11. RADIOACTIVE WASTES AND RADIATION PROTECTION

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11. RADIOACTIVE WASTES AND RADIATION PROTECTION

11.1 RADIOACTIVE WASTE HANDLING

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 ensure that plant 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 11.1-1.

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 percent 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 percent 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, Xe, Mo and Y, and a 99 percent removal efficiency for all other nuclides. All Te is assumed to plate out on the system surface. Activity levels are relatively insensitive to small changes in demineralizer efficiencies, e.g., use of 90 percent instead of 99 percent would result in only about a 10 percent 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 11.1-2.

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 for a unit containing one percent defective fuel during full power operation at the end of the second core cycle are shown in Table 11.1-3.

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. 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 plant. These include such items as coolant waste tanks, and the makeup tank which are vented to the waste gas disposal system. The activity of the gases is dependent upon the liquid activity.

11.1.1.4 Disposal Methods

Liquid wastes from the plant will be handled in two separate streams using two evaporator chains. Reactor coolant bleed will be fed through one chain; and miscellaneous wastes which may contain dirt, oil, or chemicals will be processed through the other chain. The treatment of the wastes will be in one of the following ways:

a. Reactor coolant wastes will be de-gassed by flashing, ion-exchanged, stored, and separated into concentrated boric acid and coolant make-up, both of which will be re-used.

b. Miscellaneous liquid wastes will be collected and evaporated, with the condensate re-used and the bottoms drummed for off-site disposal by an AEC-licensed disposal contractor.

TABLE 11.1-1
RADIOACTIVE WASTE QUANTITIES

Waste Source	Quantity Per Year	Assumptions and Comments
Liquid Waste:		
Reactor Coolant System:		
Startup Expansion	128,000 gal	4 cold startups
Startup Dilution	88,000 gal	2 cold startups at beginning of life and 1 cold startup at 100 and 200 full power days respectively

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TABLE 11.1-1 continued

Waste Source	Quantity Per Year	Assumptions and Comments
Lifetime Shim Bleed	176,000 gal	Dilution from 1,460 to 175 ppm
System Drain	45,000 gal	Drain to level of outlet nozzles for refueling operations
Sampling and Laboratory Drains	22,500 gal	12 samples per week at 5 gal per sample
Purification Demineralizer Sluice	160 ft^3	80 ft^3/year replacement at 2 ft^3/ft^3 resin sluice
Spent Fuel Pool Demineralizer Sluice	42 ft^3	21 ft^3/year replacement at 2 ft^3/ft^3 resin sluice
Deborating Demineralizer Regeneration and Rinse	2,500 ft^3	1 regeneration per year per demineralizer at 20 ft^3/ft^3 resin regeneration
Miscellaneous System Leakage	45,000 gal	5 gph leakage
Showers	110,000 gal	10 showers per day at 30 gal per shower
Gaseous Waste*:		
Off-Gas from Reactor Coolant System	1,350 ft^3	Degas at 25ccH_2 per liter concentration
Off-Gas from Liquid Sampling	74 ft^3	Degas at 25ccH_2 per liter concentration
Off-Gas from Makeup Tank	900 ft^3	Vent once per year
Off-Gas from Pressurizer	60 ft^3	Vent once per year
Solid Waste:		
Purification Resin	80 ft^3	Resin replacement once per year

*Excludes reactor building and plant ventilation

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TABLE 11.1-1 continued

Waste Source	Quantity Per Year	Assumptions and Comments
Spent Fuel Pool Ion-Exchanger Resin	20 ft^3	Resin replacement once per year
Evaporator Condensate Ion-Exchanger Resin	2 ft^3	Resin replacement once per year
Evaporator Bottoms	800 ft^3	Concentrated to 20 percent solids
Radwaste Ion-Exchanger Resin	100 ft^3	Four replacement beds per year

TABLE 11.1-2
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 11.1-3
REACTOR COOLANT ACTIVITIES FOR A UNIT CONTAINING
ONE PERCENT DEFECTIVE FUEL

Isotope	Activity, μCi/ml	Isotope	Activity, μCi/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	5.5
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	0.89
Xe 135	9.4	Y 91	0.29
Xe 138	0.5	Ce 144	0.0027

c. Gaseous wastes can be compressed and stored for decay if activity is high, with later release at controlled rates through filters to the environmental stack discharge, or can be diverted to the filters and stack without hold-up where activity is sufficiently low.

d. Solid wastes will be accumulated and packaged in suitable drums for later removal by an AEC-licensed disposal contractor.

