9.3.2.4 Component Cooling Loop Components

9.3.2.4.1 Component Cooling Heat Exchangers

The two component cooling heat exchangers are of the shell and straight tube type. Service water circulates through the tubes while component cooling water circulates through the shell side. Parameters are presented in Table 9.3-2.

9.3.2.4.2 Component Cooling Pumps

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

9.3.2.4.3 Auxiliary Cooling Water Pumps

The component cooling pumps do not run during the injection phase of a loss of coolant accident with loss of offsite power. The CCW circulating water pumps provide cooling for the safety injection pump motors and the auxiliary component cooling water pumps provide cooling for the recirculation pumps during this phase, with heat absorbed by the thermal inertia of the component cooling system.

Two motor-driven auxiliary component cooling water pumps are started during the injection phase to provide cooling flow to the recirculation pump motor coolers. A CCW circulating water pump is connected to the motor shaft of each safety injection pump to cool the safety injection pump bearings. Both the auxiliary component cooling water pumps and the CCW circulating water pumps are discussed in further detail in Section 6.2.

9.3.2.4.4 Component Cooling Surge Tank

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

9.3.2.4.5 Component Cooling Valves

The valves used in the component cooling loop are standard commercial valves constructed of carbon steel with bronze or stainless steel trim. Since the component cooling water is not normally radioactive, special features to prevent leakage to the atmosphere are not provided.

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

9.3.2.4.6 Component Cooling Piping

All component cooling loop piping is carbon steel with welded joints and connections except at components, which might need to be removed for maintenance. The piping has been evaluated for the most limiting component cooling water temperatures under loss of coolant accident conditions and found to be acceptable

9.3.2.5 Residual Heat Removal Loop Components

9.3.2.5.1 Residual Heat Exchangers

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

9.3.2.5.2 Residual Heat Removal Pumps

The two residual heat removal pumps are vertical, centrifugal units with special seals to prevent reactor coolant leakage to the atmosphere. All pump parts in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material. Cooling water is provided from the component cooling water system via flexible stainless steel hose.

9.3.2.5.3 Residual Heat Removal Valves

The valves used in the residual heat removal loop are constructed of austenitic stainless steel or equivalent corrosion resistant material. Stop valves are provided to isolate equipment for maintenance. Throttle valves are provided for remote and manual control of the residual heat exchanger tube side flow. Check valves prevent reverse flow through the residual heat removal pumps.

Two remotely-operated series stop valves at the inlet with a pressure interlock isolate the residual heat removal loop from the reactor coolant system. In addition the residual heat removal loop is isolated from the reactor coolant system by two series check valves and a remotely operated stop valve on the outlet lines. As depicted in Plant Drawing 227781 [Formerly UFSAR Figure 9.3-1, Sheet 1], overpressure protection in the residual heat removal loop is provided by a relief valve. Valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the waste disposal system.

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

9.3.2.5.4 Residual Heat Removal Piping

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

9.3.2.5.5 Low Pressure Purification System

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

9.3.2.6 Spent Fuel Pit Loop Components

9.3.2.6.1 Spent Fuel Pit Heat Exchanger

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

9.3.2.6.2 Spent Fuel Pit Pumps

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

9.3.2.6.3 Refueling Water Purification Pump

When it is required to clean up the refueling water storage tank water, the refueling water purification pump circulates water in a loop between the refueling water storage tank and the spent fuel pit demineralizer and filter. All wetted surfaces of the pump are austenitic stainless steel. The pump is operated manually from a local station.

9.3.2.6.4 Spent Fuel Pit Filter

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

9.3.2.6.5 Spent Fuel Pit Strainer

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

9.3.2.6.6 Spent Fuel Pit Demineralizer

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

9.3.2.6.7 Spent Fuel Pit Skimmer (Deleted)

9.3.2.6.8 Spent Fuel Pit Valves

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

9.3.2.6.9 Spent Fuel Pit Piping

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

9.3.3 System Evaluation

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

9.3.3.1 Availability And Reliability

9.3.3.1.1 Component Cooling Loop

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

Outside the containment, the residual heat removal pumps, the spent fuel heat exchanger, the component cooling pumps and heat exchangers and associated valves, piping and instrumentation are maintainable and inspectable during power operation. Replacement of one pump or one heat exchanger is practicable while the other units are in service. The wetted surfaces of the component cooling loop are fabricated from carbon steel. The component cooling water contains a corrosion inhibitor to protect the carbon steel. Welded joints and connections are used except where flanged closures are employed to facilitate maintenance. The entire system is seismic Class I and is housed in structures of the same classification. The components are designed to the codes given in Table 9.3-1 and the design pressures given in Table 9.3-2. In addition, the components are not subjected to any high pressures or stresses. Hence, a rupture or failure of the system is very unlikely.

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

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

9.3.3.1.2 Residual Heat Removal Loop

Two pumps and two heat exchangers are utilized to remove residual and sensible heat during plant cooldown. If one of the pumps and/or one of the heat exchangers is not operable, safe operation is governed by Technical Specifications and safe shutdown of the plant is not affected; however, the time for cooldown is extended. The function of this equipment following a loss-of-coolant accident is discussed in Section 6.2.

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

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

9.3.3.1.3 Spent Fuel Pit Cooling Loop

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

9.3.3.2 Leakage Provisions

9.3.3.2.1 Component Cooling Loop

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

Tube or coil leaks in components being cooled would be detected during normal plant operations by the leak detection system described in Sections 4.2.7 and 6.7. Such leaks are also detected at any time by a radiation monitor that samples the component cooling pump discharge downstream of the component cooling heat exchangers.

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

The component, which is leaking can be located by sequential isolation or inspection of equipment in the loop. If the leak is in one of the component cooling water heat exchangers it can be isolated and repaired within the limitations of the Technical Specifications. Overall leakage within the containment is limited to the value given in the Technical Specifications.

Should a large tube-side to shell-side leak develop in a residual heat exchanger, the water level in the component cooling surge tank would rise, and the operator would be alerted by a high water alarm. The atmospheric vent on the tank is automatically closed in the event of high radiation level in the component cooling loop. If the leaking residual heat exchanger is not isolated from the component cooling loop before the inflow completely fills the surge tank, the relief valve on the surge tank lifts. The discharge of this relief valve is routed to the auxiliary building waste holdup tank.

The severance of a cooling line serving an individual reactor coolant pump cooler would result in substantial leakage of component cooling water. However, the piping is small as compared to piping located in the missile-protected area of the containment. Therefore, the water stored in the surge tank after a low level alarm together with makeup flow provides ample time for the closure of the valves external to the containment to isolate the leak before cooling is lost to the essential components in the component cooling loop.

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

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

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

9.3.3.2.2 Residual Heat Removal Loop

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

Each of the two residual heat removal pumps is located in a shielded compartment with a floor drain. Piping conveys the drain water to a common sump. Two redundant sump pumps, each capable of handling the less than 50 gpm flow, which would result from the failure of a residual heat removal pump seal, discharge to the waste holdup tank.

9.3.3.2.3 Spent Fuel Pit Cooling Loop

Whenever a leaking fuel assembly is transferred from the fuel transfer canal to the spent fuel storage pool, a small quantity of fission products may enter the spent fuel cooling water. A bypass purification loop is provided for removing these fission products and other contaminants from the water.

The probability of inadvertently draining the water from the cooling loop of the spent fuel pit is exceedingly low. The only mode would be from such actions as opening a valve on the cooling line and leaving it open when the pump is operating. In the unlikely event of the cooling loop of the spent fuel pit being drained, the spent fuel storage pit itself cannot be drained and no spent fuel is uncovered since the spent fuel pit cooling connections enter near the top of the pit. With no heat removal the time for the spent fuel pit water to rise from 180°F to 212°F with a full core in storage is at least 1.8 hr. Makeup water can be supplied within this time from the primary water storage tank, the refueling water storage tank and/or the fire protection system. The maximum required makeup rate for boiloff is 62 gpm (for a full core). Spent fuel pit temperature and level instrumentation would warn the operator of an impending loss of cooling. A local flow indicator is available to support operation of the Spent Fuel Pit Pumps.

9.3.3.3 Incident Control

9.3.3.3.1 Component Cooling Loop

In the unlikely event of a pipe severance in the component cooling loop, backup is provided for postaccident heat removal by the containment fan coolers.

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

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

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

Each return line (combined oil coolers and combined thermal barrier coolers) has a common remotely operated valve outside the containment wall. The cooling water supply line to the excess letdown heat exchanger contains a check valve inside the containment wall and both supply and return lines have automatically isolated valves outside the containment wall.

Flow indication is provided on the component cooling return lines from the safety injection and residual heat removal pumps. Each of the component cooling supply lines to the residual heat exchangers has a normally closed remotely-operated valve. If one of the valves fails to open upon a safety injection signal, the valve, which does open supplies a heat exchanger with sufficient cooling to remove the heat load during long term postaccident recirculation.

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

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

9.3.3.3.2 Residual Heat Removal Loop

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

9.3.3.3.3 Spent Fuel Pit Cooling Loop

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

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

9.3.3.4 Malfunction Analysis

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

9.3.4 Minimum Operating Conditions

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

9.3.5 Tests and Inspections

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

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

REFERENCES FOR SECTION 9.3

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

TABLE 9.3-1
Auxiliary Coolant System Code Requirements

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

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

Component Cooling Pumps	Parameters
Quantity	3
Type	Horizontal centrifugal
Rated capacity (each), gpm	3600
Rated head, ft H_2O	220
Motor horsepower, hp	250
Material (pump casing)	Cast iron or Carbon steel
Design pressure, psig	150
Design temperature, °F	200
Component Cooling Heat Exchangers	
Quantity	2
Type	Shell and straight tube
Design heat transfer, Btu/hr	31.4×10^6
Shell side (component cooling water)	
Operating inlet temperature, °F	100.1
Operating outlet temperature, °F	88.2
Design flow rate, lb/hr	2.66×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Aluminum-bronze
Tube side (service water)	
Operating inlet temperature, °F	75_1
Operating outlet temperature, °F	81.9
Design flow rate, lb/hr	4.55×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Copper-nickel (90-10)

TABLE 9.3-2 (Sheet 2 of 2)

Component Cooling Loop Component Data

Component Cooling Surge Tank	
Quantity	1
Volume, gal	2000
Normal water volume, gal	1000
Design pressure, psig	100
Design temperature, °F	200
Construction material	Carbon steel
Relief valve setpoint, psig	52

Auxiliary Component Cooling Water Pumps	
Quantity	2
Type	Vertical centrifugal
Rated capacity, gpm	80
Rated head, ft H_2O	100
Motor horsepower, hp	5
Casing material	Cast steel
Design pressure, psig	150
Design temperature, °F	200

CCW Circulating Water Pumps (Safety Injection Pumps)	
Quantity	3
Type	Centrifugal
Rated capacity, gpm	20
Rated head, ft H_2O	115
Casing material	Stainless Steel
Design pressure, psig	225
Design temperature, °F	200

Component Cooling Loop Piping and Valves	
Design pressure, psig	150
Design temperature, °F	200

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

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

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

Residual Heat Removal Pumps

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

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

Residual Heat Exchangers

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

Residual Heat Removal Loop Piping and Valves

1. Isolated loop	
Design pressure, psig	600
Design temperature, °F	400
2. Loop Isolation	
Design pressure, psig	2485
Design temperature, °F	650

Notes:

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

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

Spent fuel pit heat exchanger	
Quantity	1
Type	Shell and U-tube
Design heat transfer, Btu/hrs_1	7.96×10^6
Shell side (component cooling water)	
Normal operating inlet temperature, $°F_1$	100
Normal operating outlet temperature, $°F_1$	105.7
Design flow rate, lb/hr	1.4×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Carbon steel
Tube side (spent fuel pit water)	
Normal operating inlet temperature, $°F_1$	120
Normal operating outlet temperature, $°F_1$	112.8
Design flow rate, lb/hr	1.1×10^6
Design temperature, °F	200
Design pressure, psig	150
Material	Stainless steel
Spent fuel pit skimmer pump	Retired in place
Refueling water purification pump	
Quantity	1
Type	Horizontal centrifugal
Rated capacity, gpm	100
Rated head, ft H_2O	150
Design pressure, psig	150
Design temperature, °F	200
Material	Stainless steel

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

Spent fuel pit cooling loop piping and valves	
Design pressure, psig	150
Design temperature, °F	200
Spent fuel pit skimmer loop piping and valves	Retired in place
Refueling water purification loop piping and valves	
Design pressure, psig	150
Design temperature, °F	200
Spent fuel pit pump	
Quantity	2
Type	Horizontal centrifugal
Material	Stainless steel
Rated capacity, gpm	2,300
Rated head, ft H_2O	125
Motor, hp	100
Design pressure, psig	150
Design temperature, °F	200
Spent fuel storage pool	
Volume ft^3	37,300
Typical Boron concentration, ppm boron	>2,000 min
Tech Spec Boron concentration, ppm boron	>2,000 min
Spent fuel pit filter	
Quantity	1
Internal design pressure of housing, psig	200
Design temperature, °F	250
Rated flow, gpm	100
Maximum differential pressure across filter element at rated flow (clean cartridge), psi	5
Maximum differential pressure across filter element prior to removing, psi	20
Filtration requirement	98-percent retention of particles down to 5 μ

TABLE 9.3-4 (Sheet 3 of 3)

Spent Fuel Cooling Loop Component Data

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

Notes:
1. Original design.

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

Components	Malfunction	Comments and Consequences
1. Component cooling water pumps	Rupture of a pump casing	The casing and shell are designed for 150 psi and 200°F, which exceeds maximum operating conditions. Pump is inspectable and protected against credible missiles. Rupture is not considered credible. However, each unit is isolable. Two of the three pumps are needed to carry total pumping load.
2. Component cooling water pumps	Pump fails to start	One operating pump supplies sufficient cooling water for emergency core cooling during recirculation.
3. Component cooling water pumps	Manual valve on a pump suction line	This is prevented by pre-startup and operational checks. Further, during normal operation, each pump is checked on a periodic basis, which would show if a valve is closed.
4. Component cooling water valve	Normally open valve	The valve is checked open during periodic operation of the pumps during normal operation.
5. Component cooling heat exchanger	Tube or shell rupture	Rupture is considered improbable because of low operating pressures. Each unit is isolable. Both units may be required to carry total heat load for normal operation at 95°F Service Water.
6. Demineralized water makeup line check valve	Sticks open	The check valve is backed up by the manually-operated valve. Manual valve is normally closed.
7. Component cooling heat exchanger vent or drain valve	Left open	This is prevented by pre-startup and operational checks. On the operating unit such a situation is readily assessed by makeup requirements to system. On the second unit such a situation is ascertained during periodic testing.
8. Component cooling water outlet valve to residual heat exchanger	Fails to open	There is one valve on each outlet line from each heat exchanger. One heat exchanger remains in service and provides adequate heat removal during long-term recirculation. During normal operation the cooldown time is extended.

9.3 FIGURES

9.4 SAMPLING SYSTEM

9.4.1 Design Basis

9.4.1.1 Performance Requirements

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

Primary samples include the following:

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

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

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

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

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

Sampling system discharge flows are limited under normal and anticipated fault conditions (malfunctions or failure) to preclude any fission product releases beyond the limits of 10 CFR 20. Shielding has been provided to minimize operator exposure to any radiation during the sampling procedures.

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

9.4.1.2 Design Characteristics

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

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

In addition, the system is capable of the following:

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

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

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

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

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

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

Sampling of other process coolants, such as tanks in the waste disposal system, is accomplished locally. Equipment for sampling secondary and nonradioactive fluids is separated from the equipment provided for reactor coolant samples. Leakage and drainage resulting from

the sampling operations are collected and drained to tanks located in the waste disposal system.

9.4.1.3 Primary Sampling

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

9.4.1.3.1 High Pressure-High Temperature Samples

A sample connection is provided from each of the following:

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

9.4.1.3.2 Low Pressure-Low Temperature Samples

A sample connection is provided from each of the following:

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

9.4.1.4 Expected Operating Temperatures

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

9.4.1.5 Secondary Sampling

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

A sample connection is provided from each of the following:

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

9.4.1.6 Codes and Standards

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

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

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

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

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

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

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

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

8. Provide the capability to flush the sampled lines.

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

9.4.2 System Design and Operation

9.4.2.1 Primary Sampling System

The primary sampling system consists of the high-radiation sampling system, which is shown in Plant Drawing 9321-2745. The high-radiation sampling system provides the representative samples for inline monitoring and laboratory analysis under normal or postaccident conditions. Analytical results provide guidance in the operation of the reactor coolant, auxiliary coolant, steam, and chemical and volume control systems. Analyses show both chemical and radiochemical conditions. Typical information obtained includes reactor coolant boron and chloride concentrations, fission product radioactivity level, hydrogen, oxygen, and fission gas content, corrosion product concentration, and chemical additive concentration.

The information is used in regulating boron concentration, evaluating fuel element integrity and mixed-bed demineralizer performance, and regulating additions of corrosion controlling chemicals to the systems. The high-radiation sampling system can be operated intermittently or on a continuous basis. Samples can be withdrawn under conditions ranging from full power to cold shutdown to postaccident conditions.
Reactor coolant liquid, [*Note - For postaccident conditions, the reactor coolant liquid sample may be taken from the reactor coolant system hot legs 21 and 23 or the recirculation pump*

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

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

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

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

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

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

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

The reactor coolant sample originating from the residual heat removal loop of the auxiliary coolant system has a motor-operated isolation valve located close to the sample source outside the containment. The sample line from this source intersects the sample line coming from the hot leg at a point ahead of the sample heat exchanger. This sample then follows the same flow

path as that described for the reactor coolant system hot-leg samples. See Plant Drawings 9321-2745 and 227178 [Formerly UFSAR Figure 9.4-1].

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

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

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

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

9.4.2.1.1 Components

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

9.4.2.1.1.1 Sample Heat Exchangers

Ten sample heat exchangers reduce the temperature of samples from the pressurizer steam space, the pressurizer liquid space, each steam generator, and the reactor coolant system liquid before samples reach the sample vessels and the sample sink. The tube side of the heat exchangers is austenitic stainless steel, the shell side is carbon steel.

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

9.4.2.1.1.2 Delay Coil and Restriction Orifice

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

9.4.2.1.2 Liquid Sampling Panel

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

9.4.2.1.3 Isotopic Analyzer

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

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

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

9.4.2.1.4 Boron Analyzer

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

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

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

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

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

9.4.2.1.5 Cart and Casks

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

9.4.2.1.6 Chemical Analysis Panel

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

The chemical analysis panel is divided into three major sections:

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

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

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

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

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

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

9.4.2.1.7 Chemical Monitor Panel

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

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

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

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

9.4.2.1.8 High Radiation Sampling System Collection Tank

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

9.4.2.1.8.1 Chemical Drain Tank

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

9.4.2.1.8.2 Piping and Fittings

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

9.4.2.1.8.3 Valves

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

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

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

9.4.2.2 Secondary Sampling System

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

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

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

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

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

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

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

9.4.3 System Evaluation

9.4.3.1 Availability and Reliability

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

9.4.3.2 Leakage Provisions

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

9.4.3.3 Incident Control

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

9.4.3.4 Malfunction Analysis

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

9.4.3.5 High Radiation Sampling System Evaluation

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

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

REFERENCES FOR SECTION 9.4

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

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

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

TABLE 9.4-1
Sampling System Code Requirements

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

Notes:

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

TABLE 9.4-2
Primary Sampling System Components

Sample Heat Exchanger	
Number	10
Type	Coiled tube in shell
Heat exchanged (each), Btu/hr	2.14×10^5
Surface area (each), ft^2	3.73
Shell	
Design pressure, psig	150
Design temperature, °F	350
Component cooling water flow (nominal), gpm	17
Flow, lb/hr	20,000
Component cooling water	
inlet temperature, °F	105
outlet temperature, °F	130
Material	Carbon steel
Tubes	
Tube diameter in., O.D.	3/8
Design pressure, psig	2485
Design temperature, °F	680
Flow, lb/hr	209
Inlet temperature (saturated steam), °F	653
Outlet temperature, °F	127
Material	Austenitic stainless steel

TABLE 9.4-3
Malfunction Analysis of Sampling System

Sample Chains	Malfunction	Comments and Consequences
Pressurizer steam space sample, pressurizer liquid space sample, or hot-leg sample.	Remotely operated sampling valve inside reactor containment fails to close.	Diaphragm or motor-operated valve outside the reactor containment closes automatically on containment isolation signal or by operator action from the control room.
Any sample train.	Sample line break inside containment.	Same as above.

9.4 FIGURES

9.5 FUEL HANDLING SYSTEM

The fuel handling system provides a safe, effective means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves the plant after postirradiation cooling.

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

The fuel handling system consists basically of:

1. The reactor cavity, which is flooded only during plant shutdown for refueling.

2. The spent fuel pit, which is kept full of water and is always accessible to operating personnel.

3. The fuel transfer system, consisting of an underwater conveyor that carries the fuel through an opening between the areas listed in the discussion of plant containment.

9.5.1 Design Basis

9.5.1.1 Prevention of Fuel Storage Criticality

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

During reactor vessel head removal and while loading and unloading fuel from the reactor, boron concentration is maintained at not less than that required to shutdown the core to a k_{eff} = 0.95. Periodic checks of refueling water boron concentration ensure the proper shutdown margin. The new and spent fuel storage racks are designed so that it is impossible to insert assemblies in other than the prescribed locations. The new fuel racks and spent fuel storage pit have accommodations as defined in Table 9.5-1. In addition, the spent fuel pit has the required spent fuel shipping area. The spent fuel storage pit is filled with borated water at a concentration to match that used in the reactor cavity and refueling canal during refueling. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to assure K_{eff} <1.0 even if unborated water was used to fill the pit and ≤0.95 when filled with water borated ≥ 2000 ppm boron. Limits on enrichment and burnup of fuel in the spent fuel storage pit are given in the Technical Specifications.

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

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

9.5.1.2 Fuel and Waste Storage Decay Heat

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

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

9.5.1.3 Fuel and Waste Storage Radiation Shielding

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

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

provided for waste handling and storage facilities to permit operation within requirements of 10 CFR 20.

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

9.5.1.4 Protection Against Radioactivity Release from Spent Fuel and Waste Storage

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

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

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

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

9.5.2 System Design and Operation

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

The fuel handling system may be generally divided into two areas:

The reactor cavity, which is flooded only during plant shutdown for refueling and the spent fuel pit, which is kept full of water and is always accessible to operating personnel. These two areas are connected by the fuel transfer system consisting of an underwater conveyor that carries the fuel through a fuel transfer tube, which penetrates the plant containment.

The reactor cavity is flooded with borated water from the refueling water storage tank. In the reactor cavity, fuel is removed from the reactor vessel, transferred through the water and placed in the fuel transfer system by a manipulator crane. In the spent fuel pit the fuel is removed from the transfer system and placed in storage racks with a long manual tool suspended from an overhead hoist.

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

9.5.2.1 Major Structures Required for Fuel Handling

9.5.2.1.1 Reactor Cavity

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

The reactor vessel flange is sealed to the bottom of the reactor cavity by a Presray seal, which prevents leakage of refueling water from the cavity. This seal is installed after reactor cooldown but prior to flooding the cavity for refueling operations. Following refueling operations and prior to return to power, this seal is removed. The cavity is large enough to provide storage space for the reactor upper and lower internals, the control cluster drive shafts, and miscellaneous refueling tools.

The floor and sides of the reactor cavity are lined with stainless steel.

9.5.2.1.2 Refueling Canal

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

9.5.2.1.3 Refueling Water Storage Tank

The normal duty of the refueling water storage tank is to supply borated water to the refueling canal and reactor cavity for refueling operations. In addition, the tank provides borated water for delivery to the core following either a loss-of-coolant or a steam line rupture accident. This is described in Chapter 6.

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

9.5.2.1.4 Spent Fuel Storage Pit

The spent fuel storage pit is designed for the underwater storage of spent fuel assemblies, failed fuel cans if required, and control rods after their removal from the reactor.

The pit accommodations are listed in Table 9.5-1.

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

The spent fuel storage pit is constructed of reinforced concrete and is seismic Class I design. This structure was analyzed to determine compliance with ACI-318(77), and SRP 3.8 of NUREG-0800. In addition to the mechanical loadings, the pool structure was also analyzed to

include the temperature induced loadings. For this purpose, the thermal boundary conditions were conservatively specified as 180°F pool water temperature and 0°F outside ambient. The thermal moments computed by the finite element analyses were combined with those due to mechanical loads. The results of these analyses show that there are large margins between the factored loads and corresponding design strengths.

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

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

9.5.2.1.5 Storage Rack

High density fuel storage racks have been designed to provide a maximum storage capacity of 1376 locations. The arrangement of the fuel storage racks in the spent fuel storage pool is shown in Figure 9.5-2.

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

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

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

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

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

Region 2, consisting of nine racks that use the egg-crate design, can store 1105 fuel assemblies and two failed fuel canisters. Region 2 racks consist of boxes welded into a “checkerboard” array with a storage location in each square. One Boraflex absorber panel is held to one side of each cell wall by picture frame sheathing. Region 2 racks were originally designed to store fuel assemblies that have undergone significant burnup (e.g., ≤5.0 w/o U^{235} with a burnup of at least 40,900 megawatt days per metric ton (MWD/MT)) or fuel assemblies with a relatively low initial enrichment and low burnup (i.e., ≤1.764 w/o U^{235} at zero burnup).

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

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

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

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

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

9.5.2.1.6 New Fuel Storage

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

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

9.5.2.2 Major Equipment Required for Fuel Handling

9.5.2.2.1 Reactor Vessel Stud Tensioner

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

The stud tensioner is a hydraulically-operated (oil as the working fluid) device provided to permit preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. A

stud tensioner was chosen in order to minimize the time required for the tensioning or unloading operations. Three tensioners are provided and they are normally applied simultaneously to three studs 120° apart. One hydraulic pumping unit operates the tensioners, which are hydraulically connected in parallel. The studs are tensioned to their operational load in a number of steps to prevent high stresses in the flange region and unequal loadings in the studs. A relief addition, micrometers are provided to measure the elongation of the studs after tensioning.

9.5.2.2.2 Reactor Vessel Head Lifting Device

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

9.5.2.2.3 Reactor Internals Lifting Device

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

9.5.2.2.4 Manipulator Crane

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

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

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

Safety features are incorporated in the system as follows:

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

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

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

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

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

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

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

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

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

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

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

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

The PaR Systems, Inc. Crane is a wheel-mounted platform, which spans the East-West (E-W) direction of the Spent Fuel Pool (SFP) and travels in the North-South (N-S) direction. The PaR

Crane is secured to the crane rails on the FSB El. 95'-0", via seismic hold-down brackets and associated bolting. An Encoder Tracking Device is mounted on the FSB West Walkway, El. 96'-6", which positions the crane in the N-S direction. The crane mounted computer will position the Crane Trolley-Tower Structure in the E-W direction. This equipment will position the Crane Motorized Hoist over a pre-assigned Spent Fuel Assembly (SFA) within the SFP. In addition, the PaR Crane controls interface with the existing FSB Up-Ender Control Console No. 21 (PK1). This computerized control feature provides assurance that the PaR Crane will not interfere with the FSB Up-Ender Assembly, which is located in the Fuel Transfer Canal.

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

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

9.5.2.2.6 Fuel Transfer System

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

During plant operation, the conveyor car is stored in the refueling canal inside the containment. A blank flange is bolted on the transfer tube on the reactor side and a gate valve closed on the spent fuel pit side (see Figure 5.2-5) to seal the reactor containment. The blind flange is supplied with a double o-ring seal and is pressurized by the WCCPP System during normal operation to assure containment isolation.

9.5.2.2.7 Rod Cluster Control Changing Fixture

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

releases it. The manipulator crane loads and removes the fuel assemblies into and out of the carriage.

9.5.2.2.8 Lower Internals Support Stand

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

9.5.3 System Evaluation

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

Basic provisions to ensure the safety of refueling operations are:

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

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

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

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

9.5.3.1 Incident Protection

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

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

This provision shall be satisfied with fuel in the reactor and when: 1) the reactor head is being moved, or 2) the upper internals are being moved, or 3) loading and unloading fuel from the reactor, or 4) heavy loads greater than 2300 lbs (except for installed crane systems) are being moved over the reactor with the reactor vessel head removed. If direct communication between the control room and the refueling cavity manipulator cannot be met, suspend any and all of these operations. Suspension of these operations shall not preclude completion of movement of the above components to a safe conservative position.

9.5.3.2 Malfunction Analysis

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

Inadvertently locating an unirradiated fuel assembly of 5.0-percent enrichment in a region II storage location has been analyzed. The analysis shows that the array would be subcritical even with no soluble boron poison in the water in the fuel storage pool. With a boron concentration of 350 ppm the shutdown margin would be more than 5-percent. The technical specifications require that the boron concentration be maintained at 2000 ppm or more at all times.

9.5.4 Minimum Operating Conditions

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

9.5.5 Tests and Inspections

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

9.5.6 Control of Heavy Loads

The control of heavy loads in the fuel storage building and the movement of loads over spent fuel in the spent fuel pit are discussed in the Technical Requirements Manual.

TABLE 9.5-1
Fuel Handling System Data

NEW FUEL STORAGE PIT	
Core storage capacity	1/3
Equivalent fuel assemblies	72
Center-to-center spacing of assemblies, in.	20.5
Maximum K_{eff} with unborated water	0.95
SPENT FUEL STORAGE PIT	
Equivalent fuel assemblies[1]	1376
Number of space accommodations for failed fuel cans	2
Number of space accommodations for spent fuel shipping cask	1
Center-to-center spacing of Regions 1-1,1-2 assembly storage cells, in	10.545(N-S) 10.765(E-W)
Center-to-center spacing of Regions 2-1, 2-2 assembly storage cells, in	9.04
Maximum K_{eff} with borated water (Regions 1-1, 1-2 and Regions 2-1, 2-2)	≤0.95
Maximum K_{eff} with unborated water (Regions 1-1, 1-2 and Regions 2-1, 2-2)	<1.0
MISCELLANEOUS DETAILS	
Width of refueling canal, ft	3
Wall thickness for spent fuel storage pit, ft	3 to 6
Weight of fuel assembly with rod cluster control (dry), lb	1,580
Quantity of water required for refueling, gal	300,000

Notes:
1. After reracking.

9.5 FIGURES

9.6 FACILITY SERVICE SYSTEMS

9.6.1 Service Water System

9.6.1.1 Design Basis

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

9.6.1.2 System Design and Operation

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

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

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

The service water pumps can therefore obtain water through four separate intakes each equipped with means to prevent debris from entering the pumps, and each capable of supplying all the water required for the service water pumps. Electric heaters are provided in the traveling screens 27 and 28 to prevent icing of the screens. Even if the main circulating pump intake

were 90-percent blocked, that intake alone would be capable of supplying all water required for the service water pumps at design conditions.

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

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

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

Following a simultaneous incident and loss of offsite power, the cooling water requirements for all five fan cooling units and the other essential loads can be supplied by any two of the three service water pumps on the header designated to supply the nuclear and essential secondary load supply lines. Any two of these three pumps can be powered by the emergency diesels as described in Chapter 8. These emergency powered pumps are those necessary and sufficient to meet blackout and emergency conditions. Either one of the two sets of three pumps can be placed on the diesel starting logic.

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

During normal plant operation, flow through the cooling units will normally be throttled for containment temperature control purposes by a valve on the common discharge header from the cooling units. Two independent, full-flow isolation valves open automatically in the event of a safety injection signal to bypass the control valve. Both valves fail in the open position upon loss of air pressure and either valve is capable of passing the full flow required for all five fan cooling units for accident mitigation. An 18-in. bypass line has been installed around the flow control valves in the service water return line from the fan cooler units. The line containing a flow limiting orifice and a butterfly valve can provide manual control for optimal service water flow rate through the fan cooler units during normal plant operation. Should there be a failure in the piping or valves at the header supplying water to the containment cooling coils, one of the two series header isolation valves in the center of the header can be manually closed and service will continue on the side of the header opposite the failure. The supply line attached to

this side of the header now supplies the essential loads, whether or not it did so before the failure.

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

Following a simultaneous incident and blackout, the component cooling heat exchangers are not needed during the injection phase: thus they are normally fed from the nonessential supply header. At the beginning of the recirculation phase at least one of the service water pumps on the nonessential header is manually started to supply at least 2500 gpm of service water to each of the component cooling heat exchangers.

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

9.6.1.3 Design Evaluation

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

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

In response the NRC Generic Letter 96-06, the containment fan cooler units and their associated service water piping were evaluated for susceptibility to waterhammer or two-phase flow. In the event of a loss of offsite power, the flow of essential service water will be interrupted until the emergency diesel generators start and restore power to the essential service water pumps. The pressure in the cooling coils and service water piping will drop to subatmospheric and a vapor pocket will form in the region of the fan coolers. When the essential service water pumps restart, the pocket will close and a water hammer will occur. The magnitude of waterhammer is approximately 394 psig. Dynamic analysis of the piping and supports shows that stresses meet the criteria for upset and faulted conditions, respectively.

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

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

9.6.1.4 Tests and Inspections

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

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

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

Electrical components of the service water system are tested periodically.

9.6.2 Fire Protection System

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

This criterion (GDC 3, 10 CFR 50 Appendix A) represents a revised design basis for the Indian Point Unit 2 fire protection system as was established for the original plant design and initial license application. In 1976 at the request of the NRC, Con Edison initiated a review and evaluation of the station fire protection system to this new criterion; modifications were subsequently proposed by Con Edison to the overall fire protection program. On January 31, 1979, the NRC approved the Indian Point Unit 2 overall fire protection program as providing additional assurance that safe shutdown can be accomplished and that the plant can be maintained in a safe condition during and following potential fire situations. This NRC approval was made as Amendment No. 46 to the facility operating license.

Additional fire protection regulations were issued in 10 CFR 50.48 and Appendix R to Part 50 on November 19, 1980, with an effective date of February 17, 1981. These regulations established requirements for utilities to implement a fire protection program, and backfitted certain requirements in Appendix R to all utilities. For Con Edison, these included the various separation and protection requirements contained within Section III.G, emergency lighting requirements as stipulated by III.J, and oil collection system requirements for reactor coolant pumps as contained in Section III.O. Additionally, Section III.L established performance requirements for alternative shutdown systems. Subsequent to the regulations established in 10 CFR 50.48, various NRC generic letters and guidance documents have been issued to provide clarification of the Appendix R requirements.

Reviews and evaluations of the station fire protection program and systems were performed to demonstrate conformance with the appropriate Appendix R requirements and guidance documents. A description of these reviews, the results, and subsequent NRC approvals are contained in the documents under separate cover entitled:

- IPEC Fire Protection Program Plan
- IP2 Fire Hazards Analysis
- IP2 10 CFR 50, Appendix R Safe-Shutdown Separation Analysis

These documents provide a complete description of the station fire protection program, including the definition of fire areas and barriers, descriptions of fire detection and protection systems, and the credited post-fire safe-shutdown capability, including the alternate safe-shutdown system. The IPEC Fire Protection Program Plan, IP2 Fire Hazards Analysis, and the IP2 10 CFR 50, Appendix R Safe-Shutdown Separation Analysis are considered to be part of the UFSAR by their reference herein. ***[Note: This information was relocated into the UFSAR from the Technical Specifications by NRC SER for TS Amendment 186. The requirements of the Tech Spec per the NRC Order were relocated to the UFSAR and need to remain in the UFSAR as a License Condition.]***

9.6.3 <u>City Water System</u>

The functions of the city water system are:

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

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

1. Emergency makeup to the house service boilers.
2. Cooling the house service boiler water samples.
3. General usage at the house service boilers.
4. Makeup to the expansion tank of the conventional plant closed cooling system.
5. Cooling and general usage at the steam and water analysis station.

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

1. Makeup to the expansion tanks of the diesel-generator jacket water cooling system.
2. Emergency feed to the auxiliary boiler feed pumps.
3. Makeup to the expansion tank of the instrument air compressor closed cooling system.

4. General usage via hose connections inside the primary auxiliary building and waste holdup tank pit.
5. Emergency makeup to the isolation valve seal-water supply tank.
6. Spray water to the steam-generator blowdown tank.

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

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

9.6.4 Compressed Air Systems

9.6.4.1 Instrument Air System

The instrument air system is designed such that the instrument air shall be available under all operating conditions; all essential systems requiring air during or after an accident shall be self supporting; all controls shall fail to a safe position on loss of power; and, after an accident, the air system shall be re-established. The system is shown in Plant Drawing 9321-2036 [Formerly UFSAR Figure 9.6-6].

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

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

The instrument air compressors may be operated in two modes. One mode provides for the compressors to be in standby and to come on automatically in the event of low pressure in the common air receiver. During this mode, air is supplied by the station air system. The other mode of operation provides for simultaneous running of both compressors in order to provide continuity of service to Class I areas in the event of outage of the conventional plant instrument air header. A restriction orifice is provided so as to limit the flow to the capacity of one instrument air compressor into a possible line break in the secondary plant air header.

Upon notification of this break, a valve is provided to isolate the secondary plant and prevent pressure decay in the primary plant header. Valving has been installed to provide flexible operations as related to the alternate station air supply and to maintain proper isolation capabilities.

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

9.6.4.2 Station Air System

The station air system shown in Plant Drawing 9321-2035 [Formerly UFSAR Figure 9.6-7] is supplied by a Worthington Corporation two-stage 650-scfm compressor located in the turbine building. The air is discharged through an aftercooler and moisture separator at 100 psig and 110°F. The maximum discharge pressure will be 125 psig. The cooling water for the aftercooler and compressor jacket is supplied from a closed cooling water system, which contains treated city water.

The compressor is controlled by the solenoid unloader valves, which are energized through a pressure switch arrangement in automatic or hand (manual) modes. In the automatic mode, the compressor will run in single- or two-stage operation and unload at a predetermined pressure setting with motor and compressor stopped. In manual mode, the compressor runs continuously and is loaded and unloaded at predetermined pressure settings. High-water and high-air temperature switches are connected to the control annunciator.

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

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

9.6.5 Heating System

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

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

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

Provision is made for the following heating services:

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

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

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

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

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

9.6.6 Plant Communications Systems

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

REFERENCES FOR SECTION 9.6

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

TABLE 9.6-1

Minimum Essential Service Water Requirement Under Accident Conditions

Service	Flow each (gpm)	Number	Total Flow (gpm)
Containment Recirculation Fan Coolers	1600	5	8000
Containment Recirculation Fan Coolers Motors	17	5	85
Emergency Diesel Generators	400	3	1200
Instrument Air Compressor Heat Exchangers	65	2	65
Radiation Monitor Sample Coolers	10	3	30
Service Water Pump Strainer Blowdown	100	3	300 (750)$_1$

Minimum Non Essential Service Water Requirements Post LOCA Recirculation

Service	Flow each (gpm)	Number	Total Flow (gpm)

Component Cooling Water Heat Exchangers	2500	2	5000
Service Water Pump Strainer Blowdown	100	3	300 (750)[1]

Note:
1. Each strainer is mechanically set for 225 ± 25 gpm backflush flow, 750 gpm total (max).

9.6 FIGURES

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

9.7 EQUIPMENT AND SYSTEM DECONTAMINATION

9.7.1 Design Basis

Activity outside the core can result from fission products from defective fuel elements, fission products from tramp uranium left on the cladding in small quantities during fabrication, products of n-γ or n-p reactions on the water or impurities in the water, and activated corrosion products. Fission products in the reactor coolant associated with normal plant operation and tramp uranium are generally removed with the coolant or in subsequent flushing of the system being decontaminated. The products of water activation are not long lived and may be removed by natural decay during reactor cool-down and subsequent flushing procedures. Activated corrosion products are the primary source of the remaining activity.

The corrosion products contain radioisotopes from the reactor coolant, which have been absorbed on or have diffused into the oxide film. The oxide film, essentially magnetite (Fe_30_4) with oxides of other metals including Cr and Ni, can be removed by chemical means presently used in industry.

Water from the primary coolant system and the spent fuel pit is the primary potential source of contamination outside of the corrosion film of the primary coolant system components. The contamination can be spread by various means when access is required. Contact while working on primary system components can result in contamination of the equipment, tools and clothing

of the personnel involved in the maintenance. Also, leakage from the system during operation or spillage during maintenance can contaminate the immediate areas and contribute to the contamination of the equipment, tools, and clothing.

9.7.2 Methods of Decontamination

Surface contaminates, which are found on equipment in the primary system and the spent fuel pit that are in contact with the water are removed by conventional techniques of flushing and scrubbing as required. Tools are decontaminated by flushing and scrubbing since the contaminates are generally on the surface only of nonporous materials. Personnel and their clothing are decontaminated according to the standard health physics requirements.

Those areas of the plant, which are susceptible to spillage of radioactive fluids are painted with a sealant to facilitate decontamination that may be required. Generally washing and flushing of the surface are sufficient to remove any radioactivity present.

The corrosion films generally are tightly adhering surface contaminates, and must be removed by chemical processes. The removal of these films is generally done with the aid of commercial vendors who provide both services and formulations. Since decontamination experience with reactors is continually being gained, specific procedures may change for each decontamination case.

Portable components and tools can be cleaned by the use of a liquid abrasive bead decontamination unit, an ultrasonic unit, a sandblast unit or a Freon degreaser unit installed in Unit 1.

9.7.3 Decontamination Facilities

Decontamination facilities onsite consist of an equipment pit and a cask pit located adjacent to the spent fuel storage pit. In the stainless steel-lined equipment pit, fuel handling tools and other tools can be cleaned and decontaminated.

In the cask decontamination pit, the outside surfaces of the shipping casks are decontaminated, if required, by using steam, water detergent solutions, and manual scrubbing to the extent required. When the outside of the casks are decontaminated, the casks are removed by the auxiliary building crane and hauled away.

For the personnel, a decontamination shower and washroom is located adjacent to the radiation control area locker room. Personnel decontamination kits with instructions for their use are in the radiation control area locker room.

9.8 PRIMARY AUXILIARY BUILDING VENTILATION SYSTEM

9.8.1 Design Basis

The primary auxiliary building ventilation system is designed to accomplish the following:

1. Provide sufficient circulation of filtered air through the various rooms and compartments of the building to remove equipment heat and maintain safe ambient operating temperatures.

2. Control flow direction of airborne radioactivity from low activity areas toward higher activity areas and through monitored exhaust paths.
3. Provide purging of the building to the plant vent for dispersion to the environment.

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

9.8.2 System Design and Operation

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

1. Makeup air handling system complete with fan, filters, heating coils, and supply ductwork.
2. Exhaust system complete with fans, ductwork, roughing filters, HEPA filters, and charcoal filters.
3. Outside air intake for the waste storage tank pit area.

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

Branch supply ducts direct makeup air to the various floors at the east end of the building, from where it flows to the rooms and compartments. Air is exhausted from each of the building compartments through ductwork designed to make the supply air sweep across the room as it travels to the room exhaust register. The air then flows to the exhaust fan inlet plenum, and is drawn by the operating exhaust fan through roughing filters, HEPA filters, and charcoal filters before discharge to the plant vent. The exhaust system has been designed to ensure that air flows from the "clean" end of the building through the "hot" areas.

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

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

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

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

TABLE 9.8-1
Primary Auxiliary Building Ventilation System Component Data

System	Units Installed	Units Capacity	Units Required for Normal Operation
Exhaust[1]			
Fans, standard conditions	2	55,500 cfm	1
Fan pressure	-	10.3 in. H_2O	-
Fan motors	2	125 hp	1
Plenums	2	55,500 cfm	1
Roughing filters	2	55,500 cfm	1
HEPA filters	2	55,500 cfm	1
Carbon Filters	1	55,500 cfm	1
Supply Tempering Unit (Primary Auxiliary Building)			
Fans, standard conditions	1	50,400 cfm	1
Fan pressure	1	2.5-in. H_2O	-
Fan motor	1	50 hp	1
Filters	1	50,400 cfm	1
Coils	1	50,400 cfm	1
Outside Air Intake (Waste Storage Tank Pit Area)	1	5100 cfm	1[2]

Notes:
1. These two exhaust fans are used interchangeably and/or as backup for: (1) ventilation of primary auxiliary building, (2) containment building purge system.
2. Outside Air Intake may be covered during cold weather conditions.

9.9 CONTROL ROOM VENTILATION SYSTEM

9.9.1 Design Basis

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

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

9.9.2 System Design and Operation

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

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

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

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

1. Normal Conditions

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

2. Incident Conditions

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

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

All these operations can be performed manually from the control room. However, in the event of a safety injection signal and/or high radiation signal, the control room dampers will automatically reposition and start the booster fan to place the charcoal filter unit in service, for system operating mode 2. A redundant toxic chemical and radiation monitor for central control room air intakes has been installed.

For additional discussion of this system, see Section 7.2.

9.9 FIGURES

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

9.10 FUEL STORAGE BUILDING VENTILATION SYSTEM

9.10.1 Design Basis

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

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

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

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

4. Remove normal building heat.

9.10.2 System Design And Operation

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

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

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

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

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

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

As a result of IP2 Operating License Amendment No. 229 (dated June 5, 2002), the limiting conditions for operation and the surveillance requirements for the fuel storage building air

filtration system were relocated from the Technical Specifications to the UFSAR. These relocated requirements have been modified to reflect the assumptions used for the fuel handling accidents approved by the Technical Specification Amendment 211 (July 27, 2000). These are contained in UFSAR Sections 9.10.3 and 9.10.4 below.

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

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

9.10.4 Surveillance Requirements (Fuel Storage Building Air Filtration System)

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

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

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

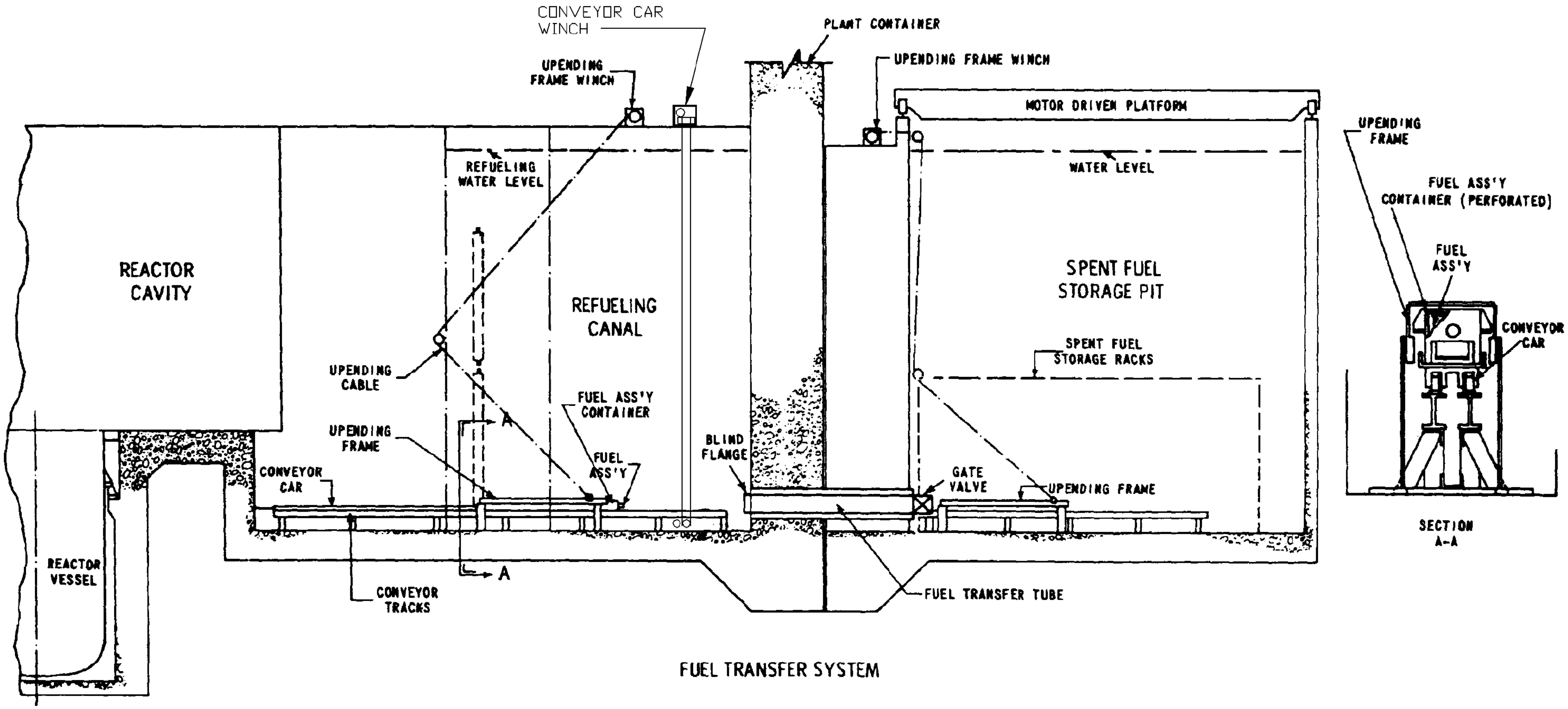
3. A visual inspection of the normal atmosphere cleanup system and all associated components should be performed in accordance with Section 5 of ASME N510-1989.

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

5. In-place leak testing, in accordance with Section 11 of ASME N510-1989, for adsorbers should be performed: at least once each 24 months; following removal of an adsorber sample for laboratory testing if the integrity of the adsorber section is affected; after each partial or complete replacement of carbon adsorber in an adsorber section; following detection of, or evidence of, penetration or intrusion of water or other material into any

portion of a normal atmosphere cleanup system that may have an adverse effect on the functional capability of the adsorbers; and, following painting, fire, or chemical release in any ventilation zone communicating with the system that may have an adverse effect on the functional capability of the system. The leak test should confirm a combined penetration and leakage (or bypass) of the adsorber section of 0.05% or less of the challenge gas at rated flow ±10%.

