11 RADIOACTIVE WASTES AND
RADIATION PROTECTION

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11.1 RADIOACTIVE WASTES

11.1.1 DESIGN BASES

11.1.1.1 Performance Objective

The Waste Disposal System is designed to provide for controlled handling and disposal of liquid, gaseous, and solid wastes from Units 1 and 2 of the Oconee Nuclear Station. The principal design criterion is to insure that station personnel and the general public are protected against excessive exposure to radioactive material in accordance with the regulations of 10 CFR 20.

11.1.1.2 Radioactive Waste Quantities

The estimated volumes of the various forms of radioactive wastes generated in the station 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 2,568 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 50 per cent removal efficiency for Cs, a zero removal efficiency for Kr and Xe, and a 99 per cent removal efficiency for all other nuclides except Te, which was assumed to deposit on the system surfaces. 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.[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.

Table 11-1

Radioactive Waste Quantities From Two Units

Waste Source	Quantity/Year, ft^3	Assumptions and Comments
Liquid Waste		
Reactor Coolant System Startup Expansion	26,200	Startup once per quarter per unit from cold condition.
Startup Dilution	8,200	One startup from cold condition at 190 and 266 full power days, respectively, per unit.
Lifetime Shim Bleed	44,000	Dilution 2,270 to 175 ppm in each unit.
System Drain	12,200	Drain of each unit to level of outlet nozzles during refueling.
Sampling and Laboratory Drains	5,900	24 samples per day at 5 gal per sample.
Purification Demin. Sluice	320	160 ft^3/year replacement at 2 ft^3/ft^3 resin sluice.
Spent Fuel Pool Demin. Sluice	84	42 ft^3/year replacement at 2 ft^3/ft^3 resin sluice.
Deborating Demin. Regen. and Rinse	5000 ~~500~~	2 regenerations per year at 20 ft^3/ft^3 resin regeneration.
Misc. System Leakage	11,700	10 gph leakage.
Laundry	14,600	300 gpd.
Showers	29,200	20 showers per day at 30 gal per shower.

Table 11-1 (Cont'd)

Waste Source	Quantity/Year, ft^3	Assumptions and Comments
Gaseous Waste[a]		
Off-Gas From Reactor Coolant System	2,700	Degas at 25cc H_2 per liter concentration.
Liquid Effluent and Misc. Leakage Off-Gas From Liquid Sampling	148	Degas at 25cc H_2 per liter concentration.
Letdown Storage Tank	900	Vent once per year.
Pressurizer	60	Vent once per year.
Solid Waste		
Purification Resin	160	Resin replacement twice per year.
Spent Fuel Pool Demin. Resin	42	Resin replacement twice per year.
Evap. Cond. Demin. Resin	4	Resin replacement twice per year.
Evaporator Bottoms	1,320	Evaporation to 25 per cent solids for reactor coolant system liquid effluent and deborating demineralizer regenerant and rinse.

[a] Excluding 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^{-10}
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 137	0.39
Sr 90	0.0028	Cs 138	0.68
Sr 91	0.057	Mo 99	1.2
Sr 92	0.018	Ba 139	0.088
Xe 131m	2.1	Ba 140	0.076
Xe 133m	3.2	La 140	0.026
Xe 133	290.0	Y 90	0.0007
Xe 135m	1.0	Y 91	0.0043
Xe 135	9.4	Ce 144	0.0027
Xe 138	0.5		

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 is 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 is generated by the evolution of radioactive gases from liquids stored in tanks throughout the station. Therefore, 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 11.1.2.5, Design Evaluation.

11.1.1.4 Disposal Methods

Liquid wastes from the station are handled in one of three ways:

a. Collected, monitored, and discharged directly to the Keowee Hydro tailrace.

b. Collected, monitored, held up for decay, and then discharged to the Keowee Hydro tailrace.

c. Collected, monitored, concentrated, packaged, and shipped offsite.

Gaseous wastes are disposed of using one of two methods:

a. Continuous dilution and discharge through waste gas filters to the station vent with sweep gas being drawn through tank voids.

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

Solid radioactive wastes are accumulated and packed in containers suitable for ICC-approved shipment offsite 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 quench tank and the reactor building sumps, all components are located in the auxiliary building. The shield design criteria for the auxiliary building is a dose rate of 2.0 mrem/hr in normally accessible areas and 15 mrem/hr in areas requiring limited access. The components of the waste disposal system will be shielded by concrete walls and floors of varying thickness 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 AND EVALUATION

11.1.2.1 Solid Waste Disposal System

Solid wastes are placed in ICC-approved containers appropriate for the waste material. Loaded containers are monitored for surface radiation levels and stored in a special area prior to shipment to an offsite disposal facility.

Evaporator concentrate from the evaporator that does not contain reusable boric acid is pumped into a shipping cask for offsite disposal. Spent resins from the demineralizers are sluiced to a spent resin storage tank, and the sluice water is transferred from the tank to the miscellaneous waste holdup tank. The spent resin storage tank can hold one complete charge of resins from the reactor auxiliary systems. Spent resin is transferred from the storage tank to a shipping cask for disposal. In the cask they are mixed with cement and vermiculite, if necessary, to form a fixed matrix.

Other miscellaneous solid wastes are disposed of using a baler and light metal shipping containers.

11.1.2.2 Liquid Waste Disposal System

Liquid wastes are accumulated in storage tanks and, depending on their composition as determined by local sampling and laboratory analysis, are either discharged through a disposal line to the tailrace of Keowee Hydro, or are processed by evaporation to concentrate impurities for ultimate disposal and to provide for return of purified water for reuse as makeup.