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 nominal 1 percent failed fuel. All components will be located in the auxiliary building. The shield design criteria for the auxiliary building is Zone II on controlled areas and Zone III in areas requiring limited access. The piping and equipment of the waste disposal system will be shielded by concrete walls and floors of varying thicknesses depending on the strength of the radiation sources therein and on the access requirements in a particular area. In some areas local shielding in the form of removable lead or concrete blocks may be utilized to facilitate maintenance or operations.

11.1.2 SYSTEM DESIGN AND EVALUATION

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 reactor coolant drains, and the other will handle all miscellaneous liquid wastes. The conceptual system flow diagram is shown in Figure 11.1-1 and 11.1-2.

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 of operational requirements for reducing or increasing reactor coolant boric acid content. After flashing and ion-exchange, it will be stored in reactor coolant waste receiver tanks and holdup tanks for concentrator feed or will be passed through deborating ion-exchangers for coolant make-up. The deborating ion-exchangers will be used only for boric acid concentrations below 1000 ppm in order to limit the frequency at which resins must be regenerated.

The reactor coolant waste receiver and holdup tanks will be sized to contain one reactor coolant system volume each. The contents of these tanks will be pumped to the concentrator. When the coolant bleed has been sufficiently concentrated, the bottoms will be pumped to the concentrated boric acid storage tank. The condensates will be collected in the concentrator condensate tank where they will be sampled to determine quality and activity level. Condensates will then be pumped through the deborating ion-exchanger to the demineralized water storage tank for re-use.

The second evaporator chain will process liquid wastes collected by the miscellaneous wastes tank and auxiliary building sump.

The miscellaneous liquid wastes will be treated as necessary to prevent foaming and samples will be taken to determine activity. Wastes will then be transferred to the miscellaneous wastes evaporator. When wastes are sufficiently concentrated, the concentrates will be collected and subsequently pumped to the waste drumming area for packaging and disposal by an AEC-licensed disposal contractor. Condensate will be collected in the evaporator condensate tank, sampled for activity, and subsequently re-used as demineralized water after passing through the evaporator condensate demineralizer.

Both evaporator chains will be designed to process wastes at a rate well in excess of the expected waste accumulation rate. As indicated above, no liquid waste will be discharged to the environment.

11.1.2.2 Solids Waste Disposal System

Evaporator concentrate will be pumped into a shipping container for off-site disposal. Spent resins from the demineralizers and ion-exchangers will be sluiced to a spent resin storage tank which will hold one complete charge of resins from the reactor auxiliary systems. Spent resin will be transferred from the storage tank to special casks or drums for disposal. Other miscellaneous solid wastes such as filters, clothing, laboratory equipment, pieces of metal, and paper will be disposed of by the use of a baler and light metal shipping containers. All solid wastes will be transported off-site in approved containers by an AEC-licensed disposal contractor.

11.1.2.3 Gaseous Waste Disposal System

Gaseous radioactive wastes are collected mainly at the flash tank and at the reactor coolant system drain tank during operation and at the purification system make-up tank during reactor shutdown degasification. Gas from each of these locations is piped to the waste gas surge tank, from where it can be either released directly through high efficiency particulate and charcoal filters to the plant vent or can be compressed and stored in the waste gas decay tanks. In the waste gas decay tanks, gases are monitored for activity, held for decay as required, and then released at a controlled rate through the plant vent. A monitor located in the gaseous discharge line will be equipped with an indicator alarm to annunciate high activity. The high activity alarm actuates an interlock to stop the discharge of these gaseous effluents from the waste gas system. Gases, diffusing from liquids at other collection tanks will be swept by continuous air purge streams into the gas discharge header and through filters to the plant vent. Because of the inherently slow nature of the diffusion process through liquids, this gas release will not cause significant ground level concentrations.

11.1.2.4 Process System Radiation Monitoring

The component cooling water system which removes heat from potentially radioactive sources will be monitored to detect accidental releases. Monitors will be provided in the component cooling water system which serves the reactor coolant pump seal return coolers, spent fuel cooler, sample coolers, pressurizer relief tank cooling coils, decay heat removal coolers, and letdown coolers. A high radiation alarm will alert the operator, and the leaking heat exchanger can be isolated.

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 plant vent will be monitored for air particulate, gaseous activity, and iodine activity.

