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Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors

PWR-GALE Code

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ABSTRACT

This report revises the original issuance of NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE-Code)" (April 1976), to incorporate more recent operating data now available as well as the results of a number of in-plant measurement programs at operating pressurized water reactors. The PWR-GALE Code is a computerized mathematical model for calculating the releases of radioactive material in gaseous and liquid effluents (i.e., the gaseous and liquid source terms). The U.S. Nuclear Regulatory Commission uses the PWR-GALE Code to determine conformance with the requirements of Appendix I to 10 CFR Part 50.

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EXECUTIVE SUMMARY

The average quantity of radioactive material released to the environment from a nuclear power reactor during normal operation including anticipated operational occurrences is called the "source term,"* since it is the source or initial number used in calculating the environmental impact of radioactive releases. The PWR-GALE (Pressurized Water Reactor - Gaseous and Liquid Effluents) Code is a computerized mathematical model for calculating the releases of radioactive material in gaseous and liquid effluents (i.e., the gaseous and liquid source terms) from pressurized water reactors. The calculations are based on data generated from operating reactors, field and laboratory tests, and plant-specific design considerations incorporated to reduce the quantity of radioactive materials that may be released to the environment during normal operation, including anticipated operational occurrences.

The U.S. Nuclear Regulatory Commission uses the PWR-GALE Code to determine conformance with the requirements of Appendix I to 10 CFR Part 50. The first issue of this NUREG report was published in April 1976. In order to use the best available data for improving the calculational models used by the Commission staff to determine conformance with Appendix I to 10 CFR Part 50, Revision 1 is being issued to update NUREG-0017. This revision incorporates more recent operation data now available and also incorporates the results of a number of in-plant measurement programs at operating pressurized water reactors.

Chapter 1 of this report gives a step-by-step procedure for using the PWR-GALE Code along with a description of the parameters which have been built into the Code for use with all PWR source term calculations. These parameters, which apply generically to all PWR's, have been incorporated into the Code to eliminate the need for their entry on input data cards. Other parameters are required to be entered on input cards used by the Code. Explanations of the data require, along with acceptable means for calculating such data, are given for each input data card.

Descriptions of the principal parameters used in source term calculations and explanations of the bases for each parameter are given in Chapter 2. The parameters have been derived from reactor operating experience where data were available. Where operating data were inconclusive or not available, information was drawn from laboratory and field tests and from engineering judgment. The bases for the source term parameters explain the reasons for choosing the numerical values listed. A list of references used in developing the parameters is also included. The source term parameters used are believed to provide a realistic assessment of reactor and radwaste system operation.

* "Source term" as discussed in this report differs from "accident source term," which deals with potential releases resulting from nuclear reactor accidents.

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Chapter 3 contains sample input data together with an explanation of the input to orient the user in making the required entries. Also included is a listing of the input data for a sample problem, a discussion of the nuclear data library used, and a FORTRAN listing of the PWR-GALE Code.

Chapter 4 lists the information needed to generate source terms for PWR's. The information is proved by the applicant and is consistent with the contents of the Safety Analysis Report (ER) of the proposed PWR. This information constitutes the basic data required in calculating the releases of radioactive material in liquid and gaseous effluents.

CHAPTER 1. PWR-GALE CODE

1.1 INTRODUCTION

In promulgating Appendix I to 10 CFR Part 50, the U. S. Nuclear Regulatory Commission indicated its desire to use the best available data for improving the calculational models used by the Commission Staff to determine conformance with the requirements of the regulation. The first issue of this NUREG Report was published in April 1976. Revision 1 is being issued to update NUREG-0017 by incorporating more recent operating data now available and also by incorporating the results of a number of in-plant measurement programs at operating pressurized water reactors (PWR's).

The PWR-GALE (Pressurized Water Reactor - Gaseous and Liquid Effluents) Code is a computerized mathematical model for calculating the releases of radioactive material in gaseous and liquid effluents from pressurized water reactors. The calculations are based on data generated from operating reactors, field and laboratory tests, and plant-specific design considerations incorporated to reduce the quantity of radioactive materials that may be released to the environment during normal operation, including anticipated operational occurrences.

The average quantity of radioactive material released to the environment from a nuclear power reactor during normal operation is called the "source term" since it is the source or initial number used in calculating the environmental impact of radioactive releases. The calculations performed by the PWR-GALE Code are based on (1) American Nuclear Society (ANS) 18.1 Working Group recommendations (Ref. 1) for adjustment factors, (2) the release and transport mechanisms that result in the appearance of radioactive material in liquid and gaseous waste streams, (3) plant-specific design features used to reduce the quantities of radioactive materials ultimately released to the environment, and (4) information received on the operation of nuclear power plants.

In a PWR, primary coolant water circulates through the reactor core where it removes the heat from the fuel elements. In the steam generators, heat from the pressurized primary coolant water is transferred to the secondary coolant water to form steam. The steam expands through the turbine and is then condensed and returned to the steam generators. The primary coolant water flows back to the reactor core. The principal mechanisms that affect the concentrations of radioactive materials in the primary coolant are: (1) fission product leakage to the coolant from defects in the fuel cladding and fission product generation in tramp uranium, (2) corrosion products activated in the core, (3) radioactivity removed in the reactor coolant treatment systems, and (4) activity removed because of primary coolant leakage. These mechanisms are described briefly in the following paragraphs.

The primary coolant is continuously purified by passing a side stream through filters and demineralizers in the reactor coolant treatment systems (RCTS). It is necessary to maintain the purity of the primary coolant to prevent fouling of heat transfer surfaces and to keep releases to the environment as low as is reasonably achievable. Chemicals are added to the primary coolant to inhibit corrosion and/or improve fuel economy. Lithium hydroxide is added for pH control to reduce corrosion.