All piping and equipment in contact with reactor coolant are constructed of corrosion-resistant material. This equipment is arranged and located to permit detection and collection of system losses and to prevent escape of any unmonitored radioactive liquid to the environment. A flow diagram of this system with the necessary instrumentation and controls for operation is shown in Figure 11-1. Component data is shown in Table 11-4. Control of equipment is from locally mounted control panels. Shielding of equipment insures operator protection.

Reactor coolant is received from the high pressure injection and purification system and is the largest single source of operational liquid waste to be handled. The liquid is received as a result of reactor coolant expansion and operational requirements for reduction of the reactor coolant boric acid content. It is 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 high pressure injection and purification system. The liquid received from reactor coolant system expansion is normally stored and reused as unprocessed borated makeup during a subsequent reactor coolant system cooldown. Liquid having a boric acid concentration above 1,000 ppm is normally conveyed to bleed storage and subsequently evaporated for removal of boron and impurities. Liquid having a boric acid concentration below 1,000 ppm is passed through the deborating demineralizers and returned to the high pressure injection and purification system as unborated makeup.

Table 11-4

Waste Disposal System Component Data

(Component quantities for two units)

Quench Tank	
Number	2
Volume Each, cu ft	1,000
Material	Carbon Steel, Corrosion-Resistant Lining
Deborating Demineralizer	
Number	3
Type	Semiautomatic Regeneration
Material	Carbon Steel, Corrosion-Resistant Lining
Reactor Coolant Bleed Holdup Tank	
Number	6
Volume Each, cu ft	11,000
Material	Aluminum
Miscellaneous Waste Holdup Tank	
Number	1
Volume, cu ft	2,700
Material	Carbon Steel, Corrosion-Resistant Lining
Waste Neutralization Tank	
Number	1
Volume, cu ft	400
Material	Carbon Steel, Corrosion-Resistant Lining
Spent Resin Storage Tank	
Number	1
Volume, cu ft	450
Material	Carbon Steel, Corrosion-Resistant Lining
Evaporator Condensate Test Tank	
Number	2
Volume Each, cu ft	400
Material	Aluminum
Waste Evaporator	
Number	1
Process Rate, lb/hr	1,000
Material	Stainless Steel

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Table 11-4 (Cont'd)

Evaporator Condenser	
Number	1
Heat Transferred, Btu/hr x 10^{-6}	1
Material, shell/tube	Stainless Steel/Stainless Steel
Reactor Coolant Water Flow, gpm	100
Condensate Demineralizers	
Number	2
Material	Stainless Steel
~~Reactor Building Sump Pump~~/ Component Drain Pump	
Number	2
Capacity Each, gpm	600
Material	Stainless Steel
Waste Transfer Pump	
Number	2
Capacity Each, gpm	100
Material	Stainless Steel
Auxiliary Building Sump Tank Pump	
Number	2
Capacity Each, gpm	50
Material	Stainless Steel
Evaporator Feed Pump	
Number	2
Capacity Each, gpm	5
Material	Stainless Steel
Evaporator Condensate Pump	
Number	2
Capacity Each, gpm	20
Material	Stainless Steel
Evaporator Concentrate Pump	
Number	2
Capacity Each, gpm	5
Material	Stainless Steel
Evaporator Vacuum Pump	
Number	1
Capacity, cfm	6
Material	Carbon Steel
Waste Gas Compressor	
Number	2
Capacity Each, cfm	20
Material	Carbon Steel

TABLE 11-4 (cont'd)

Table 11-4 (Cont'd)

Waste Gas Decay Tank	
Number	2
Volume Each, cu ft	1,350
Material	Carbon Steel
Waste Gas Filter	
Number	1
Type	Pre, Absolute, and Charcoal Filter Combination

The bleed storage capacity consists of six reactor coolant bleed holdup tanks, each sized to contain one reactor coolant system volume, and a miscellaneous waste holdup tank sized to contain one-fourth of a reactor coolant system volume.

Reactor building drainage and minor miscellaneous leakage from equipment located in the reactor building is collected in the reactor building sump. Liquid is pumped from the sump to the miscellaneous waste holdup tank located outside of the reactor building.

A quench tank, located inside the reactor building, condenses and contains any effluent from the pressurizer safety valves. The quench tank is sized to condense one normal pressurizer steam volume (700 ft3 at 2185 psig) without relieving to the reactor building atmosphere.

The miscellaneous waste holdup tank collects resin sluice water from the spent resin storage tank, liquids pumped from the reactor building sump, spent fuel storage pool overflow, and condensate drainage from the waste gas disposal equipment. Liquids from this tank are pumped to the required disposal process following sampling and laboratory analysis.

A divided auxiliary building sump tank is provided. One tank section collects low activity wastes, such as laundry and shower wastes, which are normally discharged directly to the tailrace of Keowee Hydro. The other tank collects liquids from auxiliary equipment and floor drains. The disposal method is decided by sampling and laboratory analysis.

Accumulated liquid wastes requiring processing are evaporated. They are first pumped to a waste neutralization tank, where the pH is adjusted if required; and then fed to the waste evaporator. Distillate from the evaporator is condensed and collected in an evaporator condenstate test tank, where the water quality and activity level are checked. Condensate is then pumped through a condensate demineralizer to either the station demineralized water storage tank for reuse, or to the Keowee Hydro tailrace for disposal. The evaporator train is designed to provide a decontamination factor of 10^4, and is capable of processing eight reactor coolant system volumes in 180 days of continual operation.

The flow rate and activity of all liquid discharged from this system to the Keowee Hydro tailrace are indicated and integrated to maintain release within safe limits. An alarm notifies the operator of discharge of activity above preset values. The process radiation monitoring equipment is discussed in 11.1.2.4.

