CHAPTER 9
AUXILIARY AND EMERGENCY SYSTEMS

9.0 INTRODUCTION

The auxiliary and emergency systems are supporting systems required to ensure the safe operation or servicing of the reactor coolant system (detailed in Chapter 4).

In some cases the dependable operation of several systems is required to protect the reactor coolant system by controlling system conditions within specified operating limits. Certain systems are required to operate under emergency conditions.

This section considers systems in which component malfunctions, inadvertent interruptions of system operation, or a partial system failure may lead to a hazardous or unsafe condition. The extent of information provided for each system is proportional to the relative contribution of, or reliance placed upon, each system with respect to the overall plant operational safety.

The following systems are considered under this category:

<u>Chemical and Volume Control System</u>

This system provides for boron injection, chemical additions for corrosion control, reactor coolant cleanup and degasification, reactor coolant makeup, reprocessing of primary letdown from the reactor coolant system, and reactor coolant pump seal-water injection.

<u>Auxiliary Coolant System</u>

This system provides for transferring heat from the reactor coolant during shutdown, stored spent fuel, and other components to the service water system and consists of the following three loops:

1. The residual heat removal loop removes residual and sensible heat from the core and reduces the temperature of the reactor coolant system during the second phase of plant cooldown.

2. The spent fuel pit loop removes decay heat from the spent fuel pit.

3. The component cooling loop removes residual and sensible heat from the reactor coolant system via the residual heat removal loop during plant shutdown, cools the spent fuel pit water and the letdown flow to the chemical and volume control system during power operation and provides cooling to dissipate waste heat from various primary and safety-related plant components.

<u>Sampling System</u>

This system provides the equipment necessary to obtain liquid and gaseous samples from the reactor plant systems.

<u>Facility Service Systems</u>

These systems include fire protection, service water, and auxiliary building ventilation.

Reactor Components Handling System

This system provides for handling fuel assemblies, control rod assemblies, core structural components, and material irradiation specimens.

Equipment and Decontamination Processes

These procedures provide for the removal of radioactive deposits from system surfaces.

Primary Auxiliary Building Ventilation System

This system maintains ambient operation temperatures and provides purging of the auxiliary building to the plant vent.

Control Room Ventilation System

This system maintains the required environment in the control room.

9.1 GENERAL DESIGN CRITERIA

9.1.1 Applicable Criteria

The criteria, which apply primarily to other systems discussed in other sections are listed and cross-referenced because details of directly related systems and equipment are given in this section. Those criteria, which are specific to one of the auxiliary and emergency systems are listed and discussed in the appropriate system design-basis section.

9.1.2 Related Criteria

9.1.2.1 Reactivity Control System 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)

As described in Chapter 7 and justified in Chapter 14, the reactor protection systems are designed to limit reactivity transients to maintain DNBR at or above the applicable safety analysis DNBR limit due to any single malfunction in the deboration controls.

9.1.2.2 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)

Each of the auxiliary cooling systems, which serves an emergency function provides 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.

9.1.2.3 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)

Each of the auxiliary cooling systems that 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 function.

9.2 CHEMICAL AND VOLUME CONTROL SYSTEM

The chemical and volume control system (1) adjusts the concentration of boric acid for nuclear 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 within design levels. The system is also used to fill and hydrostatically test the reactor coolant system.

This system has provisions for supplying the following chemicals:

1. Chemicals to regenerate the deborating demineralizers.
2. Hydrogen to the volume control tank.
3. Nitrogen as required for purging the volume control tank.
4. Hydrazine and lithium hydroxide, as required, via the chemical mixing tank to the charging pumps suction.

During normal plant operation, reactor coolant letdown from the intermediate leg of loop 21 flows through the shell side of the regenerative heat exchanger where its temperature is reduced by transferring heat to the charging fluid. The coolant then flows through a letdown orifice, which regulates flow and reduces the 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 (this valve essentially controls backpressure on the orifices and prevents flashing there). After passing through one of the mixed-bed demineralizers, where ionic impurities are removed, the fluid flows through the reactor coolant filter, and enters the volume control tank through a spray nozzle.

The coolant flows from the volume control tank to the charging pumps that raise the pressure above that in the reactor coolant system. The high-pressure water flows from the primary auxiliary building to the reactor containment along two parallel paths. One path returns directly to the reactor coolant system through the tube side of the regenerative heat exchanger to the cold leg of loop 21. The second path injects water into the seals of the reactor coolant pumps. A portion of this seal-water is injected into the reactor coolant system through the reactor coolant pumps labyrinth seals. The remainder of this seal-water flow returns to the volume control tank through the seal-water filter and the seal-water heat exchanger.

Concentrated boric acid, used for chemical shim or shutdown operations, is mixed in the batching tank. A transfer pump is used to transfer the batch to the boric acid storage tanks, which maintain a large inventory of concentrated boric acid solution. Small quantities of boric acid solution are metered from the discharge of the operating boric acid transfer pump for mixing with primary water in the blender to provide makeup for normal leakage or for increasing the boron concentration in the reactor coolant system.

A chemical mixing tank (primary auxiliary building - 98-ft elevation) is provided to supply small quantities of hydrazine and lithium hydroxide to the charging pump suction. However, this will generally be accomplished through the letdown relief valve exhaust line. This line has a sample header in the sampling room into which the chemicals can be added.

Equipment for processing reactor coolant for reuse of boric acid and reactor makeup water is no longer used and has been partially removed.

9.2.1 Design Bases

9.2.1.1 Redundancy of Reactivity Control

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

In addition to the reactivity control achieved by the rod cluster control as detailed in Chapter 7, reactivity control is provided by the chemical and volume control system, which regulates the concentration of boric acid solution neutron absorber in the reactor coolant system. The system is designed to prevent, under anticipated system malfunction, uncontrolled or inadvertent reactivity changes, which might cause system parameters to exceed design limits.

9.2.1.2 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 of the public. (GDC 30)

Normal reactivity shutdown capability is provided by control rods, with boric acid injection used to compensate for the long term xenon decay transient and for plant cooldown. Any time that the plant is at power, the quantity of boric acid retained in the 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 power. Boric acid can be injected by one charging pump and one boric acid transfer pump to shut the reactor down even with no rods inserted. Additional boric acid can be injected to compensate for xenon decay although xenon decay below the equilibrium operating level will not begin until approximately 12-15 hr after shutdown. Additional boric acid is employed if it is desired to bring the reactor to cold shutdown conditions. In addition, borated makeup water can

be supplied to the primary system from the refueling water storage tank in the event that availability of the boric acid transfer pumps is lost.

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

9.2.1.3 Reactivity Hot Shutdown Capability

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

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 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 rod cluster control 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.

9.2.1.4 Reactivity Shutdown Capability

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

The reactor core, together with the reactor control protection system is designed so that the minimum allowable DNBR remains at or above the applicable safety analysis DNBR limit and there is no fuel melting during normal operation including anticipated transients.

The shutdown groups of rod cluster control assemblies are provided to supplement the control group of rod cluster control assemblies to make the reactor at least 1-percent subcritical (k_{eff} = 0.99) following trip from any credible operating condition to the hot, zero power condition assuming the most reactive rod cluster control assembly remains in the fully withdrawn position.

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 boron addition via 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.

9.2.1.5 Codes and Classifications

All pressure retaining components (or compartments of components), which are exposed to reactor coolant comply with the code requirements as shown in Table 9.2-1.

The tube side on both the regenerative and excess letdown heat exchangers are designed as ASME III, Class C. This designation is based on the applicable codes at the time of construction and on the following considerations: (1) each exchanger is connected to the primary coolant system by a 3 inch line (Regenerative Heat Exchanger) or a 1 inch line (Excess Letdown Heat Exchanger), and (2) each is located inside the reactor containment. Contaminated primary coolant escaping from the primary coolant system during a break in one of these lines is confined to the reactor containment building and no public hazard results as discussed in Section 14.3.

9.2.2 System Design and Operation

The chemical and volume control system, shown in Plant Drawings 9321-2736, 208168, and 9321-2737 [Formerly UFSAR Figures 9.2-1 (Sheets 1 through 3)] provides a means for injection of control poison in the form of boric acid solution, chemical additions for corrosion control, and reactor coolant cleanup and 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 pump seals.

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. System design enables post-operational testing to applicable code test pressures. Testing is based upon requirements set forth in ASME Section XI, as discussed in Section 1.12.

During plant operation, reactor coolant is removed from the reactor coolant loop cold leg through the letdown line located on the suction side of the pump and is returned to the cold leg of the same loop 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, which reduces the coolant pressure. The cooled, low-pressure water leaves the reactor containment and enters the auxiliary building where it undergoes a second temperature reduction in the tube side of the 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 filter and enters the volume control tank though a spray nozzle.

Hydrogen is automatically supplied, as determined by pressure control, to the vapor space in the volume control tank, which is predominantly hydrogen and water vapor. The hydrogen

within this tank is, in turn, the supply source to the reactor coolant. Fission gases are periodically removed from the system by venting the volume control tank to the waste disposal system prior to a cold or refueling shutdown.

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

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

Boric acid is dissolved in hot water in the batching tank. 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 storage tank. During boric acid transfer from the batching tank when the reactor is critical the receiving storage tank is not aligned to the boric acid filter. The receiving storage tank is sampled after boric acid transfer is completed and before it is placed in service. Small quantities of boric acid solution are metered from the discharge of an operating boric acid 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 recirculation pump is provided to transfer liquid from one holdup tank to another.

Liquid effluent in the holdup tanks is processed by demineralization or as radwaste.

The deborating demineralizers can be used intermittently to remove boron from the reactor coolant near the end of the core life. When the deborating demineralizers are in operation, the letdown stream passes from the mixed-bed demineralizers 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. A booster pump is also provided in the crosstie. The pump and associated piping provide an additional capacity to provide reactor coolant system purification in a more timely manner.

9.2.2.1 Design Parameters

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

Reactor Coolant Activity Concentration

The parameters 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 9.2-5. The results of the calculations are presented in Table 9.2-4. In these calculations the defective fuel rods are assumed to be present at initial core loading and are 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. [*Note - Fuel rods containing pinholes or fine cracks.*] In 1-percent of the fuel rods is computed using the following differential equations:

For parent nuclides in the coolant:

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

for daughter nuclides in the coolant:

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

where:

- N = population of nuclide
- D = fraction of fuel rods having defective cladding
- R = purification flow, coolant system volumes per sec
- B_o = initial boron concentration, ppm
- B' = boron concentration reduction rate by feed and bleed, ppm per sec
- η = removal efficiency of purification cycle for nuclide
- λ = radioactive decay constant
- ν = escape rate coefficient for diffusion into coolant
- Subscript C refers to core
- Subscript w refers to coolant
- Subscript i refers to parent nuclide
- Subscript j refers to daughter nuclide

Tritium is produced in the reactor from ternary fission in the fuel, irradiation of boron in the burnable poison rods and irradiation of boron, lithium, and deuterium in the coolant. The deuterium contribution is less than 0.1 Ci per year and may be neglected. The parameters used in the calculation of tritium production rate are presented in Table 9.2-6.

9.2.2.2 Reactor Makeup Control

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

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 change 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 to achieve cold shutdown.

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

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

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

3. The refueling water storage tank, which supplies borated water for emergency makeup.

4. 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 from 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 placing the Makeup Control switch to “STOP”.

One primary water makeup pump 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. Part 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 of reactor coolant are required for removal of impurities and adjustment of boric acid in the reactor coolant.

9.2.2.2.1 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 Makeup Mode Selector switch and makeup stop valves are set in the "AUTO" position and the Makeup Control switch in the "START" position. At a preset low-level in the volume control tank, the automatic makeup control action is initiated as follows:

- Starts both primary water makeup pumps (if not already running)
- Starts both boric acid transfer pumps (if not already running)
- Opens the concentrated boric acid control valve (FCV-110A)
- Opens the boric acid blender to charging pumps discharge control valve (FCV-110B)
- Opens the primary water makeup control valve (FCV-111A)

The flow controllers then blend the makeup stream according to the preset concentration. Makeup addition to the charging pump suction header causes the water level in the volume control tank to rise. At a preset high-level in the volume control tank, the automatic makeup control action is ceased.

If the level in the volume control tank continues to decrease to a preset low-low level, the volume control tank outlet is isolated and the refueling water storage tank is aligned for RCS makeup as follows:

- Opens the RWST makeup to charging pumps suction stop valve (LCV-112B)
- Closes the volume control tank level control valve (LCV-112C)

9.2.2.2.2 Dilution

The dilution mode of operation permits the addition of a preselected quantity of primary water makeup at a preselected flow rate to the reactor coolant system. To prepare for dilution, the operator sets the Makeup Mode Selector switch to "DILUTE", the primary water makeup flow controller setpoint to the desired flow rate, and the primary water makeup batch integrator to the desired quantity. Placing the Makeup Control switch to “START” initiates the dilution control action as follows:

- Starts both primary water makeup pumps
- Opens the primary water makeup control valve (FCV-111A)
- Opens the boric acid blender discharge control valve (FCV-111B)

Makeup water is added to the volume control tank and then goes to the charging pump suction header. If the primary water makeup flow deviates from the preset flow rate, an alarm indicates the deviation. 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 dilution control action is ceased.

9.2.2.2.3 Boration

The boration mode of operation permits the addition of a preselected quantity of concentrated boric acid solution at a preselected flow rate to the reactor coolant system. To prepare for boration, the operator sets the Makeup Mode Selector switch to "BORATE", the concentrated boric acid flow controller setpoint to the desired flow rate, and the concentrated boric acid batch integrator to the desired quantity. Placing the Makeup Control switch to “START” initiates the boration control action as follows:

- Starts both boric acid transfer pumps
- Opens the concentrated boric acid control valve (FCV-110A)
- Opens the boric acid blender to charging pumps discharge control valve (FCV-110B)

The concentrated boric acid is added to the charging pump suction header. If the concentrated boric acid solution flow deviates from the preset flow rate, an alarm indicates the deviation. 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 boration control action is ceased.

The capability to add boron to the reactor coolant is sufficient, using the normal makeup system, so that no limitation, due to boration, is imposed on the rate for cooldown of the reactor upon shutdown.

9.2.2.2.4 Alarm Functions

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

1. Deviation of primary water makeup flow rate from the control setpoint.
2. Deviation of concentrated boric acid flow rate from the control setpoint.
3. Low-level (makeup initiation point) in the volume control tank when the reactor makeup control selector is not set for the automatic or manual makeup control mode.
4. Low-Low Level in the Volume Control Tank.

Concentrated boric acid is injected into the reactor coolant system by means of the charging pumps, which take suction from the boric acid storage tanks via the boric acid transfer pumps. The refueling water storage tank is also available to the charging pumps for injection of 2400 ppm borated water. Each operation is considered in turn:

1. Concentrated boric acid can be delivered to the suction of the charging pumps using the following paths; flow and tank level indications are available to the operator as needed for these operations (refer to Plant Drawing 9321-2736 [Formerly UFSAR Figure 9.2-1, Sheet 1)]:

   a. Through the blender and valve FCV-110B; for this operation the operator has flow indication available.

   b. Through path with manual valve 293; for this operation the operator has flow indication available.

c. In the event that neither flow paths (a) nor (b) are available, the operator would use the emergency boration path through valve MOV-333.

d. Refueling water storage tank is available to the charging pumps by closing LCV-112C and opening LCV-112B.

2. The charging pumps can deliver boric acid into the reactor coolant system via the following paths:

a. Normal charging line via flow meter FT-128.

b. Seal water supply line to the reactor pumps while bypassing the seal injection filters. If this path is used, flow indicators are available.

Facilities are provided to enable primary coolant samples to be taken from the following points:

- Pressurizer steam space
- Pressurizer liquid space
- Loop 1 hot leg, reactor coolant system
- Loop 3 hot leg, reactor coolant system
- Upstream of demineralizers (chemical and volume control system)
- Downstream of demineralizers (chemical and volume control system)

The Technical Specifications for the plant require a boron content analysis to support shutdown margin determination. Samples would normally be taken from either the loop 1 hot leg or the loop 3 hot leg for routine analysis; the sample will be analyzed for boron concentration. It is important to note, however, that the main indicator to the operator during power operation as to the requirement for boration or dilution is control rod position (see Section 14.1.5).

During startup and refueling, the main indicator to the operator of abnormal conditions is the nuclear instrumentation system source range detectors. Abnormal dilution conditions are discussed in Section 14.1.5. As for power operation, it is considered that frequent boron analysis of the primary coolant is not essential for safe operation.

For a cold shutdown, the operator borates the system prior to the start of cooldown. Boration is indicated by the flow indicators in the boric acid transfer pump discharge line. The prime indicator that sufficient boron has been added to the system is inventory from the boric acid storage tanks and reactor coolant system sample analysis.

9.2.2.3 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 charging pump is expected to be operating and the speed is modulated in accordance with pressurizer level. During load changes the pressurizer level setpoint is varied automatically to compensate partially for the expansion or contraction of the reactor coolant associated with the T_{avg} changes. T_{avg} compensates for power changes by varying the pressurizer level setpoints in conjunction with pressurizer level for charging pump control. The level setpoints are varied depending on the power level.

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 is unable to maintain the required charging rate, then a pressurizer low level alarm actuates and a second charging pump may be 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.

9.2.2.4 Components

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

9.2.2.4.1 Regenerative Heat Exchanger (containment elevation 46-ft)

The regenerative heat exchanger is designed to recover heat from the letdown flow by reheating the charging flow, to eliminate reactivity effects due to insertion of cold water, and to reduce thermal shock on the charging line penetrations to the reactor coolant loop piping.

The coolant enters the shell side of the regenerative heat exchanger (U-tube multiple pass heat exchanger) where its temperature is reduced by transferring heat to the charging flow. In order to prevent flashing, this temperature should never be allowed to exceed the saturation temperature of the letdown steam at the pressure prevailing downstream of the letdown orifices. A resistance temperature detector on the outlet of the heat exchanger provides temperature indication in the control room and a high- temperature alarm.

The unit is made of austenitic stainless steel and is of all-welded construction. The exchanger is designed to withstand 2000 step changes in shell-side fluid temperature from 100^{o}F to 560^{o}F during the design life of the unit.

9.2.2.4.2 First Stage Letdown Orifices and Control Valves (containment elevation 46-ft)

Three letdown orifices are provided to admit a predetermined coolant flow to the letdown stream and reduce to letdown pressure. They consist of two 75-gpm and one 45-gpm orifices. Normally a 75-gpm orifice is in service. The 45-gpm orifice combined with the 75-gpm orifice results in a letdown flow of 120 gpm, which is a maximum for the chemical and volume control system (greater flow will result in channeling and hence inefficient operation in the demineralizers). The second 75-gpm orifice allows for even less flow restriction for letdown during operations when the system pressure is low (excess letdown and residual heat removal connections can also be used to maintain flow during times of low system pressure). This last orifice also provides a redundant backup for the first 75-gpm orifice. The selected orifice is placed in service from the control room by remote operation of its respective letdown flow control valve. These lights and switches are located on the flight panel in the central control room. An additional switch for the three valves labeled "close-remote," is provided on the containment isolation supervisory panel in the central control room. The three letdown flow control valves will close automatically on a phase A containment isolation signal. The switch on the containment isolation panel will allow the closing of these valves manually if required. Valve position is indicated on the isolation panel by dual-colored windows. Valve position for normal plant operations and valve position for containment isolation are provided.

Orifice selection is controlled from the central control room or primary auxiliary building. Indications are also provided at this location when the valves are open. The primary auxiliary building switches will be used to control the rate of letdown when the control room is not available. The letdown inlet stop valve to the regenerative heat exchanger may also be controlled locally in the primary auxiliary building. When these switches are in use, either in the close or open position, all control from the central control room will be lost. This will be indicated in the central control room by the actuation of a category alarm (control transferred to local) and the loss of all indicating lights associated with these valves.

9.2.2.4.3 Letdown Relief Valve

Relief valve No. 203 is provided to protect the piping downstream of the letdown flow control valves and up to the low-pressure letdown valve, PCV-135. Thus, this piping will be protected in the event that the letdown flow control valves fail in the open position allowing pressure to increase up to system pressure. The relief valve is set at 600 psig or below and discharges to the pressurizer relief tank. An orifice installed just up-stream of the relief valve provides sufficient differential pressure to prevent over-cycling of the valve. A resistance temperature detector is provided on the relief valve discharge piping. This temperature is indicated in the control room and a high-temperature alarm will indicate that the relief valve is leaking or has lifted.

9.2.2.4.4 Non-regenerative (letdown) Heat Exchanger (primary auxiliary building elevation 98-ft)

The letdown stream enters the tube side of the non-regenerative heat exchanger. In passing through these U-tubes, it is cooled by the multipass flow of component cooling water in the shell side of the heat exchanger.

Temperature and pressure control of the letdown flow is accomplished automatically by means of sensors located downstream of the heat exchanger. All surfaces in contact with the reactor coolant are austenitic stainless steel, and the shell is carbon steel.

9.2.2.4.5 Demineralizers (primary auxiliary building elevation 59-ft)

All the demineralizers in this system have similar piping connections. Water enters the demineralizer at the top and flows past an impingement baffle, which prevents channels being cut in the resin bed by the water stream. After passing through the resin, the water flows through a screen and exits through the water outlet connection. This screen prevents loss of resin through the water outlet connection. In order to allow resin replacement, a resin fill and a resin discharge connection are provided on the top and bottom of the demineralizer, respectively. A vent connection, located on the top of the demineralizer, is used during resin replacement and/or regeneration and backwashing operations. The vent screen will prevent loss of resin through this connection.

Sampling connections are provided on the common inlet line to the demineralizers and on the common outlet line from the demineralizers to check on the performance of the demineralizers. When impurities begin to leak through the resin bed the demineralizer is considered exhausted. At this point it is necessary to replace or regenerate the spent resin. Regeneration will be done only with anion demineralizers; the resin beds of the cation and mixed-bed units will be replaced when depleted. (The deborating and boric acid evaporator condensate demineralizer are the only anion beds in the system.) Current procedures use only the resin replacement option.

9.2.2.4.5.1 Mixed-Bed Demineralizers

Two flushable mixed-bed demineralizers maintain reactor coolant purity by removing fission and corrosion products. The resin bed is designed to reduce the concentration of ionic isotopes in the purification stream, except for cesium, tritium, and molybdenum, by a minimum factor of 10.

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

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

9.2.2.4.5.2 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, which builds up in the coolant from the $B^{10}(n,\alpha)\ Li^{7}$ reaction. The demineralizer would be used intermittently to control cesium.

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

9.2.2.4.5.3 Chemical Control Demineralizers

There are two anion demineralizers located downstream of the cation bed demineralizer, which can be used to remove boric acid from the reactor coolant system fluid. The anion deborating demineralizers are primarily used to remove boron from the reactor coolant system near the end of a core cycle, but can be used at any time.

