CHAPTER 11

WASTE DISPOSAL AND RADIATION PROTECTION SYSTEMS

11.1 WASTE DISPOSAL SYSTEM

11.1.1 Design Bases

The General Design Criteria presented and discussed in this section are those that were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for Indian Point 3, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

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 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence. (GDC 70 of 7/11/67)

Liquid, gaseous, and solid waste disposal facilities were designed to achieve discharge of radioactive effluents and offsite shipments of radioactive materials in accordance with applicable governmental 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 quantity of radioactivity, with an isotopic breakdown if necessary. Before any attempt is made to discharge radioactive wastes, they are processed as required and then released under controlled conditions. The system design and operation were 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.

The bulk of the radioactive liquids discharged from the Reactor Coolant System are processed and retained inside the plant by the Chemical and Volume Control System recycle train. This minimizes liquid input to the Waste Disposal System that was designed to process relatively small quantities of generally low-activity level wastes via the Liquid Radwaste Processing System skid. Processed water, from which most of the radioactive material has been removed, is discharged from the Waste Disposal System through a monitored line into the circulating water discharge.

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.

Spent resins from the demineralizers, filter cartridges, and other contaminated solid wastes are packaged and stored on site until shipment for offsite disposal. Suitable containers are used to package these solids at the highest practical concentrations to minimize the numbers of containers shipped for burial.

Solid radioactive wastes in the form of dry activated waste (DAW) or solidified (or dewatered) resins may be stored onsite prior to offsite shipment.

The four steam generators originally installed in the plant were replaced during the Cycle 6/7 refueling outage in 1989. These vessels (as well as one original primary system elbow, which was also replaced) are internally contaminated, and are currently stored on site in the Replaced Steam Generator Storage Facility. The above-ground reinforced concrete storage structure provides adequate shielding to limit offsite doses from direct radiation shine and “skyshine” to less than 5 mrem/year. Building contact dose rates are low enough to permit classification outside the building perimeter as an unrestricted area. The facility is classified as non-safety and non-seismic, and is constructed of noncombustible materials. It is completely sealed with no provisions for ventilation; however, a locked locally alarmed labyrinth entrance is provided in order to permit periodic surveillance.

The four original steam generators are stored completely intact, with all openings sealed with welded steel closure plates or bolted steel covers. The replaced primary elbow is also sealed at both ends with welded steel plates. On this basis, there will be no liquid or gaseous effluents released to the environment for the duration of storage of these components. The facility is designed to house the components until the entire plant is decommissioned.

11.1.2 System Design and Operation

The Waste Disposal System Flow Diagrams are shown in Plant Drawings 9321-F-27193, Sh. 1 & 2, 9321-F-27303, -27233, and -27253, [Formerly Figures 11.1-1A, B, 11.1-2A and 11.1-3]. Typical Performance Data are given in Table 11.1-1. (The IP3 Technical Specifications, section 5.6.3, requires that a Radioactive Effluent Release Report be submitted to the NRC. The data in Table 11.1-1 is from the report submitted to the NRC on February 15, 1991. As part of the review of the Authority’s application to extend the IP3 operating license expiration date, the NRC reviewed IP3 solid waste shipments data for the period of 1986 to 1990. The NRC’s conclusions were contained in an Environmental Assessment, issued by letter dated June 25, 1992.)

The Waste Disposal System was designed to collect and process all potentially radioactive primary plant wastes for removal from the plant site, within the limitations that were established by applicable governmental regulations. During system operation, 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 and they are released under controlled conditions. A radiation monitor is provided to maintain surveillance over the release operation, but the permanent record of activity releases is provided by radio-chemical analysis of known quantities of waste. The system was designed to process wastes generated during continuous operation of the primary system assuming that fission products, corresponding to defects in one percent of the fuel, escape into the reactor coolant. The liquid inventory of the plant is maintained within acceptable limits and releases are well below the limits of 10 CFR 20.

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.

11.1.2.1 System Description

Liquid Processing

The Waste Holdup System collects low level, radioactive liquid waste from throughout the facility and holds the waste until such time that it can be processed. The system consists of three tanks: the 24,500 gallon Waste Holdup Tank No. 31, which is located in the Waste Holdup Tank Pit, and the two 62,000 gallon Waste Holdup Tanks No. 32 and No. 33, which are located in the Liquid Radwaste Storage Facility. Waste Holdup Tanks No. 32 and No. 33 are connected in parallel to tank No. 31 and are provided with a pumped recirculation/spraying system to minimize precipitation of particulates and the accumulation of crud.

The Liquid Radwaste Storage Facility that houses Waste Holdup Tanks No. 32 and No. 33 is an underground concrete structure 75 feet long x 39'-6" wide x 24'-7" high. The 62,000 gallon tanks are supported on concrete piers. A sump pit is located in one of the corners of the building. To service the water tanks, and to interconnect the building with the Waste Holdup Tank Pit, a system of platforms is provided. In addition, an opening of 2'-6" x 7'-6" through the Waste Holdup Tank Pit wall forms the entrance from the Liquid Radwaste Storage Facility to the Waste Holdup Tank Pit. An emergency exit is provided by two openings in the roof of the structure, which is protected by a concrete penthouse. The two buildings are separated by a minimum 3 inch joint filled with seismic filler. The seismic joint adequately ensures that in a seismic event both structures will react independently.

The building is supported on hard rock. The foundation consists of a rigid 2'-0" thick slab that is waterproofed. The waterproof membrane is laid upon a 4" concrete base. A 2 inch protection of concrete is placed over the waterproofing. The walls of the building are also waterproofed and they consist of reinforced concrete. The 3' thick reinforced concrete roof was poured on a steel deck and beam system.

The location of the opening in the Waste Holdup Tank Pit wall is such that it will not affect the structural integrity of the buildings. The east wall was designed to withstand a soil pressure of more than 24 feet. Locating the Liquid Radwaste Storage building adjacent to the Waste Holdup Tank Pit wall removes all of the earth pressure and the resultant stresses. The additional stresses imposed by the penetration are less than those that were imposed by the original loading condition. Therefore the net result is a safer condition of stress in the east wall.

To add operational flexibility in the event that the holdup capacity of the liquid WDS is exceeded, water from the holdup tank can be pumped to a demineralization system. This system consists of a series of pressure vessels containing activated charcoal and anion, cation and macro-reticular resins, and a pump to deliver water to the monitor tanks of the Chemical and Volume Control System. In addition, the Waste Holdup Tank pits are provided with a submersible pump tied to the inlet to waste tank No. 31.

During normal plant operation the Waste Disposal System processes liquids from the following sources:

a) Equipment drains and leaks
b) Radioactive chemical laboratory drains
c) Decontamination drains
d) Demineralizer regeneration
e) Floor drains

The system also collects and transfers liquid drained from the following sources directly to the Chemical and Volume Control System for processing:

a) Reactor coolant loops
b) Pressurizer relief tank
c) Reactor coolant pump secondary seals
d) Excess letdown during startup
e) Accumulators
f) Valve and reactor vessel flange leakoffs

The valve and reactor flange leakoff liquids flow to the Reactor Coolant Drain Tank. The reactor coolant drain tank water can drain directly to the containment sump or can be discharged directly to the CVCS holdup tanks by the reactor coolant drain pumps. These pumps also return water from the refueling canal and cavity to the Refueling Water Storage Tank. To minimize contamination of the RWST, RCDT, and RCDT Pumps resulting from refueling operations, a filter system has been provided for the refueling cavity return flow to the Refueling Water Storage Tank. (See Section 9.3)

Where plant layout permits, waste liquids drain to the waste holdup tanks by gravity flow. Other waste liquids including floor drains drain to the sump and/or sump tank and are discharged to the waste holdup tanks by pumps operated automatically by a level controller.

If the preliminary analysis by sampling indicates that the liquid is suitable for discharge, it can be pumped from the waste holdup tank to the monitor tanks of the Chemical and Volume Control System (FSAR Section 9.2). When one monitor tank is filled it is isolated, and the waste liquid is recirculated and sampled for radioactive and chemical analysis while the second tank is in service. If analysis confirms that the contents are suitable for discharge, the waste liquid contained in the monitor tank is pumped to the service water discharge; otherwise, it is returned to the waste holdup tanks for reprocessing.

Although the radiochemical analysis forms the basis for recording activity releases, the radiation monitor provides surveillance over the operation by preventing the discharge valve from opening if the liquid activity level exceeds that which can be safely discharged.

Liquids in the holdup tanks not suitable for discharge are processed through the Liquid Radwaste Processing System skid.

Sampling of the condenser inlet water and discharge water system is done continuously.

Hudson River water samples are collected continuously from the intake structure (control location) and the discharge canal (indicator location), both of which are located on site. The sampling apparatus draws water from the intake structure and from the discharge canal and pumps it into respective containers. Each container has a volume that is approximately five gallons. One sample of inlet water and one sample of discharge water are taken, at a frequency specified by the Radiological Effluent Control Program, from the containers. Each of these samples is approximately four liters (one gallon). These samples are composited for monthly gamma spectroscopy analysis (GSA) and for quarterly tritium analysis.

Gas Processing

During plant operations, gaseous wastes originate from:

a) Degasing reactor coolant and purging the volume control tank.
b) Displacement of cover gases as liquid accumulates in various tanks.
c) Equipment purging.
d) Sampling operations and automatic gas analysis for hydrogen and oxygen in cover gases.
e) Venting of actuating nitrogen for pressure control valves.

During normal operation, the Waste Disposal System supplies hydrogen from cylinders to primary plant components. Two headers are provided, one for operation, one for backup. The pressure regulator in the operating header is set for 100 psig and that in the backup header at 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 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 100 psig and the backup pressure at 90 psig.

During normal operation, the Waste Disposal System also supplies primary plant components with nitrogen for various process functions. This system, identified as Nitrogen to Nuclear Equipment (NNE) is shown on Plant Drawing 9321-F-27233 [Formerly Figure 11.1-2B]. These process functions include cover gas, calibration gas, purge gas, and gas required for operation of level instrumentation. The only safely-related function of the NNE is providing high pressure charging gas for operation of the Safety Injection Accumulators and Power Operated Relief Valves.

However, administrative controls and Technical Specifications ensure that these components maintain a minimum self-contained supply of gas at all times such that their accident mitigation functions can be implemented at anytime without reliance on the main nitrogen supply system. The NNE also has the capability to cross connect to the weld Channel and Containment Penetration Pressurization System (WCCPPS) and the Isolation Valve Seal Water System (IVSWS) for the purpose of providing an alternative nitrogen supply to those systems when those systems' respective nitrogen supplies are depleted. Again, those activities are strictly controlled by administrative procedures. The NNE may be used in conjunction with the WCCPPS and/or IVSWS for post-accident recovery operations if available. The main nitrogen supply for the NNE is derived either from standard cylinders consisting of two banks of 18 cylinders each or from a nitrogen supply trailer via a truck fill connection. Either supply is directed through a common manifold and then either through redundant, backup regulators, which reduces pressure for low pressure gas services, or through a redundant bypass for high pressure gas services.

The use of either supply source for nitrogen is acceptable and controlled by administrative procedures. Plant Drawing 9321-F-27233 [Formerly Figure 11.1-2B] depicts the flow path when configured from the nitrogen supply trailer, but only to illustrate that when the nitrogen supply trailer is utilized, the isolation valves to the cylinders must be closed to ensure that the cylinders do not deplete with the trailer gas supply and thus be unavailable as a backup gas source when the trailer supply expires. When the cylinders are in use, the nitrogen supply is divided into two independent headers/cylinder banks, one for operation and one for backup. The pressure regulator in the operating header is set for approximately 100 psig discharge, and that in the backup at approximately 90 psig.

When the operating header is exhausted, the discharge pressure will fall below 100 psig and an alarm will alert the operator. The second header/cylinder bank will come into service automatically at approximately 90 psig to ensure a continuous supply of gas. After the exhausted cylinder bank has been replenished, the operator manually sets the operating pressure back to approximately 100 psig and the back up pressure at approximately 90 psig. This redundancy is not considered necessary for the trailer gas supply due to its much greater volume and ease of replacement.

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 (0.5 psig minimum to 4.0 psig maximum) to prevent in-leakage. On the other hand, out-leakage from the system is minimized by using 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. To remove liquid waste buildup from the header, two valves permit draining into individual drain tanks. Any moisture present in the header will drain by gravity to these tanks. The drain valves on the tanks drain to the floor drain, which directs the flow to the Liquid Waste Disposal System. One of the two compressors is in continuous operation when required to be in service 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 110 psig, a pressure transmitter automatically opens the inlet valve to the backup tank, closes the inlet valves 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. Maximum decay time is allowed before releasing gas to the environment. However, the header arrangement at each tank inlet gives the operator freedom to fill, reuse or discharge gas to the environment without restricting operation of other tanks.

Six additional small gas decay tanks are supplied for use during degasing 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. With this arrangement assuming 1% defective fuel rods the activity inventory in any one tank will be less than 2.0×10^4 curies of equivalent Xe-133.

The total radioactivity content of any given gas decay tank is limited by the technical specifications to 50,000 Ci (Xe-133 dose equivalent). This specification ensures that following a postulated gas decay tank rupture, the radiation exposure at the site boundary would not exceed 500 mrem. To preclude exceeding the specification limit, the Offsite Dose Calculation Manual (ODCM) establishes a radioactivity concentration set point in the feed line to the waste gas compressors. A radiation monitor on the feed line monitors the concentration and would alarm if the ODCM set point is exceeded. This, in turn, would alert the operators for action to ensure that the total accumulated tank radioactivity does not exceed the specification limit.

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 considerable 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% by volume of oxygen. This allows him 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. Discharge 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.

When the reactor is in cold shutdown, the RCS Venting System discharges near the suction of the purge system. Vent connections are removed and the RCS venting points are capped off before the reactor is brought to hot shutdown. Liquid drains are routed to the liquid Waste Disposal System.

A gas and particulate monitor is attached to the plant vent stack to analyze the amount of radiation contained in the gas effluent (see Section 11.2).

Solids Processing

All radioactive wastes will be processed, packaged and shipped to a licensed burial facility in accordance with NRC, DOT and State regulations and burial site criteria. Overall system performance is in accordance with an established Process Control Program.

Mechanical filter cartridges are placed in high integrity containers and dewatered.