11.1.2.3 Gaseous Waste Disposal System

All components that can potentially contain radioactive off-gases are vented through collection lines to a vent header. The vent gases are subsequently drawn from this vent header by one of two waste gas compressors, which discharge through waste gas moisture separators to either the station vent or to waste gas decay tanks for holdup prior to release to the station vent. A filter bank containing a prefilter, an absolute filter, and a charcoal filter is installed in the discharge line for filtering all waste gas prior to release to the station vent.

The vent header noted above is normally separated into two sections by a remotely operated valve. One of these sections receives gases from the pressurizer, the letdown storage tank, the gas sampling lines, and the quench tank. Gases from this section of the header are normally passed through one of the two waste gas decay tanks prior to release to the station vent since their potential radioactivity level is higher than levels from other vents. All other vents are connected to the second section of the header and are normally discharged through the waste gas filters directly to the station vent. In the event of above-tolerance activity release as determined by monitoring of the discharge line to the Station vent, the air purge inlets are closed, and these vents will be diverted to the waste gas decay tanks for decay prior to release.

A flow diagram of this system with the instrumentation and controls for operation is given in Figure 11-1. Component data is contained in Table 11-4. Control of equipment is from locally mounted control panels. Shielding and adequate ventilation insure operator protection. The gases in the waste gas decay tanks are sampled and analyzed prior to discharge to evaluate the isotopes present and to establish proper dilution rates. The gaseous effluent is filtered, diluted, and discharged to the station vent. Radiation monitoring (see 11.1.2.4) of the effluent insures gaseous activity discharge rates within 10 CFR 20 limits.

11.1.2.4 Process Radiation Monitoring

Radiation monitoring of station effluent will include alarms and indications designed to provide early warning of possible equipment malfunctions or potential biological hazards. Station effluents will be monitored to insure that prescribed limits of radiation release are not exceeded. The release of gaseous and liquid effluents will be controlled within the limits of 10 CFR 20.

All gaseous radioactivity will be discharged to the atmosphere through the station vent. The source of this activity is the gaseous waste disposal system. After a delay in the holdup tanks, the gas will be analyzed prior to release. Flow in the discharge line from the waste disposal system and flow in the station vent will be monitored to insure that the system is operating correctly and that the releases are within the limits of 10 CFR 20.

A monitor located in the gaseous waste discharge line to the Station vent will be equipped with an indicator and an alarm to annunciate a high activity level. A high-level interlock will stop the discharge of gaseous effluents from the waste disposal system and direct the effluent to the waste gas decay tank.

The condenser air ejector discharge will be monitored for gaseous activity. Should leakage of reactor coolant to the steam occur and result in high activity levels, the monitor on the air ejector will initiate an alarm in the control room.

All liquid waste will be collected, stored, and analyzed for radioactive concentration prior to disposal by evaporation or discharge. If discharge to the environment is permissible, a flow indicator and appropriate valving will permit controlled release. Activity concentration will be determined by sampling of the stored liquid waste prior to release. The flow rate and activity of all liquids discharged from the waste disposal system will be indicated and recorded. An alarm notifies the operator of discharge of activity above preset values such that 10 CFR 20 will not be exceeded.

The cooling water systems that remove heat from potentially radioactive sources are monitored to detect accidental releases. Monitors will be provided on the low pressure service water outlets from the reactor coolant pump seal return coolers, spent fuel coolers, low pressure injection and decay heat removal coolers, and reactor building component coolers, and reactor building spray coolers. Radiation monitors will be located on the reactor building component cooling water lines to detect leakage of reactor coolant into the reactor building component cooling system. An alarm will alert the operator, and the heat exchanger can be isolated. In addition, a monitor will be located on the low pressure service water header upstream of its point of discharge. This monitor will serve as a backup to the preceding monitors. Alarms will alert the operator to isolate the source of release.

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 buildings and the station vent will be monitored for air particulate, gaseous, and iodine activity.

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

11.1.2.5 Design Evaluation

All analyses on liquid and gaseous waste disposal were performed on the basis of both units operating 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 safe station operation within the limits of 10 CFR 20 with quantities of radioactive fission products in the system. Thus, the adequacy of the waste disposal system design is demonstrated.

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

Table 11-5

Maximum Activity Concentrations in the Station Effluent for Both Units Operating With One Per Cent Failed Fuel

Liquid Waste

Operation	Yearly Average Concentration in Tailrace Discharge, Fraction of MPC
Lifetime Shim Bleed Including Startup Expansion and Dilution	0.05
Discharge of Miscellaneous Wastes	0.06

Gaseous Wastes

Operation	Yearly Average Concentration at Site Boundary, Fraction of MPC
Lifetime Shim Bleed	0.04
Startup Expansion and Dilution	0.03
Venting of Letdown Storage Tank	0.004
Venting of Pressurizer	0.002
Reactor Building Purge	0.006
Steam Generator Tube Leakage of 1 gpm	0.07

11.1.2.5.1 Liquid Wastes

The average coolant bleed rate per unit over a core cycle is about 30 gph, including startup expansion and letdown to storage for boric acid reduction.

The activity level in the station effluent was determined by assuming that reactor coolant system liquid was discharged throughout the entire chemical shim bleed life of 266 full power days. Letdown through the purification demineralizer was assumed to give a decontamination factor of 100 for iodine, zero for cesium, and 10 for all other elements for the purposes of this calculation. Activity levels in the reactor coolant system were those at the end of the second core cycle, i.e., the maximum levels. Discharge to the Keowee Hydro tailrace at the rate of 60 gph allows for operation of both Oconee units. No holdup or decay was assumed. Dilution through the tailrace was the average stream flow of 1,100 cfs. The results were a concentration level of 0.3 times the MPC for discharge into unrestricted areas. If the coolant bleed is held up for a period of 30 days, the concentration level is reduced to 0.05 times the MPC.