Water decomposes into oxygen and hydrogen as a result of radiolysis. The control of oxygen concentration in the primary coolant is important for corrosion control. Hydrogen, added to the primary coolant as dissolved free hydrogen, tends to force the net reaction toward the recombination of hydrogen and oxygen to water at an overall rate sufficient to maintain low primary coolant oxygen concentrations.

Boron is added to the primary coolant as a neutron absorber (shim control). As the fuel cycle progresses, boron is removed from the primary coolant through the RCTS loop (shim bleed). The shim bleed is processed through an evaporator, and the boron in the evaporator bottoms is either reused or packaged as solid waste. The evaporator distillate may be recycled to the reactor coolant system as makeup water or discharged to the environment.

Radioactive gases stripped from the primary coolant by degassification are normally collected in pressurized storage tanks and held for radioactive decay prior to recycle or release to the environment. Alternative treatment methods include charcoal delay systems and cryogenic distillation.

Because of leakage through valve stems and pump shaft seals, some coolant escapes into the containment and the auxiliary buildings. A portion of the leakage evaporates, thus contributing to the gaseous source term, and a fraction remains as liquid, becoming part of the liquid source term. The relative amount of leakage entering the gaseous and liquid phases is dependent upon the temperature and pressure at the point where the leakage occurs. Most of the noble gases enter the gas phase, whereas iodine partitions into both phases.

Leakage of primary coolant into the secondary coolant in the steam generator is the only source of radioactivity in the secondary coolant system. Water or steam leakage from the secondary system provides significant inputs to the liquid and gaseous radwaste treatment systems. Steam leakage may be significant to the gaseous source term since the radioactivity released remains in the gas phase.

In a recirculating U-tube steam generator, the nonvolatile radionuclides leaking from the primary coolant concentrate in the liquid phase in the steam generator. The degree of concentration is controlled by the steam generator blowdown rate and condensate demineralizer flow rate.

Since there is no liquid reservoir in a once-through steam generator, the primary coolant leakage boils to steam when it enters the secondary side of the steam generator. Secondary coolant purity is maintained by a condensate demineralizer system and there is no steam generator blowdown. The concentration of radioactivity in the secondary coolant is controlled by the condensate demineralizer flow rate.

Sources of radioactive wastes from the secondary system are the offgases from the turbine condenser, vent gases from the turbine gland seal, liquid and vent gases from the steam generator blowdown, and liquid and gaseous leaks into the turbine building. Liquid wastes also originate from the chemical regeneration of condensate demineralizers in feedwater/ condensate systems.

In this chapter, a step-by-step procedure for using the PWR-GALE Code is given along with a description of the parameters which have been built into the Code for use with all PWR source term calculations. These parameters, which apply generically to all PWR's, have been incorporated into the Code to eliminate the need for their entry on input data cards. Other parameters are required to be entered on input data cards used by the Code. Explanations of the data required, along with acceptable means for calculating such data, are given for each input data card. Chapter 2 gives the principal source term parameters developed for use with the PWR-GALE Code and explains the bases for each parameter. Chapter 3 contains a sample data input sheet and a Fortran IV listing of the PWR-GALE Code. Chapter 4 lists the information needed to generate source terms that an applicant is required to submit with the application.

1.2 DEFINITIONS

The following definitions apply to terms used in this report:

Activation Gases: The gases (including oxygen, nitrogen, and argon) that become radioactive as a result of irradiation in the core.

Anticipated Operational Occurrences: Unplanned releases of radioactive materials from miscellaneous actions such as equipment failure, operator error, administrative error, that are not of consequence to be considered an accident.

Chemical Waste Steam: Normally liquids that contain relatively high concentrations of decontaminants, regenerants, or chemical compounds other than detergents. These liquids originate primarily from resin regenerant and laboratory wastes.

Clean Waste System: Normally tritiated, nonaerated, low-conductivity liquids consisting primarily of liquid waste collected from equipment leaks and drains and certain valve and pump seal leakoffs. These liquids originate from systems containing primary coolant and are normally reused as primary coolant makeup water.

Decontamination Factor (DF): The ratio of the initial amount of a nuclide in a stream (specified in terms of concentration or activity of radioactive materials) to the final amount of that nuclide in a stream following treatment by a given process.

Detergent Waste Stream: Liquids that contain detergent, soaps, or similar organic materials. These liquids consist principally of laundry, personnel shower, and equipment decontamination wastes that normally have a low radioactivity content.

Dirty Waste Stream (Floor Drains): Normally nontritiated, aerated, high-conductivity, non-primary-coolant quality liquids collected from building sumps and floor and sample station drains. These liquids are not readily amenable for reuse as primary coolant makeup water.

Effective Full Power Days: The number of days a plant would have to operate 100% licensed power to produce the integrated thermal power output during a calendar year, i.e.,

$$\text{Effective Full Power Days} = \frac{\text{Integrated Thermal Power}}{\text{Licensed Power Level}} = \frac{\sum_i P_i T_i}{P_{total}}$$

where

P_i is the ith power level, in MWt;

P_{total} is the licensed power level, in MWt; and

T_i is the time of operation at power level P_i, in days.

Fission Product: A nuclide produced either by fission or by subsequent radioactive decay or neutron activation of the nuclides formed in the fission process.

Gaseous Effluent Stream: Processed gaseous wastes containing radioactive materials resulting from the operation of a nuclear power reactor.

Liquid Effluent Stream: Processed liquid wastes containing radioactive materials resulting from the operation of a nuclear power reactor.

Partition Coefficient (PC): The ratio of the concentration of a nuclide in the gas phase to the concentration of a nuclide in the liquid phase when the liquid and gas are at equilibrium.

Partition Factor (PF): The ratio of the quantity of a nuclide in the gas phase to the total quantity in both the liquid and gas phases when the liquid and gas are at equilibrium.

Plant Capacity Factor: The ratio of the average net power to the rated power capacity.