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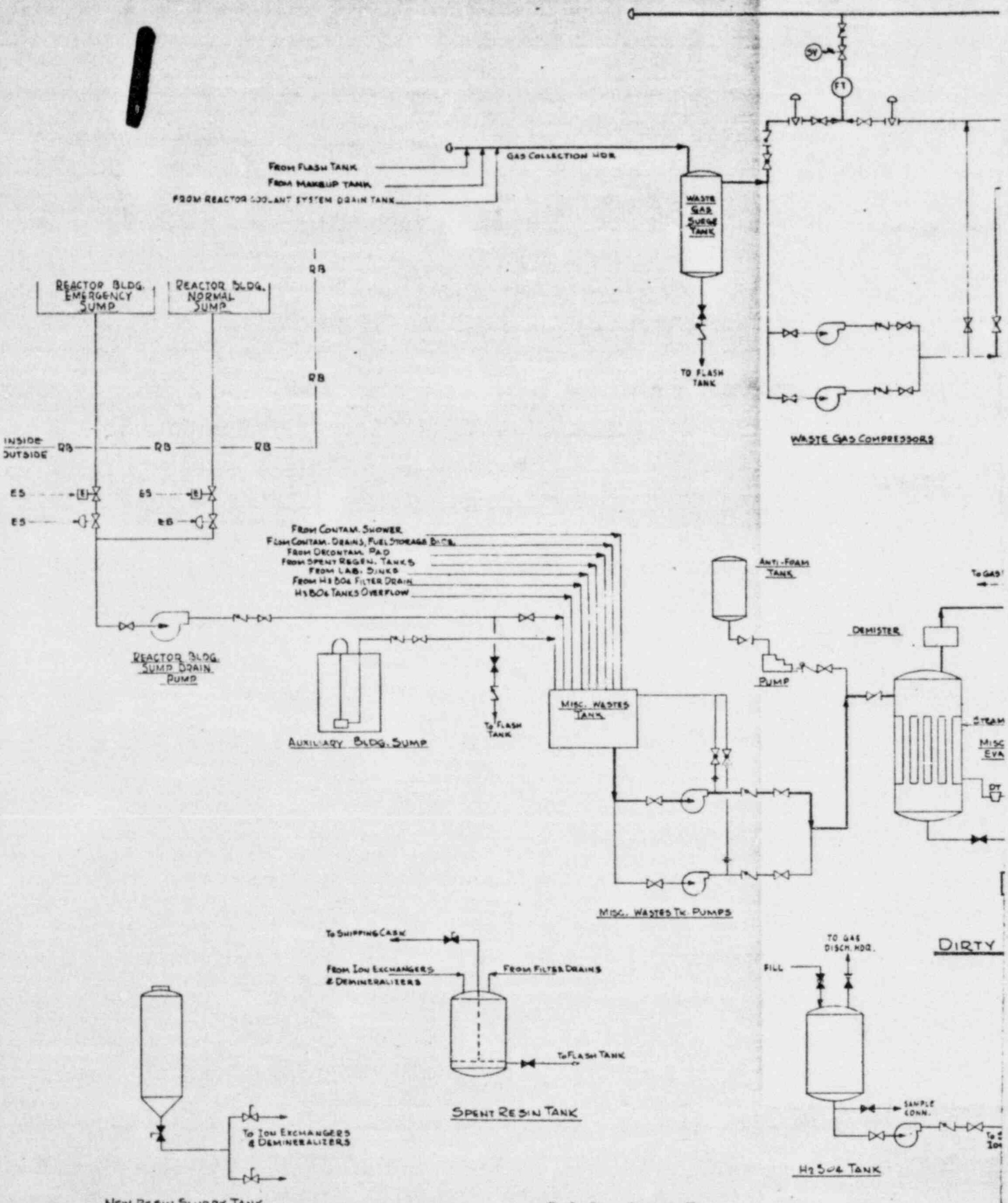
These radiation monitors are commercially-available equipment. The required characteristics will be established during detailed plant design. The minimum sensitivity of detectors, when combined with appropriate dilution factors, will ensure safe limits of release.

11.1.2.5 Design Evaluation

The possibility of a significant activity release off the site from 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 of by being removed off-site by an AEC-licensed disposal contractor. No liquids or solids will be discharged to the environment. Boric acid and coolant will be purified and re-used.

11.1.3 TESTS AND INSPECTIONS

Functional operational tests and inspections of the waste disposal system will be made as required to ensure performance consistent with the requirements of 10 CFR 20.