Spent resins are stored in the Spent Resin Storage Tank for a period of time to allow for decay of short-lived radionuclides. Resin is removed from the storage tanks first by bubbling nitrogen through the tank to agitate the resin and then pumping water through the tank at a controlled rate to sluice the slurry to either the Containment Access Facility (CAF) annex building truck bay or to the fuel storage building cask wash pit. There it is received in a shielded shipping cask fitted with a high integrity container of about 100 cu. ft. The slurry will enter the cask and be dewatered by an internal screen designed to retain the resin. Sluice water returns to the waste holdup tank. The basis for all dose rate calculations is one cycle of operation with one percent defective fuel.

Properly processed and contained spent resins may be stored in suitable locations in Radiologically Controlled Areas onsite.

Miscellaneous solid wastes, such as paper, rags, and glassware can be compressed into 55-gallon drums by a hydraulically operated baler located in the drumming room. Filled drums can be stored in a shielded area in the drumming room, if necessary. These wastes, as well as air filters and small equipment that cannot be successfully decontaminated, may also be stored within 55-gallon drums in suitable locations in Radiologically Controlled Areas onsite.

Components

Codes applying to components of the Waste Disposal System are shown in Table 11.1-2. Components summary data are shown in Table 11.1-3. Waste Disposal System components are located in the Primary Auxiliary Building except for the reactor coolant drain tank that is in the Containment, and the waste holdup tank that is in the liquid holdup tank vault.

The seismic classifications of Waste Disposal System components are included in Chapter 16.

Regenerant Tank

The regenerant tank is austenitic stainless steel and provides facility to batch the caustic solution used to regenerate anion exchange resins.

Chemical Drain Tank

The chemical drain tank is a vertical cylinder of austenitic stainless steel and collects drainage from the hot section of the chemistry laboratory. After analysis, the tank contents are pumped to the waste holdup tanks or to the waste condensate tanks.

Reactor Coolant Drain Tank

The reactor coolant drain tank (RCDT) is a horizontal cylinder with spherically dished heads. The tank is all welded austenitic stainless steel. This tank serves as a drain surge for the Reactor Coolant System and other equipment located inside the reactor containment. This tank can either drain directly into the containment sump or its contents can be pumped to the CVCS holdup tanks.

Waste Holdup Tanks

The waste holdup tanks are the central collection point for radioactive liquid waste. The tanks are stainless steel of welded construction. A pumped recirculation/sparging system is included for Waste Holdup Tanks No. 32 and No. 33 to prevent acid buildup in the tanks. Each tank can be isolated, without affecting the operation of the others.

Sump Tank and 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. All 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.

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 a period of time to allow for decay of short-lived isotopes, and then the tank is emptied. However, the contents can be removed at any time if sufficient shielded 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 decaying fission products. Resin is removed from the tank by first backflushing with nitrogen to loosen the resin bed and then flushing the resin out with water entering the bottom of the tank. The tank is all welded austenitic stainless steel.

Gas Decay Tanks

Four large and six small welded carbon steel tanks are provided to contain compressed waste gases (hydrogen, nitrogen, and fission gases). After a period for radioactive decay, these gases may be released at a controlled rate to the atmosphere through the plant vent. All discharges to the atmosphere are monitored.

Compressors

Two compressors are provided for removal of gases from equipment discharging to the vent header. These compressors are of the water-sealed centrifugal displacement type. Operation of the second compressor could be automatically controlled by radioactive waste gas vent header pressure or manually. Construction is carbon steel. A mechanical seal is provided to maintain outleakage of compressor seal water at a negligible level.

Baler

A hydraulically operated baler can be used to pack compressible solid wastes into 55-gallon drums. The baler is operated manually from a local station and is supplied with a dust shroud to prevent escape of radioactive particulate matter. The shroud vents to the exhaust system.

Nitrogen Manifold

A Nitrogen manifold is installed as one method used to provide a cover gas in the vapor space of various components. It is comprised of two manifolds discharging to a common header, each manifold consisting of a bank of standard cylinders and a pressure regulator. When one manifold depletes to a preset value, the nitrogen supply is automatically switched to the other manifold and an alarm sounds to alert operators to replenish the depleted cylinders. This is one method provided to ensure a continuous supply of gas.

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 manifold where a pressure control valve maintains a constant supply pressure. A manual bypass flowpath is provided.

Gas Analyzer

An automatic gas analyzer with a nominal one-hour recycle time is provided to monitor the concentrations of the 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.

Pumps

The wetted surfaces of pumps are stainless steel.

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.

Valves

All valves, except for diaphragm valves, exposed to gases are carbon steel. All other valves are stainless steel. All valves have stem leakage control. Globe valves are installed with flow over the seats when such an arrangement reduces the possibility of leakage.

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 tank might be overpressurized by improper operation or component malfunction. Tanks containing wastes that contain oxygen and are normally of low activity concentrations are vented into the Primary Auxiliary Building exhaust system.

11.1.3 Design Evaluation

Operating experiences demonstrate that the Waste Disposal System for the facility provides for suitable control of radioactive materials to the environment. A description of Indian Point 3 releases is submitted periodically to the NRC in the Radioactive Effluent Release Report, as described in the Technical Specifications, in accordance with the requirements of US NRC Regulatory Guide 1.21. The Indian Point 3 releases, as evidenced by the aforementioned reports, are well within the limits of 10 CFR 20.

Liquid Wastes

Liquid Wastes are generated by plant maintenance and service operations, and consequently, the quantities and activity concentrations of influents to the Waste Disposal System were, at the time of design, expected values. System loads have been greater than anticipated due to weather leaks and rainwater seepage.

The tritium concentration in a composite sample taken from every batch discharged to the river is determined periodically and used to establish the quantity of tritium released during that period.

As required by 10 CFR Part 20, every reasonable effort was made in the design of the Indian Point 3 Waste Disposal System to maintain radiation exposures and releases of radioactive materials to unrestricted areas as far below the limits specified in 10 CFR 20 as practicable.

In order to ensure that the design objectives were realized, several provisions were made for the radioactive waste processing systems. Specific design features were included in the design of the reciprocating charging pumps to collect leakage from these pumps and return it to the CVCS. The pressurizer spray valves, which are modulating in the RCS, have a live loaded packing configuration installed to mitigate the potential for valve stem leakage. These two specific features were intended to reduce the amount of primary coolant leakage, the processing load, and consequently, the amount of activity being released.

Consideration was also given to continued plant operation with the existence of primary to secondary leakage. Technical Specifications limit primary to secondary leak rate to 0.3 gpm per steam generator.

A manually operated intertie was provided from the Indian Point 3 steam generator blowdown to the Indian Point 1 Secondary Boiler Blowdown Purification System (SBBPS). Processing the Indian Point 3 blowdown through the Indian Point 1 SBBPS reduces releases by at least a factor of 10.

The requirements of 10 CFR 20 were satisfied in the design of the Indian Point Unit No. 3 Liquid Treatment System. Actual releases are reported semi-annually.

Gaseous Wastes

Gaseous Wastes consist primarily of hydrogen stripped from coolant discharged to the CVCS holdup tanks during dilution, nitrogen and hydrogen gases purged from the CVCS volume control when degasing the reactor coolant and nitrogen from the closed gas blanketing system. The gas decay capacity permits 45 days of decay for waste gas before discharge.

In the event of a pipe or tank rupture, the maximum anticipated quantity of waste gas that could be released from any one tank in the system is less than 50,000 curies of equivalent Xe-133, which would result in a dose of less than 0.5 rem beyond the site exclusion boundary.

Gaseous activity release to the plant vent on the Primary Auxiliary Building (PAB) derives from reactor coolant leakage from various system components and from periodic discharges from the gas decay tanks in the Waste Disposal System.

As part of the 10 CFR 20 compliance analysis at the time of initial license application reactor coolant leakage into the PAB was assumed to be 20 gallons per day at ambient temperature. The iodine release due to reactor coolant leakage into the PAB assumed a partition factor of 10^3 and an iodine removal efficiency of 99% for the charcoal filters in the Primary Auxiliary Building Ventilation System.

Gaseous activity releases from the Turbine Building were calculated for concurrent fuel defects and steam generator tube leakage. The turbine building gaseous releases were based on plant operation with an equivalent fuel defect percentage of 0.2% coincident with a steam generator to tube leakage of 20 gpd.

Nobel gas activities are released from the Turbine Building, primarily through the main condenser air ejector. Gaseous iodine is released from the main condenser air ejector. Gaseous iodine activity is also released from the Turbine Building as a result of steam and liquid leakage from various secondary system components into the building and exhaust from the gland seal condenser.

A system measuring the total effluent flow from the steam jet ejectors was installed in agreement with Regulatory Guide 1.97 in order to quantify the amount of radiation released through this path. The system consists of a sensor probe, mounted in the common exhaust from the after-condensers, and a remotely located electronic transmitter. The flow, ranging from 0 to 100 SCFM, may be monitored at the Plant Computer upon demand. (The Plant Computer is described in Section 7.5)

Release from the main condenser air ejector were based on a maximum discharge rate of 60 SCFM and an iodine decontamination factor of 100 in both the steam generator and the main condenser. Releases due to steam and liquid leakage in the secondary system were based on leakages of 6 gpm of (cold equivalent condensed) steam and 12 gpm of (cold, condensed) liquid. An iodine decontamination factor of 100 in the steam generator was taken into account for both types of leakages and in addition a decontamination factor 3×10^3 for iodine was used for the liquid leakage. Releases from the gland seal condenser were based on an exhaust rate of 2.6 gal/min and an iodine decontamination factor of 100 in both the steam generator and the gland seal condenser. Actual releases were reported semi-annually.

Charcoal adsorbers were installed in the containment plant vent, the Primary Auxiliary Building, and the Fuel Handling Building ventilation systems.

Solid Wastes

Solid wastes consist of spent resins, spent filter cartridges, and miscellaneous contaminated materials such as paper, rag, and glassware. All solid radioactive wastes are packaged for removal to a licensed low-level waste burial facility in accordance with applicable regulations and burial site criteria. Waste volume and activities shipped offsite are reported semi-annually.

11.1.4 Minimum Operating Conditions

Minimum operating conditions for the Waste Disposal System are dictated by the Radiological Effluent Controls, the Process Control Program (PCP), and the TRM.

TABLE 11.1-1

TYPICAL DATA

EFFLUENT AND WASTE DISPOSAL SEMI-ANNUAL REPORT (1990)
LIQUID EFFLUENTS-SUMMATION OF ALL RELEASES

	Units	Quarter 3rd	Quarter 4th	Est. Total Error %
A. Fission and activation products				
1. Total release (not including tritium, gases, alpha)	Ci	7.39E-02	1.40E-01	2.50E+01
2. Average diluted concentration during period	uCi/ml	1.50E-10	8.08E-10	
B. Tritium				
1. Total release	Ci	1.03E+02	1.83E+01	2.50E+01
2. Average diluted concentration during period	uCi/ml	2.10E-07	1.06E-07	
C. Dissolved and entrained gases				
1. Total release	Ci	4.11E-00	1.38E-02	2.50E+01
2. Average diluted concentration during period	uCi/ml	8.34E-09	7.94E-11	
D. Gross Alpha radioactivity				
1. Total release	Ci	<9.37E-05	<6.73E-05	2.50E+01
E. Volume of waste released (prior to dilution)	liters	2.34E+06	1.66E+06	1.00E+01
F. Volume of dilution water used during period	liters	4.93E+11	1.73E+11	1.00E+01
G. Percent of liquid effluent limit	%	5.69E-01	6.54E-01	2.50E+01

TABLE 11.1-1
(Cont.)

TYPICAL DATA

EFFLUENT AND WASTE DISPOSAL SEMI-ANNUAL REPORT (1990)
LIQUID EFFLUENTS-SUMMATION OF ALL RELEASES

	Units	Quarter 3rd	Quarter 4th	Est. Total Error %
A. Fission and Activation Gases				
1. Total release	Curies	3.50E+02	1.53E 00	2.50E+01
2. Average release rate for period	uCi/sec	4.40E+01	1.93E-01	
3. Percent of technical spec. limit	%	3.77E+00	2.45E-02	
B. Iodines				
1. Total Iodine - 131	Ci	7.61E-05	9.07E-06	2.50E+01
2. Average release rate for period	uCi/sec	9.58E-06	1.14E-06	
C. Particulates				
1. Particulates with half-lives >8 days	Ci	2.91E-06	2.04E-05	2.50E+01
2. Average release rate for period	uCi/sec	3.66E-07	2.56E-06	
3. Gross alpha radioactivity	Ci	<3.73E-07	<3.58E-07	
D. Tritium				
1. Total release	Ci	3.29E-01	4.58E-01	2.50E+01
2. Average release rate for period	uCi/sec	4.14E-02	5.76E-02	
E. Percent of Tech Spec Limit Iodines, Particulate, & Tritium	%	1.67E-02	4.80E-03	2.50E+01

TABLE 11.1-1
(Cont.)

TYPICAL DATA

EFFLUENT AND WASTE DISPOSAL SEMI-ANNUAL REPORT (1990)
SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

JULY 1 – DECEMBER 31, 1990

A. Solid Waste Shipped Offsite for Burial or Disposal (Not irradiated fuel)

1. Type of Waste	Unit	6 Month Period Class A	Class B	Class C	Est. total Error %
a. Spent resins, filter sludges, etc.	m^3 Ci	1.34E+1 3.51E+1	8.32E+0 6.39E+1	0 0	25
b. Dry compressible, contam. equipment for burial	m^3 Ci	3.07E+1 5.99E+0	0 0	0 0	25
c. Irradiated Components	m^3 Ci	0 0	0 0	0 0	N/A
d. Other: Dry compressible, contaminated equip. for volume reduction at offsite facility	m^3 Ci	6.14E+1 1.92E+0	0 0	0 0	25

2. Estimate of major nuclide composition (by type of waste)

NUCLIDE	UNIT	a. Resin CLASS A	a. Resin CLASS B	b. Dry Waste CLASS A	d. Vol. Red CLASS A
Cr-51	%	1.5	0.6	0	0
Mn-54	%	1.4	0.6	0	0
Fe-55	%	26	16	59	59
Co-58	%	14	4	5	5
Co-60	%	11	28	28	28
Ni-63	%	5.9	11	5	5
Cs-134	%	20	21.8	0	0
Cs-137	%	18	18	2	2

Percentage of nuclides and total activities are based on a combination of direct measurements and scaling for non-gamma emitting nuclides.

TABLE 11.1-1
(Cont.)

TYPICAL DATA

EFFLUENT AND WASTE DISPOSAL SEMI-ANNUAL REPORT (1990)
SOLID WASTE AND IRRADIATED FUEL SHIPMENT

JULY 1 – DECEMBER 31, 1990

3. Solid Waste Disposition

Number of Shipments	Mode of Transport	Destination
7	Truck	Barnwell, SC
3	Truck	SEG, Oak Ridge TN: for volume reduction.