Each anion deborating demineralizer is sized to remove the quantity of boric acid that must be removed from the reactor coolant system to maintain full power operation near the end of core life should the holdup tanks be full.

With a change in resin, either one of the two anion demineralizers could be reconfigured as a cation bed lithium control demineralizer. Either one would then be capable of removing lithium from the Reactor Coolant System, as does the normal cation bed demineralizer.

Facilities are provided for regeneration. When regeneration is no longer feasible, the resin is flushed to the spent resin storage tank.

9.2.2.4.6 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 is made of austenitic stainless steel. An additional valve at the resin fill tank is installed to reduce the need for personnel to go to the ion exchange gallery each time new resin is added, thereby reducing radiation exposure.

9.2.2.4.7 Reactor Coolant Filter (Primary Auxiliary Building elevation 98-ft)

The reactor coolant filter will remove any resin fines or particulates larger than 25 microns. A range of smaller filter micron sizes are used in accordance with industry practice to reduce reactor coolant radiation activity and, consequently, reduce personnel exposure.

When the local pressure indicators before and after the filter indicate excessive pressure drop or when the filter develops high radiation fields, the disposable filter element will be replaced. Vent and drain connections are provided for the replacement operation. The vessel is made of austenitic stainless steel.

9.2.2.4.8 Volume Control Tank

The volume control tank collects the reactor coolant surge 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. A cover of hydrogen gas is maintained in the volume control tank to control the hydrogen concentration in the reactor coolant system.

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, which 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. A bypass line, with hand-operated valve, is installed to enable water to be pumped from the holdup tanks to the volume control tank to allow faster filling of the primary system following a shutdown.

9.2.2.4.9 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.

Each pump is 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 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 charging flows. During normal operation, 8 gpm seal injection enters each reactor coolant pump in the thermal barrier region where the flow splits, with 3 gpm flowing upward through the controlled leakage seal package and returning to the chemical and volume control system. The remaining 5 gpm passes through the thermal barrier heat exchanger and into the reactor coolant system where it constitutes a portion of the reactor coolant system water makeup. In the event that normal seal cooling is lost, the component cooling water system provides adequate seal cooling by supplying flow to the thermal barrier heat exchanger.

Seal injection flow is indicated locally and in the central control room.

An alternate power supply is provided for one of the charging pumps from the 13.8-kV normal offsite power through Unit 1 switchgear. If normal offsite power is not available, this pump can be energized using the SBO / Appendix R diesel.

Any one of the three charging pumps can be used to hydrotest the reactor coolant system.

A low-pressure tank (dampener) is installed in the suction line, and a high-pressure tank is installed in the discharge line on each charging pump in order to eliminate pulsation that could potentially cause cavitation at the charging pump suction or root weld cracks on the discharge piping.

9.2.2.4.10 Chemical Mixing Tank

The primary use of the stainless steel chemical mixing tank is to prepare caustic solutions for pH control and hydrazine for oxygen scavenging. The capacity of the chemical mixing tank is more than sufficient to prepare a solution of pH control chemical for the reactor coolant system.

9.2.2.4.11 Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools reactor coolant letdown flow if letdown through the normal letdown path is blocked. The letdown stream flows through the tube side and component cooling water is circulated through the shell side. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel. All tube joints are welded. The unit is designed to withstand 2000 step changes in the tube fluid temperature from 80^{o}F to the cold-leg temperature.

9.2.2.4.12 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. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

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

9.2.2.4.13 Seal-Water Filter

The filter collects particulates larger than 25 μ 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.

9.2.2.4.14 Seal-Water Injection Filters

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

9.2.2.4.15 Boric Acid Filter

The boric acid filter collects particulates larger than 25 μ 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.

9.2.2.4.16 Boric Acid Storage Tanks

The boric acid storage tanks are sized to store sufficient boric acid solution for refueling and enough boric acid solution for a cold shutdown shortly after full power operation is achieved. In addition, sufficient boric acid solution is available for cold shutdown if the most reactive rod cluster control is not inserted. The requirements for the volume of boric acid in the tanks are contained in the Technical Requirements Manual.

The concentration of boric acid solution in storage is maintained within Technical Requirements Manual limits. Periodic manual sampling and corrective actions are taken, 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. A 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.

Each tank is provided with a low-level alarm. It is, however, optional whether the operator chooses to operate normally above the low-level alarm in both tanks. Each tank is instrumented for level indication. Indication of level is provided locally and on supervisory panel "SF" in the control room. The low-level condition is audibly annunciated in the control room.

9.2.2.4.17 Boric Acid Storage Tank Heaters

Each boric acid tank has two 100-percent capacity electric heaters, which are connected in parallel and controlled from a single controller and a single temperature sensing controller and a single temperature sensing device and are powered by a single source. The heaters maintain the boric acid solution temperature above the minimum required by the Technical Requirements Manual.

9.2.2.4.18 Batching Tank

The batching tank is used to provide makeup to the boric acid storage tanks. The tank manway is provided with a removable screen to prevent entry of foreign particles. In addition, the tank is provided with an agitator to improve mixing during batching operations. The tank is constructed of austenitic stainless steel. The tank is provided with a steam jacket for heating the boric acid solution. Original plant design also used the tank for sodium hydroxide addition for postaccident pH control inside containment. Current plant design uses sodium tetraborate for pH control. The sodium tetraborate is stored in baskets inside containment as described in Section 6.3.2.2.12.

9.2.2.4.19 Boric Acid Transfer Pumps

Two 100-percent capacity 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 storage tanks and inject boric acid into the charging pump suction header.

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. The design capacity of each pump is equal to the normal letdown flow rate. The design head is sufficient, considering line and valve losses, to deliver rated flow to the charging pump suction header when volume control tank pressure is at the maximum operating value (relief valve setting). All parts in contact with the solutions are austenitic stainless steel or other adequate corrosion-resistant material.

The transfer pumps are operated either automatically or manually from the main control room or from a local control center. The reactor makeup control operates one of the pumps automatically when boric acid solution is required for makeup or boration.

9.2.2.4.20 Boric Acid Blender

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

9.2.2.4.21 Valves

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

Isolation valves are provided at all connections to the reactor coolant system. Lines entering the reactor containment also have check valves inside the containment to prevent reverse flow from the containment. Relief valves are provided for lines and components that might be pressurized above design pressure by improper operation or component malfunction. Pressure relief for the tube side of the regenerative heat exchanger is provided by the auxiliary spray line lift check valve, which is designed to open when pressure under the seat exceeds reactor coolant pressure by 200 psi.

9.2.2.4.22 Piping

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

9.2.2.4.23 Electrical Heat Tracing

Piping containing concentrated boric acid is provided with double circuit (one circuit redundant) electrical heat tracing in conjunction with insulation to maintain the concentrated solution above the precipitation temperature.
Alarms are provided.

Exceptions are as follows:

1. Lines, which 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 storage tanks, which are provided with immersion heaters.
3. The batching tank, which is provided with a steam jacket.
4. The concentrates holding tank, which is provided with an immersion heater.
5. The boric acid transfer pumps, which are provided with strip heaters in enclosures.

Emergency power is supplied to the heat tracing circuits and electric heaters on loss of offsite power.

Each individual pipe tracing circuit has a local control cabinet containing operating, testing, and alarm devices.

Failure of the operating circuit will result in a decrease in pipe temperature and will alarm in the control room. Test and connection of the redundant circuit can be readily accomplished. Likewise, failure of any operating device in the local control cabinet will result in alarm.

9.2.2.5 Recycle Process

Boron is no longer recycled, consequently some components of the boron recycle system are no longer used, some have been removed (evaporator 22, gas stripper 22 and ion exchanger filter 22) and some (Boron Monitoring Tanks and pumps) have been retired and their inlet and outlet piping cut and capped. Heaters, pumps and level and temperature instrumentation associated with the monitor tanks have been disconnected. Reactor coolant system effluents collected in the holdup tanks are either processed through demineralizers or sent to the radwaste system for processing. The originally supplied gas stripper feed pumps have been replaced by holdup tank transfer pumps, which are used to transfer water to waste collection tanks in Unit 1.

9.2.2.5.1 Purpose

The original purpose of the recycle portion of the chemical and volume control system is to accept and process all effluents, which could be readily reused as makeup to the reactor coolant system. Boron is no longer recycled, but portions of the boron recycle system are used to collect effluents and transfer them to the waste disposal system. Effluents are initially collected in the chemical and volume control system holdup tanks. Prior to the holdup tanks, particularly if the reactor is operating with defective fuel, the letdown from the reactor coolant system is passed through the mixed-bed demineralizers. Both forms of resin remove fission products and corrosion products. As fluid enters the holdup tanks, released gases (hydrogen

and fission gases) mix with the nitrogen cover gas and are eventually drawn off to the waste gas system.

Three CVCS holdup tank transfer pumps take suction from the holdup tanks and pump the fluid through the evaporator feed ion exchangers where lithium and fission products (primarily cesium isotopes) are removed. The resin is a hydrogen form cation resin. Two ion exchangers are employed in series. Series operation is recommended to ensure prevention of breakthrough of cesium in the event of evaporation with 1-percent fuel defects. From the feed ion exchangers, the fluid is returned to the holdup tanks. The CVCS holdup tank transfer pumps are also used to transfer the holdup tank contents to the waste disposal system.

A holdup tank low pressure interlock will trip the CVCS holdup tank transfer pumps upon low pressure in the holdup tank. This interlock reduces the potential for creating a negative pressure condition in the holdup tanks during drain down of the tank.

During operation of the recycle process, samples can be taken at various positions through the system to assess the performance of the individual system components. Local samples may be obtained before and after the evaporator feed ion exchangers.

9.2.2.5.2 Holdup Tanks

Three holdup tanks contain radioactive liquid, which enters the tank 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 while another tank is being filled. The third tank is normally kept empty to provide additional storage capacity when needed.

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

9.2.2.5.3 Holdup Tank Recirculation Pump

The recirculation pump is used to mix the contents of a holdup tank and to transfer the contents of a holdup tank to another.

A holdup tank low pressure interlock will trip the holdup tank recirculation pump upon low pressure in the holdup tank. This interlock reduces the potential for creating a negative pressure condition.

The wetted surface of this pump is constructed of austenitic stainless steel.

9.2.2.5.4 Holdup Tank Transfer Pump

The three holdup tank transfer pumps originally supplied feed to the gas stripper boric acid evaporator trains from a holdup tank. They now are used to transfer water to waste collection tanks in unit 1. These centrifugal pumps are constructed of austenitic stainless steel.

9.2.2.5.5 Evaporator Feed (Cation) Ion Exchangers

Four cation flushable demineralizers remove cations (primarily cesium and lithium) from the holdup tank effluent. The demineralizer vessels are constructed of austenitic stainless steel and contain a resin retention screen.

9.2.2.5.6 Ion Exchanger Filters

These filters were originally provided to collect resin fines and particulates larger than 25 microns from the cation ion exchanger. They are no longer used. Filter 21 has been retired in place and filter 22 has been removed.

9.2.2.5.7 Gas Stripper Equipment

Two gas strippers were originally provided to remove nitrogen, hydrogen, and fission gases from the evaporator feed. They are no longer used. Gas stripper 21 has been retired in place and gas stripper 22 has been removed.

9.2.2.5.8 Boric Acid Evaporator Equipment

Two boric acid evaporators were originally provided to concentrate boric acid for reuse in the reactor coolant system. They are no longer used. Evaporator 21 has been retired in place and evaporator 22 has been removed.

9.2.2.5.9 Evaporator Condensate Demineralizers

Two anion demineralizers were originally provided to remove any boric acid contained in the evaporator condensate. These demineralizers are valved out of service and no longer used.

9.2.2.5.10 Condensate Filters

The filters were originally provided to collect resin fines and particulates larger than 25 microns from the boric acid evaporator condensate streams. These filters are no longer used.

9.2.2.5.11 Monitor Tanks

The monitor tanks have been retired in place.

9.2.2.5.12 Monitor Tank Pumps

The monitor tank pumps have been retired in place.

9.2.2.5.13 Primary Water Storage Tank

A single 165,000-gal primary water storage tank is provided to store the demineralized water used by the primary water makeup system shown in Plant Drawing9321-2724 [Formerly UFSAR Figure 9.2-2]. The storage tank is constructed of type 304 stainless steel.

Chemical addition to the tank, if required, can be accomplished via a 3-in. blind flange connection located near the top of the tank, directly off the pressure-vacuum relief valve. This connection can be used to correct the reactor coolant system water chemistry. A local sample

point is provided on the bottom of the tank in addition to a tank drain and a loop seal overflow. This loop seal will prevent the entrance of air. To ensure that this loop seal is filled with water a valved line is provided from the tank drain to the loop seal.

Besides these lines into the primary water storage tank, there are also two feeds. One comes from the monitor tank pumps, which have been retired in place, and the second comes from the primary water makeup pump recirculation. Lines carrying heating steam to and from the tank also enter it near its bottom. All of these connections and lines entering the tank are heat traced to prevent them from freezing. A large inspection port is provided on the side of the tank.

9.2.2.5.13.1 Primary Water Storage Tank Level Measurement

Level in the tank is measured and indicated locally and in the central control room. In addition, high level and low level are alarmed in the central control room.

9.2.2.5.13.2 Primary Water Storage Tank Temperature Control

Temperature in the tank is indicated locally. An additional temperature measurement is made at the tank, on the suction line to the makeup pumps.

The temperature element will sense a representative fluid temperature. This temperature measurement is used to control steam flow to the coils located at the bottom of the storage tank. The steam coils will maintain the water in the storage tank at a sufficiently high temperature to prevent freezing of the tank contents and large temperature changes in the primary water supplied to the shaft seals of the reactor coolant pumps by means of the blender. The walls of the tank are insulated and all lines connected to the tank and exposed to the environment are electrically heat traced to prevent freezing.

In addition, the external instrument cabinet is heated and weatherproofed to help ensure a controlled temperature for the tank level instrumentation. Low temperature alarms alert the operator of any instrument heat trace failure or low temperatures in the instrument enclosure.

9.2.2.5.14 Primary Water Makeup Pumps

Two primary water makeup pumps are provided and normally take their suction from the primary water storage tank. The pumps are constructed of type 316 austenitic stainless steel. Each can supply 150 gpm of water at a total dynamic head of 210-ft.

Control of both pumps is provided from the central control room. No local control of the pump is provided.

Normally one pump will be selected to run continuously; the second will be in auto. A limited flow recirculation line is provided and remains open in case makeup water is not required at a given time anywhere in the plant. An orifice in this line limits the recirculation flow.

In addition to manual operation, these pumps are also automatically controlled by the chemical and volume control system. In the event that automatic makeup to or dilution of the reactor coolant system is required, the makeup control system will send a start signal to both primary water pumps. The pump in operation will continue to run and the second pump, if in auto, will start. When this automatic start signal is removed, the pumps will return to their original operating condition. When makeup is required, the water follows the path to the boric acid

blender. In the event the pressure in the supply line to the blender falls, indicating insufficient water supply, an alarm will be annunciated in the central control room. Each pump is also provided with a discharge pressure gauge. Operation of the pumps without a suction head is prevented.

9.2.2.5.15 Concentrates Filter

A disposable synthetic cartridge-type filter was provided in the original design to remove particulates larger than 25 microns from the evaporator concentrates. This filter is no longer used and has been retired in place.

9.2.2.5.16 Concentrates Holding Tank

The concentrates holding tank was provided in the original design to hold the production of concentrates from one batch of boric acid evaporator operation. The tank is no longer used and has been retired in place.

9.2.2.5.17 Concentrates Holding Tank Transfer Pumps

Two holding tank transfer pumps were provided in the original design to discharge boric acid solution from the concentrates holding tank to the boric acid storage tanks. These pumps are no longer used and have been retired in place.

9.2.3 System Design Evaluation

9.2.3.1 Availability and Reliability

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

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

The electrical equipment of the chemical and volume control system is arranged so that multiple items receive their power form various 480-V buses (see Chapter 8). Each of the three charging pumps is powered from a separate 480-V bus. The two boric acid transfer pumps are also powered from separate 480-V buses. One charging pump and one boric acid transfer pump are capable of meeting cold shutdown requirements shortly after full power operation. In cases of loss of offsite power, a charging pump and a boric acid transfer pump can be placed on the emergency diesels, if necessary.

9.2.3.2 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 form of water. Therefore, any leakage of coolant to the containment atmosphere carries tritium in the same proportion as it exists in the coolant. Thus, the level of tritium in the containment atmosphere, when it is sealed from outside air ventilation, is a function of tritium level in the reactor coolant, the cooling water temperature at the cooling coils, which determines the

dewpoint temperature of the air, and the presence of leakage other than reactor coolant as a source of moisture in the containment air.

There are two major considerations with regard to the presence of tritium:

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.

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

Neither of these considerations is limiting in this plant.

The concentration of tritium in the reactor coolant is maintained at a level, which precludes personnel hazard during access to the containment. This is achieved by diverting the letdown flow to the Chemical and Volume Control System for processing via the Waste Disposal System.

The Annual Effluent and Waste Disposal Report shows that tritium released to the environment in this manner is well below 10 CFR 20 limits and thus no public hazard would result.

9.2.3.3 Leakage Prevention

Quality control of the material and the installation of the chemical and volume control 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, with the exception of the control valves discussed below, which are larger than 2-in. and which are designated for radioactive service at an operating fluid temperature above 212°F, are provided with a stuffing box and capped lantern leakoff connections. Leakage to the atmosphere is essentially zero for these valves. All control valves are either provided with a stuffing box and leakoff connections, a standard stuffing box suitable for the specified service, or are totally enclosed. Leakage to the atmosphere is essentially zero for these valves.

Diaphragm valves are provided where the operating pressure and the operating temperature permit the use of these valves. Leakage to the atmosphere is essentially zero for these valves.

9.2.3.4 Incident Control

The letdown line and the reactor coolant pumps seal water return 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, which 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, which 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 at least one check valve inside the reactor containment to provide isolation of the reactor containment should a break occur in these lines outside the reactor containment.

9.2.3.5 Malfunction Analysis

To evaluate system safety, failures or malfunctions were assumed concurrent with a loss-of-coolant and the consequences analyzed and presented in Table 9.2-7. As a result of this evaluation, it is concluded that proper consideration has been given to safety in the design of the system.

If a rupture were to take place between the reactor coolant loop and the first isolation valve or check valve, this incident would lead to an uncontrolled loss of reactor coolant. The analysis of loss-of-coolant accidents is discussed in Section 14.3.

Should a rupture occur in the chemical and volume control system outside the containment, or at any point beyond the first check valve or remotely operated isolation valve, actuation of the valve would limit the release of coolant and ensure continued functioning of the normal means of heat dissipation from the core. For the general case of rupture outside the containment, the largest source of radioactive fluid subject to release is the contents of the volume control tank. The consequences of such a release are considered in Section 14.2.

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 the nuclear instrumentation source range detectors, counters and count rate indicators. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to give ample time to start a corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical. The maximum dilution rate is based on the abnormal condition of three 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 value and the water volume in the system is at a minimum. This analysis is referred to as the Boron Dilution Event analysis and is discussed in Section 14.1.5.

At least two separate and independent flow paths are available for reactor coolant boration, i.e., either 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 a backup to the boration system, the operator can align the refueling water storage tank outlet to the suction of the charging pumps. A third method involves depressurization of the primary system, if necessary, and the use of the safety injection pumps.

On loss of seal injection water to the reactor coolant pump seals, seal water flow may be reestablished by manually starting a standby charging pump. Even if the seal water injection flow is not reestablished, the plant can be operated since the thermal barrier cooler has

sufficient capacity to cool the reactor coolant flow, which pass through the thermal barrier cooler and seal leakoff from the pump volute.

9.2.3.6 Galvanic Corrosion

The only types of materials, which are in contact with each other in borated water are stainless steels, Inconel, Stellite valve materials, and zircaloy fuel element cladding. These materials exhibit only and insignificant degree of galvanic corrosion when coupled to each other. As can be seen from tests, the effects of galvanic corrosion are insignificant to systems containing borated water.

Boration during normal operation to compensate for power changes will be indicated to the operator from two sources: (1) the control rod movement, and (2) 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 will indicate 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 storage tank level is another indication of boric acid injection.

9.2.4 Minimum Operating Conditions

Minimum operating conditions are specified in the Technical Requirements Manual.

9.2.5 Tests and Inspections

The minimum frequencies for testing, calibrating and/or checking instrument channels for the chemical and volume control system are specified in the Technical Requirements Manual.

TABLE 9.2-1
Chemical and Volume Control System Code Requirements

Component	Code
Regenerative heat exchanger	ASME III,[1] 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
[Deleted]	
Seal water injection filter (alternate seal injection path)	ASME III Class 2
Seal water filter	ASME III, Class C
Holdup tanks	ASME III, Class C
Boric acid filter	ASME III, Class C
Gas stripper package (Note 3)	ASME III, Class C
Boric acid evaporator package (Note 3)	ASME III, Class C
Evaporator condensate demineralizers (Note 4)	ASME III, Class C
Concentrates filter (Note 4)	ASME III, Class C
Evaporator feed (Cation) ion exchanger	ASME III, Class C
Ion exchanger filter (Note 3)	ASME III, Class C
Condensate filter (Note 4)	ASME III, Class C
Piping and valves	USAS B31.1[2]

Notes:
1. ASME III – American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.
2. USAS B31.1 – Code for Pressure Piping, and special nuclear cases where applicable.
3. Unit 21 is no longer used, Unit 22 has been physically removed.
4. No longer used.

TABLE 9.2-2
Chemical and Volume Control System Letdown Requirements[1]

[Deleted]

Normal seal water supply flow rate, gpm	32
Normal 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), gpm	87
Normal seal injection flow to reactor coolant pumps, gpm	32
Normal charging line flow, gpm	55

Notes:
1. Volumetric flow rates in gpm are based on 127°F and 15 psig.

