SECTION 11

RADIOACTIVE WASTES AND RADIATION PROTECTION

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SECTION 11

RADIOACTIVE WASTES AND RADIATION PROTECTION

11.1 GENERAL DESCRIPTION

Waste disposal systems are provided to separate, treat, and dispose of radioactive gaseous, liquid, and solid waste materials. These systems incorporate one or more of the following basic processes:

1. Filtration, to remove particulate matter.
2. Evaporation, to concentrate radioactive constituents into a smaller liquid volume.
3. Demineralization, to remove dissolved material.
4. Baling, to reduce the volume of compressible wastes.
5. Storage to provide natural decay of radioactive isotopes.
6. Dilution, to reduce concentration of effluents.

Gaseous, liquid, and solid waste materials originate in the reactor coolant system (RCS), and the auxiliary and emergency systems, as a result of operation and maintenance. Waste materials enter the waste disposal systems directly from their source or via the vent and drain system (Section 9.7).

Sampling, analysis, and monitoring of the waste disposal systems are provided to comply with the design criteria. Process radiation monitors and flow measuring equipment are provided for surveillance of various station waste process streams to ensure compliance with regulations and to provide early indication of possible malfunctions and hazardous conditions.

Shielding is provided to reduce radiation to acceptable levels for normal operation and incident conditions. Allowable dose rates are based on regulations and expected frequency and duration of exposure to radiation.

Area radiation monitoring equipment, Radiological Control facilities, environmental protection programs, and administrative controls are provided for surveillance, control of radiation, and limiting of exposure levels. These ensure adequate radiation protection for station personnel and the general public in accordance with applicable criteria.

Table 11.1-1 lists the locations of each radiation monitor.

11.2 RADIOACTIVE WASTE SYSTEMS

11.2.1 Design Basis

The waste disposal systems are designed to process wastes to meet the requirements of 10CFR20. The condition of the RCS (Section 4) primarily determines the sources and amounts of radioactivity which must be handled by the waste disposal systems.

Appendix 11A, which reflects original plant design, delineates the sources and amounts of radioactive isotopes which must be handled by the waste disposal systems. This Appendix is retained to provide historical information.

Specific effluent release rates and conditions for release and actions to be taken in the event of a failure to meet these rates and conditions are given in the Offsite Dose Calculation Manual.

11.2.2 System Design

The waste disposal and radiation monitoring systems are designed to satisfy the applicable 1971 General Design Criteria given in Appendix IA (to this FSAR). In addition, these systems are designed to protect the health of operating personnel and to limit the discharge of radioactive materials so as not to exceed the limits of 10CFR20. Transportation of radioactive materials from the station is carried out in a manner that conforms to all applicable federal, state, and local regulations.

11.2.3 Gaseous Waste Disposal System

The gaseous waste disposal system processes and monitors all waste gas streams prior to discharge to the atmosphere. The system provides the decay time for degasifier gaseous effluent. The system provides for recycling of the hydrogen present in the degasifier overheads back to the volume control tank, when the hydrogen meets Westinghouse specifications.

All gaseous waste effluent not recycled is directed to the gaseous waste disposal system for disposal. The gaseous waste disposal system is shown in Figure 11.2-1. Component design data is given in Table 11.2-1.

In addition to the waste gas system described in this section, there is also a supplementary leak collection and release system described in Section 6.6. The normal ventilation which handles the discharge to the atmosphere of low activity air streams is described in Sections 5.4.1 and 9.13.

11.2.3.1 Design Bases

The gaseous waste disposal system is designed to process effluents to meet the requirements of 10CFR20. The system provides selective holdup such that the short lived isotopes have decayed prior to release. It also provides holdup of these gases when refueling cold shutdown degassing is required.

System design provides that all the gaseous effluent from the degasifiers is directed to the gaseous waste charcoal delay subsystem for decay of most radioactive isotopes prior to compressing and discharged through the process vent. Gaseous effluent may be recycled to the volume control tank but this is not normally performed. Provision is made to direct compressed waste gas to decay tanks for control of the equilibrium activity level of the coolant fission product gas inventory and subsequent release to the atmosphere. The discharge to the atmosphere is handled by diluting the flow controlled release of waste gas with a large volume of air, discharging the air through charcoal and HEPA filters to the top of the cooling tower, approximately 500 ft above the ground. This same discharge system is also designed to handle gaseous effluent from the main condenser air ejector vents, purge and vent from the oxygen analyzers, decay tank radiation monitor aerated vents of the vent and drain system, and the gaseous discharge from the containment vacuum system. The system also handles special conditions when gases from the containment purge are vented to the top of the cooling tower.

11.2.3.2 Description

Radioactive gases enter the gaseous waste disposal system from the degasifier vent chiller of the boron recovery system and are directed by the system pressure gradient to the gaseous waste charcoal delay subsystem upstream of the overhead gas compressor. The gas is chilled to approximately 55°F to condense most of the water vapor. The compressors operate automatically in response to the suction pressure thus maintaining the degasifier's overhead components at a pressure between established limits. Radioactive gases from the degasifier vent chillers contain primarily hydrogen, water vapor and a small amount of nitrogen in the gaseous effluent. The gas is then processed through the gaseous waste charcoal delay subsystem which holds up the xenon for about 39 days, and the krypton about two days. Essentially all of the iodine is absorbed by the charcoal. This holdup assumes continuous stripping of 60 gpm of primary coolant letdown with a hydrogen concentration of 35 cc/kg.

One of the two overhead gas compressors directs the radioactive gas stream to a gas surge tank at a system pressure of about 65 psig. The gas flow is reduced in pressure and, as long as it meets Westinghouse specifications, can be returned to the volume control tank in the chemical and volume control system (Section 9.1). However, this method of gas reclamation is not normally used. A quantity of gas can be discharged from the surge tank to one of the three decay tanks at Unit 1 or the seven storage tanks at Unit 2 for eventual release to the atmosphere via the process vent on top of the cooling tower.

The decay tanks are normally operated on an alternating feed, hold, and bleed cycle. The system is operated so that one tank is on the feed portion of the cycle, a second tank is on the hold portion, and a third tank is on the bleed portion. Grab sampling of the decay tank contents is required when radioactive materials are being added to the tank and the gross concentration of the primary coolant is >100 μCi/ml.

Continuous sampling of oxygen concentration in the discharge line of the surge tank is provided by directing a sample flow to oxygen analyzers to prevent sending an undesirable oxygen concentration to the volume control tank or the decay tanks.

An automatic shutdown capacity of the overhead gas compressors in the event of a high 0_2 concentration is provided.

A shutdown-override switch is provided in order to allow for system purging. The switch position would be monitored via an alarm in the control room. This alarm is actuated from contacts of the shutdown-override switch to indicate the switch is in the override position.

Tritium sampling is normally accomplished by drawing a sample from the process stream radiation monitor during discharge from the Gaseous Waste Decay Tanks.

The three Unit 1 decay tanks, each with an internal volume of approximately 132 cu ft, are buried separately in cubicles. Each of the decay tanks is considered independently regarding tank bursts, in that each tank is shielded from any other tank. Double valving and missile protection are provided on all interconnecting piping.

After the decay tanks are sampled and authorization obtained for discharge, the flow of waste gases from the decay tanks is recorded and rapidly diluted through the use of a diffuser, (decay gas injector), with about 1,100 scfm of air in order to limit hydrogen and radioactive effluent concentration. The gases are then combined with the containment vacuum system exhaust, aerated vents of the vent and drain system, and main air ejector effluent. Gases from the containment purge (Section 5.4) may be discharged through this system. The mixture is then filtered through one of the gaseous waste disposal filters, each of which consists of a charcoal bed and a HEPA fine particulate filter. The filtered gases are then discharged by one of the gaseous waste disposal blowers to the atmosphere via a process vent on the top of the cooling tower.

The flow rate and concentration level of the stream are measured continuously to determine the rate of activity release to the atmosphere. Should the radioactivity release rate of this stream be above the Offsite Dose Calculation Manual limits, a high-high signal from the radiation monitor will stop all flow from the decay tanks. Air continues to be admitted to the system until the discharge returns to an acceptable concentration level, at which time the controls may be manually reset under strict administrative control. These controls are imposed to restrict the conditions under which further releases are allowed. Administrative controls require determining the effluent activity of the various components comprising the waste stream. This can be determined by obtaining the radiation monitor readings for the various components or grab sampling the various components.

Corrective actions that may be undertaken to resume normal operations include reestablishing the discharge at a rate that is within that permitted by the terms of a discharge permit, adding nitrogen to the decay tank for dilution, or terminating the discharge to permit more time for radioactive decay of short-lived radionuclides.

Station operating procedures define the actions to be taken in the event of high gaseous effluent activity.

Air is excluded from the decay tanks by control of surge tank pressures above atmosphere pressure, and by the use of diaphragm overhead gas compressors. Sampling and analysis of waste gas leaving the surge tank for oxygen content ensures that no combustible mixtures develop in the system. Nitrogen gas for flushing is also provided at the decay tanks.

Over-pressure protection for the underground decay tanks is provided in the form of a pressure controller followed by a rupture disc in parallel with a restriction orifice and a rupture disc. When the pressure in the decay tank reaches 100 psig, the pressure controller and the rupture disc downstream of the pressure controller will relieve gas to the release system. As soon as the pressure in the tank falls below 95 psig, the pressure controller will close. If the pressure controller fails to open at the required pressure, the rupture disc, in parallel with the pressure controller, will relieve at 110 psig. The restriction orifice upstream of this rupture disc will restrict the flow of gas to the release system to an amount (30 scfm) that can be diluted at the top of the cooling tower with a minimum of hazard. The rupture discs discharge into a common header which is isolated from the aerated gaseous waste disposal system effluent by a shallow loop seal to prevent flammable mixtures from forming in the header.

The gaseous effluent stream from the main condenser air ejectors is vented to gaseous waste disposal filters or to the Unit 2 air ejector charcoal delay beds.

Discharge of main condenser air ejector effluent through charcoal delay beds provides sufficient holdup for decay of short lived radioactive components. Prior to entering these charcoal beds, the gas-stream is chilled and dehumidified to a temperature of approximately 77°F dry bulb and 55°F wet bulb and a pressure of approximately 32.9 inches Hg. Normally, the effluent from the air ejectors is not highly contaminated. However, the possibility exists of a leakage from the reactor coolant system into the secondary system with resultant system contamination.

The adsorption coefficients used for gaseous waste charcoal decay beds are based on data from literature and activated charcoal manufacturers. In particular, the adsorption coefficients for xenon and krypton are taken from adsorption isotherms available from Pittsburgh Activated Carbon Division of Calgon Corporation. The charcoal type used was BPL coconut shell based activated carbon. The operating temperature is approximately 77°F. The adsorption isotherms cover a broad range of partial pressures varying from 0.001 mmHg to 100 mmHg which adequately covers the desired partial pressure range.

The adsorption coefficients for nitrogen are based on adsorption data from Cook & Basmadjian's paper.[(2)] The operating temperature is approximately 77°F. Partial pressure range covered is extrapolated within reason to cover the entire range.

The system is treated as a binary between xenon and nitrogen. In making this simplifying assumption, a conservative approach has been used. Krypton is lumped with xenon, and argon is lumped with nitrogen. The calculations are based on adsorption coefficients of 0.18 and 0.172 for nitrogen and xenon, respectively. At the specified composition, the adsorption capacity for xenon in presence of nitrogen is calculated as 0.036 cc(STP) of xenon per gm of charcoal. The charcoal quantity required for 30 days delay for xenon is 0.62 tons.

The calculation includes a correction for the effect of water vapor in the gas. The reduction of xenon adsorption capacity due to 41.7 percent relative humidity is estimated to be 38 percent. This is conservatively estimated from adsorption data on krypton-water vapor plotted on Figure 3 of Dragon Study - RPT - 276. Therefore, the charcoal quantity required for design conditions is 1.0 ton. The charcoal quantity provided as a safety factor is 1.55 tons, which gives additional delay time.

The gas waste decay and surge tanks are designed in compliance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Class C, with 100 percent radiography. The gas waste charcoal beds are designed in compliance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Class 3, with 100% radiography.

Process piping is designed to meet American National Standards Institute (ANSI) B31.1, Power Piping, Section 1 requirements.

The gaseous waste disposal system from the boron recovery system to an isolation valve downstream of the decay tanks is considered Seismic for design purposes. Component design data for the gaseous waste disposal system is listed in Table 11.2-1.

11.2.3.3 Evaluation

Fission product gas inventory in the reactor coolant is a function of reactor coolant system fission gas input and output. Fission gas input is determined by the reactor power level and the amount of fuel failure. Fission gas output from the reactor coolant system is determined by the amount of gas sent to the BVPS-1 decay tanks and not recycled.

Reactor coolant is letdown to adjust its chemistry and its radiation level and to provide water for seal injection.

The annual average waste gas bleed rate needed to maintain the krypton-85 inventory at acceptable levels within the reactor coolant during steady state full power operation is a small fraction of a cfm.

Operation of the gaseous waste disposal system, using charcoal delay beds selectively delaying xenon-133 for approximately 39 days, results in an annual average atmospheric emission rate which is a very small fraction of the 10 CFR 20 limit. A tabulation of nuclides and their maximum calculated annual release rate is given in Appendix 11A. Appendix 11B evaluates the radiation exposure to the general public or to individuals thereof based on the estimated radioactive nuclide releases identified in Appendix 11A. These Appendices are retained to provide historical information. Actual concentrations of radioactive material in effluents are limited in accordance with the Offsite Dose Calculation Manual.

Thirty-nine days of holdup allows time for the short lived fission gases to decay to the point where krypton-85 is the controlling isotope.

In the event of modes of fuel failure which result in abnormal concentrations of fission products in the reactor coolant, adequate storage space in the decay tanks is supplied. The tanks will be allowed to go to a higher holding pressure and will thus be able to accommodate a larger volume of gas. The higher pressure will not exceed the design pressure of the system.

When charcoal delay beds are installed for the air ejectors the activity discharged from the air ejectors is assumed to be the result of one percent failed fuel and a continuous 50 lb per hr in-leakage from all steam generators. Periodic increases in the leakage rate up to 500 lb per hr will also be considered in the design. It is considered that one percent failed fuel is an extreme condition that may never occur and the possibility of its occurrence with a leak rate of 500 lb per hr is extremely remote. If one of the steam generators should develop excessive leakage, its loop can be isolated. Therefore, the above assumption is conservative and results in the dose being over estimated. With the charcoal delay systems installed on both stripped gas and air ejector discharges the effluent is essentially krypton controlled.

11.2.3.4 Accidental Release of Waste Gases

The concentration of radioactive waste gases in the reactor coolant system and auxiliary systems is a function of the rate of fission gas release to the coolant from defective fuel and the rate of removal via the auxiliary systems.

The radiological consequence (dose) analysis for the accidental release of waste gases considers two accident scenarios. The first is a rupture of a gas decay tank with the release of its contents directly to the environment. The second is a gaseous waste system pipe rupture. The analyses were performed using conservative assumptions based on NUREG-0800, Branch Technical Position ETSB 11-5. These accidents, as described in detail below, provide the bounding conditions and resultant radiological consequences for a waste gas release.

11.2.3.4.1 Method of Analysis

Reactor coolant noble gas concentrations (taken from Table 14B-6) are based on the assumption that 1.0 percent of the fuel rods in the core develop pinhole defects, resulting in the diffusion of fission product isotopes into the coolant. The rod fission product inventories are those produced at 100.6 percent power at a maximum core thermal power of 2918 MWt.

For the decay tank rupture accident, Xe will not be present because of the relatively long holdup in the charcoal delay bed. The Kr activity that is accumulated in a tank is calculated by assuming activity transfer from the RCS at the maximum rate of 120 gpm, holdup in the charcoal delay bed then transfer to the tank for the minimum time period required to fill the tank. This methodology minimizes activity reduction by the radioactive decay process. Activity release to the environment following a tank rupture is assumed to be a puff release from the decay tank vault directly to the environment.

For the line rupture accident, the release consists of two sources: 1) 100 percent of the Xe and Kr contained in reactor coolant letdown liquid released at the maximum letdown flow rate of 120 gpm plus, 2) a fraction of the Xe and Kr that would be retained on the waste gas system charcoal delay bed during 24 hours of power operation plus 26.8 hours after reactor shutdown (the time to degas 99% of the reactor coolant system) while operating with letdown at the maximum 120 gpm. Activity release following a line rupture is assumed to be a puff release from the charcoal delay bed, and a 1 hour release of the reactor coolant letdown gases, both into the Auxiliary Building. The activity is assumed to be instantaneously released to the environment as it enters the Auxiliary Building.

Between the two accident scenarios described above, the bounding waste gas system accident is considered for the purpose of radiological consequence analysis.

Table 11.5-8 provides the significant analysis parameters for each of the accident scenarios, including relevant assumptions for calculating dose to the control room operators and at the exclusion area boundary (EAB) and the low population zone (LPZ).

11.2.3.4.2 Results

Thyroid committed dose equivalent (CDE), whole body dose (EDE) and skin beta dose equivalent (DE) for the control room operators are:

Line break: <2E-01 EDE, 3.9E+00 Skin DE
Tank Rupture: <2E-01 EDE, <1.4E+00 Skin DE

Doses calculated for offsite are provided below:

	Decay Tank Rupture		Line Rupture	
	0-2 h EAB (rem)	0-30 d LPZ (rem)	0-2 h EAB (rem)	0-30 d LPZ (rem)
Thyroid CDE	N/A	N/A	N/A	N/A
Whole Body EDE	<2E-01	<2E-01	2.3E-01	<2E-01

11.2.3.4.3 Conclusions

The maximum control room operator doses that may result due to a failure of the gaseous waste system are less than the 10 CFR Part 50 Appendix A, GDC 19 limit of 5 rem whole body, or its equivalent to any part of the body.

The maximum EAB and LPZ doses that may result due to a failure of the gaseous waste system are a small fraction of the 10 CFR Part 100.11 limits of 25 rem whole body and 300 rem thyroid, and are within the 500 mrem whole body dose specified in NUREG-0800, ETSB 11-5.

11.2.4 Liquid Waste Disposal System

The liquid waste disposal system will receive, treat, test, and dispose of all aerated liquid waste from building and equipment drain sumps (Section 9.7.1), and from laundry and contaminated shower drains.

The building and equipment sumps collect the waste from the laboratory, spent resin flush system, aerated drains from operation, decontamination and maintenance of equipment and piping, and boiler blowdown. The liquid waste disposal system is shown in Figures 11.2-3 and 11.2-4. Flow charts indicating estimated liquid waste treatment from various sources are presented in Figure 11.2-5A and 11.2-5B. The bases for these charts are discussed in Appendix 11A.

11.2.4.1 Design Basis

The system is designed so that the effluents released by the liquid waste disposal system when mixed with the cooling tower blowdown meet the requirements of 10 CFR 20.

The liquid waste system conforms to ASME Section VIII design for vessels and ANSI B.31.1 for piping.

The design is based on receiving, segregating, and batch-storing three categories of solutions: high level wastes, low level waste, and laundry and contaminated showers. The essence of this system is batch control of all liquids and recovery of primary liquids where feasible. Accommodation of the wide range of volumes and activities which may enter the system is achieved by providing a high degree of flexibility in batch operation.

The liquid radwaste treatment system (evaporator and/or demineralizer) shall be used to reduce the radioactive materials in each liquid waste batch prior to its discharge when the projected doses due to liquid effluent releases (when averaged over 31 days) would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

The potential for operating economics is provided by limited segregation of radioactive wastes. However, public safety is not compromised because the system can accommodate the full range of volumes and activities delivered to it. Suitability for discharge is determined not only by comparison of waste samples with applicable limits, but also by the opportunity afforded the station to further reduce activity with existing equipment. Modifications to limit entry of non-contaminated water sources into the waste system will be made as experience is gained through plant operation.

System design and tank sizing is provided to embrace a range of conditions which may be averaged as a 4 gpm net throughput, and at activities which correspond to operation at zero to one percent failed fuel, thus covering the various combinations of operating modes and activity contributions.

System tankage will accommodate the largest single amount of drainage from equipment which may be reasonably imposed on it.

The high and low level waste drain tankage is sized to accommodate the drainage from one reactor coolant loop and one residual heat removal exchanger. These two sources represent about 10,000 gallons of drainage, or the largest short term system accumulation. The maximum average waste drain rate is about 3 gpm.

Laundry and contaminated drains tankage is sized to accommodate approximately 1 gpm of liquid wastes.

Component design data for system equipment are listed in Table 11.2-2.

11.2.4.2 Description

Liquids resulting from operating or maintenance procedures enter the vent and drain system, which classifies these liquids for reuse or disposal. Other liquid wastes are collected, sampled for possible radioactivity and may be processed or pumped through the waste effluent filters to the cooling tower blowdown on flow control. The activity of the tank sample determines the proper waste discharge flow rate. The flow control elements, control valves, and a radiation monitor control the discharge so that no liquids having a radioactive concentration greater than the limits specified in the Offsite Dose Calculation Manual are discharged. A vented siphon break downstream of the flow control valve prevents siphoning of tank contents when the discharge pump is shut off.

Liquid effluents from the contaminated shower and laundry drains are collected in either one of the two tanks, sampled for radioactivity, filtered, and discharged via the same methods as described above for liquid wastes. A separate radiation monitor is included on the contaminated drain effluent line.

When one tank is full the valving is changed to flow to the remaining tank. Each of the drain tanks is arranged so that the batch in each tank can be thoroughly mixed by circulating the liquid through the tank using a recirculation line from the transfer pump. The tank contents are then sampled and analyzed. Normally one of the pumps in a pair, for example the "A" pump, pumps from its associated tank, the "A" tank. The tanks and pumps are piped and wired so that a pump can, if required, take suction from the alternate tank. The level instruments on the tank will control whichever pump is taking suction from it.

The waste disposal evaporator and distillate demineralizers are designed to produce a distillate suitable for discharge to the cooling tower blowdown. It is also possible to recycle distillate if it meets RCS make up specifications.

The waste disposal evaporator distillate is sampled for radioactivity, and can be discharged via the same methods as described above for liquid wastes. Noncondensable gases from the waste disposal evaporator are sent to the gaseous waste disposal system.

Demineralizer system LW-I-3 provides another method of processing liquid waste for discharge or reuse. This demineralizer system consists of a train of 4 sluiceable demineralizers and is capable of processing waste from various liquid waste tanks. Process effluent may be directed to the evaporator tanks via LW-I-1, the steam generator drain tanks, or either coolant recovery tank. Each demineralizer and the Rad Waste Reverse Osmosis (RWRO) skid are capable of being isolated and bypassed for resin changeout or maintenance, while allowing continued operation of any or all of the other vessels.

After the liquid waste tanks are sampled and authorization is obtained for discharge, discharge flow from the liquid waste disposal system is combined and mixed with the cooling tower blowdown such that the activity of the combined effluent is within the limits of the Offsite Dose Calculation Manual.

11.2.4.3 Evaluation

The liquid waste disposal system provides tankage and evaporation capacity to process all liquid wastes from operating or shutdown conditions.

The system produces processed wastes which, after dilution with the cooling tower blowdown, can be discharged in accordance with the Offsite Dose Calculation Manual.

The liquid waste disposal system has also been designed to ensure that the release of radioactivity to the environment is kept to the lowest level practicable.

Appendix 11A presents estimates of radioactive nuclide concentrations in waste disposal systems and in discharge to the environment based on the original design of the liquid waste system. This Appendix is retained to provide historical information. Actual concentrations of radioactive material in effluents are limited in accordance with the Offsite Dose Calculation Manual. Appendix 11B evaluates the radiation exposure to the general public or to individuals thereof based on the estimated radioactive nuclide releases identified in Appendix 11A.

The steam generator blowdown, although included in Appendix 11A, is normally polished by demineralization and recycled for reuse in the steam generators.

Two 50,000 gallon blowdown hold tanks are installed at Unit 2 to take the flow of blowdown and/or liquid wastes from both BVPS-1 and BVPS-2. These tanks can feed evaporators or demineralizers.

System tankage for high and low level waste drains totals 14,000 gallons or about 30 percent greater volume than the largest expected short term accumulation.

Waste disposal evaporator capacity is 100 percent larger than the maximum short term total average of 3 gpm and demineralizer system LW-I-3 capacity is more than double the evaporator capacity at nominal waste activity levels.

Administrative controls and batch handling of all waste liquids ensures positive control of all processing. System liquid level indicators, radiation monitors, flow control instrumentation, and an effluent piping siphon break prevent an inadvertent radioactivity release to the environment.

The greatest expected load on the liquid waste disposal system occurs during the refueling operation. However, administrative control of the sequence and quantity of wastes developed during this period combined with the capacity and flexibility of the waste disposal system, ensures that discharges do not exceed the limits.

The principal administrative control involves the utilization of a discharge permit procedure. This discharge permit procedure will be utilized to control and record radioactive liquid discharges. Such information as activity per unit volume of liquid to be released, maximum rate of release, dilution required, projected dose, and other information pertinent to proper control and record of the radioactive liquid release, will be indicated.

No radioactive liquid discharges will be allowed unless the waste tank has been isolated, sampled and a discharge permit issued based on analysis of the sample. Upon completion of the discharge, verification that the wastes have been discharged as prescribed is performed, or deviations are accounted for.

11.2.4.4 Accidental Release of Waste Liquid

11.2.4.4.1 Identification of Causes and Accident Description

Accidents in the auxiliary system which could result in the release of waste liquid may involve the rupture or leaking of various components.

11.2.4.4.2 Analysis of Effects and Consequences

Liquid processing components are located within the auxiliary building, the solid waste building and the fuel pool leak monitoring room. Liquid leakage from the tanks (or release from other major components) is locally collected and transferred to sumps for subsequent pumping into the liquid waste disposal system.

An exception to the above statement is the recirculation piping on liquid waste receiving tanks located within the fuel pool leak monitoring room. Specifically, liquid leakage from the recirculation piping on these tanks has potential to reach outside the building and, therefore, could result in an inadvertent radioactivity release to the environment via the catch basin/storm sewer system. Radioactivity surveillances are performed periodically on these tanks (along with other outside tanks that contain radioactive liquids) to ensure such an event (i.e., leakage to the outside areas) would not result in exceeding the radioactive concentration limits.

Curbs and floor drains to the sump are employed to minimize the effect of leakage and spills inside plant buildings. Outboard seal leakage from the charging pumps (Section 9.1) is contained in this manner. The ventilation system collects any gaseous radioactivity and discharges it to the monitored ventilation vent as discussed in Section 9.13.

The primary coolant recovery tanks are located in the yard area within building cubicles of sufficient capacity to retain the total liquid volume resulting from rupture of either primary coolant recovery tank without overflowing to areas outside the cubicles.

Piping running between the auxiliary building and the reactor containment, the auxiliary and fuel buildings, and the fuel building and the tanks in the yard area, are run in concrete trenches. Liquid released from such piping is collected and transferred to sumps and pumped into the liquid waste disposal system.

The liquid waste inventories in the various tanks are based on the mode of operation during any particular time duration. In order to determine the liquid waste inventory for the various process tanks the following programs are used:

1. Program ACTIVITY[15] calculates the primary coolant equilibrium activity for a variety of input parameters. The inputs include data such as thermal power level, volume of primary coolant, fraction of failed fuel, purification flow rate, and primary coolant letdown rate. The library of the program contains factors for each fission product nuclide such as decay constant, escape rate coefficient, purification factor, fission yield, and absorption cross section.

2. Program IONEXCHANGER[16] calculates the total accumulation of radioactive nuclides in a tank. The input data may include the feed rate, nuclide activity concentration, bleed rate, container volume, and duration of feed.

11.2.4.4.3 Conclusions

Administrative controls and batch handling of all waste liquids ensures positive control of all processing. System liquid level indicators, radiation monitors, flow control instrumentation, and piping siphon break within the plant buildings should prevent an inadvertent radioactivity release to the environment.

When accidental spillage of waste liquids does occur within the plant buildings, it is contained within the station and does not result in any significant release of activity. As previously mentioned, other tanks that have the potential to leak radioactive liquids to outside areas are monitored periodically for activity determinations.

11.2.5 Solid Waste Disposal System

11.2.5.1 Design Bases

The solid waste disposal system provides facilities for the collection and preparation of radioactive waste materials for shipment to processing and disposal facilities. The various waste streams are prepared for shipment by filtration, dewatering, solidification, segregation, compaction, packaging, and/or storage. The materials which are handled as radioactive solid waste include depleted resins from process ion exchangers, concentrated waste solutions from the evaporator bottoms hold tanks, concentrated boric acid discarded from the boron evaporator bottoms hold tank, spent filter cartridges, and miscellaneous contaminated or irradiated solid materials (other than fuel). The filling of containers, storage and shipment of radioactive solid wastes conform with 10 CFR 20, 10 CFR 61, 10 CFR 71, and the Department of Transportation Regulations, Title 49 CFR parts 171-179.

Component design data for items of equipment appear in Table 11.2-3.

11.2.5.2 Description

Evaporator Bottoms Packaging

Concentrated liquids from the boron recovery system bottoms hold tank (Figure 9.2-1) and waste evaporator (Figure 11.2-3) may be pumped through coolers to the evaporator bottoms hold tank. The evaporator bottoms hold tank also may receive the filter backflush from LW-FL-5 (Figure 11.2-4). The contents of this tank may then be processed by the solidification system described below.

Spent Resin Handling Operations

A fluid transfer system permits spent resin to be flushed from each individual ion exchanger to a spent resin holding tank, or to a high integrity container (HIC). The capability also exists to transfer spent resin from the resin waste hold tank, using the resin waste hold tank metering pump, to an HIC. Two dewatering pumps (SW-P-10 &11) are installed to dewater the HIC to the spent resin hold tank. A dewatering tank (SW-TK-11) is installed to collect and meter the dewatering water during the final stages of the HIC dewatering process.

Resin in an ion exchanger is considered spent when the decontamination factor falls below a predetermined value, or the ion exchanger surface dose approaches a predetermined limit. The ion exchanger is then isolated, and recirculated primary grade water is used to flush the spent resin into the holding tank. Spent resin remains in the holding tank and the flush liquid passes out through filter elements and discharges to the spent resin dewatering tank. Excess water in the dewatering tank is discharged to the vent and drain system. After the transfer is complete, the resin in the holding tank is dewatered. The resin is dewatered and stored as required until shipped offsite. Flanges are provided to allow connection of a portable system in the event the spent resin handling system is out of service or resin of very high radioactivity must be processed.

In the case of demineralizer system LW-I-3, spent resin is sluiced directly to a solid waste liner or HIC in the solid waste building, dewatered, and stored for shipment offsite.

Expended Filter Cartridge and Solid Waste Handling

Cartridge filter elements are removed from service when the surface dose on the filter housing reaches a predetermined level, or when the element pressure drop becomes excessive. The operation is carried out using remote tools and a filter removal shield when required. The high activity filter cartridge is raised as a unit into the shield and a bottom pan attached. These are transported by monorail crane in the auxiliary building to a transfer vehicle for moving the assembly to the solid waste area. The spent filter cartridge is lowered into a container for subsequent offsite shipment. Other solid waste may be placed in the container. Containers are stored as required until they are to be shipped offsite for ultimate disposal.

Solidification System

The system is designed so that operation of the processing equipment will be remote and performed by a single individual from a remote control area. The solid waste area 25 ton crane is used for disposable container placement and handling operations for filling, storage, retrieval and trailer loading, within the solid waste handling area.

The system may process radioactive liquid wastes from the waste disposal evaporators and the boron recovery evaporators. The waste liquids will be mostly water with a nominal 12 percent by weight boric acid with varying amounts of radioactive and stable isotopes.

The system may also process spent resins and filter cartridges which will contain radioactive crud; the spent resins may be pumped to the Solidification System as a water slurry.

The waste may be solidified or dewatered in disposable containers. The disposable containers shall be filled, stored as required, and then transported in containers which meet all applicable regulations.

The resin waste hold tank with an agitator, dewatering elements and metering pump, is used to process the resin waste. A slurry of radioactive waste and water is pumped to the hold tank, the water is removed, through a filter and re-used for making other radioactive resin waste slurries. Evaporator bottoms or flush water may also be mixed with the dewatering resin, depending upon its radioactivity, and the resin mixture will be agitated to ensure that radioactive hot spots are minimized.

The system may be used to dispose of the spent filter cartridge baskets. The baskets, shielded by the Spent Filter Removal Shield, will be put into a container having an internal basket support.

Following solidification, the container shall be decontaminated, labeled and stored, as required, until shipment from the site.

Baling Operation

Contaminated compressible materials are stored in suitably labeled containers in Radiologically Controlled Areas.

Low level radioactive materials may be transported to the solid waste baler for compression into higher density bales. Additional compressible material is added, and the contents recompacted until the container is filled. During baling operation, the air flow in the vicinity of the baler is directed by an exhauster to a filter and the building ventilation vent.

On-Site Temporary Storage (Waste Handling Building)

The Waste Handling Building (WHB), which is located in the switchyard as shown in Figure 1.2-1, is a shielded facility for temporary storage of low level radioactive waste from BVPS Unit 1 and 2. The low level waste will normally consist of dry active waste, and dewatered resins and filters.

The WHB is a reinforced concrete structure designed to hold up to 5 years of waste generated by BVPS-1 or BVPS-2. The WHB's radiation protection criteria is based upon guidance provided in NRC Generic Letter 81-38. The two-foot thick reinforced concrete walls and roof are designed to reduce radiation levels to a maximum 2 mR/hr at the exterior of the building walls. The building roof is designed to reduce the estimated radiation levels inside the building to a maximum of 100 mR/hr at the top of the roof.

The cumulative dose due to the WHB, based on continuous occupancy at the site boundary, will be maintained at less than 3 mrem/yr. When a realistic occupancy factor of 2000 hr/yr is assumed, the dose rate to any "real" member of the public will be below 1 mrem/yr.

The WHB is designed for normal wind and local seismic loadings; however, site tornado or seismic design loadings do not apply.

Waste containers will be moved with a 10-ton capacity bridge crane, which is operated remotely from the crane control room via closed circuit televisions.

Fire protection for the WHB consists of the following:

- An automatic wet pipe sprinkler system and two fire hose stations for the truck bay area.
- Smoke detection for the waste storage areas. The detection system will alarm in the WHB's crane control room.
- A yard fire hydrant is provided adjacent to the WHB.
- An automatic wet pipe sprinkler system, a fire hose station, and portable CO_2 extinguishers for the personnel area.

No fire detection or suppression system is installed in the liner storage area because no free combustibles will be stored in this area.

Radiation monitoring is achieved by a manual, periodic surveillance program and portable area radiation monitors. An area monitor can be located in the inspection cell to determine radiation levels for each liner and box prior to shipment. An area radiation monitor is centrally located in the truck bay area to alert personnel if the radiation levels exceed the acceptable limit. Periodic surveillances are conducted to assure that radiation levels in various areas are within administrative limits.

The drainage of liquid within the WHB, except sanitary waste from the personnel area, is collected in the building sump. This sump will be periodically sampled for radioactivity. If the sump becomes contaminated, the sump pump can be lined up to a portable demineralizer for cleanup and then discharged in accordance with administrative requirements.

Shipment and Disposal

All packages containing radioactive material and the procedures used to prepare these for offsite shipment conform with NRC and U.S. Department of Transportation regulations. All waste material is either transferred to a licensed disposal contractor, to a licensed waste processor, or to a common carrier for delivery to a licensed disposal contractor, as appropriate.

Decontamination

A decontamination system is provided to wash shipping containers. Decontamination uses water, and if required, a measured amount of chemicals from a chemical addition tank. High pressure water is used for washdown of liners, if needed.

The decontamination system in the solid waste area also serves as an emergency drain of the solid waste system hold tanks. The decontamination tank may be used for temporary storage of the contents of either the resin waste hold tank or the evaporator bottoms hold tank. Two pumps are provided to empty the decontamination tank. The decontamination tank is furnished with an agitator which is used to prevent coagulation.

11.2.5.3 Evaluation

The solid waste disposal system retains batchwise control over all materials entering and leaving the system. Administrative Procedures ensure that materials are handled safely and are packaged securely for offsite shipment.

Malfunctions in the system do not affect the safety of station operations.

All piping and components are designed, fabricated and inspected according to applicable code requirements. Equipment is located and monitored to prevent unacceptable activity releases to the environment. Plant procedures specify the methods of operating the solid waste system. The Process Control Program (PCP) contains the methodology and boundary conditions to assure that all activities related to waste form are controlled and directed toward an end product meeting all regulatory requirements.

11.2.5.4 Process Control Program (PCP)

11.2.5.4.1 Procedures

Written procedures shall be established, implemented, and maintained covering the implementation of the PCP.

11.2.5.4.2 Changes to the PCP

Changes to the PCP shall be documented and records of reviews performed shall be retained in accordance with the applicable record retention provision of the quality assurance program description referenced in the Updated Final Safety Analysis Report. This documentation shall contain:

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

Changes to the PCP shall become effective after review and acceptance by the PORC and the approval of the plant manager, predesignated alternate or a predesignated manager to whom the plant manager has assigned in writing the responsibility for review and approval of specific subjects.

11.2.6 Tests and Inspections

11.2.6.1 Construction and Fabrication

During the manufacturing period, the Applicant's inspectors inspected all equipment periodically, to ensure that all equipment was provided in strict accordance with specifications. Shop hydrostatic and performance tests of principal equipment was witnessed by the Applicant's inspectors.

During the construction period, all pressure systems were subjected to field hydrostatic or pneumatic tests to verify the integrity of welded connections.

During the preliminary operation period, all equipment in the waste disposal system was tested to verify compliance with specification performance requirements. All control systems and interlocks were tested and operated to ensure satisfactory functional performance and reliability.

11.2.6.2 Operation

The basic function of the waste disposal system is to control the release of radioactivity to the environment so as to ensure that there are no harmful effects on the health and safety of the general public. This is accomplished by controlling all releases such that they will not result in offsite dose rates in excess of those permitted by applicable regulatory agencies.

The following is a list of the types of radiation monitors provided and areas monitored to ensure the proper functioning of the waste disposal systems:

1. CONTINUOUS PROCESS MONITORING: As described in Section 11.3.3, process radiation monitors continuously monitor certain key systems where radioactive material may exist.

2. BATCH SAMPLE PROCESS MONITORING: As described in Section 9.6, batch samples are obtained from systems which provide information on the effectiveness of ion exchangers, filters and evaporators. This gives an indication of the effectiveness of the various waste processing subsystems. Monitoring of the gas in the waste gas holdup avoids storage of excessive activity.

3. CONTINUOUS MONITORING OF DISCHARGE EFFLUENTS: As described in Section 11.3.3, radiation monitors continuously monitor discharges from the gaseous waste disposal system, the ventilation vent, the elevated release point, the liquid waste disposal system, and the various cooling water systems. These monitors give an indication of liquid and gas activity discharges to the environment and, in certain systems, actuate valves which terminate discharge if activity levels exceed a preset level.

4. RADIATION SURVEYS ON THE OUTSIDE OF CONTAINERS: Portable radiation detectors are used to survey spent resin casks and other containers which retain radioactive materials to ensure that maximum allowable external dose rate levels will not be exceeded for shipment.

5. RADIATION SURVEY OF SOLID WASTE CONTAINERS: Radiation surveys and smear samples are taken of all shipping casks, drums, etc. that contain solid waste to ensure that such waste is properly packaged and meets transportation regulations.

6. ENVIRONMENTAL MONITORING: As described in Section 2.8, environmental samples are taken to indicate the effect of liquid and gas discharges on the environment and the compliance of these discharges with applicable regulations. Effluent monitoring will be conducted in accordance with the requirements of NRC Regulatory Guide 1.21 as described in Section 1.3.3.21.

To ensure that the performance of the waste disposal system is meeting design criteria, the following checks are made:

1. Standardized laboratory radiochemical analytical procedures are used to verify decontamination factors.

2. Radiation monitors are periodically checked with check sources and, in addition, samples can be drawn from the process streams being monitored for spot checks of the activity levels. Those monitors which shut control valves and/or trip valves by a high-high radiation signal can be tested so as to actuate these valves.

3. Portable radiation detectors are periodically calibrated with known radiation sources in accordance with station radiological control procedures.

4. Radiation levels on the outside of components, pumps, valves, and piping in the waste disposal systems are monitored on a periodic basis to identify areas of potential personnel exposure.

References to Section 11.2

1. Reference deleted, Revision 0.

2. Cook and Basmadjan, "Correlation of Adsorption Equilibria of Pure Bases on Activated Carbon", Canadian Journal of Chemical Engineering, 146-151, (August 1964).

11.3 RADIATION PROTECTION - SHIELDING AND MONITORING

11.3.1 Design Bases

Radiation protection, including radiation shielding, is designed to ensure that the criteria specified in 10CFR20 are met during normal operation and that the guidelines suggested in 10CFR50.67 are met in the event of the Design Basis Accident (DBA) or other accidents (Section 14).

Allowable dose rates are based on the expected frequency and duration of occupancy. Occupancy time and dose rates are such that no personnel shall receive in excess of that recommended in 10CFR20.

The original design basis for plant shielding was safe operation at a core power level of 2766 MWt, a one year fuel cycle length, and 1-percent fuel element defects. The design basis target dose rates in plant areas, and the associated shielding design, presented in this section are based on the above design basis.

Core power uprate to 2900 MWt and operation with an 18-month fuel cycle represents a change from the original design basis. Taking into consideration the conservative analytical techniques used to establish the original shielding design, and the plant Technical Specifications which typically restrict the reactor coolant activity to levels significantly less than 1% fuel defects, the impact of the change in the core power level and the fuel cycle length has no significant impact on plant shielding requirements and safe plant operation.

Individual worker exposure is maintained within acceptable limits by the site ALARA Program, which controls access to radiation areas. Procedural controls compensate, as necessary, for increased radiation levels to ensure that operator exposure remains ALARA and that the normal operation radiation zones are labeled and controlled for access in accordance with the requirements of 10CFR20 related to allowable operator exposure and access control.

11.3.2 Shielding Design and Evaluation

11.3.2.1 Primary Shielding

Primary shielding is provided to limit radiation emanating from the reactor vessel. The radiation consists of neutrons diffusing from the core, prompt fission gammas, fission product gammas, and gammas resulting from the slowing down and capture of neutrons.

The primary shielding is designed to:

1. Attenuate neutron flux to prevent excessive activation of components and structures.

2. Reduce the contribution of radiation from the reactor to obtain a reasonable division of the shielding function between primary and secondary shields.

3. Reduce residual radiation from the core to a level which does not limit access to the region between the primary and secondary shields at a reasonable time after shutdown.

The primary shield consists of a water-filled neutron shield tank having a radial dimension of approximately 3 ft surrounded by 4 1/2 ft of reinforced concrete. The neutron shield tank is designed to prevent overheating and dehydration of the concrete primary shield wall and to prevent activation of the components within the containment structure. A thermosiphon cooling system is provided for cooling the water in the shield tank (Section 9.4).

A 17 ft high, 2 inch thick, cylindrical shield is located beneath the neutron shield tank to protect station personnel servicing the neutron detectors during reactor shutdown.

In order to maintain the integrity of the primary shield, supplementary neutron shields fabricated from masonite Benelex 401 are provided between the reactor vessel and shield tank below the reactor coolant pipe nozzles. Borated concrete neutron shield segments are provided under the reactor cavity seal plate in the annular gap between the reactor vessel flange and the primary shield concrete above the reactor coolant pipe nozzles.

The primary shield arrangement is shown in Figures 5.1-1, 5.1-2, 5.1-3, 5.1-4, 5.1-5, 5.1-6, and 5.1-7. The shield materials and thicknesses are listed in Table 11.3-1.

11.3.2.2 Secondary Shielding

Secondary shielding consists of reactor coolant loop shielding, reactor containment shielding, fuel handling shielding, auxiliary equipment shielding, waste storage shielding, accident shielding, crane wall shielding, yard shielding and control room shielding.

Nitrogen-16 is the major source of radiation in the reactor coolant during normal operation. The shielding required to attenuate its radiation establishes the minimum combined thickness of the crane and containment walls. The design value for nitrogen-16 activity is 70 μCi/cc at the reactor vessel outlet nozzle. Activated corrosion and fission products in the reactor coolant system establish the shutdown radiation levels in the reactor coolant loop areas. Table 14B-6 lists the fission and corrosion product isotopes which are considered in designing the containment secondary shielding. Fission product activities based on a core power level of 2766 MWt, a 12 month fuel cycle and one percent failed fuel was utilized to develop the secondary shielding design. Activated corrosion and fission products from the reactor coolant system are the radioactive sources for which shielding is required in the auxiliary and waste disposal systems.

Table 14B-6 lists the fission and corrosion product activities associated with power operation at 2910 MWt, an 18 month fuel cycle and one percent failed fuel. See discussion in Section 11.3.1 regarding shielding adequacy following power uprate.

11.3.2.3 Reactor Coolant Loop Shielding

Interior shield walls separate reactor coolant loop, pressurizer, incore instrumentation, and containment access sectors. This shielding allows access to the incore instrument sector during normal operation and facilitates maintenance in all sectors during shutdown. The crane support wall provides limited access protection in the annulus between the crane wall and the reactor containment wall and provides part of the exterior shielding required during power operation. Shield walls are provided around each steam generator above the operating floor to a height required for personnel protection. The shielding arrangement is shown in Figures 5.1-1, 5.1-2, 5.1-3, 5.1-4, 5.1-5, 5.1-6, and 5.1-7. The shield materials and thicknesses are listed in Table 11.3-1.

11.3.2.4 Containment Structure Shielding

The containment shielding consists of the steel-lined, steel-reinforced concrete cylinder and hemispherical dome (as described in Section 5). This shielding, together with the crane support wall, reduces the dose rate at the outside surface of the containment to less than 0.75 mrem per hr during full power operation. In addition, it reduces doses from the assumed Design Basis Accident to acceptable levels.

11.3.2.5 Fuel Handling Shielding

Fuel handling shielding is designed to facilitate the removal and transfer of spent fuel assemblies from the reactor vessel to the spent fuel pool by protecting personnel from the radiation emitted from the spent fuel and control rod assemblies.

The refueling cavity above the reactor vessel is flooded to about El. 766 ft to provide a temporary water shield above the components being withdrawn from the reactor vessel. The water height is thus approximately 27 ft above the reactor vessel flange. This height ensures a minimum of 100 inches of water above withdrawn fuel assembly at its highest point of travel. Under these conditions, the dose rate is less than 50 mrem per hr at the water surface.

Upon removal of the fuel from the reactor vessel, it is moved to the spent fuel pool by the fuel transfer mechanism via the refueling canal.

The spent fuel pool in the fuel building is permanently flooded to provide a minimum of 100 inches of water above a fuel assembly being withdrawn from the fuel assembly transfer cask. Water height above stored fuel assemblies is a minimum of 23 ft. The sides of the spent fuel pool, three of which also form part of the fuel building exterior walls, are 6 ft thick concrete to ensure a dose rate of no more than 2.5 mrem per hr outside the building.

11.3.2.6 Auxiliary Equipment Shielding

The auxiliary components exhibit varying degrees of radioactive contamination due to the handling of various fluids. The function of the auxiliary shielding is to protect personnel working near the various auxiliary system components, such as those in the chemical and volume control system, the boron recovery system, the waste disposal systems, and the sampling system. Controlled access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded so that compartments may be entered without having to shut down and, possibly, decontaminate the entire system. Ilmenite concrete is used in certain areas where substantial shielding is required and space is at a premium, such as the mixed bed demineralizer compartments. Tungsten shielding was used above the liquid waste demineralizer system LW-I-3.

Most ion exchangers and the most highly contaminated filters are located in separate cubicles along the east wall of the auxiliary building. Each ion exchanger or filter is enclosed in a separate, shielded compartment. The concrete thicknesses provided around the shielded compartments are sufficient to reduce the surrounding area dose rate to less than 2.5 mrem per hr. The shielding thicknesses around the mixed bed demineralizers are based upon a saturation activity which could produce a contact radiation level of nearly 9,000 rem per hr.

In many areas, tornado missile protection in the form of thick concrete affords more shielding than that required for radiation protection.