B. Irradiated Fuel Shipments (Disposition)

Number of Shipments	Mode of Transport	Destination
None		

Source of Data: “Semi-Annual Report of Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents for Indian Point 3,” for the period July 1, 1990 through December 31, 1990, transmitted to the NRC by letter from J. E. Russell to T. Martin, dated February 15, 1991 (IP3-91-017).

TABLE 11.1-2

WASTE DISPOSAL 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 31	No code	
Waste Holdup Tanks 32 and 33	ASME III, (2)	Div. 2
Regenerant Tank	No code	
Waste Filter*	No code	
Piping and Valves	USAS-B31.1(3)	Section 1

NOTES:
(1) ASME III – American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section IV, Nuclear Vessels

(2) ASME-III Section VIII

(3) USAS-B31.1 (1955) – Code for pressure piping US American Standards Associations and special nuclear cases where applicable.

* Not used.

TABLE 11.1-3

COMPONENT SUMMARY DATA

Tanks	Quantity	Type	Volume	Design Pressure	Design Temperature	Material
Reactor Coolant Drain	1	Horizontal	350 gal	25 psig	267 F	ss
Regenerant Tank	1	Vertical	400 gal	Atm	180 F	ss
Chemical Drain	1	Vertical	375 gal	Atm	180 F	ss
Sump Tank	1	Vertical	375 gal	Atm	150 F	ss
Waste Holdup Tank No. 31	1	Horizontal	3300 ft^3	Atm	150 F	ss
Waste Holdup Tank No. 32 and 33	2	Horizontal	62,000 gal	Atm	150 F	ss
Spent Resin Storage	1	Vertical	300 ft^3	100 psig	150 F	ss
Gas Decay (large)	4	Vertical	525 ft^3	150 psig	150 F	cs
Gas Decay (small)	6	Vertical	40 ft^3	150 psig	150 F	cs

TABLE 11.1-3
(Cont.)

COMPONENT SUMMARY DATA

Pumps	Quantity	Type	Flow (gpm)	Head (ft)	Design Pressure	Design Temperature	Material[1]
Reactor Coolant Drain (32)	1	Horizontal Centrifugal[2]	135	175	225 psig	500 F	ss
Reactor Coolant Drain (31)	1	Horizontal Centrifugal[2]	75	175	225 psig	500 F	ss
Chemical Drain	1	Horizontal Centrifugal[2]	20	100	100 psig	180 F	ss
Regenerant	1	Horizontal Centrifugal[2]	20	100	100 psig	180 F	ss
Sump Tank	2	Horizontal Centrifugal	20	100	100 psig	180 F	ss

NOTE:
(1) Wetted surfaces only
(2) Mechanical seal provided

Miscellaneous	Quantity	Capacity	Type
Waste Evaporator	1	2 gpm	-

NOTE:
(1) Wetted surfaces only
(2) Mechanical seal provided

11.2 RADIATION PROTECTION

11.2.1 Design Bases

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's evaluations of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NCR) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.13

The NRC has concluded that the current IP3 leakage detection system capability is adequate to continue to support the technical bases cited in the NRC's March 10, 1986, SE approving Leak Before Break (LBB) for the IP3 Primary Coolant Loop piping. This position was further clarified in the IP3 Supplement to Safety Evaluation re: Leakage Detection Systems (TAC No. MB3328).

Monitoring Radioactivity Releases

Criteria: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released form normal conditions, 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 of 7/11/67).

The containment atmosphere, the plant vent, the administration building vent, the containment fan-cooler's service water discharge, the Waste Disposal System gas and liquid effluent, the condenser air ejectors, the component cooling loop liquid, the component cooling water heat exchanger Service Water discharge, the discharges from the condensate polisher waste collection tanks and the steam generator blowdown are monitored for radioactivity released during normal operations, from anticipated transients, and from accident conditions. The fuel Storage Building and waste areas have no functional air monitoring, however, the HVAC Systems for these two areas are routed to the plant vent, which is monitored.

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. All accidental spills in the auxiliary building are collected in a drain tank. Any Waste Disposal System liquid effluent discharged to the condenser circulating water canal is monitored. Any accidental spills from the Liquid Radwaste Processing System skid are collected in the Fuel Storage Building Cask Washdown area which drains to a sump and is pumped to the Waste Disposal System.

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 Health Physics office area provides adequate monitoring of accident releases. The details of the procedures and equipment to be used in the event of an accident are given in the Indian Point Energy Center Emergency Plan.

The discharges from the 20 code safety valves and the 4 power relief valves in the Steam and Power Conversion System are not monitored by the radiation monitoring system, but the activity can be estimated from plant sampling, as the mass of the steam discharged can be determined and the activity concentration in the secondary side is known from periodic sampling.

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 (GCDC 18 of 7/11/67).

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 waste release operation. The permanent record of activity releases I provided by radiochemical analysis of know quantities of waste.

There is a controlled ventilation system for the fuel storage and waste treatment areas of the auxiliary building which discharges to the atmosphere via the plant vent. Radiation monitors are in continuous service in these areas to actuate a high-activity alarm on the control board annunciator, as described in Section 11.2.3.

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 of 6/11/67).

Adequate shielding for radiation protection is ensured during reactor refueling by conducting all spent fuel transfer and storage operations under water. This permits visual control of the operation at all times while maintaining low radiation levels for periodic occupancy of the area by operating personnel. The average exposure with 0.2% failed fuel that personnel could receive from the refueling water during fuel handling operations is 0.5mr/hr. The exposure to the crane operator moving an average fuel assembly is 3.4mr/hr. These dose rates are based on the expected activity during normal refueling operations. Pit water level is indicated, and water removed from the pit must be pumped out since there are no gravity drains. Shielding is provided for waste handling and storage facilities to permit operation within requirements of 10 CFR 20.

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

Protection against Radioactivity Release from Spent Fuel and Waste Storage

Criterion: Provisions shall be made in the design of the fuel and waste storage facilities such that no undue risk to the health and safety of the public could results from an accidental release of radioactivity (GDC 69 of 7/11/67).

All fuel and waste handling and storage facilities are contained, and their related equipment were designed so that accidental releases directly to the atmosphere are monitored and do not exceed the limits of 10 CFR 100; refer to Sections 11.1.2, 14.2.2, and 14.2.3. The components of the Waste Disposal System are not subjected to any high pressure (Table 11.1-3) or stresses and are of Class I seismic design.

In addition, the tanks have a design pressure greater than atmospheric pressure, and the piping and valves of the system were designed to the codes given in Table 11.1-2. Hence, the probability of rupture or failure of the system is low.

The reactor cavity, refueling canal and present fuel storage pit are reinforced concrete structures with a steam-welded stainless steel plate liner. These structures were designed to withstand the anticipated earthquake loadings as seismic Class 1 structures so that the liner prevents leakage even in the event that the reinforced concrete develops cracks.

11.2.2 Shielding

Design Basis

Radiation shielding was designed for operation at maximum rated 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% fuel element defects.

The shielding provided was designed to ensure that in the event of a hypothetical accident, the integrated offsite exposure due to the continued activity will be below the limits established in 10 CFR 100.

A design review of Indian Point 3 was conducted in accordance with NUREG-0578, to identify areas, components and access paths which may require occupancy during post-accident recovery operation. The results of the review have been reported to NRC and remedial actions are being taken to ensure that all vital areas and equipment requiring access under post accident conditions meet the following criteria:

1) Continuous Occupancy – less than or equal to 15 mr/hr
2) Infrequent Access – less than or equal to 5 rem whole body dose, considering the required occupancy for the duration of the accident

Typical Zone 0 areas are the turbine building and turbine plant service areas and the Central Control Room. Typical Zone I areas are the offices, auxiliary building work stations and corridors, and the outer surfaces of the containment and auxiliary building. Zone II areas would include the surface of the refueling water at refueling and the operating deck of the Containment during reactor shutdown. Areas designated Zone III include the sampling room, reactor cavity area after shutdown, and reactor containment penetration areas, including ventilation, steam line and electrical penetrations.

Typical Zone IV areas include areas within the auxiliary building such as charging pump areas, evaporation area, heat exchanger areas, and valve operator areas. Typical Zone V areas are within the regions adjacent to the Reactor Coolant System at power operation and the demineralizer and volume control tank spaces.

All high radiation areas are appropriately marked and isolated in accordance with 10CFR20 and other applicable applications.

The shielding is divided into five categories according to function. These functions include the primary shielding, the secondary shielding, the accident shielding, the fuel handling shielding and the auxiliary shielding.

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 transition temperature
2) Attenuate the neutron flux sufficiently to prevent excessive 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.

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 mc/cc maximum), which is produced by neutron activation of oxygen during passage of the coolant through the core. The secondary shield was designed to limit the full power dose rate outside the Containment Building to less than 0.75 mR/hr.

Accident Shield

The main purpose of the accident shield is to ensure safe radiation levels outside the Containment Building following a maximum credible accident.

Fuel Handling Shield

The fuel handling shield was designed to facilitate the removal and transfer of present fuel assemblies and control rod clusters from the reactor vessel to the spent fuel pit. It was designed to attenuate radiation from spent fuel, control clusters, and reactor vessel internals, and together with exclusion gates, to reduce exposures to less than 2.0 mR/hr at the refueling cavity water surface and less than 0.75 mR/hr in areas adjacent to the spent fuel pit.

Auxiliary Shielding

The function of the 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 and the Sampling System.

The shielding provided for the auxiliary building was designed to limit the dose rate to less than 0.75 mR/hr in normally occupied areas, and at or below 2.0 mR/hr in intermittently occupied areas.

Shielding Design

Primary Shield

The primary shield consists of the core baffle, water annuli, core barrel (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 feet. The lower portion of the shield is a minimum thickness of 6 feet of regular concrete ($p = 2.3\ g/cm^3$) 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 feet thick, except in the area adjacent to fuel handling, where the thickness is increased to 6 feet.

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 out-of-core nuclear instrumentation. Cooling for the primary shield concrete and the nuclear instrumentation is provided by 12,000 cfm cooling air.

The primary shield calculated neutron fluxes and design parameters are listed in Table 11.2-2.

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 cylindrical portion of the reactor containment and a 3 feet concrete annular crane support wall surrounding the reactor coolant loops.

The secondary shield was designed to attenuate the radiation levels in the primary loop compartment from a value of 25 rem/hr to a level of less than 0.75 mr/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.

Accident Shield

The accident shield consists of the 4 feet -6 inches reinforced concrete cylinder capped by a hemispherical reinforced concrete dome of a 3 feet -6 inches thickness. This shielding includes supplemental shields in front of the containment penetrations.

The equipment access hatch is shielded by a 3 feet - 6 inches thick concrete shadow shield and a 1 foot 6 inches 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.

Fuel Handling Shield

The refueling cavity, flooded to elevation 93.7 feet during refueling operations, provides a temporary water shield above the components being withdrawn from the reactor vessel. The water height during refueling is approximately 24.5 feet above the reactor vessel flange. This height ensures that a minimum of 10 feet of water will be above the active fuel of a withdrawn fuel assembly. Under these conditions, the dose rate is less than 2.5 mr/hr at the water surface (Reference: NSE 00-3-039 SFPC). This presumes a minimum pool level elevation of 93.2 feet. The spent fuel pit has a nominal level of 93.7 feet, which is half-way between the minimum and maximum water level alarm setpoints.

The refueling canal is a passageway connected to the reactor cavity and extending to the inside surface of the Reactor Containment. The canal is formed by two concrete walls each 6 feet 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 feet thick, shields the spent fuel transfer tube. This shielding was designed to protect personnel from radiation during the time a spent fuel assemble 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 the concrete walls of the fuel transfer pit. An equivalent of 6 feet of regular concrete is provided to ensure a calculated maximum dose value of 0.75 mr/hr in the areas adjacent to the spent fuel pit. Exclusion gates are also provided in the fuel transfer tube.

Spent fuel is stored in the spent fuel pit which is located adjacent to the Containment Building. Shielding for the spent fuel storage pit is provided by 6 feet thick concrete walls and the pit is flooded to a level such that the water height is grater than 13 feet above the spent fuel assemblies.

The refueling shield design parameters are listed in Table 11.2-5.

Auxiliary Shielding

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 the compartments may be entered without having to shut down and, possibly, to decontaminate the adjacent system.

The shield material provided throughout the auxiliary building is regular concrete (p=2.3 g/cm^3). The principal auxiliary shielding design parameters are tabulated in table 11.2-6.

The design basis used for shield design to allow access to manual backup items (e.g. valves) was that the integrated dose to an operator, immediately after the accident will be less than 3 Rem. Shielding platforms with integrated dose to an operator, immediately after the accident will be less than 3 Rem. Shielding platforms with reach rods to such valves have bee provided.

The shielding will result in dose rates which are not significantly greater than the background dose from the containment (approximately 500 mr for one month following the accident). Doses in the vicinity of equipment located within the Primary Auxiliary Building would be much less due to the shielding afforded by the concrete walls of the Primary Auxiliary Building.

Shielding for the residual heat removal pumps is designed to limit the 8-hour integrated does to 3 Rems during maintenance of one residual pump with the adjacent pump circulating containment sump water. This was accomplished by the provision of shield walls around the pumps and associated piping and reach rods on the valves which must be manually operated.

11.2.3 Radiation Monitoring System

The radiation Monitoring System provides radiation detection equipment to ensure safe operation of the plant.

The system was designed to perform three basic functions:

1) Warn operating personnel of any radiation health hazard which develop.
2) Give early warning of a plant malfunction which might lead to a health hazard or plant damage.
3) Prevent inadvertent release of radioactivity to the environment.

Instruments are located at selected points in and around the plant to detect, indicate and record the radiation levels. If the radiation level should rise above the setpoint established for that channel, an alarm is initiated. The Radiation Monitoring System operates in conjunction with regular and special surveys and with chemical and radiochemical 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.

The only components of this system which are located in the Containment, are the detectors for certain area monitoring channels. Some of these would not be expected to operate following a major Loss-of-Coolant Accident and were not designed for this purpose. Components of all other area and process monitoring channels were designed for post-accident, as required.

The components of the original Radiation Monitoring System were designed according to the following environmental conditions:

1) Temperature – an ambient temperature range of 40 to 120°

2) Humidity – 0 to 95%*

3) Pressure – Components in the Primary Auxiliary Building and the Central Control Room were designed for normal pressure. Area monitoring system components inside the Containment were designed to withstand increased pressure.

