TABLE OF CONTENTS

TABLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

C28

TABLE OF CONTENTS
(Continued)

TABLE OF CONTENTS
(Continued)

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C28

LIST OF TABLES

C26

C26

LIST OF TABLES
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C26

C26

C26

LIST OF FIGURES

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C31

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9.0 AUXILIARY AND EMERGENCY SYSTEMS

The Auxiliary and Emergency Systems are supporting systems required to ensure the safe operation or servicing of the Reactor Coolant System (detailed in Section 4). Various components in some of the systems are shared by Unit 3 and Unit 4: Appendix A discusses this sharing and lists the shared components.

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 analyzes component malfunctions, inadvertent interruptions of system operation, or partial system failure and how the design avoids hazardous or unsafe conditions.

The systems considered under this category are:

a) Chemical and Volume Control System
b) Auxiliary Coolant System
c) Sampling System
d) Facility Service Systems
e) Fuel Handling System
f) Equipment and System Decontamination
g) Auxiliary Building Ventilation System
h) Control Room Ventilation System
i) Containment Ventilation System
j) Auxiliary Feedwater System

9.1 GENERAL DESIGN 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.

9.1.1 RELATED CRITERIA

Reactivity Control Systems Malfunction

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

As described in Section 7 and justified in Section 14, the Reactor Protection Systems are designed to limit reactivity transients to DNBR $\geq$ 1.30 due to any single malfunction in the deboration controls.

Engineered Safety Features Performance Capability

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

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 plant personnel and the public.

Containment Heat Removal Systems

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

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

9.2 CHEMICAL AND VOLUME CONTROL SYSTEM

The Chemical and Volume Control System a) adjusts the concentrations of chemical neutron absorber for chemical reactivity control, b) maintains the proper water inventory in the Reactor Coolant System, including makeup for system leakages, c) provides the required seal water flow for the reactor coolant pump shaft seals, d) processes reactor coolant letdown, e) maintains the proper concentration of corrosion inhibiting chemicals in the reactor coolant and, f) maintains the reactor coolant and corrosion activities to within design levels. The system is also used to fill and hydrostatically test the Reactor Coolant System.

9.2.1 DESIGN BASES

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 assemblies (RCCA) as detailed in Section 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 limit the rate of uncontrolled or inadvertent reactivity changes to a value which provides the operators sufficient time to correct the situation prior to system parameters exceeding design limits.

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 RCC assemblies and boric acid. The full length RCC assemblies are divided into two categories comprising control and shutdown groups.

The control group, used in combination with boric acid, provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of RCC assemblies is used to compensate for short term reactivity changes at power such as those produced due to variations in reactor power requirements or in coolant temperature. The boric acid control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup and decay.

Reactivity Shutdown Capability

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

The reactor core, together with the reactor control and protection system is designed so that the minimum allowable DNBR is no less than 1.30 and there is no fuel melting during normal operation including anticipated transients. The shutdown groups are provided to supplement the control group of RCC assemblies to make the reactor at least one per cent subcritical (k_{eff}) = 0.99) following trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCC assembly remains in the fully withdrawn position.

Sufficient shutdown capability is also provided to maintain the core subcritical for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a steam bypass or relief valve. This is achieved with combination of control rods and automatic boron addition via the Safety Injection System with the most reactive rod assumed to be fully withdrawn.

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 RCC assemblies, with boric acid injection used to compensate for the long term xenon decay transient and for cooldown. Any time that the unit is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection will always exceed that quantity required to support a cooldown to cold shutdown conditions without letdown. Under these conditions, adequate boration can be achieved simply by providing makeup for coolant contraction from a boric acid tank and the refueling water storage tank. The minimum volume maintained in the boric acid tanks, therefore, is that volume necessary to increase the RCS boron concentration during the early phase of the cooldown of each unit, such that, subsequent use of the refueling water storage tank for contraction makeup will maintain the required shutdown margin throughout the remaining cooldown. In addition, the boric acid tanks have sufficient boric acid solution to achieve cold shutdown for each unit if the most reactive RCCA is not inserted. This quantity will always exceed the quantity of boric acid required to bring the reactor to hot standby 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 any boric acid transfer pump can be operated from diesel generator power on loss of offsite power. Boric acid can C26
be injected by one charging pump and one boric acid transfer pump at a rate which shuts the reactor down with no rods inserted in less than forty minutes when a feed and bleed process is utilized (less than 30 minutes when the available pressurizer volume is utilized). In forty additional minutes, enough 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 hours after shutdown. If two boric acid pumps and two charging pumps are available, these time periods are reduced. Additional boric acid is employed if it is desired to bring the reactor to cold shutdown conditions.

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.

Codes and Classifications

All pressure retaining components (or compartments of components) which are exposed to reactor coolant, comply with the following codes:

a) System pressure vessels - ASME Boiler and Pressure Vessel Code, Section III, Class C, including para. N-2113, original equipment; Section III, Class 3 or Class 2, post-steam generator repair equipment.

b) System valves, fittings and piping - USAS B31.1, including nuclear code cases.

System component code requirements are tabulated in Table 9.2-1.

The tube and shell sides on the regenerative heat exchanger and the tube side of the excess letdown exchanger are designed to ASME Section III, Class C. This designation is based on the following considerations: (a) each exchanger is connected to the reactor coolant system by lines equal to or less than 3", and (b) each is located inside the containment. Analyses show that the accident associated with a 3" line break does not result in clad damage or failure. Reactor coolant escaping during such an accident is confined to the containment building.

9.2.2 SYSTEM DESIGN AND OPERATION

Various components of the Chemical and Volume Control System are shared by the two units. These components are shown in Table 9.2-3 and discussion concerning the sharing is given in Appendix A. The following discussion is for the Chemical and Volume Control System for one unit and applies equally to either unit.

The Chemical and Volume Control System, shown in Figures 9.2-1 through 9.2-10, provides a means for injection of boric acid, chemical additions for corrosion control, and reactor coolant cleanup and degasification. This system also adds makeup water to the Reactor Coolant System, processes water let down from the Reactor Coolant System, and provides seal water injection to the reactor coolant pump seals. Materials in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

System components whose design pressure and temperature are less than the Reactor Coolant System design limits are provided with overpressure protective devices.

System discharges from overpressure protective devices (safety valves) and system leakages are directed to closed systems. Effluents removed from such closed systems are monitored and discharged under controlled conditions.

During operation, reactor coolant flows through the letdown line from a loop cold leg on the discharge side of the pump and, after processing is returned to the cold leg of another loop on the discharge side of the pump via a charging line. An alternate charging connection is provided on a loop hot leg. An excess letdown line is also provided for removing coolant from the reactor coolant system. The largest required charging pump flow to maintain normal operation with 45 gpm letdown orifice, 7.5 gpm RCP seals runoff and 1 gpm RCS leakage is supplied by one charging pump in operation.

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 containment building 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 normally passing through one of the CVCS letdown demineralizer(s), where impurities are removed, coolant flows through the reactor coolant filters and enters the volume control tank through a spray nozzle.

Hydrogen is automatically supplied, as determined by pressure control, to the vapor space in the volume control tank, which is predominantly hydrogen and water vapor. The hydrogen supply line has an excess flow valve (Figure 9.2-11) upstream and outside of the Charging Pump Room which will automatically close if the hydrogen flow increases beyond its specific flow setting due to a downstream pipe rupture thus eliminating possible release of hydrogen into the charging pump room. The hydrogen within this tank is supplied to the reactor coolant for maintaining a low oxygen concentration. Fission gases are periodically removed from the system by venting the volume control tank to the Waste Disposal System.

During plant shutdown, dissolved hydrogen in the RCS must be removed prior to opening the RCS for maintenance or refueling. Hydrogen removal can be performed in either or a combination of the following methods: 1) The VCT level can be raised and lowered repeatedly to "burp" the hydrogen-rich VCT gas space to the waste gas system. When the level is lowered, fresh nitrogen is admitted to the gas space. 2) Hydrogen peroxide is added to the RCS to react with the dissolved hydrogen to form water.

The charging pumps take suction from the volume control tank and return the coolant to the Reactor Coolant System through the tube side of the regenerative heat exchanger.

A newly borated bed of mixed resin ($H-BO_3$ form) is used intermittently to remove excess lithium which is formed from $B^{10}(n,\alpha)Li^7$ reaction. After saturation with lithium the mixed bed ($Li-BO_3$) is ready for service as a mixed bed demineralizer for purification.

Boric acid is dissolved in the batching tank to a concentration of approximately 3.0 to 4.0 percent by weight. A transfer pump is used to transfer the batch to the boric acid tanks. Small quantities of boric acid solution are metered from the discharge of an operating transfer pump for blending with primary water as makeup for normal leakage or for increasing the reactor coolant boron concentration during normal operation. The solubility limit for 4.0 weight percent boric acid is reached at a temperature of 57°F. A 5°F room temperature measurement uncertainty is added to obtain ambient room temperature limit of 62°F. This temperature is sufficiently low that the normally expected ambient temperatures within the auxiliary building will maintain boric acid solubility. C26 C26

Excess 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 and to recirculate the contents of individual holdup tanks.

Liquid effluent in the holdup tanks is processed as a batch operation. This liquid is pumped to the waste holdup tank for processing as liquid waste.

A fresh bed of mixed resin (H-OH form) can be used intermittently to remove boron from the reactor coolant near the end of core life. When the mixed bed has been saturated with Boron ($H-BO_3$), it is ready for use in removing cesium and lithium.

During 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 filters and volume control tank. The fluid is then transferred, via the charging pump or gravity drain, through the tube side of the regenerative heat exchanger into the Reactor Coolant System.

Expected Operating Conditions

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

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 9.2-4.

The results of the calculations are presented in Table 9.2-5. In these calculations defects in one percent of the fuel rods are assumed to be present at initial core loading and are uniformly distributed throughout the core and the fission product escape rate coefficients are therefore based upon an average fuel temperature.

The fission product activity in the reactor coolant during operation with small cladding pinholes or cracks in 1% of the fuel rods is computed using the following differential equations:

For parent nuclides in the coolant,

$$\frac{dN_{wi}}{dt} = Dv_i\, N_{C_i} - (\lambda_i + Rn_i + \frac{B'}{B_o - t'B})\, N_{wi}$$

for daughter nuclides in the coolant,

$$\frac{dN_{wj}}{dt} = Dv_j\, N_{C_J} - (\lambda_j + Rn_j + \frac{B'}{B_o - t'B})\, N_{wj} + \lambda_i\, N_{w_i}$$

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
v = 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 (during initial fuel cycle only) and irradiation of boron, lithium and deuterium in the coolant. The deuterium contribution is less than 0.1 curie per year and may be neglected. The parameters used in the calculation of tritium production rate are presented in Table 9.2-6.

Reactor Makeup Control

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

Makeup for normal 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 either increase or decrease the boric acid concentration in the Reactor Coolant System. To maintain the reactor coolant volume constant, an equal amount of reactor coolant is let down to the holdup tanks. Should the letdown line be out of service during operation, sufficient volume exists in the pressurizer to accept the amount of boric acid necessary for hot standby.

Boration to the cold shutdown concentration is also achievable without letdown when boration is performed in conjunction with the plant cooldown through the required makeup for coolant contraction. Specifically, if boric acid is injected first from the boric acid tanks and then from the refueling water storage tank to maintain constant pressurizer level during the cooldown, sufficient boric acid will be added to the RCS to maintain the required shutdown margins.

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

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

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

c) The refueling water storage tank, which supplies borated water for normal or emergency makeup.

d) The chemical mixing tank, which is used to inject small quantities of solution when additions of hydrazine or pH control chemical are necessary.

Makeup is provided to maintain the desired operating fluid inventory in the Reactor Coolant System. The operator can stop the makeup operation at any time in any operating mode by remotely closing the makeup stop valves. One primary water makeup pump and one boric acid transfer pump are normally operated.

A portion of the high pressure charging flow is injected into the reactor coolant pumps between the pump impeller and the shaft seal so that the seals are not exposed to high temperature reactor coolant. Part of the flow is the shaft seal leakage flow and the remainder enters the Reactor Coolant System through a labyrinth seal on the pump shaft. The shaft seal leakage flow cools the lower radial bearing, passes through the seals, is cooled in the seal water heat exchanger, filtered, and returned to the volume control tank.

Seal water inleakage 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.

<u>Automatic Makeup</u>

The "automatic makeup" mode of operation of the reactor makeup control provides boric acid solution preset by the operator 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 operating conditions, the mode selector switch and makeup stop valves are set in the "AUTO" position. A preset low level signal from the volume control tank level controller causes the automatic makeup control action to open the makeup stop valve to the charging pump suction, open the concentrated boric acid control valve and the primary water makeup control valve.

The flow controllers adjust flow so that the concentration of the blend matches that of the preset concentration. The primary water and the boric acid streams meet and are mixed in the boric acid blender. Makeup addition to the charging pump suction header causes the water level in the volume control tank to rise. At a preset high level point, the makeup is stopped; the primary water makeup control valve closes, the concentrated boric acid control valve closes and the makeup stop valve to charging pump suction closes.

<u>Dilution</u>

The "dilute" mode of operation permits the addition of a pre-selected quantity of primary water makeup at a pre-selected flow rate to the Reactor Coolant System. The operator sets the makeup stop valves to the volume control tank and to the charging pump suction in the "auto" position, the mode selector switch to "dilute", the primary water makeup flow controller set point to the desired flow rate, and the primary water makeup batch integrator to the desired quantity. If the dilution flow deviates ±5 gpm from the preset flow rate, an alarm indicates the deviation. One primary water pump runs continuously to provide makeup water as required. Makeup water is added to the volume control tank and then goes to the charging pump suction header. Excessive rise of the volume control tank water level is prevented by automatic actuation (by the tank level controller) of a three-way diversion valve, which routes the reactor coolant letdown flow to the holdup tanks. When the preset quantity of primary water makeup has been added, the batch integrator causes the primary water makeup control valve to close.

<u>Alternate Dilute</u>

The "Alternate Dilute" mode of operation permits the addition of a pre-selected quantity of reactor makeup water at a pre-selected flow rate to the Reactor Coolant System. A primary water pump is normally operating. Before actuation of the "Start" Control Station, the operator sets the mode selector switch to "Alternate Dilute", the reactor makeup water flow controller set point to the desired flow rate, and the reactor makeup water "batch integrator" to the desired quantity.

The operator actuates the "Start" Control Station. This mode of operation is similar to the "dilute" mode except both the makeup stop valves to the Volume Control tank and charging pump suction are opened. Primary water is simultaneously added in the volume control tank and in the charging pump suction header. By adding primary water at both locations the delay time for injecting primary water is reduced and hydrogen is added to a portion of the primary water flow. Excessive water level in the volume control tank is prevented by automatic actuation of a three-way diversion valve (by the tank level controller), which routes the reactor coolant letdown flow to the hold-up tanks. When the preset quantity of reactor makeup water has been added, the batch integrator causes the reactor makeup water control valves to close. This mode of control is used when there are daily load changes. After the "Alternate Dilute" mode requirements are satisfied, the operator may return the mode selector switch back to "Dilute" to permit the addition of reactor makeup water as required.

Boration

The "borate" mode of operation permits the addition of a pre-selected quantity of concentrated boric acid solution at a pre-selected flow rate to the Reactor Coolant System. The operator sets the makeup stop valves to the volume control tank and to the charging pump suction in the "Auto" position, the mode selector switch to "borate", the concentrated boric acid flow controller set point to the desired flow rate, and the concentrated boric acid batch integrator to the desired quantity. If the boration flow deviates 1.5 gpm from the preset flow rate, an alarm indicates the deviation. Placing the reactor makeup control switch in the "start" position starts the selected boric acid transfer pump, and permits the concentrated boric acid to be added to the charging pump suction header. The total quantity added in most cases is so small that it has only a minor effect on the volume control tank level. When the preset quantity of concentrated boric acid solution has been added, the batch integrator causes the concentrated boric acid transfer pump to stop and the concentrated boric acid control valve to close.

The capability to add boron to the reactor coolant is sufficient so that no limitation is imposed on the rate of cooldown of the reactor upon shutdown. The maximum rates of boration and the equivalent coolant cooldown rates are given in Table 9.2-2. One set of values is given for the addition of boric acid from a boric acid tank with one transfer and one charging pump operating. The other set assumed the use of refueling water but with two of the three charging pumps operating. The rates are based on full power operating temperature and on the end of the core life when the moderator temperature coefficient is most negative. C26

By manual action of the operator, the boric acid transfer pump can discharge directly to the charging pump suction and bypass the blender and volume control tank.

Alarm Functions

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

a) Deviation of primary water makeup flow rate from the control set point

b) Deviation of concentrated boric acid flow rate from the control set point

c) If the reactor makeup control selector is not set for the automatic makeup control mode, a volume control tank low level alarm occurs at 12% of tank level.

Charging Pump Control

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

The speed of each pump can be controlled manually or automatically. During normal operation, only one of the three pumps is automatically controlled. During normal operation, only one charging pump is operating and the speed is modulated in accordance with pressurizer level. During load changes the pressurizer level set point is varied automatically to compensate partially for the expansion or contraction of the reactor coolant associated with the Tavg changes.

Tavg compensates for power changes by varying the pressurizer level set points in conjunction with pressurizer level for charging pump control. The level set points are varied between 22 and 60 percent of the adjustable range depending on the power level at a full load Tavg of 583°F. Charging pump speed does not change rapidly with pressurizer level variations due to the reset action of the pressurizer level controller. C26

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

Components

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

Regenerative Heat Exchanger

The regenerative heat exchanger is a multiple shell and U-tube unit which is designed to recover the heat from the letdown stream by reheating the charging stream during normal operation. This exchanger also limits the temperature rise which occurs at the letdown orifices during periods when letdown flow exceeds charging flow by a greater margin than at normal letdown conditions.

The letdown stream flows through the shell of the regenerative heat exchanger and the charging stream flows through the tubes. The unit is made of austenitic stainless steel, and is of all-welded construction.

Letdown Orifices

One of the three letdown orifices controls flow of the letdown stream during normal operation and reduces the pressure to a value compatible with the non-regenerative heat exchanger design. These orifices are used in parallel to pass maximum purification flow at normal Reactor Coolant System operating pressure.

The 45 gpm orifice is normally in service to minimize charging pump wear, and two 60 gpm orifices are normally available. The orifices are placed in and taken out of service by remote manual operation of their respective isolation valves. A combination of the standby orifices may be used in parallel with the normally operating orifice in order to increase letdown flow when the Reactor Coolant System pressure is below normal or when additional letdown is desired (e.g., for RCS cleanup). This arrangement provides a full standby capacity for control of letdown flow. Each orifice consists of bored pipe made of austenitic stainless steel.

Non-Regenerative (letdown) Heat Exchanger

The non-regenerative heat exchanger cools the letdown stream to the operating temperature of the mixed bed demineralizers. Reactor coolant flows through the tube side of the exchanger while component cooling water flows through the shell. The letdown stream outlet temperature is automatically controlled by TCV-*-144 or manually controlled by throttling manual valve *-834. The unit is a multiple-pass-tube heat exchanger. All surfaces in contact with the reactor coolant are austenitic stainless steel, and the shell is carbon steel.

CVCS Letdown Demineralizers

Five flushable demineralizers maintain reactor water chemistry. The main demineralizers are the A, B, D, and E demineralizers. The C demineralizer has smaller capacity and may be connected in series with either the A or B demineralizers.

A hydrogen ion form cation resin and a hydroxyl form anion resin are initially charged into the demineralizers. This resin bed is used to reduce RCS boron concentration (usually near the end of core life).

When saturated with boron, the resin is converted to an $H\text{-}BO_3$ form 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. In addition, each of the main demineralizers have sufficient capacity to maintain the cesium-137 concentration in the coolant below 1.0 µc/cc with one percent defective fuel. The demineralizer would be used intermittently to control cesium.

When saturated with lithium, the resin is converted to an Li^7-BO_3 form and is used to maintain reactor coolant purity. This form of resin removes both fission and corrosion products. In this form, the resin bed is designed to reduce the concentration of isotopes in the purification stream (except for cesium, yttrium, and molybdenum) by a minimum factor of 10. Each of the main demineralizers has sufficient capacity after operation for one core cycle with one percent defective fuel rods to reduce the activity of the primary coolant to refueling concentration.

With the exception of the C demineralizer, each demineralizer is sized to accommodate the maximum letdown flow. The number of demineralizers available provides flexibility and ensures standby capacity should a demineralizer become exhausted during operation. Additionally, the demineralizers may be charged with specialized resins (OH^- Anion, H^+ Cation, or Li^7-OH mixed bed) if desired. The C demineralizer is limited in use to 60 gpm letdown flow, and if used, is placed in series with either A or B demineralizers.

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

Resin Fill Tank

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

Reactor Coolant Filter

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

Volume Control Tank

The volume control tank collects the excess water released from zero power to full power, that is not accommodated by the pressurizer. It also receives the excess coolant release caused by the deadband in the reactor control temperature instrumentation. Overpressure of hydrogen gas is maintained in the volume control tank to control the hydrogen concentration in the reactor coolant at 25 to 50 cc per kg of water.

The reactor coolant hydrogen concentration can be reduced to 15 cc per kg prior to shutdown, provided that the operating period does not exceed two days and the reactor coolant hydrogen is monitored once per shift.

A spray nozzle is located inside the tank on the inlet line from the reactor coolant filter. This spray nozzle provides intimate contact of the gas and liquid phases. A vent path 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.

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 and other material of adequate corrosion resistance. Pump seal leakage is collected and routed to the waste holdup tanks for disposal. In order to minimize this leakage, which has proven to be a burden on the waste disposal systems of previous nuclear units, the pumps were modified after installation. This modification which has been successful in other projects, consists of new design plungers and seals, with a seal head tank. The pump design precludes the possibility of lubricating oil contaminating the charging flow, and the integral discharge valves act as check valves. Hydraulic accumulators are installed on the suction and discharge piping of the charging pumps to attenuate vibration and acoustically decouple this piping from the pumps.

Each pump is designed to provide the full charging flow and the reactor coolant pump seal water supply with normal seal leakage. Each pump is designed to provide rated flow against a pressure equal to the sum of the Reactor Coolant System maximum pressure (existing when the pressurizer power operated relief valve is operating) and the piping, valve and equipment pressure losses of the charging system at the design charging flows.

One of the three charging pumps can be used to hydrotest the Reactor Coolant System. The pumps are normally energized manually from the control room, and flow is automatically controlled by pressurizer level.

Chemical Mixing Tank

The chemical mixing tank is used to prepare caustic solutions for pH control, hydrazine for oxygen scavenging and hydrogen peroxide for shutdown chemistry control.

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

The chemical mixing tank is made of austenitic stainless steel.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools reactor coolant letdown flow until the flow rate is equal to the nominal injection rate through the reactor coolant pump labyrinth seal, if letdown through the normal letdown path is blocked. The unit is designed to reduce the letdown stream temperature from the cold leg temperature to 195°F. The letdown stream flows through the tube side and C26
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.

Seal Water Heat Exchanger

The seal water heat exchanger removes heat from the reactor coolant pump seal water returning to the volume control tank and reactor coolant discharge from the excess letdown heat exchanger. Reactor coolant flows through the tubes and component cooling water is circulated through the shell side.

The tubes are welded to the tube sheet because leakage could occur in either direction, resulting in undesirable contamination of the reactor coolant or component cooling water. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

The unit 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.

Seal Water Filter

The two filters collect particulates from the reactor coolant pump seal water return and from the excess letdown heat exchanger flow. The filter is designed to pass the sum of the excess letdown flow and the maximum design leakage from the reactor coolant pump seals. The vessel is constructed of austenitic stainless steel.

Seal Water Injection Filters

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

Boric Acid Filter

The boric acid filter collects particulates 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.

Boric Acid Tanks

The boric acid tank capacities are sized to store sufficient boric acid solution to support a cooldown to cold shutdown conditions without letdown. Under these conditions, adequate boration can be achieved simply by providing makeup for coolant contraction from a boric acid tank and the refueling water storage tank. The minimum volume maintained in the boric acid tanks, therefore, is that volume necessary to increase the RCS boron concentration during the early phase of the cooldown of each unit, such that, subsequent use of the refueling water storage tank for contraction makeup will maintain the required shutdown margin throughout the remaining cooldown.

In addition, the boric acid tanks have sufficient boric acid solution to achieve cold shutdown for each unit if the most reactive RCCA is not inserted. The concentration of boric acid solution in storage is maintained between 3.0 and 4.0 percent by weight. Periodic manual sampling is performed and corrective action is taken, if necessary, to ensure that these limits are maintained. Therefore, measured amounts of boric acid solution can be delivered to the reactor coolant to control the concentration. The combination overflow and breather vent connection has a water loop seal to minimize vapor discharge during storage of the solution. The tanks are constructed of austenitic stainless steel.

<u>Batching Tank</u>

The batching tank is sized to hold several days makeup supply of boric acid solution for the boric acid tank. The basis for makeup is reactor coolant leakage of 1/2 gpm at beginning of core life. The tank may also be used for solution storage. A local sampling point is provided for verifying the solution concentration prior to transferring it to the boric acid tank or for draining the tank.

The tank manway is provided with a removable screen to prevent entry of foreign particles. In addition, the tank is provided with an agitator to improve mixing during batching operations. The tank is constructed of austenitic stainless steel, and is not used to handle radioactive substances.

<u>Boric Acid Transfer Pumps</u>

Two 100% capacity centrifugal pumps per unit are used to circulate or transfer chemical solutions. The pumps circulate boric acid solution through the boric acid 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 and other adequately corrosion-resistant material.

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

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.

Recycle Process

The recycle process is common to Units 3 and 4 and the description below is of the components furnished to serve both units.

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 are processed through the waste holdup tank.

The three liquid storage tanks' capacity is approximately four Reactor Coolant System volumes. The tanks are constructed of austenitic stainless steel.

Each tank is equipped with relief devices for external and internal overpressure protection. Two vacuum breaker valves and one relief valve on each tank are set as described in Table 9.2-3.

Holdup Tank Recirculation Pump

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

Gas Stripper Feed Pumps

The three gas stripper feed pumps are used to transfer the contents of the holdup tanks to the waste holdup tank #1. One pump is normally used during the transfer operation. The non-operating pumps are maintained in standby and are available for operation in the event that the operating pump malfunctions. These centrifugal pumps are constructed of austenitic stainless steel.

Monitor Tanks

Process holding tanks that are used for bulk storage of waste water. When one tank is filled, the contents are analyzed prior to release. These tanks are constructed of stainless steel.

Monitor Tank Pumps

Two monitor tank pumps discharge water from the monitor tanks. Each pump is sized to empty a monitor tank in approximately 2.0 hours. The pumps are constructed of austenitic stainless steel.

Valves

Some valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the Waste Disposal System. Other valves may 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. Although originally designed for steam service, the source of steam to the batching tank steam jacket has since been abandoned.

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. Safety related power operated gate valves were evaluated for their susceptibility to pressure locking and thermal binding as required by NRC Generic Letters 89-10 and 95-07. The emergency boration valves (MOV-*-350) have a design feature (a hole drilled in the upstream disc to provide relief from the inter-disc space) which preclude the potential for pressure locking as described in the two generic letters.

Relief valves are provided for lines and components that might be pressurized above design pressure by improper operation or component malfunction. The excess letdown line is provided with thermal overpressure protection from post accident containment heat up (GL 96-06). Pressure relief for the tube side of the regenerative heat exchanger is provided by a bypass line around valve CV-*-310A. Relief valves settings and capacities are given in Table 9.2-3.

Turkey Point has manual operating features for selected air-operated valves as described in the Engineering Guidelines for Fire Protection for Turkey Point Units 3 & 4 (Reference 1) in the Chemical and Volume Control System. The installation of these features provides a manual means of operating these valves if the valve misoperates. C28

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.

9.2.3 SYSTEM DESIGN EVALUATION

Availability and Reliability

A high degree of functional reliability is assured in this system by providing standby components where performance is vital to safety and by assuring fail-safe response to the most probable mode of failure.

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

The electrical equipment of the Chemical and Volume Control System is arranged so that multiple items receive their power from various 480 volt buses (refer to Section 8.2). Each of the three charging pumps is powered from separate 480 volt buses.

The two boric acid transfer pumps are also powered from separate 480 volt 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 AC power, a charging pump and a boric acid transfer pump can be placed on the emergency diesels if necessary.

Control of Tritium

The Chemical and Volume Control System is used to control the concentration of tritium in the Reactor Coolant System. Essentially all of the tritium is in chemical combination with oxygen as a form of water. Therefore, any leakage of coolant to the containment atmosphere carries tritium in the same proportion, as it exists in the coolant. Thus, the level of tritium in the containment atmosphere, when it is sealed from outside air ventilation, is a function of tritium level in the reactor coolant, the cooling water temperature at the cooling coils, which determines the dew point temperature of the air, and the presence of leakage other than reactor coolant as a source of moisture in the containment air.

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

a) 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.

b) Possible public hazard due to release of tritium to the plant 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 discharging processed letdown water to the circulating water discharge

Leakage Prevention

Quality control of the material and the installation of the Chemical and Volume Control valves and piping, which are designated for radioactive service, is provided in order to essentially eliminate leakage to the atmosphere. The components designated for radioactive service are provided with welded connections to prevent leakage to the atmosphere.

However, flanged connections are provided 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. Holdup tanks are provided with threaded vacuum breakers.

The positive displacement charging pumps stuffing boxes are provided with leakoffs to collect reactor coolant before it can leak to the atmosphere. All valves which are larger than 2 inches and which are designated for radioactive service at an operating fluid temperature above 212°F were originally provided with a stuffing box and lantern leakoff connections. The leakoff connection and double set of packing may not be applicable since industry testing and experience has demonstrated better performance with a standard single packing set. Leakage to the atmosphere is essentially zero for these valves. All control valves are either provided with stuffing box and leakoff connections or are totally enclosed. Leakage to the atmosphere is essentially zero for these valves. C26

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.

<u>Incident Control</u>

The letdown line and the reactor coolant pumps seal water return line penetrate the containment. The letdown line contains three air-operated valves inside the containment and one air-operated valve outside the containment which are automatically closed by the containment isolation signal.

The reactor coolant pumps seal water return line contains motor-operated isolation valves, outside and inside the containment, which are automatically closed by the containment isolation signal.

The three seal water injection lines to the reactor coolant pumps and the charging line are inflow lines penetrating the containment. Each line contains two check valves inside the containment. The first of the two check valves from containment penetration is programmatically tested to provide isolation of the containment if a break occurs in these lines outside the containment.

Malfunction Analysis

To evaluate system safety, failures or malfunctions were assumed concurrent with a loss of coolant accident and the consequences analyzed are presented in Table 9.2-7. As a result of this evaluation, it is concluded that proper consideration has been given to unit 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. C26

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 assure continued functioning of the normal means of heat dissipation from the core. For the general case of rupture outside the containment, the largest source of radioactive fluid subject to release is the contents of the volume control tank. The consequences of such a release are considered in Section 14.

When the reactor is subcritical; i.e., during cold or hot shutdown, refueling and approach to criticality, the relative reactivity status (neutron source multiplication) is continuously monitored and indicated by BF_3 counters and count rate indicators.

Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate (See Table 9.2-2), is slow enough to give ample time to start a corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical. The maximum dilution rate is based on the abnormal condition of two charging pumps operating at full speed delivering unborated primary 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.

At least two separate and independent flow paths are available for reactor coolant boration; i.e., 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 flow path is always available for emergency boration of the reactor coolant. As backup to the boration system the operator can align the refueling water storage tank outlet to the suction of the charging pumps.

Boration during normal operation to compensate for power changes will be indicated to the operator from the flow indicators in the boric acid transfer pump discharge line. When the emergency boration path is used, three indications to the operator are available. The primary indication is a flow indicator in the emergency boration line. The charging line flow indicator will indicate boric acid flow since the charging pump suction is aligned to the boric acid transfer pump discharge for this mode of operation. The change in boric acid tank level is another indication of boric acid injection.

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 unit can be operated indefinitely since the thermal barrier cooler has sufficient capacity to cool the reactor coolant flow which would pass through the thermal barrier cooler and seal leakoff from the pump volute.

<u>Galvanic Corrosion</u>

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 have been shown[1] to exhibit only an insignificant degree of galvanic corrosion when coupled to each other.

For example, the galvanic corrosion of inconel versus 304 stainless steel
resulting from high temperature tests (575°F) in lithiated, boric acid solution C26
was found to be less than -20.9 mg/dm^2 for the test period of 9 days. Further
galvanic corrosion would be trivial since the cell currents at the conclusion
of the tests were approaching polarization. Zircaloy versus 304 stainless
steel was shown to polarize at 180°F lithiated, boric acid solution in less than C26
8 days with a total galvanic attack of -3.0 gm/dm^2. Stellite versus 304
stainless steel was polarized in 7 days at 575°F in lithiated boric acid C26
solution. The total galvanic corrosion for this couple was -0.98 mg/dm^2.

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

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(1) WCAP 1844 "The Galvanic Behavior of Materials in Reactor Coolants" D. G. Sammarone, August, 1961 Non-Proprietary.

9.2.4 REFERENCES

C28

1. STD-M-006, Engineering Guidelines for Fire Protection for Turkey Point Units 3 & 4.

TABLE 9.2-1
CHEMICAL AND VOLUME CONTROL SYSTEM CODE REQUIREMENTS

Regenerative heat exchanger	ASME III [1], Class C
Non-regenerative heat exchanger	ASME III, Class C, tube side, ASME VIII, shell side
CVCS letdown demineralizers	ASME III, Class C
Reactor coolant filters	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
Seal water injection filters	ASME III, Class C
Holdup tanks	ASME III, Class C
Boric acid filter	ASME III, Class C
Condensate filter	ASME III, Class C
Hydraulic Accumulators	ASME III, Class 2, 1977 plus Summer 77 Addenda [4]
Piping and Valves [3]	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. Aloyco valve weld ends in accordance with Westinghouse Spec. No. G-676241, Dwg. No. 498B932, hydrostatically retested at system test pressures after installation.

4. Replacement parts are procured in accordance with NRC Generic Letter 89-09, since the original manufacturer has dropped their N-stamp.

TABLE 9.2-2

NOMINAL CHEMICAL AND VOLUME CONTROL SYSTEM PERFORMANCE (1)

Unit design life, years	80	C31
Seal water supply flow rate, gpm (2)	24	
Seal water return flow rate, gpm	7.5	
Normal letdown flow rate, gpm	60	
Maximum letdown flow rate, gpm	120	
Normal charging pump flow (one pump), gpm	69	
Normal charging line flow, gpm	45	
Maximum rate of boration with one transfer and one charging pump from an initial RCS concentration of 1800 ppm, ppm/min	6.5	
Equivalent cooldown rate to above rate of boration, °F/min	1.5	
Maximum rate of boron dilution with two charging pumps from an initial RCS concentration of 2500 ppm, ppm/hour	350	
Two-pump rate of boration, using refueling water, from initial RCS concentration of 10 ppm, ppm/min(3)	6.2	
Equivalent cooldown rate to above rate of boration, °F/min	1.4	
Temperature of reactor coolant entering system at full power (design), °F	555.0	
Temperature of coolant return to reactor coolant system at full power (design), °F	493.0	
Normal coolant discharge temperature to holdup tanks, °F	127.0	
Amount of 3.0 weight percent boron solution required to meet cold shutdown requirements, at end of life with peak xenon (including consideration for one stuck rod). This value is based on a usable volume of 10,275 gallons plus 900 gallons volumetric uncertainty.	11,175	

NOTES :

1. Reactor coolant water quality is given in Table 4.2-2.
2. Volumetric flow rates in gpm are based on 130°F and 2350 psig.
3. 6.2 ppm/min remains a bounding minimum boration rate for two pumps at EPU conditions

TABLE 9.2-3

PRINCIPLE COMPONENT DATA SUMMARY

	Quantity[1]	Heat Transfer Btu/hr	Letdown Flow lb/hr	Letdown ΔT °F	Design Pressure psig,shell/tube	Design Temperature °F,shell/tube
Heat Exchangers						
Regenerative	1	8.65 x 10^6	29,826	265	2485/2735	650/650
Non-regenerative	1	14.8 x 10^6	29,826	163	150/600	250/400
Seal water	1	2.17 x 10^6	126,756	17	150/150	250/250
Excess Letdown	1	4.75 x 10^6	12,400	360	200/2485	250/650

	Quantity[1]	Type	Capacity Each gpm	Head	Design Pressure psig	Design Temperature °F
Pumps						
Charging	3	Pos.displ.	77	2385 psi	3000	250
Boric acid transfer	4*	Centrifugal	60	235 ft.	150	250
Holdup tank recirculation	1*	Centrifugal	500	100 ft.	150	200
Monitor tank	2*	Centrifugal	100	150 ft.	150	200
Gas stripper feed	3*	Canned	25	185 ft.	150	200
Gas stripper bottom	2	Centrifugal	12.5	93 ft.	75	300

	Quantity[1]	Type	Volume,Each	Design Pressure psig	Design Temperature °F
Tanks					
Volume	1	Vert.	300 ft^3	75 Int/15 Ext	250
Boric Acid	3*	Vert.	9100 gal	Atmos.	250
Chemical mixing	1	Vert.	6.0 gal	150	250
Batching	1*	Jacket Btm.	800 gal	Atmos.	250
Holdup	3*	Vert.	13,000 ft^3	15	200
RWST	1	Vert.	338,000 gal	Atmos.	200

TABLE 9.2-3

Tanks (continued)	Quantity[1]	Type	Volume	Design Pressure psig	Design Temperature °F
Monitor	2*	Vertical	10,000 gal	Atmos.	150

Demineralizer Vessels (5)	Quantity[1]	Type	Resin Volume ft^3	Flow gpm	Design Pressure psig	Design Temperature °F
	2	Flushable	30	120	200	250
	1	Flushable	20	60	200	250
	2	Flushable	43	120	200	250

Relief Valves	Quantity[1]	Relief Pressure psig	Capacity	
Charging pumps	3	2735	77 gpm	C28
Regenerative heat exchanger***	1	N/A	N/A	
Holdup tank	3	12	120 gpm	
Letdown line (intermediate pressure section)	1	600	240 gpm	
Seal water return line	1	150	165 gpm	
Batching tank heating jacket	1	20	320 lb/hr	
Volume control tank	1	75	170 gpm	
Holdup tank vacuum brks	2	1-3 in/WC**	100 scfm	
Letdown line (downstream Of non-regenerative heat exchanger)	1	450	120 gpm	
Unit 3 Excess Letdown Line****	1	2730	114 gpm	

1 Quantity per unit unless otherwise specified.

* Shared or capable of being shared by Unit 3 and Unit 4.
** Actual cracking pressure (Nominal Cracking Pressure may be higher) One vacuum breaker is capable of providing the minimum flowrate capacity shown.
*** Regenerative Heat Exchanger is protected from thermal overpressure by a bypass line around CV-*-310A instead of a relief valve.
**** Unit 4 Excess Letdown is provided with a reverse bypass line, with a check valve for thermal overpressure protection for GL 96-06.

TABLE 9.2-4

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT
FISSION PRODUCT ACTIVITIES
(PER UNIT FOR 24 MONTH CYCLE)

1.	Core thermal power, Mwt	2652	C26
2.	Fraction of fuel containing clad defects	0.01	
3.	Reactor Coolant System Water Mass (lb_m)	397,544	C26
4.	Reactor coolant average temperature, °F	576.5	
5.	Purification flow rate (nominal), gpm	60	
6.	Effective cation demineralizer flow, gpm	6	
7.	Volume control tank volumes and conditions		C26
	a. Vapor, cu ft	180	
	b. Liquid, cu ft	120	
	c Pressure, psig	18	C26
	d. Temperature, F	127	
8.	Fission product escape rate coefficients during full power operation:		
	a. Kr and Xe isotopes, sec^{-1}	6.5×10^{-8}	
	b. Br, I, Rb and Cs isotopes, sec^{-1}	1.3×10^{-8}	
	c. Te, Se, Sn and Sb isotopes sec^{-1}	1.0×10^{-9}	C26
	d. Mo, Tc 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 Pr, Nd and Pm isotopes, sec^{-1}	1.6×10^{-12}	C26
9.	Mixed bed demineralizer isotopic decontamination factors:		
	a. Kr, Xe isotopes and other isotopes	1.0	
	b. Br, I, Sr and Ba isotopes	10.0	
10.	Cation Mixed bed demineralizer isotopic decontamination factors		
	a. Cs-134, Cs-137, and Rb-86	10.0	
	b. Kr, Xe, Sr, Ba, Rb-88, Rb-89, CS-136, CS-138, Br, I, and other isotopes	1.0	

11. Volume control tank noble gas stripping fraction (closed system):

Isotope	Stripping Fraction
Kr-83	7.6×10^{-1}
Kr-85	6.3×10^{-5}
Kr-85m	5.7×10^{-1}
Kr-87	8.2×10^{-1}
Kr-88	6.8×10^{-1}
Kr-89	9.9×10^{-1}
Xe-131m	1.5×10^{-2}
Xe-133	3.2×10^{-2}
Xe-133m	7.4×10^{-2}
Xe-135	3.2×10^{-1}
Xe-135m	9.4×10^{-1}
Xe-137	9.8×10^{-1}
Xe-138	9.5×10^{-1}

12. Primary Makeup Water System Adds 200,000 gallons/year of clean water C26

Table 9.2-5

EQUILIBRIUM REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY

Security-Related Information - Withheld Under 10 CFR 2.390

C26

Table 9.2-5

EQUILIBRIUM REACTOR COOLANT SYSTEM SPECIFIC ACTIVITY

Security-Related Information - Withheld Under 10 CFR 2.390

C26

TABLE 9.2-6

TRITIUM PRODUCTION IN THE REACTOR COOLANT

(PER UNIT)

Basic Assumptions:

Unit Parameters:

1. Core thermal power, MWt	2296
2. Coolant water volume, ft^3	9,400
3. Core volume, ft^3	937.3
4. Core volume fractions	
a. UO_2	.2990
b. Zr + SS	.0933
c. H_2O	.6077
5. Unit full power operating times	
a. Initial cycle	78 weeks (18 months)
b. Equilibrium	49 weeks (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	13.4
8. Fraction of tritium in core (ternary fission + burnable boron) diffusing thru ss clad; thru Zr-clad	0.30; 0.01*
9. Ternary fission yield	8×10^{-5} atoms/fission

*The assumption that 1% of the ternary produced tritium diffuses into the coolant is based on the experience and analysis made of retention in the Shippingport, Breznau and Ginna zircaloy clad fuel.

10. Nuclear cross-sections and neutron fluxes

	σ; mb	(nv; n/cm^2-sec)
B^{10} (n, 2a,) T		
1 Mev ≤ E ≤ 5 Mev	= 31.59 (Spectrum weighted)	5.04×10^{13}
E > 5 Mev	= 75	7.4×10^{12}
Li^7 (n, na) T (99.9% purity)		
3 Mev ≤ E ≤ 6 Mev	= 39.1 (Spectrum weighted)	2.14×10^{13}
E > 6 Mev	= 0.4	2.76×10^{12}

Li^6 (n, a) T (99.9% purity Li^7)

σ = 675 barns; nv = 2.14×10^{13} n/cm^2-sec

II. CALCULATIONS (per unit)

	curies/year Initial Cycle	curies/year Equilibrium Cycle
A. Tritium from Core		
1. Ternary Fission	8,180	8,180
2. B^{10} (n, 2a) T (in ss clad poison rods)	592	N.A.
3. B^{10} (n, a,) Li^7 (n, na) T	1,110	N.A.
4. Release fraction (x 0.30, A2,3)	511	N.A.
5. Release fraction (x 0.01, A1)	82.	82.
Total release to Coolant	593.	82.
B. Tritium from Coolant		
1. B^{10} (n, 2a) T	843	582
2. Li^7 (n,na) T (limit 2.2 ppm Li)	6.6	6.6
3. Li^6 (n,a) T (purity of Li^7 = 99.9%)	6.6	6.6
4. Release Fraction (x 1.0)		
5. Total Release to Coolant	856.2	595.2
C. Total Tritium in Coolant	1,449.	677

TABLE 9.2-7

MALFUNCTION ANALYSIS OF CHEMICAL AND VOLUME CONTROL SYSTEM

Component	Failure	Comments and Consequences
1) Letdown line	Rupture in the line inside the reactor containment	The remote air-operated valve located near the coolant loop is closed on low pressurizer level to prevent supplementary loss of coolant through the letdown line rupture. The containment isolation valve in the letdown line outside the 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 these valves prevents any leakage of the containment atmosphere outside the containment.
2) Normal and alternate charging line	See above.	The check valves located near the coolant alternate loops prevent supplementary loss of coolant charging line through the line rupture. The check valves located at the boundary of the containment prevent any leakage of the containment atmosphere outside the containment.
3) Seal water return line	See above	The motor-operated isolation valve located return line both outside and inside the containment are manually closed or are automatically closed by the containment isolation signal initiated by the concurrent loss of coolant accident. The closure of these valve prevents any leakage of the containment atmosphere outside the containment.

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.2-1

REFER TO ENGINEERING DRAWING

5610-M-3046 , SHEET 1

REV.13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNITS 3 & 4**
CHEMICAL AND VOLUME CONTROL SYSTEM BORIC ACID SYSTEM **FIGURE 9.2-1**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.2-2

REFER TO ENGINEERING DRAWING

5610-M-3046 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNITS 3 & 4**
CHEMICAL AND VOLUME CONTROL SYSTEM BORON RECYCLE SYSTEM **FIGURE 9.2-2**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.2-3

REFER TO ENGINEERING DRAWING

5610-M-3046 , SHEET 3

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

CHEMICAL AND VOLUME CONTROL SYSTEM
BORON RECYCLE SYSTEM
FIGURE 9.2-3

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.2-4

REFER TO ENGINEERING DRAWING

5610-M-3046 , SHEET 4

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNITS 3 & 4**
CHEMICAL AND VOLUME CONTROL SYSTEM BORON RECYCLE SYSTEM **FIGURE 9.2-4**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.2-5

REFER TO ENGINEERING DRAWING

5613-M-3047 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNIT 3**
CHEMICAL AND VOLUME CONTROL SYSTEM CHARGING AND LETDOWN **FIGURE 9.2-5**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.2-6

REFER TO ENGINEERING DRAWING

5613-M-3047 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNIT 3**
CHEMICAL AND VOLUME CONTROL SYSTEM CHARGING AND LETDOWN **FIGURE 9.2-6**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.2-7

REFER TO ENGINEERING DRAWING

5613-M-3047 , SHEET 3

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 3

CHEMICAL AND VOLUME
CONTROL SYSTEM
SEAL WATER INJECTION TO RCP

FIGURE 9.2-7

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.2-8

REFER TO ENGINEERING DRAWING

5614-M-3047 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 4

CHEMICAL AND VOLUME CONTROL SYSTEM CHARGING AND LETDOWN

FIGURE 9.2-8

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.2-9

REFER TO ENGINEERING DRAWING

5614-M-3047 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNIT 4**
CHEMICAL AND VOLUME CONTROL SYSTEM CHARGING AND LETDOWN **FIGURE 9.2-9**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.2-10

REFER TO ENGINEERING DRAWING

5614-M-3047 , SHEET 3

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

CHEMICAL AND VOLUME
CONTROL SYSTEM
SEAL WATER INJECTION TO RCP
FIGURE 9.2-10

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.2-11

REFER TO ENGINEERING DRAWING

5610-M-3065 , SHEET 3

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNITS 3 & 4

NITROGEN & HYDROGEN SYSTEMS

HYDROGEN & CO2 SUPPLY

FIGURE 9.2-11

9.3 AUXILIARY COOLANT SYSTEM

9.3.1 DESIGN BASES

The Auxiliary Coolant System consists of three loops; the component cooling loop, the residual heat removal loop, and the spent fuel pit cooling loop as shown in Figures 9.3-1 through 9.3-9, 6.2-1, 6.2-5, 9.5-11, and 9.5-12. Each unit has a similar Auxiliary Coolant System. The description contained herein applies to both units. The spent fuel pit cooling loop and components are described in Section 9.5.

Performance Objectives

Component Cooling Loop

The component cooling loop (see figures 9.3-1 through 9.3-9) is the heat sink for the residual heat removal loop, the Chemical and Volume Control System, the spent fuel cooling loop and various Reactor Coolant System components.

The CCW system is designed with sufficient redundancy such that a single active failure will not prevent the system from accomplishing its support function of cooling safety related equipment.

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

The CCW system design satisfies performance objective requirements for Thermal Power Uprate operation as indicated in Reference 10.

Residual Heat Removal Loop

The residual heat removal loop (see figures 6.2-1 and 6.2-5) 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 cool down. During the first phase of cooldown, the temperature of the Reactor Coolant System is reduced by transferring heat from the Reactor Coolant System to the Steam and Power Conversion System.

All active loop components which are relied upon to perform their function are redundant.

The loop precludes any significant reduction in the overall design reactor shutdown margin when the loop is brought into operation for residual heat removal or for emergency core cooling by recirculation. The loop design includes provisions to enable hydrostatic testing to applicable code test pressures during shutdown.

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.

The RHR system design satisfies performance objective requirements for Extended Power Uprate operation as indicated in Reference 16.

Design Characteristics

Component Cooling Loop

One pump and three component cooling water heat exchangers are normally operated to provide cooling water for various components located in the auxiliary and containment buildings. The water is normally supplied to all components being cooled even though one of the components may be out of service.

Makeup water is taken from the primary water storage tank, as required and delivered to the surge tank. A backup source of water is provided from the water treatment plant via the demineralized water system.

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

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

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

c. A temperature and flow indicator in the outlet headers from the component cooling water heat exchangers.

d. A radiation monitor on the inlet headers to the component cooling pumps.