Six reactor coolant bleed holdup tanks, each with a capacity of 11,000 ft^3, are provided for a total storage capacity of 66,000 ft^3. The maximum quantity of coolant letdown for chemical shim, during any 30 day period in life, is about 8,000 ft^3 for both units. Thus, only two tanks, or one-third of the available storage capacity, are required to provide a 30 day holdup period for all of the coolant which is bled down for lifetime chemical shim.

The maximum volume of coolant removed from one unit during heatup and dilution to startup from a cold shutdown is 7,000 ft^3. This occurs at the end of the chemical shim period. Earlier in life the quantity removed would be less than this, e.g., prior to 190 full power days no dilution is necessary, and the volume of coolant removed from the system for startup from a cold condition is about 3,700 ft^3. Even if both units were started simultaneously, or if one unit was started up twice in a short period of time, the maximum storage capacity required would be 14,000 ft^3. Thus, a 30 day holdup period can be provided with about 22 per cent of the available storage capacity.

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 per unit 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 10 per cent of the available capacity.

It is extremely unlikely that operating conditions could occur which would require simultaneous storage for all of the above liquid wastes. However, even if simultaneous storage were required, it could be accommodated by only two-thirds of the available storage capacity. This demonstrates that the six tanks provide adequate capacity to accommodate radioactive wastes as well as providing extra capacity for liquid storage when desired.

The storage facilities for miscellaneous wastes in the dual-unit system include the miscellaneous waste holdup tank - 2,700 ft^3, the auxiliary building sump tank - 440 ft^3, and the two reactor building sumps - estimated at 1,000 ft^3 each. Activity levels in the waste holdup tank were determined by assuming that all liquid collected in the tank was reactor coolant leakage containing the maximum fission product activity (at the end of the second core cycle). Collection was assumed to take place at the rate of 240 gpd from both units for 60 days. At the end of this time the contents were discharged, without holdup, to the Keowee Hydro tailrace with a dilution flow of 1,100 cfs. The

concentration at the point of discharge averaged over 60 days was 0.8 of the MPC for unrestricted areas. Essentially all of this value is due to I-131 which has an eight day half-life. Thus, by using the auxiliary building or reactor building sumps for collection, the activity collected in the miscellaneous waste holdup tank could be held up for decay prior to discharge to give lower effluent concentrations. For example, by use of these sumps and delay of discharge from the waste holdup tank for 30 days, the average effluent concentration could be reduced to 0.06 times the MPC. If even longer holdup times are desired, one of the reactor coolant bleed holdup tanks can be employed for an additional 11,000 ft^3 of storage capacity for liquid wastes.

The above analysis demonstrates the station capability for handling large quantities of liquid wastes within allowable limits. For the reactor coolant bleed system, the purification demineralizers and the large system storage capacity 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 is 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 Gifford model for wake release as described in 14.2.2.3.6. Concentrations were calculated at the one mile exclusion distance under the long term release conditions.

The collection of gaseous activity was determined for those components representing the maximum potential radiation hazard, including the reactor building, letdown storage 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. 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 30 gph per unit. With both units releasing activity at this rate, the total fraction of the MPC (for unrestricted areas) at the exclusion distance is about 0.04.

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 startup. No coolant activity decay was assumed during the shutdown. As a result of this operation a bleed quantity equivalent to 0.8 reactor system volumes occurs from one unit. 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 1,350 ft^3, one waste gas decay tank can hold the total gas volume displaced by this quantity of coolant bleed. The approximately 7,000 ft^3 of gas displaced from the bleed holdup tank would only pressurize the waste gas decay tank to 76 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 yearly concentration at the exclusion distance, for one such operation per quarter, would be about 0.015 of the MPC for one unit. The average concentration for two units operating in this same manner would be 0.03 MPC.

The gaseous concentrations in the letdown storage 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 letdown storage 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 letdown storage tank corresponded to the reactor coolant system activity at the end of the second core cycle. It is assumed that the tank in each Unit is vented, once a year, to the waste gas decay tank. The volume of gas in the letdown storage 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. The average yearly concentration of activity at the exclusion distance is computed to be 0.004 of the MPC for both units.

Calculations similar to those used for the letdown storage 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 letdown storage tank. Venting of the pressurizer results in only about 60 ft^3 of gas, which can be released from the waste gas decay tank over a period of one week to give a yearly average concentration of less than 0.002 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 per unit. All of the Kr and Xe, and 50 per cent of the I and Cs that leaked from the reactor coolant system, were dispersed throughout the reactor building atmosphere. Activity buildup in the reactor building was computed over the last 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 averaged over 30 days was computed to be 0.003 of the MPC. Venting both reactor buildings once each 30 days would give an average yearly dose of 0.006 of the MPC.

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, and chemical shim bleed, is a maximum of about 0.04 of the MPC for one unit. Even in the remote instance of 1 per cent fuel failure in both units concurrently, the average yearly concentration at the site boundary would be about 0.08 of the MPC. The evaluation also demonstrated that equipment capacities are adequate to accommodate and store radioactive

gases as necessary. Thus, the system design is adequate to insure safe disposal of gaseous wastes.

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 air ejector exhaust to the station vent to remain within the allowable discharge 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 air ejector exhaust, the controlling isotope is xenon-133. The analysis assumed that the xenon passed directly from the reactor coolant system leak to the condenser with all the activity ultimately released to the station vent with no radioactive decay. With this conservative assumption, a reactor coolant leak rate of 1 gpm results in a concentration of 0.07 of the MPC at the exclusion distance. The analysis was based on 1 gpm tube leakage continuously over a year. Therefore, much higher tube leakages could be permitted for shorter periods of time.