4) Radiation – Process monitors are of a non-saturating design so that they “peg” full scale if exposed to radiation levels up to 100 times full-scale indication. Such process monitors are located in areas where the normal and post-accident background radiation levels will not affect their usefulness.

5) Process Monitors R20, R-62 A, B, C, D and R-63 A, B do not have this non-saturating design since they utilize a Geiger-Mueller tube. The range of these monitors is listed in Table 11.2-7, and these ranges are sufficient for these monitors to perform their functions.

*NOTE: Equipment located in the control room area or other areas in the plant with controlled environments may be specified for narrower temperature and humidity ranges because of the controlled environment provided by the heating and ventilating system.

Some monitors of the Radiation Monitoring System are required to meet the requirements of Regulatory Guide 1.97. These monitors are identified in subsections 11.2-3.1 and 11.2.3.2, and meet or exceed the design conditions stated above.

The Radiation Monitoring System is divided into the following subsystems:

1) The Process Radiation Monitoring System monitors various fluid steams for indication of increasing radiation levels.
2) The Area Monitoring System monitors area radiation in various parts of the plant.
3) The Environmental Radiation Monitoring Program monitors radioactivity in the area surrounding the plant. This program is outlined in the Off-site Dose Calculation Manual (ODCM).

In the Radiation Monitoring System, some monitors are located to detect radioactivity in several sample streams fed to a common header. These monitors read gross activity of the streams. Should a monitor detect a high gross activity, the lines into the common header may be isolated using valves located upstream of the header, and each stream read individually.

In order to assure that the sampling lines into the header do not become plugged, they are periodically inspected and tested. On certain lines of high importance (e.g., the steam generator blowdown line) a flow meter indicates any variation in flow rate that would be caused by a stoppage in one of the lines.

11.2.3.1 Process Radiation Monitoring System

This system consists of channels which monitor various fluid streams for indication of increasing radiation levels. The channels and the type of radioactivity monitored are listed in Table 11.2-7A and the measurement ranges are given in Table 11.2-7. The monitors are described below and are designed to detect the minimum concentrations of the isotopes of interest and, in monitoring gross activity, are designed to generate an alarm under abnormal conditions. Isotopic identification and concentrations are determined by grab sample analysis.

The individual channels of the Process Radiation Monitoring System are detailed below:

Containment – Air Particulate Monitor (R-11)

This monitor measures air particulate beta radioactivity in the containment and ensures that the release rate through the containment vent during purging is maintained below specified limits. Two radiation sensitive instruments provide the capability for detection of leakage from the Reactor Coolant System. The containment air particulate monitor is quite sensitive to low leak rates. The containment radiogas monitor is much less sensitive but can be used as a backup to the air particulate monitor.

High radiation level for the channel initiate closure of the containment purge supply and exhaust duct valves and pressure relief line valves.

This channel takes a continuous air sample from the containment atmosphere. The sample is drawn outside the containment in a closed a system monitored by a scintillation counter-filter paper detector assembly. The filter paper collects all particulate matter greater than 1 micron in size, on its constantly moving surface, and is viewed by a photomultiplier-plastic scintillator combination. The sample is returned to the containment, after it passes through the series connected (R-12) gas monitor.

The detector assembly is in a completely enclosed housing. A preamplifier transmits the detector pulse signal to a microprocessor which converts the signal to digital and analog outputs for display and communicated with the Radiation Monitoring System cabinets in the Control Room. The filter paper has a 25-day minimum supply at normal speed. Lead shielding is provided to reduce the background level to where it does not interfere with the detector's sensitivity. The filter paper mechanism, an electromechanical assembly which control the filter paper movement, is provided as an integral part of the detector unit.

Containment Radioactive Gas Monitor (R-12)

The monitor is provided to measure gaseous beta radioactivity in the containment to ensure that the radioactivity release rate during purging is maintained below specified limits. High gas radiation level initiates closure of the containment purge supply and exhaust duct valves and pressure line relief valves.

This channel takes a continuous air sample from the containment atmosphere, after it passes through the air particulate monitor (R-11), and draws the sample through a closed system to the gas monitor assemble. The sample is constantly circulated in the shielded, fixed, volume, where it is viewed by a plastic scintillator coupled to a heated photomultiplier tube. The sample is then returned to the containment.

The detector assembly is in a completely enclosed building. Lead shielding is provided to reduce the background level to a point where it does not interfere with the detector sensitivity. A preamplifier is mounted at the detector skid. Its output is transmitted to a microprocessor which converts the detector signal and analog outputs for display and communicates with the Radiation Monitoring System cabinet in the Central Control Room.

Plant Vent Gas Monitor (R-14)

The Plant Vent Gas Monitor detects radiation passing through the plant vent to the atmosphere. R-14 also acts as backup to R-27 as it can provide the automatic control functions to actuate diversion of the PAB exhaust through charcoal filters. It consists of a signal scintillator type detector that transmit a pulse signal to the control room.

Remote indication and annunciation are also provided on the Waste Disposal System control board. On high radiation level alarm the gas release valve in the Waste Disposal System is automatically closed, this assuring that gaseous releases from the Waste Disposal System are within the specified limits.

Condenser Air Ejector Gas Monitor (R-15)

This monitor meets the requirements of Regulatory Guide 1.97. The channel monitors the discharge from the air ejector exhaust header of the condensers for gaseous radiation which is indicative of a primary to secondary system leak. The normal gas discharge is routed to the turbine roof vent. On high radiation level alarm, the condenser exhaust gases are diverted to the Containment through a blower.

The steam jet ejectors and the primary ejectors normally exhaust to the atmosphere through a common vent stack outside the Turbine Hall. Radiation monitor channel R-15 is used to continuously monitor for high radiation in the steam jet vent line, thereby indicating a leak into the steam generator secondary water from the Reactor Coolant System.

An annunciator is provided to warn the operator of a high radiation condition. In case of a high radiation signal the following sequence of events will occur:

1) The normally closed containment isolation valves in the line to the containment building open and the control valves in the seal air line close. A selector switch and open-close lights are available for manual operation of the isolation valves.

2) The air ejector blower starts, and a three-way control valve in the air ejector effluent line diverts the effluent to the Containment Building after the blower has start.

3) A control valve to isolate the steam supply to the condenser priming ejectors is closed to prevent their operation. it should be noted that the control valve is reset locally after the high radiation signal is cleared.

The containment ventilation system provides means to manually limit, under normal conditions, containment pressure to 1 psig. However, 1 psig is not an operational limit.

If high pressure occurs in the blower suction, a pressure switch trips the blower and redirects the air ejector effluent back to the turbine hall vent.

A pressure controller and control valve are located in the steam supply to the steam jet ejectors. A solenoid valve is used to close the control valve in case of low flow through the steam jet air ejector condensers. An additional pressure control valve is located downstream of the first control valve to act as an overpressure shutoff. it is operated from a pilot valve which is set to close the valve if the pressure exceeds 175 psi.

A gamma sensitive Sodium Iodide (NaI) crystal scintillator, photomultiplier tube is used to monitor the gaseous radiation level. The radiation monitor consists of a 3" pipe section ubn series with the steam jet air ejector exhaust line, a thin walled sealed well (perpendicular to and penetrating the 3" pipe) which houses the Na/PM assembly, and employ lead shielding to reduce background radiation interference to an acceptable level.

Containment Fan Cooling Water Monitors (R-16A and R-16B)

The R-16A and R-16B channels monitor the containment fan cooling water for radiation indicative of a leak from the containment atmosphere into the cooling water. A small bypass flow form each of the heat exchangers is mixed in a common header and monitored by two adjacent-to-line scintillation detectors. Upon indication of a high radiation level each heat exchanger is individually sampled to determine which unit is leaking. This sampling sequence is achieved by manually selecting the desired unit to be monitored and allotting sufficient time for sample equilibrium to be established (approximately 15 to 20 minutes). Note that for a fan cooler unit (FCU) cooling coil failure, assumed to occur concurrently with a large break LOCA, radiological accessibility to identify and isolate the failed FCU will be possible prior to initiation of external recirculation.

Channels R-16A and R-16B use a photomultiplier tube-scintillation crystal (NaI) combination, mounted adjacent to line. Lead shielding is provided to reduce the background radiation level so it does not interfere with the detector's sensitivity. Indication and alarm are provided in the Central Control Room.

There is a 2" isolation valve and a 1/2" tap off line #12. Also, there is a 1/2" tap off line #11. These valves and taps will facilitate isolation, sampling, calibration, and urge functions. When lines #11 and #12 are isolated on either side of the radiation monitors R-16A and R-16B, using temporary hose or tubing and a portable sample pump, the radiation monitors can be calibrated then purged. These taps can also be used for local sampling in case of monitor failure.

In the event that the area of the detection assemblies for the R16-A and R16-B monitors is inaccessible (E.g., due to tornado impact on the west wall of the Vacco Filter Room, Flooding, etc.), then sampling may be achieved at the flow transmitters on the service water discharge of the fan cooler units or obtained upstream of the affected area per chemistry procedures.

The detector output signals are transmitted to a microprocessor which converts the detector signals to digital and analog outputs for display and communicates with the Radiation Monitoring Cabinet in the control room.

<u>Component Cooling Liquid Monitors (R-17A and R-17B)</u>

These channels continuously monitor the component cooling loop of the Auxiliary Coolant System for radiation indicative of a leak of reactor coolant from the Reactor Coolant System and/or the residual heat removal loop in the Auxiliary Coolant System. Each scintillation counter is located in an in-line well.

<u>Waste Disposal System Liquid Effluent Monitor (R-18)</u>

This detector monitors all Waste Disposal System liquid releases from the plant. Automatic valve closure action is initiated by this monitor to prevent further release when a high radiation level is indicated and alarmed. Remote indication and annunciation are also provided on the Waste Disposal System control board.

Channel R-18 uses a photomultiplier tube-scintillation crystal (NaI) combination, mounted in a sealed wall in an in-line fixed volume sample chamber unit for liquid effluent radiation detection. lead shielding is provided to reduce the background radiation level so it does not interfere with the detector's sensitivity. Indication and alarm are provided in the Control Room.
The sample chamber is positioned in the piping system to allow monitoring of the waste liquid during recirculation and discharge. Isolation, drain and sample valves are provided to allow purging to allow purging of the sample chamber with a clean water supply and calibration of the monitor.

The detector output signal is transmitted to a microprocessor which converts the detector signal to digital and analog outputs for display and communicates with the Radiation Monitoring Cabinet in the control room.

Steam Generator Liquid Sample Monitor (R-19)

This monitor meets the requirements of Regulatory Guide 1.97. The channel monitors the liquid phase of the secondary side of the steam generator for radiation, which would indicate a primary-to-secondary system leak, providing backup information to that of the condenser air removal gas monitor. Samples from the bottom of each of the four steam generators are mixed in a common header and the common sample is continuously monitored by one of two separate scintillation detectors. Upon indication of a high radiation level, sample and blowdown isolation valves and the blowdown tank spray valve close. Each steam generation is individually sampled in order to determine the source. This sampling sequence is achieved by manually selecting the desired unit to be monitored and allotting sufficient time for sample equilibrium to be established (approximately one minute).

The monitor is an open frame skid assembly which contains two (2) NaI(T1) Am-241 stabilized scintillation crystal photomultiplier tube detector assemblies. Each detector is housed in its own lead shielded sample chamber. The two detectors, one low range 10^{-6} to μCi/cc and one high range 10^{-3} to 10^{+2} μCi/cc are provided to meet the extended range requirements of Regulatory Guide 1.97. The monitor assembly also contains a pump assembly, sample heat exchanger, valves to direct the sample either to the low or high range detector assemblies, valves to allow purging of the sample chambers with clean water, calibration ports, and instrumentation to monitor sample flow, temperature and pressure.

The detector outputs are transmitted to a remotely mounted microprocessor which converts the detector outputs to digital and analog signals for display, generates alarms and communicates with the Central Control Room Radiation Monitoring Cabinet.

Waste Disposal System Gas Analyzer Monitor (R-20)

This channel monitors low-pressure radioactive gases in the suction line to the waste gas compressor. In effect, it measure the rate at which radioactivity is being introduced (via a common header) into a gas decay tank (large or small), and provides a means of ensuring that the accumulated radioactivity in the tank being filled does not exceed the technical specification limit. When the tank inventory limit is approached another tank is placed in service.

The monitor consists of a gamma sensitive Geiger Muller tube mounted adjacent to line. Lead shielding is provided to reduce the background radiation interference. The detector output is transmitted to a remotely located microprocessor which converts the detector output to digital and analog signals for display, generates alarms and communicates with the Control Room Radiation Monitoring Cabinet.

Radiation level indication and high radiation alarm are also provided at the Waste Disposal Panel.

Service Water from the Component Cooling Water Heat Exchanger Radiation Monitor (R-23)

The R-23 channel monitors the Service Water common return line from the Component Cooling Water Heat Exchanger for radiation indicative of a leak of Component Cooling Water into the Service Water discharge header. Upon detection of a high radiation level an alarm is actuated in the Control to alert the operators.

Channel R-23 uses a photomultiplier tube-scintillation crystal (NaI) combination detector, mounted adjacent to the common Service Water return line from the CCW Heat Exchanger. Lead shielding is provided to reduce the background radiation level so it does not interfere with the detector's sensitivity. Indication and alarm are provided in the Central Control Room.

Wide Range Plant Vent Gas Monitor (R-27)

This monitor meets the requirements of Regulatory Guide 1.l97. The detector monitors noble gas releases passing through the plant vent to the atmosphere. A wide range gas monitor is installed for this purpose. This system provides 4 channels of varying sensitivity. The lower range channel consists of an isokinetic sampling head connected by heat traced tubing to a sample conditioning module containing particulate and iodine filters. The sample then passes to the sample detection module which included a 2 cmf pump and a plastic scintillator radiation detector. The intermediate and high range detectors have a separate sampling system sized for isoknetic sampling at 0.6 cfm, including heat traced lines and shielded iodine and particulate filters. The detectors used for this portion of the system are CdTe (Cs) directly coupled to a 30 cm3 and 0.03 cm3 gas volume for the intermediate and high range detectors respectively. Both isolinetic sampling heads are located in the plant vent at elevation 164' and heat tracing maintains the temperature of the sample air streams between 80°F and 110°F depending on outside air temperature. A microprocessor controls the sample flow rate and which filter and detector channel are used as well as computing and displaying release information.

There are three ranges of indication with a minimum of one decade overlap between ranges. On high radiation alarm, gas release valve RCV-014, the containment purge supply, purge exhaust, and pressure relief isolation valves are automatically closed and PAB ventilation is diverted through charcoal filters.