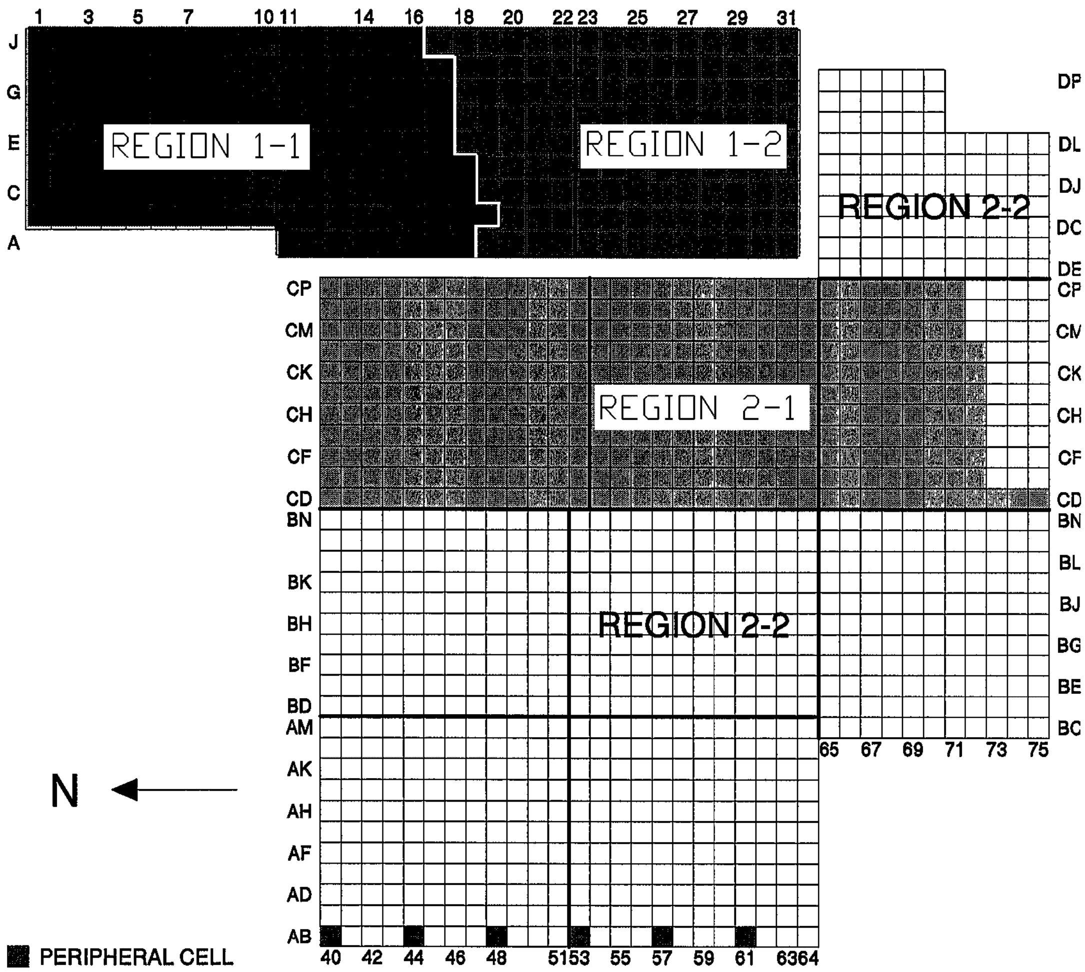
6. The efficiency of the activated carbon adsorber section should be determined by laboratory testing of representative samples of the activated carbon exposed simultaneously to the same service conditions as the adsorber section in accordance with ASTM D3803-1989 at a face velocity of 50 ft/min, a temperature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



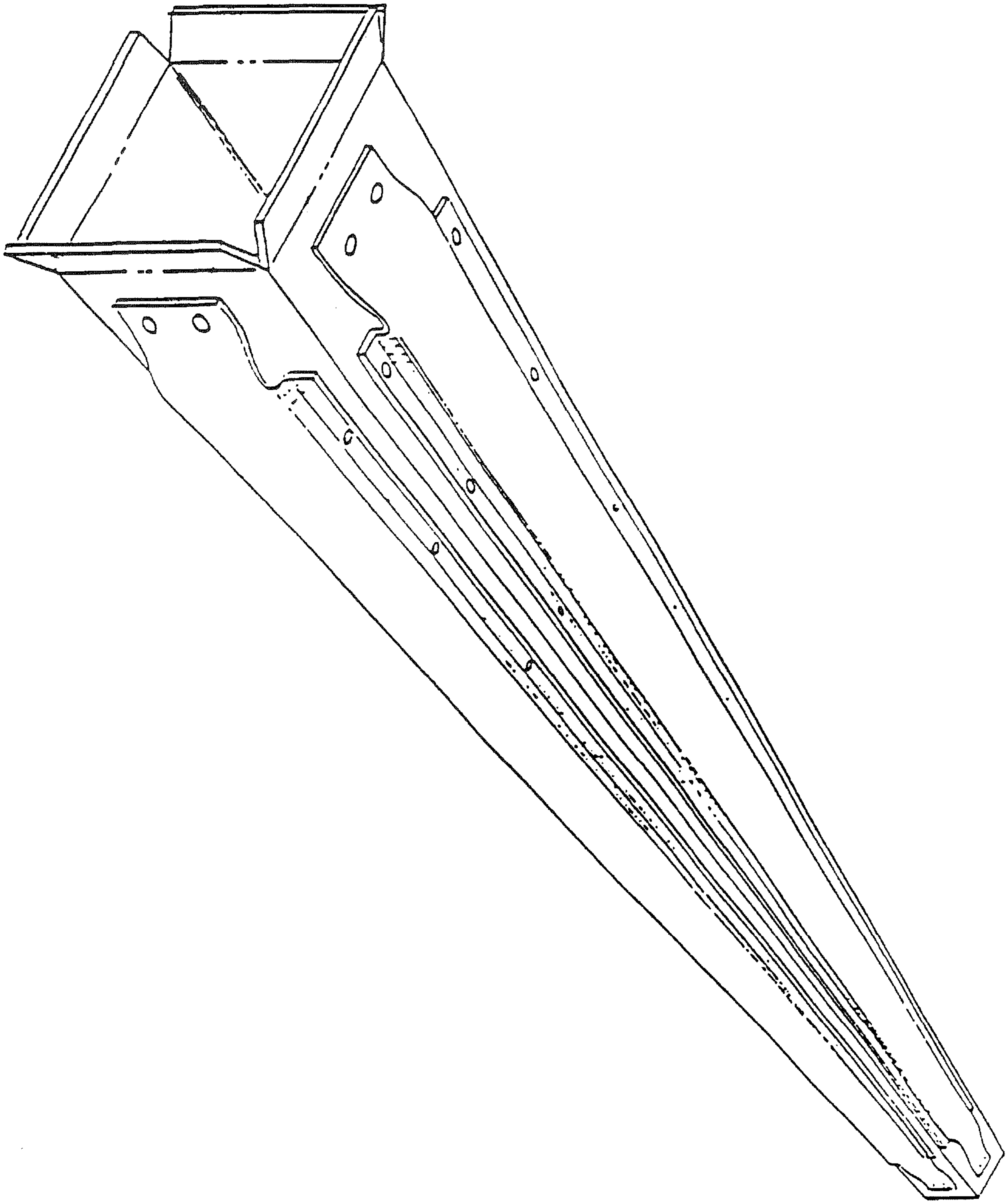
FUEL TRANSFER SYSTEM

INDIAN POINT UNIT No. 2	
UFSAR FIGURE 9.5-1 FUEL TRANSFER SYSTEM	
MIC. No. 1999MC3886	REV. No. 17B

OAGI0000215_1187

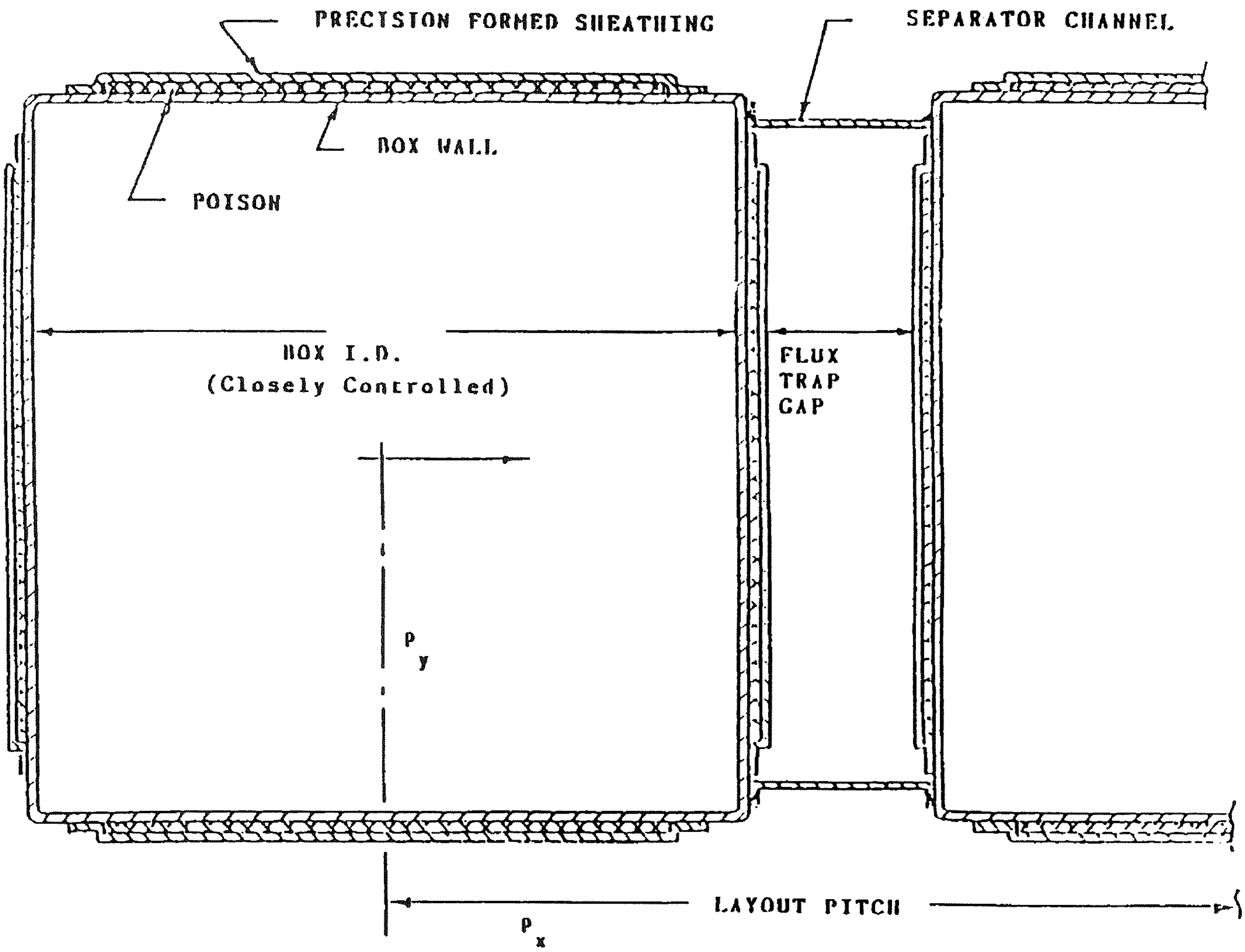


INDIAN POINT UNIT No. 2	
UFSAR FIGURE 9.5-2 SPENT FUEL STORAGE RACK LAYOUT	
MIC. No. 1999MC3887	REV. No. 17B



INDIAN POINT UNIT No. 2
UFSAR FIGURE 9.5-3 SPENT FUEL STORAGE CELL REGION I

MIC. No. 1999MC3888	REV. No. 17A

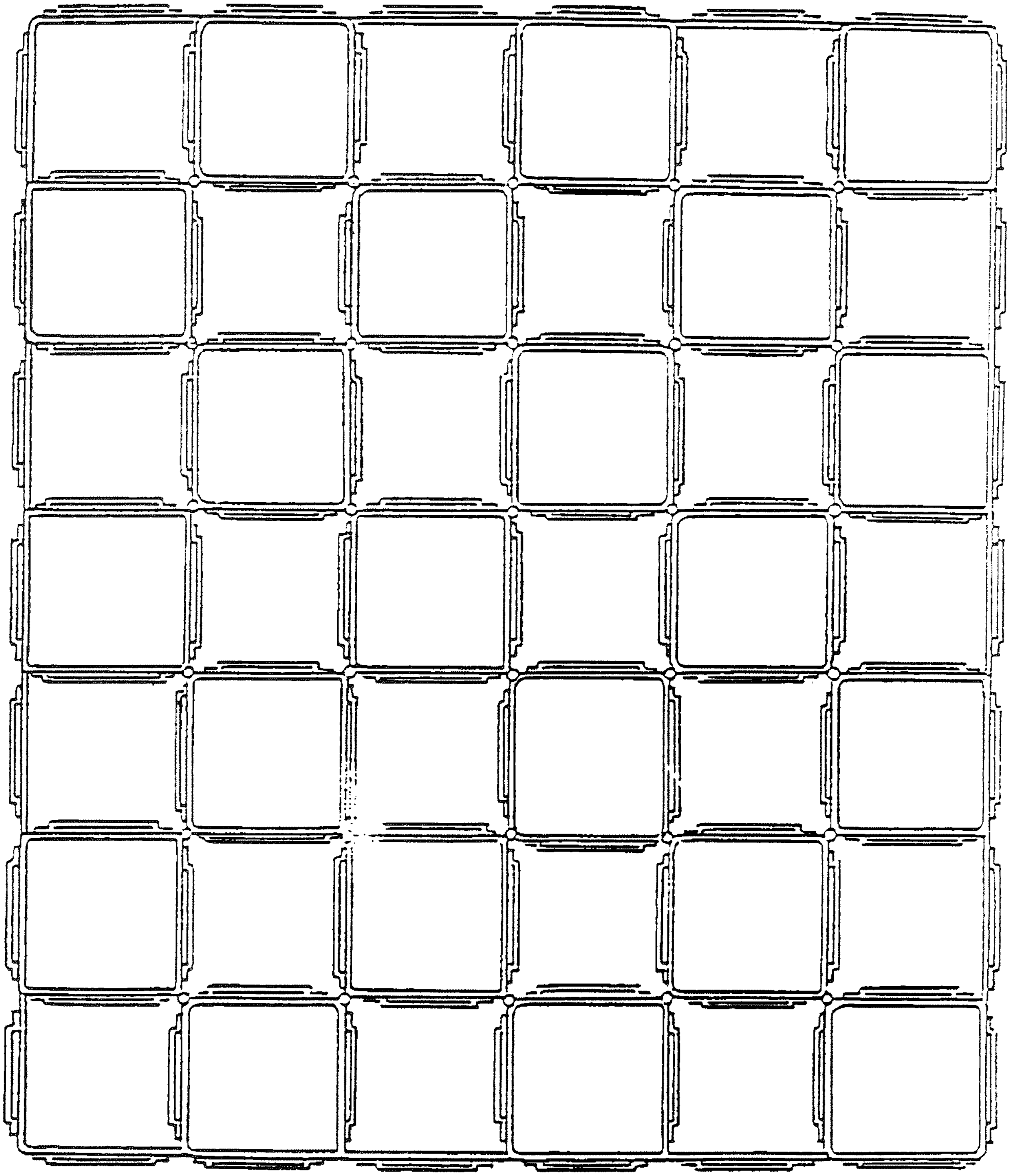


INDIAN POINT UNIT No. 2

UFSAR FIGURE 9.5-4

REGION I CELL
CROSS SECTION

MIC. No. 1999MC3889	REV. No. 17A



INDIAN POINT UNIT No. 2
UFSAR FIGURE 9.5-5
REGION II CROSS SECTION

MIC. No. 1999MC3890	REV. No. 17A

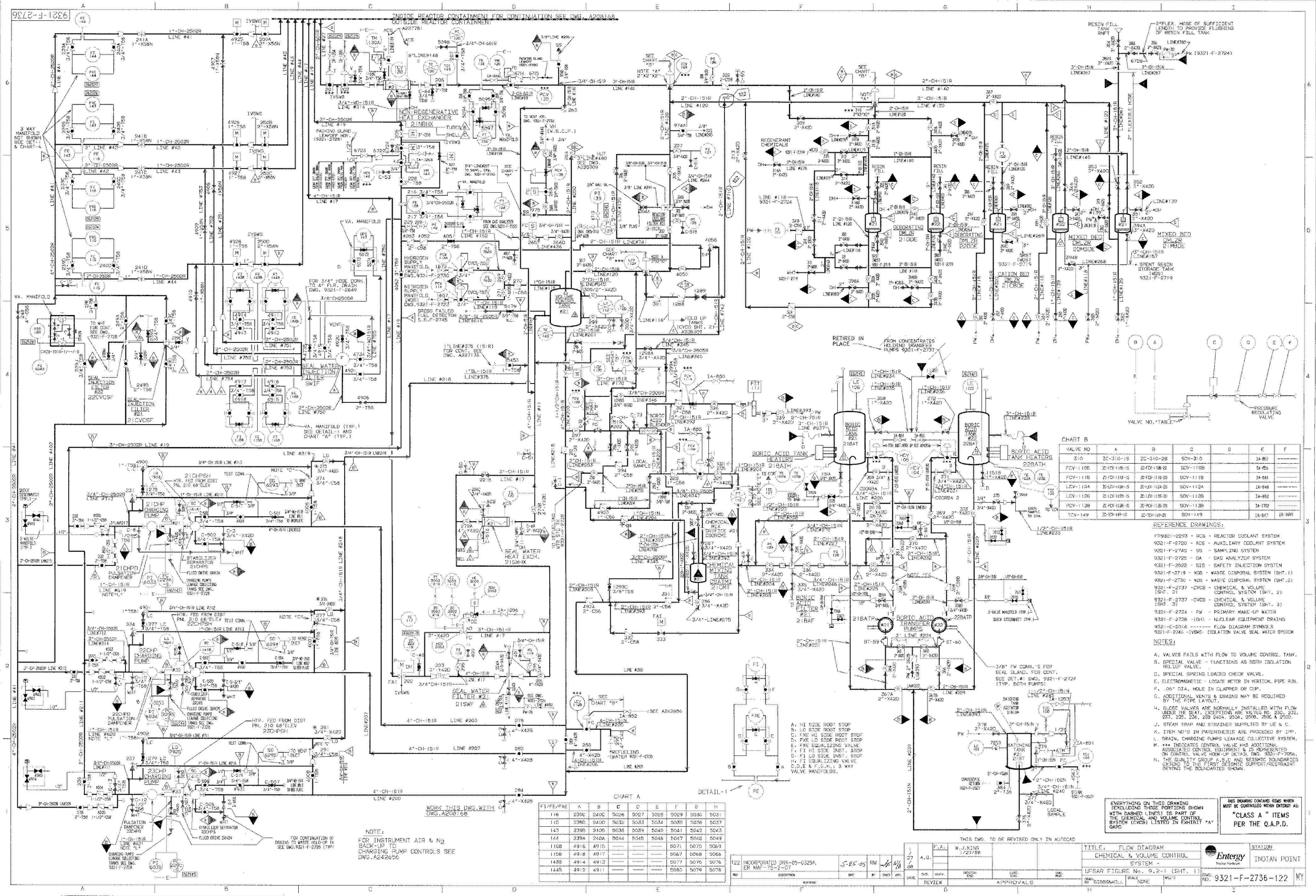


CHART A

FI/FE/FXE	A	B	C	D	E	F	G	H
116	239C	240C	5026	5027	5028	5029	5030	5031
115	239D	240D	5032	5033	5034	5035	5036	5037
143	239B	240B	5038	5039	5040	5041	5042	5043
144	239A	240A	5044	5045	5046	5047	5048	5049
116B	4916	4915	———	———	———	5071	5070	5069
115B	4918	4917	———	———	———	5067	5068	5066
143B	4914	4913	———	———	———	5077	5076	5075
144B	4912	4911	———	———	———	5080	5079	5078

CHART B

VALVE NO	A	B	C	D	E	F
310	ZC-310-1S	ZC-310-2S	SOV-310		IA-857	———
FCV-110B	ZC-FCV-110B-1S	ZC-FCV-110B-2S	SOV-110B		IA-856	———
FCV-111B	ZC-FCV-111B-1S	ZC-FCV-111B-2S	SOV-111B		IA-849	———
LCV-112A	ZC-LCV-112A-1S	ZC-LCV-112A-2S	SOV-112A		IA-848	———
LCV-112B	ZC-LCV-112B-1S	ZC-LCV-112B-2S	SOV-112B		IA-852	———
PCV-113B	ZC-PCV-113B-1S	ZC-PCV-113B-2S	SOV-113B		IA-1702	———
TCV-149	ZC-TCV-149-1S	ZC-TCV-149-2S	SOV-149		IA-847	IA-1699

REFERENCE DRAWINGS:

FP9321-2293 - RCS - REACTOR COOLANT SYSTEM
9321-F-2720 - ACS - AUXILIARY COOLANT SYSTEM
9321-F-2745 - SS - SAMPLING SYSTEM
9321-F-2725 - GA - GAS ANALYZER SYSTEM
9321-F-2522 - SIS - SAFETY INJECTION SYSTEM
9321-F-2719 - WDS - WASTE DISPOSAL SYSTEM (SHT.1)
9321-F-2730 - WDS - WASTE DISPOSAL SYSTEM (SHT.2)
9321-F-2737 -CVCS - CHEMICAL & VOLUME CONTROL SYSTEM (SHT. 2)
(SHT. 2)
9321-F-2737 -CVCS - CHEMICAL & VOLUME CONTROL SYSTEM (SHT. 3)
(SHT. 2)
9321-F-2724 - PW - PRIMARY MAKE-UP WATER
9321-F-2728 -(DH) - NUCLEAR EQUIPMENT DRAINS
9321-C-2016 ------- FLOW DIAGRAM SYMBOLS
9321-F-2746 -IVSWS- ISOLATION VALVE SEAL WATER SYSTEM

NOTES:

A. VALVES FAILS WITH FLOW TO VOLUME CONTROL TANK.
B. SPECIAL VALVE - FUNCTIONS AS BOTH ISOLATION RELIEF VALVE.
C. SPECIAL SPRING LOADED CHECK VALVE.
E. ELECTROMAGNETIC - LOCATE METER IN VERTICAL PIPE RUN.
F. .06" DIA. HOLE IN CLAPPER OR CUP.
G. ADDITIONAL VENTS & DRAINS MAY BE REQUIRED BY THE PIPE LAYOUT.
H. GLOBE VALVES ARE NORMALLY INSTALLED WITH FLOW UNDER THE SEAT. EXCEPTIONS ARE VALVES NO. 230, 232, 233, 235, 236, 238 240A, 250A, 250B, 250C & 250D.
J. STEAM TRAP AND STRAINER SUPPLIED BY UE & C.
K. ITEM NO'S IN PARENTHESIS ARE PROCEDED BY IPP.
L. DRAIN, CHARGING PUMPS LEAKAGE COLLECTIVE SYSTEM.
M. *** INDICATES CONTROL VALVE HAS ADDITIONAL ASSOCIATED CONTROL EQUIPMENT & IS REPRESENTED ON CONTROL VALVE HOOK-UP DETAIL DWG. 9321-F-7056.
N. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

EVERYTHING ON THIS DRAWING (EXCLUDING THOSE PORTIONS SHOWN WITH DASHED LINES) IS PART OF THE CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) LISTED IN EXHIBIT "A" QAPD

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS:
"CLASS A " ITEMS PER THE Q.A.P.D.

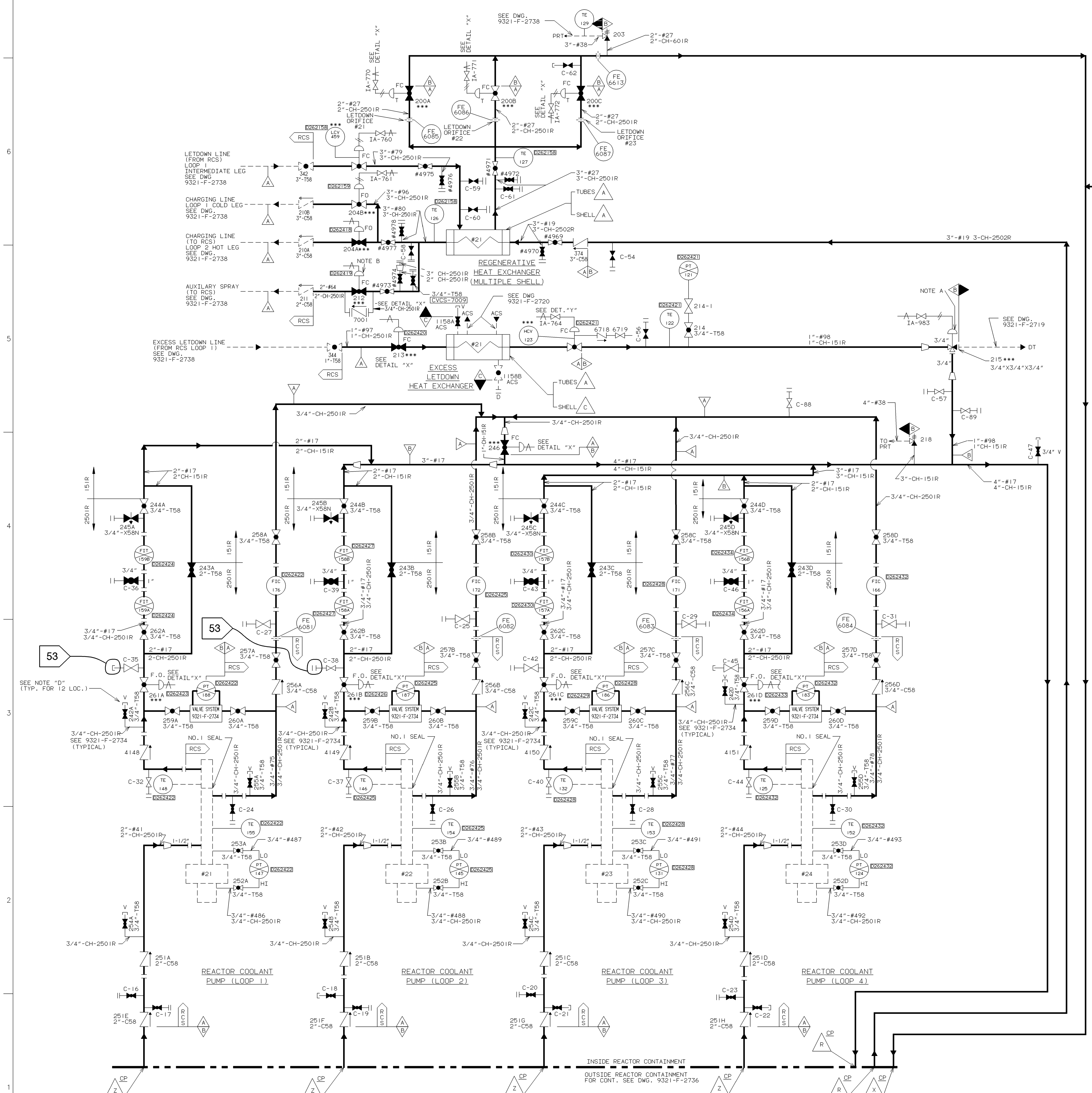
THIS DWG. TO BE REVISED ONLY IN AUTOCAD

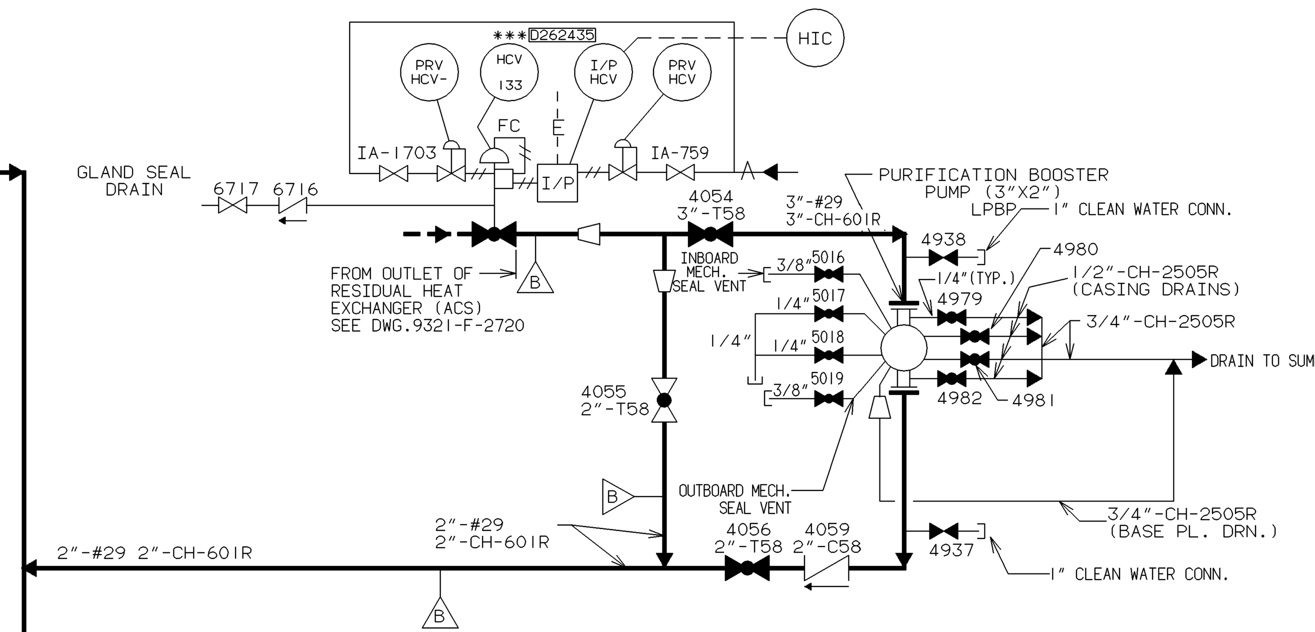
REV	DESCRIPTION	DATE	BY	CHK'D	APP.
122	INCORPORATED DRN-05-03254, ER MAF-75-2-07	5-25-05	RM		N/A

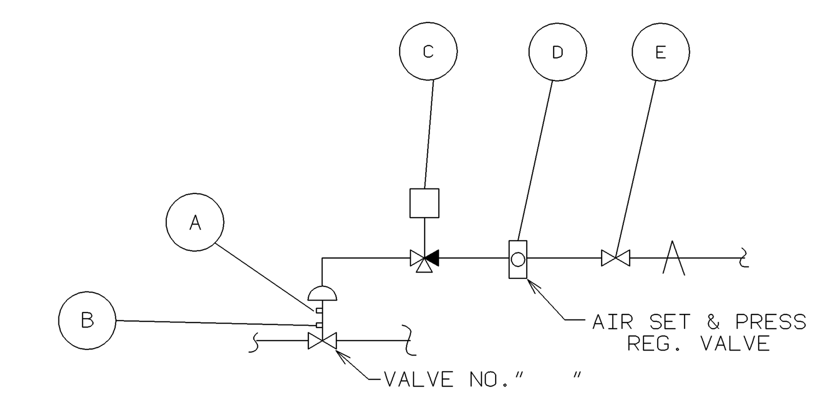
REVISIONS

DATE 1/27/88 | CKD. A.G. | SUPV. | F.A. | DESIGN ENG. W.J.KING 1/27/88 | CSSC. ENG. | ENG. MGR.
REVIEW | APPROVALS

TITLE: FLOW DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM -
UFSAR FIGURE No. 9.2-1 (SHT. 1)
DRWN. BY GIBBS&HILL | SCALE NONE
Entergy Nuclear Northeast
STATION INDIAN POINT
DWG. NO. 9321-F-2736-122 MY

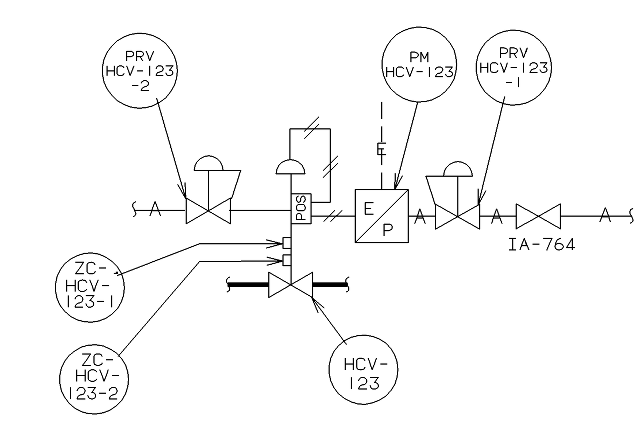






VALVE NO	A	B	C	D	E
246	ZC-246-1	ZC-246-2	SOV-246	PRV-246	IA-773
261A	ZC-261A-1S	ZC-261A-2S	SOV-261A	PRV-261A	IA-766
261B	ZC-261B-1S	ZC-261B-2S	SOV-261B	PRV-261B	IA-767
261C	ZC-261C-1S	ZC-261C-2S	SOV-261C	PRV-261C	IA-768
261D	ZC-261D-1S	ZC-261D-2S	SOV-261D	PRV-261D	IA-769
212	ZC-212-1S	ZC-212-2S	SOV-212	PRV-212	IA-1312
213	ZC-213-1S	ZC-213-2S	SOV-213	PRV-213	IA-765
200A	ZC-200A-1	ZC-200A-2	SOV-200A	PRV-200A	IA-770
200B	ZC-200B-1	ZC-200B-2	SOV-200B	PRV-200B	IA-771
200C	ZC-200C-1	ZC-200C-2	SOV-200C	PRV-200C	IA-772
215	ZC-215-1	ZC-215-2	SOV-215	PRV-215	IA-983
204A	ZC-204A-1	ZC-204A-2	SOV-204A	PRV-204A	IA-762
204B	ZC-204B-1	ZC-204B-2	SOV-204B	PRV-204B	IA-761
LCV-459	ZC-LCV-459-1	ZC-LCV-459-2	SOV-459	PRV-LCV-459	IA-760

DETAIL "X"



DETAIL "Y"

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS:
"CLASS A" ITEMS
PER THE QAPD

WORK THIS DRAWING WITH DWG. 9321-F-2734 AND DWG. 9321-F-2736

NOTES

A. VALVE FAILS WITH FLOW TO VOLUME CONTROL TANK
B. SPECIAL VALVE-FUNCTIONS AS BOTH ISOLATION & RELIEF VALVE.
C. *** INDICATES CONTROL VALVE HAS ADDITIONAL ASSOCIATED CONTROL EQUIPMENT & IS REPRESENTED ON CONTROL VALVE HOOK-UP DETAIL DRAWING 9321-F-7056.
D. FOR CONTINUATION SEE DWG. 9321-F-2734 (TYP. FOR 12 LOC.)
E. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

REF. DWG.

9321-C-2016	FLOW DIAGRAM SYMBOLS
9321-F-2538	CONTAINMENT BLDG PRIMARY COOLANT PRESSURIZER PIPING PLAN (SOUTH HALF)
9321-F-2734	PIPING AT REACTOR COOLANT PUMPS

EVERYTHING ON THIS DRAWING (EXCLUDING THOSE PORTIONS WITH DASHED LINES) IS PART OF THE CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) LISTED IN EXHIBIT A, CI-240-1 EXCEPT AS SPECIFICALLY INDICATED HEREIN
RCS REACTOR COOLANT SYSTEM

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

REVISION
USE .1 SIZE TEXT ONLY
G. BHALLA 12/10/02 53
THIS REVISION IS CLASS "A" PER THE QAPD. UPDATED DWG. PER CR #IP-2-2001-10397 AND FIELD VERIFICATION. CHANGED PLUG ON VA. C-38 TO FLG AND FLG ON VA. C-35 TO PLUG.
RELEASED AS CONSTRUCTED P.N. 69901-AF D.B./G.B.

REVISION SIGNATURES		
REV	DES	ENG
53		G. BHALLA 12/10/02

DWG. NO. A208168-53
DWG. SIZE A | DWG. TYPE COMPANY
CON EDISON | STATION INDIAN POINT
BORO: WESTCHESTER
TITLE: FLOW DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM
UFSAR FIGURE No. 9.2-1 (SHT. 2)

APPROVALS
ENGINEERING
ENGINEER MANAGER:
DISCIPLINE ENGINEER:
DESIGN
DESIGN MANAGER: W.J. KING 11-6-87
DESIGN SUPERVISOR: F.A. 8-17-90
DRAWN BY: MARZULLO 8-17-90
DESIGN CHECKER: L.H. 8-17-90
SCALE: NONE | DISCIPLINE CODE: MY

208168-53

REFERENCE DWGS:
U.E.& C.DWG.NO.
9321-F-2738------------------RCS REACTOR COOLANT SYSTEM
9321-F-2724------------------PRIMARY MAKE-UP WATER
9321-F-2720------------------ACS-AUXILIARY COOLANT SYSTEM
9321-F-2727------------------NUCLEAR EQUIP.VENT SYS.SS SAMPLING SYS.
9321-F-2016------------------FLOW DIAGRAM SYMBOLS
9321-H-2739------------------TEMP.STRAINER TABULATION
9321-F-2735------------------SIS SAFETY INJECTION SYSTEM
9321-F-2027------------------AUXILIARY STEAM SYSTEM
9321-F-2719------------------WDS WASTE DISPOSAL SYS.(SH.1)
9321-F-2730------------------WDS WASTE DISPOSAL SYS.(SH.2)
9321-F-2723------------------NITROGEN
9321-F-2736------------------CVCS-CHEMICAL & VOLUME CONT.SYS.(SH1)
9321-F-2725------(GA)--------GAS ANALYZER SYSTEM
9321-F-2728------(DH)--------NUCLEAR EQUIPMENT DRAINS

NOTES:

A. DIAPHRAGM SEAL.
B. LOOP SEAL SHOULD EXTEND ONE FOOT BELOW AND ONE FOOT ABOVE OUTLET NOZZLE.
C. SOLENOID VALVE ACTUATED BY GAS ANALYZER CHANNEL (LATER)
D. ADDITIONAL VENTS & DRAINS MAY BE REQUIRED BY THE PIPE RUN
E. GLOBE VALVES ARE NORMALLY INSTALLED WITH FLOW UNDER THE SEAT. EXCEPTIONS ARE NC.(LATER)
F. STEAM TRAP ASSY.BY U.E.&C. TRAP BY VENDOR F.P.933FD
G. TANK IS SUPPLIED WITH DIAPHRAM SEAL.
H. VOL CONTROL TK. RELIEF LINE SHOULD TAP IN CLOSE TO TANK #21.
I. SEE BELOW:
J. *** INDICATES CONTROL VALVE HAS ADDITIONAL ASSOCIATED CONTROL EQUIPMENT & IS REPRESENTED ON CONTROL VALVE HOOK-UP DETAIL DWG. 9321-F-7056.
K. ALL HEAT TRACE CIRCUITS ON THIS DRAWING ARE DISCONNECTED.
L. STRAINER INTERNALS HAVE BEEN REMOVED.

REFERENCES:

1. PROCESS FLOW DIAGRAM CVCS (SH.2) (W) E.C.DWG.540F892
2. PROCESS FLOW DIAGRAM CVCS (SH.3) (W) E.C.DWG.540F893
3. DEFINITIONS OF SYMBOLS (W) E SPEC G675176 REV.2 AND (W) E SPEC G675164 REV.0
4. INSTALLATION OF INSTR. (W) PROC.SPEC CAP 294367 REV.1
5. MATR.SPEC PIPE & FITTINGS (W) E.SPEC G569866 REV.2 AND (W) E.SPEC G676398 REV.0

EXCEPT WHERE NOTED,
EVERYTHING ON THIS DRAWING (EXCLUDING THOSE PORTIONS SHOWN WITH DASHED LINES) IS PART OF THE CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) LISTED IN EXHIBIT "A",.CI-240-1

NOTE 1 THE ITEMS SHOWN DASHED WITH IN THIS COMPONENT BOUNDARY ARE PART OF (CVCS)

LEGEND:

SRST - SPENT RESIN STORAGE TANK
VS - VENTILATION SYSTEM

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS A" ITEMS
PER CI-240-1

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

WORK THIS DRAWING WITH DWG. A235309

REVISION									
DES / G. BHALLA 02/09/01 / ENG	THIS REVISION IS NON-CLASS PER CI-240-1 UPDATED DWG. TO SHOW RETIRED BORON RECOVERY EQUIPTMENT PER CRS #199806527. RELEASE AS CONSTRUCTED. P/N 69901-AF D.B./Y.J. 2/09/01	2/8/88	A.G.	F.A.	W.J.KING 2/9/88				
		DATE	CKR.	SUPV.	DESIGN ENG	DISC. ENG.	ENG. MGR.		
			REVIEW		APPROVALS				

TITLE: FLOW DIAGRAM
CHEMICAL & VOLUME CONTROL SYSTEM - SH.NO.2 -
UFSAR FIGURE No. 9.2-1 (SHT. 3)
DRWN. BY GIBBS&HILL | SCALE NONE | REC'D
Con Edison
STATION INDIAN POINT
DWG. NO. 9321-F-2737-53 MY

A235309
RETIRED IN PLACE
REACTOR COOLANT LETDOWN CVCS (VOLUME CONTROL TANK) DWG 9321-F-2736
REACTOR COOLANT DRAIN PUMP
EVAPORATOR CONDENSATE DEMINERALIZERS
CONDENSATE FILTER ITEM NO.22
CONDENSATE FILTER ITEM NO.21
MONITOR TANK #21 RETIRED IN PLACE
MONITOR TANK #22 RETIRED IN PLACE
MONITOR TANK #23 RETIRED IN PLACE
MONITOR TANK PUMPS RETIRED IN PLACE
PRIMARY WATER STORAGE TANK DWG 9321-F-2724
PREVIOUSLY RETIRED IN PLACE
TO WASTE HOLD-UP TANK VIA #261
LINE # 794 TO PRIMARY SAMPLING SYSTEM DWG. 9321-F-2745
UNIT #2 WASTE TRANSFER PUMP DISCH.SEE DWG.A188852
HOLD-UP TANK #21
HOLD-UP TANK #22
HOLD-UP TANK #23
RECIRCULATION PUMP
CVCS HOLDUP TANK TRANSFER PUMPS
WORK THIS DRAWING WITH 9321-F-2737
EXCEPT WHERE NOTED, EVERYTHING ON THIS DRAWING (EXCLUDING THOSE PORTIONS SHOWN WITH DASHED LINES) IS PART OF THE CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) LISTED IN EXHIBIT "A",.CI-240-1
NOTE 1 THE ITEMS SHOWN DASHED WITHIN THIS COMPONENT BOUNDARY ARE PART OF (CVCS)
NOTE:
SEE DWG. 9321-F-2737 FOR ALL NOTES, REFERENCES AND LEGEND
THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.
* FOR CONTINUATION SEE DWG. 9321-2730
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS A" ITEMS
PER CI-240-1
COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED
REVISION
G. BHALLA 03/12/01 ENG
THIS REVISION IS NON-CLASS PER CI-240-1. UPDATED DWG. TO SHOW RETIRED BORON RECOVERY EQUIPMENT PER CRS #199806527. RELEASED AS CONSTRUCTED.
P/N 69901-AF D.B./G.B. 03/12/01
2/23/88 D.P.
F.A. W.J.KING 2/23/88
DATE CKR. SUPV. REVIEW
DESIGN ENG DISC. ENG. ENG. MGR. APPROVALS
TITLE: FLOW DIAGRAM
CHEMICAL & VOLUME
CONTROL SYSTEM -
UFSAR FIGURE No. 9.2-1 (SHT. 4)
DRWN. BY GIBBS&HILL
SCALE NONE
REC'D
Con Edison
STATION
INDIAN POINT
DWG. NO. A235309-33
MY

REFERENCE DRAWINGS

DWG. NO.	SYSTEM
9321-F-2738	REACTOR COOLANT
9321-F-2719	WASTE DISPOSAL SH.1
9321-F-2736	CHEMICAL VOLUME CONTROL SH.1
9321-F-2737	CHEMICAL VOLUME CONTROL SH.2
A235309	CHEMICAL VOLUME CONTROL SH.3
9321-F-2720	AUXILIARY COOLANT
9321-F-2673	ISOLATION VALVE SEAL WATER
9321-F-2728	NUCLEAR EQUIP. DRAINS
FP9321-20224	GAS STRIPPER CONTROL SCHEMATIC
9321-C-2016	FLOW DIAGRAM-SYMBOLS

EVERYTHING ON THIS DRAWING (EXCLUDING THOSE PORTIONS SHOWN WITH DASHED LINES) IS PART OF THE PRIMARY MAKEUP WATER SYSTEM (PW) LISTED IN EXHIBIT A, CI-240-1. EXCEPT AS SPECIFICALLY INDICATED HEREIN.

NOTE A
AUXILIARY STEAM SYSTEM - NOT LISTED IN CI-240-1.

NOTES:

1. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS
"CLASS A" ITEMS
PER QAPD

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

REVISION: THIS REVISION IS CLASS "A" PER THE QAPD. UPDATED DRAWING PER MOD PROCEDURE #FMX-99-12267-M TRANSFERRED DMD9321-2724-AN REV.0 AND DMD9321-2724-AO REV.0. RELEASE AS CONSTRUCTED. P.N. 12267-99 JF/DB

DES / ENG: G. BHALLA 08/24/01

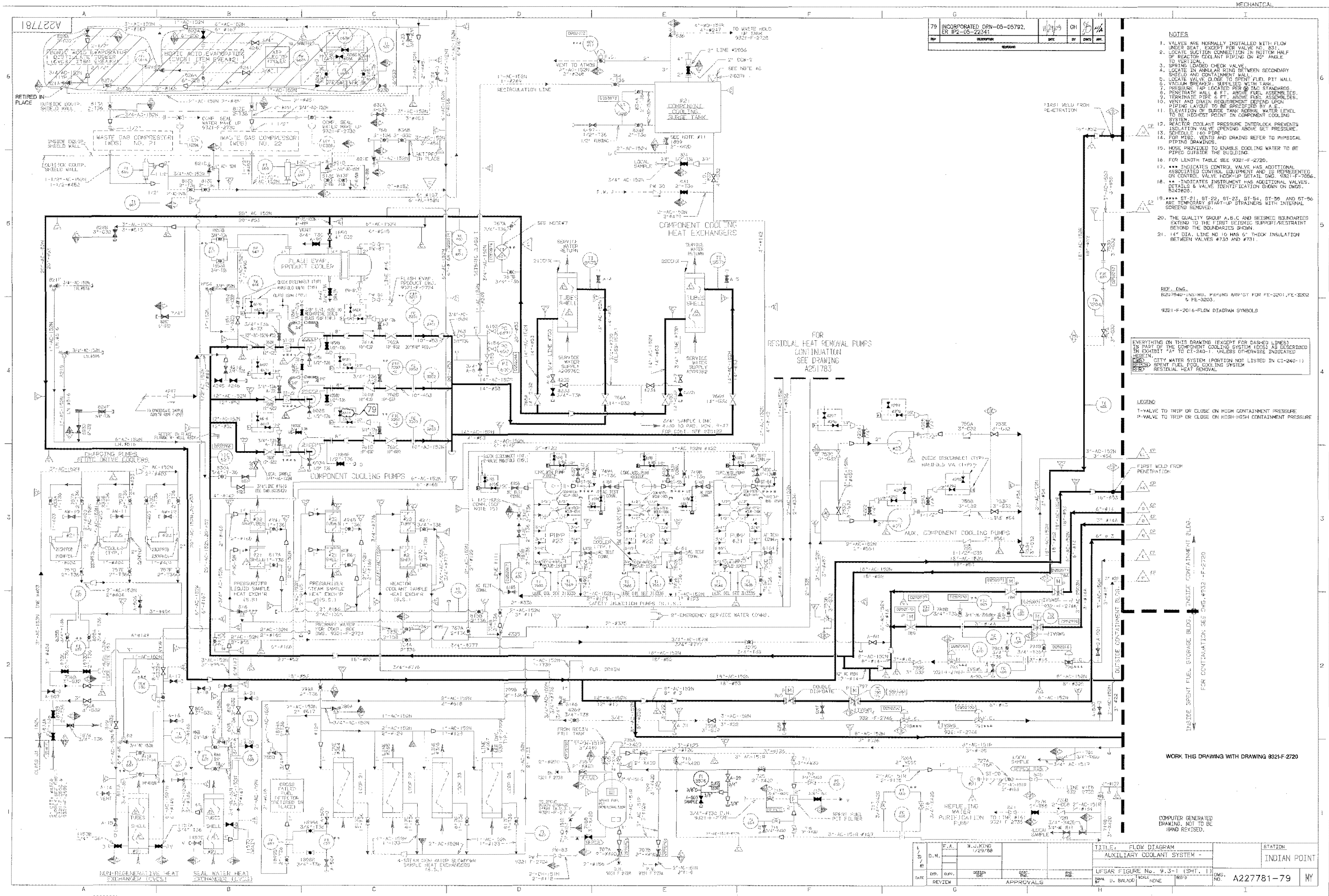
DATE: 3/28/87 — CKR.: A.G. — SUPV.: F.A. — REVIEW — DESIGN ENG: W.J.KING 3-30-87 — DISC. ENG. — ENG. MGR. — APPROVALS

TITLE: FLOW DIAGRAM
PRIMARY MAKE-UP WATER SYSTEM
NUCLEAR STEAM SUPPLY PLANT. -
UFSAR FIGURE No. 9.2-2

DRWN. BY GIBBS&HILL — SCALE NONE — REC'D

Con Edison — STATION INDIAN PO

DWG. NO. 9321-F-2724-55



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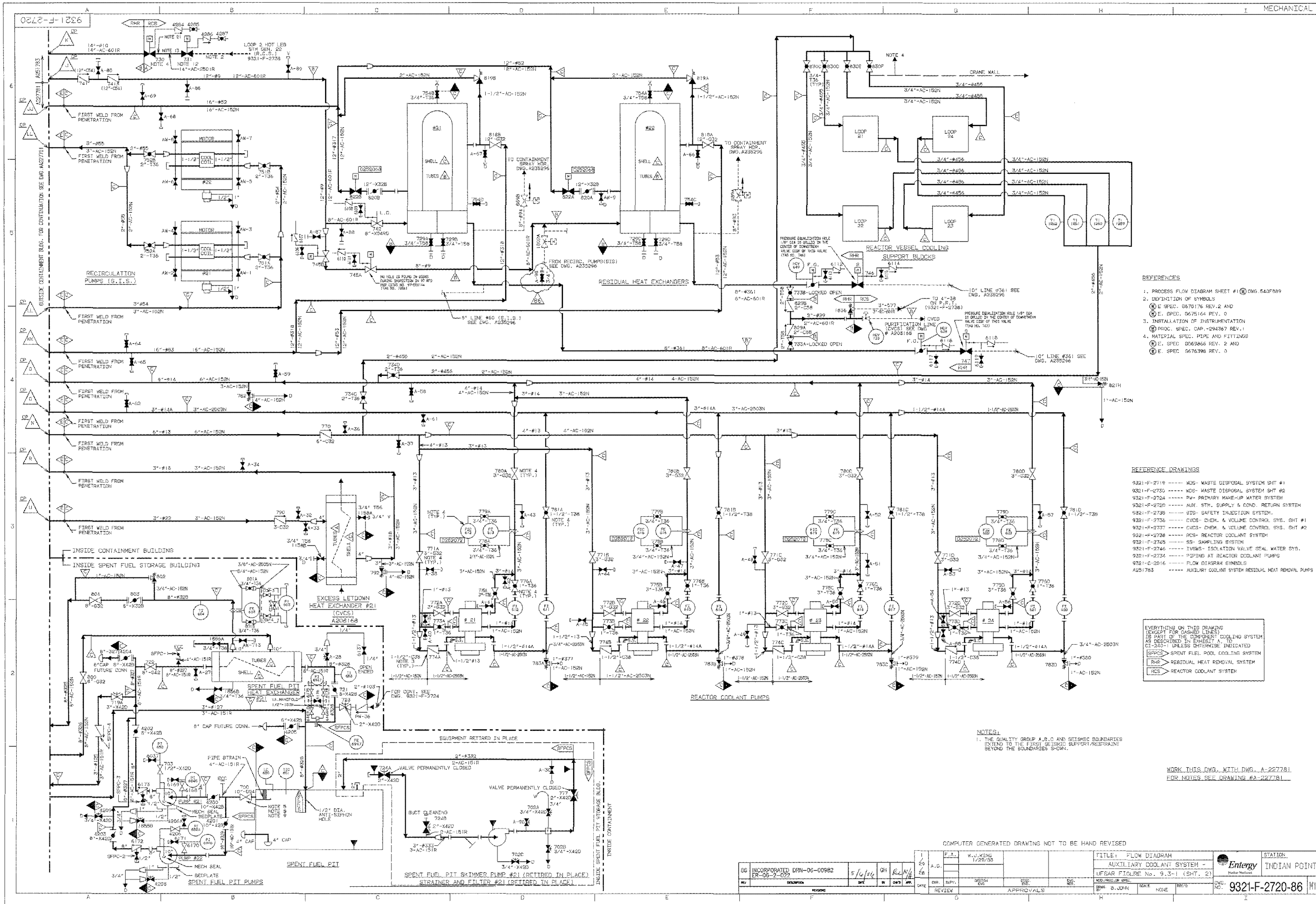

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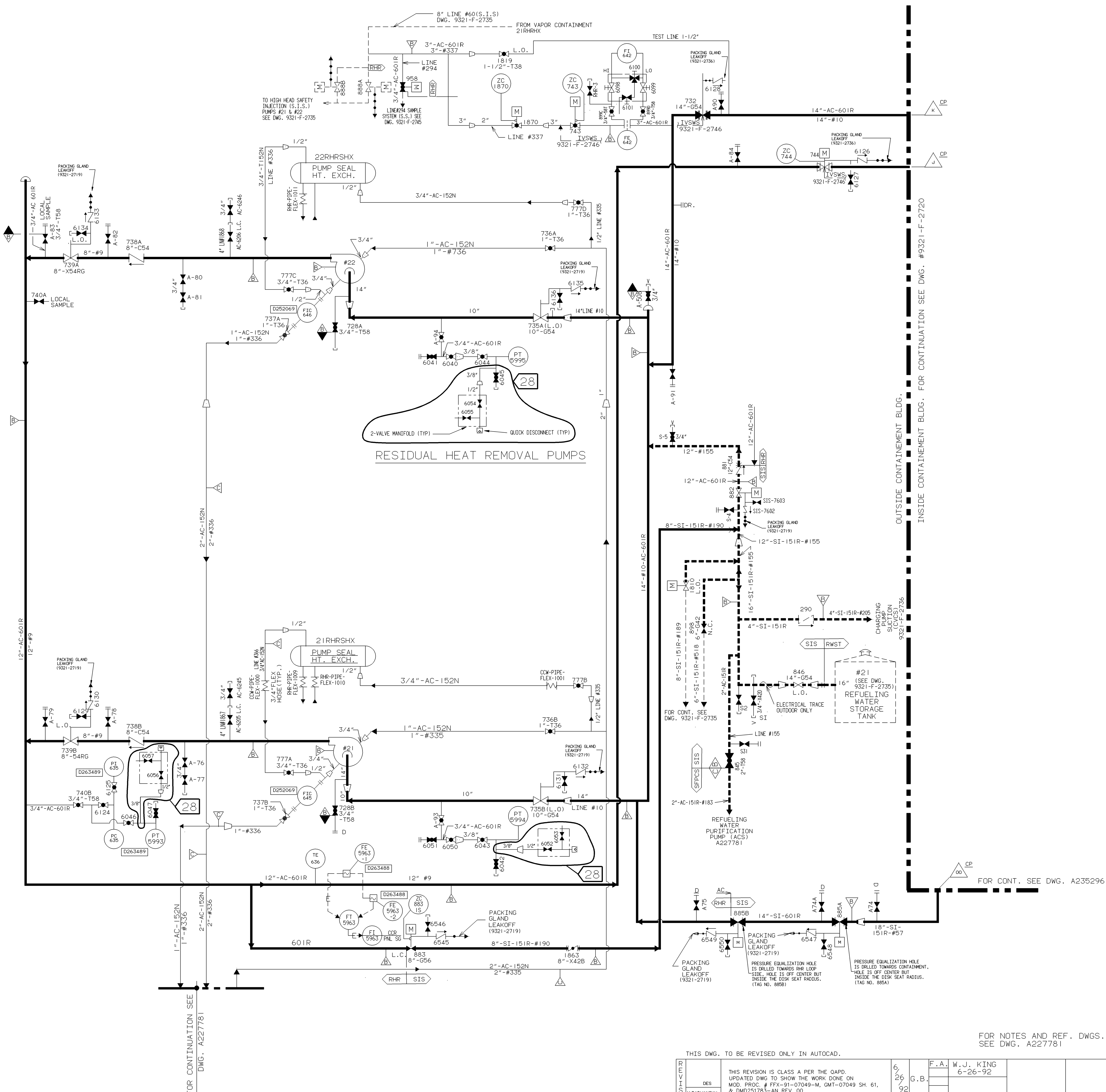
MECHANICAL

251783



NOTES:

1. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS:
"CLASS A" ITEMS PER THE QAPD

FOR NOTES AND REF. DWGS. SEE DWG. A227781

THIS DWG. TO BE REVISED ONLY IN AUTOCAD.