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 are located in shielded, controlled-access areas with provisions for maintaining contamination control in the event of spills or leakage. Solid wastes are disposed by licensed contractors in accordance with ICC regulations. Liquid wastes are sampled prior to discharge and are monitored during discharge to insure compliance with 10 CFR 20. A tabulation of potential waste disposal system failures and their consequences is presented in Table 11-6.

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

The maximum activity in a waste gas decay tank will occur following a boron dilution cycle during reactor startup just prior to switching to deborating demineralizer for boron removal. The reactor coolant water activity used for the analysis assumes prior operation for an extended period with failed fuel rods, equivalent to exposure of 1 per cent of the fuel. Approximately 0.8 of an equivalent reactor coolant system volume would be let down at this time. It is assumed that the purification demineralizers have a removal factor of only 100 for iodine, although factors of 10^3 to 10^4 have been reported in the literature. No removal of noble gases is assumed.

Table 11-6

Waste Disposal System Failure Analysis

Component	Failure	Comments and Consequences
Normal Reactor Building Sump Drain Valve (inside or outside)	Fails to close.	Backup isolation is provided on opposite side of reactor building.
Component Drain Pump Drain Valve (inside or outside)	Fails to open.	Continuous drainage is not required; the valve is located for maintenance during operation.
Reactor Building Sump Pump	Fails to operate.	Continuous operation is not required; located for maintenance during operation.
Component Drain Pump	Fails to operate.	Continuous operation is not required; located for maintenance during operation.
Quench Tank Vent Valve	Fails to open.	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.
Waste Gas Vent Filters	Rupture or lose efficiency.	High activity level monitored and alarmed if insufficient Station vent dilution is available. Waste gas is diverted to waste gas decay tanks.
Waste Gas Decay Tanks	Leak or Rupture.	Building purged to station vent ~~through filters~~. Tanks are protected by relief valves.
Reactor Coolant Bleed Holdup Tanks	Leak.	Leakage is collected in auxiliary building drain sump for process or disposal; building is continuously purged to station vent.
Evaporator Train	Fails to operate.	Continuous operation is not required; waste gas decay tanks provide for waste collection during maintenance.
Deborating 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.

The remaining gaseous activity is carried with the water to the reactor coolant bleed holdup tanks, where it is assumed that the gases are immediately released from the water and carried with the purge gas to the waste gas decay tank. This assumption is quite conservative since the gas release rate will occur due to diffusion from the surface in accordance with Henry's Law and occur over a considerable time period. Similarly, it is conservatively assumed that the gases do not undergo radioactive decay after leaving the reactor coolant system. With these assumptions, the following activity is calculated to exist in one of the waste gas decay tanks:

Isotope	Total Curies
Kr 85m	475
Kr 85	3,890
Kr 87	280
Kr 88	820
I 131	8.2
I 132	11.9
I 133	11.0
I 134	1.3
I 135	5.0
Xe 131m	540
Xe 133m	780
Xe 133	69,200
Xe 135m	260
Xe 135	1,510
Xe 138	110

The area surrounding the waste gas decay tanks is ventilated and discharges to the station vent. The discharge from the station vent is conservatively assumed to mix in the wake of the building structures rather than remain at its elevated release point. This assumption produces less favorable dilution and therefore higher ground concentrations at the exclusion distance. Also, with this assumption, the doses at the exclusion distance are essentially the same whether or not the ventilation system is operating.

The activity from a waste gas tank failure is assumed to be released as a puff from the station vent. Atmospheric dilution is calculated using the two hour meteorological model discussed in 14.2.2.3.6. The total integrated dose to the whole body at the one mile exclusion distance is 0.3 rem, and the thyroid dose at the same distance is 0.4 rem. These doses are well below the guideline values of 10 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 PROTECTION

11.2.1 PRIMARY, SECONDARY, AND REACTOR BUILDING SHIELDING

11.2.1.1 Design Bases

The shielding is designed to perform two primary functions: (1) to insure that during normal 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 insure that operating personnel are adequately protected in the event of a reactor accident so that the accident can be terminated without undue hazard to the general public. The shielding design is based on the two Oconee reactors operating at the maximum expected power level of 2,568 mwt each with system activity levels equivalent to 1 per cent failed fuel, and is governed by the following criteria for radiation levels.

Location	Dose Rate, mrem/hr
Site Boundary	0.05
Office, Control Room, and Turbine-Generator	0.5
Normal Accessible Areas in Reactor Building during Operation at Full Power	25
Inside Control Room following MHA	3 rem integrated whole body dose for 8 hour shifts over 90 days after accident.

11.2.1.2 Description

11.2.1.2.1 Primary Shield

The primary shield consists of reinforced concrete which surrounds the reactor vessel and extends upward from the reactor building floor to form the walls of the fuel transfer canal. The preliminary shield thickness is 5 ft up to the height of the reactor vessel flange, where the thickness is reduced to 4.5 ft. The primary shield is designed to meet the following objectives:

a. To attenuate the neutron flux to limit the activation of component and structural materials.

b. To limit the radiation level after shutdown to permit access to the reactor coolant system equipment.

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

11.2.1.2.2 Secondary Shield

The secondary shield is composed of reinforced concrete which surrounds ~~and covers~~ the reactor coolant equipment, including the piping, pumps, and steam generators. The preliminary thickness of structure which surrounds the coolant system is 4.~~5~~ ft, ~~and the preliminary thickness of cover over the system is 2 ft~~. The shielding is designed to reduce radiation levels from activity in the reactor coolant, and to supplement the primary shield in the attenuation of neutrons and secondary gamma rays so as to permit limited access to the reactor building during full power operation.