Primary Coolant: The fluid circulated through the reactor to remove heat. The primary coolant activity is considered to be constant over a range of power levels, coolant and cleanup flows, and coolant volumes. Radionuclide concentrations given in this NUREG are based on a recent compilation of available operating data. Therefore, the concentration values in NUREG-0017, Rev. 1 differ from the ANSI N237 values (Ref. 1). Provisions are made in the PWR-GALE Code, in accordance with the recommendations of the standard, for adjusting coolant concentrations should the plant be designed to parameters that are outside the ranges considered in the standard. The radionuclide concentrations used are considered to be representative of measured values based on the available operating data. The radionuclides are divided into the following categories:

1. Noble gases
2. Halogens (Br, I)
3. Cs, Rb
4. Water activation products
5. Tritium
6. Other nuclides (as listed in Tables 2-2 and 2-3 of Chapter 2 of this document)

Radioactive Halogens: The isotopes of fluorine, chlorine, bromine, and iodine. The radioactive isotopes of iodine are the key isotopes considered in dose calculations.

Radioactive Noble Gases: The radioactive isotopes of helium, neon, argon, krypton, xenon, and radon, which are characterized by their chemical inactivity. The radioactive isotopes of krypton and xenon are the key elements considered in dose calculations.

Radioactive Release Rate: The average quantity of radioactive material released to the environment from a nuclear power reactor during normal operation, including anticipated operational occurrences.

Secondary Coolant: The coolant converted to steam by the primary coolant in a heat exchanger (steam generator) to power the turbine. The radionuclide concentrations in the secondary coolant are obtained as discussed above in the definition of primary coolant.

Source Term: The calculated average quantity of radioactive material released to the environment from a nuclear power reactor during normal operation, including anticipated operational occurrences. The source term is the isotopic distribution of radioactive materials used in evaluating the impact of radioactive releases on the environment.

Steam Generator Blowdown: Liquid removed from a steam generator in order to maintain proper water chemistry.

Tramp Uranium: The uranium present on the cladding of a fuel rod.

Turbine Building Floor Drains: Liquids of high conductivity and low-level radioactivity primarily resulting from secondary system leakage, steam trap drains, sampling system drainage, and maintenance and waste drains.

1.3 GASEOUS SOURCE TERMS

The following sources are considered in calculating the releases of radioactive materials (noble gases, radioactive particulates, carbon-14, tritium, argon-41, and iodine) in gaseous effluents from normal operation, including anticipated operational occurrences:

1. Waste gas processing system;

2. Steam generator blowdown system;

3. Condenser air ejector exhaust;

4. Containment purge exhaust;

5. Ventilation exhaust air from the auxiliary, and turbine buildings, and the spent fuel pool area; and

6. Steam leakage from the secondary system.

The releases of radioactive materials in gaseous effluents from the following sources are calculated to be less than 1 Ci/yr of noble gases and 10^{-4} Ci/yr of iodine-131. Therefore, the following releases are considered negligible:

1. Steam releases due to steam dumps to the atmosphere and low-power physics testing and

2. Ventilation air from buildings not covered in 5. above.

The calculational model considers inputs to the waste gas processing system from both continuous stripping of the primary coolant during normal operation and from degassing the primary coolant for two cold shutdowns per year. For plants equipped with steam generator blowdown systems, the model considers iodine present in gases leaving the system vent. The PWR-GALE Code calculates the release rates of noble gases and iodine to building atmospheres based on coolant leakage rates to buildings. Radioiodine releases are related to the iodine-131 coolant concentrations for the PWR being evaluated. Particulate release rates are based on measurements at operating PWR's.

Chapter 2 provides iodine and particulate decontamination factors for removal equipment and parameters for calculating holdup times for noble gases and for calculating tritium, argon-41 and carbon-14 releases.

1.4 LIQUID SOURCE TERMS

The following sources are considered in calculating the release of radioactive materials in liquid effluents from normal operation, including anticipated operational occurrences:

1. Processed water generated from the boron recovery system to maintain plant water balance or for tritium control;

2. Processed liquid waste discharged from the dirty waste or miscellaneous waste systems;

3. Processed liquid waste discharged from the steam generator blowdown treatment system;

4. Processed liquid waste discharged from the chemical waste and condensate demineralizer regeneration system;

5. Liquid waste discharged from the turbine building floor drain sumps; and

6. Detergent waste.

The radioactivity input to the liquid radwaste treatment system is based on the flow rates of the liquid waste streams and their radioactivity levels expressed as a fraction of the primary coolant activity (PCA). The PCA is based on the recommendations of the American National Standard (ANSI N237) Source Term Specification (Ref. 1), with the changes as noted in Section 1.2 under the Primary Coolant definition.

Radionuclide removal by the liquid radwaste treatment system is based on the following parameters:

1. Decay during collection and processing and

2. Removal by the proposed treatment systems, e.g., filtration, ion exchange, evaporation, reverse osmosis, and plateout.

For PWR's using a deep-bed condensate demineralizer, the inventory of radionuclides collected on the demineralizer resins is calculated by considering the flow rate of condensate at main steam activity that is processed through the demineralizers and radionuclide removal using the decontamination factors given in Chapter 2. The activity on the condensate demineralizer resins will also include the steam generator blowdown activity if the blowdown is recycled to the condensate demineralizers. The radioactivity content of the demineralizer regenerant solution is obtained by considering that all the radioactivity is removed from the resins at the interval dictated by the regeneration frequency.

Methods for calculating collection and processing times and the decontamination factors for radwaste treatment equipment are given in this chapter. The liquid radioactive source terms are adjusted to compensate for equipment downtime and anticipated operational occurrences.

For plants using an onsite laundry, a standard detergent waste source term, adjusted for the treatment provided, is added to the adjusted source term.

1.5 INSTRUCTIONS FOR COMPLETING PWR-GALE CODE INPUT DATA CARDS

1.5.1 PARAMETERS INCLUDED IN THE PWR-GALE CODE

The parameters listed below are built into the PWR-GALE Code since they are generally applicable to all PWR source term calculations and do not require entry on input data cards.

