 24” x 2” thick
2) Frame - 304L stainless steel all welded construction
3) Filter Media - fire resistant, water-proof glass fibers reinforced with stainless steel wire cloth
4) Flow - approximately 1000 cfm per filter
5) Tests - factory and inplace tested
6) Leakage - sealed with pressure sealing tape to prevent air bypass
7) Pressure - designed for 6 inch H_2O pressure differential across filter
8) Seismic - designed to satisfy seismic design criteria

Technical Specifications (TS) require charcoal and HEPA filter testing to demonstrate operability any time a fire, chemical release or work done on the filters could alter integrity. TS surveillance testing is based upon a maximum flow of 30,800 cfm (28,000 plus 10%) giving a minimum safety factor of 2 for methyl iodide removal efficiency while allowing 1% bypass. NSE 98-3-017 HVAC demonstrates, for the purpose of TS implementation, that welding is not a fire, a chemical release or work that could alter filter integrity. The NSE also demonstrates that organic components from painting and similar activities could not alter filter integrity until the organic components are above 10% by weight and concludes that filter testing shall be performed when the organic components are greater than or equal to 2.5% by weight organics. Administrative controls are required to evaluate the percent by weight of organics when activities that could generate organics are conducted.

TABLE 9.8-1

PRIMARY AUXILIARY BUILDING VENTILATION SYSTEM COMPONENTS DATA

System	Units Installed	Unit Capacity	Units Required for Normal Operation
Exhaust*			
Fan, Standard Conditions	2	70,000 cfm	1
[Deleted]			
[Deleted]			
Plenums (Fan)	2	70,000 cfm	1
Roughing Filters	1 Bank	70,000 cfm	1
HEPA Filters	1 Bank	70,000 cfm	1
Charcoal Filters	1 Bank	70,000 cfm	1
Supply Tempering Unit (PAB)			
Fans, Standard Conditions	1	64,900 cfm	1
[Deleted]			
[Deleted]			
Filters	1	64,900 cfm	1
Coils	1	64,900 cfm	1
Tempering Unit (Waste Storage Tank Pit)			
Coil complete with motor operated dampers	1	5,100 cfm	1

* These two exhaust fans are used interchangably and/or as backup for:
1) Ventilation of Primary Auxiliary Building
2) Containment Building Purge System

9.9 CONTROL ROOM AIR CONDITIONING, HEATING AND VENTILATION SYSTEM

9.9.1 Design Basis

The Control Room Air Conditioning, Heating and Ventilation System is designed to accomplish the following (operating conditions may be different, as discussed in Sections 9.9.2 and 9.9.3):

1) Maintain 75 F D.B. and approximately 50% R.H. in the Control Room under normal operating conditions.

2) Permit cleanup of airborne particulate radioactivity that enters the Control Room through either the make-up (outside air) line or by infiltration, with $\geq$1,500 CFM make-up outside air (O.A.) for pressurization of control room circulated through a charcoal filter.

3) Sustain seismic events

Five independent air conditioning (A/C) units also exist in the Control Room to supplement the cooling capacity of the existing A/C system during normal plant operation. These A/C units can also be used during a blackout, Appendix R Safe Shutdown, or design basis accident to maintain the required Control Room temperature. However, these units are assumed operable only during normal operation. Each supplemental unit has been installed as an augmented quality related unit and has a 3-ton capacity using a split system arrangement.

9.9.2 System Design

The Control Room Air Conditioning, Heating and Ventilation System (shown in Plant Drawing 9321-F-40593 [Formerly Figure 9.9-1]) consists of the following equipment:

1) Two direct expansion, water-cooled air conditioning units complete with fans and roughing filters. Each unit sized to provide 60% of the refrigeration capacity required.

2) A filter unit consisting of casing, roughing filters, HEPA filters and charcoal filters.

3) Two charcoal filters booster fans each with a capacity of 2,000 cfm for 100% redundancy.

4) Duct system complete with dampers, controls and associated accessories to provide for three (3) different systems of air flow.

5) One kitchen and locker room exhaust fan and miscellaneous electric sill-line heaters.

6) The augmented quality related, locally mounted, supplemental cooling units are capable of supplying heat to the Control Room.

All doors in the Control Room lead to enclosed areas (turbine building, control building stairwell, locker room and pantry), not to the outside.

The exhaust fan in the kitchen and locker room ventilation system is designed to stop when the control room air conditioning system is placed in the incident mode of operation. A gravity type damper in the ductwork will prevent back flow.

The Control Room is isolated during emergencies as described in Section 9.9.3. In addition, all openings around cables are sealed airtight with a fireproof compound, and the control room door has a three-hour equivalent fire rating. As the control room is constructed of concrete, infiltration or exfiltration through walls, floor and ceiling is negligible.

A 35 cfm air make-up was calculated as the flow required to provide a slight positive pressure in the control room. It was based on a wind velocity of approximately 2.25 miles per hour and 88 linear feet of 1/16 inch wide cracks around the doors, etc., as stated in Section 14.3.5. The use of this low wind velocity is justified for this application because control room doors do not open to the outside. Normally, infiltration calculations are made on the basis that doors open to the outside and with wind velocities of 15 miles per hour. If this installation were assumed to be operating under these conditions, the make-up air requirements would become approximately 545 cfm. Revised dose calculations have assumed a $\geq$1,500 cfm of make-up air and the CCR HVAC System was modified to provide this quantity of filtered make-up air in system MODE 3.

A Control Room leakage test has shown that infiltration leakage will not exceed 700 cfm. It was determined that the dampers would better function if adjusted to provide a flow of $\geq$1,500 cfm. This ensures that a slight positive pressure is maintained in the Control Room but would not exceed the radiation (TEDE) dose limits during an accident due to increased make-up air flow.

The air conditioning system was balanced during initial construction, with the aid of dampers, so that all flows met the design requirements within $\pm$5%. During an incident, the gravity damper in the locker room exhaust system closes, and the exhaust fan in the locker room exhaust system stops. The remaining make-up air will exhaust through cracks and the relief damper, which is weighted to permit selection of control room pressure.

The fresh air intake duct supplying make-up air to the Control Room is provided with air operated dampers to direct this air through carbon filters or close off this supply to the control room completely.

All equipment, except the kitchen and locker room exhaust fan, is located in the Control Building. The air conditioning units, booster fans and carbon filter unit are located on the first floor at elevation 15’-0”. The locker room exhaust fan is located in the Control Building Fan Room on elevation 27’-0”.

The air conditioning units are normally supplied with cooling water from the essential service water header. All fans, except the locker room exhaust fan, will be powered from one of the buses serviced by the emergency diesel generators and will start automatically following a blackout. The locker room exhaust fan does not run under incident conditions. Operator action is not required to prevent unacceptable temperatures to safety-related equipment located in the Control Room.

The air conditioning system was designed so that functional capacity of the Control Room is maintained at all times, including the period during a blackout or DBA. Control Room safety

equipment is specified to a temperature of 120°F. This accounts for the temperature rise due to the enclosed cabinets. The design condition for maintaining “functional capacity” of the Control Room dictates that the ambient temperature for safety equipment located in this room shall not exceed 108.2°F for short term operation associated with a loss of one air conditioning unit. Exceptions are evaluated in NSE 95-3-032, Revision 1.