TABLE 9.2-3 (Sheet 1 of 2)
Chemical and Volume Control System Principal Component Design Data Summary

	Quantity	Heat Transfer, Btu/hr	Design Letdown Flow, 1b/hr	Letdown, ΔT °F	Design Pressure, psig, Shell/Tube	Design Temperature, °F, Shell/Tube
Heat exchangers						
Regenerative	1	10.28×10^6	37,050	257	2,485/2,735	650/650
Non-regenerative (Letdown)	1	14.8×10^6	59,700	253	150/600	250/400
Seal water	1	2.17×10^6	126,756	17	150/150	250/250
Excess letdown	1	4.75×10^6	12,400	360	150/2,485	250/650

	Quantity	Type	Capacity, gpm	Head, ft or psi	Design Pressure, psig	Design Temperature, °F
Pumps						
Charging	3	Pos. Displ.	98	2,500 psi	3,200	250
Boric acid Transfer	2	Centrifugal	75	235-ft	150	250
Holdup tank recirculation	1	Centrifugal	500	100-ft	75	200
Primary water makeup	2	Centrifugal	150	210-ft	150	Ambient
Monitor tank (Retired in place)						
Concentrates holding tank transfer (Retired in place)						
Holdup Tank Transfer Pump 22	1	Centrifugal	25	63-ft	150	200
Holdup Tank Transfer Pump 21 & 23	2	Centrifugal	25	63-ft	150	200

TABLE 9.2-3 (Sheet 2 of 2)
Chemical and Volume Control System Principal Component Design Data Summary

	Quantity	Type	Volume	Design pressure, psig	Design Temperature, °F
Tanks					
Volume control	1	Vertical	400-ft^3 [1]	75/15	250
Charging pump	3	Vertical	-	75	250
Stabilizer separator	3	Spherical	-	2735	250
Pulsation dampener					
Boric acid	2	Vertical	7,000 gal [1]	atmos.	250
Chemical mixing	1	Vertical	5.0 gal [1]	150	200
Batching	1	Jacket Btm.	400 gal [1]	atmos.	250
Holdup	3	Horizontal	8106-ft^3 [1]	15	200
Primary water storage	1	Vertical	165,000 gal	atmos.	150
Concentrates holding (Retired in Place)					
Monitor (Retired in Place)					
Resin fill	1	Open	8-ft^3 [1]	-	200

	Quantity	Type	Resin Volume, ft3	Flow, gpm	Design Pressure, psig	Design Temperature, °F
Demineralizers						
Mixed-bed	2	Flushable	30	120	200	250
Cation bed	1	Flushable	12.0	42	200	250
Evaporator feed	4	Flushable	12.0	12.5	200	250
Evaporator condensate	2	Flushable	12.0	12.5	200	250
Deborating	2	Flushable	30	120	200	250

Notes:

1. Net Internal Volume

TABLE 9.2-4
Reactor Coolant System Activities
(576°F)

Activation Products	uCi/g
Mn-54	1.60E-03
Cr-51	5.50E-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		uCi/g
Br-83	9.90E-02	Rb-86	4.55E-02	Tc-99m	7.62E-01	Ba-137m	2.48E+00
Br-84	4.86E-02	Rb-88	4.36E+00	Ru-103	6.42E-04	Ba-140	4.36E-03
Br-85	5.67E-03	Rb-89	2.00E-01	Rh-103m	6.38E-04	La-140	1.46E-03
I-127 (a)	1.53E-10	Sr-89	4.37E-03	Ru-106	3.30E-04	Ce-141	6.56E-04
I-129	8.48E-08	Sr-90	2.85E-04	Rh-106	3.30E-04	Ce-143	5.24E-04
I-130	7.08E-02	Sr-91	5.78E-03	Ag-110m	4.89E-03	Pr-143	6.37E-04
I-131	2.90E+00	Sr-92	1.28E-03	Te-125m	1.15E-03	Ce-144	4.92E-04
I-132	3.02E+00	Y-90	8.09E-05	Te-127m	3.83E-03	Pr-144	4.92E-04
I-133	4.65E+00	Y-91m	3.12E-03	Te-127	1.57E-02		
I-134	6.52E-01	Y-91	5.77E-04	Te-129m	1.16E-02		
I-135	2.57E+00	Y-92	1.12E-03	Te-129	1.50E-02		
Cs-134	5.14E+00	Y-93	3.86E-04	Te-131m	2.63E-02		
Cs-136	5.35E+00	Zr-95	6.55E-04	Te-131	1.42E-02		
Cs-137	2.62E+00	Nb-95	6.56E-04	Te-132	3.14E-01		
Cs-138	1.06E+00	Mo-99	8.22E-01	Te-134	3.13E-02		

Gaseous Fission Products

	uCi/g
Kr-83m	4.67E-01
Kr-85m	1.85E+00
Kr-85	1.36E+01
Kr-87	1.22E+00
Kr-88	3.49E+00
Kr-89	9.90E-02
Xe-131m	3.18E+00
Xe-133m	3.61E+00
Xe-133	2.57E+02
Xe-135m	5.55E-01
Xe-135	8.94E+00
Xe-137	1.93E-01
Xe-138	6.94E-01

TABLE 9.2-5
Parameters Used in the Calculation of Reactor Coolant
Fission Product Activities

1.	Core thermal power, MWt	3280.3
2.	Fraction of fuel containing clad defects	0.01
3.	Reactor coolant liquid volume, ft^3	10,620
4.	Reactor coolant average temperature, °F	573
5.	Purification flow rate (normal), gpm	75
6.	Effective cation demineralizer flow, gpm	7
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}	2.0×10^{-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	7.9×10^{-1}
Kr-85	7.5×10^{-5}
Kr-85m	6.1×10^{-1}
Kr-87	8.5×10^{-1}
Kr-88	7.1×10^{-1}
Kr-89	9.9×10^{-1}
Xe-131m	1.7×10^{-2}
Xe-133	3.9×10^{-2}
Xe-133m	8.8×10^{-2}
Xe-135	3.6×10^{-1}
Xe-135m	9.5×10^{-1}
Xe-137	9.9×10^{-1}
Xe-138	9.6×10^{-1}

TABLE 9.2-6 (Sheet 1 of 2)
Tritium Production in the Reactor Coolant

BASIC ASSUMPTIONS

Plant Parameters:

1. Core thermal power, MWt	3216
2. Coolant water volume, ft^3	12,600
3. Core volume, ft^3	1,152.5
4. Core volume fraction	
a. UO_2	0.3023
b. Zr + SS	0.1035
c. H_20	0.5942
5. Plant full power operating times	
a. Initial cycle	78 wk (18 months)
b. Equilibrium cycle	49 wk (11.3 months)
6. Boron concentrations (Peak hot full power equilibrium Xe)	
a. Initial cycle, ppm	890
b. Equilibrium cycle, ppm	825
7. Burnable poison boron content (total - all rods), kg	18.1
8. Fraction of tritium in core (ternary fission + burnable boron) diffusing through cladding	0.30_1
9. Ternary fission yield, atoms/fission	8×10^{-5}
10. Nuclear cross-sections and neutron fluxes	
B^{10} (n,2α) T σ ; mb	(nv; n/cm^2 - sec)
1 MeV $\leq$ E $\leq$ 5 MeV = 31.95 (Spectrum weighted)	5.04×10^{13}
E > 5 MeV = 75	7.4×10^{12}
Li^7 (n, nα) T (99.9-percent purity Li^7)	
3 MeV $\leq$ E $\leq$ 6 MeV = 39.1 (Spectrum weighted)	2.14×10^{13}
E > 6 MeV = 0.4	2.76×10^{12}
Li^6 (n, α) T (99.9-percent purity Li^7)	
σ = 675 barns;	2.14×10^{13}
11. Cooling water flow: 7.5×10^5 gpm = 15×10^{14} cm^3/yr	

TABLE 9.2-6 (sheet 2 of 2)
Tritium production in the Reactor Coolant

CALCULATIONS (per year)

	Initial Cycle	Equilibrium Cycle
A. Tritium from core (curies)		
1. Ternary fission	11,450	11,450
2. B^{10} (n, 2α) T (in poison rods)	800	NA
3. B^{10} (n, α) Li^7 (n, nα) T (in poison rods)	1,500	NA
4. Release fraction (0.30)		
5. Total release to coolant	4,125	3,440
B. Tritium from coolant (curies)		
1. B^{10} (n, 2α) T	1,130	780
2. Li^7 (n, nα) T (limit 2.2 ppm Li)	8.8	8.8
3. Li^6 (n, α) T (purity of Li^7 = 99.9-percent)	8.8	8.8
4. Release fraction (1.0)		
5. Total release to coolant	1,147.6	797.6
C. Total tritium in coolant (curies)	5,273	4,238

Notes:

1. The assumption that 30-percent of the ternary produced tritium diffuses into the coolant is based on the analysis made of fuel retention in the Saxton and the Yankee stainless-clad fuel. This analysis indicated that the fuel retained 68-percent of the tritium produced in the fuel. Although data is not currently available on zircaloy-clad fuel operating at the specific power anticipated for these reactors, it is reasonably certain that a significant portion of the tritium released by the fuel will not diffuse through the zircaloy possibly because of the formation of zirconium tritide. Shipping port data indicates that less than 1-percent of ternary tritium produced is released to the coolant. Although this data cannot be used directly, it does indicate that zircaloy will reduce tritium diffusion.

TABLE 9.2-7
Malfunction Analysis of Chemical and Volume Control System

Component	Failure	Comments and Consequences
Letdown line	Rupture in the line inside the reactor containment	The remote air-operated valve located near the main coolant loop is closed on low pressurizer level to prevent supplementary loss of coolant through the letdown line rupture. The containment isolation valves in the letdown line outside the reactor containment and also the orifice block valves are 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.
Normal and alternate charging lines	See above	The check valves located near the main coolant loops prevent supplementary loss of coolant through the rupture. The check valves located at the boundary of the reactor containment prevent any leakage of the reactor containment atmosphere outside the reactor containment.
Seal water return line	See above	The motor-operated isolation valve 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.

9.2 FIGURES

Figure No.	Title
Figure 9.2-1 Sh. 1	Chemical and Volume Control System - Flow Diagram, Sheet 1, Replaced with Plant Drawing 9321-2736
Figure 9.2-1 Sh. 2	Chemical and Volume Control System - Flow Diagram, Sheet 2, Replaced with Plant Drawing 208168
Figure 9.2-1 Sh. 3	Chemical and Volume Control System - Flow Diagram, Sheet 3, Replaced with Plant Drawing 9321-2737
Figure 9.2-1 Sh. 4	Chemical and Volume Control System - Flow Diagram, Sheet 4, Replaced with Plant Drawing 235309
Figure 9.2-2	Primary Water Makeup System - Flow Diagram, Replaced with Plant Drawing 9321-2724

9.3 AUXILIARY COOLANT SYSTEM

9.3.1 Design Basis

The auxiliary coolant system consists of three loops as shown in Plant Drawings 227781, 9321-2720, and 251783 [Formerly UFSAR Figure 9.3-1, Sheets 1, 2, and 3] the component cooling loop, the residual heat removal loop, and the spent fuel pit cooling loop.

9.3.1.1 Performance Objectives

9.3.1.1.1 Component Cooling Loop

The component cooling loop is designed to remove residual and sensible heat from the reactor coolant system via the residual heat removal loop during plant shutdown, 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. It also provides cooling for engineered safeguards and safe shutdown 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.

The loop design provides for detection of radioactivity entering the loop from reactor coolant source and also provides means for isolation.

9.3.1.1.2 Residual Heat Removal Loop

The residual heat removal loop is designed to remove residual and sensible heat from the core and reduce the temperature of the reactor coolant system during the second phase of plant cooldown. During the first phase of cool-down, the temperature of the reactor coolant system is reduced by transferring heat from the reactor coolant system to the steam and power conversion system.

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 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 to 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 protective devices and redundant isolation means.

9.3.1.1.3 Spent Fuel Pit Cooling Loop

The spent fuel pit cooling loop is designed to remove from the spent fuel pit the heat generated by stored spent fuel elements.

The loop design consists of two pumps, a heat exchanger, a filter, a demineralizer, piping, and associated valves and instrumentation. Alternate cooling capability can be made available under anticipated malfunctions or failures (expected fault conditions).

Loop piping is so arranged that the failure of any pipeline does not drain the spent fuel pit below the top of the stored fuel elements.

The thermal design basis for the loop provides for all fuel pool rack locations to be filled at the end of a full core discharge.

9.3.1.2 Design Characteristics

9.3.1.2.1 Component Cooling Loop

Normally one pump and two component heat exchangers are operated to provide cooling water for the components located in the auxiliary building and the reactor containment building. At elevated CCW supply temperatures two pumps may be required. The water is normally supplied to all components being cooled even though one of the components may be out of service.

Cooling is provided by at least one component cooling pump during the recirculation phase of a loss-of-coolant accident. The cooling of the recirculation pump motors is supported by the operation of at least one auxiliary component cooling water pump during the recirculation phase of a LOCA.

Makeup water is taken from the primary water treatment plant, as required, and delivered to the surge tank. A backup source of water is provided from the primary water makeup transfer pumps.

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

1. A pressure indicator on the line between the component cooling pumps and the component cooling heat exchangers.
2. A temperature indicator, flow indicator, and radiation monitor in the outlet line from the heat exchangers.
3. A temperature indicator on the main inlet line to the component cooling pumps.

9.3.1.2.2 Residual Heat Removal Loop

Two pumps and two residual heat exchangers perform the decay heat cooling functions for the reactor unit. After the reactor coolant system temperature and pressure have been reduced to approximately 350°F and below 365 psig (the upper limit to prevent RHR system overpressurization), respectively, decay heat cooling is initiated by aligning pumps to take suction from the reactor outlet line and discharge through the heat exchangers into the reactor inlet line. The normal plant cooldown times to cold shutdown and refueling entry conditions

using 95°F Service Water are given in table 9.3-3. If only one pump and one heat exchanger are available, reduction of reactor coolant temperature is accomplished at a lower rate.

The equipment used for decay heat cooling is also used for emergency core cooling during loss-of-coolant accident conditions. This is described in Chapter 6.

9.3.1.2.3 Spent Fuel Pit Cooling Loop

The spent fuel pit contains spent fuel discharged from the Unit 2 and Unit 3 reactors. Spent fuel cooling loop performance has been analyzed for operation at a core power level of 102% of 3216 MWt and at service water temperatures up to 95°F. When a refueling load of approximately 88 freshly discharged assemblies (plus previously discharged assembles) are present, the pump and spent fuel heat exchanger will handle the load and maintain a bulk pit water temperature less than 140°F. When a full core of 193 assemblies is freshly discharged, the bulk pit water temperature is maintained below 180°F.

Two criteria must be met before spent fuel can be discharged to the spent fuel pit:

1. In accordance with Technical Requirements Manual Section 3.9.A, spent fuel can not be discharged to the spent fuel pit until at least 84 hours after shutdown to satisfy the assumptions of the spent fuel handling accident analysis as discussed in Section 14.2.1.

2. An additional delay time limit prior to spent fuel discharge is administratively controlled by operating procedures to ensure that the total spent fuel heat load is within the capacity of the spent fuel cooling loop to satisfy the bulk pit water temperature limits discussed above. This is a variable time limit primarily dependant upon service water temperature, and cooling capacity without supplemental cooling.

9.3.1.3 Codes and Classifications

All piping and components of the auxiliary coolant system are designed to the applicable codes and standards listed in Table 9.3-1. 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 that contain borated water without a corrosion inhibitor.

9.3.2 System Design and Operation

9.3.2.1 Component Cooling Loop

Component cooling is provided for the following heat sources:

1. Residual heat exchangers (auxiliary coolant system).
2. Reactor coolant pumps (reactor coolant system).
3. Nonregenerative heat exchanger (chemical and volume control system).
4. Excess letdown heat exchanger (chemical and volume control system).
5. Seal-water heat exchanger (chemical and volume control system).
6. Sample heat exchangers (sampling system).
7. Waste gas compressors (waste disposal system).
8. Reactor vessel support pads.

9. Residual heat removal pumps (auxiliary coolant system).
10. Safety injection pumps (safety injection system).
11. Recirculation pump motors (safety injection system).
12. Spent fuel pit heat exchanger (auxiliary coolant system).
13. Charging pumps (chemical and volume control system), fluid drive coolers and crankcase.

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 directly to the service water system. The component cooling loop is monitored for radioactivity by a radiation monitor that samples the component cooling pump discharge downstream of the component cooling heat exchangers.

During normal full power operation, one component cooling pump and two component cooling heat exchangers accommodate the heat removal loads. Two CCW pumps are in stand-by and both heat exchangers are utilized. At elevated CCW supply temperatures two CCW pumps may be required. Three pumps and two heat exchangers can be used to remove the residual and sensible heat during plant shutdown. If one of the pumps or one of the heat exchangers is not operative, safe shutdown of the plant is not affected; however, the time for cooldown is extended. The surge tank accommodates expansion, contraction and inleakage of water, and ensures a continuous component cooling water supply until a leaking cooling line can be isolated. Makeup to the surge tank is provided from the primary water makeup system. The surge tank is normally vented to the atmosphere. In the unlikely event that the radiation level in the component cooling loop reaches a preset level above the normal background, the radiation monitor in the component cooling loop annunciates in the control room and closes a valve in the surge tank vent line. Parameters for components in the component cooling loop are presented in Table 9.3-2.

9.3.2.2 Residual Heat Removal Loop

The residual heat removal loop 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 Section 6.2. 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 is controlled by regulating the flow through the tube side of the residual heat exchangers. Two remote motor-operated control valves downstream of the residual heat exchangers are used to control flow.

Instrumentation has been provided in the control room to monitor RHR and reactor coolant system level when the system is cooled and depressurized. These instruments are provided to monitor level during draindown to assure decay heat removal capability. A channel with an intermediate range of 240 inches measures differential pressure between the top of the pressurizer and the bottom of the hot leg. A wide range channel, capable of monitoring level

from the bottom of the hot leg to top of the pressurizer, measures the pressures at those two points using redundant pressure transducer pairs. This wide range channel also has an optional third transducer pair, which may be installed on the reactor head vent line to monitor reactor water level during certain evolutions. Another level channel with a differential pressure sensor can also be used to monitor RCS level. A narrow range level channel using an ultrasonic transducer monitors level in hot leg 21. Wide and narrow range instrumentation is also provided to measure RHR system flow. Monitors are provided for RHR pump suction pressure and discharge pressure. RHR temperature sensors are located at the common discharge header of the RHR pumps and RHR Heat Exchanger outlet line. RCS temperature is monitored by core exit thermocouples whenever the reactor head and the instrumentation interface assembly are in place. The RHR pump monitors, along with the narrow range level and flow instrumentation assists the operators in avoiding air entrainment in the RHR pump suction line during periods when the reactor is shut down and water level has been lowered.

Remotely-operated, double valving is provided to isolate the residual heat removal loop from the reactor coolant system. When reactor coolant system pressure exceeds the design pressure of the residual heat removal loop, interlocks between the reactor coolant system wide range pressure channels and the residual heat removal inlet valves prevent the valves from opening. A remotely-operated normally closed valve and two check valves isolate each line to the reactor coolant system cold legs from the residual heat removal loop during power operation. Parameters for components in the residual heat removal loop are presented in Table 9.3-3.

9.3.2.3 Spent Fuel Pit Cooling Loop

The spent fuel pit cooling loop removes residual heat from fuel placed in the pit for long term storage. The loop can safely accommodate the heat load from all of the assemblies for which there is storage space available.

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 interchange of water occurs as fuel assemblies are transferred.

The spent fuel pit cooling loop consists of two pumps, a heat exchanger, filter, demineralizer, piping and associated valves and instrumentation. One of the pumps draws water from the pit, circulates it through the heat exchanger and returns it to the pit. Component cooling water cools the heat exchanger. Redundancy of this equipment is not required because of the large heat capacity of the pit and the slow heatup rate.

The clarity and purity of the spent fuel pit water is 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 draw 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.

A separate pump is used to circulate refueling water storage tank water through the same demineralizer and filter for purification.

Parameters for components in the spent fuel cooling loop are presented in Table 9.3-4.

9.3.2.4 Component Cooling Loop Components

9.3.2.4.1 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. Parameters are presented in Table 9.3-2.

9.3.2.4.2 Component Cooling Pumps

The three component cooling pumps, which circulate component cooling water through the component cooling loop are horizontal, centrifugal units. The original pumps have casings made from cast iron (ASTM 48) based on the corrosion-erosion resistance and the ability to obtain sound castings. The material thickness indicates the high quality casting practice and the ability to withstand mechanical damage and, as such, is substantially overdesigned from a stress level standpoint. Carbon steel casing material (ASTM A216) has been evaluated and approved for replacement pumps. Parameters are presented in Table 9.3-2.

9.3.2.4.3 Auxiliary Cooling Water Pumps

The component cooling pumps do not run during the injection phase of a loss of coolant accident with loss of offsite power. The CCW circulating water pumps provide cooling for the high head safety injection pumps during this situation. These pumps are connected to the motor shaft of each safety injection pump and cool the pump bearings / seals when the safety injection pumps operate. The heat is absorbed by the thermal inertia of the component cooling water system.

The auxiliary component cooling water pumps start on a Safety Injection signals and can operate during injection phase with or without Loss-Of-Offsite-Power (LOOP) by being automatically loaded onto the Emergency Diesel Generators (EDG). Their originally intended function was to provide adequate cooling flow to the recirculation pump motor coolers for both the injection and recirculation phase of the accident. An evaluation had concluded that during the injection phase when the recirculation pumps are not operating, the recirculation pump motors were qualified by testing to withstand the post-LOCA environment with no dedicated cooling. However, during the recirculation phase the function of the auxiliary component cooling water pumps is credited to ensure sufficient cooling flow to the motor coolers is available.

Both the auxiliary component cooling water pumps and the CCW circulating water pumps are discussed in further detail in Section 6.2.

9.3.2.4.4 Component Cooling Surge Tank

The component cooling surge tank, which accommodates changes in component cooling water volume is constructed of carbon steel. Parameters are presented in Table 9.3-2. In addition to piping connections, the tank has a flanged opening at the top for the addition of the chemical corrosion inhibitor to the component cooling loop.

9.3.2.4.5 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 beyond their design pressure by improper operation or malfunction.

9.3.2.4.6 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. The piping has been evaluated for the most limiting component cooling water temperatures under loss of coolant accident conditions and found to be acceptable

9.3.2.5 Residual Heat Removal Loop Components

9.3.2.5.1 Residual Heat Exchangers

The two residual 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.

9.3.2.5.2 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. Cooling water is provided from the component cooling water system via flexible stainless steel hose.

9.3.2.5.3 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 a pressure interlock isolate the residual heat removal loop from the reactor coolant system. In addition the residual heat removal loop is isolated from the reactor coolant system by two series check valves and a remotely operated stop valve on the outlet lines. As depicted in Plant Drawing 227781 [Formerly UFSAR Figure 9.3-1, Sheet 1], overpressure protection in the residual heat removal loop is provided by a relief valve. 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 and to limit the stem leakage when the valves are open.

9.3.2.5.4 Residual Heat Removal Piping

All residual heat removal loop piping is austenitic stainless steel. The piping is welded with flanged connections at the pumps and at valve 741A.

9.3.2.5.5 Low Pressure Purification System

The system is used to clean reactor coolant water when the primary system is depressurized during an outage. The system has a 100-gpm canned purification pump, a line that bypasses the volume control tank and charging pumps of the chemical and volume control system and associated valves as shown in Plant Drawing 208168 [Formerly UFSAR Figure 9.2-1, sheet 2]. The system is designed for 600 psi operation.