11.3.2.7 Waste Storage Shielding

The waste storage and processing facilities in the auxiliary building and solid waste building and the waste storage tanks in the yard are shielded to protect operating personnel in accordance with the radiation protection design bases set forth in Section 11.3.1.

Periodic surveys by radiological control personnel using portable radiation detectors ensure that radiation levels outside the shield walls meet design specifications, and establish access limitations within the shielded cubicles. In addition, continuous surveillance is provided at the auxiliary building drumming area and control area by area radiation monitors.

Area and process monitoring also ensures that any accidental radioactivity release would be detected within a reasonable period of time. The largest accidental radioactivity release from the waste disposal systems would be the rupturing one of the waste gas decay tanks. An analysis of this accident is made in Section 11.2.3.4. The gas in the waste gas decay tanks is monitored to ensure that the activity level in these tanks is never allowed to exceed the design level.

11.3.2.8 Accident Shielding

Accident shielding is provided by the containment structure, which is a reinforced concrete structure lined with steel. For structural reasons, the thicknesses of the cylindrical walls and dome are 54 inches and 39 inches, respectively. These thicknesses are more than adequate to limit the direct radiation dose at the exclusion boundary to acceptable values.

Additional shielding is provided for the main control room. This, together with the shielding afforded by its physical separation from the containment structure, ensures that an operator would be able to remain in the main control room for 30 days after an accident and not receive more than the integrated dose permitted by regulatory requirements.

11.3.2.9 Crane Wall Shielding

The crane support wall will provide limited access to the annulus between the crane wall and the containment structure wall and will provide additional exterior shielding during power operation. A removable shield is also provided to close off the opening of the crane wall opposite the containment personnel hatch.

11.3.2.10 Yard Shielding

Yard tanks requiring shielding will be shielded to a height required for personnel protection.

11.3.2.11 Control Room Shielding

The control room will be designed in accordance with 1971 General Design Criterion 19. The calculational methods and radiation sources used in designing the main control room shielding are discussed in Section 14.

11.3.2.12 Old Steam Generator Storage Facility (OSGSF)

The OSGSF is a non-safety related, non-seismic, reinforced concrete structure that provided interim storage for the original steam generators (OSG) and the Reactor Pressure Vessel Heads (RPVH). The Unit 1 OSGs and the RPVH were removed from the Reactor Containment during the Unit 1 Cycle 17 refueling outage and placed in the OSGSF. The OSGSF is designed to also house the Unit 2 RPVH if needed. The OSGSF is located outside the protected area but within the exclusion area site boundary. The general location of the OSGSF is shown on Figure 1.2-1.

The reinforced concrete walls and roof of the OSGSF and its access vestibule have been designed to ensure that the dose rates outside the facility are within the limits of 10 CFR 20 and 40 CFR 190. The interior of the OSGSF is classified as a high radiation area and is controlled in accordance with Technical Specification Administrative Controls.

The radiation dose assessment was accomplished using the SHIELD-SG and Multigroup Oak Ridge Stochastic Experiment (MORSE) computer codes. SHIELD-SG is a point-kernel program used to calculate direct doses at various distances from the source. MORSE is a Monte Carlo program that calculates direct and skyshine doses. (Reference 8)

11.3.3 Process and Effluent Radiological Monitoring System

11.3.3.1 Design Objectives

The process and effluent radiological monitoring systems will be designed in accordance with 1971 General Design Criterion 64. The process and effluent radiological monitoring system has been expanded to include those processes identified in NUREG-0737.

All normal and potential paths for release of radioactive materials during normal operation, and during anticipated operational occurrences will be monitored to ensure compliance with the requirements of 10CFR20, and NRC Regulatory Guide 1.21. The Administrative Controls section of the Technical Specifications delineates how the Offsite Dose Calculation Manual implements the requirements of Appendix I to 10CFR50.

The reactor coolant system will be monitored continuously for gross activity level. Maintaining the reactor coolant system activity within acceptable levels will ensure that the activity levels in the normally radioactive process auxiliary systems are at acceptable levels.

Normally nonradioactive systems which may become contaminated by leaks from radioactive systems will be monitored continuously to ensure that no condition hazardous to the operators or to the general public will be developed.

Selected process and effluent radiological monitoring channels will be designed as Seismic Category I to ensure their availability under accident conditions.

In the event of an accident, the process and effluent radiological monitoring system, in conjunction with the area radiation monitoring system (Section 11.3.4) will provide information on the concentration and dispersion of radioactivity throughout the unit. This will enable operating personnel to evaluate the severity and mitigate the consequences of an accident.

The following automatic actions will be initiated by the process and effluent radiological monitoring system to mitigate the consequences of postulated accidents:

1. Containment refueling ventilation may be terminated in the event of a refueling accident inside containment. Any activity released prior to accomplishing containment isolation may be directed to the Supplementary Leak Collection and Release System (SLCRS) filters.

2. Fuel building ventilation will be through the Supplementary Leak Collection and Release System (SLCRS). During fuel handling, flow may be directed through the charcoal filters.

3. Ventilation flow from all areas enclosing radioactive fluids, including loss-of-coolant accident recirculation fluids, will be diverted through charcoal filters in the event of a release of activity into that area.

4. All effluent discharges from the secondary system will be terminated in the event of a steam generator tube rupture.

5. All effluent flow to the environment will be automatically terminated or processed in the event of a high-high activity alarm condition or an ESF signal, depending on the nature of the initiating event.

The SLCRS charcoal filters are not credited in the accident analysis with respect to offsite dose analysis.

11.3.3.2 Continuous Monitoring

The monitoring system consists of separate independent channels each having:

1. Radiation detector with remotely operated check source except for steam effluent monitors: ADV, MSSV and AFTEX monitors

2. Read out device located in the main control room; alarms indicating equipment malfunction, "high" radiation level, and "high-high" radiation level

3. Independent power supply

4. Recorders to maintain a permanent record of radiological events.

Means are provided to purge each sampler in the system with clean fluid to preclude contamination of the samplers.

The operability of each channel can be checked daily with a check source which will be remotely positioned from the main control room. Tests and calibrations of the radiation monitors will be performed at intervals specified in the Offsite Dose Calculation Manual. To assist in the calibration and maintenance of the system, each channel (except the Auxiliary Feedwater Area Drain Tank Monitor) will be provided with a local read out device located near the detector. This meter will be a slave unit driven by the main control room equipment.

Reliable a-c power will be provided to the process and effluent radiological monitoring system by the vital and emergency buses. Motors which drive sample pumps in essential systems are fed by the emergency buses.

In addition to the visual alarms which are provided for each channel at the radiation monitoring panels in the main control room for equipment malfunction, "high" radiation level, and "high-high" radiation level, each channel will be provided with alarm relays for these alarms which will actuate the main control room annunciator system.

11.3.3.2.1 Locations To Be Monitored

The process and effluent radiological monitoring system channels are listed in Table 11.3-6.

Each of these channels, with the exception of the component cooling water monitor, the reactor coolant letdown gross activity monitors, and the auxiliary steam condensate monitor, will monitor potential radioactive effluent paths. The sample points will, therefore, be located downstream of the last point of possible fluid addition to the effluent being monitored.

Each detector will be located in an easily accessible area and will be provided with sufficient shielding to ensure that the required sensitivity will be achieved when the background radiation level is at the design maximum for the area.

11.3.3.2.2 Anticipated Concentration, Sensitivities and Ranges

A channel will be deemed to have the required sensitivity when the count rate indicated in the main control room with the detector exposed to the specified nuclide concentration equals or exceeds twice the count rate due to background.

Each channel will monitor gross concentrations and detector output will be measured in counts per minute (cpm). Each channel has a minimum range of three decades.

The anticipated concentrations of radionuclides in the various process and effluent streams will be at, or slightly above, normal background radiation. The sensitivity of the detectors monitoring these streams will ensure that abnormal conditions will be detected before they cause an undue hazard to the operators or the general public.

The quantities of principal nuclides in the process equipment are based on the plant operating at base load with 0.25 percent fuel defects. An estimate of the maximum expected inventory of the nuclides is shown in Table 11.3-4. This estimate is based on the original design of the liquid waste system and is retained as historical information. The average values of radiosotopic inventory are expected to be a factor of four to ten times less than the maximum expected due to periodic replacement of component internals such as resins and filters, batch operations of the evaporators, and the associated time duration to accumulate the maximum expected inventory.

The recirculation heat exchanger river water outlet radiation monitors will only be used in the event of a LOCA. They will ensure that any leaks in the recirculation heat exchanger headers leading to contamination of the river water system will be detected at levels well below those representing a hazard to the public.

The containment purge vent monitor will be exposed to the long- lived gaseous and volatile fission products that build-up in the containment atmosphere due to reactor coolant system leaks. The predominant isotope will be Kr-85, although Xe-133 and I-131 may be present in significant amounts. Tritium levels will be determined by sampling the containment atmosphere prior to purging.

The process vent monitors will be exposed to radioactive gaseous effluents from the unit. If the radioactive gaseous waste disposal equipment is operating with expected efficiency (Section 11.2), the radionuclide concentration in the process vent will be at, or slightly above, normal background levels. Krypton-85 is the only isotope expected to be present in significant quantities in the process vent effluent.

The liquid waste effluent monitor will be exposed to the normal radioactive effluents from the unit. The expected isotope composition of the liquid waste is given in Appendix 11A.

The reactor coolant letdown activity may range from negligible, when the unit is at the beginning of the fuel cycle, to about 50 μCi/cc if there is significant failed fuel. To provide monitoring over the entire range of expected radionuclide concentration in the reactor coolant, two detectors are provided.

A brief description of each of the channels in the process and effluent radiological monitoring system is provided in the following subsections. The minimum sensitivity of each channel is also discussed.

Each monitor utilized during normal plant operations is provided with a "high" alarm and a "high-high" alarm. The "high" and "high-high" alarm settings for potential effluent paths will be set in accordance with the Offsite Dose Calculation Manual. Values for these alarm settings are dependent on the maximum anticipated flow rates for each effluent. The sensitivity and ranges for each channel are indicated in Table 11.3-2. Alarm setpoints on accident monitoring equipment added as a result of NUREG 0737 are chosen to monitor doses in accordance with the Emergency Preparedness Plan.

11.3.3.3 Monitors

The process and effluent radiation monitors are described below.

11.3.3.3.1 Gaseous Waste Particulate Monitor

A sample system continuously draws a sample from the gaseous waste disposal system downstream of the gaseous waste disposal blowers. The sample flows through a moving filter paper having a collection efficiency of 99 percent for particle sizes greater than 0.3 micron. The amount of deposited activity is continuously scanned by a lead-shielded detector. This sample system is common to both the gaseous waste particulate and gas monitors and includes a pump with 1.5 hp motor, a flowmeter, an automatic flow regulator, and isolation valves. The sample system is controlled from the main control room. Normally, the sample is composed of effluent from the waste gas decay tanks and storage tank vents, and containment vacuum dilution air. A high-high activity alarm initiates automatic closure of valves downstream of the decay tanks, thus terminating the waste gas flow from the decay tank.

11.3.3.3.2 Gaseous Waste Gas Monitor

After the continuous sample described in the previous paragraph has passed through the particulate filter paper, it can be passed through a charcoal filter cartridge if desired. It is then drawn into the gaseous waste gas monitor assembly, which is a fixed lead-shielded sampler enclosing a detector. The sample activity is measured, and then it is returned to the gaseous waste disposal system. A high-high activity alarm initiates automatic closure of valves downstream of the decay tank, thus terminating the waste gas flow from the decay tanks. A purge system is integral with the gas monitoring system for flushing the sampler with clean air for purposes of calibration.

11.3.3.3.3 Ventilation Vent Particulate Monitor and Ventilation Vent Gas Monitor

The ventilation vent monitoring system draws a continuous sample from the ventilation vent exhaust line upstream of the point of discharge of the effluent to the environment. The sample flow and analysis is the same as described for the monitoring system used for the gaseous waste disposal system. All equipment is identical to that for the two gaseous waste channels. Normally, the sample is composed of effluent from the auxiliary building, and containment building which are each individually monitored. A high-high activity alarm signal from the auxiliary building or a Containment Isolation Phase A (CIA) will cause the corresponding effluent to pass through a prefilter/charcoal/HEPA filter complex in the supplementary leak collection and release system before discharge to the environment via the elevated release point.

Two high range noble gas monitoring systems continuously draw a sample from the ventilation vent exhaust line upstream of the point of discharge. Those systems are operationally and functionally similar to the Gaseous Waste high range noble gas monitors discussed in Section 11.3.3.3.23.

11.3.3.3.4 Reactor Building/Supplementary Leak Collection & Release System (SLCRS) Particulate Monitor and Gas Monitor (Discharge Point at Top of Reactor Containment Building)

The Reactor Building/Supplementary Leak Collection & Release System (SLCRS) monitoring system withdraws a continuous sample from the Reactor Building/SLCRS exhaust line upstream of the point of discharge at the top of the containment building. The sample flow and analysis is the same as described for the monitoring system used for the gaseous waste disposal system.

All equipment is identical with that of the two gaseous waste channels. Normally, the sample is composed of effluent from the leak collection areas, fuel building, and waste gas storage tank area which are each individually monitored. A high-high activity alarm signal from any of these sources will cause all of these effluents to pass through a prefilter/charcoal/HEPA filter complex in the supplementary leak collection and release system before discharge via the Reactor Building/SLCRS release point.

Two high range noble gas monitoring systems continuously draw a sample from the Reactor Building/SLCRS exhaust line upstream of the point of discharge. These systems are operationally and functionally similar to the Gaseous Waste high range noble gas monitors discussed in Section 11.3.3.3.23.

11.3.3.3.5 Auxiliary Building Ventilation Exhaust Monitors

The auxiliary building ventilation system consists of exhaust system "A" and exhaust system "B." A continuous gas sample from each exhaust system is first passed through an in-line, easily removable, charcoal filter cartridge (as desired) and then analyzed by a gas monitor similar to the gas sampler/monitor equipment described for the gaseous waste monitoring system. The building exhaust is normally discharged to the atmosphere via the ventilation vent duct. A high-high alarm will cause the exhaust effluent to be diverted through a prefilter/charcoal/HEPA filter complex in the supplementary leak collection and release system before atmospheric discharge via the elevated release point. With manual damper VS-D-7-8A normally closed during Mode 6, operator action is required to manually open the damper to accomplish diversion in the event of a high-high alarm.

The charging pump cubicle exhaust is not monitored by an auxiliary building ventilation monitor. The cubicle exhaust is discharged through the supplementary leak collection and release system release point, which is monitored by an effluent radiation monitor. A CIA signal will cause the charging pump cubicle exhaust effluent to be diverted through a prefilter/charcoal/HEPA filter complex in the supplementary leak collection and release system before atmospheric discharge.

11.3.3.3.6 Fuel Building Ventilation Exhaust Monitors

The fuel building ventilation exhaust is monitored by two redundant in-line detectors. A high-high radiation alarm may automatically divert the flow through the prefilter/charcoal/HEPA filter complex of the supplementary leak collection and release system before discharge to the elevated release duct. It will also activate the fuel building and containment evacuation alarm.

11.3.3.3.7 Containment Purge Exhaust Monitor

The containment purge exhaust is monitored by two redundant inline detectors. During the initial containment purge (prior to allowing personnel entry into containment for refueling operations) an activity alarm will signal the operator to manually actuate damper valves to divert the purge exhaust through the prefilter/charcoal/HEPA filter complex for subsequent discharge via the elevated release point. For the likely no-radiation-alarm condition, the normal purge cycle is completed and the containment is opened to atmosphere permitting entry for refueling operations. During refueling, normally the purge/exhaust system may maintain the containment at a slightly negative pressure. Flow may be directed through the SLCRS prefilter/charcoal/HEPA filter complex. A high-high activity alarm from either of the containment purge exhaust monitors, may automatically close the purge supply and exhaust isolation dampers in the containment building and activate the fuel building and containment evacuation alarm. The airborne activity in the containment atmosphere may be subsequently discharged at a controlled rate through the prefilter/charcoal/HEPA filter complex in the supplementary leak collection and release system to the elevated release point above the containment building.

11.3.3.3.8 Leak Collection Area Gas Monitor

The leak collection area exhaust is monitored by analyzing a sample continuously drawn from the discharge flow stream. The sample is passed through a removable charcoal filter cartridge (as desired) and then drawn into a fixed lead-shielded sampler enclosing the detector. The sample activity is measured and then returned to the leak collection area exhaust discharge line. A high-high activity alarm signal will cause automatic diversion of the air flow through the prefilter/charcoal/HEPA filter complex to the elevated release point above the containment. A purge system is integral with the gas monitoring system for flushing the sampler with clean air for purposes of calibration.

11.3.3.3.9 Waste Gas Tank Vault Ventilation Gas Monitor

The ventilation exhaust from the waste gas tank vaults is monitored by analyzing a sample continuously drawn from the ventilation discharge flow stream. This monitoring system and its operation are identical with the monitoring system and operation described for the leak collection area gas monitor.

11.3.3.3.10 Component Cooling Water Monitor

The component cooling water monitor continuously analyzes a composite sample drawn from the downstream leg of the primary plant component cooling water heat exchangers. The sample is drawn through an off-line sampler system containing a detector inserted in a well located in the flow stream of the off-line sampler. Detection of radioactivity indicates leakage of radioactive effluent from the reactor coolant system or an auxiliary system into the component cooling water system.

11.3.3.3.11 Liquid Waste Contaminated Drain Monitor

The liquid waste contaminated drain monitor continuously analyzes effluent downstream of the contaminated drains filter. The detector is located in an in-line, lead shielded sampler. A high-high activity alarm will automatically close the isolation valve to terminate flow. The sampler can be flushed with clean water from a flush line inlet upstream of the sample monitor and is easily removable in case of contamination.

11.3.3.3.12 Liquid Waste Effluent Monitor

The liquid waste which is discharged from the BVPS-1 is continuously analyzed by an effluent monitor. The detector is located in an in-line lead shielded sampler. The last possible point of radioactive effluent addition is upstream of the radiation detector installation. A high-high activity alarm automatically terminates flow by closing the discharge line isolation valves. The sample monitor can be flushed with clean water from a flush line inlet upstream of the sample monitor and is easily removable in case of contamination.

11.3.3.3.13 Condenser Air Ejector Vent Monitor

The condenser air ejector vent monitor continuously monitors the gaseous effluent from the condenser air ejector vent. The detector is located in a well in an in-line sampler. An alarm from this detector indicates a primary to secondary system leak. A high-high radioactivity alarm automatically diverts the discharge of noncondensables to the containment building for subsequent discharge to the environment via the elevated release point.

11.3.3.3.14 Steam Generator Blowdown Tank Discharge Monitor

The steam generator blowdown tank discharge effluent is continuously monitored by a radiation detection system. The detection system is an off-line, lead shielded sampler. Samples are taken from the cooled steam generator blowdown tank discharge pipe, and passed through the sample monitor. The sample is then returned to the steam generator blowdown tank discharge line. A high-high alarm indicates a primary to secondary system leak.

The sample monitor can be flushed with clean water from a flush line inlet to the sample monitor.

11.3.3.3.15 Steam Generator Blowdown Sample Monitor

The steam generator blowdown sample monitor analyzes a composite sample of blowdown effluent. The sample comes from a common header which acts as a manifold for effluent taken from the three steam generator bottoms. The detector is located in a well in an off-line sampler. A high-high activity alarm will alert the operator to sample each of the steam generator blowdown effluents. The individual sampling is accomplished by a valving arrangement which permits the operator to determine the source of the high-high activity. The sample monitoring system can be flushed with clean water from a flush line inlet upstream of the sample monitor.

11.3.3.3.16 Reactor Coolant Letdown Monitor

The gross activity of the reactor coolant is continuously monitored by two detectors. The samples are drawn from the reactor coolant letdown line and delayed to permit sufficient decay of the N16 isotope before they pass by the detectors. In this system, large variations in the activity levels are possible depending upon the amount of fission products leaked to the coolant. The alarm points can be set to provide graded indications of reactor coolant activity increases. This system can be flushed with clean water from a flush line inlet upstream of the sample monitor.

11.3.3.3.17 Component Cooling/Recirculation Spray Heat Exchangers River Water Monitor

This monitor analyzes a continuous sample of the river water discharged from the primary plant component cooling water heat exchangers during normal operation. Following a containment isolation phase B signal it monitors the combined river water discharge from the recirculation spray heat exchangers. The detector is located in a well in an off-line sampler. An activity measurement is indicative of a heat exchanger leak. Activity in the river water will be initially detected by either the Primary Plant Component Cooling Heat Exchanger River Water Monitor or (following a containment isolation phase B signal) the Recirculation Spray Heat Exchanger River Water Monitors, all of which are upstream of the Component Cooling/Recirculation Spray Heat Exchangers River Water Monitor, which monitors the combined flow path.

11.3.3.3.18 Recirculation Spray Heat Exchanger River Water Monitors

During normal operation, the recirculation spray heat exchangers are isolated from the river water lines. After an accident which activates the containment isolation phase B signal, river water passes through the recirculation spray coolers. During accident conditions, four detector (one for each recirculation spray heat exchanger) analyze a continuous sample obtained from each heat exchanger river water outlet. Each detector is located in a well in an off-line sampler associated with the river water line to be monitored. An activity measurement is indicative of a heat exchanger leak. A high-high alarm alerts the operator to take appropriate corrective action.

11.3.3.3.19 Auxiliary Steam Condensate Monitor

The auxiliary steam condensate monitor analyzes a continuous sample drawn from the condensate header. The detector is located in a well in an off-line sampler. An activity indication alerts the operator of a leak in any of the radioactive systems serviced by the auxiliary steam system.

11.3.3.3.20 Primary Plant Component Cooling Heat Exchanger River Water Monitor

This monitor analyzes a continuous composite sample of the river water discharged from the primary plant component cooling heat exchangers in service. The sample is drawn from a common header containing a mixture of river water from those heat exchangers in service. The detector is located in a well in an off-line sampler. A high-high alarm indicates a heat exchanger leak. A valving arrangement permits individual sampling of each heat exchanger to determine the location of the leak.

11.3.3.3.21 N-16 Main Steam Line Monitors

The N-16 main steam line monitors, one for each line, are adjacent-to-line monitors located in the main steam valve area. The purpose of these monitors is to provide an early indication of a steam generator primary to secondary tube leak that may be a precursor to a tube rupture and to identify the affected steam generator. The instrumentation cabinet, including the processing unit, display, and recorder, is located in the Service Building on 752' elevation. A Control Room annunciator signals alarm conditions. These monitors use scintillation detectors, multi channel analyzers, and microprocessor circuitry to measure the N-16 concentration in each of the main steam lines and to estimate and display the primary-to-secondary leakage (gpd) in the associated steam generator. Three alarm setpoints are provided.

Since the N-16 concentration in the reactor coolant system and the steam flow is a function of reactor power, an input to the processor adjusts the monitor response for changes in reactor power. Because of this dependency, the useful monitor range is limited to between 20% and 100% reactor power. In the event of a tube rupture that results in a reactor trip, the monitor display will not be useful. However, the recorder chart can be reviewed to determine the reading prior to the trip.

11.3.3.3.22 Spare Channels

The process monitoring system includes four spare channels which are not assigned to any specific monitoring operation. Three of these channels consist of a complete instrument chassis: one identical to the gaseous waste particulate monitor, one identical to the gaseous waste gas monitor, and one identical to the liquid waste effluent monitor. The fourth channel does not have an instrument chassis but includes cabinet space and all necessary provisions for the future addition of a channel similar to the liquid waste effluent monitor.

11.3.3.3.23 Gaseous Waste High Range Noble Gas Monitors

This monitoring system is capable of continuous monitoring and measurement of high concentrations of noble gases in gaseous waste effluent. The system is also capable of detecting low range particulates (alpha and beta) and radioiodines in the sample stream. The monitoring system obtains a sample from the gaseous waste effluent stream using separate sample and return lines. The outputs of this monitoring system are fed to a control terminal in the control room. The outputs are in units of μCi/cc for each radioactivity measurement channel and cfm for effluent flow rate.

Monitor readings are printed out upon operator request and in response to an alarm condition. The control console provides historical listings and release rate determinations. The console provides alarms and allows operators to control the pump and check source for the monitor remotely.

Provisions are made in the monitor piping for the collection of particulate and radioiodine grab samples for subsequent laboratory analysis. This capability provides for the monitoring of high activity radioiodines in the effluent stream.

A redundant high range noble gas monitoring system is also installed. The output of this monitoring system is fed to the control room via a data acquisition module and communication line isolators (CLI's). This monitoring system provides alarms, but has no control functions.

11.3.3.3.24 Atmospheric Dump and Main Steam Safety Valve Monitors

The steam generator atmospheric dump valve (ADV) and main steam safety valve (MSSV) monitor (Figure 10.3-1) consists of an externally mounted detector assembly surrounded by a lead shield. The detector assembly is located between the ADV and adjacent MSSV discharge piping risers. The lead shield is designed to allow detector operation under DBA conditions. Thus, the detector views portions of both discharge pipe risers. The pressure lift setpoint of the MSSV monitored, is the lowest lift pressure of the five MSSV's per steam generator. The effluent release flow rate is taken from flow transmitters on the main steam lines. This configuration is repeated for each of the three steam generator loops. The radiation monitor signals are displayed on count rate meters located on the radiation monitoring system racks in the control room, while the steam flow is displayed on the control room vertical boards.

11.3.3.3.25 Auxiliary Feedwater Pump Turbine Exhaust Monitor

The monitor (Figure 10.3-1) consists of an externally mounted detector assembly surrounded by a lead shield. The lead shield, designed to allow detector operation under DBA conditions, also encompasses one of the auxiliary feedwater pump turbine exhaust discharge piping risers within the assembly. The display of data for this effluent pathway is similar to that described above for the ADV and MSSV Monitors.

11.3.3.3.26 Auxiliary Feedwater Area Drain Tank Monitor

The auxiliary feedwater area drain tank discharge effluent is continuously monitored by a radiation detection system. This detection system takes a continuous sample from a receiver tank drain connection, passes the sample through a skid mounted liquid radiation monitor and returns the sample to the tank vent. A high-high activity alarm annunciates in the control room and automatically diverts the tank discharge from the yard oil interceptor to the safeguards tunnel sump. This tank normally receives influent from the auxiliary feed pump area floor drains via an oil interceptor.

11.3.3.3.27 Post-Accident Sampling System Liquid Sample Radiation Monitor

The liquid sample radiation monitor is an ion chamber-type gross gamma detector. It indicates the liquid sample radiation level and is located just upstream of the liquid sample capsule in the Post-Accident Sampling Panel. The reading is used to determine dilution size necessary to bring the reactor coolant sample radioactivity level down to the level appropriate for sample removal in the sample capsule.

11.3.3.3.28 Post-Accident Sampling System Gas Sample Stream Radiation Monitor

The gas sample stream radiation monitor is an ion chamber type gross gamma detector located downstream of the gas sample capsule in the Post-Accident Sampling Panel. This radiation monitor measures gas sample radioactivity level. The reading is used to determine dilution size necessary to bring containment atmosphere sample radioactivity level down to the level appropriate for sample removal in the sample capsule.

11.3.3.3.29 Post-Accident Sampling System Sample Box Ventilation System Radiation Monitors

The two sample box ventilation system radiation monitors are located in the sample box ventilation line. These monitors are placed parallel to each other and also with the main vent line. In the event of a PASS line failure, the radiation monitors will detect high radiation in the sample box atmosphere and will automatically actuate closure of containment isolation valves on the sample and return lines. The radiation monitors are redundant since each monitor closes one of the redundant containment isolation valves in each sample or return line that penetrates containment.

11.3.3.3.30 Blowdown Water Monitor

The blowdown water monitor is an offline gross activity gamma scintillator. It monitors steam generator blowdown activity on the discharge header of the blowdown transfer pumps to minimize the potential of radioactively contaminating the blowdown equipment. This monitor provides local and Control Room indication and alarms, but has no automatic isolation signals.

11.3.4 Area Radiation Monitoring System

11.3.4.1 General

The area radiation monitoring system reads out and records the radiation levels in selected areas throughout BVPS-1 and alarms (audible and visual) if these levels exceed a preset value or if the detector malfunctions. With the exception of the containment gas and particulate monitors, each detector reads out and alarms both in the main control room and at its unit location. Each channel is equipped with a check source remotely operated from the main control room. Strip charts produce a continuous permanent record of radiation levels while the detectors are functioning. Each channel has a regulated high voltage power supply. The area radiation monitoring system receives power from the emergency bus to ensure continuity of operation in the event of a loss of normal station power.

Area monitor locations and expected ambient radiation levels are given in Table 11.3-5.

The area radiation equipment is located near main pathways for a given building or portion thereof. In some cases, they are located at major work areas. In all cases, they provide a representative indication of the radiation level in that vicinity of the plant, but not necessarily the maximum which may be measured on contact with one of the shield walls. In those areas where radioactive material is routinely handled, such as the decontamination building, the monitors are strategically located so as to provide a representative reading.

The criteria used for obtaining a representative sample of the airborne radioactivity in a given area is to draw the sample from the exhaust duct servicing the respective area. The sample is taken from the exhaust effluent using an isokinetic nozzle which is inserted in the flow stream. The samples are continuously withdrawn, monitored, and displayed in the control room.

The airborne radioactivity sampling system supplements the inventory of radiation protection equipment, which includes portable air sampling equipment. Grab air samples are taken with the portable equipment at the discretion of the radiation protection supervisor in areas where work is scheduled to take place and a potential exists for inhalation exposure.

The fixed location air monitors are designed to remotely indicate buildup of airborne radioactivity in various areas of the plant. These airborne sampling locations have been selected on the basis of the potential for airborne activity occurring within specific work areas as determined from engineering evaluation and previous operating histories.

The routine and special survey procedures and details are documented in radiation protection procedures. In general, a control room operator will monitor the Radiation Monitoring System (which includes the multisampler) on a shift basis. This instrumentation will concurrently be monitored by the radiation protection personnel periodically for proper operation, source check, and prevailing radiation levels or trends.

The Radiation Monitoring System will be supplemented with portable (moving or fixed filter) Continuous Air Monitoring (CAM) equipment. Such equipment shall be used to provide airborne radioactivity monitoring where there is a potential for high radioactivity levels.

In addition, intermittent sampling utilizing a high volume air sampler is normally utilized to provide more positive assurance. Grab samples will be taken periodically during routine surveys, and the frequency shall be adjusted as conservatively suggested by station operating history and trends.

All monitoring is to be provided as near to the work as practicable, and in areas determined by radiation protection personnel to be either representative, or to result in conservative protective action.

The alarm setpoint of each area monitor is variable and is set at a calculated level above the normal background radiation level in the respective area.

Initial calibration of area radiation monitoring and airborne radioactivity monitoring equipment was performed by the supplier. The calibration is traceable to National Bureau of Standards (NBS) standard sources. The proper transfer source response was determined during the initial calibration. Specifications as to proper response to each transfer source will be set forth in appropriate calibration procedures. Corrective action is required if these specifications are not met. Additionally, all monitors are equipped with check sources that are normally exercised at intervals specified in the Technical Specifications, the Licensing Requirements Manual, or Offsite Dose Calculation Manual to confirm satisfactory operation.

Area monitors will be calibrated at intervals specified in the Offsite Dose Calculation Manual or after major repairs. This calibration is performed by exposure to a known transfer source. Calibration of airborne activity monitors will also normally be confirmed at intervals specified in the Offsite Dose Calculation Manual and after major repairs. Appropriate records of calibrations or checks will be maintained.

In the event of an alarm by an area monitor, the area will be evacuated and isolated as soon as possible. Access will be controlled until the cause of the alarm has been confirmed and appropriate measures have been taken. Alarm investigation will include:

1. Confirmation of monitor operability.

2. Local confirmation of the radiological status of the area as soon as practicable.

If airborne activity is known or suspected, the associated ventilation system exhausts for that area may be shut down, if possible. The shutdown system will be restored after it is established that local airborne radioactivity concentrations are at an acceptable level.

Portable area monitors and continuous airborne monitors (CAMS) which may be used to supplement the Radiation Monitoring System will also be calibrated semi-annually and after maintenance repairs. Periodic checks with a suitable source is standard practice. The frequency of these checks is stated in the radiation protection procedures.

Area monitors or airborne radioactivity monitors which do not satisfy the required specification will be removed from service and repaired as soon as practicable. Appropriate spares or backup equipment will normally be utilized, if operation of the affected monitor is required. Whenever this is not practicable, appropriate measures will be initiated to provide for the radiological safety of BVPS personnel and the general public.

The area radiation monitoring system includes particulate and gas monitors and fixed-position detectors. A brief description of the particulate and gas monitors included in the area radiation monitoring system is contained in the following text.

11.3.4.2 Containment Atmosphere Particulate Monitor

The containment atmosphere particulate monitor continuously draws a sample from the containment atmosphere into a closed, shielded system exterior to the containment. The sample is passed through a moving filter paper having a collection efficiency of 97 percent for particles greater than 0.3 micron. The amount of deposited activity is continuously scanned by a lead shielded detector having a sensitivity of 1.81×10^{-5} μCi/cc for F-18 in a background of 0.75 mr/hr with a measurement range of 10 - 10E7 cpm. The sample system, which is common to both the particulate and gas monitors, includes a pump with a 1.5 hp motor, a flowmeter, a flow regulator and isolation valves. The pump and motor are located inside the containment. A sample point is available for sampling the containment atmosphere for spectrum analysis in the laboratory.

11.3.4.3 Containment Atmosphere Gaseous Monitor

The containment atmosphere gaseous monitor takes the containment atmosphere sample after it has passed through the particulate filter paper and draws it through a sealed system to the containment gas monitor assembly. This detector is a fixed, lead shielded sampler enclosing a detector. The sensitivity of this detector is 1.3×10^{-7} μCi/cc for Ar-41 in a background of 0.75 mr/hr with a measurement range of 10 - 10E7 cpm. The sample activity is measured and returned to the containment.

A purge valve arrangement blocks the normal sample flow to permit purging the detector with a "clean" sample for calibration. Purged gases are discharged to the containment.

11.3.4.4 Multisample Particulate Monitor

The multisample particulate monitor is designed to analyze a particulate sample taken from any one of twelve locations. The sample is passed through a moving filter paper having a collection efficiency of greater than or equal to 99 percent for particles greater than 0.3 micron. The amount of deposited activity is continuously scanned by a lead shielded detector having a sensitivity of 1×10^{-9} μCi/cc for I-131 in a background of 0.75 mr/hr. The sample system, which is common to both the particulate and gas monitors, includes a pump with a 1.5 hp motor, a flowmeter, a flow regulating valve and isolation valves. This system includes isokinetic nozzles, a manifold and control valve system which permits selecting the area from which the sample is drawn.

The multisampler particulate and gas monitor is designed to be normally arranged to automatically sequence sample each of the twelve distinct controlled access locations periodically. These twelve locations are as follows:

1. One in the fuel building ventilation duct
2. Up to seven selectively distributed about the auxiliary building ventilation systems A and B
3. One in the solid waste handling area exhauster
4. One in the pipe tunnel (near safeguards area)
5. One in the main control room air ducting.
6. One in the suction of the Decontamination Building Roof Exhaust Fan.

The multisampler normally monitors the control room at all times, but can be programmed for automatic sequencing of sample points (normal operation) from the main control room, but can also be manually operated from the same location for extended monitoring of any one of the twelve sample points. In the event concurrent operations are performed in different work areas, the multisampler may be taken out of the automatic mode and be manually controlled to monitor alternately the concurrent operation or maintenance work areas. If the work environment or plant operations suggest constant radiation surveillance, based on the periodic routine of a special work permit survey, the multisampler could be positioned to monitor that location continuously, with radiological control personnel monitoring the concurrent effort utilizing portable or cart-mounted particulate and gas monitors. The multisampler particulate and gas monitor has a readout and alarm in the main control room, and also can printout on the plant computer system which has the ability to store both readings and alarm history.

A manual positioner is provided to allow continuous sampling of a chosen area. The filter can be programmed to remain in a fixed position for a specific time before advancing to the front of the detector. The valves which select the area being sampled can be synchronized to the step advance programmer to allow preset sampling time increments of 30 minutes to 24 hours for each of the twelve sample areas.

11.3.4.5 Multisample Gas Monitor

The multisample gas monitor analyzes the gaseous portion of the sample previously analyzed by the particulate monitor. The gaseous sample is drawn into a gas monitor assembly, which is a fixed, lead shielded sampler enclosing a detector. The sensitivity of this detector is 1×10^{-6} μCi/cc for Kr-85 in a background of 0.75 mr/hr.

11.3.4.6 Other Radiation Monitoring Equipment

This equipment consists of fixed-position, Geiger-Mueller and ion-chamber type gamma detectors and associated electronic equipment. These channels warn personnel of any increase in radiation level at locations where personnel may be expected to remain for extended periods of time.

The channels which comprise the area radiation monitoring system and their sensitivities and ranges are listed in Table 11.3-3.

11.3.4.7 Containment Area Monitors

The containment area high range radiation monitoring system consists of two high range area monitors installed on the crane wall above the operating floor at locations approximately 180 degrees apart. Data display are by readout modules and chart recorders located in the control room.

11.3.5 Control Areas

The walls bounding the main Control Room (described in Section 7) and the Control Room Emergency Ventilation system (described in Section 9.13.4) are designed to limit radiation dose to personnel inside the Control Room to 5 Rem TEDE for the postulated duration of all analyzed accidents with the exception of the Waste Gas System Rupture analysis, which utilizes a dose limit of 5 Rem whole body, or its equivalent to any part of the body

The potential dose to the plant operators in the Control Room is due to air intake/inleakage into the Control Room of the external cloud, and direct doses from 1) immersion in the external cloud, and 2) radiation sources in the containment and emergency systems located outside Containment including the intake filters in the Control Room. The accident analysis considers a conservative post-accident ventilation intake/exhaust rate in the Control Room to evaluate the dose from inhalation. Based on measured BVPS-1 Control Room pressurization leak test data, bounding flow rate values are used for the Control Room inleakage rate in the accident analysis.

The Control Room walls provide the necessary shielding to protect personnel from contained external sources such as the airborne radioactivity inside containment, as well as the external cloud due to Containment Building and ECCS leakage.

The BVPS-1 design basis accidents evaluated to determine the dose to the personnel in the common control room are listed below. Significant analysis assumptions are tabulated in the tables cited in parenthesis.

A. Loss of AC Powered Auxiliaries (Table 14.1-3)

B. Single RCP Locked Rotor Accident (Table 14.2-4b)

C. Fuel Handling Accident (Table 14.2-6)

D. Steam Generator Tube Rupture Accident (Table 14.2-9)

E. Main Steam Line Break Accident (Table 14.2-10)

F. Rod Ejection Accident (Table 14.2-12)

G. Loss of Coolant Accident (Table 14.3-14a)

The analyses indicate that the common BVPS-1 and BVPS-2 Control Room is habitable for all design basis accidents at BVPS-1. Postulated doses are tabulated in Table 14.1-1. For additional information regarding control room design features credited in the dose consequence analyses, see Section 14.3.5, sub-section titled "Control Room Habitability."

The integrated whole body applicable dose from the worst case radiological design basis accident was calculated to be below the criterion. Thus, the main control room walls which must be a minimum of 24 inches thick for tornado missile protection, provide more than adequate shielding from radiation.

Special consideration has been given to the design of penetrations and structural details of the main control room so as to establish an acceptable condition of leak tightness.

The air-conditioning systems are installed within the spaces served and designed to provide uninterrupted service under accident conditions. Upon a containment isolation phase B, or high radiation signal, the normal replenishment air and exhaust systems are isolated automatically from the main control room by tight closures in the ductwork. At the same time, the control room emergency pressurization system is initiated, and the outside air is introduced into the protected spaces after being cleaned by particulate and impregnated charcoal filters. This system can be used indefinitely to maintain the area pressure above atmospheric to ensure exfiltration. Analyses have determined that no provisions need to be made for radiological considerations for the ERF/TSC facility.

The radiation levels in the main control and relay rooms are measured by gamma monitors to verify safe operating conditions.

Two redundant seismically qualified gamma area radiation monitors (RM-RM-218A, B) monitor the control room area. These monitors alarm and automatically initiate control room isolation and emergency air supply upon reaching the high radiation level setpoint.

References to Section 11.3

1. Deleted by Revision 23.

2. Deleted by Revision 23.

3. Deleted by Revision 22.

4. Deleted by Revision 23.

5. Deleted by Revision 23.

6. Deleted by Revision 23.

7. Deleted by Revision 23.

11.4 RADIOACTIVE MATERIAL SAFETY

11.4.1 Materials Safety Program

11.4.1.1 General

Radiation protection procedures document the policies, procedures, guidelines and limits which are authorized, implemented and enforced as a radiation protection program. All personnel assigned to the site are required to follow these rules and procedures. These radiation protection procedures also establish the detailed program which prescribes the requirements for receipt, marking, handling and storage of all radioactive materials at BVPS. Station procedures govern each of these functions with regard to byproduct, source and special nuclear material. The station Radiation Protection Section controls the receipt, marking, handling and storage of all radioactive materials in accordance with written radiation protection procedures.

All BVPS-1 personnel who handle radioactive materials receive appropriate training and instruction. This training and instruction shall be adequate to assure that such materials are handled in an approved and safe manner.

11.4.1.2 Byproduct Material

Except for radioactive calibration sources, radiochemistry standards or material installed in instrumentation as check sources, the only other byproduct materials at BVPS are those generated as a result of reactor operations. All radioactive calibration sources and radiochemistry standards are retained in secured storage facilities when not in use. The receipt, storage, handling and use of these standards in accordance with radiation protection procedures requires authorization of the Manager, Radiation Protection and is normally limited to the Radiation Protection and Chemistry Sections. The use of unsealed reference byproduct material sources is normally limited to the radiochemistry laboratory as described in 11.4.2. Accountability procedures are established in radiation protection procedures which apply to all radioactive standards and account for any other byproduct materials such as contaminated materials and wastes, irradiated components and materials which are released from radiologically controlled areas.

The receipt possession and use of source, byproduct and special nuclear material as authorized by the Operating License will be in accordance with NRC regulations in 10 CFR Parts 30, 40 and 70, including 10 CFR Sections 30.33, 40.32, 70.23 and 70.31.

11.4.1.3 Special Nuclear Material

The use of Special Nuclear Materials at BVPS-1 is limited to the nuclear fuel, to fulfill core and reactor coolant system monitoring and surveillance requirements and as neutron sources. The initial receipt and storage of Special Nuclear Material is governed by a Special Nuclear Material license in accordance with 10 CFR 70.21 and will utilize the fuel storage building facilities for storage. Security and accountability practices consistent with NRC requirements are established and maintained in accordance with a BVPS-1 manual developed specifically for this purpose. Initial core loading and all subsequent handling of Special Nuclear Material will be in accordance with current and approved written procedures. Except for possible use of the neutron sources as calibration sources, these procedures shall be refueling or other special handling procedures and shall be limited to specific periods and/or operations.

11.4.1.4 Source Material

Procedures and practices consistent with the requirements of 10 CFR 40, related NRC guides and radiation protection procedures shall be followed for the source materials at BVPS-1. It is anticipated only limited use of source material will be experienced and that the use of all such material shall be controlled in accordance with radiation protection procedures, applicable regulations and BVPS Accountability Procedures.

11.4.2 Facilities and Equipment

The radiochemistry laboratory (RCL) is designed and equipped for safe handling of unsealed radioactive sources. Such handling shall be kept to a minimum. All sample preparation for analysis will be performed in this room. A separate chemistry laboratory is provided for all non-radioactive chemistry functions. Adequate separate laboratory equipment is provided for the two laboratories so that each is completely independent of the other. The wall, bench tops, shelving and floor surfaces of the radiochemistry laboratory (RCL) are constructed of materials designed to facilitate decontamination.

Radiochemistry laboratory (RCL) liquids are discharged to the radioactive liquid waste system. Fume hoods (at least three) are provided for handling and processing solutions. The hoods are maintained at a constant negative pressure and exhaust to a common monitored ventilation system vent through a filtering network.

Except for radioactive materials that are in use, all sources will be stored in secured storage facilities.

Radiation monitors are provided in the radiochemistry laboratory (RCL), both for general radiation surveillance of the area and for use as personnel monitors for personnel exiting the laboratory. A radioactive particulate air monitor is available for use when laboratory activities indicate the need for such additional monitoring. Personnel dosimetry equipment is worn by personnel at all times while in the radiochemistry laboratory (RCL). Protective equipment such as respirators, gloves, goggles, aprons, lab coats, coveralls and shoe covers are available in the area to be worn as needed.

11.4.3 Personnel and Procedures

The physical movement of fuel will be performed under the direction of a member of the station supervisory staff.

In addition to the basic training in radiological safety and the requirements of radiation protection procedures, each person who handles radioactive materials receives information and instructions related to the specific task so that the material can be handled in a safe manner. The information and instructions include, but are not limited to, the following:

1. Radiation and/or contamination levels which may be encountered

2. Monitoring requirements

3. Dosimetry requirements

4. Radiation exposure control procedures for the task

5. Contamination control practices for the task

6. Special techniques to be employed

7. Other precautionary steps or control measures required.

11.4.4 Required Materials

The licensee is authorized, by the BVPS-1 Facility Operating License to receive, possess, utilize and store certain special nuclear materials. These include:

1. Reactor Fuel for BVPS-1 pursuant to 10CFR70

2. Sealed Neutron Sources for reactor startup, instrumentation and monitor calibration and fission detection pursuant to 10CFR30, 40 and 70

3. Standards for sample analysis or instrument calibration, pursuant to 10CFR30, 40 and 70.

4. Byproduct and nuclear materials, such as produced by reactor operation, may be possessed and stored, but not separated, pursuant to 10CFR30 and 70.

Specific materials, their use and possession limits are given in the Unit 1 Operating License.

11.5 PERSONNEL RADIATION PROTECTION

11.5.1 Radiation Protection Program Organization, Objectives and Responsibilities

The Radiation Protection Program organization, objectives and responsibilities for Unit 1 are as provided in the Unit 2 UFSAR.

11.5.2 Dose Control

The basic criterion used for control of airborne radioactivity is that internal radiation exposure resulting from inhalation of airborne activity should be minimized to such an extent that monitoring to determine internal exposure should not be necessary consistent with maintaining Total Effective Dose Equivalent (TEDE) ALARA. Radioactivity may become airborne through operations such as welding or grinding a contaminated component, decontamination of such components, leakage from a system containing radioactive fluids or gases, or disturbing the deposited activity in various areas of the plant. An airborne sampling location is selected on the basis of potential for airborne activity within the work area.

The first defense against inhalation exposure is in the design of the ventilation system in the Radiologically Controlled Area. The design philosophy of the various ventilation systems includes consideration for minimization of airborne contamination. In addition, other steps are taken to maintain the inhalation exposure and TEDE ALARA. The first of these steps is to minimize the sources of potential airborne radioactivity. Since area or equipment contamination can become airborne, such contamination can be a significant source. This source is minimized by proper contamination control procedures. Such procedures are spelled out in the radiation protection procedures. However, a general outline is as follows:

1. Establish allowable limits for plant areas and maintain the major portion of the plant within these limits.

2. Limit the number of contaminated areas and decontaminate these periodically as warranted to maintain contamination levels in these areas as low as practicable.

3. Control all work which involves or potentially involves significant quantities of radioactive contamination, with the objective of maintaining levels as low as practicable and preventing the material from becoming airborne. Containment (enclosures around work area, glove box arrangements, etc.) will be considered for work with very high contamination.

4. Perform routine contamination surveys, checks, and work surveillance to ascertain status of areas and equipment.

5. Properly package, handle, and/or store radioactively contaminated materials.

6. Maintain stringent personnel frisking (contamination monitoring) requirements.

7. Control the movement of radioactively contaminated materials and the release of airborne radioactivity.

8. Decontaminate equipment to the maximum extent practical.

A continuing review of sources and methods of reduction of prevalent radioactive material available as potential airborne radioactivity will develop additional ways to reduce the potential for inhalation exposures. Such a review and its feedback is an integral part of the Radiation Protection Program for BVPS-1.