System Dampers A, B, D1, D2, F1, and F2 are air operated, receiving motive power from the Instrument Air System. The instrument air supply is not redundant. To permit continued operation in the event of loss of instrument air, a backup nitrogen cylinder will supply motive power for a minimum of 24 hours.

The fresh air intake for the Control Room air conditioning system is located in the east wall of the control building below the electrical tunnel between elevations 30’-0” and 18’-0”. This intake is protected from tornado generated missiles by an enclosure formed by the electrical tunnel floor above and concrete walls on the south and east sides. Make-up air enters this enclosure horizontally from the north.

With the system in the 10% incident mode, intake air from the atmosphere to the Control Room passes through charcoal filters in the control room ventilation system. The recirculation duct was blocked by the installation of a spool piece with blank off plate, therefore, the Control Room air is no longer being recirculated through the charcoal filters during the 10% incident MODE.

The charcoal filter unit for use in the control room air conditioning system, during “incident” conditions, consists of the following components designed to remain functional under earthquake conditions:

1) Air tight metal casing reinforced with structural steel shapes. The casing is provided with flanged inlet and outlet openings at each end as well as two air tight access panels. The access panels are mounted on the side of the casing, one upstream from the filter location and one downstream.

2) The filter package is made up of two roughing filters, two HEPA filters and two carbon filters. All filters are provided with gaskets or pressure sealing to prevent air from by-passing filters.

    a) The roughing filter frames are of welded construction and fabricated from stainless steel. The filter media consists of a mat of fire resistant waterproof glass fibers reinforced with stainless steel wire cloth.

    b) The HEPA filter frames are of welded construction and fabricated from stainless steel. The core of the filter is constructed by folding a continuous strip of fire resistant waterproof glass fiber filter media back and forth over corrugated aluminum or impregnated asbestos separators.

    c) Carbon filter frames are made from stainless steel. Filter cells are of the pleated type with a minimum one inch thick carbon bed.

The installed Nuclear Grade Activated Charcoal is tested in accordance with ASTM D3803-1989, per the IP3 response to Generic Letter 99-02. Technical Specification (TS) surveillance testing of the 2 inch beds is based upon a maximum flow defined by dose calculations giving a minimum safety factor of 1.81 for methyl iodide removal efficiency while allowing a 1% bypass.

On the loss of one air conditioning unit, the control room temperature will rise to approximately 106°F, with all lights, except emergency lights turned off. These temperatures are lower than the maximum tolerable upper limit of 108.2°F stated above, and show that the "functional capacity" of the Control Room will be maintained under all conditions, including the loss of one air conditioning unit.

The fire protection provisions for this system are covered in Section 9.6.

There is a remote probability that three (3) toxic chemical vapors (anhydrous ammonia, carbon dioxide, chlorine) could reach the Control Room air intake following a release of these gases to the atmosphere as a result of an accident, in sufficient concentration as to cause a potential hazard to the Control Room personnel. The control room does not see toxic limits of hazardous chemicals since regulatory guidance allows "a control room operator will take protective measures within 2 minutes (adequate time to don a respirator and protective clothing) after…detection." Air breathing apparatus is provided for Control Room personnel which consists of two (2) 220 cubic foot breathing cylinders each with a manifold for four (4) persons (two (2) cylinders for a capability of eight (8) men). The sir supply in each 220 ft^3 cylinder will last about one (1) hour for the four (4) men under normal breathing conditions. Nominally 25 backup cylinders (220 ft^3 each) are available outside the Control Room. In addition, four (4) self-contained breathing apparatus, rated at 30 minutes each are provided in the Control Room.

There are two systems that will alert the Control Room operator of the presence of these toxic chemical vapors. The first system consists of a set monitors that are located in the Control Room HVAC air intake. Separate chlorine, ammonia and oxygen probes are provided to detect the presence of these gases in the outside air intake. The oxygen monitor is used to indirectly monitor changes in carbon dioxide levels. An air sample is extracted from the outside air intake duct, analyzed in an airtight enclosure and exhausted back to the air intake. The detector and monitor are mounted in the air conditioning equipment room in the Control Building at El. 15'0". An alarm on panel SM in the control room is provided that will alarm on detection of these toxic gases, equipment trouble or loss of power. In addition, continuous digital LED readout is provided in the air conditioning equipment room for each detector.

The second system and its monitors are located in the Control Room and monitor Control Room atmosphere for low oxygen levels and for high levels of chlorine and ammonia. The system contains a central control and alarm panel that provides continuous indication of oxygen, chlorine and ammonia levels. Detectors are located near the fire display and control panel in order to monitor the atmosphere in the operator's area. Also, an additional remote oxygen detector is installed in the Central Control Room locker room.

The supplemental Control Room air conditioning system shown in Plant Drawing 9321-F-40593 [Formerly Figure 9.9.1] consists of the following equipment (1 of 5 units shown for drawing simplification):

1) Five independent direct expansion air conditioning (A/C) units each having a nominal 3-ton capacity, consisting of a wall-mounted evaporator section located in the Control Room, and an air-cooled condenser located on the Control Building roof.

2) Each evaporator section contains a cooling coil, as well as an electric heating coil, capable of providing cooling and/or heating upon demand.

3) Each evaporator drain system consists of a condensate collection sump, level switch, and drain line, with a gravity check valve.

4) Five closed loop refrigeration circuits, each consisting of copper tubing using Refrigerant 22.

9.9.3 System Operation

The Control Room Air Conditioning, Heating and Ventilation System (See Plant Drawing 9321-F-40593 [Formerly Figure 9.9-1]) will operate as follows:

1) Normal Conditions
The two air conditioning units operate in parallel to supply as required to maintain the Control Room set temperature. Approximately 1500 cfm of the circulated air is fresh outside air makeup. All circulated air bypasses the carbon filters. An exhaust fan for ventilating the locker room is interlocked to operate in conjunction with the air conditioning system.

2 Incident Conditions
a) $\geq$1,500 CFM outside air circulated through the carbon filter.
b) With 100% recirculated air.

On a Safety Injection signal, the system will automatically be placed in the incident mode of operation (2) (a) above, as follows:

One of the filter booster fans will start, damper “B” will operate to provide $\geq$1,500 cfm outdoor air to the carbon filters; the locker room exhaust fan will stop. In the event that the first booster fan fails to start, the second booster fan will start after a predetermined time delay. A firestat actuation during this incident MODE will trip the running filter booster fan and prevent the stand-by filter booster fan from starting. The system can, and should, be operated in this mode for as many hours or days required, as determined by the operator.

If for any reason it is required or desired to operate with 100% recirculated air, the system can be placed in the 100% Recirculation Mode of operation (2) (b) by local or remote manually operated switches.

The control circuit is designed to permit local control of the entire air conditioning system from elevation 15’0” in the Control Building and remote control from the Control Room proper. The control station at elevation 15’-0” will include a selector switch that will permit operation of the controls from elevation 15’-0” only when the selector switch is in the local position.