Indication is given by recorders installed in the control room Radiation Monitoring System Cabinet. The RM-23A Read Out/Control Module, also located in the Radiation Monitoring System Cabinet, gives additional control/indication of system parameters. A RM-80 micro-processor is located at elevation of 36' in the control building. it has the ability of storing past activity rates and provides automatic control of the system. There is a flow transmitter and RTD in the plant vent at elevation 164', which monitor parameters necessary for operation of the M-80. The sample conditioner and detection skids are located in the purge valve enclosure at elevation 79'.

Control Room Noble Gas Monitor (R-33)

This monitor measure gaseous beta radioactivity in the Control Room environs. The R-33 sample skid, RM-80 microprocessor and Customer Interface Junction Box (CIJB) are located in the Control Building 33 elevation. The monitoring system consists of the inlet and exhaust samples lines, sample skid, RM-80 CIJB, communications isolation junction boxes, Recorder RR- 1/33, and a RM-2A Readout/Control module.

A skid mounted diaphragm pump draws a continuous air sample from the sample probe located in the overhead above the control room operators desk and exhausts back into control room overhead. The sample is constantly circulated through the shielded, fixed volume where it is viewed by photomultiplier tube – scintillation crystal (NaI) combination. A preamplifier transmits the detector pulse signal to the RM-80 microprocessor which converts the signal to digital and analog outputs. The RM-80 also provides rate meter indication and sample skid control. The RM-80 microprocessor and adjoining CIJB provide communications between the gas sample skid and the CCR RMS Cabinet. The RM-23A Readout/Control module in the RMS Cabinet in the CCR provides remote control and indication. The Control Room RMS Cabinet communicates with the RM-80 microprocessor and displays various parameters on a flat screen located in the RMS Cabinet. The RMS performs its computer functions on the Plant Integrated Computer System (PICS) which communicates with the RMS system by a fiber optic cable. A four pen (two spare) digital strip chart Radiation Recorder RR-1/33 provides continuous trending and indication of the Control Room activity levels. The R-33 high radiation alarm actuates locally at the sample skid and on the Radiation Monitor Cabinet Annunciator Panel in the Control Room.

Auxiliary Condensate Return Activity Monitor (R-37)

This is a scintillation type detector, which monitors the auxiliary condensate radioactivity. The readout is on the 65' PAB WDS Panel and the control room receives an alarm.

Administration Building Exhaust Monitor (R-46)

This detector monitors airborne radioactivity content of the Administration Building exhaust air. The monitor is a scintillation detector and the readout is located on the panel on the fourth floor of the Administration Building and it will alarm in the Control Room. R-46 meets the requirements of Regulatory Guide 1.97 and monitors the gaseous activity.

Sewage Pipe Line Monitors (R56C)

One adjacent to line type detector is utilized to monitor the sanitary waste effluent discharges coming from Indian Point Units 1, 2 and 3. The detector is photomultiplier scintillation type.

The output of the detector is continuously monitored by a microprocessor which provides indication and alarms. A high radiation alarm is provided in the Control Room on the Radiation Monitoring System Cabinet. On detection of high high radiation levels a diverter valve is positioned to transfer discharge flow to waste holding tanks.

Channels R-56A and R-56B have been retired in place.

Radioactive Machine Shop (RAMs) Monitor (R-59)

The monitor is an open frame skid assembly. The detector is a 2 inch x 0.01 inch beta sensitive phosphor and a PM tube. Included on the skid are a rotometer flowmeter and a vacuum indicator. Air flow of 2 scfm is obtained by means of a diaphragm pump. The monitor has a microprocessor with a local control unit. readouts are provided in the Control Room and the 55' PAB. R-59 meets the requirements of Regulatory Guide 1.97.

Condensate Polisher Overboard Monitor (R-61)

This channel monitors liquid radioactivity of discharges from the LTDS or HTDS Waste Collection. Automatic valve closure is initiated by this monitor to prevent further release after a high radiation level is indicated and alarmed. This liquid monitor is part of a skid mounted, microcomputer-controlled offline sampling system containing a microprocessor, gamma sensitive scintillation detector, valves, control station and a flow switch. The detector output is transmitted to the microprocessor which converts the detector signal into digital and analog outputs for display, generates alarms and communicates with the Control Room Radiation Monitoring Cabinet. Alarms are provided in the Control Room and locally in the Condensate Polisher. The alarm trip setpoint for this process radiation monitor is established in accordance with the Indian Point 3 Offsite Dose Calculation Manual. The trip setpoint ensures that the offsite radioactive releases are kept within 10 CFR 20 limits. Manual operation of a reset switch is required to reopen the discharge valves.

Main Steam Monitors (R-62 A-D)

Four radiation detectors are externally mounted next to the main steam lines outside the containment and upstream of the safety valves. These channels monitor the noble gases released through the main steam line safety valves and atmospheric dump valves during normal and accident plant operation. Local indications/alarms are in the upper cable tunnel penetration areas as well as the control room.

The detection channels for the R-62 radiation monitor are designed to meet the range requirements of NUREG-0737. The range of these channels ($7.66E^{-03}$ to $7.66E^{+02}$ μCi/cc of ODCM instantaneous release mix) complies with a NUREG-0737 required range of $1.00E^{-01}$ to 1.00^{+0} μCi/cc of Xe-133 dose equivalent radioactivity in that the Xe-133 equivalent range for monitor R-62 is $6.04E^{-02}$ to $6.04E^{+03}$ μCi/cc.

The R-62 channels meet the intent of the range requirement imposed by Regulatory Guide 1.97 (i.e. 10^1 μCi/cc to 10^3 μCi/cc) in that they provide accurate monitoring of any radioactive releases through the main steam lines for the maximum steam line activity concentrations following a postulated design basis steam generator tube rupture accident. The actual detection range of the monitor has a low limit that is below the low limit of the range requirement imposed by Regulatory Guide 1.97 and a high limit that is above the maximum concentration of noble gases expected in the main steam lines on a design basis steam generator tube rupture accident. This range ($7.66E^{-03}$ to $7.66E^{+02}$ μCi/cc of ODCM instantaneous release mix) is displayed on the RM-23L digital display at the RM-80 microprocessor and on the RM-23A controller for the radiation monitor. The scales for the analog output from the monitor to the analog alarm and indication assemblies, the recorders and the QSPDS are $1.00E^{-03}$ to $1.00E^{+03}$ μCi/cc which bound the actual detection range.

Each of the Geiger-Mueller tube detectors is mounted in a lead shield to minimize the effect of background radiation. Each detector transmits its output signal to a common microprocessor which converts the detector outputs to digital and analog signals for display locally and communicates with a radiation monitor controller located in the control room. This controller has a digital display of radiation levels for the operator which provides output signals for strip chart recorders and the Qualified Safety Parameter Display System (QSPDS).

Gross Failed Fuel Detector (R-63A, R-63B)

The Gross Failed Fuel Detector (GFFD) is based on the principle of measuring gamma radiation from fission products in the primary coolant after having allowed decay of the seven second half-life N-16.

Delay time is obtained by the length of tubing from the core to the detector. Piping to the detector is connected to the hot leg of the reactor coolant loop (Figure 11.2-6). The fluid passes a sample cooler before it reaches the two redundant detectors. The fluid passes through a flow meter and flow controller before draining into the volume control tank. The proper delay time (about 60 seconds) to the detector can be adjusted by regulating the rate of water flow. Figure 11.2-7 shows the block diagram of the GFFD.

No reactor limitations are imposed based on operability of this detector. The recommended operator action in conjunction with the use of the gross failed fuel detector are as follows:

1) Log the gross failed fuel detector reading once per shift and report any unusual count rate increase to the shift manager.
2) Have chemistry samples taken if the concentration exceed 5 μCi/cc. This change is indicative of some possible fuel element failures occurring.

Operational requirements relating to the GFFD are included in the Technical Specifications. These monitors meet the requirements of Regulatory 1.97.

Design Containment Equilibrium Activities

During normal plant operations, Radiation Monitoring Systems Channels R-11 and R-12 provide continuous indications of the containment atmosphere gross air particulate activity and gross gaseous activity, respectively. Backup monitoring during purging is provided by Radiation Monitoring system Channels, R-14, plant vent gas monitor and R-27, plant vent wide range gas monitor. Prior to either containment purge or pressure relieving operations, containment air samples are obtained and analyzed for both particulate and gaseous activities. Table 11.2-8 lists the anticipated design equilibrium containment activities following a 16-hour operation of the containment recirculation filtration system at an iodine removal efficiency of 99%. The operating basis reactor coolant leakage into the containment of 14.4 gal/day and reactor operation with 0.2% equivalent fuel defects are assumed. Table 11.2-9 shows calculated containment activities after recirculation filtration for 16 hours at a conservative iodine removal efficiency of 90% and assuming 50 lb/day reactor coolant leak rate into the containment and 1% fuel defects.

The tritium level in the reactor coolant is monitored weekly, not to exceed 10 days between analyses. Measures are taken to ensure that during refueling, tritium activity in the refueling water is less than 1 μCi/cc. With containment purge at an assumed rate of 10,000 cfm, the maximum concentration of tritium in the containment air was calculated to be less than 1/5 of DAC.

The basis for this concentration was determined from the assumption that the refueling water evaporation rate is 100 lb/hr, the containment is purged for 2 hours at an assumed rate of 10,000 cfm prior to access, and the purge continues during the refueling operation at an assumed rate of 10,000 cfm. The containment purge isolation valves will be shut prior to going above cold shutdown to ensure closing against accident conditions.

During normal plant operation, grab samples from the auxiliary building areas are analyzed for tritium as required in the Radiological Effluent Controls section of the ODCM.

During normal operation, grab samples from the containment building are analyzed for tritium as required by 10 CFR 20 for personnel protection.

Monitoring of Radioactivity Discharges

During normal plant operation all liquids discharged from the nuclear steam supply systems of the plant are released via the Waste Disposal System. Prior to discharging from the plant, samples are taken from the monitor tanks for isotopic analysis. In addition, all liquids discharged from the Waste Disposal System, are monitored by the Waste Disposal System Liquid Effluent Monitor R-18. This monitor provides automatic closure of flow control valve RCV-019 to assure discharges of less than 10 CFR 20 limits.

Proper operation of the monitor is assured by utilization of its check source and by comparison of the monitor reading to the monitor tank sample analysis. This sample analysis is taken to establish activity in the liquid to be discharged prior to its release from the plant.

Monitoring for the occurrence of primary to secondary leakage is provided by both the Condenser Air Ejector Monitor R-15 and the Steam Generator Blowdown Sample Monitor R-19. Upon indication of leakage by either of these monitors, means have been provided to manually divert the blowdown from Indian Point 3 to the Indian Point 1 Secondary Boiler Blowdown Purification System (SBBPS). From the standpoint of rapid determination of the occurrence of primary to secondary leakage, the two monitors (R-15 and R-19) provide redundancy for this function.

Proper operation of R-19 is assured by utilization of its check source and by comparison of the monitor reading to a blowdown liquid sample analysis. During those periods of operation with primary to secondary leakage and utilization of the Indian Point 1 SBBPS, the blowdown liquid is also monitored by the radiation monitor provided for the Indian Point 1 SBBPS before release to the environment.

During those periods of plant operation with primary to secondary leakage, monitoring for the subsequent occurrence of radioactivity from such leakage in the Condensate Polishing Facility waste effluent is provided by radiation detector R-61. Automatic valve closure is initiated by this monitor to prevent further release after a high radiation level is indicated and alarmed. Readout and alarm are in the Control Room and a local alarm is provided in the Condensate Polisher.

The release rate of radioactive liquid effluents from the site must be such that the concentration of radionuclides from the circulating water discharge does not exceed the limits specified in 10 CFR 20, Appendix B, for unrestricted area.

Waste Disposal Processes

The Waste Disposal System for Indian Point 3 is described in Section 11.1. Performance data are given in Table 11.1-1.

The Indian Point 3 liquid releases include discharges from the Waste Disposal System, steam generator blowdown, and Steam and Power Conversion System liquid leakage.

A manually operated intertie is provided from the Indian Point 3 steam generator blowdown to the Indian Point 1 Secondary Boiler Blowdown Purification System (SBBPS).

The radio iodine releases from the blowdown tank vent line are estimated using partition factors from Regulatory Guide 1.42. The radio iodine release is assumed to be 5% of the radioiodine activity released from the Steam Generator Blowdown when directed to the Blowdown Flash Tank.

The Indian Point 3 gaseous releases include pressure relief operations, reactor coolant leakage in the Primary Auxiliary Building, discharges from the Waste Disposal System, steam generator blowdown, and secondary system releases from the main condenser air ejector, the gland seal condenser and Steam and Power Conversion System steam and liquid leakage. Offsite doses from gaseous tritium releases are negligible.

Plant equipment is used in conjunction with developed operating procedure to maintain surveillance of radioactive gaseous and liquid effluents produced during normal reactor operations and expected operational occurrences in an effort to maintain radioactive releases to unrestricted areas as low as practicable.

The release rate of radioactive liquid effluents from the site must be such that the concentration of radionuclides in the circulating water discharge does not exceed 10 times the limits specified in 10 CFR 20, Appendix B, for unrestricted areas. Prior to release of effluents from the radwaste system of either Indian Point 3 or Indian Point 1, a sample is taken and analyzed to provide the data necessary to assure compliance with these limits.

The release rate of gaseous effluents is limited by the Technical Specifications. The contents of the gas holdup tanks are sampled and analyzed prior to release to provide the necessary data to assure compliance with this limit. During release of gaseous effluent to the plant, the conditions stated in the Technical Specifications must be met. The inventory of noble gases in any gas tank are also limited by the Technical Specifications.

During power operation the air ejector discharge monitor may be inoperable for 48 hours. when the monitor is inoperable, samples are taken from the air ejector discharge and analyzed for gross activity on a daily basis, except that when there is indication of primary to secondary leakage, the sample is taken and analyzed for gross activity once per shift.

During the first indication of primary to secondary leakage, the partition factor for the blowdown tank, as established by the Indian Point 3 “Offsite Dose Calculation Manual”, is used. Whenever there is indication of primary to secondary leakage and any steam generator is being blow down, the blowdown line monitor must be operable. It may be inoperable. It may be inoperable for 48 hours, provided samples are taken once per shift of the blowdown effluent and analyzed for gross activity.

The discharge rate of noble gases to the plant vent from gas decay tanks are controlled by an adjustable control valve and a pressure reducing valve.

Gaseous releases from the plant vent are monitored by means of radiogas detectors. Containment atmosphere is separately monitored. On high activity in the plant vent, the monitor initiates an isolation of the containment atmosphere, and no discharge to the plant vent from this source can occur.

