AND RADIATION
PROTECTION

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11 RADIOACTIVE WASTE AND RADIATION PROTECTION

11.1 RADIOACTIVE WASTES

11.1.1 DESIGN BASES

11.1.1.1 Performance Objectives

The waste disposal system will be designed to provide controlled handling and disposal of liquid, gaseous, and solid wastes which will be generated during plant operation. The design criteria are to insure that station personnel and the general public are protected against excessive exposure to radiation from wastes in accordance with limits defined in 10 CFR 20.

11.1.1.2 Radioactive Waste Quantities

The estimated volumes of radioactive wastes generated during plant operation are listed in Table 11-1.

11.1.1.3 Waste Activity

Activity accumulation in the reactor coolant system and associated waste handling equipment has been determined on the basis of fission product leakage through clad defects in 1 per cent of the fuel. The activity levels were computed assuming full power operation of 2,535 MWt for one core cycle with no defective fuel followed by operation over the second core cycle with 1 per cent defective fuel. Continuous reactor coolant purification at a rate of one reactor system volume per day was used with a zero removal efficiency for Kr, Cs, and Xe, and a 99 per cent removal efficiency for all other nuclides. Activity levels are relatively insensitive to small changes in demineralizer efficiencies, e.g., use of 90 per cent instead of 99 per cent would result in only about a 10 per cent increase in the coolant activity.

The quantity of fission products released to the reactor coolant during steady state operation is based on the use of "escape rate coefficients" (sec -1) as determined from experiments involving purposely defected fuel elements (References 1, 2, 3, 4). Values of the escape rate coefficients used in the calculations are shown in Table 11-2.

Calculations of the activity released from the fuel were performed with a digital computer code which solves the differential equations for a five-member radioactive chain for buildup in the fuel, release to the coolant, removal from the coolant by purification and leakage, and collection on a resin or in a holdup tank. The activity levels in the reactor coolant during full power operation at the end of the second core cycle are shown in Table 11-3

The liquid waste generated by leakage, sampling, and demineralizer sluice or rinse is assumed to have an activity concentration equal to the concentration in the reactor coolant. Reactor coolant bleed will be taken from the downstream side of the purification demineralizer. It is assumed to have the same activity concentration as the reactor coolant reduced by the decontamination factor of the purification demineralizer. Laundry and shower wastes are assumed to contain negligible amounts of radioactivity.

Gaseous activity will be generated by the evolution of radioactive gases from liquids stored in tanks throughout the station. These include such items as the pressurizer and letdown storage tank which are vented to the waste gas disposal system. The activity of the gases is dependent upon the liquid activity. The assumptions for liquid activity are described above. The resulting gaseous activities are described in Section 11.1.2.5, Design Evaluation.

11.1.1.4 Disposal Methods

Liquid wastes from the station will be handled in two separate streams using two evaporator chains. Reactor coolant bleed will be fed through one chain and miscellaneous wastes, which include reactor building sump drains, reactor coolant drains, and floor drains, will be processed through the other chain. The treatment of the liquid wastes will be in one of the following ways: 2

a. Reactor Coolant Bleed 2

1. Collected, monitored, demineralized, and stored in the reactor coolant waste condensate storage tanks.

2. Collected, monitored, concentrated, and either reclaimed through boric acid recycle or discharged to waste drumming area for packaging and off-site disposal.

3. Condensate resulting from the concentration operation will either be reclaimed as demineralized water or discharged with the cooling tower blowdown effluent. 2

b. Miscellaneous Wastes

1. Collected, monitored, demineralized, and stored in the miscellaneous waste condensate storage tanks.

2. Collected, monitored, concentrated, packaged, and shipped off-site.

3. Condensate resulting from the concentration operation will be either reclaimed as demineralized water or discharged with the cooling tower blowdown effluent.

Gaseous wastes are disposed of using one of two methods:

a. Continuous dilution and discharge through waste gas filters to the station vent when activity levels permit.

b. Diversion to waste gas holdup tanks with sampling and controlled subsequent release through waste gas filters to the station vent.

Solid radioactive wastes will be accumulated and packaged in special drums suitable for ICC-approved shipment off-site to a licensed waste disposal facility.

11.1.1.5 Shielding

Shielding for the components of the waste disposal system will be designed on the basis of system activity levels with 1 per cent failed fuel. With the exception of the reactor coolant drain tank and the reactor building sumps, all components will be located in the auxiliary building. The shield design criteria for the auxiliary building is a dose rate of 1 to 2 mrem/hr in continuously accessible areas and 5 to 15 mrem/hr in areas requiring limited access. The components of the waste disposal system will be shielded by concrete walls and floors varying thicknesses depending on the magnitudes of the sources in each component and on the access requirements in a particular area. In some areas local shielding in the form of lead or removable concrete blocks will be utilized to facilitate maintenance or repair operations.

11.1.2 SYSTEM DESIGN

11.1.2.1 Liquid Waste Disposal System

Liquid waste handling will be divided into two separate waste processing chains. One chain will process the reactor coolant bleed stream and the other will handle all miscellaneous liquid wastes. The system flow diagram with necessary instrumentation and controls is shown in Figure 11-1. Waste disposal system component data is given in Table 11-4.

Reactor coolant will be received from the makeup and purification system and will be the largest single source of operational liquid waste to be handled. This liquid will be received as a result of reactor coolant expansion and operational requirements for reduction of reactor coolant boric acid content. It will be either conveyed to reactor coolant bleed holdup tanks for storage or passed through deborating demineralizers for boric acid removal and returned as unborated makeup to the makeup and purification system. The deborating demineralizers will be used only for boric acid concentrations below 1000 ppm to limit the rate at which resins are used up.

The reactor coolant bleed holdup tanks will be sized to contain one reactor coolant system volume each. The contents of each tank will be periodically sampled to determine their radioactive content. These tanks will feed the waste batch tank which in turn supplies the waste evaporator or concentrator. The contents of the waste batch tank will be pumped continuously through the evaporator using the evaporator feed pumps and returned to the batch tank in a closed loop so that the wastes in the loop become progressively concentrated. When wastes are sufficiently concentrated, the evaporator feed pumps will be shut down and the concentrated wastes will automatically drain to the waste batch tank. Residual wastes remaining in the evaporator will be flushed out by returning small amounts of condensate through the evaporator body. This backwashing will ensure a relatively clean evaporator shell and tube bundle with a minimum radiation hazard following operation of the unit. The concentrated wastes in the batch tank will be sampled to determine their content. The wastes may then be disposed of by either pumping to the waste drumming area or recycling through demineralizers to reclaim boric acid. The evaporator condensates will be collected in a condensate test tank where they are sampled to determine quality and activity level. Condensates will then be pumped through cation and evaporator condensate demineralizers to the evaporator condensate storage tanks for

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reuse or may be pumped to the river for disposal after mixing with the cooling tower blowdown effluent. Gaseous wastes will be removed continuously from both the waste batch tank and evaporator using vacuum pumps in the waste gas line and passed through moisture separators into the waste gas holdup tank. In the waste gas holdup tank, gases are monitored for activity, heldup for decay as required, and then released at a controlled rate through the station vent. A monitor located in the gaseous discharge line to the station vent will be equipped with an indicator, and alarm to annunciate a high activity level. The high level alarm actuates an interlock to stop the discharge of gaseous effluents from the waste gas system.