In addition to minimizing the sources of airborne radioactivity, a suitable monitoring program will assist in reducing inhalation exposures. In the event that any monitor or monitoring systems indicate a significant rise in contamination or radiation levels in excess of station limits, the activity will be evaluated immediately and the appropriate controls will be implemented.

If warranted by excessive levels of airborne activity, the area (or effluent) will be isolated (or ceased) as soon as possible, and access or personnel proximity will be restricted until suitable reductions in activity are effected.

To further ensure inhalation exposures are minimized, suitable respiratory equipment will be provided and worn if appropriate.

Operation of the plant, as well as certain maintenance and repair tasks, may require access to and work in all but a few areas of the plant. This includes areas which are defined by 10CFR20 as Radiation Areas, High Radiation Areas, and Very High Radiation Areas. The access time in the higher radiation fields is limited and will be of short duration. It is based on the time required to perform the immediate task and the dose accumulation of the worker(s).

The following controls are standard operating practice to minimize exposure to plant personnel:

1. Any area where potential radiological risks may be present is designated as a radiologically controlled area. In accordance with this practice, the primary plant and a portion of the Service Building are defined as the Radiologically Controlled Area. This includes Containment, the Auxiliary Building, the Fuel Building, the Decontamination Room, the PCA Shop, the Solid Waste Building, certain concrete cubicles and some adjacent service areas. Other satellite radiologically controlled areas may be established outside the Radiologically Controlled Area, if warranted. Radiation protection procedures are provided to govern the access to and the work within these areas.

2. A Radiological Work Permit is required for maintenance, repair work, or testing within radiologically controlled areas. The Radiological Work Permit will provide data on the radiological condition which will be encountered, the exposure for each member of the work party, and specify the necessary radiological controls to be followed in accomplishing this specific task. It is used to assist in exposure control. The procedure for implementation of the Radiological Work Permit is detailed in radiation protection procedures.

3. All Radiation Areas of the plant, as defined by 10 CFR 20, shall be maintained within a radiologically controlled area. It is expected that areas in the plant where radiation levels may exceed one mrem per hour will normally be within the Radiologically Controlled Area.

   Radiation areas are posted in accordance with 10 CFR 20. Radiation protection procedures define the specific posting and access requirements.

4. High Radiation Areas and Very High Radiation Areas, as defined by 10 CFR 20, will be posted in accordance with 10 CFR 20 and access to these areas will be controlled in accordance with Technical Specification 5.7 and radiation protection procedures.

5. During reactor operation, high radiation areas may result in certain areas which otherwise would not be high radiation areas. Operational, radiological, and administrative controls are in place to preclude unauthorized entry to these areas. Entries into the Reactor Containment Building or the Volume Control Tank Area during reactor operations will require authorization of the Nuclear Shift Supervisor of the applicable unit. The procedures for authorization to enter these areas are contained in the site administrative procedures and radiation protection procedures.

6. Process piping which may carry radioactive materials through habitable areas of the site are placed as shown on engineering drawings. Radiation exposure to plant personnel is minimized to as low as reasonably achievable (ALARA) by providing specifically designed radioactive pipe trenches, pipe tunnels and shielded areas for this radioactive piping and equipment. In addition, radiation exposure to plant personnel is limited by compliance with OSHA Section 1910.96, 10 CFR 20 and radiation protection procedures.

7. Dose control criteria are utilized to establish administrative control and check on the accumulated exposure of each individual. Their purpose is to ensure that all exposures are kept within 10 CFR 20 limits and as low as reasonably achievable (ALARA). These criteria are detailed in radiation protection procedures. Some of the provisions are:

   a. Each individual is authorized to receive exposure based on administrative controls prescribed in radiation protection procedures, or as required by 10 CFR 20.

   b. The individual's exposure status will be made available to the individual and his/her supervisor.

   c. Additional exposure authorizations may be granted after the thermoluminescent dosimeter (TLD) badges are processed or as prescribed in radiation protection procedures.

d. An exposure tracking system provides supervision with the individual's current radiation exposure status and/or exposure authorization. These are factored into job assignments and assist in maintaining exposures as low as reasonably achievable (ALARA).

8. Standard operating procedure requirements for dosimetry are described in the Unit 2 UFSAR Sections 12.5.2.2.5 and 12.5.3.1.1

Individuals may not exceed 10 percent of the applicable limits specified in 10 CFR 20 unless the individual's occupational radiation exposure is documented.

If non-BVPS employees are required to work in the Restricted Area, they are subject to the same rules as plant personnel.

Doses to a member of the public will be maintained within the limits set forth in 10 CFR 20.

11.5.2.1 Ventilation

In addition to providing a comfortable and breathable environment in all habitable areas of the plant, the ventilation system operates to prevent the build up of radioactive contamination in areas containing contaminated, or potentially contaminated systems.

Continuous air monitoring, intermittent sampling, and grab samples are utilized to detect airborne radioactivity in potentially contaminated areas. The actual airborne activity concentration can then be used to determine the appropriate control procedures to be implemented. See Sections 11.3.4 and 11.5.3.1.3 for more information about airborne radioactivity monitoring.

The remainder of Section 11.5.2 below describes analyses performed during station licensing to estimate potential doses and radioactivity concentrations. These analyses were based on assumptions and methodologies acceptable at that time and are part of the plant design basis. In 1994, the dose calculational and exposure control measures of 10 CFR 20.1001 -20.2402 replaced the earlier methodologies. The analyses described in this section, and the supporting tables, will not be updated since the new methodologies are not part of the design basis. Personnel exposures will be controlled pursuant to 10 CFR 20.1001 -20.2402.

Prior to initial operation of the plant, inhalation dose rates were predicted based on conservative estimates of equipment leakage. The methodology used to predict the inhalation dose rate in various plant buildings is described below.

Peak air concentrations for buildings which are expected to contain an inventory of airborne radioactivity are derived from estimates of machinery leakage and reduction factors described in Table 11.5-1. The assumptions of partition factors and reduction factors used to derive activity releases are described in Table 11.5-1. Estimates of equipment leakage used in the analyses are conservative with what experience has shown in an operating plant. The magnitude and location of plant leakages are also summarized in Table 11.5-1.

The method used for calculating the inhalation dose rate to the operator follows:

Dose Rate (REM/hr) = A (Ci/m^3) * B.R. (m^3/sec) * 3600 sec/hr * C.F. (REM/Ci)

where: A = equilibrium airborne activity concentration

B.R. = 8-hour breathing rate per NRC Regulatory Guide 1.4

C.F. = curie-dose conversion factor described in TID 14844

Specifically, expected airborne concentrations for each building and the associated inhalation dose to plant personnel are determined as follows.

11.5.2.1.1 Auxiliary Building

The ventilation air is exhausted from the auxiliary building by two 30,000 cfm each exhaust fans, and the total building free volume is approximately 1 x 10^6 cu ft. The ventilation exhaust air is normally discharged to the atmosphere through a ventilation stack projecting above the roof of the auxiliary building. The airborne nuclide concentration in the auxiliary building is shown in Table 11.5-2. The expected inhalation dose rate to plant personnel is approximately 6.5 x 10^{-2} mrem/hour.

Auxiliary Building Ventilation

The ventilation supply in the auxiliary building is to the general access areas. The ventilation exhaust is taken from shielded cubicles. A minimum air flow velocity of about 50 ft per min flows into each cubicle doorway. Areas that enclose filters, demineralizer, and pipe tunnels are shielded, inaccessible during normal operation, and covered with removable concrete slabs. In the event of a high-high radiation level in one of the two ventilation exhaust paths, the exhaust is shifted to the supplemental leak collection system main filter banks. The source of the leakage may readily be detected by the multisampler detection systems which periodically sample exhaust ducts for each elevation of the A and B ventilation systems. The samples are analyzed for radioactive particulate, gaseous, and iodine content. Two shielded area exhaust fans provide a total exhaust flow rate of 62,000 cfm and the total building volume is approximately 1 x 10^6 cu ft. The specific areas that are exhausted are listed in Table 11.5-3.

Auxiliary Building Source Terms

The estimates of equipment leakage and reduction factors are described in Table 11.5-1.

11.5.2.1.2 Turbine Building

The ventilation flow rate for the turbine building is provided by 10 exhaust fans. The summer ventilation exhaust rate is 750,000 cfm. The winter ventilation exhaust rate could be as low as 75,000 cfm. The turbine building free volume is approximately 4 x 10^6 cu ft. Internal recirculation fans mix the air from all levels. The equilibrium activity in the turbine building is based on an expected winter season exhaust rate of 75,000 cfm. The ventilation rate during the summer months could be 750,000 cfm which would reduce the equilibrium activity level and reduce the resulting inhalation dose by a factor of ten. The airborne concentrations of nuclides in the turbine building are shown in Table 11.5-4. The expected inhalation dose rate is approximately 4.1 x 10^{-3} mrem per hour.

Turbine Building Source Terms

The equilibrium activity in the turbine building is based on an expected winter season exhaust rate of 75,000 cfm. The ventilation rate during the summer months could be 750,000 cfm which would reduce the equilibrium activity level and reduce the resulting inhalation dose by a factor of 10.

11.5.2.1.3 Containment Building

The airborne nuclide activity inventory in the containment building is calculated using the parameters in Table 11.5-1. The equilibrium airborne activity concentrations in the containment atmosphere are shown in Table 11.5-5.

11.5.2.1.4 Fuel Building

The normal ventilation rate in the fuel building is 3,000 cfm (see Figure 6.6-1). The free volume is approximately 153,000 cu ft. The airborne concentration based on the parameter described in Table 11.5-1 are shown in Table 11.5-6. The average annual occupational inhalation dose rate is estimated to be 1.4 x 10^{-4} mrem per hour.

11.5.2.2 Expected Doses

An estimate of the external dose rate levels at various in-station locations and in areas internal to the building are described in Table 11.5-7. The basis is consistent with radioactive inventories expected in process equipment described in Table 11.3-4.

The expected annual doses to on-site personnel are governed by the controls imposed by the station supervision and/or radiation protection personnel as described in Section 11.3.6.1. However, dose estimates for in-station personnel for routine operation are expected to parallel those reported from operating plant experience as discussed below.

Extensive radiation shielding is provided on the basis of the maximum concentration of radioactive materials within each shielded region rather than on annual average values. For batch processes, as an example, the point of highest radionuclide concentration in the batching process (e.g., just prior to draining a tank) is assumed. The shielding and occupancy zones for normal operation are intentionally very conservative such that the normally received dose rates will likely be a fraction of the limits specified in 10 CFR 20.

The highest level of personnel exposure is anticipated to occur during shutdown and maintenance periods on systems containing items such as coolant purification filters, condensate, cleanup and radwaste demineralizers, ion exchange resins, charcoal adsorber units, and solid radwaste handling components. Since this is the case, the plant shielding and machinery locations have been designed to provide maximum laydown space, maximum working room, and minimum time required to perform operations consistent with reasonable operation of the plant. Experience gained in the operation of nuclear plants has been factored into these designs with the objective of minimizing the total man-rem exposure to plant personnel.

One survey reported by Charlesworth(1) at the April 1971 American Power Conference covered data obtained at seven operating water cooled reactor plants with a total plant worker dose of 1,700 man-rem during the previous year for an average of 244 man-rem per plant per year. In this survey, it was found that on an average 75 percent of these exposures were estimated to have been received during shutdown operations.

Another survey by Goldman(2) summarizes the results of 27 plant years of operation from operating reports. This survey indicated a range from 0.5 to 2.3 rem/year per man with limited data on number distribution of staff in several exposure categories. From these data, Goldman concluded that 19 plant years of operating data resulted in an inplant population average of 238 man-rem per plant year. These results were close to the 244 man-rem per plant year reported by Charlesworth.

Since most of the anticipated exposures are expected to occur under circumstances which do not lend themselves to analytical predication, shielding design will be based on worst case assumptions, and design features will be provided which will minimize exposures. Thus, actual exposures should by virtue of BVPS-1 design and control of personnel assignments, be consistent with the "as low as reasonably achievable (ALARA)" requirement.

11.5.3 Radiation and Contamination Surveying

Radiation and contamination surveying at BVPS-1 are performed as described in BVPS-2 UFSAR Section 12.5.3.2.

11.5.4 Radiological Safety

External dose of plant personnel and construction workers is controlled for each individual in accordance with dose control criteria rather than an arbitrary average allowable external dose rate for work areas. Dose control criteria are detailed in radiation protection procedures. The purpose of dose control criteria is to ensure that all exposures are within 10 CFR 20 limits and that the TEDE is as low as reasonably achievable (ALARA).

An exposure tracking system provides supervision with individual current radiation exposure status and/or exposure authorization. These are factored into job assignments and assist in maintaining exposures to as low as reasonably achievable (ALARA).

Occupational radiation exposure monitoring is performed as described in Sections 11.5.2 and 11.5.4.9. Radiation levels in all accessible areas are maintained as low as reasonably achievable (ALARA).

A comprehensive radiation protection program has been developed to monitor the radiation exposure received by persons working in, or making visits to the BVPS-1. This program meets all requirements established by the Code of Federal Regulations, the Commonwealth of Pennsylvania, the Department of Health and the Occupational Safety and Health Act of 1970.

An extensive environmental survey program was conducted by the NUS Corporation, Rockville, Maryland, prior to commercial operation. After commercial operation of the BVPS-1 was initiated, this program was assumed by the licensee, utilizing the services of other technical consultants.

A radiation protection program was established to survey various station areas to determine the amount and type of radiation and radioactive contamination that exists during all plant conditions. This information is used to establish working areas and type of clothing and equipment required as well as to contain radioactive materials.

Structures and systems are arranged and designed to minimize external exposures to plant personnel. Permanent and temporary shielding, individual cubicles for redundant components, physical separation of radioactive sources, adequate outdoor air supply ventilation and individual exhaust ventilation ducts for cubicles are some of the methods used to reduce the potential exposure of operating personnel.

Access to radiologically controlled areas is regulated by Administrative and radiation protection procedures which permit only authorized personnel to enter. These procedures include a Radiological Work Permit System. All work on systems or components where exposure to radiological or radioactive contamination is involved requires a Radiological Work Permit. The radiological hazards associated with the job are determined and evaluated. Radiological controls are listed on the permit as well as any other pertinent radiological data. Entry into a High Radiation Area or Very High Radiation Area is controlled in accordance with Technical Specification 5.7 and radiation protection procedures.

In the auxiliary building and service building, ventilation supply is directed to areas with the lowest potential level of contamination. Ventilation exhaust is provided at individual areas and cubicles where a higher potential for leaks or contamination exists. In this way, air flow is from areas of lower activity levels to areas of higher activity. Any leakage is exhausted from the immediate area of the leak. Flex connections are provided on ventilation ducts near filter banks and components. When these systems are opened for maintenance, hoses may be connected to the ventilation flex connections so that air may be exhausted from the areas that are opened.

Several structures are provided to assist in limiting the spread of radioactive contamination. Among these is the Fuel and Decontamination Building. This facility may be used for decontaminating plant equipment.

11.5.4.1 Radiation Protection Procedures

Exposure to operating personnel is kept "as low as reasonably achievable," and maintained within the limits of 10 CFR 20. This is accomplished by careful design and administrative procedures and controls.

Radiation protection procedures govern radiological controls and practices. Included in these procedures are a number of measures which are to be applied to work tasks involving significant radiation levels to reduce radiation exposure of individuals. These measures shall be reviewed during the preparation of work procedures involving significant radiation exposure, or prior to assignment of plant personnel to similar tasks not covered specifically by procedure, to determine if these measures are applicable to the work situation and/or environment. If these measures apply, the work situation will be evaluated to determine if significant exposure reduction will be accomplished and, if significant reduction is likely, the applicable measures will be applied with "as low as reasonably achievable" criteria being satisfied.

1. Prior Planning and Rehearsal

   Prior planning of work tasks should minimize the time spent in the radiation field, limit requirements for ingress to the work area and allow better overall control. Rehearsal should be used for portions of particularly complex tasks, so that these tasks may be completed without delay. Tryout of containment devices or enclosures, trial fit of tools and parts and similar measures on mockups will be used when practicable to reduce the total personnel exposure.

2. Shielding and Decontamination

   Attention will be given to the installation of temporary shielding for work tasks where high radiation fields exist. A corollary consideration will be the radiation exposure received during shield installation. Decontamination may also reduce radiation levels in an area and will be considered. Special beta shielding (eye protection, Anti-C clothing, etc.) shall be employed in work tasks involving high levels of beta radiation.

3. Special Tools

   Consideration will be given to the use of special tools which may speed the accomplishment of the work task or provide distance between the radiation worker and the radiation source. An evaluation will be conducted to ascertain that a special tool does not complicate the work task so that an extension in the time required to perform the task does not negate the reduction in dose due to the distance afforded by the tool. Certain procedures will specify special tools (such as refueling) whose use has proved to be radiologically effective. Such tools must be used when specified by procedure.

4. Reduction of Sources by Discharging, Flushing, Etc.

   When significant reductions in radiation levels can be achieved by flushing systems, discharging, etc., these methods shall be used where practicable.

5. Radiological Work Permit and Surveys

   Radiation surveys will be conducted prior to the start of maintenance jobs in radiation areas. The necessary radiological information will be identified on the Radiological Work Permit. Location of hot spots in the work area should be identified and marked. The permit will also specify all radiological exposure control methods which must be used during the performance of the specific assignment.

6. Work Practices

   Work practices will be specified in radiation protection procedures and followed by all personnel working in radiologically controlled areas.

7. Radiological Training

   All personnel who perform work in radiologically controlled areas will receive training in radiological control practices commensurate with their assigned task. Competency in this area will be verified for individuals granted unescorted access. Special training may be required prior to operations involving complex radiological control problems or procedures. This training will be conducted on an as-needed basis.

8. Job Debriefing

   Debriefing of involved personnel will be conducted following jobs involving new or complex problems, and a critique prepared to improve future methods of performing the job. Feedback to improve procedures will also be obtained through job supervisor and periodic general safety meetings.

11.5.4.2 Personnel Monitoring Systems

Personnel monitoring systems, as described in Unit 2 UFSAR Section 12.5.3.3, are used at Unit 1.

11.5.4.3 Personnel Protection Equipment

11.5.4.3.1 Protective Clothing

Clean Anti-contamination (Anti-C) clothing consisting of such items as underclothing, coveralls, shoe covers, cloth and rubber gloves, and hoods will be available at the access point to radiologically controlled areas. It will be worn as required by posted instructions in the work area and/or by the Radiological Work Permit which specifies radiological conditions and controls for maintenance and testing involving potential radioactive contamination. In general, unless this clothing is excessively contaminated, it will be laundered, monitored, and reused. The use of Anti-C clothing is restricted to radiologically controlled areas, unless other use is approved by radiation protection supervision. It is mandatory that all persons remove all Anti-C clothing prior to exiting radiologically controlled areas of the plant.

Additional clothing such as plastic suits (wet suits), respiratory equipment, plastic air hoods, face shields, goggles, etc., may be provided as warranted by the work to be performed.

11.5.4.3.2 Respiratory Equipment

Whenever the potential for high airborne particulate activity exists, full-face high-efficiency filter masks, supplied air masks, self-contained breathing apparatus, or evacuation of the area (if no suitable protective devices are immediately available) may be exercised. Use of respiratory protection is based on TEDE ALARA considerations.

All respiratory protective devices used for radiological purposes at the site are the full-face design sealed about the face and neck, or the completely enclosed hood type. Most types may be supplied with positive pressure of breathing quality air.

Personnel will be trained in the proper use of respiratory equipment. They will be cautioned as to the limitations of the respirators and instructed in methods to ensure a proper fit. Special pairs of spectacles designed specifically for use with the particular respiratory equipment shall be provided as required. The respirators will be handled and inspected in accordance with the requirements of Department of Labor, Occupational Safety and Health Administration rule set forth in 29 CFR 1910.134 Chapter XVII.

11.5.4.4 Change Room Area

A change room has been provided so that personnel may obtain the clean protective clothing when required prior to entry into radiologically controlled areas of the station. Facilities are also provided for personnel decontamination. These areas are routinely surveyed for contamination.

11.5.4.5 Decontamination Facility

A decontamination facility is provided at the station for decontamination of equipment and tools. No equipment or tools are shipped from the site that have not been surveyed and decontaminated to the Department of Transportation Standards as a minimum.

11.5.4.6 Access Control

In order to protect personnel from accidental ingress to high radiation areas, warning signs, audible and visual indicators, barricades, and locked doors are used as required. Administrative procedures are established to control access to radiation, high radiation, and very high radiation areas.

11.5.4.7 Laboratory Facilities

BVPS-1 includes a laboratory with adequate equipment for detecting, analyzing, and measuring the types of radiation of concern, and for evaluating radiological problems which may be anticipated. Counting equipment including Geiger-Muiller, scintillation, and scalers are provided in an appropriately designed counting room for detecting and measuring radiation as well as equipment to identify the specific energy level(s) that are present in a particular sample. Equipment and facilities to prepare and analyze certain bioassay samples are provided.

11.5.4.8 Instrumentation

Portable radiation survey instruments are provided for use by radiation protection, operating, and maintenance personnel. A sufficient number are available to allow for use, calibration, maintenance, and repair. The types of instruments used include those for detecting and measuring alpha, beta, gamma, and neutron radiation. Personnel and tool monitoring instruments are located at exits from the Radiologically Controlled Area. These instruments assist operating personnel in detecting and preventing the spread of contamination.

Permanently installed area radiation monitoring equipment is included in the facility. These detectors continuously monitor radiation levels within certain areas as discussed in Section 11.3.4. These monitors will have an indicator readout in the main control room and present an audible alarm both in the area of concern and the main control room.

11.5.4.9 Bioassay Program

The bioassay program, as described in Unit 2 UFSAR Section 12.5.3.5.1.2, is applicable at Unit 1.

11.5.4.10 Handling Methods and Special Shielding For External Radiation

In general, the plant is designed to provide adequate shielding to reduce radiation exposures in accessible areas to acceptable levels for normal operations and maintenance.

One of the steps taken to maintain plant personnel exposure as low as reasonably achievable (ALARA) is to evaluate temporary shielding requirements for work tasks in areas where high radiation fields are a potential or exist. Before commencing work in such an area, a survey will be made of that work area followed by an evaluation of the need and use of shielding as well as other protective measures. If this evaluation indicates that shielding is warranted and the installation is practicable, shielding is utilized and installed in accordance with instructions. Shielding methods may include wrapping piping or other sources of hot spots with lead sheeting, lead wool blankets, lead shot bags, lead brick, or other suitable shielding material. In some cases, the source of radiation may be isolated by construction of temporary shield walls. For major repair work involving high radiation, consideration is given to design and utilization of special temporary shielding which is tailored to suit work requirements of the job. It is a general practice to cover the lead shield whenever there is a potential to contact internal system surfaces. This is to prevent radioactive contamination of the shielding material itself and to preclude introduction of lead as a chemical contaminant to a system. If warranted, another suitable shield material may be substituted for lead in such cases.

Another method of exposure control when handling radioactive materials will be to control the time of each individual's effort to minimize exposure. Advantage will be taken of remote handling to increase the distance of personnel from the radioactive source and thus reduce external and internal exposure where practicable.

References to Section 11.5

1. D. G. Charlesworth, "Water Reactor Plant Contamination and Decontamination Requirements," A Survey Conducted by the Subcommittee on Nuclear Systems, Presented at the 33rd Annual Meeting of the American Power Conference, American Society Mechanical Engineers Research Committee on Boiler Feed Water Studies, Chicago (April 1971).

2. M. I. Goldman, "Radioactive Waste Management and Radiation Exposure," Nuclear Technology, Volume 14 (May 1972).

TABLES FOR SECTION 11

Table 11.1-1

RADIATION MONITOR LOCATIONS

Radiation Monitor	Elevation
Radiochemistry Laboratory area monitor (RM-1RM-214)	742'0" Service Bldg.
Component cooling/ recirculation spray heat exchanger river water monitor (RM-1RW-100)	697' Turbine Bldg.
Liquid waste contaminated drains monitor (RM-1LW-116)	722' 6" Aux. Bldg.
Auxiliary building area monitor (RM-1RM-211)	727' 0" Aux. Bldg.
Reactor coolant letdown high and low range monitors (RM-1CH-101A, B)	722' 0" Aux. Bldg.
Component cooling water monitor (RM-1CC-100)	722' 6" Aux. Bldg.
Auxiliary building area monitor (RM-1RM-209)	727' 0" Aux. Bldg.
Steam generator blowdown sample monitor (RM-1SS-100)	735' 6" Aux. Bldg.

Table 11.1-1 (CONT'D)

RADIATION MONITOR LOCATIONS

Radiation Monitor	Elevation
Sample room area monitor (RM-1RM-212)	740' 6" Aux. Bldg.
Component cooling heat exchanger river water monitor (RM-1RW-101)	735' 6" Aux. Bldg.
Gaseous waste particulate and gas monitor (RM-1GW-108B)	752' 6" Aux. Bldg.
Ventilation vent particulate and gas monitor (RM-1VS-101A, B)	752' 6" Aux. Bldg.
Auxiliary building ventilation exhaust system "A" gas monitor (RM-1VS-102A)	752' 6" Aux. Bldg.
Auxiliary building ventilation exhaust system "B" gas monitor (RM-1VS-102B)	752' 6" Aux. Bldg.
Leak collection area gas monitor (RM-1VS-105)	752' 6" Aux. Bldg.
Waste gas tank vault vent exhaust gas monitor (RM-1VS-106)	752' 6" Aux. Bldg.
Elevated release particulate and gas monitor (RM-1VS-107B)	752' 6" Aux. Bldg.
Containment particulate and gas monitor (RM-1RM-215A, B)	752' 6" Aux. Bldg.

Table 11.1-1 (CONT'D)

RADIATION MONITOR LOCATIONS

Radiation Monitor	Elevation
Auxiliary building area monitor (RM-1RM-210)	757' 0" Aux. Bldg.
Liquid waste effluent monitor (RM-1LW-104)	727' 0" Aux. Bldg.
Drum handling and storage area monitor (RM-1RM-208)	740' 0" Aux. Bldg.
Fuel building ventilation exhaust gross gas activity monitors (RM-1VS-103A, B)	787' 0" Fuel Bldg.
Fuel pool bridge area monitor (RM-1RM-207)	774' 0" Fuel Bldg.
New fuel storage area monitor (RM-1RM-206)	756' 0" Fuel Bldg.
Decontamination building area monitor (RM-1RM-205)	750' 0" Decon. Bldg.
Containment purge exhaust gross gas activity monitors (RM-1VS-104A & B)	795' 0" Containment
Reactor containment high range area monitor (RM-1RM-201)	773' 0" Safeguards Bldg.

Table 11.1-1 (CONT'D)

RADIATION MONITOR LOCATIONS

Radiation Monitor	Elevation
Reactor containment low range area monitor (RM-1RM-202)	773' 0" Containment
Manipulator crane area monitor (RM-1RM-203)	773' 0" Containment
Incore instrument transfer device area monitor (RM-1RM-204)	738' Containment
Recirculation spray heat exchanger river water monitors (RM-1RW-100A, B, C, & D)	722' 6" Safeguards Bldg.
Auxiliary feedwater area drain tank monitor (RM-1DA-100)	722' 6" Safeguards Bldg.
Condenser air ejector monitor (RM-1SV-100)	721' 0" Turbine Bldg.
Auxiliary building ventilation exhaust noble gas monitor SA 9/10 (RM-1VS-111)	768' 7" Aux. Bldg.
Reactor bldg/SLCRS noble gas monitor SA 9/10 (RM-1VS-112)	768' 7" Aux. Bldg.
Process vent noble gas monitor SA 9/10 (RM-1GW-110)	768' 7" Aux. Bldg.
Containment high range area monitors (RM-1RM-219A, B)	767' 10" Containment

Table 11.1-1 (CONT'D)

RADIATION MONITOR LOCATIONS

Radiation Monitor	Elevation
Main steam atmospheric & safety valves effluent monitors (RM-1MS-100A, B, C)	791' Roof of Main Steam Valve Cubicle
Auxiliary feedwater pump turbine exhaust monitor (RM-1MS-101)	791' Roof of Main Steam Valve Cubicle
Auxiliary Building Ventilation Exhaust Monitor PING206S/NGM203S (RM-1VS-109)	752' 6" Aux. Bldg.
Reactor Bldg./SLCRS Monitor PING206S/NGM203S (RM-1VS-110)	752' 6" Aux. Bldg.
Process Vent Monitor PING206S/NGM203S (RM-1GW-109)	752' 6" Aux. Bldg.
Blowdown water monitor (RM-1BD-101)	693'6" Turbine Bldg.
Unit 1 Control Room Area Radiation Monitors (RM-1RM-218A,B)	745'0" Service Bldg.

Table 11.2-1

GASEOUS WASTE DISPOSAL SYSTEM COMPONENT DESIGN DATA

Decay Tanks	GW-TK-1A, 1B & 1C
Number	3
Internal Volume, Each, ft^3	132
Design Pressure	0 psia to 100 psig
Design Temperature, F	200
Operating Pressure, psig	65
Operating Temperature, F	100
Material	Carbon Steel
Design Code	ASME III(1)

Surge Tank	GW-TK-2
Number	1
Internal Volume, ft^3	52
Design Pressure	0 psia to 100 psig
Design Temperature, F	300
Operating Pressure, psig	65
Operating Temperature, F	200
Material	ASTM Type 304 SS
Design Code	ASME III(1)

Gas Waste Charcoal Beds	GW-TK-3A, 3B, 3C & 3D
Number	4
Internal Volume, Each, ft^3	35
Charcoal Volume, Each, ft^3	25
Design Pressure	0 psia to 100 psig
Design Temperature, F	240
Operating Pressure, psig	2
Operating Temperature, F	52
Material	ASME Type 304 SS
Design Code	ASME III C(1)

(1) Maintenance block valves in the pressure relief path are administratively controlled and locked in the open position during system operation.

Table 11.2-1 (CONT'D)

GASEOUS WASTE DISPOSAL SYSTEM COMPONENT DESIGN DATA

<u>Gas Waste Chillers</u>	<u>GW-E-1A & 1B</u>	
Number	2 (Double Tubing Units)	
Total Duty, Btu/hr	100	
	Outer <u>Tube</u>	Inner <u>Tube</u>
Design Pressure, psig	150	100
Design Temperature, F	150	240
Operating Pressure, psig	93	2
Operating Temperature, In/Out, F	45/55	110/52
Material	Carbon Steel	ASTM Type 304 SS
Fluid	Water	H_2 98.5% Vol., N_2 and Trace Noble Gases (Radioactive) 1.5% Vol.
Design Code	ANSI B31.1	ANSI B31.1
<u>Tank Vent Filter</u>	<u>GW-FL-3</u>	
Number	1	
Type	HEPA, with viscous type prefilter	
Capacity, scfm	1000	
Design Pressure	Atmospheric	
Design Temperature, F	250	
Operating Temperature, F	250 Max.	
Operating Pressure, in. W.G.	-2	
<u>Dilution Air Filter</u>	<u>GW-FL-2</u>	
Number	1	
Type	HEPA, with viscous type prefilter	
Capacity, scfm	1000	
Design Pressure	Atmospheric	
Design Temperature, F	250	
Operating Temperature, F	100	
Operating Pressure, in. W.G.	-2	

Table 11.2-1 (CONT'D)

GASEOUS WASTE DISPOSAL SYSTEM COMPONENT DESIGN DATA

Gaseous Waste Disposal Charcoal Filters and Gaseous Waste Disposal Particulate Filters	GW-FL-4A & 4B GW-FL-1A & 1B
Number	2 (assemblies)
Type	Activated Charcoal, with HEPA afterfilter
Capacity, rated, scfm	1200
Design Temperature, F	115
Design Pressure	Atmospheric
Operating Temperature, F	100
Operating Pressure, in. W.G.	-5
Overhead Gas Compressor Prefilters	**GW-FL-5A & 5B**
Number	2
Type	Cartridge
Capacity, cfm	3
Design Temperature, F	240
Design Pressure, psig	15
Operating Temperature, F	52
Operating Pressure, psig	2
Fluid	Hydrogen
Gaseous Waste Disposal Blowers	**GW-F-1A & 1B**
Number	2
Type	Centrifugal
Motor Horsepower, hp	15
Capacity each, scfm	1110
Differential Pressure, in. W.G.	20.5
Suction Pressure, in. W.G.	-13
Design Pressure, psig	15
Materials	
Casing	Cast Iron
Impeller	Carbon steel
Shaft	Steel
Overhead Gas Compressors	**GW-C-1A & 1B**
Number	2
Type	Diaphragm
Motor Horsepower, hp	3
Capacity, Each, scfm	2.0
Discharge Pressure at Rated Capacity, psig	100
Materials Diaphragms and Parts Contacting Waste Gas	ASTM Types 304 & 316 SS

Table 11.2-2

LIQUID WASTE DISPOSAL SYSTEM COMPONENT DESIGN DATA

Note: Full in parenthesis indicates total volume.

Evaporator Test Tanks	LW-TK-5A &5B
Number	2
Capacity, Each, gal	3000 (full)
Design Pressure, psig	0.5
Design Temperature, F	150
Operating Pressure	Atmospheric
Operating Temperature, F	130
Material	ASTM Type 304 SS
Design Code	API 650

Laundry and Contaminated Shower Drain Tanks	LW-TK-6A & 6B
Number	2
Capacity, Each, gal	1300 (full)
Design Pressure, psig	Atmospheric
Design Temperature, F	150
Operating Pressure	Atmospheric
Operating Temperature, F	100
Material	Carbon Steel
Design Code	ASME VIII

Low Level Waste Drain Tanks	LW-TK-3A & 3B
Number	2
Capacity, Each, gal	2000 (full)
Design Pressure, psig	25
Design Temperature, F	150
Operating Pressure	Atmosphere
Operating Temperature, F	100
Material	ASTM Type 304L SS
Design Code	ASME VIII

Evaporator Distillate Accumulator	LW-TK-4
Number	1
Capacity, Each, gal	230 (full)
Design Pressure, psig	100
Design Temperature, F	300
Operating Pressure	15
Operating Temperature, F	250
Material	ASTM Type 304 SS
Design Code	ASME VIII

Table 11.2-2 (CONT'D)

LIQUID WASTE DISPOSAL SYSTEM COMPONENT DESIGN DATA

Note: Full in parenthesis indicates total volume.

High Level Waste Drain Tanks	LW-TK-2A & 2B
Number	2
Capacity, Each, gal	5000 (full)
Design Pressure, psig	25
Design Temperature, F	150
Operating Pressure	Atmospheric
Operating Temperature, F	100
Material	ASTM Type 304L SS
Design Code	ASME VIII

Demineralizer	LW-I-1
Number	1
Design Flow, gpm/ft²	7
Demineralizer Resin	H-OH Mixed Bed
Active Volume, cu ft.	35
Design Pressure, psig	175
Design Temperature, F	200
Operating Pressure, psig	40
Operating Temperature, F	130
Material	ASTM Type 304L SS
Design Code	ASME VIII

Evaporator	LW-EV-1
Number	1
Capacity, gpm	6
Design Pressure, psig	100 and Full Vac.
Design Temperature, F	338 and 100
Operating Pressure, psig	15
Operating Temperature, F	260
Materials	ASTM Type 304 SS INCOLOY-825
Design Code	ASME VIII

Table 11.2-2 (CONT'D)

LIQUID WASTE DISPOSAL SYSTEM COMPONENT DESIGN DATA

Note: Full in parenthesis indicates total volume.

Evaporator Bottoms Cooler	LW-E-4	
Number	1	
Total Duty, Btu/hr	210,000	
	Shell	Tube
Design Pressure, psig	100	100
Design Temperature, F	350	350
Operating Pressure, psig	60	25
Operating Temperature, In/Out, F	150/171	255/165
Material	Carbon Steel	Incoloy 825
Fluid	Water	Waste liquid with up to 12% boric acid and up to 25% total solids (Radioactive)
Design Code	ASME VIII	ASME VIII
Evaporator Overhead Condenser	LW-E-2	
Number	1	
Total Duty, Btu/hr	3.31×10^6	
	Shell	Tube
Design Pressure, psig	100	75
Design Temperature, F	350	350
Operating Pressure, psig	15	60
Operating Temperature, In/Out, F	250/250	105/125
Material	ASTM TYPE 304 SS	ASTM Type 304 SS
Fluid	Steam (Radioactive, Boric Acid to 1,140 ppm)	Water
Design Code	ASME VIII	ASME VIII
Evaporator Reboiler	LW-E-1	
Number	1	
Total Duty, Btu/hr	3.9×10^6	
	Shell	Tube
Design Pressure, psig	150	100
Design Temperature, F	360	360
Operating Pressure, psig	100	15

Table 11.2-2 (CONT'D)

LIQUID WASTE DISPOSAL SYSTEM COMPONENT DESIGN DATA

Note: Full in parenthesis indicates total volume.

Operating Temperature, In/Out, F	338/338	253/265
Material	Carbon Steel	Incoloy 825
Fluid	Steam	Waste Liquid with up to 12% boric acid and up to 25% total solids (Highly radioactive)
Design Code	ASME VIII	ASME VIII

Evaporator Distillate Cooler	LW-E-3	
Number	1	
Total Duty, Btu/hr	360,000	
	Shell	Tube
Design Pressure, psig	150	100
Design Temperature, F	350	350
Operating Pressure, psig	60	30
Operating Temperature, In/Out, F	105/120	250/130
Material	Carbon Steel	ASTM TYPE 304 SS
Fluid	Water	Water
Design Code	ASME VIII	AMSE VIII

Evaporator Bottoms Cooler Preheater	LW-EH-1	
Number	1 (Tubular Electric Heater)	
Total Duty, Btu/hr	20,000	
	Shell	Tube
Design Pressure, psig	100	100 (External)
Design Temperature, F	350	600
Operating Pressure, psig	60	0
Operating Temperature, In/Out, F	70/150	200/200
Material	Carbon Steel Water	Carbon Steel Electric Heating Elements
Design Code	ANSI B31.1	ANSI B31.1

Demineralizer Filter	LW-FL-3
Number	1
Retention Size, Microns	5
Filter Element Type	Wound Fiber
Capacity Normal, gpm	50
Capacity Maximum, gpm	50
Design Pressure, psig	150

Table 11.2-2 (CONT'D)

LIQUID WASTE DISPOSAL SYSTEM COMPONENT DESIGN DATA

Note: Full in parenthesis indicates total volume.

Design Temperature, F	250
Operating Pressure, psig	65
Operating Temperature, F	130
Material	ASTM Type 304 SS
Design Code	ASME VIII
Waste Effluent Filters	LW-FL-1A & 1B
Number	2 (one required)
Retention Size, Microns	5 to 25
Filter Element Type	Wound Fiber
Capacity Normal, each, gpm	30
Capacity Maximum, each, gpm	50
Design Pressure, psig	150
Design Temperature, F	250
Operating Pressure, psig	30
Operating Temperature, F	130
Material	ASTM Type 304 SS
Design Code	ASME VIII
Contaminated Drains Filters	LW-FL-2A & 2B
Number	2 (one required)
Retention Size, Microns	5
Filter Element Type	Wound Fiber
Capacity Normal, gpm	10
Capacity Maximum, gpm	20
Design Pressure, psig	150
Design Temperature, F	250
Operating Pressure, psig	65
Operating Temperature, F	100
Material	Carbon Steel
Design Code	ASME VIII

Table 11.2-2 (CONT'D)

LIQUID WASTE DISPOSAL SYSTEM COMPONENT DESIGN DATA

Evaporator Circulation Pump	LW-P-5
Number	1
Type	Horizontal Centrifugal
Motor Horsepower, hp	5
Seal Type	Double Mechanical
Capacity, gpm	700
Head at Rated Capacity, ft.	13
Design Pressure, psig	150 at 250 F
Materials	
Pump Casing	GA20
Shaft Sleeve	GA20
Impeller	GA20
Evaporator Bottoms Pump	LW-P-8
Number	1
Type	Horizontal Centrifugal
Motor Horsepower, hp	1.5
Seal Type	Mechanical
Capacity, gpm	10
Head at Rated Capacity, ft	65
Design Pressure, psig	255 at 160°F
Evaporator Bottoms Pump	LW-P-8
Materials	
Pump Casing	ASTM A296 CF8M
Shaft Sleeve	CF8M
Impeller	ASTM A296 CF8M
Low Level Waste Drain Tank Pumps	LW-P-1A & 1B
Number	2
Type	Horizontal Centrifugal
Motor Horsepower, hp	7.5
Seal Type	Mechanical
Capacity, each, gpm	30
Head at Rated Capacity, ft	200
Design Pressure, psig	150 at 350 F
Materials	
Pump Casing	Stainless Steel
Shaft Sleeve	Stainless Steel
Impeller	Stainless Steel

Table 11.2-2 (CONT'D)

LIQUID WASTE DISPOSAL SYSTEM COMPONENT DESIGN DATA

<u>High Level Waste Drain Tank Pumps</u>	<u>LW-P-2A & 2B</u>
Number	2
Type	Horizontal Centrifugal
Motor, hp	2
Seal Type	Mechanical
Capacity, each, gpm	30
Head at Rated Capacity, ft	80
Design Pressure, psig	150 at 350 F
Materials	
Pump Casing	A-351, Gr. CF8M
Shaft Sleeve	SAE 4140
Impeller	A-351, Gr. CF8M
<u>Evaporator Bottoms Cooler Circulating Pump</u>	<u>LW-P-10</u>
Number	1
Type	Horizontal Centrifugal
Motor, hp	2
Seal Type	Mechanical
Capacity, gpm	30
Head at Rated Capacity, ft	77
Design Pressure, psig	150 at 350 F
Materials	
Pump Casing	ASTM A-395, Ductile Iron
Shaft Sleeve	SAE 4140
Impeller	ASTM Type 316 SS
<u>Evaporator Distillate Pump</u>	<u>LW-P-4</u>
Number	1
Type	Horizontal Centrifugal
Motor Horsepower, hp	1
Seal Type	Mechanical
Capacity, Each, gpm	10
Head at Rated Capacity, ft	44
Design Pressure, psig	150 at 350 F
Materials	
Pump Casing	ASTM A-351, Gr. CF8M
Shaft	SAE 4140
Impeller	ASTM A-351, Gr. CF8M

Table 11.2-2 (CONT'D)

LIQUID WASTE DISPOSAL SYSTEM COMPONENT DESIGN DATA

Evaporator Test Tank Pumps	LW-P-9A & 9B
Number	2
Type	Horizontal Centrifugal
Motor Horsepower, hp	3
Seal Type	Mechanical
Capacity, Each, gpm	50
Head at Rated Capacity, ft	79
Design Pressure, psig	150 at 350 F
Materials	
Pump Casing	ASTM A-351, Gr. CF8M
Shaft	SAE 4140
Impeller	ASTM A-351, Gr. CF8M

Contaminated Drains Transfer Pumps	LW-P-6A & 6B
Number	2 (One Required)
Type	Horizontal Centrifugal
Motor Horsepower, hp	3
Seal Type	Mechanical
Capacity, Each, gpm	10
Head at Rated Capacity, ft	91
Materials	
Pump Casing	Bronze (Goulds 1103 Alloy)
Shaft	SAE 4140
Impeller	Bronze (Goulds 1103 Alloy)

Table 11.2-3

SOLID WASTE DISPOSAL COMPONENT DESIGN DATA

<u>Spent Resin Dewatering Tank</u>	<u>SW-TK-1</u>
Number	1
Capacity, gal	1400 (full)
Design Pressure, psig	40
Design Temperature, F	120
Operating Pressure	Atmospheric
Operating Temperature, F	70 to 120
Material	ASTM Type 304 SS
Design Code	ASME VIII
<u>Decontamination Tank</u>	<u>SW-TK-4</u>
Number	1
Capacity, gal	1400 (full)
Design Pressure, psig	40
Design Temperature, F	120
Operating Pressure	Atmospheric
Operating Temperature, F	70 to 120
Material	ASTM Type 304 SS
Design Code	ASME VIII
<u>Chemical Addition Tank</u>	<u>SW-TK-7</u>
Number	1
Capacity, gal	14 (full)
Design Pressure	Atmospheric
Design Temperature, F	120
Operating Pressure	Atmospheric
Operating Temperature, F	45 to 100
Material	ASTM Type 304 SS
<u>Resin Waste Hold Tank</u>	<u>SW-TK-2</u>
Number	1
Capacity Operating, cu ft	86
Capacity Full, cu ft	60
Design Pressure, psig	5
Design Temperature, F	250
Operating Pressure	Atmospheric
Operating Temperature, F	50 to 80
Material	ASTM Type 304 SS, 12 Gauge
Design Code	API App J

Table 11.2-3 (CONT'D)

SOLID WASTE DISPOSAL COMPONENT DESIGN DATA

Decontamination Filter	SW-FL-3
Number	1
Retention Size, microns	5
Filter Element, Type	Wound Filter
Capacity Normal, gpm	30
Capacity Maximum, gpm	50
Design Pressure, psig	150
Design Temperature, F	250
Housing Material	AISI Type 304 SS
Design Code	ASME VIII

Spent Resin Dewatering Tank Pump	SW-P-1
Number	1
Type	Horizontal Centrifugal
Motor Horsepower, hp	3
Seal Type	Mechanical
Capacity, gpm	50
Head at Rated Capacity, ft	81
Design Pressure, psig	150 at 350 F
Materials	
Pump Casing	ASTM A-296, Gr. CF8M
Shaft	ASTM A-276, Type 316 SS
Impeller	ASTM A-296, Gr. CF8M

East Waste Area Sump Pumps	SW-P-2A & 2B
Number	2
Type	Rotary
Motor Horsepower, hp	3/4
Seal Type	Mechanical
Capacity, each, gpm	10.5
Head at Rated Capacity, ft	60
Materials	
Pump Casing	ASTM Type 316 SS
Shaft	ASTM Type 316 SS
Rotor	H. R. Comp.

Tank Vent Filter	SW-FL-4
Number	1
Filter Element Type	HEPA
Capacity, scfm	1000
Design Pressure, inches W.G.	15
Design Temperature, F	180
Housing Material	ASTM Type 304 SS

Table 11.2-3 (CONT'D)

SOLID WASTE DISPOSAL COMPONENT DESIGN DATA

Metering Pumps	SW-P-6-7
Number	2
Type	SSR, MOYNO
Motor Horsepower, hp	1 1/2
Seal Type	Mechanical
Capacity, each, gpm	0-17
Normal Head and Capacity	10 psig at 10 gpm
Maximum Discharge Head, psig	20
Materials-Casing, Rotor, Internals	ASTM Type 316 SS
-Stator	Natural Rubber
Decontamination Pump	SW-P-4
Number	1
Type	Horizontal Centrifugal
Motor Horsepower, hp	5
Seal Type	Mechanical Seal
Capacity, each, gpm	30
Head at Rated Capacity, ft	50
Materials	
Pump Casing	ASTM Type 316 SS
Shaft	ASTM Type 316 SS
Impeller	ASTM Type 316 SS
Decontamination Pump (Resin Slurry)	SW-P-4A
Number	1
Type	Horizontal Centrifugal
Motor Horsepower, hp	7 1/2
Seal Type	Mechanical Seal
Capacity, each, gpm	70
Head at Rated Capacity, ft	60
Materials	
Pump Casing	Cast Iron A48-64 with neoprene liner
Shaft	C-1045 steel - A108
Impeller	Stell core - neoprene cover

Table 11.2-3 (CONT'D)

SOLID WASTE DISPOSAL COMPONENT DESIGN DATA

<u>Solid Waste Baler</u>	<u>SW-TK-2</u>
Number	1
Type	Hydraulic Ram With Hood and Dust Collection System
Capacity	55 gal Drums/max 1,000 lb
Material Handled	Paper, Cloth and Other Dry Waste; Possibly Radioactive
<u>Evaporator Bottoms Hold Tank</u>	<u>SW-TK-8</u>
Number	1
Capacity, Operating, cu ft	150
Capacity, full, cu ft	180
Design Pressure, psig	5
Design Temperature, F	250
Operating Pressure	Atmospheric
Operating Temperature, F	180
Material	ASTM Type 304 SS
Design Code	API-650 App J
<u>Dewatering Pump</u>	<u>SW-P-10-11</u>
Number	2
Type	Diaphragm Pump
Capacity, GPM	150 gpm at 110 psig air pressure

Table 11.2-3 (CONT'D)

SOLID WASTE DISPOSAL COMPONENT DESIGN DATA

Power Supply	Station Air
Dewatering Sample Collection Tank	SW-TK-11
Number	1
Capacity	5 gallons

Table 11.3-1

CONTAINMENT AND CONTAINMENT CONTIGUOUS
AREAS SHIELDING SUMMARY

Shield Description(2)	Material(1)	Thickness, Inches
Neutron Shield Tank	Water Steel	34 3
Primary Shield	Concrete	54
Upper Supplementary Neutron Shield	Borated Concrete	14
Lower Supplementary Neutron Shield	Benelex 401	8
Neutron Shield Tank Support	Steel Lead	1 1/2 2
Cubicle - Crane Support Wall	Concrete	33
Crane Support Wall above Operating Floor	Concrete	24
Containment Wall	Concrete	54
Containment Dome	Concrete	30
Floor (Elevation 718' -6")	Concrete	24 and 54
Operating Floor	Concrete	24
Refueling Cavity Wall	Concrete	36
Missile Shield above Refueling Cavity	Concrete	24
Refueling Cavity	Water	100 minimum
Pressurizer/Steam Generator Shield Walls above Operating Floor(3)	Concrete	12
Fuel Transfer Canal Wall	Concrete	54 to 72
Fuel Transfer Tube Shielding	Concrete Steel Block	48

Table 11.3-1 (CONT'D)

CONTAINMENT AND CONTAINMENT CONTIGUOUS AREAS SHIELDING SUMMARY

Incore Instrumentation Cubicle Wall	Concrete	42
Steam Generator Cubicle Wall	Concrete	36
Regenerative Heat Exchanger Cubicle Wall	Concrete	24
Cable Vault Wall	Concrete	24
Pump Room below Main Steam Valve Area Wall	Concrete	36
Safeguards Area Wall	Concrete	24

Notes:

1. All concrete is reinforced with steel.
2. See Figures 5.1-1, 5.1-2, 5.1-3, 5.1-4, 5.1-5, 5.1-6, and 5.1-7.
3. The removable block walls around the Steam Generators facing the access hatch are 18 inches thick.
4. The drain opening in the reactor cavity wall is arranged to minimize direct line-of-sight radiation streaming through the opening, and is provided with a permanent 3-1/4 inch thick box-shaped steel shield at the outside end of the opening.