The CCW system is flow balanced each refueling outage to ensure adequate flow is available to system components during normal power operation and post-accident conditions.

The flow balance further ensures that individual component minimum and maximum flow limits are not exceeded due to changes in the system configuration which would occur in response to a design basis accident.

Residual Heat Removal Loop

Two pumps and two residual heat exchangers perform the decay heat cooling functions for the reactor. After the Reactor Coolant System temperature and pressure have been reduced to 350°F and 450 psig respectively, decay heat cooling is initiated by aligning one pump to take suction from the reactor outlet line and discharge through the heat exchangers and into the reactor inlet line. If only one heat exchanger is available, reduction of reactor coolant temperature is accomplished but at a lower rate.

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

Codes and Classifications

All piping and components of the Auxiliary Coolant System are designed to the applicable codes and standards listed in Table 9.3-4. The component cooling loop water contains a corrosion inhibitor to protect the carbon steel piping and is subject to chemical controls to protect the austenitic stainless steel components. Austenitic stainless steel piping is used in the residual heat removal loop, which contains reactor coolant, and in the spent fuel pit cooling loop, which contains water without corrosion inhibitor.

9.3.2 SYSTEM DESIGN AND OPERATION

Component Cooling Loop

Component cooling is provided for the following heat sources:

a. Residual heat exchangers (Auxiliary Coolant System, ACS)
b. Reactor coolant pumps (Reactor Coolant System)
c. Non regenerative heat exchanger (Chemical and Volume Control System, CVCS)
d. Excess letdown heat exchanger (CVCS)
e. Seal water heat exchanger (CVCS)
f. Sample heat exchangers (Sampling System)
g. Waste gas compressors (Waste Disposal System)
h. Residual heat removal pumps (ACS)
i. Safety injection pumps (Safety Injection System, SIS)
j. Containment spray pumps
k. Spent fuel pit heat exchangers (ACS)
l. Charging pump (CVCS)
m. Normal containment coolers
n. Control rod drive coolers
o. Emergency containment coolers
p. Post Accident Sampling System (PASS)

At the reactor coolant pump, component cooling water removes the heat from the bearing oil and the thermal barrier. Since the heat is transferred from the component cooling water to the intake cooling water, the component cooling loop serves as an intermediate system between the reactor coolant and intake cooling water system. This double barrier arrangement reduces the probability of leakage of high pressure, potentially radioactive coolant to the intake cooling water system. It also reduces the probability of in-leakage of chlorides from the salt water intake to the demineralized primary system.

During normal full power operation, one component cooling water pump and two or three component cooling water heat exchangers accommodate the heat removal loads. Each of the two standby pumps provides 100% backup, during normal operation. Two pumps and three heat exchangers are utilized to remove the residual and sensible heat during unit shutdown to ensure the CCW shell side flow does not exceed established flow limits.

Higher shell side flow rates are allowed for limited periods of time. If one of the pumps or one of the heat exchangers is not operated, safe shutdown of the unit is not affected, however, the time for cooldown is extended. If one CCW heat exchanger were out of service, RCS cooldown would continue to be possible, although at a much slower rate. Analysis has shown that the Steam Generator Steam Dump to Atmosphere (SDTA) can be utilized beyond the point in time at which RHR is aligned to the RCS. In addition, when RHR Cut-in Time is achieved, the operating RCP can be secured, at which point the Maximum CCW Temperature will be allowed to rise from 125°F to 130°F. This results in an analyzed cooldown time of less than 35 hours. (Reference 15) C26

The head tank accommodates normal expansion and limited in-leakage of water. The head tank and surge tank combine to accommodate contraction and ensure a continuous component cooling water supply until a leaking cooling line can be isolated. A single inlet/outlet line connects the head tank to the top of the surge tank. Two surge lines are provided on the surge tank, so that each surge line is connected to one of the two partitioned sections of the surge tank. The lines are connected to each of the two component cooling headers in the suction side of the component cooling water pumps.

The CCW head tank is normally vented to the Waste Holdup Tank. A radiation monitor in each component cooling water pump inlet header annunciates in the control room and closes a valve in the head tank vent line in the unlikely event that the radiation level reaches a pre-set level above the normal background. The head tank vent line is also isolated on high head tank level to mitigate overflow to the Waste Holdup Tank. Redundant component cooling water headers are provided (see Figure 9.3-1).

Design Bases

The design basis of the Component Cooling Water System is to provide sufficient heat removal from the Engineered Safety Features to the ultimate heat sink (ICW System), post accident.[1,2] The system, which is normally operated in an open configuration, is designed with sufficient capability to accommodate the failure of any single, active component without resulting in undue risk to the health and safety of the public following a Maximum Hypothetical Accident (MHA). The most limiting single active failure considered was the loss of one diesel, which results in only one CCW pump starting automatically to mitigate the consequences of the MHA. This assumed single failure also results in the loss of a complete train of engineered safety features, including the inability to open the CCW isolation valve associated with one RHR heat exchanger and one Emergency Containment Cooler (ECC).

Although a complete train of engineered safety feature components will be inoperable on loss of a diesel, CCW flow to these components will continue, except as noted above in the case of an RHR heat exchanger and one ECC. In support of the Extended Power Uprate (EPU) project, detailed CCW System thermal analyses were performed to evaluate overall performance following worst-case design basis accidents. Maximum expected system operating temperatures were calculated for both the double-ended primary system pipe break and secondary (steam) pipe failure. In the thermal analyses, a consistent set of conservative cooling system operating parameters were defined for several analyzed single failure conditions. These included the failure of a diesel generator, a containment spray pump and ICW pump. To restrict CCW System post-accident operating temperatures to within acceptable ranges, a design basis change remains in effect at EPU conditions to limit the maximum number of ECCs automatically starting to no more than two, assuming when only two CCW heat exchangers are in operation. Previously, all three ECCs were allowed to auto start. The applicable Engineered Safety Features at Turkey Point Units 3 and 4 include:

a. Residual Heat Removal heat exchangers (Auxiliary Coolant System, ACS)
b. Residual Heat Removal pumps (ACS)
c. Safety Injection pumps (Safety Injection System, SIS)
d. Containment Spray pumps
e. Emergency Containment Coolers
f. Support systems for the above

The CCW system is periodically placed in a split header configuration for short periods of time to allow the performance of inservice testing of the CCW pumps. During these periods of split header configuration, the CCW system is not able to serve the needs of a fully redundant and automatic two-train fluid system. Therefore Technical Specification provisions and certain other plant operating restrictions are imposed during these CCW system configurations.

With respect to the noted safety function, the Component Cooling Water System's performance is characterized by the minimum required and maximum allowable flow rates through the Engineered Safety Features listed above and through the other required and connected loads, and the corresponding heat transfer rates. In the various EPU project post-accident thermal analyses, a range of expected CCW System flows was analyzed based on detailed hydraulic flow calculations. The system flow calculations were based on revised CCW System flow balance criteria as provided in Table 9.3-6.

It should be noted that the Engineered Safety Features (with the exception of the Emergency Containment Coolers) operate in two sequential post-accident phases. The first, the injection phase, requires operation of the Safety Injection (SI), Containment Spray (CS) and Residual Heat Removal (RHR) pumps, taking suction from the Refueling Water Storage Tank (RWST). A safety injection signal automatically starts the SI and RHR pumps and aligns them for RCS cold-leg injection, which commences when the RCS pressure falls below the shutoff head of each pump. Containment Spray pump operation is initiated by coincident High and High-High Containment pressure signals (two-out-of-three). Two Emergency Containment Coolers (ECC) will be started automatically upon receipt of a safety injection signal, and will continue to run throughout the injection and recirculation phases. To ensure that two units are in operation post-accident, the third ECC will automatically start following a failure of one of the other two units. C26

The recirculation phase is initiated after the RWST volume is depleted. This phase differs from the injection phase in that the RHR pumps are re-aligned to take suction from the containment sump. The RHR pumps discharge through the RHR heat exchangers and then to the SI pump suction. The RHR pumps can also discharge to the CS pump suction when containment conditions are elevated. C26

The CCW system has also been designed to ensure that no portion of the system can reach saturation pressure during design basis accident scenarios, precluding the potential for formation of steam voids. The CCW head tank has been designed and installed to provide sufficient static head, such that component cooling water temperatures up to 270°F will not initiate steam void formation.

The CCW system heat removal functions have been evaluated for the following operating configurations:

a. Power Operation (which includes hot shutdown and hot standby operations).

b. Residual Heat Removal (RHR) Cooldown (which includes hot shutdown, cold shutdown and refueling operations).

c. Post-Accident (which includes both the injection and recirculation phases of a LOCA and the injection mode of a MSLB).

In addition, the two CCW system headers can be periodically isolated (i.e., split header) to support special system evolutions. This configuration was also evaluated for post-accident operation.

The CCW heat exchanger heat removal capacity was relaxed to offer increased system operating margin. The capacity of the CCW heat exchangers was relaxed by increasing the total tube resistance from the design value of 0.00159 hr-ft^2-^{o}F/BTU to 0.003 hr-ft^2-^{o}F/BTU at 95^{o}F ICW inlet temperature. For added conservatism, all containment integrity analyses, which explicitly model heat transfer to the ICW System, were performed at an ICW inlet temperature of 100^{o}F. In addition, for thermal evaluation of the CCW system the total tube resistance was lowered to 0.00285 hr-ft^2-F/Btu to account for the water properties of the ultimate heat sink and align with the containment analysis performed for EPU. C28

Since the CCW system configuration can physically provide a shell side flow greater than the design maximum, plant operation must be procedurally limited to prevent such configurations. To eliminate the potential for CCW heat exchanger shell side flows to exceed the maximum specified flow limits, both procedural controls and CCW system flow balancing limitations have been established. The operating procedure restrictions limit the number of operating CCW pumps to N-1, where N is the number of in-service CCW heat exchangers. Limitations on the number of major CCW system end users (emergency containment coolers, RHR heat exchangers and normal containment coolers) have also been established and proceduralized.

For the Power Operation and RHR Cooldown configurations, thermal performance calculations were performed using standard water-to-water heat exchanger heat transfer equations and generalized heat transfer methodology. For Power Operation, thermal analyses were performed at steady-state plant operating conditions to calculate maximum expected CCW heat exchanger operating temperatures.

During postulated design basis events, the CCW system major heat loads (ECCs and the RHR heat exchangers) are variable in nature and are dependent on containment operating conditions. As such, the GOTHIC Computer Code, which was used in containment integrity analyses, was also used to conservatively calculate limiting CCW system and ICW system post-accident operating temperatures. The results of the GOTHIC computer code were used in conjunction with an AFT Fathom incompressible flow network analysis to determine CCW system piping temperatures during post accident conditions

Peak CCW system operating temperatures occur during post-accident operations due to elevated containment temperatures and unrestricted heat rejection into the CCW system. A calculated maximum CCW System supply temperature of 158.6°F is acceptable for post-accident operation, and the basis for this temperature increase over the previous design temperature of 150^{o}F is validated for the Extended Power Uprate project (Reference 16).

Numerous failure cases have been modeled to target containment integrity, CCW system and ICW system integrity. The most limiting scenarios that affect CCW system integrity were identified as the failure of one train of Emergency Power, failure of one Containment Spray pump, failure of one ICW pump during a Large Break LOCA, or during a Main Steam line break.

For the Power Operation configuration, the CCW system supply temperature (CCW heat exchanger outlet temperature) should not exceed 105°F in order to provide adequate cooling to the RCP thermal barrier and motor bearings per the manufacturer's recommended guidelines. If the RCP is operating during a period when CCW temperature is above 105°F, the RCP motor bearing and seal injection water temperatures must be continuously monitored as per the applicable plant operating procedures. For the RHR Cooldown configurations, adequate RCS cooldown performance is maintained with a CCW system supply temperature of 125°F. For post-accident operation, the following are the most critical CCW system operating temperatures:

a. CCW System heat exchanger shell side inlet (return) temperature.
b. CCW System heat exchanger Shell side outlet (supply) temperature.
c. ECC CCW System outlet temperature.
d. RHR heat exchanger CCW System outlet temperature.
e. CCW System heat exchanger ICW outlet temperature.

For the CCW system heat exchanger return temperature, the CCW system outlet temperature shall remain at or below the system design temperature (200°F) and within CCW pump Net Positive Suction Head (NPSH) limitations. The CCW system heat exchanger supply temperature shall remain within analyzed limits to ensure that equipment cooled by the system remains operable. The most limiting CCW system supply temperature calculated in the Westinghouse EPU containment analysis is 158.5°F This temperature profile is most critical C31
for the Safety Injection pump lube oil cooler. The safety injection pump lube oil cooler was evaluated at EPU for 165°F, greater than the peak supply temperature of 158.5°F. For the ECCs, the upper limiting temperature due to C31
piping stress limitations is approximately 197°F.

Pump Performance

The plant design bases specify that the operation of one CCW pump and one RHR heat exchanger during an MHA is adequate for accident mitigation. Readjustment of CCW System valve positions will change the operating point of the CCW pumps, with respect to the CCW total system pressure drop, from the original start-up configuration. The CCW pump performance at these new valve positions has been demonstrated to meet the design requirements by a combination of tests and calculations.

The two criteria evaluated were adequate flow to the CCW System and adequate NPSH to the CCW pumps. Testing and CCW System hydraulic analyses have demonstrated that one pump could provide the required flow to the entire CCW System.

The CCW system has also been designed to ensure that no portion of the system can reach saturation pressure during design basis accident scenarios, precluding the potential for formation of steam voids. The CCW head tank has been designed and installed to provide sufficient static head, such that component cooling water temperatures up to 270°F will not initiate steam void formation. The required NPSH for one pump at 15,000 gpm is approximately 46 ft. with the installed CCW head tank, the available NPSH is 123.8 ft when the maximum post accident suction temperature is 182.5°F. Therefore, sufficient NPSH is available. Installation of the CCW head tank increases available static head by a nominal 29 psig. That added NPSH will ensure that pump performance will remain unaffected by establishing added margin during elevated temperature operation. C26

Maximum Flow Considerations for ECCs

Each cooler consists of 60 five-pass tubes in parallel with an additional 60 five-pass tubes to form a ten-row coil. From the standpoint of the component cooling water, this configuration can be represented by 120 parallel tubes. When the tubing cross-section (1-1/8 inch diameter, 0.049 inch wall) is considered, an average velocity of 6.4 feet per second is produced by a flow of 2000 gpm.

Flows to the ECCs may exceed the continuous design flow rating of 2000 gpm for short periods of time based on limiting the erosion rate of the Admiralty tubes to 5 mils/year. The shorter the time frame the greater the allowable flow rate.

The relationship of allowable flow maximums versus time durations is as follows:

2000 gpm	Continuous Operation
3200 gpm	1 Month (Post-LOCA Recirculation Limit)
3600 gpm	1 Week
5000 gpm	24 Hours (Initial Safety Injection Limit)
5500 gpm	1 Hour

Pressure in the tubes is not a likely mechanism for failure, since the tubes have a design pressure well in excess of the pressure deliverable by the CCW pumps. The addition of the CCW head tank and the resultant increase in overall CCW system pressure will reduce the margin between deliverable pressure and design pressure, but will not result in pressure at the ECC exceeding the tube design pressure.

As shown above, during the injection phase of an MHA, flows to the ECCs could exceed those values cited in Table 9.3-6. This additional flow is not anticipated to adversely effect the ECCs ability to perform their safety function for the following reasons:

a. Duration of the injection phase is short in comparison to the recirculation phase.

b. The operator has the ability to decrease flow to the ECCs (i.e., aligning CCW to RHR) if he believes the ECCs are receiving excessive CCW flow.

c. The design of the ECCs, as described above, is such that additional flow above its design maximum should not degrade the coolers' ability to perform their safety related function.

Spent Fuel Pit Heat Exchanger Minimum Flow

SFP Heat Exchanger CCW flow requirements are a function of the spent fuel being stored in the pit. The minimum CCW flow rate to the Primary SFP heat exchanger has been established at 1,200 gpm during CCW System flow balancing. Actual CCW flow to the Primary SFP heat exchanger would be higher during normal operation. Calculations conclude that CCW flow to the Primary SFP heat exchanger would exceed 1850 gpm during normal plant operation. During refueling operations, CCW flow to the Primary SFP heat exchanger can be temporarily increased to 2800 gpm for each SFP heat exchanger. The supplemental SFP heat exchanger must also comply with the safety related CCW system flow balance requirements to ensure adequate flow to ESF equipment. C26

Residual Heat Removal Loop

The Residual Heat Removal (RHR) loop consists of heat exchangers, pumps, piping and the necessary valves and instrumentation. During shutdown, coolant flows from the Reactor Coolant System to the RHR pumps, through the tube side of the RHR heat exchangers and back to the Reactor Coolant System. The inlet line to the RHR loop starts at the hot leg of one reactor coolant pump and the return line normally connects to the low head Safety Injection System piping to the three cold legs. The RHR heat exchangers are used to cool the water during the latter phase of Safety Injection System operation. These duties are defined in Section 6. The heat loads are transferred by the RHR heat exchangers to the component cooling water.

During unit shutdown, the cooldown rate of the reactor coolant system is controlled by manually regulating the flow through the tube side of the RHR heat exchangers. A bypass line and an automatic flow control valve around the RHR heat exchangers are used to maintain a constant flow through the residual heat removal loop and to control cooldown.

Double, remotely operated valving in the inlet line is provided to isolate the RHR loop from the Reactor Coolant System. To protect the RHR system against excess pressure and to prevent an intersystem LOCA, the RHR inlet isolation valves, MOV-750 and MOV-751, are supervised by pressure switches which will automatically close these valves at or above 525 psig; an isolation alarm warns the operator. A pressure spike, either real or spurious, during low pressure operation could result in the closure of these valves. This event could lead to the loss of the letdown flow path, potentially leading to RCS overpressurization and/or loss of decay heat removal capability. To prevent this event from occurring, the autoclosure interlock (ACI) for MOV-750 and MOV-751 may be defeated during RHR system operation for Reactor Coolant System shutdown cooling in Modes 5 and 6. Manual override of the ACI is also provided to allow operation of the isolation valves when the overpressure modification system is in the low pressure mode. MOV-750 and MOV-751 are locked closed during normal plant operation and following a Loss of Coolant Accident to provide containment isolation. This is accomplished by locking the breakers for MOV-750 and MOV-751 open. Two remotely operated valves in parallel and two check valves in series isolate each line to the Reactor Coolant System cold legs from the RHR loop. Overpressure in the loop is prevented by a relief valve which discharges to the pressurizer relief tank.

The cooldown capability of the RHR system has been evaluated for a number of cases for each of the following scenarios:

a. RHR system cut-in evaluation
b. Normal Cooldown (all cooling equipment available)
c. Abnormal Cooldown (equipment unavailable) C28
d. Accident cases include Main Steam Line Break (MSLB), Reactor Coolant Pump Single Failure Locked Rotor (SFLR), and Steam Generator Tube Rupture (SGTR).

A "better estimate" methodology was used to determine appropriate input values for the RHR cool down analysis. This methodology provides more nominal, as opposed to extremely conservative, values of input parameters. Seasonal variability of the ICW temperature, the CCW heat exchanger fouling factor, and the auxiliary plant heat loads were considered in performing the cool down analyses listed above.

For the RHR system cut-in evaluation, the RHR system is capable of accepting the RCS heat removal function as early as 7.5 hours based on the most restrictive operating parameters and with all cooling equipment trains available. This is well within the 13-hour duration considered in the CST sizing basis.

For the Normal Cooldown evaluation, the maximum calculated duration to cool down the RCS from "no load" to 200°F is 28 hours.

C28

For the Abnormal Cooldown evaluation, RCS cooldown would continue to be possible, although at a much slower rate. In addition, the time at which the RHR system can be placed into service would also be extended.

For the MSLB, SFRF, and SGTR accident cases, the maximum calculated duration to cooldown the RCS from "No load" to 200°F ranges from 61 to 64 hours.

Component Cooling Water Components

Component Cooling Water Heat Exchangers

The three component cooling water heat exchangers are of the shell and straight tube type. Intake cooling water circulates through the tubes while component cooling water circulates through the shell side. Parameters are presented in Table 9.3-1.

Component Cooling Water Pumps

The three component cooling water pumps are horizontal, centrifugal units. The pump casings are made from cast iron (ASTM 48) which is corrosion-erosion resistant. The material thickness is dictated by high quality casting practice and ability to withstand mechanical damage and as such are substantially overdesigned from a stress level standpoint. Parameters are presented in Table 9.3-1.

Component Cooling Water Surge Tank

The component cooling water surge tank, in combination with the head tank, is designed to accommodate changes in component cooling water volume. The surge tank has piping connections to both CCW loops and the head tank, and is constructed of carbon steel. Parameters are presented in Table 9.3-1. The surge tank also has a flanged opening at the top of the tank. The flanged opening can be used to access the tank during periods of reduced inventory, but is normally inaccessible due to the level maintained in the head tank.

Chemical Pot Feeder Tank

The chemical pot feeder tank provides for the direct addition of corrosion additive to the component cooling water. Parameters are listed in Table 9.3-1.

Component Cooling Water Valves

The valves used in the component cooling water system are normally constructed of carbon steel with bronze or stainless steel trim. Self-actuated spring loaded relief valves are provided for lines and components that could be pressurized to their design pressure by improper operation or malfunction.

Turkey Point has Manual Operating features for selected air-operated valves as described in the Engineering Guidelines for Fire Protection for Turkey Point Units 3 & 4 (Reference 17) in the component cooling water system. The installation of these features provides a manual means of operating these valves if the valve misoperates.

Component Cooling Water Piping

All original component cooling water system piping is carbon steel with welded joints and connections except at components which might need to be removed for maintenance. The piping added with the CCW head tank is welded austenitic stainless steel, except at selected component connections, to provide added corrosion resistance and reduced maintenance requirements. Also, Unit 3 CCW A Supply/Return piping (4” & 3” respectively), and the Unit 4 CCW "B" Supply/Return piping (4” and 3” respectively) to Reactor Coolant Pump Thermal Barrier & Cooler located under the transfer canal (Inside Unit 3 and Unit 4 Containment EL 14’) are austenitic stainless steel (Reference 18 & 19). C31 C31

Component Cooling Water Head Tank

The component cooling water head tank provides added static head to increase CCW system pressure above saturation pressure during all design basis accidents and accommodates changes in CCW system volume in combination with the surge tank. The head tank is constructed of austenitic stainless steel to provide added corrosion resistance and reduce maintenance requirements. Parameters are presented in Table 9.3-1. In addition to piping connections to the surge tank and Waste Holdup Tank, the head tank includes vacuum breaker protection, local level indication, and a valve for chemical addition, if required.

Supplemental Cooling System

The Supplemental Cooling System (SCS) was installed to provide supplemental cooling to the Unit 3 or Unit 4 NCCs to maintain normal containment temperature within Technical Specification limits when the ICW temperature is high. The SCS is composed of two parallel loops, a SR supplemental loop and a NNS chiller loop, connected by a common plate and frame heat exchanger. The SR SCS loop circulates CCW inventory through the heat exchanger and the NNS chiller loop circulates chilled water to cool the CCW stream. The cooling function of the SCS is not a safety related function; however, because the SR supplemental loop functions as an extension of the CCW pressure boundary, it performs the safety related function of maintaining CCW pressure boundary integrity. The SCS is automatically isolated from CCW by CV-2216 and check valve 2185 on high temperature of the SCS discharge header or on low level in the CCW head tank. These safety related functions prevent adverse impact on the CCW flow balance when CCW is operating in accident mitigation mode and ensure that the CCW pressure boundary is protected from a breach in SCS piping or equipment located outside of the Auxiliary Building. The NNS chiller loop cooling function is not safety related and is not part of the CCW pressure boundary. The NNS chiller loop components are supported or restrained to prevent their failure from impacting any safety related SSC. Makeup for the NNS chiller loop is provided by the plant demineralized water system. The equipment is powered from local NNS sources. The plate and frame heat exchanger is the safety related boundary separating the supplemental cooling loop from the chiller loop. Parameters for the components are presented in Table 9.3-1.

Residual Heat Removal Loop Components

Residual Heat Exchangers

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

Residual Heat Removal Pumps

The two residual heat removal pumps are in-line, 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.

Residual Heat Removal Valves

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

Manual 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.

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. An alternate packing configuration is also used where full packing sets have been installed and leakoff lines were cut and capped.

Manually operated valves have backseats to facilitate repacking and to limit the stem leakage when the valves are open.

Turkey Point Unit 3 has Manual Operating features for selected air-operated valves as described in the Engineering Guidelines for Fire Protection for Turkey Point Units 3 & 4 (Reference 17) in the Residual Heat Removal Loop. The installation of these features provides a manual means of operating these valves if the valve misoperates. C28

Residual Heat Removal Piping

All residual heat removal loop piping is austenitic stainless steel. The piping is welded except for flanged connections at the control valves.

9.3.3 SYSTEM EVALUATION

Availability and Reliability

Component Cooling Loop

For continued cooling of the reactor coolant pumps, and the excess letdown heat exchanger, most of the piping, valves, and instrumentation are located outside the primary concrete shield at an elevation well above the anticipated post-accident water level in the bottom of the containment. (The exception is the cooling lines for the reactor coolant pumps which can be isolated by two valves in series following the accident.) In this annular area the component cooling equipment is protected against credible missiles and from being flooded during post-accident operation. Also, this location provides radiation shielding which allows for maintenance and inspections to be performed during power operation.

Outside the containment, the residual heat removal pumps, the residual heat exchangers, the spent fuel heat exchangers, the component cooling pumps and C26
heat exchangers, and associated valves, piping and instrumentation are maintainable and inspectable during power operation. System design provides for the replacement of one pump or one heat exchanger while the other units are in service.

Several of the components in 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. With the exception of portions which are normally isolated, the entire system is seismic Class 1 design. The components are designed to the codes given in Table 9.3-4. In addition the components are not normally subjected to any high pressures (See Table 9.3-1) or stresses. Hence a rupture or failure of the system is very unlikely.

During the recirculation phase following a loss-of-coolant accident, one of the three component cooling water pumps delivers flow to the shell side of one of the residual heat removal heat exchangers.

Residual Heat Removal Loop

Two pumps and two heat exchangers are available to remove residual and sensible heat during unit cooldown. If only one heat exchanger is operating, reduction of reactor coolant temperature is accomplished but at a lower rate. With this operating condition, safe operation of the unit is not affected. The function of this equipment following a loss-of-coolant accident is discussed in Section 6.

Leakage Provisions

Component Cooling Water Loop

Welded construction is used where possible throughout the component cooling loop piping, valves and equipment 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 cooling coil for the reactor coolant pump thermal barrier.

Tube or coil leaks in components being cooled would be detected during normal operation as described in Sections 4.2.7 and 6.5.

Leakage from or to the component cooling water loop can be detected by a change of level in the component cooling head tank. The rate of water level change and the area of the water surface in the head tank or subsequently in the surge tank, permits determination of the leakage rate. In-leakage from a radioactive source may also be detected by radiation monitors located on the main inlet headers. To assure adequate determination of leakage and the potential leakage source, 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, the leaking heat exchanger would be isolated and repaired. During normal operation, the leaking heat exchanger could be left in service with leakage up to the capacity of the makeup line to the system from the primary water storage tank. By manual transfer, emergency power is available for primary water pump operation.

Volumetric expansion of the CCW system due to system temperature increases has been evaluated. For the CCW system, the most limiting temperature swing would occur following a design basis accident when heat rejection to the CCW system is not manually limited. For this condition, a CCW head tank insurge would occur due to high heat loads and an overall increase in CCW system operating temperature. An increase of 100°F, which would bound any thermal swing postulated, would result in a volumetric expansion of approximately 350 gallons/header or 700 gallons total. The CCW head tank has a surge volume of less than 200 gallons. If a large thermal insurge occurred (e.g. during a LOCA), the overflow would be routed to the waste holdup tank. This would not impact safe system operation.

The component cooling water temperature increase associated with a normal plant cooldown will not overfill the component cooling water head tank; however, the heatup associated with accident heat loads may be sufficient to overflow the tank. The head tank vent line will isolate on high tank level; however, that non-safety function is subject to single failure and may permit the tank to overflow. If the head tank overflows, the loss of inventory is not critical to CCW system function since the remaining head tank and surge tank inventory is adequate to ensure continued system function.

Should a large tube side to shell side leak develop in a residual heat removal heat exchanger, the water level in the component cooling head tank would rise, and the operator would be alerted by a high water level alarm. An isolation valve in the vent line of the tank is automatically closed in the event of high radiation level at the component cooling water pump suction header. If the leaking residual heat exchanger is not isolated from the component cooling loop before the inflow completely fills the head 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. During normal operation, the water stored in the CCW head tank and 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. If a design basis leak (defined as a 50 gpm leak) were to occur coincident with a design basis LOCA, the installed automatic valves in the supply and return lines to the RCP isolate rapidly, such that the inventory remaining in the CCW head tank would be sufficient to ensure continued CCW system operability under all design basis conditions.

The relief valves on the component cooling water header downstream from each of the reactor coolant pumps are designed to relieve the thermal expansion of liquid enclosed in the piping system that can occur if the thermal barrier cooling water return piping is isolated while a heat load continues to be imposed by the thermal barrier heat exchanger. These relief valves protect the cooling water supply and return piping associated with the reactor coolant pump thermal barrier heat exchanger from overpressurization.

The relief valves on the cooling water lines downstream from the sample, excess letdown, seal water, non-regenerative, spent fuel pit and residual heat exchangers 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 is lower than the surge tank design pressure due to the installed static head tank. The relief valve set pressure prevents overpressurization of the CCW piping during design basis accident conditions. Initial protection is provided by an isolation valve which automatically closes on high flow in the event of a thermal barrier coil rupture.

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 sump is recirculated through the exterior piping system. To obtain the total radiation dose to the public due to leakage from this system, the potential leaks have been evaluated and discussed in Sections 6 and 14.

Each of the two residual heat removal pumps is located in a shielded compartment with a floor drain. In each compartment the leakage drains to a sump and is then pumped to the waste holdup tank by sump pumps. Two 75 gpm sump pumps are provided in each compartment and each is capable of handling the flow which results from the failure of a residual heat removal pump seal. The residual heat exchangers are located in a third compartment with two sump pumps.

Each sump has a level indicator which will warn the operator of high water level. Both of the lines from the containment sump to the individual residual heat removal pumps has two remotely operated isolation valves in series.

Incident Control

Component Cooling Loop

Component Cooling Containment isolation valves MOV-1417, -1418, and CV-739, are automatically closed on a Phase A containment isolation signal. Component Cooling Containment isolation valves MOV-716B, -730, and -626 are automatically closed on a Phase B containment isolation signal. The cooling water supply header to the reactor coolant pumps contains a check valve inside and two remotely operated valves outside the containment wall. The cooling water supply line to the excess letdown heat exchanger contains a check valve inside the containment which is closed during normal operation. Except for the normally closed makeup line and equipment vent and drain lines, there are no direct connections between the cooling water and other systems. The equipment vent and drain lines outside the containment have manual valves which are normally closed unless the equipment is being vented or drained for maintenance or repair operations.

Following a loss-of-coolant accident, one component cooling water pump and two component cooling water heat exchangers accommodate the heat removal loads. If either a component cooling water pump or component cooling heat water exchanger fails, one of the two remaining pumps provides 100% backup and each standby heat exchanger provides 50% backup. Valves on the component cooling water return lines from the safety injection, containment spray and residual heat removal pumps are normally open. Each of the component cooling water return lines from the residual heat removal heat exchangers has a normally closed remotely operated valve. If one of the valves fails to open at initiation of long-term recirculation, the valve which does open supplies a heat exchanger with sufficient cooling to remove the heat load.

Normally cross-connected, redundant component cooling water headers are provided for the unlikely event of a single failure in the component cooling water system following a loss of coolant accident. Header cross-connect valves are provided so that a passive failure (defined as 50 gpm leak) in the system can be isolated and cooling water flow can still be maintained to the necessary engineered safeguards equipment which require cooling water.

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. Both these valves are interlocked with reactor coolant system pressure. However, the autoclosure interlock (ACI) for these valves may be defeated during RHR system operation for Reactor Coolant System shutdown cooling in Modes 5 and 6. This action is intended to prevent a pressure spike, either real or spurious, from causing a loss of the letdown flow path which could potentially lead to RCS overpressurization and/or loss of decay heat removal capability. On the discharge side the connection is through two check valves in series and two electric motor operated gate valves in parallel. All of these are closed whenever the reactor is in the operating condition.

Malfunction Analysis

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

9.3.4 TEST AND INSPECTION CAPABILITY

The residual heat removal pumps flow instrument channels can be calibrated during shutdown.

The active components of the Auxiliary Coolant System are in either continuous or intermittent use during normal operation. Component testing in accordance with the plant IST Program is conducted, along with visual inspections and preventative maintenance.

Samples are analyzed to determine the amount of radioactivity in the reactor coolant system. If the radioactivity level is high a reactor coolant sample is analyzed and Gamma Spectroscopy is used to determine the existence of defects in fuel cladding. The frequency of sampling for gross activity and for radiochemical analysis of the reactor coolant system will be adequate to detect fuel clad defects to support operation as based on past experience.

For the Residual Heat Removal System loop to be considered operable, it must be filled with water to ensure that it can reliably perform its intended function. To address this, Generic Letter 2008-01 was issued to discuss the consequences of gas entrained in systems such as the Residual Heat Removal System loop that could compromise their operability. In response to this, a Gas Accumulation Management Program (GAMP) was established to provide long term void management. Locations in the Residual Heat Removal System loop, where gases could potentially accumulate, are periodically monitored using ultrasonic testing and/or vented to verify the system is filled. From the results of this monitoring, the GAMP ensures that gas accumulated within the Residual Heat Removal System loop is identified, evaluated, trended, and effectively controlled to prevent unacceptable degradation of performance of any structures, systems or components, ultimately to ensure system operability.

Filling and venting operations and periodic system operational and leakage tests are required to ensure that the Residual Heat Removal System loop piping and components are not damaged from water-hammer loads that may result from pump flows into voided discharge lines. The system must be periodically verified full by venting the accessible discharge piping high points. The inaccessible discharge piping high points that may be susceptible to gas accumulation are deemed appropriate to proactively provide the capability to vent each of these locations and to allow for future monitoring and trending, if it becomes necessary.

9.3.5 REFERENCES

1. Bechtel Letter SFB-2345 from G.N. Nutwell (Bechtel) to E. Preast (FPL), dated March 24, 1986.

2. NRC Letter "Component Cooling Water Flow Balancing - Turkey Point Plant Units 3 and 4," from D. G. McDonald (NRC) to C.O. Woody (FPL), dated February 5, 1987.

3. Safety Evaluation, JPE-LR-87-45, "Justification for Continued Operation for ICW System Design," Revision 3, dated March 17, 1989.

4. NRC Safety Evaluation for Amendment No. 111 and 104 to Facility Operating Licenses for Turkey Point Units 3 and 4, respectively, dated November 21, 1984.

5. FPL letter L-84-264 to the NRC,"Spent Fuel Storage Facility Expansion - Additional Information," dated October 5, 1984.

6. FPL letter L-76-178, "Proposed License Amendment to Facility Operating Licenses DPR-31 and DPR-41 Supplemental Information," dated April 30, 1976.

7. NRC letter to R. E. Uhrig (FPL) from George Lear (NRC), Re: Amendment No. 23 to License DPR-31 and Amendment No. 22 to License DPR-41, dated March 17, 1977.

8. Safety Evaluation, JPN-PTN-SENP-95-026, Revision 03, "Safety Evaluation for CCW Flow Balancing and Post-Accident Alignment Requirements to Support Current and Uprated Conditions," dated October 12, 1995.

9. Westinghouse WCAP-14276,"Turkey Point Units 3 and 4 - Uprating Licensing Report," Revision 1, dated December 1995 (Section 5.5.5).

10. Westinghouse WCAP-14291, Vols 1 - 3,"Turkey Point Units 3 and 4 - Uprating Engineering Report for Thermal Power Uprate," dated December 1995.

11. FPL letter L-2002-151 to the NRC, "Reduction of Decay Time for Core off-load and Revision of Technical Specification 3/4.9.3," dated October 21, 2002.

9.3.5 REFERENCES (cont'd)

12. NRC letter (Eva A. Brown) to FPL (J. A. Stall), "Turkey Point Units 3 & 4 - Issuance of Amendments Regarding Reduction in Decay Time From 100 Hours to 72 Hours (TAC Nos. MB6549 and MB6550), License Amendments 223/218, effective March 4, 2003.

13. Engineering Evaluation PTN-ENG-SENS-06-041, "Defeat of Residual Heat Removal System Suction Valve Autoclosure Interlock During Operation for Shutdown Cooling," Revision 0.

14. PCM 07-089, "RHR Heat Exchanger Structural Capability for OMS"

15. Westinghouse Calculation CN-SEE-I-11-15, "Turkey Point RHR Cooldown With One CCW HX Inoperable," Revision 0.

16. Westinghouse Technical Report WCAP-17152-P, "Turkey Point Units 3 and 4 Extended Power Uprate Engineering Report," August 2012.

17. STD-M-006, Engineering Guidelines for Fire Protection for Turkey Point Units 3 & 4.

18. EC-DEC 291003, "A - CCW Piping for Supply/Return to RCP Thermal Barrier & Cooler Replacement"

19. EC-DEC 292469, "B - CCW Piping for Supply/Return to RCP Thermal Barrier & Cooler Replacement" C31

TABLE 9.3-1

COMPONENT COOLING WATER
LOOP COMPONENT DATA

Component Cooling Water Heat Exchangers

Quantity	3
Type	Shell and Straight Tube
Heat Transferred, Btu/hr (shutdown condition)	14.7 x 10^6
Heat Removal Capability (Design Basis Heat Load), Btu/Hr	62.2 x $10^{6(3)}$
Shell Side (component cooling water) :	
- Inlet Temp. °F	174.6
- Outlet Temp. °F	158.6
- Maximum Allowable Unit 3 Flow Limits (gpm) :	
- Continuous Operation (TEMA limit)(1)	4,063
- Normal Plant Evolutions (testing, surveillance activities, heat exchanger cleaning, etc.; Represents manufacturer's limit without vibration concerns)	6,840
- 31 Day (erosion and vibration limit for long-term post-accident recirculation)	7,200
- Initial Safety Injection (erosion and vibration limit)	7,500
- Maximum Allowable Unit 4 Flow Limits (gpm) :	
- Continuous Operation (TEMA limit)(1)	6,756
- Normal Plant Evolutions (testing, surveillance activities, heat exchanger cleaning, etc.; Represents manufacturer's limit without vibration concerns)	8,000
- 31 day (erosion limit)	11,900
- Design Temperature, °F	200
- Design Pressure, psig	150
- Material	Carbon Steel
Tube Side (intake cooling water) :	
- Inlet Temperature, (Nominal Design), °F	95(2)
- Outlet Temperature, (Nominal Design), °F	107
- Design Flow Rate, lb/hr	4.0 x 10^6
- Design Pressure, psig	100
- Design Temperature, °F	200
- Material	Aluminum Brass

NOTES :

1. Tubular Exchanger Manufacturers Association.
2. The inlet temperature of 95°F can be exceeded if heat exchanger performance monitoring demonstrates the ability to remove postulated post accident heat loads at the elevated inlet temperature.
3. This represents the heat removal rate at the stated accident conditions. Actual heat removal will vary as a function of tube side and shell side flow rates, ICW temperature, accident conditions, and heat exchanger fouling.

Component Cooling Water Pumps	
Quantity	3
Type	Horizontal
Centrifugal	
Rated capacity, gpm, each	7500
Rated head, ft H_2O	185
Motor horsepower, hp	450
Casing material	Cast Iron
Design pressure, psig	150
Design temperature °F	200
Component Cooling Water Surge Tank	
Quantity	1
Volume, gal	2000
Normal water volume, gal.	1000
Design pressure, psig	100
Design temperature, °F	200
Construction material	Carbon Steel
Component Cooling Head Tank	
Quantity	1
Volume, gal	300
Normal water volume, gal	150 (approx.)
Design pressure, psig	100
Design temperature, °F	200
Construction material	Stainless steel
Chemical Pot Feeder Tank	
Quantity	1
Volume, gal	3
Design pressure, psig	150
Design temperature, °F	200
Component Cooling Water Loop Piping and Valves	
Design pressure (except for the excess letdown heat exchanger piping between valves *-738 and CV-*-739) and (except for the thermal relief valves RV-700A/B piping between valves 826A/D and 826B/E), psig	150
Design pressure of the piping between valves *-738 and CV-*-739), psig	200
Design pressure of thermal relief valve RV-700A piping between valves 826A and 826D, psig	200
Design pressure of thermal relief valve RV-700B piping between valves 826B and 826E, psig	200
Design temperature, °F	200

C29

C29

SCS Heat Exchanger	
- Quantity	1
- Type	Plate
Chilled Water Side:	
- Design Temperature, °F	284
- Design Pressure, psig	150
CCW Side:	
- Design Temperature, °F	284
- Design Pressure, psig	150
SR SCS Pump	
- Quantity	1
- Type	Centrifugal
- Rated Capacity, gpm	1000
- Rated Head, ft H_2O	260
- Motor Horsepower, hp	100
- Design Pressure, psig	200
- Design Temperature, °F	200
NNS SCS Pump	
- Quantity	2
- Type	Centrifugal
- Rated Capacity, gpm	1680
- Rated Head, ft H_2O	100
- Motor Horsepower, hp	60
- Design Pressure, psig	175

C29

TABLE 9.3-2

RESIDUAL HEAT REMOVAL LOOP COMPONENT DATA

Reactor coolant temperature at startup of residual heat removal, °F	350
Time to cool reactor coolant system from 350°F to 140°F, hr. (all equipment operational)	20 (NOTE 1)
Refueling water storage temperature, °F	Ambient
Decay heat generation at 20 hrs after shutdown condition, Btu/hr.	53×10^6 (NOTE 2)
Approximate refueling cavity fill time, hr.	1
Approximate refueling cavity drain time, hr.	10
H_3BO_3 concentration in refueling water storage tanks, ppm boron	$\geq$ 2400
Residual heat removal pumps	
Quantity	2
Type	Vertical Centrifugal
Rated capacity, gpm, each	3,750
Rated head, ft H_2O	240
Motor horsepower, hp	300
Material	Stainless Steel
Design pressure, psig	600
Design temperature, °F	400
Residual Heat Removal Heat Exchangers	
Quantity	2
Type	Shell and U-tube
Heat transfer, Btu/hr	29.4×10^6

C26

NOTE:

1. Cool down time for Extended Power Uprate was evaluated to be 28 hours to 200°F under the most restrictive operating parameters. The maximum calculated time to reach 140°F was 107 hours.

2. This is a typical value of decay heat generation. The actual value varies slightly between units and fuel cycles.

3. RHR heat exchangers are rated for 600 psig but analyzed for structural capability to withstand an overpressure excursion up to 720 psig under OMS conditions.

C26

Shell side (component cooling water)	
Inlet temperature, °F	108
Outlet temperature, °F	115
Design flow rate, lb/hr.	4.31×10^6
Design pressure, psig	150
Design temperature, °F	200
Material	Carbon Steel
Tube side (reactor coolant)	
Inlet temperature, °F	140
Outlet temperature, °F	124
Design flow rate, lb/hr.	1.87×10^6
Design pressure, psig	600 (NOTE 3)
Design temperature, °F	400
Material	Stainless Steel

C23

TABLE 9.3-3

DELETED

TABLE 9.3-4

AUXILIARY COOLANT SYSTEM

CODE REQUIREMENTS

Component cooling water heat exchangers	ASME VIII*
Component cooling water surge tank	ASME VIII
Component cooling water head tank	ASME VIII
Component cooling water system piping and valves	USAS B31.1**
Component cooling (SCS) heat exchanger	ASME Section VIII, 1965 edition, through Winter 1966 addenda****
Component cooling (SCS) pump	API STD 610, Eighth Edition****
Component cooling (SCS) piping	ANSI B31.1 1973 (through Winter 1976 Addenda)****
Component cooling (SCS) AOV (CV-2216)	ANSI B16.34, 1973 or Current Edition****
Component cooling (SCS) valves	ASME Section III, 1980 Edition, Summer 1982 Addenda for Seismic Class 3 valves or ASME/ANSI B16.34, either of which being consistent with the piping code****
Residual heat removal heat exchangers	
Tube side	ASME III***, Class C
Shell side	ASME VIII
Residual heat removal piping and valves	USAS B31.1
Spent fuel pit filter	ASME III, Class C
Spent fuel pit heat exchanger	
Tube side	ASME III, Class C
Shell side	ASME VIII

Supplemental spent fuel pit heat exchanger	
Tube side	ASME III, Class 3
Shell side	ASME III, Class 3
Spent fuel pit demineralizer	ASME III, Class C
Spent fuel pit loop piping and valves	USAS B31.1

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* ASME VIII - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section VIII

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

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

**** Based on manufacturer's standards selected in procurement specification for CCW SCS. Equivalent or better substitutions acceptable. C29

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 is designed for 150 psi and 200°F which exceeds maximum operating conditions. Pump is inspectable and protected against missiles. Rupture due to missiles is not considered credible. Each unit is isolable. One of the three pumps can carry the total emergency heat load.
2. Component cooling water pumps	Pump fails to start	One operating pump supplies sufficient water for emergency cooling.
3. Component cooling water pumps	Manual valve on a pump suction line closed	This is prevented by prestartup 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 pumps	Valve on discharge line sticks closed	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. Two units can carry total emergency heat load.
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 prestartup and operational checks. On the operating unit such a situation is readily assessed by makeup requirements to system. Low level annunciation on the CCW head tank is used to identify system leakage.
8. Component cooling water valve from 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.

Components	Malfunction	Comments and Consequences
9. CCW Supplemental Cooling System (SR supplemental loop)	Loss of pressure boundary (Tornado Missile Impact)	The cooling provided by the SCS is a NNS function to reduce containment temperatures during normal operation and outage conditions. The SCS pump trips and the system is automatically isolated on low level in the CCW Head Tank. C29
10. CCW Supplemental Cooling System (SR supplemental loop)	High temperature at the SCS heat exchanger discharge header	The SCS pump trips and the system is automatically isolated on high temperature in the discharge header from the SCS heat exchanger. High temperature here is indicative of CCW functioning to cool accident heat loads.

ALLOWABLE RANGE OF FLOWS THROUGH COMPONENTS
IN POST-LOCA RECIRCULATION MODE

Component	Minimum Flow GPM	Maximum Flow GPM	
Containment Spray Pump Seal Cooler	5.6	20	Note 7
RHR Heat Exchanger	2,500	10,400	Note 1
HHSI Pump Seal and Bearing Coolers	7.0	20	Note 8
RHR Pump Seal Cooler	5.6	30	
Emergency Containment Cooler	2000	3,200	Note 3
Spent Fuel Pit Heat Exchangers	1200	3,080	Note 9
Non-Regenerative Heat Exchanger	---	1,235	Notes 1, 2
Charging Pump Hydraulic Coupling Cooler	50	57	Note 6
Seal Water Heat Exchanger	---	227	Note 2
Sample Cooler	---	44	Note 2
Waste Gas Compressor Seals	---	90	Note 2
Unit 3 CCW Hx Shell Side	---	7,200	Note 4
Unit 4 CCW Hx Shell Side	---	11,900	Note 5

C26 C26 C26

NOTES:

1. The maximum RHR and Non-Regenerative heat exchanger flow rates are component flow limits and not post-LOCA recirculation mode flow limits. C26

2. Only maximum flows are of interest for the non-essential components which may not be isolated during an accident.

3. 3,200 gpm is acceptable for the Post-LOCA recirculation period of one month. Refer to Table 6.3-1 for additional time dependent acceptable flow rates.

4. 7,200 gpm is a 31 day erosion / vibration limit for long-term post-accident recirculation. 7,500 gpm is acceptable for 20 minutes, 6,840 gpm for normal plant evolutions, and 4,063 gpm for continuous operation.

5. 11,900 gpm is a 31 day erosion limit for long-term post-accident recirculation. 8,000 gpm is acceptable for normal plant evolutions, and 6,756 gpm is acceptable for continuous operation.

6. Previous values were 20 gpm min. and 60 gpm max. C26

7. (Unit 4) New FLOWSERVE seal coolers installed via PC/M 07-081 are limited to a maximum flow of 17 gpm.

8. The Safety Injection Pump Seal Cooler could encounter a CCW maximum flow rate of 36 gpm during a LOCA.

9. 3080 gpm bounds the maximum flow for each spent fuel pit heat exchanger. C26

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.3-1

REFER TO ENGINEERING DRAWING

5613-M-3030 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

COMPONENT COOLING WATER SYSTEM

FIGURE 9.3-1

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.3-2

REFER TO ENGINEERING DRAWING

5613-M-3030 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 3

COMPONENT COOLING WATER SYSTEM

FIGURE 9.3-2

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.3-3

REFER TO ENGINEERING DRAWING

5613-M-3030 , SHEET 3

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

COMPONENT COOLING WATER SYSTEM

FIGURE 9.3-3

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.3-4

REFER TO ENGINEERING DRAWING

5613-M-3030 , SHEET 4

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

COMPONENT COOLING WATER SYSTEM

FIGURE 9.3-4

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.3-5

REFER TO ENGINEERING DRAWING

5613-M-3030 , SHEET 5

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

COMPONENT COOLING WATER SYSTEM

FIGURE 9.3-5

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.3-6

REFER TO ENGINEERING DRAWING

5614-M-3030 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNIT 4**
COMPONENT COOLING WATER SYSTEM **FIGURE 9.3-6**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.3-7

REFER TO ENGINEERING DRAWING

5614-M-3030 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

COMPONENT COOLING WATER SYSTEM

FIGURE 9.3-7

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.3-8

REFER TO ENGINEERING DRAWING

5614-M-3030 , SHEET 3

REV. 13 (10/96)

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TURKEY POINT PLANT UNIT 4

COMPONENT COOLING WATER SYSTEM

FIGURE 9.3-8

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.3-9

REFER TO ENGINEERING DRAWING

5614-M-3030 , SHEET 4

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

COMPONENT COOLING WATER SYSTEM

FIGURE 9.3-9

9.4 SAMPLING SYSTEM

9.4.1 DESIGN BASES

Performance Requirements

This system provides samples for laboratory analysis to evaluate reactor coolant, and other reactor auxiliary systems chemistry during normal operation. It has no active emergency function. This system is normally isolated at the containment boundary.