In the event of fire, portable air cylinders and a manifold cylinder emergency air supply have been provided for Control Room operators (see Section 9.6.2).

Any or all five of the supplemental Control Room air conditioning units may operate to provide cooling or heating, as required.

9.9.4 Tests and Surveillance

The following test procedure was used on start-up to verify that the control room air conditioning system would operate in accordance with the design criteria and objectives:

1) Verify the operation of the air conditioning units:
    a) Verify cooling water to units

    b) Verify the installation of filters

    c) Verify operation of controls on:
        1) Cooling Cycle
        2) For various modes of operation:
            - Normal-Outside air (15%) makeup
            - Incident-Outside air (15%) makeup, filtered through carbon filters.
            - Incident-100% Recirculated air (0% through carbon filters)

    d) Verify operation of remote controls including start-stop push button motor control with selector switch

    e) Verify operation of all dampers

    f) Verify operation of kitchen and locker room exhaust fan

2) Verify operation of booster fan and filter package unit
    a) Verify the installation of roughing, HEPA and charcoal filters
    b) Verify the operation of back-up booster fan

3) Verify operation of firestat

The surveillance tests that are performed to assure proper operation and availability of emergency control room air filtration (carbon filters) system are presented in the Technical Specifications.

The surveillance test will assure the filtration system is adjusted to allow a minimum flow of greater than 1,500 cfm make-up air (damper B) for the duration of the accident (30 days) when accounting for filter degradation during that time (and an initial one (1) accumulated day of monthly functional testing time) and assuming an initial differential pressure drop across the filtration unit of 2.0” WC.

Technical Specifications (TS) require charcoal and HEPA filter testing to demonstrate operability any time a fire, chemical release or work done on the filters could alter integrity. NSE 98-3-017 HVAC demonstrates, for the purpose of TS implementation, that welding is not a fire, a chemical release or work that could alter filter integrity. The NSE also demonstrates that organic components from painting and similar activities could not alter filter integrity until

the organic components are above 10% by weight and concludes that filter testing shall be performed when the organic components are greater than or equal to 2.5% by weight organics. Administrative controls are required to evaluate the percent by weight of organics when activities that could generate organics are conducted.

9.10 BREATHABLE AIR SYSTEM INSIDE CONTAINMENT

This system has been abandoned, the components inside containment have been retired in place, the components outside containment have been removed. Refer to EC 5000041594/DCP-03-3-014.

9.10.1 Design Basis

The Breathable Air System (BAS) inside Containment is designed to provide a source of clean breathable air for maintenance personnel at selected locations inside the Containment Building. The Breathable Air System is a Non-Safety Related system except for the penetration into containment.

9.10.2 System Design

The system consists of a compressor package, refrigerant dryer, a breathing air purifier, moisture traps, air hose manifolds, and associated valves, piping and instrumentation. The air purifier assures that the air quality conforms to the standards required by OSHA. The air from the compressor is cooled by an aftercooler. High air temperature is prevented by interlocking high air temperature and low lube oil pressure signals to the Breathing Air Compressor Controls. Signals for deviation from normal operating parameters are relayed to an annunciator panel.

9.10.3 System Operation

Breathable air is provided inside containment through a spare penetration line (see Table 5.2-3), which consists of a six-inch sleeve capped by a blind flange at each end.

Prior to using the Breathable Air System, the two blind flanges must be temporarily removed and replaced with two temporary flanges that have adapters to mate with the BAS piping. After connection of the BAS piping, the system is ready to operate.

When the BAS is no longer required, the temporary flanges are replaced with the permanent blind flanges and leak tested in accordance with 10 CFR Part 50, Appendix J requirements, to satisfy the containment integrity criteria provided in the Technical Specifications.

9.10.4 Special Testing Requirements Prior to Operation

Testing was performed prior to operation in accordance with ANSI B.31.1. Functional tests were performed on alarms, systems and components.

9.11 DEMINERALIZED WATER SYSTEM INSIDE CONTAINMENT BUILDING

9.11.1 Design Basis

The Demineralized Water System Inside Containment is designed to provide a source of water for decontamination and for hydrostatic testing and flushing. The system also provides standby fire protection inside containment.

9.11.2 System Design

The piping for the system is arranged to supply water to the distribution system in containment from either of two (2) sources. Only one source will be used at any one time. The two sources are as follows:

1) From the Plant Make-up Demineralizer (Class III Line).
2) From the Fire Protection Header located in the pipe tunnel area (Class III Line).

The lines from these sources run into a 3” header, which reduces to a 2” header at the Containment Isolation Valves. These isolation valves are wired to receive a Phase “A” containment isolation signal (for details on these valves refer to Section 5.2). The line then becomes Class III at the first weld joint inside the Containment Building.

All pipe supports design and materials, both inside and outside of the containment building, are seismic Class I.

The distribution headers and lines within containment supply water to electric water heaters, to a number of hot or cold water hose connection locations, and to nine (9) fire hose racks. The system is designed to protect cables necessary for safe shutdown of the reactor.

9.11.3 System Operation

During normal operation, the system provides standby fire protection (see Section 9.6 for a description of the facility’s Fire Protection System). During refueling and maintenance, the system is utilized for decontamination (hot water) and for hydrostatic testing and flushing (cold water).

9.11.4 Special Testing Requirements Prior to Operation

Leakage testing was performed prior to operation in accordance with ANSI B.31.1.

9.12 CONTROL OF HEAVY LOADS

9.12.1 Introduction / Licensing Background

A generic letter dated December 22, 1980, required responses to the guidelines of NUREG-0612“Control of Heavy Loads at Nuclear Power Plants.’. In response, the IP3 provisions for handling and control of heavy loads at Indian Point Unit 3 were addressed by letters dated June 22, 1981, November 11, 1981, April 21, September 30, December 6, 1982 and November 15, 1983. The NRC Safety Evaluation Report in letter dated February 13, 1985, concluded that the guidelines of NUREG-0612, Sections 5.1.1 and 5.3 have been satisfied and the Phase I of this issue for Indian Point Unit 3 is acceptable.

Additional information was provided in letters dated May 13 and August 30, 1996 in response to NRC Bulletin 96-02.

NEI-08-05 R0, “Industry Initiative on Control of Heavy Loads” documented the industry initiative to address NRC staff concerns regarding the interpretation and implementation of regulatory guidance associated with heavy load lifts, was endorsed in Regulatory Issues Summary 2008-28 and has been addressed at Indian Point 3. This supersedes prior head drop analyses.

9.12.2 Safety Basis

NUREG 0612 has two basic approaches available to demonstrate compliance: demonstrate adequate load handling reliability, or demonstrate that load drop consequences are within the limits of Criteria I-IV listed in Section 5.1 of the NUREG. Both approaches have been utilized in performing the evaluations described in the following sections. The postulated drop of the Reactor Head onto the Reactor Vessel must satisfy the requirements set forth in NEI 08-05.