The second evaporator chain will process liquid wastes collected by the miscellareous waste tank and auxiliary building sump tank. The miscellaneous waste tank will collect wastes from the reactor building sump, reactor coolant drain tank, miscellaneous coolers, and liquid samples. The auxiliary building sump tank will collect a variety of liquid wastes including demineralizer rinse; chemical tank drains; waste gas moisture separator drains; and laundry, shower, and floor drains. The liquid wastes will be pumped into a neutralizing tank where the pH of the solution is adjusted as necessary to prevent foaming and samples are taken to determine activity. Wastes will then be transferred to a waste batch tank for cycling through an evaporator or may be pumped to the evaporator condensate storage tanks through the cation and evaporator condensate demineralizers. When wastes are sufficiently concentrated, the concentrates will be collected in the waste batch tank and subsequently pumped to the waste drumming area for packaging and disposal. Condensate will be collected in a condensate test tank, sampled for activity, and subsequently either reused as demineralized water or discharged to the river after mixing with the cooling tower blowdown effluent. Gaseous wastes will be removed by vacuum pumps from the evaporator and batch tank and passed to the waste gas decay tank for ultimate release through the plant vent.

Both evaporator chains will be designed to give decontamination factors higher than 10^4 and will be sized to process wastes at a rate well in excess of the expected waste accumulation rate.

As indicated above, all liquid waste will be sampled and analyzed for radioactive concentration prior to disposal. If discharge to the environment is permissible, a flow indicator and appropriate valving will permit controlled release from the evaporator condensate storage tanks. The flow rate and activity of all liquids discharged from the waste disposal system will be indicated and alarmed. The high activity alarm will actuate an interlock to stop the discharge in the event of excessive activity release.

11.1.2.2 Solids Waste Disposal System

Solid wastes will be placed in ICC-approved containers for the waste material. Loaded containers will be monitored for surface radiation levels and stored in a shielding area prior to shipment to an off-site disposal facility.

Evaporator concentrate from the evaporator that does not contain reusable boric acid will be pumped into a shipping cask for off-site disposal. Spent resins from the demineralizers will be sluiced to a spent resin

storage tank, and the sluice water will be transferred from the tank to the miscellaneous waste holdup tank. The spent resin storage tank will hold one complete charge of resins from the reactor auxiliary systems. Spent resin will be transferred from the storage tank to special drums for disposal. In the drums, radioactive resins will be mixed with concrete and vermiculite and allowed to solidify. The activity of the spent resins and the shielding capability of the drum and shipping cask will determine the mixture proportions in the drum. The insulating properties of concrete and vermuculite protect the drum from excessive gamma heating. Other miscellaneous solid wastes such as filters, clothing, laboratory equipment, pieces of metal, and paper will be disposed of using a baler and light metal shipping containers.

11.1.2.3 Gaseous Waste Disposal Syatem

Gaseous wastes will be removed continuously from both evaporator chains during the concentration operation. Vacuum pumps will maintain a constant vacuum in the batch tanks and evaporators and will draw off waste gases to the gas decay tanks via the waste gas moisture separators. Gaseous wastes in the decay tank will be monitored for activity and held up for decay and then released at a predetermined rate through absolute and charcoal filters to the plant vent. A high activity alarm and indicator will be located in the discharge line and provide automatic shut off of releases at a preselected level.

11.1.2.4 System Radiation Monitoring

The cooling water systems that remove heat from potentially radioactive sources will be monitored to detect accidental releases. A radiation monitor will be located in the intermediate cooling loop, the nuclear services closed cooling loop, the spent fuel cooling system, and in the liquid waste discharge header. In addition, a monitor will be located in the plant effluent line before final discharge into the river.

Reactor coolant letdown flow will be monitored to detect a gross fuel assembly failure. A smaller fuel assembly leak will be detected by regular laboratory analysis of reactor coolant samples. Air samples from the reactor building and the station vent will be monitored for particulate, iodine, and gaseous activity.

These radiation monitors are commercially available equipment. The required characteristics will be established during detailed station design. The maximum sensitivity of detectors when combined with appropriate dilution factors will insure safe limits of release.

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11.1.2.5 Design Evaluation

All analyses on liquid and gaseous waste disposal will be performed on the basis of operation with 1 per cent failed fuel. Although it is not expected that the number of clad defects will ever approach 1 per cent of the total fuel, the objective is to demonstrate the capability of safet station operation within the limits of 10 CFR 20 with quantities of radioactive fission products in the system.

A summary of the various operations considered in the analyses, and the total concentrations resulting in the station effluent from operation of the unit with failed fuel, are given in Table 11-5. The activity concentrations resulting are given as fractions of the Maximum Permissible Concentration (MPC) for unrestricted areas, i.e., the concentration of each radioactive nuclide has been divided by its respective MPC for discharge into unrestricted areas as set forth in 10 CFR 20.

11.1.2.5.1 Liquid Wastes

The normal mode of plant operation will be to treat contaminated wastes in the waste disposal system and store the evaporator condensates in the evaporator condensate storage tanks for reuse as demineralized water supply. However, to demonstrate safety in the event of abnormal operation of the plant, the effects of liquid waste discharges to the environment were analyzed for several situations. The first operation considered in the analysis was the continuous release of liquid wastes corresponding to reactor coolant letdown to storage for boric acid reduction. These wastes were processed through the waste disposal system without holdup or decay and the evaporator condensates were discharged to the river after mixing with the cooling tower blowdown effluent. The activity level in the station effluent was determined by assuming that reactor coolant system liquid was processed through the evaporator drain at the average coolant bleed rate of 25 gph and the condensates were discharged continuously for a period of 278 full power days. Letdown through the purification demineralizer was assumed to give a decontamination factor of zero for cesium and 100 for all other nuclides. In addition, a decontamination factor of 100 for cesium was assumed by passing the liquid waste through a cation demineralizer installed downstream of the reactor coolant evaporator. Decontamination in the evaporator unit was assumed to be 10^4. The dilution flow was the cooling tower blowdown of 2000 gpm. The resulting yearly average concentration in the station effluent was 0.005 of the MPC. This demonstrates that activity levels in the station effluent are significantly less than MPC even in the event of continuous waste releases.

In addition to the continuous release analysis, the effect of an inadvertent release of liquid waste was considered. It was assumed that the entire contents (10,000 gal.) of an evaporator condensate storage tank were pumped at 50 GPM into the cooling tower blowdown effluent due to operator error. This liquid was reactor coolant condensate which had been processed through the cation demineralizer and evaporator train. The instantaneous concentration in the cooling tower blowdown effluent was 0.5 of the MPC. An inadvertent release of raw waste from a reactor coolant bleed tank or a miscellaneous waste tank was not regarded as credible since the discharge from these tanks cannot be released to the environment without passing through the evaporator condensate storage tanks which, therefore, act as a barrier to prevent direct release to the environment.

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Three reactor coolant bleed holdup tanks, each with a capacity of 11,000 ft^3, will be provided for a total storage capacity of 33,000 ft^3. The maximum quantity of coolant letdown for chemical shim, during any 30 day period in life, will be approximately 6000 ft^3. Thus, only one tank, or one-third of the available storage capacity is required to provide a 30 day holdup period for the coolant which will be bled down over 30 days. The maximum volume of coolant removed during heatup and dilution to startup from a cold shutdown will be 14,000 ft^3. This occurs at the end of the chemical shim period. Two cold startups at this stage would generate 28,000 ft^3 of waste. Earlier in life the quantity removed would be less than this due to the smaller amount of dilution required. Two cold startups, early in life, will contribute about 3,000 ft^3 of liquid wastes. The remaining coolant removed from the reactor system is the partial drain which occurs once per year during refueling. The coolant is removed in a batch of 6,100 ft^3 and returned to the reactor coolant system upon completion of refueling. Thus, it occupies storage capacity only during the period of refueling. The required storage volume for refueling operations of 6,100 ft^3 is less than 20 per cent of the available capacity.