Basis for Monitor Trip Points

Alarm trip points for Process Radiation System monitors are established in accordance with the Indian Point 3 “Offsite Dose Calculation Manual”. These trip points ensure that offsite radioactive releases are kept within regulatory limits.

Radio Nuclides in Steam Generator Blowdown

The non-gaseous isotopic radioactive concentration in the steam generator blowdown liquid is conservatively calculated using the following set of equations:

$$S_p \text{ (uc/gm)} = (L)(F)(A_p)(1-e^{-(\lambda+B/M)t}) / M\lambda + B/M)$$
$$S_d \text{ (uc/gm)} = S_p + (L)(F)(A_d)(1-e^{-(\lambda+B/M)t}) / M\lambda + B/M)$$

where:

- S = Radioactive isotopic concentration in blowdown liquid – uc/gm (NOTE: Subscript p refers to parent isotopes, subscript d refers to daughter isotopes)
- L = Primary to secondary leak rate – gm/sec
- F = Ratio of actual percent fuel rod defects to the design defect level of 1% - dimensionless
- A = Radioactive isotopic concentration in the reactor coolant (see Chapter 9) – uc / gm
- λ = Isotope decay constant – sec^{-1}
- B = Blowdown Rate (Total 4 steam generators) – gm/sec
- M = Mass of secondary water in steam generators – gm

The noble gas fission product activity released from the secondary plant during operation with primary to secondary leakage is conservatively assumed to be released via the condenser air ejector. The noble gas release rate is calculated as follows:

$$Q = \text{(us/sec)} = (L)(F)(A)$$

where:

- Q = Noble gas activity release rate – uc/sec
- L = Primary to secondary leak rate – gm/sec
- F = Ratio of actual percent fuel rod defects to the design defect level of 1% - dimensionless
- A = Radioactive noble gas isotopic concentration in the reactor coolant (See Chapter 9) – uc/gm

Tables 11.2-10 and 11.2-11 provide the isotopic activity in the blowdown liquid and the noble gas activity emission rate from the air ejector as a function of primary to secondary leak rate. Table 11.2-10 is based on operation with 0.2% equivalent fuel rod defects and Table 11.2-11 is based on operation with 1% equivalent fuel rod defects.

Pertinent assumptions to be used in the calculations include:

1) Total mass of secondary water in steam generators (M) – 1.46×10^6 gms
2) Continuous blowdown rate (total 4 steam generators) (B) = 3.16×10^3 gm/sec

Tables Table 11.2-12 and 11.2-13 provide the noble gas isotopic concentration in the condenser air ejector discharge as a function of primary to secondary leak rate. Table 11.2-12 is based on expected operation with 0.2% equivalent fuel rod defects while Table 11.2-13 is based on design defect level of 1% equivalent fuel rod defects.

The noble gas isotopic concentrations in the condenser air ejector discharge were based on the air ejector's maximum discharge rate of 60 SCFM. In actual operation, the air ejector discharge rate will be less than 60 SCFM and the radiation monitor response time will be shorter.

Radiation monitoring channel R-15 (condenser air ejector gas monitor) has been provided to ensure that the noble gas radioactive releases from secondary plant are less than the 10 CFR 20 offsite discharge limits. Radiation monitoring channel R-19 (Steam Generator Secondary Side Liquid Monitor) has been provided to ensure the radioactive blowdown liquid releases from the secondary plant are less than 10 times the 10 CFR 20 “Effluent Concentrations”. Tables 11.2-14 and 11.2-15 provide the radiation monitor responses corresponding to the radioactivity concentration listed in Tables 11.2-10 and through Table 11.2-13.

11.2.3.2 Area Radiation Monitoring System

This system consists of channels which monitor radiation levels in various areas of the plant. These areas are listed in Table 11.27B and the measurement ranges are given in Table 11.2-7.

Control Room Area Radiation Monitor (R-1)

This monitor measure the area radiation in the Control Room and satisfies the requirements of Reg. Guide 1.97. The monitor consists of a detector, RM-80 microprocessor and Customer Interface Junction Box (box located in the Control Building 33' elevation) RM-80 CIJB, Communications Isolation Junction Boxes, recorder RR-1/33, and a RM-23A Readout Control module.

The detector is a fixed position gamma sensitive G-M tube located on the north wall on the Control room (Control Building, 53' elevation). A preamplifier transmits the detector pulse signal to the RM-80 microprocessor which converts the signal to digital and analog outputs and also provides local rate meter indication. The RM-80 microprocessor and adjoining CIJB communicate with the Control Room RMS Cabinet. An RM-23A Readout/Control module in the RMS Cabinet in the Control Room provides remote control and indication.

The RMS [Deleted] communicates with the RM-80 microprocessor and displays various parameters on a flat screen located in the RMS Cabinet. The RMS system performs its computer functions on the Plant Integrated Computer System (PICS) which communicates with the RMS system by a fiber optic cable. A two channel Strip Chart Radiation Recorder (RR-1/33) located on the RMS Cabinet provides continuous trending and indication of CCR area radiation levels. The CCR high radiation alarm actuates locally at the RM-80/CIJB and on the Radiation Monitoring Cabinet Annunciator located on the RMS Cabinet in the CCR. The high radiation alarm also automatically transfers the CCR HVAC system into the 10% Incident Mode of Operation.

Area Monitors (R-2, R-4, R-6, R-7, R-8)

Each channel consists of a fixed gamma sensitive GM tube. The detector output is amplified and the log count rate determined by the integral amplifier at the detector. The level is indicated locally at the detector and at the Radiation Monitoring System cabinets. High radiation alarms are displayed on the main annunciator the Radiation Monitoring System cabinets, and at the detector location. 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. Monitors R-4, R-6, R-7 and R-8 meet the requirements of Regulatory Guide 1.l97

Fuel Storage Building Area Radiation Monitor (R-5)

This is an extended range area monitor used to measure the area radiation fields of the Fuel Storage Building as required by Reg. Guide 1.97. It uses a GM tube detector for low range and an ionization chamber detector for high range. On a high radiation signal the bypass dampers around the charcoal filter must be manually closed, if open, the Fuel Storage Building rollup coiling truck bay door closes, the supply fans will trip, if running, and the exhaust fan is started. The inlet dampers to the charcoal filter will open. The personnel doors' inflatable seals will inflate, but this action is not required for R-5 operability.

Channel R-5 consists of 2 detectors, local indicator, annunciator, and microprocessor. The Radiation Monitoring System in the Control Room communicates with the microprocessor and displays various parameters on a flat screen.

Vapor Containment (VC) High Radiation Area Monitors (R-25, R-26)

These redundant monitors meet the requirements of Regulatory Guide 1.97 and are used to measure the area radiation fields in the VC. They can be used to follow the course of an accident by indicating the extent of gaseous and vapor fission products released from the primary system. These monitors consist of ion chamber detectors, local indicators, annunciator, and microprocessor. The Radiation Monitoring System in the Control Room communicates with each microprocessor and displays various parameters on a flat screen.

CVCS Tank Area Radiation Monitors (R-34A, R34B, R-34C)

These are ionization chamber type detectors which monitor the tank area radiation levels for CVCS Tank #31, 32 and 33, respectively. The readout of these detectors is in the Control Room.

Waste Holdup Tank (WHUT) Area Radiation Monitors (R-38A, R-38B, R-38C)

These are ionization type detectors which monitor the WHUT area radiation levels for WHUT #31, 32 and 33 respectively. The readout of these detectors is in the Control Room.

WHUT Pump Room Area Radiation Monitor (R-38D)

This is an ionization type detector which monitors the area radiation level in the WHUT Pump Room. The readout is in Rack D11 of the Control Room.

Administration Building Area Radiation Monitors (R-48 A-G, R-49, R-51)

These are all GM type detectors which readout locally and on the Victoreen Panel on the fourth floor of the Administration Building. They monitor the radiation levels in their particular areas. R-48D, R-48E, R-48F, R-49 and R-51 have been removed from service.

Radioactive Machine Shop (RAMS) Area Radiation Monitors (R-53 A-C, R-54, R-55 A-B)

These are all GM type detectors which read out locally and on the Victoreen Panel on the fourth floor of the Administration Building. They monitor the radiation levels in these particular areas. R-53A, R-53B, R-54A, R-54C and R-55A have been removed from service.

Area Monitors (R-64, R-65, R-66, R-67, R-68, R-69, R-70)

These are extended range area monitors used to measure the area radiation fields of the PAB, Fan House and Pipe Penetration Areas as required by Regulatory Guide 1.97. They utilize a GM tube detector for low range and an ionization chamber detector for high range. The monitors also have a local indicator, annunciator, and a microprocessor. The Radiation Monitoring System communicates with each microprocessor and displays various parameters on a flat screen.

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.

Calibration

A primary calibration was performed on a one time basis in the vendor's "Design Verification Tests", which utilizes typical isotopes of interest to determine proper detector response. Further primary calibrations are not required as the geometry cannot be significantly altered within the sampler. Calibration of samplers is then performed based on a known correlation between the detector responses and a secondary standard.

Maximum Offsite Concentrations During Venting

The design basis maximum concentrations of activity at the site boundary resulting from venting of the gas decay tanks correspond to 20% of MPC based on annual average meteorology and a ground level release. This site boundary concentration is based on venting at a rate which will alarm the radiation monitor in the vent line and automatically close the vent valve thereby terminating the venting process. However, during the venting process, the average site boundary concentrations is much less than this. Based on the maximum release rate and 1% defective fuel, venting of the gas decay tanks occurs less than 30% of the time in one year. However, based on the expected level of fuel defects corresponding to 0.2%, the gas decay tanks would only be vented 5% of the time throughout the year.

The Containment was expected to be purged four times per year; three purges during hot shutdown conditions and the other during the refueling shutdown. Presently, Containment purging at Indian Point 3 occurs only during cold shutdown. During the refueling cold shutdown, the major portion of the activity release is expected to occur in the first 2.5 hours. The offsite concentrations of radioactivity as a result of purging the Indian Point 3 containment is less than the effluent release limit in the Technical Specifications. The Containment Radioactive Gas Monitor (R-12) and Containment Air Particulate Monitor (R-11) can initiate automatic closure of the containment purge lines in order to assure that this limit not be exceeded. The containment purge valves will be shut prior to going above cold shutdown to ensure closure against accident pressure conditions. The circuit arrangement is such that the purge and pressure relief valves close upon a high radiation signal.

As all venting concentrations at the site boundary are below 10 CFR 20 MPC values, no limitations on releases need to be imposed by meteorological considerations.

<u>Purge and Vent During Normal Operation</u>

During normal reactor operations a “closed containment” is maintained. Containment purging normally occurs only during cold shutdown conditions.

It may be necessary, however, to provide containment pressure relief during normal operation. The flow rate and time period associated with the pressure relief are much less than that associated with the normal purging operation. Review of plant operating data for Indian Point 3 indicates that venting of the containment is necessary once every two days for approximately one and one-half hours.

Prior to either containment purge or pressure relieving operations, containment air is sampled and analyzed for both particulate and gaseous activities.

A high radiation signal from either the Containment Air Particulate Monitor or the Containment Radioactive Gas Monitor initiates automatic closure of the containment supply and exhaust duct valves and pressure relief line valves. Both these monitors would be in operation during containment purging and venting operations. The monitors are located a few feet from the containment wall in the fan house at an elevation of 54'-9".

The Containment Radioactive Gas Monitor and the Plant Vent Gas Monitor would both detect the radiation levels that would result from a fuel handling accident inside the Containment. High radiation level for the Containment Radioactive Gas Monitor initiates automatic closure of the containment purge supply and exhaust duct valves and pressure relief line valves.

Refer to the Technical Specifications and the Radiological Effluent Controls Program for applicable requirements relating to operability and periodic testing of these monitors plus related limitations placed on containment venting operations.

11.2.4 Health Physics Program

The Indian Point health physics program, medical emergency program and emergency plan are described in the “Indian Point Energy Center Emergency Preparedness Program” and the Indian Point 3 “Radiation Protection Plan.”

11.2.5 Liquid Waste Release

All liquid waste releases are assayed for radioactivity prior to release to assure compliance with the limits established in the Technical Specifications.

11.2.6 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%, 50% and 100%, at rated full power. Survey data were reviewed for conformance to design levels before increasing to the next power range.

The Off-Site Dose Calculation Manual (ODCM) specifies surveillance requirements for Technical Specification required radiation monitors. The Technical Specification required effluent monitors are tested with calibrated sources at the designated calibration frequency and are tested daily using a remotely operated check source to verify the instrument response.

11.2.7 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 11.2.7.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 11.2.7.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 microcurie of radioactive material on the test sample.

D. If the leakage test reveals the presence of 0.005 microcurie 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.

TABLE 11.2-1

PLANT ZONE CLASSIFICATIONS

Zone	Condition of Occupancy	Maximum Dose Rate (1% failed fuel) m Rem/hr
0	Unrestricted Area	Less than 0.5
I	Restricted Area	0.5-2.0
II	Low Radiation Area	2.0-5.0
III	Radiation Area	5.0-100
IV	High Radiation Area	Greater than 100
V	Exclusion Area*	Greater than 1000

*NOTE: Access to Zone V areas must be cleared with the reactor operators in the Control Room, or is controlled by watch HP.