Sampling system discharge flows are limited under normal and anticipated fault conditions (malfunctions or failure) to preclude any fission product releases beyond the 10 CFR 20 guidelines. Each unit has an identical sampling system and no equipment is shared between units except the drains and vents to the Waste Disposal System. The description contained herein is equally applicable to either unit.

Design Characteristics

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

Sampling of other process coolants, such as tanks in the Waste Disposal System, is accomplished locally. Equipment for sampling secondary and non-radioactive 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.

Two types of samples are obtained by the system: high temperature -high pressure Reactor Coolant System samples which originate inside the reactor containment, and low temperature -low pressure samples from the Chemical and Volume Control and Auxiliary Coolant Systems.

High Pressure - High Temperature Samples

A sample connection is provided from each of the following:

a) The pressurizer steam space
b) The pressurizer liquid space
c) Hot legs of loops A and B

Low Pressure - Low Temperature Samples

A sample connection is provided from each of the following:

a) The mixed bed demineralizer inlet header
b) The mixed bed demineralizer outlet header
c) The residual heat removal loop, just downstream of the heat exchangers
d) The volume control tank gas space
e) The accumulators

Expected Operating Temperatures

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

Codes and Standards

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

9.4.2 SYSTEM DESIGN AND OPERATION

The Sampling System, shown in Figures 9.4-1 and 9.4-2, provides the representative samples for laboratory analysis. Analysis results provide guidance in the operation of the Reactor Coolant, Auxiliary Coolant, Steam and Chemical and Volume Control Systems. Analyses show both chemical and radiochemical conditions. Typical information obtained includes reactor coolant boron and chloride concentrations, fission product radioactivity level, hydrogen, oxygen, and fission gas content, corrosion product concentration, and chemical additive concentration.

The information is used in regulating boron concentration adjustments, evaluating fuel element integrity and mixed bed demineralizer performance, and regulating additions of corrosion controlling chemicals to the systems. The Sampling System is designed to be operated manually, on an intermittent basis. Samples can be withdrawn under conditions ranging from full power to cold shutdown.

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

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

Reactor coolant hot leg liquid, accumulator liquid, pressurizer liquid and pressurizer steam samples originating inside the containment flow through separate sample lines to the sampling room. Each of these connections to the Reactor Coolant System has a remote operated isolation valve located close to the sample source. The samples pass through the containment to the auxiliary building, and into the sampling room, where they are cooled (pressurizer steam samples condensed and cooled) in the sample heat exchangers. The sample stream pressure is reduced by a manual throttling valve located downstream of each sample pressure vessel. The sample stream is purged to the volume control tank in the Chemical and Volume Control System or to the Waste Holdup Tank in the Waste Disposal System until sufficient purge volume has passed to permit collection of a representative sample. After sufficient purging, the sample pressure vessel is isolated and then disconnected for laboratory analysis of the contents.

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

The reactor coolant sample originating from the residual heat removal loop of the Auxiliary Coolant System has a remote operated, normally closed isolation valve located close to the sample source. The sample line from this source is connected into the sample line coming from the hot leg at a point upstream of the sample heat exchanger. Samples from this source can be collected either in the sample pressure vessel or at the sample sink as with hot leg samples.

Liquid samples originating at the Chemical and Volume Control System letdown line at the mixed bed demineralizer inlet and outlet pass directly through the purge line to the volume control tank. Samples are obtained by diverting a portion of the flow to the sample sink. If the pressure is low in the letdown line, the purge flow is directed to the chemical drain tank. The sample line from the gas space of the volume control tank delivers gas to the gas sample tree in the gas analyzer room adjacent to the primary chemistry lab. Valve alignments can be made to send gas to grab sample collection in the gas analyzer room, for isotopic analysis, or to the sample extraction board in the primary chemistry lab, for gas concentration analysis. An alternate sample line from the gas space of the volume control tank delivers gas samples to the volume control tank sample pressure vessel in the sampling room.

The sample sink, which is located in the sampling room, contains a drain line to the Waste Disposal System.

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

Components

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

Sample Heat Exchangers

Three sample heat exchangers reduce the temperature of samples to 127°F before samples reach the sample vessels and sample sink. The tube side of the heat exchangers is Alloy 600 or austenitic stainless steel as shown on Table 9.4-2, while the shell side is carbon steel.

The inlet and outlet tube sides have Swagelok high pressure fittings (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.

Delay Coil

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

Sample Pressure Vessels

The high pressure sample trains, the residual heat removal loop sample train and the volume control tank gas space sample train each contain sample pressure vessels which are used to obtain liquid or gas samples. The hot leg and the residual heat removal loop sample lines have a single sample pressure vessel in common. Integral isolation valves are furnished with the vessel and quick-disconnect coupling valves containing poppet-type check valves, are connected to nipples extending from the valves on each end. The vessels, valves and couplings are austenitic stainless steel.

Sample Sink

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

Piping and Fittings

All liquid and gas sample lines are austenitic stainless steel tubing and are designed for high pressure service. Socket weld fittings and compression fittings are used in the Sampling System, Both, inside and outside the containment. Quick disconnect couplings are used on sample vessels and on in-line analyzers. Lines are so located as to protect them from accidental damage during routine operation and maintenance.

Valves

Remotely operated stop valves are used to isolate all sample points and to route sample fluid flow inside the reactor containment. Manual stop valves are provided for component isolation and flow path control at all normally accessible Sampling System locations. Manual throttle valves are provided to adjust the sample flow rate as indicated on Figures 9.4-1 and 9.4-2.

Check valves prevent gross reverse flow of gas from the volume control tank into the sample sink.

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

An isolation valve is provided outside the containment on all sample lines leaving the containment, which fails closed and automatically closes upon actuation of the containment isolation signal.

9.4.3 SYSTEM EVALUATION

Incident Control

The system operates on an intermittent basis, and under administrative manual control.

Malfunction Analysis

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

TABLE 9.4-1

SAMPLING SYSTEM CODE REQUIREMENTS

Sample heat exchanger [3]	ASME III [1], Class C, tube side ASME VIII, shell side
Sample pressure vessels	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. All three sample heat exchangers for Unit 4 and the pressurizer liquid sample heat exchanger for Unit 3 conform to ASME III Class 3, 1977 Edition, including the Summer 1977 Addenda.

TABLE 9.4-2

SAMPLING SYSTEM COMPONENTS

Sample Heat Exchanger [1]

General

Number	3
Type	Counter flow
Design heat transfer rate (duty for 652.7°F sat. steam to 127°F liquid), each, Btu/hr	2.12×10^5

Shell

Design pressure, psig	150
Design temperature, °F	350
Component cooling water flow, gpm	40
Pressure loss at 40 gpm, psi	25
Operating cooling water temperature, inlet, °F	105
Operating cooling water temperature, outlet (maximum), °F	130
Material	Carbon steel

Tubes

Tube diameter, O.D., in.	3/8
Design pressure, psig	2485
Design temperature, °F	680
Sample flow, normal, each, lb/hr	209
Maximum allowable pressure loss, each 209 lb/hr, psi	10
Operating sample temperature, inlet (maximum), °F	652.7
Operating sample temperature, outlet (maximum), °F	127
Material	Inconel 600, Austenitic stainless steel for Unit 3 pressurizer steam heat exchanger and Unit 3 reactor coolant heat exchanger.

NOTES :

1. All three sample heat exchangers for Unit 4 and the pressurizer liquid sample heat exchanger for Unit 3 meet or exceed the criteria specified.

Sample Pressure Vessels	
Number, total	5
Volume, pressurizer steam sample, ml	75
Volume, pressurizer liquid sample, ml	75
Volume, reactor coolant hot leg sample, ml	75
Volume, volume control tank sample, ml	75
Design pressure, psig	2485
Design temperature, °F	680
Material	Austenitic Stainless
Steel	
Manual Throttle Valves	
Normal operating temperature, °F	120-130
Design pressure, psig	2485
Body design temperature, °F	680
Piping	
Liquid and gas sample line internal	
diameter, in.	0.245
Design Pressure, psig	2485
Design temperature, °F	680

TABLE 9.4-3

MALFUNCTION ANALYSIS OF SAMPLING SYSTEM

Sample Trains	Malfunction	Comments and Consequences
Pressurizer steam space sample, pressurizer liquid space sample, accumulator or hot leg sample	Remotely operated sampling valve inside containment fails to close	A diaphragm-operated valve on each line outside the containment is closed on containment high pressure signal.
Inside containment sample trains	Sample line break inside containment upstream of remotely operated valve	A diaphragm-operated valve on each line outside the containment is closed on containment high pressure signal.

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.4-1

REFER TO ENGINEERING DRAWING

5613-M-3036 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

NUCLEAR STEAM SUPPLY SYSTEM
SAMPLE SYSTEM

FIGURE 9.4-1

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.4-2

REFER TO ENGINEERING DRAWING

5614-M-3036 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNIT 4**
NUCLEAR STEAM SUPPLY SYSTEM SAMPLE SYSTEM **FIGURE 9.4-2**

9.5 FUEL STORAGE and HANDLING

The Fuel Storage and Handling System provides a safe effective means of storing, transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it is stored at the Independent Spent Fuel Storage Installation (ISFSI) or leaves the plant after post-irradiation cooling. The system is designed to minimize the possibility of malfunction that causes fuel damage and potential fission product release. C28

The Fuel Storage and Handling System includes the new and spent fuel storage racks, refueling cavity, the spent fuel pit, spent fuel pit cooling and purification system and the fuel transfer system.

The ventilation system for new and spent fuel storage is described in Section 9.8.

Dry storage of spent fuel pursuant to 10 CFR 72 is provided as discussed in Section 1.2.10.

C28

The following sections related to fuel storage and handling under 10 CFR 50.

The Technical Specification 5.5.1, “Criticality,” critical design features for the storage racks are identified in Sections 9.5.1 and 9.5.2 and associated tables. These critical design features are highlighted to better ensure continued compliance with Technical Specification 5.5.1. The identification of these critical design features of the storage racks is a corrective action to prevent recurrence from the root cause evaluation of AR 403641 and shall be maintained in the UFSAR consistent with the licensing basis criticality analysis.

9.5.1 NEW FUEL STORAGE

9.5.1.1 Design Basis

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. (1967 Proposed GDC 66)

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. (1967 Proposed GDC 69)

The new fuel storage racks are designed to :

a) store up to 54 fuel assemblies [Technical Specification 5.5.1 critical design feature];

b) provide sufficient spacing [Technical Specification 5.5.1 critical design feature] between fuel assemblies to maintain a subcritical array ($k_{eff} \leq 0.95$) during flooding with unborated water; and

c) preclude the possibility of storing a fuel assembly in other than prescribed locations [Technical Specification 5.5.1 critical design feature].

Additionally, upon implementation of the license amendments of the spent fuel pool cask racks (Reference 14), Turkey Point elected to comply (Reference 15) with the requirements of 10CFR50.68(b) (Reference 11) which includes restrictions on the reactivity of stored fresh (new) fuel.

9.5.1.2 System Description

Table 9.5.1 provides the relevant design data for new fuel storage. The new fuel storage facility is located in the Auxiliary Building in the new fuel room. New fuel is stored dry at a floor elevation of 18'6" with the top of the rack at elevation 31'3".

The new fuel storage area is sized for storage of the fuel assemblies normally associated with the replacement of one-third of a core. New fuel storage for each unit consists of an "L" shaped rack with 54 square cavities arranged in a three deep array. New fuel assemblies are stored in a vertical position within each cavity with a center-to-center spacing between the fuel assemblies of 21" [Technical Specification 5.5.1 critical design feature]. The top of each cavity is separated from the adjacent cavity by metal decking such that fuel assemblies may only be placed in the designated spaces [Technical Specification 5.5.1 critical design feature]. The size of each cavity is approximately 9.5" x 9.5" [Technical Specification 5.5.1 critical design feature].

The new fuel storage room is protected from missile impact by a concrete roof and concrete walls. The entry into the new fuel storage facility consists of a roll-up metal door, which is not missile resistant. However, the only safety related components which are exposed to missile impact through the door opening are new fuel assemblies. Since these assemblies are not irradiated, they do not present a potential for uncontrolled release of radioactivity in excess of 10CFR50.67 guidelines.

The method of transferring new fuel to the new fuel storage room and into the new fuel storage racks is discussed in Section 9.5.4

9.5.1.3 System Evaluation

The new fuel storage racks store new fuel vertically in an array that maintains $K_{eff} \leq 0.95$, even if flooded with unborated water. This is accomplished by a center-to-center spacing of 21" between the fuel assemblies when stored in the new fuel storage rack. Since the new fuel storage rack is designed with metal decking between storage cavities, the placement of fuel assemblies in other than prescribed locations is not possible.

The new fuel storage facilities are designed as Class 1 Structures, Systems and Equipment and built in accordance with applicable codes. Chapter 5, Appendix 5A details the Seismic Classification and Design Basis for the storage facility.

9.5.2 SPENT FUEL STORAGE

9.5.2.1 Design Basis

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. (1967 Proposed GDC 66)

Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities. (1967 Proposed GDC 68)

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. (1967 Proposed GDC 69)

The spent fuel storage racks are designed:

a) to allow storage of no more than 1510 fuel assemblies in the spent fuel storage pit racks (286 fuel assemblies in Region I racks and no more than 1093 fuel assemblies in Region II racks and 131 fuel assemblies in the Region I cask area rack); C26

b) to maintain subcritical conditions with a K_{eff} of less than 1.0 with unborated water in the spent fuel pit;

c) to maintain subcritical conditions with a K_{eff} of less than or equal to 0.95 with a specified level of soluble boron;

d) to preclude the possibility of storing a fuel assembly in other than prescribed locations [Technical Specification 5.5.1 critical design feature];

e) to allow spent fuel cooling by the spent fuel pit cooling system;

f) in accordance with the NRC, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978 (as amended by the NRC letter dated January 18, 1979) and SRP Section 3.8.4[3];and

g) to Seismic Category I requirements, and are classified as ANS Safety Class 3 and ASME Code Class 3 Component Support Structures.

Additionally, upon implementation of the license amendments of the spent fuel pool cask racks (Reference 14), Turkey Point elected to comply (Reference 15) with the requirements of 10CFR50.68(b) (Reference 11) which includes restrictions on the reactivity of stored spent fuel.

9.5.2.2 System Description

Auxiliary Building

The Auxiliary Building (including the spent fuel portion) is designed to Seismic Category I requirements. The spent fuel storage portion of the Auxiliary Building has reinforced concrete walls, floors and roof with interior partitions designed to provide plant personnel with the necessary radiation shielding and to protect the equipment from the effects of adverse atmospheric conditions including hurricane and tornado winds, temperature, external missiles, and corrosive environment. The walls and floors of the spent fuel pit are lined with stainless steel. The spent fuel storage structure with the augmented storage capability has been analyzed for the loads and loading combinations listed in Chapter 5, Appendix 5A for Class I structures. The increased loading due to additional storage of spent fuel elements in the pit and cask area was included in the analysis.

The analysis results confirmed that the structural integrity of the spent fuel storage area of the Auxiliary Building, when subjected to an increased number of stored fuel assemblies, including the installation of Metamic® inserts in Region II Racks, is maintained under all required load combinations (References 1 and 17).

Spent Fuel Pit

The fuel storage portion of the Auxiliary Building is shown on Figures 9.5-3, 9.5-4 and 9.5-5. The spent fuel pit is located in the Auxiliary Building and is designed for the underwater storage of up to 1535 fuel assemblies in the spent fuel pit (approximately 9 full cores) including 131 spent or fresh fuel assemblies in the cask area rack and miscellaneous fuel handling tools. The spent fuel pit contains both Region I and Region II racks. The spent fuel rack layout is shown on Figure 9.5-6. C26

The combined number of fuel assemblies loaded into the spent fuel pool storage racks and the cask pit rack is restricted to no more than the capacity of the spent fuel pool storage racks except during activities associated with reactor core offload/reload refueling activities (Reference 14).

From the 1118 storage cells in the Region II racks, 25 cells are removed from fuel storage service leaving 1093 cells available in Region II for fuel storage service. C26

These cells are located in or in close proximity to the cooling system return piping discharge flow path. Some of these cells in the discharge flow path are damaged by flow-induced vibration in such a manner that storage cells no longer conform to the center-to-center spacing requirements of the storage cells. The damaged cells and those cells in close proximity to the discharge flow path are restricted from fuel storage service (Reference 19) to assure fuel is not damaged and sub-criticality requirements are maintained.

Additional cooling capacity was added to support increased SFP heat load due to EPU. The added cooling flow required an impingement plate to be installed below the return pipe over the damaged cells. In order to reduce the return flow velocity adequately, the impingement plate extends beyond the damaged cells such that a total of 25 cells are removed from service. C26

Region I Spent Fuel Pit Racks

The spent fuel pit Region I racks consist of two (2) 8 x 11 modules and one (1) 10 x 11 module for a total of 286 storage locations. Region I is the high- enrichment, core off-load region. Region I racks in the spent fuel pit permit storage of 286 fresh and irradiated fuel assemblies.

The Region I storage racks are free-standing, seismically qualified components composed of individual storage cells made of stainless steel. These racks have a neutron absorbing material, Boraflex, which is attached to each cell. No credit is taken for Boraflex in the criticality safety analysis. The cells within a module are interconnected by grid assemblies to form an integral structure as shown in figure 9.5-7. Each Unit 3 rack module is provided with leveling screws which contact the spent fuel pool floor embedments and are remotely adjustable from above through the cells at installation. Each Unit 4 rack module has plates under the interior support pads rather than leveling screws. The modules are neither anchored to the floor nor braced to the pool walls. C26

A typical fuel rack assembly consists of three major sections which are the leveling screw, the lower and upper grid assemblies, and the cell assembly. The tops of the support plates are welded to the fuel rack base plate. The leveling screws transmit the loads to the pool floor embedments, provide a sliding contact, and provide for the leveling adjustment of the rack. The lower grid consists of box-beam members, side plates and the base plate. The bottom of the cell assembly is welded to the lower grid. The upper grid consists of box-beam members and side plates. The upper part of the cell assembly is welded to the upper grid. The upper and lower grid assemblies maintain the 10.6-inch centerline-to-centerline spacing between the cells [Technical Specification 5.5.1 critical design feature] and provide the structural connections between the cells to form a fuel rack assembly.

The major components of the cell assembly are the fuel assembly cell, the Boraflex (neutron absorbing) material for which no credit is taken in the rack criticality analysis, and the wrapper. The wrapper is attached to the outside of the cell by spot welding the entire length of the wrapper. The wrapper covers the Boraflex material and also provides for venting of Boraflex to the pool environment. Depending on the location within the rack array, some cells have a Boraflex/wrapper assembly on four sides, three sides or two sides.

Without credit for Boraflex as a neutron absorber, the placement of fuel into Region I is now controlled based on specific loading patterns, defined by four allowable 2x2 arrays. The use of RCCAs (see Table 9.5-22) as a neutron absorber is credited in two of the defined loading patterns. C26

Region II Spent Fuel Pit Racks

The Region II racks consist of three (3) 9 x 13 modules, one (1) modified 9 x 13 module, one (1) 9 x 14 module (minus 25 cells removed from fuel storage service), three (3) 10 x 13 modules and one (1) 10 x 14 module for a total storage capacity of 1093 fuel assemblies (see Figure 9.5-6). C26

The Region II storage racks are the same basic design as the spent fuel pit Region I racks with the following exceptions:

a) The cells are assembled in a checkerboard pattern with a 9.0-inch centerline-to-centerline spacing [Technical Specification 5.5.1 critical design feature] shown in Figure 9.5-8.

b) The cells are welded to the base support assembly and to one another to form an integral structure without use of grids as used in spent fuel pit Region I racks.

c) The Region II racks were manufactured with Boraflex poison panels installed; however, due to the degradation, Boraflex poison panels are no longer credited in the criticality analysis. As a consequence, criticality control within the Region II spent fuel racks is provided by following prescribed loading patterns which include specific fuel categories based upon enrichment, burnup and cooling time, along with the use of a combination of Rod Cluster Control Assemblies (RCCAs), water gaps and Metamic® inserts (see Table 9.5-23). Metamic® inserts are a metal matrix composite of aluminum and boron carbide. The Metamic® inserts are manufactured with a nominal boron carbide content of 0.0160 g/cm^2 and a minimum of 0.0150 g/cm^2 [Technical Specification 5.5.1 critical design feature]. Metamic® inserts will be installed in Region II racks cells between the fuel assembly and the inside cell wall (see Figures 9.5-14 and 9.5-15), when required by the specific loading pattern.

It is demonstrated in Reference 18 that by following the prescribed loading schemes along with the use of a combination of RCCAs, Metamic® and water gaps, Region II spent fuel rack criticality is controlled. Due to fabrication/installation issues, it is preferred that the Metamic® inserts be installed in the spent fuel pit rack resultant cells; however, there is no prohibition to installation in formed cells, when required. C26

Cask Area Rack

The cask area of the spent fuel pit is designed for the installation of a fuel transfer cask to allow fuel transfer operations. However, to provide increased fuel storage capability, a rack may be installed in the cask area when not performing fuel transfer operations. The Region I cask area rack is a 11 x 12 module with 131 storage locations (one location has been omitted to allow the placement of a fuel handling tool). The cask area rack is of the same basic design as the spent fuel pit Region I racks with the following exceptions:

a) The center-to-center spacing of the cells is 10.1” E-W and 10.7” N-S [Technical Specification 5.5.1 critical design feature](see Figure 9.5-9).

b) Boral panels are installed as a neutron absorber instead of Boraflex.

c) Bearing pads, 12” square, are installed between the rack leveling screws and the pit floor to provide sliding contact and distribute the rack weight.

When fuel transfer operations are necessary, fuel assemblies stored in the cask area rack will be relocated to the spent fuel pit racks and the cask area rack will be removed, decontaminated, as necessary, and placed in storage.

Controls for the installation of the cask area rack contain two hold points to ensure proper orientation of the racks in the spent fuel pool [Technical Specification 5.5.1 critical design feature]. Verification of the proper cask rack orientation is implemented by an authorized quality control inspector during installation of the racks to ensure consistency with associated spent fuel pool criticality analysis assumptions (Reference 18). C26

Radiation Protection

Radiation levels in the spent fuel storage area of the Auxiliary Building are controlled within limits during normal storage operations by maintaining a prescribed water level in the spent fuel pit to provide shielding. During reactor refueling, adequate shielding for radiation protection is provided by conducting all spent fuel transfer and storage operations underwater. This permits visual control of the operation at all times while maintaining low radiation levels, as low as reasonably achievable for periodic occupancy of the area by operating personnel. The spent fuel pit water level is indicated by a level transmitter which causes an audible alarm in the control room on high or low levels, and water removed from the pit must be pumped out since there are no gravity drains.

9.5.2.3 System Evaluation

Criticality Analysis

Criticality of fuel assemblies stored in the spent fuel pit storage racks is prevented in the following ways:

- In the cask area rack, a Region I storage rack, criticality is prevented by the design of the rack which limits fuel assembly interaction by fixing the separation distance between stored assemblies and/or by placing a neutron absorber panel between storage cells.

- The Region I and Region II spent fuel pit storage racks were manufactured with Boraflex poison panels; however, due to degradation, Boraflex is no longer credited in the criticality analysis. Currently, criticality is controlled in these racks by specific fuel storage patterns based on minimum burnup values for selected enrichments and cooling times and the use of a combination of RCCAs, Metamic® inserts and/or water gaps.

The design of the spent fuel pit racks incorporates the requirements of, and are in accordance with, USNRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, "April 14, 1978, as amended by the NRC letter dated January 18, 1979 and the applicable portions of the NRC Regulatory Guides, Standard Review Plan Sections and published standards as listed in Table 9.5-17.

The design criteria for preventing criticality in the spent fuel pit storage racks and the cask area rack are provided in Reference 11 and are based on the 95/95 rule, i.e., criticality calculations are performed with a 95% probability at a 95% confidence level. The criteria are as follows:

1. k_{eff} less than 1.0, without the presence of soluble boron;
2. k_{eff} less than or equal to 0.95 with the presence of a defined level of soluble boron in the spent fuel pit water.

Spent Fuel Pit Storage Racks

The criticality analysis of Reference 18 qualifies the spent fuel pit storage racks for the current and future inventory of spent fuel. Note that the cask area rack is addressed separately. The spent fuel pit storage rack analysis was based on:

- The storage of fuel assemblies in a 2 X 2 array
- Fuel categories determined by enrichment, burnup and post-irradiation cooling times
- Rules and restrictions for combining the 2 X 2 arrays and fuel categories to create acceptable loading patterns

The criticality safety analysis was performed using the three-dimensional SCALE Version 5.1 Monte Carlo code package and the two dimensional PARAGON lattice code.

C26

The methodology for depleting fuel assemblies to support burnup credit in the spent fuel pool criticality safety calculations includes the depletion of two-dimensional unit assemblies as an infinite array in reactor core geometry with PARAGON at the bounding reactor core conditions. Once the fuel assembly is depleted to a desired assembly-average burnup, it is allowed to decay to its most reactive state. The assembly-averaged isotopic concentrations from PARAGON are then brought to cold conditions, 68°F and 14.7 psi. The following assumptions are made during the depletion simulations.

- A bounding moderator temperature profile is assumed
- A bounding relative power profile is assumed
- Each fuel assembly contains a bounding amount of burnable poison absorber rods (BPRAs) and/or integral fuel burnable absorber (IFBA)
- A bounding soluble boron concentration is assumed

KENO V.a, part of the SCALE package was used for all criticality calculations. All references to KENO in this section refer to KENO V.a. In general, KENO three-dimensional calculations model a 2x2 array of cells surrounded by periodic boundary conditions. Additional KENO models with more than four cells and different boundary conditions are generated to investigate the effect of eccentric fuel assembly positioning, interfaces between racks, and to analyze accident conditions. The following assumptions are made during the criticality calculations.

- All calculations are performed using an explicit model of the fuel and storage cell geometry.
- Calculations are performed assuming an infinite radial array of fuel assemblies or assembly patterns. Specifically, all gaps between adjacent Region II rack modules are conservatively ignored, i.e., cells in neighboring Region II rack modules are assumed to be separated by a single cell wall only. The actual configuration in the Turkey Point spent fuel pool has a cell wall on each side of the Region II rack-to-rack gap. Region I rack-to-rack separation was also ignored.
- The three-dimensional KENO models assume 60 cm of unborated water above and below the active fuel length, even when soluble boron is credited in the analysis. This conservatively bounds the effect of any borated water displacement due to end fittings.
- All inserts are assumed to have the same orientation as is depicted in Figure 9.5-16. [Technical Specification 5.5.1 critical design feature]. C26
- All of the final two insert calculations are performed with the two inserts in the same row, a "parallel" arrangement. because this gives a more limiting (higher) k_{eff} results than a checkerboard arrangement.
- The normal operating temperature range of the Turkey Point spent fuel pools is accounted for in each storage array with the application of a temperature bias which includes Monte Carlo uncertainties.
- Empty cells credited in the analysis are modeled as containing only full density water [Technical Specification 5.5.1 critical design feature]. Note: Metamic inserts and RCCAs are allowed in empty cells.

Using the results of the criticality and bias and uncertainty calculations, loading curves are calculated for each 2x2 array. The analysis utilized conservative targets by ensuring that:

1. k_{eff} for all permissible storage configurations remains $\leq$0.94 when the storage racks are fully loaded with fuel of the highest permissible reactivity and the pit is flooded with borated water at the temperature corresponding to the highest reactivity.

2. k_{eff} remains < 0.99 if the pit is flooded with unborated full-density water at the temperature corresponding to the highest reactivity.

The maximum k_{eff} values are calculated with a 95% probability at a 95% confidence level and include all applicable biases and uncertainties. The acceptance criteria utilized in the development of the loading curves maintains a minimum 0.01 Δk (1000 pcm) margin to the regulatory requirements.

Applicable biases and uncertainties account for the effects of physical variations of the fuel and racks, characterization of fuel burnup, and the validation of KENO. Physical variations considered include the reactivity effects of manufacturing tolerance for the fuel, rack, and poison inserts, as well as the potential for eccentric loading of the fuel within the racks in addition to the effects of variation in pit water temperature. For storage arrays which credit fuel burnup uncertainties associated with fuel depletion, reactor records, fission product worth, and variation in reactor power operating history are also considered. A bias and bias uncertainty to validate KENO was also developed and included based on the comparison of KENO calculated results to laboratory criticality experiments. All biases and uncertainties used in the criticality analysis are discussed in detail in Section 4.3.2 of Reference 18. C26

For all Fuel Categories except I-1 and 1-2, an equation specifying the minimum required burnup as a function of the initial enrichment and post-irradiation cooling time is developed. The uncertainty in the burnup is included in the determination of the minimum burnup requirement and so it is appropriate to use the nominal burnup for comparing to the minimum required burnup determined from the loading curves. The burnup requirements are established as 3rd degree polynomial functions in the form of:

$$Bu = (A_1 + A_2*En + A_3*En^2 + A_4*En^3)* \exp [- (A_5 + A_6*En + A_7*En^2 + A_8*En^3)*Ct] + A_9 + A_{10}*En + A_{11}, *En^2 + A_{12}*En^3$$

where:

Bu = Minimum required assembly average burn up (GWd/MTU)
En = Initial Enrichment (Nominal ^{235}U Enrichment) (wt%)
Ct = Post Irradiation Cooling Time (years)
Ai = Coefficients (see Tables 9.5-19 and 9.5-20)

Separate functional relationships are developed for blanketed and non-blanketed fuel assemblies. Pre-EPU blanketed assemblies must use the EPU curves. Note that for blanketed assemblies, the enrichment to be used in the loading curve equation is the enrichment of the axial section between the blanket material (the enrichment of the axial blankets is excluded when determining the assembly enrichment for application of loading curve).

Since the loading curve is an exponential in cooling time, any cooling time between 0 and 25 years is allowed to be evaluated by the curve for blanketed fuel assemblies and between 15 years and 25 years for non-blanketed fuel assemblies. Fuel assemblies with cooling times greater than 25 years must conservatively use a value of 25 years.

The loading curves are valid for any enrichment between 2.0 and 5.0 wt.% for blanketed assemblies and between 1.8 and 4.0 wt.% for non-blanketed assemblies.

Coefficients for all loading curves, for both blanketed and non-blanketed assemblies are listed in Tables 9.5-19 and 9.5-20.

After the loading curves were developed, they were validated using confirmatory calculations. In addition to confirming the individual storage arrays, all interfaces between arrays within a region and between regions were confirmed to comply with the applicable regulatory requirements. C26

Calculations were also performed to confirm that a soluble boron concentration of 500 ppm in the spent fuel pit ensures that k_{eff} does not exceed the regulatory limit of 0.95 under normal conditions for all fuel storage arrays.

Cask Area Storage Rack

The approach taken for the cask area storage rack was different than that taken for the rest of the spent fuel pit storage racks. For the cask area storage rack, all fuel and rack dimensions are set to the worst case tolerances and positions. The fuel is modeled as fresh 5.0 wt.% ^{235}U. After the addition of the validation bias and bias uncertainty, the worst case k_{eff} is 0.9735 which is well below the regulatory limit for the unborated case. The 500 ppm soluble boron requirement was developed for the non-cask area storage arrays, which are more reactive when unborated. Therefore, it is clear that for the Cask Area Rack, when soluble boron is considered, the k_{eff} limit of $\leq$0.95 is satisfied.

Postulated Accidents Affecting Reactivity

The following reactivity increasing accidents are considered in this analysis:

- Misloaded fresh fuel assembly into incorrect storage rack location
- Inadvertent removal of an absorber insert
- Spent fuel pool temperature greater than normal operating range (150°F)
- Loss of water gap between Region I and Region II due to seismic event
- Dropped fresh fuel assembly
- Misplaced Fuel assembly in the spent fuel pit racks
- Misplaced Fuel assembly in the cask area storage rack
- Misloaded cask area storage rack
- Misloaded upender

The misloaded fresh fuel accident scenario is analyzed by placing a 5.0 wt.% ^{235}U fresh fuel assembly into the water-filled cell required by Array II-A. This is expected to be the bounding condition since the fresh assembly is being surrounded by the most reactive fuel allowed in Region II. A 6x6 model is utilized with periodic boundary conditions containing one misloaded fresh fuel assembly. A misload into one of the empty cells in Array I-A is also analyzed and found to be less limiting. This accident requires 1683 ppm of boron to maintain k_{eff} less than 0.95, including biases and uncertainties. C26

The removal of an absorber insert from an already analyzed array is bounded by the misload because the incorrectly placed assembly will be more heavily burned than the analyzed misload case, therefore this accident is covered.

The spent fuel pool is to be operated at less than 150 °F. However, under accident conditions this temperature could be higher. This condition is analyzed and found to be less limiting that the fresh misload case.

A seismic event could reduce the spacing between rack modules in the spent fuel pool. This accident scenario is analyzed by not modeling any water gap between a representative Region I and Region II interface model. Additionally, the fuel assemblies are eccentrically positioned toward the interface.

Results demonstrate a large margin to the regulatory limit, therefore the remaining interfaces are not analyzed.

During placement of the fuel assemblies in the racks, it is possible to drop the fuel assembly from the fuel handling machine. The dropped assembly could land horizontally on top of the other fuel assemblies in the rack. In this case, there is significant separation between the dropped fuel assembly and the rest of the fuel assemblies due to the top nozzle, fuel rod plenum, fuel rod end plug. and the separation between the fuel rod and the top nozzle. It is clear that the misloaded fresh fuel assembly described above is far more limiting than a single assembly lying horizontally on top of other assemblies in the rack. It is also possible that a fuel assembly could be dropped in its location with such force that the resultant fuel assembly deforms the support structure such that more of the fuel assembly is below the absorbers. The removal of an absorber insert represents 100% of the assembly below the absorber and this has already been evaluated to be non-limiting.

It is possible to misplace a fuel assembly in a location not intended for fuel. Any assembly placed outside of the racks is surrounded by water on at least two sides. The misloaded fresh assembly discussed above is surrounded by fuel on all four sides. The additional neutron leakage of the two sides not facing fuel ensures that this condition is bounded by the fresh misload event.

The Cask Area Rack has a corner where there is no storage cell box. It is possible, though very unlikely, that a fresh fuel assembly could be placed in this corner such that there is only one panel of Boral separating this misplaced fuel assembly from the fuel assemblies in the Cask Area Rack. This condition was analyzed and found to be bounded by the fresh misload event. C26

One side of the Cask Area Rack does not contain any Boral absorber because it is designed to face the pool wall. While it is considered extremely unlikely that the cask rack could be mis-positioned, if the entire rack is rotated 180 degrees, then the side with no Boral will be facing fuel assemblies in Region II. This condition was analyzed and found to be bounded by the fresh misload event.

When a fuel assembly is positioned in the upender, it possible to bring another fuel assembly in close approach to the upender. The condition of having two fully enriched fuel assemblies in direct contact with one another is bounded by the misload into the storage array I-A. While the assemblies are slightly closer together in the upender event, the I-A misload accident surrounds the misloaded assembly with fresh fully enriched fuel on all sides. The additional leakage associated with having no neighboring assemblies ensures that this accident is bounded.

For an occurrence of the postulated accident conditions, the double contingency principle of ANSI/ANS-8.1-1983 can be applied. It specifies that assumption of two unlikely, independent, concurrent events need not be considered to ensure protection against a criticality accident. Dilution of the boron concentration in the spent fuel pool and the misload of an assembly are two independent accidents. Therefore, for the accident conditions postulated, the presence of additional soluble boron in the storage pool water (above the concentration required for normal conditions) can be assumed as a realistic initial condition.

The spent fuel pool boron dilution analysis was performed to support the partial credit of soluble boron. The purpose of the analysis was to determine that sufficient time is available to detect and mitigate dilution of the spent fuel pool prior to exceeding the k_{eff} design bases limit of 0.95. The analysis utilized 1950 ppm as the spent fuel pool initial boron concentration and evaluated the dilution to 650 ppm. The dilution analysis concluded that an inadvertent or unplanned event that reduced the boron concentration below 650 ppm was not credible. The required initial spent fuel pool boron concentration has increased to 2300 ppm while the required boron concentration to maintain a k_{eff} of 0.95 has been reduced to 500 ppm. Therefore, the previous boron dilution analysis bounds the current boron limit in the Technical Specifications and the amount of soluble boron credited in the criticality safety analysis. C26

9.5.3 SPENT FUEL PIT COOLING AND PURIFICATION

9.5.3.1 Design Basis

The spent fuel pit cooling and purification system is designed to :

a) remove the residual decay heat generated by the spent fuel elements (9 full cores) stored in high-density fuel racks contained within each spent fuel pit;

b) maintain the spent fuel pool water temperature less than 150°F for planned refuelings;

c) maintain purity and optical clarity of the spent fuel pit water;

d) remain functional during and after seismic events; and

e) remain structurally intact at spent fuel pit water temperatures of 212°F.

9.5.3.2 System Description

The spent fuel pit cooling loop is designed to remove residual heat from fuel assemblies stored in the high-density storage racks contained within the spent fuel pit. Following completion of a normal refueling, up to 80 fuel assemblies are permanently discharged and stored in the spent fuel pit along with previously discharged fuel assemblies. During a typical refueling outage, the full core is off-loaded to the spent fuel pit. The cooling loop is capable of removing the decay heat from 1535 assemblies (1404 assemblies in the spent fuel pit storage racks and 131 assemblies stored in the cask area rack, when installed). The discharged spent fuel is stored in the spent fuel pit until it can be shipped off site. C26

The spent fuel pit cooling loop consists of pumps, heat exchangers, filters, a demineralizer, piping and associated valves and instrumentation. The pumps draw water from the spent fuel pit, circulate it through the heat exchangers and return it to the spent fuel pit. The component cooling water system cools the heat exchangers. C26

Spent fuel cooling loop component data is provided on Table 9.5-16. The cooling loop has been analyzed and is designed to remain functional during and following a seismic event and to structurally withstand a design temperature of 212°F. The available NPSH exceeds the pump NPSH requirement at the design flow for temperatures up to 209°F, for each of the installed pumps in the procedurally controlled normal system alignment. Loop piping is so arranged that failure of any pipeline does not drain the spent fuel pit to less than 6 feet above the top of the stored fuel elements. The clarity and purity of the spent fuel pit water is maintained by passing approximately 5% of the loop flow through filters and demineralizer. The spent fuel pit pump suction lines, penetrate 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. The return lines have a ½-inch hole in the pipe near the normal level for a siphon break function. C26

Since the spent fuel pit is located outside the containment, it is not affected by any loss-of-coolant accident in the containment. The water in the pit is isolated by a valve in the refueling canal during most of the refueling operation. Only a very small amount of interchange of water occurs as fuel assemblies are transferred during refueling.

Component Description

Spent Fuel Pit Heat Exchangers

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

Spent Fuel Pit Pumps

The spent fuel pit pumps circulate water in the spent fuel pit cooling loop. All wetted surfaces of the pumps are austenitic stainless steel, or equivalent corrosion resistant material. Pump operations are manually controlled from a local station. The cooling loop design does not incorporate redundant active cooling components (primarily pumps), because of the large heat capacity of the spent fuel pit and its corresponding slow heat-up rate. Nonetheless, two (2) 100% capacity pumps which are permanently piped into the spent fuel pit cooling system have been installed. One of these pumps running with one of the spent fuel pit heat exchangers is sufficient to cool the spent fuel pit during normal operations. During refueling activities, one or both pumps may be required based on actual conditions. An emergency pump is also connected to the suction and discharge lines of the 100% capacity pumps to be used in an emergency situations. C26 C26

Refueling Water Purification Pump

The refueling water purification pump circulates water in a loop between the refueling water storage tank and the spent fuel pit demineralizer and filters. All wetted surfaces of the pump are austenitic stainless steel. The pump is operated manually from a local station.

Spent Fuel Pit Skimmer Strainer

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

Spent Fuel Pit Filter

The three pit filters remove particulate matter from the spent fuel pit water. The filter cartridge is synthetic fiber and the vessel shell is austenitic stainless steel.

Spent Fuel Pit Demineralizer

The demineralizer is sized to pass 5% 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.

Spent Fuel Pit Skimmer

A skimmer pump and three filters are provided for surface skimming of the spent fuel pit water. Flow from this pump is returned to the spent fuel pit.

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.

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 to facilitate maintenance

9.5.3.3 System Evaluation

Thermal-Hydraulic

Thermal-hydraulic calculations have been performed in support of a 72 hour full core offload using the existing Region I and Region II spent fuel racks with specified loading patterns including the use of RCCAs, Metamic® inserts, and water cells and the installation and use of a cask area rack. The calculations determine the decay heat effects on the spent fuel pit bulk temperature, local cell/fuel assembly temperatures and time-to-boil on loss of cooling. The following discusses each of these analyses and provides the results.

Calculations were performed to demonstrate the capability of the spent fuel pit cooling system to remove the decay heat during a planned 72 hour full core offload and during an unplanned 72 hour full core offload with and without spent fuel pit cooling. The calculations considered the four different SFP heat-up cases derived from the Standard Review Plan (SRP). These cases address maximum normal and maximum abnormal heat load conditions. In support of a reduced decay time (72 hour offload), FPL has redefined the SRP cases to reflect the planned refueling practice of full core offloads.

The abnormal case is now interpreted to be an unplanned or emergency full core offload 36 days after a planned refueling shutdown. The SRP assumption of a 36 day period between shutdowns is retained in the updated analysis scenarios.

In keeping with the above, a planned refueling would offload the entire core (157 fuel assemblies) beginning at 72 hours. The postulated unplanned, forced shutdown scenario would also offload the entire core beginning at 72 hours after shutdown preceded by a 1/2 core offload (assumed to be 80 assemblies to bound one reload batch) 36 days earlier. A refueling temperature limit of 150°F for peak pool temperature was selected as the acceptance criteria during refuelings with full core off-loads. C26

SFP Bulk Temperature Analysis

The spent fuel pit bulk temperature analysis addresses the four different SFP heat-up cases derived from the Standard Review Plan (SRP).

The analysis for these cases and results are described below.

Case 1: Planned Refueling
Full core offload initiated at 72 hours after shutdown.

Case 2: Planned Operation
1/2 core offload with capacity inventory at 36 days after shutdown. C26

Case 3: Unplanned Operation with Spent Fuel Pool Cooling
Full core offload at 72 hours following a forced shutdown with 1/2 core recently offloaded (36 days after a planned refueling shutdown). C26

Case 4: Unplanned operation without Spent Fuel Pool Cooling
Full core offload at 72 hours following a forced shutdown with 1/2 core recently offloaded (36 days after a planned refueling shutdown) with loss of SFP cooling at bulk peak pool water temperature (time to boiling begins at bulk peak pool water temperature). C26

The planned refueling (Case 1) is evaluated at two different CCW temperatures. A low CCW temperature of 85°F is analyzed as Case 1a with a high fuel transfer rate of 8 fuel assemblies per hour. A second case is analyzed with CCW at a temperature of 105°F. This latter case is C29 designated Case 1b below and is analyzed with a fuel transfer rate of 6 fuel assemblies per hour. These two cases demonstrate the capability of the SFP cooling system at various CCW temperatures.

The following input parameters were used in the analysis:

Input Parameter	Value
Full Core Decay Heat Load	35.64 MBtu/hr at 72 hours 11.98 MBtu/hr at 36 days
Full Capacity (past refuelings) SFP Heat Load	5.29 MBtu/hr
CCW Inlet Temperature Case 1a Cases 1b, 2, 3, and 4	 85°F 105°F
CCW Flow Rate Cases 1a,1b, 3, and 4 Case 2	 2800 gpm (minimum) per heat exchanger 1200 gpm (minimum) per heat exchanger
SFP Cooling Flow Rate Cases 1a,1b, 3, and 4 Case 2	 2200 gpm (minimum) per heat exchanger 1600 gpm (minimum) per heat exchanger
Heat Exchanger Fouling Supplemental Heat Exchanger Fouling	0.000075 hr-ft^2-°F/Btu 0.0005 hr-ft^2-°F/Btu
Heat Exchanger Tube Plugging Allowance Supplemental Heat Exchanger Tube Plugging Allowance	0% 0.425%
Fuel Assembly Transfer Rate Case 1a Case 1b	 8 per hour 6 per hour
SFP Water Inventory	2,027,714lbm
SFP Water Make-up Rate	100gpm
Allowance for Existing Racks	1404 Fuel Assemblies
Allowance for Cask Area Rack	132 Fuel Assemblies

The analysis uses the same methodology and assumptions for heat exchanger performance as those used to support thermal power uprate (performed in 1996 under License Amendments 191 and 185 for Units 3 and 4, respectively).

The existing Heat exchanger effectiveness was quantified in 1996 to support the SFP cooling analyses at uprated conditions. Heat exchanger effectiveness was calculated using plant data obtained from the 1993 and 1994 Unit 4 refueling outages, and an empirically derived fouling factor of 0.000075 hr-ft^2-°F/Btu. The use of this fouling factor (in lieu of the design fouling factor used by the heat exchanger manufacturer for sizing purposes) is justified by the fact that tube side SFP water is continuously purified and slightly acidic and the shell side water is treated CCW. C26

Data collected during the recent 2002 Unit 4 refueling outage confirmed that there has been no observable change in heat exchanger performance compared to 1993/1994 data.

Minimum tube and shell side flow rates have been assumed in the analysis to conservatively model the existing SFP heat exchanger performance. The assumed flow rates are 10% lower than the operating flow rates during a refueling outage. This provides additional conservatism to account for potential future heat exchanger degradation, (e.g., fouling, tube plugging). No tube plugging is assumed in the heat-up analysis since no tubes are currently plugged in the original heat exchanger (after 30 years of heat exchanger operation). The supplemental heat exchanger effectiveness is based on vendor performance data. C26 C26

The decay heat values in the updated analysis were determined from the ORIGEN-S historic and projected burnup schedules. The 1/2 core is assumed to be 80 assemblies to bound one reload batch. C26

Acceptance criteria for the SFP bulk heat-up analysis:

a. The bulk maximum SFP temperature shall remain below 150°F from a full core offload during a planned refueling.

b. The bulk maximum temperature shall remain below 212°F during an unplanned offload evolution.

The 150°F acceptance criterion specified above for planned refuelings was established for the SFP cooling systems as part of the thermal power uprate. The 150°F value was based on a review of other plants' licensing requirements, and the first re-racking at Turkey Point (performed in 1977 under License Amendments 23 and 22 for Units 3 and 4, respectively).

It was applied during the analysis of the Turkey Point Unit 4, Cycle 16 (pre-uprate) full core offload. Accordingly, the 150°F temperature limitation represents a reasonable criterion for both partial and full core offloads for both Turkey Point spent fuel pits.

The 212°F acceptance criterion specified for unplanned offloads is representative of bulk SFP boiling conditions.

Results of the SFP bulk heat-up analysis:

<u>Case 1:</u> Planned Refueling

1a. The maximum expected SFP bulk temperature for a full core offload at 72 hours after shutdown is 120°F with a CCW inlet temperature of 85°F, and a transfer rate of 8 fuel assemblies per hour. C26

1b. The maximum expected SFP bulk temperature for a full core offload at 72 hours after shutdown is 139°F with a CCW inlet temperature of 105°F, and a transfer rate of 6 fuel assemblies per hour. C26

<u>Case 2:</u> Planned Operation

The maximum expected SFP bulk temperature for a 1/2 core offload with capacity inventory at 36 days after shutdown is less than or equal to 141 °F. C26

<u>Case 3:</u> Unplanned Operation with Spent Fuel Pool Cooling

The maximum expected SFP bulk temperature for a full core offload at 72 hours following a forced shutdown (36 days after a planned refueling shutdown) with 1/2 core recently offloaded is 148°F. This temperature assumes that the entire core is offloaded as a complete unit at 72 hours. The time to reach this maximum steady-state temperature with SFP cooling is 6 hours after offload . C26

<u>Case 4:</u> Unplanned Operation without Spent Fuel Pool Cooling

The maximum expected SFP bulk temperature for a full core offload at 72 hours following a forced shutdown (36 days after a planned refueling shutdown) with 1/2 core recently offloaded, with a subsequent loss of cooling, is 212°F. If SFP cooling were lost at the time of the peak pool temperature (148°F), the pool would reach boiling conditions in 2.7 hours. C26

The analysis for a planned refueling is provided below.