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It is extremely unlikely that operating conditions could occur which would require storage for excessive amounts of liquid wastes. However, even in the event of two cold startups toward the end of core life, the available storage capacity would accommodate the liquid wastes. This demonstrates that the three tanks will provide adequate capacity to accommodate all anticipated radioactive wastes as well as providing extra capacity for liquid storage when desired.

The storage facilities for miscellaneous wastes will include the miscellaneous waste holdup tank (2,700 ft^3), the auxiliary building sump tank (440 ft^3), and the reactor building sump (1,000 ft^3). Activity levels in the miscellaneous waste holdup tank were determined by assuming that all liquid collected in the tank was reactor coolant leakage. Collection was assumed to take place continuously at 25 gpd and the contents were processed through the evaporator chain without holdup. The condensates were discharged to the cooling tower blowdown effluent with a dilution flow of 2,000 gpm. The concentration at the point of discharge averaged over the year was significantly less than the MPC for unrestricted areas (as shown in Table 11-5). This concentration will normally be far lower since it is intended to reuse the evaporator condensates as a demineralized water supply.

The reactor coolant and miscellaneous waste handling systems described above will adequately process the anticipated quantities of liquid wastes. In the reactor coolant bleed system, the purification demineralizers and the large system storage capacity will provide ample means of collection and disposal for liquid wastes even in the remote case of 1 per cent fuel failure. Similarly, the miscellaneous wastes are shown to present no problem when analyzed on this conservative basis. It is concluded that the capacity of the liquid waste disposal system will be large enough to permit wide flexibility in station operations while providing a means for safe disposal of wastes with activity well below the acceptable limits.

11.1.2.5.2 Gaseous Wastes

In determining the activity concentrations in the gaseous effluent, the atmospheric dilution was computed using the model for release as described in Section 2.3. Concentrations were calculated at the exclusion distance under the long term release conditions.

The collection of gaseous activity was determined for those components representing the major sources of gaseous release, including the reactor building, makeup tank, pressurizer, and reactor coolant bleed holdup tanks.

The discharge of activity to the atmosphere as a result of reactor coolant bleed was determined for two situations: (1) continuous bleed over life, and (2) dilution and expansion following shutdown and startup.

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For the case of continuous bleed, all of the Kr, Xe, and I in the coolant letdown was assumed to come out in the void space of the reactor coolant bleed holdup tanks. The coolant activity levels were those computed at the end of the second core cycle with 1 per cent failed fuel. Before reaching the reactor coolant bleed holdup tanks, the letdown flow was taken through the demineralizers assuming a 99 per cent removal efficiency for iodine. The activity was released to the atmosphere, without holdup, at a rate equal to the average shim bleed rate over life of 25 gph. Releasing activity at this rate, the total fraction of the MPC at the exclusion distance of 2000 feet is .05. With a 30 day holdup in the reactor coolant bleed tanks, the concentration is reduced to about .005 of MPC.

In the case of unit shutdown and startup, it was postulated that a cold shutdown occurred at a time in lifetime just prior to beginning the use of the deborating demineralizer for boric acid removal. This results in the maximum quantity of coolant bleed during shutdown. As a result of this operation a bleed quantity of 9600 ft^3 is produced. Letdown through the demineralizers with a removal efficiency of 99 per cent for iodine was assumed. As the coolant is let down to the bleed holdup tanks, all of the Kr, Xe, and I is assumed to come out of the water and go into the waste gas decay tank. With a design pressure of 150 psi and a volume of 1500 ft^3, the waste decay tank can hold the total gas volume displaced by this quantity of coolant bleed. The gas displaced from the bleed holdup tank approximately 9600 ft^3 would only pressurize the waste decay tank to about 100 psig. The gaseous activity could then be discharged over a period of one week to allow dispersion in accordance with the long term atmospheric diffusion model. The average annual concentration at the exclusion distance, after a holdup of 30 days in the gas decay tank, would be about .0007 of the MPC. Two cold startups toward the end of core life would produce an average annual concentration of about .0014 of the MPC.

The gaseous concentrations in the makeup tank void were determined from Henry's Law assuming the tank gas space is in equilibrium with the reactor coolant. The fraction of activity in the reactor coolant system which collected in the makeup tank was approximately 45 per cent for Kr, 30 per cent for Xe, and 0.14 per cent for I. The activity levels used for sources in the makeup tank correspond to the reactor coolant system activity at the end of the second core cycle. It is assumed that the tank will be vented once a year to the waste decay tank. The volume of gas in the makeup tank is about 300 ft^3 at 45 psia. This gas would only increase the waste gas decay tank pressure 10 psi. This gas can be discharged to the atmosphere over a period of one week to ensure dispersion in accordance with the long-term atmospheric diffusion model to give an average annual concentration of about .0005 of the MPC at the exclusion distance with no allowance for decay.

Calculations similar to those used for the makeup tank were performed to determine the activity in the pressurizer. It was found that the activity in the pressurizer was approximately one-third the activity in the makeup tank. Venting of the pressurizer results in only about 60 ft^3 of gas, which can be released from the waste decay tank over a period of one week to give a yearly average concentration of about .0001 of the MPC at the exclusion distance.

The activity level in the reactor building atmosphere was computed assuming a reactor coolant system leakage to the reactor building air of 10 gpd. All of the Kr and Xe, and 50 per cent of the I and Cs that leaked from the reactor coolant system, was dispersed throughout the reactor building atmosphere.

Activity buildup in the reactor building was computed over the 30 days of fuel leakage, i.e., it was assumed that no purge had been made for 30 days. This quantity of activity was then discharged to the atmosphere, without decay, by way of the reactor building purge system. The concentration at the exclusion distance of 2000 feet averaged over 30 days was computed to be 0.02 of the MPC. Venting the reactor building once each 30 days would give an average yearly concentration of 0.02 of the MPC, at the exclusion boundary. This calculation was based on the use of the short term dispersion model discussed in Section 2 (Table 2-4).

A preliminary analysis has been made to examine the consequences of reactor operation with steam generator tube leakage and 1 per cent failed fuel rods. The analysis considered the direct dose at various locations in the steam and condensate systems and also the activity release to the environment. The limiting concentration was established by the activity carried with the vacuum pump exhaust to the station vent to remain within the allowable dischar limits of 10 CFR 20. At this limiting concentration, the direct dose rate from the condenser is below the permissible value for continued access.

In the vacuum pumps exhaust, the controlling isotope is xenon-133. The analys assumed that the xenon passed directly from the reactor coolant system leak to the condenser with all the activity ultimately released to the off-gas vent with no radioactive decay. With this conservative assumption, a reactor coolant leak rate of 1 gpm results in a concentration of 0.06 of the MPC at the exclusion distance. The analysis was based on 1 gpm tube leakage continuously over a year.

This evaluation demonstrates that the total yearly average concentration of activity at the exclusion distance from all modes of release, including pressurizer vent, reactor building purge, venting of the letdown storage tank, startup expansion and dilution, chemical shim bleed, and steam generator tube leakage is a maximum of about 0.09 of the MPC. The evaluation also demonstrat that equipment capacities are adequate to accommodate and store radioactive gases as necessary. Thus, the system design is adequate to insure safe dispos of gaseous wastes.