SV
FT
FROM FLASH TANK
FROM MAKEUP TANK
FROM REACTOR COOLANT SYSTEM DRAIN TANK
GAS COLLECTION HDR
WASTE GAS SURGE TANK
TO FLASH TANK
WASTE GAS COMPRESSORS
RB
REACTOR BLDG. EMERGENCY SUMP
REACTOR BLDG. NORMAL SUMP
INSIDE
OUTSIDE
ES
FROM CONTAM. SHOWER
FROM CONTAM. DRAINS, FUEL STORAGE BLDG.
FROM DECONTAM. PAD
FROM SPENT REGEN. TANKS
FROM LAB. SINKS
FROM H3 BO3 FILTER DRAIN
H3 BO3 TANKS OVERFLOW
REACTOR BLDG. SUMP DRAIN PUMP
AUXILIARY BLDG. SUMP
TO FLASH TANK
MISC. WASTES TANK
ANTI-FOAM TANK
PUMP
DEMISTER
TO GAS
STEAM
MISC
EVA
DT
MISC. WASTES TK. PUMPS
TO SHIPPING CASK
FROM ION EXCHANGERS & DEMINERALIZERS
FROM FILTER DRAINS
TO FLASH TANK
SPENT RESIN TANK
TO GAS DISCH. HDR.
FILL
SAMPLE CONN.
H2 SO4 TANK
DIRTY
TO ION EXCHANGERS & DEMINERALIZERS
NEW RESIN SLURRY TANK

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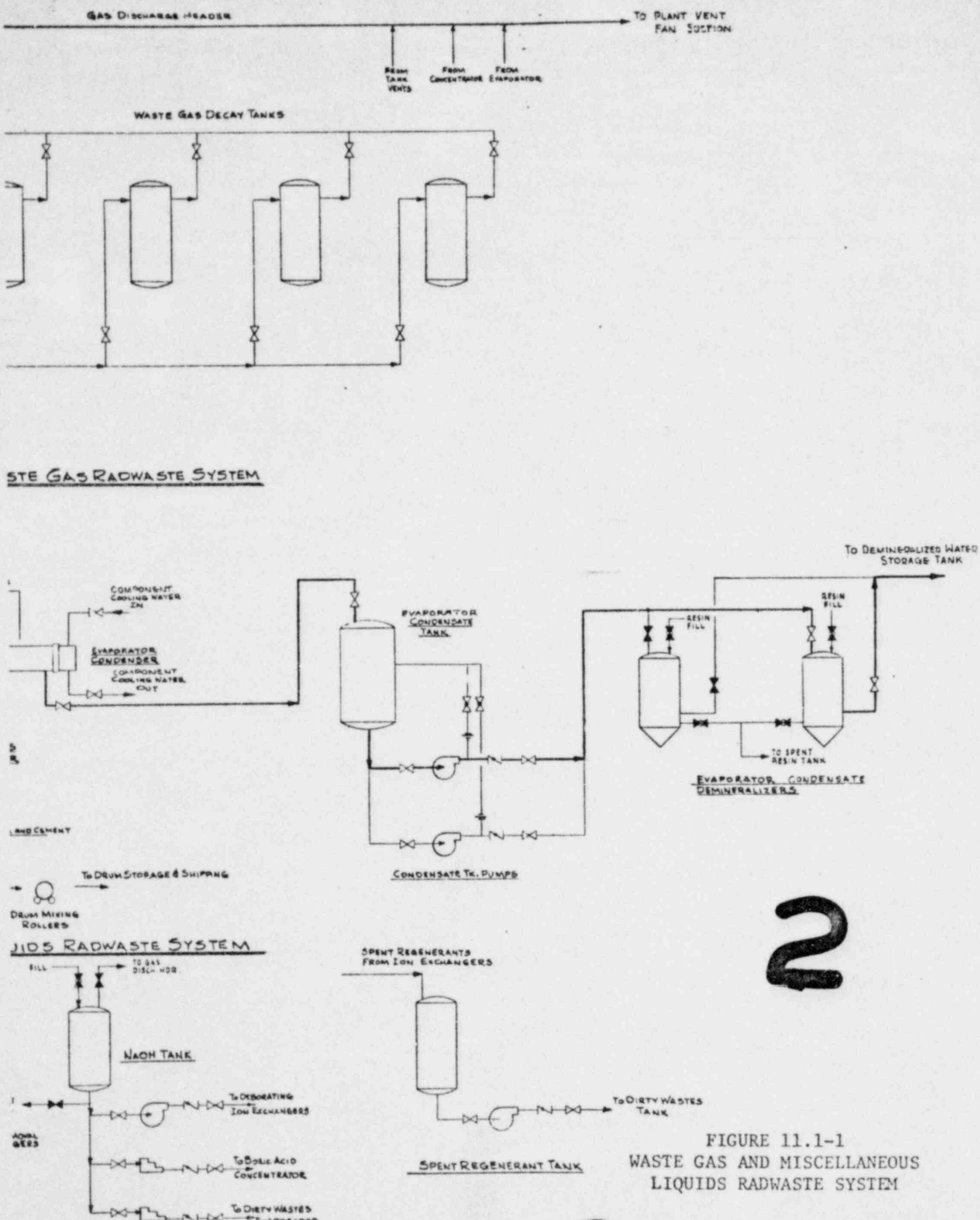