REVISION			DATE	CKR.	SUPV.				
DES M.RADVANSKY 04/10/03 ENG	THIS REVISION IS CLASS A PER THE QAPD. UPDATED DWG TO SHOW THE WORK DONE ON MOD. PROC. # FFX-91-07049-M, GMT-07049 SH. 61, & DMD251783-AN REV. 00 RELEASED AS CONSTRUCTED. P.N. 07049-91	GH/MR	6/26/92	G.B.	F.A.	W.J. KING 6-26-92	DESIGN ENG.	DISC. ENG.	ENG. MGR.
				REVIEW			APPROVALS		

TITLE: FLOW DIAGRAM
AUXILIARY COOLANT SYSTEM
RESIDUAL HEAT REMOVAL PUMPS -
UFSAR FIGURE No. 9.3-1 (SHT. 3)

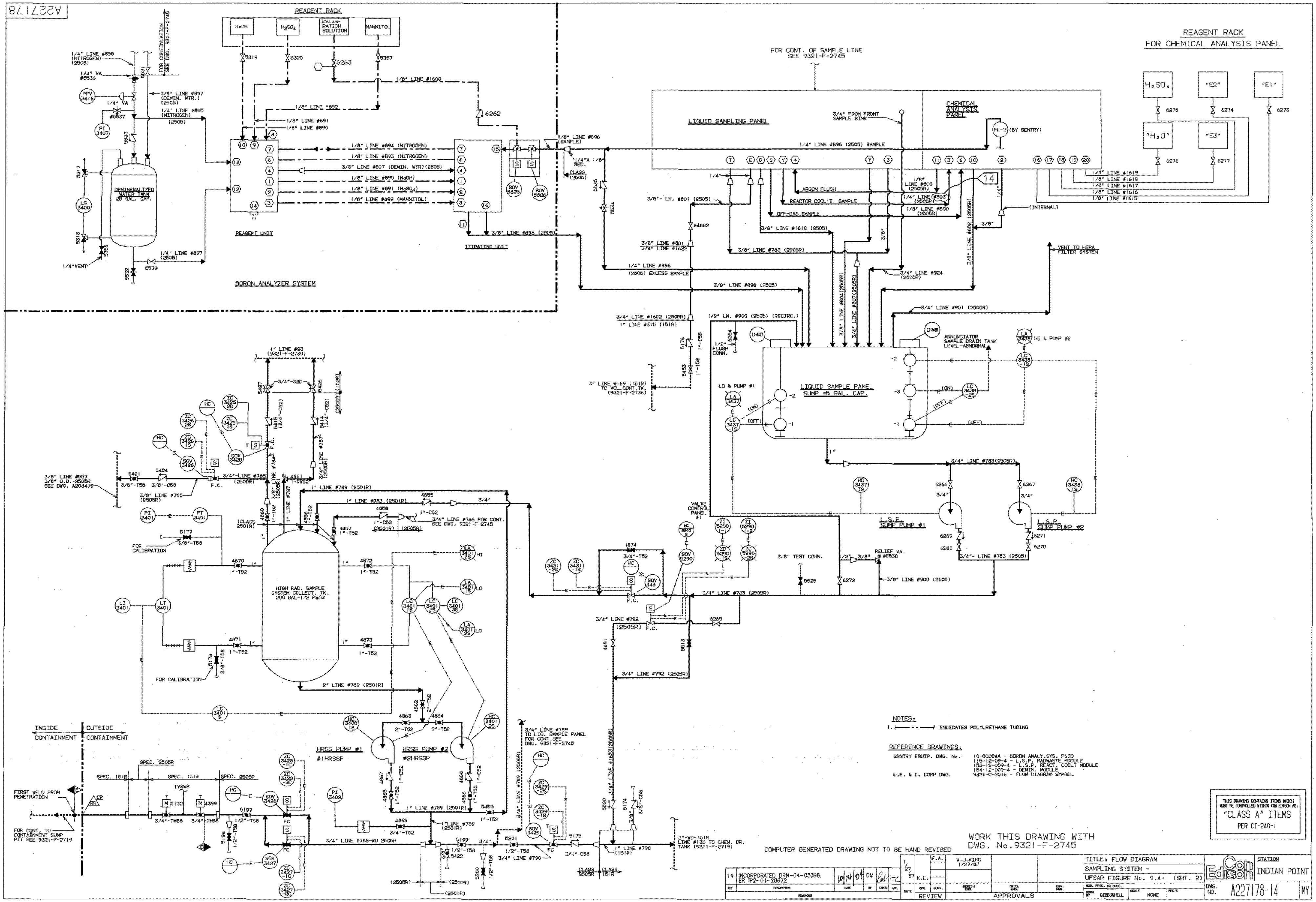
DRWN. BY J.SESE | SCALE NONE | REC'D

Con Edison | STATION INDIAN POINT

DWG. NO. A251783-28 | MY

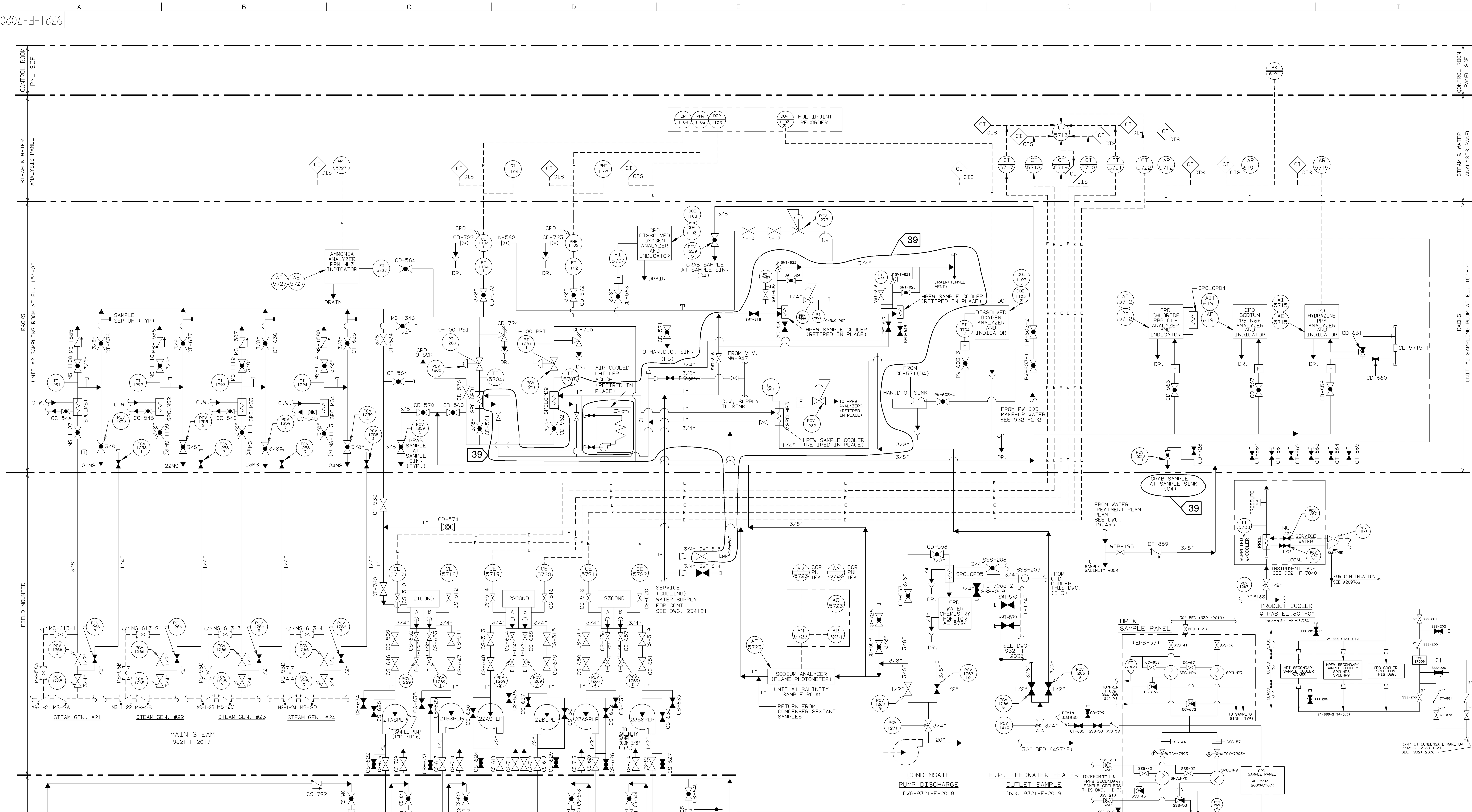
OAGI0000215 1199


MECH.
9321-F-2745
NOTES:
1. ALL TUBING IS 3/8" O.D. EXCEPT AS DETAILED.
2. ALL VALVES INSTALLED WITH FLOW UNDER SEAT EXCEPT VALVE NUMBERS 962A, 962B, 962C, 968, 970A, 978, 980.
3. 3/4" NOMINAL PIPE O.D. X 3/8" O.D. TUBING INSERT.
4. ADDITIONAL VENTS AND DRAINS MAY BE REQUIRED BASED ON PIPE LAYOUT.
5. ALL CHECK VALVES SHOULD BE "Y" TYPE LIFT CHECK.
6. DELAY COIL --- 125 FT. X 3/8" O.D. TUBING X 0.245 I.D. OR EQUIVALENT.
7. *** INDICATESS CONTROL VALVE HAS ADDITIONAL ASSOCIATED EQUIPMENT AND IS REPRESENTED ON CONTROL VALVE HOOK-UP DETAIL DRAWING 9321-F-7056.
8. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.
REFERENCE DRAWING:
CVCS - CHEMICAL VOLUME CONTROL SYSTEM U.E.& C. DWG. ------ 9321-F-2736
U.E.& C. DWG. ------ 9321-F-2737
RCS - REACTOR COOLANT SYSTEM U.E.& C. DWG. ------ 9321-F-2738
STEAM GEN. BLOWNDOWN SYS. U.E.& C. DWG. ------ 9321-F-2729
ACS - AUXILIARY COOLANT SYSTEM U.E.& C. DWG. ------ 9321-F-2720
WDS - WASTE DISPOSAL SYSTEM U.E.& C. DWG. ------ 9321-F-2719
U.E.& C. DWG. ------ 9321-F-2730
SIS - SAFETY INJECTION SYSTEM U.E.& C. DWG. ------ 9321-F-2735
IVSWS - ISOLATION VALVE SEAL WATER SYS. U.E.& C. DWG. ------ 9321-F-2746
REFERENCES:
PROCESS FLOW DIAGRAM W-DWG.540F902
DEFINITION OF SYMBOLS:U.E.&C. SPEC. PART B. 9321-01-248-18
INSTALLATION OF INSTRUMENTATION W-PROC.SPEC. CAP - 294367 REV.1
MATERIAL SPEC. PIPE AND FITTINGS; U.E.&C. SPEC. 9321-01-248-18
9321-F-2016 FLOW DIAGRAM SYMBOLS
NUCLEAR EQUIPMENT DRAINS 9321-F-2738
SS-SAMPLING SYSTEM DWG.A227178
POST ACCIDENT CONTAINMENT AIR SAMPLING SYSTEM DWG.A208479
LEGEND
-INDICATES INTERFACE WITH PANEL MFR.
VS -VENTILATION SYSTEM
INSIDE CONTAINMENT
OUTSIDE CONTAINMENT
PRESSURE STM.SPACE SAMPLE HT.EXCH.NO.21
PRESSURE STM.SPACE SAMPLE HT.EXCH.NO.22
PRESSURE LIQUID SPACE SAMPLE HT.EXCH.NO.21
PRESSURE LIQUID SPACE SAMPLE HT.EXCH.NO.22
REACTOR COOLANT SAMPLE HEAT EXCHANGER NO.21
REACTOR COOLANT SAMPLE HEAT EXCHANGER NO.22
COMPONENT COOLING WATER (ACS) REF DWG NO. 9321-F-2720
INLINE ISOTOPIC ANALYZER (CANBERRA) IIAN
LIQUID NITROGEN
DEMINERALIZER UNIT
DETAIL "A"
RAW WATER CHARCOAL FILTER
ION EXCHANGE FILTER
FINAL FILTER
GROSS FAILED FUEL DETECTOR
(1) ARGON BOTTLES
(3) NITROGEN BOTTLES
LIQUID SAMPLING PANEL
CHEMICAL ANALYSIS PANEL
WORK THIS DWG WITH DWG A227178
NOTE A
THIS BOUNDS THE PORTION OF SAMPLING SYSTEM AS DESCRIBED IN EXHIBIT A. CI-240-1
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS; "CLASS A" ITEMS PER CI-240-1
THIS DWG. TO BE REVISED ONLY IN CUSTOMIZED AUTOPLANT.
THIS REVISION IS NON-CLASS PER QAPD. UPDATED DRAWING PER MOD PROCEDURE #FPX-96-83574-F.DMD9321-2745-AI REV.0 AND DMD9321-2745-AJ REV.0. RELEASE AS CONSTRUCTED. P.N. 83574-96 D.B./J.F. 5/29/01
G. BHALLA 5/29/01
W.J.KING 2/9/88
TITLE: FLOW DIAGRAM SAMPLING SYSTEM -
UFSAR FIGURE No. 9.4-1 SHT. 1
DRWN. BY GIBBS&HILL SCALE NONE
Con Edison
STATION INDIAN POINT
DWG. NO. 9321-F-2745-58 MY



KARL-3898
A227178
REAGENT RACK
NaOH
H_2SO_4
CALIBRATION SOLUTION
MANNITOL
REAGENT RACK FOR CHEMICAL ANALYSIS PANEL
"E1"
"E2"
"E3"
"H_2O"
FOR CONT. OF SAMPLE LINE SEE 9321-F-2745
LIQUID SAMPLING PANEL
CHEMICAL ANALYSIS PANEL
3/4" FROM FRONT SAMPLE SINK
1/4" LINE #896 (2505) SAMPLE
DEMINERALIZED WATER TANK 25 GAL. CAP.
REAGENT UNIT
TITRATING UNIT
BORON ANALYZER SYSTEM
ARGON FLUSH
REACTOR COOL'T. SAMPLE
OFF-GAS SAMPLE
VENT TO HEPA FILTER SYSTEM
LIQUID SAMPLE PANEL SUMP =5 GAL. CAP.
ANNUNCIATOR SAMPLE DRAIN TANK LEVEL-ABNORMAL
LO & PUMP #1
HI & PUMP #2
L.S.P. SUMP PUMP #1
L.S.P. SUMP PUMP #2
VALVE CONTROL PANEL #1
3/8" TEST CONN.
RELIEF VA.
HIGH RAD. SAMPLE SYSTEM COLLECT. TK. 200 GAL.=1/2 PSIG
FOR CALIBRATION
HRSS PUMP #1
HRSS PUMP #2
INSIDE CONTAINMENT
OUTSIDE CONTAINMENT
FIRST WELD FROM PENETRATION
FOR CONT. TO CONTAINMENT SUMP PIT SEE 9321-F-2719
NOTES:
INDICATES POLYURETHANE TUBING
REFERENCE DRAWINGS:
SENTRY EQUIP. DWG. No. 10-00004A - BORON ANALY.SYS. P&ID
115-12-09-4 - L.S.P. RADWASTE MODULE
153-12-09-4 - L.S.P. REACT. COOLT MODULE
154-12-009-4 - DEMIN. MODULE
U.E. & C. CORP DWG. 9321-C-2016 - FLOW DIAGRAM SYMBOL
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON NO.
"CLASS A" ITEMS
PER CI-240-1
WORK THIS DRAWING WITH DWG. No.9321-F-2745
COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED
14 INCORPORATED DRN-04-03398, ER IP2-04-28672
W.J.KING 1/27/87
REVIEW
APPROVALS
TITLE: FLOW DIAGRAM
SAMPLING SYSTEM -
UFSAR FIGURE No. 9.4-1 (SHT. 2)
Con Edison
STATION INDIAN POINT
DWG. NO. A227178-14

9321-F-7020



REFERENCE DWG'S.

9321-F-7021	STEAM & WATER SAMPLE PIPING
9321-F-7044	STEAM & WATER ANALYSIS PANEL
9321-F-2021	FLASH EVAPORATOR FLOW DIAGRAM
9321-F-2729	STEAM GENERATOR BLOWDOWN DIAGRAM
9321-F-2597	STEAM GENERATOR BLOWDOWN SAMPLE PIPING
9321-F-2120	SERVICE BOILERS FLOW DIAGRAM
9321-F-7040	P.A. BLDG. INSTRUMENT ARRANGEMENT
9321-F-2724	PRIMARY MAKE-UP WATER FLOW DIAGRAM
A215116	COND. #21,22, & 23 SEXTANT SAMPL'G, PIP'G ARRG'T
B226920	COND.WTR CHLORIDE ANALY QUICK RESP. INST. FLOW DIAG.
2000MC6024	HW SEXTANT PANEL P& ID (JOHNSON MARCH SYS. INC DWG. D-250561-M01)
2000MC5873	HPFW SAMPLE PANEL P&ID (SENTRY EQUIP. CORP.)

NOTE:

1. COOLING WATER IS SUPPLIED FROM CLOSED COOLING WATER SYSTEM (9321-F-2033) EXCEPT WHERE NOTED OTHERWISE.
2. ALL SAMPLE TUBING SHALL BE STN. STL. TYP. TP304 INN ACCORDANCE WITH SPECIFICATION 9321-01-248-18 CLASS M-1
3. PROVIDE GATE & CHECK VALVES AT ALL TAKE-OFFS FROM CITY WATER LINES

SYMBOLS

— SAMPLE COOLER (S.C.)

CE — CONDUCTIVITY ELEMENT

1 BULKHEAD, SEE DWG.9321-F-7044

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

REVISION — DES: R.MEIKLE 05/12/03 — ENG

THIS REVISION IS NON-CLASS PER THE QAPD.
UPDATED DWG. PER CR-IP2-2003-01626
RELEASED FOR RECORD
P.N 69901-AF
GH/RAM

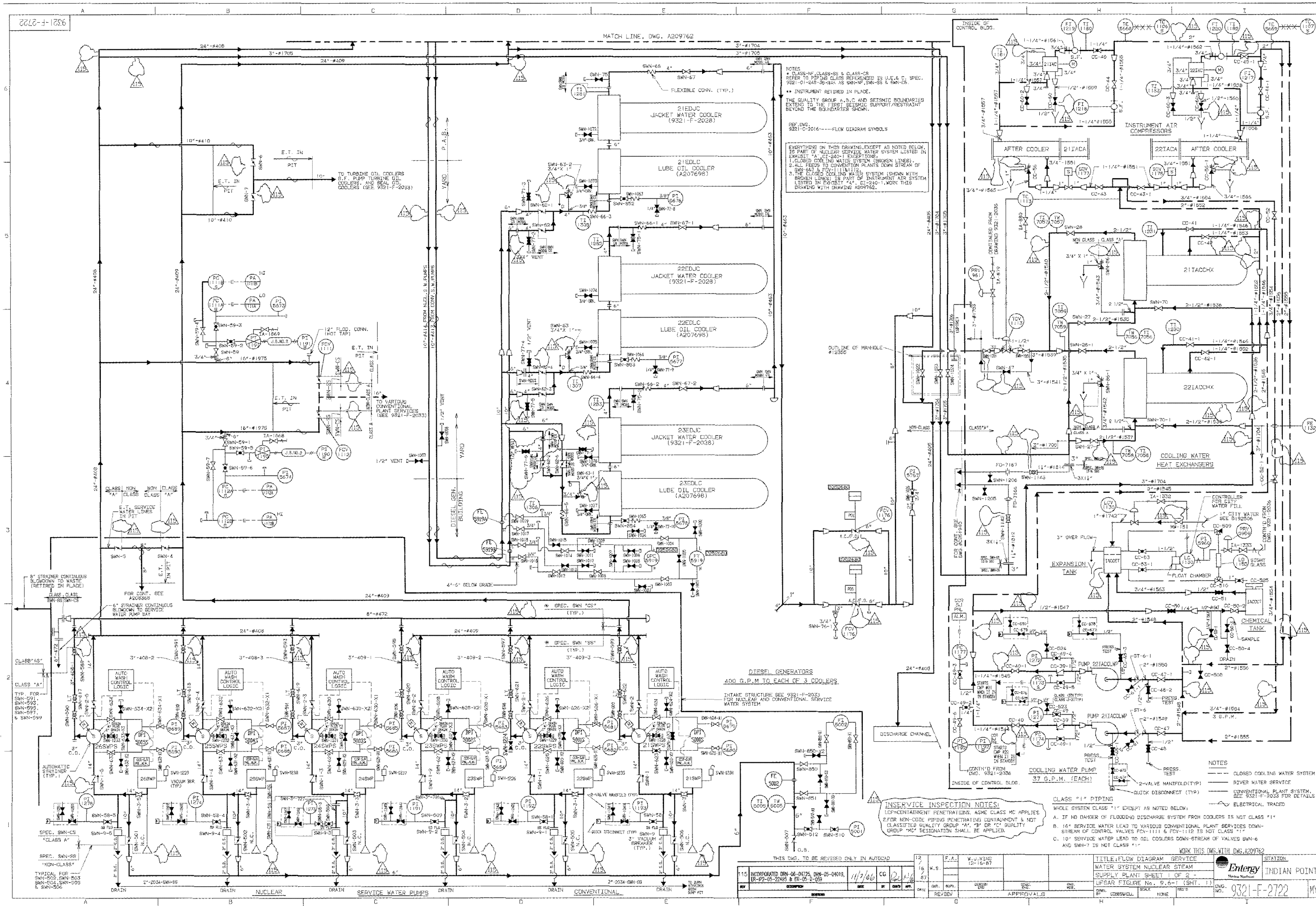
DATE	CKR.	SUPV.	DESIGN ENG	DISC. ENG.	ENG. MGR.
2/8/88	L.H.	F.A.	W.J.KING 2/9/88		
	REVIEW		APPROVALS		

TITLE: STEAM & WATER ANALYSIS SYSTEM SAMPLING DIAGRAM INSTRUMENTATION -
UFSAR FIGURE No. 9.4-2

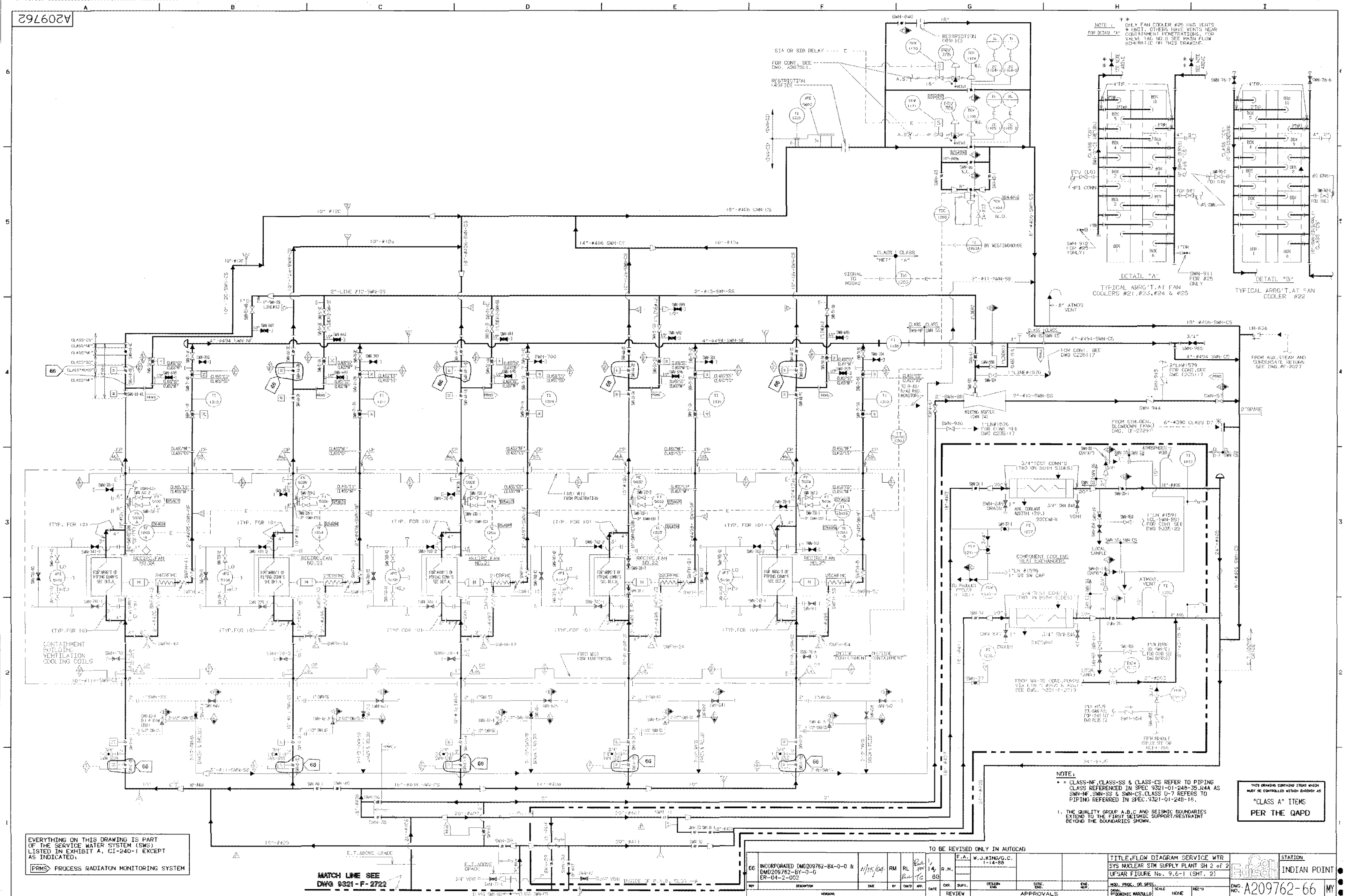
DRWN. BY | SCALE NONE | REC'D

Con Edison — STATION: INDIAN POINT

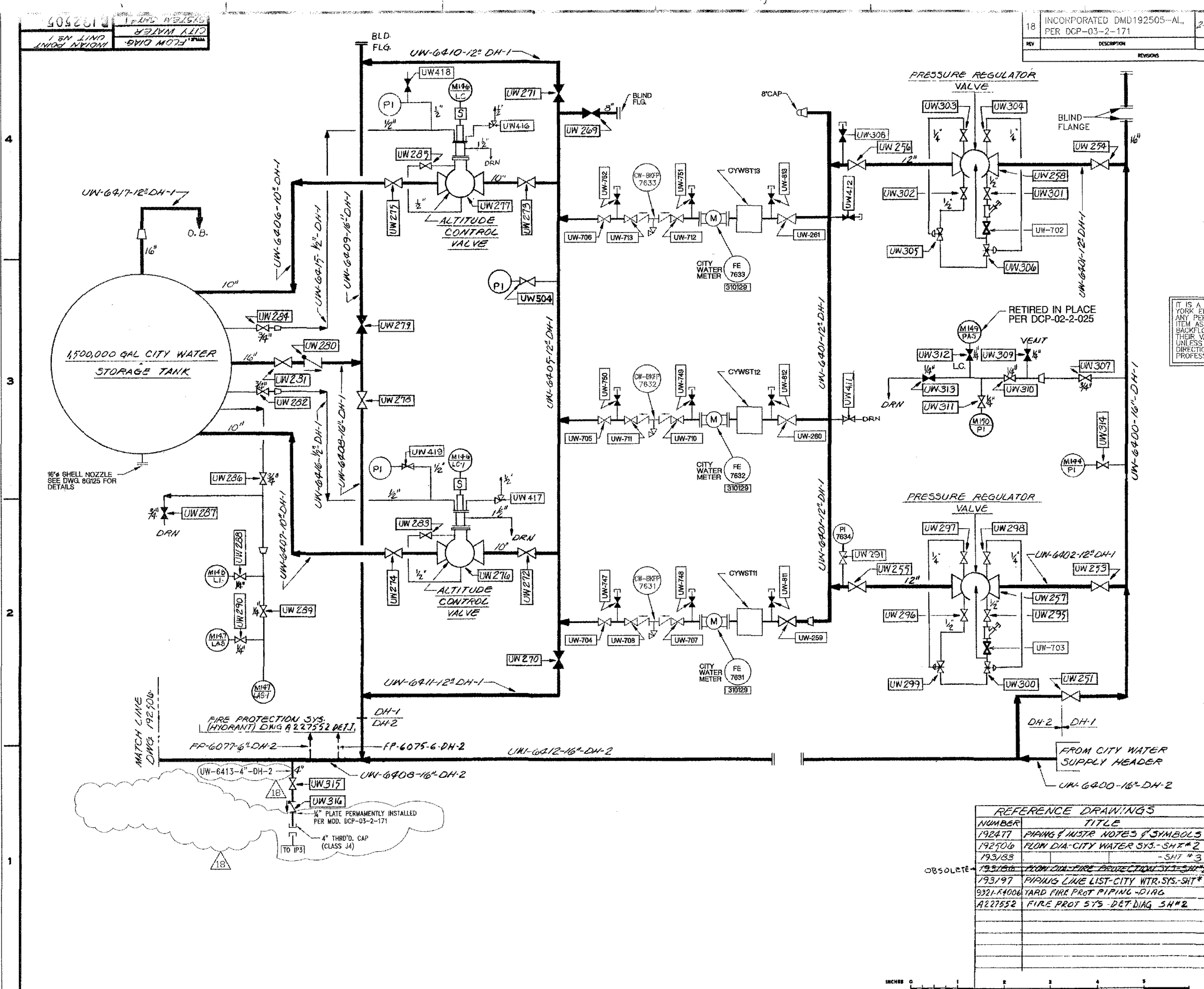
DWG. NO. 9321-F-7020-39 MY



OAGI0000215_1203



OAGI0000215_1204



REV	DESCRIPTION	DATE	BY	CHK'D	APP.
18	INCORPORATED DMD192505-AL, PER DCP-03-2-171	2-8-2006	DM		

REVISIONS

M.RADVANSKY 05/02/03 — 17

THIS REV.IS NON CLASS PER THE QAPD. UPDATED DWG. TO SHOW THE WORK DONE ON MOD. PROC. NO. FMX-94-10478-M, DMD192505-AE,REV.03 DMD192505-AF,REV.04 DMD192505-AG,REV.00 DMD192505-AH,REV.00 DMD192505-AI,REV.00 DMD192505-AJ,REV.00 RELEASE AS CONSTRUCTED P.N.10478-94 GH/MR

REFERENCE DRAWINGS

NUMBER	TITLE
192477	PIPING & INSTR. NOTES & SYMBOLS
192506	FLOW DIA-CITY WATER SYS.-SH#2
193183	-SHT #3
~~193186~~	~~FLOW DIA-FIRE PROTECTION SYS-SH#2~~ (OBSOLETE)
193197	PIPING LINE LIST-CITY WTR. SYS.-SHT#1
9321-F4006	YARD FIRE PROT PIPING-DIAG
A227552	FIRE PROT SYS-DET. DIAG SH#2

B 192505-18

Entergy Nuclear Northeast

PREPARED BY: TREADWELL CORP. N.Y.

STATION LOCATION: INDIAN POINT

UNIT Nº 1

PIPING FLOW DIAGRAM

CITY WATER SYSTEM SHEET No. 1 - UFSAR FIGURE No. 9.6-5 (SHT. 1)

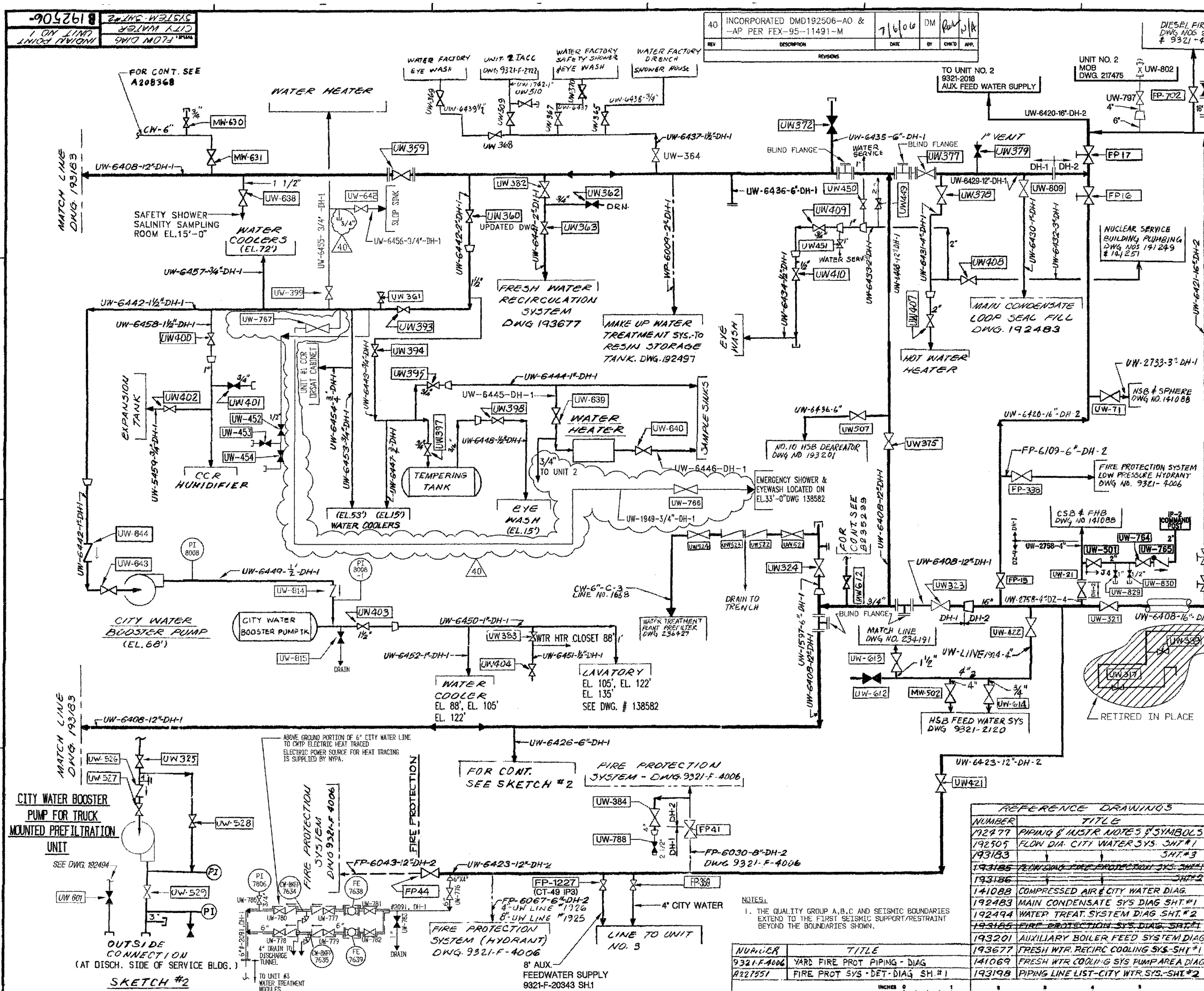
SCALE: NONE — DRAWN BY: W. NATER — RECORDED: 1-6-75

J.O. 3948 — CHECKED BY: H.M. — ENG.: VK

APPROVALS

TC-#2300594

OAGI0000215_1205



REV	DESCRIPTION	DATE	BY	CHK'D	APP.
40	INCORPORATED DMD192506-A0 & -AP PER FEX-95-11491-M	7/6/06	DM		N/A

REVISIONS

NOTES:

1. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

NUMBER	TITLE
9321-F-4006	YARD FIRE PROT PIPING - DIAG
A227551	FIRE PROT SYS - DET - DIAG SH. #1

REFERENCE DRAWINGS

NUMBER	TITLE
192477	PIPING & INSTR. NOTES & SYMBOLS
192505	FLOW DIA. CITY WATER SYS. SHT #1
193183	SHT #3
~~193185~~	~~FLOW DIAG FIRE PROTECTION SYS. SHT #1~~
~~193186~~	~~SHT #2~~
141088	COMPRESSED AIR & CITY WATER DIAG.
192483	MAIN CONDENSATE SYS DIAG SHT #1
192494	WATER TREAT. SYSTEM DIAG SHT #2
~~193185~~	~~FIRE PROTECTION SYS. DIAG. SHT #1~~
193201	AUXILIARY BOILER FEED SYSTEM DIAG.
193677	FRESH WTR. RECIRC COOLING SYS-SHT #1
141069	FRESH WTR COOLING SYS PUMP AREA DIAG.
193198	PIPING LINE LIST-CITY WTR. SYS.-SHT #2

B192506-40

Entergy
Nuclear Northeast

PREPARED BY: TREADWELL CORP. N.Y.

STATION LOCATION: INDIAN POINT

UNIT NO. 1

PIPING FLOW DIAGRAM CITY WATER

SYSTEM SHEET No. 2 -

UFSAR FIGURE No. 9.6-5 SHT. 2

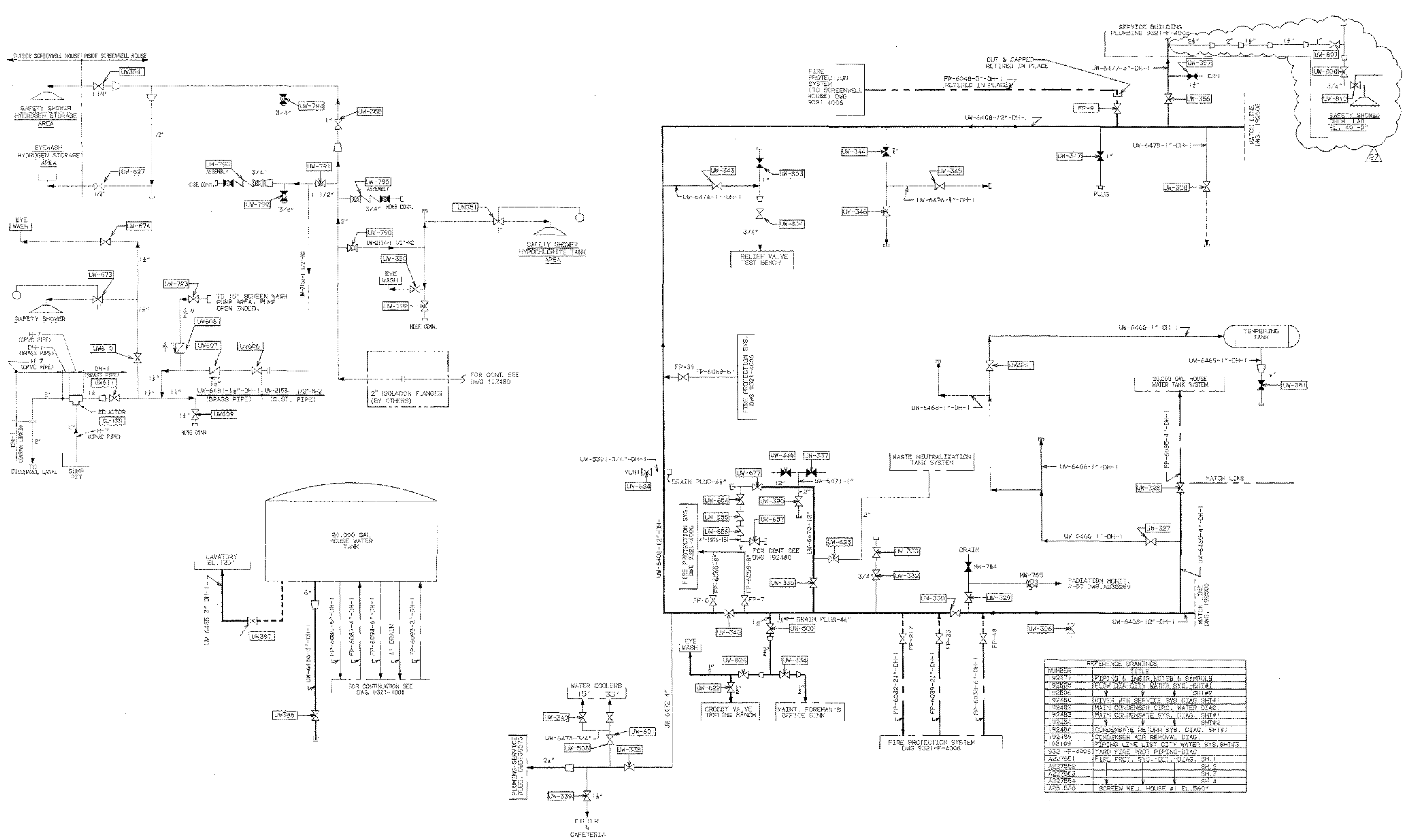
SCALE: NONE — DRAWN BY: W. NATER — J.O. 3946 — CHECKED BY: H.M.

TITLE: FLOW DWG INDIAN POINT UNIT NO. 1 CITY WATER SYSTEM - SHT #2 — B 192506-

TC #2300595

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
27	INCORPORATED DMD193183-AP, PER FPX-91-06585-F	7/12/06	DM	Rc	N/A

REVISIONS



REFERENCE DRAWINGS	
NUMBER	TITLE
192477	PIPING & INSTR.NOTES & SYMBOLS
192505	FLOW DIA-CITY WATER SYS.-SHT#1
192506	-SHT#2
192480	RIVER WTR SERVICE SYS DIAG.SHT#1
192482	MAIN CONDENSER CIRC. WATER DIAG.
192483	MAIN CONDENSATE SYS. DIAG. SHT#1
192484	SHT#2
192486	CONDENSATE RETURN SYS. DIAG. SHT#1
192489	CONDENSER AIR REMOVAL DIAG.
193199	PIPING LINE LIST CITY WATER SYS.SHT#3
9321-F-4006	YARD FIRE PROT PIPING-DIAG.
A227551	FIRE PROT. SYS.-DET.-DIAG. SH.1
A227552	SH.2
A227553	SH.3
A227554	SH.4
A251568	SCREEN WELL HOUSE #1 EL.560"

DWG. NO. 193183-27

Entergy

DWG. TYPE MECHANICAL

DWG. SIZE A

LOCATION: INDIAN POINT

TITLE: UNIT No. 1 PIPING FLOW DIAG. CITY WATER SYSTEM SHEET No. 3 UFSAR FI. No. 9.6-5 (SHT. 3)

– APPROVALS –

ENGINEERING MANAGER:

PROJECT ENGINEER:

DESIGNER:

DRAWN BY: M. HEREDIA

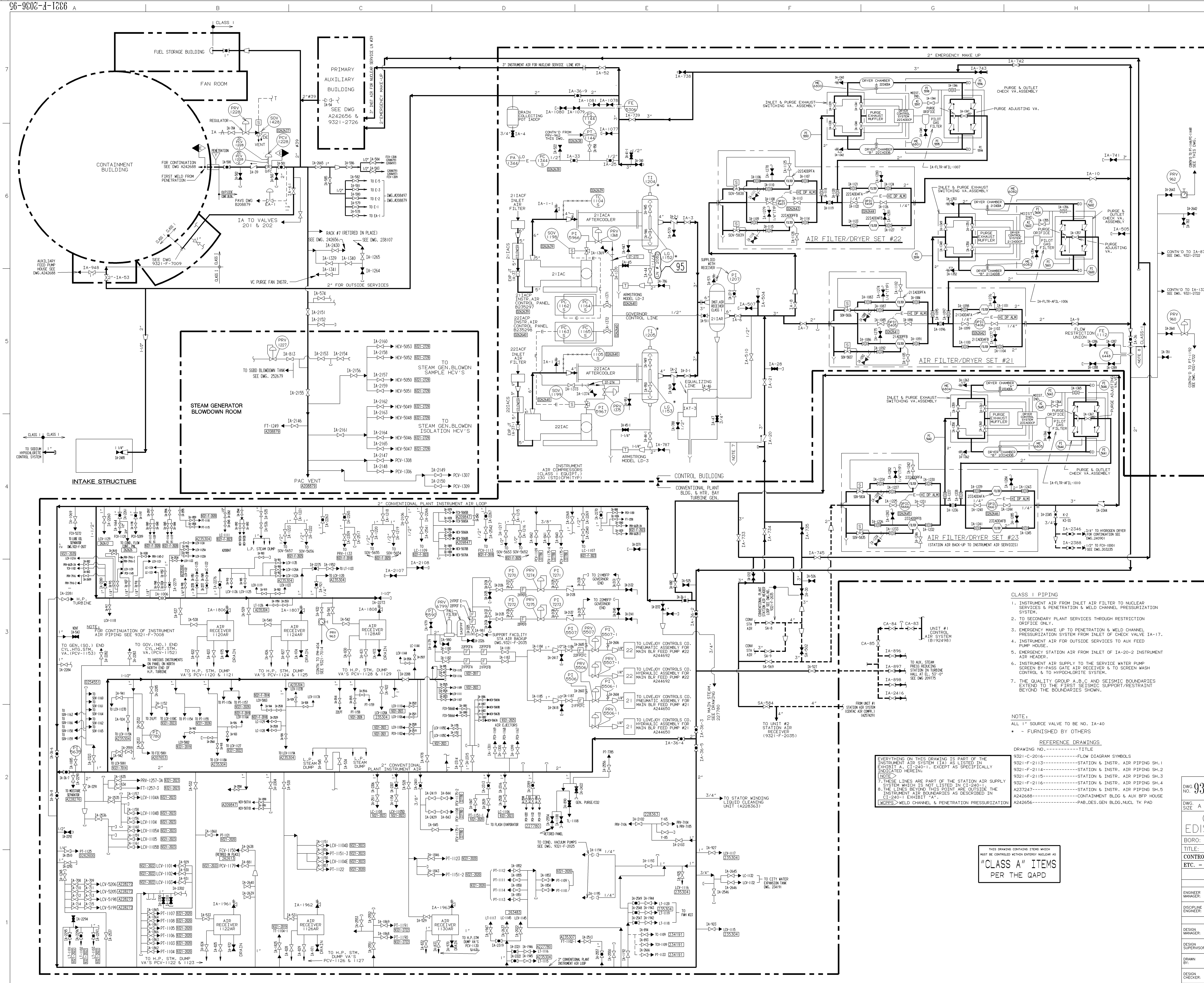
SCALE:

DISCIPLINE CODE: MB

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

193183-27

OAGI0000215_1207



CLASS I PIPING

1. INSTRUMENT AIR FROM INLET AIR FILTER TO NUCLEAR SERVICES & PENETRATION & WELD CHANNEL PRESSURIZATION SYSTEM.
2. TO SECONDARY PLANT SERVICES THROUGH RESTRICTION ORIFICE ONLY.
3. EMERGENCY MAKE UP TO PENETRATION & WELD CHANNEL PRESSURIZATION SYSTEM FROM INLET OF CHECK VALVE IA-17.
4. INSTRUMENT AIR FOR OUTSIDE SERVICES TO AUX FEED PUMP HOUSE.
5. EMERGENCY STATION AIR FROM INLET OF IA-20-2 INSTRUMENT AIR HEADER.
6. INSTRUMENT AIR SUPPLY TO THE SERVICE WATER PUMP SCREEN BY-PASS GATE AIR RECEIVER & TO SCREEN WASH CONTROL & TO HYPOCHLORITE SYSTEM.
7. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

NOTE:

ALL 1" SOURCE VALVE TO BE NO. IA-40

* ~ FURNISHED BY OTHERS

REFERENCE DRAWINGS

DRAWING NO.	TITLE
9321-C-2016	FLOW DIAGRAM SYMBOLS
9321-F-2113	STATION & INSTR. AIR PIPING SH.1
9321-F-2114	STATION & INSTR. AIR PIPING SH.2
9321-F-2115	STATION & INSTR. AIR PIPING SH.3
9321-F-2116	STATION & INSTR. AIR PIPING SH.4
A237247	STATION & INSTR. AIR PIPING SH.5
A242688	CONTAINMENT BLDG & AUX BFP HOUSE
A242656	PAB,DES.GEN BLDG,NUCL TK PAD

EVERYTHING ON THIS DRAWING IS PART OF THE INSTRUMENT AIR SYSTEM (IA) AS LISTED IN EXHIBIT A, CI-240-1. EXCEPT AS SPECIFICALLY INDICATED HEREIN:

NOTE

7. THESE LINES ARE PART OF THE STATION AIR SUPPLY SYSTEM WHICH IS NOT LISTED IN CI-240-1.
8. THE LINES BEYOND THIS POINT ARE OUTSIDE THE INSTRUMENT AIR BOUNDARIES AS DESCRIBED IN CI-240-1 EXHIBIT "A".

WCPPS WELD CHANNEL & PENETRATION PRESSURIZATION

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLED WITHIN ENTERGY NUCLEAR AS

"CLASS A" ITEMS PER THE QAPD

REVISION

USE .1 SIZE TEXT ONLY

V. MYERS 8/04/03 — 94

THIS REVISION IS CLASS "A" PER THE QAPD. UPDATED DRAWING PER MOD. PROC. # FFX-91-07049-M, DMD9321-2036-BC-00. RELEASE AS CONSTRUCTED. P/N 07049-91 SW/VM

V. MYERS 08/29/03 — 95

THIS REVISION IS CLASS "A" PER THE QAPD. DRAWING REVISED TO SHOW WORK COMPLETED IN FIELD PER MOD. PROC. No. MFI-88-01408-M GMT SHT. 86. INCORPORATED DMD9321-2036-BG. RELEASE "AS-CONSTRUCTED".