9.3.2.6 Spent Fuel Pit Loop Components

9.3.2.6.1 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.

9.3.2.6.2 Spent Fuel Pit Pumps

One of two spent fuel pit pumps circulates water in the spent fuel pit cooling loop. The second pump is on standby. All wetted surfaces of the pumps are austenitic stainless steel, or equivalent corrosion resistant material. The pumps are operated manually from a local station.

9.3.2.6.3 Refueling Water Purification Pump

When it is required to clean up the refueling water storage tank water, 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.

9.3.2.6.4 Spent Fuel Pit Filter

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

9.3.2.6.5 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.

9.3.2.6.6 Spent Fuel Pit Demineralizer

The demineralizer is sized to pass 5-percent of the loop circulation flow, to provide adequate purification of the fuel pit water for unrestricted access to the working area, and to maintain optical clarity. In addition, it is used for purification of the refueling water storage tank water.

9.3.2.6.7 Spent Fuel Pit Skimmer [Deleted]

9.3.2.6.8 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.

9.3.2.6.9 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 are used at the pump, heat exchanger, and filter to facilitate maintenance.

9.3.3 System Evaluation

System performance has been evaluated for service water temperatures up to 95°F for normal operating modes, loss of offsite power and loss of coolant accident conditions.

9.3.3.1 Availability And Reliability

9.3.3.1.1 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 postaccident 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 postaccident 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 practicable while the other units are in service. The wetted surfaces of the component cooling loop are 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. The entire system is seismic Class I and is housed in structures of the same classification. The components are designed to the codes given in Table 9.3-1 and the design pressures given in Table 9.3-2. In addition, the components are not subjected to any high pressures or stresses. Hence, a rupture or failure of the system is very unlikely.

In the event of a loss-of-offsite power, the plant emergency diesel generators are immediately started and the component cooling water pumps are automatically loaded (in sequence) onto the emergency buses and started. Component cooling water to the reactor coolant pump thermal barrier heat exchanger is thus automatically restored to provide reactor coolant pump seal cooling and prevent seal failure for at least a 2-hr period following a loss-of-offsite power.

An alternate power supply is also provided for one of the component cooling water pumps from the 13.8-kV normal offsite power through Unit 1 switchgear. If normal offsite power is not available, this pump can be energized using the SBO / Appendix R diesel. During the recirculation phase following a loss-of-coolant accident, one of the three component cooling water pumps is required to deliver flow to the shell side of one of the residual heat exchangers.

9.3.3.1.2 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 operable, safe operation is governed by Technical Specifications and safe shutdown 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 Section 6.2.

Alternate power can be supplied to one residual heat removal pump from the 13.8-kV normal outside power through Unit 1 switchgear.

The time to cool down using the alternate safe shutdown components (1 RHR pump and heat exchanger, 1 component cooling pump, and 1 service water pump supplying flow to non-essential header) has been determined[1 & 2]. Conditions assumed were an initial core power of 102% of 3216 MW and service water temperature of 95°F. The analysis shows that the RCS can be brought to the cold shutdown mode (temperature less than 200°F) within 72 hours.

9.3.3.1.3 Spent Fuel Pit Cooling Loop

This manually controlled loop may be shut down safely for time periods, as shown in Section 9.3.3.2.3, for maintenance or replacement of malfunctioning components.

9.3.3.2 Leakage Provisions

9.3.3.2.1 Component Cooling Loop

Water leakage from piping, valves, and equipment in the system inside the containment is not considered to be generally detrimental unless the leakage exceeds the makeup capability. With respect to water leakage from piping, valves, and equipment outside the containment, welded construction is used where possible to minimize the possibility of leakage. The component cooling water could become contaminated with radioactive water due to a leak in any heat exchanger tube in the chemical and volume control, the sampling, or the auxiliary coolant systems, or a leak in the thermal barrier cooling coil for the reactor coolant pumps.

Tube or coil leaks in components being cooled would be detected during normal plant operations by the leak detection system described in Sections 4.2.7 and 6.7. Such leaks are also detected at any time by a radiation monitor that samples the component cooling pump discharge 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 tank. The rate of water level fall and the area of the water surface in the tank 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 one of the component cooling water heat exchangers it can be isolated and repaired within the limitations of the Technical Specifications. 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 tank would rise, and the operator would be alerted by a high water alarm. The atmospheric vent on the tank is automatically closed in the event of high radiation level in the component cooling loop. If the leaking residual heat exchanger is not isolated from the component cooling loop before the inflow completely fills the surge tank, the relief valve on the surge tank lifts. The discharge of this relief valve is routed to the auxiliary building waste holdup tank.

The severance of a cooling line serving an individual reactor coolant pump cooler would result in substantial leakage of component cooling water. However, the piping is small as compared to piping located in the missile-protected area of the containment. Therefore, 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.

The relief valves on the component cooling water lines downstream from each reactor coolant pump protect the downstream piping and thermal barrier cooling coils from overpressure should cooling water be isolated to the thermal barrier coil when the reactor coolant pumps are still operating. The valves set pressure equals the design pressure of the reactor coolant system.

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

The relief valve on the component cooling surge tank is sized to relieve the maximum flow rate of water, which enters the surge tank following a rupture of a reactor coolant pump thermal barrier cooling coil. The set pressure will allow the component cooling system to be a closed system under accident conditions, even at 100-percent of containment design pressure. The over-pressurization incident, which results from a passive failure of a reactor coolant pump seal cooling coil coincident with the failure of the high flow cutoff valve would result in a maximum component cooling water pressure of 185 psig. This pressure is allowed in the component cooling water system in accordance with its design code of B31.1, 1967 edition, par 102.2.4(2), addressing permissible variation and allowable stress value for a limited time.

9.3.3.2.2 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 containment 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 a common sump. Two redundant sump pumps, each capable of handling the less than 50 gpm flow, which would result from the failure of a residual heat removal pump seal, discharge to the waste holdup tank.

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 Generic Letter 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 used in the HHSI and RHR pumps.

9.3.3.2.3 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 bypass purification loop is provided for removing these fission products and other 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 mode would be from such actions 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. With no heat removal the time for the spent fuel pit water to rise from 180°F to 212°F with a full core in storage is at least 1.8 hr. Makeup water can be supplied within this time from the primary water storage tank, the refueling water storage tank and/or the fire protection system. The maximum required makeup rate for boiloff is 62 gpm (for a full core). Spent fuel pit temperature and level instrumentation would warn the operator of an impending loss of cooling. A local flow indicator is available to support operation of the Spent Fuel Pit Pumps.

9.3.3.3 Incident Control

9.3.3.3.1 Component Cooling Loop

In the unlikely event of a pipe severance in the component cooling loop, backup is provided for postaccident heat removal by the containment fan coolers. Pipe severance is a passive failure and is assumed to occur 24 hours or greater after event initiation.

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 from 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 2 hr.

If the break occurs inside the containment on a cooling water line to a reactor coolant pump, the leak can be isolated. Each of the cooling water supply lines to the reactor coolant pumps contains a check valve inside and a common remotely operated valve outside the containment wall.

Each return line (combined oil coolers and combined thermal barrier coolers) has a common remotely operated valve outside the containment wall. The cooling water supply line to the excess letdown heat exchanger contains a check valve inside the containment wall and both supply and return lines have automatically isolated valves outside the containment wall. Should the break occur inside containment and the leak can not be isolated the residual heat removal pumps and safety injection pumps, if required, are employed to recirculate uncooled spilled water to the core that is removed from the core by boiling off of the water to the containment with the fan coolers being used to condense the resulting steam.

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 has a normally closed remotely-operated valve. If one of the valves fails to open upon a safety injection signal, the valve, which does open supplies a heat exchanger with sufficient cooling to remove the heat load during long term postaccident recirculation.

The portion of the component cooling loop located outside the containment is considered to be a part of the reactor building isolation barrier.

Except for the normally closed makeup line the primary water and city water emergency cooling lines, and equipment vent and drain lines, there are no direct connections between the cooling water and other systems. The primary water make-up and SIS/RHR Emergency Cooling Lines have manual valves that are normally closed unless required for their design function or testing. The city water emergency cooling line contains two normally closed isolation valves with an open tell-tale connection between them. The tell-tale prevents the potential contamination of a potable water source with component cooling water corrosion inhibitor chemicals. 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.

9.3.3.3.2 Residual Heat Removal Loop

The residual heat removal loop is connected to the reactor outlet line on the suction side and to the reactor inlet line on the discharge side. On the suction side the connection is through two electric motor-operated gate valves in series with both valves independently interlocked with reactor coolant system pressure. On the discharge side the connection is through two check valves in series with an electric motor-operated gate valve. All of these are closed whenever the reactor is in the operating condition.

9.3.3.3.3 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 is also 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.

Instrumentation is provided that will activate an alarm in the control room if the level in the spent fuel pit is at a preset level deviation above or below normal. Operators normally observe the level in the pool on a regular basis.

9.3.3.4 Malfunction Analysis

A failure analysis of pumps, heat exchangers and valves is presented in Table 9.3-5.

9.3.4 Minimum Operating Conditions

Minimum operating conditions for the auxiliary coolant system are specified in the Technical Specifications.

9.3.5 Tests and Inspections

Tests and inspections of the auxiliary coolant system are specified in the Technical Specifications.

The portion of the Residual Heat Removal System that is outside of containment, and not tested in accordance with Technical Specifications, shall be tested at least once each 24 months either by use in normal operation or by hydrostatically testing at 350 psig. The piping, between the residual heat removal pump suction and the containment isolation valves in the residual heat removal pump suction line from the containment sump, shall be hydrostatically tested once each 24 months at no less than 100 psig. Visual inspection of the system components shall be performed during these tests and any significant leakage shall be measured by collection and weighing or by another equivalent method. Repairs or isolation shall be made as required to maintain leakage from the Residual Heat Removal System components located outside of the containment per Technical Specification 5.5.2.

REFERENCES FOR SECTION 9.3

1. Letter (with attachment, WCAP-12312) from S. Bram, Con Edison, to NRC, Subject: Application for License Amendment to Increase the Design Basis Inlet Temperature of the Service Water System, dated July 13, 1989.

2. Westinghouse calculation CN-SEE-03-5, Indian point Unit 2 RHR Cooldown Analysis for the 5% Power Uprate Program, Rev. 0.

TABLE 9.3-1
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 side ASME VIII, shell side	ASME III, Class C, tube
Residual heat removal piping and valves	USAS B31.1
Spent fuel pit filter	ASME III, Class C
Spent fuel heat exchanger side ASME VIII, shell side	ASME III, Class C, tube
Spent fuel pit loop piping and valves	USAS B31.1

TABLE 9.3-2 (Sheet 1 of 2)
Component Cooling Loop Component Data

Component Cooling Pumps	Parameters
Quantity	3
Type	Horizontal centrifugal
Rated capacity (each), gpm	3600
Rated head, ft H_2O	220
Motor horsepower, hp	250
Material (pump casing)	Cast iron or Carbon steel
Design pressure, psig	150
Design temperature, °F	200

Component Cooling Heat Exchangers	
Quantity	2
Type	Shell and straight tube
Design heat transfer, Btu/hr	31.4×10^6
Shell side (component cooling water)	
Operating inlet temperature, °F	100.1
Operating outlet temperature, °F	88.2
Design flow rate, lb/hr	2.66×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Aluminum-bronze
Tube side (service water)	
Operating inlet temperature, °F	75_1
Operating outlet temperature, °F	81.9
Design flow rate, lb/hr	4.55×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Copper-nickel (90-10)

TABLE 9.3-2 (Sheet 2 of 2)

Component Cooling Loop Component Data

Component Cooling Surge Tank	
Quantity	1
Volume, gal	2000
Normal water volume, gal	1000
Design pressure, psig	100
Design temperature, °F	200
Construction material	Carbon steel
Relief valve setpoint, psig	52
Auxiliary Component Cooling Water Pumps	
Quantity	2
Type	Vertical centrifugal
Rated capacity, gpm	80
Rated head, ft H_2O	100
Motor horsepower, hp	5
Casing material	Cast steel
Design pressure, psig	150
Design temperature, °F	200
CCW Circulating Water Pumps (Safety Injection Pumps)	
Quantity	3
Type	Centrifugal
Rated capacity, gpm	40
Rated head, ft H_2O	110
Casing material	Stainless Steel
Design pressure, psig	225
Design temperature, °F	200
Component Cooling Loop Piping and Valves	
Design pressure, psig	150
Design temperature, °F	200

Notes:
1. Operation is acceptable up to 95°F.

TABLE 9.3-3 (Sheet 1 of 2)
Residual Heat Removal Loop Component Data

Reactor coolant temperature at startup of heat removal, °F	350
Time to cool reactor coolant system from	
350°F to 200°F, hr (all equipment operational)	48 [1]
350°F to 140°F, hr (all equipment operational)	113.6 [1]
Refueling water storage temperature, °F	Ambient
Decay heat generation at 10 hrs after shutdown	
condition, Btu/hr	85.6×10^6 [1]
Reactor cavity fill time, hr	1
Reactor cavity drain time, hr	4

Residual Heat Removal Pumps

Quantity	2
Type	Vertical centrifugal
Rated capacity (each), gpm	3000
Rated head, ft H_2O	350
Motor, hp	400
Material	Stainless steel
Design pressure, psig	600
Design temperature, °F	400

TABLE 9.3-3 (Sheet 2 of 2)
Residual Heat Removal Loop Component Data

Residual Heat Exchangers	
Quantity	2
Type	Shell and U-tube
Design heat transfer (each), Btu/hr	30.8×10^6
Shell side (component cooling water)	
Operating inlet temperature, °F	88.3
Operating outlet temperature, °F	100.8
Design flow rate, lb/hr	2.46×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Carbon steel
Tube side (reactor coolant)	
Operating inlet temperature, °F	135
Operating outlet temperature, °F	113.5
Design flow rate, lb/hr	1.44×10^6
Design temperature, °F	400
Design pressure, psig	600
Material	Stainless steel
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

Notes:

1. Aligned to RHR system at 20 hours after shutdown, 95°F Service Water

TABLE 9.3-4 (Sheet 1 of 3)
Spent Fuel Cooling Loop Component Data

Spent fuel pit heat exchanger	
Quantity	1
Type	Shell and U-tube
Design heat transfer, Btu/hrs$_1$	7.96×10^6
Shell side (component cooling water)	
Normal operating inlet temperature, °F$_1$	100
Normal operating outlet temperature, °F$_1$	105.7
Design flow rate, lb/hr	1.4×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Carbon steel
Tube side (spent fuel pit water)	
Normal operating inlet temperature, °F$_1$	120
Normal operating outlet temperature, °F$_1$	112.8
Design flow rate, lb/hr	1.1×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Stainless steel
Spent fuel pit skimmer pump	Retired in place
Refueling water purification pump	
Quantity	1
Type	Horizontal centrifugal
Rated capacity, gpm	100
Rated head, ft H_2O	150
Design pressure, psig	150
Design temperature, °F	200
Material	Stainless steel

TABLE 9.3-4 (Sheet 2 of 3)
Spent Fuel Cooling Loop Component Data

Spent fuel pit cooling loop piping and valves	
Design pressure, psig	150
Design temperature, °F	200
Spent fuel pit skimmer loop piping and valves	Retired in place
Refueling water purification loop piping and valves	
Design pressure, psig	150
Design temperature, °F	200
Spent fuel pit pump	
Quantity	2
Type	Horizontal centrifugal
Material	Stainless steel
Rated capacity, gpm	2,300
Rated head, ft H_2O	125
Motor, hp	100
Design pressure, psig	150
Design temperature, °F	200
Spent fuel storage pool	
Volume ft^3	37,300
Typical Boron concentration, ppm boron	>2,000 min
Tech Spec Boron concentration, ppm boron	>2,000 min
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 filter element prior to removing, psi	20
Filtration requirement	98-percent retention of particles down to 5 μ

TABLE 9.3-4 (Sheet 3 of 3)

Spent Fuel Cooling Loop Component Data

Spent fuel pit strainer	
Quantity	1
Rated flow, gpm	2,300
Maximum differential pressure across the strainer element at rated flow (clean), psi	1
Perforation, in.	~0.2
Spent fuel pit demineralizer	
Quantity	1
Type	Flushable
Design pressure, psig	200
Design temperature, °F	250
Flow rate, gpm	100
Resin volume, ft^3	30
Spent fuel pit skimmers	Deleted
Spent fuel pit skimmer strainer	Retired in place
Spent fuel pit skimmer filter	Retired in place

Notes:
1. Original design.

TABLE 9.3-5
Failure Analysis of Pumps, Heat Exchangers, and Valves

	Components	Malfunction	Comments and Consequences
1.	Component cooling water pumps	Rupture of a pump casing	The casing and shell are designed for 150 psi and 200°F, which exceeds maximum operating conditions. Pump is inspectable and protected against credible missiles. Rupture is not considered credible. However, each unit is isolable. Two of the three pumps are needed to carry total pumping load.
2.	Component cooling water pumps	Pump fails to start	One operating pump supplies sufficient cooling water for emergency core cooling during recirculation.
3.	Component cooling water pumps	Manual valve on a pump suction line	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 low operating pressures. Each unit is isolable. Both units may be required to carry total heat load for normal operation at 95°F Service Water.
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 valve	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 outlet 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-term recirculation. During normal operation the cooldown time is extended.

9.3 FIGURES

9.4 SAMPLING SYSTEM

9.4.1 Design Basis

9.4.1.1 Performance Requirements

This system provides for analysis of liquid and gaseous samples obtained during normal operation and postaccident conditions. The containment atmosphere postaccident sampling system is discussed in Sections 6.8.2.2 and 6.8.2.3. Sampling of the primary and secondary coolant systems is discussed below.

Primary samples include the following:

1. Reactor coolant system hot-leg loops 21 and 23.
2. Pressurizer steam space and liquid space.
3. Residual heat removal loop.
4. Safety injection system accumulators 21, 22, 23, and 24.
5. Recirculation pumps 21 and 22 discharge.
6. Chemical and volume control system letdown lines at demineralizer inlet and outlet.
7. Holdup tanks.
8. CVCS holdup tank transfer pumps discharge.
9. Chemical drain pump 21 discharge.
10. Waste evaporator feed pump 21 discharge.
11. High-radiation sampling system collection tank discharge.

These samples are obtained at the high-radiation sampling system panels and evaluated by the online analysis systems or manual analysis. Secondary samples are obtained from the secondary sampling system, which is separate from the high–radiation sampling system. Postaccident sampling of the primary system is an extension of the use of the high-radiation sampling system for routine sampling.
The NRC approved[3] the removal of the requirements and administrative controls for the postaccident sampling system from the Technical Specifications and accepted regulatory commitments to maintain:

1. contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere;

2. the capability for classifying fuel damage events at the Alert threshold within the Emergency Plan Implementing Procedures (EPIPs); and

3. the capability for monitoring radioactive iodines that have been released to offsite environs within the EPIPs.

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. Shielding has been provided to minimize operator exposure to any radiation during the sampling procedures.

The primary coolant sampling system was evaluated by the NRC against the criteria in Item II.B.3 of NUREG-0737 and found acceptable.[1,2]

9.4.1.2 Design Characteristics

The design characteristics of the high-radiation sampling system include the following:

1. Control of background radiation and operator exposure to radiation.
2. Rapid sampling and analysis.
3. Sampling and transfer of undiluted samples.

In addition, the system is capable of the following:

1. The system can be used for both routine and postaccident sampling and has the capability to obtain an undiluted reactor coolant sample under accident conditions for transport offsite for independent analyses.

2. Inline measurement of the reactor coolant specific conductivity, pH, and dissolved oxygen, hydrogen, chlorides, and boron under both routine and postaccident conditions.

3. Additional sample connections are available for more flexibility in selecting sample points; redundant sample connections allow for further expansion if needed to ensure sample acquisition under postaccident conditions.

4. Methods for cooling and depressurizing all high temperature-high pressure fluids for gas sampling and inline analyses.

5. Specially designed shielded transfer casks minimize operator radiation exposure when obtaining diluted and undiluted liquid samples. A small aliquot of reactor coolant system liquid or containment air samples is transferred as required to designated areas for analyses using a holder to maintain adequate distance and provide low operator radiation exposure.

Flow paths are also provided for boron concentration, and isotopic inline analysis.

Sampling of other process coolants, such as tanks in the waste disposal system, is accomplished locally. Equipment for sampling secondary and nonradioactive fluids is 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.

9.4.1.3 Primary Sampling

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

9.4.1.3.1 High Pressure-High Temperature Samples

A sample connection is provided from each of the following:

1. The pressurizer steam space.
2. The pressurizer liquid space.
3. Hot legs of loops 21 and 23.
4. Blowdown from each steam generator.

9.4.1.3.2 Low Pressure-Low Temperature Samples

A sample connection is provided from each of the following:

1. The letdown demineralizers inlet and outlet header.
2. The residual heat removal loop, just downstream of the heat exchangers.
3. The volume control tank gas space.
4. The (safety injection system) accumulators 21, 22, 23, and 24.
5. Recirculation pumps 21 and 22 discharge.

9.4.1.4 Expected Operating Temperatures

The high pressure-high temperature samples and the residual heat removal loop samples leaving the sample heat exchangers are cooled to minimize the generation of radioactive aerosols.

9.4.1.5 Secondary Sampling

The secondary sampling system provides continuous sampling and analysis of the plant's secondary systems. This ensures the maintenance of proper water chemistry conditions in the secondary side piping and equipment.

A sample connection is provided from each of the following:

1. Each of the four main steam lines.
2. Each condenser hotwell section.
3. Condensate pump discharge.
4. Outlet of the 26 feedwater heaters.
5. Drains collection tank inlet from primary water.

9.4.1.6 Codes and Standards

System code requirements are given in Table 9.4-1. In addition, the high radiation sampling system was designed and installed to meet the provisions of NUREG-0737. These provisions include the following:

1. Provide postaccident sampling and analysis capability. The combined time for sampling and analysis is 3 hr or less from the time a decision is made to take a sample.

2. Provide capability to obtain and analyze a sample without radiation exposure to any individual exceeding the criteria of GDC 19 (10 CFR Part 50, Appendix A).

3. Provide means of measuring pH, conductivity, chlorides, dissolved hydrogen, dissolved oxygen, inline isotopic analysis, and boron analysis.

4. Provide means of safely obtaining pressurized samples, depressurized samples, and diluted and undiluted samples for laboratory analysis.

5. Safely store the sampled fluid until its disposal is determined.

6. Provide means of diverting to the containment the stored sample fluid.

7. Provide the capability to use the system on a continuous day-to-day basis.

8. Provide the capability to flush the sampled lines.

9. Provide the capability of drawing samples even when the reactor coolant system is depressurized (reactor coolant system, residual heat removal, and recirculation lines).