In situations where a demonstration of handling system reliability was employed, the guidelines of NUREG 0612, Section 5.1.6, “Single-Failure Proof Handling Systems,” were utilized.

In situations where a demonstration of limited load drop consequences was employed, a combination of system analyses and structural analyses was utilized. The specific approach chosen was based on the completeness of the available information, and a preliminary assessment of the likelihood of success of the possible approaches.

Auxiliary Feedwater Pump (AFP) Monorails / Plant Auxiliary Building (PAB) Monorail

The two AFP Monorails and the PAB monorail were determined to meet the intent of the NRC guidance by demonstrating that adequate load handling reliability will be achieved.

Containment Polar Crane

The Containment Polar Crane was evaluated using a combination of the approaches. The principal approach was systems evaluations to demonstrate that sufficient redundancy and separation are available to maintain core cooling even in the unlikely event of a heavy load drop inside containment.

The containment was subdivided into 10 regions of interest. For all regions, with the exception of the Reactor Vessel and the annulus outside the Crane Wall, the postulated load drops were evaluated using systems evaluations. The evaluations established if the load drop could cause loss of the primary cooling mode or, if the primary cooling mode is lost, if backup cooling modes could be lost from the same drop.

In one area, the annulus region between the crane wall and the containment, it was necessary to demonstrate that sufficient load handling reliability of the auxiliary hoist will be available. This precludes the need for evaluating the consequences of load drops in this region.

In the Reactor Vessel area, a Reactor Head drop analysis was performed to show that the structural integrity of the critical components is maintained such that the core cooling will not be compromised and the core will remain covered.

9.12.3 Scope of Heavy Load Handling Systems

The following cranes and hoists were determined to be capable of handling heavy loads based on the criteria of NUREG-0612:

- Containment Polar Crane (175/28-ton) Note the Auxiliary Hook is derated to 28 Tons
- 2-ton Plant Auxiliary Building (PAB) monorail (55 ft and 73 ft elev.)
- (2) 5-ton Auxiliary Fuel Pump (AFP) building monorails
- Fuel Storage Building crane (40/5-ton).

9.12.4 Control of Heavy Loads Program

The following discuss the results of our evaluations and submittals and are controlled using commitments COM-73-01078, COM-81-02284, 02285, 02287, 02288, 02370, 02371, 02373, and COM-04-00005.

9.12.4.1 Response to NUREG 0612, Phase I Elements

A defense-in-depth approach was used to ensure that all load handling systems are designed and operated so that their probability of failure is appropriately small. The
basis for the approach was the Staff guidelines tabulated in Section 5 of NUREG-0612.and the program initiated to ensure that these guidelines are implemented These guidelines consist of the following criteria from Section 5.1.1 of NUREG-0612:

Guideline 1 - Safe Load Paths
Guideline 2 - Load Handling Procedures
Guideline 3 - Crane Operator Training
Guideline 4 - Special Lifting Devices
Guideline 5 - Lifting Devices (Not Specially Designed)
Guideline 6 - Cranes (Inspection, Testing, and Maintenance)
Guideline 7 - Crane Design

Satisfaction of these guidelines for the Polar Crane and the Fuel (Storage) Handling Crane is shown in Table 9.12-1.

Guideline 1 - Safe Load Paths

To ensure that crane operators remain knowledgeable of load handling precautions, annual refresher training is conducted to identify exclusion areas and to review load handling procedures.

In addition to the above procedures, additional structural and systems analyses were performed to determine the consequences of a load drop indicate that suitable system redundancy and structural integrity exist so that the consequences of a load drop would not exceed the criteria of NUREG-06l2, Section 5.1. The specific requirements for the polar crane and fuel handling crane are below:

Polar Crane

The containment building polar crane is utilized to remove and replace heavy loads during refueling operations. These include:

1) Control rod drive missile shield
2) Reactor vessel head
3) Reactor internals

All standard modes of failure were considered in the design of the polar crane. These modes of failure were provided for by utilization of a minimum safety factor of 5 based on the ultimate strength of the material used in the design of cables, shafts and keys, gear teeth and brakes. All crane equipment was sized to handle the single heaviest load realized during plant operation. All lifts are made by qualified personnel. The equipment is properly maintained and periodically inspected by qualified personnel. An analysis of impact loading on the reactor vessel due to dropping the reactor vessel head is described in Section 9.12.4.2.

For the Indian Point Unit 3 polar crane, operating procedures define three areas over which loads are not allowed to be carried with the exception of certain pre-identified load movements. These areas are as follows:

1. directly over the reactor vessel, where no heavy loads are allowed to be carried (with the exception of movements of the reactor vessel head, upper internals, missile shields, and inservice inspection (ISI) tool into and out of the area).

2. When the reactor vessel head is off and there is fuel in the reactor vessel, Class 3 loads, including CRDM missile shields, CRDM support beams, and the reactor vessel head stud tensioners, cannot be moved over the reactor cavity.

3. over residual heat removal (RHR) heat exchanger NO. 32, which may be exposed to overhead load drops.

No unidentified loads are moved over either exclusion area at any time. For certain loads (identified by procedures) which must be moved in and out of the reactor vessel area, the loads are moved by the most direct route to pre-designated lay down areas. A load handling supervisor or person in charge is present to ensure that procedures are followed and that exclusion area boundaries are not violated.

Fuel Storage Building Crane

The fuel storage building crane is utilized to move loads not exceeding 2000 lbs during normal refueling operations. These loads include:

1) Irradiated specimens
2) Neutron source
3) Crane load block
4) Burnable poison rod and handling tool.

Electrical limit switches incorporated on the bridge rails and trolley rails of the fuel storage building crane limit crane travel so that loads are not inadvertently moved over any region of the spent fuel pit which contains irradiated fuel. Removable mechanical stops are available for installation on the bridge rails of the fuel storage building crane to backup the bridge rail electrical limit switches and prevent the bridge of the crane from traveling further north than a point directly over the spot in the spent fuel pit that is reserved for the spent fuel cask. With

the bridge rail removable mechanical stops in use, it will be impossible to carry any object over the spent fuel storage areas north of the spot in the pit that is reserved for the cask with either 40 or 5-ton hook of the fuel storage building crane.

With the bridge rail removable mechanical stops in use and if the trolley rail electrical limit switches are bypassed or out of service, it is possible for the fuel storage building crane to carry objects over the spent fuel storage areas that are directly east of the spot in the pit that is reserved for the cask. However, the FSAR and plant procedures prevent any object weighing more than 2,000 pounds from being moved over any region of the spent fuel pit containing irradiated fuel, unless a technical analysis has been performed consistent with the requirements of NUREG-0612 establishing the necessary controls to assure that a load drop accident could damage no more than a single fuel assembly. Administrative and procedural controls to protect fuel and fuel racks may include a path selection to prevent loads from passing over or near fuel. For cases in which very heavy loads (>30,000 pounds) are transported over the spent fuel pit, the loads cannot under any circumstances pass over fresh or irradiated fuel. In all cases where loads >2,000 pounds are carried over the pit, the ventilation system must be operable.