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FIGURE 11.1-1
WASTE GAS AND MISCELLANEOUS
LIQUIDS RADWASTE SYSTEM

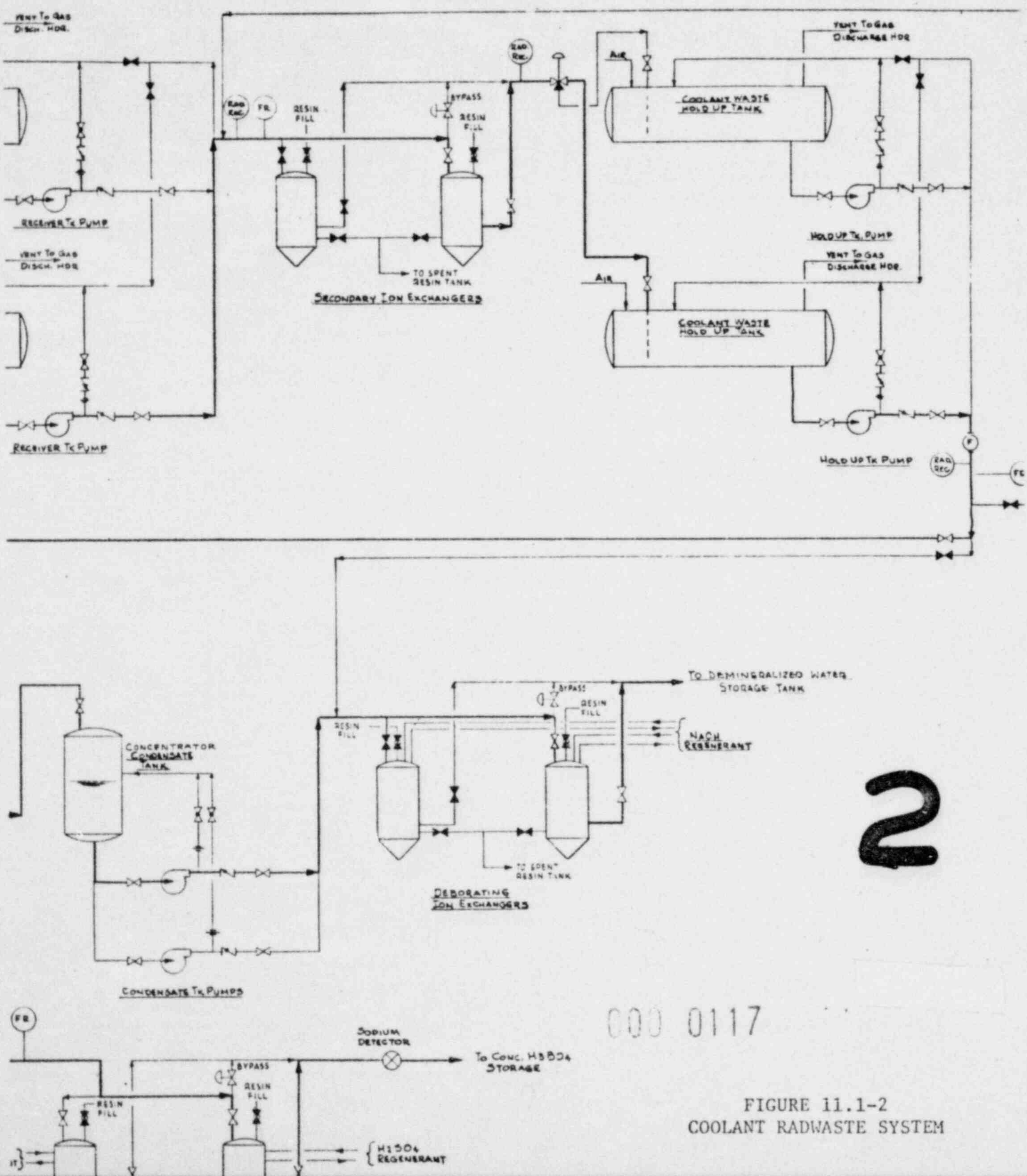

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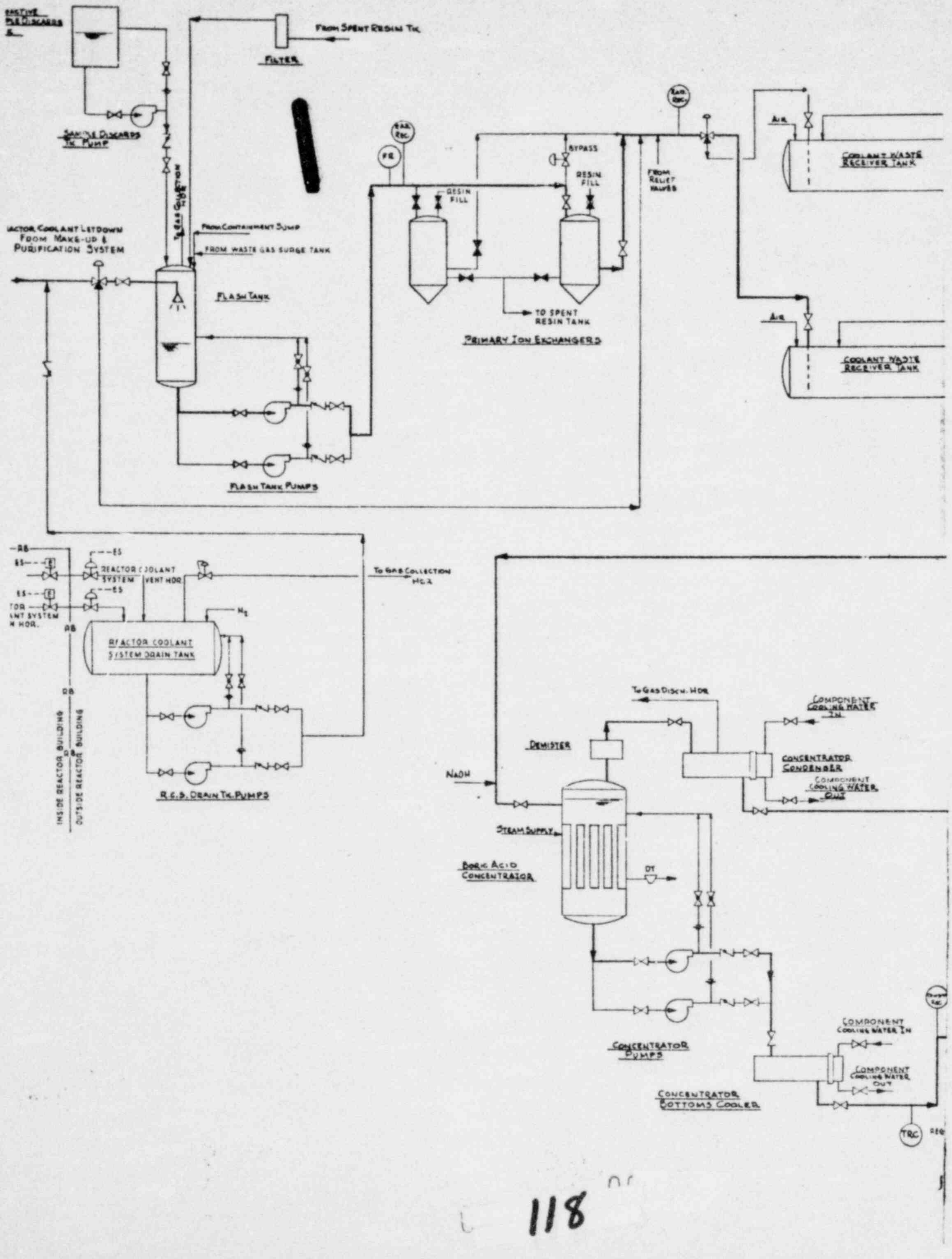
FIGURE 11.1-2
COOLANT RADWASTE SYSTEM


FROM SPENT RESIN TK.
FILTER
SAMPLE DISCARDS TK. PUMP
TO GAS COLLECTION HDR
FROM CONTAINMENT SUMP
FROM WASTE GAS SURGE TANK
COOLANT LETDOWN FROM MAKE-UP & PURIFICATION SYSTEM
FLASH TANK
FR
BAR. REC.
RESIN FILL
BYPASS
RESIN FILL
FROM RELIEF VALVES
TO SPENT RESIN TANK
PRIMARY ION EXCHANGERS
AIR
COOLANT WASTE RECEIVER TANK
AIR
COOLANT WASTE RECEIVER TANK
FLASH TANK PUMPS
RB
ES
REACTOR COOLANT SYSTEM VENT HDR.
ES
TO GAS COLLECTION HDR.
N_2
REACTOR COOLANT SYSTEM DRAIN TANK
R.C.S. DRAIN TK. PUMPS
INSIDE REACTOR BUILDING
OUTSIDE REACTOR BUILDING
TO GAS DISCH. HDR
COMPONENT COOLING WATER IN
DEMISTER
CONCENTRATOR CONDENSER
COMPONENT COOLING WATER OUT
NAOH
STEAM SUPPLY
BORIC ACID CONCENTRATOR
DT
CONCENTRATOR PUMPS
COMPONENT COOLING WATER IN
COMPONENT COOLING WATER OUT
CONCENTRATOR BOTTOMS COOLER
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11.2 RADIATION SHIELDING