9.4.2 System Design and Operation

9.4.2.1 Primary Sampling System

The primary sampling system consists of the high-radiation sampling system, which is shown in Plant Drawing 9321-2745. The high-radiation sampling system provides the representative samples for inline monitoring and laboratory analysis under normal or postaccident conditions. Analytical 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, evaluating fuel element integrity and mixed-bed demineralizer performance, and regulating additions of corrosion controlling chemicals to the systems. The high-radiation sampling system can be operated intermittently or on a continuous basis. Samples can be withdrawn under conditions ranging from full power to cold shutdown to postaccident conditions.

Reactor coolant liquid, [***Note*** *- For postaccident conditions, the reactor coolant liquid sample may be taken from the reactor coolant system hot legs 21 and 23 or the recirculation pump discharge or the residual heat removal loop.*], which is normally inaccessible or which requires frequent sampling, is sampled by means of permanently installed piping leading to either the inline isotopic analyzer, or the liquid sampling panel located in the sentry high-radiation sampling system room (formerly the waste evaporator room) at the 80-ft level of the primary auxiliary building. A seismic Class I concrete wall surrounds the high-radiation sampling system panel and a combination of lead shot and steel composes the shielding for the panel itself. These materials provide the shielding necessary to allow access to the high-radiation sampling system during and following accident conditions. Most of the primary sampling equipment is located in the sentry high-radiation sampling system room although some of it is located in other areas such as the pipe trench area of the 51-ft elevation and the 68-ft elevation of the mezzanine within the primary auxiliary building. The delay coils and remotely operated valves on the reactor coolant system hot-leg sample lines are located inside the reactor containment. Containment isolation valves are located immediately outside containment and are controlled, in an accident, from either the central control room or the sample system valve control panel. A line from the makeup water system has been installed to provide water for flushing of the sample lines.

Reactor coolant hot-leg liquid, pressurizer liquid, and pressurizer steam samples originating inside the reactor containment flow through separate sample lines to the sentry liquid sampling panel. The samples pass through the reactor containment to the auxiliary building where they are cooled (pressurizer steam samples recondensed and cooled) in the sample heat exchangers.

The reactor coolant samples are then routed through the inline isotopic analyzer where specific nuclides are identified.

All samples then go to the sentry high-radiation sampling system panel. This consists of a liquid sampling panel, which is subdivided into a reactor coolant module, which includes the capability for dissolved gas analysis, a demineralizer sampling module, and a radwaste sampling module. Associated with the liquid sampling panel is the chemical analysis panel. These modules are discussed in detail later.

The chemical analysis panel analytical results register on the chemical monitor panel in the sentry high-radiation sampling system room. There are remote readouts for the boron analysis in the radio chemistry laboratory and nuclear service building 1.

Reactor coolant and demineralizer samples from the chemical and volume control system are depressurized and degasified in the reactor coolant module and demineralizer modules, respectively. From there they are sent to the chemical analysis panel, which can analyze for hydrogen, oxygen, chlorides, pH, and conductivity.

Provisions are included in the primary sampling system to allow each sample to be purged through the sample lines and panel to ensure that representative samples are obtained. The sample volumes are routed to the high-radiation sampling system collection tank or chemical drain tank after completion of the task.

The reactor coolant sample originating from the residual heat removal loop of the auxiliary coolant system has a motor-operated isolation valve located close to the sample source outside the containment. The sample line from this source intersects the sample line coming from the

hot leg at a point ahead of the sample heat exchanger. This sample then follows the same flow path as that described for the reactor coolant system hot-leg samples. See Plant Drawings 9321-2745 and 227178 [Formerly UFSAR Figure 9.4-1].

A steam-generator sample line is taken from each blowdown line outside containment. The sample lines are routed to the blowdown tank room adjacent to the primary auxiliary building where the samples are cooled and are then passed through a radiation monitor as well as routed to cell 2 of the support facilities.

These sample streams pass through additional local heat exchangers in cell 2 and subsequently through radiation, pH, conductivity, and chloride monitors. The sample waste under normal conditions is then routed to the river. Samples not suitable for release are diverted to the support facilities contaminated drain tank and waste disposal system.

In the event of primary-to-secondary coolant leakage in one or more of the steam generators, the blowdown will be diverted to the support facilities secondary boiler blowdown purification system flash tank. This system cools the blowdown and either stores it in the support facilities waste collection tanks or purifies it. The purification process consists of filtering and demineralizing the blowdown. The filters will remove undissolved material of 25 μ or greater. Mixed-bed demineralizers, which utilize cation and anion resin, remove isotopic cations and anions as well as nonradioactive chemical species. The effluents of the demineralizers are monitored and the specific activity is recorded on a two-pen recorder in the support facilities chemical system control room.

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

9.4.2.1.1 Components

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

9.4.2.1.1.1 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 before samples reach the sample vessels and the 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 through the tube side and component cooling water from the auxiliary coolant system circulates through the shell side.

9.4.2.1.1.2 Delay Coil and Restriction Orifice

The high-pressure reactor coolant sample line, which contains a delay coil consisting of coiled tubing and a restriction orifice, will provide at least 40 sec sample transit time within the containment and an additional 20 sec transit time from the reactor containment to the sampling station. This allows for decay of short-lived isotopes to a level that permits normal access to the sampling room.

9.4.2.1.2 Liquid Sampling Panel

The liquid sampling panel valves and components are arranged in three modules installed in a common panel shield:

1. Module 1 - Reactor coolant sampling module (RC).
2. Module 2 - Demineralizer sampling module (DM).
3. Module 3 - Radwaste sampling module (RW).

Sample tubing and components are mounted behind the shielded panel within a plenum. Any gas leakage is vented to a local prefilter and HEPA filters and finally to existing ventilation ducts containing charcoal filters. A vessel at the bottom of the plenum collects any minor liquid leakage, which is pumped to radwaste. This provides containment of radioactivity during sampling operations.

As a safety measure, the liquid sampling panel has a hooded splash box to contain any accidental liquid spill or gaseous release during normal sampling of pressurized reactor coolant or liquid grab sampling from all three modules.

Each system can be purged through the sample lines and panel to ensure representative samples will be obtained. The purge flow can be directed back to the containment to chemical drain tank 21 and the associated waste disposal system or to the shielded high-radiation sampling system waste collection tank.

All lines of the liquid sampling panel can be flushed with demineralized water following each sampling operation. Provisions are included for eliminating water from the gas expansion vessel and drying the gas lines of the panel.

Included as part of the liquid sampling panel are carts, shielded casks, and other specialized equipment for sampling under accident conditions. After sampling, the shielded casks can be removed to provide samples for backup in-house analyses or stored for subsequent offsite analysis. The viewing window and sampling compartment for alignment of the cart and cask are located in the lower right section of the liquid sampling panel.

The types of samples that can be obtained from the liquid sampling panel during normal operation are:

1. Undiluted, depressurized liquid grab samples from the reactor coolant, demineralizer, and radwaste modules.
2. Removable 75-ml pressurized liquid samples from the reactor coolant module, for subsequent analysis in the chemical analysis panel.
3. Inline pressurized liquid samples from the reactor coolant module.

Additional functions of the liquid sampling panel during normal operation include:

1. Purging of lines with sample to ensure representative samples will be obtained.
2. Reduction of pressure and control of flow rate of the primary coolant as it flows to the chemical analysis panel.
3. Routing of stripped gas from the pressurized liquid sample to the chemical analysis panel gas chromatograph.

The types of samples that can be obtained from the liquid sampling panel during accident conditions are:

1. Undiluted liquid samples from the reactor coolant and radwaste modules in cart/cask.
2. Diluted (1 to 1000) liquid samples from the reactor coolant and radwaste modules in cart/cask.
3. Inline pressurized liquid sample from the reactor coolant module.
4. Diluted (1 to 15,000) stripped gas sample from the reactor coolant pressurized liquid sample.

Additional functions of the liquid sampling panel during accident conditions include:

1. Purging of lines with sample to ensure representative samples will be obtained.
2. Capability for back-flushing the inline filters of the reactor coolant and radwaste modules.
3. Capability for flushing all lines and sample bottles on an individual section basis to control radiation levels as necessary.
4. Routing of stripped gas from the pressurized reactor coolant sample to the chemical analysis panel gas chromatograph.
5. Reduction of pressure and control of flow rate of the primary coolant as it flows to the chemical analysis panel.

9.4.2.1.3 Isotopic Analyzer

Isotopic analyses may be performed on the following samples obtained from the liquid sampling panel:

1. Pressurized reactor coolant sample (gas and liquid) in removable sample flask for normal sampling.
2. Undiluted grab samples from the reactor coolant, demineralizer and radwaste modules of the liquid sampling panel for normal sampling.
3. Diluted liquid samples from the reactor coolant and radwaste modules of the liquid sampling panel for accident sampling.
4. Undiluted liquid samples from the reactor coolant and radwaste modules of the liquid sampling panel for offsite analyses during accident conditions.
5. Diluted stripped gas samples from the reactor coolant module of the liquid sampling panel for accident sampling.

Isotopic analyses are performed using a Ge(Li) detector gamma spectroscopy system using previously established counting geometries.

9.4.2.1.4 Boron Analyzer

Backup boron analyses may be performed on the following samples from the liquid sampling panel for analysis in the onsite laboratory.

1. Undiluted grab samples from the reactor coolant, demineralizer, and radwaste modules of the liquid sampling panel for normal sampling.

2. Diluted liquid samples obtained from the liquid sampling panel shielded cart/cask from the reactor coolant and radwaste modules of the liquid sampling panel for accident sampling.

The primary sampling system provides that both the routine and accident sample analyses of undiluted samples are performed online using a mannitol titration boron analyzer. It periodically samples an identical line from the chemical analysis panel from which conductivity, dissolved oxygen, and pH are measured.

The range of the accident procedure is from 0.5 to 6.0 ppm boron. The estimated precision at the 95-percent confidence level is +13-percent, -3.3-percent at the 2-ppm boron level.

9.4.2.1.5 Cart and Casks

The cart and casks associated with the liquid sampling panel are used for removal of samples obtained from the reactor coolant and radwaste modules during accident conditions. The shielded casks are mounted on a cart, which moves the cask into position for sampling from the liquid sampling panel. The carts permit access to the casks to obtain a laboratory sample or for storage in a remote area upon completion of the sampling operation.

9.4.2.1.6 Chemical Analysis Panel

The chemical analysis panel receives an undiluted liquid sample stream and stripped gas from the reactor coolant module of the liquid sampling panel.

The chemical analysis panel is divided into three major sections:

1. Flow control and cell section, consisting of the appropriate tubing, valves, and sensing elements.
2. Chromatograph section, containing two ion chromatographs for liquid analysis and a gas chromatograph for gas analysis.
3. Calibration section, where the solutions required for calibrating the pH, specific conductivity, and dissolved oxygen monitors, and ion chromatograph are available for use.

Valves, tubing, cells, and transmitters are mounted on the back of the panel shield within a plenum. Any gas leakage from the liquid sampling panel, chemical analysis panel, or boron analyzer is vented to a pre-filter and HEPA filter and subsequently to the primary auxiliary building ventilation ducts containing charcoal filters. Drip pans are mounted beneath the flow control/cell section and ion chromatograph to collect any minor leakage and to protect other equipment.

The ion and gas chromatographic equipment, which contacts radioactive liquid or gas is mounted behind the shield to minimize operator exposure during the sampling/analysis process. The chemical analysis panel gas chromatograph and ion chromatograph sampling operations are controlled from the chemical monitor panel.

The chemical analysis panel provides the capability for inline determination of the pH, specific conductivity, dissolved oxygen, temperature, and chloride content of a reactor coolant sample flowing from the liquid sampling panel during normal or accident conditions. In addition, the gas chromatograph permits determination of the hydrogen concentration of the stripped gas from

the reactor coolant. Remote readouts of the instrumentation measuring the chemical parameters are on the chemical monitor panel.

Flushing lines are provided to flush all internal liquid and gas panel lines, and sample lines connecting the chemical analysis panel to the liquid sampling panel. Reagent calibration tanks may be flushed with nitrogen.

9.4.2.1.7 Chemical Monitor Panel

The chemical monitor panel is an auxiliary recorder/monitor panel, which contains the indicating and recording equipment for the cells and analyzers, which are mounted in the chemical analysis panel. The panel permits the operator to work with and observe indicating and recording equipment from a remote location, to reduce exposure under accident conditions.

Prior to sampling, the operator performs instrument zero and calibration adjustments of the monitors and evaluates chromatograms during the process of calibrating the instrumentation. This is accomplished prior to the chemical analysis panel receiving reactor coolant liquid or stripped gas from the liquid sampling panel.

The monitor indicator readings include conductivity, pH, and dissolved oxygen measurements. The dissolved oxygen monitor, for low level routine analysis, includes a meter indication while the oxygen/temperature monitor provides a recording during accident conditions for higher levels of dissolved oxygen.

A three-channel recorder records the chromatograms from the ion and gas chromatograph. The ion chromatogram is evaluated to determine the chloride concentration in the reactor coolant. Dissolved hydrogen concentration in the reactor coolant is determined by evaluating the gas chromatogram. Control of the sample injection to the chromatographs is provided by controls on the front of the panel.

9.4.2.1.8 High Radiation Sampling System Collection Tank

After analysis, the liquid and gaseous samples are routed to the high radiation sampling system collection tank. A nitrogen line to the tank provides a pressurized noncombustible atmosphere. A vent line is provided for the venting of excess gases. There is also a line running back to the high radiation sampling system panel for analysis of the contents of the tank. If the level of radiation is too high following an accident the samples in the tank can be routed back to containment; otherwise the samples will be routed to the chemical drain tank.

9.4.2.1.8.1 Chemical Drain Tank

During normal operation the liquid and gaseous samples are routed to the chemical drain tank. This tank is then pumped to the Unit 2 waste holdup tank. A sample can be directed to the radwaste module, if analysis is required prior to transfer.

9.4.2.1.8.2 Piping and Fittings

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

9.4.2.1.8.3 Valves

Remotely-operated stop valves are used to isolate all sample points and to route sample fluid flow inside the reactor containment. Manual or motor-operated stop valves are provided for component isolation and flow path control at all normally accessible sampling system locations. Manual throttle valves are provided to adjust the sample flow rate.

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

Isolation valves are provided outside the reactor containment, which trip closed upon generation of the containment isolation signal.

9.4.2.2 Secondary Sampling System

The secondary sampling system is shown in Plant Drawing 9321-7020 [Formerly UFSAR Figure 9.4-2]. This system is used to determine steam and condensate/feedwater quality and chemical addition requirements.

The steam and water analysis station is located in the turbine building. It consists of a local panel where various controls, alarms, recorders and indicators are located; racks for the sample coolers and analyzers; and, a sample sink where grab samples can be obtained.

The main steam can be analyzed for various additives, contaminants or isotopes.

The condensate and/or feedwater can be analyzed for salinity, pH, conductivity, dissolved oxygen, residual hydrazine, and various additives and contaminants. High salinity is indicative of river water leakage into the condenser or makeup carryover.

Conductivity is measured to determine the degree of possible dissolved solids entrainment into the systems.

Because of its corrosive effects, dissolved oxygen is measured and recorded and used as a guide in determining the proper amount of hydrazine to be added to the condensate.

The six individual condenser hotwells are provided with specific conductivity analyzers. These instruments are used to identify the specific condenser sextant that has salt water ingress.

9.4.3 System Evaluation

9.4.3.1 Availability and Reliability

Automatic action is not required of the sampling system during an emergency or to prevent an emergency condition. In a postaccident situation, after proper safeguards are instituted between the central control room and the liquid sample control panels 1 and 2, permission could be granted for operators to activate specific valve combinations on these panels. This would permit selective use of the inline isotopic analyzer and associated high radiation sampling system liquid sampling panel.

9.4.3.2 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 containment fan coolers. Leakage of radioactive material from the most likely places outside the containment is collected by running a ventilation line from the high radiation sampling system panel to an existing exhaust duct in the old sampling room. This duct has a diffuser with a damper. During normal operation, air from the room is taken in through the diffuser; during accident conditions the damper is closed and air is taken into the ventilation system only from the high radiation sampling system panel ventilation. The gases from the panel pass through a pre-filter, a HEPA filter, a 450 cfm exhaust fan, and then into the existing ventilation system, which contains a charcoal filter. This system is seismic Class I. Liquid leakage from the sentry liquid sampling panel, chemical analysis panel, and boron analyzer valves within the common vented system is drained to the liquid sampling panel sump and pumped to either the chemical drain tank 21 or the high radiation sampling system collection tank.

9.4.3.3 Incident Control

The system operates on a continuous basis for isotopic analysis, conductivity, dissolved oxygen, pH, and during steam-generator blowdown sampling. The inline dissolved hydrogen, chloride and boron concentrations can be obtained periodically from the sentry high radiation sampling system room.

9.4.3.4 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.

9.4.3.5 High Radiation Sampling System Evaluation

The high radiation sampling system is an independent system to provide information to plant operators. It is separate from other safety and non-safety systems. It is located in an area served by the primary auxiliary building ventilation system.

The high radiation sampling system has the capability of handling both low and high radiation sampling without exceeding personnel exposure guidelines. Sufficient shielding is provided on the high radiation sampling system liquid sampling panel to allow personnel access for postaccident sampling.

REFERENCES FOR SECTION 9.4

1. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Postaccident Sampling at the Indian Point Unit 2, Safety Evaluation Report, dated June 28, 1984.

2. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Postaccident Sampling at the Indian Point Unit 2, Safety Evaluation Report, dated December 12, 1984.

3. Letter from P.D Milano, NRC to M.R. Kansler, Entergy, Subject: Indian Point Nuclear Generating Unit No. 2 – Amendment Re: Deletion of Technical Specifications for the Post Accident Sampling System (PASS) using the Consolidated Line Item Improvement Process (TAC No. MB2991). Dated January 30, 2002.

TABLE 9.4-1
Sampling System Code Requirements

	Code
Sample heat exchanger	ASME III,[1] Class C, tube side ASME VIII, shell side
Piping and valves	USAS B31.1[2]

Notes:
1. ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.
2. USAS B31.1 - Code for pressure piping and special nuclear cases where applicable.

TABLE 9.4-2
Primary 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
Shell	
Design pressure, psig	150
Design temperature, °F	350
Component cooling water flow (nominal), gpm	17
Flow, lb/hr	20,000
Component cooling water	
inlet temperature, °F	105
outlet temperature, °F	130
Material	Carbon steel
Tubes	
Tube diameter in., O.D.	3/8
Design pressure, psig	2485
Design temperature, °F	680
Flow, lb/hr	209
Inlet temperature (saturated steam), °F	653
Outlet temperature, °F	127
Material	Austenitic stainless steel

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.	Remotely operated sampling valve inside reactor containment fails to close.	Diaphragm or motor-operated valve outside the reactor containment closes automatically on containment isolation signal or by operator action from the control room.
Any sample train.	Sample line break inside containment.	Same as above.

9.4 FIGURES

Figure No.	Title
Figure 9.4-1 Sh. 1	Primary Sampling System - Flow Diagram, Sheet 1, Replaced with Plant Drawing 9321-2745
Figure 9.4-1 Sh. 2	Primary Sampling System - Flow Diagram, Sheet 2, Replaced with Plant Drawing 227178
Figure 9.4-2	Secondary Sampling System - Flow Diagram, Replaced with Plant Drawing 9321-7020

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 is designed to minimize the possibility of mishandling or maloperations that could cause fuel damage and potential fission product release.

The fuel handling system consists basically of:

1. The reactor cavity, which is flooded only during plant shutdown for refueling.
2. The spent fuel pit, which is kept full of water and is always accessible to operating personnel.
3. 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.

9.5.1 Design Basis

9.5.1.1 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)

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. Periodic checks of refueling water boron concentration ensure the proper shutdown margin. The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The new fuel racks and spent fuel storage pit have accommodations as defined in Table 9.5-1. In addition, the spent fuel pit has the required spent fuel shipping area. The spent fuel storage pit is filled with borated water at a concentration to match that used in the reactor cavity and refueling canal during refueling. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to assure K_{eff} <1.0 even if unborated water was used to fill the pit and $\leq$0.95 when filled with water borated $\geq$ 2000 ppm boron. Limits on enrichment and burnup of fuel in the spent fuel storage pit are given in the Technical Specifications.

Both IP2 and IP3 irradiated fuel assemblies may be handled and stored in the IP2 spent fuel storage pit. The above stated spent fuel storage requirements are being applied to both IP2 and IP3 irradiated fuel assemblies in the IP2 spent fuel storage pit. Detailed instructions are available for use by personnel handling irradiated fuel assemblies. 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 the irradiated fuel handling operations that would result in a hazard to public health and safety.

In lieu of maintaining a monitoring system capable of detecting a criticality as described in 10CFR70.24, IP2 has chosen to comply with the seven criteria of 10CFR50.68(b).

9.5.1.2 <u>Fuel and Waste Storage Decay Heat</u>

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 of the public. (GDC 67)

The refueling water provides a reliable and adequate cooling medium for spent fuel transfer and heat removal from the spent fuel pit. Overall this is provided by an auxiliary cooling system. Natural radiation and convection is adequate for cooling the holdup tanks.

9.5.1.3 <u>Fuel and Waste Storage Radiation Shielding</u>

Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities. (GDC 68)

Adequate shielding for radiation protection is provided during reactor refueling by conducting all spent fuel transfer and storage operations underwater. This permits visual control of the operation at all times while maintaining radiation levels as low as reasonably achievable for the period of occupancy of the area by operating personnel. Pit water level is indicated, and water removed from the pit must be pumped out since there are no gravity drains. Shielding is provided for waste handling and storage facilities to permit operation within requirements of 10 CFR 20.

Gamma radiation is continuously monitored in the auxiliary building. A high level signal is alarmed locally and is annunciated in the control room.

9.5.1.4 Protection Against Radioactivity Release from Spent Fuel and Waste Storage

Criterion: Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity. (GDC 69)

All fuel and waste storage facilities are contained and equipment designed so that accidental releases of radioactivity directly to the atmosphere are monitored and do not exceed the applicable limits.

The reactor cavity, refueling canal and spent fuel storage pit are reinforced concrete structures with a seam-welded stainless steel plate liner. These structures are designed to withstand the anticipated earthquake loadings as seismic Class I structures so that the liner prevents leakage even in the event the reinforced concrete develops cracks.

All vessels in the waste disposal system, which are used for waste storage are designed as seismic Class I equipment.

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 generally 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 a fuel transfer tube, which penetrates the plant containment.

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 system 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.

New fuel assemblies are received and stored in racks in the new fuel storage area. New fuel is delivered to the reactor by lowering it into the spent fuel pit and taking it through the transfer system. The new fuel storage area is sized for storage of the fuel assemblies and inserts normally associated with the replacement of one-third of a core.