P/N: 01408-88

S. O'BRIEN 08/29/03

REVISION SIGNATURES

REV	DES	ENG
95	S. O'BRIEN 08-29-03	V. MYERS 08-29-03

DWG. NO. 9321-F-2036-95

DWG. SIZE A | DWG. TYPE COMPANY

CON EDISON | STATION: INDIAN POINT

BORO: WESTCHESTER

TITLE: FLOW DIAG. INSTR. AIR CONTROL BLDG. CONVENTIONAL PLANT, ETC. - UFSAR FIGURE No. 9.6-6

APPROVALS

ENGINEERING

ENGINEER MANAGER:

DISCIPLINE ENGINEER:

DESIGN

DESIGN MANAGER: W.J. KING 7-30-87

DESIGN SUPERVISOR: F.A. 7-30-87

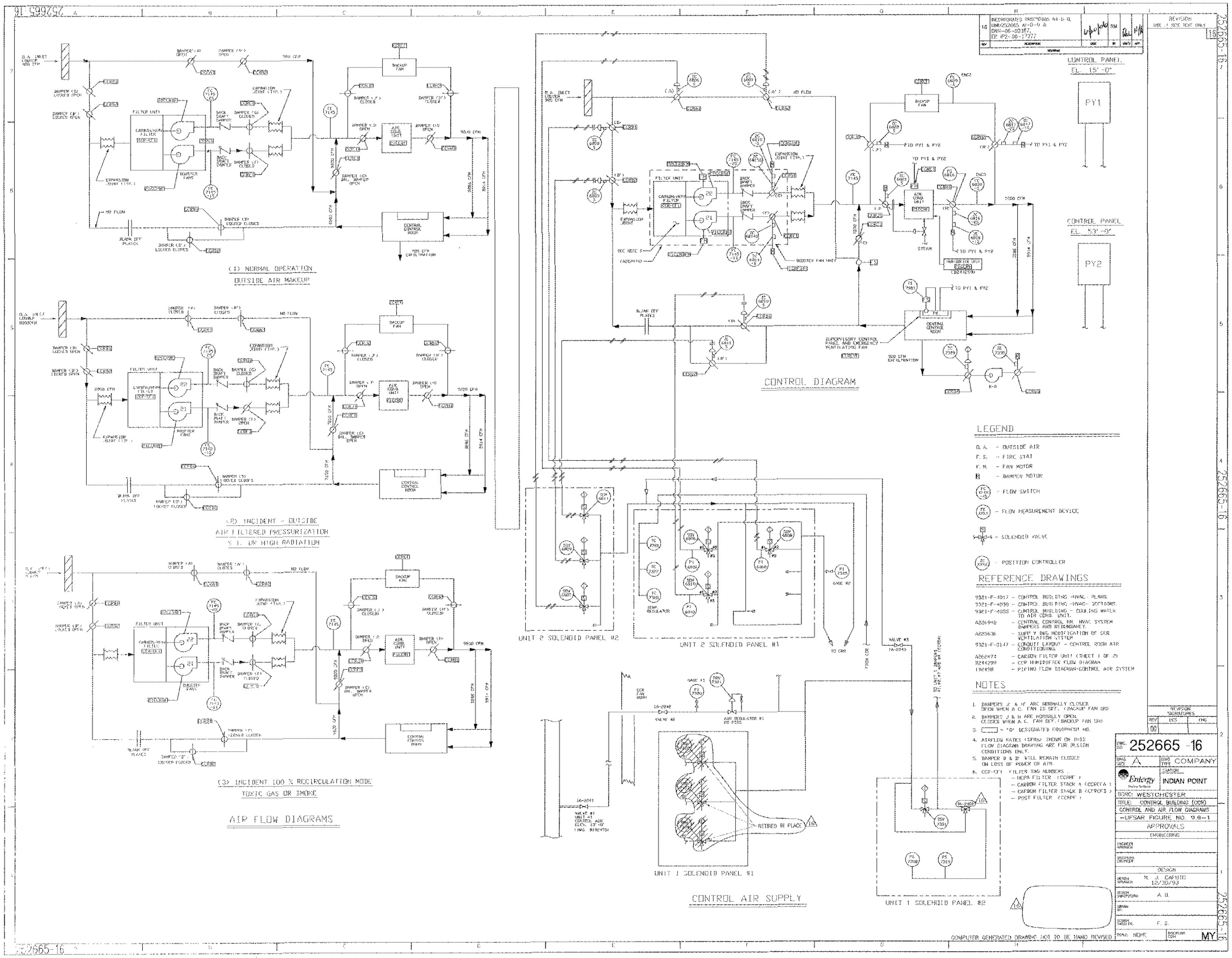
DRAWN BY: GIBBS & HILL

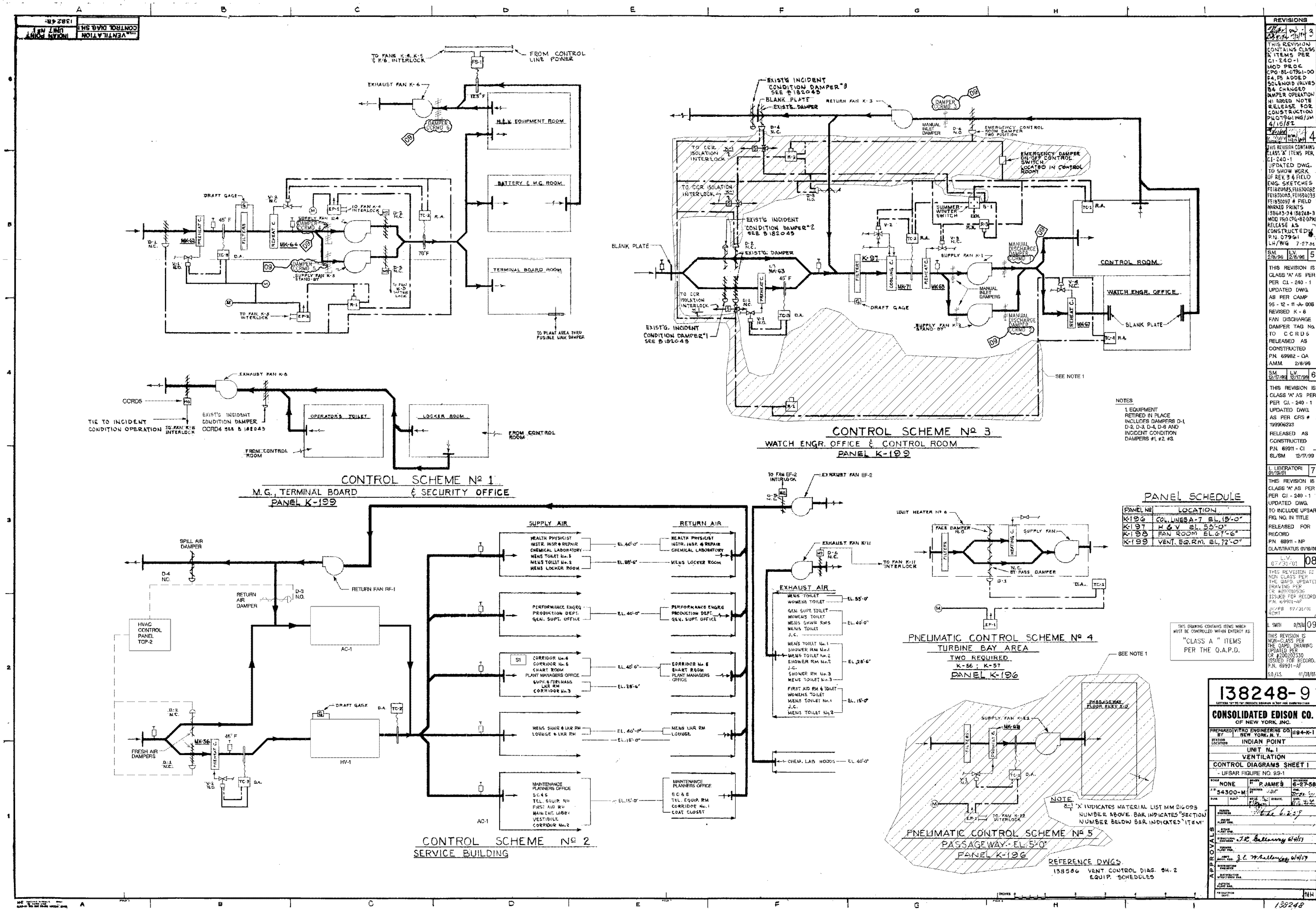
DESIGN CHECKER: L.H. 7-30-87

SCALE: NONE | DISCIPLINE CODE: MY

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED


9321-F-2035
DETAIL "A"
1/4" PNEUMATIC SEAL
M&D BLDG
FSB
PRV SA-1063
PRV SA-1064
TO RAD.MONITOR #5976 SEE DWG.B241548
TO SANDBLASTER DEGREASER & DECONTAM.ROOM
M & O BLDG.
FUEL STORAGE BUILDING
1/2" FOR CONT. SEE DETAIL "A"
FAN ROOM
CONTAINMENT BUILDING
FIRST WELD FROM PENETRATION
CLASS I
WASTE HOLD UP TANK PIT
PRIMARY AUXILIARY BUILDING
68' EL. PAB PIPE TUNNEL
CONTROL BUILDING
AUX. FEED PUMP BLDG.
CONVENTIONAL PLANT STATION AIR HEADER
650 SCFM STATION AIR COMPRESSOR (2 STAGE)
AFTER COOLER
INTERCOOLER
STATION AIR RECEIVER
INLET AIR FILTER
EXPANSION JOINT (SA-22)
TO PALL FILTER INST.AIR BACK UP FOR MBFP CONTROLS DWG.9321-F-2036
4" STATION AIR TIE TO UNIT #1 DWG.A257429
2" STATION AIR SUPPLY FROM UNIT #1 DWG.A257429
3" EMERGENCY SUPPLY TO INSTRUMENT AIR COMPRESSOR SYS. & EMERGENCY INSTRUMENT AIR SYSTEM. SEE DWG. 9321-F-2036
FOR CONT. SEE 9321-F-2036 (BACK-UP AIR DRYER)
CONVENTIONAL PLANT-STATION AIR HEADER
CONVENTIONAL PLANT T.G. BLDG. & HTR. BAY
INTAKE STRUCTURE
CLASS I PIPING
DOUBLE ISOLATION VALVE THROUGH CONTAINMENT INCLUDING ISOLATION VALVE SEAL WATER SUPPLY.
THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.
REFERENCE DRAWINGS
U.E.&C. DWG. NO.
DESCRIPTION
9321-F-2113 TURB. BLDG. & HTR. BAY-STA.& INSTR. AIR PIP'G SHT. #1
9321-F-2114 TURB. BLDG. & HTR. BAY-STA.& INSTR. AIR PIP'G SHT. #2
9321-F-2115 TURB. BLDG. & HTR. BAY-STA.& INSTR. AIR PIP'G SHT. #3
9321-C-2016 FLOW DIAGRAM SYMBOLS
9321-F-2687 CONTAINMENT BLDG SERVICE AIR CITY WATER PIPING
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS A" ITEMS
PER CI-240-1
COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED
L.A. 12/20/99
THIS REVISION IS NON CLASS PER CI-240-1
UPDATED DWG. PER CRS #199904407,
OIR FILE NO. 0-2602
RELEASED AS BUILT
P/N 69982-CD
GB/SW 12/20/99
W.J.KING 7-30-87
REVIEW
APPROVALS
TITLE: FLOW DIAGRAM
STATION AIR -
UFSAR FIGURE No. 9.6-7
GIBBS & HILL
SCALE NONE
Con Edison
STATION
INDIAN POINT
DWG. NO. 9321-2035-43


252665-16
(1) NORMAL OPERATION
OUTSIDE AIR MAKEUP
(2) INCIDENT - OUTSIDE
AIR FILTERED PRESSURIZATION
S.I. OR HIGH RADIATION
(3) INCIDENT 100 % RECIRCULATION MODE
TOXIC GAS OR SMOKE
AIR FLOW DIAGRAMS
CONTROL DIAGRAM
CONTROL PANEL
EL. 15'-0"
PY1
CONTROL PANEL
EL. 53'-0"
PY2
UNIT 2 SOLENOID PANEL #2
UNIT 2 SOLENOID PANEL #1
UNIT 1 SOLENOID PANEL #1
UNIT 1 SOLENOID PANEL #2
RETIRED IN PLACE
CONTROL AIR SUPPLY
LEGEND
O.A. - OUTSIDE AIR
F.S. - FIRE STAT
F.M. - FAN MOTOR
- DAMPER MOTOR
- FLOW SWITCH
- FLOW MEASUREMENT DEVICE
- SOLENOID VALVE
- POSITION CONTROLLER
REFERENCE DRAWINGS
9321-F-4017 - CONTROL BUILDING -HVAC- PLANS
9321-F-4050 - CONTROL BUILDING -HVAC- SECTIONS
9321-F-4055 - CONTROL BUILDING - COOLING WATER TO AIR COND. UNIT.
A226948 - CENTRAL CONTROL RM. HVAC SYSTEM DAMPERS AND REDUNDANCY.
A225636 - SUPPLY DWG MODIFICATION OF CCR VENTILATION SYSTEM
9321-F-3147 - CONDUIT LAYOUT - CONTROL ROOM AIR CONDITIONING
A262474 - CARBON FILTER UNIT (SHEET 1 OF 2)
B244299 - CCR HUMIDIFIER FLOW DIAGRAM
192498 - PIPING FLOW DIAGRAM-CONTROL AIR SYSTEM
NOTES
1. DAMPERS J' & H' ARE NORMALLY CLOSED. OPEN WHEN A.C. FAN IS OFF. (BACKUP FAN ON)
2. DAMPERS J & H ARE NORMALLY OPEN. CLOSED WHEN A.C. FAN OFF. (BACKUP FAN ON)
3. = "O" DESIGNATED EQUIPMENT NO.
4. AIRFLOW RATES (CFMs) SHOWN ON THIS FLOW DIAGRAM DRAWING ARE FOR DESIGN CONDITIONS ONLY.
5. DAMPER B & B' WILL REMAIN CLOSED ON LOSS OF POWER OR AIR.
6. CCR-CF1 FILTER TAG NUMBERS:
- HEPA FILTER (CCRHF)
- CARBON FILTER STACK A (CCRCFA)
- CARBON FILTER STACK B (CCRCFB)
- POST FILTER (CCRPF)
REVISION SIGNATURES
REV
00
DWG. NO. 252665-16
DWG. SIZE A
DWG. TYPE COMPANY
Entergy
STATION INDIAN POINT
BORO: WESTCHESTER
TITLE: CONTROL BUILDING (CCR)
CONTROL AND AIR FLOW DIAGRAMS
-UFSAR FIGURE NO. 9.9-1
APPROVALS
ENGINEERING
DESIGN
M. J. CAPUTO
12/20/93
A. B.
F. B.
SCALE: NONE
MY
COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED



OAGI0000215_1211

CHAPTER 10
STEAM AND POWER CONVERSION SYSTEM

10.1 DESIGN BASIS

10.1.1 Performance Objectives

The turbine-generator systems consist of components of conventional design acceptable for use in large power stations. The equipment is arranged to provide high thermal efficiency without sacrificing safety. The component design parameters are given in Table 10.1-1.

The steam and feedwater system is designed to remove heat from the reactor coolant in the four steam generators and produce steam for use in the turbine-generator. It can receive and dispose of, in its cooling systems and through atmospheric relief valves, the total heat existent or produced in the reactor coolant system following an emergency shutdown of the turbine-generator from a full-load condition.

The heat balance diagram at 1,078,200 kWe, maximum calculated; is shown on Figure 10.1-1. The stretch rating heat balance diagram, Figure 10.1-2A for 1,007,838 kWe incorporates the new electrical generator, uprated HP element and ruggedized LP element.

The system design monitors and restricts radioactivity discharge to normal heat sinks or the environment so that the limits of 10 CFR 20 are not exceeded under normal operating conditions or in the event of anticipated system malfunctions.

One steam turbine- and two electric motor-driven auxiliary feedwater pumps are provided to ensure that adequate feedwater is supplied to the steam generators for removing reactor decay heat under all circumstances, including loss of power and normal heat sink (e.g., condenser isolation, loss of circulating water flow). Feedwater flow can be maintained until either power is restored or reactor decay heat removal can be accomplished by other systems. Auxiliary feedwater pumps and piping are designed as seismic Class I components.

10.1.2 Load Change Capability

Load changes up to step increases of 10-percent and ramp increases of 5-percent per min within the load range of 15 to 100-percent and with manual rod control can be accommodated without reactor trip subject to possible xenon limitations late in core life. Similar step and ramp load reductions are possible within the range of 100 to 15-percent of full load. The reactor coolant system will accept a complete loss of load from full power with reactor trip. In addition, the turbine bypass and steam dump systems make it possible to accept a turbine load decrease of up to 25- to 50-percent of full power at a maximum turbine unloading rate of 200%/minute without reactor trip (see Section 7.3.3.1). The plant is normally in base-loaded operation.

10.1.3 Functional Limits

The system design incorporates backup means (power relief and code safety valves) of heat removal under any loss of normal heat sink (e.g., condenser isolation, loss of circulating water flow) to accommodate reactor shutdown heat rejection requirements. System atmospheric discharges under normal operation are made only if the releases are within the acceptable limits

of 10 CFR 20. All discharges to the atmosphere that may contain nonnegligible contributions to the offsite radiation environment are monitored to ensure acceptable radiation levels.

10.1.4 Secondary Functions

The steam and power conversion system provides steam for the turbine-driven auxiliary feedwater pump and for the operation of the air ejectors. The turbine bypass system is designed to dissipate the heat in the reactor coolant following a full-load trip. This heat is removed by the steam bypass of the turbine generator to the condenser circulating water and by the steam dump through the atmospheric power relief valves and safety valves in the event of loss of vacuum in the condenser.

TABLE 10.1-1

Steam and Power Conversion System
Component Design Parameters

Turbine Generator	
Turbine type	Four-element, tandem-compound, six-flow exhaust
Turbine capacity (MWe)	
Initial license application	906.6
At current licensed Reactor Power	1078.2
Generator rating (kVA)	1,439,200 (0.91pf; 75 psig H_2)
Turbine speed (rpm)	1,800
Condensers	
Type	RADIAL FLOW, SINGLE-PASS, DIVIDED WATER BOX, DEAERATING
Number	3
Condensing capacity (pounds of steam per hour, total)	7,243,971 (plus BFPT)
Condensate pumps	
Type	Eight-stage, vertical, pit-type, centrifugal
Number	3
Design capacity, each (gpm)	7,860
Motor type	Vertical, induction
Motor rating (hp)	3,000
Feedwater pumps	
Type	High-speed, barrel casing, single-stage, centrifugal
Number	2
Design capacity, each (gpm)	15,300
Pump drive	Horizontal steam turbine
Drive rating, each (hp)	8,350
Auxiliary feedwater pumps	
Number	3 (one steam-turbine- driven, two electric- motor-driven)
Design capacity (gpm)	800 (turbine-driven) 400 (each, motor-driven pump)
Auxiliary feedwater source	360,000-gal ensured reserve in 600,000-gal condensate tank Alternate supply from 1,500,000-gal city water tank

10.1 FIGURES

10.2 SYSTEM DESIGN AND OPERATION

10.2.1 Main Steam System

The main steam system, which is designed for a pressure of 1085 psig at 600°F, conducts steam from the four steam generators, which are located inside the containment structure, to the turbine generator unit, located in the Turbine Generator Building. The system, shown in Plant Drawings 227780, 9321-2017, and 235308 [Formerly UFSAR Figure 10.2-1, sheets 1, 2 and 3], has four 28-inch main steam pipes, one from each steam generator to the turbine stop and control valves. The four lines are interconnected local to the turbine. Each steam pipe has a swing disk type main steam isolation value (MSIV) and a swing disk type nonreturn valve located outside the containment. The MSIVs were redesigned to better withstand the dynamic forces associated with rapid closure in the event of a steam line rupture and thus reduce the likelihood of damage. The material for the valve discs was upgraded to stainless steel and the design of the disc arms was improved to reduce valve strains. In their Safety Evaluation Report (SER) dated September 15, 1976, the NRC determined that these modifications would satisfy General Design Criteria 4 of 10 CFR 50, Appendix A. A flow venturi upstream of the isolation valve measures steam flow, providing flow signals used by the automatic feedwater control system (see Section 7.3.3.3). The venturi also limits the steam flow rate in the event of a steam line break downstream of the venturi. Steam pressure is also measured upstream of the isolation valve.

The MSIVs each contain a free-swinging disk that is normally held out of the main steam flow path (valve open) by a solenoid controlled air piston. On receipt of a signal from the steam line break protection system described in Section 7.2.3.2.3.7, the solenoid valves are energized, releasing air from the piston and thereby allowing the MSIV to close. The MSIVs are designed to close in 5 seconds or less.

The nonreturn valves are activated on reverse flow of steam in case of accidental pressure reduction in any steam generator or its piping.

The system is classified as Class I for seismic design up to and including the isolation valves.

The steam line break incident is analyzed in Section 14.2.5.

10.2.1.1 Turbine Steam Bypass System

Excess steam generated by the reactor coolant system is bypassed, during conditions described below, from the four 28-in. main steam lines ahead of the turbine stop valves directly to the condensers by two 20-in. main steam bypass lines that run on either side of the turbine. From each 20-in. line, six 8-in. lines are taken, each with an 8-in. bypass control valve installed. Each bypass valve discharges into a 10-in. pipe that is connected by a manifold with one other 8-in. bypass valve and discharges into a 12-in. manifold. Each 12-in. manifold is taken to a separate section of the condenser where it discharges into the condenser through a perforated diffuser. Each bypass valve has a maximum capacity of 505,000 lb/hr and is rated at 442,000 lb/hr with 650 psia inlet pressure. The total capacity of all 12 bypass valves when operated with 765 psia in the steam generators (stretch rated load of 1078.2 MWe) is approximately 5,561,500 lb/hr (40-percent of the steam generator steam flow). The large number of small-size valves installed limit the uncontrolled steam flow to less than that of a steam generator/main steam safety valve should one valve stick open. Thus, a stuck open bypass valve will not result in a plant cooldown in excess of the steam line rupture/malfunction cases analyzed in UFSAR section 14.2.5. Additionally, local manually operated isolation valves are provided at each control valve.

On a turbine trip with reactor trip, the pressure in the steam generators rises. To prevent overpressure without main steam safety valve operation, the 12 turbine steam bypass valves open and discharge to the condenser for several minutes. The operation of the valves is initiated by a signal from the reactor coolant average temperature instrumentation. In the event of a turbine trip, all valves open fully in 3 sec. After the initial opening, the valves are modulated by the T_{avg} signal to reduce the average temperature and to maintain it at the no-load value. This is described further in Section 7.3.3.

After a normal orderly shutdown of the turbine generator leading to plant cooldown, the operator may select pressure control for more accurate maintenance of no-load conditions using the bypass valves to release steam generated by the residual heat. Plant cooldown, programmed to minimize thermal transients and based on residual heat release, is effected by a gradual manual adjustment of this pressure setpoint until the cooldown process is transferred to the residual heat removal system.

During startup, hot standby service, or physics testing, the bypass valves may be controlled manually from the pressure controllers located on the main control board.

The 12 bypass valves open on temperature control on turbine trip or large load rejection. All 12 valves are prevented from opening on loss of condenser vacuum. They are also blocked on trip of the associated condenser circulating water pump.

10.2.1.2 Steam Dump to Atmosphere

If the condenser heat sink is not available during a turbine trip, excess steam, generated as a result of reactor coolant system sensible heat and core decay heat, is discharged to the atmosphere.

There are four 6-in. by 10-in. and one 6-in. by 8-in. code safety valves located on each of the four 28-in. main steam lines outside the reactor containment and upstream of the isolation and nonreturn valves. Discharge from each of the 20 safety valves is carried to the atmosphere through individual vent stacks. The five safety valves in each main steam line are set to relieve

at 1065, 1080, 1095, 1110, and 1120 psig. The total relieving capacity of all 20 valves is 15,108,000 lb/hr.

In addition, four 6-in. power-operated relief valves are provided, which are capable of releasing steam to the atmosphere to dissipate the sensible and core decay heat. These valves are automatically controlled by pressure or may be manually operated from the main control board and are capable of releasing 10-percent of the equivalent rated steam flow (1,390,375 lb/hr of steam at 1020 psig pressure). One power-operated relief valve is located on each main steam line upstream of the swing disk isolation valve. Discharge from each of the four power relief valves is carried to the atmosphere through individual muffled (silencer-fitted) vent stacks. In addition, the power-operated relief valves may be used to release the steam generated during reactor physics testing and plant hot standby operation if the main condenser is not available.

10.2.1.3 Low-Pressure Steam Dump System

A low pressure steam dump system is provided to bypass steam from the exhaust lines from the high-pressure turbine directly to the condenser. The system is provided to minimize turbine speedup immediately following a turbine trip or generator breaker opening.

The low-pressure steam dump system consists of six dump valves, which connect the high-pressure turbine exhausts to the condensers through individual breakdown orifices. An isolation valve is provided for each dump valve. At any generator breaker opening, turbine trip, or overspeed trip with the isolation valves open and dump valves closed, the dump valves would be activated. This would divert approximately 25-percent of the steam available to overspeed the turbine to the condensers, thus reducing the potential maximum turbine speed.

10.2.1.4 Steam for Auxiliaries

The steam for the turbine-driven auxiliary feedwater pump is obtained from two of the 28-in. steam-generator outlet mains upstream of the swing disk isolation valves. A pressure-reducing control valve reduces the steam to 550 psig for the auxiliary turbine.

Auxiliary steam for the turbine gland steam supply control valve, the three steam-jet air ejectors, the reheater section of the six moisture separator-reheaters, the three priming ejectors, and supplementary steam for the main feed pump turbines is obtained from branches on the steam lines ahead of the turbine stop valves. Pressure-reducing stations are used for the priming and main air ejectors. Reheater temperature control valves are located in the steam line to the reheaters. The design pressure and temperature for this system are 1085 psig and 600°F. Steam from six extraction openings in the turbine casings is piped to the shells of the three parallel strings of feedwater heaters. The first extraction point originates at the high-pressure turbine casing and supplies steam to the shells of the No. 26 A/B/C (high-pressure) feedwater heaters. The second extraction point originates in the moisture preseparators located in the high-pressure turbine exhaust piping ahead of the moisture separators and supplies steam to the No. 25 A/B/C (low-pressure) feedwater heaters. The third, fourth, fifth, and sixth extraction points all originate at the low-pressure turbine casings and supply steam to the Nos. 24 A/B/C, 23 A/B/C, 22 A/B/C, and 21 A/B/C (all low-pressure) feedwater heaters, respectively.

Nonreturn valves are provided in all but the two lowest pressure extraction steam lines to prevent turbine overspeed from the backflow of flashed condensate from the heaters after a turbine trip. All of these valves are air-cylinder operated and close automatically on turbine trip. Two of these valves are installed in each of the steam lines to heater Nos. 25 and 26 and also in

the extraction line from each moisture preseparator. One of these valves is installed in the steam lines to heater Nos. 23 and 24. The low-pressure fifth and sixth point extraction lines are located entirely in the condenser shells and do not contain nonreturn valves.

10.2.1.5 Steam Generator Blowdown

Each steam generator is provided with two 2½ in. bottom blowdown connections to control the shell solids concentration. The two connections are at the same level but are on opposite sides of the shell. Piping from 2½ to 2 in. reducing inserts at each of the two connections join to form a 2-in. blowdown header for each steam generator. The bottom of each steam generator is also provided with a drain connection, except in the case of the steam-generator No. 21 drain, which has been blanked off.

Each blowdown line has two diaphragm-operated trip valves acting as isolation valves and a hand shutoff valve. The isolation valves are solenoid controlled and open when their individual solenoid is energized. The isolation valves will fail shut on loss of air or power. Each valve is provided with position indicating lights in the control room. In addition to the isolation valves, each line includes a manually operated needle-type flow control valve for blowdown flow or sample flow adjustment and an air-operated valve acting as a fluid trap valve. The steam-generator sample line is taken off from the blowdown line outside containment downstream of the isolation valves. A small flow from each sample line is combined and is monitored for radiation. In the event of a high-radiation signal, both isolation diaphragm valves in the blowdown lines close automatically. They also shut on a phase A containment isolation signal and on an automatic start signal for the motor-driven auxiliary feedwater pumps. The two isolation valves and the fluid trap valve are electrically interlocked to preclude water hammer during closure of the valves on an isolation signal. On an open signal, the isolation valves open prior to the fluid trap valve.

Blowdown from all four steam generators passes to the blowdown flash tank. The flashed vapor is discharged to the atmosphere while the condensate drains by gravity through a service water discharge line into the circulating water discharge canal.

If drains from the blowdown flash tank become contaminated, or in the event of primary to secondary coolant leakage in one or more of the steam generators, the blowdown may be manually diverted to the support facilities (Unit 1 site) secondary boiler blowdown purification system flash tank. This system cools the blowdown and either stores it in the support facilities waste collection tanks or purifies it.

The normal full-load blowdown rate from four steam generators is approximately 57,455 lb/hr or 0.41-percent of feedwater flow. The design basis blowdown flow for four steam generators is 265,200 lb/hr. The maximum limit for blowdown flow is 198,900 lb/hr per steam generator for short periods of operation, not to exceed one year cumulative over the life of the steam generator. This provides for occasionally higher blowdown rates should they be required to reduce solids concentration, and/or sodium carry over via the feedwater system in case of small condenser leaks.

10.2.2 Turbine Generator

The original turbine generator had a guaranteed capability of 1,021,793 kWe at 1.5-in. Hg absolute exhaust pressure with zero percent makeup and six stages of feedwater heating. The unit currently operates at 1800 rpm with steam supplied ahead of the main stop valves at 737

psia, 509°F, and enthalpy of 1200 BTU/lb. Steam is admitted to the turbine through four stop valves and four control valves. The expected throttle flow at 1078 MWe is 12,971,500 lb of steam per hour.

The turbine (TC6F-45) is a four-casing, tandem-compound, six-flow exhaust unit with 45-in. last row blades and consists of one double-flow high-pressure element in tandem with three double-flow low-pressure elements. The low pressure rotors are of the fully-integral design, which eliminates the separate discs (with their bores and keyways) of the earlier design. Steam, after passing through the stop and control valves, passes through the high-pressure turbine, then through the moisture preseparators, through the moisture separator reheaters, and then to the low-pressure turbines as shown in Plant Drawings 227780, 9321-2017, and 235308 [Formerly UFSAR Figure 10.2-1, sheets 1, 2 and 3].

There are four moisture preseparators and six horizontal-axis, cylindrical shell, combined moisture separator/steam reheater assemblies. Steam from the exhaust of the high-pressure turbine element passes through the preseparators and enters each reheater assembly at one end. Internal manifolds in the lower section distribute the wet steam. The steam then rises through a chevron moisture separator where the moisture is removed and drained to a drain tank. The steam leaving the chevron separator flows over a tube bundle where it is reheated. This reheated steam leaves through nozzles in the top of the assemblies and flows to the low-pressure turbines. The tube bundle is supplied with main steam from ahead of the turbine throttle valves, which condenses in the tubes and leaves as condensate. Condensate from the reheater assemblies flows to the high-pressure heaters. The turbine-generator building general arrangement, operating floor, and cross section are shown in Plant Drawings 9321-2004 and 9321-2008 [Formerly UFSAR Figures 10.2-2 and 10.2-3].

The turbine oil system consists of a high-pressure hydraulic control system and a low-pressure lubrication system. Oil is also used to seal the generator shaft seals to prevent hydrogen leakage from the generator into the turbine building. The oil pump mounted on the main turbine shaft normally supplies all oil requirements. A motor-driven auxiliary oil pump supplies the oil required during turbine startup and whenever there is low pressure in the bearing oil header. The auxiliary unit is a centrifugal pump driven by a 150-hp motor. Oil is supplied to the hydraulic control mechanisms at 300 psig. A motor-driven bearing oil pump is also provided to supply oil whenever there is a low pressure in the bearing oil header. This is a centrifugal-type pump with a 75-hp motor. During startup, these auxiliary oil pumps supply all the oil while the main pump acts against a closed check valve. An alternating current motor-driven oil pump is provided for turning gear and emergency operation. A direct current motor-driven oil pump, operated from a station battery, provides additional backup to ensure a supply of lubricating oil to the machine. An alternating current motor-driven generator seal oil pump is furnished for normal operation with a direct current motor-driven backup pump to ensure confinement of the hydrogen within the generator.

A continuous bypass turbine oil purification system removes contaminants from the oil.

To maintain shaft alignment while the unit is down, a motor-driven turning gear is provided.

In 1987, the original generator was replaced with a generator of larger capacity. The new generator has a hydrogen cooled rotor and a water cooled stator, and is rated at 1,439,000 kVA at 75 psig hydrogen pressure. It has sufficient capability to accept the gross kilowatt output of the steam turbine with its control valves wide open, at a reactor power of 3216 MWt.

10.2.3 Turbine Controls

High-pressure steam enters the turbine through four stop valves and four governing control valves. The four main stop valves are designed for the specific operating conditions. Each stop valve is a single-seated, oil-operated, spring-closing valve controlled primarily by the turbine overspeed trip device. The turbine overspeed trip pilot is actuated by one of the following to close the stop valves:

1. Turbine thrust bearing trip.
2. Low bearing oil pressure trip.
3. Low condenser vacuum.
4. Solenoid trip.
5. Overspeed trip.
6. Manual trip.

Each stop valve has limit switches that operate position lights on the main control board.

Test switches on the main control board permit test closure of each valve. The valve position can be observed at the turbine. Periodic tests exercise the stop valves and ensure their ability to close during an emergency. The turbine steam stop and control valves shall be tested at a frequency determined by the methodology presented in WCAP-11525 “Probabilistic Evaluation of Reduction in Turbine Valve Test Frequency,” and in accordance with established NRC acceptance criteria for the probability of a missile ejection incident at IP-2. In no case shall the test interval for these valves exceed one year.

Before a stop valve can be opened, the pressure across the valve must be equalized. This is done by opening a small bypass valve around each of the stop valves.

Electrical interlocks (e.g. circuit breaker position contacts, instrument contacts, relay contacts, valve limit switch contacts) are utilized in control circuits that actuate the turbine trip auxiliary relays. This will initiate a reactor trip.

Four hydraulically operated control valves of the single-seated plug type open and close in sequence to control steam admission to the turbine. They are actuated by the turbine speed governor, which is responsive to turbine speed.
It includes:

1. A speed changer or synchronizing device.
2. A load limit device that must be reset after operation of the overspeed trip before the control valves can be opened.
3. A second load limit device without reset, furnished to give redundancy of load cutback following a rod drop.
4. The governing emergency trip valve, actuated by loss of low pressure auto stop oil pressure.
5. An auxiliary governor, responsive to the rate of turbine speed increase to close the control valves.

Each control valve has a motor-controlled hydraulic pilot valve to test the operation of the valve. Test switches with indicating lights are provided on the main control board turbine section. Removable strainers are located in each control valve body to protect the valves and turbine from foreign material in the steam.

The normal governing devices that operate through hydraulic relays to operate the control valves are as follows:

1. The governor handwheel at the unit.
2. The governor synchronizing motor, which is controlled by a switch on the electrical section of the main control board and is used for raising or lowering turbine speed or load.
3. The load limit handwheel at the unit.
4. The load limit motor, which is controlled by a switch on the turbine section of the main control board and by a reactor control rod drop runback signal (this is described further in Chapter 7).

The preemergency device functions similarly to the normal governing devices by operating the control valves in case of abnormal operating conditions in the auxiliary governor. This preemergency device closes the control valves on rapid increase in turbine speed. The control valves will be actuated by either the speed governor or load limit. The device delivering the lowest oil pressure will be in control. Pressure gauges on the main control board indicate the oil pressure from these devices.

The emergency devices that will trip the stop valves, the control valves, and the air relay dump valves are as follows:

1. Solenoid trip.
2. Low condenser vacuum trip.
3. Low bearing oil trip.
4. Thrust bearing trip.
5. Manual trip at the unit.
6. Overspeed trip.

The solenoid trip is produced directly by the following:

1. Reactor trip breakers opening.
2. Turbine generator primary lockout relay.
3. Turbine generator backup lockout relay.
4. Manual trip push button at control board.
5. Vibration.
6. Main steam isolation valve closure.
7. Steam generator high-high level.
8. Deleted
9. Safety injection.
10. Differential expansion
11. AMSAC trip
12. IEOP overspeed signal
13. Loss of stator or rectifier cooling

The solenoid trip signals and logic are shown in Plant Drawing 225096 [Formerly UFSAR Figure 7.2-3].

The mechanical overspeed trip mechanism consists of an eccentric weight mounted in the end of the turbine shaft that is balanced in position by a spring until the speed reaches the point at which the trip is set to operate.

The centrifugal force overcomes the restraining spring and the eccentric weight flies out striking a trigger that trips the overspeed trip valve and releases the autostop fluid to drain. The resulting decrease in autostop pressure causes the governing emergency trip valve to release the control oil pressure.

This closes the main stop and control valves. An air pilot valve used to control the extraction lines nonreturn valves is also actuated by the autostop pressure.

The independent electrical overspeed protection system (IEOPS), which is not required by the Technical Specifications, utilizes the output of Hall Effects probes mounted around the turning gear to detect and measure turbine shaft speed. The system generates tripping logic signals on overspeed in three redundant speed channels. These signals are used in a 2/3 logic matrix to energize redundant control relays that energize redundant solenoid-operated dump valves attached to the actuator of each turbine stop and control valve (two dump valves per actuator). Opening either dump valve drains the oil under the piston of the actuator, thus closing the corresponding turbine stop and/or control valve. Since the turbine stop and control valve in each line are in series, closure of either one will stop steam flow in that line.

The autostop valve is also tripped when any one of the protective devices is actuated. The protective devices include low bearing oil pressure, solenoid, thrust bearing, and low vacuum trips. These devices are all included in a separate assembly, but they are connected hydraulically to the overspeed trip valve. An additional protective feature includes a turbine trip following a reactor trip.

When the unit load is at or above P-8, trip of the turbine generator requires a reactor trip.

A loss of one main feedwater pump initiates automatic turbine load cutback. This is described further in Chapter 7.

10.2.4 Circulating Water System

Hudson River water is used for the condenser circulating water. River water flows under the floating debris skimmer wall, through traveling screens, and into six separate screenwells. The traveling screens, which operate continuously, are designed to reduce the potential for fish and debris from entering the circulating water pumps. Each screenwell is provided with stop logs to allow dewatering of any individual screenwell for maintenance purposes.

The water from each individual screenwell flows to a motor-driven, vertical, mixed flow condenser circulating water pump. Each of the six condenser circulating water pumps provides 140,000 gpm and 21-ft total dynamic head when operating at 254 rpm and 84,000 gpm and 15-ft total dynamic head when operating at 187 rpm. Each pump is located in an individual pump well, thus tying a section of the condenser to an individual pump. The circulating water is piped to the condensers and is discharged back into the river far enough away from the intake to minimize recirculation. To protect the traveling screens against ice during freezing water conditions, bar grates with ice shields are installed upstream of the traveling screens at the inlet of the intake bays. Heating elements located in the traveling screen head section prevents ice from forming on the screens.

Sodium hypochlorite, is available for injection into the circulating water to prevent the buildup of bacterial slime on the traveling water screens, condenser tubes, and piping. Sodium hypochlorite may be stored in two 4000-gal tanks in the hypochlorite room of the Unit 1 screenwell house. One of the tanks, #12 Sodium Hypochlorite Tank, has been isolated and permanently closed. The remaining tank, #11 Sodium Hypochlorite Tank, supplies a 500 gallon day tank on 15 ft. elevation to supply the hypochlorite feed pump skid.

10.2.5 Condenser and Auxiliaries

Three surface-type, single-pass, radial flow condensers with bolted divided water boxes at both ends are provided. Fabricated steel water boxes and shell construction is used. Hotwell design is for at least 4-min storage while operating at maximum turbine throttle flow with free volume for condensate surge protection. The hotwells are longitudinally divided to facilitate the detection of condenser tube leakage. Each half is provided with separate conductivity measurement devices. In the event of high conductivity (high salinity) in a hotwell, it will be manually isolated. The condensate will be dumped overboard instead of being used to provide suction for the condensate pumps. The deaerating hotwells reduce the residual oxygen in the condensate to less than 0.01 cm^3/l. Condensers 21, 22 and 23 use titanium tubes and tube sheets. Water box manholes are provided for access. Provision is made steam turbine bypass condensing arrangements to condense turbine bypass steam for controlled startup and to condense residual and decay heat steam following a shutdown.

Three motor-driven, eight-stage, one-third capacity, vertical, pit-type, centrifugal condensate pumps are provided, each taking suction from the condenser hotwells. The condensate pumps discharge into three separate parallel streams of feedwater heaters and provide the suction supply to the feedwater pumps.

Each condenser has one four-element, two-stage air ejector with separate intercondensers and common aftercondensers as shown in Figure 10.2-4. The ejectors function by using steam from the main steam system supplied through a pressure-reducing valve. Motor driven vacuum pumps are also provided. Air removed from the condenser is monitored for radioactivity. In the event of a steam-generator leak and the subsequent presence of radioactive contaminated steam in the secondary system, the radioactive noncondensable gases that concentrate in the air ejector effluent will be detected by this radiation monitor. A high activity level signal automatically diverts the exhaust gases from the vent stack to the containment.

For initial condenser shell side air removal, three noncondensing priming ejectors are provided. Each has a capacity of 900 cfm. This apparatus may be used during periods of plant shutdown where decay heat is involved. The main ejectors will also be operated at the same time to ensure that the effluent is monitored for radioactivity.

Examinations of condensers are conducted regularly during scheduled outages in accordance with engineering recommendations. Examinations typically include visual inspections and eddy current tests.

For startup operation two full size motor driven vacuum pumps with all ancillary equipment are installed to reduce oxygen levels in the feedwater and condensate prior to and during start-up. The pumps are also capable of being used for the normal holding operation in lieu of the air ejector system or as a backup to the air ejector system. For the start-up operation steam from the house boiler is used for turbine gland sealing.

10.2.6 Condensate and Feedwater System

The condensate and feedwater system is designed to supply a total of 13,957,950 lb of feedwater per hour to the four steam generators at a turbine load of 1078 MW(e). This system is composed of:

1. A condensate system that collects and transfers condensed steam and the drains from five feedwater heaters through five stages of feedwater heating to the suction of the main feedwater pumps.

2. A condensate makeup and surge system that maintains a normal water level in the condenser hot wells.

3. A heater drain system that collects and transfers the drains from Nos. 25 and 26 feedwater heaters, the moisture preseparators and the six moisture separator/reheaters to the suction of the main feedwater pumps.

4. A feedwater system that delivers the condensate and heater drains through the final stage of feedwater heating to the steam generators.

5. An auxiliary feedwater system that provides a flow of water from the condensate storage tank to the steam generators when the main feedwater pumps are unavailable. The flow is equivalent to that required for makeup because of reactor core decay heat removal requirements.

10.2.6.1 Condensate System

The condensate system transfers condensate and low-pressure heater drains from the condenser hotwell through five stages of feedwater heating to the suctions of the main feedwater pumps. The system flow diagram is shown in Plant Drawings 9321-2018 and 235307 [Formerly UFSAR Figure 10.2-5].

Three one-third size condensate pumps, arranged in parallel, take suction from the bottoms of the condenser hotwells. The pumps discharge into a common header that carries a portion of the condensate through three steam jet air ejector condensers, arranged in parallel, and through one gland steam condenser. The remaining portion flows in parallel with the first flow path, bypassing the steam jet air ejectors and the gland steam condenser. The second flow path rejoins the first in the header downstream of the gland steam condenser.

The condensate pumps are eight-stage, vertical, pit-type pumps. Each pump is rated at 7860 gpm and 1150-ft total dynamic head when operating at 1185 rpm. A standard packed stuffing box is used for shaft sealing. The pump bearings are lubricated by the pumped liquid. Each pump is driven through a solid coupling by a 3000-hp, vertical, solid shaft, induction motor that has an open drip-proof enclosure. The condensate pumps are operated by manual control on the main control board. To maintain the condenser vacuum and turbine steam seals during startup, shutdown, and at very low loads, an 8-in. condensate recirculation line, containing a diaphragm-operated valve, is provided to maintain minimum flow through the air ejector condensers and gland steam condenser. The recirculation line originates at the condensate header downstream of the gland steam condenser and terminates at the condenser hotwell.

The diaphragm-operated recirculation valve is automatically controlled by the minimum flow required by the air ejector condensers.

The 24-in. header divides into three 14-in. lines downstream of the gland steam condenser. From these lines, the condensate passes through the tube sides of three parallel strings of two low-pressure feedwater heaters. The flow from these heaters is combined in another 24" pipe, and then divided to go to the remaining three strings of three low-pressure heaters. After the No.25 feedwater heater, the three condensate lines join into a common header. The heater drain pump discharge enters this header and then continues on to the suction of the main feedwater pumps.

Each parallel string of feedwater heaters may be taken out of service by closing a manual gate valve at the inlet to the string of heaters and at the outlet of the string of heaters.

The condensate makeup and surge systems maintain normal water level in the condenser hotwell.

The makeup system connects the 600,000-gal capacity condensate storage tank to a diffusing pipe in the condenser shell. This line contains a diaphragm-operated valve that can automatically open on low level in the condenser hotwell to pass makeup water from the tank to the condenser. This valve may be operated manually or automatically. An isolating valve will close the condenser makeup before the condensate storage tank level reaches its Technical Specification minimum capacity. This will ensure a reserve of condensate for the auxiliary feedwater pumps that will hold the plant at hot shutdown for 24 hr following a trip at full power.

The condensate surge system connects the condensate pump discharge header to the condensate storage tank. This line contains a diaphragm-operated valve that automatically opens on high level in the condenser hotwell to pass excess condensate from the condensate pump discharge header to the condensate storage tank.

Hotwell levels are indicated on the main control board. Should the automatic makeup valve or the surge valve become inoperative, it may be isolated from its respective system and the hotwell level controlled from the control room by remote manual positioning. The condenser hot-wells contain 114,000 gal, which is equal to approximately 5.63-min condensate flow at 1078 MWe load.

The drains from the No. 26 A/B/C feedwater heaters flow to the heater drain tank. Normal condensate level is maintained in the No. 26 heaters by diaphragm-operated level control valves.

The drains from the No. 25 A/B/C feedwater heaters flow by gravity directly to the heater drain tank. There are no level control valves in the drains from these heaters.

Two half-size heater drain pumps pump the drains from the drain tank into the condensate header upstream of the main feedwater pumps. Both pumps discharge through diaphragm-operated level control valves.

The heater drain pumps are 14-stage, vertical, enclosed suction-type pumps. Each pump is rated at 4150 gpm and 720-ft total dynamic head when operating at 1170 rpm. Each pump is driven through a solid coupling by a 1000-hp, vertical, solid shaft, induction motor that has an open drip-proof enclosure.

The heater drain pumps are operated by manual controls on the main control board. A heater drain pump is automatically stopped on low drain tank level or if the flow falls below a set minimum. After the pump has stopped, the water level in the heater drain tank will increase. An alarm sounds in the control room on both tank low level and pump low flow.

When a high level occurs in the heater drain tank, diaphragm-operated valves open to discharge the excess condensate from the heater drain tank directly to the shell of a condenser. An alarm sounds in the control room. The heater drain tank has a 5660-gal storage capacity at normal water level or approximately 0.64-min storage of drains at the normal full load of 1078 MWe.

Drains from the Nos. 24, 23, and 22 feedwater heater strings normally flow through diaphragm-operated level control valves to the shells of the next lowest pressure feedwater heater. On high level in any heater, a separate high-level drain from the heater discharges directly to the condenser.

Drains from the No. 21 feedwater heaters normally flow through diaphragm-operated level control valves to the condenser. When a high level occurs in the heaters, a separate high-level drain for each heater discharges to the condenser.

10.2.6.2 Main Feedwater System

Two half-size steam-driven main feedwater pumps increase the pressure of the condensate for delivery through the final stage of feedwater heating and then the feedwater regulating valves to the steam generators. The system flow diagram is given in Plant Drawing 9321-2019 Figure 10.2-7.

The main feedwater pumps are single-stage, horizontal, centrifugal pumps with barrel casings. Each pump is rated at 15,300 gpm and 1700-ft total dynamic head when operating at 4740 rpm. Seal-water injection is used for shaft sealing. Bearing lubrication for both the pump and its turbine drive is accomplished by an integral lubricating oil system. Normal circulation of the lubricating oil is by a motor-driven pump. The lubricating oil system includes a reservoir, a cooler, and two motor-driven oil pumps. Each main feedwater pump is driven through a flexible coupling by an 8350-hp horizontal steam turbine that uses steam from the discharge of the three reheater moisture separators on one side of the turbine hall. The main feedwater pumps are operated automatically by the feed control system. Manual controls are also provided on the main control board for remote operation and testing during normal operation. During normal startup of the plant, these pumps are started locally. A minimum flow control system is provided to ensure that each pump is handling at least a 3000-gpm flow at all times.

Above a preset turbine power, the operator may arm the condensate pump auto-start circuit (MBFP trip or MBFP low suction pressure or running condensate pump trip).

Low suction pressure starts any idle condensate pumps (if armed) and reduces the feed pump turbine speed to maintain suction pressure. Normal speed is regained when the suction pressure and flow is reestablished. High discharge pressure reduces turbine speed to prevent excessive pressure in the feed piping.

In the original design, a bypass was provided around the low pressure heaters, which was to be used to provide sufficient suction pressure at the feed pumps during a transient when flashing

might occur in the heater drain tank and affect the performance of the heater drain pumps. The bypass valve was retired in place when operating experience proved that it was not required to perform this function.

High main feedwater pump bearing temperatures are alarmed in the main control room. However, they do not automatically stop the pump.

The two parallel main feedwater pumps operate in series with the condensate pumps and discharge through check valves and motor-operated gate valves into a common header. The feedwater then flows through the three parallel, high-pressure feedwater heaters into a common header. Four parallel 18-in. lines containing the feedwater metering and regulating stations feed the four steam generators.

Shutoff valves at the inlets and outlets of the feedwater heaters permit a heater to be taken out of service. Bypass lines are provided around the heaters to allow operation when a heater is out of service for maintenance.

A long loop recirculation line, from the high pressure feedwater header, leading back through an installed particulate removal filter and portable demineralizers to the condenser, is available for secondary coolant cleanup during plant outages.

The steam-generator feedwater metering and regulating stations measure, indicate, record, and control the water level in each of the four steam generators. A conventional three-element system receives flow and load signals from the reactor protection system through isolation amplifiers and compares the difference between steam and feedwater flows to adjust the level setpoint. The deviation of level measurement from this setpoint positions the feedwater control valve accordingly. Totalized steam flow controls the speed of the main feedwater pump turbines.

Low-flow feedwater regulating valves bypass the main control valves for the control of low-load feedwater flow.

On trip of one main feedwater pump above a preselected turbine power, the following actions are automatically initiated to prevent a trip of the reactor and turbine-generator.

a. The turbine load limit is run back to reduce the steam demand.
b. Any idle condensate pumps are started. (if armed)
c. Non tripped pump to pick up additional load.

A reactor trip is actuated on a coincidence of steam flow-feedwater flow mismatch, coupled with a low level in the corresponding steam generator. A reactor trip is also initiated on a coincidence of two-out-of-three low-low water-level signals from any one steam generator. Whenever this reactor trip occurs, the main feedwater valves move to the fully open position in response to an increased level demand signal from the feedwater control system. This provides an additional heat sink for the reduction of reactor coolant temperature to the no-load average temperature value. The feedwater regulating valves close on one of the following conditions:

1. High-high steam generator water level.
2. Reactor trip coincident with low T_{avg} signal.
3. Safety injection signal.

In the case of reactor trip coincident with low T_{avg} signal, the low flow feedwater bypass valve closure may be delayed by means of an installed timer to allow main feedwater to moderate the cooler auxiliary feedwater before it enters the steam generators. The feedwater control system is an electronic analog instrumentation system.

Readout and control equipment is as follows:

1. Wide and narrow range level shown on recorder calibrated for cold conditions in the steam generator, permits observation of the level essentially over the full height of each steam-generator shell.

2. Visual indication is provided in the main control room of feedwater flows in pounds per hour for each steam generator.

3. A leading edge flow meter in each steam-generator feedline provides feedwater flow data for thermal power calculations.

4. Each flow channel and each narrow-range level channel is indicated on the main control board.

5. Each feedwater controller has one manual control station. The unit consists of an auto/manual transfer switch and an analog output control, which serves as the valve position signal when in "Manual." The "Automatic" setpoint is preset, but adjustable in the instrument rack.

6. Other manual control stations are used to position auxiliary feedwater regulating valves.

10.2.6.3 Auxiliary Feedwater System

This system is used for normal startup. The auxiliary feedwater system supplies high-pressure feedwater to the steam generators to maintain a water inventory. This is needed to remove decay heat energy from the reactor coolant system by secondary-side steam release in the event that the main feedwater system is inoperable. The head generated by the pumps is sufficient to deliver feedwater into the steam generators at safety valve pressure. Diverse auxiliary feedwater supplies are provided by using two pumping systems using different sources of motive power for the pumps. The system flow diagram is given in Figure 10.2-7.

The capacity of each system is set so that all four steam generators can be supplied with auxiliary feedwater. Under limiting conditions, at least two steam generators will not boil dry nor will the primary side relieve water through the pressurizer relief/safety valves following a loss of main feed-water flow. Further details are given in Section 14.1.9.

One system uses a steam-turbine-driven pump with the steam capable of being supplied from two of the steam generators. This system is designed to supply up to 800 gpm of feedwater (200 gpm to each steam generator). The estimated (expected) design performance characteristic of the pump is given in Figure 10.2-8. The technical specification requirement is that this pump be capable of supplying at least 380 gpm.

Steam to drive the turbine is supplied from two of the main steam lines upstream of the isolation valves at steam-generator outlet pressure and is reduced to within the 550-psig turbine design

pressure by a pressure-reducing control valve (PCV-1139). The turbine is started by opening the pressure-reducing valve between the turbine supply steam header and the main steam lines. The turbine sleeve journal bearings are ring oil-lubricated, water cooled. The pump uses oil slinger lubricated ball bearings. The drive is a single-stage turbine, capable of quick starts from cold standby, and is directly connected to the pump.

The speed of the turbine can be adjusted manually via a remote pneumatic speed controller (HC-1118). It is normally set at zero percent (i.e., minimum setting of approximately 3200 rpm). Upon generation of an automatic start signal for the turbine-driven pump, PCV-1139 will open, and the turbine will start and run. The pump itself will only operate on recirculation flow since the auxiliary feedwater regulating valves in its discharge are normally closed. In order to deliver flow to the steam generators using this pump, the operator must open one or more of the associated auxiliary feedwater regulating valves, and manually adjust the speed controller for the turbine. Both of these actions can be performed from the central control room control board or locally at the valves. The auxiliary feedwater regulating valves are pneumatically operated. PCV-1139 opens fully on loss of control air. All pneumatic instruments and valves associated with the auxiliary feedwater system requiring instrument air for their safety function have automatic nitrogen back-up.

Since the single failure criterion for loss of normal feedwater events can be satisfied by one motor-driven auxiliary feedwater pump providing flow for a sufficiently long period of time before an operator action is taken to align the turbine-driven auxiliary feedwater pump, manual alignment of the turbine-driven pump is acceptable. Further details are given in Section 14.1.9.

The other system uses two motor-driven pumps with lubricated ring oiled ball bearings. Each pump has a design capacity of 400 gpm, and the discharge piping is arranged so that each pump supplies two of the four steam generators. The estimated design performance characteristic for these pumps is given in Figure 10.2-9. The technical specification requirement is that each pump be capable of supplying at least 380 gpm.

The motors are of open drip-proof design with ball bearings. In the event of complete loss of power, electrical power is automatically obtained from the diesel generators. Each motor-driven pump is provided with a discharge pressure sustaining control system to prevent the pump from "running out" on its curve. The Regulating valves are pneumatically operated and have an automatic nitrogen bottle backup system to maintain operability in the event that control air is lost. A recirculation line and control system are provided for each pump to maintain a minimum flow when it is running.

Upon generation of an automatic start signal for the motor-driven auxiliary feedwater pumps, both pumps will start and each will deliver at least 380 gpm. The regulating valves for each motor-driven pump are controlled such that each steam generator receives approximately 190 gpm. An additional restriction on auxiliary feedwater flow when a steam generator feed ring has been uncovered for an extended period of time provides added assurance against a potentially damaging water hammer upon initiation of cold auxiliary feedwater to the steam generators. This restriction limits auxiliary feedwater flow to the affected steam generator(s) until an increase in steam generator level can be seen. The accident analyses in Sections 14.1.9 (Loss of Normal Feedwater) and 14.1.12 (Loss of All AC Power to the Station Auxiliaries) assume that only one motor-driven pump starts one minute after accident initiation and delivers 380 gpm (nominal 190 gpm to each of two steam generators). Further, operator action is credited at 10 minutes after reactor trip to start the second motor-driven pump or to align the steam-driven-turbine pump.

The auxiliary feedwater pumps are located in an enclosed room in the auxiliary feedwater building, which houses the area of the main steam and feedwater penetrations immediately outside the reactor containment.

Safety-grade flow measurement devices are installed in the feedwater supply to each steam generator with indicators on the main control board. In addition, wide-range and safety-grade narrow-range steam-generator level indications are provided in the main control room. These provide the operator with the information necessary to route auxiliary feedwater discharge flow through the remote manual discharge regulating valves.

The distribution piping is seismic Class I throughout. It is designed to ensure that a single fault will not restrict the system function.

The overall seismic qualification of the auxiliary feedwater system was reviewed and found acceptable by NRC Safety Evaluation Reports issued September 7, 1982 and September 29, 1987.

The water supply source for this system is redundant. The main source is by gravity feed from the condensate storage tank. This tank is sized to meet the normal operating and maintenance needs of the turbine cycle systems. However, a minimum water level will be maintained, equivalent to the steam generation from 24 hr of residual heat generation at hot shutdown conditions. The condensate storage tank is considered the safety grade source for the auxiliary feedwater system.

The auxiliary feedwater pumps can draw from an alternative supply of water to provide for long-term cooling. This alternative supply is from the 1.5 million gal city water storage tank. This supply is manually aligned to the auxiliary feedwater pumps in the event of unavailability of the condensate storage tank.

The auxiliary feedwater pumps are automatically started on receipt of any of the following signals:

1. Steam-driven auxiliary feedwater pump:
   a. Low-low water level in any two of the four steam generators.
   b. Loss of offsite power concurrent with a unit trip and with no safety injection signal present.

2. Motor-driven auxiliary feedwater pumps:
   a. Low-low water level in any steam generator.
   b. Automatic trip of main feedwater pumps [*Note - One main feedwater pump trip automatically sends a demand start signal to both motor-driven auxiliary feedwater pumps.*] as indicated by loss of main feed pump control oil pressure after manual control switch was last operated to the "start" position.
   c. Safety injection signal.
   d. Loss of outside power concurrent with a unit trip.

The auxiliary feedwater system automatic initiation signals and circuits meet safety-grade requirements. Interfacing AMSAC signals and circuits which are not safety-grade are provided with Class 1E isolation devices.

In the event of a complete loss of offsite power, the electrical power is supplied by the diesel generators as described in Chapter 8.

In addition, the steam-driven and the motor-driven auxiliary feedwater pumps can be started manually from the control room and locally at the pumps.

In the event of a loss of the condensate storage tank supply (e.g., one or both condensate storage tank discharge valves are closed), immediately place the auxiliary feedwater pump controls in the manual mode. Within 1 hour either the valve(s) shall be reopened or the valves from the alternate city water supply shall be opened and the auxiliary feedwater pump controls restored to the automatic mode.

10.2.6.4 System Chemistry

Steam-generator water chemistry is maintained within the required water quality limits. A nitrogen blanket in the condensate storage tank minimizes oxygen ingress. During outages, as part of the wet lay-up process, nitrogen is introduced as a sparging gas to displace air from the steam generators. Hydrazine is added to the condensate for oxygen control and ammonium hydroxide and/or volatile amines are added to maintain the pH at the optimum value for the materials of construction for the system.

No radiation shielding is required for the components of the steam and power conversion system. During normal operation, continuous access to the components of this system outside of containment is possible.

Under normal operating conditions, no radioactive contaminants are present in the steam and power conversion system. It is possible for this system to become contaminated through steam-generator tube leaks. In this event, any contamination is detected by monitoring the steam-generator shell-side blowdown sample points and the condenser air ejector discharge. Operation with a steam-generator tube leak is discussed in Chapter 14. Radiation monitors are installed in the main steam lines outside of the containment wall to provide continuous readout on recorders in the control room.

Steam generator feedwater is monitored at the main condensers. The condensate is analyzed for the major chemical constituent of river water (sodium) and is monitored for total dissolved solids.

10.2.7 Codes and Classifications

The pressure-retaining components or compartments of components comply, as a minimum, with the codes detailed in Table 10.2-1.

TABLE 10.2-1
Codes and Classifications

System pressure vessels and$_3$ pump casing	ASME Boiler and Pressure Vessel Code, Section VIII
Steam-generator vessel (shell side)	ASME Boiler and Pressure Vessel Code, Section III, Class C_1 (required)
System valves, fittings, and piping$_2$	USAS Section B31.1 Power Piping Code (1955) ASA, USAS, ANSI

Pressure Testing of Repairs and Modifications	USAS Section B31.1 Power Piping Code (1992)

Notes:
1. The shell side of the steam generator conforms to the requirements for Class A vessels (Actual) and is so stamped as permitted under the rules of Section III.
2. Except piping supplied by Westinghouse as part of the Turbine generator package, which was designed and fabricated to Westinghouse proprietary standards. This includes crossover, crossunder and lube oil piping.
3. Nos. 26A and 26B feedwater heater extraction steam inlet nozzles were modified in 1995 under the provisions of ASME Section VIII and were inspected and accepted under the provisions of the licensee's 10 CFR 50 Appendix B Quality Assurance Program.