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	7.4×10^8	3.9×10^2
5.53 Kev < E ≤ 1 Mev	1.3×10^{10}	8.9×10^2
0.625 ev ≤ E ≤ 5.53 Kev	7.4×10^9	1.6×10^3
E < 0.625 ev	1.9×10^9	1.3×10^5

Design Parameters

Core thermal power	3216 MW(t)
Active core height	144 in
Effective core diameter	132.7 in
Baffle wall thickness	1.125 in
Barrel wall thickness	2.285 in
Thermal shield wall thickness	2.80 in
Reactor vessel I.D.	173.0 in
Reactor vessel wall thickness	8.625 in
Reactor coolant cold leg temperature	542 F
Reactor coolant hot leg temperature	601 F
Maximum thermal neutron flux exiting primary concrete	$< 10^6$ n/cm^2 sec
Reactor shutdown dose exiting primary concrete	<15 mr/hr

TABLE 11.2-3

SECONDARY SHIELD DESIGN PARAMETERS

Core power density	98.5 watt/cm^3
Reactor coolant liquid volume	12,600 ft^3
Reactor coolant transit times:	
Core	0.817 sec
Core exit to steam generator inlet	2.001 sec
Steam generator inlet channel	0.592 sec
Steam generator tubes	3.220 sec
Steam generator tubes to vessel inlet	2.758 sec
Vessel inlet to core	2.167 sec
Total out of core	10.738 sec
Full power dose rate outside secondary shield	<0.75 mr/hr

TABLE 11.2-4

ACCIDENT SHIELD DESIGN PARAMETERS

Core thermal power	3216 Mw(t)
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
Clean-up rate following accident	0
Maximum integrated direct dose (one week exposure) in the control room	<1.5 rem
Maximum integrated direct dose (one week 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 mr/hr
Maximum dose rate at water surface	2.0 mr/hr

TABLE 11.2-6

PRINCIPAL AUXILIARY SHIELDING

Component	**Concrete Shield Thickness. Ft-In**
Demineralizers	4 – 0
Charging pumps	2 – 6
Liquid waste holdup tanks	2 – 6
Volume control tank	3 – 6
Reactor Coolant filter	3 – 6
Gas decay tanks	3 – 6
Gas Compressor	2 – 0

Design parameters for the auxiliary shielding include:

Core thermal power	3216 MW(t)
Fraction of fuel rods containing small clad defects	0.01
Reactor coolant liquid volume	12,600 ft^3
Letdown flow (normal purification)	75 pgm
Effective cesium purification flow	7 gpm
Cut-in concentration deborating demineralizer	150 gpm
Dose rate outside auxiliary building	0.75 mr/hr
Dose rate in the building outside shield walls	0.75 mr/hr

TABLE 11.2-7

RADIATION MONITORING SYSTEM CHANNEL RANGES

Channel	Range*	Units	Nuclides Detected
R-1	0.1 - 1E+4	mR/hr	NG
R-2	0.1 - 1E+4	mR/hr	NG, FP, CP
R-4	0.1 - 1E+4	mR/hr	CP
R-5	1E-4 - 1E+4	R/hr	FP, CP
R-6	0.1 - 1E+4	mR/hr	FP, CP
R-7	0.1 - 1E+4	mR/hr	NG, FP, CP
R-8	0.1 - 1E+4	mR/hr	FP, CP
R-11	1E-11 – 1E-5	μCi/cc	Rb-88, FP CP
R-12	1E-7 – 1E-1	μCi/cc	NG
R-14	1E-6 to 1E-1	μCi/cc	NG
R-15	!e-6 – 1E0	μCi/cc	NG
R-16A, B	1E-7 –1E-1	μCi/ml	NG, FP, CP
R-17A, B	1E-6 to 1E-1	μCi/cc	NG, FP, CP
R-18	1E-7 – 1E-1	μCi/ml	FP, CP
R-19	1E-6 – 1E+2	μCi/ml	NG, FP, CP
R-20	1E-2 – 1E+3	μCi/cc	NG
R-23	1E-7 – 1E-1	μCi/cc	NG, FP, CP
R-25, 26	1 – 1E+8	R/hr	NG
R-27	10 – 1E+13	μCi/sec	NG
R-33	10 – 1E+8	cpm	NG
R-34A, B, C	0.1 – 1E+7	mR/hr	FP, CP
R-37	10 – 1E+6	cpm	FP, CP
R-38A, B, C, D	0.1 – 1E+7	mR/hr	FP, CP

NG = Noble Gases, e.g., Xe-133, Xe-135, Kr-87, Kr-88
FP = Fission Products, e.g., Cs-137, Cs-134
CP = Corrosion Products, e.g., Co-60, Co-58, Mn-54, Cr-51
*Range when stated in units of activity is a function of detection system counting rate range and the count rate to activity conversion factors associated with the expected radio nuclide mix present in the sampled medium.

TABLE 11.2-7
(Cont.)

RADIATION MONITORING SYSTEM CHANNEL RANGES

Channel	Range*	Units	Nuclides Detected
R-46	10 - 1E+6	cpm	NG
R-48A – C, G	0.01 - 1E+3	mR/hr	FP, CP
R-53C	0.01 - 1E+3	mR/hr	FP, CP
R-54B	0.01 - 1E+3	mR/hr	FP, CP
R-55B	0.01 - 1E+3	mR/hr	FP, CP
R-56C	1E-7 – 1E-1	μCi/cc	FP, CP
R-59	1E-6 – 1E+2	μCi/cc	NG
R-61	1E-7 - 1E-1	μCi/cc	NG, FP, CP
R-62A – D	7.66E-03 to 7.66E+02	μCi/cc	NG, ODCM Mix (digital range)
	1.00E-03 to 1.00E+03	μCi/cc	NG, IDCM Mix (analog range)
R-63A	1-2E+4	μCi/cc	NG
R-63B	1-2E+4	μCi/ml	NG
R-64	0.1-1E+7	mR/hr	CP, FP, NG
R-65	0.1-1E+7	mR/hr	CP, FR, NG
R-66	0.1-1E+7	mR/hr	CP, FR, NG
R-67	0.1-1E+7	mR/hr	CP, FR, NG
R-68	0.1– 1E+7	mR/hr	CP, FR, NG
R-69	0.1– 1E+7	mR/hr	CP, FR, NG
R-70	0.1 – 1E+7	mR/hr	CP, FR, NG

NG = Noble Gases, e.g., Xe-133, Xe-135, Kr-87, Kr-88
FP = Fission Products, e.g., Cs-137, Cs-134
CP = Corrosion Products, e.g., Co-60, Co-58, Mn-54, Cr-51
Iodines, e.g., I-131, 1-133, 1-135
*Range when stated in units of activity is a function of detection system counting rate range and the count rate to activity conversion factors associated with the expected radio nuclide mix present in the sampled medium.

TABLE 11.2-7A

PROCESS RADIATION MONITORING SYSTEM

I. Gaseous Radiation Monitoring

Channel	Monitor Location
R-12	Fan House - 54'-9'
R-14	Plant Vent -124'
R-15	Turbine Bldg - 53'
R-20	Primary Auxiliary Bldg - 55'
R-27	Purge Valve - 80'
R-33	Control Bldg - 33' (Cable Spreading Room)
R-46	Administration Bldg – 4th floor
R-59	RAMS - 55'
R-62A	ABFB
R-62B	ABFB
R62-C	ABFB
R62-D	ABFB

II. Particulate Radiation Monitoring

Channel	Monitor Location
R-11	Fan House - 54'-9'
R-13	Pipe Pen -67' (RETIRED)

TABLE 11.2-7A
(Cont.)

PROCESS RADIATION MONITORING SYSTEM

IV. Radiation Monitoring

Channel	Monitor Location
R-16A	Primary Auxiliary Bldg - 15'
R-16B	Primary Auxiliary Bldg - 15'
R-17A	Primary Auxiliary Bldg - 41' (CCW Heat Exchanger Discharge)
R-17B	Primary Auxiliary Bldg - 41' (CCW Heat Exchanger Discharge)
R-18	Primary Auxiliary Bldg - 41'
R-19	Service Water Access - 43'
R-23	Primary Auxiliary Bldg - 41'
R-37	Auxiliary Condensate Return
[Deleted]	
[Deleted]	
R-56C*	Sewage Life Station – Middle Parking Lot
R-61	CPF – 1st Fl. mezz.
R-63A	Pipe Pen - 67'
R-63B	Pipe Pen - 67'

* Channels R-56A and R-56B have been retired in place

TABLE 11.2-7B

Area Radiation Monitoring System

Channel	Detector Location
R-1	Control Room – 33”
R-2	Vapor Containment - 80'
R-4	Charging Pump Room
R-5	Fuel Storage Building - 95'
R-6	Chemistry Sampling Room
R-7	In-Core Instrumentation Room
R-8	Drumming Station
R-25	Vapor Containment - 95' West
R-26	Vapor Containment - 95' East
R-34A	31 CVCS Tank Area
R-34B	32 CVCS Tanks Area
R-34C	33 CVCS Tank Area
R-38A	31 Waste Holdup Tank Area
R-38B	32 Waste Holdup Tank Area
R-38C	33 Waste Holdup Tank Area
R-38D	Waste Holdup Tank Pump Room
R-48A	Radiochemistry Lab – 4th Floor Admin Bldg
R-48B	Radiochemistry Lab – 4th Floor Admin Bldg
R-48C	Chemistry Counting Room – 4th Floor Admin Bldg
R-48D	Respiratory Maintenance Room – REMOVED FROM SERVICE
R-48E	HVAC Room – 4th Floor Admin Bldg – REMOVED FROM SERVICE
R-48F	Controlled Passage – 4th Floor Admin Bldg – REMOVED FROM SERVICE
R-48G	Source Vault – 4th Floor Admin Bldg
R-49	Liquid Waste Disposal Area - 47' Admin Bldg – REMOVED FROM SERVICE
R-51	Laundry Room - 47' Admin Bldg – REMOVED FROM SERVICE
R-53	RAMS Sump Stairwell – REMOVED FROM SERVICE
R-53B	RAMS Outside Filter Area - 41' – REMOVED FROM SERVICE
R-53C	RAMS Radiation Area - 41'
R-54A	RAMS Decon Room - 54' – REMOVED FROM SERVICE
R-54B	RAMS Disassembly Area - 54'
R-54C	RAMS Storage Room - 54' – REMOVED FROM SERVICE
R-55A	RAMS Radiation Area - 73' Near Fence – REMOVED FROM SERVICE
R-55B	RAMS Radiation Area - 73' Near Tool Room
R-64	Primary Auxiliary Bldg - 55'
R-65	Primary Auxiliary Bldg - 73'
R-66	Primary Auxiliary Bldg - 34'
R-67	Primary Auxiliary Bldg - 41'
R-68	Primary Auxiliary Bldg - 15'
R-69	Pipe Pen - 54'
R-70	Fan House - 80'

TABLE 11.2-8

EQUILIBRIUM CONTAINMENT AIR ACTIVITIES FOLLOWING RECIRCULATION FILTRATION AT FULL POWER OPERATION

(14.4 gal/day Reactor Coolant Leak, 0.2% Equivalent Fuel Rod Defects)

Isotope	Equilibrium Containment Activity Curies
Kr-85M	0.006
Kr-85 (Peak)	6.77
Kr-87	1×10^{-3}
Kr-88	0.006
Xe-133M	0.106
Ke-133	22.0
Xe-135M	$<10^{-3}$
Xe-135	0.036
I-131	0.0036
I-132	0.0006
I-133	0.0043
I-134	0.0002
I-135	0.0020

TABLE 11.2-9

EQUILIBRIUM CONTAINMENT AIR ACTIVITIES FOLLOWING RECIRCULATION FILTRATION AT FULL POWER OPERATION

(50 lb/day Reactor Coolant Leak, 1% Equivalent Fuel Rod Defects)

Isotope	Equilibrium Containment Activity Curies
Kr-85M	0.012
Kr-85 (Peak)	14.1
Kr-87	$2x\ 10^{-3}$
Kr-88	0.013
Xe-133M	0.22
Xe-133	45.9
Xe-135M	$<10^{-3}$
Xe-135	0.075
I-131	0.00925
I-132	0.0013
I-133	0.01
I-134	$<10^{-3}$
I-135	0.0045

TABLE 11.2-10

ACTIVITY DISTRIBUTION IN THE SECONDARY PLANT
AS A FUNCTION OF PRIMARY TO SECONDARY LEAK RATE

(Expected Operation With 0.2% Equivalent Fuel Rod Defects)

Secondary Plant Noble Gas Release - μCi/sec

Primary to Secondary Leak Rate

Isotope	0.014 gpm	0.1 gpm	0.5 gpm	1.0 gpm	5.0 gpm
Kr-85	1.18	8.43	42.14	84.29	421.43
Kr-85M	0.34	2.43	12.14	24.29	121.43
Kr-87	0.20	1.43	7.14	14.29	71.43
Kr-88	0.60	4.29	21.43	43.86	214.29
Xe-133	46.1	329.29	1646.43	3292.86	16464.29
Xe-133M	0.51	3.64	18.21	36.43	182.14
Xe-135	1.02	7.29	36.43	72.86	364.29
X3-135M	0.03	0.21	1.07	2.14	10.71

Steam Generator Blowdown Liquid Concentration - μCi/gm

Primary to Secondary Leak Rate

Isotope	0.014 gpm	0.1 gpm	0.5 gpm	1.0 gpm	5.0 gpm
Mo-99	2.44×10^{-4}	1.62×10^{-3}	8.13×10^{-3}	1.62×10^{-2}	8.12×10^{-2}
I-131	1.19×10^{-4}	7.93×10^{-4}	3.97×10^{-3}	7.93×10^{-3}	3.97×10^{-2}
I-132	9.46×10^{-6}	6.42×10^{-5}	3.15×10^{-4}	6.31×10^{-4}	3.15×10^{-3}
I-133	1.42×10^{-4}	9.47×10^{-4}	4.73×10^{-3}	9.47×10^{-3}	4.73×10^{-2}
I-134	2.57×10^{-6}	1.71×10^{-5}	8.57×10^{-5}	1.71×10^{-4}	8.57×10^{-4}
I-135	4.70×10^{-5}	3.13×10^{-4}	1.57×10^{-3}	3.13×10^{-3}	1.57×10^{-2}
Cs-134	1.48×10^{-5}	9.87×10^{-5}	4.93×10^{-4}	9.87×10^{-4}	4.93×10^{-3}
Cs-137	7.34×10^{-5}	4.89×10^{-4}	2.45×10^{-3}	4.89×10^{-3}	2.45×10^{-3}

TABLE 11.2-11

ACTIVITY DISTRIBUTION IN THE SECONDARY PLANT
AS A FUNCTION OF PRIMARY TO SECONDARY LEAK RATE

(Expected Operation With 1.0% Equivalent Fuel Rod Defects)

Secondary Plant Noble Gas Release - µCi/sec

Primary to Secondary Leak Rate

Isotope	0.014 gpm	0.1 gpm	0.5 gpm	1.0 gpm	5.0 gpm
Kr-85	5.90	42.14	210.70	421.40	2107.00
Kr-85M	1.70	12.14	60.70	121.40	607.00
Kr-87	1.00	7.14	35.70	71.40	357.00
Kr-88	3.00	21.43	107.15	214.30	1071.50
Xe-133	230.5	1646.43	8232.15	16464.30	82321.50
Xe-133M	2.55	18.21	91.05	182.10	910.50
Xe-135	5.10	36.43	182.15	364.30	1821.50
X3-135M	0.15	1.07	5.35	10.70	53.50