11.2.1.2.3 Reactor Building Shield

The reactor building shield is a reinforced, prestressed concrete structure with ~~3.5~~ 3.75-ft-thick cylindrical walls and a ~~3~~ 2.5-ft-thick dome. In conjunction with the primary and secondary shields, it will limit the radiation level outside the reactor building from all sources inside the reactor building to no more than 0.5 mrem/hr at full power operation. The shielding is also designed to protect station personnel from radiation sources inside the reactor building following the Maximum Hypothetical Accident. Additional shielding is provided around the control room to insure that exposure to operating personnel in the control room is within the design limits following the MHA.

11.2.1.2.4 Materials

The material used for the primary, secondary, and reactor building shields is ordinary concrete with a density of approximately 140 lbs/ft^3. Since the primary and secondary shielding walls serve as the refueling structure, give support for the reactor coolant components under pipe rupture conditions, and provide missile shielding, they are reinforced and designed to be self-supporting. The concrete primary shielding immediately surrounding the reactor vessel will require supplemental cooling to remove internal gamma heat. This is accomplished by water-cooled coils embedded in the concrete in this area.

11.2.1.3 Evaluation

11.2.1.3.1 Radiation Sources

Gamma-ray yield and spectral distributions from prompt fission and gross fission product activity are based on the information in Volume III, Part B, of the Reactor Handbook. The yield and spectral data for capture gammas are taken from ANL-5800, Reactor Physics Constants, and the Reactor Handbook. Data on activation product gamma rays are derived primarily from the Review of Modern Physics, Vol 30, No. 2, April 1958. The production of N-16 in the reactor coolant is calculated with a code by The Babcock & Wilcox Company which computes the integral of the O-16 (n,p) N-16 cross section over the neutron flux in a water-cooled reactor, subject to variables in coolant flow and density and in neutron flux spectra and magnitude. The O-16(n,p) N-16 cross section used is that reported in WAPD-BT-25. Activities of individual fission products in the core, reactor, coolant, and reactor auxiliary systems are determined by a B&W computer code designed to predict activities from a five-member radioactive chain at any point in the core history. Fission product leakage from the core to the coolant, and removal from the coolant by purification and leakage, are calculated.

11.2.1.3.2 Calculation Methods

Neutron penetration in shield regions is 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 are used to generate 4-group fluxes in the radial and axial directions from the core.

Gamma ray attenuation is calculated using the Taylor exponential form of build-up with the gamma source strengths divided into 1 mev energy intervals between 1 and 10 mev. The equations for the flux from the simpler geometric sources (line, disc, truncated cone, and cylinder) are solved by the B&W Basic Geometry Code. Equations and data for the generation and attenuation of secondary gamma rays in a laminated, semi-infinite shield array are contained in the B&W Secondary Gamma Program. For the more complex source-shield configurations where nonuniform source distributions may exist, BONGO, a B&W kernel integration code, is used. The program utilizes a point kernel attenuation along a line-of-sight from the source point to the dose point, and computes the flux by summing over the source distribution. A description of the aforementioned B&W codes and techniques can be found in IDO-24467.

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 (2,568 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

All parameters governing the shield design, including heating and dose rate profiles, temperature distributions, and coolant flow requirements, will be performed during the detailed design of the station.

11.2.1.3.5 Radiation Surveys

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

11.2.2 AREA RADIATION MONITORING SYSTEM

11.2.2.1 Design Bases

The area radiation monitoring system will be designed to indicate and alarm high radiation levels inside the station. Indication from the beta-gamma detectors located in selected areas of the station will be used in conjunction with operating procedures to assure that personnel exposure does not exceed 10 CFR 20 limits. The control room and Auxiliary Building ventilation systems will be monitored.

11.2.2.2 Description

Beta-gamma detectors are located as follows:

a. One detector 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 space inside the Reactor Building.

d. Fuel handling bridge in Auxiliary Building.

e. Auxiliary Building pump area.

f. Auxiliary Building near sample sink.

g. Auxiliary Building cask decontamination and loading area.

h. Auxiliary Building in shutdown cooler area.

i. Auxiliary Building near Reactor Building component cooling water coolers.

j. Chemistry laboratory.

k. Cable and computer room.

l. Contaminated machine shop.

m. Control room.

Readout for each detector will be provided in the control room. High radiation alarm signals for each detector will be furnished to the control room and to each remote detector location. Sources will be available to allow the overall system performance to be verified at regular intervals.

Detector ranges will be determined depending upon the normal background at the detector locations and the expected radiation levels for abnormal conditions.

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.

11.2.3 HEALTH PHYSICS

The station superintendent is responsible for radiation protection and contamination control for Oconee. 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.

The administration of the radiation protection program will be the responsibility of the station Health Physicist. It will be the responsibility of the Health Physics section to train station personnel in radiation safety; to locate, measure and evaluate radiological problems; and to make recommendations for control or elimination of radiation hazards. The Health Physics section will function in an advisory capacity to assist all personnel in carrying out their radiation safety responsibilities and to audit all aspects of station operation and maintenance to assure safe conditions and compliance with AEC and other federal and state regulations concerning radiation protection.

Administrative controls will be established to assure that all procedures and requirements relating to radiation protection are followed by all station personnel. The procedures that control radiation exposure will be subject to the same review and approval as those that govern all other station procedures (see Section 12.5, Administrative Control). These procedures will include a Radiation Work Permit system. All work on systems or locations where exposure to radiation or radioactive materials is or may be involved will require an appropriate Radiation Work Permit initiated by Health Physics and approved by cognizant supervisors before work can begin. The radiological hazards associated with the job will be determined and evaluated prior to issuing the permit. The work permit will list the precautions to be taken, the protective clothing to be worn and any other radiation control and safety precautions that may be required.

