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 <u>Design Evaluation</u>

11.1.3.1 <u>Liquid Wastes</u>

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 <u>Gaseous Wastes</u>

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 through 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
V = Vertical
C = Centrifugal
CC = Centrifugal canned
CC = Carbon Steel
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
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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 dose 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 non-occupational 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 Shielding Design

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 Radiation Monitoring System

11.2.3.1 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 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

[Deleted] EC-31455 removes R48 from service and retires it in place.

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

Steam Generator Secondary System radiation monitors (R-55A, R-55B, R-55C, R-55D) have been Interim Abandoned in place in accordance with Entergy procedure EN-DC-162 via Engineering Change EC-26250, and are no longer in service.

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 pre-calibrated 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 three channels, which monitor radiation levels in various Unit 1 locations. These areas are listed below:

Channel	Area Monitor
ARM-1	Drum Storage Area Corridor
[Deleted]	
ARM-3	Nuclear Service Building SBBPS HX Room
[Deleted]	
ARM-5	Fuel Handling Floor

Channels ARM-1, ARM-3 and 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 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 NUREG-0737 Monitors

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} μCi/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 [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 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 [Deleted].

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.

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
	[Deleted]	
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*	(Interim abandoned per EC-26250)	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
	[Deleted]	
ARM-3	Unit 1 Nuclear Services Building Valve Room	1.0×10^3 mR/hr
	[Deleted]	
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* R-26*	Containment High Range Radiation	1.0×10^{10} mR/hr 1.0×10^{10} mR/hr
R-37-1 R-37-2 R-37-3	Post Accident Sampling System	1.0×10^7 mR/hr 1.0×10^7 mR/hr 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

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Table and Title

11B-1 Concentrations of Primary Coolant Isotopes to the Hudson River at Indian Point and Chelsea

11B-2 Concentrations of Radioisotopes in the Hudson River at Indian Point and Chelsea

LIST OF FIGURES

Figure and Title

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11B-2 Iodine-131 Concentration at Chelsea vs Days After Burst Release From Indian Point for 1 Curie Release

11B-3 Maximum Concentration vs Distance Upstream for 1 Curie Release

11B-4 Maximum Concentration at Chelsea vs Half-Life for 1 Curie Release

11B-5 Time to Reach Peak Concentration at Chelsea vs Half-Life for 1 Curie Release

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 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 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	2.3×10^{-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}	$2.24 \text{X} 10^{-14}$	$1.12 \text{X} 10^{-9}$	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 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	$8.62x10^{-2}$	$1.04x10^{6}$	$3.07x10^{-8}$	$1x10^{-1}$	$1.35x10^{-9}$	$4.5x10^{-3}$	22.7
Te-132	$0.9x10^{-2}$	$1.10x10^{5}$	$8.08x10^{-9}$	$2.7x10^{-4}$	$2.38x10^{-12}$	$7.94x10^{-7}$	3400
I-132	7.2	$3.56x10^{5}$	$1.18x10^{-9}$	$1.5x10^{-4}$	$1.63x10^{-19}$	$2.03x10^{-14}$	$7.25x10^{9}$
I-133	0.81	$8.05x10^{5}$	$7.97x10^{-9}$	$8x10^{-3}$	$2.82x10^{-12}$	$2.82x10^{-6}$	2830
Te-134	23	$1.16x10^{4}$	$21.6x10^{-12}$	-	-	-	-
I-134	19	$2.12x10^{5}$	$4.34x10^{-10}$	$2.2x10^{-5}$	$7.70x10^{-26}$	$3.85x10^{-21}$	$5.64x10^{15}$
Cs-134	$0.93x10^{-3}$	$1.36x10^{5}$	$1.47x10^{-8}$	$1.6x10^{-3}$	$4.01x10^{-9}$	$4.46x10^{-4}$	3.67
I-135	2.39	$8.05x10^{5}$	$4.58x10^{-9}$	$1.1x10^{-3}$	$5.88x10^{-15}$	$1.47x10^{-9}$	$7.8x10^{5}$
Cs-136	$5.14x10^{-2}$	$1.32x10^{4}$	$4.95x10^{-10}$	$6x10^{-6}$	$3.49x10^{-11}$	$3.88x10^{-7}$	14.2
Cs-137	$6.3x10^{-4}$	$5.76x10^{5}$	$6.34x10^{-8}$	$3.2x10^{-3}$	$1.91x10^{-8}$	$9.55x10^{-4}$	3.32
Cs-138	32	$2.62x10^{4}$	$41.8x10^{-12}$	-	-	-	-
Ba-140	$5.4x10^{-2}$	$3.56x10^{2}$	$12.1x10^{-12}$	$4x10^{-7}$	$9.09x10^{-13}$	$3.03x10^{-8}$	13.3
La-140	0.415	$3.70x10^{2}$	$5.1X10^{-12}$	$2.5X10^{-7}$	$1.33X10^{-14}$	$6.65X10^{-10}$	384
Ce-144	$2.44x10^{-3}$	$1.25x10^{3}$	$122.5x10^{-12}$	$1.2x10^{-5}$	$3.05x10^{-11}$	$3.05x10^{-6}$	4.02
Pr-144	$5.13x10^{-2}$	$1.37x10^{6}$	$5.13x10^{-8}$	-	-	-	-
Tritium		$1.49x10^{6}$	$1.74x10^{-7}$	$5.8x10^{-5}$	$4.75x10^{-8}$	$1.59x10^{-5}$	3.66
	Total	$9.15x10^{6}$					