1.5.1.1 The Plant Capacity Factor

0.80 (292 effective full power days per year).

1.5.1.2 Radionuclide Concentrations in the Primary Coolant, Secondary Coolant, and Main Steam

See Section 2.2.3 of Chapter 2 of this document.

1.5.1.3 Radioiodine Releases from Building Ventilation Systems Prior to Treatment

See Table 1-1. For a discussion of the normalization techniques see Section 2.2.4.

1.5.1.4 Radioactive Particulate Releases from Building Ventilation Systems Prior to Treatment

See Table 1-2.

1.5.1.5 Noble Gas Releases from Building Ventilation Systems

Noble Gas Releases from the containment building are based on a leakage rate of 3%/day of primary coolant noble gas inventory. Releases from the auxiliary building are based on 160 lb/day primary coolant leakage. Releases from the turbine building are based on 1700 lb/hr steam leakage.

TABLE 1-1†*

RADIOIODINE RELEASES FROM BUILDING VENTILATION SYSTEMS PRIOR TO TREATMENT
($Ci/yr/\mu Ci/g$)

	Containment Building	Auxiliary Building**	Turbine Building ***
Annual Normalized* Iodine Release Rate			
Power Operation	8.0×10^{-4}††	0.72†	3.8×10^{3}
Refueling/Maintenance Outages	0.32**	2.59	4.2×10^{2}

†* The values in this table come from Tables 2-13 through 2-16.

* The normalized release rate, during different modes of operation, represents the effective leak rate for radioiodine. It is the combination of the reactor water leakage rate into the building and the partitioning of the radioiodine between the water phase in the leakage and the gas phase where it is measured. For the turbine building the effective leak rate must consider the carryover for radioiodine from water to steam in the steam generator.

** To obtain the actual iodine release from these buildings in Ci/yr, multiply the normalized release by the iodine coolant concentration in $\mu Ci/g$.

*** To obtain the actual iodine release from the turbine building in Ci/yr, multiply the normalized release by the secondary coolant concentration in $\mu Ci/g$ and by the partition coefficient (NS) from Table 2-6.

† Includes contribution from the fuel pool area.

†† This release rate is expressed in %/day of leakage of primary coolant inventory of iodine and represents the effective leak rate for radioiodine. It is the combination of the reactor water leakage rate into the buildings, and the partitioning of the radioiodine between the water phase in the leakage and the gas phase where it is measured. In order to obtain the releases in curies/year during power operations from the containment building of a particular PWR, the normalized leak rates in Table 1-1, are multiplied in the PWR-GALE Code by the iodine concentration in the reactor coolant for that particular PWR, and then this leak rate is considered along with the containment purging method for that particular PWR.

TABLE 1-2

RADIOACTIVE PARTICULATE RELEASES FROM BUILDING VENTILATION SYSTEMS PRIOR TO TREATMENT*
(Ci/yr)/Unit

Nuclide	Containment	Auxiliary Building	Fuel Pool Area	Waste Gas System
Cr-51	9.2(-3)†	3.2(-4)	1.8(-4)	1.4(-5)
Mn-54	5.3(-3)	7.8(-5)	3.0(-4)	2.1(-6)
Co-57	8.2(-4)	NA	NA	NA
Co-58	2.5(-2)	1.9(-3)	2.1(-2)	8.7(-6)
Co-60	2.6(-3)	5.1(-4)	8.2(-3)	1.4(-5)
Fe-59	2.7(-3)	5.0(-5)	NA	1.8(-6)
Sr-89	1.3(-2)	7.5(-4)	2.1(-3)	4.4(-5)
Sr-90	5.2(-3)	2.9(-4)	8.0(-4)	1.7(-5)
Zr-95	NA	1.0(-3)	3.6(-6)	4.8(-6)
Nb-95	1.8(-3)	3.0(-5)	2.4(-3)	3.7(-6)
Ru-103	1.6(-3)	2.3(-5)	3.8(-5)	3.2(-6)
Ru-106	NA	6.0(-6)	6.9(-5)	2.7(-6)
Sb-125	NA	3.9(-6)	5.7(-5)	NA
Cs-134	2.5(-3)	5.4(-4)	1.7(-3)	3.3(-5)
Cs-136	3.2(-3)	4.8(-5)	NA	5.3(-6)
Cs-137	5.5(-3)	7.2(-4)	2.7(-3)	7.7(-5)
Ba-140	NA	4.0(-4)	NA	2.3(-5)
Ce-141	1.3(-3)	2.6(-5)	4.4(-7)	2.2(-6)

NA - No release observed from this source. Release assumed to be less than 1.0% of total.

† 9.2(-3) = 9.2×10^{-3}.

* The values in this table come from Tables 2-17 through 2-21.

1.5.1.6 Containment Building Purge Frequency

Two purges at cold shutdown per year plus a continuous purge specified by the applicant in his containment design.

1.5.1.7 Primary System Volumes Degassed per Year

Two coolant volumes per year for cold shutdowns plus volumes degassed due to continuous stripping.

1.5.1.8 Steam Generator Partition Coefficient (PC)

Once-through	PC
Iodine	1.0
Nonvolatiles	1.0
Recirculation U-Tube	
Iodine	0.01
Nonvolatiles	0.005

1.5.1.9 Radioiodine Releases from the Main Condenser Air Ejector Exhaust Prior to Treatment

The normalized release rate of radioiodine from the main condenser air ejector exhaust prior to treatment is 1.7×10^3 Ci/yr/μCi/g. The normalized release rate represents the effective release rate for radioiodine. It is the combination of the steam flow to the main condenser, the partitioning of radioiodine between the main condenser and the air ejector exhaust where it is measured, and the partition coefficient for radioiodine from water to steam in the steam generator. To obtain the actual iodine release from the main condenser air ejector exhaust in Ci/yr, multiply the normalized release by the secondary coolant concentration in μCi/g and by the iodine partition coefficient (NS) from Table 2-6.