11.2.3.1 Personnel Monitoring Systems

Personnel monitoring equipment consisting of film badges or their equivalent will be assigned by the Health Physics section and worn by all personnal at Oconee. In addition, those persons who ordinarily work in restricted areas or whose job requires frequent 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 occasion 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 will be divided into "clean" and "contaminated" sections. The "contaminated" section of the change rooms will be used for the removal and handling of 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.

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 multichannel 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 Oconee. 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 present 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 Oconee. 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.

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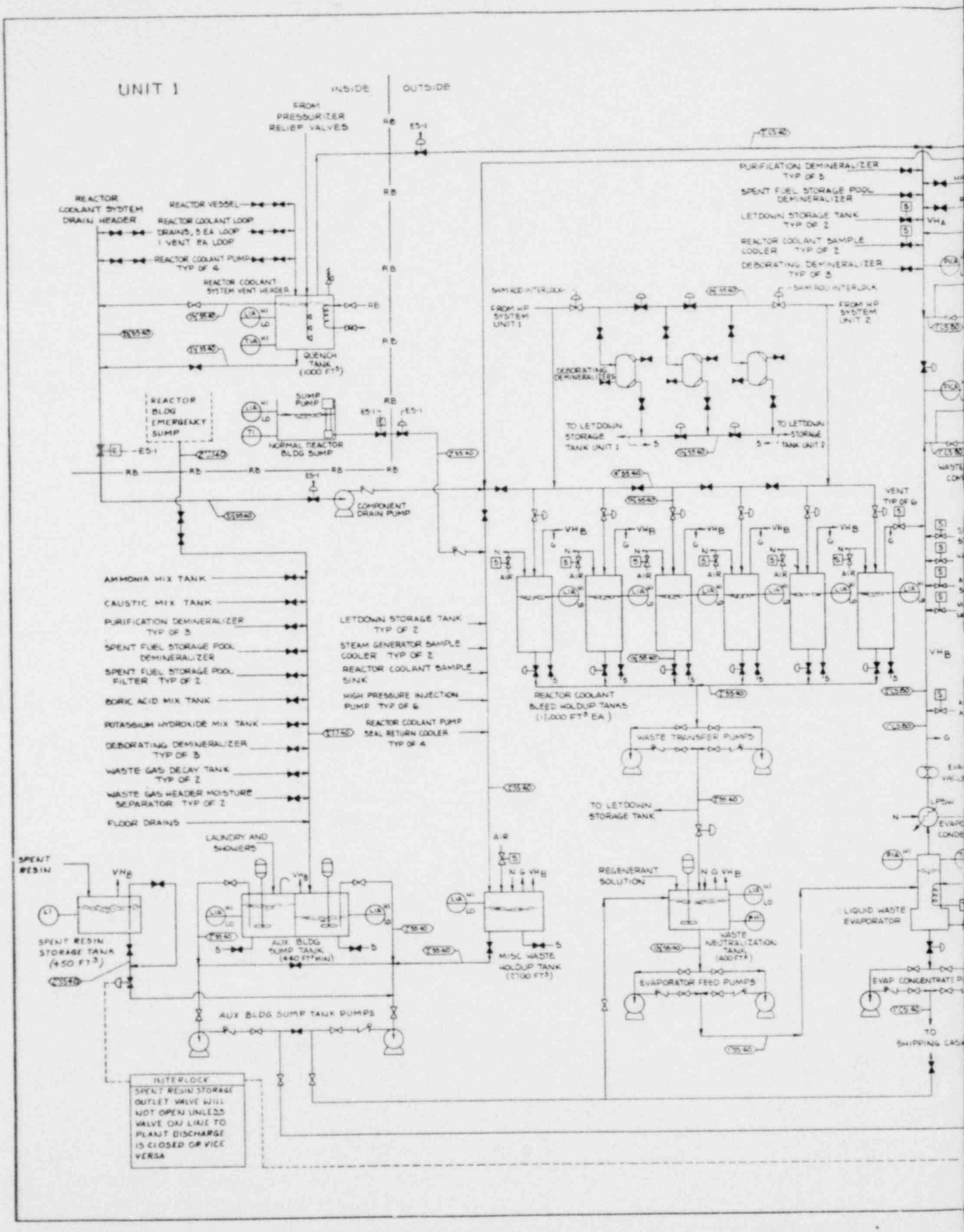
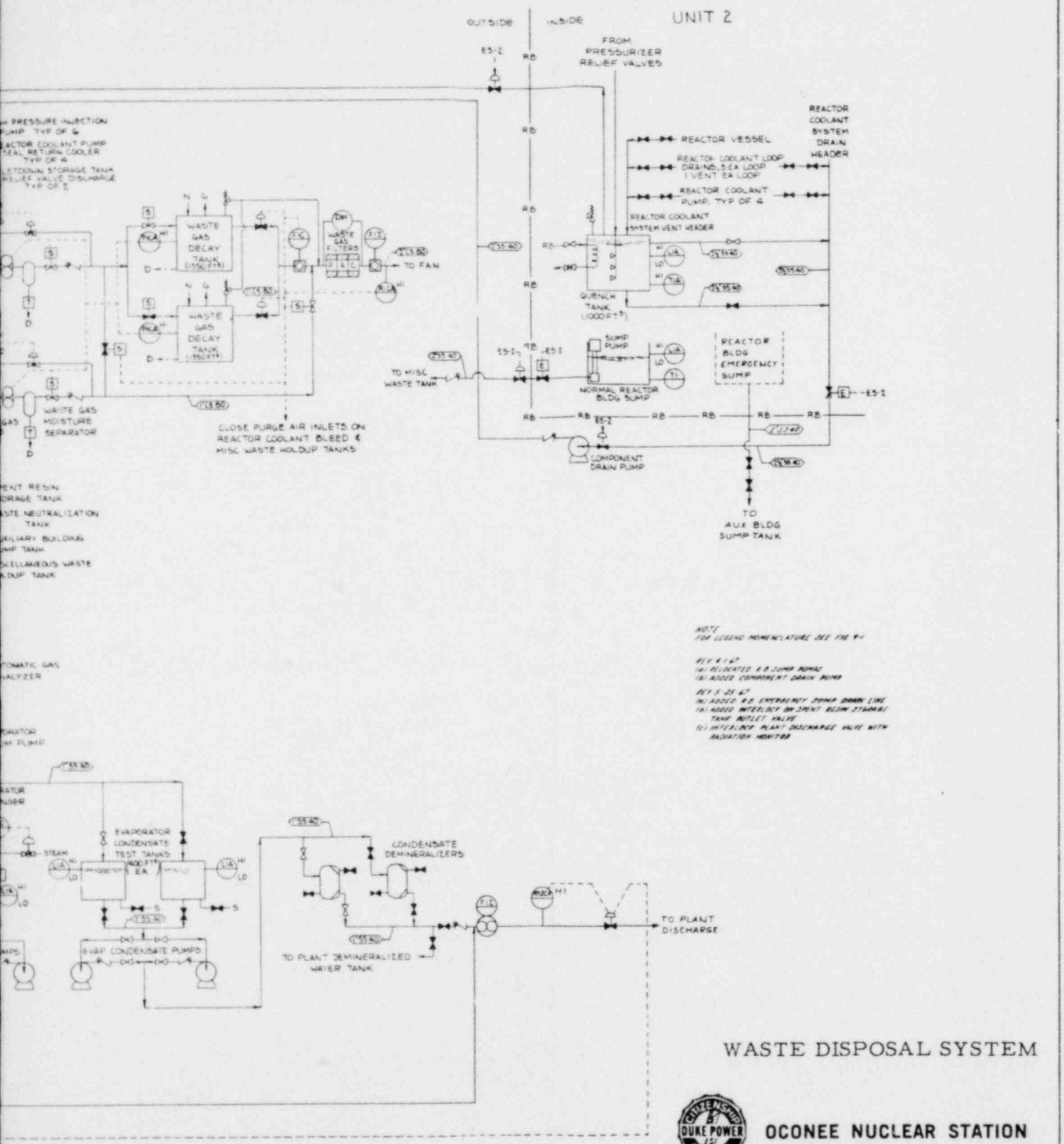

