CHAPTER 4
WASTE DISPOSAL AND RADIATION PROTECTION SYSTEM

4.1 Waste Disposal System

4.1.1 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. 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 and the filter cartridges 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.

4.1.2 System Design and Operation

The waste disposal system process flow diagrams are shown in Figure 4.1-1, Sheets 1 and 2 (replaced with Drawings 9321-2719 and 9321-2730), and performance data are given in the Annual Radioactive Effluent Release Report.

The waste disposal system collects and processes all potentially radioactive facility wastes for removal from the facility 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.

As secondary functions, system components 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 Radioactive Effluent Release Report.

4.1.2.1 System Description

4.1.2.1.1 Liquid Processing

The waste disposal system processes liquids from the following sources:

1. Equipment drains and leaks.
2. Chemical laboratory drains.
3. Decontamination drains.
4. Floor drains.

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

1. Reactor coolant loops.
2. Refueling Canal Drain
3. Containment Spray Header Recirculation Lines

The fluid pumped by the reactor coolant drain pumps is sent to the waste holdup tank. 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. Equipment drains.
6. Chemical drain tank pump.

7. Relief valve discharge from the component cooling surge tank and the chemical and volume control system holdup tanks.
8. Maintenance and Operation Building floor drains.
9. Primary Auxiliary Building sump pumps.

Where facility 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.

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 processed water produced by the demineralizer water processing is collected in two distillate storage tanks. Each storage tank is vented to the unit 1 ventilation system. 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 ODCM.

4.1.2.1.2 Gas Processing

Gaseous waste will originate from:

1. Displacement of cover gases as liquid accumulates in various tanks.
2. Equipment purging.
3. Sampling operations and automatic gas analysis for hydrogen and oxygen in cover gases.

The waste disposal system supplies nitrogen to 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 site personnel. 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, site personnel manually sets the operating pressure back to 110 psig and the backup pressure at 90 psig.

Most of the gas received by the waste disposal system 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. 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 (0.5 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 utilized with the second unit instrumented to act as backup. 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 site personnel 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 personnel of this event so that he may select a new backup tank. Pressure indicators are supplied to aid site personnel 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 site personnel 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. 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 site personnel 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.

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 manually 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 actual amounts of gas activity (by isotope) released to the environment is given in the Annual Radioactive Effluent Release Report.

4.1.2.1.3 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 4.1.2.2.5. 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 facility, 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.

4.1.2.2 Components

Codes applying to components of the waste disposal system are shown in Table 4.1-1. Component summary data is shown in Table 4.1-2. 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.

4.1.2.2.1 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.

4.1.2.2.2 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.

4.1.2.2.3 Waste Holdup Tank

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

4.1.2.2.4 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.

4.1.2.2.5 Spent Resin Storage Tank

The spent resin storage tank retains resin discharged from the facility 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.

4.1.2.2.6 Gas Decay Tanks

Four large (525-ft^3) welded, vertical, carbon steel tanks are provided to hold radioactive waste gases for decay. Four tanks are provided so that 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 site personnel 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 site personnel 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.

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 6.3.8).

4.1.2.2.7 Compressors

Two compressors are provided for continuous removal of gases from equipment discharging to the facility 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 out-leakage of compressor seal-water at a negligible level.

4.1.2.2.8 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.

4.1.2.2.9 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.

4.1.2.2.10 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 site personnel.

4.1.2.2.11 Pumps

Pumps used throughout the system for draining tanks and transferring liquids are shown on Figure 4.1-1 sheets 1 and 2 (replaced with Drawings 9321-2719 and 9321-2730).

The wetted surfaces of all pumps are stainless steel.

4.1.2.2.12 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.

4.1.2.2.13 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.

4.1.3 <u>Design Evaluation</u>

4.1.3.1 <u>Liquid Wastes</u>

Liquid wastes are primarily generated by facility operations. The Annual Radioactive Effluent Release Report provides the total liquid effluent activity released by isotope.

Appendix 4B 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 retained for historical purposes.

4.1.3.2 Gaseous Wastes

Gaseous wastes consist primarily of nitrogen and hydrogen gases purged from the chemical and volume control system tank, nitrogen from the closed gas blanketing system. 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 Radioactive Effluent Release Disposal Report.

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

4.1.3.3 Solid Wastes

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

Spent resins and filters are 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.

4.1.4 Minimum Operating Conditions

Minimum operating conditions for the waste disposal system are enumerated in the ODCM.

TABLE 4.1-1
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
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 4.1-2 (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
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 4.1-2 (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 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

4.1 FIGURES

4.2 Radiation Protection

4.2.1 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 facility betterment activities are subjected to ALARA reviews to ensure dose reduction actions are taken. ALARA is taught in Radiation Worker Qualification courses

4.2.1.1 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 plant vent and the waste disposal system liquid effluent are monitored for radioactivity during normal operations and accident conditions.

All gaseous effluent from possible sources of accidental releases of radioactivity (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. The details of the procedures and equipment to be used in the event of an accident are specified in Section 4.2.5, the procedures, and the emergency plan. The formulation of these details considers the requirements for notification of personnel, the utility load dispatcher, and local authorities.