The electrical limit switches may be bypassed and the mechanical stops may be removed under administrative controls and the crane moved over spent fuel storage areas, provided that the fuel storage building ventilation system is operable, the spent fuel boron concentration is at least 1,000 ppm and there is no heavy load carried. This allows operations over the spent fuel pit with the 5-ton hoist. The 40-ton hoist may not carry any heavy load over the spent fuel storage areas.

All standard modes of failure have been considered in the design of the fuel storage building crane. These modes of failure were provided for by utilization of a minimum safety factor of 5 based on the ultimate strength of the material used in the design of cables, shafts and keys, gear teeth and brakes.

All crane equipment was sized to handle the single heaviest load realized during plant operation. All lifts are made by qualified personnel. The equipment is properly maintained and periodically inspected by qualified personnel. An analysis of impact loading of the spent fuel cask into the spent fuel storage pool is provided in Section 9.5.3

<u>Guideline 2 - Load Handling Procedures</u>

A series of operating procedures have been developed for operation of load handling equipment at Indian Point Unit 3.

Load handling procedures provide for the movement of all heavy loads in the vicinity of irradiated fuel or systems and equipment required for safe shutdown and decay heat removal, and that load designation was based on the generic load identified in Table 3-1 of NUREG-0612. Further, these procedures contain the precautionary information required by NUREG-0612, Guideline 2. These procedures comply with the commitments made for safe load handling.

Guideline 3 – Crane Operator Training

A qualification program for the qualification and training of crane operators at Indian Point Unit 3 have been developed to meet the provisions of ANSI 830.2-1976, with no exceptions taken. Crane operator training and qualification is addressed in the qualification program and include precautions and instructions to assure proper operator conduct.

A qualification program for crane operators was established requiring the following:

a) Certification by a company physician that the crane operator met the physical standards set forth in ANSI B30.2.0-1967, Article 2.3.1.2.

b) Successful completion of an oral and practical examination given by a designated experienced Crane Instructor.

c) Certification by the crane operator that he has read, understands and will comply with the operational and safety requirements set forth by OSHA and ANSI B30.2.0-1967.

This qualification program meets the requirements of Chapter 2-3 of ANSI B30.2.0-1967, “Operation – Overhead and Gantry Cranes”, as developed by the American National Safety Code for Cranes, Derricks, Hoists, Jacks and Slings.

Guideline 4 - Special Lifting Devices

The following special lifting devices are subject to compliance with the requirements of NUREG 0612, Guideline 4:

- reactor vessel head lifting rig (also described in Section 9.5.2)
- internals lift rig (also described in Section 9.5.2)
- reactor vessel ISI tool.

All three devices were designed and manufactured prior to the existence of ANSI N14.6-1978. Based on review of ANSI criteria, detailed evaluation of these devices has been limited to Sections 3.2 (Design Criteria) and 5 (Acceptance. Testing, Maintenance, and Assurance of Continuing Compliance). Detailed comparison of each of the devices indicates that the devices comply with ANSI criteria with limited exceptions.

The designer verified that each device was originally designed with a factor of safety of 5:1 on ultimate strength and that suitable margins to yield exist for all components. Consideration of dynamic effects is not necessary since the maximum dynamic load has been calculated to be less than 5.5% of the static load and does not significantly affect the load handling reliability of these devices.

Although only one of the devices was originally load tested to 150% of rated load or greater, adequate documentation exists to document proof of workmanship of these devices. The internals lift rig has been load tested to over 200% of the heavy load of concern (the upper internals). The ISI tool has been load tested to 137% of rated load. The reactor vessel head lift rig was only lifted 100% of rated load on various occasions with no signs of deformation or overstress.

To ensure continued load handling reliability, these devices are inspected by qualified personnel at regular intervals (12 months or prior to use) except for the 50.59 Evaluation Number 05-0294-PR-00-RE which relaxed the Reactor Head Lifting Rig and Upper Internals Lifting Rig inspection interval for NDE and dimensional examination from 12 months, or prior to use, to 5 year intervals.

Guideline 5 - Lifting Devices (Not Specially Designed)

Plant procedures require that sling selection and use for all ·loads requiring sling lifting devices be in accordance with ANSI B30.9-197l.

As noted for special lifting devices, calculations indicate that the maximum dynamic load experienced is only 2.1% of the maximum static load for the main hoist and 5.5% for the auxiliary hoist. Addition of these dynamic loads does not significantly affect load handling reliability and therefore dynamic loads have not been considered in selection of slings.

Guideline 6 - Cranes (Inspection, Testing, and Maintenance)

A program for inspection, testing, and maintenance of the polar crane has been developed that satisfies the criteria in ANSI B30.2-1976 Chapter 2-2, with no exceptions noted. The criteria of ANSI B30.2-1976 are not easily applied to such handling systems as monorails and hand-driven hoists. Accordingly, a procedure has been developed based on the criteria of ANSI B30.11-1973, "Monorail Systems and Underhung Cranes", with no exceptions noted from the criteria of the standard.

Guideline 7 - Crane Design

A design analysis of each handling system using the design criteria of the applicable standards has been performed. The polar crane has been evaluated in accordance, with ANSI B30.2-1976, while the AFP building monorail has been evaluated in accordance with ANSI B30.11, ‘Monorail Systems and Underhung Cranes,’ and ANSI B30.16, ‘Overhead Hoists’.

The polar crane was built prior to the issuance of ANSI B30.2-1976 and CMAA-70. However, a detailed point-by-point comparison has been performed, comparing information from the manufacturer with the criteria of these standards. Analysis was performed for only those components that are load bearing or are necessary to prevent conditions which could lead to a load drop. This review indicates that the polar crane complies with all requirements with the exception of Specification 3.2 of CMAA-70 and Section 2.1.4.1 of ANSI B30.2-1976. These specifications require that welding be performed in accordance with AWS D1.1, ‘Structural Welding Code’, and AWS D14.1, ‘Specifications for Welding Industrial and Mill Cranes’. The welding procedures used are equivalent to current welding criteria based on the following:

a) welding was performed in accordance with the then-current code AWS DI,I, ‘Structural Welding Code’

b) practices and procedures used for welding are equivalent to those in AWS D14.I, which was not issued at the time

c) welders were qualified to existing AWS criteria

d) all welds were visually inspected

e) structural integrity was demonstrated when the polar crane was used to perform a 450-ton (250% of rated capacity) construction lift.

Section 9.5.1, Design Bases, provides more detail on the fuel storage building crane under the heading Design Codes and Criteria.

In the AFP building, no hoist is permanently attached to the monorail system. Hoist selection criteria for the AFP and PAB monorails comply with the requirements of ANSI B30.16-1973 and have been included in procedures. Review of monorail design indicates that the AFP and PAB monorails comply with the criteria of ANSI B30.11-1973.

Additional specific information concerning design compliance with the more restrictive requirements of CMAA-70 is contained in the safety evaluation report.

9.12.4.2 Reactor Pressure Vessel Head (RPVH) Lifting Procedures

In response to NEI 08-05, an evaluation of a head drop was performed. For this drop scenario, it was postulated that during removal or installation of the closure head assembly, in which the closure head is lifted or lowered directly above the reactor vessel (RV), the polar crane fails and the closure head assembly falls and impacts flat and concentrically with the RV flange.