11.1.2.5.3 Radioactive Waste Disposal System Failures

The possibility of a significant activity release off the site from accidents in either the solid or the liquid waste disposal equipment is extremely remote Both of these systems will be located in shielded, controlled-access areas with provisions for maintaining contamination control in the event of spills or leakage. Solid wastes will be disposed by licensed contractors in accordance with ICC regulations. Liquid wastes will be sampled prior to discharge and will be monitored during discharge to insure compliance with 10 CFR 20. A tabulation of potential waste disposal system failures and their consequence is presented in Table 11-6.

Radioactive gases will be sampled and discharged in compliance with the requirements of 10 CFR 20. In the event of waste decay tank failure, these gases would be released to the decay tank compatment, and then released to the station vent via the normal ventilation system.

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he maximum activity in a waste gas decay tank will occur following a boron ilution cycle during reactor startup just prior to switching to deborating emineralizer for boron removal. The reactor coolant water activity used or the analysis assumes prior operation for an extended period with failed uel rods, equivalent to exposure of 1 per cent of the fuel. Approximately .600 ft^3 of reactor coolant would be let down at this time. It is assumed hat the purification demineralizers have a removal efficiency of 99 per ent for iodine and zero removal efficiency for noble gases. The remaining aseous activity will be carried with the water to the reactor coolant bleed oldup tanks, where it is assumed that the gases are immediately released rom the water and carried with the purge gases to the waste gas decay tank. his assumption is quite conservative since the gas release rate will occur ue to diffusion from the surface in accordance with Henry's Law and occur ver a considerable time period. Similarly, it is conservatively assumed hat the gases do not undergo radioactive decay after leaving the reactor oolant system. With these assumptions, the following activity is calculated o exist in the waste gas decay tank:

Isotope	Total Curies
Kr 85M	544.0
Kr 85	4,200.0
Kr 87	300.0
Kr 88	1,000.0
I 131	9.0
I 132	13.4
I 133	12.2
I 134	1.5
I 135	5.7
Xe 131m	570.0
Xe 133m	870.0
Xe 133	79,000.0
Xe 135m	272.0
Xe 135	2,550.0
Xe 138	136.0

he room containing the waste gas decay tanks will be ventilated, and the ischarge will be to the station vent. The activity from a waste gas tank ailure is assumed to be released from the waste gas tank room at a low ontrolled rate to the station vent. The discharge from the station vent s conservatively assumed to mix in the wake of the building structures. tmospheric dilution is calculated using the two hour meterological model iscussed in Section 2 (Table 2-4). The total integrated dose to the whole ody at the exclusion distance is 0.3 rem, and the thyroid dose at the same istance is 0.4 rem. These doses are well below the guideline values of 0 CFR 100.

11.1.3 TESTS AND INSPECTIONS

Functional operational tests and inspections of the Waste Disposal System will be made as required to insure performance consistent with the requirements of 10 CFR 20.

11.2 RADIATION SHIELDING

11.2.1 PRIMARY, SECONDARY, REACTOR BUILDING, AND AUXILIARY SHIELDING

11.2.1.1 Design Criteria

Plant operating personnel and the general public must be protected by radiation shielding wherever radiation hazards exist. Protection will be in accordance with limits on radiation exposure as outlined in 10 CFR 20. The shielding will be designed to perform two primary functions: (1) to insure that during normal plant operation the radiation dose to operating personnel and to the general public is within the limits set forth in 10 CFR 20 and (2) to provide the necessary protection of operating personnel following a reactor accident so that the accident may be terminated without excessive radiation exposure to the operators or to the general public.

To comply with limits specified in 10 CFR 20, the shielding will be designed to give the following radiation dose rate levels throughout the plant:

Full Power Operation Conditions (1% failed fuel)

Location	Dose Rate, mrem/hr
Office, Control Room, and Turbine Building	0.50
Reactor Building: Accessible Areas	25.0
Auxiliary Building: Accessible Areas	1.0 - 2.0

Maximum Hypothetical Accident Conditions

Location	Dose Rate, mrem/hr
Inside Control Room	3 rem integrated whole body dose over 90 days
Outside Reactor Building	See Section 14 (Safety Analysis) for Integral Dose Rate Curves
Site Boundary	See Section 14 (Safety Analysis) for Integral Dose Rate Curves

..2.1.2 Description of Shielding

..2.1.2.1 Primary Shield

ıe primary shield will be a large mass of reinforced concrete surrounding ıe reactor vessel and extending upward from the reactor building floor) form the walls of the fuel transfer canal. The preliminary shield ıickness is 5 ft up to the height of the reactor vessel flange where ıe thickness is reduced to 4.5 ft. The primary shield will meet the)llowing objectives:

a. To reduce, in conjunction with the secondary shield, the radiation level from sources within the reactor vessel and reactor coolant system to allow limited access to the reactor building during normal full power operation.

b. To limit the radiation level after shut down from sources within the reactor vessel to permit limited access to the reactor coolant system equipment.

c. To limit neutron flux activation of component and structural materials.

ıe neutron and gamma-ray heating of the primary shield will be dissipated ' the concrete shield cooling system. The primary shield concrete will be)oled to maintain temperatures less than 150 F.

..2.1.2.2 Secondary Shield

ıe secondary shield will be a reinforced concrete structure surrounding the ıactor coolant equipment including piping, pumps, and steam generators. This ıield will protect personnel from the direct gamma radiation resulting from ıactor coolant activation products and fission products carried away from ıe core by the reactor coolant. In addition, the secondary shield will supplement ıe primary shield by attenuating neutron and gamma radiation escaping from ıe primary shield. The secondary shield will be sized to allow limited access) the reactor building during full power operation. The preliminary thickness ' secondary shield walls is 4.5 ft.

..2.1.2.3 Reactor Building Shield

ıe reactor building shield will be a reinforced, prestressed concrete containment ıructure which completely surrounds the nuclear steam supply system. At ıll power operation, this shield will attenuate any radiation escaping from ıe primary-secondary shield complex such that radiation levels outside the ıactor building will be less than 0.5 mr/hr. In addition, the reactor building ıructure will shield personnel from radiation sources inside the building)llowing a Maximum Hypothetical Accident (MHA). The shielding will be of ıfficient thickness to allow personnel a reasonable time period in which) evacuate the immediate vicinity of the reactor building following the MHA ıthout excessive radiation exposure. The curves in Section 14 (Safety Analysis) ıdicate an integrated direct dose of 9 rem over a period of two hours immediately ıtside the reactor building following the MHA. Preliminary thicknesses of e reactor building wall and dome are 3.5 ft and 3 ft respectively.

11.2.1.2.4 Control Room Shield

The control room shielding will be designed for continuous occupancy for essential control room personnel following a Maximum Hypothetical Accident. This would enable full control and shutdown procedures to be carried out without hazard to the control room operators. Preliminary thickness of the control room shielding is 2 ft. This ensures that the integrated whole body dose over 90 days following the MHA will not exceed 3 rems. Ventilation of the control room under post-accident conditions will be controlled as described in Section 9.8.2.

11.2.1.2.5 Auxiliary Shield

Auxiliary shielding will include all concrete walls, covers, and removable blocks which will shield the numerous sources of radiation occurring in the radioactive waste disposal, makeup and purification, chemical addition and sampling systems. Typical components which require shielding include waste holdup tanks, boric acid and waste evaporators, makeup tank, waste decay tanks, demineralizers, makeup pumps, waste drumming area, reactor coolant drain tank, and reactor building sump pump.