9.5.2.1 Major Structures Required for Fuel Handling

9.5.2.1.1 Reactor Cavity

The reactor cavity is a reinforced concrete structure that forms a pool above the reactor when it is filled with borated water for refueling. The cavity is filled to a depth that limits the radiation at the surface of the water during fuel assembly transfer.

The reactor vessel flange is sealed to the bottom of the reactor cavity by a Presray seal, which prevents leakage of refueling water from the cavity. This seal is installed after reactor cooldown but prior to flooding the cavity for refueling operations. Following refueling operations and prior to return to power, this seal is removed. 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.

9.5.2.1.2 Refueling Canal

The refueling canal is a passageway extending from the reactor cavity to the inside surface of the reactor containment. The canal is formed by two concrete shielding walls, which extend upward to the same elevation as the reactor cavity. The floor of the canal is at a lower elevation than the reactor cavity to provide the greater depth required for the fuel transfer system 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. Canal wall and floor linings are similar to those for the reactor cavity.

9.5.2.1.3 Refueling Water Storage Tank

The normal duty of the refueling water storage tank is to supply borated water to the refueling canal and reactor cavity 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 minimum volume of water and the minimum amount of boration of the water in the refueling water storage tank is defined in the Technical Specifications. Heating is provided to maintain the temperature above freezing. The tank design parameters are given in Chapter 6.

9.5.2.1.4 Spent Fuel Storage Pit

The spent fuel storage pit is designed for the underwater storage of spent fuel assemblies, failed fuel cans if required, control rods and other non-fuel hardware inserts after their removal from both the Unit 2 reactor and the Unit 3 reactor.

The pit accommodations are listed in Table 9.5-1.

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 and is seismic Class I design. This structure was analyzed to determine compliance with ACI-318(77), and SRP 3.8 of NUREG-0800. In addition to the mechanical loadings, the pool structure was also analyzed to include the temperature induced loadings. For this purpose, the thermal boundary conditions were conservatively specified as 180°F pool water temperature and 0°F outside ambient. The thermal moments computed by the finite element analyses were combined with those due to mechanical loads. The results of these analyses show that there are large margins between the factored loads and corresponding design strengths.

The pit is lined with a leak-proof stainless steel liner. All welds were vacuum-box tested during construction to assure a leaktight membrane. The effect of a thermal gradient would be to compress the liner. A review of the stress factors resulting from the finite element analyses demonstrates that an adequate design margin exists for the spent fuel pit liner walls and basemat.

Storage racks are provided to hold spent fuel assemblies and are erected on the pit floor. Fuel assemblies are held in a square array, and placed in vertical cells. Fuel inserts are stored in place inside the spent fuel assemblies from both Units 2 and 3.

9.5.2.1.5 Storage Rack

High density fuel storage racks have been designed to provide a maximum storage capacity of 1374 locations. The arrangement of the fuel storage racks in the spent fuel storage pool is shown in Figure 9.5-2.

The fuel storage rack arrangement contains two types of storage rack arrays.

Region 1, consisting of three racks that use the flux trap design, can store 269 new or irradiated fuel assemblies. The flux trap design used in Region 1 uses spacer plates in the axial direction to separate the cells. Boraflex absorber panels are held in place adjacent to each side of the cell by picture-frame sheathing. The spacer plates between cells form a flux trap between the boraflex absorber panels. Region 1 racks were originally designed to store new fuel with enrichments up to 5.0 w/o U^{235}. Region 1 is subdivided into two regions (Region 1-1 and Region 1-2):

Region 1-1 is assumed to have sustained a 100% loss of Boraflex (i.e., none of the boraflex in the panels is assumed to be available). Technical Specifications show the fuel assembly criteria that will meet the requirements of 10 CFR 50.68(b)(4) if stored in Region 1-1. The maximum initial enrichment that can be stored in Region 1-1 with no burnup is 1.95 w/o U^{235}.

Region 1-2 is assumed to have sustained a 50% loss of Boraflex (i.e., 50% of the boraflex in the panels is assumed to be available). Region 1-2 can accommodate unirradiated fuel up to 5.0 w/o U^{235} assuming the presence of a minimum number of IFBA rods. The maximum initial enrichment that can be stored in Region 1-2 when there are no IFBA rods is 4.50 w/o U^{235}.

Each Region I storage cell, as shown in Figure 9.5-3, is a square box with an 8.75 inch inside dimension. Boraflex poison is held in place adjacent to each side of the box by "picture-frame" sheathing. The boxes are assembled into racks with an east-west pitch of 10.765 inches (center-to-center) and a north-south pitch of 10.545 inches, as shown in Figure 9.5-4. A 1/2 inch thick base plate is provided at the bottom of the rack. Adjustable leg supports are welded to the underside of the base plate. A six-inch diameter flow hole is provided in the base plate for each storage cell, and two one-inch holes are provided for cross flow at the bottom of each cell.

Region 2, consisting of nine racks that use the egg-crate design, can store 1105 fuel assemblies and two failed fuel canisters. Region 2 racks consist of boxes welded into a “checkerboard” array with a storage location in each square. One Boraflex absorber panel is held to one side of each cell wall by picture frame sheathing. Region 2 racks were originally designed to store fuel assemblies that have undergone significant burnup (e.g., $\leq$5.0 w/o U^{235} with a burnup of at least

40,900 megawatt days per metric ton (MWD/MT)) or fuel assemblies with a relatively low initial enrichment and low burnup (i.e., ≤1.764 w/o U^{235} at zero burnup).

Region 2 is subdivided into two regions (Region 2-1 and Region 2-2):

Region 2-1 is assumed to have sustained a 100% loss of Boraflex (i.e., none of the boraflex in the panels is assumed to be available). The maximum initial enrichment that can be stored in Region 2-1 with no burnup is 1.06 w/o U^{235}.

Region 2-2 is assumed to have sustained only a 30% loss of Boraflex (i.e., 70% of the boraflex in the panels is assumed to be available).

“Peripheral” Cells, consisting of six select cells along the SFP west wall in Region 2-2, may be used to store fuel that meets the requirements for storage in any other location in the SFP. Cells between and adjacent to the “peripheral” cells may be filled with fuel assemblies that meet the requirements for storage in Region 2-2). The two prematurely discharged fuel assemblies meet the requirements and qualify for storage in the “peripheral” cells.

The storage racks are positioned on the floor so that adequate clearances are provided between racks and between the rack and pool structure to avoid impacting of the sliding racks during seismic events. The horizontal seismic loads transmitted from the rack structure to the pool floor are only those associated with friction between the rack structure and the pool liner. The vertical deadweight and seismic loads are transmitted directly to the pool floor by the support feet.

9.5.2.1.6 New Fuel Storage

New fuel assemblies and control rods are stored in a separate area with a location that facilitates the unloading of new fuel assemblies or control rods from trucks. This storage vault is designed to hold new fuel assemblies in specially constructed racks and is utilized primarily for the storage of the replacement fuel assemblies.

Criticality analyses have been performed assuming the fully loaded racks are flooded with water. The analyses demonstrated that K_{eff} is less than 0.95 for fuel with Integral Fuel Burnable Absorbers (IFBA) and enrichments in the range 4.5 w/o to 5.0 w/o. K_{eff} is also less than 0.95 for fuel enriched to 4.5% or less with no absorbers.

9.5.2.2 Major Equipment Required for Fuel Handling

9.5.2.2.1 Reactor Vessel Stud Tensioner

Stud tensioners are used to make up the head closure joint and during this process all studs are stretch tested to more than nominal working loads at every refueling.

The stud tensioner is a hydraulically-operated (oil as the working fluid) device provided to permit preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. A stud tensioner was chosen in order to minimize the time required for the tensioning or unloading operations. Three tensioners are provided and they are normally applied simultaneously to three studs 120° apart. One hydraulic pumping unit operates the tensioners, which are hydraulically connected in parallel. The studs are tensioned to their operational load in a number of steps to prevent high stresses in the flange region and unequal loadings in the studs.

A relief addition, micrometers are provided to measure the elongation of the studs after tensioning.

9.5.2.2.2 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. The lifting device is permanently attached to the reactor vessel head.

9.5.2.2.3 Reactor Internals Lifting Device

The reactor internals lifting device is a fixture providing the means to grip the top of the reactor internals package 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 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.

9.5.2.2.4 Manipulator Crane

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 out of the mast to grip the fuel assembly. The gripper tube is long enough so 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.

Controls for the manipulator crane are located inside the control console mounted on the trolley platform. Bridge, trolley and hoist positions are electronically displayed via encoders on the control console. The drives for the bridge, trolley and hoist are variable speed. Crane interlocks and limit switches are monitored by a Programmable Logic Controller (PLC). In an emergency the bridge trolley and hoist can be operated manually.

An electronic load cell located on the trolley platform monitors the suspended weight on the gripper tool. This load cell sends a low voltage signal to a PLC and to a display located on the control console. This load is electronically displayed on the control console. An overload condition stops the hoist drive from moving in the up direction. The gripper is interlocked through a weight-sensing device and also a mechanical spring lock so that it cannot be opened when supporting a fuel assembly.

Safety features are incorporated in the system as follows:

1. Encoders provide feedback pertaining to the bridge, trolley and hoist positions. Bridge, trolley, and hoist positions are displayed to the operator on the control console.

2. Only the bridge and trolley are allowed to simultaneously operate at the same time. Bridge and trolley motion will be prohibited if hoist is in motion. Likewise, hoist motion will be prohibited if the bridge and trolley are already in motion.

3. Encoders determine the position of the mast, which will prohibit bridge and trolley movement based on the gripper height. The hoist also has a mechanical limit switch serving as a redundant mast “full up” limit.

4. A mechanical weight actuated lock in the gripper prevents operation of the gripper under load even if air pressure is applied to the operating cylinder. As backup protection to the mechanical interlock, an electrical interlock prevents the opening of a solenoid valve in the air line to the gripper except when the gripper is unloaded as indicated by a load cell.

5. Hoist load monitoring components detect overload conditions which will prohibit hoist raise motion when loading is excessive.

6. The PLC monitors the status of the gripper selector switch. Crane motion will not be allowed if the gripper indicator shows that the gripper is in transition or both conditions are activated (between OPEN and CLOSED).

7. The systems encoders along with the Crane’s PLC will establish a boundary zone within the pool area. Crane motion is prohibited through these established boundary zones unless the bypass mode has been selected. Motion speeds will be decreased when operating in the bypass mode.

8. When the gripper is loaded with an assembly the mast must be in the full up position before bridge and trolley motion are allowed. With an empty gripper, bridge and trolley motion are prohibited until the “Gripper in Mast” elevation is present (full up is not required to traverse with an empty gripper).

9. Hoist load monitoring components detect underload conditions which will prohibit hoist lower motion. This prevents continued hoist motion if an assembly is hung up while being inserted between other fuel assemblies.

10. An encoder positioning system displays to the operator the precise position of the manipulator crane over each row of core coordinates for bridge, trolley and hoist movement over the reactor and the transfer canal.

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 design basis earthquake.

9.5.2.2.5 <u>FSB Fuel Handling Bridge Crane</u>

The PaR Systems, Inc. Crane is a wheel-mounted platform, which spans the East-West (E-W) direction of the Spent Fuel Pool (SFP) and travels in the North-South (N-S) direction. The PaR Crane is secured to the crane rails on the FSB El. 95’-0”, via seismic hold-down brackets and associated bolting. An Encoder Tracking Device is mounted on the FSB West Walkway, El. 96’-6”, which positions the crane in the N-S direction. The crane mounted computer will position the Crane Trolley-Tower Structure in the E-W direction. This equipment will position the Crane Motorized Hoist over a pre-assigned Spent Fuel Assembly (SFA) within the SFP. In addition, the PaR Crane controls interface with the existing FSB Up-Ender Control Console No. 21 (PK1).

This computerized control feature provides assurance that the PaR Crane will not interfere with the FSB Up-Ender Assembly, which is located in the Fuel Transfer Canal.

The Motorized Hoist-Sheave Assembly is attached to a Trolley Structure, which is located on the wheel-mounted platform. The Motorized Hoist design incorporates a single lifting cable, which has a safety factor of 11.49:1. This safety factor exceeds the design criteria (10:1) for single lifting cables, as outlined in NUREG-0612. The Tower Structure is mounted on a Motorized Trolley, which travels in the E-W direction on the wheel-mounted work platform. The Motorized Hoist-Sheave Assembly, which has a 1-Ton rated capacity, will transfer SFAs within the SFP via long-handled tools suspended from the hoist hook. The hoist travel and tool length are designed to limit the maximum lift of a SFA and maintain a safe shield depth below the water surface of the SFP. A load weighing system will sense overload and underload conditions. This system will stop the upward movement of a SFA when it senses a load greater than a programmed set-point. In addition, this system will stop the downward movement of a SFA when it senses a slack cable condition.

A 480V, 3-phase, 50 AMP power feed (normal supply) is provided from Distribution Panel No. EP57 to the PaR Crane. In addition, a 480V, 3-phase, 100 AMP power feed (alternate supply) is provided from MCC27 to the PaR Crane. Transfer Switch No. EDA57 is provided so that the reliable power feeds can be provided by Distribution Panel No. EP57 or MCC27.

9.5.2.2.6 Fuel Transfer System

The fuel transfer system, shown in Figure 9.5-1, is a cable driven system that traverses the conveyor car carriage on tracks extending from the refueling canal through the transfer tube and into the spent fuel pit. The conveyor car receives a fuel assembly in the vertical position from the manipulator crane. The fuel assembly is then 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 inside the containment. A blank flange is bolted on the transfer tube on the reactor side and a gate valve closed on the spent fuel pit side (see Figure 5.2-5) to seal the reactor containment. The blind flange is supplied with a double o-ring seal and is pressurized by the WCCPP System during normal operation to assure containment isolation.

9.5.2.2.7 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.

9.5.2.2.8 Lower Internals Support Stand

A support stand for the lower internals package is installed in the lower internals laydown area at the east end of the refueling canal. The stand is to be used to rest the lower internals package to facilitate access to the internal surfaces of the reactor vessel.

9.5.2.2.9 Shield Transfer Canister (STC) and HI-TRAC Transfer Cask

The NRC has issued Amendment 268 for the inter-unit transfer of spent fuel from Unit 3 to Unit 2 (Reference 1). The Amendment is based on evaluations conducted for each aspect of the inter-unit transfer of fuel as documented in the Licensing Report (Reference 2). The non-proprietary version of the Licensing Report is “incorporated by reference” in 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 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 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 the 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.

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 a reliable cooling medium for removal of decay heat.

Basic provisions to ensure the safety of refueling operations are:

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

2. Violation of containment integrity is not permitted when the reactor vessel head is removed unless the shutdown margin is maintained greater than 5-percent Δk/k.

3. Whenever fuel is added to the reactor core, a reciprocal curve of source neutron multiplication is recorded to verify the sub-criticality of the core.

4. A Boraflex surveillance program was established when the high density racks utilizing Boraflex were installed. This program now includes coupon surveillance and monitoring of silica level (which is indicative of Boraflex degradation) in the spent fuel pit water.

9.5.3.1 Incident Protection

Direct communication between the control room and the refueling cavity manipulator crane operator 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.

This provision shall be satisfied with fuel in the reactor and when:

1) the reactor head is being moved, or
2) the upper internals are being moved, or
3) loading and unloading fuel from the reactor, or
4) heavy loads greater than 2300 lbs (except for installed crane systems) are being moved over the reactor with the reactor vessel head removed.

If direct communication between the control room and the refueling cavity manipulator cannot be met, suspend any and all of these operations. Suspension of these operations shall not preclude completion of movement of the above components to a safe conservative position.

9.5.3.2 Malfunction Analysis

Various potential failures, which could create paths for drainage from the refueling cavity have been considered. A plant procedure defines actions to deal with these postulated events. All credible failures result in drainage to safe storage. An analysis evaluating the environmental consequences of a fuel handling incident is presented in Section 14.2.1.1.

Inadvertently locating an unirradiated fuel assembly of 5.0-percent enrichment in a region II storage location has been analyzed. The analysis shows that the array would be subcritical even with no soluble boron poison in the water in the fuel storage pool. With a boron concentration of 350 ppm the shutdown margin would be more than 5-percent. The technical

specifications require that the boron concentration be maintained at 2000 ppm or more at all times.

9.5.4 Minimum Operating Conditions

Minimum operating conditions are specified in the facility Technical Specifications. In addition, when fuel is in the reactor vessel and the reactor head bolts are less than fully tensioned the reactor Tavg shall be less than or equal to 140°F.

9.5.5 Tests and Inspections

During preoperational testing, the Presray seal (which seals the reactor vessel flange to the bottom of the reactor cavity) was deflated with a full head of water in the cavity. No leakage was observed.

9.5.6 Control of Heavy Loads

9.5.6.1 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 IP2 provisions for handling and control of heavy loads at Indian Point Unit 2 were addressed by letters June 22, 1981, September 30, 1982, January 31, 1983, and January 20, 1984. The NRC Safety Evaluation Report in letter dated February 19, 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 2 is acceptable.

The NRC Safety Evaluation Report in letter dated November 21, 2005 authorized the use of a single-failure-proof gantry crane for spent fuel cask handling operations up to 110 tons in weight.

Additional information was provided in letter dated July 12, 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 2. This supersedes prior head drop analyses.

9.5.6.2 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. The Ederer crane for cask handling was designed as a single failure proof crane.

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) Monorail / Plant Auxiliary Building (PAB) Monorail

The AFP Monorail 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 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 one area, the annulus region between the crane wall and the containment, load handling reliability can not be demonstrated for the Equipment Hatch. This is the area over which the containment equipment hatch door/airlock is carried. The hatch door weighs approximately 25tons. Accordingly, a systems evaluation of the potential consequences of a drop of the equipment hatch door into region demonstrates successful core cooling.

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.

9.5.6.3 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 (175/35-ton)
- 7-ton Plant Auxiliary Building (PAB) monorail
- 5-ton Auxiliary Fuel Pump (AFP) building monorail
- 2-ton diesel generator building overhead crane
- Fuel Handling crane (40/5-ton)
- 110t Ederer crane (110-ton)

9.5.6.4 Control of Heavy Loads Program

The following discuss the results of our evaluations and submittals and are controlled using commitments A-873, A-887, A-1010, A-1015, A-1207, A-2467, A-2491, A-2492, A-2493, A-3174, A-3175, A-3176, A-3179, A-3180 and A-3465.

9.5.6.4.1 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 Handling Crane is shown in Table 9.5-2.

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-0612, 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

For the Indian Point Unit 2 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. WHEN RV head is removed and fuel is in the Reactor Vessel, THEN NO heavy loads shall be moved over the RV area, with the exception of the RV head, the Upper Internals, the RV weld ISI inspection tool and their associated lifting devices.
2. WHEN RV head is removed and fuel is in the RV, THEN NO heavy loads shall be moved over the Reactor Cavity Area, with the exception of the RV head, the Upper Internals, the RV weld ISI inspection tool, the Polar Crane Load Block, the Regenerative Heat Exchanger Concrete Floor Plug, the auxiliary bridge, and their associated lifting devices.
3. WHEN fuel is in the RV, THEN NO loads including the Polar Crane Load Block shall be moved over the #22 RHR HX area.

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.

Since the geometry of the Indian point reactor is the same as that analyzed in NUREG-0612 therefore the maximum expected increase in Keff would be about 0.02 and a criticality condition will not occur as a result of fuel crushing. Thus, criterion II of NUREG-0612, Section 5.1 is satisfied.

Auxiliary Hoist of the Polar Crane

The Polar Crane Auxiliary Hoist has a capacity of 35 tons and has a hook travel that can service the Annulus Region between the containment wall and the crane wall. For the purpose of addressing the NUREG 0612 guidelines for this region of the containment, the load handling reliability of the Auxiliary Hoist has been evaluated against the criteria of Section 5.1.6. , Based on the discussion below, adequate load handling reliability of the Auxiliary Hoist in the Annulus Region is demonstrated and, therefore, with one exception, load drops into this region have not been postulated.

The auxiliary hoist is mounted on the trolley frame and fully satisfies the criteria in CMAA-70-1975 and ANSI B30.2-1976. For most load handling operations, the auxiliary hoist satisfies the intent of Section 5.1.6 of NUREG 0612 (i.e., dual load path or increased safety factors of 10:1 in lieu of normal 5: 1).

One load carried over the open reactor vessel that could potentially damage fuel in the vessel, is the Reactor Vessel Weld ISI tool (5 tons). For this particular lift, by the Auxiliary Hoist, adequate load handling reliability is assured because the hoist was designed to and fully satisfies current industry standards and was built with a safety factor of 5:1. Therefore, the safety factor for the load of concern is far greater than 10:1. Adequate load handling reliability will be assured on the same basis as for loads lifted by the Auxiliary Hoist in the Annulus Region.

The auxiliary hoist components are designed with a 5:1 design safety factor on ultimate strength. For loads of less than 17.5 tons, the design safety factor for the hoist will be better than 10:1. With the exception of the containment equipment hatch door/airlock, all loads typically carried in the Annulus Region are less than 17.5 tons. The equipment hatch door weighs approximately 25 tons. For this reason an evaluation of the consequences of a

postulated drop of this door into the Annulus Region has been performed. This is the area over which the containment equipment hatch door/airlock is carried.

The very conservative assumption that all equipment within the region is lost was made for evaluating the consequences of a load drop for the equipment hatch door. Systems analyses demonstrate that the consequences of this load drop will not preclude the ability to maintain core cooling.

Fuel Storage Building Ederer Crane

The Ederer 110-ton design rated gantry crane is used to move spent fuel casks up to 110 tons into and out of the spent fuel pit by lifting a fully loaded Holtec HI-TRAC® 100 spent fuel transfer cask and its associated components. The HI-STORM® cask system utilizes the HI-TRAC® 100 transfer cask for transporting a multi-purpose canister (MPC) from the spent fuel pit, and for inter-cask MPC transfers required for on-site storage. However, this crane is restricted from handling casks over spent fuel in the spent fuel pit and will only be utilized for other loading activities in the FSB.

Safe load paths have been determined, analyzed and documented in procedures for control of heavy loads handled by the Ederer gantry crane. Deviations from the safe load paths will require written alternative procedures reviewed and approved in accordance with IP2 procedures.

The Ederer gantry crane (by design) is unable to move spent fuel casks over any area of the spent fuel pit where the spent fuel is stored.

Fuel Handling Crane

The Fuel handling Crane is utilized over the Spent Fuel Pit to move fresh fuel to the New Fuel Elevator. Also, it may be used to transport equipment, such as inspection rigs or electronics, to the Spent Fuel Pit area. For equipment handling, the crane is utilized to transport loads of no greater than 2000 lbs over the pool area. For fuel handling, the crane may carry a load no heavier than the weight of a fuel assembly containing a control rod assembly, plus the tool and small load block.