The stresses and strains caused by the postulated impact were evaluated to demonstrate that structural integrity of the critical components is maintained such that core cooling will not be compromised and the core will remain covered.

The analysis used a conservative weight for the Reactor Head of 350,000 pounds, which matches the Polar Crane Main Hook capacity and exceeds the weight of the Reactor Head and Lifting Rig, and considered a drop through air from a height of 32 feet above the Reactor Vessel flange, the maximum height as controlled by plant procedures. The finite element model was prepared using ANSYS, the Reactor Head drop and impact was simulated using LS-DYNA, and the response of the Reactor Vessel, Reactor Vessel support components, and main loop piping were obtained with the postprocessor LS-PREPOST.

The maximum primary stress intensity in the Reactor Vessel shell at the Inlet nozzle is 60,255 psi, versus a 72,000 psi allowable. The maximum primary stress intensity in the Reactor Vessel shell at the Outlet nozzle is 60,766 psi, versus a 72,000 psi allowable. The maximum primary stress intensity at the inlet and outlet nozzles was found to be 68,388 psi and 63,936 psi, respectively, against a 72,000 psi allowable.

The analysis concludes that the structural integrity of the critical components is maintained such that core cooling is not compromised and the core remains covered.

The Polar Crane and Reactor Pressure Vessel Head lifts procedures are used to control the lift and replacement of the reactor pressure vessel head. These procedures incorporate the 32 feet limit on the lift height of the Reactor Vessel Head assembly which weighs less than 350,000 pounds.

One other load is carried over the open reactor vessel that could potentially damage spent fuel in the vessel. This is the Reactor Vessel Weld ISI tool. Its weight is approximately 5 tons. For this particular lift, which is performed by the Auxiliary Hoist, adequate load handling reliability will be assured on the same basis as for loads lifted by the Auxiliary Hoist in the Annulus Region. This basis is described in Section 9.12.4.1.

9.12.4.3 Single Failure Proof Cranes for Spent Fuel Casks

The original fuel storage building crane furnished by Whiting Corporation did not meet the guidelines of NRC NUREG-0612 and NUREG-0554 (Single-Failure-Proof Cranes for Nuclear Power Plants, May 1979) for designation as "single-failure-proof". The original crane has a main hook rated for 40 tons and an auxiliary hook rated at 5 tons. To support spent fuel cask handling activities without the necessity of having to postulate the drop of a spent fuel cask, the Whiting crane was replaced in 2010 with a single-failure-proof Morris Material Handling crane. The replacement crane has a main hook rated at 40 tons that complies with current guidelines for designation as single-failure-proof, including the applicable guidelines of NRC NUREG-0554 and the applicable requirements of American Society of Mechanical Engineers ASME NOG-1-2004, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder). The replacement crane has an auxiliary hook rated at 5 tons that was not upgraded to single-failure-proof guidelines.

9.12.5 Safety Evaluation

The controls implemented to address NUREG- 0612 Phase 1 elements make the risk of a load drop very unlikely. The use of increased safety factors for load path elements makes the risk of a load drop extremely unlikely and acceptably low. In the event of a postulated load drop, the consequences are acceptable, as demonstrated by system analyses or the load drop analysis. Restrictions on load height, load weight, and medium under the load are reflected in plant procedures. The risk associated with the movement of heavy loads is evaluated and controlled by station procedures.

Table 9.12-1

NUREG-0612 Compliance Matrix

Heavy Loads	Weight or Capacity (tons)	Guideline 1 Safe Load Paths	Guideline 2 Procedures	Guideline 3 Crane Operator Training	Guideline 4 Special Lifting Devices	Guideline 5 Slings	Guideline 6 Crane – Test and Inspection	Guideline 7 Crane Design	Interim Measure 1 Technical Specifications	Interim Measure 6 Special Attention
1. Containment Polar Crane	175	++	++	C	++	++	C	R	++	R
Reactor Vessel Head	169	R	C	++	R	++	++	++	++	R
Upper Internals (Plenum)	67	R	C	++	R	++	++	++	++	R
Inservice Inspection Tool	5	R	C	++	R	++	++	++	++	R
Reactor Coolant Pumps	32	R	C	++	R	++	++	++	++	R
Missile Shields	7.5	R	C	++	++	C	++	++	++	R
Crane Load Block	4.5	R	C	++	++	C	++	++	++	R
Concrete Hatch Cover	7.3	R	C	++	++	C	++	++	++	R
Pressurizer Missile Shield	7.5	R	C	++	++	C	++	++	++	R
2. Fuel Handling Crane	40	R	R	C	++	++	C	R	C	++

C - Action complies with NUREG-0612 Guideline.
R - Revisions/modifications designed to comply with NUREG-0612 Guideline.
++ - Not applicable.

9.13 Backup Spent Fuel Pool Cooling System

9.13.1 Design Bases

The Backup Spent Fuel Pool Cooling System (BSFPCS) has been installed to allow maintenance and repair and to operate in parallel with the normal SFP Cooling System (SFPCS) to improve spent fuel pool (SFP) conditions during refueling activities. The BSFPCS is designed to extend the period of time the SFPCS could be taken out of service during periods of maximum heat load (a full-core offload completed no earlier than 254 hours subcritical).

9.13.2 System Description

The Backup Spent Fuel Pool Cooling System (BSFPCS) consists of a primary loop and a secondary loop. The primary loop handles the SFP water. The secondary loop is the system heat sink. The primary loop components are located in the Fuel Storage Building (FSB). The secondary loop components are located both inside the FSB and routed outside to the waste holdup tank pad where the cooling towers are located.

The primary loop takes suction from the SFP using one of two BSFPC pumps (primary loop), flows through the BSFPC plate heat exchanger (HX) and returns to the SFP. The BSFPC pumps (primary loop) are 100 percent capacity, 25 horsepower primary loop pumps designed for 1500 gpm at 45 feet of head. The primary loop pumps are locally controlled and the local control station has red and green status lights. There are duplicate status lights at the secondary loop pump control station. Disconnect switches for the motors have padlocks to ensure only one motor is energized at a time. System piping is stainless steel.

The secondary loop takes suction from the two BSFPC open-circuit evaporative cooling towers, flows through the BSFPC HX and returns to the cooling towers. The secondary loop has two 100 percent capacity, 200 horsepower secondary loop pumps rated for 2500 gpm each at 192 feet of head. There are red and green status lights at the control station. Disconnect switches for the motors have padlocks to ensure only one motor is energized at a time.

The BSFPCS heat exchanger is a plate and frame type. The secondary loop is maintained at a higher pressure than the primary loop to prevent leakage from the SFP into the secondary loop. Each primary pump receives a trip signal when the differential pressure is less than 10 psid. The secondary loop pump remains running to preclude a leak path from the primary loop into the secondary loop. Differential pressure less than 10 psid also illuminates yellow status lights at the primary and secondary loop pump control stations.