11.2.1 PRIMARY, SECONDARY, REACTOR BUILDING, AND AUXILIARY SHIELDING

11.2.1.1 Design Criteria

Plant operating personnel must be protected by radiation shielding wherever a potential radiation hazard may exist. The shielding must perform two primary functions: ensure that under normal operation the potential radiation dose to operating personnel and the general public is within the limits of 10 CFR 20, and ensure that operating personnel are adequately protected in the event of a power plant accident so that the accident can be terminated without undue hazard to the general public.

All plant areas which can be occupied by personnel are classified according to anticipated access, as follows:

- Zone I - Normal continuous occupancy with radiation levels not exceeding 1 mrem/hr.
- Zone II - Periodic occupancy with radiation levels not exceeding 2.5 mrem/hr.
- Zone III - Limited occupancy with radiation levels not exceeding 15 mrem/hr., when occupied.
- Zone IV - Restricted occupancy consistent with measured radiation levels exceeding 15 mrem/hr.

The turbine structure, offices, turbine plant service areas, and the control room areas are designated Zone I. Areas such as the local control space in the auxiliary building and the waste disposal area, operating deck of the spent fuel storage building, and the operating deck of the reactor plant during shutdown would generally be designated Zone II. Intermittently occupied work areas are designated ás Zone III. Typical Zone IV areas include, reactor loop areas after shutdown, drumming areas, and
makeup and purification tank areas. The reactor building is accessible 1
for limited times in certain areas such as the in-core instrumentation control centers during normal plant operation.

The radiation sources which provide the basis for the shield design are sub-divided into four categories according to their origin or location.

a. The reactor core, internals, and reactor vessel.

b. The reactor coolant.

c. Auxiliary systems equipment.

d. Radioactive materials released during accidents.

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The radiation emanating from the reactor vessel consists of neutrons leaking from the core, prompt fission gammas, fission product gammas, and gammas resulting from the interaction of neutrons with steel and water.

Nitrogen 16 is the major radiation source in the reactor coolant during normal operation and it establishes the secondary shield thickness. The shutdown radiation levels in the reactor loop areas are established by activated corrosion products and by assuming that 1% of the reactor fuel is defective allowing fission products to escape into the coolant.

The sources in the auxiliary systems and waste disposal systems are established by both the corrosion product activities and the assumed fission product activities from 1% defective fuel circulating in the reactor coolant.

The accident sources are established on the basis of the fission products released to the reactor containment following a hypothetical loss of reactor coolant, coupled with a failure of the safety injections system and subsequent core melting.

11.2.1.2 Description of Shielding

11.2.1.2.1 Primary Shield

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

11.2.1.2.2 Secondary Shield

The secondary shield will be a reinforced concrete structure surrounding the reactor coolant equipment, including piping, pumps, and steam generators. This shield will protect personnel from the direct gamma radiation resulting from reactor coolant activation products and fission products

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carried away from the core by the reactor coolant. In addition, the secondary shield will supplement the primary shield by attenuating neutron and gamma radiation escaping from the primary shield. The secondary shield will be sized to allow limited access to the reactor building during full power operation. The preliminary thickness of secondary shield walls will be approximately 4.0 feet.

11.2.1.2.3 Reactor Building Shield

The reactor building shield will be a reinforced, prestressed concrete containment structure which completely surrounds the nuclear steam supply system. At full power operation, this shield will attenuate any radiation escaping from the primary-secondary shield complex such that radiation levels outside the reactor building will be less than 1 mrem/hr. In addition, the reactor building structure will shield personnel from radiation sources inside the reactor building following a maximum hypothetical accident (MHA). The shielding will be of sufficient thickness to allow personnel a reasonable time period in which to evacuate the immediate vicinity of the reactor building following the MHA without excessive radiation exposure. The curves in Section 14 (Safety Analysis) indicate an integrated direct dose of less than 1 mrem over a period of two hours immediately outside the reactor building following the MHA. Preliminary thicknesses of the reactor building wall and dome are 3.75 ft and 3.25 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 1 ft. This ensures that the integrated whole body dose over 90 days following the MHA will not exceed 25 rems. Ventilation of the control room under post-accident conditions will be controlled as described in Section 9.7.2.

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 concentrator and miscellaneous waste evaporators, makeup tank, waste gas decay tanks, demineralizers, makeup pumps, waste drumming area, reactor coolant system drain tank, and reactor building sump pump.