No object weighing more than 2,000 pounds may be moved over any region of the spent fuel pit when the pit contains spent 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 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.

All standard modes of failure have been considered in the design of the Fuel Handling 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

Mechanical stops incorporated on the bridge rails of the Fuel Handling Crane make it impossible for the bridge of the crane to travel further north than a point directly over the spot in the spent fuel pit that is reserved for the spent fuel cask. Therefore, 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 the 40 or 5-ton hook of the Fuel Handling Crane. However, to further minimize the potential for a heavy load impacting irradiated fuel in the spent fuel pit, load paths will be defined in procedures and shown on equipment layout drawings.

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 pit boron concentration is at least 2000 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 load over the SFP since the load block is a one-ton load and has not been fully evaluated for heavy loads.

The existing 40-ton non-single-failure-proof Fuel Handling Crane does not have the capacity to handle the HI-TRAC® 100 transfer cask and its associated components. Performance of the crane satisfies the objectives of NUREG-0612 and the intent of NUREG-0554 with regard to maintaining the potential for a load drop extremely small.

Guideline 2 - Load Handling Procedures

A series of operating procedures have been developed for operation of load handling equipment at Indian Point Unit 2.

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.

The 110T Ederer gantry crane operating procedures utilized for cask and cask component lifts include: identification of required equipment; inspection and acceptance criteria required before load movement; the steps and proper sequence to be followed in handling the load; defining the safe load path; and other precautions. A specific cask loading and handling procedure will provide additional details for controlled movement during cask handling operations.

Guideline 3 – Crane Operator Training

A qualification program for the qualification and training of crane operators at Indian Point Unit 2 have been developed to meet the provisions of ANSI B30.2-1976, 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.

This qualification program meets the requirements of Chapter 2-3 of ANSI B 30.2-1967, "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 9.5.2.2)
- internals lift rig (also described in Section 9.5.2.2)
- 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. The Licensee stated that further 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, the Licensee stated that 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). Inspection and NDE on the devices is performed at extended intervals (5 years) since annual inspection is impractical; these extended intervals are justified on the basis of the limited frequency of use and the controlled storage and handling of these devices.

The HI-TRAC® lifting yoke used with the Ederer crane is the only special lifting device that is required to meet the guidelines of ANSI N14.6-1993 and the additional guidelines of NUREG-0612, Section 5.1.6(1)(a).

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.

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.

Other lift components utilized with the Ederer Crane and HI-STORM® 100 cask system meet ANSI B30.9-1971 requirements, including the additional guidelines of NUREG-0612, Section 5.1.6(1)(b).

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.

The 110T Ederer gantry crane is inspected, tested and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976 and the additional guidance contained in NUREG-0612, Section5.1.1(6) regarding frequency of inspections and test.

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 and the 40ton Fuel Handling Crane were 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 both cranes comply 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 9.5.7 Fuel Storage Building (FSB) Dry Cask Storage (DCS) Operations, provides more detail on the FSB 110-Ton Ederer Single Failure Proof Gantry Crane and section 9.5.2.2.5 provides more detail on the FSB Fuel Handling Bridge crane.

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.

The 110t Ederer gantry crane is installed on a crane rail system. The crane rail system for the Ederer crane consists of crane rail, rail pad, rail clip, sole plate assembly, and sole plate anchor embedments. The sole plate assembly consists of 2" thick steel plate which is held to the concrete slab with 1" diameter rod anchor embedments. The crane rail is attached to the sole plate assembly by rail clips with a rail pad between the crane rail and the sole plate assembly. The crane rail and the concrete slab of the reconstructed truck bay are designed and built to withstand seismic loads, as well as the static loads.

The Ederer gantry crane was designed with a telescopic tower and automated folding cantilever arms to avoid interference with either the existing overhead crane or the refueling bridge crane. During dry cask loading operations the gantry crane will be in its raised position and the existing overhead crane will remain in the south position and de-energized to prevent accidental movement. Once the cask loading operation is completed, the gantry crane will be stored in its far west position, with the tower lowered and the arms folded. This will allow unobstructed use of both the existing overhead and refueling bridge cranes.

The cantilevered girder for the main hoist trolley will extend over the spent fuel pit cask laydown area. The girders are equipped with a retraction mechanism, accomplished via lead screw actuators that allow them to be folded back in order to permit unobstructed use of the existing overhead and refueling bridge cranes. Because of the cantilevered design, the gantry crane requires provisions to ensure stability against overturning. This is accomplished via a floor anchorage system with fixed-in-place hold down features that oppose crane uplift forces. To provide a foundation system capable of resisting these uplift forces, the design includes a steel ballast box filled with steel plates that will act as a counterbalance. The ballast box foundation consists of a 2-foot thick reinforced concrete slab founded on bedrock, and its primary function is to transmit all bearing loads from the weight of the ballast box directly to the underlying bedrock.

The Ederer gantry crane movements are governed by a series of limit and proximity switches that are controlled by a programmable logic controller (PLC) which ensures that: (1) movement of the trolley towards the spent fuel pit is only permitted if turnbuckles are attached to the crane tie down points, cantilever arms are extended and locked in place, and the main transfer hoist is operating at an elevation that allows the HI-TRAC® to clear the south wall of the spent fuel pit; (2) limit switches on the trolley rails limit excessive movement of the trolley to the north and prohibit lowering of the load until a minimum northward travel is reached; and (3) main transfer hoist operation is prohibited until the Ederer gantry crane tower is in its raised position and pinned in place.

9.5.6.4.2 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 9.5.6.2, Safety Basis.

9.5.6.4.3 Single Failure Proof Cranes for Spent Fuel Casks

Sections 9.5.6.4.1, Response to NUREG 0612, Phase I Elements and 9.5.7, Fuel Storage Building (FSB) Dry Cask Storage (DCS) Operations, provide more detail on the FSB 110-Ton Ederer Single Failure Proof Gantry Crane.

9.5.6.5 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.

The design and use of the Ederer single-failure-proof gantry crane is in accordance with NUREG-0554 and satisfies the guidelines of NUREG-0612. The crane enables the use of the HI- TRAC® transfer cask and associated components with very low risk to irradiated fuel stored in the spent fuel pit or to redundant trains of safe shutdown equipment during spent fuel transfer activities. The use of the Ederer single-failure-proof gantry crane for cask handling operations for loads up to 110 tons is approved.

9.5.7 Fuel Storage Building (FSB) Dry Cask Storage (DCS) Operations

The 100-Ton Dry Cask Storage System (HI-TRAC, Multi-Purpose Canister (MPC) and HI-STORM Overpack), FSB 110-Ton Single Failure Proof Gantry Crane, FSB Low Profile Transporter (LPT) System, and Vertical Cask Transporter (VCT) facilitate removal of Spent Fuel Assemblies (SFAs) from the Spent Fuel Pool (SFP). During FSB Dry Cask Storage Operations, SFAs are transferred from the SFP with the HI-TRAC / MPC, inserted into the HI-STORM Overpack. The LPT System transfers the HI-STORM Overpack from the FSB to the east side of the PAB / MOB Crossover Walkway. The Vertical Cask Transporter transports the HI-STORM Overpack to the IPEC Independent Spent Fuel Storage Installation (ISFSI) Facility.

9.5.7.1 FSB 110-Ton Ederer Single Failure Proof Gantry Crane

The 110-Ton Ederer Crane was designed to withstand normal operating loads, rated loads, seismic loads and extraordinary loads. The design of the 110-Ton Ederer Crane satisfies the design requirements and safety factors of CMAA-70, NUREG-0554 and Regulatory Guide 1.29. The 110-Ton Ederer Crane conforms to the single failure proof requirements addressed in NRC NUREG-0554 and will retain and control a suspended critical load during and following a Safe Shutdown Earthquake (SSE).

The 110-Ton Ederer Crane is normally located on the west side of the FSB Truck Bay Floor, EL 77'-6" in its lowered position and rests on rails that are installed within the FSB Truck Bay Floor. In order to lift and transfer loaded and unloaded spent fuel casks into and out of the SFP, the 110-Ton Ederer Crane is raised to its upper position. Once in the upper position, the East and West Crane Girder Assemblies, and the North Tie-End Crane Girder Assembly are fully-extended (cantilevered position) out over the SFP. The 110-Ton Ederer Crane is then in position for visually controlled lifting and transferring of loaded and unloaded spent fuel casks into and out of the SFP Cask Pit Area.

The 110-Ton Ederer Crane is connected, via tie-down turnbuckles, to the 30-Ton Ballast Box, which is embedded within the FSB Truck Bay Floor. The Ballast Box is filled with ~200-Tons of counter weight steel plates. The Ballast Box and tie-down turnbuckles provide stability and restraints for the 110-Ton Ederer Crane during a seismic event. The 110-Ton Ederer Crane is

provided with counter weight to reduce the uplift (tension) loads in the tie-down turnbuckles, when the Crane Trolley travels onto the East and West Crane Girder Assemblies towards the SFP Cask Pit Area.

The 110-Ton Ederer Crane is provided with numerous limit switches, which control movement of the 110-Ton Ederer Crane. The limit switches are mounted on the Trolley, Girder Assemblies, Jacking Screw System, and Gantry Crane Structure. The limit switches control the speed of the equipment, limit travel distances and provide interlock signals to defeat some crane functions, when the 110-Ton Ederer Crane is not in the correct configuration. In addition, a Seismic Accelerometer automatically de-energizes the power feed to the 110-Ton Ederer Crane during a seismic event.

9.5.7.2 FSB Low Profile Transporter (LPT) System

The FSB LPT System was designed to withstand normal operating loads, maximum vertical – lateral track loads, maximum static stack-up loads, maximum transit loads, and Safe Shutdown Earthquake (SSE) loads. The design of the FSB LPT System satisfies the design requirements and safety factors of AISC, IPEC and Hilman-Rollers.

The FSB Gantry Crane transports the fully-loaded HI-TRAC / MPC towards the east for stake-up onto the Hi-STORM Overpack. The LPT Assembly, Chain Drive Assembly and Chain Drive Control Panel control internal FSB movements in the East-West direction.

The FSB LPT System transports the fully-loaded HI-STORM Overpack towards the east and south, so that, the HI-STORM Overpack can exit the FSB. The LPT Transporter Assembly, Transfer Table, Hydraulic Cylinders, and Transfer Table Hydraulic Cart control internal FSB movements in the North-South direction.

The FSB LPT System transports the fully-loaded HI-STORM Overpack through the FSB Truck Bay Roll-Up Door Opening, into the FSB Alleyway Trench and to the east side of the IP2 PAB / MOB Crossover Walkway. The HI-STORM Overpack is in position to be lifted and transported by the Vertical Cask Transporter to the IPEC ISFSI Facility. The LPT Assembly, Track Assemblies, Guide Bars and Aircraft Tugger control external FSB movements in the East-West direction. Empty HI-STORM Overpacks will be transported into the FSB in the reverse order from above.

9.5.8 Inter-Unit Spent Fuel Transfer Operations

The NRC has issued Amendment 268 for the inter-unit transfer of spent fuel from Unit 3 to Unit 2 (Reference 1). The Amendment is based on evaluations conducted for each aspect of the inter-unit transfer of fuel as documented in the Licensing Report (Reference 2). The non-proprietary version of the Licensing Report is “incorporated by reference” into the UFSAR.

In preparation for inter-unit spent fuel transfer operations between the Spent Fuel Pool (SFP) in the IP3 Fuel Storage Building (FSB) and the SFP in the IP2 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 IP2 Low Profile Transporter (LPT), or using Air Pads at IP3.

THE VCT moves the HI-TRAC containing the empty STC outside the IP3 FSB truck bay door. The HI-TRAC is lowered onto Air Pads and the VCT releases the HI-TRAC. The IP3 FSB truck bay door is opened and the HI-TRAC is positioned inside the IP3 FSB truck bay beneath the FSB cask handling crane using the IP3 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.

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 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 testing 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 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 2000 ppm 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 in the SFP racks in accordance with the requirements of 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 on 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 are 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. 268 to Facility Operating License No. DPR-26, 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 System Data

NEW FUEL STORAGE PIT	
Core storage capacity	1/3
Equivalent fuel assemblies	72
Center-to-center spacing of assemblies, in.	20.5
Maximum K_{eff} with unborated water	0.95
SPENT FUEL STORAGE PIT	
Equivalent fuel assemblies[1]	1374
Number of space accommodations for failed fuel cans	2
Number of space accommodations for spent fuel shipping cask	1
Center-to-center spacing of Regions 1-1,1-2 assembly storage cells, in	10.545(N-S) 10.765(E-W)
Center-to-center spacing of Regions 2-1, 2-2 assembly storage cells, in	9.04
Maximum K_{eff} with borated water (Regions 1-1, 1-2 and Regions 2-1, 2-2)	$\leq$0.95
Maximum K_{eff} with unborated water (Regions 1-1, 1-2 and Regions 2-1, 2-2)	<1.0
MISCELLANEOUS DETAILS	
Width of refueling canal, ft	3
Wall thickness for spent fuel storage pit, ft	3 to 6
Weight of fuel assembly with rod cluster control (dry), lb	1,580
Quantity of water required for refueling, gal	300,000

Notes:
1. After re-racking.

TABLE 9.5-2
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	++	++
Reactor Vessel Head	169	R	C	++	R	++	++	++	++	++
Upper Internals (Plenum)	67	R	C	++	R	++	++	++	++	++
Inservice Inspection Tool	5	R	C	++	R	++	++	++	++	++
Reactor Coolant Pumps	32	R	C	++	R	C	++	++	++	++
Missile Shields	7.5	R	C	++	++	C	++	++	++	++
Crane Load Block	4.5	R	C	++	++	C	++	++	++	++
Concrete Hatch Cover	7.3	R	C	++	++	C	++	++	++	++
2. Fuel Handling Crane	40	R	R	C	++	C	C	R	R	++
3. 110t Ederer Crane	110	C	C	C	C	C	C	C	++	++

C - Action complies with NUREG-0612 Guideline.
R - Revisions/modifications designed to comply with NUREG-0612 Guideline.
++ - Not applicable.

9.5 FIGURES

9.6 FACILITY SERVICE SYSTEMS

9.6.1 Service Water System

9.6.1.1 Design Basis

The service water system is designed to supply cooling water from the Hudson River to various heat loads in both the primary and secondary portions of the plant. Provision is 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 and accident conditions. Sufficient redundancy of active and passive components is provided to ensure that cooling is maintained to vital loads for short and long periods. The design of the essential header is to provide cooling water in the event of a single failure of any active component used during the injection phase of a loss-of-coolant accident. The system also provides water required for cleaning the traveling screens.

9.6.1.2 System Design and Operation

The service water system flow diagram is shown in Plant Drawings 9321-2722 and 209762 [Formerly UFSAR Figure 9.6-1, sheets 1 and 2]. Six identical vertical, centrifugal sump-type pumps, each having a capacity of at least 5000 gpm at 220-ft total design head, supply service water to two independent discharge headers; each header may be supplied by three of the pumps. Two pumps are required for design flow in each header. A rotary-type strainer is in the discharge of each pump, and is designed to remove solids down to 1/16-in. diameter. 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 nonessential loads. The essential loads are those, which must have an assured supply of cooling water in the event of a loss of offsite power and/or a loss-of-coolant accident. The cooling water for these loads is supplied by the designated essential service water header. The nonessential loads are those, which are supplied with cooling water from the designated nonessential service water header by manually starting a service water pump when required following a loss-of-coolant accident. The essential and nonessential service water requirements are listed in Table 9.6-1.

The nonessential loads are the component cooling heat exchangers, the turbine lube oil coolers, the main boiler feed pump lube oil coolers, and the remaining steam generation plant services. By manual valve operation, the essential loads can be transferred to the supply line carrying the nonessential loads and vice versa. Connections have been provided so the turbine generator lube oil coolers and other non-safety related loads can be supplied from the Unit 1 river water system.

Water is drawn from the river and passes under a debris wall, through two racks in parallel and finally two traveling screens. Each pump in the circulating water system is installed in an individual chamber while the service water pumps are in a common chamber with two intakes. Each intake is provided with a traveling screen. Openings are also provided between the main circulating water pump chambers and the service water pump chamber. These two openings can be closed by gates. One gate is normally open.

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. Electric heaters are provided in the traveling screens 27 and 28 to prevent icing of the screens. Even if the main circulating pump intake were 90-percent blocked, that intake alone would be capable of supplying all water required for the service water pumps at design conditions.

Service water is chlorinated by the addition of sodium hypochlorite solution as required to control micro-organism fouling of the system.

The intake structure is designed as seismic Class I, and is therefore not subject to collapse under earthquake loading.

During normal operation, the essential loads are supplied by at least one of the three pumps provided and the nonessential loads are normally supplied by two of the three pumps provided.

Following a simultaneous incident and loss of offsite power, the cooling water requirements for all five fan cooling units and the other essential loads can be supplied by any two of the three service water pumps on the header designated to supply the nuclear and essential secondary load supply lines. Any two of these 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 redundant motor-operated shutoff valves and drain valves. Similarly, each discharge line from the cooler is provided with redundant motor-operated shutoff valves and a manual balancing valve. This allows each cooler to be isolated individually for leak testing of the system or to be drained and maintained open to atmosphere during the integrated leak tests of containment. The ventilation cooler and motor cooler discharge lines will be monitored for radioactivity by routing a small bypass flow from each through redundant radiation monitors. Upon indication of radioactivity in the effluent, each cooler discharge line would be monitored individually to locate the defective cooling coil. This feature has been incorporated into the design since the service water system pressure at locations inside the containment with the system in the incident mode alignment could be below the containment post-accident design pressure of 47 psig. Thus, there could be outleakage of radioactivity to the environment if a break occurred in the service water system. However, since the cooling coils and service water lines are completely closed inside the containment, no contaminated leakage is expected into these units. The service water system pressure at locations inside the containment with the system in the incident mode alignment is below the containment design pressure of 47 psig.

During normal plant operation, flow through the cooling units will normally be throttled for containment temperature control purposes by a valve on the common discharge header from the cooling units. Two independent, full-flow isolation valves open automatically in the event of a safety injection 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 for accident mitigation. Should there be a failure in the piping or valves at the header supplying water to the containment cooling coils, one of the two series header isolation valves in the center of the header can be manually closed and service will continue on the side of the header opposite the failure. The supply line attached to this side of the header now supplies the essential loads, whether or not it did so before the failure.

Likewise, operation of at least one component cooling heat exchanger is ensured 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 blackout, the component cooling heat exchangers are not needed during the injection phase: thus they are normally fed from the nonessential supply header. At the beginning of the recirculation phase at least one of the service water pumps on the nonessential header is manually started to supply at least 2400 gpm of service water to each of the component cooling heat exchangers.

The emergency diesel-driven generator units are supplied with cooling water from the essential supply line on a continuous basis. One of the two parallel modulating control valves in the common discharge line from the diesel coolers is flow-controlled during normal operation, and on a safety injection signal, both valves open fully to ensure a sufficient supply of cooling water to the diesels. The inlet valving is arranged so that each of the three diesels can be served by either of the supply headers and, furthermore, the failure of a single passive or active component will not result in the loss of all diesel power.

After an approximately eight (8) hour Appendix R event, cooling water supply to the Appendix R Diesel Generator Heat Exchanger is realigned from city water to service water.

9.6.1.3 Design Evaluation

The nonessential portion of the service water system is not required for the maintenance of plant safety immediately following an accident. The essential portion of the service water system is designed to provide cooling water in the event of a single failure of any active component used during the injection phase of the safety injection system (Section 6.2).

Sufficient pump capacity is included to provide design service water flow under all conditions and the headers are arranged in such a way that even loss of a complete header does not jeopardize plant safety.

In response the NRC Generic Letter 96-06, the containment fan cooler units and their associated service water piping were evaluated for susceptibility to waterhammer or two-phase flow. In the event of a loss of offsite power, the flow of essential service water will be interrupted until the emergency diesel generators start and restore power to the essential service water pumps. The pressure in the cooling coils and service water piping will drop to subatmospheric and a vapor pocket will form in the region of the fan coolers. When the essential service water pumps restart, the pocket will close and a water hammer will occur. The magnitude of

waterhammer is approximately 394 psig. Dynamic analysis of the piping and supports shows that stresses meet the criteria for upset and faulted conditions, respectively.

In the case of loss of offsite power and a loss of coolant accident, water trapped in the tubes and piping will be heated and vaporized. When the service water pumps are restarted, rapid condensation of trapped steam and collapsing of the void causes a waterhammer pressure pulse, with a magnitude less than that discussed in the preceding paragraph.

The potential for two-phase flow conditions has also been evaluated. If it is assumed that there is no fouling of the fan cooler tubes, there will be flashing and two-phase flow in the discharge piping. However, analyses show that, although the flow will be reduced, the clean fan cooler units will exchange enough heat to meet required removal rates.

9.6.1.4 Tests and Inspections

Each service water pump underwent a hydrostatic test in the shop in which all wetted parts were subjected to a hydrostatic pressure of one-and-one-half times the shutoff head of the pump. In addition, the normal capacity versus head tests were made on each pump.

Valves in the portions of the service water system essential to safety underwent a shop hydrostatic test of 250 psi on the body and 175 psi on the seat. The service water system design pressure is 150 psig.

All service water piping was hydrostatically tested in the field at 225 psig or one-and-one-half times design. The welds in shop-fabricated service water piping were liquid penetrant or magnetic particle inspected in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII.

Electrical components of the service water system are tested periodically.

9.6.2 Fire Protection System

Criterion: Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and the control room. Fire detection and protection systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components. (GDC 3, Appendix A to 10 CFR 50)

This criterion (GDC 3, 10 CFR 50 Appendix A) represents a revised design basis for the Indian Point Unit 2 fire protection system as was established for the original plant design and initial license application. In 1976 at the request of the NRC, Con Edison initiated a review and evaluation of the station fire protection system to this new criterion; modifications were subsequently proposed by Con Edison to the overall fire protection program. On January 31, 1979, the NRC approved the Indian Point Unit 2 overall fire protection program as providing additional assurance that safe shutdown can be accomplished and that the plant can be

maintained in a safe condition during and following potential fire situations. This NRC approval was made as Amendment No. 46 to the facility operating license.