The secondary loop cooling towers are two 50 percent capacity, open-circuit evaporative cooling towers. Each tower has a fan that pulls air over the water as it is sprayed over the fill material to assist cooling by evaporation. The fans cycle independently, depending on the temperature at the outlet of the cooling towers. The first fan will start at 71°F and stop at 70°F. The second fan will start at 76°F and stop at 75°F. The tower also functions as a container for the water in the secondary loop. Makeup water is supplied from the Demineralized Makeup

Water System (DMWS) line going to the Primary Water Storage Tank (PWST). Emergency Make up is from the fire protection system. There is hard piping from both the DMWS and fire protection piping for the cooling tower makeup. The water treatment system, common to both IP2 and IP3, provides makeup to the DWMS. Fill requirements are 1600 gallons with a maximum required makeup of 72 gpm. The secondary system is stainless steel to prevent corrosion.

Pressure and temperature indicators are provided locally at pump discharges and on the plate heat exchanger discharges to evaluate proper operation. Their ranges are consistent with maximum operating conditions. A temperature element in the cooling tower basin and associated controls will regulate cooling tower fans as necessary for efficient operation. The spent SFP local temperature indication is also available to monitor pool conditions.

Normal power to the BSFPCS is supplied from 480 VAC MCC E1 via Power Panel PP-582 (primary pumps) and 480 VAC MCC E2 (secondary pumps). There is a transfer switch that allows power to the BSFPCS to be supplied from a portable diesel generator.

The system will be kept in dry layup most of the time and has been designed to accommodate this. Typically, the BSFPCS will only be used during refueling outages in order to supplement the SFPCS by reducing the SFP temperature to as low as feasible. During maintenance and repairs the BSFPCS may be used as the sole source of cooling. This is expected to be infrequent.

9.13.3 Safety Evaluation

The BSFPCS is not designed seismic I or for a tornado event. The system is designed to preclude seismic interaction with the spent fuel pool and its components. The primary loop has Seismic II supports. The secondary loop is designed in the same manner from the heat exchanger to the first isolation valves on the inlet and outlet piping. The cooling towers also have seismic II supports.

Failures in the system piping due to seismic events, tornados or moderate energy line breaks do not cause unacceptable interactions. The lines are moderate energy because the operating pressures are about 17 to 85 psig and the temperatures are all below 142^{o}F, except for the primary pump discharge to the heat exchanger at 190^{o}F. Primary and secondary side piping has a design pressure and temperature of 210 psig and 300^{o}F (the secondary side fill and makeup supplies are 150 psig and 500^{o}F). Also, pump shutoff head pressures are less than 100 psig. A moderate energy line break would not affect equipment. A moderate energy piping break on the pump discharge would be different than one on the SFPCS. The BSFPCS does not have an anti-siphoning hole in its suction line at the 93 feet 1 inch elevation like the SFPCS, but the bottom of the suction line is at 87 feet 8 inch which is well above the SFPCS suction line at 74 feet 4-3/4 inch. This would allow an additional SFP reduction of 5 feet 5 inches but would not affect the ability to provide makeup to preclude pool boiling.

The system is normally run as backup to the SFPCS to provide enhanced cooling during outages. Under these conditions the BSFPCS is an enhancement and not credited to meet the design basis cooling requirements. During periods that the SFPCS is taken out of service for necessary maintenance or repair the BSFPCS may be used to cool the SFP as a stand-alone system provided the limitations for such use, described below, are met.

The potential for loss of offsite power during periods when the BSFPCS is the only cooling in place will be accommodated by a backup diesel that would be temporarily brought in. The backup diesel will be required for those cases where the scheduled work exceeds the time to raise the temperature to boiling without cooling.

The BSFPCS can maintain the SFP bulk temperature at 175°F per calculation IP-CALC-10-00121. Sample heat loads and wet bulb temperatures shown below:

Wet Bulb Temperature [°F]	Heat Load [BTU/Hr] at 175°F
40	36,451,000
45	35,622,000
50	34,938,000
55	34,146,000
60	33,425,000
65	32,525,000
70	31,696,000
75	31,048,000

Administrative controls will be the measures used to compensate for the reduced reliability to improve the availability and reliability of the BSFPCS and its makeup system when the BSFPCS is the only means of forced cooling. A specific Defense in Depth Plan will be implemented in accordance with station procedures and authorized by senior site management to ensure actions are taken. These, including past proposed controls, are as follows:

1. Work will be scheduled when the SFP heat load is at a reduced value rather than the design value.

2. A backup diesel generator will be tested and made available to power the BSFPCS where the scheduled work exceeds the time to raise the temperature to boiling without cooling.

3. The SFP will be brought to a temperature as low as reasonable, but not lower than the design temperature, prior to starting the work.

4. The ambient wet bulb temperature will be assured to be at or below the temperature, with 5°F of margin, which is calculated to keep the SFP below 150°F (for online work) and 175°F (for core offload work) given the residual heat load at the time work starts. Additional margin will be provided by keeping recovery times within the time the SFP would reach 190°F. The recovery time is the time to detect BSFPCS failure (i.e., based on monitoring times) added to the time to restart the main or redundant pumps on the

backup diesel (assumes offsite power lost) given the residual heat load at the time work starts. The time to start the diesel and pumps will be demonstrated in the field in order to determine time required. The wet bulb temperature will be taken about noon on the day work is to commence (or the day before) in order to allow for daily temperature swings.

5. The approximate time for recovery actions defined for normal types of events (e.g., loss of power, loss of makeup water, pump failure, loss of suction, etc) will be identified and, if there is a loss of BSFPCS, those which could be implemented before 196^{o}F is reached will be used. Table 9.13-1 provides a failure analysis to define these.

6. Monitoring of the wet bulb temperature, BSFPCS operation and contractor water system supply will be performed at a frequency consistent with the time for the SFP to rise by 5^{o}F if the BSFPCS were to be lost. For example, if the SFP temperature rise was 10^{o}F per hour, then the wet bulb temperature will be taken every 30 minutes and the proper operation of the BSFPCS and the makeup water supply will be validated every 30 minutes.

7. The SFP High Temperature alarm set point will assure that the alarm alerts the control room that the calculated temperature is exceeded.

8. The normal water supply and backup water supply to the secondary loop of the BSFPCS and the sources of SFP makeup (Primary Water Storage Tank, RWST through the plant demineralizers and from the Fire Water Tank) will be protected when the BSFPCS is the only source of cooling.

9.13.4 Inspection and Testing Requirements

The BSFPCS is put into layup and taken out of layup in accordance with plant procedures. The system is functionally tested prior to being placed into service.