1.5.1.10 Containment Internal Cleanup System

For systems using an internal cleanup system, the PWR-GALE Code calculates the iodine concentration in the containment atmosphere based on 16 hours of system operation prior to purging, an iodine removal efficiency for the charcoal adsorbers corresponding to Table 1-5, a particulate DF of 100 for HEPA filters and an internal mixing efficiency of 70%.

1.5.1.11 Detergent Wastes

The radionuclides listed in Table 2-27 of Chapter 2 are assumed to be released unless treatment is provided or laundry is not processed on site.

1.5.1.12 Tritium Releases

The tritium releases through the combined liquid and vapor pathways are 0.4 Ci/yr per MWt. The quantity of tritium released through the liquid pathway is based on the calculated volume of liquid released, excluding secondary system wastes, with a primary coolant tritium concentration of 1.0 μCi/ml up to a maximum of 0.9 of the total quantity of tritium calculated to be available for release. It is assumed that the remainder of the tritium produced is released as a gas from building ventilation exhaust systems.

1.5.1.13 Argon-41 Releases

The annual quantity of argon-41 released from a pressurized water reactor is 34 Ci/yr. The argon-41 is released to the environment via the containment vent when the containment is vented or purged.

1.5.1.14 Carbon-14 Releases

The annual quantity of carbon-14 released is 7.3 Ci/yr, of which the releases from the containment, auxiliary building and waste gas system are 1.6, 4.5 and 1.2 Ci/yr, respectively.

1.5.1.15 Decontamination Factors for Condensate Demineralizer

Demineralizer	Anion	Cs, Rb	Other Nuclides
Deep Bed	10	2	10
Powdex	10	2	10

Note: For a system using filter/demineralizers (Powdex), a zero is entered for a regeneration frequency as explained later in Section 1.5.2.10.

1.5.1.16 Primary Coolant Purification System Demineralizers

Demineralizer	Anion	Cs, Rb	Other Nuclides
Mixed Bed	100	2	50
Cation	1	10	10

1.5.1.17 Releases of Radioactive Material in Liquid Waste from the Turbine Building Floor Drain System

7200 gal/day at main steam activity.

1.5.1.18 Regeneration of Condensate Demineralizers

Flow rates and concentrations of radioactive materials routed to the liquid radwaste treatment system from the chemical regeneration of the condensate demineralizers are based on the following parameters:

1. Liquid flow to the demineralizer is based on the radioactivity of the main steam and the fraction of radioactivity which does not bypass the condensate demineralizer if there is pumped forward flow. The steam generator blowdown radioactivity is added to the condensate radioactivity if the blowdown is processed through the condensate demineralizer.

2. All radionuclides removed from the secondary coolant by the demineralizer resins are removed from the resins during chemical regeneration. The radioactivity in the regenerant wastes is adjusted for radionuclide decay during demineralizer operation.

1.5.1.19 Adjustment to Liquid Radwaste Source Terms for Anticipated Operational Occurrences

1. The calculated source term is increased by 0.16 Ci/yr per reactor using the same isotopic distribution as for the calculated source term to account for anticipated occurrences such as operator errors resulting in unplanned releases.

2. Evaporators are assumed to be unavailable for two consecutive days per week for maintenance. If a two-day holdup capacity or an alternate evaporator is available, no adjustment is needed. If less than a two-day capacity is available, the waste excess is assumed to be handled as follows:

   a. Clean or Dirty Waste - Processed through an alternative system (if available) using a discharge fraction consistent with the lower purity system.

   b. Chemical Waste - Discharged to the environment to the extent holdup capacity or an alternative evaporator is not available.

1.5.2 PARAMETERS REQUIRED FOR THE PWR-GALE CODE

Complete the cards designated in the sections below by "(SAR/ER)" from information given in the Safety Analysis and Environmental Reports. Complete the remaining cards (i.e., those not designated below as "(SAR/ER)" cards), using the principal source term parameters specified below and discussed in Chapter 2 of this document.

1.5.2.1 Card 1: Name of Reactor (SAR/ER)

Enter in spaces 33-60 the name of the reactor.

Enter in spaces 78-80 the type of reactor, i.e., PWR.

1.5.2.2 Card 2: Thermal Power Level (SAR/ER)

Enter in spaces 73-80 the maximum thermal power level (in MWt) evaluated for safety considerations in the Safety Analysis Report.

1.5.2.3 Card 3: Mass of Coolant in Primary System (SAR/ER)

Enter in spaces 73-80 the mass of coolant (in 10^3 lb) in the primary system at operating temperature and pressure.

1.5.2.4 Card 4: Primary System Letdown Rate (SAR/ER)

Enter in spaces 73-80 the average letdown rate (gal/min) from the primary system to the purification demineralizers.

1.5.2.5 Card 5: Letdown Cation Demineralizer Flow Rate (SAR/ER)

Enter in spaces 73-80 the annual average flow rate (gal/min) through the cation demineralizers for the control of cesium in the primary coolant. The average flow rate is determined by multiplying the average letdown rate (value entered on Card 4) by the fraction of time the cation demineralizers are in service to obtain the average cation demineralizer flow rate.

1.5.2.6 Card 6: Number of Steam Generators (SAR/ER)

Enter in spaces 73-80 the number of steam generators.

1.5.2.7 Card 7: Total Steam Flow (SAR/ER)

Enter in spaces 73-80 the total steam flow (in 10^6 lb/hr) for all steam generators.

1.5.2.8 Card 8: Mass of Liquid in Each Steam Generator (SAR/ER)

Enter in spaces 73-80 the mass of liquid (in 10^3 lb) in each steam generator.

1.5.2.9 Card 9: Steam Generator Blowdown Rate and Blowdown Treatment Method (SAR/ER)

Enter in spaces 37-44 the steam generator blowdown rate as given in the applicants SAR or ER.

Enter total blowdown rate in thousands of lb/hr in spaces 37-44. For a once-through steam generator, leave spaces 37-44 blank.