Table 11.3-2

PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEM

Channel	No.	Medium	Limiting Isotope	Sensitivity(1) Microcuries/cc	Range (2) Decades	Maximum(3) Back-ground, mr/hr	Medium Temp, F	Medium Pressure, (Max) psig	Ambient Temp, F	Ambient Pressure psig
PING206S/NGM203S (Gaseous Waste, Ventilation, Reactor Building/SLCRS Release)	3	Gas, Air								
Alpha Particulate			Pu-239	2.7×10^{-13}	8	1.0	90-150	Negligible	50-104 23-131(8)	0
Beta Particulate			SR-90/ Y-90	2.7×10^{-11}	6	1.0	90-150	Negligible	50-104 23-131(8)	0
Iodine			I-131	1×10^{-10}	6	1.0	90-150	Negligible	50-104 23-131(8)	0
Noble Gas (Low Range)			XE-133	1×10^{-6}	8	1.0	90-150	Negligible	50-104 23-131(8)	0
Noble Gas (High Range)			XE-133 Kr-85	2.7×10^{-5} 1.0×10^{-4}	9	1.0	90-150	Negligible	50-104 23-131(8)	0
SA-9/SA-10 (Gaseous Waste Ventilation, Reactor Bldg/SLCRS Release)	3	Gas, Air								
Noble Gas SA-10			XE-133	2.3×10^{-5}	6	1.0	90-150	Negligible	50-120	0
Noble Gas SA-9			XE-133	8.6×10^{-2}	6	1.0	90-150	Negligible	50-120	0

Table 11.3-2 (CONT'D)

PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEM

Channel	No.	Medium	Limiting Isotope	Sensitivity(1) Microcuries/cc	Range (2) Decades	Maximum(3) Back-ground, mr/hr	Medium Temp, F	Medium Pressure, (Max) psig	Ambient Temp, F	Ambient Pressure psig
SGADV/MSSV Effluent	3	Steam	Mixed	2.2×10^{-2} (Note 6)	5	Note 5	510	N/A	-20 to 140	0
Aux. FWP Turbine Exhaust	1	Steam	Mixed	4.0×10^{-2}	5	Note 5	240	N/A	-20 to 140	0
Gaseous Waste Particulate	1	Gas, Air	I-131	1×10^{-9} (I-131)	3	1.0	90-150	Negligible	50-120	0
Gaseous Waste Gas	1	Gas, Air	I-131,Xe-133,Kr-85	1×10^{-6} (Kr-85)	4	1.0	90-150	Negligible	50-120	0
Ventilation Vent Particulate	1	Air	I-131	1×10^{-11} (I-131)	3	1.0	90-150	Negligible	50-120	0
Ventilation Vent Gas	1	Air	I-131,Xe-133,Kr-85	1×10^{-6} (Kr-85)	3	1.0	90-150	Negligible	50-120	0
SLCRS										
Particulate	1	Air	I 131	1×10^{-10} (I 131)	3	1.0	50-120	0	50-120	0
Gas	1	Air	I-131,Xe-133,Kr-85	1×10^{-6} (Kr 85)	3	1.0	50-120	0	50-120	0
Auxiliary Building Ventilation Exhaust System "A & B" Gas	2	Air	I-131,Xe-133,Kr-85	1×10^{-6} (Kr-85)	3	1.0	90-150	Negligible	50-120	0
Fuel Building Ventilation Exhaust Gross Activity	2	Air	I-131,Xe-133,Kr-85	1×10^{-6} (Kr-85)	4	1.0	90-100	Negligible	50-120	0
Containment Purge Exhaust Gross Activity	2	Air	I-131,Xe-133,Kr-85	1×10^{-6} (Kr-85)	3	2.5	90-150	-5 to 40	50-120	0

Table 11.3-2 (CONT'D)

PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEM

Channel	No.	Medium	Limiting Isotope	Sensitivity(1) Microcuries/cc	Range (2) Decades	Maximum(3) Back-ground, mr/hr	Medium Temp, F	Medium Pressure, (Max) psig	Ambient Temp, F	Ambient Pressure psig
Leak Collection Area Gas	1	Air	I-131,Xe-133,Kr-85	1 x 10^{-6} (Kr-85)	3	0.75	90-150	Negligible	50-120	0
Waste Gas Tank Vaults Ventilation Gas	1	Air	I-131,Xe-133,Kr-85	1 x 10^{-5} (Kr-85)	3	0.75	90-150	Negligible	50-120	0
Component Cooling Water	1	Water	Co-60, Cs-137	1 x 10^{-6} (Cs-137)	3	2.5	60-125	100	50-120	0
Liquid Waste Contaminated Drain	1	Water	Co-60, Cs-137	1 x 10^{-6} (Cs-137)	3	2.5	140	50	50-120	0
Liquid Waste Effluent	1	Water	Co-60, Cs-137	1 x 10^{-6} (Cs-137)	3	2.5	140	50	50-120	0
Condenser Air Ejector Vent	1	Vapor	I-131,Xe-133,Kr-85	1 x 10^{-4} (Kr-85)	4	0.75	40-165	Negligible	50-120	0
Steam Generator Blowdown Tank Discharge	2	Water	Co-60, Cs-137	1 x 10^{-6} (Cs-137)	3	0.75	120	200	50-120	0
Steam Generator Blowdown Sample	1	Water	Co-60, Cs-137	1 x 10^{-6} (Cs-137)	3	2.5	120	200	50-120	0
RC Letdown High Range	1	Water	Co-60, Cs-137	1 x 10^{-1} (Co-60)	4	2.5	120	200	50-120	0
*RC Letdown Low Range	1	Water	Co-60, Cs-137	1 x 10^{-4} (Co-60)	4	2.5	120	200	50-120	0

* Can be configured as a High Range or a Low Range Monitor

Table 11.3-2 (CONT'D)

PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEM

Channel	No.	Medium	Limiting Isotope	Sensitivity(1) Microcuries/cc	Range (2) Decades	Maximum(3) Back-ground, mr/hr	Medium Temp, °F	Medium Pressure, (Max) psig	Ambient Temp, °F	Ambient Pressure psig
Component Cooling/ Recirc. Spray Heat Exchanger River Water	1	Water	Co-60,Cs-137	1 x 10^{-6} (Cs-137)	3	0.75	40-125	150	50-120	0
Recirc. Spray Heat Exchanger River Water	4	Water	Co-60,Cs-137	1 x 10^{-6} (Cs-137)	3	5.0	40-125	150	50-120	0
Aux. Steam Condensate	1	Water	Co-60,Cs-137	1 x 10^{-6} (Cs-137)	3	2.5	50-125	150	50-120	0
Component Cooling Heat Exchanger River Water	1	Water	Co-60,Cs-137	1 x 10^{-6} (Cs-137)	3	0.75	40-125	150	50-120	0
Auxiliary Feedwater Area Drain Tank	1	Water	Cs-137,Co-60	1 x 10^{-6} (Cs-137)	4	2.5	40-120	150	50-120	0
Blowdown Water	1	Water	Co-60,Cs-137	1 x 10^{-7} (Cs-137)	6	0.75	140-325	120	50-104 23-131(8)	0
Main Steam Monitors	3	Steam	N-16	(7)	3/6	(7)	510-560	N/A	50-131	0

(1) Count rate to equal or exceed 100 percent of background.
(2) The range is the number of decades above the sensitivity. The log ratemeter shall be on scale throughout the range.
(3) Background from 1 Mev gammas.
(4) All process monitors shall have a channel accuracy of ±10 percent of the signal leaving the detector.
(5) Background based on U_{238} dopping of 10-25 cpm.
(6) Represents the SGADV and the first MSSV being open.
(7) Under optimum conditions an N-16 monitor provides indication of a primary-to-secondary leak as low as approximately 0.1 kilograms per hour in about one to two minutes. Background radiation level is not limiting due to the detector energy range.
(8) Long term temperature range is +50˚F to +104˚F, and periodic temperature range is +23˚F to +131˚F.

Table 11.3-3

AREA RADIATION MONITORING LOCATIONS AND RANGES[(3)]

Channel[(4)]	No.	Medium	Limiting Isotope	Sensitivity Microcuries/cc	Range (2) Decades	Maximum[(3)] Back-ground mr/Hr	Medium Temp, F	Medium Pressure, (Max) psig	Ambient Temp, F	Ambient Pressure psig
Containment Particulate	1	Air	F-18	1.81×10^{-5}	3	.75	50-120	-5.2 to 2	50-120	-5.2 to 2
Containment Gas	1	Air	Ar-41	1.30×10^{-7}	3	.75	50-120	-5.2 to 2	50-120	-5.2 to 2
Multisample Particulate	1	Air	I-131,Xe-133,Kr-85	1×10^{-9} (I-131)	3	.75	50-120	0	50-120	0
Multisample Gas	1	Air	I-131,Xe-133,Kr-85	1×10^{-6} (Kr-85)	3	.75	50-120	0	50-120	0

Channel[(4)]	No.	Range, mr/Hr	Ambient Temp, F	Ambient Pressure, psig
Containment High Range Gamma	1	10^{-1} - 10^{7}	50-120	0
Containment High Range Gamma (Post Accident)	2	10^{3} - 10^{10}	(See Note 6)	
Manipulator Crane*	1	10^{-1} - 10^{4}	50-120	-5.2 to 43
Reactor Containment Area*	1	10^{-1} - 10^{4}	50-120	-5.2 to 43
Incore Inst. Transfer Device*	1	10^{-1} - 10^{4}	50-120	-5.2 to 43
Decontamination Building	1	10^{-1} - 10^{4}	55-100	0

Table 11.3-3 (CONT'D)

AREA RADIATION MONITORING LOCATIONS AND RANGES[3]

Channel[4]	No.	Range, mr/Hr	Ambient Temp, F	Ambient Pressure, psig
New Fuel Storage Area	1	10^{-1} - 10^{4}	55-100	0
Fuel Pool Bridge	1	10^{-1} - 10^{4}	55-100	0
Auxiliary Building	3	10^{-1} - 10^{4}	55-100	0
Drum Handling and Storage Area	1	10^{-1} - 10^{4}	55-100	0
Sample Room	1	10^{-1} - 10^{4}	55-100	0
Main Control Room	2	10^{-2} - 10^{3}	65-90	0
Laboratory	1	10^{-1} - 10^{4}	55-95	0
Spares	4	---	---	-

(1) Count rate to equal or exceed 100 percent of background.

(2) The range is the number of decades above the sensitivity. The log ratemeter shall be on a scale throughout the range.

(3) Background from 1 Mev gammas.

(4) a) Channels shown with an asterisk are located within the containment building.

b) Sample pumps for the containment particulate and gas channels is located within the containment. Motor shall be totally enclosed-fan cooled or equivalently protected.

Table 11.3-3 (CONT'D)

AREA RADIATION MONITORING LOCATIONS AND RANGES[3]

(5) All area monitors shall have a channel accuracy of $\pm$15 percent of the signal leaving the detector (Except post accident monitors - meter accuracy $\pm$20 percent of photons of 0.1 - 3 MEV)

(6) Post Accident Conditions

Temperature	Pressure
350 F (5 min.)	45 psig (0-60 min.)
280 F (5-60 min.)	-1.2 psig (>60 min.
140 F (>60 min.)	

Table 11.3-4 - [HISTORICAL]

MAXIMUM EXPECTED QUANTITY
OF PRINCIPAL NUCLIDES IN
PROCESS EQUIPMENT

Nuclide	Liquid Waste Tanks (μCi)	Liquid Waste Evaporator Test Tank (μCi)	Liquid Waste Effluent Filter (μCi)	Liquid Waste Demineralizer Filter (μCi)
Sr-89	1.90 + 04[(1)]	2.95 - 01	2.89 - 04	4.24 + 00
Sr-90	4.47 + 02	6.94 - 03	6.82 - 06	2.68 - 01
Sr-91	8.74 + 03	1.36 - 01	1.33 - 04	1.69 - 02
Zr-95	3.10 + 03	4.81 - 02	4.73 - 05	8.26 - 01
Nb-95	3.15 + 03	4.89 - 02	4.81 - 05	1.21 + 00
Te-132	1.20 + 06	1.86 + 01	1.83 - 02	1.86 + 01
La-140	6.77 + 03	1.05 - 01	1.03 - 04	2.32 + 01
Ce-144	1.63 + 03	2.52 - 02	2.48 - 05	7.94 - 01
Y-90	5.40 + 02	8.38 - 03	8.23 - 06	2.70 - 01
Y-91	3.03 + 03	4.71 - 02	4.63 - 05	7.89 - 01
Y-92	3.27 + 03	5.07 - 02	4.98 - 05	2.34 - 03
Mo-99	1.50 + 07	2.32 + 01	2.28 - 02	2.12 + 01
Cs-137	5.84 + 06	9.05 + 01	8.89 - 02	3.15 + 03
Ba-140	3.80 + 05	5.90 + 00	5.80 - 03	2.32 + 01
Cr-51	1.75 + 04	6.77 - 02	6.65 - 05	5.72 - 01
Mn-54	1.44 + 04	5.57 - 02	5.48 - 05	1.77 + 00
Mn-56	5.40 + 05	2.05 + 00	2.11 - 03	6.79 - 02
Fe-59	1.92 + 04	7.47 - 02	7.34 - 05	9.72 - 01
Co-58	4.62 + 05	1.79 + 00	1.76 - 03	3.26 + 01
Co-60	1.39 + 04	5.38 - 02	5.28 - 05	2.03 + 00
I-131	1.18 + 07	1.83 + 02	1.80 - 01	4.55 + 02
I-132	4.06 + 06	6.29 + 01	6.18 - 02	2.04 + 01
I-133	1.83 + 07	2.84 + 02	2.79 - 01	7.65 + 01
I-134	2.52 + 06	3.90 + 01	2.71 - 02	4.38 - 01
I-135	9.73 + 06	1.51 + 02	1.48 - 01	1.30 + 01

(1) 1.90 + 04 = 1.90 x 10^4; typical for all values.

Table 11.3-4 (CONT'D) - [HISTORICAL]

MAXIMUM EXPECTED QUANTITY
OF PRINCIPAL NUCLIDES IN
PROCESS EQUIPMENT

Nuclide	Primary Drain Transfer Tank (μCi)	Boron Injection Tank (μCi)	Liquid Waste Evaporator (μCi)	Liquid Waste Evaporator Bottoms Cooler (μCi)
Sr-89	1.14 + 03	1.31 + 04	5.22 + 05	1.04 + 04
Sr-90	2.69 + 01	3.23 + 02	1.38 + 04	2.75 + 02
Sr-91	5.25 + 02	5.25 + 02	8.84 + 03	1.76 + 02
Zr-95	1.86 + 02	2.15 + 03	8.69 + 04	1.73 + 03
Nb-95	1.89 + 02	2.26 + 03	9.59 + 04	1.91 + 03
Te-132	7.20 + 04	4.47 + 05	9.50 + 06	1.89 + 05
La-140	4.07 + 02	9.79 + 03	3.79 + 05	7.55 + 03
Ce-144	9.16 + 01	1.16 + 03	4.90 + 04	9.75 + 02
Y-90	3.24 + 01	5.66 + 02	1.44 + 04	2.86 + 02
Y-91	1.82 + 02	4.71 + 03	8.62 + 04	1.72 + 03
Y-92	1.96 + 02	2.18 + 02	2.16 + 03	4.28 + 01
Mo-99	8.98 + 05	1.14 + 07	1.04 + 08	2.07 + 06
Cs-137	3.50 + 05	9.28 + 06	1.79 + 08	3.57 + 06
Ba-140	1.17 + 03	1.17 + 03	3.86 + 05	7.70 + 03
Cr-51	1. 9 + 02	3.16 + 03	1.08 + 05	2.15 + 03
Mn-54	1. 6 + 02	2.82 + 03	1.08 + 05	2.15 + 03
Mn-56	6. 0 + 03	2.36 + 03	3.62 + 04	7.19 + 02
Fe-59	2. 2 + 02	3.58 + 03	1.30 + 05	2.59 + 03
Co-58	5. 1 + 03	8.83 + 04	3.26 + 06	6.48 + 04
Co-60	1. 6 + 02	2.73 + 03	1.06 + 05	2.11 + 03
I-131	7.10 + 05	6.38 + 06	1.87 + 08	3.72 + 06
I-132	2.44 + 05	5.01 + 05	1.05 + 07	2.09 + 05
I-133	1.10 + 06	2.36 + 06	4.00 + 07	7.98 + 05
I-134	1.51 + 05	1.42 + 04	2.40 + 05	4.79 + 03
I-135	5.84 + 05	4.02 + 05	6.79 + 06	1.33 + 05

Table 11.3-4 (CONT'D) - [HISTORICAL]

MAXIMUM EXPECTED QUANTITY
OF PRINCIPAL NUCLIDES IN
PROCESS EQUIPMENT

.

Nuclides	Nonregenerative Heat Exchanger (μCi)	Mixed Bed Demineralizer (μCi)	Boric Acid Filter (μCi)	Reactor Coolant Filter (μCi)
Sr-89	3.47 + 02	2.56 + 06	7.42 + 01	2.56 + 04
Sr-90	8.19 + 00	6.38 + 04	1.82 + 00	6.38 + 02
Sr-91	1.60 + 02	7.96 + 04	2.98 + 00	7.96 + 02
Zr-95	5.67 + 01	4.22 + 05	1.22 + 01	4.22 + 03
Nb-95	5.77 + 01	4.49 + 05	1.29 + 01	4.49 + 03
Te-132	2.19 + 04	7.52 + 07	2.53 + 03	7.52 + 05
La-140	1.24 + 02	1.95 + 06	5.55 + 01	1.95 + 04
Ce-144	2.98 + 01	2.30 + 05	6.56 + 00	2.30 + 03
Y-90	9.87 + 00	3.95 + 04	1.11 + 00	3.95 + 02
Y-91	5.55 + 01	7.35 + 03	1.20 + 01	7.35 + 01
Y-92	5.97 + 01	8.45 + 03	4.44 + 00	8.45 + 01
Mo-99	2.73 + 05	negligible	2.92 + 05	negligible
Cs-137	1.07 + 05	negligible	2.39 + 05	negligible
Ba-140	3.57 + 02	2.20 + 06	6.63 + 01	2.20 + 06
Cr-51	3.18 + 02	2.23 + 06	1.26 + 02	2.23 + 04
Mn-54	2.63 + 02	2.03 + 06	1.07 + 02	2.03 + 04
Mn-56	1.00 + 04	1.30 + 06	9.39 + 01	1.30 + 04
Fe-59	3.53 + 02	2.56 + 06	1.44 + 02	2.56 + 04
Co-58	8.44 + 02	6.31 + 07	3.52 + 03	6.31 + 05
Co-60	2.54 + 02	1.98 + 06	1.09 + 02	1.98 + 04
I-131	2.16 + 05	1.17 + 09	3.63 + 04	1.17 + 07
I-130	7.43 + 04	8.35 + 07	2.85 + 03	8.35 + 05
I-133	3.35 + 05	3.60 + 08	1.35 + 03	3.60 + 06
I-134	4.61 + 04	2.16 + 06	8.06 + 01	2.16 + 04
I-135	1.78 + 05	6.11 + 07	2.29 + 03	6.11 + 05

Table 11.3-4 (CONT'D) - [HISTORICAL]

MAXIMUM EXPECTED QUANTITY
OF PRINCIPAL NUCLIDES IN
PROCESS EQUIPMENT

Nuclides	Boric Acid Tanks (μCi)		Cesium Removal Ionexchanger (μCi)	Boron Recovery Evaporator (μCi)
Sr-89	1.04 + 05	9.43 + 04	2.01 + 5	7.42 + 3
Sr-90	2.55 + 03	2.55 + 03	1.26 + 4	1.82 + 2
Sr-91	4.15 + 03	negligible	7.96 + 2	2.98 + 2
Zr-95	1.70 + 04	1.58 + 04	3.89 + 4	1.22 + 3
Nb-95	1.79 + 04	1.77 + 04	5.70 + 4	1.29 + 3
Te-132	3.53 + 06	7.93 + 05	8.75 + 5	2.53 + 5
La-140	7.73 + 04	7.13 + 04	5.87 + 4	5.55 + 3
Ce-144	9.18 + 03	9.03 + 03	3.74 + 4	6.56 + 2
Y-90	4.50 + 03	2.88 + 03	1.56 + 4	1.11 + 2
Y-91	3.73 + 04	3.45 + 04	3.66 + 5	1.20 + 3
Y-92	1.72 + 03	negligible	1.19 + 3	4.44 + 2
Mo-99	9.05 + 07	1.64 + 07	9.52 + 7	2.92 + 7
Cs-137	7.35 + 07	7.35 + 07	1.65 + 9	2.39 + 7
Ba-140	9.25 + 04	6.33 + 04	5.62 + 4	6.63 + 3
Cr-51	2.50 + 04	2.10 + 04	2.69 + 4	1.26 + 4
Mn-54	2.23 + 04	2.19 + 04	8.35 + 4	1.07 + 4
Mn-56	1.87 + 04	negligible	3.26 + 3	9.39 + 3
Fe-59	2.85 + 04	2.55 + 04	4.58 + 4	1.44 + 4
Co-58	6.98 + 05	6.53 + 05	1.53 + 6	3.52 + 5
Co-60	2.16 + 04	2.16 + 04	9.53 + 4	1.09 + 4
I-131	5.05 + 07	2.78 + 07	2.77 + 4	3.63 + 6
I-132	3.95 + 06	8.15 + 05	8.75 + 5	2.85 + 5
I-133	1.87 + 07	7.30 + 04	negligible	1.35 + 5
I-134	1.13 + 05	negligible	9.43 + 2	8.06 + 3
I-135	3.18 + 06	negligible	negligible	2.29 + 5

Table 11.3-4 (CONT'D) - [HISTORICAL]

MAXIMUM EXPECTED QUANTITY
OF PRINCIPAL NUCLIDES IN
PROCESS EQUIPMENT

Nuclides	Boron Evaporator Distillate Accumulator (μCi)	Boron Evaporator Bottoms Cooler (μCi)	Cation Bed Demineralizer (μCi)	Deborating Demineralizer (μCi)
Sr-89	8.75 - 3	1.74 + 2	negligible	negligible
Sr-90	2.06 - 4	4.26 + 0	"	"
Sr-91	4.03 + 3	6.97 + 0	"	"
Zr-95	1.43 - 3	2.85 + 1	"	"
Nb-95	1.45 - 3	3.01 + 1	"	"
Te-132	5.52 - 1	5.91 + 3	"	"
La-140	3.12 - 3	1.30 + 2	"	"
Ce-144	7.49 - 4	1.53 + 1	"	"
Y-90	2.49 - 4	2.61 + 0	3.23 + 3	"
Y-91	1.40 - 3	2.81 + 1	3.30 + 5	"
Y-92	1.50 - 3	1.04 + 1	1.10 + 3	"
Mo-99	6.88 + 0	6.83 + 5	8.64 + 7	"
Cs-137	2.68 + 0	5.59 + 5	1.24 + 7	"
Ba-140	9.00 - 3	1.55 + 2	negligible	"
Cr-51	8.0 - 3	2.94 + 2	"	"
Mn-54	6.6 - 3	2.50 + 2	"	"
Mn-56	2.5 - 1	2.20 + 2	"	"
Fe-59	8.9 - 3	3.37 + 2	"	"
Co-58	2.13 - 1	8.22 + 3	"	"
Co-60	6.38 - 3	2.55 + 2	"	"
I-131	5.45 + 0	8.49 + 4	"	2.36 + 9
I-132	1.87 + 0	6.66 + 3	"	9.63 + 6
I-133	8.43 + 0	3.15 + 3	"	3.96 + 8
I-134	1.16 + 0	1.89 + 2	"	2.27 + 6
I-135	4.49 + 0	5.35 + 3	"	6.74 + 7

Table 11.3-4 (CONT'D) - [HISTORICAL]

MAXIMUM EXPECTED QUANTITY
OF PRINCIPAL NUCLIDES IN
PROCESS EQUIPMENT

Nuclide	Degasifier (μCi)
Sr-89	5.53 + 3
Sr-90	1.30 + 2
Sr-91	2.47 + 3
Zr-95	9.00 + 2
Nb-95	9.15 + 2
Te-132	3.48 + 5
La-140	2.00 + 3
Ce-144	4.73 + 2
Y-90	1.56 + 3
Y-91	8.80 + 3
Y-92	8.85 + 3
Mo-99	4.33 + 7
Cs-137	1.69 + 7
Ba-140	5.68 + 3
Cr-51	1.27 + 3
Mn-54	1.04 + 3
Mn-56	3.53 + 4
Fe-59	1.40 + 3
Co-58	3.35 + 4
Co-60	1.01 + 3
I-131	3.43 + 6
I-132	1.09 + 6
I-133	5.25 + 6
I-134	5.58 + 5
I-135	1.27 + 7
Kr-83m	5.05 + 5
Kr-85m	2.40 + 6
Kr-85	1.46 + 7
Kr-87	1.31 + 6
Kr-88	3.88 + 6
Kr-89	2.09 + 4
Xe-131m	2.24 + 4
Xe-133m	4.13 + 6
Ke-133	3.58 + 7
Xe-135m	1.11 + 6
Xe-135	4.28 + 6
Xe-137	4.04 + 4
Xe-138	4.43 + 5

Table 11.3-4 (CONT'D) - [HISTORICAL]

MAXIMUM EXPECTED QUANTITY
OF PRINCIPAL NUCLIDES IN
PROCESS EQUIPMENT

Nuclide	Fuel Pool Filter (μCi)	Fuel Pool Ion Exchanger (μCi)	Seal Water Filter (μCi)	Seal Water Injection Filter (μCi)	Seal Water Heat Exchanger (μCi)
Cr-51	6.63 + 00	6.63 + 02	1.17 + 05	1.98 + 05	1.11 + 01
Mn-54	5.67 + 00	5.67 + 02	3.64 + 05	6.15 + 05	9.21 + 00
Mn-56	4.62 - 03	4.62 - 01	1.42 + 04	2.40 + 04	3.51 + 02
Fe-59	7.44 + 00	7.44 + 02	2.00 + 05	3.37 + 05	1.24 + 01
Co-58	1.88 + 02	1.80 + 04	6.69 + 06	1.13 + 07	2.96 + 02
Co-60	5.49 + 00	5.49 + 02	4.16 + 05	7.02 + 05	8.88 + 00
Sr-89	6.76 + 00	6.76 + 02	2.18 + 05	3.69 + 05	1.22 + 01
Sr-90	1.62 - 01	1.62 + 01	1.38 + 04	2.33 + 04	2.87 + 01
Sr-91	1.83 - 01	1.83 + 01	8.64 + 02	1.46 + 03	5.60 + 00
Zr-95	1.10 + 00	1.10 + 02	4.24 + 04	7.16 + 04	1.98 + 00
Nb-95	1.14 + 00	1.14 + 02	6.22 + 04	1.05 + 05	2.02 + 00
Te-132	3.04 + 02	3.04 + 04	9.53 + 05	1.61 + 06	7.67 + 02
Ca-145	4.59 + 00	4.59 + 02	6.39 + 04	1.08 + 05	4.34 + 00
Ce-144	6.47 + 00	6.47 + 02	4.07 + 04	6.88 + 04	1.04 + 00
Y-90	1.79 + 00	1.79 + 02	1.43 + 04	2.41 + 04	3.45 + 00
Y-91	1.34 + 01	1.34 + 03	8.94 + 04	1.51 + 05	1.94 + 01
Y-92	1.66 - 02	1.66 + 00	3.59 + 02	6.06 + 02	2.09 + 01
Mu-99	4.39 + 04	4.39 + 06	2.31 + 01	3.90 + 07	9.57 + 04
Cs-137	2.59 + 04	2.59 + 06	3.99 + 08	6.74 + 08	3.73 + 04
Ba-140	6.47 + 00	6.47 + 02	6.10 + 04	1.03 + 05	1.25 + 01
I-131	3.71 + 03	3.71 + 05	2.34 + 07	3.95 + 07	7.57 + 03
I-132	3.14 + 02	3.14 + 04	1.05 + 06	1.77 + 06	2.60 + 03
I-133	1.77 + 03	1.77 + 05	3.92 + 05	6.63 + 06	1.17 + 04
I-134	Negligible	Negligible	2.35 + 04	3.97 + 04	1.61 + 03
I-135	5.64 + 01	5.64 + 03	6.69 + 05	1.13 + 06	6.24 + 03

Note: The information presented in the above table was developed in support of the original license and is considered historical.

Table 11.3-5

Ambient Radiation Levels

The expected normal ambient radiation level in locations where area radiation monitoring equipment will be operating is:

Location	No.	Ambient Radiation[(1)] mr/hr
Containment High Range Gamma (Post-Accident)	2	21.0
Containment High Range Gamma	1	0.05
Manipulator Crane	1	34.0
Reactor Containment Area	1	21.0
In-core Inst. Transfer Device	1	2.0
Decontamination Building	1	0.2
New Fuel Storage Area	1	0.2
Fuel Pool Bridge	1	0.9
Auxiliary Building	3	0.1
Drum Handling and Storage Area	1	5.0
Sample Room	1	0.6
Main Control Room	2	0.05
Laboratory	1	0.2

(1) Design values for monitor location under normal operating conditions and/or occupancy requirements.

Table 11.3-6

PROCESS & EFFLUENT RADIOLOGICAL MONITORING
SYSTEM CHANNELS

No. of Monitors	Radiation Monitors
2	Gaseous Waste Particulate Monitor
3	Gaseous Waste Gas Monitor
2	Ventilation Vent Particulate Monitor
3	Ventilation Vent Gas Monitor
2	Auxiliary Building Ventilation Exhaust Systems A and B Gas Monitors
2	Fuel Building Ventilation Exhaust Gross Activity Monitors
2	Containment Purge Exhaust Gross Activity Monitors
1	Leak Collection Area Gas Monitor
1	Waste Gas Tank Vaults Ventilation Exhaust Gas Monitor
1	Component Cooling Water Monitor
1	Liquid Waste Effluent Monitor
1	Liquid Waste Contaminated Drains Monitor
1	Condenser Air Ejector Monitor
1	Steam Generator Blowdown Sample Monitor
1	Steam Generator Blowdown Tank Discharge Monitor
1	Steam Generator Blowdown Water Monitor
2	Reactor Coolant Letdown Gross Activity Monitors
1	Auxiliary Steam Condensate Monitor

Table 11.3-6 (CONT'D)

PROCESS & EFFLUENT RADIOLOGICAL MONITORING
SYSTEM CHANNELS

No. of Monitors	Radiation Monitors
1	Component Cooling/Recirculation Spray Heat Exchanger River Water Monitor
4	Recirculation Spray Heat Exchanger River Water Monitors
1	Component Cooling Heat Exchanger River Water Monitor
2	Reactor Building/SLCRS Particulate Monitor
3	Reactor Building/SLCRS Gas Monitor
4	Spares
3	Atmospheric Steam Dump and Main Steam Safety Valve Monitors
1	Auxiliary Feedwater Pump Turbine Exhaust Monitor
1	Auxiliary Feedwater Area Drain Tank Monitor
1	Post-Accident Sampling Gas Monitor
1	Post-Accident Sampling Liquid Monitor
2	Post-Accident Sampling Ventilation Monitors

Table 11.5-1 - [HISTORICAL]

AIRBORNE RADIOACTIVITY FROM EQUIPMENT LEAKAGE

Estimates of equipment leakage and the associated reduction factors used to develop expected airborne radioactivity levels within plant buildings are as follows:

A. Containment Building

(1) 40 gal at 120°F, per day of hot primary coolant leakage

(2) Factor of two reduction for iodine nuclides due to plateout

B. Auxiliary Building

(1) 2.5 gal at 120°F, per day of hot primary coolant leakage

(2) Factor of 100 reduction for iodine nuclides due to phase change

C. Turbine Building

(1) Secondary system leakage of 100 lb/hr

(2) Primary-to-secondary leakage of 20 gal per day

(3) Internal steam generator iodine partition factor of 100

(4) Steam generator blowdown rate of 15 gpm

D. Fuel Building

(1) Spent fuel escape rate coefficients

Noble gases - 6.5×10^{-13} ℓ/sec
Iodines - 1.3×10^{-13} ℓ/sec

(2) Fuel pool cleanup rate of 150 gpm

(3) Fuel pool demineralizer iodine reduction factor DF=10[1]

(4) Noble gas pool-to-air DF=1

(5) Iodine pool-to-air DF=100

(6) Six month retention of full core inventory

(7) 100 hr decay of fuel prior to transfer of fuel to the fuel pool

[1] DF = Dilution Factor

Note: The above table supports Section 11.5.2.1 which describes analyses performed during station licensing to estimate airborne radioactivity concentrations in various buildings. These analyses were based on assumptions and methodology acceptable at that time and are part of the plant design basis.

Table 11.5-2 - [HISTORICAL]

AIRBORNE NUCLIDE CONCENTRATION
IN AUXILIARY BUILDING

Nuclide	Concentration (μCi/cc)
Kr-83m	2.01 - 10[1]
Kr-85m	1.26 - 9
Kr-85	1.04 - 8
Kr-87	4.59 - 10
Kr-88	1.80 - 9
Kr-89	2.57 - 11
Xe-131m	1.34 - 11
Xe-133m	2.87 - 9
Xe-133	2.49 - 8
Xe-135m	1.23 - 10
Xe-135	2.54 - 9
Xe-137	5.10 - 12
Xe-138	8.13 - 11
I-131	2.42 - 11
I-132	4.56 - 12
I-133	3.47 - 11
I-134	1.63 - 12
I-135	1.56 - 11

[1] 2.01 - 10 = 2.01 x 10^{-10}; typical for all values.

Note: The above table is based on Section 11.5.2.1 which describes analyses performed during station licensing to estimate airborne radioactivity concentrations in various buildings. These analyses were based on assumptions and methodology acceptable at that time and are part of the plant design basis.

Table 11.5-3

AUXILIARY BUILDING CUBICLES EXHAUSTED
TO SUPPLEMENTARY LEAK COLLECTION SYSTEM

Cubicles Exhausted	Duct Elevation
Seal water, non regenerative heat exchanger	747' 10"
Pipe chase and valve area	745' 0"
Sample room (hood exhaust)	746' 9"
Primary drains transfer pumps and tank	722' 6"
Degasifier recirc pump A	732' 4"
Degasifier recirc pump B	732' 4"
Condensate receiver pump A and condensate receiver tank	729' 9"
Gaseous waste compressor A	731' 5"
Gaseous waste compressor B	731' 5"
Liquid waste tank A	750' 3"
Liquid waste tank B	749' 3"
Liquid waste pump A	728' 3"
Liquid waste pump B	728' 3"
Non regenerative heat exchanger	747' 10"
Charging pump A	731' 0"
Charging pump B	731' 0"
Charging pump C	731' 0"
Boric acid tank A	761' 8"
Boric acid tank B	761' 8"
Volume control tank	763' 1"
Boric acid pump A	764' 5"
Boric acid pump B	764' 5"
Degasifier vent chiller and vent condenser A	760' 4"
Degasifier vent chiller and vent condenser B	760' 4"
Charcoal delay beds	765' 0"
Primary component cooling surge tank	768' 0"

Normally Closed Areas-access through removable floor slabs - El. 752' 6"

Boric acid filter
Fuel pool filter
Coolant recovery filter
Cesium demineralizer
Mixed bed demineralizer
Cation demineralizer
Deborating demineralizer
Seal water filter
Seal water injection filter
Reactor coolant filter
Pipe tunnels

Table 11.5-4 - [HISTORICAL]

AIRBORNE NUCLIDE CONCENTRATION
IN TURBINE BUILDING

Nuclide	Concentration (μCi/cc)
Kr-83m	1.71 - 14
Kr-85m	8.91 - 14
Kr-85	5.75 - 13
Kr-87	4.35 - 14
Kr-88	1.39 - 13
Kr-89	4.37 - 16
Xe-131m	7.53 - 16
Xe-133m	1.63 - 13
Xe-133	1.42 - 12
Xe-135m	1.75 - 14
Xe-135	1.61 - 13
Xe-137	8.57 - 16
Xe-138	1.15 - 14
I-131	1.82 - 12
I-132	1.58 - 13
I-133	1.35 - 12
I-134	8.37 - 15
I-135	3.05 - 13

Note: The above table is based on Section 11.5.2.1 which describes analyses performed during station licensing to estimate airborne radioactivity concentrations in various buildings. These analyses were based on assumptions and methodology acceptable at that time and are part of the plant design basis.

Table 11.5-5 - [HISTORICAL]

AIRBORNE NUCLIDE CONCENTRATION
IN CONTAINMENT BUILDING

Nuclide	Concentration (μCi/cc)
Kr-83m	4.39 - 8
Kr-85m	3.37 - 7
Kr-85	2.61 - 3
Kr-87	6.87 - 8
Kr-88	3.94 - 7
Kr-89	2.35 - 10
Xe-131m	2.22 - 5
Xe-133m	7.71 - 6
Xe-133	1.62 - 4
Xe-135m	2.00 - 7
Xe-135	1.94 - 6
Xe-137	4.71 - 10
Xe-138	8.34 - 9
I-131	1.10 - 5
I-132	4.92 - 7
I-133	1.84 - 6
I-134	1.11 - 8
I-135	3.14 - 7

Note: The above table is based on Section 11.5.2.1 which describes analyses performed during station licensing to estimate airborne radioactivity concentrations in various buildings. These analyses were based on assumptions and methodology acceptable at that time and are part of the plant design basis.

Table 11.5-6 - [HISTORICAL]

AIRBORNE NUCLIDE CONCENTRATIONS
IN FUEL BUILDING

Nuclide	Concentration[1] (μCi/cc)
Kr-83m	Negligible
Kr-85m	2.10 - 15
Kr-85	1.58 - 9
Kr-87	Negligible
Kr-88	Negligible
Kr-89	Negligible
Xe-131m	2.12 - 8
Xe-133m	2.75 - 9
Xe-133	1.83 - 7
Xe-135	Negligible
Xe-137	Negligible
Xe-138	Negligible
I-131	1.99 - 10
I-132	1.11 - 10
I-133	1.65 - 12
I-134	Negligible
I-135	Negligible

(1) Concentrations <10^{-15} μCi/cc considered negligible

Note: The above table is based on Section 11.5.2.1 which describes analyses performed during station licensing to estimate airborne radioactivity concentrations in various buildings. These analyses were based on assumptions and methodology acceptable at that time and are part of the plant design basis.

Table 11.5-7 - [HISTORICAL]

ESTIMATE OF DOSE RATE LEVELS FOR SELECTED STATION LOCATIONS

Location	Dose Rate (mrem/hr)
Gate House	1.6×10^{-4}
Site Boundary (2,000 ft)	1.4×10^{-6}
Control Room	2.3×10^{-2}
Lunch and Meeting Room (Service Building)	1.7×10^{-2}
Clean Shop (Service Building)	3.9×10^{-4}
Field Office (Service Building)	5.7×10^{-1}
Turbine Building Sample Room	1.5×10^{-6}
Auxiliary Building Access Areas	2.5×10^{-6}

Note: The above table is based on radioactive inventories expected in process equipment described in Table 11.3-4 and is considered historical.

TABLE 11.5-8

PARAMETERS USED IN RADIOLOGICAL ANALYSIS
OF THE WASTE GAS SYSTEM FAILURE ACCIDENT[1]

	Design Case
Power, MWt	2918
Fraction of fuel with defects	0.01
Offsite AC power	Lost
Letdown flow rate, gpm	120
Analysis cases	
Rupture of Waste Gas Storage Tank (WGST) Rupture of Gaseous Waste System line upstream of charcoal delay beds	
Volume of WGST, ft^3	132
WGST feed rate, scfm	2
Gaseous Waste System operating pressure, psig	65
Gaseous Waste System operating temperature, °F	100
Charcoal bed holdup time, days	
Krypton	2.1
Xenon	38.7
Control Room volume, ft^3	1.73E+5
Control Room normal intake, cfm	500
Control Room isolation	None[2]
Control Room purging	None
Control Room χ/Q values, sec/m^3	
Auxiliary Building 0-8 hours (line rupture case)	Table 2.2-12
Gaseous Waste Vault 0-8 hours (WGST rupture case)	Table 2.2-12
Offsite χ/Q values, sec/m^3	Tables 2.2-11a, 2.2-11b
Analysis references	

Notes:

1. This analysis was originally performed in 1987 in support of plant modifications converting the Unit 1 Control Room to a common Unit 1 - Unit 2 facility. The analysis inputs, assumptions, and methodologies are based on regulatory requirements that existed at the time the analysis was performed.

2. The dose calculation conservatively assumed that the control room radiation monitors will not initiate control room isolation.