11.2.1.2.6 Spent Fuel Shielding

Shielding will be provided for protection during all phases of spent fuel removal and storage. Operations requiring shielding of personnel are spent fuel removal from reactor, spent fuel transfer through refueling canal and transfer tubes, spent fuel storage, and spent fuel shipping cask loading prior to transportation. Since all spent fuel removal and transfer operations will be carried out under borated water, minimum water depths above the tops of the fuel assemblies will be established to provide radiation shielding protection. Water depths during handling are a minimum of 10 ft in the reactor cavity and fuel transfer canal and 13 ft over stored assemblies in the spent fuel storage area. The dose rates at the water surface will be less than 10 mrem/hr. The concrete walls of the fuel transfer canal and spent fuel pit will supplement the water shielding and will limit the maximum continuous radiation dose levels in working areas to less than 2.5 mrem/hr.

The refueling water and concrete walls also provide shielding from activated control rod clusters and reactor internals which will be removed at refueling times. Although dose rates will generally be less than 2.5 mrem/hr in working areas, certain manipulations of fuel assemblies, rod clusters, or reactor internals may produce short term exposures in excess of 2.5 mrem/hr. However, the radiation levels will be closely monitored during refueling operations to establish the allowable exposure times for plant personnel in order not to exceed the integrated doses specified in 10 CFR 20.

11.2.1.2.7 Materials and Structural Requirements

The material used for the primary, secondary, reactor building, and auxiliary shields will be ordinary concrete with density of approximately 140 lb/ft^3. Since the primary and secondary shielding walls serve as the

efueling structure, give support for the reactor coolant components under ipe rupture conditions, and provide missile shielding, they will be reinorced and designed to be self-supporting.

imes of occupancy in restricted areas will vary depending on measured adiation levels in each area. Such areas as containment operating floor, eactor vessel head prior to refueling, primary loop compartments after hutdown, and spent fuel handling areas will be surveyed prior to access nd a time-limited work schedule will be set up.

1.2.1.3 Evaluation

1.2.1.3.1 Radiation Sources

he shielding will be designed to attenuate neutron and gamma radiation anating from the following basic sources:

a. Reactor Core, Internals, and Reactor Vessel

b. Reactor Coolant Loops

c. Radioactive material released during accidents

d. Auxiliary Systems Equipment

e. Spent Fuel Elements

ource magnitudes are determined for the reactor operating at the aximum expected power level of 2535 MWt with reactor coolant ctivity levels corresponding to 1 per cent failed fuel. Gamma-ray ield and spectral distributions from prompt fission and gross fission roduct activity are based on information in Volume III, Part B, of he Reactor Handbook. The yield and spectral data for capture gammas re taken from ANL-5800, Reactor Physics Constants, and the Reactor andbook. Data on activation product gamma rays are derived primarily rom the Review of Modern Physics, Vol. 30, No. 2 (April 1958). The roduction of N-16 in the reactor coolant is calculated with a B&W ode which computes the integral of the O-16 (n,p) N-16 cross section ver the neutron flux in a water-cooled reactor, subject to variables in oolant flow and density and in neutron flux spectra and magnitude. The -16 (n,p) N-16 cross section used is that reported in WAPD-BT-25. ctivities of individual fission products in the core, reactor, coolant, nd reactor auxiliary systems are determined by a B&W computer code esigned to predict activities from a five-member radioactive chain t any point in the core history. Fission product leakage from the ore to the coolant and removal from the coolant by purification and eakage are calculated.

1.2.1.3.2 Calculation Methods

eutron and Gamma Shields

he primary shield preliminary thickness is based on work performed for he Oconee Nuclear Station using Babcock and Wilcox computer codes which olve the neutron and gamma-ray attenuation equations for the multi-layer

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source-shield complex. Neutron penetration in shield regions was calculated using the B&W LIFEX code as a coefficient generator to provide input data into either the TOPIC code or MIST code. TOPIC (IDO-16968) and MIST (IDO-16856) are programs which solve the transport equation using the Carlson S_N method in cylindrical and slab geometries respectively, and were used to generate 4-group fluxes in the radial and axial directions from the core. Gamma-ray attenuation was calculated using the Taylor exponential form of buildup with the gamma source strengths divided into 1 Mev energy intervals between 1 and 10 Mev. The equations for the direct gamma flux from the simpler geometric sources (line, disc, truncated cone, and cylinder) were solved by a Basic Geometry Code. For the more complex source-shield configurations where non-uniform source distributions may exist, a kernel integration code was used. This program uses a point kernel attenuation along a line-of-sight from the source point to the dose point and computes the gamma flux by summing over the source distribution. Secondary gamma-ray penetration was calculated using a Secondary Gamma program for a laminated, semi-infinite shield array. The aforementioned B&W codes and techniques are described in IDO-24467. Gilbert Associates, Inc. will perform the shielding calculations for final sizing of all radiation shielding.

11.2.1.3.3 MHA Dose Calculation

The thickness of the reactor building shielding, in accordance with the design dose rate criteria, is based upon radiation levels due to fission product release following a reactor accident. For the calculations it was assumed that 100 per cent of the gases, 50 per cent of the halogens, and 1 per cent of the solid fission products were instantaneously released to the reactor building following a buildup period in the core of 600 full power (2,544 MWt) days.

The fission product activity was assumed to be uniformly dispersed throughout the reactor building volume, and the reactor building was represented by a cylindrical source for the dose calculations. The integrated dose over various time intervals was computed as a function of distance from the reactor building. The results are given in 14.2.2.4.

11.2.1.3.4 Operating Limits

The radiation shielding design, including heating and dose rate profiles, temperature distributions, and coolant flow requirements, will be evaluated during the detailed design of the plant to establish the operative limits.

11.2.1.3.5 Radiation Surveys

Neutron and gamma radiation surveys will be performed in all accessible areas of the plant as required to determine shielding integrity. Plans and procedures for radiation surveys during operation and following shutdown will be formulated during the detailed plant design.

11.2.2 AREA RADIATION MONITORING SYSTEM

11.2.2.1 Design Bases

The fixed radiation monitoring system for the Three Mile Island Nuclear Reactor Facilities will be designed to indicate, record, and alarm high radiation monitoring levels throughout the station. The system will be comprised of radiation detectors and allows for visual presentation of readings, recorded presentation, and an audible/visible alarm at the detector location and the Control Room. All instrumentation for the radiation monitoring system will obtain its voltage supply from the vital instrumentation bus and each detector will have a "Loss of Power" alarm. The normal radiation alert alarm setpoint will be above the normal operational reading of the detector. A maximum alarm point will be set to correspond to the MPC value specified in 10 CFR 20. The maximum alarm point set at 10 CFR 20 values could be either an actual value or a calculated number corresponding to 10 CFR 20 limits.

11.2.2.2 Description

Area Gamma Monitoring

Detectors are located as follows:

a. On each of the fuel handling bridges inside the Reactor Building.

b. Inside the Reactor Building near the personnel access hatch.

c. Near incore instrument termination space inside the Reactor Building.

d. On fuel handling bridge in Auxiliary Building.

e. Auxiliary Building decay heat removal pump area.

f. Auxiliary Building near reactor coolant waste pump area.

g. Auxiliary Building near makeup tank.

h. Auxiliary Building near intermediate cooling pumps.

i. Radio-Chemistry laboratory.

j. Cable Relay Room.

k. Contaminated machine shop.

l. Control room.

m. Sample Sink.

n. Reactor Building Dome

Atmospheric Monitoring

Detectors are located as follows:

a. Reactor Building Purge Duct

b. Auxiliary and Fuel Handling Building Exhaust Duct

c. Control Room Ventilation Duct

d. Fuel Handling Ventilation Duct

e. Auxiliary Building Ventilation Duct

f. Reactor Building Air Sample Line

g. Condenser Vacuum Pump Exhaust

h. Site Monitors (2) at 2000 ft site boundary

i. Sample Sink

j. Radio-Chemical Laboratory

k. Spent Fuel Area

l. Waste Gas Decay Tank Discharge

Liquid Monitoring

Detectors are located as follows:

a. Letdown Coolant

b. Intermediate Cooling Water

c. Nuclear Services Closed Cooling Water

d. Spent Fuel Cooling Water

e. Plant Liquid Effluent Line

f. Liquid Waste Discharge Header

D E L E T E D

Detector ranges will be determined depending upon the normal background at the detector locations and the calculated levels for abnormal conditions. Radioactive test sources will be available to allow the overall system performance to be verified at regular intervals.