Table 9.13-1
Failure Analysis – Backup Spent Fuel Pit Cooling (BSFPC)

Component	Malfunction	Consequences/ Comments	How to fix/ How long to fix	How to limit pool temp <196°F
Primary BU SFP Pump (31-BSFPP Pri.) (32-BSFPP Pri.)	1. Fails to start 2. Fails to run 3. Inadequate flow 4. Loss of power 5. Interlock intervention	1-3. Should this pump fail to operate, a second primary pump is available as a backup. 4. MCC E1 is the normal power supply. If this fails, an auxiliary diesel generator can be utilized. 5. This will prevent either primary pump from operating to prevent contamination of the environment through the cooling towers.	1-3. Switch to the other primary pump. (Note 1) 4. Transfer power to the auxiliary diesel generator through the transfer switch. 5. BSFPC is not available. Normal SFPC would have to be put back in service. (Note 2)	1-5. Once the backup pump or power supply is in service, the pool will no longer be in any additional danger to approach 196°F. Should neither pump be available, utilize 3-AOP-SF-01 Loss of Spent Fuel Pit Cooling.
Secondary BU SFP Pump (31-BSFPP Sec.) (32-BSFPP Sec.)	1. Fails to start 2. Fails to run 3. Inadequate flow 4. Loss of power	1-3. Should this pump fail to operate, a second secondary pump is available as a backup. 4. MCC E2 is the normal power supply. If this fails, an auxiliary diesel generator can be utilized.	1-3. Switch to the other secondary pump. 4. Transfer power to the auxiliary diesel generator through the transfer switch.	1-4. Once the backup pump or power supply is in service, the pool will no longer be in any additional danger to approach 196°F. Should neither pump be available, utilize 3-AOP-SF-01 Loss of Spent Fuel Pit Cooling.
Heat Exchanger (BSFP Plate HX)	1. Tube Rupture	1. A tube rupture in this heat exchanger would create a leakage path for contaminated SFP water to leak to the environment.	1. BSFPC is not available. Normal SFPC would have to be put back in service. (Note 2)	1. Utilize 3-AOP-SF-01 Loss of Spent Fuel Pit Cooling.

Table 9.13-1
Failure Analysis – Backup Spent Fuel Pit Cooling (BSFPC)

Component	Malfunction	Consequences/ Comments	How to fix/ How long to fix	How to limit pool temp <196^{o}F
31/32-BSFP Cooling Towers 31/31-BSFP Fan Cooling	1. Fan failure 2. Loss of power 3. Loss of 1 cooling tower. 4. Loss of both cooling towers. 5. Makeup water unavailable.	1,2. The fans are set to run when the ambient air temperature reaches certain set points. If these fans were to fail, the heat removal capabilities would be limited. 3. The cooling towers are both rated for 50% capacity. Therefore, they cannot be utilized independently with full cooling capacity intact. Losing 1 tower, while still operable, would greatly reduce the cooling capacity. 4. Loss of both cooling towers would render the BSFPCS system inoperable. 5. Normal makeup is provided through from the PW system. If this is unavailable, Fire water, taken from FPS and the FWST is available as a backup for makeup water.	1,2,3. The SFP temperature would have to be monitored and the normal SFPC system would have to be put back in service eventually. 4. BSFPCS is not available. Normal SFPC would have to be put back in service. (Note 2) 5. The water flow rates and temp/press would have to be monitored, eventually requiring the normal SFPC system to be placed back in service.	1-5. Utilize 3-AOP-SF-01 Loss of Spent Fuel Pit Cooling.
DPIS-4469	1. Fails to indicate <10 psid.	1. Upon indication of <10 psid, indicating a potential leak in the heat exchanger, this unit sends a trip signal to the primary pumps. This prevents contaminated water on the primary side from leaking to the secondary side. The secondary pump remains in operation to prevent this leakage.	1. This switch takes signal from pressure indicators on the primary and secondary sides. Comparing these local gauges would indicate a problem with the switch, and would require manual trip of the primary pump.	1. Utilize 3-AOP-SF-01 Loss of Spent Fuel Pit Cooling.

Table 9.13-1
Failure Analysis – Backup Spent Fuel Pit Cooling (BSFPC)

Component	Malfunction	Consequences/ Comments	How to fix/ How long to fix	How to limit pool temp <196°F
			Normal SFPC would have to be put back in service. (Note 2)	
Secondary Side Strainers (AC-ST-1946) (AC-ST-1945)	1. Too much debris causing a >5 psid drop. 2. DPI-4473/4472 fail to indicate the pressure drop.	1,2. AC-ST-1945 / AC-ST-1946 are the strainers located upstream of the primary secondary-side pumps. If this were to cause a large enough pressure drop, cavitation could occur.	1,2. The baskets should be cleaned out repeatedly until the pressure drop is satisfactory. If this can't be achieved, a backup secondary-side pump is available. If they cannot be cleaned out to a satisfactory level, normal SFPC would have to be kept in service or restored.	1,2. These strainers are only used on startup to remove any damaging particles that could have built up. If they cannot be cleaned out to a satisfactory level, utilize 3-AOP-SF-01 Loss of Spent Fuel Pit Cooling.
Check Valves (AC-1900) (AC-1901) (AC-1957) (AC-1958)	1. Failing to close. 2. Failing to open.	1. On sections with redundant equipment, a check valve that doesn't close properly can allow backflow into the unused section. This can affect performance of the system. Typically, this system has normally closed valves on the upstream sides of the unused redundant pumps, which would limit the performance loss. 2. A check valve failing to open would stop flow in the correct direction.	1,2. An evaluation of the loss of performance would determine if the malfunctioning check valve will cause BSFPCS to become inoperable. If it is determined to be inoperable, normal SFPC would have to be put back in service. (Note 2)	1,2. If performance isn't affected by a lot, the system can function but at a reduced capacity. If the condition becomes too degraded, utilize 3-AOP-SF-01 Loss of Spent Fuel Pit Cooling.

Table 9.13-1
Failure Analysis – Backup Spent Fuel Pit Cooling (BSFPC)

Component	Malfunction	Consequences/ Comments	How to fix/ How long to fix	How to limit pool temp <196°F
Gate Valves (AC-1923) (AC-1924) (AC-1939) (AC-1940) Butterfly valve (AC-1947)	1. Failing to open. (Pump isolation) 2. Throttling valve failure (AC-1947)	1. Gate valves are supplied to stop flow through redundant pumps. These valves are all manipulated manually. If one were to fail, it would render the backup redundant pumps inoperable. 2. AC-1947 is a butterfly valve that is used to achieve the 2500 gpm necessary by throttling the flow downstream of the heat exchanger. If this were to fail, the flow rate would have to be maintained by throttling some other valve, as long as this valve didn't fail closed.	1. These normally closed gate valves are used to isolate the backup pumps. If they cannot be opened, the evolution would not be started. 2. If another valve could be throttled and this valve doesn't hinder flow, BSFPCS would be operable. If flow cannot be throttled, the heat exchanger capacity must be determined at the available flow rate.	1,2. As long as these valves don't challenge flow of the system, BSFPCS can be run at limited capacity with proper oversight. If one of these valves fails leaving BSFPCS inoperable, utilize 3-AOP-SF-01 Loss of Spent Fuel Pit Cooling.

Note 1: Transferring pumps is a short term process (10-60 minutes)

Note 2: The time to place the normal SFPC depends on the maintenance activity that is being performed.

- SFPC Heat Exchanger: 13 Hours
- SW Valves: Not yet defined because repair or replace not decided.