UNIT 1
INSIDE
OUTSIDE
FROM PRESSURIZER RELIEF VALVES
REACTOR COOLANT SYSTEM DRAIN HEADER
REACTOR VESSEL
REACTOR COOLANT LOOP DRAINS, 3 EA LOOP 1 VENT EA LOOP
REACTOR COOLANT PUMP TYP OF 4
REACTOR COOLANT SYSTEM VENT HEADER
QUENCH TANK (1000 FT³)
REACTOR BLDG EMERGENCY SUMP
SUMP PUMP
NORMAL REACTOR BLDG SUMP
COMPONENT DRAIN PUMP
PURIFICATION DEMINERALIZER TYP OF 3
SPENT FUEL STORAGE POOL DEMINERALIZER
LETDOWN STORAGE TANK TYP OF 2
REACTOR COOLANT SAMPLE COOLER TYP OF 2
DEBORATING DEMINERALIZER TYP OF 3
SHIM ROD INTERLOCK
FROM HP SYSTEM UNIT 1
FROM HP SYSTEM UNIT 2
DEBORATING DEMINERALIZERS
TO LETDOWN STORAGE TANK UNIT 1
TO LETDOWN STORAGE TANK UNIT 2
VENT TYP OF 6
AMMONIA MIX TANK
CAUSTIC MIX TANK
PURIFICATION DEMINERALIZER TYP OF 3
SPENT FUEL STORAGE POOL DEMINERALIZER
SPENT FUEL STORAGE POOL FILTER TYP OF 2
BORIC ACID MIX TANK
POTASSIUM HYDROXIDE MIX TANK
DEBORATING DEMINERALIZER TYP OF 3
WASTE GAS DECAY TANK TYP OF 2
WASTE GAS HEADER MOISTURE SEPARATOR TYP OF 2
FLOOR DRAINS
LETDOWN STORAGE TANK TYP OF 2
STEAM GENERATOR SAMPLE COOLER TYP OF 2
REACTOR COOLANT SAMPLE SINK
HIGH PRESSURE INJECTION PUMP TYP OF 6
REACTOR COOLANT PUMP SEAL RETURN COOLER TYP OF 4
REACTOR COOLANT BLEED HOLDUP TANKS (11,000 FT³ EA)
WASTE TRANSFER PUMPS
TO LETDOWN STORAGE TANK
REGENERANT SOLUTION
WASTE NEUTRALIZATION TANK (400 FT³)
EVAPORATOR FEED PUMPS
LIQUID WASTE EVAPORATOR
EVAP CONCENTRATE PUMPS
TO SHIPPING CASK
SPENT RESIN
SPENT RESIN STORAGE TANK (450 FT³)
LAUNDRY AND SHOWERS
AUX BLDG SUMP TANK (440 FT³ MIN)
AUX BLDG SUMP TANK PUMPS
MISC WASTE HOLDUP TANK (2700 FT³)
INTERLOCK
SPENT RESIN STORAGE OUTLET VALVE WILL NOT OPEN UNLESS VALVE ON LINE TO PLANT DISCHARGE IS CLOSED OR VICE VERSA



WASTE DISPOSAL SYSTEM

OCONEE NUCLEAR STATION

FIGURE 11-1