11.2.2.3 Evaluation

Area radiation monitor detectors will be located on each of the fuel handling bridges to warn personnel if a high radiation level is approached during refueling operations.

A wide range detector will be mounted near the access hatch of the Reactor Building to indicate radiation levels inside the hatch before it is opened. The upper range of the detector will be sufficiently high to indicate the accessibility of the Reactor Building following a serious accident inside.

The incore instrument area will be monitored, and a local alarm will be provided to warn if a high radiation level exists or is created while incore assemblies are being manipulated.

The sample sink area in the Auxiliary Building will be equipped with a detector to alarm an abnormal condition in connection with system sampling.

Alarms will be actuated in the control room and at the detectors if an abnormal change in radiation background occurs.

The radiation monitoring system shall be checked and calibrated at least once per month. When any portion of the radiation monitoring system requires maintenance, that unit shall be completely checked and calibrated immediately after completion of maintenance.

11.2.3 HEALTH PHYSICS

The station superintendent is responsible for radiation protection and contamination control for the Three Mile Island Facility. This responsibility is, in turn, shared by all supervisors. All personnel assigned to the station and all visitors will be required to follow rules and procedures established by administrative control for protection against radiation and contamination.

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ne administration of the radiation protection program will be the responsi-
ility of the station Health Physicist. It will be the responsibility of
ne Health Physics section to train station personnel in radiation safety;
o locate, measure, and evaluate radiological problems; and to make recom-
endations for control or elimination of radiation hazards. The Health
hysics section will function in an advisory capacity to assist all personnel
n carrying out their radiation safety responsibilities and to audit all
spects of station operation and maintenance to assure safe conditions
nd compliance with the AEC and other federal and state regulations concer-
ing radiation protection.

dministrative controls will be established to assure that all procedures
nd requirements relating to radiation protection are followed by all station
ersonnel. The procedures that control radiation exposure will be subject
o the same review and approval as those that govern all other station
rocedures (see Section 12.5, Administrative Control). These procedures
ill include a Radiation Work Permit system. All work on systems or locations
here exposure to radiation or radioactive materials is or may be involved
ill require an appropriate Radiation Work Permit.

RADIATION WORK PERMITS

Radiation Work Permit shall be obtained by all personnel prior to entering
Control Area or performing any work on radioactive or contaminated material
r equipment.

n the event that the safety of the plant or its personnel are endangered,
ntry may be made into a Control Area simultaneously with monitoring per-
onnel. A Radiation Work Permit shall be completed as soon as possible
fter correction of the casualty.

adiation Work Permits shall be issued routinely by the Shift Foreman.

hese permits shall show:

a. The nature of the work to be performed.

b. Expected duration of work.

c. Names of persons to perform the work.

d. Signature of authorizing Foreman.

e. Signature of an individual from the Health Physics Group who shall ensure that:

   1. Designated personnel are within their permissible exposure limits.

   2. The area has been adequately surveyed prior to entry.

   3. Adequate protective clothing and supplies are available at the control point.

   4. Monitors are available for the work.

ll such permits shall be filed by Health Physics for future reference.

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11.2.3.1 Personnel Monitoring System

The personnel monitoring program for the Three Mile Island Nuclear Station shall insure that the recommendations and regulations of the Atomic Energy Commission are followed for all involved personnel. All personnel entering a Control Area shall wear a film badge or its equivalent. Exposures shall be maintained within the limits established in 10 CFR 20. In addition those persons who ordinarily work in restricted areas or whose job requires frquent access to these areas will have pocket chambers, self-reading dosimeters, pocket high-radiation alarms, wrist badges, and finger tabs readily available for use, when required by station conditions. This personnel monitoring equipment will also be available on a day-to-day basis for those persons, employees, or visitors not assigned to the station who have occassic to enter Restricted Areas or to perform work involving possible exposure to radiation. Records of radiation exposure history and current occupational exposure will be maintained by the Health Physics section for each individual for whom personnel monitoring is required. The external radiation dose to personnel will be determined on a daily and/or weekly basis, as required, by means of the pocket chamber and dosimeter. Film badges will be processed monthly or more frequently when conditions indicate it is necessary.

11.2.3.2 Personnel Protective Equipment

Special "protective" or "anticontamination" clothing will be furnished and worn as necessary to protect personnel against contact with radioactive contamination. Change Rooms will be conveniently located for proper utilization of this protective clothing. Respiratory protective equipment will also be available for the protection of personnel against airborne radioactive contamination and will consist of full face filter masks, self-contained air-breathing units, or air-supplied masks and hoods. The first line of defense against airborne contamination in the work area is the ventilation system. However, respiratory protective equipment will be provided should its use become necessary.

Maintenance of the above equipment will be in accordance with the manufacturer's recommendations and rules of good practice such as those published by the American Industrial Hygiene Association in its "Respiratory Protective Devices Manual." The use and maintenance of this equipment will be under the direct control of the Health Physics section, and personnel will be trained in the use of this equipment before using it in the performance of work.

11.2.3.3 Change Room Facilities

Change room facilities will be provided where personnel may obtain clean protective clothing required for station work. These facilities wi divided into "clean" and "contaminated" sections. The "contaminated section of the change rooms will be used for the removal and handling c contaminated protective clothing after use. Showers, sinks, and necessary monitoring equipment also will be provided in the change areas to aid in the decontamination of personnel.

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Equipment decontamination facilities will also be provided at the station for large and small items of plant equipment and components.

Provision will also be made for decontamination of work areas throughout the station.

Appropriate written procedures will govern the proper use of protective clothing; where and how it is to be worn and removed, and how the change room and decontamination facilities for personnel, equipment, and station areas are to be used.

In order to protect personnel from access to high radiation areas that may exist temporarily or semipermanently as a result of station operations and maintenance, warning signs, audible and visual indicators, barricades, and locked doors will be used as necessary. Administrative procedures will also be written to control access to high radiation areas. The Radiation Work Permit System will also be utilized to control access to high radiation areas.

11.2.3.4 Health Physics Laboratory Facilities

The station will include a Health Physics Laboratory with facilities and equipment for detecting, analyzing, and measuring all types of radiation and for evaluating any radiological problem which may be anticipated. Counting equipment (such as G-M, scintillation, and proportional counters) will be provided in an appropriate shielded counting room for detecting and measuring all types of radiation as well as equipment (such as a multi-channel analyzer) for the identification of specific radionuclides. Equipment and facilities for analyzing environmental survey and bioassay samples will also be included in the Health Physics Laboratory. Maintenance and use of the Health Physics Laboratory facilities and equipment will be the responsibility of the Health Physics section.