Heat-up Analysis results – Planned Refueling

Analysis	CCW Temperature	Peak SFP Temperature
72-hour Offload Case 1a	85°F	120°F
72-Hour Offload Case 1b	105°F	139°F

C26

As shown above, the analysis for a planned refueling predicts peak SFP temperatures < 150°F for a full core offload with CCW temperature at 105°F. C26

Procedural Controls

Plant procedures control the allowable off-load start time, fuel assembly off-load rate, and administrative SFP bulk temperature limit required to maintain pool temperature below 150°F. As indicated, administrative controls are used to ensure SFP bulk water temperature does not exceed 150°F during planned refuelings. The requisite controls include minimum off-load start time, maximum SFP bulk temperature, maximum fuel assembly transfer rate, the number of pumps and heat exchangers in operation, and the system operating alignment. These controls are incorporated into plant procedures governing reactor refueling. C26

The evaluated off-load schedule is also affected by CCW temperature. This variable, along with off-load start time and fuel assembly transfer rate, affects the SFP heat-up and the peak SFP water temperature. To address this process variable, plant operating procedures controlling minimum start time, maximum fuel assembly transfer rate, and the number of pumps and heat exchangers in operation relate these parameters to CCW temperature required to maintain SFP temperature below the 150°F limit. C26 An administrative bulk pool temperature limit is imposed to ensure that the 150°F limit is not exceeded after completion of off-load activities due to the inherent lag in SFP heat-up. The specified administrative limit will maintain SFP temperature below 150°F without intervening operator action. Bounding values are provided in procedures with an option to use cycle-specific values prior to commencing off-load activities.

Due to the many variables that can have an impact on SFP temperature, Turkey Point may elect to use a cycle-specific off-load start time and fuel assembly off-load rate in lieu of the bounding restrictions. Consideration will be given to the actual core power history, scheduled off-load start time, actual CCW temperature, predicted SFP heat exchanger performance and planned fuel assembly off-load rate in the establishment of the specific control values.

Regardless of whether cycle-specific or bounding off-load parameters are used for a particular refueling, plant procedures require that fuel transfer to the SFP be suspended if the administrative temperature limit is reached during off-load. Resumption of off-load activities would occur when the bulk SFP temperature decreases below the administrative limit and additional administrative requirements are satisfied. Note that Turkey Point may elect to perform a partial in-core shuffle in conjunction with administrative temperature controls, to complete a planned off-load without interruption and maintain bulk SFP water temperature below 150°F.

SFP Local Temperature Analysis

An analysis was performed to determine if the water in the storage racks will remain subcooled with the decay heat associated with 72-hour offload conditions.

Acceptance criteria for the SFP local thermal-hydraulic analysis:

a. The local maximum SFP temperature shall remain below the local saturation temperature of the water.

b. The maximum fuel cladding temperature in the SFP shall remain below the local saturation temperature of the water. If the maximum fuel cladding temperature exceeds the local saturation temperature of the water, a departure from nucleate boiling shall not occur.

In the SFP storage rack cells, decay heat from the fuel induces a natural circulation of water upward through the fuel assembly. Cooler water is supplied to the bottom of the rack cells through various flow holes. Water gaps or plenums between the racks and the SFP floor and walls allow the water from the area above the rack to flow to the inlet of the rack cells.

Fluid flows and temperatures within a rack cell loaded with fuel having a 72-hour decay time were determined by rigorous computational fluid dynamics (CFD) analysis. The CFD analysis was performed using the FLUENT™ fluid flow and heat transfer modeling program. A single bounding case was evaluated that includes the highest bulk temperature (150°F) and decay heat load, and conservative hydraulic parameters.

Key assumptions of the SFP local thermal-hydraulic analysis include:

- No downcomer flow exists between the individual storage rack modules.
- The hydraulic resistance of every rack cell in the SFP includes the hydraulic resistance that would result from a dropped Metamic® insert lying across the top of the rack which is more conservative than a dropped fuel assembly.
- A fouling of 0.0005 hr-ft^2-°F/Btu is imposed on the outside of the fuel rods to account for any crud layer.
- The maximum local water temperature (at the fuel rack exit) and peak fuel assembly heat flux (typically near the mid-height of the active fuel) occur coincidentally.

- The Metamic® inserts are assumed to be installed such that the flow area around the fuel assembly is minimized.
- The rack cell inlet temperature is equal to the SFP bulk temperature of 150°F.
- The flow area in the gaps between the Metamic® insert and the storage rack cell wall is not credited in the analysis.

Results of the SFP local thermal-hydraulic analysis:

Parameter	Value
Peak Local Water Temperature	185.5°F
Peak Fuel Cladding Temperature	252.2°F

C26

The saturation temperature of water in the SFP increases with increasing depth (pressure). The critical location for localized boiling in the fuel racks is at the top of the active fuel height. The minimum depth of water at the top of the active fuel height is 26.8 feet. At this water depth, the saturation temperature of water is 242.1°F. The calculated peak local water temperature is below the local saturation temperature; however, the calculated peak cladding temperature was determined to be greater than the calculated local saturation temperature; therefore, further evaluation was required to show that departure from nucleate boiling (DNB) did not occur. The evaluation determined that the maximum fuel assembly heat flux is approximately 3200 watt/meter^2. The minimum heat flux to cause DNB to occur is approximately 10^6 watts/meter^2; consequently, DNB will not occur and the local water temperature analysis was acceptable. C26

Time-To-Boil Analysis

Upon a loss of spent fuel pit cooling, the spent fuel pit will begin to heat-up due to decay heat. An analysis was performed to determine the time available before the spent fuel pit reaches 212°F (boiling) upon loss of spent fuel pit cooling following a 72 hour full core off-load. The acceptance criteria for that analysis was as follows:

a. The time to heat the SFP to 212°F after loss of SFP cooling during an unplanned off-load evolution shall be sufficient to permit alignment of available make-up sources.

b. The required make-up rate to replace water due to boiling shall be less than the existing rate of 100 gpm.

NOTE: the credited make-up source for the spent fuel pits is 100 gpm from the demineralizer water system. In the event that the demineralized water system is not available, alternate makeup can be provided via the seismic Category I refueling water storage tank, or via temporary (non-Category I) connections from the fire water system or primary water storage tank.

The analysis concluded that the time-to-boil for an unplanned full core off-load at 72 hours following forced shutdown (36 days after a planned refueling shutdown) with 1/2 core recently off-loaded was 2.7 hours assuming that SFP cooling is lost at the time of the peak pool temperature (148°F). If spent fuel pit cooling is lost, 2.7 hours is considered sufficient time for corrective action to be taken to initiate alternate cooling and/or makeup. The maximum boil-off (makeup) at 212°F for this condition is 99 gpm. This make-up rate is within the 100 gpm acceptance criteria established for the SFP bulk heat-up analysis. C26

In addition to borated makeup water available from the RWST, available unborated water sources for makeup to the SFP have been previously discussed in Attachment 6 to FPL letter L-99-176 in support of Amendments 206 and 200 for Turkey Point Units 3 and 4, respectively. As documented in L-99-176, the following unborated makeup sources satisfy the 100 gpm acceptance criterion: Demineralized Water System, Primary Water System (direct connection or local hose station) and the fire hose station outside the SFP.

Spent Fuel Pool Makeup

The increased spent fuel storage capability does not affect the design basis or functional requirements for makeup to the spent fuel pool. Makeup requirements for unexpected leakage are as summarized in the AEC (NRC) Final Safety Evaluation Report, Section 9.6.

A normal makeup flow rate to the spent fuel pool is 100 gpm provided from the demineralized water system.

In the event of a loss of the normal demineralized water makeup supply to the spent fuel pool, alternate makeup can be provided via the Seismic Class I refueling water storage tank or via temporary connections from the fire water system or the primary water storage tank. The supply piping from these sources is not seismically installed, since Turkey Point is located in a seismically inactive area far from any recorded damaging shocks (Reference: FSAR Section 2.11). There are approximately 180,000 gallons of water in the pool above the top of the spent fuel under normal conditions. If a leak should occur and the normal demineralized water supply becomes inoperable, a long period of time would be available before the water level would lower to the top of the spent fuel. For example, if the assumed leak were 100 gpm, 30 hours would be available to connect an alternate supply or to repair the normal source. In addition, should all sources of fresh or demineralized water become unavailable, sufficient time exists to install a temporary source using portable pumps and fire hoses using salt water as a makeup source. C26

Sufficient time is available to assure the viability of this alternate approach to that of the Regulatory Guides 1.13 and 1.29.

An analysis of the spent fuel pool in the boiling regime using salt water indicates that corrosion problems are not significant. At boiling temperatures, the oxygen content of water is extremely low such that stress corrosion is reduced greatly because necessary shielding corrosion products to form cathodic areas in the cracks cannot form due to lack of oxygen. That is, of the cathodic reactions on the passive surface that are required to sustain the anodic reaction within the crack, namely

$$2H_2O + 2e^- \text{------------->} H_2 + 2OH^-$$

$$1/2O_2 + H_2O + 2e^- \text{--------->} 2OH^-$$

the latter reaction is essentially eliminated.

The absence of the oxygen related cathodic reaction greatly reduces the anodic reactions within the crack, namely

$2CR + 3H_2O$-----------> $CR_2O_3 + 6H^+ + 6e^-$

$3H^+ + e^-$ ----------> $H(Fe) + H_2$ (companion cathodic reaction)

Propagation of cracks in the boiling water environment will be minimal.

Use of sea water will not result in unacceptable corrosion of the Zircalory-4 fuel cladding or structural components. Integrity of the fuel rod cladding and containment of fuel material and fission products is therefore assured. With regard to stainless steel structural components, it is unlikely that any localized corrosion cracking can result in loss of structural integrity of these components. Equipment and techniques can be made available to recover fuel rods.

Leakage Provisions

Whenever a failed fuel assembly is transferred from the fuel transfer canal to the spent fuel storage pool, a small quantity of fission products may enter the spent fuel cooling water. A small purification loop is provided for removing these fission products and other contaminants from the water. The probability of inadvertently draining the water from the cooling loop of the spent fuel pit is exceedingly low. The only means of draining the cooling loop is through 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 spent fuel pit cooling loop 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. The temperature and level indicators in the spent fuel pit would warn the operator of the loss of cooling. The slow heat-up rate of the spent fuel pit, as indicated in Table 9.5-16, would allow sufficient time to take any necessary action to provide adequate cooling using the emergency cooling connections provided while the cooling capability of the spent fuel pit cooling loop is being restored.

Incident Control

The most serious potential failure of the spent fuel pit cooling loop is complete loss-of-water in the storage pit when fuel is in the pit. To protect against this possibility, the cooling pump suction lines penetrate the pit wall and terminates near the normal water level so that a break in the pipe will not gravity drain the pit. The pit drain piping penetrates the pit wall at an elevation 6 feet above the top of the fuel assemblies. Complete siphon draining of the pit by a break in this line is prevented by a normally closed valve located near the pit wall at the same elevation as the penetration. A break in this line upstream of the valve will only drain the pool to an elevation 6 feet above the fuel assemblies. One of the cooling water return lines (8”) penetrates the pit wall about one foot below the normal water level, and the other return line (10”) goes into the pit from above the normal water level. To prevent siphon draining the pit a 1/2” inch hole is drilled in the 8” and 10” lines. C26

In the event of failure of one of the 100% capacity spent fuel pit cooling pumps the other 100% capacity pump is available to replace the failed pump. If the failure occurs during refueling, which requires both 100% pumps to be in operation, the emergency pump, which is connected to the suction and discharge lines of the 100% capacity pumps, is available to provide additional cooling. C26

Spent Fuel Pit Structural Analysis for Thermal Conditions

In support of spent fuel pit re-racking, analyses were performed to demonstrate the structural adequacy of the spent fuel pool and liner to the possible temperature gradients that might exist at elevated temperatures within the pool. The following provides an insight into and a summary of the analyses:

A temperature gradient (not a temperature difference) of 150°F (180°F inside pool and 30°F outside air) was considered in the structural analysis of the pool. Additional structural analyses have been conducted on the pool for a temperature gradient reflecting a 212°F water temperature and 30°F air temperature.

As part of the EPU SFP analysis due to higher fuel enrichment, the potential for Gamma heating of the concrete was evaluated. No additional heating of the pool concrete will occur provided freshly discharged fuel is not stored in the outer fuel rack row adjacent to the SFP Liner. C26

Except for the changes in water temperature, the analysis was identical to that described in response to NRC Questions 9, 10, and 11 (submitted to Florida Power and light via letter of August 13, 1984 - Reference 13).

The analysis conservatively assumes that sufficient time elapses to allow thermal equilibrium to be reached; the results are therefore independent of time. For the load case which includes thermal considerations but ignores the seismic event, the pool would be expected to remain at 212°F for only a short time before corrective action would result in reduced temperatures. During this time period, the likelihood of a seismic event, particularly at this site, is considered extremely remote. This load case results in stresses which are within the original design criteria allowables.

When both seismic and thermal conditions are assumed to occur simultaneously, there occur localized instances of reinforcing steel stresses slightly exceeding the allowable stress of 36 ksi. However, these minor localized occurrences (which take place only if seismic and thermal accident conditions occur simultaneously) cannot be construed as causing a loss of function or loss of structural integrity. The criteria included in the Updated FSAR (Appendix 5A, Section II) recognizes this fact by allowing such types of occurrences (including localized yielding) under certain conditions.

Additionally, the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsection CC-3000 (including Summer 1981 Addenda) recognizes the self-limiting nature of thermal stress in reinforced concrete and allows reinforcing strains in excess of yield strain under thermal accident conditions. Although this reference does not apply to Turkey Point Units 3 and 4 as a design basis, it does represent an approach to thermal loads considered acceptable by the engineering community.

Liner plate integrity was investigated in a separate analysis. This analysis conservatively considered both the difference between the concrete temperature (average of 212°F and 30°F) and temperature of the liner plate (212°F) as well as the difference in thermal coefficients of expansion of the two materials. The analysis evaluated the liner plate, as well as stresses in the welds and embeds associated with it. The results of the analysis showed that there would be no loss of function.

9.5.3.4 Test and Inspection Capability

In accordance with the requirements of ASME XI, Subsection IWP (Inservice Testing of Pumps in Nuclear Power Plants), instrumentation is provided for the Units 3 and 4 Spent Fuel Pit Cooling Pumps. The Unit 3 and 4 SFP Cooling System is provided with a pitot tube (annubar) flow element with a local flow indicator in the suction lines to the SFP cooling water pumps. The SFP heat exchangers are provided with temperature and flow indicators for performance testing.

9.5.4 Fuel Handling System

The Fuel Handling System consists of the refueling cavity, refueling transfer canal, manipulator crane, the fuel transfer equipment, spent fuel pit bridge crane, containment polar crane, spent fuel cask handling crane, new fuel bridge crane and the new fuel elevator. The general arrangement of the Fuel handling System is shown in Figure 9.5-1.

The spent fuel pit is provided with a spent fuel pit cooling and purification system which is discussed in Subsection 9.5.3.

9.5.4.1 Design Basis

The Fuel Handling System is designed for handling and storage of new and spent fuel assemblies, providing an area for RCCA change out and for the required assembly, disassembly, and storage of reactor internals and the reactor vessel closure head. The fuel handling equipment includes interlocks, travel limiting features, and other protective devices to minimize the possibility of mishandling or equipment malfunction that could result in inadvertent damage to a fuel assembly and potential fission product release.

All 10 CFR 50 spent fuel transfer and storage operations are designed to be conducted underwater to ensure adequate shielding during refueling and to permit visual control of the operation at all time. C28

9.5.4.2 System Description

The reactor is refueled with equipment designed to handle the spent fuel under water from the time it leaves the reactor vessel until it is placed in a transfer cask for either storage at the site Independent Spent Fuel Storage Installation (ISFSI), or shipment from the site. Boric acid is added to the water to ensure subcritical conditions during refueling. C28

The Fuel Handling System is divided between two areas: The Auxiliary Building and the containment. The refueling cavity in containment and the transfer canal in the Auxiliary Building are flooded only during shutdown for refueling. The spent fuel pit is continuously full of water during refueling and when spent fuel is stored in the spent fuel pit. The cavity and canal are connected by the Fuel Transfer System consisting of an underwater conveyor that carries the fuel through an opening in the containment. The refueling cavity is flooded with borated water from the refueling water storage tank. In the refueling 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-handled manual tool suspended from the spent fuel pit bridge crane. Both the manipulator crane and the long-handled tool can handle only one assembly at a time.

For fuel cask transfer operations, any fuel stored in the cask area rack is first removed and placed in the other spent fuel pit racks. The cask area rack is then removed, decontaminated as required, and placed in dry storage. After satisfying a required decay period for offloaded spent fuel, the cask is lowered into the cask pit area and spent fuel that meets dry fuel storage requirements in 10 CFR 72 is removed from the spent fuel racks and loaded into the cask. The cask is then removed from the spent fuel pool cask pit by the cask handling crane located above the Auxiliary Building and either transferred to the ISFSI (transfer cask) or shipped off site. C28

New fuel assemblies are received and stored in racks in the new fuel storage area using the new fuel bridge crane. The new fuel bridge crane unloads the new fuel assembly shipping containers and also removes the assemblies from the containers and stores them in the new fuel storage racks. During refueling one new fuel assembly at a time is removed from the storage racks by the new fuel bridge crane and placed in a specially assigned rack from where it is picked up by the new fuel mono-rail hoist and placed in the new fuel elevator in the spent fuel pit fuel transfer canal. The new fuel elevator lowers the new fuel assembly into the canal where it is picked up by the long handled tool attached to the spent fuel bridge crane. The fuel assembly is then placed in the transfer carriage which delivers it to the refueling canal, or it may be temporarily stored in the spent fuel pit.

The new fuel storage area is sized for storage of the fuel assemblies normally associated with the replacement of one-third of a core.

Cathodic Protection

The Unit 4 Fuel Handling Building (FHB) is provided with an Impressed Current Cathodic Protection (ICCP) system to mitigate corrosion. The ICCP system provides a controlled amount of DC current to the outer and inner layers of the reinforcing steel of the exterior wall reinforcement, via a multi-channel Local Rectifier Unit (LRU). The LRU is remotely controlled by a Main Control Unit (MCU) that monitors and regulates the DC current to the system to prevent the electrochemical action of galvanic corrosion before it begins.

C31

The ICCP system consists of a mixed metal oxide (MMO) coated titanium mesh ribbon anode system that uses 1/2” wide ribbon anode strips distributed horizontally over the surface of the building. The areas of the FHB exterior walls with ICCP are divided into four zones per wall for a total of 16 zones; identified as zones 1, 2, 3 and 4 for each of the four (4) exterior walls. ICCP current is delivered to the reinforcing steel by an array of ½” MMO coated titanium mesh ribbon. Two titanium conductor bars are installed in each zone, with each conductor bar connected to two positive anode feeders. The anode maximum voltage limit is 15VDC.

Major Structures Required for Fuel Handling

Refueling Cavity

The refueling 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 as low as reasonably achievable during fuel assembly transfer.

The reactor vessel flange is sealed to the bottom of the refueling cavity by a clamped, gasketed seal ring which prevents leakage of refueling water from the cavity. This seal is placed and secured after reactor cooldown but prior to flooding the cavity for refueling operations. The installed cavity seal ring assembly for both Units 3 and 4 is shown in Figure 9.5-2.

An alternative segmented cavity seal ring is used in place of the gasketed seal ring shown in Figure 9.5-2. The segmented seal ring is sectioned into five sections that can be installed in reduced time without the need of the polar crane to place or remove the seal, thereby reducing refueling activity critical path time. The alternative segmented cavity seal ring assembly for both Units 3 and 4 is shown in Figure 9.5-2a. The use of the alternative segmented cavity seal ring as the primary cavity seal is necessary due to the installation of the Integrated Head Assembly. The original, single piece cavity seal ring cannot be lifted from or lowered into position with the head in place because of interferences with components of the Integrated Head Assembly.

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

An approved coating (e.g. Instacote) is applied to the refueling cavity and refueling canal prior to flooding the cavity for fuel movement to improve water tightness of the stainless steel liner. The coating is removed upon completion and drain down of the cavity. Unit 4 refueling canal has been evaluated to leave the applied coating for an extended period of time as a supplemental barrier to improve leak tightness and minimize degradation of the stainless steel liner.

Refueling Canal

The refueling canal is a passageway extending from the refueling cavity to the inside surface of the containment. The canal is formed by two concrete shielding walls, which extend upward to the same elevation as the refueling cavity. The floor of the canal is at a lower elevation than the refueling cavity to provide the greater depth required for the fuel transfer system tipping device located in the canal. The transfer tube enters the reactor containment and protrudes through the end of the canal. The canal walls and floor are lined with stainless steel.

Refueling Water Storage Tank

The normal duty of the refueling water storage tank is to supply borated water to the refueling canal for refueling operations. In addition, the tank provides borated water for delivery to the core following either a loss-of coolant or a steam line rupture accident. This is described in Chapter 6. The capacity of the tank is based upon the requirement for filling the refueling cavity and refueling canal. The water in the tank is borated to a concentration which assures reactor shutdown by at least 5% d k/k when all RCC assemblies are inserted and the reactor is cooled down for refueling. The tank design parameters are given in Chapter 6.

Spent Fuel Storage Pit

The spent fuel storage pit is designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor. The pit design parameters are listed in Table 9.5-1. Control rods are stored in fuel assemblies.

Spent fuel assemblies are handled by a long-handled tool suspended from the spent fuel pit bridge overhead crane and manipulated by an operator standing on the movable bridge over the pit. The spent fuel storage pit is constructed of reinforced concrete having thick walls and is Class I seismic design. The entire interior pit face and transfer canal is lined with stainless steel plate. The probability of rupture of the pit is exceedingly low. The pit rests on compacted rock which has an extremely low permeability. Loss of water through an area equal to the floor area with a full pool head of water is less than 1 gpm.

Storage racks are provided to hold spent fuel assemblies and are erected on the pit floor. Fuel assemblies are held in a rectangular array, and placed in vertical cells. The racks are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations, thereby ensuring the necessary spacing between assemblies.

The spent fuel pit is provided with continuous water level indication with alarm setpoints. Exceeding the setpoint causes an audible alarm in the control room. Low water level is made up by normal sources. Excess water is pumped out since there are no gravity drains.

New Fuel Storage

New fuel assemblies and control rods are stored in a separate area whose location facilitates the unloading of new fuel assemblies or control rods from trucks. This storage vault is designed to hold a reload region in racks in the storage area listed in Table 9.5-1. The assemblies which make up the remaining part of the first core are stored in the spent fuel pit which otherwise remains unused until the time of first refueling.

Major Equipment Required for Fuel Handling

Reactor Vessel Stud Tensioner

The stud tensioner is a hydraulically operated (oil as the working fluid) devise provided to permit preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. Stud tensioners were chosen in order to minimize the time required for tensioning or unloading operations. Three tensioners are provided and they are applied simultaneously to three studs approximately 120° apart. A compact hydraulic power unit is positioned above each of the three utilized tensioners. The studs are tensioned to their operational load in a predetermined sequence to prevent high stresses in the flange region and unequal loadings in the studs. Relief valves are provided on each tensioner to prevent over tensioning of the studs due to excessive pressure. Charts indicating the stud elongation and load for a given oil pressure are included in the tensioner operating instructions. In addition, gauges are provided to measure the elongation of the studs after each tensioning step.

Reactor Vessel Head Lifting Device

The Integrated Head Assembly (IHA) includes lifting components that provide an integral means for lifting the reactor vessel head without the need to remove and/or reinstall a lifting tripod. These lift components are made up of three clevises, three lifting rods, the tripod assembly, and the integral missile shield, and are mounted to the lift lugs on the RVCH.

Reactor Internals Lifting Device

The reactor internals lifting device is a fixture provided to remove the upper reactor internals package and to move it to a storage location in the refueling canal. 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 turned by long torque tubes extending up to an operating platform on the lifting device. Bushings on the fixture engage guide studs mounted on the vessel flange to provide close guidance during removal and replacement of the internals package. This lifting device can also be used to remove the lower internals once the vessel has been cleared of all fuel assemblies.

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 refueling cavity and runs on rails set into the floor along the edge of the refueling 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 from the mast to grip the fuel assembly. 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. The manipulator can lift only one fuel assembly at a time.

All controls for the manipulator crane are mounted on a removable console on the trolley which are removed from containment during operation. The bridge is positioned with the aid of a TV monitoring system. A TV camera on the bridge truck and trolley provides position information to the operator via a monitor. The drives for the bridge, trolley, and winch are variable speed and include separate inching control on the winch. Electrical interlocks and limit switches on the bridge and trolley drives protect the equipment. In an emergency, the bridge, trolley, and winch can be operated manually using a handwheel on the motor shaft.

The suspended weight on the gripper tool is monitored by an electric load cell indicator mounted on the control console. A load in excess of 110 percent of a fuel assembly weight stops the winch 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:

a) Bridge, trolley, and winch drives which are interlocked to prevent simultaneous operation of any two drives.

b) Two redundant position safety switches, the GRIPPER TUBE UP position switches, which prevents bridge and trolley main motor drive operation except when it is actuated, or when the below listed conditions are met:

   1. Gripper switch in unlatch position.

   2. Gripper 12” up into upper mast with no weight indicated on load cell.

   3. Gripper up disengaged (G.U.D.) switch in operate mode.

c) An interlock which prevents the opening of a solenoid valve in the air line to the gripper when the suspended weight indicated by an electric load cell is greater than the weight of the empty gripper tool, or the tube down disengaging permissive circuit limit switch is open. As back-up protection for this interlock, the mechanical weight actuated lock in the gripper prevents operation of the gripper under load even if air pressure is applied to the operating cylinder.

d) The backup OVERLOAD switch, which opens the hoist drive circuit in the up direction when the loading is excessive.

e) An interlock on the hoist drive circuit in the up direction, which permits the hoist to be operated only when either the OPEN or CLOSED indicating switch on the gripper is actuated.

f) Hoist slow zones operating through limit switches on the hoist cut out the main hoist control and transfer control to the inching control at critical elevations of hoist travel.

g) Bridge and trolley drives are interlocked by travel limit switches to limit the outer mast to a path of travel that clears the guide studs in the reactor vessel, the upper internals in their storage position and the brackets and tool racks on the wall of the transfer canal.

h) An underload circuit in the fuel loading hoist of the manipulator crane which prevents damage to fuel assemblies as they are being inserted into the core.

i) A dual cable hoist to retain a load under single cable failure.

j) A hoist load indicator system to provide continuous load readout. Suitable restraints are provided between the bridge and trolley structures and their respective rails to prevent derailing and the manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper in the event of a maximum hypothetical earthquake.

Spent Fuel Handling Crane

The spent fuel handling crane is a traveling bridge with a top-running trolley mounted on an overhead structure. The trolley is equipped with two hoists, one on each side of the bridge. The crane spans the spent fuel pit so that it can transfer fuel from the fuel transfer canal into the storage racks located in the pit. The fuel assemblies are moved from the fuel transfer canal and within the spent fuel pit by means of a long-handled tool suspended from one of the hoists on the trolley. The hoist travel and tool length are designed to limit the maximum lift of active fuel to a safe shielding depth. Control consoles are roller mounted an each of the bridge handrails. All operator actions, including bridge, trolley and hoist travel can be performed from the control consoles, which in turn can be moved from one end of the bridge to the other. A fixed scale and pointer location indication system for the bridge and trolley is provided. Hoist position is also indicated.

The hoists are provided with limit switches, overload sensors and other safety features to withstand two-blocking, load hang-ups and other overloading, mis-reeving, and single cable failures. The capacity of each hoist is two tons. The crane has weight sensors that are interlocked to limit the total load on both hoists combined to a maximum of two tons.

In addition, an in-line weight sensing system is provided for each hoist to limit the lifting load to preclude accidental fuel damage should binding occur. When lifting over spent fuel, the total load is limited to 1 ton by procedures, limit switches and load sensors.

Spent Fuel Cask Handling Crane

The spent fuel cask handling crane is a traveling bridge crane arranged to serve both spent fuel pits. Limit switches prevent movement of the cask beyond the laydown area in the bottom of the pit, prevent interference of the cask crane bridge, trolley and hoist with fuel racks or building structures, and restrict vertical lift of the cask to an elevation of about six inches above the top of the pit wall. The interlocks are administratively initiated by a selector switch located in the control cab. The crane has hurricane latches.

Fuel Transfer System

The Unit 4 fuel transfer system, shown in Figure 9.5-1, is an underwater conveyor car that runs 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 lowered to a horizontal position for passage through the tube, and then is raised by the upending machine to a vertical position in the spent fuel pit.

The fuel transfer system for both units utilizes a transverse cable drive operated by two electric winches located on the Spent Fuel Operating Floor (elev. 58’) to drive the conveyor car. During operation, the conveyor car is stored in the refueling canal. The gate valve is closed and a blind flange is bolted on the transfer tube to seal the containment.

Reactor Cavity Filtration System

In order to assure that the borated water in the reactor refueling cavity will be clear and allow good visibility for observation during refueling operations, a portable reactor cavity filtration system is installed and utilized only for the duration of refueling. Various portable systems/configurations may be used; however, each is assessed for compatibility prior to use.

Refueling Operation Activities

The following refueling activities and their sequence are governed by approved plant procedures and schedules. Below is the typical sequence.

Preparation

a) The reactor is shutdown and cooling to ambient conditions begins.

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

c) Deleted

d) Cooling air ducts between the IHA air plenum and the CRDM air coolers on the operating floor are disconnected and removed to storage. CRDM cables are disconnected from the bulkheads and stored on the IHA.

e) The incore thimble tubes are withdrawn.

f) Reactor vessel head insulation is removed and instrument leads are disconnected. The segmented cavity seal ring is installed: the seal segments are removed from the storage box, assembled in place and the captured j-bolts are clamped to effect the seal.

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

h) The reactor vessel head studs are removed to storage or left on the head and removed to storage after the head is on the storage pedestal.

i) The refueling cavity drain valves are closed.

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

k) Guide studs are installed in two holes 180° apart and the remainder of the stud holes are plugged.

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

m) The fuel transfer tube flange is removed.

n) The reactor vessel head is unseated and raised approximately one inch with the polar bridge crane and a level check is made.

o) The reactor vessel head is slowly lifted to clear the drive shafts.

p) The reactor vessel head is taken to the storage pedestal.

q) The old seal rings are removed from the reactor vessel head.

r) The refueling cavity is filled with water to the vessel flange. The water is pumped into the refueling cavity by the Residual Heat Removal pumps from the refueling water storage tank. The normal Residual Heat Removal System inlet valves from the Reactor Coolant System are closed. The reactor cavity filtration system is started as required. When the cavity is filled, the residual heat removal loop is restored to normal operation.

s) The full length control rod drive shafts are unlatched.

t) The reactor vessel internals lifting rig is lowered into position by the polar bridge crane and latched to the support plate.

u) The reactor vessel upper internals are lifted out of the vessel and placed on the underwater storage rack.

v) The core is now ready for refueling.

<u>Refueling</u>

The refueling sequence is now started utilizing the manipulator crane. The sequence for fuel assemblies in non-control positions is as follows:

a) Spent fuel is removed from the core and placed into the fuel transfer carriage for relocation to the spent fuel pit. This could involve either a full or a partial core off-load during the refueling outage (Reference 1).

b) An insert shuffle (RCCAs and spent WET Annular Burnable Absorber) is performed either in the reactor core or the spent fuel pit.

c) New and reused fuel assemblies are brought in from the spent fuel pit through the fuel transfer system and loaded into the core.

d) Whenever any fuel is being added to the reactor core, or is being relocated, a reciprocal curve of source neutron multiplication is recorded to verify the subcriticality of the core.

A transfer of the RCCAs between fuel assemblies is required when a fuel assembly containing a RCCA is spent or relocated to an unrodded position in the core. The RCCA change tool is used to transfer the RCCA to a fuel assembly which will be located in a rodded position in the core.

Reactor Reassembly

a) The fuel transfer car is parked and the fuel transfer tube isolation valve closed.

b) The reactor vessel internals package is picked up by the polar bridge crane and replaced in the vessel.

c) The full length control rod drive shafts are relatched to the RCCAs.

d) The manipulator crane is parked.

e) The water level is lowered by opening a valve at the Residual Heat Removal pump discharge and water is pumped from the refueling cavity into the refueling water storage tank until the level is slightly below the vessel flange. The Residual Heat Removal line is throttled. If in operation, the reactor cavity filtration system is shutdown, and the refueling cavity is completely drained. When the water in the refueling cavity reaches the vessel flange level, the valve at the Residual Heat Removal pump is closed.

   The Residual Heat Removal operation is restored and the remaining water in the refueling cavity is drained into the reactor coolant drain tank via the low point in the canal drain. The water is then pumped back into the refueling water storage tank by the Reactor Coolant Drain Tank pumps. The flange surface is manually cleaned.

f) The reactor vessel head flange seal ring grooves are cleaned and the new seal rings are installed in the reactor vessel head.

g) The reactor vessel head is picked up by the polar bridge crane and positioned over the reactor vessel. Fifty six (56) head studs may be attached to the head when the head is moved over the reactor vessel.

h) The reactor vessel head is slowly lowered.

i) The reactor vessel head is seated.

j) The guide studs are removed to their storage rack. The stud hole plugs are removed.

k) All head studs are replaced and retensioned.

l) The refueling canal drain valves are opened and the fuel transfer tube flange is replaced.

m) The segmented cavity seal ring is removed: the seal segments are unclamped, disassembled, placed in their storage box, and removed for storage outside containment.

n) Vessel head insulation and instrumentation leads are replaced.

o) Electrical cables are reconnected to the bulkheads. Cooling air ducts are reconnected to the IHA air plenum and CRDM air coolers on the operating floor.

p) Flux map thimble tubes are inserted and high pressure seals connected.

q) The system is brought up to operating temperature and pressure and the reactor vessel closure is leakage tested.

r) Control rod drives are checked.

s) Deleted

t) Pre-operational tests are performed.

9.5.4.3 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:

a) Gamma radiation levels in the containment, control room and fuel storage areas are continuously monitored (see Section 11.2.3). These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. Continuous monitoring of reactor neutron flux provides immediate indication and alarm in the control room of an abnormal core flux level.

b) With the exception of the personnel and equipment hatches, containment integrity is maintained when core alterations or movement of irradiated fuel occurs inside the containment.

c) Whenever any fuel is being added to the reactor core or is being relocated, a reciprocal curve of source neutron multiplication is recorded to verify the subcriticality of the core.

Incident Protection

Direct communication between the control room and the refueling cavity manipulator crane is required whenever changes in core geometry which affect criticality are taking place. This provision allows the control room operator to inform the manipulator crane operator of any impending unsafe conditions detected from the control board indicators during fuel movement.

Malfunction Analysis

Operation of a single residual heat removal loop is permitted for decay heat removal when fuel is in the reactor vessel and the refueling cavity is filled. A loss of this single residual heat removal loop has been evaluated to ensure adequate natural circulation cooling can be maintained for decay heat removal. The analysis utilized GOTHIC thermal hydraulic analysis software to evaluate natural circulation cooling conditions with both the reactor vessel upper internals installed and removed. In each case, adequate natural circulation conditions will be maintained to prevent fuel damage.

For the internals in case, the natural circulation flow path modeled is up from the core to the vessel upper plenum to the refueling cavity via the holes in the upper support plate, the CRDM guide tubes, the head spray flow nozzles, the upper internals hold-down spring gap, and the hot leg gap, with return flow to the core via the downcomer and barrel/baffle bypass. A direct flow path for natural circulation from the core to the vessel upper plenum to the refueling cavity and back to the core via the downcomer and barrel/baffle bypass exists for the internals out case.

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

9.5.4.4 Test and Inspection Capability

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

9.5.5 REFERENCES

1. FPL Letter L-2002-214 to the NRC, dated November 26, 2002, Proposed License Amendments, "Addition of Cask Area Spent Fuel Storage Racks."

2. FPL Letter L-84-206 to the NRC, "Proposed Amendment to Spent Fuel Storage Facility Expansion - Additional Information," dated August 14, 1984.

3. Newmyer, W.D., Westinghouse Spent Fuel Rack Criticality Analysis Methodology," WCAP-14416-NP-A, Revision 1, November 1996.

4. American Nuclear Society, "American National Standard Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," ANSI/ANS-57.2-1983, October 7, 1983.

5. U.S. Nuclear Regulatory Commission, Standard Review Plan, NUREG-0800, Section 9.1.2, "Spent Fuel Storage," Rev. 1, July 1981.

6. American Nuclear Society, "American National Standard for Nuclear Criticality Safety in Operation with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983, October 7, 1983.

7. American National Standard, ANS-51.1/N18.2-1973, "Nuclear Safety Criteria for the Design of Pressurized Water Reactor Plants," August 6, 1973, Section 5.7" Fuel Handling System. C26

8. Deleted.

9. FPL Letter L-2002-151 to the NRC, dated October 21, 2002, Proposed License Amendment for "Reduction of Decay Time for Core Off-load and Revision of Technical Specification 3/4.9.3."

10. NRC Letter (Eva A. Brown) to FPL (J.A. Stall), "Turkey Point Units 3 & 4- Issuance of Amendments Regarding Reduction in Decay Time From 100 Hours to 72 Hours (TAC Nos. MB6549 and MB6550)," License Amendments 223/218, effective March 4, 2003.

11. 10CFR50.68, "Criticality Accident Requirements."

12. “Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants,” L. Kopp (USNRC) to T. Collins (USNRC), August 19, 1998.

13. NRC Letter (S.A. Varga) to FPL (J.W. Williams), “Proposed Spent Fuel Pool Expansion for Turkey Point Units 3 and 4 – Request for Additional Information,” dated August 13, 1984.

14. NRC Letter to FPL, “Turkey Point Units 3 and 4 – Issuance of Amendments Regarding Temporary Spent Fuel Pool Cask Racks (TAC Nos. MB6909 and MB6910),” License Amendments 226/222, November 24, 2004.

15. FPL Letter L-2003-213 to the NRC, “Turkey Point Units 3 and 4 - RAI Response for Addition of Spent Fuel Pool Cask Area Rack Amendment,” September 8, 2003.

16. Engineering Evaluation PTN-ENG-SENS-07-032, “RHR System Operation with the Reactor Cavity Filled and the Vessel Upper Internals in Place,” Revision 1. C27

17. FPL Letter L-2005-247 to the NRC, dated January 27, 2006, Proposed License Amendments, Spent Fuel Pool Boraflex Remedy.

18. WCAP-17094-P, Revision 3, “Turkey Point Units 3 and 4 New Fuel Storage Rack and Spent Fuel Pool Criticality Analysis,” February 2011.

19. Deleted

TABLE 9.5-1

FUEL HANDLING DATA

New Fuel Storage Area	
Core storage capacity	1/3
Equivalent fuel assemblies	54
Center-to-center spacing of assemblies, in.	21**
Maximum K_{eff}, if flooded with unborated water	≤0.95
Spent Fuel Storage Pit	
Core storage capacity	Approx 9
Equivalent fuel assemblies	1379 (1510*)
Number of space accommodations for spent fuel transfer casks	1 (0*) C28
Center-to-center spacing of assemblies, in.	
Region 1 - Spent Fuel Pit Storage Racks	10.6**
Region 1 - Cask Area Storage Rack	10.1 E-W, 10.7 N-S**
Region 2 - Spent Fuel Pit Storage Racks	9.0**
Maximum K_{eff} when flooded with water borated to the level identified in section 9.5.2.3	≤0.95
Maximum K_{eff} if flooded with unborated water	<1.0
Miscellaneous Details	
Width of refueling canal, ft.	3
Wall thickness of spent fuel storage pit, ft.	3 to 6
Weight of fuel assembly with RCCA (dry), lb.	Approx 1,634
Capacity of refueling water storage tank, gal.	338,000
Minimum contents of refueling water storage tank for Safety Injection or Spray System	
Operability, gal.	320,000
Quantity of water required for refueling, gal.	285,000
Minimum water required for post MHA sump recirculation pump N.P.S.H. protection including allowance for possibility of drainage delay in containment	239,000

* With Cask Area Storage Rack installed

** Technical Specification 5.5.1 Critical Design Feature

TABLE 9.5-2

CRANES, HOISTS & ELEVATORS

New Fuel Bridge Crane

Number		1 per unit
Capacity		4 tons

New Fuel Monorail Hoist

Number		1 per unit
Capacity		1 ton

New Fuel Elevator

Number		1 per unit
Capacity		1 fuel assembly at one time

Spent Fuel Cask Handling Crane

Number		1
Capacity		main hoist: 130 tons auxiliary hoist: 25 tons

C25

Spent Fuel Pit Bridge Crane

Number		1 crane per unit; 2 hoists per crane
Capacity		2 tons/hoist [1]

Containment Polar Crane

Number		1 per unit
Capacity	Main Hook (Unit 4)	205 tons
	Main Hook (Unit 3)	205 tons
	Aux. Hook	35 tons

Notes:

[1] Limit switches and procedures limit load to 1 ton over spent fuel areas.

TABLE 9.5-3

Nominal Fuel Assembly Specification used in the Region I and Region II Rack Criticality Analysis

Parameter	Value	
Assembly type	OFA/DRFA	LOPAR
Rod Array Size	15x15	
Rod Pitch, Inches	0.563	
Active Fuel Length, Inches	144	
Stack Density (%TD)	97.5	
Maximum Nominal Enrichment, wt%	5.0	
Total Number of Fuel Rods	204	
Fuel Cladding Outer Diameter, Inches	0.422	
Fuel Cladding Inner Diameter, Inches	0.3734	
Fuel Cladding Thickness, Inches	0.0243	
Pellet Diameter, Inches	0.3659	
Number of Guide/Instrument Tubes	20 / 1	
Guide/Instrument Tube Outer Diameter, Inches	0.533	0.546
Guide/Instrument Tube Inner Diameter, Inches	0.499	0.512
Guide/Instrument Tube Thickness, Inches	0.017 min.	

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Revised 04/17/2013

TABLE 9.5-4

Region I Spent Fuel Storage Cell and
Fuel Nominal Parameters

Rack Parameter	Unit	Value
Rack Cell Inner Dimension	inch	8.75
Rack Cell Pitch	inch	10.60[2]
Rack Material		Stainless Steel
Rack Wall Thickness	inch	0.075[2]
Wrapper Material		Stainless Steel
Wrapper Plate Thickness	inch	0.020[2]
Poison Panel Thickness	inch	0.078
Poison Cavity Thickness	inch	0.090
Poison Panel Width	inch	7.5
Poison Panel Length	inch	139.4
B-10 Loading in Boraflex*	gm/cm^2	0.020
Bottom of Boraflex Above Support Pad	inch	6.16
Fuel Parameter		
maximum Axial Blanket at Each End of the Rods	inch	0.0 - 8.0
Maximum Nominal Fuel Enrichment of the Central Length	%	5.0
Blanket Region Enrichment	%	0.79 - 2.6
Theoretical Density	%	97.5[1]

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(1) As per Reference 18, criticality analyses remain valid up to a maximum region average fuel pellet density of 97.5% of theoretical.

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* Original Rack values, Boraflex no longer credited in the criticality analysis

(2) Technical Specification 5.5.1 Critical Design Feature

TABLE 9.5-5

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TABLE 9.5-6

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TABLE 9.5-7

Region II - Spent Fuel Storage Cell Nominal Parameters

Parameter	Unit	Value
Rack Cell Inner Dimension	inch	8.80
Rack Cell Pitch	inch	9.0**
Rack Material		Stainless Steel
Rack Wall Thickness	inch	0.075**
Wrapper Material		Stainless Steel
Wrapper Plate Thickness	inch	0.020
Poison Panel Thickness	inch	0.051
Poison Cavity Thickness	inch	0.064
Poison Panel Width	inch	7.5
Poison Panel Length	inch	139.4
B-10 Loading in Boraflex*	gm/cm^2	0.012
Bottom of Boraflex Above Support Pad	inch	6.16

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* Original Rack values, Boraflex no longer credited in the criticality analysis

** Technical Specification 5.5.1 Critical Design Feature

TABLE 9.5-8

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TABLE 9.5-9

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TABLE 9.5-10

Spent Fuel Pool Pit Soluble Boron Requirements

Normal (ppm)	For Misloaded Assembly (ppm)
500	1683

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TABLE 9.5-11

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TABLE 9.5-12

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TABLE 9.5-13

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TABLE 9.5-14

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TABLE 9.5-15

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TABLE 9.5-16

Spent Fuel Cooling Loop Component Data

System Cooling Capacity, Btu/hr		
Racks Full with Full Core Off-Load at 72 Hours (Equilibrium Temperature of 120°F) (1)		35.64×10^6
Racks Full with Full Core Off-Load (4)(5) (Equilibrium Temperature of 139°F)		47.62×10^6
Time for Pool Water Boiling (No Heat Removal) from the Equilibrium Temperature, Hours		
Racks Full with Full-Core Off-Load (3) (Equilibrium Temperature of 148°F)		2.7
Spent Fuel Pit Heat Exchanger		
Quantity		1
Type		Shell and U-tube
Heat Transfer, Btu/hr(2)		7.96×10^6
Shell side (Component Cooling Water)		
Inlet Temperature, °F		100
Outlet Temperature, °F		106
Design Flow rate, lb/hr		1.4×10^6
Design Pressure, psig		150
Design Temperature, °F		200
Material		Carbon Steel
Tube side (Spent Fuel Pit Water)		
Inlet Temperature, °F		120
Outlet Temperature, °F		113
Design Flow Rate, lb/hr		1.1×10^6
Design Pressure, psig		150
Design Temperature, °F		200
Material		Stainless Steel
Supplemental Spent Fuel Pit Heat Exchanger		
Quantity		1
Type		Shell and U-tube
Heat Transfer, Btu/hr(6)	Modes 1-5:	11.5×10^6
	Mode 6:	20.9×10^6
Shell side (Component Cooling Water)		
Inlet Temperature, °F		105
Outlet Temperature, °F	Modes 1-5:	124.2
	Modes 6:	120
Design Flow rate, lb/hr	Modes 1-5:	0.6×10^6
	Mode 6:	1.4×10^6
Design Pressure, psig		150
Design Temperature, °F		200
Material		Carbon Steel
Tube side (Spent Fuel Pit Water)		
Inlet Temperature, °F		140
Outlet Temperature, °F	Modes 1-5:	129.5
	Modes 6:	121
Design Flow Rate, lb/hr		1.1×10^6
Design Pressure, psig		150
Design Temperature, °F		212
Material		Stainless Steel

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TABLE 9.5-16

Spent Fuel Cooling Loop Component Data

Notes:

1. The pool equilibrium temperature at 72 hours after shutdown with pool cooling system operating, off-load rate of 8 Fuel Assemblies / hr and cooling water to heat exchanger at 85°F. C26

2. Assumed pool water to heat exchanger at 120°F and cooling water to heat exchanger at 100°F.

3. This heat load case also assumes that 1/2 core discharge (assumed to be 80 assemblies to bound one reload batch) precedes the full core off-load by 36 days. C26

4. The pool equilibrium temperature at 72 hours after shutdown with SFP cooling system operating, off-load rate of 6 assemblies/hr hours, and cooling water to heat exchanger at 105°F. C26

5. Total assumes fuel is also stored in the 25 cells removed from service. C26

6. Heat Transfer values are based on the specified temperature and flow values for the supplemental spent fuel pit heat exchanger.