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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 tube, 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. Minimum allowed water depths during handling are approximately 10 feet in the reactor cavity and fuel transfer canal and approximately 13 feet over stored assemblies in the spent fuel storage area. The dose rates at the water surface with a minimum water coverage of 10 feet will be less than 15 mrem/hr. While 13 feet of water limits the dose rate to less than 2.5 mrem/hr at the surface. 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 refueling structure, give support for the reactor coolant components under pipe rupture conditions, and provide missile shielding, they will be reinforced and designed to be self-supporting.

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

11.2.1.3 Evaluation

11.2.1.3.1 Radiation Sources

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

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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

Source magnitudes are determined for the reactor operating at the maximum expected power level of 2568 Mwt with reactor coolant activity levels corresponding to 1 percent failed fuel. Gamma-ray yield and spectral distributions from prompt fission and gross fission product activity are based on 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 B&W code which computes the integral of the 0-16 (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 0-16 (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 Neutron and Gamma Shield

The preliminary estimates for primary shielding requirements will be subsequently confirmed by Bechtel's diffusion and transport computer programs.

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 percent of the gases, 50 percent of the halogens, and 1 percent 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 with the GRACE II computer code as a function of distance from the reactor building. The results are given in Section 14.

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 will be designed to indicate and
alarm high radiation monitoring levels throughout the plant. An audible/
visible alarm at both the detector location and the control room will be
1 provided. Recorded presentation may be required. All instrumentation
for the radiation monitoring system will obtain its voltage supply from
the 120 volt a-c essential service buses and each detector will have a
loss-of-power alarm. The normal high radiation alarm setpoint will be
10 percent 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

Beta-gamma detectors are located as follows.

a. One portal monitor and one hand and foot monitor at control access door

b. Inside the reactor building near the personnel access hatch

c. Near incore instrument space inside the reactor building

d. Fuel handling bridge in spent fuel building

e. Auxiliary building sump pump area

f. Auxiliary building near sample sink

g. Steam generator blowdown

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h. Auxiliary building in decay heat cooler area

i. Near component cooling water heat exchangers

j. Radio-chemistry laboratory

k. Primary loop

l. Control room

m. Three battary powered monitors in reactor building

n. Condenser air ejector

Air particulate and radio gas detectors to be mounted in the following places.

a. Plant vent

b. Inside reactor building

c. Radio-chemistry laboratory

d. Plant boundary

e. Control room and auxiliary building

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 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 plant superintendent is responsible for radiation protection and contamination control. All personnel assigned to the plant and all visitors will be required to follow rules and procedures established by administrative control for protection against radiation and contamination.

Under supervision of the plant superintendent, the administration of the radiation protection program will be the responsibility of the radiation protection engineer. It will be the responsibility of the health physics section to train plant 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 plant operation and maintenance to assure safe conditions and compliance with the 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 plant personnel. The procedures that control radiation exposure will be subject to the same review and approval as those that govern all other plant procedures (see Section 12.6, 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.

11.2.3.1 Radiation Work Permits

A radiation work permit shall be obtained by all personnel prior to entering a control area or performing any work on radioactive or contaminated material or equipment.

In the event that the safety of the plant or its personnel are endangered, entry may be made into a control area simultaneously with monitoring personnel. A radiation work permit shall be completed as soon as possible after correction of the situation.

Radiation work permits shall be issued routinely by the shift supervisor. These permits shall show:

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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 shift supervisor.

e. Signature of an individual from the health physics groups 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.

All such permits shall be filed with the Health Physics group for future reference.

11.2.3.2 Personnel Monitoring System

The personnel monitoring program shall ensure 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 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 plant 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 plant 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 group 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.

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11.2.3.3 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 group, and personnel will be trained in the use of this equipment before using it in the performance of work.

11.2.3.4 Change Room Facilities

Change room facilities will be provided where personnel may obtain clean protective clothing required for plant 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.

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 plant 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 plant 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.

11.2.3.5 Health Physics Facilities

The plant will include a health physics facility 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 group.

11.2.3.6 Health Physics Instrumentation

Portable radiation survey instruments will be provided for use by the health physics group 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 plant. 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 plant.

The plant 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.

11.2.3.7 Medical Programs

Facilities for screening personnel for contamination will be available on site with outside services utilized as backup and support for this program. A medical examination program appropriate for radiation workers will be conducted to establish and maintain records of the physical status of each employee. 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 group will be responsible for the program and will assist the physicians to preserve the health of the employees concerned and to confirm the radiation control methods employed at the plant.

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11.3 REFERENCES

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