Describe the Blowdown Treatment Method as follows:

1. Enter 0 in space 80 if the blowdown is recycled to the condensate system after treatment in the blowdown system whether or not there are condensate demineralizers.

2. Enter 1 in space 80 if the steam generator blowdown is recycled directly to condensate system demineralizers without prior treatment in the blowdown system.

3. Enter 2 in space 80 if the steam generator blowdown is not recycled to the condensate system.

If the plant has once-through steam generators, leave space 80 blank.

1.5.2.10 Card 10: Condensate Demineralizer Regeneration Time

For deep-bed condensate demineralizers which do not use ultrasonic resin cleaner, use a 1.2-day regeneration frequency. Multiply the frequency by the number of demineralizers and enter the calculated number of days in spaces 73-80; for deep-bed condensate demineralizers which use ultrasonic resin cleaning, use an 8-day regeneration frequency. For filter/demineralizers (Powdex) or if condensate demineralizers are not used, enter zeros in spaces 73-80.

1.5.2.11 Card 11: Fraction of Feedwater Through Condensate Demineralizer (SAR/ER)

Enter in spaces 73-80 the fraction of feedwater to the steam generator processed through the condensate demineralizers. If condensate demineralizers are not used, enter 0.0 in spaces 73-80.

1.5.2.12 Cards 12-29: Liquid Radwaste Treatment System Input Parameters

Six liquid radwaste inlet streams are considered in the PWR-GALE Code:

1. Shim Bleed, Cards 12-14.

2. Equipment Drain Waste, Cards 15-17.

3. Clean Waste, Cards 18-20.

4. Dirty Waste, Cards 21-23.

5. Blowdown Waste, Cards 24-26.

6. Regenerant Wastes, Cards 27-29.

Three input data cards are used to define the major parameters for each of the six waste streams. Essentially the same information is needed on the three input data cards used for each of the six waste streams. The instructions given in this section are applicable to all six waste streams with the following exception: The inlet waste activity is not entered for Cards 12, 24, and 27 for the shim bleed, blowdown wastes, or regenerant wastes since that activity for these wastes is calculated by the PWR-GALE Code.

Cards 12-14 are used only for the shim bleed stream. For reactor designs that combine the shim bleed with other reactor grade wastes prior to processing, the other wastes are entered as equipment drain wastes on Cards 15-17.

The entries required on the first card (12, 15, 18, 21, 24, and 27) for each of the six waste streams, respectively, considered in the PWR-GALE Code are outlined below and described in more detail in Section 1.5.2.15.1.

1. Enter in spaces 17-39 the name of the waste stream (Card 24 spaces 17-44).

2. Enter in spaces 42-49 the flow rate (in gal/day) of the inlet stream (except on Cards 24 and 27).

3. Enter in spaces 57-61 the activity of the inlet stream expressed as a fraction of primary coolant activity (PCA) (except on Cards 12, 24 and 27).

The second card (13, 16, 19, 22, 25, and 28) for each waste stream contains the overall system decontamination factors for the three categories of radionuclides, as follows:

1. Enter in spaces 21-28 the DF for iodine.

2. Enter in spaces 34-41 the DF for cesium and rubidium.

3. Enter in spaces 47-54 the DF for other nuclides.

The following entries are required on the third card (14, 17, 20, 23, 26, and 29) for each waste stream:

1. Enter in spaces 28-33 waste collection time (in days) prior to processing.

2. Enter in spaces 48-53 waste processing and discharge times (in days).

3. Enter in spaces 72-77 the average fraction of wastes to be discharged after processing.

Cards 24-26 are for waste inputs due to steam generator blowdown.

1. Card 24

    a. For recirculating U-tube steam generator systems, enter the fraction of the blowdown stream processed in spaces 73-80. The PWR-GALE Code will calculate releases based on steam generator blowdown wastes.

    b. For once-through steam generator systems, leave spaces 73-80 blank.

2. Card 25

    a. If the steam generator blowdown is not recycled to the condensate system, enter blowdown system DF's as explained for Card 13.

b. If the steam generator blowdown is recycled directly to the condensate system demineralizers without prior treatment in the blowdown system, enter DF of 1.0 for iodine in spaces 21-28, DF of 1.0 for cesium and rubidium in spaces 34-41, and DF of 1.0 for other nuclides in spaces 47-54.

c. If the steam generator blowdown is recycled to the condensate system demineralizers after treatment in the blowdown system, enter blowdown system DF's as explained for Card 13.

3. Card 26

Complete Card 26 as explained for Card 14.

Cards 27-29 are for waste inputs due to regenerant wastes.

1. Card 27

a. For recirculating U-tube steam generator systems that do not utilize condensate demineralizers in the secondary system, leave spaces 73-80 blank.

b. For once-through steam generator systems and for recirculating U-tube steam generator systems that utilize condensate demineralizers in the secondary system, enter the regeneration solution waste flow (gal/day) in spaces 73-80. The inlet waste activity is not needed since the activity is calculated by the PWR-GALE Code.

2. Cards 28 and 29

Complete Cards 28 and 29 as explained for Cards 13 and 14.

The following sections explain in more detail the use of the parameters in this report and the information given in the SAR/ER to make the data entries on Cards 12-92 listed above.

1.5.2.12.1 Liquid Waste Flow Rates and Activities (Cards 12, 15, 18, 21, 24 and 27)

Flow rates and activity are calculated, using the waste volumes and activities given in Table 1-3. To the input flow rates given in the table, add expected flows and activities more specific to the plant design as given in the SAR/ER. With the exception of the shim bleed, the individual streams are combined based on the radwaste treatment system described in the SAR/ER.