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) 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
I-131	622	$3.8x10^{-6}$	12.2	9.8	$4.99x10^{-8}$	$1.7x10^{-1}$	$7.2x10^{1}$
Te-132	65.7	$4.14x10^{-7}$	$1.88x10^{-2}$	18+	$1.3x10^{-8}$	$4x10^{-4}$	$3.5x10^{1}$
I-132	195	$3.35x10^{-8}$	$4.18x10^{-3}$	1.3	$9.7x10^{-16}$	$1x10^{-10}$	$4.2x10^{7}$
I-133	485	$2.06x10^{-6}$	2.06	3.8	$8.03x10^{-10}$	$8x10^{-4}$	$2.6x10^{3}$
Te-134	6.94	$6.73x10^{-13}$	-	0.8	$5.6x10^{-24}$	-	$1.2x10^{11}$
I-134	127	$6.04x10^{-11}$	$3.02x10^{-6}$	0.8	$3.45x10^{-21}$	$2x10^{-16}$	$1.5x10^{10}$
Cs-134	81.5	$5.17x10^{-7}$	574	21.1	$2.01x10^{-8}$	$2.2x10^{-3}$	$2.6x10^{6}$
I-135	485	$9.3x10^{-7}$	$2.3x10^{-1}$	2.2	$6.62x10^{-12}$	$1.6x10^{-6}$	$1.4x10^{5}$
Cs-136	7.9	$5.0x10^{-8}$	$5.55x10^{-4}$	11.5	$8.98x10^{-10}$	$1x10^{-5}$	$5.6x10^{1}$
Cs-137	348	$2.20x10^{-6}$	$1.10x10^{-1}$	21.6	$8.73x10^{-8}$	$4.4x10^{-3}$	$2.5x10^{1}$
Cs-138	15.7	$1.09x10^{-15}$	-	0.7	$5.23x10^{-26}$	-	$2.1x10^{10}$
Ba-140	0.212	$1.35x10^{-9}$	$4.50x10^{-5}$	11.5	$2.3x10^{-11}$	$8x10^{-8}$	$5.6x10^{2}$
La-140	0.22	$1.15x10^{-9}$	$5.75x10^{-5}$	5.2	$1.95x10^{-12}$	$1x10^{-7}$	$5.8x10^{2}$
Ce-144	0.075	$4.74x10^{-10}$	$4.75x10^{-5}$	20.3	$1.78x10^{-11}$	$2x10^{-7}$	$2.4x10^{2}$
Pr-144	0.082	$5.19x10^{-10}$	-	11.7	$9.65x10^{-12}$	-	$5.4x10^{1}$
Tritium	890	$5.36x10^{-6}$	$1.79x10^{-3}$	21.8	$2.22x10^{-7}$	$8x10^{-4}$	$2.2x10^{0}$

11B FIGURES

Figure No.	Title
Figure 11B-1	Iodine-131 Concentration vs Days After Burst Release From Indian Point for 1 Curie Release
Figure 11B-2	Iodin-131 Concentration vs Chelsea vs Days After Burst Release From Indian Point for 1 Curie Release
Figure 11B-3	Maximum Concentration vs Distance Upstream for 1 Curie Release
Figure 11B-4	Maximum Concentration at Chelsea vs Half-Life for 1 Curie Release
Figure 11B-5	Time to Reach Peak Concentration at Chelsea vs Half-Life for 1 Curie Release

Appendix 11C
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Appendix 11D
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TABLE 11D-1
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11D FIGURES

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Figure 11D-1	Deleted
Figure 11D-2	Deleted

Appendix 11E
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11E FIGURES

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