11.2.3.5 Health Physics Instrumentation

Portable radiation survey instruments will be provided for use by the Health Physics section as well as for operating and maintenance personnel. A variety of instruments will be selected to cover the entire spectrum of radiation measurement problems anticipated at the Three Mile Island Nuclear Station. Sufficient quantities will be obtained to allow for use, calibration, maintenance, and repair. This will include instruments for detecting and measuring alpha, beta, gamma, and neutron radiation. In addition to the portable radiation monitoring instruments, fixed monitoring instruments, i.e., count rate meters, will be located at exits from restricted areas. These instruments are intended to prevent any contamination on personnel, material, or equipment from being spread into unrestricted areas. Appropriate monitoring instruments will also be available at various locations within the restricted areas for contamination control purposes. Portal monitors will also be utilized, as appropriate, to control personnel egress from restricted areas or from the station.

The station will have a permanently installed remote radiation and radioactivity monitoring system for locations where significant levels can be expected. This system will monitor airborne particulate and gaseous radioactivity as well as external radiation levels. This system will presen an audible alarm and radiation level indication in the area of concern in addition to the control room.

11.2.3.6 Medical Programs

Facilities and counting equipment for screening personnel for internal exposure will be available on site with outside services utilized as backup and support for this program.

A comprehensive medical examination program appropriate for radiation workers will be conducted to establish and maintain records of the physical status of each employee at the Three Mile Island Nuclear Station. Subsequent medical examinations will be held as determined necessary for radiation workers. Medical doctors, preferably in the local area, will be used for this program. The Health Physics section will be responsible for the program and will assist the physicians in maintaining medical control of personnel. This program will be designed to preserve the health of the employees concerned and to confirm the radiation control methods employed at the station.

11.3 REFERENCES

(1) Frank, P. W., et al., Radiochemistry of Third PWR Fuel Material Test - X-1 Loop NRX Reactor, WAPD-TM-29, February 1957.

(2) Eichenberg, J. D., et al., Effects of Irradiation on Bulk UO_2, WAPD-183, October 1957.

(3) Allison, G. M. and Robertson, R. F. S., The Behavior of Fission Products in Pressurized-Water Systems. A Review of Defect Tests on UO_2 Fuel Elements at Chalk River, AECL-1338, 1961.

(4) Allison, G. M. and Roe, H. K., The Release of Fission Gases & Iodines From Defected UO_2 Fuel Elements of Different Lengths, AECL-2206, June 19

(5) Duke Power Company, Preliminary Safety Analysis Report, Volume II, 1966.

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Table 11-1
Radioactive Waste Quantities

Waste Source	Quantity per Year	Assumptions and C
Liquid Waste	Ft^3	
Reactor Coolant System:		
Startup Expansion	17,000	4 Cold Startups
Startup Dilution	11,700	2 Cold Startups at begin life and 1 cold startup and 200 full power days tively.
Lifetime Shim Bleed	23,500	Dilution from 1460 to 17
System Drain	6,100	Drain to level of outlet for refueling operations
Sampling and Laboratory Drains	3,000	12 samples per week at 5 per sample
Purification Demineralizer Sluice	160	80 ft^3/year replacement 2 ft^3/ft^3 resin sluice.
Spent Fuel Pool Demineralizer Sluice	42	21 ft^3/year replacement ft^3 resin sluice
Deborating Demineralizer Regeneration and Rinse	2,500	1 Regeneration per year 20 ft^3/ft^3 resin regener
Miscellaneous System Leakage	6,000	5 gph leakage
Laundry	7,300	150 gpd
Showers	15,000	10 showers per day at 30 per shower
Gaseous Waste (a)		
Off-Gas from Reactor Coolant System	1,350	Degas at 25ccH_2 per lite concentration
Off-Gas from Liquid Sampling	74	Degas at 25ccH_2 per lite concentration
Off-Gas from Letdown Storage Tank	900	Vent once per year
Off-Gas from Pressurizer	60	Vent once per year

Table 11-1 (Cont'd)

lid Waste		
urification Resin	80	Resin replacement twice per year
pent Fuel Pool Demineralizer Resin	21	Resin replacement twice per year
aporator Condensate Demineralizer Resin	2	Resin replacement twice per year
aporator Bottoms	800	Concentrated to 20 per cent solid

Excludes reactor building and station ventilation.

Table 11-2

Escape Rate Coefficients for Fission Product Release

Element	Escape Rate Coefficient, sec^{-1}
Xe	1.0×10^{-7}
Kr	1.0×10^{-7}
I	2.0×10^{-8}
Br	2.0×10^{-8}
Cs	2.0×10^{-8}
Rb	2.0×10^{-8}
Mo	4.0×10^{-9}
Te	4.0×10^{-9}
Sr	2.0×10^{-10}
Ba	2.0×10^{-11}
Zr	1.0×10^{-11}
Ce and other rare earths	1.0×10^{-11}

Table 11-3

Reactor Coolant Activities From One Per Cent Defective Fuel

Isotope	Activity, μc/ml	Isotope	Activity, μc/ml
Kr 85m	2.0	I 131	3.3
Kr 85	15.5	I 132	4.9
Kr 87	1.1	I 133	4.5
Kr 88	3.7	I 134	0.55
Rb 88	3.7	I 135	2.1
Sr 89	0.057	Cs 136	0.81
Sr 90	0.0028	Cs 137	77.0
Sr 91	0.057	Cs 138	0.74
Sr 92	0.018	Mo 99	1.2
Xe 131m	2.1	Ba 139	0.088
Xe 133m	3.2	Ba 140	0.076
Xe 133	290.0	La 140	0.026
Xe 135m	1.0	Y 90	0.0007
Xe 135	9.4	Y 91	0.0043
Xe 138	0.5	Ce 144	0.0027

Table 11-4

Waste Disposal System Component Data

eactor Coolant Drain Tank

Number	1
Volume, cu ft	1000
Material	Carbon Steel, Corrosion-Resistant Lining

eborating Demineralizer

Number	2
Type	Semiautomatic Regeneration
Material	Carbon Steel, Corrosion-Resistant Lining

eactor Coolant Bleed Holdup Tank

Number	3
Volume Each, cu ft	11,000
Material	Carbon Steel, Corrosion-Resistant Lining

iscellaneous Waste Holdup Tank

Number	1
Volume, cu ft	2700
Material	Carbon Steel, Corrosion-Resistant Lining

aste Neutralization Tank

Number	1
Volume, cu ft	400
Material	Carbon Steel, Corrosion-Resistant Lining

ent Resin Storage Tank

Number	1
Volume, cu ft	450
Material	Carbon Steel, Corrosion-Resistant Lining

aporator Condensate Storage Tank
(Reactor Coolant Condensate)

Number	2
Volume, cu ft	1500
Material	Carbon Steel, Corrosion-Resistant Lining

Table 11-4 (Cont'd)

Evaporator Condensate Storage Tank (Miscellaneous Waste Condensate)	
Number	2
Volume, cu ft	500
Material	Carbon Steel Corrosion-Resistant Lining
Waste Evaporator (Reactor Coolant Waste)	
Number	1
Process Rate gpm	7.5
Material	Stainless Steel
Waste Evaporator (Miscellaneous Waste)	
Number	1
Process Rate, gpm	2
Material	Stainless Steel
Evaporator Condensate Demineralizers	
Number	2
Material	Stainless Steel
Reactor Building Sump Pump	
Number	1
Capacity, gpm	200
Material	Stainless Steel
Waste Transfer Pump (Reactor Coolant Waste)	
Number	2
Capacity Each, gpm	100
Material	Stainless Steel
Waste Transfer Pump	
Number	2
Capacity Each, gpm	50
Material	Stainless Steel
Auxiliary Building Sump Tank Pump	
Number	2
Capacity Each, gpm	50
Material	Stainless Steel