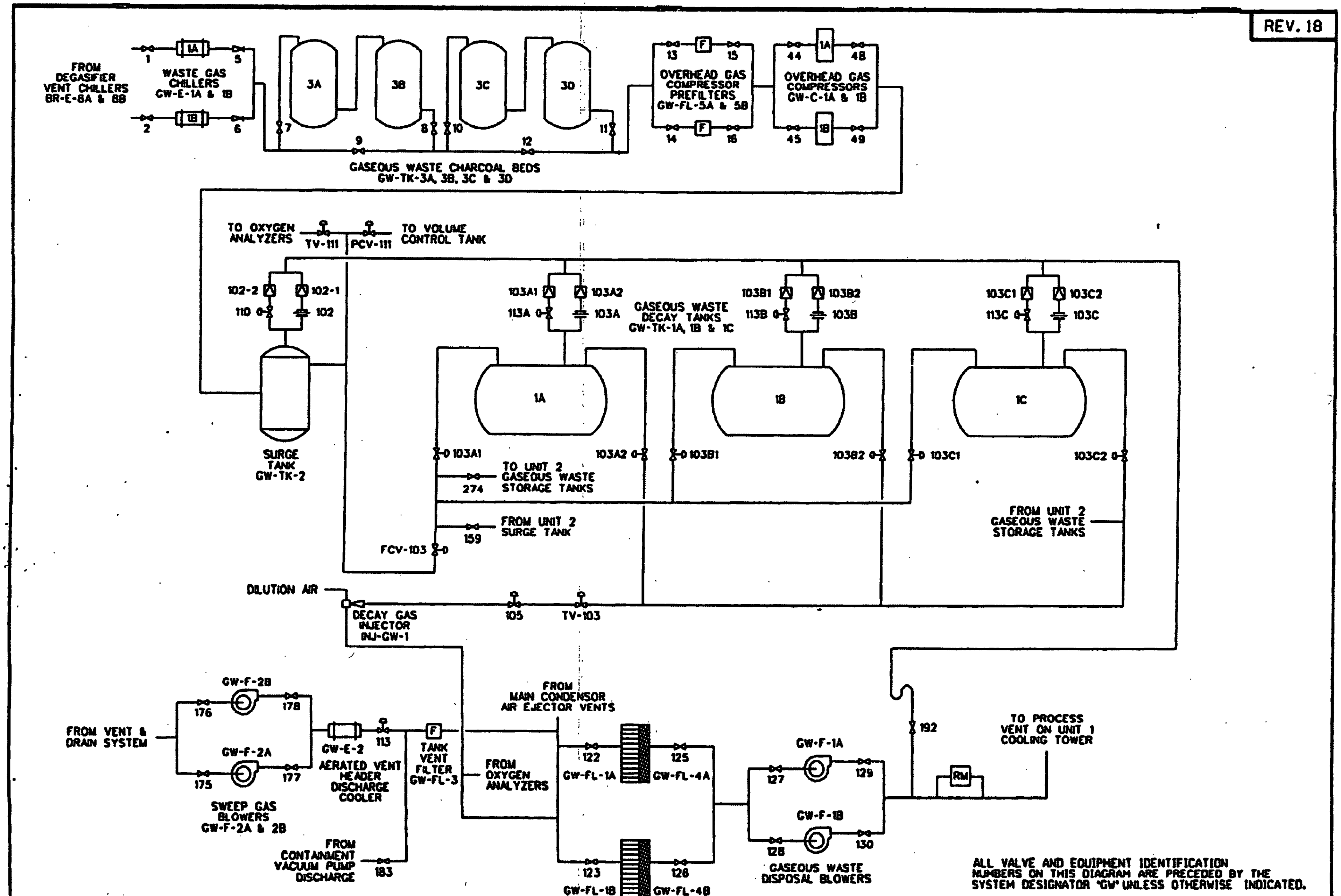

FIGURE 11.2-1
GASEOUS WASTE DISPOSAL SYSTEM
REFERENCE: STATION DRAWING RM-419
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

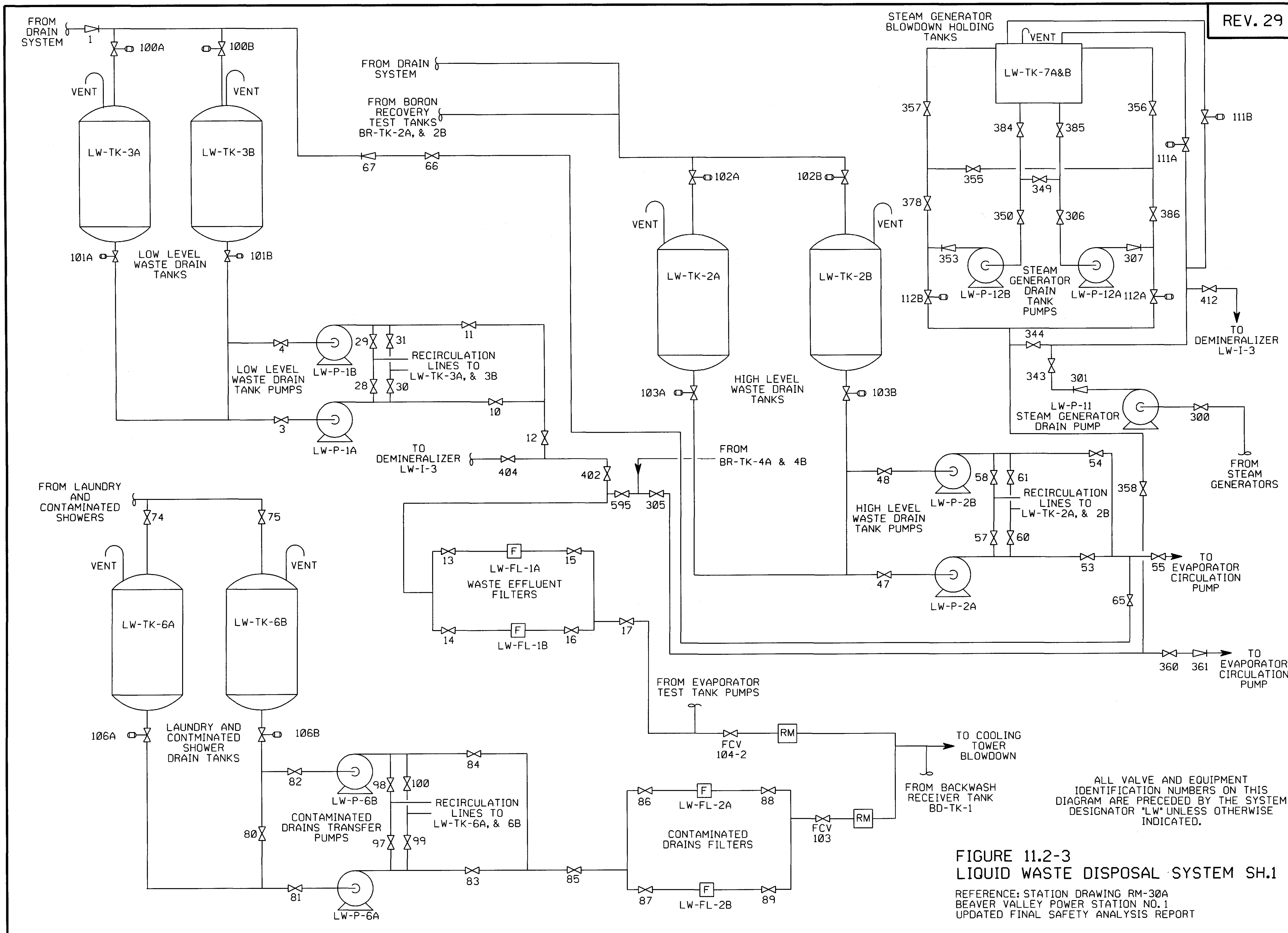

FIGURE 11.2-3
LIQUID WASTE DISPOSAL SYSTEM SH.1

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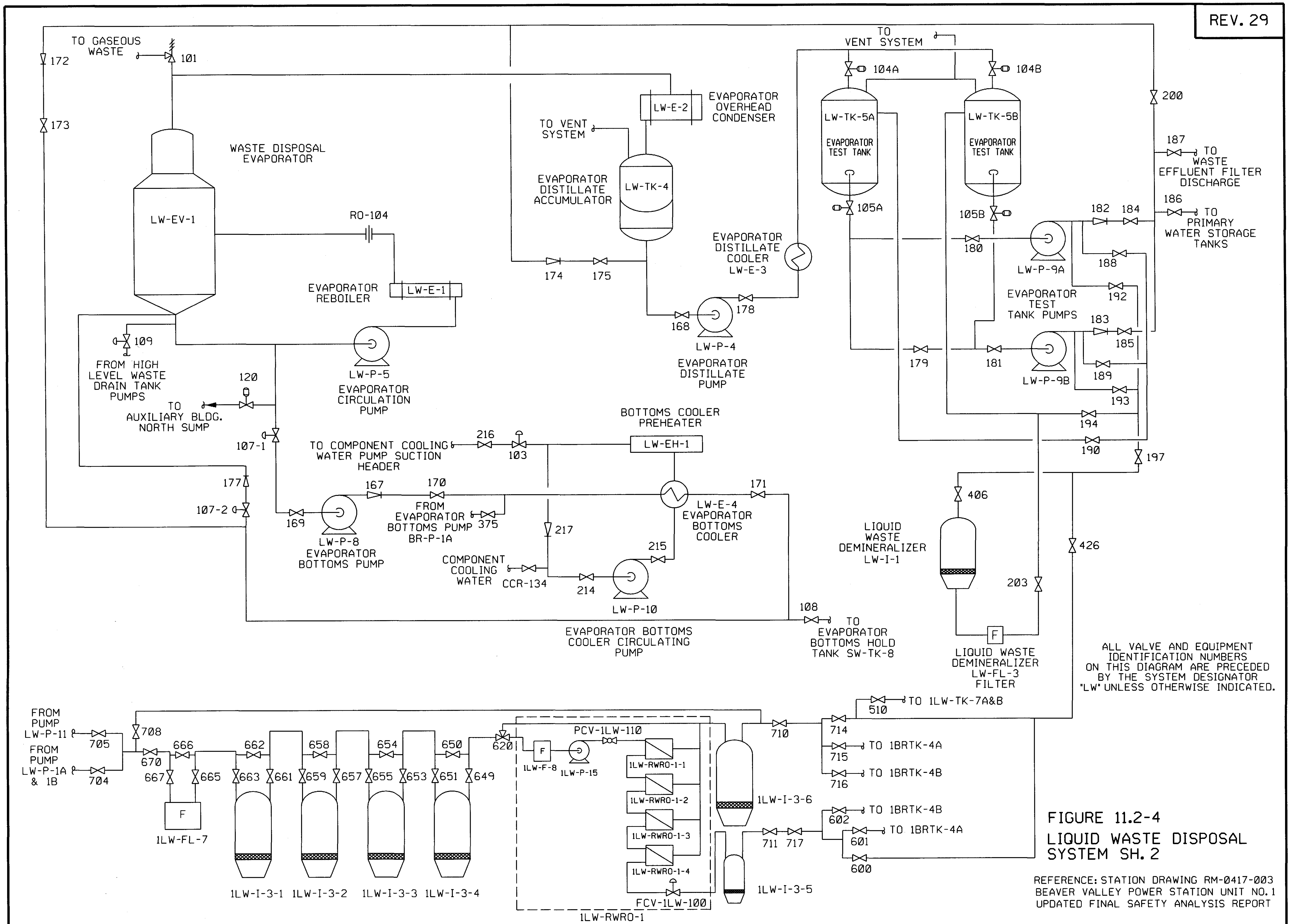

FIGURE 11.2-4
LIQUID WASTE DISPOSAL SYSTEM SH. 2

REFERENCE: STATION DRAWING RM-0417-003
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

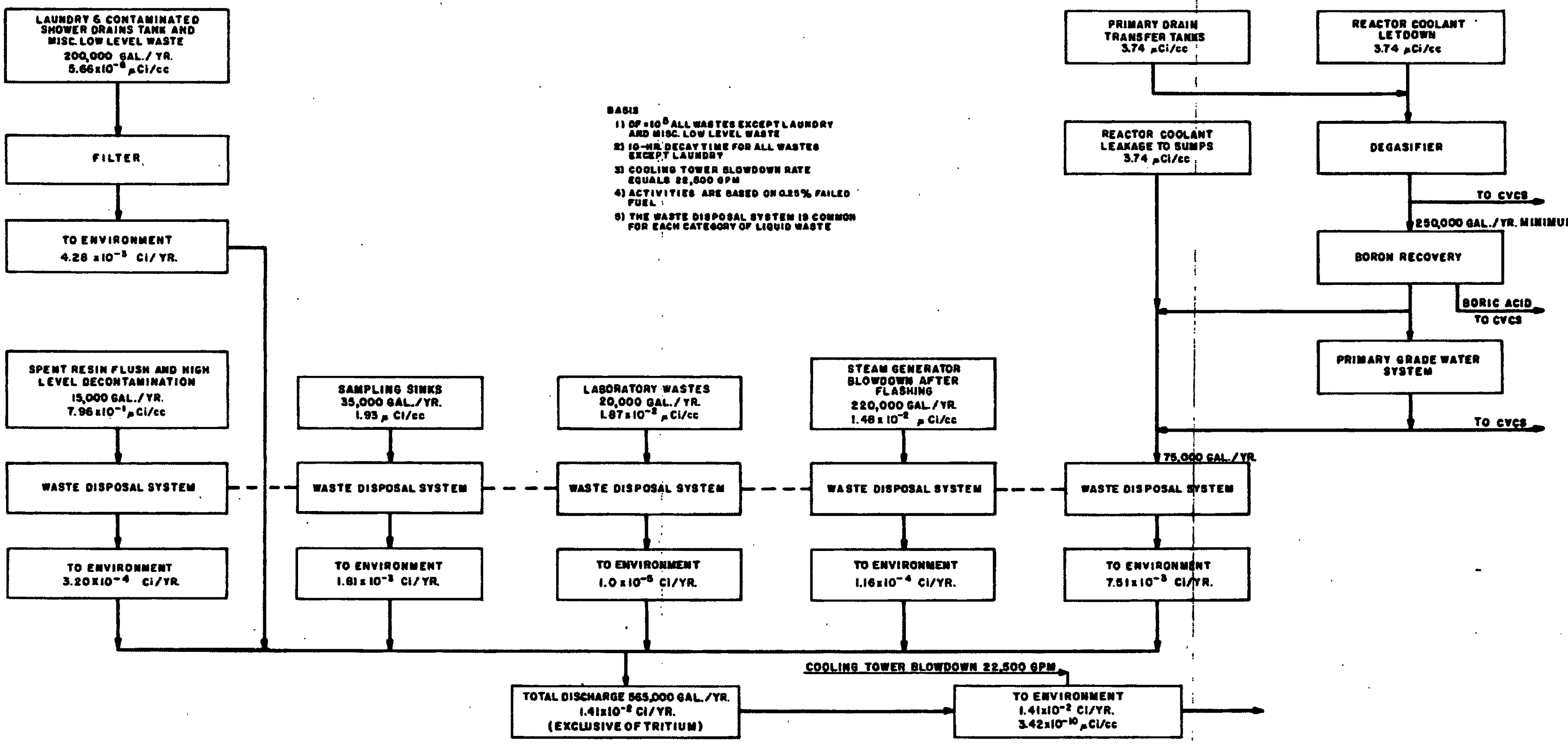

FIGURE 11.2-5A
FLOW CHART - ESTIMATED LIQUID WASTE FROM SYSTEM SOURCES MAXIMUM EXPECTED
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

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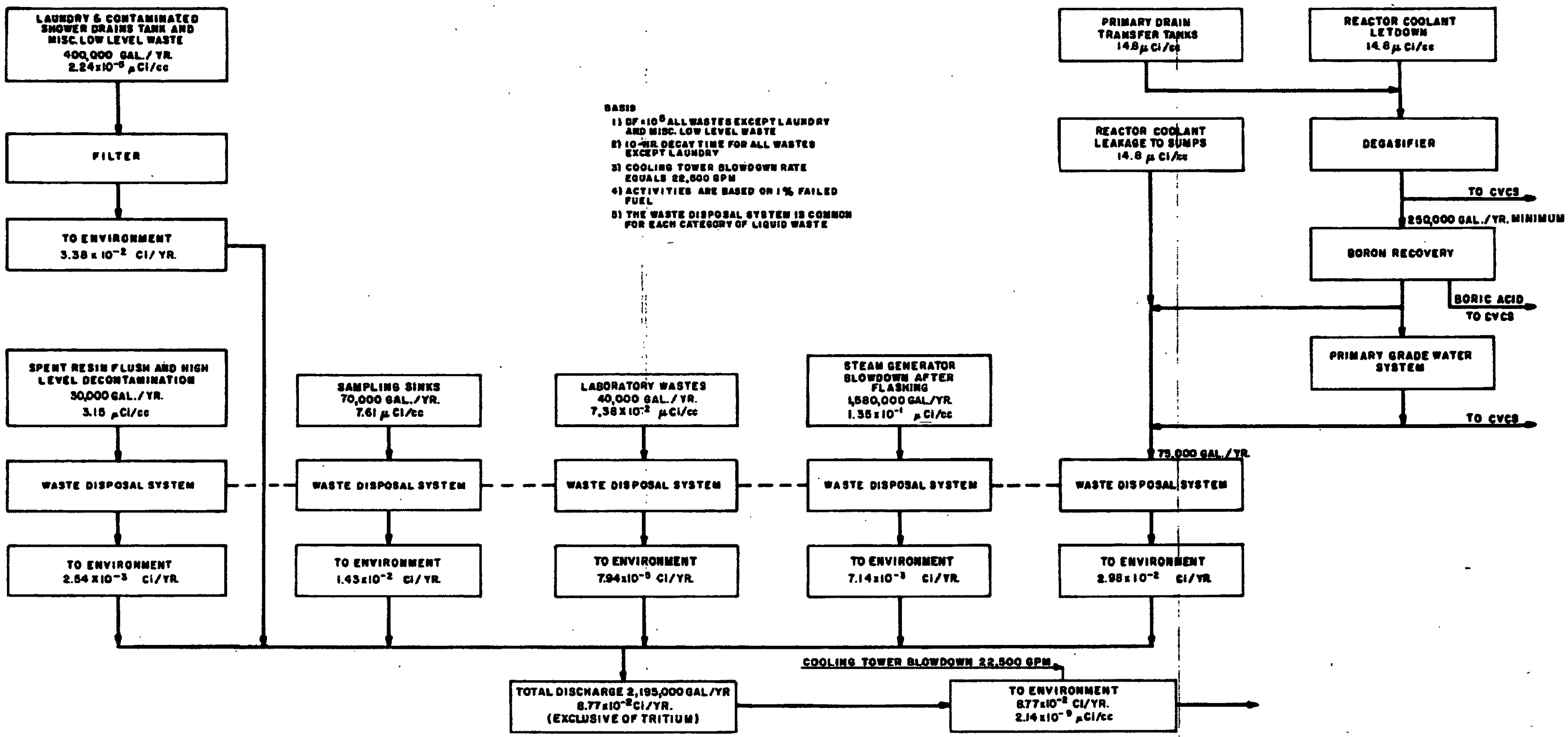

FIGURE 11.2-5B
FLOW CHART - ESTIMATED LIQUID
WASTE FROM SYSTEM SOURCES
SYSTEM DESIGN
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

APPENDIX 11A - [HISTORICAL]

ESTIMATED RADIOACTIVE NUCLIDE CONCENTRATIONS IN WASTE DISPOSAL SYSTEMS AND IN DISCHARGE TO THE ENVIRONMENT

This Appendix presents estimates of radioactive nuclide concentrations in waste disposal systems and in discharge to the environment based on the original design of the liquid waste system. Appendix 11A has been retained to provide historical information. Actual concentrations of radioactive material in effluents are limited in accordance with the Offsite Dose Calculation Manual.

The radioactive waste effluents which are expected to be processed and discharged to the environment from the BVPS-1 are divided into liquid wastes, gaseous wastes and solid wastes. These wastes are collected and processed by the various methods which are described in Section 11.2.

The tables presented in this appendix are for two bases. The first is the maximum that can reasonably be expected to be discharged to the environment under normal conditions of unit operation. The second is the assumptions that have been used for determining the maximum system design capability.

The conditions upon which the maximum expected and system design are based for liquids are as follows:

Liquid Waste Conditions

Condition	Maximum Expected	System Design
Failed Fuel	0.25 percent	1.0 percent
Primary to Secondary Leakage	20 gpd	144 gpd
Primary Leakage to Containment	40 gpd	200 gpd
Primary Leakage to Auxiliary Building	2.5 gpd	2.5 gpd

Other conditions upon which the quantities and concentrations of radioactive nuclides in the liquid effluents were based are:

1. Steam blowdown was based on 500 ppm boron in the primary coolant, maintaining 125 ppm solids in the steam generator. The amount of liquid given in the tables in this appendix is the liquid remaining after approximately 1/3 of blowdown has flashed off as vapor. Plant modifications will provide for complete recycling of the steam generator blowdown to eliminate this source from the liquid waste system. The modified, separate steam generator blowdown treatment system will be designed to handle blowdown resulting from primary to secondary leakage of 1,400 gpd for up to seven days.

2. Discharge of tritium to maintain equilibrium level in the reactor coolant.

3. Dilution of discharge waste effluent with 22,500 gpm blowdown from cooling tower. (Operating experience has shown 15,000 gpm to be the blowdown for normal operations).

4. Effective decay time of 10 hours within the liquid waste disposal system except for laundry and contaminated showers for which no decay time is included.

5. Decontamination factor of 10^5 via the use of evaporators and demineralizers.

6. An operating period of 330 days per year.

Estimated quantities and activities of the maximum expected liquid wastes are given in Table 11A-1; system design wastes are given in Table 11A-2. Annual average radionuclide inventory in the cooling tower blowdown are given in Table 11A-3 for maximum expected, and Table 11A-4 for system design.

Figures 11.2-5A and 11.2-5B give the estimated liquid wastes from system sources for the maximum expected and system design conditions.

The expected quantities and concentrations for each liquid waste processing stream are described in Table 11A-7 and reflect BVPS-1 and BVPS-2 operating with 0.25 percent failed fuel. The assumptions used to derive the maximum expected discharge quantities were:

1. Liquids waste disposal system decontamination factor of 10^5 for all sources except contaminated laundry and showers, miscellaneous low level wastes and turbine building drains, for which no decontamination credit is included.

2. Effective decay time of 10 hours for all sources except the contaminated laundry and showers, miscellaneous low level wastes and turbine building drains for which no decay is included.

3. Recycling of steam generator blowdown effluent.

4. An operating period of 330 days per year.

5. Turbine drains assumed to have 0.001 the concentration of steam generators secondary side.

Subsequent to processing, the liquid waste effluents are diluted by the cooling tower blowdown. Although the cooling tower blowdown rate has been as low as 9000 gpm during discharges, this rate can be increased to 22,500 gpm if necessary.

The contribution of each of the liquid effluent streams are shown in Table 11A-8, 11A-9, 11A-10, 11A-11, 11A-12, 11A-13, 11A-14, 11A-15 and 11A-16 inclusive.

The principal nuclide concentrations in the turbine building liquid drains are based on an assumed leak rate of approximately 0.3 gpm. The building floor drains nuclide inventory are assumed to be 0.001 times the liquid concentration of the secondary system. The nuclide concentrations are shown in Table 11A-17.

Gaseous Waste Conditions

The gaseous effluent sources discharged from the turbine building were based on the following parameters:

1. 0.25 percent failed fuel

2. Primary-to-secondary leakage of 20 gpd

3. Steam generator iodine partition factor of 100

4. Secondary system leakage of 100 lbs per hour

5. Secondary system blowdown of 15 gpm

The expected quantity of gaseous nuclides discharged to the environment from the turbine building is contained in Table 11A-18.

The bases for the gas effluent nuclides from other sources on the site were:

1. The gases associated with the conditions for liquid effluent described under liquid waste conditions.

2. Gases from degasification of reactor coolant letdown associated with fuel consumption during core life.

3. Direct release via charcoal/HEPA filters of gases from the air ejector.

4. A minimum of 30 days decay of gases derived from degasification of the reactor coolant system.

5. Processing of gas with charcoal filters having an iodine removal efficiency of 99.0 percent. It has been demonstrated that charcoal filters used in the gaseous waste disposal system actually have an iodine removal efficiency of 99.9 percent.

In addition, for the Maximum Expected bases, the gaseous release derived from primary system leakage within the containment is recirculated through charcoal filters until the containment atmosphere is 10 MPC of equivalent I-131. Subsequently, the containment atmosphere is purged to the environment via the ventilation vent system.

For the System Design bases, 90 percent of the containment atmosphere is vented through the charcoal and HEPA filters in the gaseous waste disposal system to the process vent on top of the cooling tower. The remaining 10 percent is vented through the main filter banks to the elevated release point on top of the containment.

The quantities of gaseous radioactive nuclides that will be discharged from the station are given in Table 11A-5 for the Maximum Expected basis and Table 11A-6 for the System Design basis.

Solid Waste

The exhausted resins from the ion exchangers are transferred to a disposable liner and dewatered or solidified.

Approximately 30 truck shipments of solid wastes will be required per year. The following amounts of solid waste material were used to estimate the shipments:

1. Evaporator Bottoms - 2000 ft^3
2. Spent Resin - 525 ft^3
3. Filters - 18
4. Compacted Trash - 6000 ft^3

TABLES FOR APPENDIX 11A

Table 11A-1

LIQUID WASTE QUANTITIES
AND INITIAL ACTIVITIES - MAXIMUM EXPECTED

	Source	Maximum Expected Yearl Volume, Gal	Initial Activity (μCi/cc)	Radwaste System Decontamination Factor (Including Reprocessing i Required)
1.	Contaminated Laundry and Showers	100,000		
	Misc. Low Level Waste	100,000		
		200,000	5.66-6	1.0
2.	Sampling Sinks	35,000	1.93-0	1.0+5
3.	Resin Flushing	7,500		
	High Level Decontamination Solutions	7,500		
		15,000	7.96-1	1.0+5
4.	Laboratory Wastes	20,000	1.87-2	1.0+5
5.	Primary Coolant System Discharge	75,000	3.74-0	1.0+5
6.	S.G. Blowdown	220,000	1.48-2	1.0+5
	Total	565,000		

Note: Activities do not include tritium.
Exponents are indicated as + or -;
eg. 5.66-6 = 5.66 x 10^{-6}.

Table 11A-2

LIQUID WASTE QUANTITIES
AND INITIAL ACTIVITIES - SYSTEM DESIGN

	Source	Radwaste System Design Yearly Volume, Gal	Initial Activity (μCi/cc)	Radwaste System Decontamination Factor (Including Reprocessing if Required)
1.	Contaminated Laundry and Showers	200,000		
	Misc. Low Level Waste	200,000		
		400,000	2.24-5	1.0
2.	Sampling Sinks	70,000	7.61-0	1.0+5
3.	Resin Flushing	15,000		
	High Level Decontamination Solutions	15,000		
		30,000	3.15-0	1.0+5
4.	Laboratory Wastes	40,000	7.38-2	1.0+5
5.	Primary Coolant System Discharge	75,000	1.48+1	1.0+5
6.	S.G. Blowdown	1,580,000	1.35-1	1.0+5
	Total	2,195,000		

Note: Activities do not include tritium.
Exponents are indicated as + or -;
eg. 2.24-5 = 2.24 x 10^5.

Table 11A-3

ANNUAL AVERAGE RADIONUCLIDE INVENTORY IN COOLING TOWER BLOWDOWN FOR THE MAXIMUM EXPECTED BASES

Nuclide	Discharge Rate (Ci/yr)	Concentration (μCi/cc)	MPC[1] (μCi/cc)	Fraction of MPC
H3	5.06+2[2]	1.23-5	3.0-3	4.11-3
Mn54	3.79-6	9.24-14	1.0-4	9.24-10
Mn56	4.04-5	9.83-13	1.0-4	9.83-9
Co58	1.22-4	2.97-12	9.0-5	3.31-8
Fe59	5.06-6	1.23-13	5.0-5	2.47-9
Co60	3.71-6	9.04-14	3.0-5	3.01-9
Sr59	5.00-6	1.22-13	3.0-6	4.06-8
Sr90	1.20-7	2.91-15	3.0-7	9.71-9
Sr91	1.38-6	3.35-14	5.0-5	6.70-10
Y90	1.32-7	3.22-15	2.0-5	1.61-10
Y91	8.00-7	1.95-14	3.0-5	6.49-10
Y92	2.94-7	7.15-15	6.0-5	1.19-10
Zr95	8.21-7	2.00-14	6.0-5	3.33-10
Nb95	0.37-7	2.04-14	1.0-4	2.04-10
Mo99	3.58-3	8.71-11	4.0-5	2.18-6
I131	2.99-3	7.27-11	3.0-7	2.43-4
I132	2.89-4	7.03-12	8.0-6	8.79-7
I133	3.71-3	9.03-11	1.0-6	9.03-5
I134	1.55-4	3.77-12	2.0-5	1.89-7
I135	1.28-3	3.12-11	4.0-6	7.80-6
Te132	2.91-4	7.08-12	2.0-5	3.54-7
Cs137	1.56-3	3.80-11	2.0-5	1.90-6
Ba140	5.02-6	1.22-13	2.0-5	6.11-9
La140	1.60-6	3.89-14	2.0-5	1.95-9
Ce144	4.35-7	1.06-14	1.0-5	1.06-9
Cr51	4.55-6	1.11-13	2.0-3	5.54-11
Total With H3	5.06+2	1.23-5		4.45-3
Total Without H3	1.41-2	3.42-10		3.47-4

(1) Maximum permissible concentration 10CFR20, Appendix B, Table 2

(2) 5.06+2 = 5.06 x 10^2

Table 11A-4

ANNUAL AVERAGE RADIONUCLIDE INVENTORY IN COOLING TOWER BLOWDOWN FOR THE SYSTEM DESIGN BASES

Nuclide	Discharge Rate (Ci/yr)	Concentration (μCi/cc)	MPC[1] (μCi/cc)	Fraction of MPC
H3	5.06+2[2]	1.23-5	3.0-3	4.11-3
Mn54	6.28-6	1.53-13	1.0-4	1.53-9
Mn56	7.52-5	1.83-12	1.0-4	1.83-8
Co58	2.00-4	4.87-12	9.0-5	5.41-8
Fe59	8.28-6	2.02-13	5.0-5	4.03-9
Co60	6.07-6	1.48-13	3.0-5	4.92-9
Sr89	3.28-5	7.99-13	3.0-6	2.66-7
Sr90	7.83-7	1.91-14	3.0-7	6.35-8
Sr91	8.53-6	2.08-13	5.0-5	4.15-9
Y90	8.61-7	2.10-14	2.0-5	1.05-9
Y91	5.26-6	1.28-13	3.0-5	4.27-9
Y92	2.07-6	5.05-14	6.0-5	8.42-10
Zr95	5.37-6	1.31-13	6.0-5	2.13-9
Nb95	5.49-6	1.34-13	1.0-4	1.34-9
Mo99	2.21-2	5.38-10	4.0-5	1.35-5
I131	1.91-2	4.66-10	3.0-7	1.55-3
I132	2.20-3	5.35-11	8.0-6	6.69-6
I133	2.24-2	5.45-10	1.0-6	5.45-4
I134	1.24-3	3.02-11	2.0-5	1.51-6
I135	8.20-3	2.00-10	4.0-6	4.99-5
Te132	1.80-3	4.39-11	2.0-5	2.20-6
Cs137	1.02-2	2.49-10	2.0-5	1.24-5
Ba140	3.25-5	7.91-13	2.0-5	3.96-8
La140	1.20-5	2.93-13	2.0-5	1.46-8
Ce144	2.85-6	6.93-14	1.0-5	6.93-9
Cr51	7.44-6	1.81-13	2.0-3	9.06-11
Total With H3	5.06+2	1.23-5		6.29-3
Total Without H3	8.77-2	2.14-9		2.18-3

(1) Maximum permissible concentration 10CFR20, Appendix B, Table 2 Column 2

(2) 5.06+2 = 5.06×10^2

Table 11A-5

RADIOACTIVE NUCLIDE INVENTORY IN GASEOUS
WASTE EFFLUENT FOR THE MAXIMUM EXPECTED BASES

Process Vent Release

Nuclide	Primary System Leakage Outside Containment (Ci)	Primary System Leakage Via Steam Generators Air Ejector (Ci)	Waste Gas Release (Ci)	Primary Systems Degas (Ci)	Total (Ci)	Ventilation Vent Release (Ci)
Kr83M	0.322+0	2.68+0	0.648-2	-	3.01+0	2.24-3(1)
Kr85M	1.50+0	1.21+0	0.695-1	-	1.37+1	1.74-2
Kr85	0.853+1	6.88+1	1.21+3	0.518+3	1.72+3	1.33+2
Kr87	0.933+0	7.55+0	1.82-2	-	8.50+0	3.52-3
Kr88	2.50+0	2.02+1	0.488-1	-	2.27+1	2.01-2
Kr89	0.783-1	6.32-1	1.38-3	-	7.12-1	1.20-5
Xe121M	1.17-2	8.99-2	0.913+0	0.135+0	1.14+0	1.13+0
Xe133M	2.45+0	1.97+1	0.558-1	0.0175+0	2.22+1	3.93-1
Xe133	2.10+1	1.69+2	0.823+2	24.8+0	2.97+2	8.25+0
Xe135M	0.848+0	6.84+0	1.63-2	-	7.70+0	1.02-2
Xe135	0.255+1	2.04+1	0.495-1	-	2.30+1	9.88-2
Xe137	1.27-1	1.02+0	2.27-3	-	1.15+0	2.40-5
Xe138	0.520+0	4.21+0	1.00-2	-	4.74+0	4.25-4
I131	2.01-4	1.65-5	0.393-9	-	2.18-4	4.36-3
I132	0.690-4	8.75-7	0.263-9	-	6.99-5	1.95-4
I133	0.313-3	3.55-6	0.610-9	-	3.17-4	7.34-4
I134	0.428-4	2.19-8	0.833-10	-	4.28-5	4.39-6
I135	1.66-4	6.16-7	0.0325-9	-	1.67-4	1.24-4

(1) 2.24-3 = 2.24×10^{-3}

Table 11A-6

RADIOACTIVE NUCLIDE INVENTORY IN GASEOUS
WASTE EFFLUENT FOR THE SYSTEM DESIGN BASIS

Process Vent Release

Nuclide	Primary System Leakage Outside Containment (Ci)	90% Primary System Leakage Inside Containment (Ci)	Primary System Leakage Via Steam Generators To Air Ejector (Ci)	Waste Gas Release (Ci)	Primary System Degas (Ci)	Total (Ci)	Elevated Release From Containment (Ci)
Kr83M	1.33+0(1)	4.01-2	7.71+1	2.59-2	-	7.85+1	4.45-3
Kr85M	5.99+0	3.07-1	3.47+2	2.78-1	-	3.54+2	3.41-2
Kr85	3.41+1	2.37+3	1.98+3	4.48+3	2.07+3	1.09+4	2.63+2
Kr87	3.73+0	6.25-2	2.17+2	7.29-2	-	2.21+2	6.94-3
Kr88	9.99+0	3.59-1	5.81+2	1.95-1	-	5.92+2	3.99-2
Kr89	3.13-1	2.14-4	1.82+1	5.50-3	-	1.85+1	2.38-5
Xe131M	4.67-2	2.02+1	2.59+0	3.65+0	0.48+0	2.71+1	2.24+0
Xe133M	9.80+0	6.99+0	5.68+2	2.23-1	0.07+0	5.85+2	7.77-1
Xe133	8.38+1	1.48+2	4.87+3	3.29+2	1.02+2	5.53+3	1.64+1
Xe135M	3.39+0	1.53-1	1.97+2	6.52-2	-	2.01+2	2.03-2
Xe135	1.02+1	1.76+0	5.88+2	1.98-1	-	6.00+2	1.96-1
Xe137	5.06-1	4.28-4	2.94+1	9.08-3	-	2.99+1	4.75-5
Xe138	2.08+0	7.59-3	1.21+2	4.01-2	-	1.23+2	8.43-4
I131	8.04-4	0.999-1	1.87-4	1.57-9	-	1.02-1	1.11-2
I132	2.76-4	0.449-2	1.53-5	1.05-9	-	4.83-3	4.99-4
I133	1.25-3	0.168-1	8.41-5	2.44-9	-	1.82-2	1.87-3
I134	1.71-4	0.101-3	6.25-7	3.33-10	-	2.74-4	1.12-5
I135	6.65-4	0.285-2	1.66-5	1.30-9	-	3.56-3	3.17-4

(1) 1.33+0 = 1.33×10^0

Table 11A-7

LIQUID WASTE QUANTITIES AND INITIAL ACTIVITIES FOR BVPS-1 OPERATION

(Maximum Expected)

	Source	Annual Volume (gal)	Initial Activity to Waste System (μCi/cc)	Radwaste System Decontamination Factor (including Reprocessing if required)
1)	Contaminated Laundry and Showers	100,000	5.66-6(1)	1.0
2)	Miscellaneous Low Level Wastes	100,000	5.66-6	1.0
3)	Sampling Sinks	35,000	1.93-0	1.0 + 5
4)	Spent Resin Flush	7,500	7.96-1	1.0 + 5
5)	High Level Wastes	7,500	7.96-1	1.0 + 5
6)	Laboratory Wastes	20,000	1.87-2	1.0 + 5
7)	Primary Coolant System Discharge	75,000	3.74-0	1.0 + 5
8)	Turbine Building Drains	135,000	1.92-6	1.0

(1) 5.66-6 = 5.66×10^{-6}
Activities do not include tritium

Table 11A-8

ACTIVITY FROM LAUNDRY DRAINS

DF Of Waste Disposal System for this source =	1.0
Decay Time in Waste Disposal System (Hours) =	0.0
Flow Rate (Gal/Yr) =	1.00+05

Nuclide	Initial Activity (μCi/cc)	Activity After Treatment (μCi/cc)	Discharge Rate From W.D. System (Ci/yr)
Mn-54	1.169-09[(1)]	1.169-09	4.424-07
Mn-56	4.383-08	4.383-08	1.659-05
Co-58	3.757-08	3.757-08	1.422-05
Fe-59	1.565-09	1.565-09	5.925-07
Co-60	1.127-09	1.127-09	4.266-07
Sr-89	1.544-09	1.544-09	5.846-07
Sr-90	3.631-11	3.631-11	1.375-08
Sr-91	7.096-10	7.096-10	2.686-07
Y-90	4.383-11	4.383-11	1.659-08
Y-91	2.463-10	2.463-10	9.322-08
Y-92	2.650-10	2.650-10	1.003-07
Zr-95	2.525-10	2.525-10	9.559-08
Nb-95	2.567-10	2.567-10	9.717-08
Mo-99	1.217-06	1.217-06	4.606-04
I-131	9.600-07	9.600-07	3.634-04
I-132	3.297-07	3.297-07	1.248-04
I-133	1.488-06	1.488-06	5.633-04
I-134	2.045-07	2.045-07	7.742-05
I-135	7.931-07	7.931-07	3.002-04
Te-132	9.767-08	9.767-08	3.697-05
Cs-137	4.737-07	4.737-07	1.793-04
Ba-140	1.590-09	1.590-09	6.020-07
La-140	5.531-10	5.531-10	2.094-07
Ce-144	1.325-10	1.325-10	5.017-08
Cr-51	1.419-09	1.419-09	5.372-07
TOTAL	5.656-06	5.656-06	2.141-03

(1) 1.169-09 = 1.169×10^{-9}

Table 11A-9

ACTIVITY FROM SAMPLING SINKS

DF of Waste Disposal System for This Source = 1.00+05[(1)]
Decay Time in Waste Disposal System (Hours) = 1.00+01
Flow Rate (Gal/Yr) = 3.50+04

Nuclide	Initial Activity (μCi/cc)	Activity After Treatment (μCi/cc)	Discharge Rate From W.D. System (Ci/yr)
Mn-54	3.978-04	3.974-09	5.265-07
Mn-56	1.492-02	1.017-08	1.348-06
Cu-58	1.279-02	1.273-07	1.687-05
Fe-59	5.328-04	5.294-09	7.013-07
Co-60	3.836-04	3.835-09	5.081-07
Sr-89	5.257-04	5.227-09	6.925-07
Sr-90	1.236-05	1.236-10	1.638-08
Sr-91	2.415-04	1.184-09	1.569-07
Y-90	1.429-05	1.339-10	1.773-08
Y-91	8.382-05	8.341-10	1.105-07
Y-92	9.021-05	1.315-10	1.742-08
Zr-95	8.595-05	8.557-10	1.134-07
Nb-95	8.737-05	8.666-10	1.148-07
Mo-99	4.141-01	3.735-06	4.948-04
I-131	3.268-01	3.153-06	4.177-04
I-132	1.122-01	5.515-08	7.306-06
I-133	5.065-01	3.641-06	4.824-04
I-134	6.961-02	2.530-10	3.352-08
I-135	2.699-01	9.606-07	1.273-04
Te-132	3.324-02	3.042-07	4.030-05
Cs-137	1.612-01	1.612-06	2.136-04
Ba-140	5.413-04	5.292-09	7.011-07
La-140	1.882-04	1.583-09	2.099-07
Ce-144	4.511-05	4.506-10	5.970-08
Cr-51	4.830-04	4.780-09	6.333-07
TOTAL	1.925-00	1.363-05	1.806-03

(1) 1.00+05 = 1.00 x 10^5

Table 11A-10

ACTIVITY FROM SPENT RESIN FLUSH AND HIGH LEVEL WASTES

DF of Waste Disposal System for This Source = 1.00+05(1)
Decay Time in Waste Disposal System (Hours) = 1.00+01
Flow Rate (Gal/Yr) = 1.50+04

Nuclide	Initial Activity (μCi/cc)	Activity After Treatment (μCi/cc)	Discharge Rate From W.D. System (Ci/yr)
Mn-54	1.645-04	1.643-09	9.330-08
Mn-56	6.168-03	4.205-09	2.388-07
Co-58	5.287-03	5.265-08	2.990-06
Fe-59	2.203-04	2.189-09	1.243-07
Co-60	1.586-04	1.586-09	9.004-08
Sr-89	2.173-04	2.161-09	1.227-07
Sr-90	5.110-06	5.110-11	2.902-09
Sr-91	9.986-05	4.896-10	2.780-08
Y-90	6.168-06	5.534-11	3.142-09
Y-91	3.466-05	3.449-10	1.958-08
Y-92	3.730-05	5.436-11	3.086-09
Zr-95	3.554-05	3.538-10	2.009-08
Nb-95	3.613-05	3.583-10	2.034-08
Mo-99	1.712-01	1.544-06	8.768-05
I-131	1.351-01	1.303-06	7.401-05
I-132	4.640-02	2.280-08	1.295E-06
I-133	2.094-01	1.505-06	8.547-05
I-134	2.878-02	1.046-10	5.939-09
I-135	1.116-01	3.972-07	2.255-05
Te-132	1.375-02	1.258E-07	7.141-06
Cs-137	6.667-02	6.667-07	3.785-05
Ba-140	2.238-04	2.188-09	1.242-07
La-140	7.783-05	6.550-10	3.719-08
Ce-144	1.865-05	1.863-10	1.058-08
Cr-51	1.997-04	1.976-09	1.122-07
TOTAL	7.959-01	5.637-06	3.201-04

(1) 1.00+05 = 1.00 x 10^5

Table 11A-11

ACTIVITY FROM LABORATORY WASTES

DF of Waste Disposal System for This Source =	1.00+05(1)
Decay Time in Waste Disposal System (Hours) =	1.00+01
Flow Rate (Gal/Yr) =	2.00+04

Nuclide	Initial Activity (μCi/CC)	Activity After Treatment (μCi/CC)	Discharge Rate From W.D. System (Ci/yr)
Mn-54	3.858-06	3.855-11	2.918-09
Mn-56	1.447-04	9.865-11	7.469-09
Co-58	1.240-04	1.235-09	9.352-08
Fe-59	5.167-06	5.134-11	3.887-09
Co-60	3.721-06	3.720-11	2.816-09
Sr-89	5.099-06	5.070-11	3.838-09
Sr-90	1.199-07	1.199-12	9.076-11
Sr-91	2.343-06	1.149-11	8.695-10
Y-90	1.447-07	1.298-12	9.829-11
Y-91	8.130-07	8.090-12	6.125-10
Y-92	8.750-07	1.275-12	9.654-11
Zr-95	8.337-07	8.300-12	6.284-10
Nb-95	8.475-07	8.405-12	6.363-10
Mo-99	4.017-03	3.623-08	2.743-06
I-131	3.169-03	3.058-08	2.315-06
I-132	1.089-03	5.349-10	4.050-08
I-133	4.913-03	3.531-08	2.674-06
I-134	6.752-04	2.454-12	1.858-10
I-135	2.618-03	9.317-09	7.054-07
Te-132	3.225-03	2.950-09	2.234-07
Cs-137	1.564-03	1.564-08	1.184-06
Ba-140	5.250-06	5.133-11	3.886-09
La-140	1.826-06	1.537-11	1.163-09
Ce-144	4.375-07	4.371-12	3.309-10
Cr-51	4.685-06	4.637-11	3.510-09
TOTAL	1.867-02	1.322-07	1.001-05

(1) 1.00+05 = 1.00 x 10^5

Table 11A-12

ACTIVITY FROM PRIMARY COOLANT SYSTEM LEAKAGE

DF of Waste Disposal System for This Source =	1.00+05(1)
Decay Time in Waste Disposal System (Hours) =	1.00+01
Flow Rate (Gal/Yr) =	7.50+04

Nuclide	Initial Activity (μCi/cc)	Activity After Treatment (μCi/cc)	Discharge Rate From W.D. System (Ci/yr)
Mn-54	7.722-04	7.715-09	2.190-06
Mn-56	2.896-02	1.974-08	5.606-06
Co-58	2.482-02	2.472-07	7.019-05
Fe-59	1.034-03	1.028-08	2.918-06
Co-60	7.447-04	7.445-09	2.114-06
Sr-89	1.020-03	1.015-08	2.881-06
Sr-90	2.399-05	2.399-10	6.812-08
Sr-91	4.689-04	2.299-09	6.526-07
Y-90	2.896-05	2.599-10	7.377-08
Y-91	1.627-04	1.619-09	4.597-07
Y-92	1.751-04	2.552-10	7.246-08
Zr-95	1.669-04	1.661-09	4.716-07
Nb-95	1.696-04	1.682-09	4.776-07
Mo-99	8.040-01	7.250-06	2.058-03
I-131	6.343-01	6.120-06	1.738-03
I-132	2.179-01	1.071-07	3.039-05
I-133	9.832-01	7.068-06	2.007-03
I-134	1.351-01	4.911-10	1.394-07
I-135	5.240-01	1.865-06	5.294-04
Te-132	6.454-02	5.905-07	1.676-04
Cs-137	3.130-01	3.130-06	8.887-04
Ba-140	1.051-03	1.027-08	2.917-06
La-140	3.654-04	3.076-09	8.732-07
Ce-144	8.757-05	8.748-10	2.484-07
Cr-51	9.377-04	9.280-09	2.635-06
TOTAL	3.737-00	2.647-05	7.514-03

(1) 1.00+05 = 1.00×10^5

Table 11A-13

ACTIVITY FROM MISCELLANEOUS LOW LEVEL WASTES

DF of Waste Disposal System for This Source =	1.0
Decay Time in Waste Disposal System (Hours) =	0.0
Flow Rate (Gal/Yr) =	1.00+05[(1)]

Nuclide	Initial Activity (μCi/cc)	Activity After Treatment (μCi/cc)	Discharge Rate From W.D. System (Ci/yr)
Mn-54	1.169-09	1.169-09	4.424-07
Mn-56	4.383-08	4.383-08	1.659-05
Co-58	3.757-08	3.757-08	1.422-05
Fe-59	1.565-09	1.565-09	5.925-07
Co-60	1.127-09	1.127-09	4.266-07
Sr-89	1.544-09	1.544-09	5.846-07
Sr-90	3.631-11	3.631-11	1.375-08
Sr-91	7.096-10	7.096-10	2.686-07
Y-90	4.383-11	4.383-11	1.659-08
Y-91	2.463-10	2.463-10	9.322-08
Y-92	2.650-10	2.650-10	1.003-07
Zr-95	2.525-10	2.525-10	9.559-08
Nb-95	2.567-10	2.567-10	9.717-08
Mo-99	1.217-06	1.217-06	4.606-04
I-131	9.600-07	9.600-07	3.634-04
I-132	3.297-07	3.297-07	1.248-04
I-133	1.488-06	1.488-06	5.633-04
I-134	2.045-07	2.045-07	7.742-05
I-135	7.931-07	7.931-07	3.002-04
Te-132	9.767-08	9.767-08	3.697-05
Cs-137	4.737-07	4.737-07	1.793-04
Ba-140	1.590-09	1.590-09	6.020-07
La-140	5.531-10	5.531-10	2.094-07
Ce-144	1.325-10	1.325-10	5.017-08
Cr-51	1.419-09	1.419-09	5.372-07
TOTAL	5.656-06	5.656-06	2.141-03

(1) 1.00+5 = 1.00×10^5

Table 11A-14

ACTIVITY FROM TURBINE BUILDING DRAINS

DF of Waste Disposal System for This Source = 1.0
Decay Time in Waste Disposal System (Hours) = 0.0
Flow Rate (Gal/Yr) = 1.35+05[(1)]

Nuclide	Initial Activity (μCi/cc)	Activity After Treatment (μCi/cc)	Discharge Rate From W.D. System (Ci/Yr)
Mn-54	7.014-10	7.014-10	3.585-07
Mn-56	2.227-09	2.227-09	1.138-06
Co-58	2.227-08	2.227-08	1.138-05
Fe-59	9.190-10	9.190-10	4.697-07
Co-60	6.784-10	6.784-10	3.467-07
Sr-89	9.139-10	9.139-10	4.670-07
Sr-90	2.189-11	2.189-11	1.119-08
Sr-91	1.101-10	1.101-10	5.625-08
Y-90	2.509-11	2.509-11	1.282-08
Y-91	1.485-10	1.485-10	7.588-08
Y-92	3.085-11	3.085-11	1.576-08
Zr-95	1.498-10	1.498-10	7.653-08
Nb-95	1.549-10	1.549-10	7.915-08
Mo-99	5.184-07	5.184-07	2.649-04
I-131	5.056-07	5.056-07	2.584-04
I-132	5.491-08	5.491-08	2.806-05
I-133	3.840-07	3.840-07	1.962-04
I-134	3.891-09	3.891-09	1.989-06
I-135	9.190-08	9.190-08	4.697-05
Te-132	4.314-08	4.314-08	2.204-05
Cs-137	2.854-07	2.854-07	1.459-04
Ba-140	8.768-10	8.768-10	4.481-07
La-140	5.568-10	5.568-10	2.845-07
Ce-144	7.936-11	7.936-11	4.056-08
Cr-51	8.205-10	8.205-10	4.193-07
TOTAL	1.918-06	1.918-06	9.801-04

(1) 1.35+05 = 1.00 x 10^5

Table 11A-15

ACTIVITY IN WASTE DISPOSAL SYSTEM

Calculated Maximum Allowable Discharge Rate (CI/YR) = 5.46-02[(1)]
Calculated Total Discharge Flow Rate (GPM) = 0.91

Nuclide	Actual Activity (μCi/cc)	Actual Disch. Rate (Ci/yr)
H3	2.785-01	5.060-02
Mn54	2.232-09	4.056-06
Mn56	2.285-08	4.152-05
Co58	7.153-08	1.300-04
Fe59	2.973-09	5.402-06
Co60	2.154-09	3.915-06
Sr89	2.937-09	5.336-06
Sr90	6.944-11	1.262-07
Sr91	7.897-10	1.432-06
Y90	7.746-11	1.407-07
Y91	4.693-10	8.527-07
Y92	1.703-10	3.095-07
Zr95	4.307-10	8.734-07
Nb95	4.881-10	8.869-07
Mo99	2.108-06	3.830-03
I131	1.770-06	3.217-03
I132	1.743-07	3.167-04
I133	2.146-06	3.900-03
I134	8.641-08	1.570-04
I135	7.305-07	1.327-03
Te132	1.713-07	3.113-04
Cs137	9.058-07	1.646-03
Ba140	2.971-09	5.398-06
La140	1.004-09	1.825-06
Ce144	2.531-10	4.599-07
Cr51	2.684-09	4.877-06
Total	2.785E-01	5.060-02
Total (Non-Tritium)	8.207E-06	1.491-02

(1) $5.46\text{-}02 = 5.46 \times 10^{-2}$

Table 11A-16

ACTIVITY IN COOLING TOWER BLOWDOWN

Discharge Canal Flow Rate (GPM) = 2.06-04[(1)]
Tritium Discharge Rate (Curies/Year) = 5.06-02

Nuclide	MPC (UC/CC)	Half Life (Days)	Actual Activity (UC/CC)	Ratio (UC/CC) /MPC
H-3	3.000-03	4.481-03	1.232-05	4.106-03
MN-54	1.000-04	3.121-02	9.876-14	9.876-10
MN-54	1.000-04	1.075-01	1.011-12	1.011-08
CO-58	9.000-05	7.161-01	3.164-12	3.516-08
FE-59	5.000-05	4.506-01	1.315-13	2.630-09
CO-60	3.000-05	1.919-03	9.531-14	3.177-09
SR-89	3.000-06	5.109-01	1.299-13	4.330-08
SR-90	3.000-07	1.044-04	3.072-15	1.024-08
SR-91	5.000-05	4.051-01	3.485-14	6.971-10
Y-90	2.000-05	2.665-00	3.427-15	1.713-10
Y-91	3.000-05	5.898-01	2.076-14	6.920-10
Y-92	6.000-05	1.499-01	7.535-15	1.256-10
ZR-95	6.000-05	6.521-01	2.126-14	3.544-10
NB-95	1.000-04	3.503-01	2.159-14	2.159-10
MO-99	4.000-05	2.795-00	9.324-11	2.331-06
I-131	3.000-07	8.061-00	7.832-11	2.611-04
I-132	8.000-06	9.583-02	7.711-12	9.639-07
I-133	1.000-06	8.747-01	9.495-11	9.495-05
I-134	2.000-05	3.646-02	3.823-12	1.911-07
I-135	4.000-06	2.795-01	3.232-11	8.079-06
TE-132	2.000-05	3.247-00	7.579-12	3.789-07
CS-137	2.000-05	1.088-04	4.007-11	2.004-06
BA-140	2.000-05	1.279-01	1.314-13	6.571-09
LA-140	2.000-05	1.674-00	4.442-14	2.221-09
CE-144	1.000-05	2.854-02	1.120-14	1.120-09
CR-51	2.000-03	2.775-01	1.187-13	5.937-11
Total			1.232-05	4.476-03

(1) 2.06-04 = 2.06×10^{-4}

Table 11A-17

ESTIMATED ANNUAL INVENTORY OF NUCLIDES RELEASED FROM THE TURBINE BUILDING LIQUID DRAINS

Nuclide	Activity (Ci/yr)
Mn-54	7.17-07[(1)]
Mn-56	2.23-06
Co-58	2.23-05
Co-60	5.93-07
Fe-59	9.39-07
Sr-89	9.34-07
Sr-90	2.24-08
Sr-91	1.13-07
Y-90	2.56-08
Y-91	1.52-07
Y-92	3.15-08
Zr-95	1.53-07
Nb-95	1.58-07
Mo-99	5.30-04
I-131	5.17-04
I-132	5.61-05
I-133	3.93-04
I-134	3.98-06
I-135	9.39-05
Te-132	4.41-05
Cs-137	2.92-04
Ba-140	8.96-07
La-140	5.69-07
Ce-144	8.11-08
Cr-51	8.39-07

(1) 7.17-07 = 7.17×10^{-7}

Table 11A-18

ESTIMATED ANNUAL INVENTORY OF
GASEOUS NUCLIDES RELEASED FROM
THE TURBINE BUILDING

Nuclide	Activity (Ci/yr)
Kr-83M	2.27-05[(1)]
Kr-85M	1.02-04
Kr-85	5.80-04
Kr-87	6.40-05
Kr-88	1.71-04
Kr-89	5.35-06
Xe-131M	7.60-07
Xe-133M	1.66-04
Xe-133	1.43-03
Xe-135M	5.80-05
Xe-135	1.73-04
Xe-137	8.65-06
Xe-138	3.56-05
I-131	1.84-03
I-132	2.00-04
I-133	1.40-03
I-134	1.42-05
I-135	3.35-04

(1) 2.27-05 = 2.27×10^{-5}

APPENDIX 11B

EVALUATION OF DOSES DUE TO NORMAL OPERATION

NOTE: Appendix 11B consists entirely of a report prepared for Duquesne Light Company by the NUS Corporation in 1972.(1) This report has been updated to include the responses to several AEC Questions but otherwise it was left in its original form and retained in the Updated FSAR to provide the bases of the BVPS-1 Environmental Surveillance Program. The current provisions of the Environmental Surveillance Program are in the Offsite Dose Calculation Manual. An Annual Environmental Report containing the results and conclusions of the Environmental Surveillance Program is filed with the NRC at the end of each calendar year. These reports cover each calendar year since the beginning of commercial operation in 1976.