Spent Fuel Cooling Loop Component Data

Spent Fuel Pit Pump Data	
Quantity	2
Type	Horizontal Centrifugal
Flow, gpm	2300
Minimum Developed Head, ft H_2O	125
Motor Horsepower	100
Design Pressure, psig	150
Design Temperature, °F	200
Material	Stainless Steel
Spent Fuel Pit	
Volume, ft^3 (Nominal)	40,000
Boron Concentration, ppm Boron	≥2300 C26
Spent Fuel Pit Filter	
Quantity	3
Type	Replaceable Cartridge
Internal Design Pressure of Housing, psig	150
Design Temperature, °F	200
Rated Flow, gpm	150
Maximum Differential Pressure Across Filter Element at Rated Flow (Clean Cartridge), psi	5
Maximum Differential Pressure Across Filter Element Prior to Cartridge Replacement, psi	20
Spent Fuel Pit Demineralizer	
Quantity	1
Type	Flushable
Design Pressure, psig	200
Design Temperature, °F	250
Design Flow rate, gpm	100
Resin Volume, cu. ft.	30
Vessel Volume, cu. Ft.	43

Spent Fuel Cooling Loop Component Data

Spent Fuel Pit Skimmers	
Quantity	2
Flow Per Unit, gpm	50
Vertical Fluctuation Range:	
Floating, inch	4
Manual Adjustment, feet (nominal)	4
Spent Fuel Pit Skimmer Strainer	
Quantity	1
Type	Basket
Rated Flow, gpm	100
Design Pressure, psig	50
Design Temperature, °F	200
Maximum Differential Pressure Across Strainer Element At Rated Flow (Clean), psi	1
Perforation, inch	1/8
Spent Fuel Pit Skimmer Filter	
Quantity	3
Type	Replaceable Cartridge
Internal Design Pressure, psig	50
Design Temperature, °F	200
Rated Flow Per Cartridge, gpm	50
Filter System Rated Flow, gpm	150
Maximum Differential Pressure Across Filter Element At Rated Flow (Clean), psi	5
Maximum Differential Pressure Across Filter Element Prior To Removing, psi	20

TABLE 9.5-16

Spent Fuel Cooling Loop Component Data

Spent Fuel Pit Skimmer Pump	
Quantity	
Type	Horizontal Centrifugal
Design Flow Rate, gpm	100
Developed Head, ft H_2O	50
Design Pressure, psig	50
Design Temperature, °F	200
Refueling Water Purification Pump	
Quantity	1
Type	Horizontal Centrifugal
Design Flow Rate, gpm	100
Developed Head, ft H_2O	150
Design Pressure, psig	150
Design Temperature, °F	200
Spent Fuel Pit Cooling Loop Piping and Valves	
Design Pressure, psig	150
Design Temperature, °F	212
Spent Fuel Pit Skimmer Loop Piping and Valves	
Design Pressure, psig	150
Design Temperature, °F	200
Refueling Water Purification Loop Piping and Valves	
Design Pressure, psig	150
Design Temperature, °F	200

Table 9.5-17

Applicable Design and Fabrication Codes, Standards and Specifications for Spent Fuel Storage Racks

Spent Fuel Pit Storage Racks

1. NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, April 14, 1978, as amended by the NRC letter dated January 18, 1979.
2. Not used.
3. NRC Regulatory Guides
   a. 1.13, Rev. 2, Spent Fuel Storage Facility Design Basis, Dec. 1981 (Draft).
   b. 1.25, Rev.0 Assumptions Used for Evaluating the Potential Radiological Consequence of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, March 1972.
   c. 1.26, Rev. 3, Quality Group Classifications and Standards for Water Steam and Radioactive Waste Containing Components of Nuclear Power Plants, Feb. 1976.
   d. 1.29, Rev. 3, Seismic Design Classification, Sept. 1978.
   e. 1.92, Rev. 1, Combining Modal Responses and Spatial Components in Seismic Response Analysis, Feb. 1976.
   f. 1.124, Rev. 1, Service Limits and Load Combinations for Class I Linear Type Component Supports, Jan. 1978.
4. Standard Review Plan - NUREG-0800, Rev. 1, July 1981.
   a. Section 3,7.1, Seismic Design, Rev. 1, July 1981.
   b. Section 3.8.4, Other Seismic Category I Structures, Rev. 3, July 1981.
   c. Section 9.1.2, Spent Fuel Storage, Rev. 1, July 1981.
   d. Section 9.1.3, Spent Fuel Pool Cooling System, Rev.1, July 1981.
5. NRC Branch Technical Position ASB 9-2, Residual Decay Energy for Light Water Reactors for Long Term Cooling,
   Rev. 2, July 1981.
6. Industry Codes and Standards
   a. ANSI N16.1-75, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.
   b. ANSI N16.9-75, Validation of Calculational Methods for Nuclear Criticality Safety.
   c. ANSI N210-76, Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations.
   d. ASME Section III-80, Nuclear Power Plant Components (through Summer 1982 Addendum).
   e. ACI 318-63, Building Code Requirements for Reinforced Concrete.
   f. DSS-ISG-2010-1, “Staff Guidance Regarding the Nuclear Criticality Safety Analysis for Spent Fuel Pools”, September 2011. C26

Cask Area Storage Rack

1. NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, April 14, 1978, as
   amended by the NRC letter dated January 18, 1979.
2. NUREG 0612, Control of Heavy Loads at Nuclear Power Plants, USNRC, July 1980
3. NRC Regulatory Guides
   a. 1.13, Rev.2, Spent Fuel Storage Facility Design Basis, Dec. 1981 (Draft).
   b. 1.25, Rev. 0, Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, March 1972.
   c. 1.29, Rev. 2, Seismic Design Classification, Feb. 1976.
   d. 1.92, Rev. 1, Combining Modal Responses and Spatial Components in Seismic Response Analysis, Feb. 1976.
   e. 1.124, Rev. 1, Service Limits and Load Combinations for Class 1 Linear- Type Component Supports, Jan. 1978.
4. Standard Review Plan - NUREG-0800, June 1987
   a. Section 3.7.1, Seismic Design, Rev.1, July 1981.
   b. Section 3.8.4, Other Seismic Category I Structures, Rev. 1, July 1981.
5. Industry Codes and Standards
   a. ASME Boiler and Pressure Vessel Code, Section II Parts A and C, Section III, 1989 Edition.
   b. ACI 318-63, Building Code Requirements for Reinforced Concrete.
   c. ANSI N16.9-75, Validation of Calculational Methods for Nuclear Criticality Safety.
   d. ANSI/ANS 8.1, Criticality Safety in Operations with Fissionable Materials Outside Reactors.
   e. ANSI/ANS 8.17, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.

TABLE 9.5-18

Fuel Category Ranked by Reactivity

Region I	I-1	High Reactivity
	I-2	
	I-3	
	I-4	Low Reactivity
Region II	II-1	High Reactivity
	II-2	
	II-3	
	II-4	
	II-5	Low Reactivity

Notes:

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1. Fuel Category is ranked by decreasing order of reactivity without regard for any reactivity-reducing mechanisms, e.g., Category I-2 is less reactive than Category I-1, etc. The more reactive fuel categories require compensatory measures to be placed in Regions I and II of the SFP, e.g., use of water filled cells, Metamic inserts, or full length RCCAs.

2. Any higher numbered fuel category can be used in place of a lower number fuel category from the same Region.

3. Category I-1 is fresh unburned fuel up to 5.0wt% U-235 enrichment.

4. Category I-2 is fresh unburned fuel that obeys the IFBA requirement of Table 9.5-21 or contains an equivalent amount of another burnable absorber.

5. All Categories except I-1 and I-2 are determined from Tables 9.5-19 or 9.5-20.

Table 9.5-19

Blanketed Fuel - Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

Coeff.	Fuel Category						
	I-3	I-4	II-1	II-2	II-3	II-4	II-5
A1	5.66439153	-14.7363682	-7.74060457	-7.63345029	24.4656526	8.5452608	26.2860949
A2	-7.22610116	11.0284547	5.13978237	10.7798957	-20.3141124	-4.47257395	-18.0738662
A3	2.98646188	-1.80672781	-0.360186309	-2.81231555	6.53101471	2.09078914	5.8330891
A4	-0.287945644	0.119516492	0.0021681285	0.29284474	-0.581826027	-0.188280562	-0.517434342
A5	-0.558098618	0.0620559676	-0.0304713673	0.0795058096	-0.16567492	0.157548739	-0.0614152031
A6	0.476169245	0.0236575787	0.098844889	-0.0676341983	0.243843226	-0.0593584027	0.134626308
A7	-0.117591963	-0.0088144551	-0.0277584786	0.0335130877	-0.0712130368	0.0154678626	-0.0383060399
A8	0.0095165354	0.0008957348	0.0024057185	-0.0040803875	0.0063998706	-0.0014068318	0.0033419846
A9	-47.1782783	-20.2890089	-21.424984	14.6716317	-41.1150	-0.881964768	-12.1780
A10	33.4270029	14.7485847	16.255208	-10.0312224	43.9149156	9.69128392	23.6179517
A11	-6.11257501	-1.22889103	-1.77941882	5.62580894	-9.6599923	-0.18740168	-4.10815592
A12	0.490064351	0.0807808548	0.127321203	-0.539361868	0.836931842	0.0123398618	0.363908736

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the “minimum burnup” (GWd/MTU) given by the curve fit for the assembly “cooling time” and “initial enrichment.” The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$Bu = (A_1 + A_2*En + A_3*En^2 + A_4*En^3)*\exp[-(A_5 + A_6*En + A_7*En^2 + A_8*En^3)*Ct] + A_9 + A_{10}*En + A_{11}*En^2 + A_{12}*En^3$$

2. Initial enrichment, En, is the nominal central zone U-235 enrichment. Axial blanket material is not considered when determining enrichment. Any enrichment between 2.0 and 5.0 may be used.

3. Cooling time, Ct, is in years. Any cooling time between 0 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.

4. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment.

5. Category I-2 is fresh unburned fuel that obeys the IFBA requirements in Table 5.5-4 or contains an equivalent amount of another burnable absorber.

6. This Table applies for any blanketed fuel assembly.

C26

Table 19.5-20

Non-Blanketed Fuel - Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

Coeff.	Fuel Category						
	I-3	I-4	II-1	II-2	II-3	II-4	II-5
A1	2.04088171	-27.6637884	-11.2686777	20.7284208	29.8862876	-83.5409405	35.5058622
A2	-4.83684164	26.1997193	2.0659501	11.9673275	-37.0771132	94.7973724	-30.1986997
A3	2.59801889	-7.2982252	2.66204924	-14.4072388	16.3986049	-31.9583373	11.0102438
A4	-0.300597247	0.723731768	-0.513334362	2.83623963	-2.1571669	3.55898487	-1.27269125
A5	-0.610041808	0.401332891	-0.0987986108	-1.49118695	1.02330848	0.299948492	1.34723758
A6	0.640497159	-0.418616707	-0.0724198633	1.75361041	-1.21889631	-0.312341996	-1.19871392
A7	-0.219000712	0.144304039	0.106248806	-0.659046438	0.467440882	0.107463895	0.352920811
A8	0.0252870451	-0.0154239536	-0.0197359109	0.080884618	-0.0560129443	-0.0108814287	-0.0325155213
A9	-4.48207836	-5.54507376	-1.34620551	-245.825283	12.1549	39.4975573	-5.2576
A10	-2.12118634	-5.76555416	-10.1728821	243.59979	-22.7755385	-50.5818253	10.1733379
A11	2.91619317	6.29118025	8.71968815	-75.7805818	14.3755458	23.3093829	0.369083041
A12	-0.196645176	-0.732079719	-1.14461356	8.10936356	-1.80803352	-2.69466612	0.0443577624

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the “minimum burnup” (GWd/MTU) given by the curve fit for the assembly “cooling time” and “initial enrichment.” The specific minimum burnup required for each fuel assembly is calculated from the following equation: C26

   $$Bu = (A_1 + A_2*En + A_3*En^2 + A_4*En^3)*\ exp\ [\ -\ (A_5 + A_6*En + A_7*En^2 + A_8*En^3)*Ct\ ] + A_9 + A_{10}*En + A_{11}*En^2 + A_{12}*En^3$$

2. Initial enrichment, En, is the nominal U-235 enrichment. Any enrichment between 1.8 and 4.0 may be used.

3. Cooling time, Ct, is in years. Any cooling time between 15 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.

4. This Table applies only for pre-EPU non-blanketed fuel assemblies. If a non-blanketed assembly is depleted at EPU conditions, none of the burnup accrued at EPU conditions can be credited (i.e., only burnup accrued at pre-EPU conditions may be used as burnup credit).

TABLE 9.5-21

Maximum k_{eff} for each 2 X 2 Array with No Soluble Boron

Array	Maximum k_{eff} (1)
I-A	0.95335
I-B/I-C	0.98971
I-D	0.98938
II-A	0.98962
II-B	0.98978
II-C	0.98995
II-D	0.98998

C26

(1) Maximum k_{eff} values include the sum of biases and uncertainties from Tables 5-2 and 5-3 of Reference 18.

TABLE 9.5-22

Nominal RCCA Specifications

Parameter	Value
Clad Inner Diameter, inches	0.4005
Clad Outer Diameter, inches	0.439
Poison Outer Diameter, inches	0.3900*
Material content, wt%	Silver 80, Indium 15, Cadmium 5
Poison Density (gm/cm^3)	10.17*
Clad Material	SS

*Technical Specification 5.5.1 critical design feature

C26

Revised 04/17/2013

TABLE 9.5-23

<u>Nominal Metamic® Inserts Specifications</u>

Parameter	Value
Material	Al-B_4C
B-10 Loading	Nominal 0.0160 g/cm^2 Minimum 0.0150g/cm^2*
Thickness, Inches	Nominal 0.073*
Inside Width, Inches	Nominal 8.35*
Formed insert Length, measured from top of the assembly Inches	Approx. 155*
Welded insert Length, measured from top of the assembly Inches	Approx. 152*

C26

*Technical Specification 5.5.1 critical design feature

TABLE 9.5-24

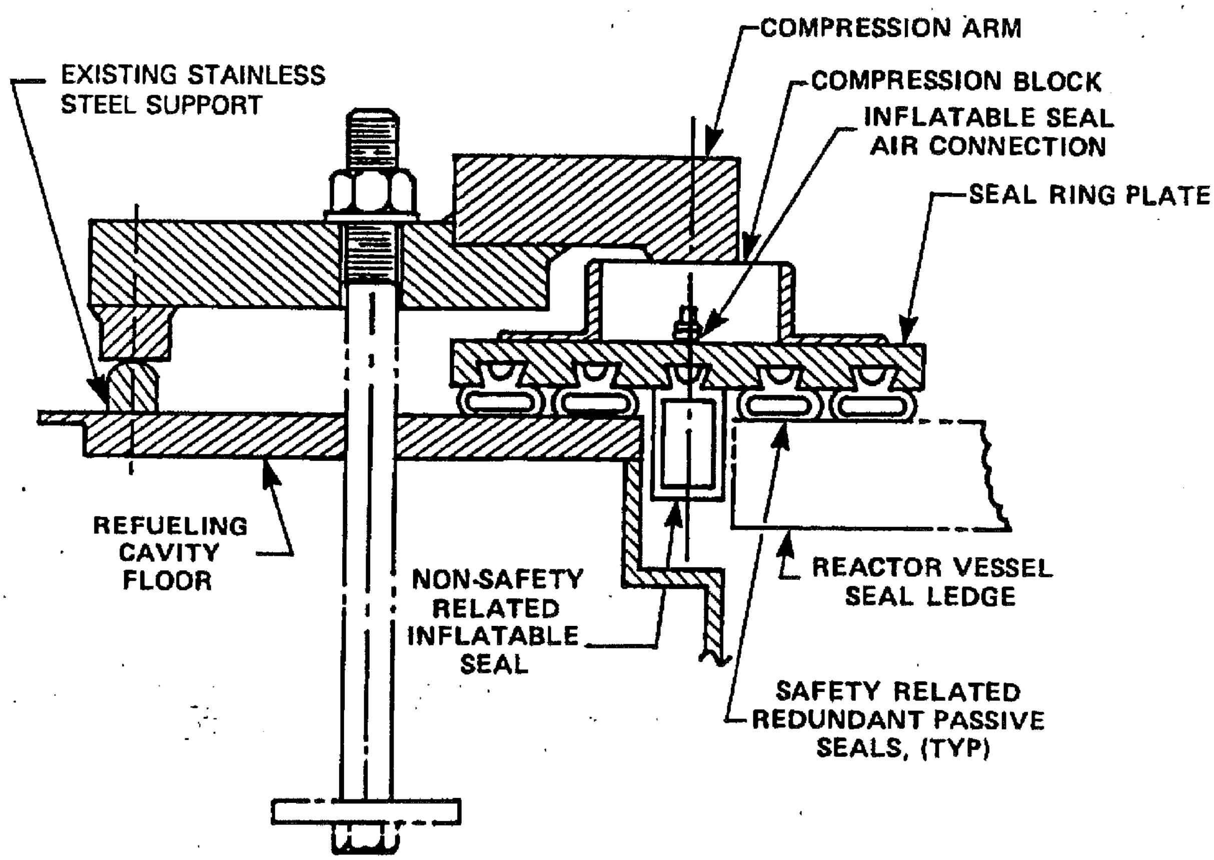
IFBA Requirements for Fuel Category I-2

Nominal Enrichment (wt% U-235)	Minimum Required Number of IFBA Pins
Enr. $\leq$ 4.3	0
4.3 < Enr. $\leq$ 4.4	32
4.4 < Enr. $\leq$ 4.7	64
4.7 < Enr. $\leq$ 5.0	80

C26

Security-Related Information - Withheld Under 10 CFR 2.390

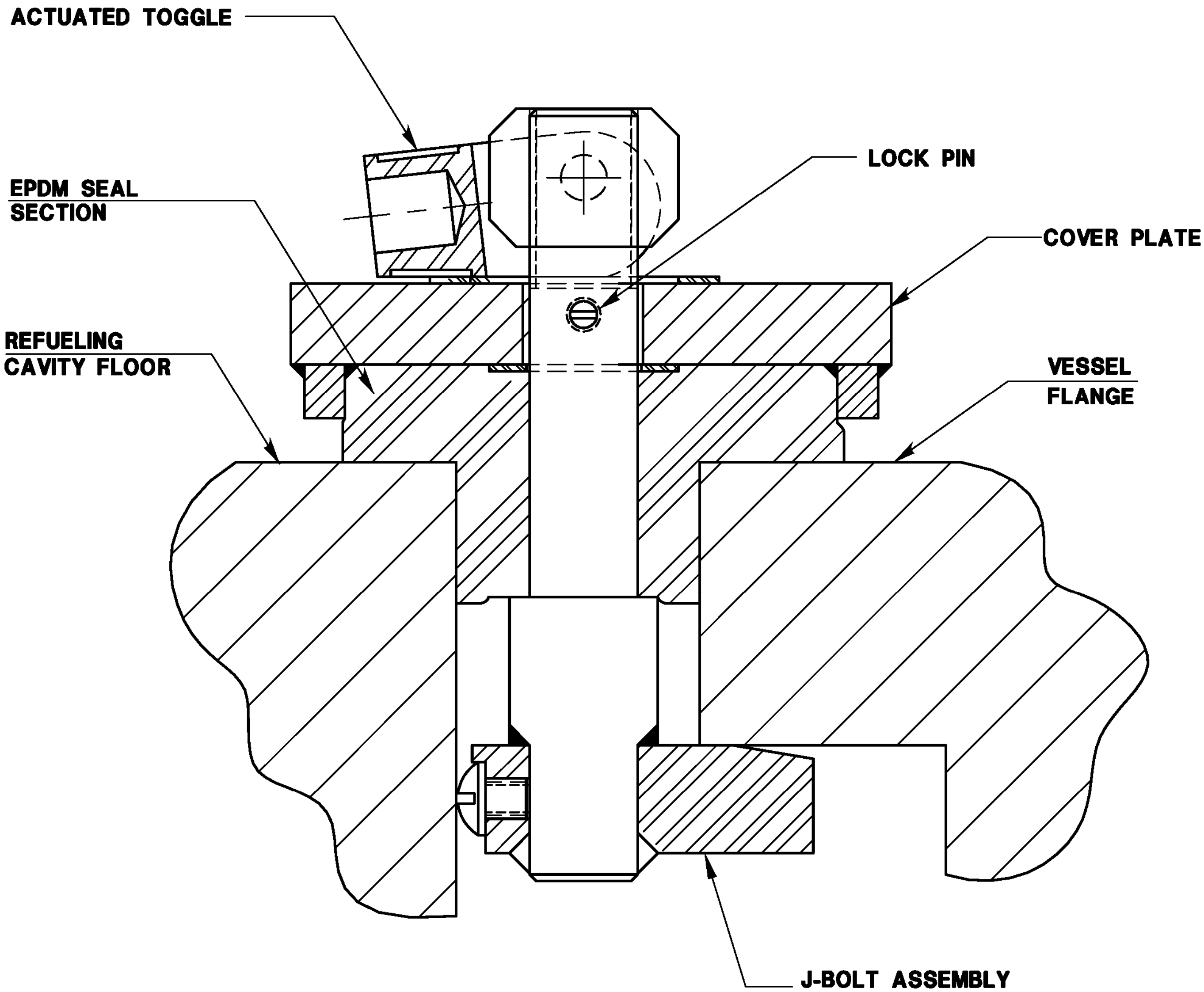
FUEL TRANSFER SYSTEM
FIG. 9.5-1



TYPICAL SECTION OF INSTALLED ASSEMBLY

REV. 9 (7/91)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNITS 3 & 4**
REACTOR CAVITY SEAL **FIGURE 9.5-2**



ALTERNATIVE SEGMENTED CAVITY SEAL

06/05/2001

FLORIDA POWER & LIGHT COMPANY TURKEY POINT NUCLEAR UNITS 3 & 4
REACTOR CAVITY SEAL FIGURE 9.5-2a

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.9-5

REFER TO ENGINEERING DRAWING

5610-C-249, for Unit 3

5610-C-250, for Unit 4

C28

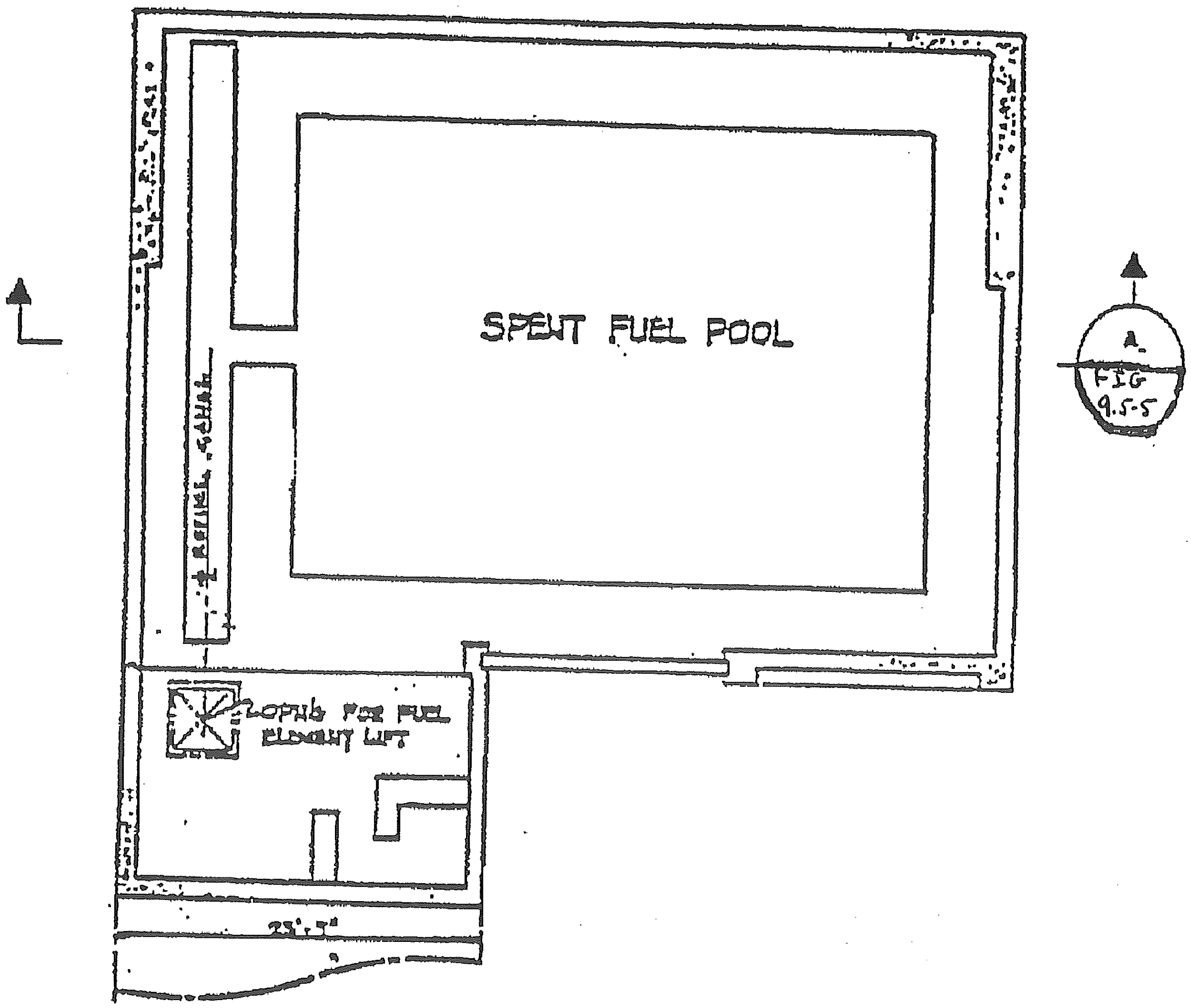
Revised 09/20/2016

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNITS 3 & 4

SPENT FUEL POOL PLAN AT ELEVATION 18’ 0”

FIGURE 9.5-3



UNIT 3 — AS SHOWN
UNIT 4 — OPPOSITE HAND

Revised 09/29/2005

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

SPENT FULE POOL PLAN AT ELEVATION 58' 0"

FIGURE 9.5.4

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.5-5

REFER TO ENGINEERING DRAWING

5610-C-257 , SHEET 1

C26

Revised 02/09/12

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNITS 3 & 4

SECTION A-A of SPENT FUEL POOL

FIGURE 9.5-5

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.5-6

REFER TO ENGINEERING DRAWING

5610-C-45-29

SHEET 2 (Unit 3)

SHEET 7 (Unit 4)

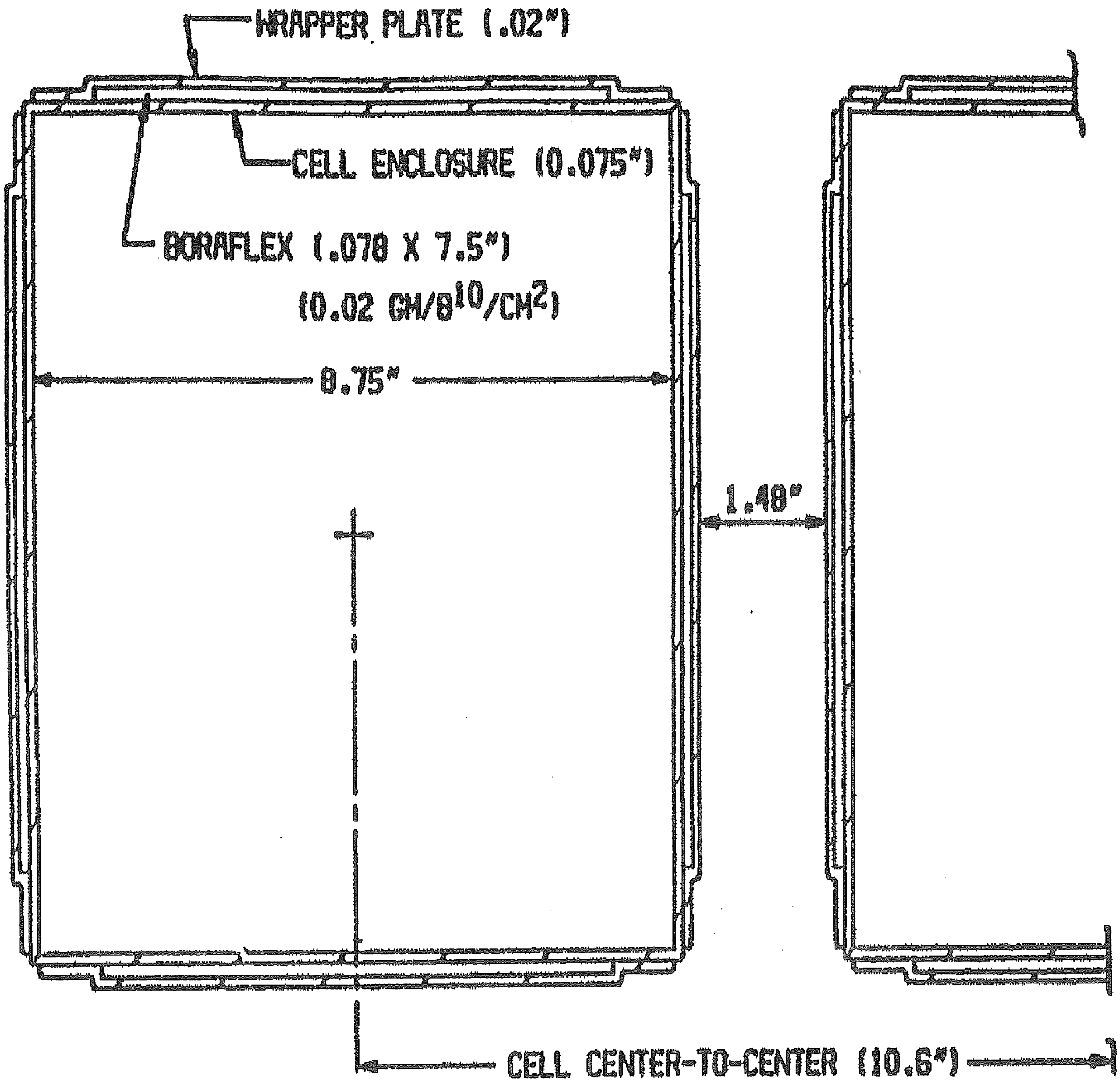
C26

Revised 02/09/12

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

SPENT FUEL POOL COOLING SYSTEM

FIGURE 9.5-6

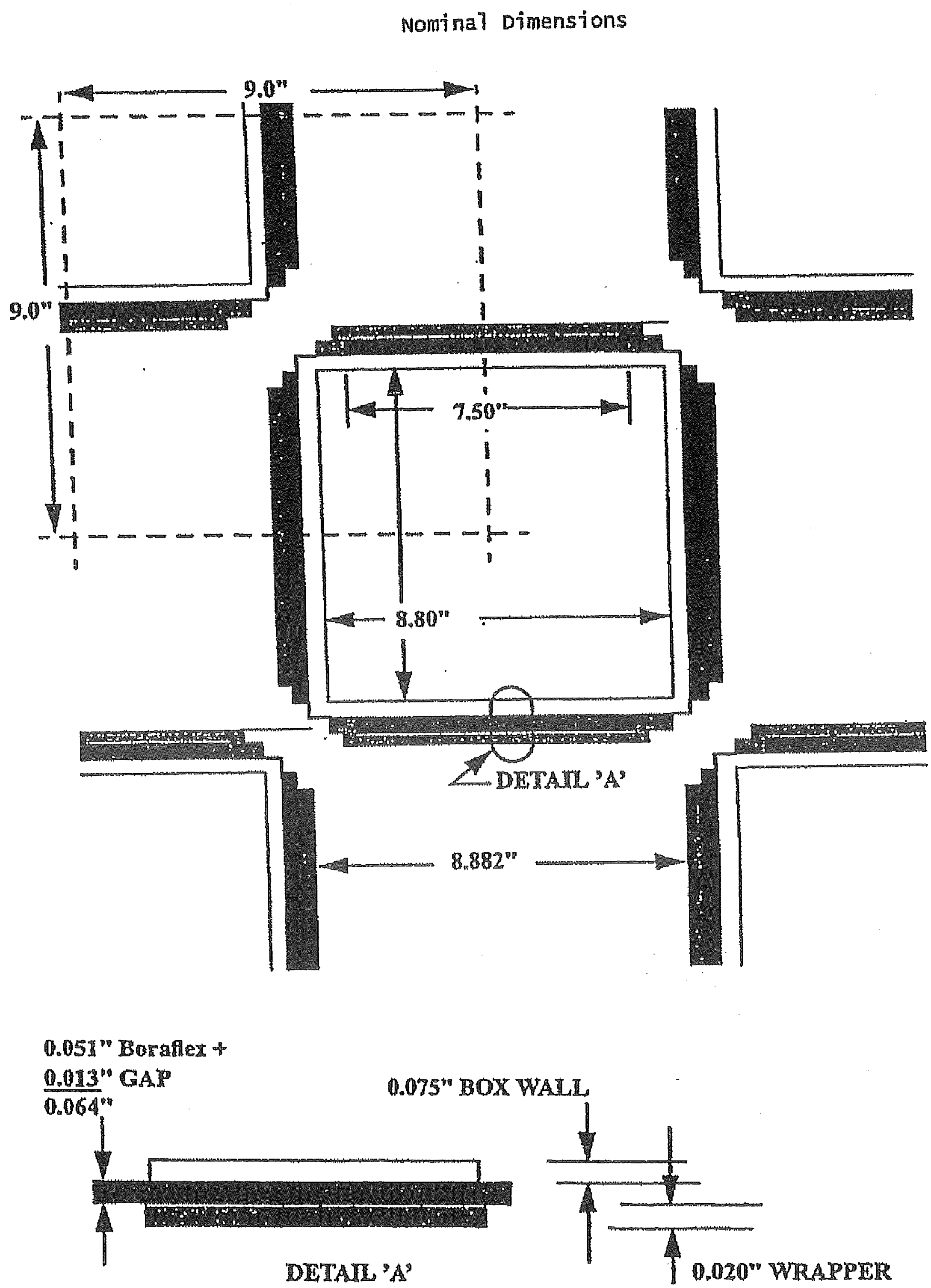


Revised 09/29/2005

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

NOMINAL DIMENSIONS FOR THE REGION I SPENT FUEL PIT STORAGE CELLS

FIGURE 9.5.7



Revised 09/29/2005

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

NOMINAL DIMENSIONS FOR THE REGION II SPENT FUEL PIT STORAGE CELLS

FIGURE 9.5.8

Security-Related Information - Withheld Under 10 CFR 2.390

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

NOMINAL DIMENSIONS FOR THE REGION I CASK AREA RACK STORAGE CELLS

FIGURE 9.5.9

FIGURE 9.5-10

DELETED

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FINAL SAFETY ANALYSIS REPORT

FIGURE 9.5-11

REFER TO ENGINEERING DRAWING

5613-M-3033, SHEET 1

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT
UNIT 3

SPENT FUEL POOL COOLING SYSTEM

FIGURE 9.5-11

Revised 09/29/2005

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.5-12

REFER TO ENGINEERING DRAWING

5614-M-3033, SHEET 1

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT
UNIT 4

SPENT FUEL POOL COOLING SYSTEM

FIGURE 9.5-12

Revised 09/29/2005

FIGURE 9.5-13

DELETED

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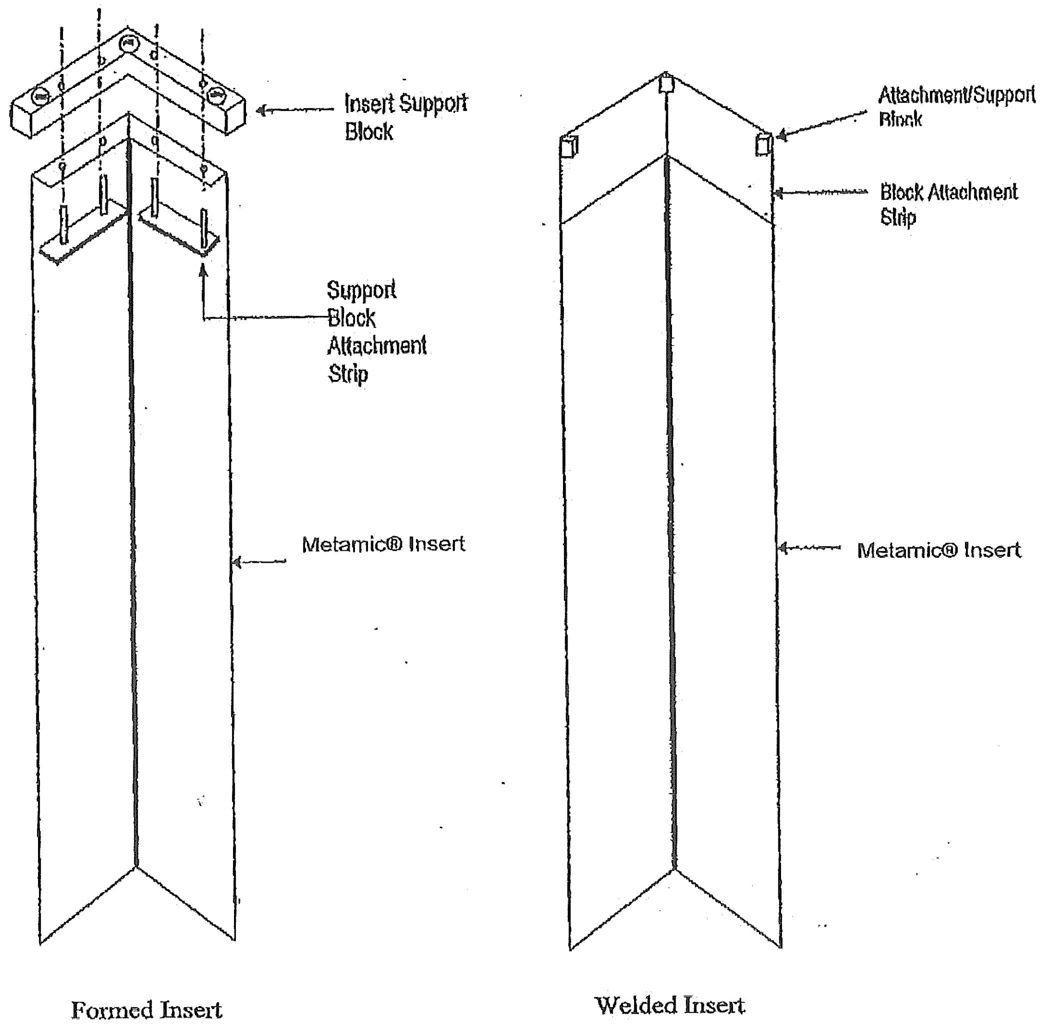

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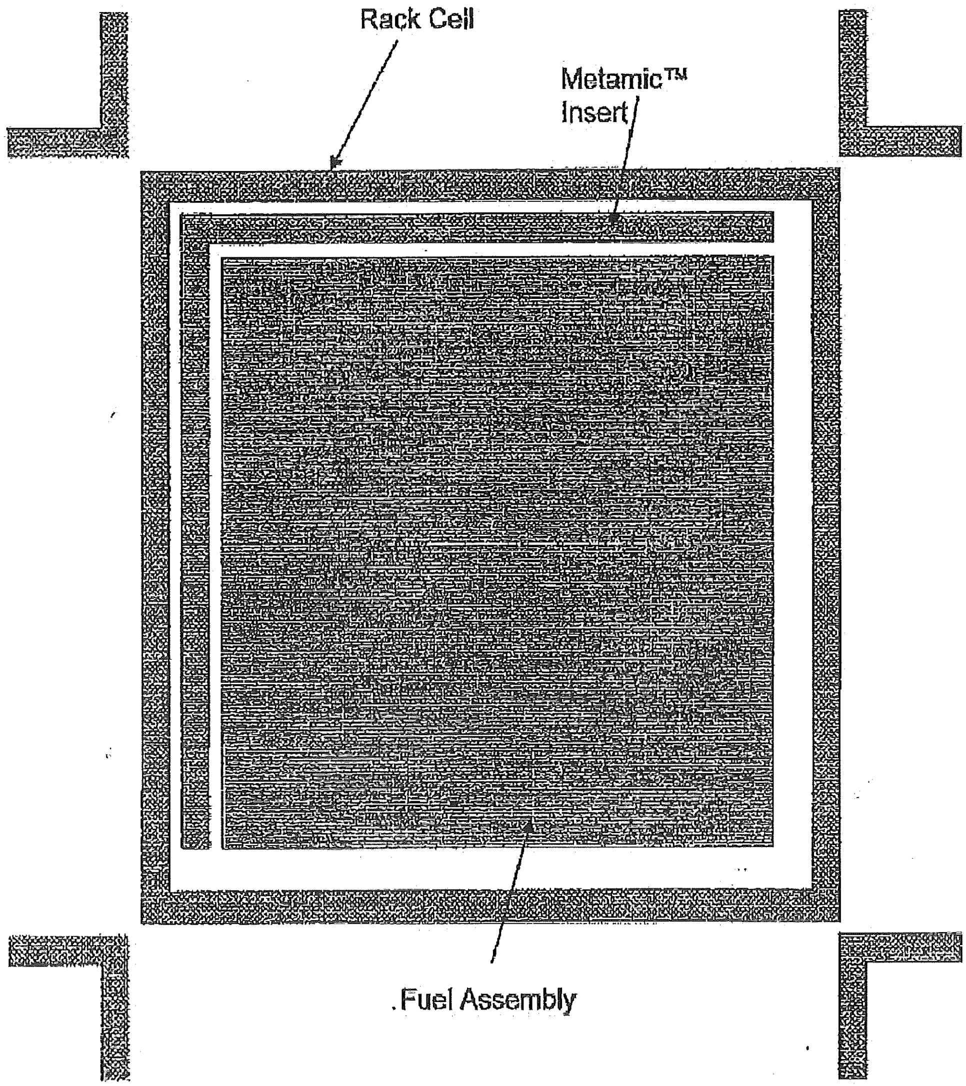
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT
UNITS 3 & 4

METAMIC® INSERTS REGION II RACKS

FIGURE 9.5-14



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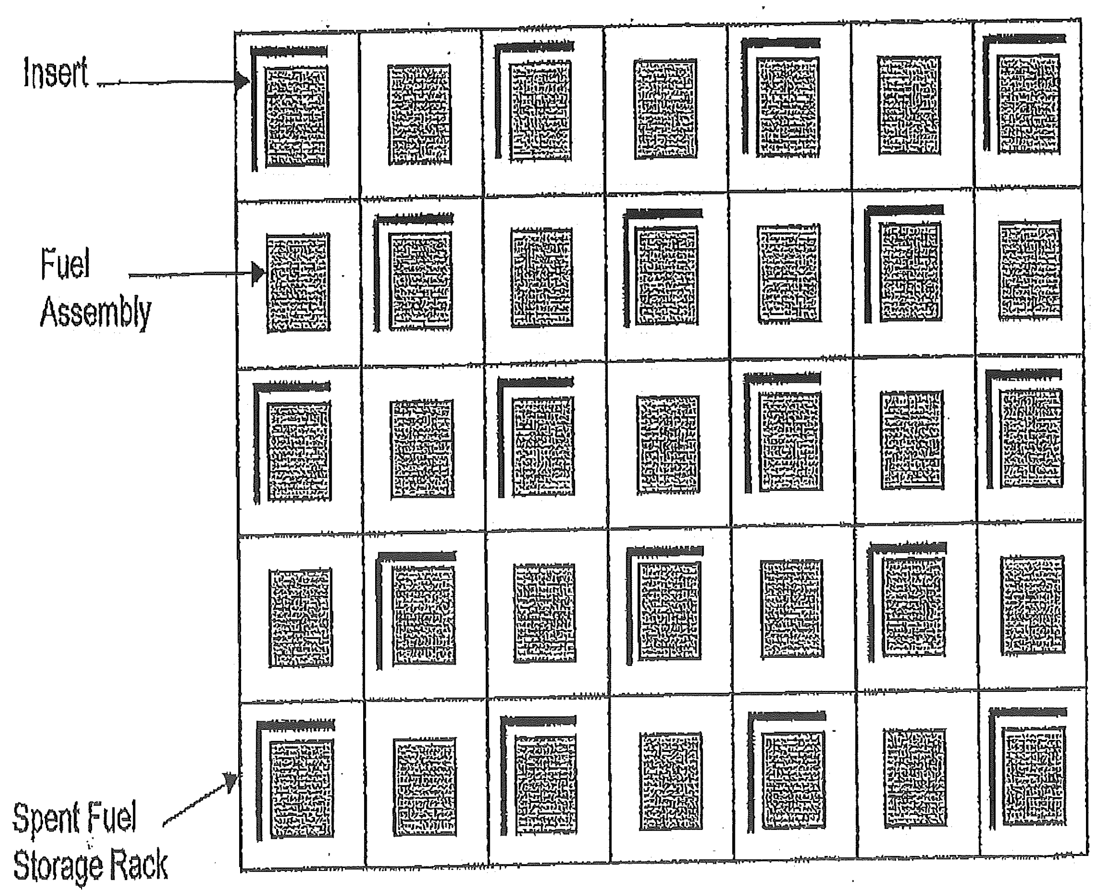
FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT
UNITS 3 & 4

REGION II RACK CELL with METAMIC®
INSERTS and FUEL ASSEMBLY

FIGURE 9.5-15



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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT
UNITS 3 & 4

ILLUSTRATION SHOWING SPATIAL ORIENTATION of METAMIC® INSERTS

FIGURE 9.5-16

Loading Step 1

II-B	II-B	II-B	II-B	II-B	II-B
II-B	II-B		II-B	II-B	II-B
II-B	II-B		II-B	II-B	II-B
II-B	II-B		II-B	II-B	II-B

Loading Step 2

II-B	II-B	II-B	II-B	II-B	II-B
II-B	II-B		II-B	II-B	II-B
II-B	II-B	II-B	II-B	II-B	II-B
II-B	II-B		II-B	II-B	II-B

Loading Step 3

II-B	II-B	II-B	II-B	II-B	II-B
II-B	II-B		II-B	II-B	II-B
II-B	II-B	II-B	II-B	II-B	II-B
II-B	II-B		II-B	II-B	II-B

Loading Step 4

II-B	II-B	II-B	II-B	II-B	II-B
II-B	II-B	II-B	II-B	II-B	II-B
II-B	II-B	II-B	II-B	II-B	II-B
II-B	II-B		II-B	II-B	II-B

Loading Step 5

II-B	II-B	II-B	II-B	II-B	II-B
II-B	II-B	II-B	II-B	II-B	II-B
II-B	II-B	II-B	II-B	II-B	II-B
II-B	II-B	II-B	II-B	II-B	II-B

C25

Notes:
1) Numbers in the above cells are Array numbers of the assemblies in the cells.
2) Shaded cells contain an insert.
3) Cells with a cross contain only water.
4) The reverse sequence is an example of Unloading steps for a single assembly in Array II-B

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT
UNITS 3 & 4

ILLUSTRATION SHOWING SPATIAL ORIENTATION of METAMIC® INSERTS

FIGURE 9.5-17

9.6 FACILITY SERVICES

9.6.1 FIRE PROTECTION PROGRAM

The fire protection program is based on the NRC requirements and guidelines, Nuclear Electric Insurance Limited (NEIL) Property Loss Prevention Standards and related industry standards. With regard to NRC criteria, the fire protection program meets the requirements of 10 CFR 50.48(c), which endorses, with exceptions, the National Fire Protection Association's (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants" – 2001 Edition. Turkey Point Nuclear Plant Units 3 and 4 has further used the guidance of NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)" as endorsed by Regulatory Guide 1.205, "Risk-Informed, Performance Fire Protection for Existing Light-Water Nuclear Power Plants."

Adoption of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", 2001 Edition in accordance with 10 CFR 50.48(c) serves as the method of satisfying 10 CFR 50.48(a) and General Design Criterion 3. Prior to adoption of NFPA 805, General Design Criterion 3, "Fire Protection" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," was followed in the design of safety and non-safety related structures, systems, and components, as required by 10 CFR 50.48(a) C28

NFPA 805 does not supersede the requirements of GDC 3, 10 CFR 50.48(a), or 10 CFR 50.48(f). Those regulatory requirements continue to apply. However, under NFPA 805, the means by which GDC 3 or 10 CFR 50.48(a) requirements are met may be different than under 10 CFR 50.48(b). Specifically, whereas GDC 3 refers to SSCs important to safety, NFPA 805 identifies fire protection systems and features required to meet the Chapter 1 performance criteria through the methodology in Chapter 4 of NFPA 805. Also, under NFPA 805, the 10 CFR 50.48(a)(2)(iii) requirement to limit fire damage to SSCs important to safety so that the capability to safely shut down the plant is satisfied by meeting the performance criteria in Section 1.5.1 of NFPA 805.

A Safety Evaluation was issued on May 28, 2015 by the NRC, that transitioned the existing fire protection program to a risk-informed, performance-based program based on NFPA 805, in accordance with 10 CFR 50.48(c).

9.6.1.1 DESIGN BASIS SUMMARY

9.6.1.1.1 Defense-In-Depth

The fire protection program is focused on protecting the safety of the public, the environment, and plant personnel from a plant fire, and its potential effect on safe reactor operations. The fire protection program is based on the concept of defense-in-depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:

(1) Preventing fires from starting,

(2) Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage,

(3) Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

9.6.1.1.2 NFPA 805 Performance Criteria

The design basis for the fire protection program is based on the following nuclear safety and radiological release performance criteria contained in Section 1.5 of NFPA 805: C28

Nuclear Safety Performance Criteria. Fire protection features providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria shall be met.

(a) Reactivity Control. Reactivity control shall be capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity inserting shall occur rapidly enough such that fuel design limits are not exceeded.

(b) Inventory and Pressure Control. With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of controlling coolant level such that subcooling is maintained such that fuel clad damage as a result of a fire is prevented for a PWR.

(c) Decay Heat Removal. Decay heat removal shall be capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition.

(d) Vital Auxiliaries. Vital auxiliaries shall be capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.

(e) Process Monitoring. Process monitoring shall be capable of providing the necessary indication to assure the criteria addressed in (a) through (d) have been achieved and are being maintained.

Radioactive Release Performance Criteria. Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, Limits.

Chapter 2 of NFPA 805 establishes the process for demonstrating compliance with NFPA 805.

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features.

C28

Chapter 4 of NFPA 805 establishes the methodology to determine the fire protection systems and features required to achieve the nuclear safety performance criteria outlined above. The methodology shall be permitted to be either deterministic or performance-based. Deterministic requirements shall be “deemed to satisfy” the performance criteria, defense-in-depth, and safety margin and require no further engineering analysis. Once a determination has been made that a fire protection system or feature is required to achieve the nuclear safety performance criteria of Section 1.5, its design and qualification shall meet the applicable requirement of Chapter 3.

9.6.1.1.3 Codes of Record

The codes, standards and guidelines used for the design and installation of plant fire protection systems are listed in 5610-016-DB-001, Fire Protection System NFPA 805 Design Basis document.

9.6.1.2 SYSTEM DESCRIPTION

9.6.1.2.1 Required Systems

Nuclear Safety Capability Systems, Equipment, and Cables

Section 2.4.2 of NFPA 805 defines the methodology for performing the nuclear safety capability assessment. The systems, equipment, and cables required for the nuclear safety capability assessment are contained in the 5610-M-722A, Nuclear Safety Capability Assessment Basis Document, 5610-M-723A, Essential Equipment List, and 5610-E-2000A, Essential Cable List.

Fire Protection Systems and Features

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features. Compliance with Chapter 3 is documented in the Fire Protection System NFPA 805 Design Basis document.

Chapter 4 of NFPA 805 establishes the methodology and criteria to determine the fire protection systems and features required to achieve the nuclear safety performance criteria of Section 1.5 of NFPA 805. These fire protection systems and features shall meet the applicable requirements of NFPA 805 Chapter 3. These fire protection systems and features are documented in Fire Protection System NFPA 805 Design Basis document.

Fire protection system and features plant configuration information is documented in STD-M-006, Engineering Guidelines for Fire Protection for Turkey Point Units 3 & 4. C28

Radioactive Release

Structures, systems, and components relied upon to meet the radioactive release criteria are documented in Fire Protection System NFPA 805 Design Basis document.

9.6.1.2.2 Definition of “Power Block” Structures

Where used in NFPA 805 Chapter 3 the terms “Power Block” and “Plant” refer to structures that have equipment required for nuclear plant operations. For the purposes of establishing the structures included in the fire protection program in accordance with 10 CFR 50.48(c) and NFPA 805, the plant structures listed in the Fire Protection System NFPA 805 Design Basis document are considered to be part of the ‘powerblock’.