10.2 FIGURES

10.3 SYSTEM EVALUATION

10.3.1 Safety Features

Trips, automatic control actions, and alarms will be initiated by deviations of system variables within the steam and power conversion system. Appropriate corrective action is taken as required to protect the reactor coolant system. The more significant malfunctions or faults that cause trips, automatic actions, or alarms in the steam and power conversion system are:

1. Turbine trip (see Section 10.2.3 for further discussion of trip actions):

a. Generator/electrical faults.
b. Low condenser vacuum.
c. Thrust bearing failure.
d. Low lubricating oil pressure.
e. Turbine overspeed.
f. Reactor trip.
g. Manual trip.
h. Main steam isolation valve closure.

2. Automatic control actions (see Chapter 7 for a further discussion of trip actions):
   a. High level in steam generator stops feedwater flow.
   b. Normal and low level in steam generator modifies feedwater flow by continuous proportional control.

3. Principal alarms:
   a. Low vacuum in condenser.
   b. Thrust bearing failure.
   c. Low lubricating oil pressure.
   d. Turbine overspeed.
   e. Low level in steam generator.
   f. High level in steam generator.
   g. Condenser hotwell high and low levels.

A reactor trip from power requires the removal of core decay heat. Immediate decay heat removal requirements are satisfied by the steam bypass to the condensers. Thereafter, core decay heat can be continuously dissipated by the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption.

Normally, the capability to return feedwater flow to the steam generators is provided by the operation of the turbine-cycle feedwater system. In the unlikely event of a complete loss of offsite electrical power to the station and concurrent reactor trip, decay heat removal would be ensured by the single turbine-driven and two motor-driven (by emergency diesel-generator power) auxiliary feedwater pumps, and steam dump to atmosphere by the main steam safety and/or power relief valves. Further details are given in Section 14.1.12. In this case, feedwater from the condensate storage tank is available by gravity feed to the auxiliary feedwater pumps. The minimum 360,000 gal of water in the condensate storage tank is adequate for decay heat removal at hot shutdown conditions for at least 24 hr. A backup source of feedwater is available from the city water storage tank.

The analysis of the effects of loss of full load on the reactor coolant system is discussed in Section 14.1.8.

10.3.2 Secondary-Primary Interactions

Following a turbine trip, the control system reduces reactor power output immediately by a reactor trip. Steam is bypassed to the condenser, and there is no lifting of the main safety valves. In the event of failure of a main feedwater pump, a motor-driven auxiliary feedwater pump is automatically started and the second main feedwater pump remaining in service will carry approximately 65-percent of full-load feedwater flow. If both main feedwater pumps fail, the reactor will be tripped as a result of steam-generator low-low level or steam-feedwater flow

mismatch and the auxiliary feedwater pumps will start. Not withstanding the anticipatory reactor trip on turbine trip, if reactor coolant system conditions reach trip limits, the reactor will trip.

Pressure relief is required at the main steam system design pressure of 1085 psig. The first safety valve is set to relieve at 1065 psig. Additional safety valves are set at pressures up to 1120 psig (see Section 10.2.1.2), as allowed by the ASME Code. The pressure relief capacity is greater than the steam generation rate at maximum calculated conditions.

The evaluation of the capability to isolate a steam generator to limit the release of radioactivity in the event of a steam-generator tube leak is presented in Section 14.2.4. The steam break accident analysis is presented in Section 14.2.5.

10.3.3 Single Failure Analysis

Table 10.3-1 presents the results of a single failure analysis of selected components in the system.

TABLE 10.3-1
Single-Failure Analysis

Component or System	Malfunction	Comments and Consequences
Auxiliary feedwater system	Auxiliary feedwater pump fails to start (following loss of main feedwater)	The auxiliary feedwater system comprises one turbine-driven and two motor-driven pumps. The turbine pump has twice the capacity of a motor-driven pump. A single motor-driven pump has sufficient capacity to allow time for an operator action to align the turbine-driven train and prevent relief of water through the primary side safety/relief valves. Thus adequate redundancy of auxiliary feedwater pumps is provided, as described in UFSAR 14.1.9.
Steam line isolation system	Failure of steam line isolation valve to close (following a main steam line rupture)	Each steam line contains an isolation valve and a non-return check valve in series. Hence, a failure of an isolation (or nonreturn) valve will not permit the blowdown of more than one steam generator irrespective of the steam-line rupture location, as described in UFSAR section 14.2.5.

Turbine bypass system	Bypass valve sticks open (following operation of the bypass system resulting from a turbine trip)	The turbine bypass system comprises 12 bypass valves, each with a steam flow capacity less than a steam generator/main steam safety valve. Thus, the uncontrolled steam flow from a stuck open bypass valve will not result in a plant cooldown in excess of the bounding steam line rupture / malfunction cases analyzed in UFSAR section 14.2.5.

10.4 TESTS AND INSPECTIONS

The main steam isolation valves are tested at least at refueling intervals and a maximum closure time of 5 sec is verified.

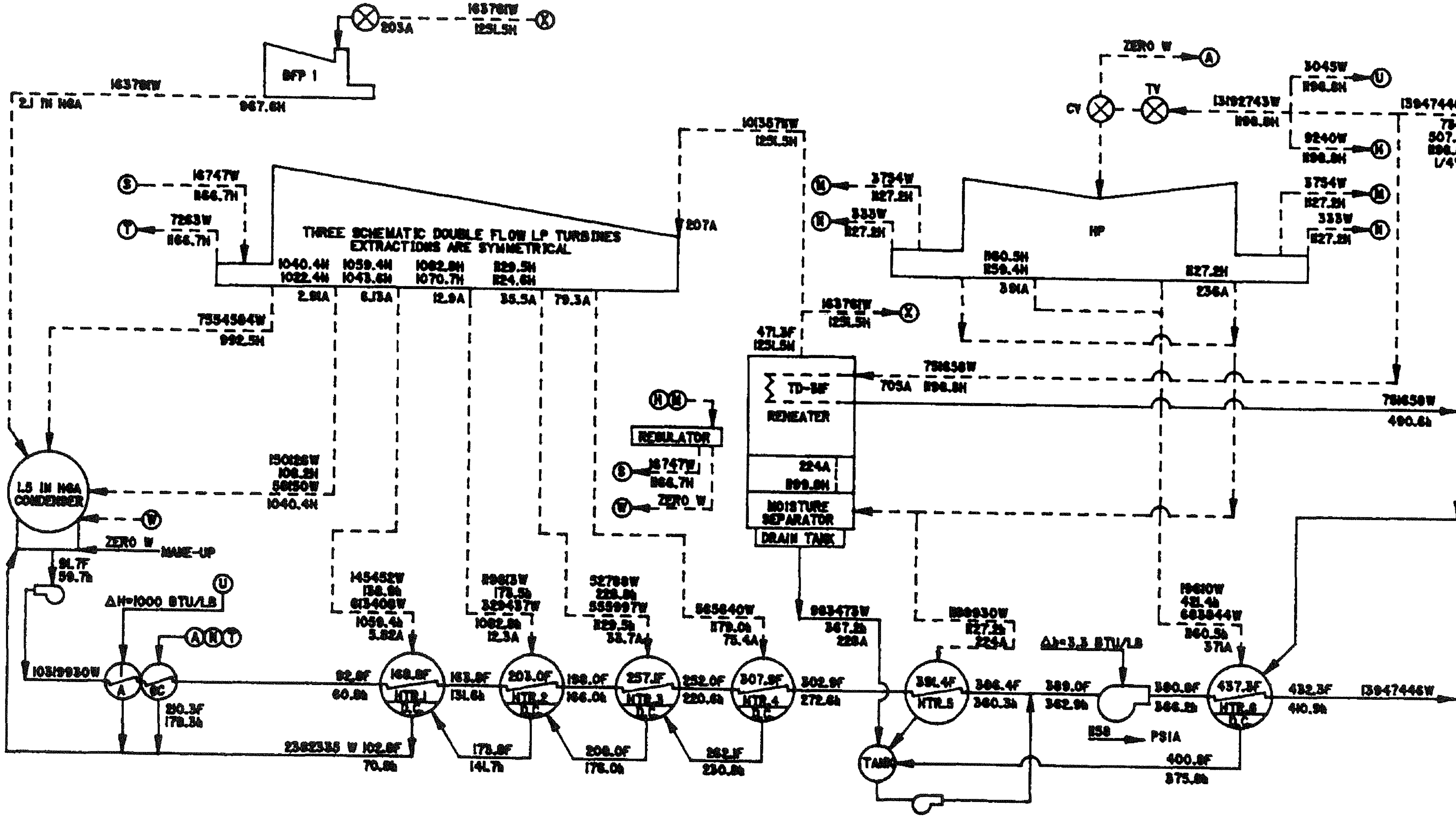
The main steam isolation valves serve to limit an excessive reactor coolant system cooldown rate and resultant reactivity insertion following a main steam break incident. Their ability to close upon signal is verified at periodic intervals. A closure time of 5 sec from receipt of closing signal was selected as being consistent with expected response time for instrumentation as detailed in the steam line break analysis. Further details are given in Section 14.2.5.

The auxiliary feedwater pumps are tested at regular intervals. Verification of correct operation is made both from instrumentation within the main control room and by direct visual observation of the pump. In addition, during reactor startup and shutdown, the auxiliary feedwater pumps (normally the motor-driven pumps) are used to deliver water from the condensate storage tank through its feedwater control valves to the feedwater line to the steam generators.

In response to NRC IE Bulletin 87-01, an inspection program has been established for piping and fittings in the extraction steam, turbine crossunder, heater drain pump discharge, condensate, feedwater and auxiliary feedwater systems. UT inspections are utilized to evaluate wall thickness at locations considered to be most susceptible to erosion/corrosion. Additional information is given in reference 1.

REFERENCES FOR SECTION 10.4

1. Letter from Murray Selman (Con Edison), to William Russell, NRC, dated 9/11/87.



NET HEAT RATE = $\frac{13947446\ (1198.8-410.9)}{1068701}$ = 10283 BTU/KW HR

(1)CALCULATIONS ARE BASED ON NO RADIATION LOSSES TO HEATERS OR EXTRACTION PIPING LOCATED IN THE CONDENSER NECK.
(2)PRIMARY VALVE AND ABOVE HEAT RATES ARE CALCULATED ON LOCUS OF VALVE POINTS.

STEAM GEN. FLOW AT MAX. CALC. IS NOT GUARANTEED.

MAX. GUAR. S.G. FLOW = 13283282 LB/HR. MAX. CALC. S.G. FLOW = 13947446 LB/HR.

TEP = 992.3 BTU/LB
ELEP = 973.4 BTU/LB
MECH LOSS = 3646 KW
ELECT LOSS = 11940 KW
0.90PF 75#H2
FWP POWER = 13624 KW
FWP EFF = 85%

1021793KW TURB-GEN.UNIT TC6F-44IN.
730PSIA- 507.8F 1.5IN.HGA
1125600 KVA 0.90PF 22000 VOLTS 75#H2

1068701 KW(e)NET LOAD HEAT BALANCE
MAXIMUM CALCULATED - NOT GUARANTEED

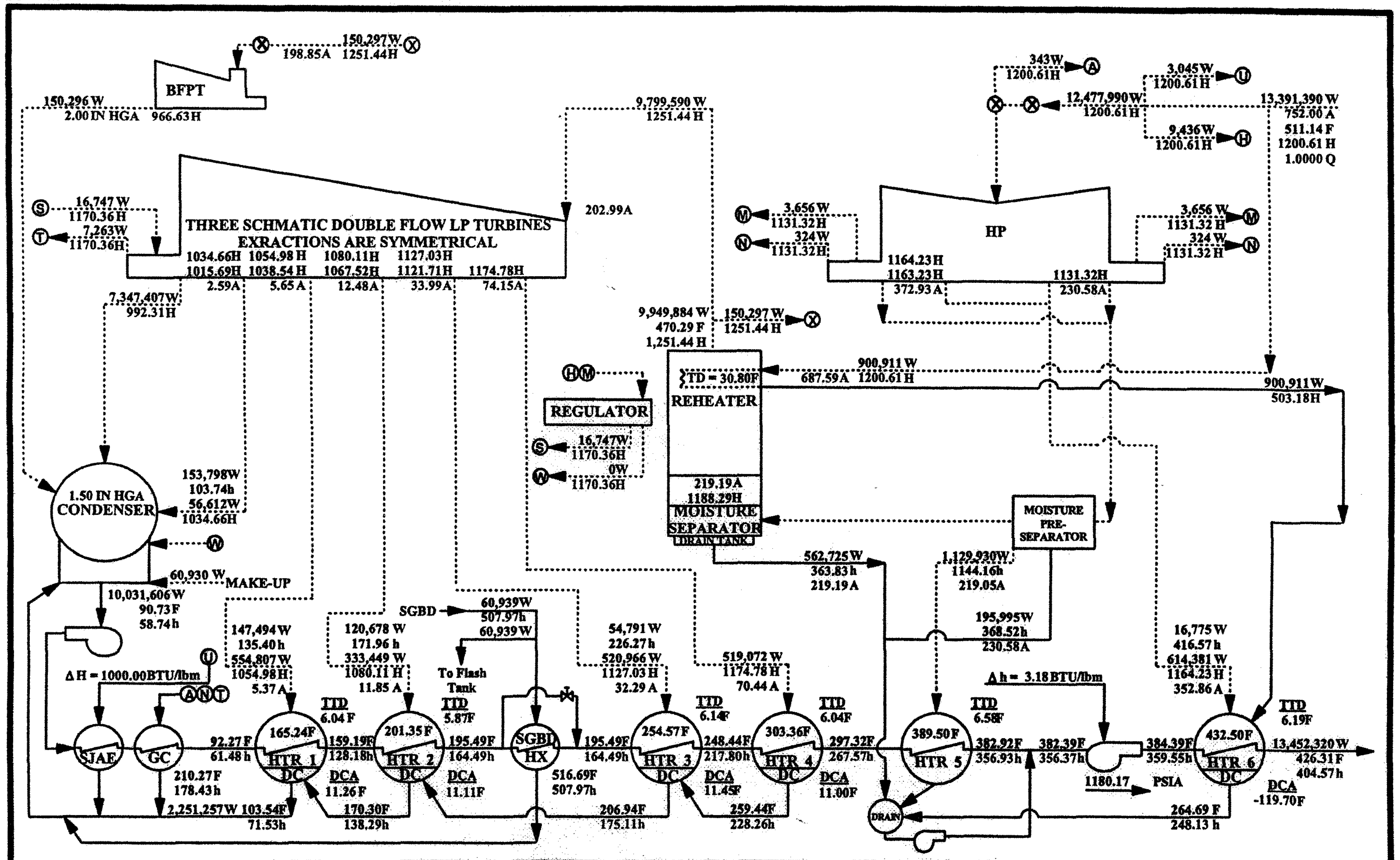
INDIAN POINT UNIT No. 2

UFSAR FIGURE 10.1-1

LOAD HEAT BALANCE DIAGRAM
AT 1,068,701 KWE

MIC. No. 1999MC3911 | REV. No. 17A

OAGI0000215_1236



NET HEAT RATE = $\frac{13,452,320(1200.61 - 404.57) - 60,939(1200.61 - 507.97)}{1,034,072}$ = 10,314.92BTU / KW HR	TEP = 992.08 BTU/lbm ELEP = 974.56 BTU/lbm MECH LOSS = 3.656 KW ELECT LOSS = 8.565 KW 0.91 PF 75# H2 FWP POWER = 12,545 KW FWP EFF = 85.00 %

1) Calculations are based on no radiation losses to heaters or extraction piping located in the condenser neck.
2) Primary valve and above heat rates are calculated on locus of valve points.

1028612 KW TURB - GEN UNIT TC6F - 45 IN.
668 PSIA - 497.9 F 1.5 IN.HGA
1439200 KVA 0.91 PF 22000 VOLTS 75# H2

1,034,072 KW NET LOAD HEAT BALANCE
101.40% Rx Power

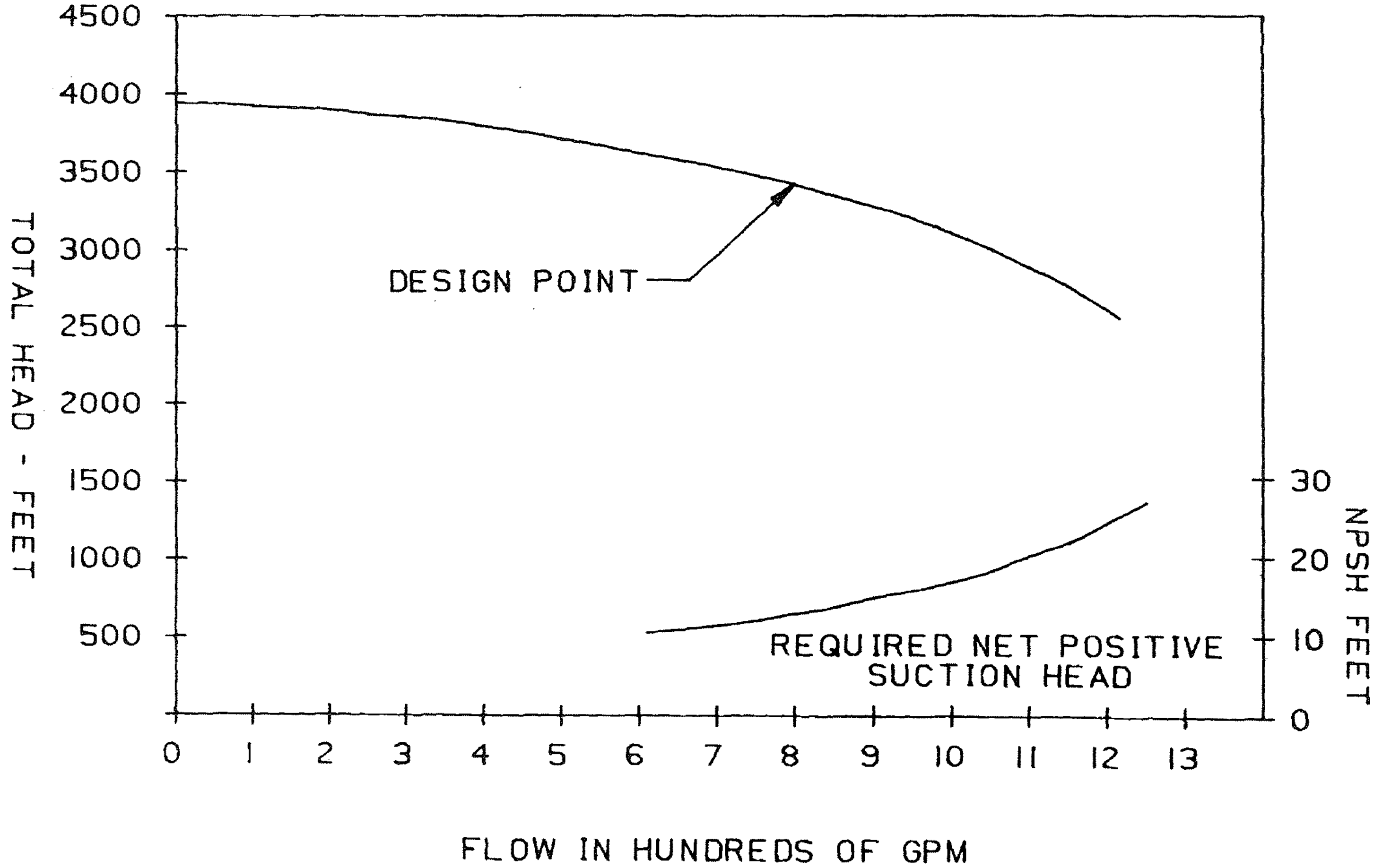
INDIAN POINT UNIT No. 2

UFSAR FIGURE 10.1-7

LOAD HEAT BALANCE DIAGRAM
AT 1,034,072 KWE

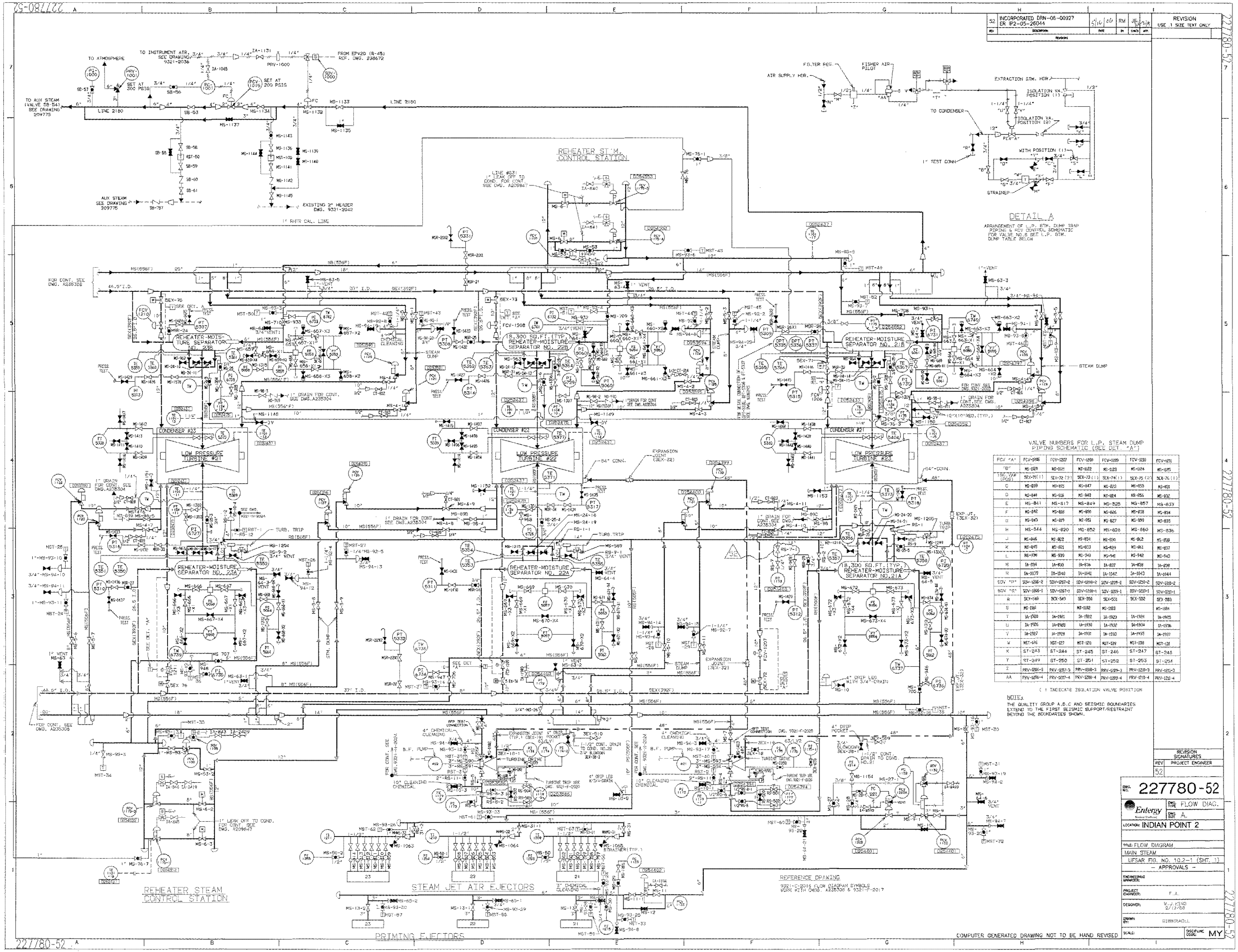
UFSAR FIGURE 10.1-7	REV. No. 19

OAGI0000215_1237



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 10.2-8 STEAM TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP ESTIMATED PERFORMANCE CHARACTERISTICS	
MIC. No. 1999MC3918	REV. No. 17A

OAGI0000215_1238



VALVE NUMBERS FOR L.P. STEAM DUMP PIPING SCHEMATIC (SEE DET. "A")

FCV "A"	FCV-1206	FCV-1207	FCV-1208	FCV-1209	FCV-1210	FCV-1211
"B"	MS-1120	MS-1121	MS-1122	MS-1123	MS-1124	MS-1125
ISO. VA# (POS)	SEX-71(1)	SEX-72(2)	SEX-73(1)	SEX-74(1)	SEX-75(2)	SEX-76(1)
C	MS-839	MS-815	MS-847	MS-823	MS-855	MS-831
D	MS-840	MS-816	MS-848	MS-824	MS-856	MS-832
E	MS-841	MS-817	MS-849	MS-825	MS-857	MS-833
F	MS-842	MS-818	MS-850	MS-826	MS-858	MS-834
G	MS-843	MS-819	MS-851	MS-827	MS-859	MS-835
H	MS-844	MS-820	MS-852	MS-828	MS-860	MS-836
J	MS-846	MS-822	MS-854	MS-830	MS-862	MS-838
K	MS-845	MS-821	MS-853	MS-829	MS-861	MS-837
L	MS-938	MS-939	MS-940	MS-941	MS-942	MS-943
M	IA-834	IA-835	IA-836	IA-837	IA-838	IA-839
N	IA-1039	IA-1040	IA-1041	IA-1042	IA-1043	IA-1044
SOV "P"	SOV-1206-2	SOV-1207-2	SOV-1208-2	SOV-1209-2	SOV-1210-2	SOV-1211-2
SOV "R"	SOV-1206-1	SOV-1207-1	SOV-1208-1	SOV-1209-1	SOV-1210-1	SOV-1211-1
R	SEX-548	SEX-549	SEX-550	SEX-551	SEX-552	SEX-553
S	MS-1181		MS-1182	MS-1183		MS-1184
T	IA-1920	IA-1921	IA-1922	IA-1923	IA-1924	IA-1925
U	IA-1926	IA-1928	IA-1930	IA-1932	IA-1934	IA-1936
V	IA-1927	IA-1929	IA-1931	IA-1933	IA-1935	IA-1937
W	MST-126	MST-127	MST-128	MST-129	MST-130	MST-131
X	ST-243	ST-244	ST-245	ST-246	ST-247	ST-248
Y	ST-249	ST-250	ST-251	ST-252	ST-253	ST-254
Z	PRV-1206-3	PRV-1207-3	PRV-1208-3	PRV-1209-3	PRV-1210-3	PRV-1211-3
AA	PRV-1206-4	PRV-1207-4	PRV-1208-4	PRV-1209-4	PRV-1210-4	PRV-1211-4

( ) INDICATE ISOLATION VALVE POSITION

NOTE:
THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
52	INCORPORATED DRN-06-00927 ER IP2-05-26044	5/16/06	RM	JB	MR

REVISION
USE .1 SIZE TEXT ONLY

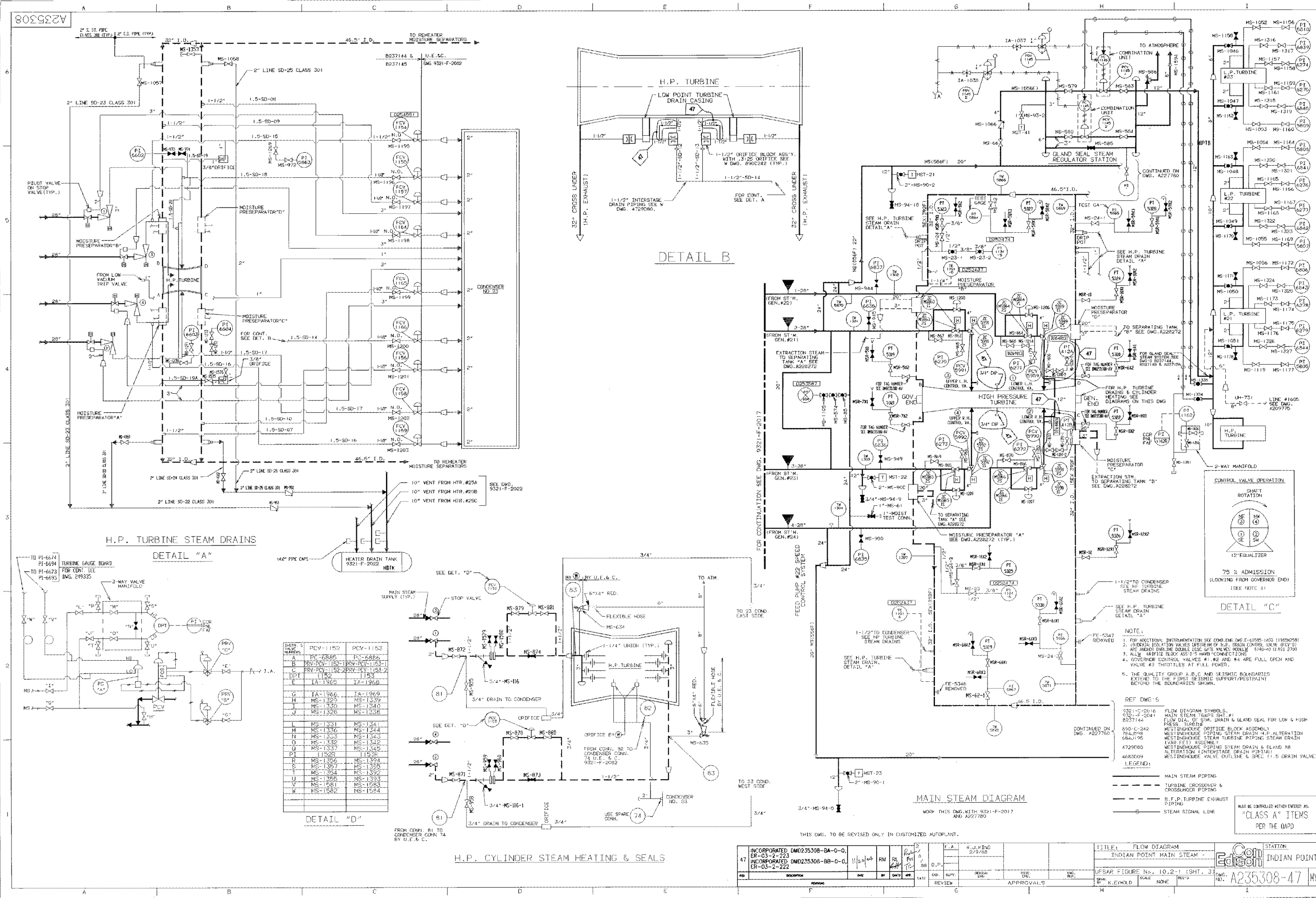
REVISION SIGNATURES

REV	PROJECT ENGINEER
52	

DWG. NO. 227780-52
Entergy
DWG. TYPE FLOW DIAG.
DWG. SIZE A
LOCATION: INDIAN POINT 2
TITLE: FLOW DIAGRAM
MAIN STEAM
UFSAR FIG. NO. 10.2-1 (SHT. 1)
– APPROVALS –
ENGINEERING MANAGER:
PROJECT ENGINEER: F.A.
DESIGNER: W.J.KING 2/12/88
DRAWN BY: SIBBSHILL
SCALE:
DISCIPLINE CODE: MY

227780-52

9321-F-2017
STEAM GENERATORS
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
FOR VALVES ARRANGEMENT SEE DWG 206108(TYP)
SILENCER #22
SILENCER #21
SILENCER #23
SILENCER #24
AIR ACCUMULATOR (ATTACHED TO VA. CYLINDER TYPICAL)
DRIP PAN ON 10" VENT PIPE (TYP.)
14" FOR 8" OUTLET OF SAFETY VALVES (TYP)
16" TYP. FOR 10" OUTLET OF SAFETY VALVES (TYP)
PUMP NO.22 (CLASS I EQPT.)
AUX.F.W. TURBINE DRIVE (SEE 9321-F-2019)
FOR CONTROL LINKAGE SEE B238709
FOR CONT. OF DRAINS & TRAP ARRG'T SEE DWG.9321-F-2041
FOR CONTINUATION SEE DWG# A235308
FLR. DRAIN
NOTE:
CLASS I FROM PUMP TO 28'HDR.
CLASS I PIPING
1. MAIN STEAM HEADERS FROM EACH STEAM GENERATOR THROUGH MAIN STEAM NON RETURN VALVES INCLUDING ALL CONNECTED VENTS DRAINS TRAP PIPING TO & INCLUDING TRAP & TRAP BY-PASS VALVES INSTRUMENTATION
2. STEAM GENERATOR SAFETY RELIEF VALVES MS-45 THROUGH MS-49 & ATMOSPHERIC DUMP VALVES PCV-1134 THROUGH PCV-1137 INCLUDING DISCHARGE PIPING THEREFROM
3. STEAM SUPPLY TO TURBINE DRIVEN AUXILIARY FEEDWATER PUMP INCLUDING ALL CONNECTED VENTS DRAINS TRAP PIPING TO & INCLUDING TRAP & TRAP BY-PASS VALVE, INSTRUMENTATION, RELIEF VALVE WITH DISCHARGE PIPING AND TURBINE EXHAUST PIPING
4. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.
REFERENCE DWGS
9321-C-2016 - FLOW DIAGRAM SYMBOLS
9321-F-2049 - YARD AREA WEST OF CONTM BLD'G MAIN STEAM PIPING SHT.#1
9321-F-2041 - FLOW DIAGRAM MAIN STEAM TRAPS,SHT.#1
B235623 - ATMOSPHERIC STEAM DUMP VALVE PANEL
NOTE A
THIS BOUNDS THE PORTION OF THE MAIN STEAM SYSTEM SHOWN ON THE DRAWING WHICH IS INCLUDED IN THE SECONDARY COOLANT SYSTEM DESCRIPTION IN EXHIBIT "A" CI-240-1.
NOTE:
FOR TRAP "T" DIAGRAM SEE DWG. 9321-F-2041
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS A" ITEMS
PER CI-240-1
THIS DWG. TO BE REVISED ONLY IN CUSTOMISED AUTOPLANT.
REVISION
D.A.
SGT
L.V.
SGR
THIS REVISION IS CLASS A PER CI-240-1.
UPDATED DWG. PER MOD.#FMX-00-52429-A
& PER DMD9321-2017-AP
RELEASED FOR RECORD.
P/N 52429-00
WM
BGA ENGINEERING
1/13/88
K.E.
DATE
CKR.
SUPV.
REVIEW
F.A.
W.J.KING/G.C. 1-13-88
DESIGN ENG
DISC. ENG.
ENG. MGR.
APPROVALS
TITLE: FLOW DIAGRAM
MAIN STEAM -
UFSAR FIGURE No. 10.2-1 (SHT. 2)
DRWN. BY GIBBS&HILL
SCALE
REC'D
Con Edison
STATION
INDIAN POINT
DWG. NO. 9321-F-2017-83
MY

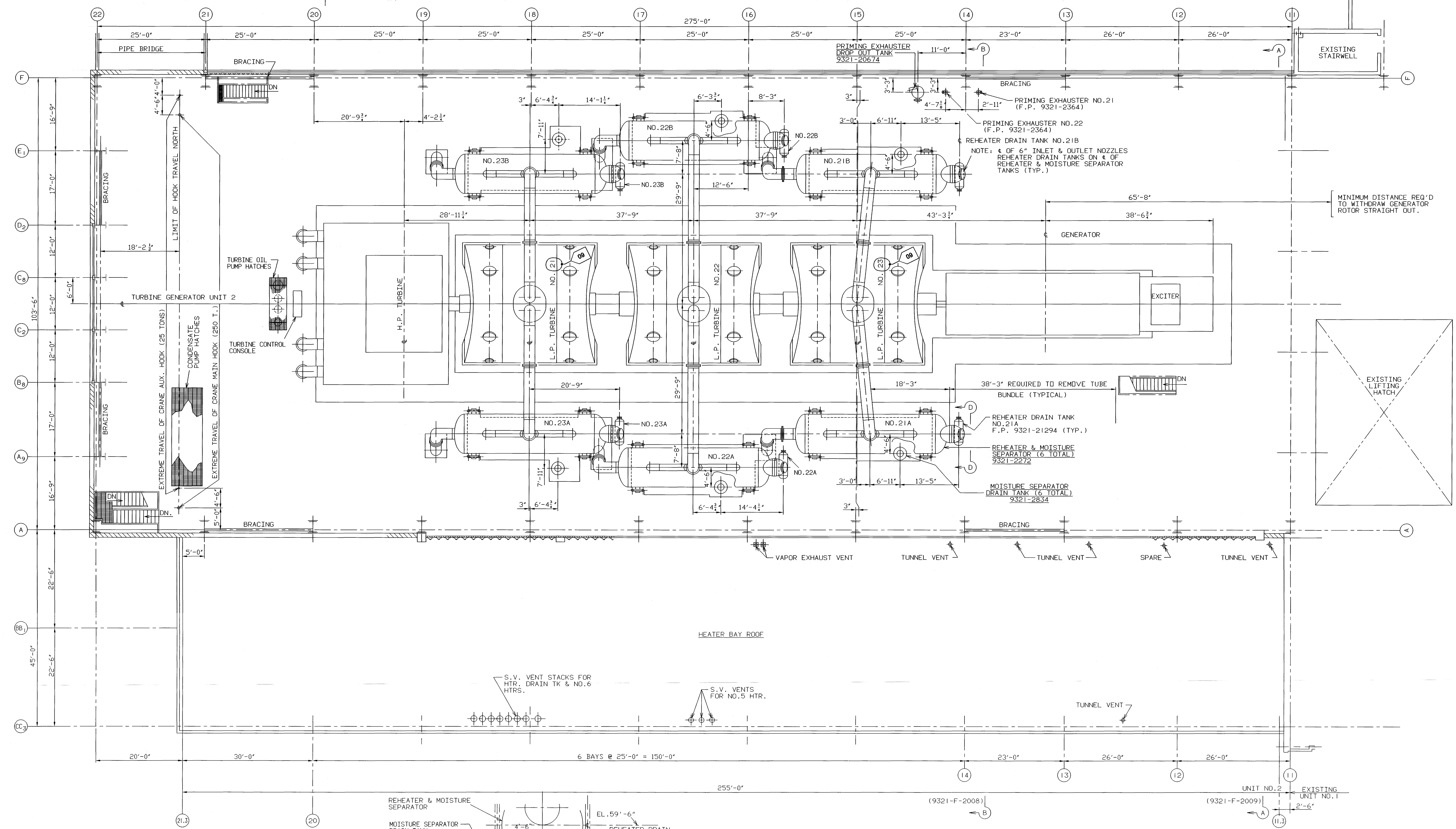


OAGI0000215_1241

9321-F-2004

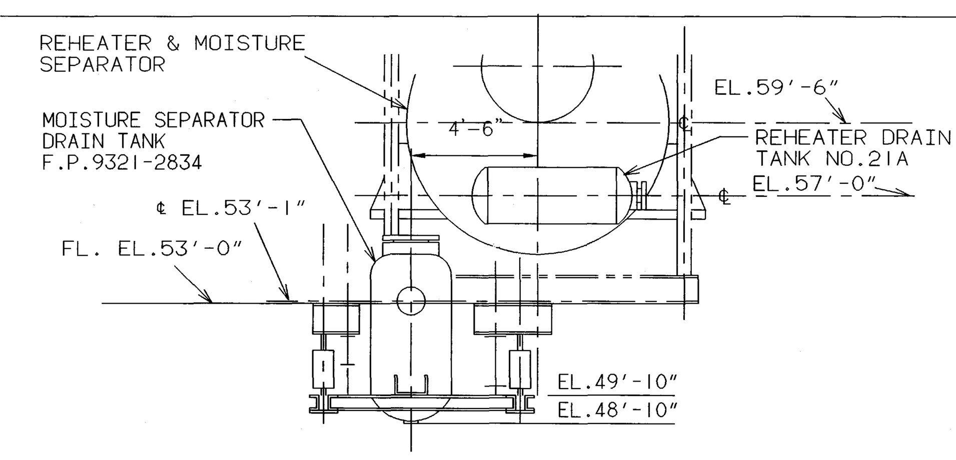
REV	DESCRIPTION	DATE	BY	CHK'D	APP.
9	INCORPORATED DRN-04-02056, IP2-ER-04-15884	8/18/04	RH		

REVISIONS



PLAN

SCALE: 1/16" = 1'-0"



SECTION "D-D"

SCALE: 1/4" = 1'-0"

FOR GENERAL NOTES & REFER. DWGS. SEE DWG. NO.9321-F-2006

(CON.ED.CO.DWG.NO. A200352)
COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

REVISION: THIS REVISION IS NON-CLASS PER CI-240-1. UPDATED DWG. TO INCLUDE UFSAR FIGURE NO. IN TITLE. RELEASED FOR RECORD P/N 69911-NP DLA/4/26/99

DES V.I. 04/26/99 — ENG STRATUS ENGINEERING

F.BERNARD SMITH 9-11-73

DATE | CKR. | SUPV. | REVIEW | DESIGN ENG | DISC. ENG. | ENG. MGR. | APPROVALS

TITLE: TURBINE BLDG. & HTR. BAY GEN. ARR'GMT. OPERATING FLR.-PLAN AT EL. 53'-0"
UNIT NO.2 – UFSAR FIGURE No. 10.2-2

STATION: INDIAN PT

Con Edison

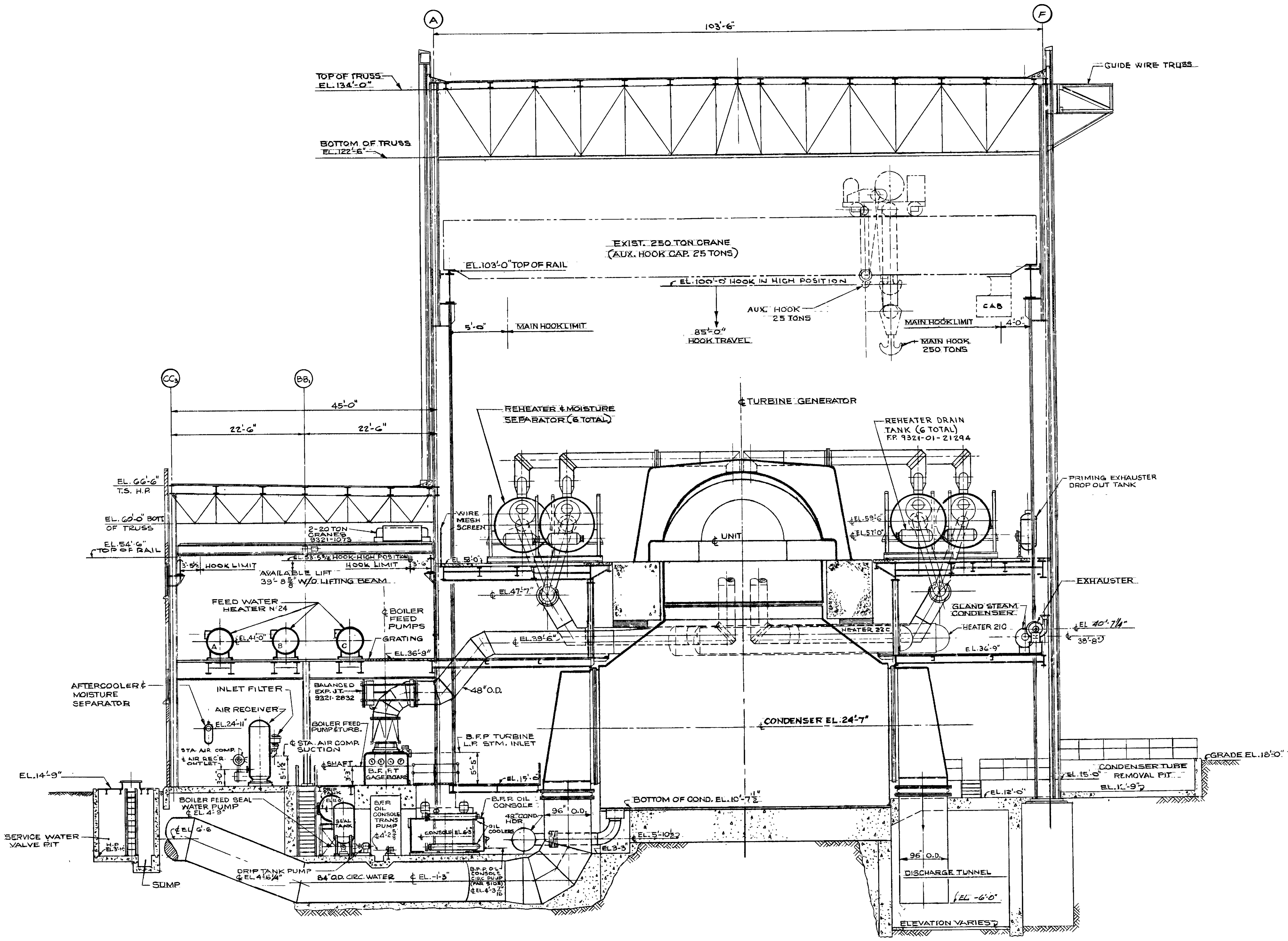
DRWN. BY R. HARRIS — SCALE 1/16" = 1'-0"

DWG. NO. 9321-F-2004-9 — MA

OAG10000215_1242

INDIAN POINT UNIT NO 2

208356



SECTION "B-B"
9321-F-2004
SCALE: 1/8"=1'-0"

07

V.I. 4/26/99	07

THIS REVISION IS NON-CLASS PER CI-240-1 UPDATED DRAWING TO INCLUDE UFSAR FIGURE No. IN TITLE P/N 69911-NP RELEASED FOR RECORD

WESTINGHOUSE ELECTRIC CORPORATION

TURBINE BUILDING & HEATER BAY GENERAL ARRANGEMENT CROSS SECTION

UFSAR FIGURE No. 10.2-3

FOR

CONSOLIDATED EDISON COMPANY

INDIAN POINT GENERATING STATION

UNIT NO. 2

MA

W.E.&C. DWG NO. | CON. ED. CO DWG NO.

9321-F-2025

REFERENCE DWG'S.

UE&C.DWG.NO.	DESCRIPTION
9321-C-2016	FLOW DIAGRAMS SYMBOLS
9321-F-2091	CONDENSER AIR REMOVAL PIPING PLAN-SHT.NO.1
9321-F-2092	CONDENSER AIR REMOVAL PIPING SECTIONS-SHT.NO.2
9321-F-2140	MISCELLANEOUS PIPING-SHT.NO.3
9321-F-2087	WATERBOX PRIMING-PIPING PLAN-SHT.NO.1
9321-F-2088	WATERBOX PRIMING-PIPING SECT.&ELEV.-SHT.NO.2

CLASS I PIPING

DOUBLE ISOLATION VALVES PCV-1229 & PCV-1230 THROUGH CONTAINMENT BUILDING INCLUDING PRESSURIZATION AIR SYSTEM

NOTE:

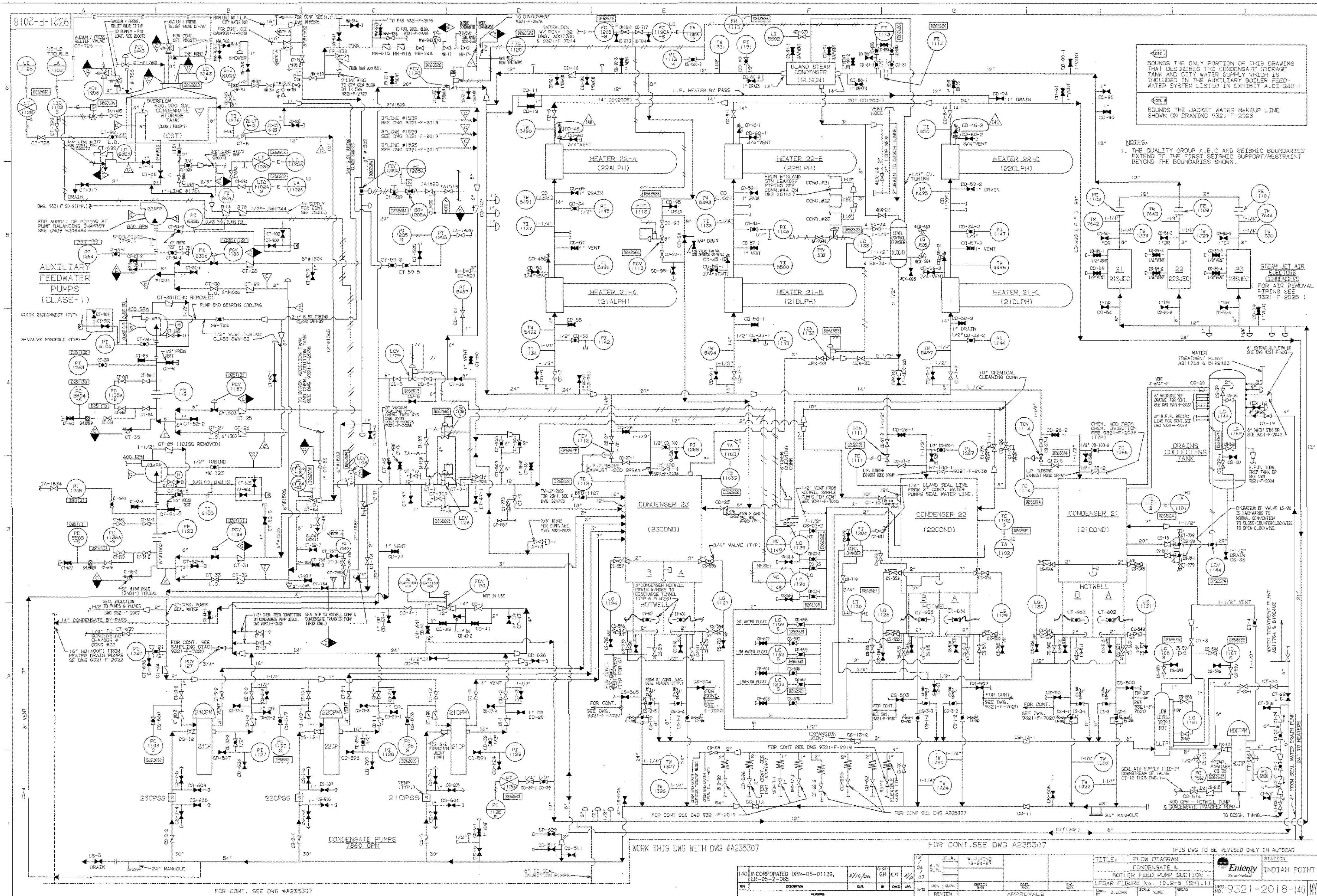
1). N.C. - NORMALLY CLOSED
2). CVP - CONDENSER VACUUM PUMP
3) THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

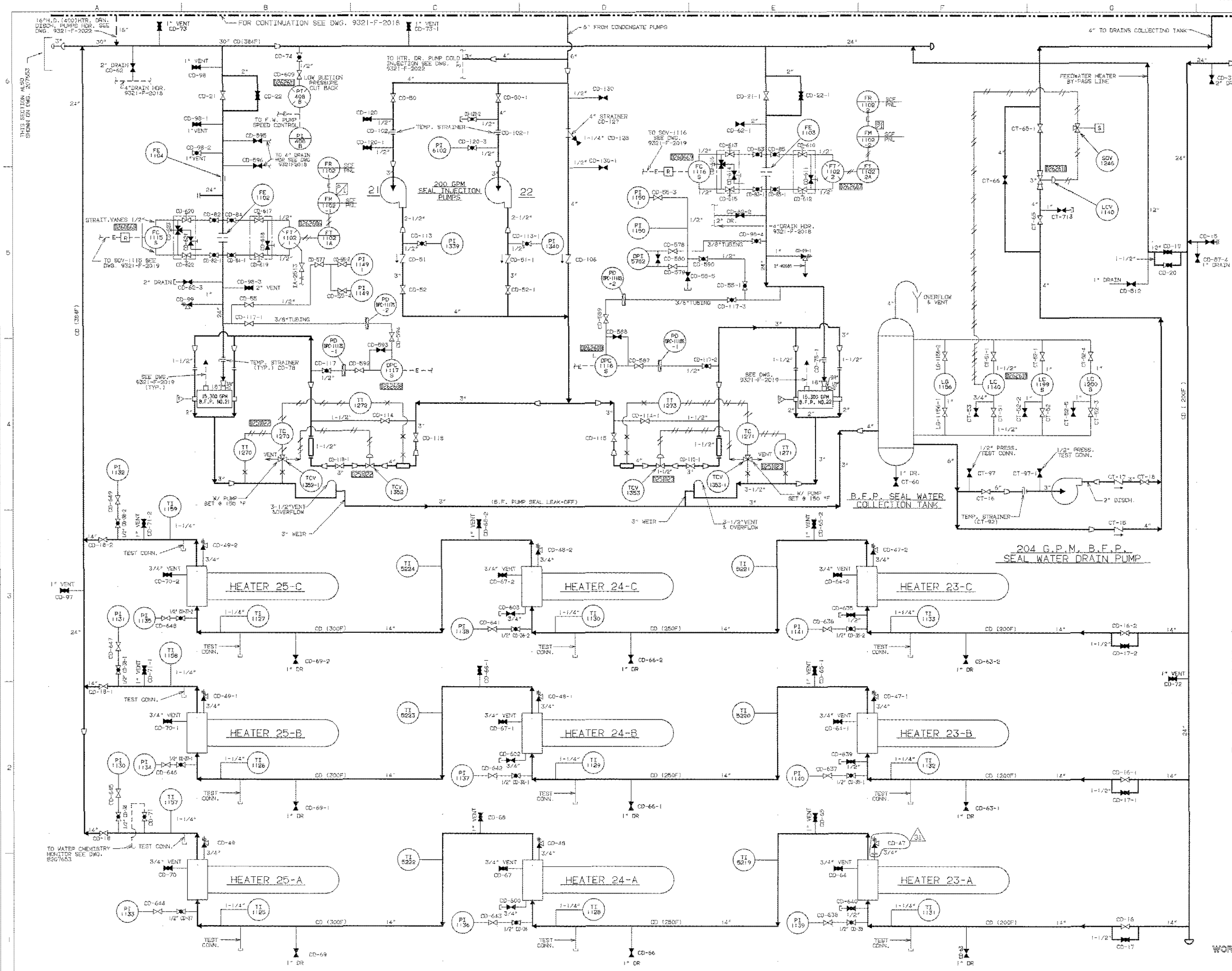
THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN ENTERGY AS:
"CLASS A " ITEMS
PER THE QAPD

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

REVISION							
M.RADVANSKY 04/08/03	THIS REV. IS NON-CLASS PER QAPD. UPDATED DRAWING TO SHOW THE WORK DONE ON MOD. PROC. # DCP-200110550-P, & DMD9321-2025-BA REV.00 RELEASE AS CONSTRUCTED P.N. 10550-01 GH/MR	6/3/87	W.S.	F.A.	W.J.KING 6/3/87		
ENG		DATE	CKR.	SUPV.	DESIGN ENG	DISC. ENG.	ENG. MGR.
			REVIEW		APPROVALS		

TITLE: FLOW DIAGRAM
CONDENSER AIR REMOVAL AND WATERBOX PRIMING (UFSAR FIG NO. 10.2-4)
MOD.PROC.OR SPEC.
DRWN. BY K. EYMOLD | SCALE NONE | REC'D
Con Edison
STATION INDIAN POINT
DWG. NO. 9321-F-2025-55 MY


9321-F-2018
NOTE A BOUNDS THE ONLY PORTION OF THIS DRAWING THAT DESCRIBES THE CONDENSATE STORAGE TANK AND CITY WATER SUPPLY WHICH IS INCLUDED IN THE AUXILIARY BOILER FEED-WATER SYSTEM LISTED IN EXHIBIT A.CI-240-1
NOTE B BOUNDS THE JACKET WATER MAKEUP LINE SHOWN ON DRAWING 9321-F-2028
NOTES:
1. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.
OVERFLOW 600,000 GAL CONDENSATE STORAGE TANK (CLASS I EXCEPT) (CST)
AUXILIARY FEEDWATER PUMPS (CLASS-1)
GLAND STEAM CONDENSER (GLSCN)
L.P. HEATER BY-PASS
HEATER 22-A (22ALPH)
HEATER 22-B (22BLPH)
HEATER 22-C (22CLPH)
HEATER 21-A (21ALPH)
HEATER 21-B (21BLPH)
HEATER 21-C (21CLPH)
STEAM JET AIR EJECTORS CONDENSERS FOR AIR REMOVAL PIPING SEE 9321-F-2025
21SJEC
22SJEC
23SJEC
DRAINS COLLECTING TANK
WATER TREATMENT PLANT A211784 & B192483
CONDENSER 23 (23COND)
CONDENSER 22 (22COND)
CONDENSER 21 (21COND)
HOTWELL
CONDENSATE PUMPS 7860 GPM
23CPSS
22CPSS
21CPSS
24" MANHOLE
FOR CONT. SEE DWG #A235307
WORK THIS DWG WITH DWG #A235307
FOR CONT. SEE DWG A235307
THIS DWG TO BE REVISED ONLY IN AUTOCAD
140 INCORPORATED DRN-06-01129, ER-05-2-065 5/16/06
TITLE: FLOW DIAGRAM CONDENSATE & BOILER FEED PUMP SUCTION - UFSAR FIGURE No. 10.2-5 (SH1.1)
Entergy
STATION INDIAN POINT
DWG NO. 9321-2018-140



CLASS I PIPING

1. CITY WATER FROM UNIT NO.1 TIE IN TO AUX. FEEDWATER PUMPS SUCTION INCLUDING ALL BRANCHES,VENT,DRAIN,INSTRUMENTATION.
2. CONDENSATE STORAGE TANK & ALL ATTACHMENTS.
3. CONDENSATE MAKE-UP LINE FROM THE TANK UP TO & INCLUDING VALVE LCV-1158
4. CONDENSATE FROM MAKE-UP LINE TO EACH AUX. FEEDWATER PUMPS INCLUDING ALL CONNECTED VENTS,DRAINS & INSTRUMENTS
5. FILL LINE FROM INLET TO CHECK VALVE CD-109 TO CONDENSATE STORAGE TANK
6. AUX. FEED PUMP RECIRCULATION SYSTEM COMPLETE.
7. THE QUALITY GROUP A,B,C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

REFERENCE DWGS.