Additional fire protection regulations were issued in 10 CFR 50.48 and Appendix R to Part 50 on November 19, 1980, with an effective date of February 17, 1981. These regulations established requirements for utilities to implement a fire protection program, and backfitted certain requirements in Appendix R to all utilities. For Con Edison, these included the various separation and protection requirements contained within Section III.G, emergency lighting requirements as stipulated by III.J, and oil collection system requirements for reactor coolant pumps as contained in Section III.O. Additionally, Section III.L established performance requirements for alternative shutdown systems. Subsequent to the regulations established in 10 CFR 50.48, various NRC generic letters and guidance documents have been issued to provide clarification of the Appendix R requirements.
NRC approval of the Indian Point Unit 2 Fire Protection Program including safe shutdown capability is provided in the Fire Protection Safety Evaluations Reports (SERs) dated:

- November 30, 1977
- February 3, 1978
- January 31, 1979
- October 31, 1980
- August 22, 1983
- March 30, 1984
- October 16, 1984
- September 16, 1985
- November 13, 1985
- March 4, 1987
- January 12, 1989
- March 26, 1996

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. 186. 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 2 Fire Protection Program description is provided separately in the following documents:

- IPEC Fire Protection Program Plan
- IP2 Fire Hazards Analysis Report
- IP2 Safe Shutdown Analysis Report

These documents provide a complete description of the Indian Point 2 Fire Protection Program including a description of fire areas, fire suppression and detection as well as other fire protection features credited to limit the effects of fires and the credited safe shutdown capability include the Alternate Safe Shutdown System (ASSS).

9.6.3 City Water System

The functions of the city water system are:

1. To provide the water supply for the fire protection system.
2. To provide an emergency supply of water to the suction of the auxiliary boiler feed pumps.
3. To provide makeup water to various systems.
4. To provide cooling water to various components.
5. To provide water to areas where hose connections are located for general usage.
6. To provide cooling water to the SBO / Appendix R Diesel Generator Heat Exchanger.

City water for the Indian Point Unit 2 comes from the city water main on Broadway via the Unit 1 mains and storage tanks and is under cathodic protection where the piping crosses the Algonquin Gas pipes. Unit 2 is tied to this system primarily through piping connections at two locations on the low pressure header (see Plant Drawings 192505, 192506, and 193183 [Formerly UFSAR Figure 9.6-5]). One connection is in the vicinity of the Unit 1 superheater building on the south side of the header. This connection provides water for:

1. Emergency makeup to the house service boilers.
2. Cooling the house service boiler water samples.
3. General usage at the house service boilers.
4. Makeup to the expansion tank of the conventional plant closed cooling system.
5. Cooling and general usage at the steam and water analysis station.
6. Cooling to the Appendix R Diesel Generator Heat Exchangers (Appendix R event only.

The second connection is at the north side of the header. This connection provides water for:

1. Makeup to the expansion tanks of the diesel-generator jacket water cooling system.
2. Emergency feed to the auxiliary boiler feed pumps.
3. Makeup to the expansion tank of the instrument air compressor closed cooling system.
4. General usage via hose connections inside the primary auxiliary building and waste holdup tank pit.
5. Emergency makeup to the isolation valve seal-water supply tank.
6. Spray water to the steam-generator blowdown tank.

A backup water supply is also provided for the circulating water pump seals and bearings.

There are also emergency city water connections in the primary auxiliary building that can be used for the charging pumps, residual heat removal pumps, and safety injection pumps.

9.6.4 Compressed Air Systems

9.6.4.1 Instrument Air System

The instrument air system is 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; all controls shall fail to a safe position on loss of power; and, after an accident, the air system shall be re-established. The system is shown in Plant Drawing 9321-2036 [Formerly UFSAR Figure 9.6-6].

To meet the design criteria the following design features have been incorporated. Duplicate compressors are installed with duplicate dryers and filters throughout. In addition, alternate supplies are provided from the Unit 2 station air system, and Unit 1 station air system. A connection has been provided in the station air system to allow a backup supply of air from portable compressed air equipment. Those items essential for safe operation and safe cooldown are provided with air reserves or gas bottles. These supplies 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, filters and air dryers are located on the ground floor of the control building, a seismic Class I structure, and they, along with other essential sections of the air supply system, have been designed to operate after a seismic event. In the event of a break in the non-essential portion of the system, a flow restrictor in the supply line to the non-essential portion will limit flow to the capacity of one instrument air compressor.

The system is served by two 225-scfm Worthington teflon-ring compressors, which discharge into a common air receiver. The instrument air from the receiver passes through one of two full-capacity heatless dryers. These heatless dryers are rated at 750 scfm, dewpoint compatible with the lowest expected outdoor temperature, and are dual-tower type dryers, with one of the dryers in service and one on standby. However, in the event that the transfer mechanism should fail during cycling of the dryer, the other dryer can be brought in to service. Each dryer is basically a stand-alone system, with dual prefilter, dryer and afterfilter units, and with local alarms and category alarms to the control room. An alternate air supply line from the station air system is provided, and has its own pair of full-capacity heatless regenerative dryers.

The instrument air compressors may be operated in two modes. One mode provides for the compressors to be in standby and to come on automatically in the event of low pressure in the common air receiver. During this mode, air is supplied by the station air system. The other mode of operation provides for simultaneous running of both compressors in order to provide continuity of service to Class I areas in the event of 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 provided to isolate the secondary plant and prevent pressure decay in the primary plant header. Valving has been installed to provide flexible operations as related to the alternate station air supply and to maintain proper isolation capabilities.

All air and oil filters are dual type to provide maintenance during operation.

9.6.4.2 Station Air System

The station air system shown in Plant Drawing 9321-2035 [Formerly UFSAR Figure 9.6-7] is supplied by a Worthington Corporation two-stage 650-scfm 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 aftercooler and compressor jacket is supplied from a closed cooling water system, which contains treated city water.

The 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 runs continuously and is loaded and unloaded at predetermined pressure settings. High-water and high-air temperature switches are connected to the control annunciator.

This system is alternatively supplied by the Unit 1 service air system through a manually operated valve interconnection to the Unit 2 air receiver. The size of the connection is equal to the Unit 2 supply pipe.

The station air system can also serve as an alternate supply to the Unit 2 instrument air system. In addition, an automatic emergency supply is supplied to the containment building weld channel and penetration pressurization system. Valve position lights in the control room advise the operator as to the status of emergency makeup control valve PCV-1140. A manual local reset solenoid valve is provided at the emergency valve.

9.6.5 Heating System

The heating system for Unit 2 represents an extension of the heating system for the Indian Point Unit 1.

Package boilers have been installed to supply steam for Unit 2 and are interconnected with the distribution header of the boilers for Unit 1. The main steam header from these boilers links the existing steam header to Unit 2 and also to Unit 3, so that output from any of the package boilers may be made available for the heating requirements of Unit 1, Unit 2, or Unit 3.

With respect to Unit 2, there are separate piping circuits for the unit heater steam supply to the east side and the west side of the turbine hall, including the heater bay. An extension from the circuit to the east side of the turbine hall serves the turbine oil storage tanks for both clean and dirty oil storage. Other heating services extend to the fan room, the fuel storage building, the containment building, the primary auxiliary building, the primary water storage tank, and the refueling water storage tank.

Provision is made for the following heating services:

1. Containment building.
    a. Steam unit heaters.
    b. Valves with hose bibs for maintenance purposes.

2. Primary auxiliary building.
   a. Electric strip heaters.
   b. Steam unit heaters.
   c. Air makeup steam tempering units.

3. Purge system containment building.
   a. Air makeup steam tempering units.

4. Fuel storage building.
   a. Steam unit heaters for standby heating.
   b. Air makeup steam tempering units. (Steam supply isolated)

5. Fan room.
   a. One steam unit heater.

9.6.6 Plant Communications Systems

For discussion of the facility communications systems, see Section 7.7.4.

REFERENCES FOR SECTION 9.6

1. Letter from Donald S. Brinkman, NRC, to Stephen B. Bram, Con Edison, Subject: Emergency Amendment to Increase the Service Water Temperature Limit to 90^{o}F (TAC 73764), dated August 7, 1989.

TABLE 9.6-1

Minimum Essential Service Water Requirement
Under Accident Conditions

Service	Flow each (gpm)	Number	Total Flow (gpm)
Containment Recirculation Fan Coolers	1600	5	8000
Containment Recirculation Fan Coolers Motors	17	5	85
Emergency Diesel Generators	350[1]	3	1050[1]
Instrument Air Compressor Heat Exchangers	65	2	65
Radiation Monitor Sample Coolers	2.5	2	5
Service Water Pump Strainer Blowdown	100	3	300

Minimum Non Essential Service Water Requirements
Post LOCA Recirculation

Service	Flow each (gpm)	Number	Total Flow (gpm)
Component Cooling Water Heat Exchangers	2400	2	4800
Service Water Pump Strainer Blowdown	100	3	300

1. With a design 10% tube plugging limit the Emergency Diesel Generator Coolers require a minimum flow of 296 gpm to remove the design heat load. The minimum required flow per cooler was set to 350 gpm to provide margin for heat removal yet allow for hydraulic limitations.

9.6 FIGURES

Figure No.	Title
Figure 9.6-1 Sh. 1	Service Water System - Flow Diagram, Sheet 1, Replaced with Plant Drawing 9321-2722
Figure 9.6-1 Sh. 2	Service Water System - Flow Diagram, Sheet 2, Replaced with Plant Drawing 209762
Figures 9.6-2 Through 9.6-4	Deleted
Figure 9.6-5 Sh. 1	City Water System - Flow Diagram, Sheet 1, Replaced with Plant Drawing 192505
Figure 9.6-5 Sh. 2	City Water System - Flow Diagram, Sheet 2, Replaced with Plant Drawing 192506
Figure 9.6-5 Sh. 3	City Water System - Flow Diagram, Sheet 3, Replaced with Plant Drawing 193183
Figure 9.6-6	Instrument Air - Flow Diagram, Replaced with Plant Drawing 9321-2036
Figure 9.6-7	Station Air - Flow Diagram, Replaced with Plant Drawing 9321-2035

9.7 EQUIPMENT AND SYSTEM DECONTAMINATION

9.7.1 Design Basis

Activity outside the core can 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-γ 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 being decontaminated. The products of water activation are not long lived and may be removed by natural decay during reactor cool-down and subsequent flushing procedures. Activated corrosion products are the primary source of the remaining activity.

The corrosion products contain radioisotopes from the reactor coolant, which have been absorbed on or have diffused into the oxide film. The oxide film, essentially magnetite (Fe_30_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 components. The contamination can be spread by various means when access is required. Contact while working on primary system components can 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 can contaminate the immediate areas and contribute to the contamination of the equipment, tools, and clothing.

9.7.2 Methods of Decontamination

Surface contaminates, which 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 nonporous materials. Personnel and their clothing are decontaminated according to the standard health physics requirements.

Those areas of the plant, which 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 surface 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 and tools can be cleaned by the use of a liquid abrasive bead decontamination unit, an ultrasonic unit, a sandblast unit or a Freon degreaser unit installed in Unit 1.

9.7.3 Decontamination Facilities

Decontamination facilities onsite 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 the 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 locker room. Personnel decontamination kits with instructions for their use are in the radiation control area locker room.

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 building 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 and through monitored exhaust paths.
3. Provide purging of the building to the plant vent for dispersion to the environment.

The air exhausted by the system is filtered, monitored, and diluted so that offsite dose during normal operation will not exceed Offsite Dose Calculation Manual (ODCM).

9.8.2 System Design and Operation

The primary auxiliary building ventilation system (See Plant Drawing 9321-4022 [Formerly UFSAR Figure 5.3-1]) is composed of the following systems:

1. Makeup air handling system complete with fan, filters, heating coils, and supply ductwork.
2. Exhaust system complete with fans, ductwork, roughing filters, HEPA filters, and charcoal filters.
3. [Deleted]

Design parameters for the system components are given in Table 9.8-1.

Branch supply ducts direct makeup air to the various floors at the east end of the building, from where it flows to the rooms and compartments. Air is exhausted from each of the building 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 filters, HEPA filters, and charcoal filters before discharge to the plant vent. The exhaust system has been designed to ensure that air flows from the "clean" end of the building through the "hot" areas.

Ventilating air exhausted from the waste storage tank pit is arranged to bypass the primary auxiliary building system and flow directly into the exhaust fan inlet plenum.

There are four fans in the containment building purge system and primary auxiliary building ventilation system. The two exhaust fans (containment building purge and/or primary auxiliary building exhaust fans 21 and 22) are common to both the containment building purge system and primary auxiliary building ventilation system. The supply fan in each of the ventilation systems operates only in its individual ventilation system.

The primary auxiliary building supply fan normally runs, along with either or both of the exhaust fans. The containment building purge supply fan runs with either of the exhaust fans, with the other exhaust fan as a backup. All four fans may also run simultaneously. The interlocking for the fans is such that in no event will the number of supply fans operating be greater than the number of exhaust fans operating. However, operation of an exhaust fan without a supply fan running is acceptable.

Fans are manually selected. All four fans can be started and stopped by four discrete control switches located on the fan room control panels. Each fan has indicating lights on the fan room control panel and in the main control room. An auto trip alarm is also provided. In addition, each of the fans have a "jog" pushbutton located on the fan room control panel for testing.

TABLE 9.8-1
Primary Auxiliary Building Ventilation System Component Data

System	Units Installed	Units Capacity	Units Required for Normal Operation
Exhaust[1]			
Fans, standard conditions	2	55,500 cfm	1
Fan pressure	-	10.3 in. H_2O	-
Fan motors	2	125 hp	1
Plenums	2	55,500 cfm	1
Roughing filters	2	55,500 cfm	1
HEPA filters	2	55,500 cfm	1
Carbon Filters	1	55,500 cfm	1
Supply Tempering Unit (Primary Auxiliary Building)			
Fans, standard conditions	1	50,400 cfm	1
Fan pressure	1	2.5-in. H_2O	-
Fan motor	1	50 hp	1
Filters	1	50,400 cfm	1
Coils	1	50,400 cfm	1
Deleted			

Notes:
1. These two exhaust fans are used interchangeably and/or as backup for: (1) ventilation of primary auxiliary building, (2) containment building purge system.
2. Deleted

9.9 CONTROL ROOM VENTILATION SYSTEM

9.9.1 Design Basis

The control room heating, ventilation, and air conditioning system is designed to accomplish the following:

1. Maintain 75°F dry bulb and approximately 50-percent relative humidity in the control room at outside design conditions at 93°F dry bulb and 75°F wet bulb.
2. Permit cleanup of airborne particulate radioactivity entering the control room with normal makeup air flow and by infiltration.

9.9.2 System Design and Operation

The Unit 2 control room ventilation system is composed of the following equipment:

1. A direct expansion air conditioning unit complete with fan, steam heating coil and roughing filter. The design capacity of the unit is 9200 cfm. A backup fan of the same design capacity has been installed in parallel with the air conditioning unit.
2. A filter unit consisting of case, HEPA filters, charcoal filters, post-filters and booster fans with a capacity of approximately 2000 cfm.
3. Duct system complete with dampers and controls to allow three system operating modes.

The Unit 1 control room ventilation equipment for the central control room has been modified for recirculation mode only.

The control room ventilation systems are shown on Plant Drawings 252665 and 138248 [Formerly UFSAR Figure 9.9-1]. The Unit 2 control room ventilation system can be operated as follows:

1. Normal Conditions

   a. With outside air makeup will supply cooling or heating for the control room atmosphere as required, using fresh outside air makeup and with the charcoal filter unit bypassed. (Mode 1)

2. Incident Conditions

   a. On safety injection and/or high radiation signal, with outside air makeup filtered the booster fan will start and dampers will be positioned to permit outside air to flow through the charcoal filter unit. (Mode 2)

   b. On toxic gas and/or smoke signal, the outside makeup air will be isolated and the carbon filter booster fan will not operate, the system will be in 100% recirculation mode. (Mode 3)

All these operations can be performed manually from the control room. However, in the event of a safety injection signal and/or high radiation signal, the control room dampers will automatically reposition and start the booster fan to place the charcoal filter unit in service, for

system operating mode 2. A redundant toxic chemical and radiation monitor for central control room air intakes has been installed.

Evaluations of postulated hazardous chemical release scenarios and their impact on the Control Room Envelope (CRE) conducted over the past decade in accordance with NRC Regulatory Guide 1.78, have determined that Control Room (CCR) occupants are not affected by changes to shipments or stationary sources of chemicals. The CCR would be adequately protected from current licensing basis chemical releases during applicable events by the Control Room Ventilation System being aligned to 100% recirculation mode operation (i.e., no outside air being supplied) and, if necessary, by donning available protective breathing apparatus within two (2) minutes of toxic gas detection. The protective breathing system consists of at least eight (8) functional SCBA units available in the CCR, at least twenty-two (22) 1/2-hour capable bottles for SCBA use in the CCR, and a minimum of one-hundred-and-six (106) 1/2-hour capable bottles readily accessible to the CCR crew outside the CRE. A sixty-four (64) hour supply of breathable air is available and dedicated for CCR personnel use at all times.

For additional discussion of this system, see Section 7.2.

9.9 FIGURES

Figure No.	**Title**
Figure 9.9-1	Central Control Room HVAC (Heating, Ventilation, and Air Conditioning), Replaced with Plant Drawings 252665 & 138248

9.10 FUEL STORAGE BUILDING VENTILATION SYSTEM

9.10.1 Design Basis

The fuel storage building ventilation system is designed to perform the following functions:

1. Maintain the fuel storage building at negative pressure so as to prevent unmonitored releases.

2. Provide sweep ventilation of the building, across the spent fuel pool, from areas of low potential contamination to areas of higher potential contamination.

3. Filter particulates and iodine through HEPA and charcoal filters to reduce the postulated offsite dose, which may result from a dropped fuel rod. NRC SER dated July 27, 2000 approved a fuel handling accident analysis that took no credit for filtration to reduce offsite dose so this design feature is no longer required for accident mitigation.

4. Remove normal building heat.

9.10.2 System Design And Operation

The fuel storage building ventilation system, shown in Figure 5.3-1, consists of two air supply units (whose fans have been retired in place) and one exhaust system. In addition, an axial spot cooling fan circulates 3000 cfm of air to the spent fuel pit heat exchanger room.

The power and control circuits for the fuel storage building (FSB) air supply fans and dampers, and dampers for the FSB exhaust fan, have been retired-in-place. Each supply unit has manually-operated outlet dampers that allow the exhaust fan to draw air through the building. Each also has a tempering (heating) coil which have been retired in place. Steam supply to the heating coils have been isolated and retired in place and the condensate line isolated.

The exhaust system consists of registers, ductwork, a filter bank, and a fan. Three exhaust registers are located near the pool surface level, at the north end, and a fourth is near the ceiling at the north end of the building. The registers near the pool surface are intended to provide a sweep flow over the pool.

Air from the registers is ducted to a plenum chamber, which contains the filter banks. It flows sequentially through filter banks, consisting of roughing filters, HEPA filters, and charcoal filters, and then to the exhaust fan. Air from the exhaust fan is discharged to the plant vent.

The exhaust fan is the centrifugal type, belt-driven by 100 hp 480-V motor.

The system provides an air flow rate of nominally 20,000 cfm. The system is balanced to divide the exhaust air flow equally between the exhaust registers and to maintain the building at a slight negative pressure. The exhaust fan is operated and controlled from a single local control room.

As a result of IP2 Operating License Amendment No. 229 (dated June 5, 2002), the limiting conditions for operation and the surveillance requirements for the fuel storage building air filtration system were relocated from the Technical Specifications to the UFSAR. These relocated requirements have been modified to reflect the assumptions used for the fuel handling accidents approved by the Technical Specification Amendment 211 (July 27, 2000). These are contained in UFSAR Sections 9.10.3 and 9.10.4 below.

9.10.3 Limiting Conditions for Operation (Fuel Storage Building Air Filtration System)

The fuel storage building ventilation system is assumed to be operating whenever spent fuel movement is taking place within the spent fuel storage areas, allowed after the fuel has had a continuous 100 hour decay period.

9.10.4 Surveillance Requirements (Fuel Storage Building Air Filtration System)

Amendment 211 recognized the fuel storage building ventilation system would be operating for an accident even though the assumptions were to release the source term over a 2 hour period at ground level (FSAR Section 14.2). The fuel storage building ventilation system does not have to be demonstrated operable in the assumed configuration each refueling, prior to refueling operations, and prior to handling fuel. The fuel storage building air filtration system shall be periodically tested (a 25% allowance is allowed consistent with the philosophy of Technical Specification SR 3.0.2) to assure continued compliance with 10 CFR 50, Appendix I and design criteria in accordance with ASME N510-1989, as follows:

1. verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches water gauge while operating the system at ambient conditions and at a flow rate of 20,000 cfm ±10% at least once each 24 months during aerosol or leak rate system tests.

2. verifying that the system maintains the spent fuel storage pool area at a pressure less than that of the outside atmosphere during system operation at least once each 24 months.

3. A visual inspection of the normal atmosphere cleanup system and all associated components should be performed in accordance with Section 5 of ASME N510-1989.

4. In-place aerosol leak tests, in accordance with Section 10 of ASME N510-1989, for HEPA filters upstream from the carbon adsorbers in normal atmosphere cleanup systems should be performed: at least once each 24 months; after each partial or complete replacement of a HEPA filter bank; following detection of, or evidence of, penetration or intrusion of water or other material into any portion of a normal atmosphere cleanup system that may have an adverse effect on the functional capability of the filters; and, following painting, fire, or chemical release in any ventilation zone communicating with the system that may have an adverse effect on the functional capability of the system. The leak test should confirm a combined penetration and leakage (or bypass) of the normal atmosphere cleanup system of less than 0.05% of the challenge aerosol at rated flow ±10%. A filtration system satisfying this condition can be considered to warrant a 99% removal efficiency for particulates.

5. In-place leak testing, in accordance with Section 11 of ASME N510-1989, for adsorbers should be performed: at least once each 24 months; following removal of an adsorber sample for laboratory testing if the integrity of the adsorber section is affected; after each partial or complete replacement of carbon adsorber in an adsorber section; following detection of, or evidence of, penetration or intrusion of water or other material into any portion of a normal atmosphere cleanup system that may have an adverse effect on the functional capability of the adsorbers; and, following painting, fire, or chemical release in any ventilation zone communicating with the system that may have an adverse effect on the functional capability of the system. The leak test should confirm a combined penetration and leakage (or bypass) of the adsorber section of 0.05% or less of the challenge gas at rated flow ±10%.

6. The efficiency of the activated carbon adsorber section should be determined by laboratory testing of representative samples of the activated carbon exposed simultaneously to the same service conditions as the adsorber section in accordance with ASTM D3803-1989 at a face velocity of 50 ft/min, a temperature of 89F, and a 95% relative humidity. Sampling and analysis should be performed: at intervals of approximately 24 months; following painting, fire, or chemical release in any ventilation zone communicating with the system that may have an adverse effect on the functional capability of the carbon media; and, following detection of, or evidence of, penetration of water or other material into any portion of the filter system that may have an adverse effect on the functional capability of the carbon media. The acceptance criteria is a methyl iodide penetration of less than 7.5%.