Waste streams processed with the shim bleed are entered as equipment drain wastes on Cards 15-17. Input activities are based on the weighted average activity of the composite stream entering the waste collection

TABLE 1-3

PWR LIQUID WASTES

SOURCE	EXPECTED DAILY AVERAGE INPUT FLOW RATE (in Gal/day) — Type of treatment of blowdown recycled to secondary system (U-tube steam generator plants) or type of treatment of condensate (once-through steam generator plants): Deep-bed cond. demineralizers with ultrasonic resin cleaner	Deep-bed cond. demineralizers without ultrasonic resin cleaner	Filter-demineralizer	Plant with blowdown treatment. Product not recycled to condenser or secondary coolant system	FRACTION OF PRIMARY COOLANT ACTIVITY (PCA)
1. REACTOR CONTAINMENT					
a. Primary coolant pump seal leakage	20	20	20	20	0.1
b. Primary coolant leakage, miscellaneous sources	10	10	10	10	1.67*
c. Primary coolant equipment drains	500	500	500	500	0.001
2. PRIMARY COOLANT SYSTEMS (OUTSIDE OF CONTAINMENT)					
a. Primary coolant system equipment drains	80	80	80	80	1.0
b. Spent fuel pit liner drains	700	700	700	700	0.001
c. Primary coolant sampling system drains	200	200	200	200	0.05
d. Auxiliary building floor drains	200	200	200	200	0.1

TABLE 1-3 (Continued)

3. SECONDARY COOLANT SYSTEMS					
a. Secondary coolant sampling system drains	1400	1400	1400	1400	10^{-4}
b. Condensate demineralizer rinse and transfer solutions	3000	12000	-	-	10^{-8}
c. Condensate demineralizer regenerant solutions	850	3400	-	-	Calculated in GALE Code
d. Ultrasonic resin cleaner solutions	15000	-	-	-	10^{-6}
e. Condensate filter-demineralizer backwash	-	-	8100	-	2×10^{-6}
f. Steam generator blowdown	-	-	-	Plant dependent**	Plant dependent**
g. Turbine building floor drains	7200	7200	7200	7200	Calculated in GALE Code
4. DETERGENT AND DECONTAMINATION SYSTEMS					
a. On-site laundry facility	300	300	300	300	See Table 2-26
b. Hot showers	Negligible	Negligible	Negligible	Negligible	-
c. Hand wash sink drains	200	200	200	200	See Table 2-26
d. Equipment and area decontamination	40	40	40	40	See Table 2-26
TOTALS	29,700	26,300	19,000	10,000	

* About 40 percent of the leakage flashes, resulting in PCA fraction of the leakage greater than 1.0.
** Input parameter.

tanks. For example, if the inlet streams A, B, and C enter the dirty waste collector tank at average rates and PCA as listed below,

Stream A	1,000 gal/day at 0.01PCA
Stream B	2,000 gal/day at 0.1PCA
Stream C	500 gal/day at 1.0PCA

the composite A, B, C activity would be calculated as follows:

$$\frac{(1{,}000\ \text{gal/day})(0.01\text{PCA}) + (2{,}000\ \text{gal/day})(0.1\text{PCA}) + (500\ \text{gal/day})(1.0\text{PCA})}{(1{,}000\ \text{gal/day} + 2{,}000\ \text{gal/day} + 500\ \text{gal/day})} = 0.2\text{PCA}$$

The entries on Card 21 for this example would then be: spaces 17-33, "Dirty Waste"; spaces 42-49, 3500.; spaces 57-61, "0.2".

The input flow rates and activities are entered in units of gal/day and fractions of PCA, respectively.

1.5.2.12.2 Decontamination Factors for Equipment Used in the Liquid Radwaste Treatment System (Cards 13, 16, 19, 22, 25, and 28)

The decontamination factors (DF's) given in this document are used in the PWR-GALE Code. The DF's represent the expected equipment performance averaged over the life of the plant, including downtime. The following factors should be considered in calculating the overall decontamination factors for the various systems:

1. DF's are categorized by one of the following types of radionuclides:

    a. Halogens

    b. Cs, Rb

    c. Other Nuclides

Note: A DF of 1 is assumed by the PWR-GALE Code for tritium. Noble gases and water activation products, e.g., N-16, are not considered in the liquid code.

2. The system DF for each inlet stream is the product of the individual equipment DF's in each of the subsystems.

3. Equipment that is used optionally (as required) and not included in the normal flow scheme should not be considered in calculating the overall system DF.

Table 1-4 shows the decontamination factors to be used for PWR systems.

TABLE 1-4

DECONTAMINATION FACTORS FOR PWR LIQUID WASTE TREATMENT SYSTEMS

TREATMENT SYSTEM	DECONTAMINATION FACTOR		
Demineralizer	Anion	Cs, Rb	Other Nuclides
Mixed Bed			
Primary coolant letdown (CVCS)	100	2	50
Radwaste (H^+OH^-)	10^2(10)*	2(10)	10^2(10)
Evaporator condensate polishing	5	1	10
Boron recycle	10	2	10
Steam generator blowdown	10^2(10)	10(10)	10^2(10)
Cation bed (any system)	1(1)	10(10)	10(10)
Anion bed (any system)	10^2(10)	1(1)	1(1)
Powdex (any system)	10(10)	2(10)	10(10)

Evaporators	All Nuclides Except Iodine	Iodine
Miscellaneous radwaste	10^3	10^2
Boric acid recovery	10^3	10^2

Reverse Osmosis	All Nuclides
Laundry wastes	30
Other liquid wastes	10
Filters	DF of 1 for all nuclides

* For demineralizers in series, the DF for the second demineralizer is given in parentheses.