9321-C-2016 - FLOW DIAGRAM SYMBOLS
9321-F-2064 - TURBINE BLDG. & HEATER BAY-COND. & B.F.P. SUCTION PIPING SHT. NO.1
9321-F-2036 - FLOW DIAG. INSTRUMENT AIR CONTROL BLDG., CONTENTIONAL PLANT, ETC.

WORK THIS DWG. WITH DWG. 9321-F-2018

THIS DWG. TO BE REVISED ONLY IN AUTOCAD

REV	DESCRIPTION	DATE	BY	CHK'D	APP.
31	INCORPORATED DRN-05-01130, ER-05-2-065	5/16/06	GH		

REVISIONS

W.J.KING 9/28/67

DESIGN ENGR. | SUPERV. ENG. | ENG. MGR.

APPROVALS

TITLE: FLOW DIAGRAM CONDENSATE & BOILER FEED PUMP SUCTION - UFSAR FIGURE No. 10.2-5 (SHT. 2)

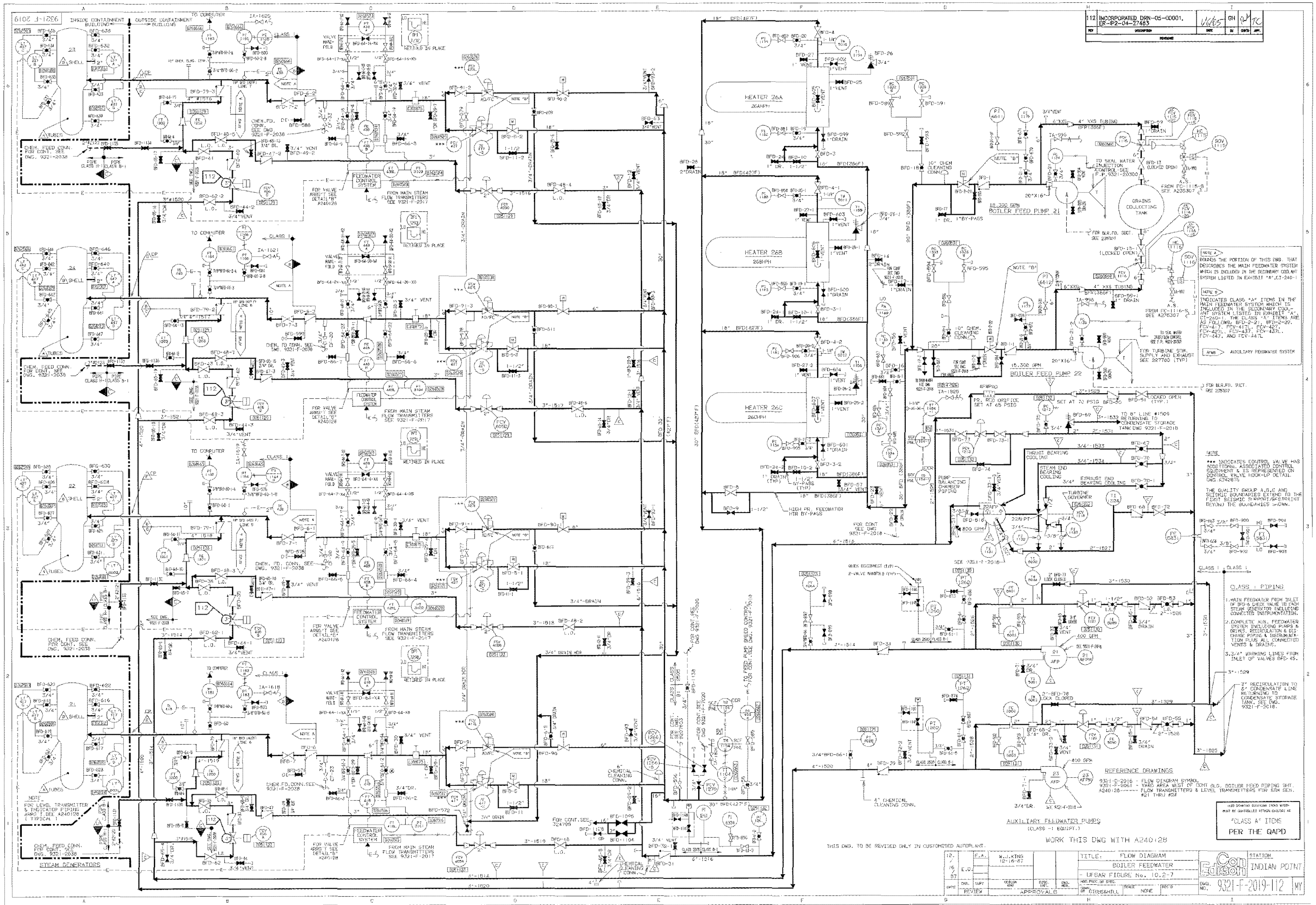
DRWN BY B. JOHN | SCALE NONE | REC'D

Entergy

STATION: INDIAN POINT

DWG. NO. A235307-31 MY

OAGI0000215_1246



OAGI0000215_1247

CHAPTER 11
WASTE DISPOSAL AND RADIATION PROTECTION SYSTEM

11.1 WASTE DISPOSAL SYSTEM

11.1.1 Design Bases

Control of Releases of Radioactivity to the Environment

Criterion: The facility design shall include those means necessary to maintain control over the plant radioactive effluents whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control must be justified (a) on the basis of 10 CFR 20 requirements, for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dose level guidelines for potential reactor accidents of exceedingly low probability of occurrence (GDC 70).

Liquid, gaseous, and solid waste processing and handling facilities are designed so that the discharge of effluents and offsite disposal shipments are in accordance with applicable government regulations.

Radioactive fluids entering the waste disposal system are collected in sumps and tanks until determination of subsequent treatment can be made. They are sampled and analyzed to determine the concentration of radioactivity, with an isotopic breakdown if necessary. Before any attempt is made to discharge radioactive waste, it is processed as required. The processed water from waste disposal, from which most of the radioactive material has been removed, is discharged through a monitored line into the circulating water discharge. The system design and operation are characteristically directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude releases in excess of the limits of 10 CFR 20.

Radioactive gases are pumped by compressors through a manifold to one of the gas decay tanks where they are held a suitable period of time for decay. Cover gases in the nitrogen blanketing system are reused to minimize gaseous wastes. During normal operation, gases are discharged intermittently at a controlled rate from these tanks through the monitored plant vent. The system is provided with discharge controls so that the release of radioactive effluents to the atmosphere is controlled within the limits set in the Technical Specifications.

The spent resins from the demineralizers, the filter cartridges, and the concentrates from the evaporators are packaged and stored onsite until shipment offsite for disposal. Suitable containers are used to package these solids at the highest practical concentrations to minimize the number of containers shipped for burial.

All solid waste is placed in suitable containers and stored onsite until shipped offsite for disposal.

The application of the NUREG-1465 alternative source term methodology for Indian Point Unit 2 includes verification that the dose limits specified in 10 CFR 50.67 are met for low probability accidents.

11.1.2 System Design and Operation

The waste disposal system process flow diagrams are shown in Figure 11.1-1, Sheets 1 and 2, and performance data are given in the Annual Effluent and Waste Disposal Report.

The waste disposal system collects and processes all potentially radioactive primary plant wastes for removal from the plant site within limitations established by applicable government regulations. Fluid wastes are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary, before any attempt is made to discharge them. They are then released under controlled conditions. A radiation monitor is provided to maintain surveillance over the release operation, but the permanent record of activity release is provided by radiochemical analysis of known quantities of waste. The original system design was based on processing all wastes generated during continuous operation of the primary system assuming that fission products, corresponding to defects in 1-percent of the fuel cladding, escape into the reactor coolant.

As secondary functions, system components supply hydrogen and nitrogen to primary system components as required during normal operation, and provide facilities to transfer fluids from inside the containment to other systems outside the containment.

The Offsite Dose Calculation Manual (ODCM) provides the methodology to calculate radiation does rates and dose to individual persons in unrestricted areas in the vicinity of Indian Point due to the routine release of liquid effluents to the discharge canal. The ODCM also provides setpoint methodology that is applied to effluent monitors and optionally to other process monitors.

Activity release due to tritium is given in the Annual Effluent and Waste Disposal Report.

11.1.2.1 System Description

11.1.2.1.1 Liquid Processing

During normal plant operation the waste disposal system processes liquids from the following sources:

1. Equipment drains and leaks.
2. Chemical laboratory drains.
3. Decontamination drains.
4. Demineralizer regeneration.
5. Floor drains.
6. Steam generator blowdown.

The reactor coolant drain tank collects and transfers liquid drained from the following sources:

1. Reactor coolant loops.
2. Pressurizer relief tank.
3. Reactor coolant pump secondary seals.

4. Excess letdown during startup.
5. Accumulator drains
6. Valve and reactor vessel flange leakoffs.
7. Refueling Canal Drain
8. Containment Spray Header Recirculation Lines

The valve and reactor flange leakoff liquids flow to the reactor coolant drain tank and are discharged directly to the chemical and volume control system holdup tanks by the reactor coolant drain pumps, which are designed to operate automatically by a level controller in the tank.

Since the fluid pumped by the reactor coolant drain pumps is of high quality and can be reused, the discharge of these pumps will normally be routed to the holdup tanks of the chemical and volume control system. If the fluid is considered unsuitable for reuse, it will be sent to the waste holdup tank. The discharge of the reactor coolant drain pumps can also be routed to the refueling water storage tank. This path will be used when pumping down the containment refueling canal during return from refueling operations. In the event the reactor coolant drain pumps are unavailable, the contents of the reactor coolant drain tank or the pressurizer relief tank can be dumped to the containment sump.

The waste holdup tank serves as the collection point for liquid wastes. It collects fluid directly from the following sources:

1. Reactor coolant drain tank pumps
2. Containment sump pumps.
3. Holdup tank pit sump pump.
4. Sump tank pump (from primary auxiliary building).
5. Spent regenerant chemicals from demineralizers.
6. Equipment drains.
7. Chemical drain tank pump.
8. Relief valve discharge from the component cooling surge tank and the chemical and volume control system holdup tanks.
9. Waste condensate pumps.
10. Maintenance and Operation Building floor drains.
11. Primary Auxiliary Building sump pumps.

Where plant layout permits, waste liquids drain to the waste holdup tank by gravity flow. Other waste liquids, including floor drains, drain to the sump tank or to the primary auxiliary building sump. The liquid wastes are pumped to the waste holdup tank. The liquid waste holdup tank is processed by sending its contents to the Unit 1 waste collection system.

Capability exists to transfer the waste holdup tank contents to the waste condensate tank. If used, sampling indicates that the liquid is suitable for discharge and the waste liquid can be pumped from the waste holdup tank to the waste condensate tanks. There it’s activity can be determined for recording by isolation sampling and analyzing before it would be discharged through the radiation monitor to the condenser circulating water.

The Indian Point Unit 1 waste collection system has four tanks with a capacity of 75,000 gal each. From there the liquid can also be processed by use of sluiceable demineralizer vessels.

A portable demineralization system is being used in the Unit 1 Chemical System Building. The system employs a number of in-line ion exchanger resin beds and filters to remove radionuclides and chemicals as required from the waste stream. The demineralization/filtration system processes liquid waste from the unit 1 waste collection tanks and discharges the clean water to the distillate storage tanks.

Spent resins from the portable system are sluiced from the vessels into a high integrity container, which is dewatered and then transported to the burial site without solidification. Spent filters can also be placed in the high integrity container.

The distillate produced by the demineralizer water processing is collected in two distillate storage tanks. Each storage tank is vented to the unit 1 ventilation system. Normally one tank is filling while the other is sampled and discharged. When a distillate storage tank is ready for discharge, it is isolated and sampled to determine the allowable release rate. If the contents of the tank are not suitable for release, they are returned to waste collection tanks for reprocessing. If analysis confirms that the activity level is suitable for release, the distillate is discharged to the river. A radiation detector and high radiation trip valve are provided in the release line to prevent an inadvertent release of activity at concentrations in excess of the setpoint derived from the technical specifications. In the event of primary-to-secondary coolant leakage, the affected steam generator blowdown can be manually diverted to the support facilities secondary boiler blowdown purification system flash tank. This system cools the blowdown and either stores it in the support facilities waste collection tanks or purifies it. The purification process consists of filtering and demineralizing the blowdown. The filters will remove undissolved material of 25 microns or greater. Mixed-bed demineralizers, which utilize cation and anion resin, remove isotopic cations and anions, as well as nonradioactive chemical species. The effluents of the demineralizers are monitored and the specific activity is recorded. Section 10.2.1 provides further discussion of the steam generator blowdown.

Also, in the event of primary-to-secondary leakage, potentially contaminated water that collects in secondary-side drains may be collected and routed to a collection point in the auxiliary boiler feedwater building for eventual processing. The path is an alternative to the normally used path to the drains collection tank.

11.1.2.1.2 Gas Processing

During plant operations, gaseous waste will originate from:

1. Degassing the reactor coolant and purging the volume control tank.
2. Displacement of cover gases as liquid accumulates in various tanks.
3. Equipment purging.
4. Sampling operations and automatic gas analysis for hydrogen and oxygen in cover gases.

During normal operation, the waste disposal system supplies nitrogen and hydrogen to primary plant components. Two headers are provided, one for operation and one for backup. The pressure regulator in the operating header is set for 110 psig discharge and that in the backup header for 90 psig. When the operating header is exhausted, its discharge pressure will fall below 100 psig and an alarm will alert the operator. The second tank will come into service automatically at 90 psig to ensure a continuous supply of gas. After the exhausted header has been replaced, the operator manually sets the operating pressure back to 110 psig and the

backup pressure at 90 psig This operation is identical for both the nitrogen supply and the hydrogen supply.

Most of the gas received by the waste disposal system during normal operation is cover gas displaced from the chemical and volume control system holdup tanks as they fill with liquid. Since this gas must be replaced when the tanks are emptied during processing, facilities are provided to return gas from the decay tanks to the holdup tanks. A backup supply from the nitrogen header is provided for makeup if return flow from the gas decay tanks is not available. Since the hydrogen concentration may exceed the combustible limit during this type of operation, components discharging to the vent header system are restricted to those containing no air or aerated liquids and the vent header itself is designed to operate at a slight positive pressure (O.5 psig minimum to 2.0 psig maximum) to prevent inleakage. On the other hand, outleakage from the system is minimized by using Saunders patent diaphragm valves, bellows seals, self-contained pressure regulators, and soft-seated packless valves throughout the radioactive portions of the system.

Gases vented to the vent header flow to the waste gas compressor suction header. One of the two compressors is in continuous operation with the second unit instrumented to act as backup for peak load conditions. From the compressors, gas flows to one of the four large gas decay tanks. The control arrangement on the gas decay tank inlet header allows the operator to place one large tank in service and to select a second large tank for backup. When the tank in service becomes pressurized to a predetermined pressure, a pressure transmitter automatically opens the inlet valve to the backup tank, closes the inlet valve to the filled tank, and sounds an alarm to alert the operator of this event so that he may select a new backup tank. Pressure indicators are supplied to aid the operator in selecting the backup tank. Gas held in the decay tanks can either be returned to the chemical and volume control system holdup tanks, or discharged to the atmosphere if the activity concentration is suitable for release. Generally, the last tank to receive gas will be the first tank emptied back to the holdup tanks in order to permit the maximum decay time for the other tanks before releasing gas to the environment. However, the header arrangement at the tank inlet gives the operator freedom to fill, reuse, or discharge gas to the environment simultaneously without restriction by operation of the other tanks.

Six additional small gas decay tanks are supplied for use during degassing of the reactor coolant prior to a cold shutdown. The reactor coolant fission gas activity inventory is distributed equally among the six tanks through a common inlet header.

A radiation monitor in the sample line to the gas analyzer checks the gas decay tank activity inventory each time a sample is taken for hydrogen-oxygen analysis. An alarm warns the operator when the inventory limit is approached so that another tank may be placed in service.

Before a tank can be emptied to the environment, its contents must be sampled and analyzed to verify sufficient decay and to provide a record of the activity to be released, and only then discharged to the plant vent at a controlled rate through a radiation monitor in the vent. Samples are taken manually by opening the isolation valve to the gas analyzer sample line and permitting gas to flow to the gas analyzer where it can be collected in one of the sampling system gas sample vessels. After sampling, the isolation valve is closed. During release, a trip valve in the discharge line is closed automatically by a high activity level indication in the plant vent.

During operation, gas samples are drawn periodically from tanks discharging to the waste gas vent header as well as from the particular large gas decay tank being filled at the time, and

automatically analyzed to determine their hydrogen and oxygen content. The hydrogen analysis is for surveillance since the concentration range will vary considerably from tank to tank. There should be no significant oxygen content in any of the tanks, and an alarm will warn the operator if any sample shows 2-percent by volume of oxygen. This allows time to isolate the tank before the combustible limit is reached. Another tank is placed in service while the operator locates and eliminates the source of oxygen. Discharged gases are released from the plant vent and diluted in the atmosphere due to the turbulence in the wake of the containment building in addition to the effects of normal dispersion.

The maximum expected annual gaseous release by isotope is given in the Annual Effluent Release and Waste Disposal Report.

11.1.2.1.3 Solids Processing

Solid waste processing is controlled by the Process Control Program in the ODCM.

Resin is normally stored in the spent resin storage tank for decay; this tank is described in section 11.1.2.2.6. Resin is removed from the storage tank to a high integrity container, which is dewatered and prepared for transportation in accordance with the Process Control Program. Spent filters can be placed in the high integrity containers.

Miscellaneous solid wastes such as paper, rags and glassware, are processed in accordance with the Process Control Program. When possible, solid waste is sent to a licensed incineration and volume reduction center, or to a material recovery center. This process is controlled by the Process Control Program.

The unit 1 containment has been modified for use as an interim onsite storage facility for dry active waste.

The Original Steam Generators (OSGs) are stored in the Original Steam Generator Storage Facility (OSGSF). Storage in this building is limited to the OSGs. The OSGSF is a reinforced concrete structure measuring approximately 150 feet by 54 feet (not including the labyrinth entryways). The building is located on the eastern side of the plant, between Electrical Tower 3 and the Buchanan Service Center access road. This location is within the Owner Controlled Area outside the Protected Area. The structure is constructed of cast-in-place concrete. Except for the South wall, which consists of pre-cast stackable concrete blocks. Use of pre-cast blocks provide access to install the OSGs and for removal of the OSGs at a later date. The roof is covered with a single-ply membrane roofing system.

The walls of the OSGSF are 3’-0" thick and the roof is tapered from 2’-6" in the center of the building to 2’-0" at the east and west walls. The slab is 3’-0" thick with a thickened perimeter that is 5’-0" thick. Personnel access doors with labyrinth entryways are provided at each end of the building to prevent radiation streaming through the door. The walls of the labyrinth entryway are 3’-0" thick with the roof over the labyrinth entryway tapered from 1’-2" to 1’-0". Two locked steel doors in each entryway will provide access to the building after the pre-cast concrete blocks are put in place, one in the exterior wall opening and one in the labyrinth wall.

The OSGSF is designed to contain contaminated materials and facilitate decontamination should such an action become necessary. Waterstops are used at all construction joints to prevent both the intrusion of water into the facility and the escape of contaminated water from the facility. The floor of the facility is sloped to provide adequate drainage to a sump. Protective

coatings are applied to the floor slab and lower portion of the walls to ease decontamination, if required. A passive HEPA filter system is provided to allow venting of the OSGSF while containing any airborne contamination.

An electrical system provides interior and exterior lighting, 110-volt AC outlets, and a remote alarm system on each entryway. Two locked steel doors secure the building and a security fence is installed around the perimeter of the building.

11.1.2.2 Components

Codes applying to components of the waste disposal system are shown in Table 11.1-6. Component summary data is shown in Table 11.1-7. Waste disposal system components are located in the auxiliary building except for the reactor coolant drain tank, which is in the containment and the waste holdup tank, which is in the liquid holdup tank vault.

11.1.2.2.1 Deleted

11.1.2.2.2 Chemical Drain Tank

The chemical drain tank is a vertical cylinder of austenitic stainless steel and collects drainage from the chemistry sampling station. The tank contents are pumped to the waste holdup tanks.

11.1.2.2.3 Reactor Coolant Drain Tank

The reactor coolant drain tank is a horizontal cylinder with spherically dished heads. The tank is all welded austenitic stainless steel. This tank serves as a drain surge tank for the reactor coolant system and other equipment located inside the reactor containment. The water collected in this tank is transferred to the chemical and volume control system holdup tanks, the refueling water storage tank, or the waste holdup tank.

11.1.2.2.4 Waste Holdup Tank

The waste holdup tank is the central collection point for radioactive liquid waste. The tank is stainless steel of welded construction.

11.1.2.2.5 Sump Tank and Sump Tank Pumps

The sump tank serves as a collecting point for waste discharged to the basement level drain header. It is located at the lowest point in the auxiliary building. Floor drains enter this tank through a loop seal to prevent back flow of gas from the tank. Two horizontal centrifugal pumps transfer liquid waste to the waste holdup tank. All wetted parts of the pumps are stainless steel. The tank is all-welded austenitic stainless steel.

11.1.2.2.6 Spent Resin Storage Tank

The spent resin storage tank retains resin discharged from the primary plant demineralizers. Normally, resins are stored in the tank for decay of short-lived isotopes. However, the contents can be removed at any time, if sufficient shielding is provided for the spent resin shipping vessel. A layer of water is maintained over the resin surface as a precaution against resin degradation due to heat generation by radioactive decay. Resin is removed from the tank by first sparging with nitrogen to loosen the resin and then pressurizing the tank with nitrogen to

approximately 60 psig to force the resin slurry out of the tank. If desired, the primary water supply can be used instead of nitrogen for agitating the resin before discharging it from the tank. The tank is all-welded austenitic stainless steel.

11.1.2.2.7 Gas Decay Tanks

Four large (525-ft^3) welded, vertical, carbon steel tanks are provided to hold radioactive waste gases for decay. This arrangement is adequate for operation with 1-percent fuel defects (as discussed in Section 14.2.3). Four tanks are provided so that during normal operation, sufficient time is available for decay but release is allowed at any time providing the activity is within limits. Normally one of the large gas decay tanks will be in service receiving waste gas while a second tank will be selected to provide backup. When the pressure in the tank receiving gas reaches a predetermined pressure, the fill valve on the tank in service will close and the fill valve on the standby tank will open. A connection is provided on the bottom the tank to allow any water collected in the tank to be removed to the drain header. A nitrogen supply is available for purging the tank.

The large gas decay tanks are sampled periodically by the gas analyzer. Only the tank in the process of being filled will be sampled; the other tanks will be bypassed. A radiation monitor in the gas analyzer line will indicate its reading in the Central Control Room. An alarm is provided so the operator can stop the filling operation before the 6000 Ci limit on the tank is reached. The Offsite Dose Calculation Manual provides the methodologies used to determine the alarm setpoint of the radiation monitor. An administrative maximum of 6000 Ci of equivalent Xe-133 is allowed in any one tank to minimize impacts of accidental release from equipment or tank failure and is well below the ODCM limit.

Gas held in the decay tanks can either be returned to the chemical and volume control system holdup tanks, or discharged to the atmosphere if the activity concentration is suitable for release. The header arrangement at the tank inlet gives the operator freedom to fill, reuse, or discharge gas to the environment simultaneously without restriction by operation of the other tanks.

Six small (40-ft^3), welded carbon steel, vertical tanks are provided to hold waste gases released during degassing of the reactor coolant prior to a cold shutdown.

A connection is provided on the bottom of the tank to allow any water collected in the tank to be removed to the drain header. A nitrogen supply is available for purging the tank.

The small gas decay tanks have the same administrative activity limit, 6000 Ci, as the large tanks. Since the activity of the gases collected during the degassing operation will be much higher than that collected during normal operation, a smaller tank volume is required to stay below the limit of 6000 Ci. This is the reason the tanks provided to collect the gas from the degassing operation are smaller than the tanks provided for normal operation and why the large gas decay tanks cannot be used for this degassing operation.

No sampling connections are provided on the small tanks. Prior to degassing the reactor coolant system, the total gaseous activity of the coolant should be determined. The fission gas activity inventory will be distributed equally among the six tanks through a common inlet header. With this arrangement, assuming typical coolant concentrations, the activity inventory in any one tank will be less than the normal administrative limit of 6000 Ci of equivalent Xe-133 (as

discussed in Section 14.2.3). Assuming operation with up to 1% fuel defects, the inventory in each small gas decay tank would be greater than this but less than the ODCM limit.

11.1.2.2.8 Compressors

Two compressors are provided for continuous removal of gases from equipment discharging to the plant vent header. These compressors are of the water-sealed centrifugal displacement type. Operation of each of the compressors is controlled by a selector switch allowing one compressor to operate at any one time. Construction is cast iron, bronze fitted. A mechanical seal is provided to maintain outleakage of compressor seal-water at a negligible level.

11.1.2.2.9 Waste Evaporator Package

Waste Evaporator Package has been retired.

11.1.2.2.10 Distillate Storage Tanks

Two distillate storage tanks are provided.

The tanks are horizontal, cylindrical type with standard flanged and dished heads. Each tank is provided with heaters for cold weather temperature control.

11.1.2.2.11 Waste Condensate Tanks

Two 1000-gal waste condensate tanks are provided to collect liquid wastes that are suitable for direct release to the river. The tanks are vertical, cylindrical types with one standard flanged and dished head and one flat head. They are located on the 80-ft elevation of the primary auxiliary building and are constructed of austenitic stainless steel.

11.1.2.2.12 Baler

The balers have been retired and removed from the facility.

11.1.2.2.13 Nitrogen Manifold

Nitrogen, used as cover gas in the vapor space of various components, is supplied from a dual manifold. Pressure control valves automatically switch from one manifold to the other, to ensure a continuous supply of gas.

11.1.2.2.14 Hydrogen Manifold

Hydrogen is supplied to the volume control tank to maintain the hydrogen concentration in the reactor coolant. The hydrogen is supplied from a dual manifold. Pressure control valves automatically switch from one manifold to the other to ensure a continuous supply of gas.

11.1.2.2.15 Gas Analyzer

An automatic gas analyzer with a nominal 1-hr recycle time is provided to monitor the concentrations of oxygen and hydrogen in the cover gas of tanks discharging to the radiogas vent header. Upon indication of a high oxygen level, an alarm sounds to alert the operator.

11.1.2.2.16 Pumps

Pumps used throughout the system for draining tanks and transferring liquids are shown on Figure 11.1-1 sheets 1 and 2.

The wetted surfaces of all pumps are stainless steel.

11.1.2.2.17 Piping

Piping carrying liquid wastes is stainless steel while all gas piping is carbon steel. Piping connections are welded except where flanged connections are necessary to facilitate equipment maintenance.

11.1.2.2.18 Valves

All valves exposed to gases are carbon steel. All other valves are stainless steel.

Stop valves are provided to isolate each piece of equipment for maintenance, to direct the flow of waste through the system, and to isolate storage tanks for radioactive decay.

Relief valves are provided for tanks containing radioactive waste if the tanks might be over-pressurized by improper operation or component malfunction. Tanks containing wastes, which contain oxygen and are normally of low activity concentrations are vented into the auxiliary building exhaust system.

11.1.3 Design Evaluation

11.1.3.1 Liquid Wastes

Liquid wastes are primarily generated by plant operations. The Annual Effluent and Waste Disposal Report provides the total liquid effluent activity released by isotope.

Appendix 11B presents the results of an original plant preoperational assessment of river water dilution factors between the Indian Point site and the nearest public drinking water intake and is being retained for historical purposes.

11.1.3.2 Gaseous Wastes

Gaseous wastes consist primarily of hydrogen stripped from coolant discharged to the chemical and volume control system holdup tanks during boron dilution, nitrogen and hydrogen gases purged from the chemical and volume control system tank when degassing the reactor coolant, nitrogen from the closed gas blanketing system, and controlled depressurization of the containment atmosphere. The gas decay tanks will permit decay of waste gas before discharge in accordance with the ODCM. The annual gaseous release to atmosphere is given in the Annual Effluent Release and Waste Disposal Report.

Compliance of gaseous effluent releases to regulatory requirements is reflected in the plant's Technical Specifications.

11.1.3.3 Solid Wastes

Solid wastes consist of solidified waste liquid concentrates and sludges, spent resins and filters, and miscellaneous materials such as paper and glassware.

Waste liquid concentrates and sludges are solidified in liners. Spent resins and plant filters are also packaged in liners, which are placed in waste casks for removal to a burial facility. Miscellaneous wastes are packaged in 52 or 55-gal drums. When possible, solid waste is sent to a licensed incinerator, volume reduction center, or material recovery center. Preparation of solid radwastes for shipment and offsite disposal is conducted in accordance with a process control program. Certain activities such as inspections and verifications are considered to be Quality Control activities.

Changes to operations and design were implemented during 1981 to reduce the amount of solid radioactive waste packaged at the plant. The solid radwaste associated with liquid radwaste processing has been reduced by a significant factor since 1981. This was accomplished by using sluicable ion exchange demineralizers instead of evaporators and solidification of concentrate bottoms. It is intended to continue with the use of demineralizers as the prime method of liquid waste processing with evaporation and solidification as the backup method.

Sandblasters are available to remove fixed radioactivity from non-compressible items such as gas bottles, I-beams, angle irons, steel plates, and various tools and equipment. A very low volume of contaminated sand (grit) is being generated. This sand is used to fill voids in non-compactable waste containers.

To further reduce solid waste volumes a liquid abrasive bead decontamination unit, an ultrasonic unit and a solvent degreaser unit have been installed in 1985 to remove loose and fixed contamination from equipment. This equipment can then be reused in the controlled area or released for uncontrolled use. Also, offsite supercompaction and licensed incineration methods are available and used to reduce total burial volumes.

11.1.4 Minimum Operating Conditions

Minimum operating conditions for the waste disposal system are enumerated in the ODCM.

TABLES 11.1-1 thruogh 11.1-5
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TABLE 11.1-6
Waste Disposal System Components Code Requirements

Component	Code
Chemical drain tank	No code
Reactor coolant drain tank	ASME III,[1] Class C
Sump tank	No code
Spent resin storage tanks	ASME III,[1] Class C
Gas decay tanks	ASME III,[1] Class C
Waste holdup tank	No code
Water condensate tank	No code
Distillate storage tank No code	
Waste filter	No code
Piping and valves	USAS-B31.1,[2] Section 1

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

TABLE 11.1-7 (Sheet 1 of 2)
Component Summary Data

Tanks	**Quantity**	**Type**	**Volume**	**Design Pressure**	**Design Temperature F°**	**Material**
Reactor Coolant drain	1	H	350 gal	25 psig	267	ss
Chemical drain	1	V	375 gal	Atm	180	ss
Sump	1	V	375 gal	Atm	150	ss
Waste holdup	1	H	3300-ft^3	Atm	150	ss
Spent resin Storage	1	V	300-ft^3	100 psig	150	ss
Waste condensate	2	V	1000 gal	Atm	180	ss
Distillate storage	2	H	25000 gal	17 psig	250	cs
Gas decay (large)	4	V	525-ft^3	150 psig	150	cs
Gas decay (small)	6	V	40-ft^3	150 psig	150	cs

Pumps	**Quantity**	**Type**	**Flow gpm**	**Head ft**	**Design Pressure psig**	**Design Temperature F°**	**Material** 1
Reactor coolant drain (A)	1	H, CC	50	175	100	267	ss
Reactor coolant drain (B)	1	H, CC	150	175	100	267	ss
Chemical drain	1	H, C_2	20	100	100	180	ss

TABLE 11.1-7 (Sheet 2 of 2)
Component Summary Data

Pumps	Quantity	Type	Flow gpm	Head Ft	Design Pressure psig	Design Temperature F°	Material[1]
Sump tank	2	H, C[2]	20	100	150	180	ss
Waste condensate	2	H, C[2]	20	100	150	180	ss
Waste evaporator feed	1	H, C[2]	20	100	150	180	ss
Waste transfer	1	H, C[2]	30	215	105	70	ss
Distillate recirculation	2	H, C[2]	200	100	43[3]	120[4]	ss
Reactor cavity pit (2RCPP)	1	Sub-merge V, C	100	50	150	120	ss
Reactor cavity pit (1RCPP)	1	Sub-merge V, C	20	62	150	120	ss

Miscellaneous	Quantity	Capacity	Type
Waste gas compressors	2	48 f^3/min	H, C[2]

Key:

H = Horizontal	C = Centrifugal	CC = Carbon Steel
V = Vertical	CC = Centrifugal canned	SS = Stainless Steel

Notes:
1. Wetted surfaces only.
2. Mechanical seal provided.
3. 43 psig is the operating differential pressure of the pump.
4. 120°F is the maximum operating temperature of the pump

TABLE 11.1-8
DELETED

11.1 FIGURES

Figure No.	Title
Figure 11.1-1 Sh. 1	Waste Disposal System Process Flow Diagram, Sheet 1, Replaced with Plant Drawing 9321-2719
Figure 11.1-1 Sh. 2	Waste Disposal System Process Flow Diagram, Sheet 2. Replaced with Plant Drawing 9321-2730

11.2 RADIATION PROTECTION

11.2.1 Design Bases

Radiation protection at Indian Point 2 incorporates a program for maintaining radiation exposures as low as reasonably achievable (ALARA). The ALARA program is part of all normal and special work processes. Procedures, designs, modifications, work packages, inspections, surveillances, maintenance activities and plant betterment activities are subjected to ALARA reviews to ensure close reduction actions are taken. Operational and design ALARA training programs are provided to station and support engineering and technical groups. ALARA is taught in Radiation Worker Qualification courses

11.2.1.1 Monitoring Radioactivity Releases

Criterion: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive. (GDC 17)

The containment atmosphere, the plant vent, the containment fan cooler service water discharge, the waste disposal system liquid effluent, the condenser air ejectors, and steam generator blowdown are monitored for radioactivity during normal operations, from anticipated transients, and from accident conditions.

All gaseous effluent from possible sources of accidental releases of radioactivity external to the reactor containment (e.g., the spent-fuel pit and waste handling equipment) will be exhausted from the plant vent, which is monitored. Any contaminated liquid effluent discharged to the condenser circulating water canal is monitored. For the case of leakage from the reactor containment under accident conditions the plant area radiation monitoring system supplemented by portable survey equipment to be kept in the Health Physics office area should provide adequate monitoring of accident releases. The details of the procedures and equipment to be used in the event of an accident are specified in Section 11.2.5, the plant procedures, and the plant emergency plan. The formulation of these details considers the requirements for notification of plant personnel, the utility load dispatcher, and local authorities.

11.2.1.2 Monitoring Fuel and Waste Storage

Criterion: Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels. (GDC 18)

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect inadequate cooling and to detect excessive radiation levels. Radiation monitors are provided to maintain surveillance over the release operation, but the permanent record of activity releases is provided by radiochemical analysis of known quantities of waste.

The spent fuel pit temperature and level are monitored to assure proper operation, as discussed in Section 9.3.3.2.3.

A controlled ventilation system removes gaseous radioactivity from the atmosphere of the fuel storage and waste treating areas of the auxiliary building and discharges it to the atmosphere via the plant vent. Radiation monitors are in continuous service in these areas to actuate high-activity alarms on the control board annunciator, as described in Section 11.2.3.

11.2.1.3 Fuel and Waste Storage Radiation Shielding

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

Auxiliary shielding for the waste disposal system and its storage components is designed to limit the dose rate to levels not exceeding 0.75 mrem/hr in normally occupied areas, to levels not exceeding 2.0 mrem/hr in intermittently occupied areas, and to levels not exceeding 15 mrem/hr in limited occupancy areas.

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

11.2.1.4 Protection Against Radioactivity Release From Spent Fuel and Waste Storage

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

All waste handling and storage facilities are contained and equipment designed so that accidental releases directly to the atmosphere are monitored and will not exceed applicable limits; refer also to Sections 11.1.2, 14.2.2, and 14.2.3. The components of the waste disposal system are designed to the pressures given in Table 11.1-7 and the codes given in Table 11.1-6. Hence, the probability of a rupture or failure of the system is exceedingly low.

11.2.2 Shielding

11.2.2.1 Design Basis

Radiation shielding is designed for reactor operation at maximum calculated thermal power and to limit the normal operation radiation levels at the site boundary below those levels allowed for

continuous nonoccupational exposure. The plant is capable of continued safe operation with 1-percent fuel element defects (as discussed in Section 14.2.3).

In addition, the shielding provided ensures that in the event of a hypothetical accident, the integrated offsite exposure due to the contained activity does not result in any offsite radiation exposures in excess of applicable limits.

Operating personnel at the plant are protected by adequate shielding, monitoring, and operating procedures. When additional shielding is suggested, and permitted as a function of reactor operating mode, it will be evaluated in the context of the station ALARA program and temporary shielding procedures. Modifications to existing structures or shields, which may alter personnel or equipment qualification dose will be evaluated in the design review process. The permanent large and significant shielding arrangement is shown on Figures 1.2-5, 5.1-3, 5.1-4, 5.1-6 and 5.1-7. Shielding arrangements may be altered consistent with the radiation protection plan and the ALARA program station administration orders.

Detailed and periodic surveys of all restricted area radiation levels are performed. All high radiation areas are appropriately marked and access controlled in accordance with 10 CFR 20 and other applicable regulations and station procedures as well as the Technical Specifications.

In accordance with NUREG-0737, Item II.B.2, each power reactor licensee was required to perform a radiation and shielding design review of spaces around systems that may, as a result of an accident, contain highly radioactive material. Additionally, each licensee was required to provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedure controls. Indian Point Unit 2 shielding design review and corrective action were reviewed during an NRC inspection in May 1983. The inspection report[1] and a safety evaluation report[2] concluded that the requirements of NUREG 0737, Item II.B.2 were met at Indian Point Unit 2.

The shielding is divided into five categories according to function. These functions include the primary shielding, the secondary shielding, the accident shielding, the fuel transfer shielding, and the auxiliary shielding.

11.2.2.1.1 Primary Shield

The primary shield is designed to:

1. Reduce the neutron fluxes incident on the reactor vessel to limit the radiation induced increase in nil ductility transition temperature.

2. Attenuate the neutron flux sufficiently to limit activation of plant components.

3. Limit the gamma fluxes in the reactor vessel and the primary concrete shield to avoid excessive temperature gradients or dehydration of the primary shield.

4. Reduce the residual radiation from the core, reactor internals, and reactor vessel to levels, which will permit access to the region between the primary and secondary shields after plant shutdown.

5. Reduce the contribution of radiation leaking to obtain optimum division of the shielding between the primary and secondary shields.

11.2.2.1.2 Secondary Shield

The main function of the secondary shielding is to attenuate the radiation originating in the reactor and the reactor coolant. The major source in the reactor coolant is the Nitrogen-16 activity (83 $\mu Ci/cm^3$ maximum), which is produced by neutron activation of oxygen during passage of the coolant through the core. The secondary shield will limit the full power dose rate outside the containment building to less than 0.75 mrem/hr.

11.2.2.1.3 Accident Shield

The main purpose of the accident shield is to ensure radiation levels outside the containment building are within applicable limits following a maximum credible accident.

11.2.2.1.4 Fuel Handling Shield

The fuel handling shield is designed to facilitate the removal and transfer of spent fuel assemblies and control rod clusters from the reactor vessel to the spent-fuel pit. It is designed to attenuate radiation from spent fuel, control clusters, and reactor vessel internals to less than 2.0 mrem/hr at the refueling cavity water surface and less than 0.75 mrem/hr in areas adjacent to the spent-fuel pit.

11.2.2.1.5 Auxiliary Shielding

The function of the auxiliary shielding is to protect personnel working near various system components in the chemical and volume control system, the residual heat removal system, the waste disposal system, the sampling system and the high radiation sampling system sentry panels. The shielding provided for the auxiliary building is designed to limit the dose rates to less than 0.75 mrem/hr in normally occupied areas, and at or below 2.0 mrem/hr in intermittently occupied areas during normal operation. Under accident conditions, samples are diverted to a shielded high radiation sampling system tank. Liquid can be pumped from this tank back into the containment.

An additional room has been constructed in the primary auxiliary building (elevation 98-ft) to provide additional shielding protection for operators. The walls are seismically qualified to avoid damage to the equipment in the room after a design-basis accident. In order to reduce personnel exposure during accident conditions, all gas sample lines to the gas analyzers have been provided with a nitrogen purge capability. This system purges all the sampled gases from the sample lines and returns them to their source.

11.2.2.2 <u>Shielding Design</u>

11.2.2.2.1 Primary Shield

The primary shield consists of the core baffle, water annuli, barrel-thermal shield (all of which are within the reactor vessel), the reactor vessel wall, and a concrete structure surrounding the reactor vessel.

The primary shield immediately surrounding the reactor vessel consists of an annular reinforced concrete structure extending from the base of the containment to an elevation of 69-ft. The lower portion of the shield is a minimum thickness of 6-ft of regular concrete ($q = 2.3\ g/cm^3$) and

is an integral part of the main structural concrete support for the reactor vessel. It extends upward to join the concrete cavity over the reactor. The reactor cavity, which is approximately rectangular in shape, extends upward to the operating floor with vertical walls 4-ft thick, except in the area adjacent to fuel handling, where the thickness is increased to 6-ft. A shielding collar is provided at each point where the eight reactor coolant pipes penetrate the primary shield.

The primary concrete shield is air cooled to prevent overheating and dehydration from the heat generated by radiation absorption in the concrete. Eight "windows" have been provided in the primary shield for insertion of the ex-core nuclear instrumentation. Cooling for the primary shield concrete and the nuclear instrumentation is provided by 12,000 cfm cooling air.

The primary shield neutron fluxes and design parameters are listed in Table 11.2-2.

11.2.2.2.2 Secondary Shield

The secondary shield surrounds the reactor coolant loops and the primary shield. It consists of the annular crane support wall, the operating floor, and the reactor containment structure. The containment structure also serves as the accident shield.

The lower portion of the secondary shield above grade consists of the 4-ft 6-in. thick cylindrical portion of the reactor containment and a 3-ft concrete annular crane support wall surrounding the reactor coolant loops. The secondary shield will attenuate the radiation levels in the primary loop compartment from a value of 25 rem/hr to a level of less than 0.75 mrem/hr outside the reactor containment building. Penetrations in the secondary shielding are protected by supplemental shields.

The secondary shield design parameters are listed in Table 11.2-3.

11.2.2.2.3 Accident Shield

The accident shield consists of the 4-ft 6-in. thick reinforced concrete cylinder capped by a hemispherical reinforced concrete dome of a 3-ft 6-in. thickness. This shielding includes supplemental shields in front of the containment penetration.

The equipment access hatch is shielded by a 3-ft 6-in. thick concrete shadow shield and 1-ft 6-in. thick concrete roof to reduce scattered dose levels in the event of loss of reactor coolant accident accompanied by a complete core meltdown.

The accident shield design parameters are listed in Table 11.2-4.

11.2.2.2.4 Fuel Handling Shield

The refueling cavity, flooded to approximately elevation 93.7-ft during refueling operations, provides a temporary water shield above the components being withdrawn from the reactor vessel. The water height during refueling is approximately 24.50-ft above the reactor vessel flange. This height ensures that a minimum of 10.50-ft of water will be above the active fuel of a withdrawn fuel assembly. Under these conditions, the dose rate is less than 2.0 mrem/hr at the water surface.

The fuel transfer canal is a passageway connected to the reactor cavity extending to the inside surface of the reactor containment. The canal is formed by two concrete walls each 6-ft thick,

which extends upward to the same height as the reactor cavity. During refueling, the canal is flooded with borated water to the same height as the reactor cavity.

The spent fuel assemblies and control rod clusters are remotely removed from the reactor containment through the horizontal spent fuel transfer tube and placed in the spent fuel pit. Concrete, 6-ft thick, shields the spent fuel transfer tube. This shielding is designed to protect personnel from radiation during the time a spent fuel assembly is passing through the main concrete support of the reactor containment and the transfer tube. Radial shielding during fuel transfer is provided by the water and concrete walls of the fuel transfer pit. An equivalent of 6-ft of regular concrete is provided to ensure a maximum dose value of 0.75 mrem/hr in the areas adjacent to the spent fuel pit.

Spent fuel is stored in the spent fuel pit, which is located adjacent to the containment building. Shielding, above grade elevation, for the spent fuel storage pit is provided by concrete walls 6-ft thick and is flooded to a level such that the water height is greater than 13-ft above the spent fuel assemblies.

The refueling shield design parameters are listed in Table 11.2-5.

11.2.2.2.5 Auxiliary Shield

The auxiliary shield consists of concrete walls around certain components and piping, which process reactor coolant. In some cases, the concrete block walls are removable to allow personnel access to equipment during maintenance periods. Periodic access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded so that compartments may be entered without having to shut down and, possibly, to decontaminate the adjacent system.

The shielding material provided throughout the auxiliary building is regular concrete (r = 2.3 g/cm^3). The principal auxiliary shielding provided is tabulated in Table 11.2-6.

11.2.3 <u>Radiation Monitoring System</u>

11.2.3.1 <u>Design Bases</u>

The radiation monitoring system is designed to perform two basic functions:

1. Warn of any radiation health hazard, which might develop.
2. Give early warning of a plant malfunction, which might lead to a health hazard or plant damage.

Instruments are located at selected points in and around the plant to detect, compute, and record the radiation levels. In the event the radiation level should rise above a desired setpoint, an alarm is initiated in the control room. The automatic radiation monitoring system operates in conjunction with regular and special radiation surveys and with chemical and radio-chemical analyses performed by the plant staff. Adequate information and warning is thereby provided for the continued safe operation of the plant and assurance that personnel exposure does not exceed 10 CFR 20 limits.

11.2.3.2 Radiation Monitoring Betterment Program

A new system has been installed to replace the original process radiation monitoring system. Each of the original monitors is removed from service after installation and testing of the new monitor. The new system is described below as it currently exists.

The process radiation monitoring system is a digital system with the following major components: individual radiation monitoring units for each monitored process line; a minicomputer unit located in the technical support center; a CRT display and printer located in the central control room; and annunciators located in the central control room.

The minicomputer unit includes a console with CRT and typer, disk drive and magnetic tape drive. It communicates digitally with the individual radiation monitoring units, and processes, records, and displays data.

Table 11.2-7 shows the process streams monitored by the individual radiation monitor units, along with the normal maximum channel output. Each monitor unit monitors a sample of the process fluid, which is piped through a bypass loop. The sample is cooled if required. To facilitate maintenance and calibration, the bypass loop can be isolated and purged.

The liquid and airborne monitors utilize an off-line sampler(s) and a gamma or beta scintillation detectors to measure radioactivity present in a sample. Each monitor has a micro-processor, which communicates with the minicomputer.

Each monitor will activate an annunciation alarm in the event of failure, high radiation, or high temperature where applicable.

The minicomputer and the CRT/printer unit are powered from a battery-backed inverter. As discussed below, several monitor units receive power from MCC-26A and MCC-26BB, which are powered by an emergency diesel generator in the event of loss of other power sources.

Information on specific monitors is given in the following sections.

11.2.3.2.1 Service Water from Component Cooling Heat Exchangers Monitors

Monitors R39 and R40 monitor the service water from component cooling heat exchangers 21 and 22, respectively. Radioactivity in these streams would indicate a component cooling heat exchanger leak when there is radioactivity in the component cooling loop. These monitors are powered from MCC-26A. They are wired to a control room annunciator, independent of their communications loop through the minicomputer.

11.2.3.2.2 Containment Air Monitors

Monitors R41 and R42 monitor the containment atmosphere for particulate and gaseous activity, respectively. These monitors are seismically qualified, and their power supplies are class IE. Either monitor, on detection of a high activity level, will initiate containment ventilation isolation, consisting of closure of the two containment purge supply valves, the two containment purge exhaust valves, and the containment pressure relief valves. Although IP2 plant design has always included isolation of these valves upon detection of high radioactivity in the containment atmosphere, this function has also been analyzed and credited for IP2 compliance with

NUREG-0737, Item II.E.4.2.7 (Reference 24). Their signals are provided to control room indicators and recorders and to the safety assessment system.

11.2.3.2.3 Plant Vent Air Monitors

R43 monitors the air in the plant vent for particulate and iodine activity, while R44 monitors for gaseous activity. They are seismically qualified, and their power supplies are class IE. On detection of a high activity level, R44 initiates containment ventilation isolation as described in the preceding section, and also initiates closure of the gas discharge valve in the waste gas disposal system. Their signals are provided to control room indicators and recorders and to the safety assessment system. Additionally, an indicator for monitor R44 is located at the waste disposal panel.

11.2.3.2.4 Condenser Air Ejector Discharge Monitor

The gas removed from the condenser by the air ejector is monitored for gaseous radioactivity (which is indicative of steam generator tube leakage) by monitor R45. On the detection of high radiation, the condenser exhaust gas is diverted from the atmospheric discharge to the containment. A control room alarm is provided independent of the communications loop. The monitor, which receives power from a highly reliable source backed up by the emergency diesel generators, is capable of functioning after a steam generator tube rupture coincident with loss of offsite power.

11.2.3.2.5 Service Water Return from Containment Fan Cooler Units

Two redundant monitors, R46 and R53, monitor the service water return from all containment fan cooler units. Small bypass flows from each of the heat exchangers and from the fan motor coolers are mixed in a common header and monitored. During a loss of coolant accident, radioactivity at this point would indicate a leak from the containment atmosphere into the cooling water. Upon indication of a high radiation level, each heat exchanger is sampled to determine, which unit is leaking. Each of these channels is hardwired to a safety-related display unit, a recorder and an annunciator, all in the control room. The communications link through the minicomputer is isolated from each of these channels by an isolation device. The channels receive power from MCC-26A. These monitors, the display units and the connecting piping are designed to be capable of functioning after a safe shutdown earthquake.

11.2.3.2.6 Component Cooling Radiation Monitor

This channel, R47, monitors the component cooling loop for radioactivity, which would indicate a leak of reactor coolant from the reactor coolant system and/or the residual heat removal loop. An interlock initiates closure of a valve in the component cooling surge tank vent line in the event a high radiation level is detected. Closure of this valve will prevent gaseous activity release. Component cooling activity is recorded and displayed in the control room, and high activity initiates a control room annunciator. The display unit, recorder and annunciator are independent of the minicomputer communications loop. The monitor is isolated from the communications loop by an isolation device. This monitor is powered from MCC-26A, and is designed to be capable of functioning after a safe shutdown earthquake.

11.2.3.2.7 Waste Condensate Tank Discharge Line

This channel, R48, monitors liquid releases from the Waste Condensate Tanks. Automatic valve closure is initiated by this monitor to prevent further release after a high radiation level is detected. This monitor is hardwired to a control room chart recorder. It receives power from MCC-26A.