Table 11-4 (Cont'd)

uxiliary Building Sump Tank		2
Number		
Volume		
Material	Carbon Steel, Corrosion-Resistant Lining	
vaporator Feed Pump (Reactor Coolant Waste)		
Number	2	
Capacity Each, gpm	7.5	
Material	Stainless Steel	
vaporator Feed Pump (Miscellaneous Waste)		
Number	2	
Capacity Each, gpm	2	
Material	Stainless Steel	
vaporator Condensate Pump (Reactor Coolant Waste)		
Number	2	
Capacity Each, gpm	20	
Material	Stainless Steel	
vaporator Condensate Pump (Miscellaneous Waste)		
Number	2	
Capacity Each, gpm	10	
Material	Stainless Steel	
vaporator Vacuum Pump		
Number	2	
Capacity, cfm	6	
Material	Carbon Steel	
aste Gas Compressor		
Number	2	
Capacity Each, cfm	20	
Material	Carbon Steel	
aste Gas Decay Tank		
Number	2	
Volume Each, cu ft	1500	
Material	Carbon Steel	

Table 11-4 (Cont'd)

Waste Gas Filter	
Number	1
Type	Pre, Absolute, and Charcoal Filter Combination
Cation Demineralizers	
Number	2
Material	Carbon Steel, Corrosion-Resistant Lining

Table 11-5

Maximum Activity Concentrations in the Station Effluent With One Per Cent Failed Fuel

Liquid Waste

Operation	Yearly Average Concentrations in Circulating Water Discharge, Fraction of MPC
Lifetime Shim Bleed	0.0050
Discharge of Miscellaneous Wastes	0.0002

Gaseous Wastes

Operation	Yearly Average Concentrations at Site Boundry Fraction of MPC
Lifetime Shim Bleed	0.0050
Startup Expansion and Dilution	0.0014
Venting of Makeup Tank	0.0005
Venting of Pressurizer	0.0001
Reactor Building Purge	0.0200
Steam Generator Tube Leakage of 1 gpm	0.0600

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Table 11-6

Waste Disposal System Failure Analysis

Component	Failure	Comments and Consequences
eactor Building Sump Drain alve (inside or outside)	Fails to close	Backup isolation is provided on opposite side of reactor building.
eactor Building Drain Line alve (inside or outside)	Fails to open	Continuous drainage is not require the valve is located for maintenance during operation.
eactor Building Sump Pump	Fails to operate	Continuous operation is not requir located for maintenance during operation.
eactor Coolant Drain ank Vent Valve	Fails to operate	Continuous venting is not required relief protection is provided for tank
	Fails to close	Vent gas is conveyed to waste gas decay tank and discharged through filters to station vent.
aste Gas Vent Filters	Rupture or lose efficiency	High activity level monitored and alarmed if unsufficient station vent dilution is available. Waste gas is diverted to waste gas decay tanks.
aste Gas Decay Tanks	Leak or rupture	Building purged to station vent through filters. Tanks are protected by relief valves.
eactor Coolant Bleed oldup Tanks	Leak	Leakage is collected in auxiliary building drain sump for process or disposal; building is continuously purged to station vent.
vaporator Train	Fails to operate	Continuous operation is not requir waste gas decay tanks provide for waste collection during maintenanc
eborating Demineralizers	Exhausted resin	Spare unit placed in service while original unit is regenerated. Startup time is increased near end-of-life depending on balance between rod worth and boric acid required.

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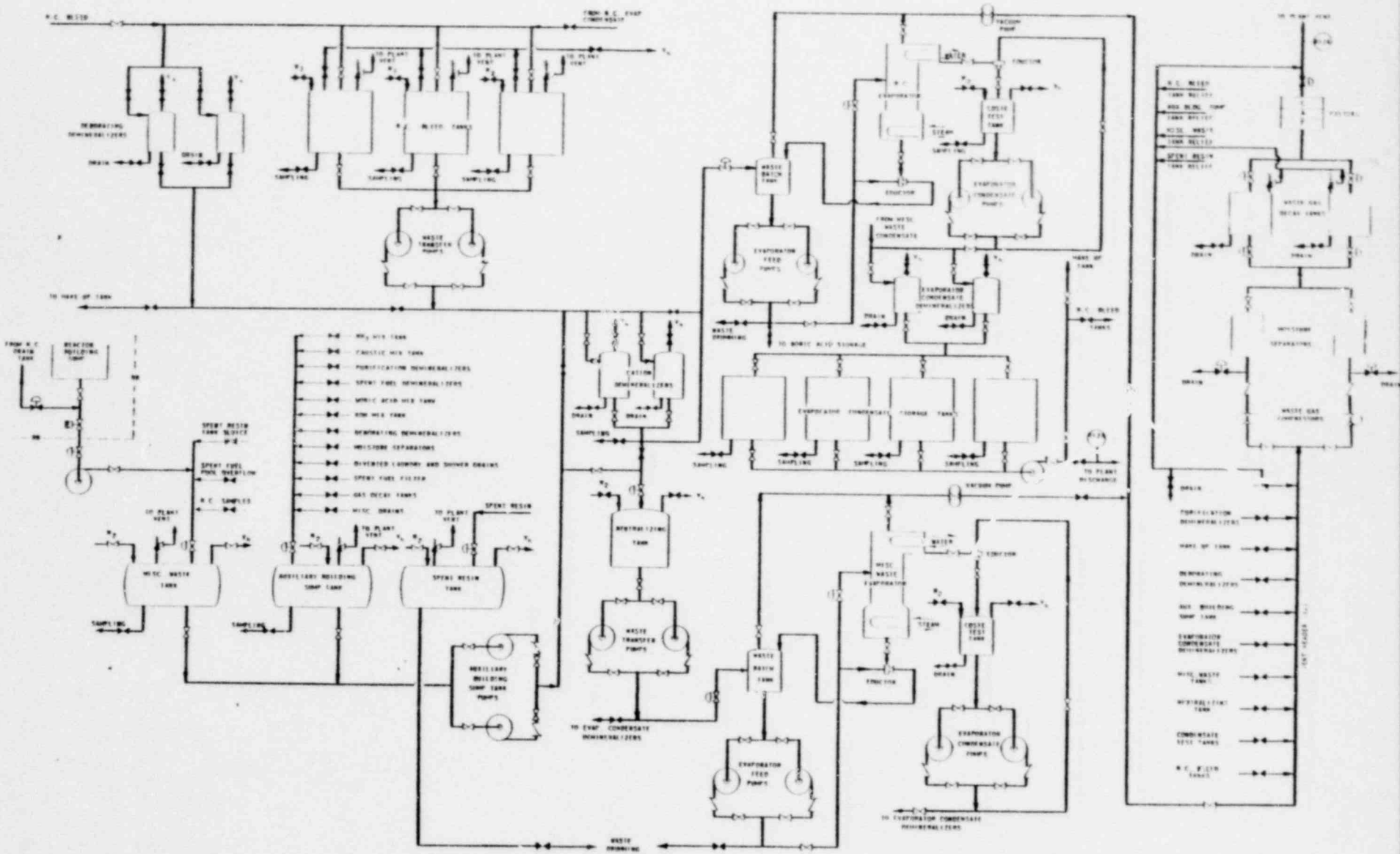

WASTE DISPOSAL SYSTEM

Metropolitan Edison

FIGURE 11-1

THREE MILE ISLAND NUCLEAR STATION

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