Steam Generator Blowdown Liquid Concentration - µCi/gm

Primary to Secondary Leak Rate

Isotope	0.014 gpm	0.1 gpm	0.5 gpm	1.0 gpm	5.0 gpm
Mo-99	1.22×10^{-3}	8.13×10^{-3}	4.07×10^{-2}	8.13×10^{-2}	4.07×10^{-1}
I-131	5.95×10^{-4}	3.97×10^{-3}	1.99×10^{-2}	3.97×10^{-2}	1.99×10^{-1}
I-132	4.73×10^{-4}	3.15×10^{-3}	1.58×10^{-2}	3.15×10^{-2}	1.58×10^{-1}
I-133	7.10×10^{-4}	4.73×10^{-3}	2.37×10^{-2}	4.73×10^{-2}	2.37×10^{-1}
I-134	1.29×10^{-6}	8.57×10^{-6}	8.29×10^{-5}	8.57×10^{-5}	4.29×10^{-4}
I-135	2.35×10^{-4}	1.57×10^{-3}	7.85×10^{-3}	1.57×10^{-2}	7.85×10^{-2}
Cs-134	7.40×10^{-5}	4.93×10^{-4}	2.47×10^{-3}	4.93×10^{-3}	2.47×10^{-2}
Cs-137	3.67×10^{-4}	2.45×10^{-3}	1.23×10^{-2}	2.45×10^{-2}	1.23×10^{-1}

TABLE 11.2-12

NOBLE GAS ACTIVITY CONCENTRATIONS IN THE CONDENSER
AIR EJECTOR AS A FUNCTION OF PRIMARY TO SECONDARY LEAK RATE

(Operation With Expected 0.2% Equivalent Fuel Rod Defects)

Air Ejector Discharge Concentration - μ Ci/sec

Primary to Secondary Leak Rate

Isotope	0.014 gpm	0.1 gpm	0.5 gpm	1.0 gpm	5.0 gpm
Kr-85	4.17×10^{-5}	2.98×10^{-4}	1.49×10^{-3}	2.98×10^{-3}	1.49×10^{-2}
Kr-85M	1.20×10^{-5}	8.58×10^{-5}	4.29×10^{-4}	8.58×10^{-4}	4.29×10^{-3}
Kr-87	7.06×10^{-6}	5.05×10^{-5}	2.52×10^{-5}	5.05×10^{-4}	2.52×10^{-3}
K-88	2.12×10^{-5}	1.51×10^{-4}	7.57×10^{-4}	1.51×10^{-3}	7.57×10^{-3}
Xe-133	1.63×10^{-3}	1.16×10^{-2}	5.81×10^{-2}	1.16×10^{-1}	5.81×10^{-1}
Xe-133M	1.80×10^{-5}	1.29×10^{-4}	6.43×10^{-4}	1.29×10^{-3}	6.43×10^{-3}
Xe-135	3.60×10^{-5}	2.57×10^{-4}	1.29×10^{-3}	2.57×10^{-3}	1.29×10^{-2}
X3-135M	1.06×10^{-6}	7.42×10^{-6}	3.78×10^{-5}	7.42×10^{-5}	3.78×10^{-4}

TABLE 11.2-13

NOBLE GAS ACTIVITY CONCENTRATIONS IN THE CONDENSER AIR EJECTOR AS A FUNCTION OF PRIMARY TO SECONDARY LEAK RATE

(Operation With Expected 0.2% Equivalent Fuel Rod Defects)

Air Ejector Discharge Concentration - μ Ci/sec

Primary to Secondary Leak Rate

Isotope	0.014 gpm	0.1 gpm	0.5 gpm	1.0 gpm	5.0 gpm
Kr-85	2.08×10^{-4}	1.49×10^{-3}	7.44×10^{-3}	1.49×10^{-2}	7.44×10^{-2}
Kr-85M	6.00×10^{-5}	4.29×10^{-4}	2.14×10^{-3}	4.29×10^{-3}	2.14×10^{-2}
Kr-87	3.53×10^{-5}	2.52×10^{-4}	1.26×10^{-3}	2.52×10^{-3}	1.26×10^{-2}
K-88	1.06×10^{-4}	7.57×10^{-4}	3.78×10^{-3}	5.57×10^{-3}	3.78×10^{-2}
Xe-133	8.14×10^{-3}	5.81×10^{-2}	2.91×10^{-1}	5.81×10^{-1}	2.91×10^{-0}
Xe-133M	9.00×10^{-5}	6.43×10^{-4}	3.22×10^{-3}	6.43×10^{-3}	3.22×10^{-2}
Xe-135	1.80×10^{-4}	1.29×10^{-3}	6.43×10^{-3}	1.29×10^{-2}	6.43×10^{-2}
X3-135M	5.30×10^{-6}	3.78×10^{-5}	1.89×10^{-4}	3.78×10^{-4}	1.89×10^{-3}

TABLE 11.2-14

RADIATION MONITORING SYSTEM CHANNEL R-15 (CONDENSER AIR EJECTOR) AND CHANNEL R-19 (STEAM GENERATOR SECONDARY LIQUID) RESPONSE FOR EXPECTED PLANT OPERATION WITH 0.2% EQUIVALENT FUEL ROD DEFECTS

Primary to Secondary Leak Rate

RMS Channel	0.014 gpm	0.1 gpm	0.5 gpm	1.0 gpm	5.0 gpm
#15 Output-CPM	3.4×10^{4}	2.2×10^{5}	9.7×10^{5}	1.0×10^{6}	1.0×10^{6}
#15 Output-μci/cc	1.7×10^{-3}	1.2×10^{-2}	7.0×10^{-2}	1.2×10^{-1}	7.0×10^{-1}
#19 Output-μci/cc	7.3×10^{-4}	5.2×10^{-3}	2.6×10^{-2}	5.2×10^{-2}	2.6×10^{-1}

TABLE 11.2-15

RADIATION MONITORING SYSTEM CHANNEL R-15 (CONDENSER AIR EJECTOR) AND CHANNEL R-19 (STEAM GENERATOR SECONDARY LIQUID) RESPONSE FOR OPERATION WITH 1% EQUIVALENT FUEL ROD DEFECTS

Primary to Secondary Leak Rate

RMS Channel	0.014 gpm	0.1 gpm	0.5 gpm	1.0 gpm	5.0 gpm
#15 Output-CPM	1.7×10^{5}	9.3×10^{5}	1.0×10^{6}	1.0×10^{6}	1.0×10^{6}
#15 Output-μci/cc	8.5×10^{-3}	6.0×10^{-2}	3.0×10^{-1}	6.0×10^{-1}	3.0×10^{-0}
#19 Output-μci/cc	3.6×10^{-3}	2.6×10^{-2}	1.3×10^{-1}	2.6×10^{-1}	1.3×10^{-0}

TABLE 11.2-16
DELETED

TABLE 11.2-17

TIME-INTEGRATED ACTIVITY CONCENTRATION OUTSIDE CONTROL ROOM DUE TO CONTAINMENT LEAKAGE (μci/C.C.) - HR

Isotope	0-2 hr	2-8 hr	8-24 hr	1-4 days	4-30 days
Kr-85m	1.63-3	1.33-3	7.72-4	1.53-5	0
Kr-85	9.26-5	1.39-4	3.70-4	3.97-4	3.42-3
Kr-87	2.24-3	5.65-4	2.23-5	0	0
Kr-88	4.08-3	2.43-3	7.02-4	0	0
Xe-133m	2.43-3	3.48-3	8.14-3	5.24-3	3.56-3
Xe-133	9.77-3	1.45-2	3.77-2	3.45-2	7.56-2
Xe-135m	1.42-3	1.89-3	2.86-3	3.05-4	0
Xe-135	2.74-3	4.40-3	9.51-3	1.79-3	1.53-5
I-131	1.08-4	3.10-5	1.53-5	1.53-7	0
I-132	1.42-4	1.44-5	9.58-7	0	0
I-133	2.38-4	6.10-5	2.39-5	1.53-7	0
I-134	2.04-4	5.11-6	0	0	0
I-135	2.09-4	4.12-5	9.58-6	0	0

TABLE 11.2-18

TIME-INTEGRATED ACTIVITY CONCENTRATION OUTSIDE CONTROL ROOM
DUE TO ESF LEAKAGE (μci/C.C.) - HR

Isotope	30 min - 2 hr	2-8 hr	8-24 hr	24 hr – 4 day	4-30 days
I-131	6.25-6.08-4	1.23-5	3.16-5	5.84-5	1.77-4
I-132	6.62-6	4.91-6	9.81-7	4.08-9	----
I-133	1.35-5	2.39-5	4.46-5	2.78-5	2.82-6
I-134	6.47-6	1.41-6	1.23-6	----	----
I-135	1.12-5	1.55-5	1.48-5	1.72-6	1.11-9

TABLE 11.2-19

TIME-INTEGRATED ACTIVITY CONCENTRATION INSIDE CONTROL ROOM
DUE TO CONTAINMENT LEAKAGE (μci/C.C.) - HR

Isotope	0-2 hr	2-8 hr	8-24 hr	1-4 days	4-30 days
Kr-85m	1.63-3	1.33-3	7.72-4	1.53-5	0
Kr-85	9.26-5	1.39-4	3.70-4	3.97-4	3.42-3
Kr-87	2.24-3	5.65-4	2.23-5	0	0
Kr-88	4.08-3	2.43-3	7.02-4	0	0
Xe-133m	2.43-3	3.48-3	8.14-3	5.24-3	3.56-3
Xe-133	9.77-3	1.45-2	3.77-2	3.45-2	7.56-2
Xe-135m	1.42-3	1.89-3	2.86-3	3.05-4	0
Xe-135	2.74-3	4.40-3	9.51-3	1.79-3	1.53-5
I-131	2.39-6	6.85-7	3.38-7	3.38-9	0
I-132	3.14-6	3.18-7	2.12-8	0	0
I-133	5.26-6	1.35-6	5.28-7	3.38-9	0
I-134	4.51-6	1.13-7	0	0	0
I-135	4.62-6	9.11-7	2.12-7	0	0

TABLE 11.2-20

TIME-INTEGRATED ACTIVITY CONCENTRATION OUTSIDE CONTROL ROOM
DUE TO ESF LEAKAGE (μci/C.C.) - HR

	30 min - 2 hr	2-8 hr	8-24 hr	24 hr – 4 day	4-30 days
I-131	2.38-7	4.69-7	1.20-6	2.23-6	6.74-6
I-132	2.53-7	1.87-7	3.74-8	1.56-10	----
I-133	5.14-7	9.11-7	1.70-6	1.06-6	1.08-7
I-134	2.47-7	5.37-8	4.69-10	----	----
I-135	4.28-7	5.92-7	5.66-7	6.57-8	4.25-11

TABLE 11.2-21

CONTAINMENT AIR SOURCE STRENGTH AT
VARIOUS TIMES FOLLOWING A LOCA (TID-14844 RELEASE FRACTION)

SOURCE STRENGTH (Ci)

ISOTOPE	0	2 HRS.	8 HRS.	24 HRS.	4 DAYS	30 DAYS
Kr-85M	3.57+7	2.6-+7	1.00+7	7.85+5	8.37	0
KR-85	1.74+6	1.74+6	1.74+6	1.74+6	1.73+6	1.70+6
Kr-87	6.85+7	2.36+7	9.63+5	1.90+2	0	0
Kr-88	9.74+7	5.91+7	1.32+7	2.41+5	4.67-3	0
Xe-133m	4.63+7	4.52+7	4.20+7	3.46+7	1.41+7	5.39+3
Xe-133	1.83+8	1.83+8	1.81+8	1.73+8	1.28+8	4.55+6
Xe-135m	2.71+7	2.62+7	2.08+7	7.48+6	1.43+4	0
Xe-135	4.94+7	5.35_7	5.50+7	3.23+7	3.20_5	0
I-131	2.02+7	7.26+5	2.09+5	1.39+4	7.19-2	0
I-132	3.07+7	6.14+5	3.04+4	1.84+1	0	0
I-133	4.53+7	1.54+6	3.71+5	1.54+4	9.48-3	0
I-134	5.29+7	3.94+5	1.00+3	2.21-4	0	0
I-135	4.10+7	1.21+6	1.93+5	2.67+3	1.15-5	0

TABLE 11.2-22

CONTAINMENT SUMP AND RECIRCULATION PIPINGS
OUTSIDE CONTAINMENT SOURCE STRENGTHS AT
VARIOUS TIMES FOLLOWING A MAXIMUM CREDIBLE
ACCIDENT (TID-14844 RELEASE FRACTION)

SOURCE STRENGTH (MeV/cc-sec)

Er. MeV	0	0.5 HR.	2.0 HRS.	8.0 HRS.	1 DAY	7 DAYS	30 DAYS
0.2-0.4	3.12+09	1.09+09	8.76+08	7.59+08	5.65+08	2.34+08	2.92+07
0.4-0.9	1.17+10	7.79+09	4.28+09	1.64+09	7.98+08	1.50_08	7.01+07
0.9-1.35	7.01+09	3.31+09	1.95+09	8.57+08	1.95+08	8.76+06	2.34+06
1.35-1.8	6.82+-0	3.31+09	1.73+-0	6.04+08	1.48+08	4.48+07	1.29+07
1.8-2.2	2.92+09	1.69+09	9.93+08	2.34+08	1.25_07	1.79+06	7.20+05
2.2-2.6	3.31+09	2.14+09	1.19+09	2.53+08	1.31+07	2.73+06	7.79+05
2.6-3.0	1.58+09	2.53+08	1.17+08	1.67+07	3.70+05	4.67+04	1.34+04
3.0-4.0	1.11+09	1.44+08	4.87+-7	7.01+06	1.48+05	1.83+04	5.26+03
4.0-5.0	8.18+08	6.23+06	5.-6+06	0	0	0	0
5.0-6.0	3.60+06	4.67+04	0	0	0	0	0

TABLE 11.2-23

RADIATION DOSES IN THE CONTROL ROOM INTEGRATED OVER 30 DAYS AFTER LOCA

Control Room	Dose Guidelines per S.R.P. 6-4	Containment Air	Containment Sump Water	Activity Inside Rm. From Makeup Air Intake	Activity Outside Rm. Thru Plume Release	Activity Inside Rm. From Makeup Air Intake	Activity Outside Rm. Thru Plume Release	Activity Radiation From Recirc. Pipes	TOTAL
a. Whole body gamma	5 Rem	$1x10^{-3}$ Rem	$1.5x10^{-3}$ Rem	1.75 Rem	$1.4x10^{-2}$ Rem	NIL	NIL	$2.1x10^{-3}$ Rem	1.8 Rem
b. Thyroid	30 Rem	-	-	19.2 Rem	-	9.6 Rem	-	-	28.8 Rem
c. Beta Skin	30 Rem	-	-	28.2 Rem	-	NIL	-	-	28.2 Rem