(1) D.E. Martin, "Evaluation of the Doses from Radiation Exposure Due to Normal Operation of Unit 1 of the Beaver Valley Power Station," NUS Corporation, Rockville, Maryland (1972).

NUS-958

EVALUATION OF DOSES FROM RADIATION EXPOSURE

DUE TO NORMAL OPERATION OF UNIT 1

OF THE BEAVER VALLEY POWER STATION

Prepared for

DUQUESNE LIGHT COMPANY

By

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August, 1972

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I. INTRODUCTION

The Beaver Valley Power Station will consist of two light-water- cooled, pressurized water reactors having a combined net electrical generating capability of approximately 1,700 megawatts. The station site consists of 449 acres located on the south bank of the Ohio River in Shippingport Borough, Beaver County, Pennsylvania. In the year 1990 the population within 50 miles of the station is projected to total 4.27 million persons.(1) The object of this report is to evaluate the potential radiological impact upon this population of the normal operation of Unit 1 of the Beaver Valley Power Station. The combined radiological impact of Units 1 and 2 is discussed fully in a separate report, NUS-907.

During normal operation, small amounts of liquid and gaseous radioactive materials will be released to the environment under closely supervised and controlled conditions. These discharges are limited by Title 10, Part 20, of the Code of Federal Regulations (10CFR20), Appendix I, and comply with applicable Technical Specifications. The purpose of this study is to quantify and assess the potential radiation exposure to the public resulting from these anticipated radioactive emissions. This evaluation considers the total dose to the population within a 50 mile radius of the plant as well as the per capita dose. The study also includes the dose to hypothetical "maximum" individuals who may: reside full time at the nearest site boundary; or obtain their entire daily drinking water requirement from the undiluted cooling tower blowdown discharge; or consume 50 grams daily of fish grown in waters always containing discharge concentrations of all radionuclides. All significant environmental pathways by which the activity released from the plant could conceivably reach the public have been evaluated in detail. A visual illustration is presented in Figure 11B-1, "Exposure Pathways".

Since the applicable regulations are based on a consideration of average annual dose, this study is directed at determining the doses averaged over a year. The estimated doses to individuals and the population are compared to existing and proposed dose guidelines, to normal background radiation doses, and to other ordinarily acceptable radiation exposure levels.

Calculational methods employed are those suggested or recommended by such bodies as the International Atomic Energy Agency, the NRC or the Internal Conference on Radiation Protection where such methods are available. In situations where the lack of knowledge has precluded accuracy, conservative assumptions have been made in an attempt to eliminate the possibility of under-estimating the particular exposure level.

The maximum quantities of radioactivity expected to be released annually from BVPS Unit 1 under normal operation have been calculated by Stone & Webster Engineering Corporation and tabulated elsewhere in the report. Allowances have been made for significant failed fuel cladding (0.25%), a continuous reactor coolant system to steam system leak rate of 0.1 gpm, continuous reactor coolant leakage inside and outside of the containment structure and annual containment purging as well as nominal additional releases due to laboratory spills and other incremental releases. Design basis releases, based on 1.0% failed fuel cladding have also been provided to NUS by Stone & Webster for radiological impact evaluation. Throughout the text of this report, doses calculated on the basis of 1.0% defective cladding will be presented in parenthesis immediately after doses calculated on the basis of the maximum expected releases, when a significant numerical difference occurs. Only a single number is presented when the difference in fuel failure does not appreciably affect the result, as in the case of the water ingestion whole body exposure, due almost entirely to tritium.

II. SUMMARY AND CONCLUSIONS

This report evaluates the maximum expected radiation exposure to the general public or to individuals thereof resulting from the operation of Unit 1 of the Beaver Valley Power Station. The quantities of the various radioisotopes released from the station in gaseous or liquid form were obtained from an assumed combination of station operating conditions which would provide maximum expected annual average discharges. These assumptions include 0.25% defective fuel.* In addition, conservative environmental dilution and concentration factors have been used so that the annual average doses calculated in this study are the maximum values which could reasonably be expected to occur due to the normal operation of the Beaver Valley Power Station.

All calculated human exposure levels resulting from the operating of Beaver Valley Unit 1 have been found to amount to only small fractions of normally accepted levels of radiation exposure. The limits set by law in 10CFR20 for maximum permissible exposure have not even been remotely approached. Likewise, all exposure levels fall within the limits proposed in the new Appendix I to 10CFR50.

A. Maximum Individual Exposure

The most significant source of whole body radiation exposure is the external dose received due to release of the noble gases krypton and xenon. The maximum potential dose from this source is calculated to be 0.27 (0.74) mrem per year. About 97% (71%) of this dose is due to the releases from the primary auxiliary building vent although only 6.3% (1.4%) of the total annual noble gas activity is released from these points. To derive this entire dose, an individual would have to continually remain outdoors at a position on the site boundary about 2,160 feet to the northeast of the Unit 1 containment. A person so situated would receive a thyroid inhalation dose due to the released radioiodines of 0.05 (0.12) mrem per year.

If an individual were to depend continually and exclusively upon the cooling tower discharge flow for his source of potable water, he would receive an annual whole body dose of 0.63 mrem. The associated maximum organ dose from this source is calculated to be 1.10 (1.51) mrem per year to the thyroid.

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* Radiation exposure levels have also been evaluated on the design basis criteria of 1.0% defective fuel.

The ingestion of 50 grams daily of fish grown in waters containing discharge concentrations of all effluent radionuclides would result in an annual whole body dose of 0.05 (0.21) mrem. The maximum organ dose would be 0.11 (0.51) mrem to the liver.

The external radiation that would be received if an individual were to swim in the undiluted effluent yields a calculated whole body exposure rate of 6.10×10^{-7} (3.85×10^{-6}) mrem per hour. Because tritium, the most predominant isotope, can only irradiate the skin, the skin dose rate exceeds the whole body dose rate. The skin dose rate is calculated to be 0.0001 mrem per hour.

Gaseous radioiodine releases may be deposited on grass, ingested by a grazing cow and subsequently secreted in commercially available milk supplies. The maximum dose from this source occurs if the receptor is a young child. Assuming no delay between production and consumption and no dilution with other milk supplies, the potential annual dose from this source is calculated to be 0.15 (0.42) mrem at the nearest known dairy farm.

The maximum total whole body dose that an individual could potentially receive from plant contributed sources is 0.95 (1.58) mrem per year. This dose includes the maximum dose from water ingestion, fish ingestion, swimming 200 hours in the river and external exposure from the released radiogases. In comparison, the annual average whole body exposure from naturally occurring background radiation in the United States is 130 millirem.(2) The average in Pennsylvania is 125 millirem per year.(2)

The maximum potential individual dose from Beaver Valley is also small in comparison with other ordinary and acceptable individual radiation exposure levels.

In the immediate area of the Beaver Valley site, the dose rate from naturally occurring terrestrial radiation has been found by actual survey to vary spatially about 50 mrem per year.(3) The variation increases even more as the area of examination increases and approaches 90 mrem per year for the Pittsburgh area in general. The exposure to an individual from a single chest X-ray may be as much as 170 mrem.(2) Wearing a particular kind of watch may yield an incremental annual whole body dose of 4 mrem(2). The inhabitation of a stone or masonry dwelling rather than a frame house has been calculated to yield an average additional annual whole body dose of 40 mrem and a maximum annual incremental exposure of over 500 mrem(4).

Thus, the maximum potential incremental exposure to man from plant contributed radioactivity represents only a small fraction of exposure increments common to ordinary experience.

B. Population Exposure

The largest source of population exposure from the expected plant effluents is due to the external radiation received from the released radiogases.

The population projected to exist on the area surrounding the Beaver Valley plant in 1990 is expected to receive 4.90 (37.3) man-rem of whole body exposure from this source. Of this total, 3.69 (34.9) man-rem is due to releases from the elevated release point above the cooling tower and 1.21 (2.4) man-rem is due to the containment vent releases.

The total population dose from water ingestion has been estimated using 1970 census data and all known municipal water intakes downstream of the plant within a 50 mile radius. The estimated total whole body dose exposure from this source is 0.49 man-rem annually.

A total annual population exposure from fish ingestion has been calculated and is 4.48×10^{-5} (1.78×10^{-4}) man-rem.

The total plant contributed whole body population exposure from all sources amounts to 5.39 (37.8) man-rem per year. Based on the estimate of 4.27 million persons living within 50 miles of the station by 1990, the per capita dose is 1.26×10^{-3} (8.85×10^{-3}) mrem per year. This total may be compared to the population exposure received annually from naturally occurring background radiation which amounts to an estimated 555,100 man-rem per year within 50 miles of the station (based on the U.S. average background exposure of 130 mrem per year per capita[2]).

C. Exposure of Aquatic Organisms

The maximum possible radiation exposure of several varieties of biota has been calculated on an extremely conservative basis. These exposure levels, presented in Table 11B-11, are based on the continual presence in the Ohio River of the annual average discharge concentrations of all radionuclides. Also, it has been assumed that radioactivity concentrations in the river sediment may be as much as ten thousand times those in the ambient water. It was further assumed that a particular organism, or the deep roots of plants, may exist at sediment depths of tens of inches. These assumptions maximize the potential exposure level to aquatic biota and minimize the probability of occurrence. The resulting exposure rate is about 57 (344) mrad per year.

The dose to continually bottom feeding fish is calculated to be about 29 (173) mrad per year, using the same conservative basis. Most of this exposure is due to the high levels of radioactivity assumed to potentially exist in the river sediment. For a fish which does not normally dwell at the river bottom the annual exposure from all internal and external sources is calculated to be 2.52 (4.50) mrad.

If credit is taken for dilution of the discharge of the river, the dose estimates are significantly reduced. The maximum exposure in sediments is then about 0.10 (0.58) mrad per year. Bottom feeding fish are calculated to receive about 0.05 (0.29) mrad per year and free-swimming fish are calculated to receive only 0.004 (0.008) mrad per year. These exposure levels will exist downstream from the station and are much more realistic in terms of probability of occurrence than those presented earlier for undiluted discharge water.

The analysis has resulted in no estimated exposure level which exceeds the guideline in 10CFR20[(5)] for the maximum permissible radiation exposure of man (500 mrem/yr.). In view of this fact and the extreme conservatism of the models and assumptions employed in the derivation of the dose it is concluded that no significant radiological hazard to the aquatic population of the Ohio River will occur due to operation of Unit 1 of the Beaver Valley Power Station.

D. Conclusions

The maximum radiation exposure of the general public or any individual thereof has been evaluated. The highest calculated per capita dose to the population within 50 miles of the power station is 1.26×10^{-3} (8.85×10^{-3}) mrem per year. This amounts to less than one (six) one-thousandth(s) of one percent of the allowable dose of 170 mrem/year under Title 10, Part 20 of the Code of Federal Regulations.[(5)]

A hypothetical "maximum" individual receiving the maximum individual dose from all significant pathways would receive an annual radiation exposure dose of 0.95 (1.58) mrem to the whole body. The maximum organ dose from this exposure would occur in the thyroid and amount to 1.51 (2.67) mrem annually. The whole body dose, the thyroid dose and the dose to blood forming organs, such as the liver, are limited by 10CFR20 to 500 mrem per year to any individual of the general public. Thus, the maximum dose to this hypothetical "maximum" individual amounts to only 0.3% (0.5%) of the maximum permissible dose.

Since no Federal limit or regulation regarding radiation exposure of the public will be exceeded or even approached, it is concluded that no significant radiological hazard will be posed by the normal operation of Unit 1 of the Beaver Valley Power Station.

III. RADIATION EXPOSURE FROM GASEOUS EFFLUENTS

There are a number of potential pathways through which local populations may be exposed to the radioactive effluents from nuclear power plant operations. These are illustrated in Figure 11B-1. Three general pathways may be identified for the gaseous effluents. (1) direct radiation exposure, (2) inhalation exposure and (3) exposure through food chains. The importance of these individual pathways is determined by the spectrum and relative abundance of the various radioisotopes in the gaseous effluent. Thus, the first step in estimating population exposure is a quantitative identification of the constituents in the projected gaseous effluent.

A. Projected Gaseous Effluents

The maximum expected quantities of gaseous radioactivity to be released from Unit 1 of the Beaver Valley Power Station on an annual basis are shown in Table 11B-1, "Estimated Gaseous Effluents". Allowance has been made for 0.25% fuel defects, the highest value expected to be encountered in normal operation, continuous reactor coolant system leakage and annual containment purging. Release rates based on the design basis criteria of 1.0% fuel defects are also presented.

All normally processed waste gas will be released through an elevated release stack situated about fifteen feet above the top of the Unit 1 cooling tower. This position is about 160 meters above ground level according to the present design. Gases exhausting at this point will originate from reactor coolant system degasification, main condenser air ejectors and reactor coolant leakage within the containment.

Due to the necessity for annual refueling, the containments are expected to be fully purged once per year. The released gas from this source are planned to be released directly to the atmosphere via the exhaust vent above the primary auxiliary building. However, in the unanticipated occurrence of 1% fuel defects about 90% of the containment air would be filtered through a long term delay charcoal adsorption system prior to exhaust through the elevated release point on the Unit 1 cooling tower.

TABLE 11B-1

ESTIMATED GASEOUS EFFLUENTS

Isotope	Primary Auxiliary Building Releases(6) curies per year 0.25% ff*	Primary Auxiliary Building Releases(6) curies per year 1.0% ff	Elevated Cooling Tower Releases(6) curies per year 0.25% ff	Elevated Cooling Tower Releases(6) curies per year 1.0% ff
Kr-83m	2.24-3**	4.45-3	3.01	7.85+1
Kr-85m	1.72-2	3.41-2	1.37+1	3.54+2
Kr-85	1.33+2	2.63+2	1.72+3	1.09+4
Kr-87	3.50-3	6.94-3	8.50	2.21+2
Kr-88	2.01-2	3.99-2	2.27+1	5.92+2
Kr-89	1.20-5	2.38-5	7.12-1	1.85+1
Xe-131m	1.13	2.24	1.14	2.71+1
Xe-133m	3.93-1	7.77-1	2.22+1	5.85+2
Xe-133	8.25	1.64+1	2.97+2	5.53+3
Xe-135m	1.02-2	2.03-2	7.70	2.01+2
Xe-135	9.88-2	1.96-1	2.30+1	6.00+2
Xe-137	2.40-5	4.75-5	1.15	2.99+1
Xe-138	4.25-4	8.43-4	4.74	1.23+2
Total Noble Gases	1.43+2	2.83+2	2.13+3	1.93+4
I-131	4.36-3	1.11-2	2.18-4	1.02-1
I-132	1.95-4	4.99-4	6.99-5	4.83-3
I-133	7.34-4	1.87-3	3.17-4	1.82-2
I-134	4.39-6	1.12-5	4.28-5	2.74-4
I-135	1.24-4	3.17-4	1.67-4	3.56-3

* The abbreviation “ff” is used to denote “failed fuel”.

** 2.24-3 = 2.24 x 10^{-3}; typical for all numeric values.

The validity of the assumptions employed in estimating the gaseous activity release is illustrated by comparison with the gaseous activities which have been released from operating pressurized water reactors to date. The annual average release rate experience[7] is summarized in Table 11B-2. Based on this experience (with the exception of the first year of Ginna experience which is atypical), the BVPS Unit 1 extrapolated release rate would range from a minimum of 21 Ci/yr to a maximum of 2,562 Ci/yr. The total release rate estimate for Beaver Valley, upon which this report is based, is 2,273 (19,583) curies per year. Approximately 82%, (57%) of this total is due to Kr-85.

B. External Exposure From Gaseous Effluents

As has been mentioned earlier, there are three general pathways through which humans may be exposed to the gaseous radioactivity released from a nuclear plant. The importance of each of the pathways is determined by the quantity and chemical nature of the gases released. It may be observed in Table 11B-1 that the primary constituents in the gaseous effluent are the noble gases, krypton and xenon. Since the noble gases do not react chemically with other substances under normal conditions, there is no physical basis for either transport through food chains or reconcentration within the human body for these gases. Thus, the most significant exposure pathway for released noble gases is direct external radiation to the whole body.

The opposite is true of the released radioisotopes of iodine for which inhalation and food chain transport are critical pathways.

TABLE 11B-2

PWR WASTE GAS RELEASE EXPERIENCE

Plant	Annual Average Release Rate μCi/(sec)/1000 Mwe-hr	Extrapolated Expected Beaver Valley Release Rate* Ci/yr
Indian Point #1 (265 Mwe)	1.2×10^{-2} (1963-1970)	2,562 (Maximum)
Yankee Rowe (175 Mwe)	1.0×10^{-4} (1964-1970)	21 (Minimum)
San Onofre (430 Mwe)	5.0×10^{-3} (1967-1970)	1,067
Connecticut Yankee (600 Mwe)	3.0×10^{-3} (1967-1970)	641
Ginna (420 Mwe)	1.4×10^{-1} (1970)	29,891 (Atypical Maximum)

* Based on a power level of 857 Mwe in Unit 1 and operation at an annual load factor of 90%.

External radiation from iodine is generally insignificant in comparison with the internal dose derived through inhalation.

1. Maximum Individual External Exposure

Considering both the magnitude of the resulting exposure and the probability of occurrence, the external radiation exposure from passing clouds of released radiogases is considered to be the most significant exposure pathway for human radiation from the Beaver Valley Power Station. Only doses from direct and continuous ingestion of the cooling tower blowdown have been calculated to be higher.

Due to the variation in height above ground at which the two types of gaseous releases are made, two different methods have been used to compute external dose rates.

Since the dose calculations include atmospheric diffusion parameters, meteorological information was required. The latest data compilation covering the period from 9/5/70 to 9/5/71 was derived from data obtained from the on-site meteorological station. The data reduction was accomplished using the NUS computer program, WINDVANE. The WINDVANE output was adequate for the determination of the exposure rate from the auxiliary building releases, which were assumed to occur at ground level, and for the beta component of the exposure rate from the elevated releases. For the accurate calculation of the exposure rate due to gamma radiations from gases released above the cooling tower, it was necessary to make use of the NUS computer code, WINDOR.

The WINDOR* code, utilizing the atmospheric data summary obtained from WINDVANE, the source data presented in Table 11B-1, and pertinent disintegration data, produces tabulated whole body ground level gamma exposure rates. The output provides dose rates at distances from the source of 100 to 88,000 meters, in 16 directions. The code accounts for radioactive decay, attenuation in air, and the buildup of radioactive daughter products. The degree of accuracy involved in the use of this code is considered excellent, both absolutely and in comparison with other calculational alternatives. Figure 11B-2 presents a visual illustration of

* The WINDOR code basically performs a volume integration over a spatial source distribution computed from the WINDVANE output. The point kernel used to calculate the differential gamma dose rate is very similar to that presented in TID 24190, Meteorology and Atomic Energy".[(8)]

the WINDOR results for the east-northeast direction. As can be seen from the illustration, the maximum site boundary exposure rate from this source is very small, amounting to 8.5×10^{-3} (2.1×10^{-1}) mrem per year.

External doses from the remaining sources were calculated using the ICRP[9] "infinite semispherical cloud" model;[8] that is, the exposed individual is assumed to be located at the center of an infinitely large semispherical cloud of uniform radioactivity concentration equal to that of the centerline, or maximum concentration level of the plume at the specified distance. A detailed description of this calculational model (and others used to derive doses due to gaseous effluents) is contained in Appendix A, "Computational Methods for Doses Resulting from Gaseous Effluents". Table 11B-3 presents the necessary disintegration energies of the effluent noble gases.

The ICRP method for calculating whole body dose assumes that beta radiation with a maximum energy of 0.1 MeV or greater is considered as contributing to the external whole body to the same extent as gamma radiation. This is a conservative assumption and in the case of nuclides which are primarily beta emitters, such as Kr-85, results in a substantial overestimate of the genetically significant dose.[11]

After a comparison of the combined external dose rate from all sources of radiogases at various off-site locations around and about the Beaver Valley Station it was found that the highest dose rate on land occurred at the site boundary, about 2,160 feet northeast of the center of the Unit 1 containment. This location is estimated to be 1,860 feet northeast of the auxiliary building and about 1,380 feet ENE of the elevated release point above the Unit 1 cooling tower. The total dose from all sources is estimated at 0.27 (0.74) mrem per year. The individual components of the dose, and the χ/Q values used to obtain them are presented in Table 11B-4. Nearly all of this dose, over 96% (70%), arises from the auxiliary building releases, conservatively assumed to be at ground level. Less than 4% (30%) of the dose is due to releases occurring above the cooling tower, although that is where almost 94% (99%) of the annual activity release is expected to take place.

TABLE 11B-3

BETA, GAMMA AND TOTAL DISINTEGRATION ENERGIES OF EFFLUENT NOBLE GASES[10]

Isotope	Beta Energy MeV/dis	Gamma Energy MeV/dis	Total Energy MeV/dis
Kr-83m	0.034	0.005	0.039
Kr-85m	0.101	0.156	0.257
Kr-85	0.228	0.0021	0.2301
Kr-87	1.014	0.856	1.870
Kr-88	0.307	2.00	2.307
Kr-89	1.001	2.219	3.220
Xe-131m	0.135	0.022	0.157
Xe-133m	0.191	0.0239	0.2149
Xe-133	0.126	0.0728	0.1988
Xe-135m	0.09	0.44	0.53
Xe-135	0.303	0.248	0.551
Xe-137	1.37	0.192	1.562
Xe-138	0.565	0.932	1.497

2. External Population Exposure

The annual average external population exposure from released radiogases has been calculated separately for the anticipated auxiliary building releases and releases from the elevated position above the cooling tower. Additionally, the population exposure due to the elevated releases has been separated into beta and gamma components.

For each of these separate sources population has been computed for each of the population ring sectors shown in Figures 2.2-1 and 2.2-2 of the Environmental Report, Operating License Stage for BVPS Unit 1. The population dose was calculated by multiplying the average dose for a sector by the sector population projected for the year 1990, and summing over all sectors out to 50 miles from the site. The results are presented in Table 11B-5, Parts A, B, C and D. The total external noble gas population dose from the various sources amounts to about 4.90 (37.3) man-rem per year, distributed as follows on page 27:

TABLE 11B-4

DATA PERTAINING TO THE CALCULATION OF MAXIMUM INDIVIDUAL EXPOSURE FROM GASEOUS RELEASES

Release Location	Relative Position of Nearest Site Boundary	Value of χ/Q at Nearest Site Boundary	Calculated Dose Rate rem/yr 0.25% ff*	Calculated Dose Rate rem/yr 1.0% ff*
Primary Auxiliary Building Vent	1860 ft. NE	3.108×10^{-5} sec/m^3	2.626×10^{-4}	5.194×10^{-4}
Elevated Release Point Above the Unit 1 Cooling Tower	1380 ft. ENE	3.057×10^{-9} sec/m^3		
Beta Component			3.695×10^{-7}	3.223×10^{-6}
Gamma Component			8.479×10^{-6}	2.135×10^{-4}
	Total Annual Noble Gas Exposure Rate at Nearest Site Boundary is		2.714×10^{-4}	7.361×10^{-4}

* In this report “ff” denotes “failed fuel”.

TABLE 11B-5

POPULATION EXPOSURE FROM RELEASED NOBLE GASES

PART A: EXPOSURE FROM THE PRIMARY AUXILIARY BUILDING VENT RELEASES

PART B: BETA EXPOSURE FROM THE ELEVATED RELEASES

PART C: GAMMA EXPOSURE FROM THE ELEVATED RELEASES

PART D: POPULATION EXPOSURE FROM ALL GASEOUS RELEASES

TABLE 11B-5

PART A: EXPOSURE FROM THE PRIMARY AUXILIARY BUILDING VENT RELEASES

EXTERNAL WHOLE BODY POPULATION RADIATION DOSES, BEAVER VALLEY POWER STATION, UNIT 1, DUQUESNE LIGHT CO./NUS CORP., 8/72

THE POPULATION DOSES PRESENTED ON THIS PAGE ARE THOSE ESTIMATED TO OCCUR IN THE YEAR 1990.

THE DOSES PRESENTED ARE IN UNITS OF MAN-REM.

AUXILIARY BUILDING RELEASE, GAMMA AND BETA DOSE, 0.25% FAILED FUEL.

DIR.	0-1 MILES	1-2 MILES	2-3 MILES	3-4 MILES	4-5 MILES	5-10 MILES	10-20 MILES	20-30 MILES	30-40 MILES	40-50 MILES	DIRECTION TOTAL
NNE	0.	1.100E-02	3.829E-03	2.394E-03	1.908E-03	4.969E-02	3.305E-02	2.918E-02	4.399E-02	2.199E-02	1.970E-01
NE	2.134E-02	3.587E-03	4.840E-03	6.234E-03	1.481E-03	5.564E-02	3.120E-02	8.818E-03	1.188E-02	5.410E-03	1.504E-01
ENE	5.816E-03	9.720E-04	8.998E-04	6.521E-04	3.311E-04	1.983E-02	1.152E-02	1.282E-02	9.980E-03	3.281E-03	6.610E-02
E	5.466E-04	5.811E-04	2.149E-03	1.941E-03	2.639E-04	1.721E-02	1.052E-02	1.934E-02	1.343E-02	4.043E-03	7.002E-02
ESE	2.814E-03	2.481E-03	7.465E-04	9.822E-04	6.006E-04	1.063E-02	1.179E-02	4.077E-02	1.854E-02	6.728E-03	9.609E-02
SE	2.586E-04	9.946E-04	1.117E-03	2.792E-04	7.965E-05	2.107E-03	1.014E-02	5.006E-02	1.884E-02	7.528E-03	9.141E-02
SSE	1.554E-04	9.978E-05	1.442E-04	1.311E-04	2.782E-04	8.368E-04	4.086E-03	1.642E-02	6.822E-03	3.300E-03	3.227E-02
S	0.	1.546E-04	7.119E-04	2.573E-04	3.357E-04	4.184E-04	2.488E-03	2.307E-03	2.527E-03	2.418E-03	1.162E-02
SSW	2.410E-04	1.242E-04	1.384E-04	1.663E-04	8.330E-05	5.342E-04	1.625E-03	1.287E-03	8.680E-04	8.382E-04	5.905E-03
SW	0.	1.420E-04	5.295E-04	1.654E-04	8.055E-05	6.902E-04	1.856E-03	1.349E-03	5.399E-04	5.283E-04	5.881E-03
WSW	0.	3.671E-05	3.745E-04	1.128E-04	6.317E-05	3.593E-03	2.027E-03	1.195E-03	6.872E-04	1.288E-03	9.377E-03
W	0.	8.081E-05	1.468E-04	1.290E-04	7.293E-05	8.598E-03	1.832E-03	4.718E-04	8.317E-04	3.837E-03	1.600E-02
WNW	1.177E-03	3.081E-04	6.929E-05	1.063E-03	1.212E-03	1.019E-02	3.972E-03	3.818E-03	5.912E-03	9.609E-03	3.733E-02
NW	2.658E-02	7.901E-02	6.416E-03	3.773E-03	1.912E-03	4.229E-03	7.097E-03	1.118E-02	1.496E-02	1.376E-02	1.689E-01
NNW	2.404E-03	5.629E-03	8.526E-04	4.613E-04	7.724E-04	7.745E-03	7.433E-03	1.517E-02	2.297E-02	1.557E-02	7.901E-02
N	0.	7.918E-04	2.020E-03	4.155E-03	2.167E-03	1.363E-02	1.674E-02	3.894E-02	5.979E-02	3.020E-02	1.684E-01
Annulus Total	6.134E-02	1.060E-01	2.499E-02	2.290E-02	1.164E-02	2.056E-01	1.574E-01	2.531E-01	2.326E-01	1.303E-01	1.206E+00

TABLE 11B-5

PART A: (Continued)

EXTERNAL WHOLE BODY POPULATION RADIATION DOSES, BEAVER VALLEY POWER STATION, UNIT 1, DUQUESNE LIGHT CO./NUS CORP., 8/72

THE POPULATION DOSES PRESENTED ON THIS PAGE ARE THOSE ESTIMATED TO OCCUR IN THE YEAR 1990.

THE DOSES PRESENTED ARE IN UNITS OF MAN-REM.

AUXILIARY BUILDING RELEASE, GAMMA AND BETA DOSE, 1.0% FAILED FUEL.

DIR.	0-1 MILES	1-2 MILES	2-3 MILES	3-4 MILES	4-5 MILES	5-10 MILES	10-20 MILES	20-30 MILES	30-40 MILES	40-50 MILES	DIRECTION TOTAL
NNE	0.	2.176E-02	7.573E-03	4.736E-03	3.775E-03	9.829E-02	6.538E-02	5.772E-02	8.702E-02	4.350E-02	3.898E-01
NE	4.221E-02	7.095E-03	9.575E-03	1.233E-02	2.929E-03	1.101E-01	6.171E-02	1.744E-02	2.350E-02	1.070E-02	2.976E-01
ENE	1.150E-02	1.923E-03	1.780E-03	1.290E-03	6.550E-04	3.922E-02	2.279E-02	2.536E-02	1.974E-02	6.491E-03	1.308E-01
E	1.081E-03	1.149E-03	4.251E-03	3.840E-03	5.220E-04	3.404E-02	2.081E-02	3.825E-02	2.656E-02	7.998E-03	1.385E-01
ESE	5.567E-03	4.907E-03	1.477E-03	1.943E-03	1.188E-03	2.104E-02	2.332E-02	8.065E-02	3.667E-02	1.331E-02	1.901E-01
SE	5.115E-04	1.967E-03	2.210E-03	5.523E-04	1.576E-04	4.168E-03	2.007E-02	9.903E-02	3.726E-02	1.489E-02	1.808E-01
SSE	3.074E-04	1.974E-04	2.852E-04	2.594E-04	5.502E-04	1.655E-03	8.082E-03	3.247E-02	1.349E-02	6.528E-03	6.383E-02
S	0.	3.057E-04	1.408E-03	5.090E-04	6.641E-04	8.277E-04	4.922E-03	4.563E-03	4.999E-03	4.783E-03	2.298E-02
SSW	4.767E-04	2.457E-04	2.738E-04	3.289E-04	1.648E-04	1.057E-03	3.214E-03	2.545E-03	1.717E-03	1.658E-03	1.168E-02
SW	0.	2.808E-04	1.047E-03	3.272E-04	1.593E-04	1.365E-03	3.672E-03	2.669E-03	1.068E-03	1.045E-03	1.163E-02
WSW	0.	7.262E-05	7.407E-04	2.230E-04	1.249E-04	7.108E-03	4.010E-03	2.363E-03	1.359E-03	2.548E-03	1.855E-02
W	0.	1.598E-04	2.903E-04	2.553E-04	1.443E-04	1.701E-02	3.625E-03	9.332E-04	1.645E-03	7.590E-03	3.165E-02
WNW	2.329E-03	6.095E-04	1.371E-04	2.102E-03	2.397E-03	2.015E-02	7.857E-03	7.552E-03	1.169E-02	1.901E-02	7.384E-02
NW	5.258E-02	1.563E-01	1.269E-02	7.464E-03	3.781E-03	8.366E-03	1.404E-02	2.212E-02	2.960E-02	2.721E-02	3.341E-01
NNW	4.756E-03	1.113E-02	1.686E-03	9.125E-04	1.528E-03	1.532E-02	1.470E-02	3.001E-02	4.544E-02	3.079E-02	1.563E-01
N	0.	1.566E-03	3.996E-03	8.219E-03	4.286E-03	2.696E-02	3.311E-02	7.702E-02	1.183E-01	5.973E-02	3.332E-01
Annulus Total	1.213E-01	2.097E-01	4.942E-02	4.529E-02	2.303E-02	4.066E-01	3.113E-01	5.007E-01	4.600E-01	2.578E-01	2.385E+00

TABLE 11B-5

PART B: BETA EXPOSURE FROM THE ELEVATED RELEASES

EXTERNAL WHOLE BODY POPULATION RADIATION DOSES, BEAVER VALLEY POWER STATION, UNIT 1, DUQUESNE LIGHT CO./NUS CORP., 8/72

THE POPULATION DOSES PRESENTED ON THIS PAGE ARE THOSE ESTIMATED TO OCCUR IN THE YEAR 1990.

THE DOSES PRESENTED ARE IN UNITS OF MAN-REM.

COOLING TOWER RELEASE, BETA DOSE ONLY, 0.25% FAILED FUEL.

DIR.	0-1 MILES	1-2 MILES	2-3 MILES	3-4 MILES	4-5 MILES	5-10 MILES	10-20 MILES	20-30 MILES	30-40 MILES	40-50 MILES	DIRECTION TOTAL
NNE	0.	1.761E-04	1.727E-04	2.776E-04	4.122E-04	2.651E-02	3.544E-02	3.840E-02	6.481E-02	3.153E-02	1.977E-01
NE	7.699E-05	9.522E-05	3.953E-03	1.143E-03	4.632E-04	3.868E-02	4.046E-02	1.367E-02	2.026E-02	8.947E-03	1.277E-01
ENE	3.592E-05	3.850E-05	1.363E-04	2.624E-04	2.478E-04	3.603E-02	3.951E-02	5.206E-02	4.467E-02	1.450E-02	1.875E-01
E	4.095E-06	4.376E-05	5.227E-04	8.817E-04	2.936E-04	4.596E-02	5.276E-02	1.153E-01	8.888E-02	2.680E-02	3.315E-01
ESE	2.148E-05	2.912E-04	2.979E-04	8.939E-04	9.336E-04	3.660E-02	7.239E-02	2.925E-01	1.462E-01	5.319E-02	6.034E-01
SE	3.409E-06	2.254E-04	7.082E-04	3.532E-04	1.594E-04	8.510E-03	6.881E-02	3.880E-01	1.581E-01	6.297E-02	6.878E-01
SSE	1.630E-06	1.886E-05	8.050E-05	1.481E-04	4.979E-04	3.016E-03	2.477E-02	1.142E-01	5.147E-02	2.478E-02	2.190E-01
S	0.	2.514E-05	3.209E-04	2.468E-04	5.383E-04	1.467E-03	1.556E-02	1.690E-02	2.039E-02	1.959E-02	7.503E-02
SSW	5.853E-06	3.017E-05	8.264E-05	1.941E-04	1.544E-04	2.028E-03	1.051E-02	9.572E-03	7.017E-03	6.761E-03	3.636E-02
SW	0.	3.525E-05	3.661E-04	2.180E-04	1.616E-04	2.634E-03	1.149E-02	9.487E-03	4.088E-03	3.973E-03	3.246E-02
WSW	0.	7.027E-06	1.985E-04	1.196E-04	1.063E-04	1.227E-02	1.174E-02	7.966E-03	4.941E-03	9.284E-03	4.664E-02
W	0.	9.438E-06	5.440E-05	1.060E-04	1.011E-04	2.604E-02	9.729E-03	2.880E-03	5.499E-03	2.511E-02	6.953E-02
WNW	1.981E-06	1.153E-05	1.107E-05	4.765E-04	1.034E-03	2.153E-02	1.574E-02	1.770E-02	2.997E-02	4.807E-02	1.345E-01
NW	1.926E-04	1.927E-03	7.409E-04	1.362E-03	1.375E-03	7.903E-03	2.606E-02	4.952E-02	7.398E-02	6.763E-02	2.307E-01
NNW	2.942E-06	1.841E-04	8.901E-05	1.329E-04	4.300E-04	1.102E-02	2.096E-02	5.195E-02	8.809E-02	5.911E-02	2.320E-01
N	0.	1.288E-05	1.494E-04	8.462E-04	8.251E-04	1.273E-02	3.096E-02	8.829E-02	1.522E-01	7.549E-02	3.615E-01
Annulus Total	3.469E-04	3.132E-03	7.885E-03	7.662E-03	7.733E-03	2.929E-01	4.869E-01	1.268E+00	9.605E-01	5.377E-01	3.573E+00

TABLE 11B-5

PART B: (Continued)

EXTERNAL WHOLE BODY POPULATION RADIATION DOSES, BEAVER VALLEY POWER STATION, UNIT 1, DUQUESNE LIGHT CO./NUS CORP., 8/72

THE POPULATION DOSES PRESENTED ON THIS PAGE ARE THOSE ESTIMATED TO OCCUR IN THE YEAR 1990.

THE DOSES PRESENTED ARE IN UNITS OF MAN-REM.

COOLING TOWER RELEASE, BETA DOSE ONLY, 1.0% FAILED FUEL.

DIR.	0-1 MILES	1-2 MILES	2-3 MILES	3-4 MILES	4-5 MILES	5-10 MILES	10-20 MILES	20-30 MILES	30-40 MILES	40-50 MILES	DIRECTION TOTAL
NNE	0.	1.536E-03	1.506E-03	2.421E-03	3.595E-03	2.312E-01	3.090E-01	3.349E-01	5.652E-01	2.749E-01	1.724E+00
NE	6.714E-04	8.303E-04	3.448E-02	9.969E-03	4.039E-03	3.373E-01	3.529E-01	1.192E-01	1.767E-01	7.802E-02	1.114E+00
ENE	3.132E-04	3.357E-04	1.189E-03	2.289E-03	2.161E-03	3.142E-01	3.445E-01	4.539E-01	3.895E-01	1.265E-01	1.635E+00
E	3.571E-05	3.816E-04	4.558E-03	7.689E-03	2.560E-03	4.008E-01	4.601E-01	1.006E+00	7.750E-01	2.337E-01	2.890E+00
ESE	1.873E-04	2.540E-03	2.598E-03	7.795E-03	8.141E-03	3.192E-01	6.313E-01	2.551E+00	1.275E+00	4.638E-01	5.262E+00
SE	2.973E-05	1.965E-03	6.176E-03	3.080E-03	1.390E-03	7.421E-02	6.001E-01	3.383E+00	1.378E+00	5.492E-01	5.998E+00
SSE	1.421E-05	1.644E-04	7.020E-04	1.292E-03	4.342E-03	2.630E-02	2.160E-01	9.959E-01	4.489E-01	2.161E-01	1.910E+00
S	0.	2.192E-04	2.799E-03	2.152E-03	4.694E-03	1.279E-02	1.356E-01	1.474E-01	1.778E-01	1.708E-01	6.543E-01
SSW	5.104E-05	2.631E-04	7.207E-04	1.693E-03	1.346E-03	1.768E-02	9.167E-02	8.347E-02	6.119E-02	5.895E-02	3.170E-01
SW	0.	3.074E-04	3.192E-03	1.901E-03	1.409E-03	2.297E-02	1.002E-01	8.273E-02	3.564E-02	3.465E-02	2.830E-01
WSW	0.	6.128E-05	1.731E-03	1.043E-03	9.271E-04	1.070E-01	1.024E-01	6.947E-02	4.309E-02	8.096E-02	4.067E-01
W	0.	8.230E-05	4.744E-04	9.241E-04	8.816E-04	2.271E-01	8.484E-02	2.511E-02	4.796E-02	2.189E-01	6.063E-01
WNW	1.728E-05	1.005E-04	9.652E-05	4.155E-03	9.017E-03	1.878E-01	1.372E-01	1.544E-01	2.614E-01	4.192E-01	1.173E+00
NW	1.679E-03	1.681E-02	6.460E-03	1.187E-02	1.199E-02	6.892E-02	2.272E-01	4.318E-01	6.451E-01	5.898E-01	2.012E+00
NNW	2.565E-05	1.606E-03	7.762E-04	1.159E-03	3.750E-03	9.614E-02	1.828E-01	4.530E-01	7.682E-01	5.154E-01	2.023E+00
N	0.	1.123E-04	1.303E-03	7.379E-03	7.195E-03	1.110E-01	2.700E-01	7.699E-01	1.327E+00	6.583E-01	3.152E+00
Annulus Total	3.025E-03	2.731E-02	6.876E-02	6.681E-02	6.743E-02	2.554E+00	4.246+00	1.106E+01	8.376E+00	4.689E+00	3.116E+01

TABLE 11B-5

PART C: GAMMA EXPOSURE FROM THE ELEVATED RELEASES

EXTERNAL WHOLE BODY POPULATION RADIATION DOSES, BEAVER VALLEY POWER STATION, UNIT 1, DUQUESNE LIGHT CO./NUS CORP., 8/72

THE POPULATION DOSES PRESENTED ON THIS PAGE ARE THOSE ESTIMATED TO OCCUR IN THE YEAR 1990.

THE DOSES PRESENTED ARE IN UNITS OF MAN-REM.

COOLING TOWER RELEASE, GAMMA DOSE ONLY, 0.25% FAILED FUEL.

DIR.	0-1 MILES	1-2 MILES	2-3 MILES	3-4 MILES	4-5 MILES	5-10 MILES	10-20 MILES	20-30 MILES	30-40 MILES	40-50 MILES	DIRECTION TOTAL
NNE	0.	3.920E-04	1.363E-04	8.360E-05	6.630E-05	1.540E-03	7.776E-04	3.357E-04	3.380E-04	1.122E-04	3.782E-03
NE	7.400E-04	2.448E-04	3.560E-04	4.794E-04	1.125E-04	3.760E-03	1.380E-03	1.870E-04	1.587E-04	4.560E-05	7.464E-03
ENE	5.555E-04	1.824E-04	1.773E-04	1.296E-04	6.265E-05	3.234E-03	1.162E-03	6.090E-04	3.055E-04	7.100E-05	6.489E-03
E	5.400E-05	1.240E-04	5.047E-04	3.633E-04	6.794E-05	4.275E-03	1.900E-03	1.792E-03	8.120E-04	1.696E-04	1.006E-02
ESE	2.254E-04	5.520E-04	2.152E-04	3.102E-04	2.085E-04	3.741E-03	3.040E-03	5.632E-03	1.739E-03	4.080E-04	1.607E-02
SE	2.550E-05	2.990E-04	4.278E-04	1.250E-04	3.922E-05	1.054E-03	3.990E-03	1.078E-02	2.772E-03	8.307E-04	2.034E-02
SSE	2.800E-05	4.560E-05	9.052E-05	9.600E-05	2.295E-04	7.800E-04	3.003E-03	6.084E-03	1.873E-03	7.035E-04	1.293E-02
S	0.	5.060E-05	3.160E-04	1.410E-04	2.129E-04	3.075E-04	1.500E-03	7.955E-04	6.375E-04	5.005E-04	4.462E-03
SSW	7.020E-05	7.200E-05	1.151E-04	1.739E-04	1.916E-04	8.284E-04	2.232E-03	1.283E-03	7.032E-04	5.012E-04	6.081E-03
SW	0.	1.113E-04	4.962E-04	1.809E-04	9.976E-05	9.425E-04	2.303E-03	9.900E-04	3.150E-04	2.600E-04	5.699E-03
WSW	0.	1.710E-05	1.938E-04	6.360E-05	3.872E-05	2.200E-03	9.688E-04	2.820E-04	1.088E-04	1.539E-04	4.027E-03
W	0.	1.820E-05	3.445E-05	3.160E-05	1.782E-05	1.870E-03	3.300E-04	4.700E-05	5.460E-05	1.650E-04	2.569E-03
WNW	5.280E-05	2.880E-05	6.732E-06	1.054E-04	1.144E-04	8.883E-04	2.700E-04	1.557E-04	1.701E-04	1.837E-04	1.976E-03
NW	6.864E-04	5.376E-03	5.066E-04	3.200E-04	1.609E-04	3.135E-04	4.550E-04	4.440E-04	4.312E-04	2.728E-04	8.966E-03
NNW	5.940E-05	3.480E-04	6.373E-05	3.520E-05	5.832E-05	5.580E-04	4.440E-04	5.256E-04	5.726E-04	2.479E-04	2.913E-03
N	0.	3.080E-05	9.102E-05	1.941E-04	1.059E-04	6.450E-04	6.720E-04	8.625E-04	8.070E-04	2.988E-04	3.707E-03
Annulus Total	2.497E-03	7.893E-03	3.731E-03	2.833E-03	1.697E-03	2.694E-02	2.443E-02	3.081E-02	1.180E-02	4.924E-03	1.175E-01

TABLE 11B-5

PART C: (Continued)

EXTERNAL WHOLE BODY POPULATION RADIATION DOSES, BEAVER VALLEY POWER STATION, UNIT 1, DUQUESNE LIGHT CO./NUS CORP., 8/72

THE POPULATION DOSES PRESENTED ON THIS PAGE ARE THOSE ESTIMATED TO OCCUR IN THE YEAR 1990.