4.2.1.2 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 3.3.3.2.2.

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 4.2.3.

4.2.1.3 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.

4.2.1.4 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 4.1.2, 6.3, and 6.4. The components of the waste disposal system are designed to the pressures given in Table 4.1-2 and the codes given in Table 4.1-1. Hence, the probability of a rupture or failure of the system is exceedingly low.

4.2.2 Shielding

4.2.2.1 Design Basis

Radiation shielding is designed to limit the normal radiation levels at the site boundary below those levels allowed for continuous non-occupational exposure.

Site personnel at the facility are protected by adequate shielding, monitoring, and procedures. When additional shielding is suggested, and permitted, 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 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.

The shielding is divided into the following categories according to function: (1) fuel transfer shielding; and (2) the auxiliary shielding.

4.2.2.1.1 Fuel Handling Shield

The fuel handling shield 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.

4.2.2.1.2 Auxiliary Shielding

The function of the auxiliary shielding is to protect personnel working near various system components in the chemical and volume control system and 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 activities. Samples may be 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 site personnel. 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.

4.2.2.2 Shielding Design

4.2.2.2.1 Fuel Handling Shield

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 fuel handling shield design parameters are listed in Table 4.2-1.

4.2.2.2.2 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.

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 4.2-2.

4.2.3 Radiation Monitoring System

4.2.3.1 Design Bases

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 facility malfunction, which might lead to a health hazard or facility damage.

Instruments are located at selected points in and around the facility 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 facility staff. Adequate information and warning are thereby provided for the continued safe maintenance of the facility and assurance that personnel exposure does not exceed 10 CFR 20 limits.

4.2.3.2 Radiation Monitoring Betterment Program

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 4.2-3 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 the Appendix R / SBO diesel generator in the event of loss of other power sources.

Information on specific monitors is given in the following sections.

4.2.3.2.1 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.

4.2.3.2.2 Plant Vent Air Monitor

R44 monitors for gaseous activity. It was historically seismically qualified, and it’s power supplies was historically class IE. On detection of a high activity level, R44 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.

4.2.3.2.3 Component Cooling Radiation Monitor

This channel, R47, monitors the component cooling loop for radioactivity. 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.

4.2.3.2.4 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.

4.2.3.2.5 Liquid Waste Effluent 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 tank discharges upon detecting high activity.

4.2.3.2.6 Unit 1 Stack Radiation Monitor

R60 monitors for gaseous activity in the air in the Unit 1 stack. Particulates and iodines are collected on filters and analyzed in the count room.

4.2.3.2.7 Sphere Foundation Drain Sump Liquid Effluent

Monitor R-62 monitors the activity of the liquid discharge from the Unit 1 Sphere Foundation Drain Sump drainage. This monitor alarms of the common process radiation monitor panel for high radiation.

4.2.3.3 Original Radiation Monitoring System

4.2.3.3.1 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.

4.2.3.3.2 Monitor Channel Output

The maximum channel output of the radiation monitors is given in Table 4.2-3.

4.2.3.3.3 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 post-accident 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 facility.

3. Environmental radiation monitoring system, which monitors radiation in the area surrounding the facility.

4.2.3.3.4 Original Area Radiation Monitoring System

The Unit 2 area radiation monitoring system consists of two 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-5	Spent fuel building

Channels R-1 and R-5 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.

4.2.3.4 NUREG-0737 Monitors

The following monitors were installed in conformance with NUREG-0737, "Clarification of TMI Action Plan Requirements":

4.2.3.4.1 Wide Range Gas Monitor (R-27)

The wide range 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.

4.2.3.4.2 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.

4.2.4 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.

4.2.5 Radiation Protection and Medical Programs

4.2.5.1 Personnel Monitoring

The official and permanent record of accumulated external radiation exposure received by individuals is obtained principally from a Dosimeter of Legal Record (DLR). Direct reading and electronic dosimeters provide day-by-day indication of external radiation exposure.

Special or additional DLRs are issued as may be required under unusual conditions. These devices are issued as directed by the Radiation Protection department.

The DLRs are processed on a routine basis, typically at 6-month intervals.

Annual reports of personnel monitoring are submitted to the NRC in accordance with 10 CFR 20.2206.

4.2.5.2 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 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.

4.2.5.3 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 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 are defined in Technical Specifications 5.7.1 and 5.7.2 of the IP2 PDTS.

4.2.5.4 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 locations as dictated by the 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.

4.2.5.5 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.

4.2.5.6 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.

4.2.5.7 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.

4.2.5.8 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.

4.2.5.9 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.

4.2.6 Evaluation of Radiation Protection

All liquid waste releases will be assayed for radioactivity to comply with the limits (one-tenth of 10 CFR 20) for unrestricted areas specified.

4.2.7 Tests and Inspections

The gas and particulate effluent monitors shall be tested at every two years 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 every two years with calibrated sources and normal response of each monitor shall be tested daily using a remotely-operated test source to verify the instruments response.

4.2.8 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).

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.

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.