9.6.1.3 SAFETY EVALUATION

The 5610-016-DB-001, Fire Protection System NFPA 805 Design Basis document documents the achievement of the nuclear safety and radioactive release performance criteria of NFPA 805 as required by 10 CFR 50.48(c). This document fulfills the requirements of Section 2.7.1.2 “Fire Protection Program Design Basis Document” of NFPA 805. The document contains the following:

- Identification of significant fire hazards in the fire area. This is based on NFPA 805 approach to analyze the plant from an ignition source and fuel package perspective.
- Summary of the Nuclear Safety Capability Assessment (at power and non-power) compliance strategies.
  - Deterministic compliance strategies.
  - Performance-based compliance strategies (including defense-in-depth and safety margin).
- Summary of the Non-Power Operations Modes compliance strategies.
- Summary of the Radioactive Release compliance strategies.
- Summary of the Fire Probabilistic Risk Assessments.
- Key analysis assumptions to be included in the NFPA 805 monitoring program.

9.6.1.4 FIRE PROTECTION PROGRAM DOCUMENTATION, CONFIGURATION CONTROL AND QUALITY ASSURANCE

In accordance with Chapter 3 of NFPA 805 a fire protection plan documented in 0-ADM-016, Fire Protection Program, defines the management policy and program direction and defines the responsibilities of those individuals responsible for the plan’s implementation. 0-ADM-016:

- Designates the senior management position with immediate authority and responsibility for the fire protection program. C28
- Designates a position responsible for the daily administration and coordination of the fire protection program and its implementation.
- Defines the fire protection interfaces with other organizations and assigns responsibilities for the coordination of activities.
- Identifies the appropriate authority having jurisdiction for the various areas of the fire protection program.
- Identifies the procedures established for the implementation of the fire protection program, including the post-transition change process and the fire protection monitoring program.
- Identifies the qualifications required for various fire protection program personnel.
- Identifies the quality requirements of Chapter 2 of NFPA 805.

Detailed compliance with the programmatic requirements of Chapters 2 and 3 of NFPA 805 are contained in the Fire Protection System NFPA 805 Design Basis document.

9.6.2 MISCELLANEOUS WATER SYSTEMS

Flow diagrams for the primary water, demineralized water, intake cooling water, chlorination and circulating water systems are shown in Figures 9.6-1 through 9.6-18.

Intake Cooling Water System

The Intake Cooling Water System (see Figures 9.6-1 through 9.6-7) is provided with normally cross-connected, redundant headers, such that the heat exchangers in the Auxiliary Coolant System and the Turbine Plant Cooling System (see Figures 9.6-8 through 9.6-9) normally receive flow from both headers. The design includes provision for isolation of turbine plant cooling water heat exchangers following a loss-of-coolant accident. Necessary isolation valves between the 100% capacity headers and the three pumps are also provided. The supply headers are redundant, but the return merges to a non-redundant discharge header that returns water to the discharge canal. The redundant ICW supply headers addresses the design for passive failure.

The system is analyzed to ensure adequate heat removal with the highest expected temperature of cooling water, maximum loadings and leakage allowances. Each of the CCW heat exchangers for Unit 3 have the capability to be periodically cleaned by chemical injection upstream of the heat exchangers, in order to minimize tube-side fouling and preserve heat transfer capability of the heat exchangers.

System Design

The intake cooling water system supplies salt water to the tube side of the component cooling water heat exchangers. The intake cooling water system also supplies salt water to the cold side of the turbine area cooling water heat exchangers. The redundant header system is provided with isolation valves that can be shut so that failure of one loop does not require immediate shutdown of the unit.

Three intake cooling water pumps are provided for each unit. One, two, or three pumps are operated as required to support normal plant operating conditions. However, only one pump is required following a MHA. The A and B pumps are powered by 4160 volt buses which can be powered by each train's associated emergency diesel generator. The C pump is powered by a swing 4160 volt safety related bus which can be powered, through aligning the bus manually, by either the train A or train B emergency diesel generator associated with the same unit. This pump is interlocked, such that, it is started on a loss of offsite power or safety injection signal, if the supply breaker for the A or B ICW pump (associated with the A or B 4KV Bus to which it is aligned) is open and racked out.

To ensure pump operation under flood conditions, the motors are installed above the maximum flood level. The pump section columns are also installed to provide sufficient submergence under minimum water level conditions.

The intake cooling water system provides sufficient redundancy so that at least one intake cooling water pump will continue to operate to handle heat loads from design basis accidents following a postulated single active failure. A single intake cooling water pump, however, is limited in its ability to supply the required cooling water to the component cooling water heat exchangers during an accident when flow is also allowed to continue through the turbine plant cooling water (TPCW) heat exchangers.

In order to maximize the available intake cooling water flow to the CCW heat exchangers, two fail-closed, pneumatically operated valves, POV-*-4882 and POV-*-4883, are installed on the two supply headers to the TPCW heat exchangers. The POVs will automatically isolate the TPCW heat exchangers in the event that instrument air or electrical (DC) power is lost. However, accumulators are provided on the instrument air inlet to maintain the valves open on a loss of instrument air for a sufficient duration to protect turbine plant equipment. Additionally, in the event of a safety injection actuation signal, the POVs will automatically isolate the TPCW heat exchangers. These features ensure that sufficient intake cooling water flow to the component cooling water heat exchangers will be available. These valves do not automatically isolate upon loss of offsite power (LOOP). Should a loss of offsite power event occur which is accompanied by a coincident single failure affecting the availability of one intake cooling water pump, a condition could exist where one intake cooling water pump may operate in a condition where $NPSH_r$ exceeds $NPSH_a$. To mitigate this potential condition, a second ICW pump could be started, or controls that are provided locally and in the control room could be used to initiate manual closure of the POVs and isolate the TPCW heat exchangers within 30 minutes. In addition, these valves are provided with the capability to remain open for a minimum of 2 hours following a loss of instrument air to avoid potential damage to turbine plant equipment.

The intake cooling water system is designed as Class I.

System Evaluation

During normal operations two Intake Cooling Water pumps provide flow to the three Component Cooling Water heat exchangers and to the Turbine Plant Cooling Water heat exchangers. During an accident, one or two ICW pumps can provide flow to two or three CCW heat exchangers. Measures are in place to ensure the heat transfer capability of the CCW heat exchangers meets the accident heat load. This heat transfer capability is ensured even if a single active failure resulted in a partial flow of intake cooling water through the turbine plant cooling water heat exchangers. The measures include performance monitoring, and periodic cleaning of the component cooling water heat exchangers. Periodic cleaning of the Unit 3 CCW heat exchangers by chemical injection can be performed to minimize tube-side fouling, thus preserving the heat transfer capability of the heat exchangers.

The turbine plant cooling water heat exchangers are provided with cathodic protection by means of sacrificial anodes located in the heat exchanger inlet and outlet channel heads. The component cooling water heat exchangers are provided with cathodic protection by means of sacrificial anodes located in the heat exchanger inlet and outlet channel heads.

Water Treatment System

A Water Treatment Plant provides the demineralized water of the required quality for Units 3 and 4 (Reference FSAR Figures 9.6-10 through 9.6-13). A well water supply connection, a backup raw water supply connection and a return treated water connection are provided (Reference FSAR Figures 9.6-10 and Figure 9.6-11). The purpose of these connections is to allow use of a complete vendor supplied water purification system typically furnished as a turnkey service. The Water Treatment Plant purification equipment including all auxiliaries such as carbon pre-filters, two-pass reverse osmosis, chemical feed, deionization, de-gas membrane, ultraviolet reduction, ion exchange, as well as a supplemental containerized single pass reverse osmosis system are provided by a vendor. C29

Wastes from the Water Treatment Plant demineralizer regeneration process from a previous common water treatment system, which has since been demolished, which had a pH between 2.0 and 12.5 were transferred to the neutralization basin where they were neutralized and discharged. If the pH of the demineralized regeneration waste was outside this range, it used to be sent to the waste neutralization tank where it would be neutralized and then sent to the neutralization basin. The current method of processing demineralized water, however, utilizes a Reverse Osmosis System which obviates the need for the Neutralization tank. The Neutralization Tank has been isolated from the Water Treatment Plant and renamed as the CWP Lube Water Storage Tank. It is currently used in the Circulating Water Pumps Lube Water System. C29

An automatic isolation valve is provided on the Water Treatment Plant (WTP) discharge line to prevent discharge of high conductivity water from the WTP to the storage tanks. A drain line with a manual isolation valve is located upstream and close to the automatic isolation valve. C29

Since the water treatment system is not required for safe shutdown of the units following a MHA, it is designated as Class III.

C29

Primary Water Make-up System

Adequate primary water storage is provided to fulfill the water requirements load fluctuations and leakage in the reactor coolant system during normal unit operation.

The primary water is unborated, deaerated, demineralized water suitable for in the reactor coolant system. Boric acid may be added to this water in the desired concentration before it is used as the reactor coolant.

System Design

Demineralized water for the primary water make-up system is supplied from the Water Treatment Plant. Operation of the primary water deaerator may be required to remove dissolved oxygen from the supply stream. (See Figures 9.6-15 through 9.6-18). C29

One primary water storage tank and one deaerator is provided for each unit.

Two primary water make-up pumps per unit are provided. One pump is normally in operation and supplies primary water to various loads, including to the chemical and volume control system where boric acid may be added before its injection (via charging) as reactor coolant. The systems may be operated in a cross-connected arrangement or aligned to support their respective unit.

The primary water make-up system is designated as Class III, since it is not required for the safe shutdown of the unit following a MHA.

9.6.3 SYSTEM DESIGN EVALUATION

Malfunction Analysis

The intake cooling water system is designed to prevent a component failure from curtailing normal unit operation. Two normally cross-tied, parallel headers with necessary cross connections and isolation valves provide redundant flow paths. In addition to the header isolation valves, each component also has individual isolation valves to permit removing any piece of equipment from the system.

A malfunction in the water treatment system or primary water make-up system does not create any abnormal condition in reactor operation; therefore, a component failure or temporary outage is not of primary importance for reactor safety.

Minimum Operating Conditions

Minimum operating requirements of the intake cooling water system are met by one pump and one header. The remote operated control valve permits isolation of the non-essential services for one-pump operation within 30 minutes.

Tests and Inspections

All intake cooling water system components are hydrostatically tested prior to unit start-up and are accessible for periodic inspections during operation. all electrical components, transfer, and starting controls are tested.

9.6.4 REFERENCES

1. FPL Safety Evaluation JPE-LR-87-45,"Justification for Continued Operation of Turkey Point Unit 3 for ICW System Design," Revision 3, dated March 17, 1989.

2. Safety Evaluation by the Office of Nuclear Reactor Regulation for Turkey Point Nuclear Generating Unit. Nos. 3 and 4 - Issuance of Amendments Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with Title 10 of the Code of Federal Regulations Section 50.48(C), dated May 28, 2015 (ML15061A237).

3. License Amendment Request, June 28, 2012, Transition to 10 CFR 50.48(c) - NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition.

4. National Fire Protection Association Standards, NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition.

5. Regulatory Guide 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants, Revision 1, December 2009. C28

6. NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c), Revision 1, September 2005.

7. FAQ 12-0062, Updated Final Safety Analysis Report (UFSAR) Standard Level of Detail, Revision 1, May 21, 2012

8. 5610-016-DB-001, Fire Protection System NFPA 805 Design Basis.

9. 0-ADM-016, Fire Protection Program.

10. 5610-M-722A, Nuclear Safety Capability Assessment Basis Document.

11. 5610-M-723A, Essential Equipment List.

12. STD-M-006, Engineering Guidelines for Fire Protection for Turkey Point Units 3 & 4

13. 5610-E-2000A, Essential Cable List.

TABLE 9.6-1

EQUIPMENT DESIGN PARAMETERS

Intake Cooling Water Pumps

Number	3
Capacity GPM, each	16,000
T.D.H. Feet	60
Pumped Fluid	Sea Water

Water Treatment System

Number	1 shared by Units 3 & 4
Capacity, GPM (nominal)	400
Capacity, GPM (max nominal)	800
Capacity, GPM (maximum capacity with supplemental equipment)	1000

C29

Primary Water Storage Tank

Primary Water Storage Tank	
Number	1
Capacity, gal. (nominal)	150,000
Materials of Construction	Carbon steel, epoxy lined with floating diaphragm.
Design Pressure	Atmospheric

Deaerator

Number	1
Maximum Capacity, GPM	less than 150

Primary Water Make-Up Pumps

Number	2
Capacity, GPM, each	150
T.D.H. feet	350
Pumped Fluid	Primary Water

Basket Strainers

Number	4
Material	
Body & Cover	Carbon Steel, Epoxy Lined (SA515 or SA516, Gr 70)
Strainer Sections	SS316

Note: Numbers of components are per unit unless otherwise indicated.

TABLE 9.6-2

INTAKE COOLING WATER SYSTEM-CODE REQUIREMENTS

Component	Codes
Pumps	Hydraulic Institute Standards
Piping	Above Ground: ASA-A21.6-1955
	Below Ground: ASA-A21.8-1955
Basket Strainers Original Replacement	ANSI B31.1 ASME Section VIII - Division 1 1983 Summer 1985

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-1

REFER TO ENGINEERING DRAWING

5613-M-3019 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

INTAKE COOLING WATER SYSTEM

FIGURE 9.6-1

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-2

REFER TO ENGINEERING DRAWING

5613-M-3019 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

INTAKE COOLING WATER SYSTEM

FIGURE 9.6-2

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-3

THIS FIGURE HAS BEEN DELETED.

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

INTAKE COOLING WATER SYSTEM
TUBE CLEANING FOR CCW
HEAT EXCHANGERS
FIGURE 9.6-3

FIGURE 9.6-4 (DELETED)

INTAKE COOLING WATER SYSTEM
TUBE CLEANING FOR
TPCW HEAT EXCHANGERS

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-5

REFER TO ENGINEERING DRAWING

5614-M-3019 , SHEET 1

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

INTAKE COOLING WATER SYSTEM

FIGURE 9.6-5

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-6

REFER TO ENGINEERING DRAWING

5614-M-3019 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

INTAKE COOLING WATER SYSTEM

FIGURE 9.6-6

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-7

THIS FIGURE HAS BEEN DELETED.

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

INTAKE COOLING WATER SYSTEM
TUBE CLEANING FOR CCW
HEAT EXCHANGERS
FIGURE 9.6-7

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-8

REFER TO ENGINEERING DRAWING

5613-M-3008 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

TURBINE PLANT COOLING
WATER SYSTEM

FIGURE 9.6-8

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-9

REFER TO ENGINEERING DRAWING

5614-M-3008 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 4

TURBINE PLANT COOLING
WATER SYSTEM

FIGURE 9.6-9

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-10

REFER TO ENGINEERING DRAWING

5610-M-3021 , SHEET 1

Revised 04/06/2018

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNITS 3 & 4

WATER TREATMENT PLANT SYSTEM

C29

FIGURE 9.6-10

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-11

REFER TO ENGINEERING DRAWING

5610-M-3021 , SHEET 2

Revised 04/06/2018

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNITS 3 & 4

WATER TREATMENT PLANT SYSTEM

C29

FIGURE 9.6-11

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-12

REFER TO ENGINEERING DRAWING

5610-M-3021 , SHEET 4 C29

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FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNITS 3 & 4

WATER TREATMENT PLANT SYSTEM C29

FIGURE 9.6-12

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-13

REFER TO ENGINEERING DRAWING

5610-M-3021 , SHEET 5 C29

Revised 04/06/2018

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

WATER TREATMENT PLANT SYSTEM
SAMPLING SYSTEM C29

FIGURE 9.6-13

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-14

DELETED C29

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

DELETED C29

FIGURE 9.6-14

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-15

REFER TO ENGINEERING DRAWING

5613-M-3020 , SHEET 1

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

PRIMARY WATER MAKEUP SYSTEM

FIGURE 9.6-15

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-16

REFER TO ENGINEERING DRAWING

5613-M-3020 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

PRIMARY MAKEUP WATER SYSTEM

FIGURE 9.6-16

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-17

REFER TO ENGINEERING DRAWING

5614-M-3020 , SHEET 1

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FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 4

PRIMARY WATER MAKEUP SYSTEM

FIGURE 9.6-17

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.6-18

REFER TO ENGINEERING DRAWING

5614-M-3020 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 4

PRIMARY MAKEUP WATER SYSTEM

FIGURE 9.6-18

APPENDIX 9.6A

FIRE PROTECTION PROGRAM REPORT

Note:
The Fire Protection Program Report Appendix 9.6A has been deleted.
Refer to Section 9.6.1 for more Details

9.7 EQUIPMENT AND SYSTEM DECONTAMINATION

9.7.1 DESIGN BASIS

Activity outside the core could result from fission products from defective fuel elements, fission products from tramp uranium left on the cladding in small quantities during fabrication, products of n - Y or n - p reactions on the water or impurities in the water, and activated corrosion products. Fission products in the reactor coolant associated with normal operation and tramp uranium are generally removed with the coolant or in subsequent flushing of the system to be decontaminated. The products of water activation are not long lived and may be removed by natural decay during reactor cooldown and subsequent flushing procedures. Activated corrosion products are the primary source of the remaining activity.

The corrosion products contain radioisotopes from the reactor coolant which have been absorbed on or have diffused into the oxide film. The oxide film, essentially magnetite (Fe_3O_4) with oxides of other metals including Cr and Ni, can be removed by chemical means presently used in industry.

Water from the Reactor Coolant System and the spent fuel pit is the primary potential source of contamination outside of the corrosion film of the reactor coolant system. Contact while working on primary system components could result in contamination of the equipment, tools and clothing of the personnel involved in the maintenance. Also, leakage from the system during operation or spillage during maintenance could contaminate the immediate areas and could contribute to the contamination of the equipment, tools, and clothing.

9.7.2 METHODS OF DECONTAMINATION

Surface contaminants which are found on equipment in the reactor coolant 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 contaminants are generally on the surface only of non-porous materials. Personnel and their clothing are decontaminated according to the standard health physics requirements.

Those areas 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 surfaces are sufficient to remove any radioactivity present.

The corrosion films generally are tightly adhering surface contaminants, and must be removed by chemical processes. The removal of these films is generally done with the aid of commercial vendors who provide both services and formulations. Since decontamination experience with reactors is continually being gained, specific procedures may change for each decontamination case.

Portable components may be cleaned with a combination of chemical and ultrasonic methods if required.

9.7.3 DECONTAMINATION FACILITIES

Permanent Equipment decontamination facilities on site include a CO_2 decontamination facility located in the Dry Storage Warehouse and a Cask Handling Facility for each unit. Portable enclosures may also be used for decontaminating tools and equipment. Fuel handling tools and other tools can be cleaned and decontaminated in the facility, the portable enclosures or in the refueling canal area. C26

The Cask Handling Facility is used to decontaminate surfaces of large equipment and tools. In the Cask Handling Facility, the outside surfaces of the spent fuel casks and radiological material shipping containers are decontaminated before they are hauled away. The cask exterior is decontaminated before it is hauled away. Prior to hauling a spent fuel cask or radiological material shipping container to the ISFSI or offsite, the outside surface is inspected for contamination in accordance with plant procedures. C26

A personnel decontamination shower facility is located adjacent to the dress facility east of the radwaste building. It is designed to prevent potential dose rates from exceeding the limits specified in the Code of Federal Regulations. The ventilation system filters water vapor and room air before being released to the outside environment, and water from the decontamination shower facility is drained to the radwaste holdup tank.

9.8 AUXILIARY BUILDING HVAC SYSTEMS AND CONTAINMENT PURGE SYSTEMS

This section describes the design and operation of the Auxiliary Building Ventilation System, Containment Purge System, and Electrical Equipment Room HVAC System.

9.8.1 AUXILIARY BUILDING VENTILATION AND CONTAINMENT PURGE SYSTEMS

9.8.1.1 DESIGN BASIS

The Auxiliary Building Ventilation System is designed to meet the following principal criteria:

a) Ensure adequate heat removal from equipment rooms and open areas during non-design basis accident conditions.

b) Control direction of flow of potential airborne radioactivity from areas of low activity through areas of higher activity, to the common ventilation exhaust during non-design basis accident conditions.

The Containment Purge System is designed to purge the containment atmosphere for unlimited access during shutdown periods.

9.8.1.2 SYSTEM DESIGN AND OPERATION

The Auxiliary Building Ventilation System provides clean air to the operating areas of the Auxiliary Building. The Auxiliary Building Ventilation System is shown schematically in Figure 9.8-1. The system exhausts air from the equipment rooms and open areas of the Auxiliary Building and Unit 4 spent fuel storage pit, through a closed system. The Unit 4 Spent Fuel Pool and New Fuel Storage Area ventilation is shown schematically in Figure 9.8-4. The exhaust system includes a 100 percent capacity bank of high efficiency particulate air (HEPA) filters, and two 100 percent capacity fans discharging to the atmosphere via the plant vent. A separate fan exhausts air from the Unit 3 spent fuel area through HEPA filters to its own vent and is not connected to the Auxiliary Building Ventilation System. The Unit 3 Spent Fuel Pool and New Fuel Storage Area ventilation is shown schematically in Figures 9.8-3 and 9.8-4. Radiation monitoring is provided to monitor gases and particles discharged from the spent fuel vents. Additionally, the Unit 3 Cask Handling Facility ventilation system exhausts air into the Unit 3 Spent Fuel Pit Vent, as shown schematically in Figures 9.8-3; the Unit 4 Cask Handling Facility ventilation system exhausts air into the Plant Vent, as shown schematically in Figures 9.8-4; and the Unit 3 and 4 Steam Jet Air Ejector (SJAE) exhausts into the plant vent, as shown schematically in Figures 9.8-7 and 9.8-8. These arrangements ensure the proper direction of air flow for removal of potential airborne radioactivity from the Auxiliary Building, and spent fuel areas.

The Auxiliary Building Ventilation System provides a minimum of five air exchanges per hour for each of the rooms and open areas of the building. The minimum of five air exchanges per hour no longer applies to the laundry room per blank off of air inlets and installation of split unit a/c system. This assures adequate heat removal from operating equipment.

Operation of this system would be interrupted by a loss of normal power supplies, as the main supply and exhaust fans are not required for operation of engineered safety features. These exhaust fans can be manually loaded on to the emergency diesel generators. C26

Additionally, the high head safety injection pump room has louvered doors installed to ensure long term cooling of the high head safety injection pumps following an accident. C26

The Containment Purge System is independent of the Auxiliary Building Ventilation System and includes provisions for handling both supply and exhaust air. The Containment Purge System ventilation is shown schematically in Figures 9.8-5 and 9.2-6. The supply system includes an outside air connection to roughing filters, a fan duct system and a supply penetration with two quick-closing butterfly valves for bubble tight shut-off. The exhaust system includes an exhaust penetration with two quick-closing butterfly valves similar to those above, a duct system, fan, and roughing filters with connection to the plant vent. Both supply and exhaust systems include filters and two fans, one for each containment, with power-operated, fail-closed isolating dampers. The full flow rate is 35,000 CFM per containment which is equivalent to 1.33 air changes per hour. However, as discussed in the note to Table 9.8-1, the purge supply and exhaust valves have been adjusted to limit flow to approximately 7000 CFM. The quick closing butterfly valves act as purge isolation valves and are capable of closing in less than five seconds on receipt of the containment isolation signal or high activity signal from the air particulate and gas monitor. In each case, there are two valves in series, one inside and one outside the containment.

Prior to purging the containment, the air particulates and gas monitor and the Area Radiation Monitor indications of the closed containment activity levels will be used to guide the operator in use of the purge system. When purging the containment, releases from the plant vent are continuously monitored.

When the plant is in a state other than cold shutdown or refueling shutdown, the opening angles of the purge valves are mechanically restrained to less than or equal to 30° (54-inch purge return valves) and less than or equal to 33° (48-inch purge supply valves), hence the pre-access time period is increased to reduce containment humidity and allow habitation. Stress reports on the two sizes of valves by the supplier indicate that the margins between calculated stresses for the various parts and allowables are adequate to qualify the valves (in blocked position) for closure during a DBA-LOCA.

Debris screens are located inside containment inboard of the containment supply and exhaust purge valves. The debris screens and the pipe between the screens are seismically designed. The debris screens are designed to withstand the peak containment differential LOCA pressure. The debris screens will preclude material from blocking the containment isolation valves following a LOCA.

9.8.2 ELECTRICAL EQUIPMENT ROOM HVAC SYSTEM

9.8.2.1 DESIGN BASIS

The Electrical Equipment Room HVAC System is designed to accomplish the following:

1. To remove heat dissipated by all equipment in the Electrical Equipment Room during normal plant operation and emergency conditions.
2. To provide a redundant, reliable, and independent system supplied from emergency power to maintain a temperature controlled environment for the safety related equipment located within the Electrical Equipment Rooms.

9.8.2.2 SYSTEM DESIGN AND OPERATION

The Electrical Equipment Room HVAC System is designed to provide cooling and ventilation to the Electrical Equipment Room to prevent room temperature from exceeding the 104°F limitation for the safety related equipment. The system is comprised of one non-safety related chilled water air conditioning system and two safety related air conditioning (A/C) systems. Both safety related and non-safety related systems are controlled by wall mounted thermostats in the Electrical Equipment Room. The non-safety related system consists of one air handler with two redundant chiller units located on the roof of the Auxiliary Building. The non-safety related system is designed to maintain the Electrical Equipment Rooms below 77°F during normal plant operation with one chiller in operation. If one chiller unit fails, the other chiller will automatically start and maintain the room temperature within design limits. In the event that both chiller units fail, or the air handler fails during normal operation, the two safety related air conditioning systems are designed to auto-start on the predetermined thermostat temperature settings and prevent room temperature from exceeding the 104°F limitation.

The safety related system consists of two separate split A/C trains. One of the safety related air conditioning condensing units is located at the 18 foot elevation to the west and south of the Electrical Equipment Room, while the redundant condensing unit is located on the roof of the penetration area at the 42 foot elevation. The air handling units are located in the Electrical Equipment Room. Each A/C train is capable of maintaining the Electrical Equipment Room at a temperature of no more than 104°F. Because the safety related A/C units are required to remove emergency heat loads, they must be capable of operating during a loss of offsite power. Both safety related A/C trains are powered from vital motor control centers and, therefore, can be powered from the emergency diesel generators.

9.8.2.3 SAFETY ANALYSIS

The Electrical Equipment Room HVAC System is designed with sufficient redundancy such that any single component failure, up to the malfunction of the non-safety related A/C system and one safety related A/C train, will not prevent the system from performing its intended cooling function. The design of the safety related air conditioning units provides protection against tornado generated missiles for outdoor components. This protection is based on separation and location of redundant components. In addition, the safety related air conditioning units have been designed to meet the requirements for Seismic Category I components.

TABLE 9.8-1

AUXILIARY BUILDING FAN DATA SUMMARY

System	Units Installed Per Containment	Unit Capacity Each	Units Required for Normal Operation	
Purge System:				
(1) Purge Supply				
Fans	1	35,000 CFM*	1	
Fan static pressure	--	3 in. W.G.	--	
Fan motor (460 V, 3 PH, 60 Hz)	1	20 HP	1	
Air Filters, roughing	1	7,000 CFM*	1	C26
(2) Purge Exhaust				
Fans	1	35,000 CFM*	1	
Fan static pressure	--	2 in W.G.	--	
Fan motor (460 V, 3 PH, 60 Hz)	1	20 HP	1	
Air Filters, Roughing	20	35,000 CFM	20	
(3) Aux. Bldg., Exhaust				
Fans	1	40,000 CFM	1	
Fan static pressure	--	4 1/2 in. W.G.	--	
Fan motor	1	40 HP	1	
Air filters, HEPA	20	40,000 CFM	20	

*The containment purge supply and exhaust valves have been adjusted to limit air flow to approximately 7000 CFM. See FPL letter to NRC L-83-120, R. E. Uhrig to A. Schwencer, dated March 4, 1983, "Containment Purge Operability".

RADWASTE SOLIDIFICATION BUILDING FAN DATA SUMMARY

System	Units Installed Per Containment	Unit Capacity Each	Units Required for Normal Operation
(1) Radwaste Bldg. Exhaust			
Fans	2	9,210 CFM	1
Fan Static Press	--	9.54 in. W.G.	--
Fan Motor	2	30 HP	1

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.8-1

REFER TO ENGINEERING DRAWING

5610-M-3060 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

AUXILIARY BUILDING VENTILATION

FIGURE 9.8-1

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.8-2

DELETED

C25

Revised 06/30/2011

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

AUXILIARY BUILDING VENTILATION
LAUNDRY DRYERS EXHAUST

FIGURE 9.8-2

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.8-3

REFER TO ENGINEERING DRAWING

5613-M-3034 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNIT 3**
SPENT FUEL POOL AND NEW FUEL STORAGE AREA VENTILATION **FIGURE 9.8-3**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.8-4

REFER TO ENGINEERING DRAWING

5614-M-3034 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

SPENT FUEL POOL AND
NEW FUEL STORAGE AREA
VENTILATION
FIGURE 9.8-4

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.8-5

REFER TO ENGINEERING DRAWING

5613-M-3053 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

CONTAINMENT PURGE SYSTEM
AND
PENETRATION COOLING SYSTEM
FIGURE 9.8-5

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.8-6

REFER TO ENGINEERING DRAWING

5614-M-3053 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

CONTAINMENT PURGE SYSTEM
AND
PENETRATION COOLING SYSTEM
FIGURE 9.8-6

C31

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.8-7

REFER TO ENGINEERING DRAWING

5613-M-3014, SHEET 3

Revised 05/17/2021

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNIT 3**
CONDENSER SYSTEM **FIGURE 9.8-7**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.8-8

REFER TO ENGINEERING DRAWING

5614-M-3014, SHEET 3

C31

Revised 05/17/2021

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

CONDENSER SYSTEM

FIGURE 9.8-8

9.9 CONTROL BUILDING VENTILATION SYSTEM

9.9.1 CONTROL ROOM

9.9.1.1 DESIGN BASIS

Criterion: *GDC 19–Control Room.* A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

C26

PTN has committed to 10 CFR 50, Appendix A, GDC 19 - Control Room as part of the change to the Alternate Source Term (AST) methodology for dose analysis as approved in Amendments 244/240. 10 CFR 50.67 was issued by the NRC to permit holders of facility operating licenses to revise the traditional accident source term (TID 14844) that is used in the design basis accident radiological consequence analyses with one derived from the AST methodology and requires an evaluation of the consequences of affected design basis accidents. The Control Room Ventilation System (CRVS) is designed to maintain doses to the operators within the regulatory limits of GDC-19 following the radiological consequences of a Loss-of-Coolant Accident (LOCA), Main Steam, Line Break (MSLB), Steam Generator Tube Rupture (SGTR), Locked Rotor (LR), Rod Cluster Control Assembly (RCCA) Ejection, Fuel Handling Accident (FHA), and Waste Gas Decay Tank (WGDT) Rupture accidents. Each of the radiological release accidents was evaluated following regulatory guidance for implementation of the AST provided in NRC Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors".

Clarification Item III.D.3.4, “Control Room Habitability Requirements," of NUREG 0737 required all licensees to assure that control room operators would be adequately protected against the effects of an accidental release of toxic or radioactive gases such that the unit(s) could be safely operated or shut down under design basis accident conditions. C26

The function of the CRVS, including the Control Room Emergency Ventilation System (CREVS), is to provide a controlled environment for the comfort and safety of control room personnel and to assure the operability of control room components during normal operating, anticipated operational occurrences, and design basis accident conditions. C26

The design basis of the system with respect to radiological emergencies is to be capable of automatically starting under accident conditions to initiate emergency control room pressurization and filtration, assuming the occurrence of a single active damper or supply fan failure. C26

The design basis of the system with respect to other emergencies that could affect the control room environment is to be capable of manual actuation. Additionally, multiple self-contained breathing apparatus units are in and near the control room for use by the control room personnel during accidental releases of toxic gases.

As part of adopting the Alternative Source Term (AST) methodology, a Compensatory Filtration Unit has been added to the CREVS, capable of manual actuation as a qualified backup to the CREVS Recirculation Filter train. Both of these filtration systems take suction through dual CREVS emergency air intake ducts which are located in diverse wind directions, designed to seismic criteria, and protected from environmental effects including tornado generated missiles. The AST dose analyses are based on having balanced outside air makeup from both emergency air intakes. Balanced flow is achieved by throttling valves provided in each intake branch and manual isolation valves are provided for system maintenance and designed to limit inleakage into the system. C26

9.9.1.2 SYSTEM DESIGN AND OPERATION

The control room atmosphere is filtered, and conditioned as required by a separate ventilation system as shown on Figures 9.9-1, 9.9-2, 9.9-4 and 9.9-5. This system circulates air from the control room and control room offices through return air ducts to three 100% capacity air handling units located in the mechanical equipment room, adjacent to the cable spreading room. Outside air is drawn into the air handling units through roughing filters, and cooled as required. Conditioned air is then directed back to the rooms through a supply air duct system. The associated air conditioning condensing units are located on top of the Control Building roof.

The CRVS, which normally draws in fresh air from the outside, also has the capability to go into a recirculation mode that is part of the CREVS. In the recirculation mode, fresh air provided from the CREVS intake piping and recirculated air from the Control Room is processed through the Recirculation Filters [High Efficiency Particulate Air (HEPA) filters and charcoal filters], and Supply Fans to maintain an acceptable control room environment during adverse radiological conditions. The CREVS Compensatory Filtration Unit is provided as a qualified backup to the CREVS Recirculation Filter train in the unlikely event of its inoperability. C26

The CREVS intake piping is shared by both the CREVS Recirculation Filter train and Compensatory Filtration Unit. Provisions have been made to allow isolation, as necessary, of either of the two duct branches that feed CREVS Recirculation Filter train and CREVS Compensatory Filtration Unit. C26

All three HVAC units are powered by swing power sources, each of which can be powered by the emergency diesel generators. One HVAC unit is powered by MCC 3D, one unit by MCC 4D, and the third unit is powered via a transfer switch which automatically transfers between MCCs 3B and 4B. This configuration precludes the loss of more than one HVAC unit for any postulated single failure. Control room equipment is designed to operate in an environment of 120°F and 95% relative humidity. If two of three units were inoperative, the third would maintain the environment within these limits.

The Control Room Envelope (CRE) is established as a component of the Control Room Emergency Ventilation System (CREVS). The CRE consists of the Control Room and the Mechanical Equipment Room (located in the southwest corner of the Cable Spreading Room) including the Control Room's offices, rack area, kitchen, and lavatory. Both rooms are considered part of the envelope because both are serviced and pressurized by the control room's air handlers through common ductwork. C26

The boundaries of the envelope are the floors, walls, ceilings, dampers, doors and ductwork of the two rooms and are designed to limit the unfiltered inleakage to less than or equal to 100 cfm when operating in the emergency mode. The CRE boundary integrity is required for CREVS operability.

The CREVS Compensatory Filtration Unit located in the Purge Fan Room and the associated ductwork are Safety-Related and designed to meet seismic criteria. It includes a supply fan, carbon filter, pre-filter, upstream and downstream HEPA filters. The supply fan motor is powered from the MCC by emergency diesel generators and has the same rating as the CREVS Supply Fans.

9.9.1.3 NORMAL OPERATION

During normal operation, fresh makeup air is admitted to this system through an intake louver and two dampers in series located in the west wall of the Control Building. This system maintains a positive pressure in the control room envelope greater than that in the cable spreading room in order to prevent smoke from a hypothesized fire in the cable spreading room from entering the control room. All control room penetrations are designed for leak tightness standards, including doors per NFPA 80. Since the control room is maintained at slightly more than atmospheric pressure, the infiltration of contaminated air into the control room is negligible.

Two radiation monitors located in the normal air intake ducting continuously monitor for radiation in the incoming air. In the unlikely event of a maximum hypothetical accident (MHA), the Control Room ventilation will automatically be placed in a recirculation mode as described further in Section 9.9.1.5.

9.9.1.4 CONTROL ROOM EMERGENCY OPERATION DESIGN

9.9.1.4.1 CREVS Recirculation Filter Train

Two emergency modes of operation exist:

(1) one automatic, upon receipt of applicable signals associated with a potential radiological exposure; and

(2) the other manually initiated.

The automatically initiated mode, described in detail in Subsection 9.9.1.5, provides pressurization using a limited quantity of outside air through a HEPA and charcoal filter system. Without pressurization, in-leakage in excess of radiological limits could occur. No requirement currently exists for the complete isolation provided by the manually initiated mode, since no concerns related to chemical releases have been identified for the site.

To ensure that the Control Room operators are not impaired by an ammonia storage tank spill at Turkey Point Unit 5, a layer of floating (special surface blanketing) balls has been installed in the impoundment basin below the ammonia storage tanks. These balls will automatically arrange themselves into a close packed formation if a spill occurs and reduce the release of ammonia to the atmosphere. Consequence modeling demonstrates that the concentration of ammonia in the control room will remain below the Occupational Safety and Health Administration Permissible Exposure Levels (OSHA – PEL) without operator action. These levels are significantly less than the limits to which Turkey Point committed in RG 1.78, Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Chemical Hazardous Chemical Release. C26

9.9.1.4.2 CREVS Compensatory Filtration Unit

The emergency mode of operation of the CREVS Compensatory Filtration Unit is initiated manually only.

When the CREVS Recirculation Filter train is in standby or in operation, the CREVS Compensatory Filtration Unit is turned off and isolated from the associated CREVS intake branch by an isolation damper. C26

The CREVS Compensatory Filtration Unit is manually aligned for operation when the CREVS Recirculation Filter train is declared inoperable. Alignment of the CREVS Compensatory Filtration Unit for operation requires the closing of the isolation damper in the intake branch that feeds the CREVS Recirculation Filter train and the opening of the isolation damper in the intake branch to CREVS Compensatory Filtration Unit.

9.9.1.5 AUTOMATIC EMERGENCY OPERATION

Security-Related Information - Withheld Under 10 CFR 2.390

C26

9.9.1.6 CONTROL ROOM HABILITY PROGRAM

The administrative site procedure, "Control Room Habitability Program", provides the administrative controls to ensure and verify that the Control Room Envelope (CRE) meets the requirements for operability and habitability during normal, emergency, and post-accident conditions. The CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Ventilation System (CREVS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under Design Basis Accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 REM Total Effective Dose Equivalent (TEDE) for the duration of the accident.

The program shall include the following elements: C26

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) measuring the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement of CRE pressure with respect to all areas adjacent to the CRE boundary at designated accessible locations.

9.9.2 CONTROL BUILDING ANNEX

9.9.2.1 DESIGN BASIS

The DC Equipment/Inverter Room HVAC System is designed to accomplish the following:

1. Provide a redundant, reliable, independent means of maintaining the room temperatures below the qualified operability temperature of the equipment located in the DC Equipment/Inverter rooms.
2. Maintain the battery rooms at a temperature above that at which the battery capacity must be de-rated below its required capacity.
3. Maintain adequate ventilation to ensure that hydrogen concentration remains below the lower limit for flammability.
4. Maintain room temperature during normal plant operation below the continuous operation qualification temperature of the equipment location in the rooms.

9.9.2.2 SYSTEM DESIGN AND OPERATION

The DC Equipment/Inverter Room HVAC system provides cooling ventilation in the Control Building Annex. This system provides cooling to the equipment in the inverter rooms, the DC equipment rooms and the battery rooms which comprise the annex. The HVAC system for these rooms consists of a common split A/C unit and two packaged A/C units. Each packaged unit is dedicated to the north or south equipment rooms. The common unit can provide air to both the north and south rooms. Refer to Figure 9.9-3, sheets 1 and 2.

The system design also incorporates a supplemental cooling system consisting of portable fans and administrative controls. This supplemental cooling system will be used to enhance ventilation in the room and also to draw cooler air from adjacent rooms to maintain temperatures in a range compatible with equipment operation. When not in use the dedicated fans are stored in seismically designed restraints in close proximity to the equipment rooms.

All of the ventilating and air conditioning equipment is capable of being powered from an emergency diesel generator (EDG). The common split A/C unit is automatically loaded on the EDGs following a loss of offsite power. The north and south units are powered from vital buses and may be manually started after a loss of offsite power. Special dedicated receptacles have been provided in the rooms to power the portable fans. These fixtures are 120 VAC fed from an EDG-backed source.

This system circulates air from these rooms to air conditioning units or the air handling unit and returns cool air into the rooms. Each unit is controlled by a thermostat. These units are designed to maintain the temperature in the rooms below 104°F.

In the event of a fire or failure of one of the HVAC units, the room temperature may increase. Routine surveillance of these rooms is performed to confirm a suitable environment for the equipment is maintained. Should increasing temperatures be noted, supplemental cooling may be initiated using the portable fans. The batteries have been shown to be operable at temperatures up to 110°F and the other safety related equipment is operable at temperatures up to 135°F (for short time periods). The supplemental cooling system is capable of maintaining temperatures below these limits.

9.9.2.3 SAFETY ANALYSIS

The DC Equipment/Inverter Room HVAC system is designed with sufficient redundancy such that component failures up to and including malfunction or unavailability of two of the three cooling units, will not prevent the system from performing its cooling and ventilation functions. This capability credits the administrative controls and operator actions to monitor the room temperatures and align the portable fans if conditions warrant their use. Analyses have demonstrated that the supplemental cooling system can maintain the room temperatures within an acceptable range for electrical equipment operability. Analysis (JPN-PTN-SEMP-93-010) also shows that hydrogen concentrations will be maintained below flammable limits. The portable fans are stored in seismically designed racks, and provided with EDG backed power receptacles so that neither loss of offsite power nor earthquake will cause loss of cooling capacity. Lead blankets are provided to seismically restrain the fans during free standing operations(i.e. when removed from storage racks).

Based on the preceding, the adverse effects of any credible event or system failure will not compromise the ability to provide cooling to the DC Equipment/Inverter rooms.

9.9.3 COMPUTER/CABLE SPREADING ROOM HVAC SYSTEM

9.9.3.1 DESIGN BASIS

The Computer/Cable Spreading Room HVAC System is designed to accomplish the following:

1. To remove heat dissipated by all equipment in the Computer and Cable Spreading Rooms During normal plant operation and emergency conditions.
2. To provide a redundant, reliable, and independent system supplied from emergency power to maintain a temperature controlled environment for the safety-related equipment located within the Computer and Cable Spreading Rooms.

9.9.3.2 SYSTEM DESIGN AND OPERATION

The Computer/Cable Spreading Room HVAC System provides cooling and ventilation to the equipment located in the Computer Room and Cable Spreading Room. The system is designed to maintain temperatures in the rooms below the 104°F limit for the safety related equipment.

The system is comprised of two independent chilled water A/C trains. Each train consists of 100% capacity chiller package located on the Control Building roof and three air handling units. Two 50% capacity air handling units for each train are located in the Computer Room. One 100 % capacity air handling unit for each train and a common duct run are located in the Cable Spreading Room. Each train is capable of providing 100% cooling for both rooms during normal and emergency conditions.

During a loss of offsite power (LOOP), the system is not automatically loaded on the Emergency Diesel Generator (EDG). The system is manually loaded on the EDG by administrative procedures. Supplemental cooling system discussed in the Control Building Annex section will be used to enhance ventilation in the room and also to draw cooler air from adjacent rooms to maintain temperatures in a range compatible with equipment operation. When not in use the dedicated fans are stored in seismically designed restraints in close proximity to the equipment rooms. Temperature indicator located in the Control Room provide indication to allow operators to load the system prior to exceeding the temperature limitations.

9.9.3.3 SAFETY ANALYSIS

The Computer/Cable Spreading Room HVAC System is designed with sufficient redundancy such that any single component failure, up to malfunction of a complete train, will not prevent the system from performing its intended cooling function. The Computer/Cable Spreading Room HVAC System is designed and installed to Seismic Category I requirements. In addition, the chilled water equipment located on the roof is designed to withstand the effects of a tornado missile.

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.9-1

REFER TO ENGINEERING DRAWING

5610-M-86

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

CONTROL BUILDING HVAC
EL 42'-0"

FIGURE 9.9-1

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.9-2

REFER TO ENGINEERING DRAWING

5610-M-87 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNITS 3 & 4

CONTROL BUILDING HVAC

EL. 30'-0"

FIGURE 9.9-2

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.9-3, SHEET 1

REFER TO ENGINEERING DRAWING

5610-M-85 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNITS 3 & 4

DC EQUIPMENT/INVERTER ROOMS

HVAC

SHEET 1

FIGURE 9.9-3

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.9-3, SHEET 2

REFER TO ENGINEERING DRAWING

5610-M-85 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

DC EQUIPMENT/INVERTER ROOMS
HVAC SECTIONS
SHEET 2
FIGURE 9.9-3

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.9-4

REFER TO ENGINEERING DRAWING

5610-M-3025 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNITS 3 & 4**
CONTROL BUILDING VENTILATION CONTROL ROOM HVAC **FIGURE 9.9-4**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.9-5

REFER TO ENGINEERING DRAWING

5610-M-3025 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

CONTROL BUILDING VENTILATION
COMPUTER FACILITY/CABLE
SPREADING ROOM HVAC
FIGURE 9.9-5

9.10 NORMAL CONTAINMENT VENTILATION SYSTEM

9.10.1 DESIGN BASIS

Performance Objectives

The normal containment ventilation system is designed to accomplish the following:

a) Remove the normal heat lost from all equipment and piping in the containment during operation and to maintain the temperature at or below a normal ambient of 120°F. Operation with containment bulk temperatures above 120°F has been evaluated. The evaluation (Reference 1) concludes that operation at containment bulk ambient temperatures above 120°F but not exceeding 125°F for a cumulative period of two weeks (i.e., 336 temperature dependent equivalent hours which is defined consistent with NRC accepted methodology [Arrhenius] for determining the qualified life of equipment to meet 10 CFR 50.49) per year is acceptable with no adverse impact on plant safety.

b) Provide sufficient air mixing and circulation throughout all containment areas to permit maintenance and/or refueling operations after reactor shutdown.

In order to accomplish these objectives, the following systems have been provided:

a) Normal Containment Cooling System
b) Control Rod Drive Mechanism Cooling System

Design Description

The design characteristics of the equipment required in the containment for cooling during normal operation are presented in Table 9.10-1. The Containment Ventilation System is shown schematically in Figure 9.10-1. The Containment Ventilation System diesel generator loading is presented in Section 8.2. C26

9.10.2 SYSTEM DESIGN

The Normal Containment Cooling fans are of the centrifugal type, belt drive. Control rod drive mechanism cooling fans are direct driven, vane-axial. Each of the Normal Containment Coolers is provided with a motor trip alarm which alerts the operator of a failed fan and reduced air flow. C26

The dampers are provided with limit switches to indicate damper position.

<u>Containment Ventilation</u>

Turkey Point Unit 3 and Unit 4 each have four (4) fan-coil units that discharge and distribute cooled air through containment via common headered, discharge ducting. Each cooling coil in the fan-coil units is designed to transfer up to 1.94 x 10^6 Btu/hr to the Component Cooling Water system to accomplish adequate containment cooling during normal operation. Table 9.10-1 provides a summary of normal containment ventilation data. C26

Each air handling unit consists of the following equipment arranged so that, during normal operation, air flows through the assembly in the following sequence: (1) cooling coils, (2) centrifugal fan, and (3) supply header. The normal air flow rate per fan-coil unit is 120,000 SCFM. C26

The air handling units are located in the space between the reactor coolant loop (secondary) shield wall and the containment wall above the refueling floor. The shielded location makes inspection of the equipment possible at power operation under controlled access conditions and after shutdown. (See Figure 6.3-4).

The Control Rod Drive Mechanism (CRDM) Cooling System supplements the normal containment cooling system and can be used to remove heat from the reactor vessel head during natural circulation cooldown initiated by a loss of offsite power. This system consists of fans, cooling coil and ductwork to draw air through the control rod drive mechanism baffle and eject it to the containment.

The CRDM Cooling Fans are non-safety related but are fed from vital motor control centers. This allows the operator to manually load the fans onto the Emergency Diesel Generators under specified conditions. The loading of the CRDM Cooling Fans onto the Emergency Diesel Generators will be governed by strict administrative control.

The Emergency Diesel Generators loading is described in Section 8.2.

9.10.3 REFERENCES

1. "Safety Evaluation for Containment Bulk Ambient Temperatures," JPN-PTN-SENJ-88-052, Rev. 3, April 13, 1989.

TABLE 9.10-1

NORMAL CONTAINMENT VENTILATION DATA SUMMARY

SYSTEM	UNITS INSTALLED PER CONTAINMENT	UNIT CAPACITY (EACH)	UNITS REQUIRED FOR NORMAL OPERATION
CONTAINMENT Normal Cooling			
Fan-Coil Units	4	120,000 Scfm	3(1)
Cooling Coil	4	1.94 x 10^6 Btu/hr	3(1)
Fan Static Pressure	---	2.74 in. W.G.	---
Fan Motor (460 V, 3 ph, 60 Hz)	4	100 HP	3(1)
CONTROL ROD DRIVE Mechanism Cooling			
Fan-Coil Units	2	32,000 cfm	1
Cooling Coil	2	1.60 x 10^6 Btu/hr	1
Fan Static Pressure	---	6.3 in. W.G.	---
Fan Motor (460 V, 3 ph, 60 Hz)	2	60 HP	1

(1) High CCW supply temperature during summer months may require operation of all fan-coils.