THE DOSES PRESENTED ARE IN UNITS OF MAN-REM.

COOLING TOWER RELEASE, GAMMA DOSE ONLY, 1.0% FAILED FUEL.

DIR.	0-1 MILES	1-2 MILES	2-3 MILES	3-4 MILES	4-5 MILES	5-10 MILES	10-20 MILES	20-30 MILES	30-40 MILES	40-50 MILES	DIRECTION TOTAL
NNE	0.	1.011E-02	3.456E-03	2.200E-03	1.721E-03	4.088E-02	1.852E-02	1.813E-02	9.802E-03	1.321E-03	1.061E-01
NE	1.909E-02	5.998E-03	9.040E-03	1.241E-02	2.960E-03	1.018E-01	3.484E-02	1.035E-02	4.036E-03	4.598E-04	2.010E-01
ENE	1.444E-02	4.598E-03	4.511E-03	3.384E-03	1.686E-03	8.732E-02	2.821E-02	3.277E-02	7.194E-03	7.277E-04	1.848E-01
E	1.370E-03	3.131E-03	1.285E-02	9.376E-03	1.770E-03	1.127E-01	4.636E-02	9.523E-02	2.072E-02	1.770E-03	3.053E-01
ESE	5.658E-03	1.409E-02	5.477E-03	8.249E-03	5.395E-03	9.752E-02	7.600E-02	2.970E-01	4.366E-02	4.471E-03	5.575E-01
SE	6.600E-04	7.279E-03	1.093E-02	3.225E-03	1.002E-03	2.830E-02	9.702E-02	5.390E-01	6.583E-02	8.705E-03	7.620E-01
SSE	7.300E-04	1.180E-03	2.319E-03	2.554E-03	6.140E-03	2.000E-02	6.716E-02	3.001E-01	3.987E-02	7.354E-03	4.474E-01
S	0.	1.258E-03	8.129E-03	3.680E-03	5.515E-03	8.100E-03	3.712E-02	3.956E-02	1.481E-02	5.369E-03	1.235E-01
SSW	1.911E-03	1.924E-03	2.953E-03	4.513E-03	2.649E-03	2.169E-02	6.287E-02	6.061E-02	1.620E-02	5.585E-03	1.809E-01
SW	0.	2.729E-03	1.283E-02	4.808E-03	2.597E-03	2.525E-02	5.194E-02	4.323E-02	6.237E-03	2.782E-03	1.524E-01
WSW	0.	4.149E-04	4.947E-03	1.675E-03	9.944E-04	5.830E-02	2.318E-02	1.545E-02	2.856E-03	1.648E-03	1.095E-01
W	0.	4.303E-04	8.692E-04	8.137E-04	4.679E-04	5.270E-02	8.280E-03	2.524E-03	1.703E-03	2.145E-03	6.993E-02
WNW	1.326E-03	6.912E-04	1.680E-04	2.706E-03	3.578E-03	2.400E-02	7.000E-03	8.114E-03	5.395E-03	2.705E-03	5.519E-02
NW	1.700E-02	1.317E-01	1.260E-02	8.000E-03	4.160E-03	8.920E-03	1.169E-02	2.246E-02	1.332E-02	4.057E-03	2.339E-01
NNW	1.483E-03	8.700E-03	1.581E-03	9.416E-04	1.624E-03	1.586E-02	1.156E-02	2.808E-02	1.759E-02	3.634E-03	9.104E-02
N	0.	7.656E-04	2.275E-03	5.106E-03	2.764E-03	1.733E-02	1.638E-02	4.514E-02	2.892E-02	4.341E-03	1.230E-01
Annulus Total	6.367E-02	1.950E-01	9.494E-02	7.364E-02	4.442E-02	7.207E-01	5.981E-01	1.558E+00	2.982E-01	5.707E-02	3.703E+00

TABLE 11B-5

PART D

POPULATION EXPOSURE FROM ALL GASEOUS RELEASES,
0.25% Failed Fuel Cladding

Direction	Auxiliary Building Releases man-rem/yr	Elevated Releases Beta Component man-rem/yr	Elevated Releases Gamma Component man-rem/yr	Total From all Sources man-rem/yr
N	1.684-1*	3.615-1	3.707-3	5.34-1
NNE	1.970-1	1.977-1	3.782-3	3.98-1
NE	1.504-1	1.277-1	7.464-3	2.86-1
ENE	6.610-2	1.875-1	6.489-3	2.60-1
E	7.002-2	3.315-1	1.006-2	4.12-1
ESE	9.609-2	6.034-1	1.607-2	7.16-1
SE	9.141-2	6.878-1	2.034-2	8.00-1
SSE	3.227-2	2.190-1	1.293-2	2.64-1
S	1.162-2	7.503-2	4.462-3	9.11-2
SSW	5.905-3	3.636-2	6.081-3	4.83-2
SW	5.881-3	3.246-2	5.699-3	4.40-2
WSW	9.377-3	4.664-2	4.027-3	6.00-2
W	1.600-2	6.953-2	2.569-3	8.81-2
WNW	3.733-2	1.345-1	1.976-3	1.74-1
NW	1.689-1	2.307-1	8.966-3	4.09-1
NNW	7.901-2	2.320-1	2.913-3	3.14-1
Totals	1.206	3.573	1.175-1	4.90

NOTE: The total annual population noble gas exposure is computed to be 4.90 man-rem within 50 miles of BVPS for 0.25% failed fuel.

* 1.684-1 = 1.684 x 10^{-1}, typical for all numeric values.

TABLE 11B-5

PART D, CONTINUED

POPULATION EXPOSURE FROM ALL GASEOUS RELEASES,
1.0% Failed Fuel Cladding

Direction	Auxiliary Building Releases man-rem/yr	Elevated Releases Beta Component man-rem/yr	Elevated Releases Gamma Component man-rem/yr	Total From all Sources man-rem/yr
N	3.332-1*	3.152	1.230-1	3.61
NNE	3.898-1	1.724	1.061-1	2.22
NE	2.976-1	1.114	2.010-1	1.61
ENE	1.308-1	1.635	1.848-1	1.95
E	1.385-1	2.890	3.053-1	3.33
ESE	1.901-1	5.262	5.575-1	6.01
SE	1.808-1	5.998	7.620-1	6.94
SSE	6.383-2	1.910	4.474-1	2.42
S	2.298-2	6.543-1	1.235-1	8.01-1
SSW	1.168-2	3.170-1	1.809-1	5.10-1
SW	1.163-2	2.830-1	1.524-1	4.47-1
WSW	1.855-2	4.067-1	1.095-1	5.35-1
W	3.165-2	6.063-1	6.993-2	7.08-1
WNW	7.384-2	1.173	5.519-2	1.30
NW	3.341-1	2.012	2.339-1	2.58
NWN	1.563-1	2.023	9.104-2	2.27
Totals	2.385	3.116+1	3.703	3.73+1

NOTE: The total annual population noble gas exposure is computed to be 3.73 man-rem within 50 miles of BVPS for 1.0% failed fuel.

* 3.332-1 = 3.332 x 10^{-1}, typical for all numeric values.

1. Auxiliary building releases, man-rem per yr.
   1.21 (2.39)

2. Elevated releases, man-rem per yr.
   β only: 3.57 (31.16)

3. Elevated releases, man-rem per yr.
   γ only: 0.12 (3.70)

The sector average dose rate was taken to be the dose rate of the geometric mid point of the sector. For auxiliary building releases and the beta dose from the elevated releases, the ICRP infinite hemisphere model was used. WINDOR results were used for the gamma dose from the elevated release. Since the separation of these release points is small with respect to the distances at which the bulk of the population exposure occurs, all sources were assumed to originate at a common location except with respect to height.

It is notable that over 25% (6%) of the population exposure is due to the auxiliary building releases which constitute only about 6.3% (1.4%) of the total annual activity release. Thus, a curie of activity released at ground level is more dose-effective than a curie released from the cooling tower by about a factor of four. In terms of maximum individual exposure at the site boundary, this factor is even higher.

C. Internal Exposure From Gaseous Effluents

Released radiogases (or their radioactive daughter products) must be either inhaled or ingested in order to yield internal radiation exposure. Ingestion requires the physical transport of the radioactive gases through some form of food chain. This is possible for the isotopes of iodine and for the noble gas particulate daughter products. Inhalation is a significant pathway only for radioiodine.

1. Internal Exposure from Released Noble Gases

Since the noble gases do not react chemically with other substances, there is no physical basis for either food chain transport or reconcentration within the human body.

In terms of continued inhalation and absorption in the body, both krypton and xenon may develop in physical solution, chiefly in the body water and fat.[12] Several human exposure experiments revealed that inhalation of relatively large amounts of radioactive noble gases resulted in very low tissue exposures.[13,14] In general, it may be estimated that the internal dose from radioactive noble gases dissolved in body tissue following inhalation from a cloud is negligible, i.e., less than 1% of the associated external whole body dose.[11] The resultant doses from exposure to noble gases, therefore, are considered to be external whole body doses only.

In the consideration of possible food chain transport of noble isotope is Cs-135, formed by the decay of Xe-135. The decay chain of interest is:[15]

$$\text{Xe}-135 \xrightarrow[\text{hour}]{9.2} \text{Cs}-135 \xrightarrow[\text{year}]{2\text{x}10^6} \text{Ba}-135 \qquad \text{(11B-1)}$$

The amount of Xe-135 estimated to be released from Unit 1 of the Beaver Valley Power Station, as shown in Table 11B-1, is relatively small, 23.1 (600) Ci/year. Assuming that all of this activity decayed immediately to Cs-135, the equivalent Cs-135 release rate is 1.21×10^{-8} (3.15×10^{-7}) Ci/year.

This maximum annual production rate can be compared to the maximum permissible body burden for individuals in the general population by assuming that the permissible body burden for members of the general population is one-tenth of that for occupational exposure.[9] The maximum permissible total body burden, for occupational exposure, for Cs-135 is 300 micro-curies. Thus the total body burden limit for members of the general population is 30 micro-curies. At the projected rate of normal gaseous waste release, Beaver Valley Unit 1 will produce one body burden of Cs-135 about every 2,500 (95) years. Therefore, this exposure pathway is not felt to be a significant hazard either to any individual, or the population as a whole.

2. Internal Exposure from Released Radioiodine

A small amount of radioactive iodine in addition to the noble gases will be released with the gases from the Beaver Valley Power Station. Iodine is an insignificant contributor to the external whole body dose but may produce potentially significant internal doses due to the preferential concentration of iodine in the human thyroid gland. Iodine may enter the body either through inhalation or by ingestion. The most critical pathway for environmental transport of the routine release of radioiodine is the pasture-cow-milk-man pathway.

a. Iodine Inhalation Thyroid Dose

As can be seen in Table 11B-1, the total annual release of all radioactive isotopes of iodine from BVPS Unit 1 is estimated to be much less than one curie per year. The annual iodine induced thyroid exposure resulting from continuous residence of the nearest site boundary (2,160 feet NE of Unit 1) has been calculated to be 0.049 (0.124) millirem.

The largest portion of this exposure is due to the single isotope I-131. This reflects its relative abundance in comparison to the other released isotopes of iodine and its greater dose effectiveness, which is best illustrated by the following figures:

Iodine Isotopes	Rem per curie Inhaled[16]
131	1.48×10^6
132	5.35×10^4
133	4.00×10^5
134	2.50×10^4
135	1.24×10^5

The greater dose-effectiveness of I-131 is primarily because of its relatively long 8.05 day half-life.

b. Iodine Ingestion Thyroid Dose

Although the radioiodine released from Beaver Valley will be in gaseous form initially, it may be deposited on pasture which is ingested by grazing cows. The cows may then transfer the radioiodine to milk which is consumed as human foodstuff.

Various mathematical models have been devised to estimate the dose to the thyroid through this route. In all cases the exposure is inversely proportional to the mass of the thyroid gland. The most sensitive receptor in the population, in terms of total thyroid dose per unit intake, is thus a young child or infant who would have a very small thyroid. Also, the relative radiosensitivity of the thyroid decreases markedly with age. Since the rate of milk ingestion is important in determining the dose, the most critical receptor is not a newborn infant, but is more likely to be a child six months to one year in age.

The site nearest to BVPS at which milk was known to be produced from grazing cows was Meyer's dairy farm located about 1.5 miles to the southwest of the station. Meyer's Dairy ceased production of milk in 1977. The analysis of exposures at Meyer's Dairy is retained in the Updated FSAR since it represents a conservative analysis when more distant dairies are considered.

The meteorological conditions are computed from the WINDVANE output to be as follows:

Release Point	χ/Q at Meyer's Dairy, sec/m^3
Primary Auxiliary Building vent	3.480×10^{-7}
Elevated Release Point	5.579×10^{-9}

These values of χ/Q and the dose conversion parameters recommended by the ICRP[9] and the Federal Radiation Council[15] were used to calculate the maximum potential dose from this source. Conservatively assuming that the cows graze on grass the entire year and that a child may drink an average of one liter of milk per day, the resulting annual thyroid exposure is 0.145 (0.423) millirem per year.

The effect of delay between the production and consumption of the milk was not included nor was the effect of possible dilution by other milk supplies. The child thyroid dose model is shown in detail in Appendix A.

IV. RADIATION EXPOSURE FROM LIQUID EFFLUENTS

This section of the report is directed at an evaluation of the potential radiation exposure occurring as a result of the normal release of radioactive liquid waste from Unit 1 of the Beaver Valley Power Station. Liquid radwaste will be discharged via injection into the Unit 1 cooling tower blowdown flow of 22,500 gpm. This will result in substantial initial dilution which is fully accounted for in the following analysis. Further dilution will occur in the Ohio River, into which the cooling tower discharge flows. For the purpose of calculating a maximum potential dose through the various pertinent pathways no credit is taken for river dilution. These doses are calculated by assuming an individual to drink, take fish from and swim in waters containing annual average discharge concentrations of effluent radionuclides.

A. Estimated Liquid Effluents

Table 11B-6, "Estimated Liquid Effluents and Concentrations", presents the pertinent data from which doses presented in this section have been calculated. The entire table is based on data supplied by Stone and Webster Engineering Corporation[6] which appears in the first two columns, "Curies Per Year Discharged". The sources of plant waste accounted for by Stone and Webster are as follows:[6]

1. Contaminated laundry and showers
2. Sampling sinks
3. Resin transfer system
4. Laboratory wastes
5. Reactor coolant system leaks and drains.

The remaining columns present the annual average concentrations assumed to exist in the Ohio River at point of discharge, opposite the city of Midland and at all points further downstream than Midland. The annual average discharge concentrations are calculated using the Unit 1 cooling tower blowdown flow of 22,500 gpm.

TABLE 11B-6

ESTIMATED LIQUID EFFLUENTS AND CONCENTRATIONS

	CURIES PER YEAR Discharged(6)		ANNUAL AVERAGE RADIOACTIVITY CONCENTRATIONS, μCi/cc					
			At Discharge		At Midland		Below Midland	
Isotope	0.25% ff	1.0% ff	0.25% ff	1.0% ff	0.25% ff	1.0% ff	0.25% ff	1.0% ff
H-3	5.06+2*	5.06+2	1.13-5	1.13-5	1.13-6	1.13-6	1.89-8	1.89-8
Cr-51	4.55-6	7.44-6	1.02-13	1.66-13	1.02-14	1.66-14	1.70-16	2.78-16
Mn-54	3.79-6	6.28-6	8.47-14	1.40-13	8.47-15	1.40-14	1.41-16	2.34-16
Mn-56	4.04-5	7.52-5	9.03-13	1.68-12	9.03-14	1.68-13	1.51-15	2.81-15
Co-58	1.22-4	2.00-4	2.73-12	4.47-12	2.73-13	4.47-13	4.55-15	7.47-15
Co-60	3.71-6	6.07-6	8.29-14	1.36-13	8.29-15	1.36-14	1.38-16	2.27-16
Fe-59	5.06-6	8.28-6	1.13-13	1.85-13	1.13-14	1.85-14	1.89-16	3.09-16
Sr-89	5.00-6	3.28-5	1.12-13	7.33-13	1.12-14	7.33-14	1.87-16	1.22-15
Sr-90	1.20-7	7.83-7	2.68-15	1.75-14	2.68-16	1.75-15	4.48-18	2.92-17
Sr-91	1.38-6	8.53-6	3.08-14	1.91-13	3.08-15	1.91-14	5.15-17	3.18-16
Y-90	1.32-7	8.61-7	2.95-15	1.92-14	2.95-16	1.92-15	4.93-18	3.21-17
Y-91	8.00-7	5.26-6	1.79-14	1.18-13	1.79-15	1.18-14	2.99-17	1.96-16
Y-92	2.94-7	2.07-6	6.57-15	4.62-14	6.57-16	4.62-15	1.10-17	7.73-17
Zr-95	8.21-7	5.37-6	1.83-14	1.20-13	1.83-15	1.20-14	3.06-17	2.00-16
Nb-95	8.37-7	5.49-6	1.87-14	1.23-13	1.87-15	1.23-14	3.12-17	2.05-16
Mo-99	3.58-3	2.21-2	8.00-11	4.94-10	8.00-12	4.49-11	1.34-13	8.25-13
I-131	2.99-3	1.91-2	6.68-11	4.27-10	6.68-12	4.27-11	1.12-13	7.13-13
I-132	2.89-4	2.20-3	6.46-12	4.91-11	6.46-13	4.91-12	1.08-14	8.21-14
I-133	3.71-3	2.24-2	8.29-11	5.00-10	8.29-12	5.00-11	1.38-13	8.36-13
I-134	1.55-4	1.24-3	3.46-12	2.77-11	3.46-13	2.77-12	5.79-15	4.63-14
I-135	1.28-3	8.20-3	2.86-11	1.83-10	2.86-12	1.83-11	4.78-14	3.06-13
Te-132	2.91-4	1.80-3	6.50-12	4.02-11	6.50-13	4.02-12	1.09-14	6.72-14
Cs-137	1.56-3	1.02-2	3.49-11	2.28-10	3.49-12	2.28-11	5.82-14	3.81-13
Ba-140	5.02-6	3.25-5	1.12-13	7.26-13	1.12-14	7.26-14	1.87-16	1.21-15
La-140	1.60-6	1.20-5	3.57-14	2.68-13	3.57-15	2.68-14	5.97-17	4.48-16
Ce-144	4.35-7	2.85-6	9.72-15	6.37-14	9.72-16	6.37-15	1.62-17	1.06-16
Total	5.06+2	5.06+2	1.13-5	1.13-5	1.13-6	1.13-6	1.89-8	1.89-8
Total Non-tritium	1.41-2	8.77-2	3.14-10	1.96-9	3.14-11	1.96-10	5.25-13	3.27-12

* 5.06 + 2 = 5.06 x 10^2; typical for all values.

To guarantee compliance with all state and federal regulations concerning thermal discharges, Duquesne Light Company has commissioned Alden Research Laboratories to perform various analyses using scale models of the Ohio River.[17,18,19] These tests indicate extremely rapid dilution of the plant effluents upon discharge to the river. The results of the most appropriate analyses performed are presented graphically in Figure 11B-3. The drawing presents the average temperature increase, at a depth of one foot, 3,000 feet downstream of Phillis Island. This location, which is several thousand feet upstream of the municipal water intakes for the city of Midland, is shown to experience only about a one degree fahrenheit temperature rise due to the combined thermal discharges of Beaver Valley Units 1 and 2 and the Shippingport Power Station. The assumed discharge from Beaver Valley is 20°F above ambient at 36 cubic feet per second for each unit. This temperature rise must be assumed to be primarily due to the thermal discharge from the Shippingport reactor which is about 14.5°F above ambient at 254 cfs. Thus, in this instance, Beaver Valley is assumed to contribute only about 29% of the total heat flow to the river in this situation.

Assuming, however, that all of the heat rise at the specified location is due to Beaver Valley and that the I F temperature rise would be doubled were it not for heat loss to the atmosphere, a dilution factor of about ten is conservatively obtained. Since this point is well upstream from the water intakes at Midland, the concentrations of all radionuclides in the river opposite Midland City are assumed to be a factor of ten less than those at the point of discharge.

Since over six miles must be traversed from Midland downstream to the next known potable water intakes at East Liverpool, Ohio and Chester, West Virginia, full mixing in the river is assumed to occur prior to intake downstream from Midland. The river dilution is based on the annual average Ohio River flow rate at the Beaver Valley site which is about 30,000 cfs.[1]

B. Maximum Individual Radiation Exposure from Liquid Effluents

In the region of the Beaver Valley site, and downstream from there, the Ohio River contains acid mine drainage as well as substantial amounts of industrial and municipal waste water. In the immediate vicinity of the site (at the point of discharge) it is not expected that any individual will consume as potable water the untreated cooling tower blowdown either before or after discharge to the Ohio River. Nor is it expected that any individual will swim in these same waters, although this is very much more likely than the water ingestion mentioned above. However, for the purpose of establishing the maximum potential individual radiation exposure due to the normal operation of Unit 1 of the Beaver Valley Power Station, a "maximum individual" is hypothesized who does the following:

1. The maximum individual is assumed to consume 1.2 liters daily of the Unit 1 cooling tower blowdown flow. This water ingestion represents the total water intake that will adequately sustain an average adult male, as determined by the ICRP;[(9)]

2. The maximum individual is assumed to consume 50 grams daily of fish grown since birth in waters containing annual average discharge concentrations of all radionuclides; and

3. The maximum individual is assumed to swim 200 hours per year in waters containing the annual average discharge concentrations of all radionuclides.

Other less significant exposure pathways are also considered.

1. Internal Exposure from Water Ingestion

Although it is extremely unlikely that any individual will ever ingest water from the cooling tower blowdown, maximum individual exposure is calculated on this basis and presented as an upper limit to the potential drinking water ingestion dose. The assumed intake rate of 1.2 liters per day represents the entire daily drinking water requirement of a "standard man", as recommended by the ICRP.[(9)]

The ingestion of radionuclides will generally cause an uneven distribution of dose within the human body. Some elements, hydrogen in particular, become rather evenly distributed. Others, such as iodine or cesium, are preferentially taken up by certain body organs. This phenomena produces particular body organ doses which can be either higher or lower than the associated whole body dose.

The array of particular body organ doses due to the hypothesized water ingestion has been calculated using the NUS computer code DOSCAL. The results of this calculation are summarized in Table 11B-7, “Internal Body Organ Exposure from Ingestion of Water and Fish”. Calculational methods used in DOSCAL are those advocated by such bodies as ICRP,[9] the IAEA[20] (International Atomic Energy Agency) and the U.S. Atomic Energy Commission.

The highest dose from this route of exposure, to the maximum individual is calculated to be 1.0 (1.51) mrem per year to the thyroid. Of this total almost 93% (68%) is due to tritium. The remainder is almost entirely due to the five isotopes of iodine. The total annual body exposure from this route is calculated to be 0.63 mrem of which over 99% is due to tritium.

TABLE 11B-7

INTERNAL BODY ORGAN EXPOSURE FROM INGESTION OF WATER AND FISH

	Exposure From Water Ingestion, rem/year		Exposure From Fish Ingestion, rem/year	
Body Organ	0.25% ff	1.0% ff	0.25% ff	1.0% ff
Whole Body	6.27-4*	6.31-4	5.21-5	2.07-4
Body Water	1.02-3	1.03-3	6.73-5	2.22-4
Lungs	1.02-3	1.02-3	4.76-5	9.29-5
Liver	1.02-3	1.03-3	1.11-4	5.06-4
Spleen	1.02-3	1.03-3	9.58-5	4.09-4
Kidney	1.02-3	1.03-3	6.58-5	2.12-4
Bone	1.52-6	9.87-6	5.23-5	3.42-4
Muscle	1.02-3	1.03-3	9.07-5	3.75-4
Pancreas	1.02-3	1.03-3	6.73-5	2.22-4
Thyroid	1.10-3	1.51-3	7.06-5	2.43-4
Small Intestine	1.02-3	1.02-3	4.40-5	6.99-5
Upper Large Intestine	1.02-3	1.03-3	6.73-5	2.22-4
Lower Large Intestine	1.02-3	1.04-3	8.22-5	3.08-4

Notes about this table: Doses presented here are based on the undiluted annual average radioactivity concentration of the cooling tower discharge. Liquid exposure is based on an intake of 1.2 liters per day. Fish ingestion is postulated to be 50 grams per day. Other body organs for which doses were calculated but not found to substantially differ from those presented are the testes, ovaries, prostate, heart, brain, skin and stomach.

* 6.27 - 4 - 6.27 x 10^{-4}; typical for all values.

It should be noted that the nearest known potable water intake, from the point of discharge, is located at Midland about 1.5 miles downstream and on the opposite side of the river.[21] Concentrations of radionuclides at the water intakes of Midland City have been very conservatively estimated using a dilution factor of only 10. Thus, the actual expected doses from water ingestion are at least a factor of 10 lower than the calculated maximum potential doses. Also, it is known that standard water treatment facilities have the capacity to reduce radioactive content in water prior to distribution to the public. No credit is assumed either for this or for radioactive decay in transit.

2. Internal Exposure from Fish Ingestion

A number of factors are required to compute the internal doses from ingestion of fish which might conceivably contain radioactivity from the plant liquid effluents. In addition to the activity concentrations in the waters of interest, the concentration of the activity in the organism and the amount consumed must be established.

Aquatic organisms, through biological processes, have the ability to concentrate radionuclides released from the plant. This concentration of activity in aquatic organisms, which is in turn ingested by man, must be considered in determining the possible dose to man. The ratio of the concentration of a radionuclide in an aquatic organism to that in the ambient water is known as the concentration factor (CF). The concentration factor varies among the different species of aquatic life and, for a given specie, varies with the different radionuclides. Also, the concentration may vary considerably between different organs of an organism. For the dose calculations in this report, appropriate concentration factors were used for the edible portions of the fish[22] and are shown in Table 11B-8.

TABLE 11B-8

CONCENTRATION FACTORS FOR EFFLUENT RADIONUCLIDES IN FISH

Element	Concentration Factor in Freshwater Fish*
H	0.926
Cr	200
Mn	25
Co	500
Fe	300
Sr	40
Y	100
Nb	30,000
Zr	100
Mo	100
I	1
Te	400+
Cs	1,000
Ba	10
La	100
Ce	100

* All concentration factors used in this report for freshwater organisms are those compiled by Chapman, Fisher and Pratt and presented in UCRL-50564, "Concentration Factors of Chemical Elements in Edible Aquatic Organisms".[22]

\+ Due to lack of published information regarding CF of tellurium in freshwater fish, the CF for tellurium was taken as equal to that of selenium due to the chemical similarity.

In order to determine the dose to humans, the quantity of fish eaten must be estimated. The dose model used postulates that the maximum individual consumes 50 grams of fish flesh every day. This is about equal to the seafood consumption reported for commercial fisherman[23] and about four times the annual per capita consumption of seafood in the United States.[24]

The various maximum doses to an individual have been calculated by the NUS computer code DOSCAL and are presented in Table 11B-7. The ingested fish was assumed to have concentrations of all radionuclides in the edible flesh equal to the annual average discharge concentrations times the appropriate concentration factors, as presented in Table 11B-8. The highest dose so derived amounts to 0.11 (0.51) mrem per year to the liver. The liver dose is 64% (91%) due to the single effluent isotope Cs-137. The thyroid exposure is less than the liver exposure, amounting to 0.07 (2.24) mrem per year. The whole body exposure is calculated to total 0.05 (0.21) mrem per year, primarily due to Cs-137. These doses and the others calculated from this exposure pathway are presented in Table 11B-7.

Since various thermal analyses have demonstrated the quick dispersion of the heated discharges from Beaver Valley,[17,18,19] and since fish are highly mobile in general, it is estimated that the actual annual average concentrations to which the fish are exposed will be reduced by at least a factor of 10. Thus, the maximum expected individual doses from fish ingestion should be reduced by at least this factor of 10 in comparison to the maximum potential individual doses as described above.

TABLE 11B-9

EFFECTIVE SUBMERSION ENERGIES

Isotope	Effective Submersion Energy MeV	Data Source
H-3	6.00-3*	3
Cr-51	2.90-2	3
Mn-54	8.35-1	3
Mn-56	2.54	3
Co-58	1.163	1
Co-60	2.611	1
Fe-59	1.304	3
Sr-89	4.88-1	2
Sr-90	1.82-1	2
Sr-91	1.242	2
Y-90	9.300-1	2
Y-91	5.18-1	2
Y-92	1.367	2
Nb-95	8.18-1	2
Zr-95	8.61-1	2
Mo-99	5.390-1	1
I-131	5.84-1	2
I-132	2.711	2
I-133	1.039	2
I-134	3.066	2
I-135	1.932	2
Te-132	3.48-1	2
Cs-137	7.58-1	2
Ba-140	4.96-1	2
La-140	2.872	2
Ce-144	1.335	3

Data Sources

1. Yong S. Kim, "NEEP-3, NUS Effective Energy Program", NUS-884, NUS Corporation.

2. Meek and Gilbert, "Summary of Beta and Gamma Energy and Intensity Data", NEDO 12037.[10]

3. Lederer, Hollander and Perlman, Calculated from disintegration data presented in, "Table of Isotopes".

* 6.00-3 = 6.00 x 10^{-3}; typical for all values.

3. External Exposure from Swimming

The external exposure of an individual by submersion in waters containing the radioactive effluents of the Beaver Valley Power Station is presented only to insure the completeness of this report. Not only is such exposure extremely unlikely, but the direct irradiation of humans while swimming presents only a minor exposure pathway. The radionuclide that will be present in the cooling tower blowdown in the highest concentration is tritium which yields only weak beta radiation upon decay. The beta particles emitted from decaying tritium nuclides do not have sufficient energy to penetrate the human skin and, therefore, cannot contribute to a whole body dose. All other nuclides are assumed to irradiate the whole body.

The exposure to swimmers may be conservatively estimated by assuming that the swimmer is completely immersed in an infinite medium of uniform concentration and receives the same dose as the water itself. The expression for the dose rate is given for each radionuclide by:

$$D \frac{\text{rem}}{\text{hour}} = C_i \frac{\mu\text{Ci}}{\text{cc}_w} \quad 1.0 \frac{\text{cc}}{\text{gm}} \quad E \frac{\text{MeV}}{\text{dis}} \quad 3600 \frac{\text{sec}}{\text{hr}} \quad 3.7 \times 10^4 \frac{\text{dis}}{\text{sec}-\mu\text{Ci}}$$

$$1.6 \times 10^{-8} \frac{\text{gm}-\text{rad}}{\text{MeV}} \quad \frac{1\text{rem}}{1\text{rad}} \qquad \text{(11B-2)}$$

where C_i = the annual average discharge concentration of the ith radionuclide is given in Table 11B-6, and where E = the effective submersion energy per disintegration.

This relationship applies only to an infinite medium where the energy released per unit volume by the radionuclides is equal to the energy absorbed per unit volume of the medium. It is a conservative calculation and will tend to over-estimate the dose.

Application of the above method, and the assumption that the maximum individual might spend as much as 200 hours per year swimming in the discharge yields an annual skin dose from tritium of 0.03 mrem and an annual whole body dose due to all other radioisotopes of 0.0001 (0.0008) mrem.

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* For the purposes of this report, the effective submersion energies for external exposure were taken from several sources, including NUS-884, "NEEP-3, NUS Effective Energy Program", by Yong S. Kim. These energies are presented in Table 11B-9.

Since the Ohio River at or near the plant is not known to be used for the purpose of swimming on a regular basis, the expected maximum individual exposure from this source is essentially zero.

4. Radiation Exposure from Other Sources

Two remaining exposure pathways for liquid radwaste are discussed in this section. These pathways are of lesser significance than those previously discussed. Other exposure pathways found to be of negligible significance and therefore not discussed further are:

1. The uptake of river water in animal flesh which is subsequently ingested by man;
2. The uptake of river water in crops which are subsequently ingested by man; and
3. The seepage of river water into the ground water table from which water may be taken via wells.

These exposure pathways are ignored due to their relative insignificance in comparison to the pathways considered.

A potential external exposure pathway exists for persons sunbathing or walking along the river's edge. The radionuclides in the liquid wastes discharged via the cooling tower blowdown flow at the Beaver Valley Power Station can be expected to accumulate to some degree on the bottom sediments and the shoreline sand. The affected area would be limited to approximately the area of the beach between the low and high water marks.

The concentrations of the radionuclides on the shore are influenced by the dilution that occurs in the river, the chemical composition of the effluent as well as the aquatic environment and the sorptive capacity of the shoreline soil, which is usually expressed as a concentration factor between water and soil. The ability of soils to concentrate radioactive materials differs widely from one element to another. Radionuclides are removed from solution primarily by adsorption and ion exchange. Generally, the fine grained bottom sediments are most effective sorbers of radionuclides than are the coarser grained sands.

The largest discharges of radioactive liquids to the environment have been made from the Windscale site in England and at Hanford in the United States. Smaller discharges have been made from the Oak Ridge National Laboratory into the Clinch River and from the Savannah River Plant. At Windscale, where the principal radioactive effluents are fission products derived from the reprocessing of spent fuel elements, the discharges are made directly to the marine environment whereas the Hanford wastes are discharged to the Columbia River some several hundred miles from the Pacific Ocean. An extensive environmental monitoring program at Windscale has resulted in the development of concentration factors for a few radionuclides on shoreline sand.[25,26] The highest concentration factor for cesium in shoreline sand is 65, and ten times that figure for cesium in bottom silt. However, Saddington suggests a CF of 1,000 for cesium in bottom sediment,[26] and although this is not an experimental value, for conservatism, it is used here not only for cesium but for all other radionuclides, except tritium. Tritium, released in the chemical form of water, cannot concentrate in shoreline sand.

It was assumed that an individual lying on the shore would receive half the dose that he would receive if he were completely immersed in the sand. A calculation similar to the swimming dose calculation with sand as the infinite medium and with the concentration factors applied results in a maximum potential whole body dose of 3.05×10^{-4} (1.92×10^{-3}) mrem per hour for sunbathing along the river. Assuming an exposure period of 200 hours per year, the annual dose to a sunbather from the Beaver Valley Power Station would be 6.10×10^{-2} (3.85×10^{-1}) mrem. The significance of this dose can be kept in perspective by consideration of the extremely small probability of occurrence. Also, it is calculated on the basis of annual average discharge concentrations involving an overestimation of the actual dose (should one occur) by at least a factor of 10.

The remaining pathway to consider involves the transfer of activity ingested in river water by a cow to the secreted milk, which is then ingested by a man. In order to evaluate this route of exposure, it is assumed that the milking cattle obtain all of their drinking water from the cooling tower blowdown flow. It can be estimated that the water requirement (exclusive of milk) for a cow is approximately 50 milliliters per kilogram of body weight.[27] For a 1,000 pound animal this yields a water intake of about 22.7 liters/day. Considering the lactating cow to produce about 18.2 liters of milk per day and taking the milk to be about 87% water, the total water intake for a lactating animal is estimated to be approximately 38.5 liters/day.

Of the radionuclides present in the water only the alkaline earths, alkali metals and halogens need to be considered. The other elements are effectively discriminated against by the cow.[28] The concentration in the milk is estimated as the product of the concentration in water, times the cow's intake of water, times the coefficient of transfer. The coefficient of transfer is defined as the fraction of the daily intake secreted per liter of milk.[28] The values used are listed below:

Element	Coefficient of Transfer[28]
Sr	.001
I	.01
Cs	.025
Ba	.002

The concentration of tritium in the secreted milk is assumed to be 0.87 times the concentration in the water ingested by the cow. The highest transfer coefficient is that of Cs (0.25). Since a person may be assumed to ingest a maximum of one liter of milk daily, on the average, the daily activity intake of Cs in micro-curies will be given by:

$$C_w \quad Cs \frac{\mu Ci}{cc} \quad 38.5 \times 10^3 \frac{cc}{day} \quad 0.25 \frac{\mu Ci/l}{\mu Ci/day} 1.0\ell/day$$

$$= 962.5 \frac{cc}{day} \times C_w \frac{\mu Ci}{cc} \qquad (11B\text{-}3)$$

where $C_w \, Cs \frac{\mu Ci}{cc}$ is the concentration of cesium activity in the water ingested by the cow.

This can be compared to the daily cesium intake due to a 1.2 liter/day ingestion of the same water the cow drinks, which is given by:

$$1200 \frac{cc}{day} \times C_w (\mu Ci/cc)$$

The brief analysis demonstrates that the dose from the ingestion of milk from a cow drinking from the Ohio River flow is necessarily less than the dose from drinking the water directly. Because of this and because a one liter per day rate of milk ingestion should not be arbitrarily superimposed on the maximum individual liquid ingestion rate of 1.2 liters per day discussed earlier, the doses from this hypothetical pathway are not presented.

C. Population Radiation Exposure from Liquid Effluents

In regard to the radiological well being of the general public, the total integrated population exposure, in terms of man-rem per year, is the most pertinent measure of the impact of the liquid effluents projected to be released by Unit 1 of the Beaver Valley Power Station. Population exposure may accrue from all the maximum individual exposure pathways previously considered. However, only two of these pathways are considered significant in terms of population exposure from liquid effluents. Detailed consideration is given to the population exposure derived from potable water intakes and fish ingestion.

Sources of exposure not considered significant in terms of population exposure are:

1. Swimming in, sunbathing near or walking along the river

2. Ingestion of milk from cows drinking from the river

3. Other exposure pathways not considered in terms of maximum individual exposure.

The exclusion of these three categories of radiation exposure is primarily based on the polluted condition of the Ohio River which is not known to be extensively used for purposes other than municipal and industrial water supplies.[1]

1. Population Exposure from Water Ingestion

Table 11B-10 presents a list of the municipal water intakes now existing downstream of the site and within a 50 mile radius. The names and 1970 populations of the cities served by these intakes are included in the table as well as the dose rates estimated to occur at each one.

TABLE 11B-10

MUNICIPAL WATER INTAKES DOWNSTREAM OF BVPS AND WITHIN A FIFTY MILE RADIUS(21)

Distance Downstream Miles	City or Town	1970 Population	Applicable Dilution Factors	Annual Whole Body Exposure From Water Ingestion: Population Dose, man-rem 0.25% ff*	Population Dose, man-rem 1.0% ff	Per Capita Dose, mrem 0.25% ff	Per Capita Dose, mrem 1.0% ff
1.7	Midland, Pennsylvania	5,271	10	3.30-1**	3.33-1	6.27-2	6.31-2
8	East Liverpool, Ohio	20,020	598	2.10-2	2.11-2	1.05-3	1.06-3
8	Chester, West Virginia	3,614	598	3.79-3	3.81-3	1.05-3	1.06-3
23	Toronto, Ohio	7,705	598	8.08-3	8.13-3	1.05-3	1.06-3
32	Weirton, West Virginia	27,131	598	2.84-2	2.86-2	1.05-3	1.06-3
32	Steubenville, Ohio	30,771	598	3.23-2	3.25-2	1.05-3	1.06-3
36	Mingo Junction, Ohio	5,278	598	5.53-3	5.57-3	1.05-3	1.06-3
61	Wheeling, West Virginia	48,188	598	5.05-2	5.08-2	1.05-3	1.06-3
65	Bellaire, Ohio	9,655	598	1.01-2	1.02-2	1.05-3	1.06-3
	Totals	157,633		4.90-1	4.93-1		

* ff is used to abbreviate "failed fuel".
** 3.30-1 = 3.30 x 10^{-1}; typical of all values.

As previously described, the concentrations of the effluent radionuclides at the city of Midland are assumed to be one tenth of the annual average discharge concentrations. Assuming that the entire 1970 population of the city of Midland consumes 1.2 liters daily of water taken from the Ohio River, the whole body water ingestion population dose for the city of Midland is calculated to be 0.33 man-rem per year.* The per capita exposure at Midland from this source is estimated at 0.06 mrem per year. Not considered were the effects of radioactive decay in transit and the cleansing action of the water treatment facilities.

Since it is a full six miles downstream from Midland to the next municipal water intake, complete mixing in the river of the cooling tower discharge is assumed to occur prior to that point. This assumption is justified because if "streamlining" occurs, as it may during periods of high river flow, the effluent is expected to be preferentially swept down the central portions of the river path, thus creating lower than average concentrations to the sides of the river where the potable water intakes are located. Assuming full dilution, and using the annual average river flow of 30,000 cfs, a dilution factor of about 598 is calculated which is applicable to all potable water intakes listed in Table 11B-10 with the exception of those located at Midland.

The whole body population dose from water ingestion at the remaining known intakes (excluding Midland) is calculated to be 0.16 man-rem per year. The per capita exposure rate at these intakes is found to be 1.05×10^{-3} (1.06×10^{-3}) mrem annually.

The total population exposure from water ingestion is the total of that occurring at Midland and the other locations listed in Table 11B-10. The total is 0.49 man-rem per year, which is over an order of magnitude smaller than the population doses calculated to result from gaseous releases.

The basis for the liquid effluent dilution factors of 10 and 598 shown in Table 11B-10 is discussed below.

A dilution factor of 598 was used to obtain annual average radioactivity concentrations at the water intakes of the city of East Liverpool and at all other intakes further downstream. The

* Because the predominant population exposure from Beaver Valley is from external whole body radiation due to gaseous releases, and because the whole body dose is the "genetically significant" dose, only population exposure of the whole body is presented here.

factor 598 is the ratio of the annual average river flow at this location (about 30,000 cfs) and the BVPS blowdown discharge flow rate (22,500 gpm).

$$598 = \frac{30{,}000\ \mathrm{ft}^3/\mathrm{sec}\ \mathrm{x}\ 7.481\ \mathrm{gal}/\mathrm{ft}^3\ \mathrm{x}\ 60\ \mathrm{sec}/\mathrm{min}}{22{,}500\ \mathrm{gal}/\mathrm{min}} \qquad (11B\text{-}4)$$

The assumption of full dilution of the station discharge with the entire annual average river flow is first made at a location about 8 miles downstream from the plant. This assumption is justified because if "streamlining" occurs, as it may during periods of high river flow, the effluent is expected to be preferentially swept down the central portions of the river, creating lower than average concentrations to the sides of the river where the potable water intakes are located.

A dilution factor of 10 was used to obtain annual average radioactivity concentrations at the potable water intakes of Midland, about 1.7 miles downstream of the station. This small amount of dilution is justified by various analyses of the thermal plume resulting from the combined Beaver Valley and Shippingport thermal discharge[34][35][36]. Figure 11B-3 is a reproduction of a figure summarizing the results of thermal analyses conducted by the Alden Research Laboratories.[36] The figure is a graph of the average increase in temperature (at a depth of one foot), at a position 3,000 ft downstream of the end of Phillis Island, as a function of the flow and temperature of the BVPS blowdown discharge. Each data point on the graph includes the effects of the Shippingport discharge, assumed to be 14.5 F above ambient at a flow rate of 234 cfs.

The BVPS blowdown flow rate of 22,500 gpm is nearly equivalent to 50 cfs. Figure 11B-3 presents the results of analyses in which it was assumed that the BVPS discharge flow was equal to 72 cfs and was 20 F and 40 F above ambient. The resulting average temperature increases are about 2.7 F and 3.3 F for a minimal river flow of only 5,000 cfs. Since about 2.5 F of each increase is due to Shippingport (for a 5,000 cfs river flow), the actual contribution to the average temperature from BVPS amounts to only 0.2 F and 0.8 F, respectively. In each case, the average temperature increase is less than 10 percent of the initial temperature above ambient. Since loss of heat by evaporation is not significant during initial dilution, the data supports a dilution factor of 10 during a very small river flow of only 5,000 cfs. Since the actual annual average river flow is 30,000 cfs, and since Midland is substantially further downstream than the location considered in Figure 11B-3, a dilution factor of only 10 is probably very conservative.

It should be noted that when the average 30,000 cfs river flow is present, it is only necessary for the BVPS discharge to mix with about 3 percent of the river flow to achieve dilution by a factor of 10. It would seem highly unlikely that the discharge from BVPS could cross from one side of the river to the side on which the Midland intakes are located without mixing with at least 3 percent of the volume flow of the river.

2. Population Exposure from Fish Ingestion

At the present time the condition of the Ohio River in the vicinity of Beaver Valley is not conducive to a great deal of fishing.(1) The plant is situated on the New Cumberland Pool approximately 19.6 miles upstream from the New Cumberland Lock and Dam, and about 3.1 miles downstream from Montgomery Lock and Dam.(1) The New Cumberland Pool, which is not fished commercially, is estimated to yield only about 1,000 lb. per year of fish caught for sport.(1) The 1,000 pounds per year fish catch was obtained from a Pennsylvania fish warden who estimated the catch for the New Cumberland pool. Fish catch data are not readily available for the pool; therefore, the best estimate of fish caught was obtained from the district fish warden.

If it was assumed that this entire 1,000 lb. per year catch was grown in waters containing one tenth of the annual average discharge concentrations of all radionuclides, the whole body population exposure from fish ingestion from this pool can be calculated. The resulting dose is found to be 4.34×10^{-5} (1.72×10^{-4}) man-rem per year. It has been assumed that only one third of the annual catch is actually edible.(29)

The population exposure due to fish ingestion from the remainder of the river downstream of the plant and within a 50 mile radius has been estimated. Over the 22.7 miles of the New Cumberland Lock and Dam approximately 1,000 pounds of fish are caught per year. Extrapolating this catch rate over the as yet unaccounted for portion of the river within the 50 mile radius, and assuming full dilution of the radionuclide concentrations (dilution factor of 598) the remaining dose can be estimated, completing the calculation. The remaining fish ingestion population dose within 50 miles of the plant amounts to an estimated 1.35 x 10^{-6} (5.38 x 10^{-6}) man-rem per year.

Thus, the total population exposure per year from fish ingestion is calculated to be 4.48 x 10^{-5} (1.78 x 10^{-4}) man-rem, only about 0.01% (0.04%) of the population dose calculated to result from the ingestion of water. All population and individual doses are summarized in the final section of this report.

D. Radiation Exposure of Aquatic Bio-Systems

This section of the report is directed towards an evaluation of the radiological impact of liquid effluents upon the various life forms inhabiting the Ohio River. The long range effect of continuous small releases of radioactive materials from Beaver Valley will be an incremental increase in the normally occurring environmental levels. These increases will be most pronounced in situations where the radionuclides may tend to concentrate.

An organism inhabiting the river downstream from the plant will be subject to plant contributed radiation from three different sources. Internally, radiations from radioisotopes which may have concentrated within the organism will contribute to the total annual exposure. Externally, this organism will be affected by the radiations from the surrounding water body, and possibly also from the river bottom where large concentration factors (CF's) are sometimes the case.

The receptor organisms at risk from this exposure are various in size, feeding habits, habitual location and other characteristics important in the determination of dose. Calculations have shown that the maximum radiation exposure received by any aquatic organism will be that delivered to life forms inhabiting the river bottom where sediment concentrations of the released radioisotopes may be orders of magnitude above those in the ambient water.

In the immediate vicinity of the discharge, stationary biota may be subject to annual average concentrations in water of the released radioisotopes approaching in magnitude those projected for the discharge itself. This situation can only occur in a very small region of the river habitat owing to the rapid dilution afforded by the usually large river flow. Since, as previously mentioned, the concentrations of radionuclides are generally substantially higher in fine grained bottom sediments than in the coarser grained shoreline sands, a higher CF is assumed. The CF assumed for shoreline sands earlier in this report is 10^3. Hence, for bottom sediments, a factor of 10^4 is assumed for all radioisotopes except tritium (released in the chemical form of water, prohibiting concentration). This increase is admittedly arbitrary and reflects not only the desire for conservatism in these calculations, but the existing lack of adequate information on this subject. However, the maximum potential external exposure rate for an organism completely imbedded in the bottom sediments under the point of discharge is calculated (using a CF in sand of 10^4) to be about 53 (337) mrad per year.*

Organisms which may be subjected to this exposure rate are those which burrow into or take deep root in bottom sediments. As the depth in the sediment decreases this external dose rate will also decrease becoming approximately 27 (169) mrad per year at the sediment surface (river bottom). The reduction by a factor of two is entirely due to geometry, since the radiation from the sediments is now only from half the total solid angle.