BIBLIOGRAPHY FOR SECTION 4.2

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 4.2-1
Fuel Handling Shield Design Parameters

Maximum dose rate adjacent to spent fuel pit	0.75 mrem/hr
Maximum dose rate at water surface	2.0 mrem/hr

TABLE 4.2-2
Principal Auxiliary Shielding

Component	Concrete Shield Thickness
Demineralizers	4-ft - 0-in.
Gas decay tanks	3-ft - 6-in.
Gas compressor	2-ft - 0-in.
Motor control centers and support equipment	1-ft – 0-in.
Design parameters for the auxiliary shielding include:	
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 4.2-3
Radiation Monitoring Channel Data

Effluent Monitors

Channel	Stream Monitored	Normal Maximum Channel Output
R-27	Wide Range Gas	1.0×10^{5} uCi/cc
R-39 R-40	Service Water from Component Cooling Heat Exchangers	1.0×10^{7} CPM 1.0×10^{7} CPM
R-44	Plant Vent Air Gaseous	1.0×10^{7} CPM
R-50	Waste Gas Decay Tanks	5.0×10^{4} uCi/cc
R-54	Liquid Waste Effluent	1.0×10^{7} CPM
R-60	Unit 1 Stack Air Gaseous	1.0×10^{7} CPM
R-62	Unit 1 Sphere Foundation Drain Sump	1.0×10^{7} CPM

Process Monitors

Channel	Stream Monitored	Normal Maximum Channel Output
R-47	Component Cooling Water	1.0×10^{7} CPM
R-38-1 R-38-2	Control Room Air Intake	1.0×10^{3} mR/hr 1.0×10^{3} mR/hr

Area Monitors

Channel	Stream Monitored	Normal Maximum Channel Output
R-1	Control Room	1.0×10^{4} mR/hr
R-5	Spent Fuel Building	1.0×10^{4} mR/hr

Note: Radiation monitors listed as Effluent Radiation Monitors in Table 4.2-3 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 biennial frequencies.

Appendix 4B – HISTORICAL INFORMATION

DETERMINATION OF RIVER WATER DILUTION FACTORS BETWEEN THE INDIAN POINT SITE AND THE NEAREST PUBLIC DRINKING WATER INTAKES

LIST OF TABLES

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Appendix 4B – HISTORICAL INFORMATION

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 4B-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 continuous 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 4B-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 4B-1 through 4B-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 4B-1, the peak concentrations at Chelsea and Castle Point are for the purpose of this discussion essentially the same.

2. Referring to Figure 4B-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 4B-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 4B-3 and 4B-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.

TABLE 4B-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	2.3×10^{-3}	1.54×10^{2}	15.25×10^{-12}	1.5×10^{-7}	3.99×10^{-12}	3.99×10^{-8}	3.82
Mn-56	6.3	3.33×10^{4}	118.5×10^{-12}	1.2×10^{-6}	5.5×10^{-20}	5.5×10^{-16}	2.16×10^{9}
Co-58	0.97×10^{-2}	4.62×10^{3}	332×10^{-12}	3.3×10^{-6}	6.35×10^{-11}	5.35×10^{-7}	5.22
Fe-59	1.5×10^{-2}	1.07×10^{2}	6.77×10^{-12}	1.1×10^{-7}	1.05×10^{-12}	1.75×10^{-8}	6.45
Co-69	3.6×10^{-4}	5.45×10^{2}	61.8×10^{-12}	1.2×10^{-6}	1.73×10^{-11}	3.45×10^{-7}	3.58
Br-84	3.15×10^{-3}	1.63×10^{4}	1530×10^{-12}	-	-	-	-
Rb-88	5.6×10^{-3}	1.54×10^{4}	1.28×10^{-7}	-	-	-	-
Rb-89	6.48×10^{-3}	3.56×10^{4}	2870×10^{-12}	-	-	-	-
Sr-89	1.37×10^{-2}	1.20×10^{3}	76.4×10^{-12}	2.5×10^{-5}	1.25×10^{-11}	4.28×10^{-6}	6.11
Sr-90	0.69×10^{-4}	0.81×10^{2}	9.35×10^{-12}	3.1×10^{-5}	2.68×10^{-12}	8.92×10^{-6}	3.49
Y-90	2.6×10^{-4}	1.66×10^{2}	2.88×10^{-12}	1.4×10^{-7}	$2.24 \mathrm{X} 10^{-14}$	$1.12 \mathrm{X} 10^{-9}$	352
Sr-91	1.73	7.82×10^{2}	5.32×10^{-12}	0.8×10^{-7}	6.1×10^{-17}	8.70×10^{-13}	8.72×10^{4}
Y-91	1.2×10^{-2}	3.56×10^{2}	23.9×10^{-12}	8×10^{-7}	4.27×10^{-12}	1.34×10^{-7}	5.60
Mo-99	2.5×10^{-1}	1.96×10^{6}	3.47×10^{-8}	1.7×10^{-4}	2.84×10^{-10}	1.42×10^{-6}	122

TABLE 4B-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}	$5.1 X 10^{-12}$	$2.5 X 10^{-7}$	$1.33 X 10^{-14}$	$6.65 X 10^{-10}$	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}					

4B FIGURES