11.2.3.2.8 Steam Generator Blowdown Monitor

This monitor, R49, monitors the liquid blowdown from the secondary side of the steam generators. Radioactivity in this stream would indicate a primary-to-secondary leak, providing information to back up the condenser air removal gas monitor. Samples from the bottoms of all four steam generators are mixed in a common header and the common sample is monitored. Upon indication of high activity, an interlock from monitor R49 closes all steam generator blowdown containment isolation valves and the city water supply to the steam generator blowdown tank spray. Each steam generator is individually sampled to determine the source. Due to the location of monitor R49, the sample travel time from the sample point to the monitor is 90 seconds to 2 minutes (as discussed in Section 14.2.4). The sample point is downstream of the blowdown line containment isolation valves, which close on Phase A containment isolation signal. The signal from R49 is one of the parameters available to the operator to diagnose a steam generator tube rupture backing up the indication from the condenser air ejector monitor. Initiation of safety injection and Phase A isolation, in response to a steam generator tube rupture, could prevent R49 from seeing the increase in activity resulting from the steam generator tube rupture. R49 is not a primary indication to the operator of steam generator tube rupture, thus the ability of the operator to respond to steam generator tube rupture will not be adversely affected.

Monitor R49 receives power through MCC-26BB and is designed to be capable of operating after a safe shutdown earthquake. It will annunciate in the control-room independent of its communication loop through the minicomputer. The monitor is hardwired to a recorder in the control room.

11.2.3.2.9 Waste Gas Decay Tank

This monitor, R50, indicates activity in the waste gas decay tanks. It is hardwired to a recorder in the control room and also annunciates in the control room, independent of the communication loop through the minicomputer. It receives power from MCC-26A.

11.2.3.2.10 Secondary Boiler Blowdown Purification System

This monitor, R51, indicates activity in the system effluent and the Unit 1 North Curtain Drain sump discharge. It enables plant operators to take corrective action in the event of high activity. It is powered from a Unit 1 motor control center. It alarms in the control room independent of its communications loop through the minicomputer.

11.2.3.2.11 Steam Generator Blowdown Purification System Cooling Water Monitor

This monitor, R52, monitors the cooling water from the Unit 1 secondary boiler blowdown purification system, which can be used to process steam generator blowdown effluents from Unit 2. It actuates an alarm in the control room. It is not required to function in the event of an earthquake.

11.2.3.2.12 Liquid Waste Distillate Radiation Monitor

This monitor, R54, is powered from a Unit 1 motor control center. It alarms in the central control room independent of the communications loop through the minicomputer. This monitor terminates the distillates tank discharges upon detecting high activity.

11.2.3.2.13 Steam Generator Secondary System Monitors

There are four monitors (R55A, R55B, R55C, and R55D) for activity in the secondary systems of the steam generators. A small flow from each is cooled and depressurized, and then monitored.

11.2.3.2.14 Effluent Discharge to ENIP3

This monitor, R57, is not required to function to mitigate any postulated accident. It monitors the contents of the sewage ejector pit, located in Unit 1 and trips the ejector pumps if high activity is detected. A central control room alarm is provided, independent of the communications loop. Power to monitor R57 is supplied from a Unit 1 source. This monitor terminates sewage transfer upon detecting high activity.

11.2.3.2.15 House Service Boilers

This monitor, R59, is powered from a Unit 1 motor control center. It indicates any activity that may be present in the condensate return. It alarms in the control room.

11.2.3.2.16 Stack Radiation Monitor

R60 has monitors for gaseous, particulate, and iodine activity in the air in the stack.

11.2.3.2.17 Maintenance and Outage Building Ventilation Exhaust

The air exhausted from elevation 95' of the Maintenance and Outage Building is monitored by R-5976 for particulates and gases. This monitor is integrated into the process monitoring system.

11.2.3.2.18 Sphere Foundation Sump Liquid Effluent

Monitor R-62 monitors the activity of the liquid discharge from the Unit 1 Sphere Foundation Sump drainage. This monitor alarms of the common process radiation monitor panel for high radiation.

11.2.3.2.19 Main Steam/Steam Generator Tube Leakage

Nitrogen-16 monitors R-61A, R-61B, R-61C, and R-61D are located near the main steam lines in the Auxiliary Boiler Feed Pump Building and when a steam generator tube leaks sufficiently the N-16 monitor will alarm.

11.2.3.3 Original Radiation Monitoring System

11.2.3.3.1 Control Room Cabinet

Most of the control room system equipment is centralized in three cabinets. High reliability and ease of maintenance are emphasized in the design of this system. Sliding channel drawers are used for rapid replacement of units, assemblies, and entire channels. It is possible to remove the various chassis completely from the cabinet after disconnecting the cables from the rear of these units.

Radiation recorders and associated preamplifiers for channels R-11, R-12, R-13, R-14, R-15, R-16, R-17, R-18, R-19, R-20, and R-23 have been installed in a new radiation recorder panel SA-1, which is adjacent to Panel SA in the central control room. This installation allows for continuous monitoring and trending of these channels during emergencies. The new panel includes a 36-point annunciator panel and eleven recorders, one for each parameter indicated above.

11.2.3.3.2 Monitor Channel Output

The maximum channel output of the radiation monitors is given in Table 11.2-7.

11.2.3.3.3 Operating Conditions

Where fluid temperature is too high for the monitor, a cooling device with temperature indication is included. The different operating temperature ranges are within the design limits of the sensors.

The relation of the radiation monitoring channels to the systems with which they are associated is given in the sections describing those systems. Routine test and recalibrations will ensure that the channels operate properly.

The components of the radiation monitoring system are designed according to the following environmental conditions:

1. Temperature - an ambient temperature range of 40°F to 120°F.

   [Note - *Equipment located in the control room area may be specified for smaller temperature and humidity ranges because of the controlled environment provided by the heating and ventilating system.*]

2. Humidity - 0 to 100-percent relative humidity.

   [Note - *Equipment located in the control room area may be specified for smaller temperature and humidity ranges because of the controlled environment provided by the heating and ventilating system.*]

3. Pressure - components in the auxiliary building and control room are designed for normal atmospheric pressure. Area monitoring system components inside the containment are designed to withstand test pressure.

4. Radiation - process and area radiation monitors are of a nonsaturating design so that they "peg" full-scale if exposed to radiation levels up to 100 times full scale indication. Process monitors are located in areas where the normal and postaccident background radiation levels will not affect their usefulness.

The radiation monitoring system is divided into the following subsystems:

1. The process radiation monitoring system, which monitors various fluid streams for indication of increasing radiation levels.

2. The area monitoring system, which monitors area radiation in various parts of the plant.

3. Environmental radiation monitoring system, which monitors radiation in the area surrounding the plant.

Portable alarming area radiation monitors and continuous area monitors are used in the Unit 1 area utilized for interim storage of dry active waste.

11.2.3.3.4 Original Process Radiation Monitoring System

This system monitors radiation levels in various plant operating systems. The output from each channel detector is transmitted to the radiation monitoring system cabinets located in the control room area where the radiation level is indicated by a meter and recorded by a multipoint recorder. High radiation level alarms are annunciated on the main control room board and indicated on the radiation monitoring system cabinets.

The installed monitoring systems are not designed to determine the nature and amount of radioactivity in the systems being monitored, but are designed to detect the concentrations of the isotopes in their respective streams or areas as indicated in Table 11.2-7. These systems monitor gross activity and are designed to generate an alarm under abnormal conditions and in most cases generate automatic responses. Isotopic identification and concentrations are determined by grab sample analysis.

Each channel contains a completely integrated modular assembly, which includes the following:

1. Level amplifier

   Amplifies the energy of the radiation pulse to provide a discriminated output to the log level amplifier.

2. Log level amplifier

   Accepts the shaped pulse of the level amplifier output, performs a log integration, (converts total pulse rate to a logarithmic analog signal) and amplifies the resulting output for suitable indication and recording.

3. Power supplies

   Power supplies are contained in each drawer for furnishing the positive and negative voltages for the transistor circuits, relays and alarm lights, and for providing the high voltage for the detector.

4. Test-calibration circuitry

   These circuits provide a precalibrated analog signal to perform channel test, and a solenoid-operated radiation check source to verify the operation of the channel. An annunciator light on the main control board indicates when the channel is in the test-calibrate mode.

5. Radiation level meter

   This meter, mounted on the drawer, has a scale calibrated logarithmically in counts per minute from 10^1 to 10^4, and 10^1 to 10^6. The level signal is also recorded by the recorder.

6. Indicating lights

   These lights indicate high-radiation alarm levels and circuit failure. An annunciator on the main control board is actuated on high radiation.

7. Bistable circuits

   Two bistable circuits are provided, one to alarm on high radiation (actuation point may be set at any level within the range of the instruments), and one to alarm on loss of signal (circuit failure).

8. A remotely-operated long-half-life radiation check source is furnished in each channel. The energy emission ranges are similar to the radiation energy spectra being monitored. The source strength is sufficient to cause approximately mid-range indication of the detector unit.

The process radiation monitoring system consists of the radiation monitoring channels, which are discussed in the following pages.

11.2.3.3.4.1 Containment and Plant Vent Air Particulate Monitors (R-11 and R-13)

These monitors are no longer functional.

11.2.3.3.4.2 Containment Radioactive Gas Monitor (R-12)

Information in this paragraph is being retained for historical perspective. During normal plant operation, the tritium level in the reactor coolant will be limited to a sufficient level to ensure an acceptable tritium activity in the refueling water. With a containment purge rate of 40,000 cfm, the maximum concentration of tritium in the containment air will be less than 1/5 of MPC. The basis for this concentration is determined from the assumption that the refueling water evaporation rate is 100 1b/hr, the containment is purged for 2 hr at the rate of 40,000 cfm prior to access, and that the purge continues during the refueling operation at 40,000 cfm.

During normal plant operation, grab samples from the containment and auxiliary building area will be analyzed for tritium as required.

11.2.3.3.4.3 Plant Vent Gas Monitor (R-14)

This monitor is no longer functional.

11.2.3.3.4.4 Condenser Air Ejector Gas Monitor (R-15)

This monitor is no longer functional.

11.2.3.3.4.5 Containment Fan Cooling Water Monitors (R-16 and R-23)

These monitors are no longer functional.

11.2.3.3.4.6 Component Cooling Loop Liquid Monitor (R-17)

This monitor is no longer functional.

11.2.3.3.4.7 Waste Disposal System Liquid Effluent Monitor (R-18)

This monitor is no longer functional.

11.2.3.3.4.8 Waste Disposal System Gas Analyzer Monitor (R-20)

This monitor has been replaced by R-50.

11.2.3.3.4.9 Steam Generator Liquid Sample Monitor (R-19)

This monitor is no longer functional.

11.2.3.3.4.10 Gross Failed Fuel Detector

This detector is no longer functional.

11.2.3.3.4.11 Iodine-131 Monitors

These monitors are no longer functional

11.2.3.3.4.12 Calibration of Process and Effluent Monitors

Liquid and gaseous sources, similar to those expected during normal plant operation, will not be used to verify proper installation and operating capability of the detectors. A check source, installed in the sampler, will be used to verify that the detectors are operating and properly installed.

A primary calibration was performed on a one-time basis in the vendor's design verification test. Further primary calibrations are not required since the geometry cannot be significantly altered within the sampler. The design verification test utilizes typical isotopes of interest to determine proper detector response.

Secondary standard calibrations are performed with a radiation source of known activity. These single point calibrations are used to verify the original vendor calibration. Cesium sources are used for both gaseous and liquid effluent monitors. The secondary standard calibrations are performed by removing the detector and placing the source on the sensitive area of the detector. The secondary standard calibrations are performed at each refueling outage.

An additional secondary calibration of each monitor is performed periodically by manually sampling the system involved and analyzing for composition and activity using gamma spectrometry. The knowledge of the isotopes present is then used for proper instrument calibration.

There are no specific routine maintenance procedures for the radiation monitoring system monitors. If background buildup is observed, decontamination procedures will be performed.

11.2.3.3.5 Original Area Radiation Monitoring System

The Unit 1 area radiation monitoring system consists of five channels, which monitor radiation levels in various Unit 1 locations. These area are listed below:

Channel	Area Monitor
ARM-1	Drum Storage Area Corridor
ARM-2	Pedestrian Tunnel
ARM-3	Nuclear Service Building SBBPS HX Room
ARM-4	Evaporator Bottom Pumps Room Corridor
ARM-5	Fuel Handling Floor

Channels ARM-1 through ARM-5 consist of a fixed position gamma sensitive sodium iodide detector. The detector output is amplified and shaped locally, and displayed both locally and in the control room. Both local and control room logarithmic meters span the range from 0.1 mR/hr to 1000 mR/hr. The control room annunciator is common to all five units.

The Unit 2 area radiation monitoring system consists of six channels, which monitor radiation levels in various areas of Unit 2. These areas are listed as follows:

Channel	Area Monitor
R-1	Control Room
R-2	Containment
R-4	Charging pump room
R-5	Spent fuel building
R-6	Sampling room
R-7	Incore instrument area

Channels R-1, R-2 and R-4 through R-7 consist of a fixed position gamma sensitive Geiger-Mueller tube detector. The detector output is amplified and the log count-rate is determined by the integral amplifier at the detector. The radiation level is indicated locally at the detector and at the radiation monitoring system (RMS) cabinets. The RMS signals are also logged and trended (recorded) by the plant computer. High radiation alarms are displayed on the main annunciator, the radiation monitoring cabinets, and at the detector location. When radiation

levels drop below the high level alarm setpoint, the "high" alarms on the monitors are reset automatically. The automatic reset procedure also exists for the "low" alarms.

The control room annunciator provides a single window, which alarms for any channel detecting high radiation. Verification of which channel has alarmed is done at the radiation monitoring system cabinets. A remotely-operated, long half-life radiation check source is provided in each channel. The source strength is sufficient to produce indication of detector response.

A meter is mounted on the front of each computer-indicator module and is calibrated logarithmically from 0.1 mrem/hr to 10 rem/hr.

A remote meter calibrated logarithmically from 0.1 mrem/hr to 10 rem/hr, is mounted at the detector assembly.

Radiation monitoring system cabinet alarms consist of a red indicator light for high radiation and an amber light to annunciate detector or circuit failure. The remote meter and alarm assembly at the detector contains a red indicator light and a buzzer type alarm annunciator actuated on high radiation.

11.2.3.4 <u>NUREG-0737 Monitors</u>

The following monitors were installed in conformance with NUREG-0737, "Clarification of TMI Action Plan Requirements":

11.2.3.4.1 Containment High Range Radiation Monitors (R-25 and R-26)

Installed within the containment building are two ion chamber type radiation detectors. These detectors are wired to receiving units located on the accident assessment panel. Analog type ratemeters display rem/hr values from 10^0 to 10^7. These values will be continuously recorded on separate strip chart recorders. Computer outputs are also provided as well as alarm output contacts for annunciation of high radiation inside of the containment building. A check feature is also provided for periodic system verification. Pushbuttons for check initiation and reset are provided on the front of each ratemeter.

One of the high-range radiation detectors is installed at the top of the pressurizer and the other on the steam generator wall in such a way that they can monitor dose rates within the containment building. These monitors are intended to provide information about the imminence or extent of a breach of a fission-product barrier.

No control features are provided with this system.

11.2.3.4.2 High-Range, Noble Gas Monitor (R-27)

The high-range noble gas monitor is installed in the boric acid evaporator building on the 84-ft elevation along with a sample station. The monitor is intended to provide information about the magnitude of releases of radioactive materials, should they occur.

The monitor is skid-mounted and fixed in place by anchor bolts; the various parts of the sample station are similarly secured to the wall and floor. Connections have been installed for data processors and displays and to supply electrical power and a nitrogen purge capability. The

display for this monitor is located on the accident assessment panel in the common Units 1 and 2 central control room.

11.2.3.4.3 Main Steam Line Radiation Monitors (R-28, R-29, R-30, and R-31)

Each of the four steam lines is monitored for gross activity by an individual Geiger-Mueller detector assembly, which is positioned next to the lines upstream of the pressure relief valves. The readouts for these detectors are located in the control room on the accident assessment panel. The sensitivity of these monitors is from 10^{-1} to 10^{3} $\mu Ci/cm^3$. Each meter has an alarm output for high radiation. The four separate outputs are wired to independent alarms for each main steam line radiation monitor located on the accident assessment panel in the common Units 1 and 2 control room. Each meter also has recorder output, which is wired to a common multipoint recorder. These monitors are used in combination with the total steam flow from the low range flow meter as a backup method of determining the magnitude of the estimated releases through the atmospheric dump valves and the steam generator safety valves.

Each detector assembly includes a constant depleted uranium source giving a fixed readout. This feature takes the place of the usual electrically activated check source mechanism.

No control features are provided with this system.

11.2.3.4.4 Dual Channel Gas and Particulate Monitor (Deleted)

11.2.3.4.5 PAB Breaker Service Access Area Radiation Monitor R-5987

Area Monitor channel R-5987 is provided to indicate habitability of the primary auxiliary building area between motor control centers 26AA and 26BB. Post-accident access to this area may be required to service accident mitigation equipment. The monitor, which uses a Geiger-Mueller detector, has a range of 0.1 mrem/hr to 10 rem/hr. It provides indication and alarm both locally and in the central control room, and provides input data to the plant computer. It receives power from an instrument bus and is designed to the category 3 criteria of regulatory guide 1.97, rev. 2.

11.2.3.4.6 Post Accident Sampling System Monitors

There are three area radiation monitors, R-37-1, R-37-2, R-37-3, installed for the Post Accident Sampling System. The detectors are Ionization Chambers with readout/alarms located on one of the local Sampling System control panels. There are no control features associated with these monitors.

11.2.3.4.7 Control Room Air Intake

Process radiation monitors R-38-1 and R-38-2 are installed near the intake ducts in the northern and southern sections of the Control Room's fan room. The southern detector is located on the intake air stream for the Unit 1 area of the Control Building excluding the Control Room. The northern detector is near the Unit 2 intake duct where the duct penetrates the north wall of the fan room. If a high radiation condition is sensed entering from either south or north of the Control Building the Control Room Ventilation will switch to the “Incident – Outside Air Filtered Pressurization Mode (Mode 2).”

11.2.4 Environmental Monitoring Program

Environmental monitoring is discussed in section 2.8 and requirements are set forth in the ODCM. The environmental monitoring program and results are described in the Annual Radiological Environmental Operating Report.

11.2.5 Radiation Protection and Medical Programs

In response to an Order Modifying License[18], Con Edison developed a comprehensive action plan[19,20] to upgrade station radiological controls. The action plan was approved by the NRC in Reference 21. Con Edison's plan to maintain program effectiveness was submitted to the NRC in Reference 22. The NRC determined in 1986 that the implementation of the action plan was thorough and complete, and all terms of the order have been satisfactorily completed (Reference 23).

11.2.5.1 Personnel Monitoring

The official and permanent record of accumulated external radiation exposure received by individuals is obtained principally from a thermoluminescent dosimeter (TLD). Direct reading dosimeter provides day-by-day indication of external radiation exposure.

Special or additional TLDs are issued as may be required under unusual conditions. These devices are issued as directed by the environmental health and safety personnel.

The TLDs are processed on a routine basis, usually at monthly intervals.

Annual reports of personnel monitoring are submitted to the NRC in accordance with 10 CFR 20.2206 and Technical Specifications.

11.2.5.2 Personnel Protective Equipment

All personnel are required to wear appropriate protective clothing as specified by a radiation work permit. The nature of the work to be done is the governing factor in the selection of protective clothing to be worn by individuals. The most common protective apparel available is shoe covers, head covers, gloves, and coveralls. Additional items of specialized apparel such as plastic suits, face shields, and respirators are available. In all cases, radiation protection personnel evaluate the radiological conditions and specify the required items of protective clothing to be worn. Respiratory protective devices are available in any situation arising from plant operations in which an airborne radioactive area exists or is expected to exist in excess of applicable limits. In such cases, the airborne concentrations are monitored by radiation protection personnel and the necessary protective devices are specified according to concentration and type of airborne contaminants present.

Respiratory devices available for use include:

1. Full-face respirator (filter or gas canister, negative pressure).
2. Atmosphere supplying respirators (pressure demand, or continuous flow).
3. Airhood.
4. Self-contained breathing apparatus.

Self-contained breathing apparatus will be used in any situation involving oxygen deficient atmospheres.

The appropriate type of respiratory protection equipment required will be determined from 10 CFR, 20.1701-1704.

11.2.5.3 Facilities and Access Provisions

The radiologically controlled area is a portion of an area to which access is limited and additional steps are applied for purposes of occupational dose control and loose radioactive material control. A Radiation Area is an area accessible to personnel in which there exists radiation at such levels that a major portion of the body could receive in any 1 hr a dose in excess of 5.0 mrem at 30 cm from the source. The Radiologically Controlled Areas of IP2 are established, identified, and controlled through plant procedures.

Any area in which radioactive material and radiation are present shall be surveyed, classified, and conspicuously posted with the appropriate radiation caution sign as specified in 10 CFR 20.1902.

The general arrangement of the control point facilities is designed to provide access control to the RCA and it also provides a change location for personal clothing.

Friskers and/or Personnel Contamination Monitors are located at all authorized personnel exits from the radiologically controlled area. All personnel will survey themselves before leaving the controlled area.

Personnel decontamination equipment is available in the controlled area decontamination and first aid rooms.

Administrative and physical security measures are employed to prevent unauthorized entry of personnel to any high radiation area. These measures include the following:

1. Areas in which radiation levels are so high that individuals might receive doses in excess of 100 mrem at 30 cm in 1 hr shall be barricaded and conspicuously posted as "high radiation areas." Administrative controls require the issuance of a radiation work permit prior to entry to any high radiation area.

2. Locations where the above value exceeds 1 rem at 30 cm in 1 hr are conspicuously posted, and in addition, locked doors are provided to prevent unauthorized entry. Keys to these doors are kept under special administrative control. The locks and administrative controls on these doors are arranged so that personnel cannot be prevented from leaving high radiation areas.

The limits for removable surface contamination in the controlled area are as follows:

Removable Radioactive Surface Contaminations

Beta-gamma	500 dpm/100 cm^2
Alpha (uranium and thorium)	20 dpm/100 cm^2
Alpha (other)	20 dpm/100 cm^2

When the above levels are exceeded, the area is posted as a "Contaminated Area." Additionally, all personnel are required to wear appropriate protective clothing for entry. The areas involved will be decontaminated as soon as possible to prevent the spread of contamination. Decontamination will be carried out under the direction of Radiation Protection personnel.

11.2.5.4 Radiation Instrumentation

Laboratory facilities are provided for the radiation protection and chemistry sections. These facilities include both laboratory and calibration rooms. A health physics control station is equipped to analyze routine air samples and contamination swipe surveys. The control station also serves as a central location for portable radiation survey instruments.

"Friskers" and other type personnel monitors are located at appropriate plant locations as dictated by the plant radiation protection program.

A beta-gamma portal monitor is located at all authorized personnel exits from the radiologically controlled area as a final check on personnel leaving the controlled area.

The types of portable radiation survey instruments available for routine monitoring functions are controlled and placed by Health Physics and governed by procedures.

Survey instruments are included in a formal maintenance program to ensure that they are normally calibrated. Calibration and maintenance records are provided for each instrument.

Portable radiation survey instruments are available for use offsite during and following any possible accidental release of radioactivity from the facility. The equipment available and required are controlled by the Emergency Plan and Health Physics procedures.

11.2.5.5 Onsite Treatment Facilities, Equipment and Supplies

Onsite treatment facilities consist of a Decontamination Room and an Examination Room located in the Unit 1 Nuclear Services Building adjacent to the Containment Sphere but outside the external concrete biological shield. An alternate location for the treatment of injured and/or contaminated personnel and for the storage of supplies is the Medical Bureau Examination Room located in the Buchanan Service Center.

Onsite equipment and supplies for the treatment of injured and/or contaminated personnel are controlled by Health Physics Procedures and the Emergency Plan and its Implementing Procedures.

11.2.5.6 Treatment Procedures and Techniques

The procedure and techniques used to treat injured and/or contaminated personnel are addressed by Health Physics procedures and the Emergency Plan and its Implementing Procedures.

11.2.5.7 Qualifications of Medical Personnel

Arrangements with local hospitals with qualified personnel to provide medical services for injured and/or contaminated personnel are included in the Emergency Plan and its Implementing Procedures.

Onsite Emergency Medical Technicians are certified by New York State. First Aid responders are certified by the American Red Cross, the American Heart Association or other certified First Aid / CPR training association. Health Physics technicians receive personnel decontamination training.

11.2.5.8 Transport of Injured Personnel

Arrangements for ambulance service to transport injured and/or contaminated personnel to local hospitals are included in the Emergency Plan and its Implementing Procedures.

11.2.5.9 Hospital Facilities

Arrangements with local hospitals with qualified personnel to provide medical services for injured and/or contaminated personnel are included in the Emergency Plan and its Implementing Procedures.

11.2.6 Evaluation of Radiation Protection

In the event of an accident involving a major release of core activity to the containment (e.g., the large break Loss-of-Coolant Accident with core degradation), the shielding provided by the containment protects the personnel in the control room from receiving excessive doses from the activity inside the containment. The dose to control room operators following the postulated large break LOCA includes the dose from the activity entering the control room, the direct dose from the cloud of activity outside the control room, and the direct dose from the radiation emanating from the containment. The control room doses are discussed in Section 14.3.6.5.

Liquid Waste Release

All liquid waste releases will be assayed for radioactivity to comply with the limits (one-tenth of 10 CFR 20) for unrestricted areas specified.

11.2.7 Tests and Inspections

Complete radiation surveys were made throughout the plant containment and auxiliary building during initial phases of plant startup. Survey data were taken and compared to design levels at power levels of 10-percent, 50-percent, and 100-percent of rated full power. Survey data were reviewed for conformance to design levels before increasing to the next power range.

The gas and particulate effluent monitors shall be tested at each refueling shutdown with calibrated sources and normal response of each monitor shall be tested daily using a remotely-operated test source to verify the instruments response. Liquid effluent monitors shall be tested at each refueling shutdown with calibrated sources and normal response of each monitor shall be tested daily using a remotely-operated test source to verify the instruments response.

11.2.8 Handling and Use of Sealed Special Nuclear, Source and By-Product Material

A. Tests for leakage and / or contamination shall be performed as follows:

1. Each sealed source, with a half-life greater than thirty days, shall be tested for leakage and / or contamination at intervals not to exceed six months (see 11.2.8.A.2 for testing of sealed sources that are stored and not being used).

[Note: Does not apply to startup sources subject to core flux, tritium, and material in gaseous form.]

2. Sealed sources that are stored and not being used shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Startup sources shall be leak tested prior to being subjected to core flux and following repair or maintenance to the source.

B. Sealed sources are exempt from 11.2.8.A when the source contains:

1. Less than or equal to 100 microcuries of beta and / or gamma emitting material, or
2. Less than or equal to 5 microcuries of alpha emitting material.

C. The leakage test shall be capable of detecting the presence of 0.005 microcuries of radioactive material on the test sample.

D. If the leakage test reveals the presence of 0.005 microcuries or more of removable contamination, the sealed source shall immediately be withdrawn from use and either decontaminated and repaired, or be disposed of in accordance with USNRC regulations.

E. If the leakage test reveals the presence of 0.005 microcuries or more of removable contamination, a special report shall be prepared and submitted to the Commission within 30 days.

REFERENCES FOR SECTION 11.2

1. Letter from W. Starostecki, NRC, to J. D. O'Toole, Con Edison, Subject: Inspection 50-247/83-14, dated July 5, 1983.

2. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Indian Point Unit 2 - NUREG 0737, Item II.B.2.2, Corrective Actions for Access to Vital Areas, dated October 26, 1983.

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18. Letter from R. C. DeYoung, NRC, to A. Hauspurg, Con Edison, Subject: Notice of Violation and Order Modifying License (NRC Inspection Nos.50-247/84-13 and 50-247/84-22), dated September 27, 1984.

19. Letter from J. D. O'Toole, Con Edison, to T. E. Murley, NRC, Subject: Response to Order Modifying License - Radiation Protection Plan Improvements, dated November 21, 1984.

20. Letter from J. D. O'Toole, Con Edison, to T. E. Murley, NRC, Subject: Revised Radiation Protection Oversight Committee Charter, dated February 14, 1985.

21. Letter from T. E. Murley, NRC, to J. D. O'Toole, Con Edison, Subject: Approval of Radiation Protection Action Plan, dated April 12, 1985.

22. Letter from M. Selman, Con Edison, to T. E. Murley, NRC, Subject: Plan for Maintaining Effectiveness of Radiation Protection Upgrade Programs, dated January 8, 1986.

23. Letter from T.E. Murley, NRC, to A. Hauspurg, Con Edison, Subject: Completion of Requirements of Order Modifying License, dated August 18, 1986.

24. Letter from S. A. Varga, NRC, to J. D. O'Toole, Con Edison, Subject: Completion of Review of NUREG-0737, Item II.E.4.2.6 and II.E.4.2.7 (with attached Safety Evaluation Report), dated November 9, 1982.

BIBLIOGRAPHY FOR SECTION 11.2

Comprehensive Public Water Supply Study for the New York City of New York and County of Westchester, Report CPWS-27, (submitted by Metcalf and Eddy, Hazen and Sawyer, and Malcolm Pirnie Engineers to the New York State Department of Health), August 1967.

TABLE 11.2-1
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TABLE 11.2-2
Primary Shield Neutron Fluxes and Design Parameters

Calculated Neutron Fluxes

Energy Group	Incident Fluxes (n/cm^2- sec)	Leakage Fluxes (n/cm^2 - sec)
E > 1 MeV	7.2×10^8	2.6×10^2
5.3 KeV $\leq$ E $\leq$ 1 MeV	1.0×10^{10}	5.9×10^2
.625 eV $\leq$ E $\leq$ 5.2 KeV	5.3×10^9	1.1×10^3
E < .625 eV	1.5×10^9	8.8×10^4

Design Parameters

Core thermal power	3216 MWt
Active core height, in.	144
Effective core diameter, in.	132.7
Baffle wall thickness, in.	1.125
Barrel wall thickness, in.	2.25
Thermal shield wall thickness, in.	2.75
Reactor vessel I.D., in	173.0
Reactor vessel wall thickness, in.	8.625
Reactor coolant cold-leg temperature	555°F
Reactor coolant hot-leg temperature	613°F
Maximum thermal neutron flux exiting primary concrete	10^6 n/cm^2-sec
Reactor shutdown dose exiting primary concrete	< 15 mrem/hr

TABLE 11.2-3
Secondary Shield Design Parameters

Core power density	98.5 w/cm^3
Reactor coolant liquid volume	12,600-ft^3
Reactor coolant transit times (sec):	
Core	0.817
Core exit to steam generator inlet	2.001
Steam generator inlet channel	0.592
Steam generator tubes	3.220
Steam generator tubes to vessel inlet	2.758
Vessel inlet to core	2.167
Total out of core	10.738
Full power dose rate outside secondary shield	<0.75 mrem/hr

TABLE 11.2-4
Accident Shield Design Parameters

Core thermal power	3216 Wt
Minimum full power operating time	1000 days
Equivalent fraction of core melting	1.0
Fission product fractional releases:	
Noble gases	1.0
Halogens	0.5
Remaining fission product inventory	0.01
Cleanup rate following accident	0
Maximum integrated direct dose (1-wk exposure) in control room	<1.5 rem
Maximum integrated direct dose (1-wk exposure) at the site boundary	<350 mrem

TABLE 11.2-5
Refueling Shield Design Parameters

Total number of fuel assemblies	193
Minimum full power exposure	1000 days
Minimum time between shutdown and fuel handling	56 hours
Maximum dose rate adjacent to spent fuel pit	0.75 mrem/hr
Maximum dose rate at water surface	2.0 mrem/hr

TABLE 11.2-6
Principal Auxiliary Shielding

Component	Concrete Shield Thickness
Demineralizers	4-ft - 0-in.
Charging pumps	2-ft - 6-in.
Liquid waste holdup tank	2-ft - 6-in.
Volume control tanks	3-ft - 6-in.
Reactor coolant filter	3-ft - 6-in.
Gas stripper	2-ft - 6-in.
Gas decay tanks	3-ft - 6-in.
Gas compressor	2-ft - 0-in.
Waste evaporator	2-ft - 0-in.
High radiation sampling system sentry panels	1-ft - 6-in.[1]
Motor control centers and support equipment	1-ft – 0-in.

Design parameters for the auxiliary shielding include:

Core thermal power	3216 MWt
Fraction of fuel rods containing small cladding defects	0.01
Reactor coolant liquid volume	12,600-ft^3
Letdown flow (normal purification)	75 gpm
Effective cesium purification flow	7 gpm
Cut-in concentration deborating demineralizer	150 ppm
Dose rate outside auxiliary building	0.75 mrem/hr
Dose rate in the building outside shield walls	0.75 mrem/hr

Notes:
1. This represents shielding minimum for the panels. The panels themselves contain 7 in. lead shot shielding sandwiched between two steel plates. The base of the panels (up to a height of 2-ft 9-in.) is also shielded by lead shot shielding sandwiched between two steel plates.

TABLE 11.2-7 (Sheet 1 of 3)
Radiation Monitoring Channel Data

Effluent Monitors

Channel	Stream Monitored	Normal Maximum Channel Output
R-27*	High Range, Noble Gas	1.0×10^5 uCi/cc
R-39*	Service Water from	1.0×10^7 CPM
R-40*	Component Cooling Heat Exchangers	1.0×10^7 CPM
R-43*	Plant Vent Air Particulate	1.0×10^7 CPM
	Plant Vent Air Iodine	1.0×10^7 CPM
R-44	Plant Vent Air Gaseous	1.0×10^7 CPM
R-45	Condenser Air Ejector Discharge	1.0×10^5 uCi/cc
R-46	Service Water Returns from	1.0×10^7 CPM
R-53	Containment Fan Cooler Units	1.0×10^7 CPM
R-48	Waste Condensate Tank Discharge Line	1.0×10^7 CPM
R-49	Steam Generator Blowdown	1.0×10^7 CPM
R-50	Waste Gas Decay Tanks	5.0×10^4 uCi/cc
R-51	Secondary Boiler Blowdown Purification System	1.0×10^7 CPM
R-52*	Secondary Boiler Blowdown Purification System Cooling Water	1.0×10^7 CPM
R-54	Liquid Waste Distillate	1.0×10^7 CPM
R-55A*	Steam Generator Blowdown Secondary System	1.0×10^7 CPM
R-55B*		1.0×10^7 CPM
R-55C*		1.0×10^7 CPM
R-55D*		1.0×10^7 CPM
R-57	Sewage Effluent Discharge	1.0×10^7 CPM
R-60*	Stack Air Gaseous	1.0×10^7 CPM
	Stack Air Particulate*	1.0×10^7 CPM
	Stack Air Iodine*	1.0×10^7 CPM
R-62	Unit 1 Sphere Foundation Sump	1.0×10^7 CPM
R-5976*	Maintenance & Outage Building Gaseous	1.0×10^7 CPM
	Maintenance & Outage Building Particulate	1.0×10^7 CPM

TABLE 11.2-7 (Sheet 2 of 3)
Radiation Monitoring Channel Data

Process Monitors

Channel	Stream Monitored	Normal Maximum Channel Output
R-41	Containment Air Particulate	1.0×10^{7} CPM
R-42	Containment Air Gaseous	1.0×10^{7} CPM
R-47	Component Cooling Water	1.0×10^{7} CPM
R-59	House Service Boiler Condensate	1.0×10^{7} CPM
R-28*	Main Steam Line High Radiation	1.0×10^{6} CPM
R-29*		1.0×10^{6} CPM
R-30*		1.0×10^{6} CPM
R-31*		1.0×10^{6} CPM
R-38-1	Control Room Air Intake	1.0×10^{3} mR/hr
R-38-2		1.0×10^{3} mR/hr
R-61A*	Main Steam Line, N-16	1.0×10^{4} CPM
R-61B*		1.0×10^{4} CPM
R-61C*		1.0×10^{4} CPM
R-61D*		1.0×10^{4} CPM

TABLE 11.2-7 (Sheet 3 of 3)
Radiation Monitoring Channel Data

Area Monitors

Channel	Stream Monitored	Normal Maximum Channel Output
ARM-1	Unit 1 Drum Storage	1.0×10^{3} mR/hr
ARM-2	Unit 1 Pedestrian Tunnel	1.0×10^{3} mR/hr
ARM-3	Unit 1 Nuclear Services Building Valve Room	1.0×10^{3} mR/hr
ARM-4	Unit 1 Evaporator Bottom Room	1.0×10^{3} mR/hr
ARM-5	Unit 1 Fuel Handling Floor	1.0×10^{3} mR/hr
R-1	Control Room	1.0×10^{4} mR/hr
R-2	Containment by Personnel Hatch	1.0×10^{4} mR/hr
R-4	Charging Pump Room/PAB Area	1.0×10^{4} mR/hr
R-5	Spent Fuel Building	1.0×10^{4} mR/hr
R-6	Sample Room	1.0×10^{4} mR/hr
R-7	Incore Instrument Area in Containment	1.0×10^{4} mR/hr
R-25*	Containment High Range Radiation	1.0×10^{10} mR/hr
R-26*		1.0×10^{10} mR/hr
R-37-1	Post Accident Sampling System	1.0×10^{7} mR/hr
R-37-2		1.0×10^{7} mR/hr
R-37-3		1.0×10^{7} mR/hr
R-5987	PAB Breaker Service Area	1.0×10^{4} mR/hr

Note:

* This listing does not apply the requirements of the Technical Specifications, Technical Requirements Manual, or Offsite Dose Calculation Manual (ODCM) to any radiation monitor that was not installed as a result of an NRC requirement, but was installed as an enhancement or as a means of providing additional information to plant personnel, such as the R-61A through R-61D radiation monitors.

Radiation monitors listed as Effluent Radiation Monitors in UFSAR Table 11.2-7 and not specifically listed in Technical Requirements Manual Table 3.3.G-1, ODCM Table D 3.3.1-1, ODCM Table D 3.3.2-1, or Unit 1 Technical Specifications Section 5.2.5 will continue to maintain surveillance requirements imposed by ODCM Table D 3.3.1-1 or ODCM Table D 3.3.2-1 for daily, monthly, quarterly and refueling frequencies.

TABLE 11.2-7a
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TABLES 11.2-8 through 11.2-13
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11.2 FIGURES

Figure No.	Title
Figure 11.2-1	Deleted
Figure 11.2-2	Deleted
Figure 11.2-3	Deleted
Figure 11.2-4	Deleted
Figure 11.2-5	Deleted
Figure 11.2-6	Deleted

Appendix 11A
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Appendix 11B
DETERMINATION OF RIVER WATER DILUTION FACTORS BETWEEN THE INDIAN POINT SITE AND THE NEAREST PUBLIC DRINKING WATER INTAKES

LIST OF TABLES

LIST OF FIGURES

Appendix 11B
DETERMINATION OF RIVER WATER DILUTION FACTORS BETWEEN THE INDIAN POINT SITE AND THE NEAREST PUBLIC DRINKING WATER INTAKES

The analytical techniques used to analyze the dispersion of continuous and burst releases of liquids are discussed in detail in "Transport of Contaminants in the Hudson River above Indian Point Station," which is referenced in Section 2.5.

There are two potential sources of drinking water in the Hudson River, namely, New York City's Chelsea Pumping Station and the Castle Point Veteran's Hospital. The city of New York's Chelsea Pumping Station is located about 1 mile north of Chelsea, New York, on the east bank of the Hudson River. The pumping station is 22 miles upriver from Indian Point measured along the centerline of the river. The Castle Point Veteran's Hospital is a relatively small intake located approximately 21 miles upriver from the proposed site.

Analyses have been conducted to determine the difference in concentration at Chelsea and Castle Point Veteran's Hospital. The difference in concentration is small; hence, the discussion

of the potential intake, namely, Chelsea, is sufficient. (See Reference 3 of Section 2.5 for continuous and burst releases.)

The River drought conditions analyzed have been characterized in terms of salinity because the operation of the Chelsea Station is dependent on the level of salt at the station. Consider the following five drought conditions, i.e., salinities at Chelsea:

Salt Concentration in ppm		Runoff (cfs)	Dispersion Coefficient (Square miles/day)
At Chelsea	At Indian Point		
200	2300	5000	5.24
300	2800	4600	5.28
500	4000	4400	5.43
1000	5500	4000	6.00
2000	7000	3500	7.16

The first two drought conditions correspond to concentrations of salinity at Chelsea, at which the New York City Department of Water Resources would begin to be concerned about using Chelsea for New York City's water supply.

The third condition, a salinity of 500 ppm, corresponds to the "midthousand" level, which might constitute the maximum level at which Chelsea operation would be stopped. This also corresponds to the Public Health Service drinking water standard for total dissolved solids.

The fourth condition, a salinity of 1000 ppm, represents the maximum level at which Chelsea operation would be stopped.

The fifth condition, a salinity of 2000 ppm, corresponds to the highest levels of salinity known to have occurred at Chelsea and represents the most conservative river conditions used in this analysis. This concentration of salinity at Chelsea was reached in late November 1964 at the end of 6 months of Hudson River low flows. Support that the 1964 drought was the worst on record after regulation of the Hudson River is given in a recent report concerning the potential of the Hudson River supplementing New York City's water supply system.*

[Note:
**"Comprehensive Public Water Supply Study for the New York City of New York and County of Westchester" - Report CPWS-27 submitted by Metcalf and Eddy, Hazen and Sawyer, and Malcolm Pirnie Engineers to the New York State Department of Health, August 1967.]*

The upstream movement of salt is the result of a rather delicate balance, which is struck between the salinity-induced density currents, which tend to drive the salt itself up the estuary, and fresh water flow, which tends to hold back the salt movement. The river's dispersion characteristics are strongly influenced by this phenomenon, so that salinity profiles become the chief means of estimating the longitudinal dispersion coefficient in the river.

Calculation of dispersion coefficients requires a knowledge of the salinity changes between two fixed points and the river's flow. The essential point, however, is that the behavior of a conservative substance is identical to the salt behavior, which is well-defined; hence, the salinity at Chelsea is an excellent indicator of the upstream movement of any pollutant introduced to the river below the station. This is explained as follows:

1. If salt is not present at Chelsea, then neither will any other pollutant, discharged many miles below Chelsea, be present at Chelsea.

2. When salt is present at Chelsea, the ratio between the salt concentrations at Indian Point and Chelsea is a measure of the "mechanical dilution," i.e., dilution due to the river's flow and dispersion characteristics for non-decaying pollutants.

Hence, for the five drought conditions cited above, the mechanical dilution factors between Indian Point Station and Chelsea may be obtained directly from the ratio of salinity at these two points and are as follows:

Runoff (cfs)	Mechanical Dilution
5000	11.5
4600	9.4
4400	8.0
4000	5.5
3500	3.5

To obtain the concentrations of decaying radionuclides at Chelsea, simple ratios of the salt concentrations at Indian Point and Chelsea are not used. Rather, a material balance on each isotope is struck over any segment of the river by considering the transport mechanisms of net flow and longitudinal dispersion, and the radioactive decay mechanism. The longitudinal dispersion coefficient is obtained from salt profiles. The approach is described in the reference cited above in Section 2.5.

To show how the significant parameters, namely, the salinity and the half-life affect the river's ability to reduce concentration of introduced pollutants, a study was made assuming a normalized continuous release rate for each isotope of 1 Ci/day and a normalized burst release for each isotope of 1 Ci. Since the concentrations at Chelsea are directly proportional to the source term, the normalized curves can be used to determine quickly the concentration at Chelsea due to a known burst or continuous release from Indian Point, or to determine dilution factors.

Continuous Release

A hypothetical case where primary coolant with 1-percent failed fuel being released directly to the discharge canal was considered so that the behavior of all isotopes of possible concern in the river could be presented. The activity is released at a constant rate, the value of which is set so that the MPC of the mix will not be exceeded in the discharge water. The most severe drought conditions have been utilized; for the continuous release, these consist of a long-term steady upstream runoff of 3500 cfs, which causes the salt concentration at Chelsea to reach 2000 ppm.

Other pertinent river parameters used in the analysis are as follows:

1. Longitudinal dispersion coefficient, "E" = 7.16 mi^2/day
2. Average cross-sectional area, "A" = 140,000-ft^2

The results of this analysis are presented in Table 11B-1 and the computational procedure follows:

1. Column 1 - Unit 3 PSAR, Column 2, Part B, Table 16 (E-3.1).
2. Column 2 - 0.693 divided by half-life in days.
3. Column 3 - allowable release rate based on MPC of mix in discharge canal.
4. Column 4 through 7 - computation procedure for continuos release, QL and M report to Con Edison on Chelsea concentrations (May 1966), and included in both Units 2 and 3 submittals. (Analyses appended to Section 2.5.)
5. Column 8 - concentration at Chelsea divided by concentration at Indian Point.

The minimum dilution factors for all isotopes of concern are given in column 8 of Table 11B-1.

For the effect of all three units at Indian Point releasing radioactivity to the river under the conditions described above, the corresponding Chelsea and Indian Point concentrations can be computed by multiplying the concentrations in these tables by 1,960,000/840,000 or 2.34, the ratio of the total condenser flow to the Units 2 or 3 condenser flow. This assumes that the mix distribution from each unit is the same.
Burst Release

The results of the normalized burst release studies are presented in Figures 11B-1 through 11B-5. They are based on a 1 Ci burst release of each isotope. The following conclusions can be reached from these Figures.

1. Referring to Figure 11B-1, the peak concentrations at Chelsea and Castle Point are for the purpose of this discussion essentially the same.

2. Referring to Figure 11B-2, variations in drought conditions, i.e., changes in low runoff values do not appreciably affect the peak concentrations at Chelsea.

3. Referring to Figure 11B-5, the runoff does not appreciably affect the time for an isotope to reach a peak concentration at Chelsea; the time to the peak is a weak function of half-life for isotopes with half-lives less than 100 days, and the time to the peak is not sensitive to half-life for isotopes with half-lives greater than 100 days.

4. Referring to Figures 11B-3 and 11B-4, short-lived (less than 1 day) isotopes will not reach Chelsea; peak concentrations of intermediate isotopes (1 day to 100 days) are strongly dependent on the half life.

The river dilution factor between Indian Point and Chelsea for the burst release is a nonapplicable concept. When the maximum radioactivity effect of each isotope occurs at Chelsea, the corresponding concentration of that isotope at Indian Point will be very low. Furthermore, Chelsea will not see the maximum concentration of each isotope at the same time. For these reasons, for the burst release, the concentration in the Hudson River is considered for Indian Point one-half day after the release and at Chelsea at the time when the concentration of the given isotope is maximum at that point. Zero time cannot be used at Indian Point because the equations used will yield infinity for the concentration at $x = 0$, $t = 0$. One-half day later was used because this corresponds to one tidal cycle, the minimum time necessary to provide the river mixing, which these equations presume.

Based on the above definition of dilution factor for the burst release, the minimum dilution factors for the burst release were determined for the drought condition resulting in 2000 ppm of salt at Chelsea. The hypothetical case where the entire primary coolant with fission product inventory due to operation with 1-percent failed fuel was dumped into the river was used to arrive at the dilution factors for all isotopes of concern. The results of this analysis are given in Table 11B-2 and the computational procedure is as follows:

1. Columns 1 and 2 - Taken from Table 9.2-5 (Unit No. 3 PSAR), entitled "Reactor Coolant System Equilibrium Activities," and computed using a primary coolant volume of 3.56×10^{8} ml. Tritium activity of 890 Ci added later.

2. Columns 3 through 7 - Computation procedure for accidental release, QL and M report to Con Edison on Chelsea, May 1966, and included in Units 2 and 3 submittals (as appended to Section 2.5).