In the immediate area of the plant, and in the remainder of the New Cumberland Pool, benthic conditions have been found to be poor due to previous and current pollution.[(1)] Examples of benthic organisms known to exist in this pool are midge flies, flatworms, snails and other varieties of aquatic worms. These organisms and deep roots of plants will be subject to an external exposure rate from sand estimated to be in the range of 53 (337) to 27 (169) mrad per year, representing the variation from the dose very deep in contaminated sediments (tens of inches) to the dose rate estimated to occur at the sediment surface.

At the sediment surface, the dose from the surrounding water mass amounts to about 2.67×10^3 (1.69×10^{-2}) mrad per year considering all isotopes except tritium. The dose from tritium, which emits only weak beta radiation, will be only a "skin" dose for all organisms except plankton. For plankton the additional dose due to tritium is 0.63 mrad per year. These doses will apply to biota commonly habitating the river bottom such as mosses, algae, ferns, bottom feeding fish, etc. For free swimming fish not generally found at the river bottom, and for plankton at locations several feet or more from the bottom, the only external exposure will come from the surrounding water mass, and not from the sediment. In this case, the external dose rate for fish, at annual average discharge concentrations, amounts to 5.34×10^{-3} (3.37×10^{-2}) mrad per year. The dose is 1.27 (1.30) mrad per year for plankton.

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* This dose and all other external doses to aquatic organisms are calculated using the ICRP recommended infinite semisphere (or sphere) model.

The third form of exposure that must be considered for aquatic biota is that occurring internally due to known reconcentration processes. Due to the very small dimensions involved, the internal radiation of plants and plankton is not felt to be significant. This source of exposure for larger species must be considered, primarily for fish but also for invertebrate creatures. Internal exposure was calculated by utilization of the following formula:

$$D_{internal}^{(rad/year)} = \sum_i \left(C_i \frac{\mu Ci}{cc}\right) CF_i \left(\frac{\mu Ci/gram}{\mu Ci/cc}\right) 3.7x10^4 \left(\frac{dis}{sec - \mu Ci}\right)$$

$$1.6x10^{-8}\left(\frac{gm - rad}{MeV}\right) 3.16x10^7 \left(\frac{sec}{year}\right) 1.0E_i \left(\frac{MeV}{dis}\right) \qquad (11B\text{-}5)$$

where:

C_i	=	the annual average discharge concentration of the ith radionuclide.
CF_i	=	the CF in the organism of interest of the ith radionuclide.[22].
E_i	=	the effective energy of disintegration for a 10 cm effective radius.*

The internal exposure so calculated amounts to 3.48 (6.52) mrad per year for invertebrates and 2.51 (4.47) mrad per year for vertebrates.

The doses from all the sources considered are compiled and presented in Table 11B-11, "Radiation Exposure of Aquatic Biota". These doses were calculated using the annual average concentrations at discharge for all radionuclides and take no credit for any dilution of the discharge in the river. These dose estimates are thus maximum potential exposure rates at the point of discharge. The annual average discharge concentrations cannot be realistically assumed to exist more than a few hundred feet from the discharge sluice. For this reason Table 11B-11 also includes expected dose rates to aquatic biota based on full dilution of the cooling tower discharge with the annual average river flow of 30,000 cfs.

In the evaluation of the magnitude of the dose rates presented, the conservatism of the calculational procedures and assumptions should be kept in mind. Also to be considered is the relative radiosensitivity of the aquatic population. Within the aquatic population, the more primitive forms, such as unicellular organisms and macrophytes, are the most radioresistant;

fish and amphibians are more radiosensitive. Eggs and larvae in their early developmental stages are the most radiosensitive, and therefore, constitute the most critical group in the aquatic population[30].

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* Values of E_i were abstracted from ICRP II.[9]

TABLE 11B-11

RADIATION EXPOSURE OF AQUATIC BIOTA
0.25% Failed Fuel Cladding

Receptor	Maximum Potential Annual Radiation Exposure, mrad				Expected Annual Exposure, mrad
	From Sediment	From Water	Internal Exposure	From all Sources	From all Sources
Deep Roots of Aquatic Plants	53.4	~0	~0	~53.4	0.089
Burrowing Invertebrates	53.4	~0	3.48	~56.9	0.095
Sediment Level Micro-Organisms	26.7	0.63	~0	~27.3	0.046
Bottom-Feeding Fish	26.7	0.003	2.51	~29.2	0.049
Bottom Dwelling Invertebrates	26.7	0.003	3.48	~30.2	0.051
Free Floating Micro-organisms	~0	1.27	~0	~1.27	0.002
Free Swimming Fish	~0	0.005	2.51	~2.52	0.004
Free Swimming Invertebrates	~0	0.005	3.48	~3.49	0.006

Note: Maximum potential doses are calculated using the annual average discharge concentrations. The expected exposure rates account for full dilution of the cooling tower blowdown in the river.

TABLE 11B-11 Continued

RADIATION EXPOSURE OF AQUATIC BIOTA
1.0% Failed Fuel Cladding

Receptor	Maximum Potential Annual Radiation Exposure, mrad From Sediment	From Water	Internal Exposure	From all sources	Expected Annual Exposure, mrad From all Sources
Deep Roots of Aquatic Plants	337	~0	~0	~337	0.56
Burrowing Invertebrates	337	~0	6.52	~344	0.58
Sediment Level Micro-Organisms	169	0.65	~0	~170	0.28
Bottom-feeding Fish	169	0.02	4.47	~173	0.29
Bottom Dwelling Invertebrates	169	0.02	6.52	~176	0.29
Free Floating Micro-organisms	~0	1.30	~0	~1.30	0.002
Free Swimming Fish	~0	0.03	4.47	~4.50	0.008
Free Swimming Invertebrates	~0	0.03	6.52	~6.55	0.011

Note: Maximum potential doses are calculated using the annual average discharge concentrations. The expected exposure rates account for full dilution of the cooling tower blowdown in the river.

Studies have been made to determine the effect on fish eggs and larvae chronically irradiated by immersion in low-level radionuclide solutions. The lowest radionuclide concentration reported to have had a detectable effect on the development of an aquatic organism (anchovy eggs) was a 0.1 μCi/ml solution of Sr-90 and Y-90.[31]

This exposure, which amounts to about 500 rad per year, resulted only in a slight decrease in the birth rate of the exposed fish eggs.[31]

The relatively radiosensitivity of fish in comparison to man is further illustrated by this comment from Auerbach:[32]*

> One of the current concerns over the increasing installation of nuclear power stations is the potential impact of radioactive waste releases on local ecosystems. In particular, the question has been raised whether waste releases at maximum permissible concentration (MPC) levels would cause ecological problems due to the radioactivity. Hypothetical annual submersion dose rates from water assumed to be maintained at the occupational MPC x 1/30 were calculated for organisms living continuously in these waters. These hypothetical doses are used as a basis for comparisons in a variety of ecological studies of low doses of ionizing radiation and are analyzed and evaluated in terms of detectability of biological effects at MPC levels. Present knowledge based on these and similar studies of the ecological effects of low-level chronic doses, such as could result from routine reactor releases under current standards, guidelines, and operational experience, indicates that any possible biological effects would be undetectable. Although the data in support of this contention are limited, they consistently point to this conclusion.

* Stanley I. Auerbach is the Director of the Environmental Sciences Division at the Oak Ridge National Laboratory.

V. DOSE TOTALS AND COMPARISON WITH FEDERAL REGULATIONS AND NATURAL BACKGROUND

The doses calculated to occur due to the operation of Unit 1 of the Beaver Valley Power Station may best be brought into perspective by a comparison with exposure levels already present from naturally occurring background radiation.

The Environmental Protection Agency[(2)] has conducted a state by state survey of natural background radiation levels and has made the results available to the public. The report issued through the Special Studies Group of the Division of Criteria and Standards, Office of Radiation Programs, lists state-wide average whole body radiation doses from three sources; 1) cosmic ray radiation, 2) naturally occurring terrestrial radiation and 3) internal radiation from naturally occurring radioisotopes incorporated into the human body. For the state of Pennsylvania, the respective annual doses from these sources are:[(2)] 1) 45 mrem, 2) 55 mrem and 3) 25 mrem. Thus in Pennsylvania, the average whole body dose from natural background radiation amounts to 125 mrem annually, slightly lower than the national average of 130 mrem per year.[(2)]

In other states the annual whole body background dose is reported to range upwards from 100 mrem/year in Louisiana to 250 mrem/year in Colorado. The Special Studies Group has also found that medical exposure in the USA averaged 90 mrem/year in 1970.[(2)] The annual average exposure from television sets is reported to be 0.1 mrem.[(2)] The total whole body population exposure occurring as a result of increased cosmic radiation during commercial air travel is estimated at 90,000 man-rem in the USA during 1969.[(33)]

Title 10 Part 20 of the Code of Federal Regulation limits the allowable exposure to individuals of the general population to 500 millirem annually. The per capita dose to any substantial section of the population is not allowed to exceed one third of this value, about 170 millirems annually. These legal guidelines together with the above exposure levels due to natural and man-made radiation provide a basis on which comparison of the incremental doses due to the operation of the Beaver Valley Power Station may be adequately drawn. Such a comparison is illustrated by Table 11B-12 which provides the total maximum individual dose, the total population dose and the total per capita dose to the projected year 1990 population within 50 miles of the plant.

All doses in Table 11B-12 are whole body doses. Organ doses to the maximum individual for all significant organs from all significant sources are presented and totaled in Table 11B-13. It is extremely improbable that any of these maximum individual exposure rates should ever occur. The probability that they should all occur simultaneously to a single individual is entirely negligible. The totals are presented only as certain upper limits of the potential exposure.

TABLE 11B-12

DOSE TOTALS AND COMPARISON WITH FEDERAL REGULATIONS AND NATURAL BACKGROUND
0.25% Failed Fuel Cladding

	Organ of Reference	Maximum Individual Exposure mrem/yr	Population Exposure* In 50 miles man-rem/yr	Per Capita Exposure* In 50 miles mrem/yr
I. Exposure Due to Beaver Valley Gaseous Releases				
1. From Noble Gas Immersion	Whole Body	0.27	4.90	1.15(-3)**
2. From Radioiodine Inhalation	Thyroid	0.05	-	-
3. From Radioiodine Ingestion, via cow-milk pathway	Thyroid	0.15	-	-
II. Exposure Due to Beaver Valley Liquid Releases				
1. From Ingestion of Water	Whole Body	0.63	0.49	1.15(-4)
2. From Ingestion of Fish	Whole Body	0.05	4.48(-5)	1.05(-8)
3. From Swimming	Whole Body	1.22(-4)	-	-
III. Dose Totals				
1. Exposure Due to Beaver Valley Operation	Whole Body	0.95	5.39	1.26(-3)
2. Exposure Due to Natural Background	Whole Body	125	5.34(+5)	125
3. Maximum Allowable Exposure Under 10CFR20	Whole Body	500	7.26(+5)	170

* Population and per Capita Exposure is based on the projected population within 50 miles of the plant for the year 1990 and 4.27×10^6 persons.

** 1.15(-3) = 1.15×10^{-3}; typical for all values.

TABLE 11B-12 Continued

DOSE TOTALS AND COMPARISON WITH FEDERAL REGULATIONS AND NATURAL BACKGROUND
1.0% Failed Fuel Cladding

	Organ of Reference	Maximum Individual Exposure mrem/yr	Population Exposure* In 50 miles man-rem/yr	Per Capita Exposure* In 50 miles mrem/yr
I. Exposure Due to Beaver Valley Gaseous Releases				
1. From Noble Gas Immersion	Whole Body	0.74	37.3	8.74(-3)**
2. From Radioiodine Inhalation	Thyroid	0.12	-	-
3. From Radioiodine Ingestion, via cow-milk pathway	Thyroid	0.42	-	-
II. Exposure Due to Beaver Valley Liquid Releases				
1. From Ingestion of Water	Whole Body	0.63	0.49	1.15(-4)
2. From Ingestion of Fish	Whole Body	0.21	1.78(-4)	4.17(-8)
3. From Swimming	Whole Body	7.70(-4)	-	-
III. Dose Totals				
1. Exposure Due to Beaver Valley Operation	Whole Body	1.58	37.8	8.85(-3)
2. Exposure Due to Natural Background	Whole Body	125	5.34(+5)	125
3. Maximum Allowable Exposure Under 10CFR20	Whole Body	500	7.26(+5)	170

* Population and per Capita Exposure is based on the projected population within 50 miles of the plant for the year 1990 and 4.27×10^6 persons.

** 8.74(-3) = 8.74×10^{-3}; typical for all values.

TABLE 11B-13

WHOLE BODY AND BODY ORGAN DOSE TOTALS FOR THE MAXIMUM INDIVIDUAL
0.25% Failed Fuel Cladding

	mrem/yr to the Whole Body	mrem/yr to the Body Water	mrem/yr to the Thyroid	mrem/yr to the Liver	mrem/yr to the Muscle
I. Exposure Due to Beaver Valley Gaseous Releases					
1. From Noble Gas Immersion	0.271	0.271	0.271	0.271	0.271
2. From Radioiodine Inhalation	n	n	0.049	n	n
3. From Radioiodine Ingestion,* via cow-milk pathway	n	n	0.016	n	n
II. Exposure Due to Beaver Valley Liquid Releases					
1. From Ingestion of Water	0.627	1.020	1.098	1.021	1.021
2. From Ingestion of Fish	0.052	0.067	0.071	0.111	0.091
3. From Swimming	1.220(-4)**	1.220(-4)	1.220(-4)	1.220(-4)	1.220(-4)
III. Dose Totals From all Sources	0.950	1.358	1.505	1.403	1.383

n denotes negligible

* The thyroid ingestion dose used here is that applicable to an adult, for the purpose of compatibility.

** 1.220(-4) = 1.220×10^{-4}; typical for all values.

TABLE 11B-13 Continued

WHOLE BODY AND BODY ORGAN DOSE TOTALS FOR THE MAXIMUM INDIVIDUAL
1.0% Failed Fuel Cladding

		mrem/yr to the Whole Body	mrem/yr to the Body Water	mrem/yr to the Thyroid	mrem/yr to the Liver	mrem/yr to the Muscle
I.	Exposure Due to Beaver Valley Gaseous Releases					
	1. From Noble Gas Immersion	0.736	0.736	0.736	0.736	0.736
	2. From Radioiodine Inhalation	n	n	0.124	n	n
	3. From Radioiodine Ingestion,* via cow-milk-man pathway	n	n	0.046	n	n
II.	Exposure Due to Beaver Valley Liquid Releases					
	1. From Ingestion of Water	0.631	1.025	1.515	1.033	1.028
	2. From Ingestion of Fish	0.207	0.222	0.243	0.506	0.375
	3. From Swimming	7.70(-4)**	7.70(-4)	7.70(-4)	7.70(-4)	7.70(-4)
III.	Dose Totals From all Sources	1.575	1.984	2.665	2.276	2.140

n denotes negligible

* The thyroid ingestion dose used here is that applicable to an adult, for the purpose of compatibility.

** 7.70(-4) = 7.70×10^{-4}; typical for all values.

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APPENDIX 11B.A

COMPUTATIONAL METHODS FOR DOSES RESULTING FROM GASEOUS EFFLUENTS

A. Whole Body Noble Gas Immersion Dose

As indicated in the body of the report, the gaseous radioactive effluents consist primarily of the noble gases krypton and xenon. Exposure of a man to an atmosphere contaminated with radioactive isotopes of these elements results only in an external, whole body dose, from submersion in the radioactive cloud. Since these elements are not incorporated into the human body to a significant degree, there are no resultant internal doses.

For releases at ground level and for the beta component of elevated releases, the resulting dose is proportional to the ground level concentration of radioactivity and can be computed using the ICRP recommended semi-infinite sphere model.[(1)]

The external, whole body, population dose within a sector, s, resulting from the release of Q_i curies of the each radionuclide present per year from the power station was computed by means of the following equation:

$$D_{s,i}(\text{man-rem/year}) = \frac{1}{2} P_s \times (\chi/Q)_s \times Q_i \times (3.7 \times 10^4) \times E_i \times 1.6 \times 10^{-6} \times 1.13 \times \frac{1}{1.293 \times 10^{-3}} \times \frac{1}{10^2} \qquad \text{(11B.A-1)}$$

where: 1/2 = Geometry factor. The body is assumed to be irradiated from half the solid angle by the radioactive cloud of large volume, that is, it is assumed to be surrounded by an infinite, semispherical radioactive cloud.

P_s = Estimated number of persons living within sector "s".

$(\chi/Q)_s$ = Factor computed from atmospheric dispersion equations for the distance of the midpoint sector "s" from the station, and expressed in units of sec/meter^3.

Q_i = Number of curies of the i^{th} radionuclide released from the station per year.

$3.7x10^4$ = dis/sec-μCi

E_i = Effective energy (of β's and γ's) per disintegration of the i^{th} radionuclide, in MeV.

1.13 = Stopping power of tissue relative to air, for β's and secondary electrons produced by x- and γ-radiation.(1)

$1.293x10^{-3}$ = Density of air, gm/cm^3.

10^2 = ergs/gm-rem.

It should be noted that for all the gaseous radionuclides expected to be released, the biological factors for converting rads to rem (the Quality Factor QF and the relative damage factor "n") are taken to be unity. Therefore, the ratio of rem to rads is unity and the dose equation given above may be expressed in units or either rad or rem.

To facilitate the calculations, the terms in the above equation were grouped as follows:

$$D_{s,\,i}(mn-rem/yr) = F_s \, x F_i \qquad (11B.A\text{-}2)$$

where: $F_s = (P_s)(\chi/Q)_s$

$F_i = (0.26)(Q_i)(E_i)$

As such, F_s is a function of population, direction and distance from the station; and F_i is a function of a specific radionuclide.

Values used for P_s were those estimated for the year 1990 population as presented in Figures 2.2-1 and 2.2-2 of the Environmental Report Operating License Stage, BVPS Unit 1. In these figures, ten concentric circles of varying radii are drawn about the site, forming annuli and population data are given for each annulus by compass sectors. These values of P_s have been used in all external noble gas population dose calculations.

The average value of χ/Q for a specific sector was taken to be that for the distance of the midpoint of that sector from the station. For example, the average χ/Q for a sector 10-20 miles in a given direction was taken to be that for a distance of 15 miles from the station in the given direction. The numerical values for χ/Q used in this evaluation were obtained using the NUS computer code WINDVANE. The Q_i values used in the dose calculations are given in Table 11B-1 in the body of the report.

The remaining factor in the dose equation is E_i, the effective energy (β's and γ's) per disintegration of the i^{th} radionuclide. The value of E_i for each radionuclide of interest was obtained from NEDO-12037, "Summary of Gamma and Beta Energy and Intensity Data", by Meek and Gilbert (January 1970).

The population dose (man-rem/year) within the entire area (considered to be within a 50 mile radius of the station) resulting from the annual release of radionuclides from the source being considered is given by the equation:

$$D(\text{man-rem/yr}) = \sum_{s,\,i} F_s F_i \qquad \text{(11B.A-3)}$$

Division of D (man-rem/yr) by the total population within 50 miles will result in the per capita dose within 50 miles from the particular source.

For the Beaver Valley, population doses have been evaluated using this method for the auxiliary building releases and for the beta component of the dose from the elevated release.

Population and individual exposure due to the gamma component of the elevated release has been evaluated using dose rate data obtained by the NUS computer code, WINDOR. A full explanation of the calculational procedures used in WINDOR is beyond the scope of this report, and is therefore not included.

B. Thyroid Inhalation Dose

A small amount of radioactive iodine might be released from Beaver Valley Unit 1 during operation. The external whole body resulting from submersion in a cloud of radioactive iodine is negligible; however, iodine which is taken into the body produces an internal dose as the iodine is preferentially concentrated in the thyroid gland. The thyroid dose was calculated by the following equation:

$$D(rem/yr) = (\chi/Q) \times (BR) \times (DCF) \times (S) \qquad (11B.A\text{-}4)$$

where:

	χ/Q	=	The applicable annual average atmospheric diffusion factor, in sec/m^3.
	S*	=	The total I-131 "dose equivalent" release rate, in Ci/yr.
	BR	=	Breathing rate, $2.31 \times 10^{-4} m^3/sec$ for the "standard man".[1]
	DCF	=	Dose conversion factor, 1.48×10^6 rem/curie I-31 inhaled.[2]

* For other iodine isotopes than I-131, a "dose equivalent" release rate of I-131 is obtained by the method outline in TID-14844.[2]

C. Child Thyroid Milk Ingestion Dose Model

A small amount of gaseous radioiodines will be released from the Beaver Valley Station in addition to the noble gases. The critical pathway for iodine ingestion is the pasture-cow-milk-man pathway with the thyroid being the critical organ. The most sensitive receptor in the population in terms of a thyroid dose from milk ingestion is a young child six months to one year old. The following model[3,4] was used to compute the child thyroid milk dose from I-131.

$$D(\mathrm{mrem/yr}) = \chi/Q \quad V_g K_c I_d \quad \frac{D\infty}{A\tau} \quad \frac{S}{\lambda g} \quad x \quad 10^3 \frac{\mathrm{mrem}}{\mathrm{rem}} \tag{11B.A-5}$$

where:

X/Q = The applicable annual average atmospheric diffusion factor at Meyer's dairy farm.

V_g = 0.1 m/s: deposition velocity of iodine onto pasture(4)

K_c = 0.9: milk/grass activity ratio(4)
(μCi/l)/(μCi/m)

I_d = 0.1 l/day: child milk ingestion rate(5)

$\frac{D\infty}{A\tau}$ = 13.97 x 10^6 rads/Ci: dose activity ratio for child I-31 ingestion(6,7)

$(\lambda_g)^{-1}$ = 7.19 days: mean lifetime for I-131 on the ground

S = The annual average I-131 release (Ci/yr).

The maximum expected and design basis release rates of radioiodines are presented in Table 11B-1.

These annual average release rates can be combined with applicable values of χ/Q to obtain annual average concentrations in air at the location of interest.

For calculational purposes, the nearest potential offsite grazing area has been taken to be at a position on the site boundary that is about 1,860 ft northeast of the primary auxiliary building vent and about 1,380 ft east-northeast of the elevated release point above the Unit 1 cooling tower. This is the same location assumed for the purpose of calculating maximum external exposure from gaseous releases. The pertinent data on distance direction and χ/Q values for this location are presented in Table 11B-4.

The location of the nearest cow and the nearest dairy are identical; no cows are known to exist at a distance from the plant which is less than 1.5 miles, the approximate distance to Meyer's dairy farm. Pertinent data regarding Meyer's dairy farm and others in the vicinity of the plant are presented in Table 11B.A-1. The χ/Q data have been extracted from the WINDVANE computer reduction (as presented in Section 2.2.5) of met data collected at the 50 ft level during the period 9/5/70 to 9/5/71.

For a given isotope with a radiological halflife short in comparison to the natural "weathering" and biological (in the thyroid) halflives, the dose to the child is approximately proportional to the square of the radiological halflife. In consideration of this fact and the relative abundances of the released isotopes, only I-131 needs to be considered in the dose analysis. To calculate the potential annual child thyroid exposure via this pathway it is first necessary to calculate the annual activity intake. The necessary formula is:

$$A\ (\mathrm{Ci/yr}) = \chi/Q\ S\ V\ K\ \ T\ (\mathrm{ground}) I\ /0.693 \qquad (11B.A\text{-}6)$$

where:

where:	χ/Q	=	The applicable atmospheric diffusion parameter, sec/m^3
	S	=	The release rate of I-131 Ci/yr
	V	=	0.01 m/sec, the average deposition velocity[8]
	K	=	0.09 (Ci/l)/(Ci/m^2), the milk to grass activity ratio[9]
	T(ground)	=	The effective halflife on grass of I-131, days
	I	=	1.01/day, the average daily milk intake[2]

The value of the effective halflife on grass is obtained by combining the radiological halflife with the "weathering" halflife which is taken to be 14 days[3]. The resulting value of T (ground) is 5 days for I-131[9][10]. The thyroid exposure per unit activity ingested must also be evaluated. The following expression can be used:

R(rem/ci) = (f T(thyroid)/0.693) (3.7 x 10^{10} dis/sec-ci)
...(8.64 x 10^{4} sec/day)(E)(1.6 x 10^{-8} gram-rad MeV(1 rem/rad)
...(1/m)

(11B.A-7)

where: f = 0.3, the fraction of iodine ingested which reaches the thyroid (see References 9, 11, 12, 13, 14 and 15)

T(thyroid) = The effective halflife in the thyroid of I-131 days

E = The effective disintegration energy in the thyroid of I-131, MeV

m = 2.0 grams[(9)(15)] the mass of the child thyroid gland

The effective halflife of I-131 in the thyroid of a small child is 4.0 days[(16)], found by combining the radiological halflife with an 8 day biological halflife for iodine in the thyroid. The appropriate value of the effective disintegration energy is 0.21 MeV for I-131[(16)].

The equation for annual thyroid dose can be found by combining the expressions for annual activity intake and dose effectiveness. The resulting formula is:

D (rem/yr) = AR (11B.A-8)

= 1.065 x 10^{8} χ/Q (V K I f/m)ST(ground)
...T(thyroid)E

Simplify to obtain:

D (rem/yr) = 6.04 x 10^{4} χ/Q S (11B.A-9)

The formula immediately above was used in conjunction with the given release rates and χ/Q values to obtain potential child thyroid dose rates at the site boundary and at all dairies listed in Table 11B.A-1. The dose results are presented in Table 11B.A-2. It should be noted that the calculational model used here is identical to that used to calculate thyroid doses in Appendix 11-B, however, the value for the effective halflife in the child thyroid has been changed. No correction has been made to account for the fact that cows in the region of concern graze outdoors for only a fraction of the year.

TABLE 11B.A-1

INFORMATION ON DAIRY FARMS IN THE VICINITY OF THE BEAVER VALLEY SITE

	Meyers[3] Dairy	Searight Dairy	Brunton Dairy	Sherman Dairy	Nichols Dairy
Approximate Miles:					
from PAB Vent	1.4	2.3	3.5	9.0	8.5
from Unit 1 Cooling Tower	1.5	2.4	3.5	9.0	8.4
Direction from Plant	SW	SSW	SE	N	NE
Number of Milk Cows	76	33	60	30	60
Total Herd	92	64	160	50	80
Type of Cow	Holstein	Holstein	Holstein	Holstein	Holstein
Feed	Stored feed in winter (corn silage, hay, oats) plus supplemental grains, fresh crop in summer plus supplemental grains.[1]				
Living Habits Winter	inside barn	inside barn	inside barn	inside barn	inside barn
Summer	graze outside	graze outside	inside barn	graze May-October	graze outside
χ/Q Values, sec/m^3					
Release height = 0	3.5-7[2]	1.9-7	2.6-7	2.9-7	2.3-7
Release height = 160m	5.5-9	6.6-9	2.3-8	2.4-8	1.3-8

1. Agway brand or equivalent includes soybean, bran, vitamin, minerals, etc.
2. 3.5-7 = 3.5×10^{-7}; typical for all values.
3. Meyer's Dairy ceased milk production in 1977. It is retained in the Updated FSAR because it represents a conservative analysis when more distant dairies are considered.

TABLE 11B.A-2

POTENTIAL CHILD THYROID MILK INGESTION EXPOSURE RATES

Location	Annual Child Thyroid Exposure Maximum Expected Releases, rem/yr			Annual Child Thyroid Exposure Design Basis Releases, rem/yr			χ/Q Values sec/m^3	
	PAB Releases	Elevated Releases	Total Dose	PAB Releases	Elevated Releases	Total Dose	PAB Releases	Elevated Releases
Site boundary	8.16-3[1]	4.08-8	8.16-3	2.08-2	1.91-5	2.08-2	3.1-5	3.1-9
Meyers Dairy[2]	9.22-5	7.24-8	9.22-5	2.35-4	3.39-5	2.69-4	3.5-7	5.5-9
Searight Dairy	5.00-5	8.69-8	5.01-5	1.27-4	4.07-5	1.68-4	1.9-7	6.6-9
Brunton Dairy	6.85-5	3.03-7	6.88-5	1.74-4	1.42-4	3.16-4	2.6-7	2.3-8
Sherman Dairy	7.84-5	3.16-7	7.67-5	1.94-4	1.48-4	3.42-4	2.9-7	2.4-8
Nichols Dairy	6.06-5	1.71-7	6.07-5	1.54-4	8.01	2.34-4	2.3-7	1.3-8

1. 8.16-3 = 8.16 x 10^{-3}; typical for all values.
2. Meyer's Dairy ceased milk production in 1977. It is retained in the Updated FSAR, because it represents a conservative analysis when more distant dairies are considered.

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APPENDIX 11B.B

COMPUTATIONAL METHODS FOR DOSES RESULTING FROM LIQUID EFFLUENTS

A. Whole Body and Body Organ Exposure from Water Ingestion

The ICRP has established the maximum average liquid intake of the standard man at 1.2 liters per day.[1] The standard man is assumed to have a total body mass of 70 kilograms,[1] hence, the 1.2 liter daily liquid intake represents a daily renewal of 1.71% of the body mass.

The daily intake of the ith radionuclide, A_i (μCi/day) from the hypothesized 1.2 liter daily ingestion of cooling tower blowdown water, can be found from the maximum expected equilibrium concentration, C_i (μCi/cc), by the following formula:

$$A_i\ (\mu Ci/day) = 1{,}200\ (cc/day) \times C_i\ (\mu Ci/cc) \qquad (11B.B\text{-}1)$$

For the whole body, and for all body organs of interest for the particular element, the equilibrium body (organ) burden Q (μCi), can be computed from the daily intake and other information.

$$Q(\mu Ci) = \frac{Af_w}{\lambda_{eff}} = \frac{Af_w T_{eff}}{.693} \qquad (11B.B\text{-}2)$$

where:

A	=	intake rate, μCi/day
f_w	=	fraction of intake reaching organ of interest[1]
λ_{eff}	=	effective (radioactive decay plus biological elimination rate) decay constant,[1] days^{-1}
T_{eff}	=	effective half-life,[1] days

This can be combined with the general equation for determining dose rate, R mrem/year, to an organ from an organ burden Q:

$$R = \Sigma EF\,(RBE)n \frac{MeV}{dis} \frac{3.7x10^4 dis}{sec-\mu Ci} \frac{1}{Mgrams} Q\mu Ci$$

$$\frac{1.6 \times 10^{-6}\ ergs}{MeV} \frac{gm-rem}{100\ ergs} \frac{3.16 \times 10^{7}\ sec}{year} \frac{10^{3}\ mrem}{rem}$$

$$R = \frac{1.87x10^7 \Sigma EF\,(RBE)n\,Q}{M} \text{ mrem/year} \qquad (11B.B\text{-}3)$$

where:	ΣEF (RBE)n	=	effective energy, MeV, for the body organ of interest
	M	=	mass of body organ of interest

Therefore, for the total body with a mass of 70,000 grams,[(1)] the result is:

$$R = 3.845 \times 10^2\ \Sigma EF\ (RBE)n\ f_w\ T_{eff}\ A \qquad (11B.B\text{-}4)$$

For a given radioisotope, the values of the effective energy, effective half-life, fraction reaching the organ, and the organ mass vary with the particular organ under consideration. Table 11B.B-1 lists the particular organs considered for each element. Table 11B.B-2 lists the ICRP values of mass and effective radius of the body organs considered in this report.

Using data given in the ICRP Report of Committee II[(1)] for values of ΣEF (RBE)n, T_{eff} and f_w the annual dose rate due to each ingested radioisotope was determined for the whole body and all the body organs of interest for the particular element, with the exception of the gastrointestinal tract.

Doses to the GI-tract were calculated by the method of "MPC ratioing" wherein the ratio of the actual concentration and the MPC[(1)] are multiplied by the dose derived by the GI-tract from ingestion of water at MPC levels (15 rem/year).

TABLE 11B.B-1
ORGANS OF INTEREST FOR EFFLUENT RADIONUCLIDES

Element	Whole Body	Body Water	Lungs	Liver	Spleen	Kidney	Bone	Muscle	Pancreas	Thyroid	Testes	Ovaries	Prostate	Stomach	Small Intestine	Upper Lg. Intestine	Lower Lg. Intestine
Tritium	X	X															
Chromium	X		X			X				X			X				X
Manganese	X			X		X			X								X
Cobalt	X			X	X	X			X								X
Iron	X		X	X	X		X										X
Strontium	X						X									X	X
Yttrium	X						X									X	X
Niobium	X						X										X
Zirconium	X						X										X
Molybdenum	X			X		X											X
Iodine	X			X	X	X	X			X	X			X	X		X
Tellurim	X			X	X	X	X			X	X						X
Cesium	X		X	X	X	X	X	X							X		
Barium	X		X	X	X	X	X	X			X	X					X
Lanthanum	X			X			X										X
Cerium	X			X		X	X										X

TABLE 11B.B-2

ICRP VALUES FOR ORGAN MASS AND EFFECTIVE RADIUS[1]

Body Organ	Organ Mass grams	Effective Radius cm
Whole Body	70,000	30
Body Water	43,000	Not Given
Lungs(2)	1,000	10
Liver	1,700	10
Spleen	150	7
Kidneys(2)	300	7
Bone	7,000	5
Muscle	30,000	30
Pancreas	70	5
Thyroid	20	3
Testes(2)	40	3
Ovaries(2)	8	3
Prostate	20	3
Stomach	250	10
Small Intestine	1,100	30
Upper Large Intestine	135	5
Lower Large Intestine	150	5

Once the above outlined calculations are accomplished, the doses may be totaled. The dose to a particular organ is made up of the following three components:

1. The body water dose from tritium, unless the organ of consideration is the bone, in which case the dose from tritium is zero.
2. The dose to the particular organ from those isotopes for which the particular organ dose is calculable.
3. The whole body dose from all other isotopes.

Doses to the population as a whole from liquid ingestion, D_{pop} (man-rem/year), were calculated as explained in the body of the report. A dilution factor of ten was assumed to water consumed at Midland. Full dilution of the discharge in the river was assumed for all potable water intakes further downstream than Midland.

B. Whole Body and Body Organ Exposure from Fish Ingestion

Aquatic life forms will concentrate various elements within their bodies in proportion to the concentration of the element in the water in which they live. The ratio of the concentration of the element in the organism to the concentration in water is defined as the concentration factor, or CF. The CF generally depends on the element being concentrated, the species of organism and the environment in which it lives. Average CF's for fish, shellfish and plants in seawater and fresh water have been tabulated[(2)] and are used here to estimate the maximum activity of individual radioisotopes in fish.

For the purpose of estimating the maximum dose an individual might obtain from eating fish it was assumed that an individual might consume a maximum average of 50 grams of fish daily. Thus, the daily activity intake of the ith radionuclide would be given by:

$$A_i\ (\mu Ci/day) = C_i\ (\mu Ci/cc)\ (CF)_i\ 1.0\ (cc/gram)\ (50\ grams/day) \qquad (11B.B\text{-}5)$$

Once the daily activity intake is established, the equilibrium body (organ) burden and the concurrent whole body and organ doses are calculated in the same manner as presented in Part A of this appendix.

The calculation of a population dose from fish ingestion conservatively assumes an average dilution factor of ten to exist in the entire New Cumberland Pool, where it is estimated that 1,000 pounds per year of freshwater fish are caught.(3) Of this thousand pounds, approximately one third will be edible flesh.(4) Since the assumed fish ingestion rate for the maximum individual amounts to about 40 pounds per year, the population dose can be found from the following formula:

$$D_{pop}\,(man-rem/yr)_1 = D_{ind}\,(mrem/yr) \times 10^{-3}\,\frac{rem}{mrem} \times \frac{1,000}{30 \times 40 \times 10}$$

(11B.B-6)

Approximately 65 miles of river downstream of the station exist within a radius of 50 miles. To estimate the total remaning fish catch within 50 miles, the New Cumberland Pool figures were extrapolated. Over the 22.7 miles of the New Cumberland Pool one thousand pounds are caught annually, thus, over the remaining downstream river mileage an estimated 1,863 pounds are caught annually. Assuming the full flow dilution factor of 598, the extra population exposure from this source is computed as follows:

$$D_{pop}\,(man-rem/yr)_2 = D_{ind}\,(man-rem/yr) \times 10^{-3}\,\frac{rem}{mrem} \times \frac{1,863}{3 \times 40 \times 598}$$

(11B.B-7)

And the total population exposure from fish ingestion is given by:

D_{pop} (total) = $D_{pop,1}$ + $D_{pop,2}$ (11B.B-8)

APPENDIX B

REFERENCES

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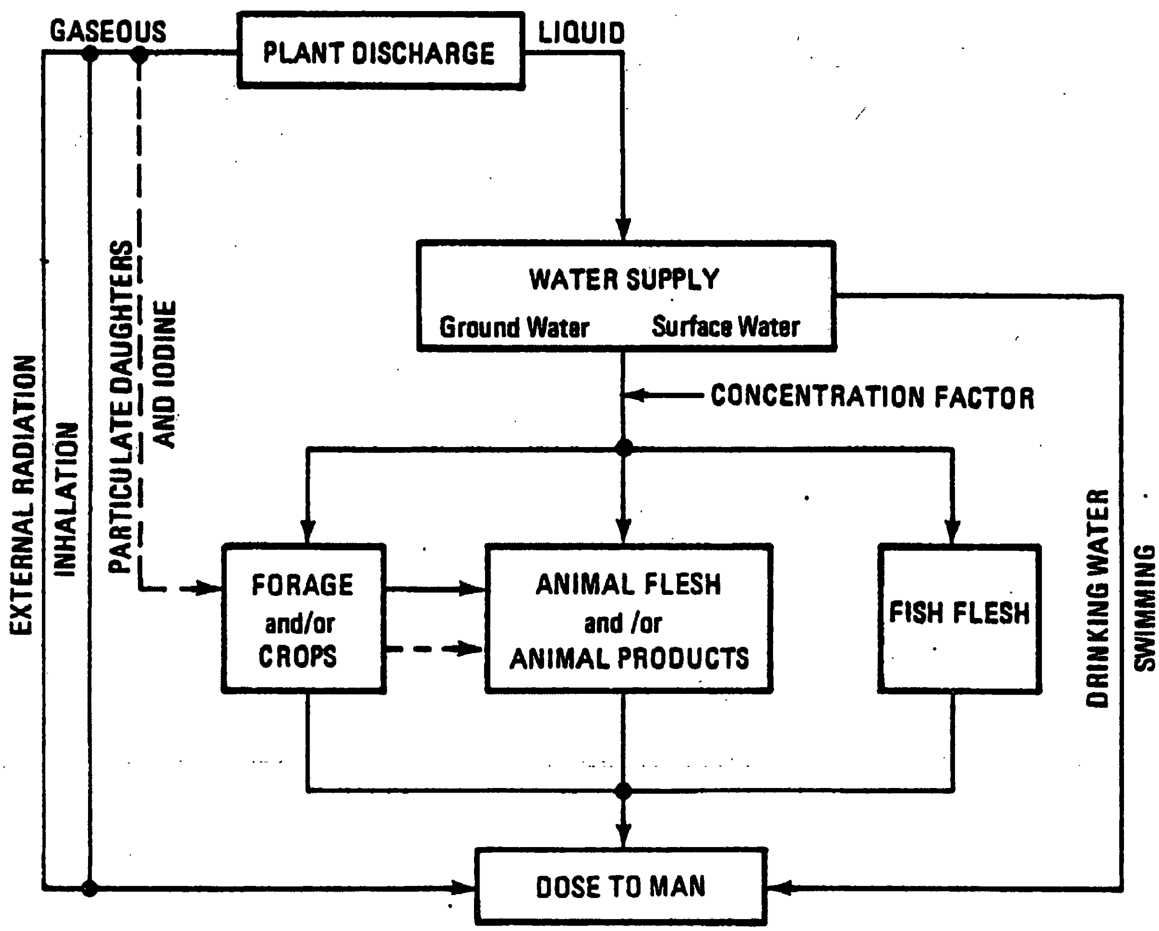


Figure 11B-1 EXPOSURE PATHWAYS

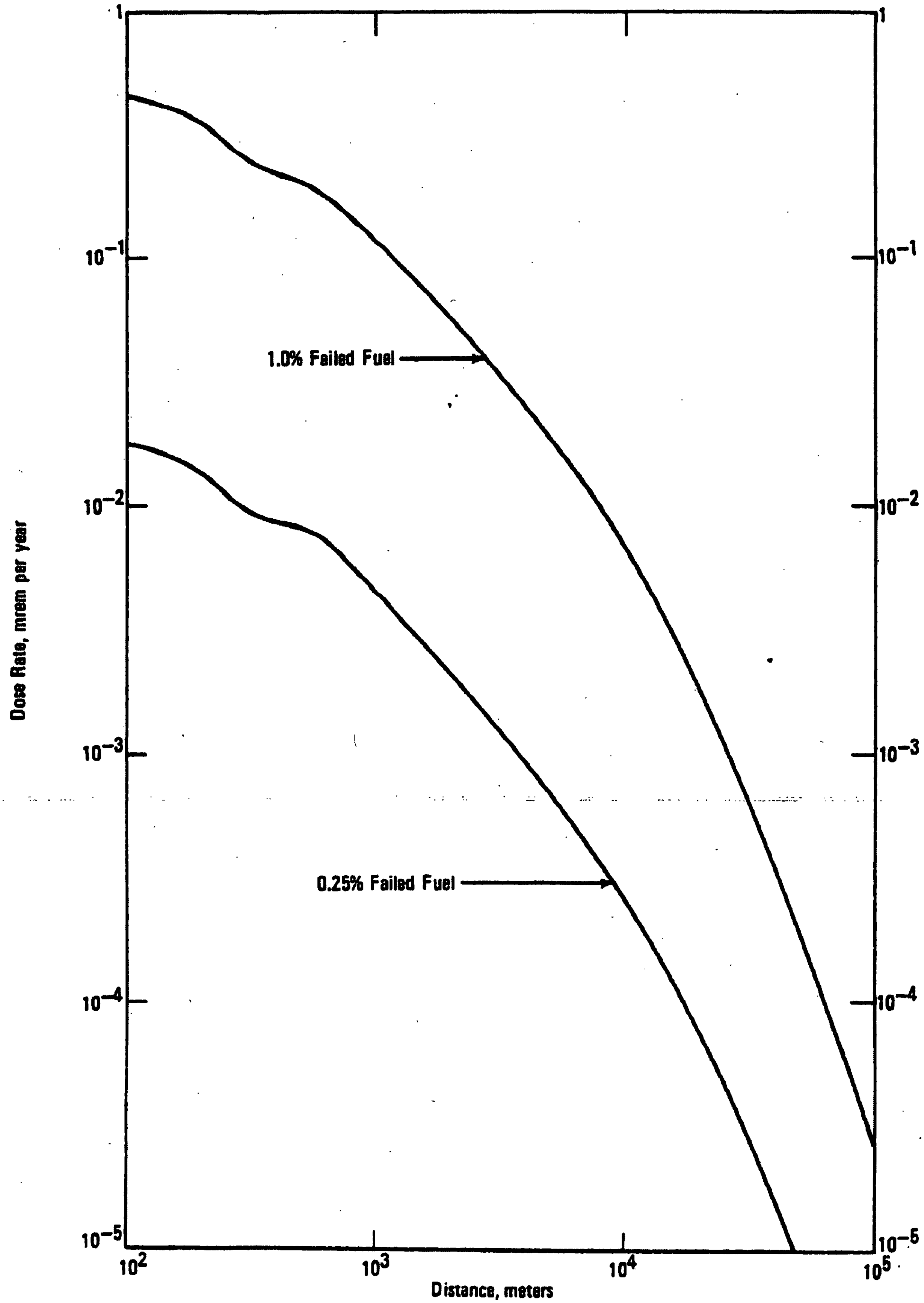
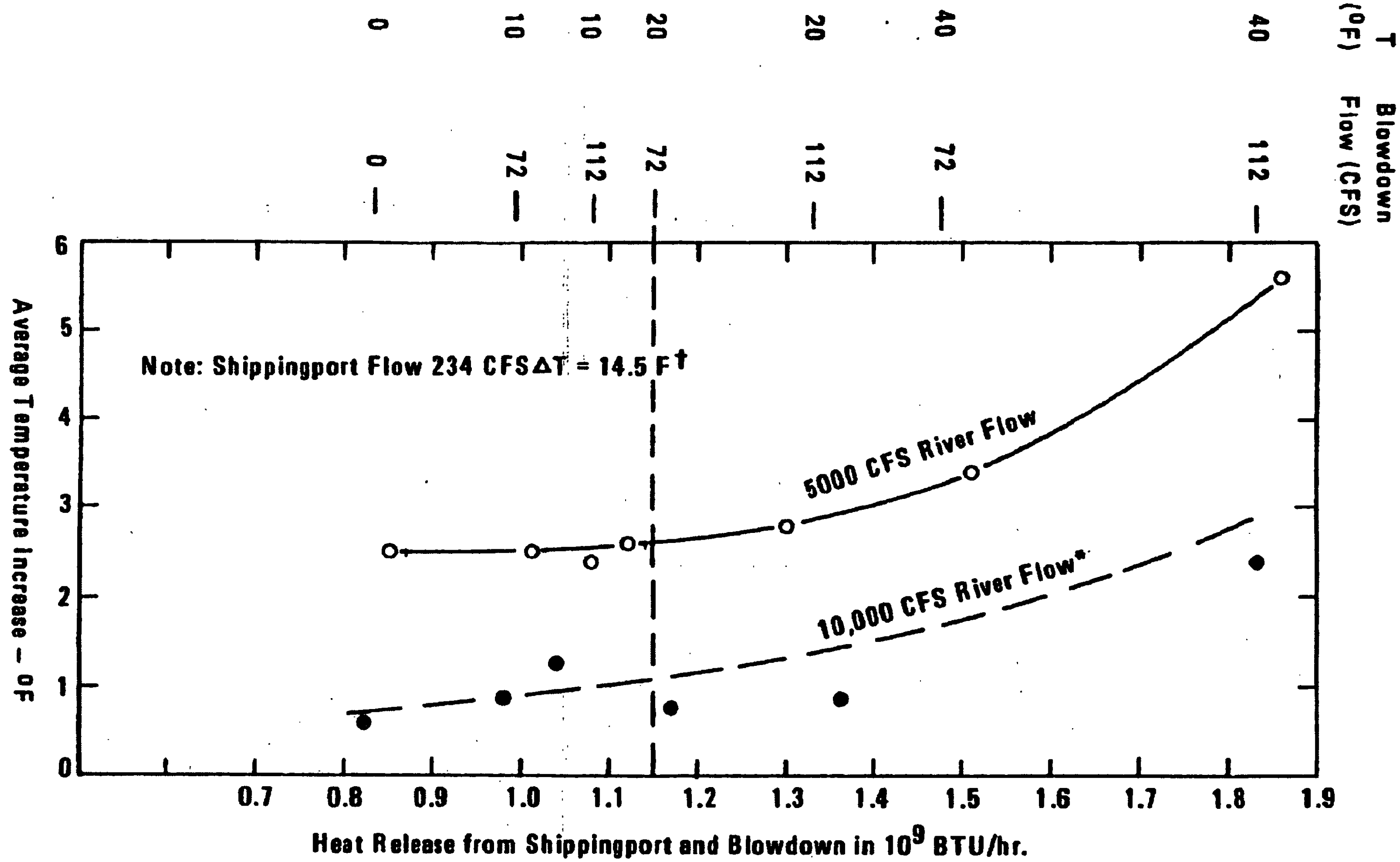


Figure 11B-2 EXTERNAL GAMMA EXPOSURE FROM BVPS UNIT 1 ELEVATED RELEASE, ENE DIRECTION



* The annual average river flow at Beaver Valley is 30,000 CFS

† If the Beaver Valley thermal discharge is 72 CFS (32,000GPM) at 20^0 above ambient, the Beaver Valley heat input to the river amounts to only 29% of the combined heat input from Shippingport and the BVPS.

Figure 11B-3 AVERAGE INCREASE IN TEMPERATURE AT ONE FOOT DEPTH, 3000 FT. DOWNSTRE^M OF THE END OF PHILLIS ISLAND