C26

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.10-1

REFER TO ENGINEERING DRAWING

5613-M-3057 , SHEET 1

5614-M-3057 , SHEET 1

C26

Revised 04/17/2013

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3 & 4

C26

CONTAINMENT NORMAL AND
EMERGENCY COOLING SYSTEMS

FIGURE 9.10-1

9.11 AUXILIARY FEEDWATER SYSTEM

9.11.1 DESIGN BASIS

The Auxiliary Feedwater System is designed to:

1) Sustain operation, following a loss of offsite power, for a period of 13 hours to include maintaining the unit in Hot Standby for 4 hours followed by a 9 hour cooldown to Residual Heat Removal System entry conditions. Maximum hot standby is for 18 hours.

2) Supply auxiliary feedwater to the steam generators within 95 seconds of a start signal on low-low steam generator water level during loss of normal feedwater events in which offsite power is available. C28

3) Supply auxiliary feedwater to the steam generators within 95 seconds of a start signal on low-low steam generators water level during loss of normal feedwater events in which offsite power is not available.

9.11.1.1 SYSTEM OPERATION

The Auxiliary Feedwater System is shown in Figures 9.11-2 through 9.11-9. Upon initiation of the Auxiliary Feedwater System, the turbine steam isolation valves open and actuate position switches. These position switches actuate the solenoid valves mounted on the Auxiliary Feedwater control valves. The instrument air, in conjunction with a signal generated from the controllers in the Main Control Room is then permitted to modulate the control valves. Manual and automatic modes of operation are provided and may be selected from the Main Control Room. Flow indication is provided locally and in the Main Control Room. Auxiliary Feedwater Flow Control Valves position are provided locally.

9.11.1.2 Auxiliary Feedwater Flow Control Valves

Each steam generator auxiliary feedwater line, Train 1 and Train 2, has a flow element, flow transmitter, and flow control valve. When the AFW system is in the standby mode, controllers in the control room are set to a predetermined flow rate as delineated in the Technical Specifications and plant operating procedures.

The controller pneumatic output signal to the air operated flow control valve is blocked by a solenoid valve, thus the control valve is closed. Upon receipt of an AFW steam supply MOV open signal, either auto or manual, the solenoid valves associated with the flow control valves to the steam generators on the affected unit will be energized to open. This permits the predetermined output signal from the controller to be applied to the flow control valve, supplying AFW to the affected unit's steam generators. The flow control is normally set to supply at least 125 gpm to each steam generator. A travel limit stop is installed on each flow control valve and is adjusted to limit the stem travel to 95% and 100% of full open, limiting AFW flow to a Steam Generator to 1318.5 gpm, should one of the two flow control valves fail to the full open position.

Flow indicators for AFW Trains 1 and 2 are installed under the main feedwater platform for use during manual operation of the flow control valves. Flow indicators are also provided on the Main Feedwater Platform.

9.11.1.3 NITROGEN BACKUP SYSTEM TO THE AUXILIARY FEEDWATER CONTROL VALVES

The Auxiliary Feedwater System flow control valves are normally operated using motive force from the Instrument Air System. As a backup to this source, a nitrogen system is available (see Figures 9.11-6 and 9.11-9) via automatic transfer on low instrument air pressure. The nitrogen backup system, consisting of two racks of five bottles each, supplies nitrogen at approximately 80 psig via a seismically qualified regulator to the AFW Flow Control Valves for an independent AFW train. Three out of five bottles are normally open to the header which will provide at least two hours of nitrogen supply. When the pressure of the three nitrogen bottles decreases to 750 C30
psig, a low pressure alarm alerts the control room operator for "valving in" the two spare (off-line) bottles within 45 minutes after the initiation of the alarm, in order to maintain automatic operation of the AFW flow control valves.

The nitrogen bottle racks are designed for hurricane, tornado wind and seismic loads. These are provided with a missile shield to prevent any damage to plant equipment and the redundant train due to internally generated missiles.

9.11.2 AUXILIARY FEEDWATER PUMPS

Three quick starting steam turbine driven, auxiliary feedwater pumps are provided for Turkey Point Units 3 and 4. Each pump is capable of delivering 624.8 gpm to the steam generators between 1085 psig at 5900 rpm and 120 psig at 3200 rpm.

The three pumps are installed such that each supplies auxiliary feedwater to either Unit 3 or 4, with any single pump supplying the total feedwater requirement of either unit. Two pumps (B&C) are normally aligned to AFW Train 2 and the third (A) is normally aligned to AFW Train 1.

The turbine driven pumps are supplied with steam from the unit which has lost its normal feedwater supply. RPM indicators are provided locally and in the control room to provide indication that the AFW pump/turbine is running. The turbines have an atmospheric exhaust. Steam can also be supplied from the unit having normal feedwater supply or from the unit's auxiliary steam supply. The supply valves will automatically open by any one of the following five signals. C28

1. Safety Injection.
2. Low-Low Level in any of the three steam generators.
3. Loss of both feedwater pumps under normal operating conditions.
4. Bus Stripping.
    a. Loss of voltage on either the A or B 4.16 KV bus.
    b. Degraded voltage on one 480V load center (instantaneous) coincident with safety injection and the diesel generator breaker open.
    c. Degraded voltage on one 480V load center (delayed) coincident with the diesel generator breaker open.
5. AMSAC signal.

The turbine casing is provided with a sentinel type relief valve for warning purposes only.

Impulse type steam traps are provided upstream of the Steam Supply MOVs and drain to the condenser. Additional impulse type steam traps are provided upstream of the trip and throttle valves and drain to an adjacent drain trough. The orifice type steam traps, the turbine casing drains, the exhaust pipe drain, the gland seal drain, the governor valve and the HP and LP steam leakoffs in the throttle trip valve drain to an adjacent drain trough. The pump recirculation is controlled by an orifice in the recirculation piping. The pumps can continue to supply reduced amounts of water to the steam generators until steam pressure is reduced to 85 psig. The pump output in pounds per hour is greater than the steam consumption until the 85 psig point is reached. However, at 120 psig, the Residual Heat Removal System is started and the auxiliary feedwater pumps are shutdown. Operation of the AFW pumps under low flow conditions is administratively controlled in plant procedures.

Cooling water is supplied to the safety related auxiliary feedwater pump oil coolers from the second stage of the auxiliary feed pump and is discharged to the condensate storage tanks.

Standby Feedwater Pumps

Two non-safety grade standby steam generator feedwater pumps (SSGFP) are provided; refer to Figure 10.2-21. The standby steam generator feedwater pumps are normally used to supply feedwater to the steam generators during normal start-up, shutdown, and hot standby conditions. The pumps take suction from the demineralized water storage tank and discharge into the main feedwater header upstream of the feedwater regulating valves. These pumps can be operated from the control room or from the local control panel. One pump is motor driven and normally powered from the 4160 volt C-Bus. The other pump is diesel engine driven with an integral fuel tank and electric starting system.

C30

The Standby Feedwater System is shared by Unit Nos. 3 and 4 such that any one SSGFP can supply either or both units as necessary to meet feedwater demand. The system provides a shutdown function but not a safety-related or emergency function. In case of loss of offsite power, the normal safety supply of feedwater to the steam generators is provided by the steam turbine driven auxiliary feedwater pumps. However in the event the AFW system does not function properly, the Standby Feedwater System can be used as a backup water supply, during which the SSGFPs can be manually started, aligned, and controlled by the operator, as needed. In the event of a loss of offsite power in which AFW is unavailable, feedwater can be supplied by the diesel engine-driven SSGFP (Reference 1). For fires affecting the AFW pumps, credit is taken for the Standby Feedwater System. C30

License Amendments 282 and 276 to the Turkey Point Unit 3 and 4 operating licenses, respectively, relocated the Standby Feedwater System requirements from the Technical Specifications to UFSAR Section 12.12.3 and applicable licensee controlled documents (Reference 2). Plant surveillance and maintenance procedures implement the applicable requirements previously specified in the Technical Specifications for evaluating Standby Feedwater System capability to provide makeup water to the steam generators. Related procedures specify appropriate compensatory and corrective actions in the event the system, or portions thereof, become non-functional.

SSGFP functionality is verified by starting and operating the pumps in the recirculation mode. Each SSGFP is also periodically started and aligned to provide flow to the nuclear unit's steam generators. This surveillance regimen demonstrates functionality of the entire flow path, backup non-safety grade power supply and pump associated with a unit at least each refueling outage. The pump, motor driver, and normal power supply availability are typically demonstrated by operation of the pumps in the recirculation mode monthly on a staggered test basis. The functionality of the diesel engine driver for the B SSGFP is periodically verified. In addition, an inspection on the diesel is periodically performed in accordance with procedures prepared in conjunction with manufacturer recommendations. The inspection ensures that the diesel driver is maintained in good operating condition. C30

The Standby Steam Generator Feedwater System is expected to perform with high reliability. FPL intends to maintain the system in good operating condition with regard to structures, supports, component maintenance, calibrations, etc. The Standby Feedwater System requirements assure system and component functionality. Changes to the Standby Feedwater System requirements are subject to the regulatory controls of 10 CFR 50.59.

9.11.3 CONDENSATE STORAGE TANKS

Normal water supply to the auxiliary feedwater pumps is from the two 250,000 gallon (nominal) condensate storage tanks, through locked open gate valves and check valves. Each tank contains a 210,000 gallon minimum indicated volume which assures a minimum usable volume of 195,331 gallons of demineralized water for the auxiliary feedwater pumps.

The condensate storage tank design sizing is based on allowing each unit to be taken from full power to hot standby following a loss of offsite power, and:

1. Kept at hot standby for 4 hours and then cooled to 350°F in 9 hours, at which point the Residual Heat Removal System will be put in service, or

2. Kept at hot standby for about 18 hours.

Cooldown rates for using either forced or natural circulation are governed by plant operating procedures.

An additional auxiliary feedwater supply can be provided from the water treatment system. Demineralized water at a maximum rate of 200 gpm per unit will be available from the water treatment system. The auxiliary feedwater requirement to remove decay heat after thirteen hours is less than 128 gpm. The condensate storage tanks are interconnected so that each of the pumps can take suction from either tank.

An alternate, non-safety source of water is the demineralized water storage tank (DWST). The DWST is a 500,000 gallon, non-safety related source of demineralized water that is considered part of the primary makeup demineralized water system. The DWST is the main source of water for the Standby Feedwater System and is the alternate source of water for the AFW system. The DWST minimum allowable inventory ensures that adequate water is available to provide reactor decay heat removal for either or both nuclear units in the event the AFW system is unavailable. C30

C30

A supply of 77,000 gallons from the DWST for the Standby Steam Generator Feedwater Pumps is sufficient water to remove decay heat from the reactor for six (6) hours for a single unit or two (2) hours for two units. The 77,000 gallons of water in the non-safety grade DWST is judged to provide sufficient time for restoring the AFW System or establishing make-up to the DWST.

The DWST maintains a minimum indicated volume of 145,000 gallons, which consists of an allowance for level indication instrument uncertainties (approximately 15,000 gallons), for water deemed unusable because of tank discharge line location and vortex formation (approximately 50,200 gallons), and the minimum usable volume (77,000 gallons). The minimum indicated volume corresponds to a water level of 9.2 feet in the DWST.

Amendments 282 and 276 to the Turkey Point Unit 3 and 4 operating licenses, respectively, relocated the DWST requirements from the Technical Specifications to UFSAR Section 12.12.3 and licensee controlled documents (Reference 2). Plant surveillance and maintenance procedures implement the applicable requirements previously specified in the Technical Specifications for verifying DWST capability to maintain an alternative source of makeup water to the steam generators. Related plant procedures specify appropriate compensatory and corrective actions in the event the DWST becomes non-functional. Changes to the DWST requirements are subject to the regulatory controls of 10 CFR 50.59. C30

9.11.4 REFERENCES

1. NRC Letter from Richard Croteau to J. H. Goldberg Dated May 20, 1994, TURKEY POINT UNITS 3 & 4 - ISSUANCE OF AMENDMENTS RE: ELIMINATION OF CRANKING DIESEL GENERATORS (TAC NOS. M87662 AND M87663).
2. NRC Letter dated September 11, 2018, Turkey Point Nuclear Generating Unit Nos. 3 and 4 – Issuance of Amendments Regarding Technical Specifications Pertaining to Explosive Gas Monitoring, Gas Decay Tanks, and Standby Feedwater System (CAC Nos. MG0143 and MG0144; EPID L-2017-LLA-0272), (ML18214A125). C30

TABLE 9.11-1

AUXILIARY FEEDWATER PUMP/TURBINE CONSTRUCTION DATA

Auxiliary Feedwater Pump

Ingersoll-Rand 2-1/2 CNTAM4, four stage centrifugal, horizontally split casing.

Performance Data	Nominal Speed	Min. Speed	
Capacity, gpm	600	600	
Temperature, F.	100	100	
N. P. S. H. required, feet	36	21	
Suction	Flooded	Flooded	
Total Dynamic Head, feet	2775(Nominal)	410	C22
Steam Pressure, psig	1085	120	
Steam Temperature, F.	556	540	
Steam Conditions	Saturated, 1/4% moisture	Saturated, 1% moisture	
Speed, rpm	5900	3200	
Turbine Back Pressure, psig	10	10	
Brake Horsepower	758(Max)	155	C22

AUXILIARY FEEDWATER PUMP/TURBINE CONSTRUCTION DATA

Pump Construction	
Casing	5% Chrome Stainless Steel
Shaft	410 Stainless Steel
Impeller	11-13% Chrome Steel
Wear Rings	Stainless Steel
Turbine Construction	
Casing	Cast Steel
Shaft	Alloy Steel
Bucket Wheels	Forged Steel
Design Pressure	1185 psig
Blades	Buckets Milled on Forged Steel Wheel
Packing	Carbon Rings
Lubrication	Forced Feed
Pump	Shaft Driven

TABLE 9.11-2

AUXILIARY FEEDWATER SYSTEM-CODE REQUIREMENTS

Component	Codes
Pumps	Hydraulic Institute Standards
Turbines	NEMA SM-22-1970
Valves & Piping	ASA-B31.1-1955

DELETED

C26

Revised 04/17/2013

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

CONDENSATE STORAGE
REQUIREMENTS
AFTER
LOSS OF OFFSITE POWER

FIGURE 9.11-1

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.11-2

REFER TO ENGINEERING DRAWING

5610-M-3075 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNITS 3 & 4

AUXILIARY FEEDWATER SYSTEM

TURBINE DRIVE FOR AFW PUMPS

FIGURE 9.11-2

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.11-3

REFER TO ENGINEERING DRAWING

5610-M-3075 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

AUXILIARY FEEDWATER SYSTEM
AUXILIARY FEEDWATER PUMPS

FIGURE 9.11-3

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.11-4

REFER TO ENGINEERING DRAWING

5613-M-3075 , SHEET 1

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

AUXILIARY FEEDWATER SYSTEM
STEAM TO AUXILIARY FEEDWATER
PUMP TURBINES
FIGURE 9.11-4

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.11-5

REFER TO ENGINEERING DRAWING

5613-M-3075 , SHEET 2

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

AUXILIARY FEEDWATER SYSTEM
AUXILIARY FEEDWATER TO
STEAM GENERATORS
FIGURE 9.11-5

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.11-6

REFER TO ENGINEERING DRAWING

5613-M-3075 , SHEET 3

REV.13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

AUXILIARY FEEDWATER SYSTEM
NITROGEN SUPPLY TO
AFW CONTROL VALVES
FIGURE 9.11-6

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.11-7

REFER TO ENGINEERING DRAWING

5614-M-3075 , SHEET 1

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FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

AUXILIARY FEEDWATER SYSTEM
STEAM TO AUXILIARY FEEDWATER
PUMP TURBINES
FIGURE 9.11-7

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.11-8

REFER TO ENGINEERING DRAWING

5614-M-3075 , SHEET 2

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FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 4

AUXILIARY FEEDWATER SYSTEM
AUXILIARY FEEDWATER TO
STEAM GENERATORS

FIGURE 9.11-8

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.11-9

REFER TO ENGINEERING DRAWING

5614-M-3075 , SHEET 3

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNIT 4**
AUXILIARY FEEDWATER SYSTEM NITROGEN SUPPLY TO AFW CONTROL VALVES **FIGURE 9.11-9**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.11-10

REFER TO ENGINEERING DRAWING

5614-M-3018 , SHEET 1

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FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 4

CONDENSATE STORAGE SYSTEM

FIGURE 9.11-10

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.11-11

REFER TO ENGINEERING DRAWING

5613-M-3018 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 3

CONDENSATE STORAGE SYSTEM

FIGURE 9.11-11

9.12 POST-ACCIDENT HYDROGEN CONTROL

Turkey Point Units 3 and 4 received an exemption from the hydrogen control requirements of 10 CFR 50.44 and 10 CFR 50, Appendix A, General Design Criteria 41, 42, and 43, in December 2001. The exemption was based in part on NRC resolution of Generic Issue 121, “Hydrogen Control for PWR Dry Containments”, and the Turkey Point Individual Plant Examination (IPE). These evaluations demonstrated that large dry containment building designs such as those at Turkey Point can withstand the effects of hydrogen combustion during design basis accidents without hydrogen concentration control.

The exemption also considered the impact of hydrogen combustion during severe accidents. Severe accidents can result in large quantities of hydrogen being released over short periods of time. The exemption acknowledged that the hydrogen control systems necessitated by 10 CFR 50.44 would likely be overwhelmed under severe accident conditions in which there is a significant amount of core damage. Thus, operation of such systems would provide no benefit in limiting the effects of hydrogen combustion, and hence would not be needed for severe accident mitigation. All discussions relating to the post accident containment ventilation system and the hydrogen recombiners will be deleted in its entirety. C29

9.12.1 DELETED

9.12.2 DELETED

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.12-1

REFER TO ENGINEERING DRAWING

5613-M-3094 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNIT 3**
POST-ACCIDENT CONTAINMENT VENT AND SAMPLING SYSTEM FLOW DIAGRAM **FIGURE 9.12-1**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.12-2

REFER TO ENGINEERING DRAWING

5614-M-3094 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 4

POST-ACCIDENT CONTAINMENT

VENT AND SAMPLING SYSTEM

FLOW DIAGRAM

FIGURE 9.12-2

9.13 POST ACCIDENT SAMPLING SYSTEM

The capability to obtain and analyze post-accident samples of reactor coolant and containment atmosphere was originally provided as a post-TMI modification. The system was provided with in-line analyzers and radiation detectors to maintain continuous sampling capability. The system was intended to provide information about the radionuclides existing post-accident to support emergency response decision making during the initial phases of an accident.

In light of the significant improvements that have been made since the TMI accident in the areas of defining realistic source terms, understanding fission product behavior, and severe accident management, the NRC staff has concluded that many of the original Post Accident Sampling System (PASS) sample requirements are either unnecessary or effectively provided by other indications of process parameters or measurement of radiation levels. Thus, from a plant risk standpoint, there is no longer a compelling need to maintain dedicated equipment for prompt analysis of post accident samples for emergency planning purposes.

In keeping with the above NRC staff position, much of the original PASS instrumentation has been abandoned in-place.

The current regulatory commitments applicable to PASS include:

a) The capability to obtain and analyze highly radioactive samples of reactor coolant, containment sump, and containment atmosphere must be provided. The use of dedicated onsite equipment to obtain and analyze these samples is not required.

b) The capability of classifying fuel damage events at the alert level threshold must be provided. This capability may utilize the normal sampling system and/or correlations of sampling or letdown line dose rates to coolant concentrations.

c) The capability to monitor radioactive iodines that have been released to offsite environs must be provided.

Sampling capability is provided for the Reactor Coolant System the containment atmosphere and the containment sump contents as shown on Figure 9.4-1, and Figure 9.4-2

The reactor coolant samples are tapped off of the normal sample lines, downstream of the containment isolation valves.

The containment sump content sample is tapped off of the normal sample lines via RHR in the recirculation mode, and is provided with reach rods on the isolation valve.

The containment atmosphere sample is tapped off of the sample line used for the containment hydrogen monitors. This sample line is provided with reach rods on the isolation valves.

9.14 POST ACCIDENT HYDROGEN MONITORING SYSTEM

The containment post-LOCA hydrogen monitoring equipment (Figures 9.12-1 and 9.12-2) will provide reliable and accurate indication of the concentration of hydrogen gas in the containment atmosphere following a loss-of-coolant accident. Two completely independent systems are provided to monitor for free gaseous hydrogen in the range of 0 to 10 percent by volume in the containment atmosphere, with a system recorder accuracy of ± 2.5% full scale.

The sampling will be either educted or received under pressure based on the condition of the containment post-LOCA environment. These systems are complete closed loops; that is the sample is returned back to the containment atmosphere. Both recording and indicating devices are provided and located in the Control Room. Channel 1 of the containment post-LOCA hydrogen monitors will be located in the Post Accident Sampling Room. Channel 2 will be located in the access area to the waste gas hold up tanks.

The hydrogen monitor sample connections tie-in to the existing post accident containment ventilation system outside containment and return via the containment atmospheric sample Plant Radiation Monitoring System (PRMS) return line. The PACVS sample ports are placed in two separate pipe headers near the containment dome.

9.15 EMERGENCY DIESEL GENERATOR AUXILIARIES

The original configuration of Turkey Point Units 3 and 4, as licensed by the NRC, utilized two emergency diesel generators (EDGs), currently labeled 3A and 3B, that were shared between the two units. In 1990 and 1991, as part of an upgrade of the Emergency Power System (EPS), two additional EDGs, labeled EDGs 4A and 4B, were added to the plant. These two new EDGs were designed, to the extent possible, to the latest standards while maintaining a consistent design approach with and avoiding extensive redesign and rework of existing Emergency Power System structures/components.

9.15.1 EMERGENCY DIESEL GENERATOR FUEL OIL STORAGE AND TRANSFER SYSTEM

9.15.1.1 DESIGN BASIS

The EDG fuel oil storage and transfer system consists of three separate systems. One system is associated with EDGs 3A and 3B, while the other two systems are associated with EDGs 4A and 4B (one system dedicated to each of the Unit 4 EDGs).

9.15.1.1.1 EDGs 3A AND 3B

The system associated with EDGs 3A and 3B is designed to:

1. Provide diesel oil storage capacity for at least seven days for one EDG.

2. Maintain diesel oil supply to at least one EDG, assuming a single failure in the system.

3. Meet the requirements for Class I systems/components in accordance with Appendix 5A.

4. Withstand the maximum flood levels in accordance with Appendix 5G, and winds in accordance with Appendix 5A without loss of function.

9.15.1.1.2 EDGs 4A AND 4B

The systems associated with EDGs 4A and 4B are designed in accordance with ANSI Standard N195-1976, as endorsed by Regulatory Guide 1.137. These systems, in conjunction, are designed to:

1. Provide diesel oil storage capacity for at least seven days for two EDGs.

2. Maintain diesel oil supply to at least one EDG, assuming a single failure in the systems coincident with a loss of offsite power.

3. Meet Seismic Category I requirements.

4. Withstand the maximum flood levels and tornado winds and missiles without loss of function by locating critical components inside the Unit 4 EDG Building.

5. Prevent the failure of nonseismic structures or components from affecting the safety related functions of the system.

9.15.1.2 SYSTEM DESCRIPTION

9.15.1.2.1 EDGs 3A AND 3B

The system associated with EDGs 3A and 3B stores diesel oil in an onsite diesel oil storage tank, transfers the diesel oil to either one of the two diesel oil day tanks (one tank is associated with each EDG), and transfers the diesel oil from each EDG's diesel oil day tank to its associated skid mounted tank. Operating diagrams representing the emergency diesel generator fuel oil storage and transfer system, which are associated with EDG 3A and 3B, are presented in Figures 9.15-3 and 9.15-4. The major design features of this system are:

1. Diesel Oil Storage Tank - The diesel oil storage tank has a capacity of 64,000 gallons, which provides sufficient storage capacity to permit one EDG (3A or 3B) to operate at its "168 hour rating" for at least 7 days. This tank is designed as a Seismic Class I structure.

2. Diesel Oil Transfer Pumps - Two diesel oil transfer pumps are provided to transfer diesel oil from the diesel oil storage tank to the diesel oil day tanks. Either of these pumps can be manually aligned to provide diesel oil flow to either or both of the diesel oil day tanks associated with EDGs 3A and 3B. In addition, the discharge lines from these pumps and the pumps associated with EDGs 4A and 4B are interconnected to provide additional diesel oil supply alignment flexibility.

3. Diesel Oil Day Tanks - Each EDG has a 4000 gallon diesel oil day tank. These tanks are designed as Class I equipment and are separated from each other by a concrete wall. Each diesel oil day tank has an alternate fill connection which is suitable for a tie-in from a mobile tank unit. These fill lines provide an alternate fill path for the diesel oil day tanks should the normal supply via the diesel oil transfer pumps become unavailable.

4. Solenoid Valves - Solenoid valves in the transfer lines from the diesel oil day tanks to the skid mounted tanks, in conjunction with level switches in the skid mounted tanks, prevent the skid mounted tanks from overflowing. The solenoid valves are provided with manual bypass valves and associated piping. This arrangement provides capability to fill the skid tank should the solenoid valve fail to open when required due to a design basis rain event. C26

5. Air-Operated Valves - Air-operated valves in the transfer lines from the diesel oil storage tank to the day tank automatically open in response to signals developed by logic circuitry incorporating tank level and pump control switch positions. The valves can be locally opened using a separate air source in the event normal instrument air is not available.

6. Skid Mounted Tanks - Each EDG has a 275 gallon skid mounted fuel tank.

7. Diesel Oil - The diesel oil is No. 2 fuel oil complying with the tests, limits, and applicable ASTM Standards specified in the diesel fuel oil testing program described in the Technical Specifications. The diesel oil is chemically treated with a biocide to prevent the deleterious effects of biological activity upon diesel oil quality and a stabilizing compound to enhance diesel oil long term storage.

8. Truck Fill Connection - Additional diesel oil can be delivered to the plant site by truck to replenish the fuel supply system during normal operation or following an accident by way of a truck fill connection. In addition, the truck fill connection to the systems for EDGs 4A and 4B may be used to fill this system's diesel oil storage tank.

Diesel oil is normally transferred from the diesel oil storage tank automatically to maintain diesel oil day tank level by the electric motor driven diesel oil transfer pump associated with the particular diesel oil day tank. Each EDG's (3A and 3B) diesel oil day tank gravity feeds though a solenoid valve to the associated skid mounted tank. Diesel oil is then drawn from the skid mounted tank, as required, through a duplex filter to the fuel manifold line. From here the diesel oil goes on to each injector inlet filter and into the injector.

As described in Section 9.15.1.4, the diesel oil storage tank is drained every ten years for sediment removal and cleaning. During this maintenance evolution, a temporary storage and transfer system may be used to provide a minimum 7-day supply of fuel oil to either the 3A or 3B EDG to satisfy the system design basis. Additional storage requirements are imposed on the Unit 3 day and skid tanks and the Unit 4 diesel oil storage tanks when this temporary system is utilized. The additional storage requirements are contained in plant procedures.

9.15.1.2.2 EDGs 4A AND 4B

The systems associated with EDGs 4A and 4B provide each of the EDGs with a completely independent diesel oil storage and transfer system. Each of these systems consists of a diesel oil storage tank, a diesel oil transfer pump, and a diesel oil day tank which, in turn, supplies fuel oil to the EDG diesel engine. Representative diagrams of the emergency diesel generator fuel oil storage and transfer system, which are associated with EDG 4A and 4B, are presented in Figures 9.15-9 and 9.15-10. The major design features of these systems are:

1. Diesel Oil Storage Tanks - Each EDG (4A and 4B) has a diesel oil storage tank which contains approximately 40,400 gallons of diesel oil; 34,700 gallons of diesel oil is sufficient to operate one EDG for at least 7 days per ANSI N195-1976 methodology. These reinforced concrete tanks are steel lined and are designed, tested, and inspected in accordance with the requirements of ASME B&PV Code, Section VIII, 1986 Edition and Addenda (ASME Section VIII) and meet Seismic Category I requirements. Materials or coating containing aluminum and/or zinc are not used for the construction or coating of any tank surface that may be in contact with the diesel oil. C22

   The diesel oil transfer pump suction connection of each tank is located six inches above the bottom of the tank to minimize the possibility of drawing any sediment and/or water into the pump's suction. The diesel oil transfer pump rooms are sized to contain the contents of a diesel oil storage tank.

Overflow lines have been provided for each of the diesel oil storage tanks. These lines are piped to an equipment drain which goes to a sump located inside the associated diesel oil transfer pump room. Also, each diesel oil storage tank has a vent line and flame arrestor connection located at the top of the tank. This vent line and flame arrestor is piped to the outside of the building and is designed to prevent entrance of water during adverse weather conditions. The exposed line and flame arrestor is not designed to withstand a tornado missile since the almost complete crimping, which would be required as a result of the missile to result in a vacuum being formed in the diesel oil storage tank, is not considered credible.

The fill lines for the diesel oil storage tanks enter near the top of the tanks. Therefore, if one of the lines were impacted by a tornado missile, there would be no significant loss in fuel inventory of the tank. Additionally, in the event that a fill line was impacted by a tornado missile, manual alignment of systems can provide alternate flow paths to fill an affected diesel oil storage tank.

Sample connections for these tanks are provided in accordance with ASTM-D270-1975, Petroleum and Petroleum Products Sampling.

2. <u>Diesel Oil Transfer Pumps</u> - Each EDG (4A and 4B) has an associated diesel oil transfer pump. Diesel oil transfer system piping can be manually aligned such that the two pumps can take suction from either or both of the diesel oil storage tanks associated with EDGs 4A and 4B and discharge into either or both of the diesel oil day tanks associated with these EDGs. In addition, the discharge lines from these pumps and the pumps associated with EDGs 3A and 3B are interconnected to provide additional diesel oil supply alignment flexibility. Each of these two pumps has sufficient capacity to supply diesel oil to both EDGs 4A and 4B at their continuous load rating.

   To prevent any suspended sediment from being transferred to the diesel oil day tanks, each of these diesel oil transfer pumps has a duplex strainer located in its suction path.

These pumps are designed in accordance with the requirements of the ASME B&PV Code, Section III, 1983 Edition, Summer 1984 Addenda, (ASME Section III) for Class 3 components and meet Seismic Category I requirements. Inservice testing capability for these pumps is provided in accordance with ASME Section XI.

The motor for each diesel oil transfer pump is designated Class 1E and is powered from its associated train of the emergency power system. The pumps start and stop automatically on low and high level signals, respectively, from their associated diesel oil day tank. Also, these pumps can be started and stopped manually, if required.

3. Diesel Oil Day Tanks - Each EDG has its own 650 gallon diesel oil day tank. These tanks are designed in accordance with the requirements of the ASME Section III for Class 3 components and meet Seismic Category I requirements. Materials or coating containing aluminum and/or zinc are not used for the construction or coating of any tank surface that may be in contact with the diesel oil.

4. Piping and Valves - Piping which performs a safety function external to the engine skid is designed in accordance with the requirements of the ASME Section III for Class 3 components and meets Seismic Category I requirements. Diesel oil transfer piping has been designed for a maximum of 350°F and 125 psig, which exceeds the system operating requirements. The safety related engine skid mounted piping, as a minimum, is designed and analyzed to meet the stresses specified by ANSI/ASME B31.1, Power Piping, 1986 edition (ANSI B31.1, Power Piping).

5. Diesel Oil – The diesel oil is No. 2 fuel oil complying with the tests, limits, and applicable ASTM Standards specified in the diesel fuel oil testing program described in the Technical Specifications. The oil is chemically treated with a biocide to prevent the deleterious effects of biological activity upon diesel oil quality and a stabilizing compound to enhance diesel oil long term storage.

6. Truck Fill Connection - Additional diesel oil can be delivered to the plant site by truck to replenish the fuel supply system during normal operation or following an accident. These two systems have a common truck fill connection which goes to a valve header. Depending on the valve line-up in each diesel oil transfer pump room, oil may be transferred to either diesel oil storage tank. Diesel oil from the truck fill connection passes through a one micron filter prior to entering the diesel oil storage tanks. In addition, the truck fill connection for the system associated with EDGs 3A and 3B may be used to fill these systems' diesel oil storage tanks. The truck fill connection performs no safety function.

Normally, diesel oil is automatically supplied to the diesel oil day tank for each EDG from its associated diesel oil storage tank by its diesel oil transfer pump through a solenoid valve. Diesel oil is then drawn from the diesel oil day tank, as required, by the fuel pumps.

The diesel oil from the fuel pump (fuel priming pump or engine driven fuel pump) is directed through a duplex filter and into the engine fuel header. Here the fuel oil goes into each injector inlet filter and on into the injector. Any excess fuel is then returned to the diesel oil day tank.

System pressure is maintained below 65 psig by safety relief valves located downstream of the diesel oil transfer pumps and the DC fuel priming pumps.

9.15.1.3 SAFETY EVALUATION

9.15.1.3.1 EDGs 3A AND 3B

The design of the EDG fuel oil system associated with EDGs 3A and 3B ensures that diesel oil storage capacity for at least seven days is provided and that diesel oil can be supplied to at least one EDG, assuming a single failure in the system coincident with a loss of offsite power. Sufficient time exists for providing an alternative air source for opening the day tank fill isolation valves should instrument air fail before the day tank is emptied. In addition, the discharge lines of the Unit 3 diesel oil transfer pumps and the diesel oil transfer pumps associated with EDGs 4A and 4B are interconnected which provides the capability to transfer diesel oil from the Unit 4 diesel oil storage tanks to either the Unit 3 diesel oil storage tank or to the Unit 3 diesel oil day tanks. Also, the diesel oil storage tank and each of the diesel oil day tanks have an alternate fill connection which is suitable for a tie-in from a mobile tank unit. These alternate fill lines provide a fill path for the diesel oil day tanks should the normal supply via the diesel oil transfer pumps become unavailable. Refer to Appendix 5A concerning the seismic and wind loading design of structures, Appendix 5E concerning missile protection criteria, Appendix 5F concerning internal flooding, and Appendix 5G concerning external flooding for this system.

9.15.1.3.2 EDGs 4A AND 4B

There is complete redundancy of the components in the diesel oil storage and transfer systems associated with EDGs 4A and 4B, with the exception of a common fill station. These systems, in conjunction, ensure that diesel oil storage capacity for at least seven days for one EDG is provided and can be supplied to at least one EDG, assuming a single failure in one of the systems coincident with a loss of offsite power.

If continuous operation of a diesel generator is required for an extended period of time, additional fuel oil can be delivered to the plant site by truck and the diesel oil storage tank(s) can be filled to enable each EDG to supply uninterrupted power. Design provisions are included to minimize the possibility of degrading EDG performance due to the potential resuspension of sediment during the refilling process. The tanks are sufficiently sized to provide ample time to be available to refill a diesel oil storage before it reaches a very low level. The diesel oil transfer pumps' suction lines are located six inches from the bottom of the diesel oil storage tanks and have low flow velocity to minimize any suction turbulence. Duplex strainers are also located downstream of the tanks to remove any sediment prior to the diesel oil entering the EDG. Additionally, the volume of diesel oil stored in the diesel oil day tank would provide time for the sediment to settle prior to pumping oil from the diesel oil storage tank to the diesel oil day tank. Also, the diesel oil storage tank not currently being refilled could be used to supply the required diesel oil to the EDG, allowing additional time for sediment to settle in the diesel oil storage tank being refilled.

The Safety Related portions of these systems are located inside Seismic Category I structures (the design of these structures is discussed in

Section 5.3.4) or otherwise protected from the effects of natural phenomena and external missiles. In addition, since each of these EDGs and their associated diesel fuel systems are independent and physically separated from the other by a concrete wall, an internally generated missile will not result in the failure of more than one EDG or its associated diesel fuel system. Also, the systems are not affected by the effects of high-energy and moderate-energy pipe breaks, since there is no high-energy or moderate-energy piping in the Unit 4 EDG Building.

Any single failure in the diesel oil storage and transfer systems is bounded by the loss of an EDG. A failure modes and effects analysis associated with the loss of an EDG has been performed and the results are discussed in Section 8.3.

9.15.1.4 TESTS AND INSPECTIONS

During preoperational testing of the systems associated with EDGs 4A and 4B, the diesel oil transfer systems are checked for proper operation. This testing is also required periodically by the Technical Specifications.

At least once per ten years each diesel oil storage tank will be drained, accumulated sediment removed, and the tank cleaned. In addition, the storage and transfer systems associated with EDGs 4A and 4B will have a pressure test in accordance with ASME Section XI performed on those portions designed to ASME Section III, subsection ND.

The quality of oil is tested and verified prior to being put into the diesel oil storage tanks, and periodically thereafter, per plant procedure in accordance with the appropriate ASTM Standards specified in the diesel fuel oil testing program as described in the Technical Specifications.

9.15.1.5 INSTRUMENTATION APPLICATION

9.15.1.5.1 EDGs 3A AND 3B

The following instrumentation and controls are available for the system associated with EDGs 3A and 3B:

1. Local diesel oil level indication is provided for the diesel oil storage tank.

2. Diesel oil level switches on the diesel oil day tanks provide alarms in the Main Control Room. Additional level switches are also used to start and stop the diesel oil transfer pumps and for opening and closing the solenoid valve at the inlet of each diesel oil day tank.

3. A multi-purpose level switch on each skid tank actuates the solenoid valve in the line between the diesel oil day tank and the skid tank allowing the skid tank to be filled by gravity from the diesel oil day tank. This switch also provides local alarms on low and high skid tank level and an interlock to prevent EDG start and extinguish the EDG ready to start lamp in the Main Control Room if the skid tank level is below acceptable levels.

4. Local diesel oil level indication and alarms are provided for the each skid tank.

5. Local manual control is available for the diesel oil transfer pumps.

9.15.1.5.2 EDGs 4A AND 4B

The following instrumentation and controls are available for the systems associated with EDGs 4A and 4B:

1. Local diesel oil level and temperature indication is provided for each diesel oil storage tank. Diesel oil storage tank oil level switches provide annunciation via an EDG trouble alarm locally and in the Main Control for diesel oil low and high level.

2. Diesel oil level switches for low, low-low and high levels in the diesel oil day tanks provide annunciation via an EDG trouble alarm locally and in the Main Control Room. Additional level switches are also used to start and stop the diesel oil transfer pumps and for opening and closing the solenoid valve at the inlet of each diesel oil day tank. Local diesel oil level indication is provided for each diesel oil day tank

3. Oil pressure indication is provided at the diesel oil transfer pumps discharge.

4. Pressure switches for low diesel oil pressure provide annunciation via an EDG trouble alarm locally and in the Main Control Room

5. Local manual control is available for the diesel oil transfer pumps.

6. Annunciation is provided via an EDG trouble alarm locally and in the Main Control Room for the diesel oil transfer pump in a tripped/off condition.

7. Annunciation is provided via an EDG trouble alarm locally and in the Main Control Room for high differential pressure across the fuel oil filter or strainer.

8. Annunciation is provided via an EDG trouble alarm locally and in the Main Control Room for of fuel priming pump in a no power, overload, or timed out condition.

9.15.2 EMERGENCY DIESEL GENERATOR COOLING WATER SYSTEM

9.15.2.1 DESIGN BASES

Each EDG has a dedicated cooling water system.

9.15.2.1.1 EDGs 3A AND 3B

Each of the cooling water systems associated with EDGs 3A and 3B are designed to:

1. Cool its associated EDG to permit proper operation under all EDG loading conditions.

2. Function independently from other cooling water systems to assure that no single failure can affect the cooling of more than one EDG.

3. Automatically maintain the diesel engine and lubricating oil's temperature, by means of a heater, in a prewarmed condition to minimize diesel engine wear.

4. Include provisions to bypass protective interlocks of the systems during emergency conditions.

5. Perform its function under the same environmental conditions as the EDG which it serves.

6. Meet the requirements for Class I systems/components in accordance with Appendix 5A.

7. Withstand the maximum flood levels in accordance with Appendix 5G, winds in accordance with Appendix 5A, and missiles in accordance with Appendix 5E without loss of function.

9.15.2.1.2 EDGs 4A AND 4B

Each of the cooling water systems associated with EDGs 4A and 4B are designed to:

1. Cool its associated EDG to permit proper operation under all EDG loading conditions for seven days without additional cooling water being added.

2. Function independently from other cooling water systems to assure that no single failure can affect the cooling of more than one EDG.

3. Automatically maintain the diesel engine and lubricating oil's temperature by means of a heater in a prewarmed condition to minimize diesel engine wear.

4. Include provisions to bypass protective interlocks of the systems during emergency conditions.

5. Perform its function under the same environmental conditions as the EDG which it serves.

6. Meet Seismic Category I requirements.

7. Withstand the maximum flood levels and tornado winds and missiles without loss of function by locating critical component inside the Unit 4 EDG Building or by protecting them with elevation and missile barriers.

8. Prevent the failure of nonseismic structures or components from affecting the safety related functions of the system.

9.15.2.2 SYSTEM DESCRIPTION

9.15.2.2.1 EDGs 3A AND 3B

Each EDG (3A and 3B) has a self-contained cooling system, which consists of a forced circulation cooling water loop, to cool the engine directly, rejecting heat through an air cooled radiator. Refer to Figures 9.15-5 and 6 for an operating diagram representative of these EDG 3A and 3B systems. The major design features of each of these systems are:

1. <u>Pumps</u> - Each cooling loop contains two gear driven centrifugal pumps to circulate water through the closed loop.

2. <u>Immersion Heaters</u> - Each cooling loop contains an electrical immersion heater which circulates hot water, by convection, through the oil cooler (described in Section 9.15.4) to provide standby heating of the engine and lubricating oil when the EDG is not operating to reduce diesel engine wear.

3. <u>Expansion Tanks</u> - Each cooling loop contains an expansion tank to allow for the expansion and contraction of the water in the cooling loop due to changes in the water's temperature.

4. <u>Cooling Fans</u> - Each cooling loop contains two cooling fans, driven by belts from their associated EDG, which circulate air over the radiators.

5. Radiators - Each cooling loop contains an air cooled radiator.

When a diesel generator is not operating, the engine block and lubricating oil are maintained in a warmed condition to provide for reliable starting and reduced wear. This is accomplished by heating the cooling water with the electrical immersion heater and the water circulating through the system due to convection. A temperature regulator valve allows the water to bypass the radiator when operating in this mode.

When the EDG is operating, the cooling water system removes heat from the intake air in the turbocharger after cooler, engine water jackets, and the lube oil system, and transfers the heat to the radiator where two fans remove the heat by forced air to the atmosphere.

9.15.2.2.2 EDGs 4A AND 4B

Each EDG (4A and 4B) has a self-contained cooling system which consists of a forced circulation cooling water loop, to cool the engine directly, rejecting heat through an air cooled radiator. A representative diagram of diesel generator cooling water system as associated with either EDG 4A or 4B is presented by Figures 9.15-11 and 12. The major design features of each of these systems are:

1. Pumps - Each cooling loop contains two centrifugal pumps, which are driven by a coupling to their associated EDGs, to circulate water through the closed loop. These pumps meet Seismic Category I requirements.

2. Immersion Heaters - Each cooling loop contains an electrical immersion heater which recirculates hot water, by convection, through the oil cooler (described in Section 9.15.4) to provide standby heating when the EDG is not operating. This heater is designated Class 1E and is supplied power via Class 1E Motor Control Centers. This heater meets Seismic Category I requirements.

3. Expansion Tanks - Each cooling loop contains an expansion tank with a 100 gallon capacity to allow for the expansion and contraction of the

water in the cooling loop due to changes in the water's temperature. This tank contains approximately 60 gallons of water, when the cooling system is in standby conditions, leaving room in the expansion tank for the approximately 21 gallons of water expansion during system heatup to operating conditions. The 60 gallons of water in the expansion tank assures the cooling water pumps' net positive suction head, while allowing for expected system leakage for seven days without water being added to the system. This tank is located at the highest point of the cooling system to allow for venting of the system. This tank also provides the capability to add water to the cooling loop. These expansion tanks are designed in accordance with the requirements of the ASME Section VIII, Division 1, 1986 Edition and Addenda, and meet Seismic Category I requirements.

4. Piping - Piping which performs a safety function external to the EDG skid is designed in accordance with the requirements of ASME Section III for Class 3 components and meet Seismic Category I requirements. Piping internal to the EDG skid is designed and analyzed to meet the stresses specified by ANSI B31.1 and meets Seismic Category I requirements. Cooling water piping has been designed for a maximum of 350°F and 125 psig, which exceeds the system operating requirements.

5. Cooling Fans - Each cooling loop contains three electric, direct coupled, motor driven cooling fans which circulate air over the radiators. Two fans provide sufficient air circulation with ambient air conditions up to 90°F for diesel engine loading up to 100% of rated continuous load or up to 98.5% loading with 95°F ambient air condition. These fans are designed to meet Seismic Category I requirements and their motors are designated Class 1E and are provided power via Class 1E Motor Control Centers.

6. Radiators - Each cooling loop contains an air convection cooled radiator. These radiators are designed in accordance with the requirements of the ASME Section VIII, Division 1 and meet the Seismic Category I requirements.

When a diesel generator is not operating, the engine block and lubricating oil are maintained in a warmed condition to provide for reliable starting and reduced wear. This is accomplished by heating the cooling water with the electrical immersion heater and the water circulating through the system due to convection. A temperature regulator valve allows the water to bypass the radiator when operating in this mode.

When a diesel generator is operating, cooled water is drawn through the lube oil cooler by the two cooling water pumps. One pump supplies cooling water to the right side inlet manifold, ten cylinders, and one side of the discharge manifold and the other pump supplies cooling water to the other side of the manifolds and to the other ten cylinders. A temperature regulator valve then allows the water to pass through the radiator, to reject the necessary heat, as required to maintain the correct temperature.

9.15.2.3 SAFETY EVALUATION

9.15.2.3.1 EDGs 3A AND 3B

Each EDG 3A and 3B has its own independent cooling water system. This independence ensures that only one EDG will be affected by a single failure in one of these cooling water systems. Refer to Appendix 5A concerning the seismic design of structures, Appendix 5E concerning missile protection criteria, Appendix 5F concerning internal flooding, and Appendix 5G concerning external flooding for these systems.

9.15.2.3.2 EDGs 4A AND 4B

Each EDG 4A and 4B has its own independent cooling water system. This independence ensures that only one EDG will be affected by a single failure in one of these cooling water systems.

System components and piping have sufficient physical separation or shielding to protect the system from externally generated missiles. In addition, since each of these EDGs and their associated cooling water systems are independent and physically separated from the other by a concrete wall, an internally generated missile will not result in the failure of more than one EDG or its associated cooling water system. Also, the systems are not affected by the effects of high-energy and moderate-energy pipe breaks, since there is no high-energy or moderate-energy piping in the Unit 4 EDG Building.

These systems are located inside Seismic Category I structures (the design of these structures is discussed in Section 5.3.4) and, therefore, protected from the effects of natural phenomena and external missiles. The radiators are located in the outside wall of the Unit 4 EDG Building and are designed to withstand the effects of the design basis tornado. The radiators are protected from tornado generated missiles by a missile shield.

In the event of a cooling system piping connection failure, the resultant loss of cooling system fluid would eventually force the affected EDG out of service. However, the possibility for this event is very low for the following reasons:

1. The cooling system operates at relatively low pressure and temperature (7 psig and 215°F maximum) as compared with the design pressure and temperature (125 psig and 350°F) of the system.

2. The entire cooling system is analyzed for all normal and postulated loads including dead weight, thermal and seismic conditions, and is designed accordingly.

3. The cooling system design is consistent with the manufacturer's standard practice which has historically proven to be very reliable with respect to cooling system piping failures.

Although highly unlikely, should such a failure occur, available alarms and indication would make plant operators aware of the possible degradation in the cooling system's performance so that appropriate action could be taken. In addition, during non-emergency use, automatic EDG shutdown would occur when the cooling water temperatures reached 215°F.

Any single failure in one of these cooling water systems is bounded by the loss of an EDG. A failure modes and effects analysis associated with the loss of an EDG has been performed and the results are discussed in Section 8.3.

9.15.2.4 TESTS AND INSPECTIONS

System components of the systems associated with EDGs 4A and 4B were inspected and tested by the manufacturer. Following installation and prior to declaring the systems operable these cooling water systems were inspected, tested, and operated. Testing was performed to verify system operability in accordance with plant Technical Specification requirements, manufacturer's recommendations, and applicable codes and standards. Additional information on the testing and inspection of the EDGs is described in Section 8.2.

9.15.2.5 INSTRUMENTATION APPLICATION

9.15.2.5.1 EDGs 3A AND 3B

The necessary instrumentation and controls are provided with each of the cooling water systems associated with EDGs 3A and 3B to maintain the EDG engine jacket at the proper temperature in all modes of EDG operation. The following instrumentation and controls are available for the systems associated with EDGs 3A and 3B:

1. Abnormal cooling water temperature is monitored and alarmed at the local control panel and at a common EDG trouble alarm in the Main Control Room.

2. The EDG is interlocked to shutdown, during non-emergency operation, if the cooling water pressure or temperature goes beyond its shutdown limit.

9.15.2.5.2 EDGs 4A AND 4B

The necessary instrumentation and controls are provided for each of the cooling water systems associated with EDGs 4A and 4B to maintain the EDG engine jacket at the proper temperature in all modes of EDG operation. The following instrumentation and controls are available for the systems associated with EDGs 4A and 4B:

1. Level switches for low expansion tank water level provide annunciation via an EDG trouble alarm locally and in the Main Control Room.

2. Temperature switches for high-high and, high cooling water temperature provide annunciation via an EDG trouble alarm locally and in the Main Control Room.

3. Pressure switches for low-low and low cooling water pressure provide annunciation via an EDG trouble alarm locally and in the Main Control Room.

4. The EDG is interlocked to shutdown, during non-emergency operation, on low-low cooling water pressure or high-high cooling water temperature.