3. Column 8 - Based on burst release dilution factor definition cited above.

TABLE 11B-1 (Sheet 1 of 2)
Concentrations of Primary Coolant Isotopes in the
Hudson River at Indian Point and Chelsea

Hypothetical Continuous Release, One Percent Failed Fuel

MPC in Discharge Canal

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)
			Behavior At				
			Indian Point		Chelsea		
Isotope	Decay Rate (day^{-1})	Discharge Rate (µCi/day)	Concentration (µCi/ml)	Fraction of MPC	Concentration (µCi/ml)	Fraction of MPC	River Dilution Between Indian Point - Chelsea
Mn-54	$2.3x10^{-3}$	$1.54 x10^{2}$	$15.25x10^{-12}$	$1.5x10^{-7}$	$3.99x10^{-12}$	$3.99x10^{-8}$	3.82
Mn-56	6.3	$3.33x10^{4}$	$118.5x10^{-12}$	$1.2x10^{-6}$	$5.5x10^{-20}$	$5.5x10^{-16}$	$2.16x10^{9}$
Co-58	$0.97x10^{-2}$	$4.62x10^{3}$	$332x10^{-12}$	$3.3x10^{-6}$	$6.35x10^{-11}$	$5.35x10^{-7}$	5.22
Fe-59	$1.5x10^{-2}$	$1.07x10^{2}$	$6.77x10^{-12}$	$1.1x10^{-7}$	$1.05x10^{-12}$	$1.75x10^{-8}$	6.45
Co-69	$3.6x10^{-4}$	$5.45x10^{2}$	$61.8x10^{-12}$	$1.2x10^{-6}$	$1.73x10^{-11}$	$3.45x10^{-7}$	3.58
Br-84	$3.15x10^{-3}$	$1.63x10^{4}$	$1530x10^{-12}$	-	-	-	-
Rb-88	$5.6x10^{-3}$	$1.54x10^{4}$	$1.28x10^{-7}$	-	-	-	-
Rb-89	$6.48x10^{-3}$	$3.56x10^{4}$	$2870x10^{-12}$	-	-	-	-
Sr-89	$1.37x10^{-2}$	$1.20x10^{3}$	$76.4x10^{-12}$	$2.5x10^{-5}$	$1.25x10^{-11}$	$4.28x10^{-6}$	6.11
Sr-90	$0.69x10^{-4}$	$0.81x10^{2}$	$9.35x10^{-12}$	$3.1x10^{-5}$	$2.68x10^{-12}$	$8.92x10^{-6}$	3.49
Y-90	$2.6x10^{-4}$	$1.66x10^{2}$	$2.88x10^{-12}$	$1.4x10^{-7}$	$2.24X10^{-14}$	$1.12X10^{-9}$	352
Sr-91	1.73	$7.82x10^{2}$	$5.32x10^{-12}$	$0.8x10^{-7}$	$6.1x10^{-17}$	$8.70x10^{-13}$	$8.72x10^{4}$
Y-91	$1.2x10^{-2}$	$3.56x10^{2}$	$23.9x10^{-12}$	$8x10^{-7}$	$4.27x10^{-12}$	$1.34x10^{-7}$	5.60
Mo-99	$2.5x10^{-1}$	$1.96x10^{6}$	$3.47x10^{-8}$	$1.7x10^{-4}$	$2.84x10^{-10}$	$1.42x10^{-6}$	122

TABLE 11B-1 (Sheet 2 of 2)
Concentrations of Primary Coolant Isotopes in the
Hudson River at Indian Point and Chelsea

Hypothetical Continuous Release, One Percent Failed Fuel

MPC in Discharge Canal

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)
			Behavior At				
			Indian Point		Chelsea		
Isotope	Decay Rate (day^{-1})	Discharge Rate (μCi/day)	Concentration (μCi/ml)	Fraction of MPC	Concentration (μCi/ml)	Fraction of MPC	River Dilution Between Indian Point - Chelsea
I-131	8.62×10^{-2}	1.04×10^{6}	3.07×10^{-8}	1×10^{-1}	1.35×10^{-9}	4.5×10^{-3}	22.7
Te-132	0.9×10^{-2}	1.10×10^{5}	8.08×10^{-9}	2.7×10^{-4}	2.38×10^{-12}	7.94×10^{-7}	3400
I-132	7.2	3.56×10^{5}	1.18×10^{-9}	1.5×10^{-4}	1.63×10^{-19}	2.03×10^{-14}	7.25×10^{9}
I-133	0.81	8.05×10^{5}	7.97×10^{-9}	8×10^{-3}	2.82×10^{-12}	2.82×10^{-6}	2830
Te-134	23	1.16×10^{4}	21.6×10^{-12}	-	-	-	-
I-134	19	2.12×10^{5}	4.34×10^{-10}	2.2×10^{-5}	7.70×10^{-26}	3.85×10^{-21}	5.64×10^{15}
Cs-134	0.93×10^{-3}	1.36×10^{5}	1.47×10^{-8}	1.6×10^{-3}	4.01×10^{-9}	4.46×10^{-4}	3.67
I-135	2.39	8.05×10^{5}	4.58×10^{-9}	1.1×10^{-3}	5.88×10^{-15}	1.47×10^{-9}	7.8×10^{5}
Cs-136	5.14×10^{-2}	1.32×10^{4}	4.95×10^{-10}	6×10^{-6}	3.49×10^{-11}	3.88×10^{-7}	14.2
Cs-137	6.3×10^{-4}	5.76×10^{5}	6.34×10^{-8}	3.2×10^{-3}	1.91×10^{-8}	9.55×10^{-4}	3.32
Cs-138	32	2.62×10^{4}	41.8×10^{-12}	-	-	-	-
Ba-140	5.4×10^{-2}	3.56×10^{2}	12.1×10^{-12}	4×10^{-7}	9.09×10^{-13}	3.03×10^{-8}	13.3
La-140	0.415	3.70×10^{2}	$5.1 \mathrm{X} 10^{-12}$	$2.5 \mathrm{X} 10^{-7}$	$1.33 \mathrm{X} 10^{-14}$	$6.65 \mathrm{X} 10^{-10}$	384
Ce-144	2.44×10^{-3}	1.25×10^{3}	122.5×10^{-12}	1.2×10^{-5}	3.05×10^{-11}	3.05×10^{-6}	4.02
Pr-144	5.13×10^{-2}	1.37×10^{6}	5.13×10^{-8}	-	-	-	-
Tritium		1.49×10^{6}	1.74×10^{-7}	5.8×10^{-5}	4.75×10^{-8}	1.59×10^{-5}	3.66
	Total	9.15×10^{6}					

TABLE 11B-2 (Sheet 1 of 2)
Concentrations of Radioisotopes the Hudson River at Indian Point and Chelsea

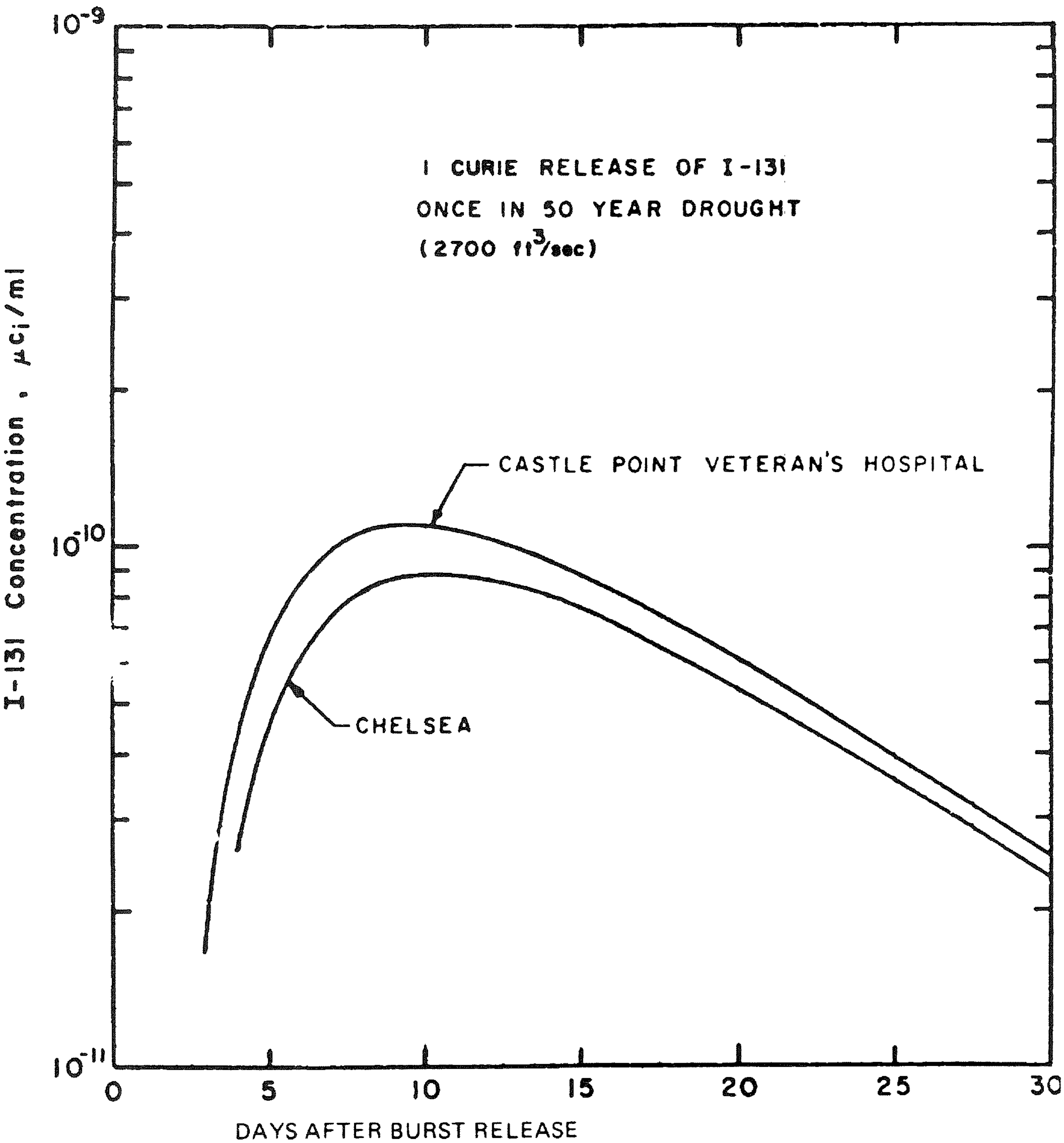
Accidental Loss of Entire Primary Coolant (One Percent Failed Fuel) in a Burst Release

(1) Isotope	(2) Equilibrium Activity in the Primary Coolant (Ci)	(3) River Concentrations at Indian Point One-Half Day After Release µCi/ml	(4) Fractions of MPC	(5) Time for Maximum Concentrations to Reach Chelsea (days)	(6) Maximum River Concentrations at Chelsea µCi/ml	(7) Fractions of MPC	(8) River Dilution Between Indian Point - Chelsea
Mn-54	0.092	$5.83x10^{-9}$	$5.83x10^{-5}$	20.4	$2.22x10^{-11}$	$2x10^{-7}$	$2.9x10^{2}$
Mn-56	19.9	$2.26x10^{-10}$	$2.26x10^{-6}$	1.4	$2.68x10^{-16}$	$3x10^{-12}$	$7.5x10^{5}$
Co-58	2.78	$1.76x10^{-10}$	$1.76x10^{-6}$	17.6	$4.75x10^{-12}$	$5x10^{-8}$	$3.7x10^{1}$
Fe-59	0.064	$4.05x10^{-10}$	$6.75x10^{-6}$	16.2	$1.21x10^{-12}$	$2x10^{-7}$	$3.4x10^{1}$
Co-60	0.29	$1.84x10^{-9}$	$3.68x10^{-5}$	21.4	$8.18x10^{-11}$	$2x10^{-6}$	$1.8x10^{1}$
Br-84	9.65	$6.1x10^{-8}$	-	0.7	$2.87x10^{-26}$	-	$2.1x10^{18}$
Rb-88	920	$5.81x10^{-6}$	-	0.5	$2.3x10^{-30}$	-	$2.5x10^{24}$
RB-89	1.95	$2.39x10^{-8}$	-	0.5	$3.89x10^{-34}$	-	$6.2x10^{25}$
Sr-89	0.91	$5.73x10^{-9}$	$1.91x10^{-3}$	16.5	$1.94x10^{-10}$	$6x10^{-5}$	$6.2x10^{1}$
Sr-90	0.049	$3.1x10^{-10}$	$1.0x10^{-3}$	21.6	$1.2x10^{-11}$	$4x10^{-5}$	$2.5x10^{1}$
Y-90	0.099	$4.84x10^{-10}$	$2.42x10^{-4}$	6.3	$2.11x10^{-12}$	$1x10^{-7}$	$2.4x10^{2}$
Sr-91	0.469	$1.25x10^{-9}$	$1.79x10^{-5}$	2.7	$4.25x10^{-14}$	$6x10^{-10}$	$3x10^{4}$
Y-91	19.9	$1.20x10^{-7}$	$4.0x10^{-3}$	17.0	$4.01x10^{-9}$	$1x10^{-4}$	$4x10^{1}$
Mo-99	1170	$6.56x10^{-6}$	$3.28x10^{-2}$	6.4	$2.61x10^{-8}$	$1x10^{-4}$	$3.3x10^{2}$

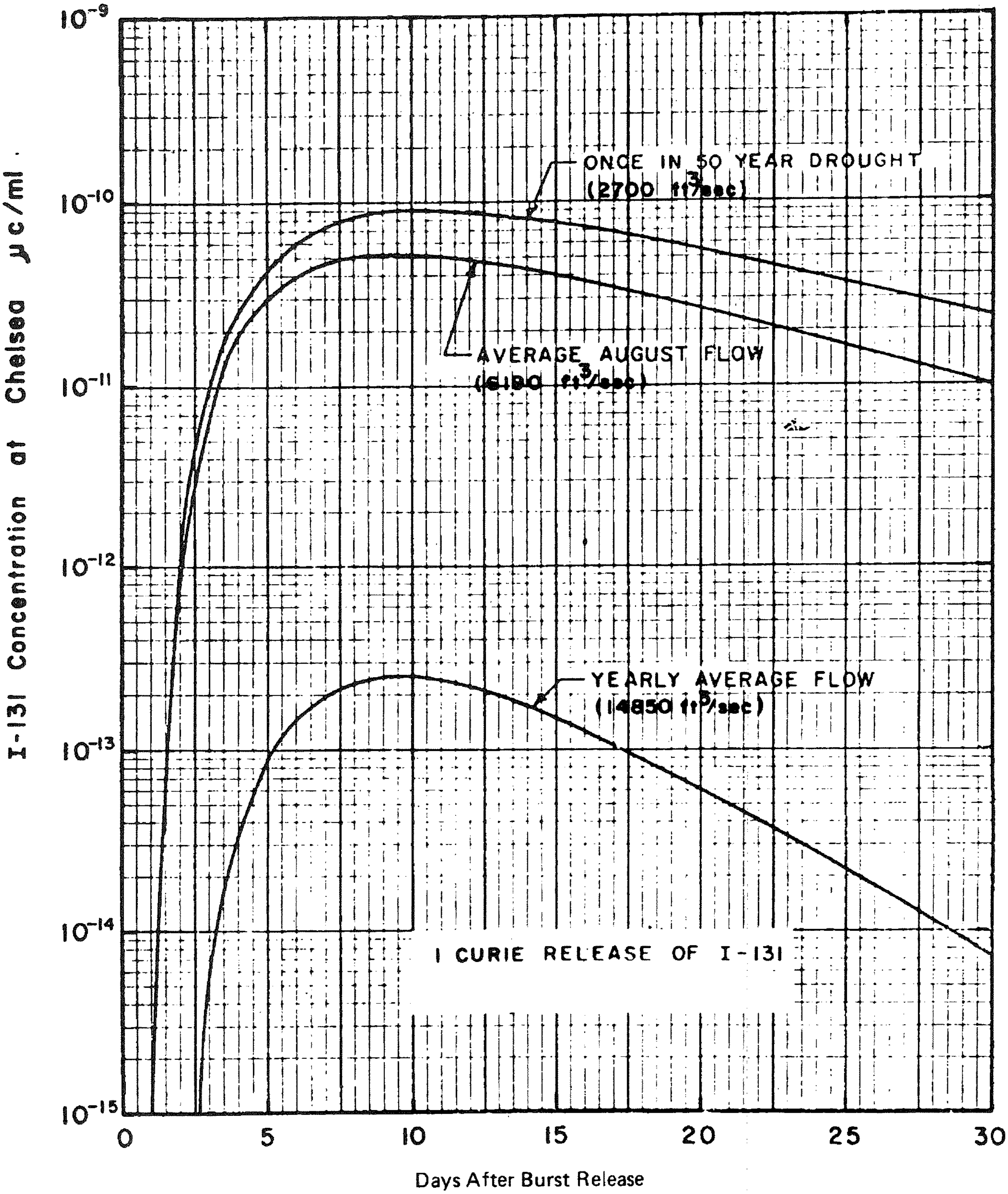
TABLE 11B-2 (Sheet 2 of 2)
Concentrations of Radioisotopes in the Hudson River at Indian Point and Chelsea

Accidental Loss of Entire Primary Coolant (One Percent Failed Fuel) in a Burst Release

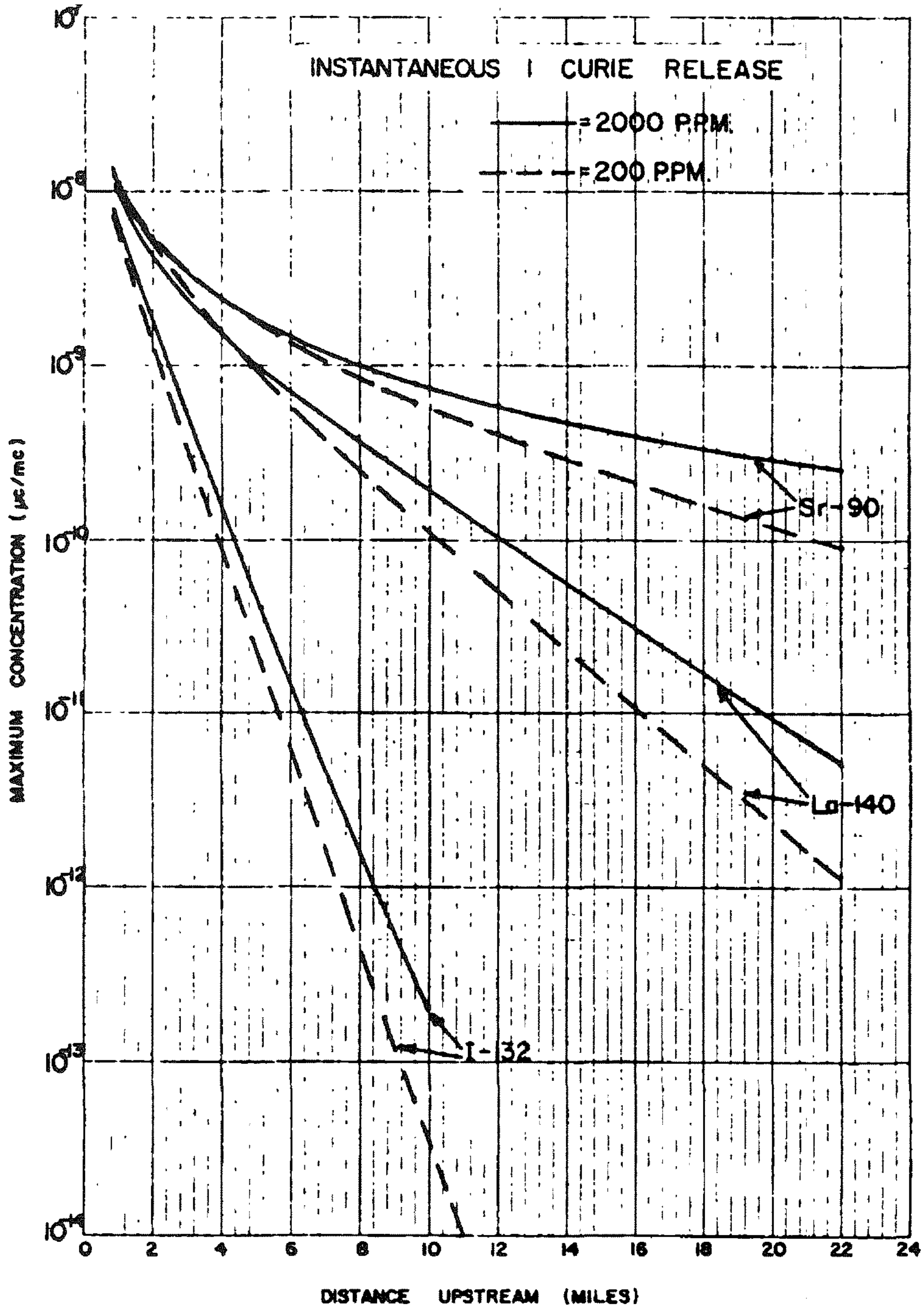
(1) Isotope	(2) Equilibrium Activity in the Primary Coolant (Ci)	(3) River Concentrations at Indian Point One-Half Day After Release: μCi/ml	(4) River Concentrations at Indian Point One-Half Day After Release: Fractions of MPC	(5) Time for Maximum Concentrations to Reach Chelsea (days)	(6) Maximum River Concentrations at Chelsea: μCi/ml	(7) Maximum River Concentrations at Chelsea: of MPC	(8) River Dilution Between Indian Point - Chelsea
I-131	622	3.8×10^{-6}	12.2	9.8	4.99×10^{-8}	1.7×10^{-1}	7.2×10^{1}
Te-132	65.7	4.14×10^{-7}	1.88×10^{-2}	18+	1.3×10^{-8}	4×10^{-4}	3.5×10^{1}
I-132	195	3.35×10^{-8}	4.18×10^{-3}	1.3	9.7×10^{-16}	1×10^{-10}	4.2×10^{7}
I-133	485	2.06×10^{-6}	2.06	3.8	8.03×10^{-10}	8×10^{-4}	2.6×10^{3}
Te-134	6.94	6.73×10^{-13}	-	0.8	5.6×10^{-24}	-	1.2×10^{11}
I-134	127	6.04×10^{-11}	3.02×10^{-6}	0.8	3.45×10^{-21}	2×10^{-16}	1.5×10^{10}
Cs-134	81.5	5.17×10^{-7}	574	21.1	2.01×10^{-8}	2.2×10^{-3}	2.6×10^{6}
I-135	485	9.3×10^{-7}	2.3×10^{-1}	2.2	6.62×10^{-12}	1.6×10^{-6}	1.4×10^{5}
Cs-136	7.9	5.0×10^{-8}	5.55×10^{-4}	11.5	8.98×10^{-10}	1×10^{-5}	5.6×10^{1}
Cs-137	348	2.20×10^{-6}	1.10×10^{-1}	21.6	8.73×10^{-8}	4.4×10^{-3}	2.5×10^{1}
Cs-138	15.7	1.09×10^{-15}	-	0.7	5.23×10^{-26}	-	2.1×10^{10}
Ba-140	0.212	1.35×10^{-9}	4.50×10^{-5}	11.5	2.3×10^{-11}	8×10^{-8}	5.6×10^{2}
La-140	0.22	1.15×10^{-9}	5.75×10^{-5}	5.2	1.95×10^{-12}	1×10^{-7}	5.8×10^{2}
Ce-144	0.075	4.74×10^{-10}	4.75×10^{-5}	20.3	1.78×10^{-11}	2×10^{-7}	2.4×10^{2}
Pr-144	0.082	5.19×10^{-10}	-	11.7	9.65×10^{-12}	-	5.4×10^{1}
Tritium	890	5.36×10^{-6}	1.79×10^{-3}	21.8	2.22×10^{-7}	8×10^{-4}	2.2×10^{0}



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 11B-1 IODINE-131 CONCENTRATION VS DAYS AFTER BURST RELEASE FROM INDIAN POINT FOR 1 CURIE RELEASE	
MIC. No. 1999MC3945	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 11B-2 IODINE-131 CONCENTRATION AT CHELSEA vs DAYS AFTER BURST RELEASE FROM INDIAN POINT FOR 1 CURIE RELEASE	
MIC. No. 1999MC3946	REV. No. 17A

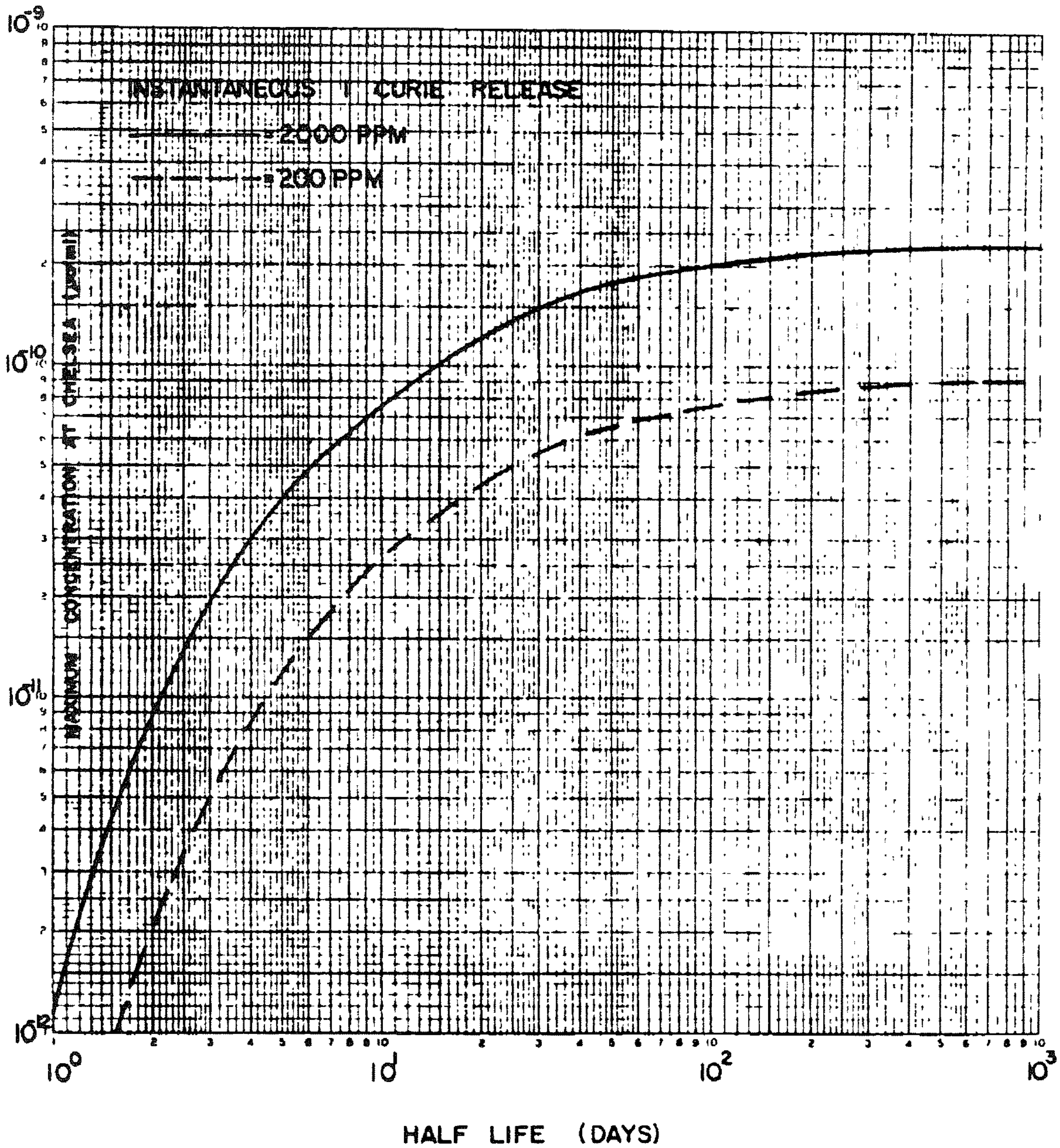


INDIAN POINT UNIT No. 2

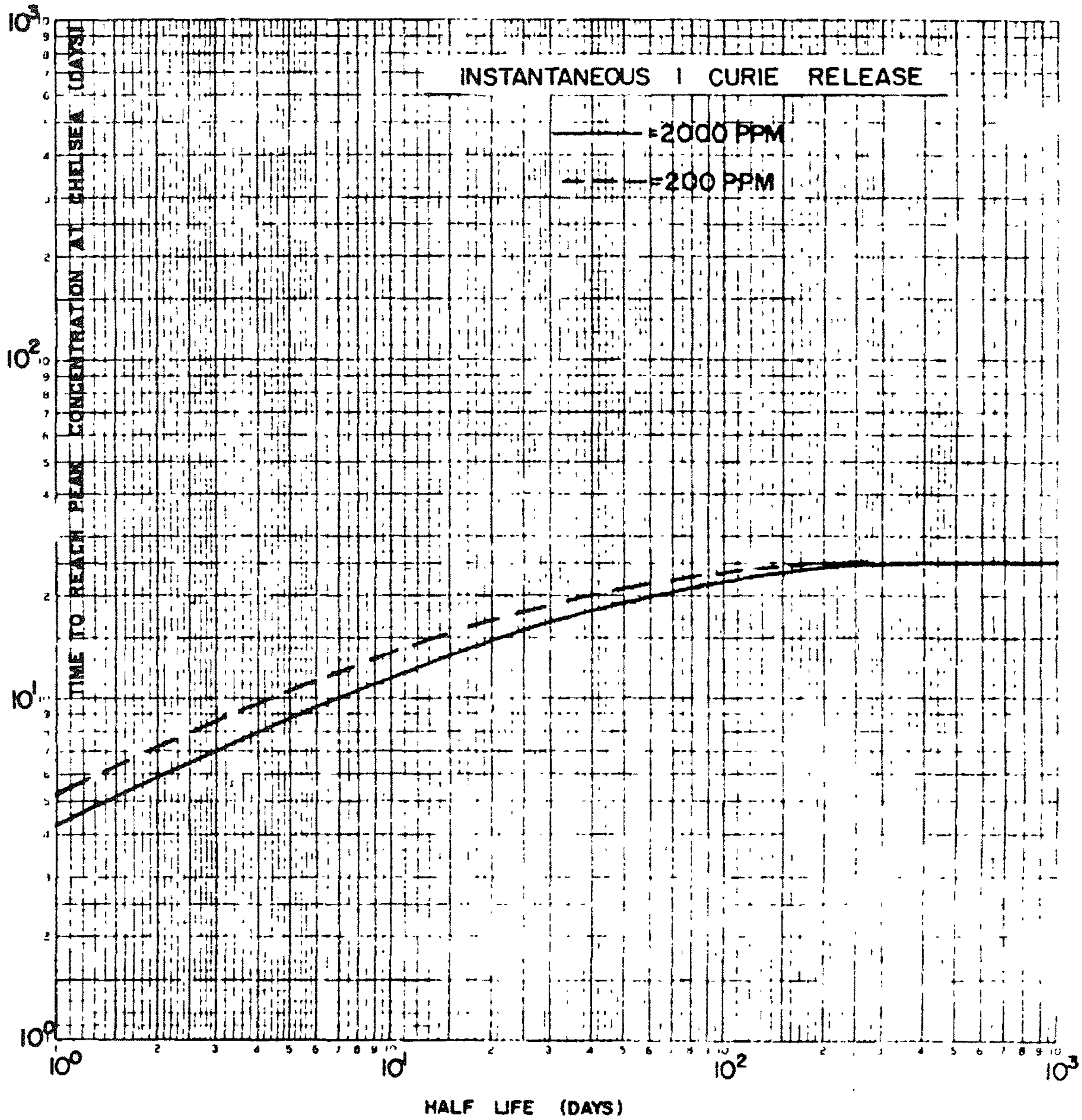
UFSAR FIGURE 11B–3

MAXIMUM CONCENTRATION vs DISTANCE UPSTREAM FOR 1 CURIE RELEASE

MIC. No. 1999MC3947	REV. No. 17A



INDIAN POINT UNIT No. 2	
UFSAR FIGURE 11B–4 MAXIMUM CONCENTRATION AT CHELSEA vs HALF–LIFE FOR 1 CURIE RELEASE	
MIC. No. 1999MC3948	REV. No. 17A

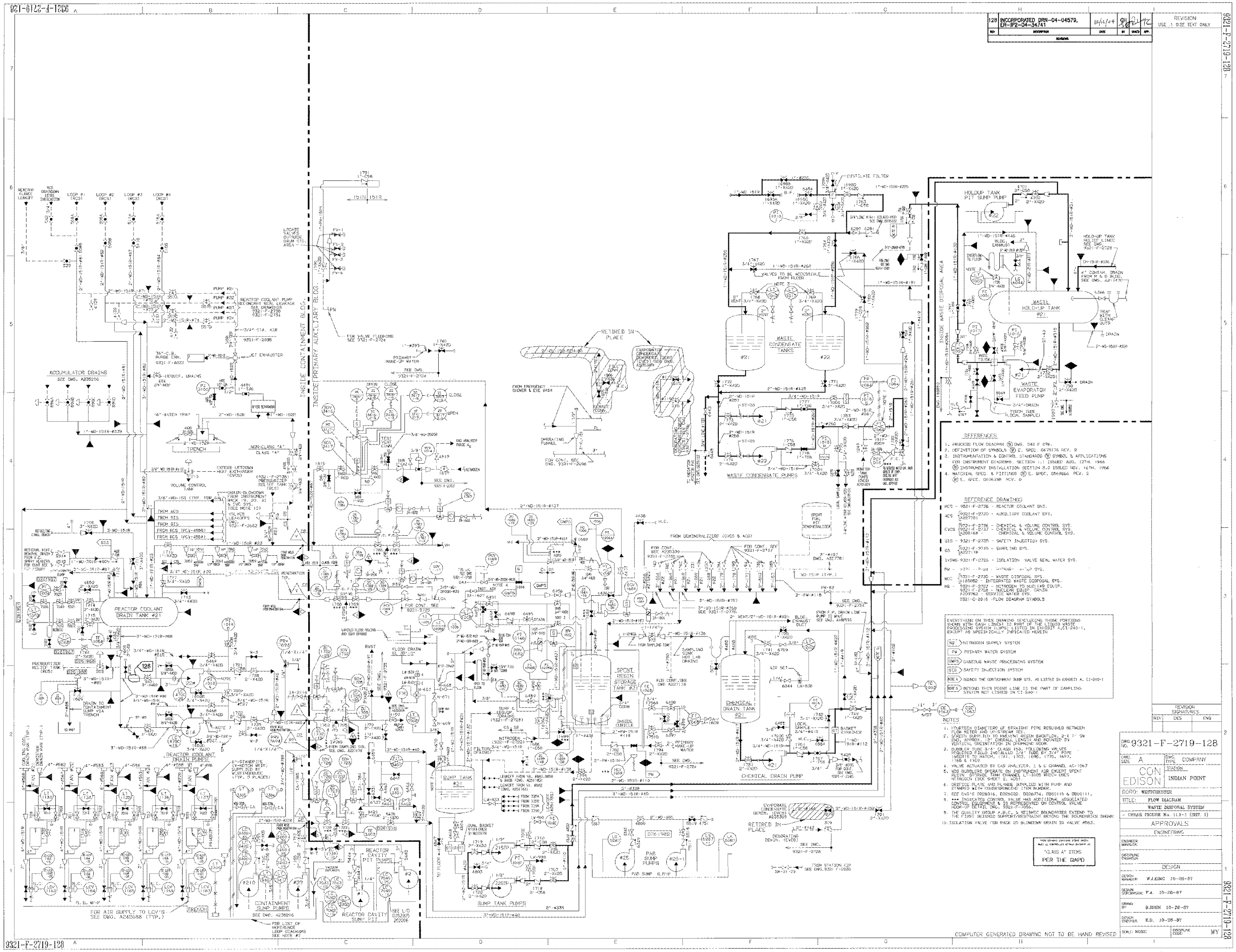


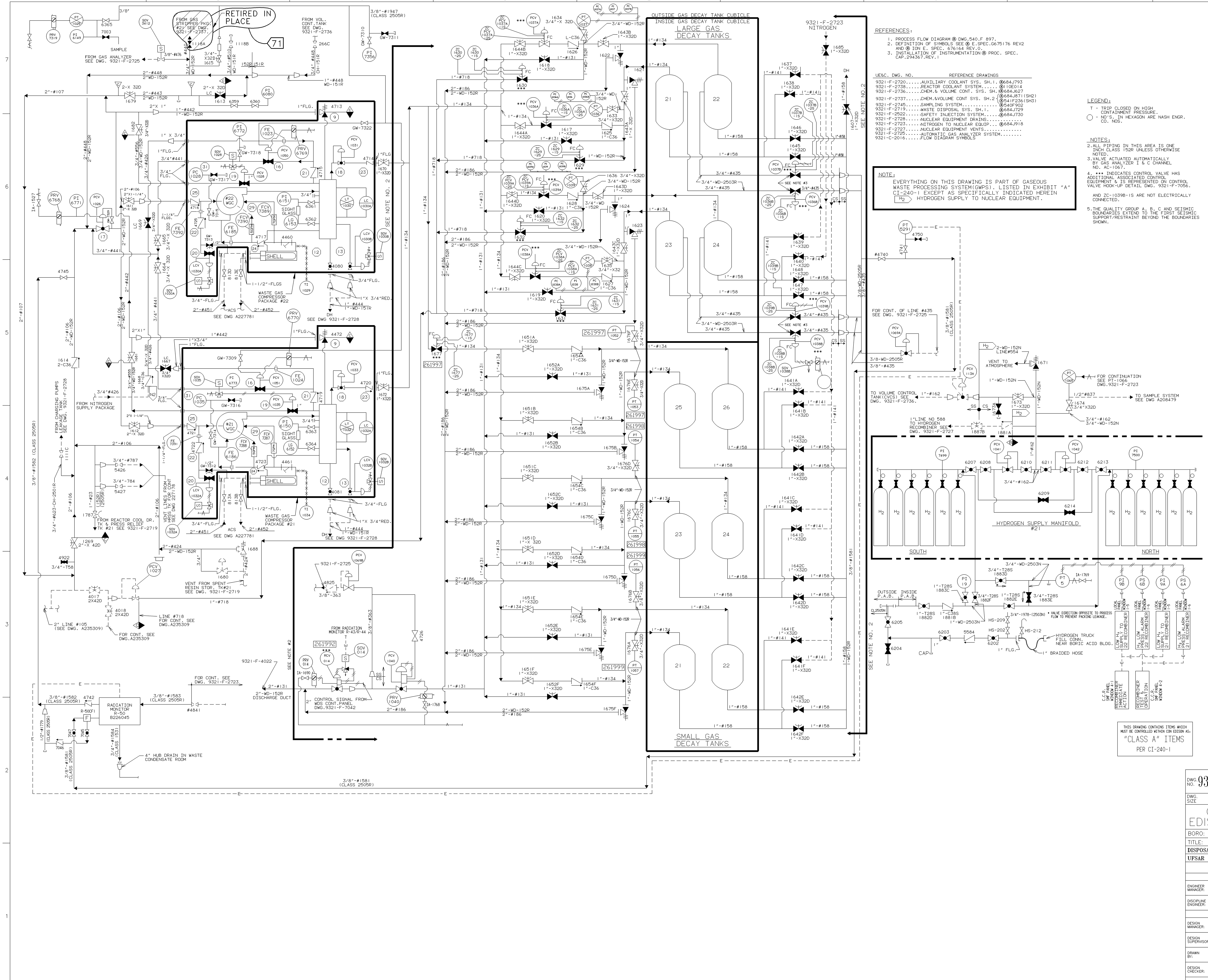
INDIAN POINT UNIT No. 2

UFSAR FIGURE 11B-5

TIME TO REACH PEAK CONCENTRATION AT CHELSEA vs HALF-LIFE FOR 1 CURIE RELEASE

MIC. No. 1999MC3949	REV. No. 17A


9321-F-2719-128
128 INCORPORATED DRN-04-04579, ER-IP2-04-34741
REVISION
USE .1 SIZE TEXT ONLY
INSIDE CONTAINMENT BLDG.
INSIDE PRIMARY AUXILIARY BLDG.
INSIDE WASTE DISPOSAL AREA
HOLDUP TANK PIT SUMP PUMP
WASTE HOLD-UP TANK #21
WASTE EVAPORATOR FEED PUMP
WASTE CONDENSATE TANKS
WASTE CONDENSATE PUMPS
DISTILLATE FILTER
RETIRED IN PLACE
REACTOR COOLANT DRAIN TANK #21
REACTOR COOLANT DRAIN PUMPS
ACCUMULATOR DRAINS
CONTAINMENT SUMP PUMPS
REACTOR CAVITY PIT SUMP PIT
SPENT RESIN STORAGE TANK #21
CHEMICAL DRAIN TANK #21
CHEMICAL DRAIN PUMP
SUMP TANK #21
SUMP TANK PUMPS
PAB SUMP PUMPS
FOR AIR SUPPLY TO LCV'S SEE DWG. A242688 (TYP.)
REFERENCES
1. PROCESS FLOW DIAGRAM Ⓢ DWG. 9321-F-2719 ...
2. DEFINITION OF SYMBOLS Ⓢ E. SPEC. G676176 REV. 2
3. INSTRUMENTATION & CONTROL STANDARDS Ⓢ SYMBOL & APPLICATIONS FOR INSTRUMENT DIAGRAMS. SECTION 1.1 ISSUED AUG. 12TH, 1966
Ⓢ INSTRUMENT INSTALLATION SECTION 2.0 ISSUED NOV. 16TH, 1966
4. MATERIAL SPEC. & FITTINGS Ⓢ E. SPEC. G569666 REV. 8
Ⓢ E. SPEC. G676398 REV. 0
REFERENCE DRAWINGS
RCS - 9321-F-2738 - REACTOR COOLANT SYS.
ACS - 9321-F-2720 - AUXILIARY COOLANT SYS.
CVCS - 9321-F-2736 - CHEMICAL & VOLUME CONTROL SYS.
SIS - 9321-F-2735 - SAFETY INJECTION SYS.
SS - 9321-F-2745 - SAMPLING SYS.
IVSWS - 9321-F-2726 - ISOLATION VALVE SEAL WATER SYS.
WDS - 9321-F-2730 - WASTE DISPOSAL SYS.
NS - 9321-F-2723 - NITROGEN TO NUCLEAR EQUIP.
9321-C-2018 - FLOW DIAGRAM SYMBOLS
NS NITROGEN SUPPLY SYSTEM
PW PRIMARY WATER SYSTEM
GWPS GASEOUS WASTE PROCESSING SYSTEM
SIS SAFETY INJECTION SYSTEM
NOTES
COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED
"CLASS A" ITEMS PER THE QAPD
DWG. NO. 9321-F-2719-128
SIZE A
CON EDISON
INDIAN POINT
BORO: WESTCHESTER
TITLE: FLOW DIAGRAM WASTE DISPOSAL SYSTEM
APPROVALS
ENGINEERING
DESIGN
W.J.KING 10-26-87
P.A. 10-26-87
B.JOHN 10-26-87
K.D. 10-26-87
SCALE: NONE
MY



REFERENCES:

1. PROCESS FLOW DIAGRAM Ⓦ DWG.540.F 897.
2. DEFINITION OF SYMBOLS SEE Ⓦ E.SPEC.G675176 REV2 AND Ⓦ ION E. SPEC. 676164 REV.0.
3. INSTALLATION OF INSTRUMENTATION Ⓦ PROC. SPEC. CAP.294367.REV.1

UE&C. DWG. NO.	REFERENCE DRAWINGS	
9321-F-2720	AUXILIARY COOLANT SYS. SH.1.	Ⓦ684J793
9321-F-2738	REACTOR COOLANT SYSTEM	Ⓦ110E014
9321-F-2736	CHEM.& VOLUME CONT. SYS. SH.1	Ⓦ684J627
9321-F-2737	CHEM.&VOLUME CONT SYS. SH.2	Ⓦ684J871(SH2) Ⓦ541F236(SH3)
9321-F-2745	SAMPLING SYSTEM	Ⓦ540F902
9321-F-2719	WASTE DISPOSAL SYS. SH.1.	Ⓦ684J729
9321-F-2522	SAFETY INJECTION SYSTEM	Ⓦ684J730
9321-F-2728	NUCLEAR EQUIPMENT DRAINS	
9321-F-2723	NITROGEN TO NUCLEAR EQUIP...	Ⓦ684J918
9321-F-2727	NUCLEAR EQUIPMENT VENTS	
9321-F-2725	AUTOMATIC GAS ANALYZER SYSTEM	
9321-C-2016	FLOW DIAGRAM SYMBOLS	

NOTE:
EVERYTHING ON THIS DRAWING IS PART OF GASEOUS WASTE PROCESSING SYSTEM(GWPS). LISTED IN EXHIBIT "A" CI-240-1 EXCEPT AS SPECIFICALLY INDICATED HEREIN
H_2 HYDROGEN SUPPLY TO NUCLEAR EQUIPMENT.

LEGEND:

T - TRIP CLOSED ON HIGH CONTAINMENT PRESSURE.
⬡ - NO'S. IN HEXAGON ARE NASH ENGR. CO. NOS.

NOTES:

2. ALL PIPING IN THIS AREA IS ONE INCH CLASS 152R UNLESS OTHERWISE NOTED.
3. VALVE ACTUATED AUTOMATICALLY BY GAS ANALYZER I & C CHANNEL NO. AC-1067.
4. *** INDICATES CONTROL VALVE HAS ADDITIONAL ASSOCIATED CONTROL EQUIPMENT & IS REPRESENTED ON CONTROL VALVE HOOK-UP DETAIL DWG. 9321-F-7056.

AND ZC-1039B-1S ARE NOT ELECTRICALLY CONNECTED.

5. THE QUALITY GROUP A, B, C AND SEISMIC BOUNDARIES EXTEND TO THE FIRST SEISMIC SUPPORT/RESTRAINT BEYOND THE BOUNDARIES SHOWN.

REVISION
USE .1 SIZE TEXT ONLY

71
THIS REVISION IS NON CLASS PER CI-240-1. UPDATED DRAWING TO SHOW RETIRED BORON RECOVERY EQUIPTMENT PER CRS #199806527. RELEASED AS CONSTRUCTED P.N. 69901-AF.
D.B./Y.J. 02/09/01

THIS DRAWING CONTAINS ITEMS WHICH MUST BE CONTROLLED WITHIN CON EDISON AS:
"CLASS A" ITEMS
PER CI-240-1

REVISION SIGNATURES

REV	DES	ENG
71		G. BHALLA 02/09/2001

DWG. NO. 9321-F-2730-71

DWG. SIZE A	DWG. TYPE COMPANY
CON EDISON	STATION INDIAN POINT
BORO: WESCHESTER	
TITLE: FLOW DIAGRAM WASTE DISPOSAL SYSTEM SHT. No. 2	
UFSAR FIG. No. 11.1-1 (SHT. 2)	
APPROVALS	
ENGINEERING	
ENGINEER MANAGER:	
DISCIPLINE ENGINEER:	
DESIGN	
DESIGN MANAGER: W.J. KING/G.C. 1-13-88	
DESIGN SUPERVISOR: F.A. 1-13-88	
DRAWN BY: GIBBS & HILL	
DESIGN CHECKER: A.G. 1-13-88	
SCALE: NONE	DISCIPLINE CODE: MY

COMPUTER GENERATED DRAWING NOT TO BE HAND REVISED

9321-F-2730-71

CHAPTER 12
CONDUCT OF OPERATIONS

12.1 ORGANIZATION AND RESPONSIBILITY

Operation and maintenance of the Indian Point Unit 2 facility is the responsibility of the Entergy Nuclear organization. The management organization and functional responsibilities as they relate to the operation and maintenance of the Indian Point facility are discussed in Section 1.10.3 and in the Quality Assurance Program Manual (QAPM).

12.1.1 Facility Staff

The corporate officer with direct responsibility for the plant shall be responsible for overall facility activities and shall delegate in writing the succession to this responsibility during his absence.

The Plant Manager is responsible for overall unit safe operation and has control over those onsite activities necessary for safe operation and maintenance of the plant.

The facility organization, duty shift composition, control room occupancy, and other requirements for reactor operational and refueling personnel are in accordance with the Technical Specifications.

A fire brigade is maintained on the site at all times. The organization, operation and training of the fire brigade is discussed in the document under separate cover entitled, "IPEC Fire Protection Program Plan."

12.1.2 Facility Staff Qualifications

Each member of the facility staff meets or exceeds the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for (1) the Operations Manager and the Assistant Operation Manager's SRO license requirement which shall be in accordance with Technical Specification 5.2.2.e, and (2) the Radiation Protection Manager who meets or exceeds the minimum qualifications of Regulatory Guide 1.8, September 1975.

The Plant Manager meets or exceeds the minimum qualifications specified for Plant Manager in ANSI N18.1-1971.

Each Watch Engineer has a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

TABLE 12.1-1
DELETED

12.1 FIGURES

Figure No.	Title
Figure 12.1-1	Deleted
Figure 12.1-2	Deleted

12.2 TRAINING

A retraining and replacement training program for the facility staff is maintained under the direction of the Nuclear Training Manager and meets or exceeds the requirements and recommendations of Section 5.5 of ANSI N18.1-1971, 10 CFR Part 55 and the requirements of the Technical Specifications.

Other areas of operator training are included in the overall plant training program. These specific areas are the training or retraining of plant personnel on specific procedures in accordance with the TMI Lessons Learned implementation schedule and the modification of reactor operator qualifications relating to experience and training. Details of these additional areas of training are included in References 1 and 2.

The training program for the fire brigade is described in the document under separate cover entitled, "IPEC Fire Protection Program Plan."

An emergency plan training program is maintained to cover licensee and non-licensee individuals or groups assigned to the various functional areas of emergency activity.

Radiation protection training is given to personnel requiring unescorted access to controlled areas of the plant.

The initial and requalification training programs for reactor operators and senior reactor operators include instruction in heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a degraded core as required by NUREG-0737.

Operating personnel from the Plant Manager through the operations chain to the reactor operators and watch engineers receive training in the use of installed systems to control or mitigate accidents that severely damage the core as required by NUREG-0737.

Training requirements for the security force are set forth in the "Security Force Training and Qualification Plan, Indian Point Units 1 and 2."

REFERENCES FOR SECTION 12.2

1. Letter from P. Zarakas, Con Edison, to H Denton, NRC, Subject: Actions Taken To Comply With 30 Day Requirement in the NRC Confirmatory Order of February 11, 1980, dated March 11, 1981.

2. Letter from J. D. O'Toole, Con Edison, to D. G. Eisenhut, NRC, Subject: RC Interim Staffing Criteria, dated January 7, 1981.

12.3 WRITTEN PROCEDURES

Written procedures and administrative policies are established, implemented, and maintained in accordance with the Quality Assurance Program Manual (QAPM).

12.3.1 Emergency Operating Procedures

Emergency operating procedures (EOPs) in use at Indian Point 2 were systematically developed through a program, which included phases of validation, verification, training and

operator feedback. This program met the requirements of NUREG-0737 and utilized the guidance of NUREG-0899, NRC Standard Review Plan 13.5.2, and the Westinghouse Owners Group (WOG) Emergency Response Guidelines. These generic WOG Emergency Response Guidelines were evaluated by the NRC in a December 26, 1985 Supplemental Safety Evaluation Report[1]. The resulting EOPs are symptom oriented and based upon acceptable technical guidelines derived from approved analyses of transients and accidents. Implementation of the procedure development program included analyses of the operator's tasks to identify the instrumentation and controls necessary for the operator to perform the functions specified in the technical guidelines. A writer's guide ensured a consistent method of preparing EOPs to satisfy objectives of being usable, accurate, complete, readable and acceptable to control room personnel. Validation and verification assured they are technically correct and usable, follow the writer's guide, correspond to the control room and plant hardware, and are compatible with the minimum number, qualifications, training and experience of the operating staff. The training and operator feedback phases resulted in the understanding by the operators of the philosophy behind the approach to the EOPs, their mitigative strategy and technical bases. These phases also ensured that the operators are capable of executing the EOPs under expected conditions. EOP training program includes guidance against misuse or misapplication of the EOPs during normal operating events.

In accordance with NRC Generic Letter 82-33, Supplement 1 to NUREG-0737 and NUREG-0899, each licensee is required to have plant specific Procedures Generation Package (PGP) for preparing, implementing and maintaining upgraded Emergency Operating Procedures (EOPs). The PGP is to embody the programmatic elements of the EOP maintenance program including plant specific technical guidelines, a writers guide, the verification and validation programs, the EOP training program, and maintenance of the EOPs consistent with updated generic WOG Emergency Response Guidelines. Con Edison described the Indian Point Unit No. 2 PGP processes and procedures in submittals to the NRC[2,3]. The NRC provided their review and recommendations by NRC Safety Evaluation dated October 16, 1989[4].

REFERENCES FOR SECTION 12.3

1. Letter from T. Novak (NRC) to D. Butterfield (WOG) dated December 26, 1985 forwarding “Supplemental Safety Evaluation Report by the Office of Nuclear Reactor Regulation in the Matter of Westinghouse Owners Group Emergency Response Guidelines”.

2. Letter from J. O’Toole (Con Edison) to D. Eisenhut (NRC), dated June 4, 1984

3. Letter from M. Selman (Con Edison) to Document Control Desk (NRC), dated February 11, 1987

4. Letter from D. Brinkman (NRC) to S. Bram (Con Edison) dated October 16, 1989 forwarding “Safety Evaluation Regarding the Procedures Generation Package for Indian Point Unit 2 (TAC No. 44309).”

12.4 RECORDS

Records concerning facility operations are maintained in the form of logbooks, charts, and other such internal reports as may be needed to document pertinent operating conditions.

The principal logs to be maintained are those in the central control room, in the senior watch supervisor's office, by the shift chemist, and by the shift health physics technician. These logs include descriptions of the operating conditions that exist at the time, descriptions of significant operational efforts accomplished during the shift, and such operating events or circumstances as are deemed pertinent to maintain proper continuity of knowledge and understanding of such matters as responsibility in those areas is passed on from shift to shift.

A record of radiation safety conditions, internal and environmental, is maintained in the form of appropriate log entries, and continuous recording chart information in those functional systems and areas provided with radiation survey instruments. In addition, Radiation Work Permit survey information provides the necessary record of radiation exposure conditions prior to job commencement. Actual personnel radiation exposure information is maintained. Records of controlled radiation releases to the environment are maintained by the station chemical and health physics groups, and all necessary information describing specific radioactivity concentrations, total volumes released, along with any dilution requirements, are entered on the Radioactive Waste Release Permit prepared for each release.

All abnormal occurrences that occur during the course of facility operations are recorded in the senior watch supervisor's logbook and, where appropriate, in the logbooks maintained by the licensed operator in the main control room, the shift chemist, and the shift health physics technician.

Plant modification records (e.g., procedures, drawings, specifications) are maintained on file.

Detailed records of total uranium, U-235, Pu-239, and Pu-241 for all fuel in use or in storage are maintained. Records of fuel transfers are maintained via proper execution of NRC forms. Specific locations for all fuel assemblies in the reactor core or in the fuel storage pools are maintained on appropriate core or fuel storage pool arrangement drawings.

Record maintenance and retention is in accordance with the requirements of the Quality Assurance Program Manual (QAPM). Records are maintained on paper, microfilm/aperture cards, or optical disk storage media. Procedures for maintenance of optical disk records comply with the guidance of NRC Generic Letter 88-18 “Plant Record Storage on Optical Disks.”

12.5 REVIEW AND AUDIT OF OPERATIONS

Matters such as design changes to the facility which require a license amendment, changes to operating procedures, or changes to the Technical Specifications, are conducted in accordance with the requirements of 10 CFR 50 and the Quality Assurance Program Manual (QAPM). To assist in this function, Entergy has chartered two committees specifically for the review of safety-related items. These committees (i.e., the On-Site Safety Review Committee and the Safety Review Committee) function in accordance with the requirements of the Quality Assurance Program Manual (QAPM).

A continuing review of facility operations is performed by the station operating staff and at the executive level.

12.5.1 On-Site Safety Review Committee (OSRC)

The On-Site Safety Review Committee functions to advise on all matters related to nuclear safety in accordance with the requirements of the Quality Assurance Program Manual (QAPM).

12.5.2 Safety Review Committee (SRC)

The Safety Review Committee functions to provide independent review and audit of designated activities and plant operations in accordance with the requirements of the Quality Assurance Program Manual (QAPM).

12.5.3 Qualification of Inspection, Examination, Testing, and Audit Personnel

Entergy's commitments and exceptions related to the qualification of inspection, examination, testing, and audit personnel are described in the Quality Assurance Program Manual (QAPM).

REFERENCES FOR SECTION 12.5

1. Letter from Con Edison to NRC, Subject: Con Edison Response to Generic Letter 81-01, dated July 31, 1981.

2. Letter from S.A. Varga, NRC, to J.D. O'Toole, Con Edison, Subject: NRC Review of Con Edison's Response to Generic Letter 81-01, dated September 27, 1982.

12.6 PLANT SECURITY

The program for ensuring the physical security of the Indian Point Unit 2 station has been reviewed by the NRC and found acceptable.[1] The fully implemented security plan provides the protection needed to meet the general performance requirements of 10 CFR 73.55(a) and the objectives of the specific requirements of 10 CFR 73.55, paragraphs (b) through (h), without impairing the ability to operate the plant safely. The approved plant security program, titled "Indian Point Station Unit Nos. 1 and 2, Physical Security Plan," is addressed in the facility operating license. The approved security plan documents and the NRC Security Plan Evaluation Report have been withheld from public disclosure pursuant to 10 CFR 2.790(d).

Access to Indian Point Unit 1 and 2 areas for all persons is controlled under approved procedures administered by the Station Security Section.

REFERENCES FOR SECTION 12.6

1. Letter from A. Schwencer, NRC, to W. Cahill, Con Edison, Subject: Amendment 50 to Indian Point Unit 2 Operating License and the Facility Physical Security Plan, dated February 27, 1979.

12.7 EMERGENCY PREPAREDNESS

12.7.1 Emergency Plan

In accordance with 10 CFR 50.54(q), a document titled Indian Point Energy Center Emergency Plan was submitted by Entergy to the NRC.[1]

12.7.2 Emergency Response Facilities

The emergency response facilities concept is part of the implementation plan for Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability," as requested by Generic Letter 82-33.

The Emergency Operations Facility provides for the management of overall emergency response, coordination of radiological and environmental assessments, and determination of recommended public protective actions. An alternate Emergency Operations Facility is located outside of the 10-mile emergency planning zone.

The Emergency News Center is a separate facility located at the Hudson Valley Traffic Management Center in Hawthorne, N.Y. The Emergency News Center will be used for information dissemination to the public via the news media.

The Technical Support Center is an onsite facility located adjacent to the control room that would provide plant management and technical support to the reactor operating personnel located in the control room during emergency conditions.

The Operational Support Center is an onsite area, separate from the control room and the Technical Support Center, where support personnel would assemble in an emergency.

In developing the facilities, NRC guidance in regard to facilities, location, space requirements, environmental control, radiological monitoring, reliable communications, site status data, records, and staffing was taken into consideration.

The emergency response facilities became fully functional on March 8, 1983. Their functional capability was initially demonstrated on March 9, 1983, at a full-scale Federal Emergency Management Agency exercise.

REFERENCES FOR SECTION 12.7

1. Letter from J.T. Herron, Entergy, to NRC, Document Control Desk, Subject: Combined Emergency Plan for the Indian Point Energy Center, dated September 26,2002.

CHAPTER 13
TESTS AND OPERATIONS

13.0 INTRODUCTION

[Historical Information] The testing and startup operation of the plant systems prior to full power operation of the unit included tests made prior to the initial reactor fuel loading, precritical tests, zero power tests, and power level escalation, plus tests made as part of the zero power and power ascension program inherent with each core loading cycle and periodic test requirements of the Technical Specifications.

The purpose of the program has been to test and operate the reactor and its various systems (1) to make certain that the equipment has been installed and will operate in accordance with the design requirements, (2) to provide procedures for safe initial fuel loading or fuel reloading and to determine zero power values of core parameters significant to the design and operation, and (3) to bring the unit to its rated capacity in a safe and orderly fashion.

Prior to initial full-power operation of Indian Point Unit 2, the plant underwent a thorough, systematic testing program that successively demonstrated the capability and safety of the plant to proceed to each following stage of testing until full power was achieved and maintained. WEDCO, a wholly owned subsidiary of Westinghouse, had the overall responsibility for engineering, construction management, and initial startup testing. The initial startup tests were subdivided into several stages, each to be completed before the next stage was undertaken. Following the startup and testing program, periodic system and plant performance tests are performed as described in the Technical Specifications.

Detailed procedures stating the test purpose, conditions, precautions, and limitations are prepared for each test. The procedures include a delineation of administrative procedures and test responsibility, equipment clearance procedures, and an overall sequence of startup operations. The procedures specify the sequence of tests and measurements to be conducted and conditions under which each is to be conducted to ensure both safety of operation and the relevancy and consistency of the results obtained. If significant deviations from design predictions should exist, unacceptable behavior be revealed, or apparent anomalies develop, testing is suspended and the situation reviewed by the licensee and technical advisors as appropriate to determine whether a question of safety is involved and what corrective action is to be taken prior to resumption of testing. The ultimate responsibility for these determinations rests with the licensee.

The test objectives incorporate testing of redundant equipment where it is involved. Abnormal plant conditions may be simulated during testing when such conditions do not endanger personnel or equipment, or contaminate clean systems. Where predicted emergency or abnormal conditions are involved in the testing program, the detailed operation is provided in the test procedure.

Acceptance criterion for all components and systems is that the test results are acceptable when the test objectives are met within the design specification limits and within the applicable Technical Specifications.

The test program described in the following sections is based upon the reference plant design and experience gained during startup of other units. The detailed procedures include expected