5. Annunciation is provided via the EDG trouble alarm locally and in the Main Control Room for a cooling fan in a tripped or off condition.

6. Annunciation is provided via the EDG trouble alarm locally and in the Main Control Room for the immersion heater in a tripped or off condition or not under automatic control.

7. Temperature switches for cooling water temperature provide automatic immersion heater control.

8. Temperature indicators provide local cooling water temperature indication.

9.15.3 EMERGENCY DIESEL GENERATOR AIR START SYSTEM

9.15.3.1 DESIGN BASES

Each EDG has its own associated air start system.

9.15.3.1.1 EDGs 3A AND 3B

The air start systems associated with EDGs 3A and 3B are designed to:

1. Store and provide sufficient starting air to ensure starting of its associated EDG.

2. Function with sufficient independence from other diesel generator air start systems to assure that no single failure can affect the provision of sufficient starting air to more than one EDG.

3. Perform its function under the same environmental conditions as the EDG which it serves.

4. Meet the requirements for Class I systems/components in accordance with Appendix 5A.

5. Withstand the maximum flood levels in accordance with Appendix 5G, wind loadings in accordance with Appendix 5A, and missiles in accordance with Appendix 5E without loss of function.

9.15.3.1.2 EDGs 4A AND 4B

The air start systems associated with EDGs 4A and 4B are designed to:

1. Store and provide sufficient starting air to ensure starting of its associated EDG, assuming a single failure.

2. Function with sufficient independence from other diesel generator air start systems to assure that no single failure can affect the provision of sufficient starting air to more than one EDG.

3. Contain provisions for the periodic blowdown of accumulated moisture and foreign material in the air receiver(s).

4. Perform its function under the same environmental conditions as the EDG which it serves.

5. Meet Seismic Category I requirements.

6. Withstand the maximum flood levels and tornado winds and missiles without loss of function by locating critical components inside the Unit 4 EDG Building.

7. Prevent the failure of nonseismic structures or components from affecting the safety related functions of the system.

9.15.3.2 SYSTEM DESCRIPTION

9.15.3.2.1 EDGs 3A AND 3B

Each EDG (3A and 3B) has an independent air start system capable of starting its associated EDG. Refer to Figure 9.15-2 for diagrams of EDG 3A and 3B starting air systems. The major design features of these systems are:

1. Air Motors - Four air motors, two independent 100% capacity sets consisting of an upper and lower air motor, are provided for cranking each EDG.

2. Air Receivers - Each EDG's air start system contains two sets of two air receivers. Air receiver tanks A & B provide an air supply to one set of air start motors and air receiver tanks C & D provide an air supply to the second set of air start motors. The air receivers are sized to ensure that the four air receivers have sufficient capacity, at the design pressure setpoint, for four unsuccessful (based on nominal 2 second duration) attempts and one successful EDG starting attempt without the need for recharging with air. In order to deliver sufficient compressed air to both sets of air start motors for a minimum of five starting attempts, all four air receiver tanks A, B, C and D are valved-in to the air start system.

3. Electric Motor Driven Air Compressors - Each EDG's air start system contains an electric motor driven air compressor to provide charging air to the air receivers. The discharge of the air compressors associated with EDGs 3A and 3B may be cross connected to allow either air compressor to supply either EDG's air receivers.

4. Air Quality Package Units - Each EDG's air start system contains an air quality package unit, which is located between the EDG air compressor and air receiver tanks. The air quality unit consists of a membrane air dryer downstream of the air compressor, an air-cooled aftercooler upstream of the air dryer, a moisture separator and pre-filters between the after cooler and air dryer C26

complete with associated piping, valves and instrumentation. The air quality package units have bypass capability which will not interrupt service of the starting air compressor to maintain the air storage tanks at their design pressure.

When the air start system associated with EDGs 3A and 3B is in normal standby, there is approximately 200 psig of air pressure in the piping up to the air start solenoid valve. The electrical start signal will open the air start solenoid valve and admit starting air to engage the pinions of the air start motors. When both pinions are engaged, the control valve is then actuated which admits the starting air to the air start motors. When the starting air enters the air start motor assemblies, it first pressurizes the pinion which engages the air start motor to the engine flywheel; then the air turns the air start motor internals and the engine flywheel together.

9.15.3.2.2 EDGs 4A AND 4B

Each EDG (4A and 4B) has an independent air start system capable of starting its associated EDG. Representative diagram of the EDG air start system as associated with either EDG 4A or 4B is shown in Figures 9.15-7 and 9.15-8. The major design features of these systems are:

1. <u>Air Receivers</u> - Each EDG's air start system contains two sets of two air receivers. The air receivers are sized to ensure that the system has sufficient capacity, with a pressure of 185 psig in all four air receivers and four air start motors available or 195 psig with only two air receivers and two air start motors available, for four unsuccessful (up to five seconds each) attempts and one successful EDG starting attempt without the need for recharging. These values are for the receivers sizing criteria, and are not the minimum pressure required to start the EDG. If either set of air receivers is out of service, the other set of air receivers with either set of air start motors or both sets of air start motors is capable of starting the associated EDG. Redundant air start subsystems are not required for operability of the EDG. C24 C24

   The air receivers are designed in accordance with the requirements of ASME Section III for Class 3 components and meet Seismic Category I requirements. Each air receiver is protected from over pressure by ASME Section III, Class 3 relief valves.

2. Electric Motor Driven Air Compressors - Each EDG's air start system contains an electric motor driven air compressor to provide charging air to the air receivers. Each air compressor is capable of recharging a set of two air receivers in no more than 30 minutes following five successive EDG starting attempts. A filter is provided at the compressor suction to prevent dust and other foreign matter from entering the system. While these compressors are provided power via Class 1E Motor Control Centers, they perform no safety function, since they are not required to function to start the EDG. The receivers contain sufficient air reserves to successfully start the EDG.

3. Diesel Engine Driven Air Compressors - Each EDG's air start system contains a diesel engine driven air compressor to provide charging air to the air receivers. This diesel engine driven air compressor provides
a backup to and can be isolated from each system's electric motor driven air compressor. Each air compressor is capable of recharging a set of two air receivers in no more than 30 minutes following five successive EDG starting attempts. A filter is provided at the compressor suction to prevent dust and other foreign matter from entering the system. These air compressors, like the motor driven air compressors, perform no safety function.

4. Air Motors - Four air motors, two 100% sets consisting of an upper and lower air motor each, are provided for cranking each EDG. Each set of two motors is supplied with starting air from a separate set of air receivers. Cross connections in the piping allows either set of air receivers to supply either set of air start motors. The air motors are designed and analyzed to meet the stresses specified by ANSI B31.1.

   For a normal EDG start all four air motors are used to roll over the engine. If two air start motors are inoperable, the other two have sufficient torque to start the EDG in the required time interval.

C23

5. <u>Piping and Valves</u> - Piping which performs a safety function and is external to the EDG skid is designed in accordance with the requirements of ASME Section III for Class 3 components and meet Seismic Category I requirements. Piping internal to the EDG skid is designed and analyzed to meet the stresses specified by ANSI B31.1. Air start system piping has been designed for a maximum temperature of 350°F and 300 psig, which meets or exceeds the system operating requirements.

6. <u>Air Dryers</u> - Each EDG's air start system contains a membrane type air dryer installed between the air compressors and the air receivers. These air dryers ensure starting air is dried to a dew point of less than 40°F. The air dryer performs no safety function. C26

When the air start systems associated with EDGs 4A and 4B are in normal standby, there is approximately 200 psig of air pressure in the piping up to the two redundant air start solenoid valves. The electrical start signal will open both of the air start solenoid valves and admit starting air to engage the pinions of the air start motors. When both pinions are engaged, the control valve is then actuated which admits the starting air to the air start motors. The starting air enters the air start motor assemblies, first pressurizing the pinion and engaging the air start motor to the engine flywheel; then the air turns the air start motor internals and the engine flywheel together.

When the air pressure downstream of the air start motors builds to a preset value due to the air start motors rotational velocity, a pneumatic pressure sensing line sends air to the governor servo-booster pump and simultaneously to the top side of the air start solenoid valve. This allows the governor to obtain faster control of the EDG and shuts the air start solenoid valve when the EDG has reached the necessary starting speed so the fuel racks can maintain and accelerate the engine to the desired revolutions per minute (rpm).

9.15.3.3 SAFETY EVALUATION

9.15.3.3.1 EDGs 3A AND 3B

The air start systems for EDGs 3A and 3B are independent to ensure that a single failure in one EDG's air start system will not result in the unavailability of more than one EDG. Refer to Appendix 5A concerning the seismic design of structures, Appendix 5E concerning missile protection criteria, Appendix 5F concerning internal flooding, and Appendix 5G concerning external flooding for these systems.

9.15.3.3.2 EDGs 4A AND 4B

Each EDG 4A and 4B has its own independent air start system; therefore, a failure in one EDG's air start system will not result in the unavailability of more than one EDG. In addition, each EDG's air start system has sufficient redundancy to ensure that sufficient starting air will be provided to each EDG following a single failure in its associated air start system.

The systems are contained in structures designed to meet Seismic Category I requirements (the design of these structures is discussed in Section 5.3.4) and, therefore, system components and piping have sufficient physical separation or barriers to protect the system from externally generated missiles. In addition, since each EDG and its associated air start system is independent and physically separated from the other via a concrete wall, an internally generated missile will not result in the failure or more than one air start system. Also, the systems are not affected by the effects of high-energy and moderate-energy pipe breaks, since there is no high-energy or moderate-energy piping in the Unit 4 EDG Building.

Functional capability of the air start systems are not adversely affected due to a maximum probable flood because of their location in the Unit 4 EDG Building.

Any single failure in the air start systems is bounded by the loss of an EDG. A failure modes and effects analysis associated with the loss of an EDG has been performed and the results are discussed in Section 8.3.

9.15.3.4 TESTS AND INSPECTIONS

System components associated with EDGs 4A and 4B were inspected and tested by the manufacturer. Following installation and prior to declaring the systems operable, the air start systems were inspected, tested, and operated.

Testing was performed to verify system operability, as part of the testing of their respective EDGs, in accordance with plant Technical Specification requirements, manufacturer's recommendations, and applicable codes and standards. This testing included but was not limited to testing that each air start system, consisting of four air receivers, at a minimum pressure of 185 psig and four air start motors (or two air receivers at a minimum pressure of 195 psig and two air start motors), was capable of cranking its associated EDG, at standby conditions, a minimum of five times (four unsuccessful start-ing attempts followed by a successful start) without recharging the air receivers. These values are for the receivers sizing criteria, and are not the minimum pressure required to start the EDG. Since redundant air start subsystems are not required for operability of the EDG, the minimum air receiver pressure value for operability with full qualification is set to 185 psig corresponding to the minimum air pressure required for four receivers and four air start motors. Additional information on the testing and inspection of the EDGs is described in Section 8.2. C24

9.15.3.5 INSTRUMENTATION APPLICATION

9.15.3.5.1 EDGs 3A AND 3B

The following instrumentation and controls are available for the systems associated with EDGs 3A and 3B:

1. Automatic and manual control of each EDG's air compressor.

2. Pressure switches provide input to the automatic control of the air compressors and alarms on low starting air pressure in the air receivers both locally and at an EDG trouble alarm in the Main Control Room.

9.15.3.5.2 EDGs 4A AND 4B

The following instrumentation and controls are available for the systems associated with EDGs 4A and 4B:

1. Automatic controls are provided to start and stop the electric motor driven compressors and manual controls are provided for the diesel driven compressors.

2. Pressure switches provide input to the automatic control of the air compressors and on low or high starting air pressure in the air receivers provide annunciation via an EDG trouble alarm locally and in the Main Control Room.

3. Moisture detection devices monitor the air receivers for moisture buildup (should moisture accumulate in the air receivers) and provide annunciation via an EDG trouble alarm locally and in the Main Control Room.

4. Annunciation is provided via the EDG trouble alarm locally and in the Main Control Room for the air barring device in a engaged condition. The air barring device is only engaged for EDG maintenance.

5. Annunciation is provided via the EDG trouble alarm locally and in the Main Control Room for the motor driven air compressors in a tripped or off condition.

6. Local annunciation is provided if the start pinion of an air start motor failed to engage or recycle control during an EDG start attempt.

The air start system has no interlocks which will shutdown its associated EDG.

9.15.4 EMERGENCY DIESEL GENERATOR LUBRICATION SYSTEM

9.15.4.1 DESIGN BASES

Each EDG has its own associated lubrication system.

9.15.4.1.1 EDGs 3A AND 3B

The systems associated with EDGs 3A and 3B are designed to:

1. Provide each EDG with a dedicated lubricating oil system which includes measures to provide lubrication to the diesel engine during standby conditions and/or normal and emergency operations.

2. Function independently from other EDG lubrication systems to assure that no single failure can affect the operation of more than one EDG.

3. Automatically maintain the lubricating oil's temperature in a prewarmed condition to minimize diesel engine wear.

4. Provide measures for cooling the system and removing the system heat load to permit proper EDG operation under all loading conditions.

5. Perform its function under the same environmental conditions as the EDG which it serves.

6. Meet the requirements for Class I systems/components in accordance with Appendix 5A.

7. Withstand the maximum flood levels in accordance with Appendix 5G, wind loadings in accordance with Appendix 5A, and missiles in accordance with Appendix 5E without loss of function.

9.15.4.1.2 EDGs 4A AND 4B

The systems associated with EDGs 4A and 4B are designed to:

1. Provide each EDG with a dedicated lubricating oil system which includes measures to provide lubrication to the diesel engine wearing parts during standby conditions and/or normal and emergency operations for seven days without additional lubricating oil being added to the system.

2. Function independently from other EDG lubrication systems to assure that no single failure can affect the operation of more than one EDG.

3. Automatically maintain the lubricating oil's temperature in a prewarmed condition to minimize diesel engine wear.

4. Provide measures for cooling the system and removing the system heat load to permit proper EDG operation under all loading conditions.

5. Perform its function under the same environmental conditions as the EDG which it serves.

6. Meet Seismic Category I requirements.

7. Withstand the maximum flood levels and tornado winds and missiles without loss of function by locating critical components inside the Unit 4 EDG Building.

8. Prevent the failure of nonseismic structures or components from affecting the safety related functions of the system.

9.15.4.2 SYSTEM DESCRIPTION

9.15.4.2.1 EDGs 3A AND 3B

A combination of four subsystems provide lubrication for EDGs 3A and 3B. These subsystems are the scavenging oil system, main lube oil system, piston cooling oil system, and soak back oil system; refer to Figures 9.15-5 and 9.15-6. The major design features of these subsystems are described below:

1. Engine Driven Scavenging Pumps - Each EDG's lubrication system has an engine driven scavenging pump. This pump is part of the scavenging oil system and forces the oil through the oil filter and lube oil cooler. This pump is driven by the accessories gear train at the front of its associated EDG.

2. Engine Driven Main Pressure Pumps - Each EDG's lubrication system has an engine driven main pressure pump. This pump is part of the main lube oil system and supplies oil under pressure to most of the moving parts of the engine. This pump is driven by the accessories gear train at the front of its associated EDG. Although this pump is separate from the engine driven piston cooling pump, it is located in the same housing.

3. Engine Driven Piston Cooling Pumps - Each EDG's lubrication system has an engine driven piston cooling pump. This pump is part of the piston cooling oil system and supplies oil to the two piston cooling oil manifolds. This pump is driven by the accessories gear train at the front of its associated EDG.

4. Soak Back Oil Pumps - Each EDG's lubrication system has a motor driven soak back oil pump. This pump is part of the soak back oil system and

runs when the engine is shutdown. The pump is powered by a Class 1E Motor Control Center.

5. <u>Backup Soak Back Pump</u> - Each EDG's lubrication system has a backup soak back pump. This pump is part of the soak back oil system and is redundant to the soak back pump. The pump motor is powered by a Class 1E Motor Control Center. A manual transfer switch allows power to be transferred between the primary and backup soak back pumps such that only one pump will be capable of operating at a time.

6. <u>Lube Oil Coolers</u> - Each EDG's lubrication system has a lube oil cooler. When the EDG is operating lube oil is circulated through this cooler by the scavenging oil system and heat is removed from it by the cooling water system. When the EDG is in a standby condition, lube oil is circulated through lube oil cooler by the soak back oil system and the oil is heated by the cooling water system's immersion heater and natural circulation.

<u>Scavenging Oil System</u>

The scavenging oil system pump draws oil through the scavenging oil strainer from the oil sump. The pump then forces the oil through the oil filter and oil cooler. Oil then returns to the strainer housing to supply the main lube oil pump and piston cooling oil pump with cooled filtered oil. Excess oil spills over a dam in the strainer housing and returns to the oil sump.

<u>Main Lube Oil System</u>

The main lubrication system supplies oil, under pressure, to most of the moving parts of the engine. The main lube oil pump takes oil from its strainer in the strainer housing. The majority of the moving parts receive their oil from this subsystem.

<u>Piston Cooling Oil System</u>

The piston cooling oil system receives oil from its strainer and delivers oil to the two piston cooling oil manifolds extending the length of the engine, one on each side. A piston cooling oil pipe at each cylinder directs a stream of oil to cool the underside of the piston crown and the ring belt. Some of this oil enters the oil grooves in the piston pin bearing while the remainder drains out through holes in the carrier skirt to the sump.

Soak Back Oil System

The soak back system pump runs when the engine is shutdown to supply lube oil to the turbocharger bearing for proper lubrication during emergency starts and coasting down of the engine, and also to maintain flow through the main lube oil filter and cooler to keep the engine warm during standby conditions. The oil is warmed in the lube oil cooler due to the immersion heater contained in the diesel generator cooling water system as described in Section 9.15.2. The warmed oil is circulated though the engine to facilitate "cold" starts and reduce engine wear. To prevent possible overheating of the turbocharger, oil is automatically supplied to the turbocharger after stopping the engine. The system design ensures continuous oil flow through each of the supply headers.

9.15.4.2.2 EDGs 4A AND 4B

A combination of four subsystems provide lubrication for EDGs 4A and 4B. These subsystems are the scavenging oil system, main lube oil system, piston cooling oil system, and soak back oil system. Representative diagrams of the these subsystems associated with EDG 4A and 4B are shown in Figures 9.15-11 and 9.15-12. The major design features of these subsystems are described below:

1. Engine Driven Scavenging Pumps - Each EDG's lubrication system has an engine driven scavenging pump. This pump is part of the scavenging oil system and forces the oil through the oil filter and lube oil cooler. This pump is driven by the accessories gear train at the front of its associated EDG. The pump meets Seismic Category I requirements.

2. Engine Driven Main Pressure Pumps - Each EDG's lubrication system has an engine driven main pressure pump. This pump is part of the main lube oil system and supplies oil under pressure to most of the moving parts of the engine. Although this pump is separate from the engine driven piston cooling pump, it is located in the same housing. This pump is driven by the accessories gear train at the front of its associated EDG. The pump meets Seismic Category I requirements.

3. Engine Driven Piston Cooling Pumps - Each EDG's lubrication system has an engine driven piston cooling pump. This pump is part of the piston cooling oil system and supplies oil to the two piston cooling oil

manifolds. This pump is driven by the accessories gear train at the front of its associated EDG. The pump meets Seismic Category I requirements.

4. <u>Turbo Oil AC Pumps</u> - Each EDG's lubrication system has a turbo oil AC pump. This pump is part of the soak back oil system and supplies lube oil to the turbocharger bearing for proper lubrication during EDG emergency start and coast down. The pump motor is designated Class 1E and is powered by a Class 1E Motor Control Center. The pump meets Seismic Category I requirements.

5. <u>Backup Turbo Oil DC Pumps</u> - Each EDG's lubrication system has a backup turbo oil DC pump. This pump is part of the soak back oil system and is redundant to the turbo oil AC pump. The pump motor is designated Class 1E and is powered by a Class 1E DC bus. The pump meets Seismic Category I requirements.

6. <u>Circulating Oil AC Pumps</u> - Each EDG's lubrication system has a circulating oil AC pump. This pump is part of the soak back oil system and maintains flow through the main lube oil filter and cooler to keep the engine warm during standby conditions. The pump motor is designated Class 1E and is powered by a Class 1E Motor Control Center. The pump meets Seismic Category I requirements.

7. <u>Backup Circulating Oil DC Pumps</u> - Each EDG's lubrication system has a backup circulating oil DC pump. This pump is part of the soak back oil system and is redundant to the circulating oil AC pump. The pump motor is designated Class 1E and is powered by a Class 1E DC bus. The pump meets Seismic Category I requirements.

8. <u>Piping</u> - Piping is designed and analyzed to meet the stresses specified by ANSI B31.1 and meets Seismic Category I requirements.

9. <u>Lube Oil Coolers</u> - Each EDG's lubrication system has a lube oil cooler. When the EDG is operating, lube oil is circulated through this cooler by the scavenging oil system and heat is removed from it by the cooling water system. When the EDG is in a standby condition, lube oil

is circulated through the lube oil cooler by the soak back oil system and the oil is heated by the cooling water system's immersion heater and natural circulation. The lube oil cooler is designed and analyzed to meet the stresses specified by ANSI B31.1 and meets Seismic Category I requirements.

10. Lube Oil Sumps - Each EDG's lubrication system has a lube oil sump. This lube oil sump contains sufficient inventory to assure the lube oil system's pumps have net positive suction head while allowing for expected system leakage for seven days without lube oil being added to the system. The lube oil sump meets Seismic Category I requirements.

Scavenging Oil System

The scavenging oil system pump draws oil through the scavenging oil strainer from the oil sump. The pump then forces the oil through the engine oil filter and oil cooler. Oil then returns to the strainer housing to supply the main lube oil pump and piston cooling oil pump with cooled filtered oil. Excess oil spills over a dam in the strainer housing and returns to the oil sump.

Main Lube Oil System

The main lubrication system supplies oil, under pressure, to most of the moving parts of the engine. The main lube oil pump draws oil through the engine oil strainer from the lube oil sump. Oil from the pump goes into the main oil manifold which is located above the crankshaft, and extends the length of the engine. The majority of the moving parts receive their oil from passages directly off the manifold.

Piston Cooling Oil System

The piston cooling oil system receives oil from the lube oil sump through the engine oil strainer and delivers the oil to the two piston cooling oil manifolds extending the length of the engine, one on each side. A piston cooling oil pipe at each cylinder directs a stream of oil to cool the underside of the piston crown and the ring belt. Some of this oil enters the oil grooves in the piston pin bearing while the remainder drains out through holes in the carrier skirt to the sump.

Soak Back Oil System

The soak back system's pumps run continuously to supply lube oil to the turbocharger bearing for proper lubrication during emergency starts and coasting down of the engine, and also to maintain flow through the main lube oil filter and lube oil cooler to keep the engine warm during standby conditions. The oil is warmed in the lube oil cooler by the immersion heater contained in the diesel generator cooling water system as described in Section 9.15.2. The warmed oil is circulated though the engine to facilitate "cold" starts and reduce engine wear. To prevent possible overheating of the turbocharger, oil is automatically supplied to the turbocharger after stopping the engine. The system design ensures continuous oil flow through each of the supply headers.

9.15.4.3 SAFETY EVALUATION

9.15.4.3.1 EDGs 3A AND 3B

Each EDG 3A and 3B has its own independent lubrication system. This independence ensures that only one EDG will be effected by a single failure in one of these lubrication systems. Refer to Appendix 5A concerning the seismic design of structures, Appendix 5E concerning missile protection criteria, Appendix 5F concerning internal flooding, and Appendix 5G concerning external flooding for these systems.

9.15.4.3.2 EDGs 4A AND 4B

Each EDG 4A and 4B has its own independent lubrication system, therefore, a failure in one EDG's lubrication system will not result in the unavailability of more than one EDG.

The systems are contained in structures designed to meet Seismic Category I requirements (the design of these structures is discussed in Section 5.3.4) and, therefore, system components and piping have sufficient physical separation or barriers to protect the system from externally generated missiles. In addition, since each EDG and its associated lubrication system is independent and physically separated from the other via a concrete wall, an internally generated missile will not result in the failure of more than one lubrication system. Also, the systems are not affected by the effects of high-energy and moderate-energy pipe breaks, since there is no high-energy or moderate-energy piping located in the Unit 4 EDG Building.

Functional capability of the lubrication systems are not adversely affected due to a maximum probable flood because of their location in the Unit 4 EDG Building.

Any single failure in the lubrication systems is bounded by the loss of an EDG. A failure modes and effects analysis associated with the loss of an EDG has been performed and the results are discussed in Section 8.3.

9.15.4.4 TESTS AND INSPECTIONS

System components associated with EDGs 4A and 4B were inspected and tested by the manufacturer. Following installation and prior to declaring the subsystems operable these diesel generator lubrication oil subsystems were inspected, tested, and operated. Testing was performed to verify system operability, as part of the testing of their respective EDG, in accordance with plant Technical Specification requirements, manufacturer's recommendations, and applicable codes and standards. Additional information on the testing and inspection of the EDGs is described in Section 8.2.

9.15.4.5 INSTRUMENTATION APPLICATION

9.15.4.5.1 EDGs 3A AND 3B

The following instrumentation and controls are available for the systems associated with EDGs 3A and 3B:

1. Main oil system pressure switches provide an alarm on low pressure and an engine trip on low-low pressure.

2. Scavenging oil system temperature switch provide an alarm in the Main Control Room on low temperature.

3. Soak back oil system pressure switch extinguish the EDG ready to start lamp in the Main Control Room on low pressure.

9.15.4.5.2 EDGs 4A AND 4B

The following instrumentation and controls are available for the systems associated with EDGs 4A and 4B:

1. Temperature indicators are provided for the lube oil temperature at the engine inlet and outlet.

2. Temperature switches for low, high-high and high lube oil temperature provide annunciation via an EDG trouble alarm locally and in the Main Control Room. Temperature switches for high-high temperature provide an EDG shutdown function during non-emergency operation.

3. Pressure switches for main manifold lube oil low pressure, piston cooling oil low pressure and crankcase high pressure provide annunciation via an EDG trouble alarm locally and in the Main Control Room and provide an EDG shutdown function during nonemergency operation.

4. Pressure switches for lube oil low pressure and level switches for lube oil sump low level provide annunciation via an EDG trouble alarm locally and in the Main Control Room.

5. Differential pressure switches for high differential pressure across the lube oil strainer and filter, and the lube oil cooler provide annunciation via an EDG trouble alarm locally and in the Main Control Room.

6. Pressure switches are provided for control of the backup circulating and turbo oil DC pumps.

7. Annunciation is provided via an EDG trouble alarm locally and in the Main Control Room for the backup circulating oil DC pumps or the backup turbo oil DC pumps in a no power or overload condition or running.

8. Annunciation is provided via an EDG trouble alarm locally and in the Main Control Room for the backup turbo oil DC pump not in the automatic control.

9. The Lube Oil Metal Detector and associated alarms have been abandoned in place by EC 242379 (PC/M 08-011). C26

9.15.5 EMERGENCY DIESEL GENERATOR COMBUSTION AIR INTAKE/EXHAUST SYSTEM

9.15.5.1 DESIGN BASES

Each EDG has its own associated combustion air intake/exhaust system.

9.15.5.1.1 EDGs 3A AND 3B

The systems associated with EDGs 3A and 3B are designed to:

1. Provide combustion air to its associated EDG and exhaust the combustion products to the atmosphere.

2. Function independently from other EDG combustion air intake/exhaust systems to assure that no single failure can affect the operation of more than one EDG.

3. Preclude degradation of the EDG's power output due to exhaust gases and other contaminants entering the air intake. The systems are physically arranged, such that, no degradation of engine function is experienced when an EDG is required to operate continuously at the maximum power output.

4. Perform its function under the same environmental conditions as the EDG which it serves.

5. Meet the requirements for Class I systems/components in accordance with Appendix 5A.

6. Withstand the maximum flood levels in accordance with Appendix 5G, winds in accordance with Appendix 5A.

9.15.5.1.2 EDGs 4A AND 4B

The systems associated with EDGs 4A and 4B are designed to:

1. Provide combustion air to its associated EDG and exhaust the combustion products to the atmosphere.

2. Function independently from other EDG combustion air intake/exhaust systems to assure that no single failure can affect the operation of more than one EDG.

3. Preclude degradation of the EDG's power output due to exhaust gases and other contaminants entering the air intake. The systems are physically arranged, such that, no degradation of engine function is experienced when an EDG is required to operate continuously at the maximum power output.

4. Perform its function under the same environmental conditions as the EDG which it serves.

5. Meet Seismic Category I requirements.

6. Withstand the maximum flood levels and tornado winds and missiles without loss of function by locating critical components inside the Unit 4 EDG Building or by protecting them by elevation and missile barriers.

7. Prevent the failure of nonseismic structures or components from affecting the safety related functions of the system.

9.15.5.2 SYSTEM DESCRIPTION

9.15.5.2.1 EDGs 3A AND 3B

Each EDG (3A and 3B) has an independent combustion air intake and exhaust system. The major design features of these systems are:

1. Air Intake - The air intake for each EDG consists of a short section of duct work and a filter assembly and is located inside the Unit 3 EDG Building. The air intake is provided with multiple turns to prevent air entrained water from entering the building.

2. Silencers - Each EDG has its own silencer.

3. Exhaust Piping - The exhaust piping is designed to be suitable for the combustion gases handled.

9.15.5.2.2 EDGs 4A AND 4B

Each EDG (4A and 4B) has an independent combustion air intake and exhaust system. Representative diagrams of the EDG combustion air intake and exhaust system as associated with the EDG 4A and 4B are shown in Figure 9.15-7 and 9.15-8, respectively. The major design features of these systems are:

1. Air Intake - The air intake for each EDG is located inside the Unit 4 EDG Building. The air flow path from the building air intake is provided with multiple turns to prevent air entrained water from entering the EDG's air intake. The Unit 4 EDG Building is provided with a large weather protected air intake which is located approximately 32 feet above grade. This intake is designed in accordance with AISC Manual of Steel Construction, 8th edition. The building air intake is located on the opposite side of the building from the EDG exhaust which virtually precludes the possibility of combustion products being introduced into the building or the EDG air intake.

2. Silencers - Each EDG has its own silencer. The silencer has a maximum pressure drop of one inch of water across it. The silencers are

designed and analyzed to meet the stresses specified by ANSI B31.1 and to meet Seismic Category I requirements.

3. Exhaust Piping - The exhaust piping is designed and analyzed to meet the stresses specified by ANSI B31.1 and meets Seismic Category I requirements. The piping is made of carbon steel and designed for temperatures up to 800°F and is suitable for the combustion gases handled.

9.15.5.3 SAFETY EVALUATION

9.15.5.3.1 EDGs 3A AND 3B

Each EDG 3A and 3B has its own independent combustion air intake/exhaust system. This independence ensures that only one EDG will be effected by a single failure in one of these systems. Refer to Appendix 5A for information on the seismic design of structures, Appendix 5E concerning missile protection criteria, Appendix 5F concerning internal flooding, and Appendix 5G concerning external flooding for these systems.

9.15.5.3.2 EDGs 4A and 4B

Each EDG 4A and 4B has its own independent combustion air intake/exhaust system, therefore, a single failure in one EDG's combustion air intake/exhaust system will not result in the unavailability of more than one EDG.

The exhaust system for the diesel engine is indoors, except for the discharge which is through an exterior wall and protected by a tornado missile shield. Except for the exhaust discharge, as previously discussed, these systems are located inside a Seismic Category I structure (the design of this structure is discussed in Section 5.3.4). Since the air intake is above maximum water level, these systems are protected from the effects of natural phenomena and external missiles.

In addition, since each of these EDGs and their associated combustion air intake/exhaust systems are independent and physically separated from the other by a concrete wall, an internally generated missile will not result in the failure of more than one EDG. Also, the systems are protected from the effects of high-energy and moderate-energy pipe breaks, since there is no high-energy or moderate-energy piping located in the Unit 4 EDG Building.

Due to the location of the air intakes and the exhausts, as described in 9.15.5.2.2 above, there will be no significant contamination of the intake air by exhaust gases.

The building's air intake is located approximately 32 feet above ground level and the air intake path contains several turns resulting in a minimum amount of dust or other deleterious material being introduced into the diesel engine rooms. Although this air is reasonably clean, there will be some dust, etc. entrained in it, and therefore, all switches in the diesel engine rooms have been enclosed to protect them from dust and moisture, and the diesel engines' air intakes utilize a filter to minimize the introduction of these contaminants into the diesel engines. The greater quantity of switches, relays and other electrical contacts are located in the local EDG control rooms directly above the EDGs. These rooms are separate from the engine rooms and are climate controlled, i.e., air entering these rooms is filtered and cooled via air conditioning units. Not only does this arrangement prevent dust and other deleterious material from fouling the electrical equipment, it prevents excess moisture from accumulating in the room that could cause rusting or corroding of electrical contacts or the drifting of setpoints of instruments and meters due to temperature variations. The concrete walls and floors in the Unit 4 EDG building are sealed and/or painted to preclude dust from being generated internally.

Any single failure in the diesel generator combustion air intake/exhaust systems is bounded by the loss of an EDG. A failure modes and effects analysis associated with the loss of an EDG has been performed and the results are discussed in Section 8.3.

9.15.5.4 TESTS AND INSPECTIONS

System components associated with EDGs 4A and 4B were inspected and tested by the manufacturer. Following installation and prior to declaring the systems operable these diesel generator combustion air intake/exhaust systems were inspected, tested, and operated. Testing was performed to verify system operability in accordance with plant Technical Specification requirements, manufacturer's recommendations, and applicable codes and standards.

9.15.5.5 INSTRUMENTATION APPLICATION

9.15.5.5.1 EDGs 3A AND 3B

The diesel generator combustion air exhaust temperature for EDGs 3A and 3B is indicated at the local control panels.

9.15.5.5.2 EDGs 4A AND 4B

The following instrumentation and controls are available, at the local control panels, for the systems associated with EDGs 4A and 4B:

1. Temperature switches provide high-high and high exhaust temperature, combustion air high temperature, and turbocharger high differential temperature annunciation via an EDG trouble alarm locally and in the Main Control Room. Temperature switches also provide an EDG shutdown function, during nonemergency operation, on high-high exhaust temperature.

2. Pressure switches provide high exhaust pressure and combustion air low pressure annunciation via an EDG trouble alarm locally and in the Main Control Room.

3. Indicators provide air box pressure, exhaust pressure, and engine exhaust temperature for each cylinder.

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.15-1

REFER TO ENGINEERING DRAWING

5613-M-3022 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

EMERGENCY DIESEL
ENGINE AND OIL SYSTEM
DG 3A AIR STARTING SYSTEM
FIGURE 9.15-1

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.15-2

REFER TO ENGINEERING DRAWING

5613-M-3022 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 3

EMERGENCY DIESEL
ENGINE AND OIL SYSTEM
DG 3B AIR STARTING SYSTEM

FIGURE 9.15-2

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.15-3

REFER TO ENGINEERING DRAWING

5613-M-3022 , SHEET 3

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

EMERGENCY DIESEL
ENGINE AND OIL SYSTEM
DG 3A FUEL OIL
FIGURE 9.15-3

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.15-4

REFER TO ENGINEERING DRAWING

5613-M-3022 , SHEET 4

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

EMERGENCY DIESEL ENGINE
AND OIL SYSTEM
DG 3B FUEL OIL
FIGURE 9.15-4

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.15-5

REFER TO ENGINEERING DRAWING

5613-M-3022 , SHEET 5

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

EMERGENCY DIESEL
ENGINE AND OIL SYSTEM
DG 3A LO & COOLING WATER
FIGURE 9.15-5

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.15-6

REFER TO ENGINEERING DRAWING

5613-M-3022 , SHEET 6

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 3

EMERGENCY DIESEL

ENGINE AND OIL SYSTEM

DG 3B LO & COOLING WATER

FIGURE 9.15-6

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.15-7

REFER TO ENGINEERING DRAWING

5614-M-3022 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

EMERGENCY DIESEL
ENGINE AND OIL SYSTEM
EDG 4A AIR STARTING SYSTEM
FIGURE 9.15-7

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.15-8

REFER TO ENGINEERING DRAWING

5614-M-3022 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 4

EMERGENCY DIESEL

ENGINE AND OIL SYSTEM

EDG 4B AIR STARTING SYSTEM

FIGURE 9.15-8

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.15-9

REFER TO ENGINEERING DRAWING

5614-M-3022 , SHEET 3

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 4

EMERGENCY DIESEL
ENGINE AND OIL SYSTEM
EDG 4A FUEL SYSTEM

FIGURE 9.15-9

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.15-10

REFER TO ENGINEERING DRAWING

5614-M-3022 , SHEET 4

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 4

EMERGENCY DIESEL
ENGINE AND OIL SYSTEM
EDG 4B FUEL SYSTEM

FIGURE 9.15-10

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.15-11

REFER TO ENGINEERING DRAWING

5614-M-3022 , SHEET 5

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

EMERGENCY DIESEL
ENGINE AND OIL SYSTEM
DG 4A LO & COOLING WATER
FIGURE 9.15-11

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.15-12

REFER TO ENGINEERING DRAWING

5614-M-3022 , SHEET 6

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

EMERGENCY DIESEL
ENGINE AND OIL SYSTEM
DG 4B LO & COOLING WATER
FIGURE 9.15-12

9.16 LOAD CENTER AND SWITCHGEAR ROOMS AIR CONDITIONING SYSTEM

9.16.1 DESIGN BASIS

The Load Center and Switchgear Rooms Air Conditioning System is designed to accomplish the following:

1. Remove the heat dissipated by all equipment in the Load Center and Switchgear Rooms during normal plant operation and emergency conditions, maintaining room temperatures below 95°F with an outdoor air temperature of 95°F. However, the design limit for the equipment in the Load Center Room is 104°F, while the design limit for the equipment in Switchgear Room is 104°F. It should be noted, that if in single train operation during emergency operations (i.e. one chiller unit during LOOP/SI), operator action may be required to prevent exceeding the Load Center and Switchgear Rooms equipment design temperatures within 7 days. C26

2. Provide a redundant, reliable, and independent system supplied from emergency power to maintain a temperature controlled environment for the safety related equipment located within the Load Center and Switchgear Rooms.

The air conditioning system itself does not perform a safety related function.

9.16.2 SYSTEM DESIGN AND OPERATION

The operating diagrams for the Load Center and Switchgear Rooms Air Conditioning System are shown schematically in Figure 9.16-1 through 9.16-4.

The Load Center and Switchgear Rooms Air Conditioning System consists of two redundant trains. Each train is provided with two 50 percent capacity chillers, two chilled water pumps, one air separator, one expansion tank, piping, instrumentation, and four 100 percent capacity air handling units (one each for the Load Center and Switchgear Rooms). Each train is capable of providing 100 percent cooling for the rooms during normal and emergency conditions. Each of the rooms is served by both trains. A total of 200 percent cooling can be provided for each room during normal and emergency (i.e., loss of offsite power) conditions with both trains operating.

During normal plant operation, two chillers are required to provide 100 percent cooling for all the rooms. The chillers of the same or opposite trains can be used to satisfy the normal demand. However, only one chiller of either train is required during emergency conditions since the heat loads are substantially reduced. This arrangement permits removal of a chiller unit or train from service for maintenance while providing the required cooling for the rooms. It should be noted, that if in single train operation during emergency operations (i.e. one chiller unit during LOOP/SI), operator action may be required to prevent exceeding the Load Center and Switchgear Rooms equipment design temperatures within 7 days.

There are three modes of operation for each chiller that correspond to its associated chilled water pump control switch positions: STANDBY, AUTO, and ON. Following a loss of offsite power (LOOP) with the chiller control switch in the ON position, the chiller will automatically load onto its respective emergency diesel generator (EDG) in approximately five minutes. As a result, both chillers of each train will be loaded on their EDG if each is selected in the ON mode of operation. Administrative controls will be imposed to limit the operation of the chillers of the same train in the ON mode/control switch position. This mode will be limited to maintenance activities at which time continuous surveillance will be required. Therefore, manual actions may be taken to change the chiller control switch position necessary to limit EDG loading.

When the chillers are in either the AUTO or STANDBY mode of operation, only the lead chiller for each train will be loaded automatically on the EDGs approximately five minutes following a LOOP. The lead chiller is determined from the position of the Chilled Water Pump Selector Switch. Whichever Chilled Water Pump is selected (1 or 2,) that chiller is the lead, the other is the lag. The Air Handler Units will load automatically on the EDGs in response to room temperature regardless of the chiller mode of operation.

The Load Center and Switchgear Rooms Air Conditioning System is designed to maintain a room temperature during normal and emergency plant conditions of 95°F or lower with an outdoor air temperature of 95°F. The system is designed to automatically control room temperature below the design limits.

9.16.3 SAFETY ANALYSIS

The Load Center and Switchgear Rooms Air Conditioning System is designed with sufficient redundancy such that any component failure, up to malfunction of a complete cooling train, will not prevent the system from performing its cooling function. It should be noted, that if in single train operation during emergency operations (i.e. one chiller unit during LOOP/SI), operator action may be required to prevent exceeding the Load Center and Switchgear Rooms equipment design temperatures within 7 days. The system is seismically designed and backed by emergency power so that neither loss of offsite power nor earthquake will cause loss of cooling function capability.

Exposures to other credible hazards have also been accommodated to ensure that the required Load Center and Switchgear Room cooling is provided. The hazards include fire, high winds, missiles, high energy pipe breaks (Unit 4 only) and heavy load drops. The risk from these hazards is considered to be sufficiently low such that the effects would not compromise the system cooling function capability. Nevertheless, additional cooling function reliability and independence is provided by wall-mounted fans located in the west wall of the building. In the extremely unlikely event that the entire air conditioning system is rendered non-functional, an operator could open the doors and start the fan locally to provide once-through ventilation. The once-through ventilation concept is consistent with the original plant design.

Based on the preceding, the adverse effects of any credible event or system failure will not compromise the ability to provide cooling to the Load Center and Switchgear Rooms.

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.16-1

REFER TO ENGINEERING DRAWING

5613-M-3070 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

TURBINE BUILDING VENTILATION
LOAD CENTER & SWGR ROOMS
CHILLED WATER SYSTEM-TRAIN A
FIGURE 9.16-1

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.16-2

REFER TO ENGINEERING DRAWING

5613-M-3070 , SHEET 2

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FLORIDA POWER & LIGHT COMPANY

TURKEY POINT PLANT UNIT 3

TURBINE BUILDING VENTILATION
LOAD CENTER & SWGR ROOMS
CHILLED WATER SYSTEM-TRAIN B

FIGURE 9.16-2

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.16-3

REFER TO ENGINEERING DRAWING

5614-M-3070 , SHEET 1

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FLORIDA POWER & LIGHT COMPANY **TURKEY POINT PLANT UNIT 4**
TURBINE BUILDING VENTILATION LOAD CENTER & SWGR ROOMS CHILLED WATER SYSTEM-TRAIN A **FIGURE 9.16-3**

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.16-4

REFER TO ENGINEERING DRAWING

5614-M-3070 , SHEET 2

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

TURBINE BUILDING VENTILATION
LOAD CENTER & SWGR ROOMS
CHILLED WATER SYSTEM-TRAIN B
FIGURE 9.16-4

9.17 INSTRUMENT AIR SYSTEM

9.17.1 DESIGN BASIS

The Instrument Air System is designed to accomplish the following:

1. Provide motive power and control air to air operated components such that they operate in a predictable manner and perform their functions reliably.

2. Provide instrument air to achieve and maintain cold shutdown (Mode 5) during fires that require Control Room evacuation, with or without concurrent loss of offsite power.

3. Provide instrument air to achieve and maintain hot standby (Mode 3) during fires not requiring Control Room evacuation, with or without concurrent loss of offsite power.

9.17.2 SYSTEM DESIGN AND OPERATION

A system diagram for the Instrument Air System (IAS) is shown schematically in Figures 9.17-1 and 9.17-2.

The IAS consists of air compressors, receiver tanks, aftercoolers, moisture separators, desiccant dryers, particulate filters, instrumentation and interconnecting piping and valves. The system is designed to provide a continuous reliable air source during normal plant operation.

Each Unit has its own air supply system which is cross connected through 4" cross-tie piping. Two compressors are provided for each Unit - one diesel driven compressor and one motor driven compressor - with the rest of the air supply system comprising a receiver tank, moisture separator, desiccant dryer and particulate filters. The system also contains a “swing” dryer that can be aligned to either unit’s air system. An instrument air header downstream of each unit's air supply equipment provides air to the various end users such as air operated valves. A backup air source from the service air system C28
is also provided. This source of air is normally isolated and requires manual line-up when needed.

Each compressor is sized for sufficient capacity to provide instrument air for both Unit 3 and Unit 4 combined. The compressors have a capacity that is C26 greater than 750 SCFM. The system uses the motor driven compressor for its normal source of compressed air with the two compressors operating LEAD/LAG such that upon loss of pressure in one Unit, the opposite Unit's motor driven compressor will load and provide compressed air as required. Control of the load and unload feature of the motor driven compressors is from on-skid C26 pressure transducers.

The diesel driven compressors are capable of supplying the required capacity without reliance on external power sources. Each diesel driven compressor will automatically start on when the on-skid pressure transducer senses low- C26 low system pressure, a low voltage on the power busses to the motor driven compressors, or from a manual start switch at the compressor panel. During fires coincident with a loss of offsite power, instrument air availability is assured because the diesel driven compressors are located in separate fire zones. In the event that one of the diesel compressors is damaged by fire, the opposite Unit's compressor can supply both Units through the crosstie piping.

The quality of the air provided by the IAS is important to proper functioning of the end-use components. The primary method of removing particulates and debris from the air stream is by filters installed at each end-use component. In addition to this, particulate filters in the dryer outlet stream remove particulates before the air gets into the IAS piping. This reduces the load on local filters at the end-use devices and decreases their maintenance requirements.

The method used to reduce moisture is by centrifugal moisture separators at the compressor after coolers and at the air receiver as well as desiccant dryers at the outlet of the air receiver. The desiccant dryers are capable of lowering the dewpoint of the instrument air to -40°F which satisfies the ISA standard of 18°F below minimum ambient temperature (Reference 1). The dryers require electrical power to regenerate the desiccant. Power to the C26 Unit 3 and 4 dryers is supplied from the 'A' channel vital AC power busses from the associated unit which gives the dryer heaters a second source of power in case of a loss of offsite power. The swing dryer is powered from a nonvital source. In addition, should power not be available to the dryers, crosstie connections from the opposite Unit's instrument air supply is available.

9.17.3 SAFETY ANALYSIS

The IAS does not perform a safety related function, however it does provide air to safety related components. Equipment which uses instrument air and which are necessary for the plant to mitigate the consequences of an accident or to achieve and maintain safe shutdown, are designed to either fail in the required safe position or are provided with safety related backup air or nitrogen sources or have other means to accomplish their safety functions.

The IAS is designed to provide a continuous and reliable source of instrument air. The design incorporates sufficient redundancy such that any active component failure will not prevent the system from performing its function. Both the motor-driven and diesel driven compressors are independent from their opposite unit counterpart. To achieve a continuous reliable source of instrument air, a single compressor is sized for the expected instrument air demand of both units. Because the IAS is operated cross-connected between units, the two systems will function as a single system. Normally a single motor driven compressor will supply all the instrument air required. In the event the system pressure decreases to the LAG compressor's setpoint, the second (LAG) motor driven compressor in the other unit will load and supplement the air supply. Further reduction of system pressure or a loss of power to the motor driven compressors will automatically start the first (Standby-Lead) diesel driven compressor. If further reduction of system pressure occurs the second (Standby-Lag) diesel driven compressor will start, providing a third and fourth level of reliability. An additional level of redundancy is provided by a backup air supply from the service air system. C28

9.17.4 REFERENCES

1. Instrument Society of America Standard ISA S7.3, “Quality Standard for Instrument Air,” November 16, 1981.

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.17-1

REFER TO ENGINEERING DRAWING

5613-M-3013 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 3

INSTRUMENT AIR SYSTEM

FIGURE 9.17-1

FINAL SAFETY ANALYSIS REPORT

FIGURE 9.17-2

REFER TO ENGINEERING DRAWING

5614-M-3013 , SHEET 1

REV. 13 (10/96)

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNIT 4

INSTRUMENT AIR SYSTEM

FIGURE 9.17-2

9.18 TECHNICAL SUPPORT CENTER

An on-site Technical Support Center (TSC) serves as a work area for use by technical and management personnel in order to provide support to Control Room personnel during an emergency. The TSC is outside of the Control Room Envelope and located in a building that is separate from the Control Building at the back of the site property near the Circulating Water Inlet Bay. Like the Control Room, the TSC is equipped with a dedicated ventilation system that provides for a controlled environment for the comfort and safety of the TSC occupants and the protection against excessive radiological exposure during a design basis accident. The TSC ventilation system emergency mode is actuated by the Control Room Emergency Ventilation System B Channel. The TSC and its ventilation system maintain occupants' exposure below 10 CRF 50 Appendix A GDC-19, i.e., 5 rem Total Effective Dose Equivalent (TEDE) limits (Reference 1 and 2). C30

Additionally, a removable shield block wall in front of the Unit 3 Containment equipment hatch mitigates accident dose to the TSC. However, shine dose analysis FPL-085-CALC-001, "Technical Support Center LOCA Radiological Analysis" (Reference 9.18.1, 1), demonstrates that this wall is not required to be in place to keep TSC exposure below these TEDE limits in the event of a limiting DBA.

9.18.1 REFERENCES

1. FPL Calculation FPL-085-CALC-001, "Technical Support Center LOCA Radiological Analysis", Rev. 1.

2. FPL Calculation NAI-1396-048, "Turkey Point EPU TSC Direct Shine Doses for LOCA", Rev. 6A.