The following example illustrates the calculation of the decontamination factor for a dirty waste treatment system: Assume that dirty wastes are collected; processed through a filter, an evaporator, and a mixed-bed polishing demineralizer; and collected for sampling. If required to meet discharge criteria, the contents of the waste sample (test) tank are processed through a mixed-bed demineralizer for additional radionuclide removal. This example may be summarized graphically as:

Demineralizer 2

Dirty waste collector tank — Filter — Evaporator — Demineralizer 1 — Waste sample tank

Extracting from Table 1-4 gives the following values for the example:

	Filter	Evaporator	Demineralizer 1	Demineralizer 2	Product
Iodine	1	10^2	5	1	5×10^2
Cs, Rb	1	10^3	1	1	10^3
Other Nuclides	1	10^3	10	1	10^4

These values are obtained as follows:

- A DF of 1.0 is applied to all nuclides for the filter.
- A DF of 10^2 for iodine and 10^3 for Cs, Rb, and other nuclides is applied for the radwaste evaporator.
- A DF of 5 is applied for iodine, a DF of 1 for Cs, Rb and a DF of 10 for the evaporator condensate polishing demineralizer.
- A DF of 1 is applied to the second demineralizer since this demineralizer's used is optional, and it is not used for normal operations.
- The product of the DF's is obtained by multiplication of the first four columns for each nuclide.

Thus on Card 22, the following would be entered: in spaces 21-28, "500.0"; in spaces 34-41, "1000.0"; and in spaces 47-54, "10000.0".

1.5.2.12.3 Collection Time for Liquid Wastes (Cards 14, 17, 20, 23, 26, and 29 -- Spaces 29-33)

Collection time prior to processing is based on the input flow calculated above. Where redundant tanks are provided, assume the collection tank to be filled to 80% design capacity. If only one tank is provided, assume the tank to be filled to 40% design capacity. For example, if flow from a 1,000-gal/day floor drain is collected in two 20,000-gallon tanks prior to processing, collection time would be calculated as follows:

$$\text{Collection time } (T_c) = \frac{(0.8)(20{,}000 \text{ gal})}{(1{,}000 \text{ gal/day})} = 16 \text{ days}$$

Then, for example, "16.0" should be entered in spaces 29-33 on Card 23.

1.5.2.12.4 Processing and Discharge Time (Cards 14, 17, 20, 23, 26, and 29 -- Spaces 48-53)

Decay during processing and discharge of liquid wastes is shown graphically as follows:

$$\text{Tank A} - \text{B } R_b - \overset{R_o}{\text{Tank C}} - R_c - \text{Discharge Canal}$$

where

A is the capacity of initial tank in flow scheme, in gal;

B is the limiting process based on equipment flow capacity, dimensionless;

C is the capacity of final tank in flow scheme prior to discharge, in gal;

R_b is the equipment flow capacity of Process B, in gal/day;

R_c is the flow capacity of Tank C discharge pump, in gal/day; and

R_o is the rate of flow of additional waste inputs to Tank C, in gal/day.

T_p, the process time credited for decay, is calculated as follows, in days:

$$T_p = \frac{0.8A}{R_b} \text{ for redundant tanks, or } T_p = \frac{0.4A}{R_b} \text{ for a single tank}$$

T_d, the discharge time (50% credited for decay), is calculated as follows, in days:

$$T_d = \frac{0.8C}{R_c} \text{ for redundant tanks, or } T_d = \frac{0.4C}{R_c} \text{ for a single tank.}$$

After performing the above two calculations, calculate whether credit may be taken for decay during discharge by determining whether

$$0.8C > T_p(R_b + R_o) \text{ for redundant tanks, or}$$

$$0.4C > T_p(R_b + R_o) \text{ for a single tank.}$$

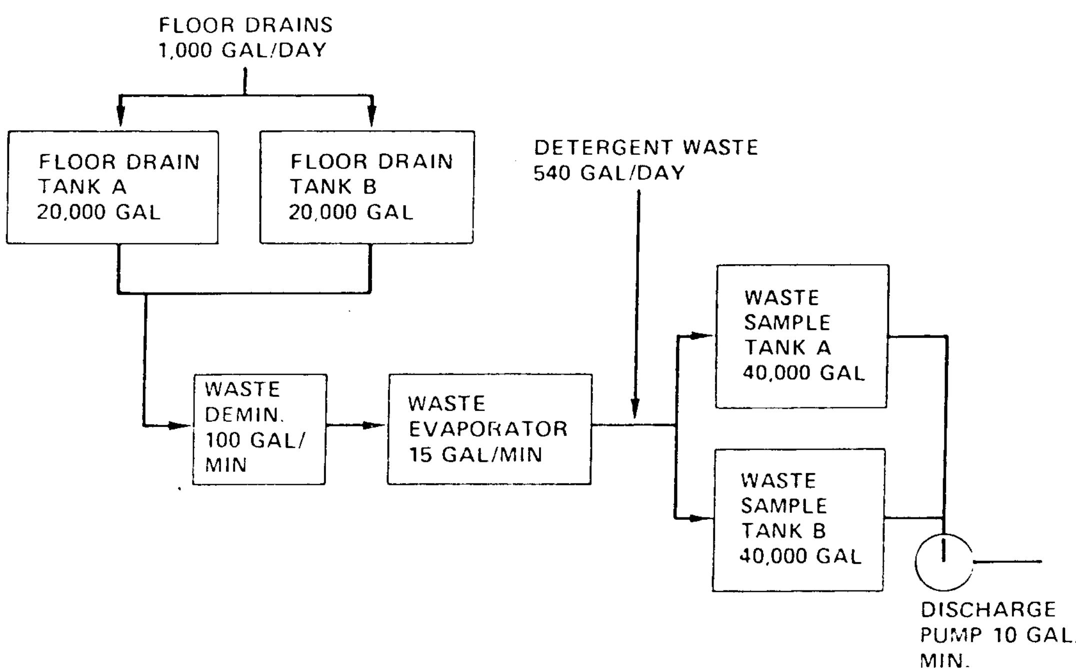
If so, then

$$\text{Decay} = T_p + 0.5T_d$$

where "Decay" is the new processing and discharge time to be entered in spaces 48-53 of the third card for each input stream (Cards 14, 17, 20, 23, 26, and 29).

If, however, 0.8C (or 0.4C, as appropriate) $\leq T_p(R_b + R_o)$, T_p is used for the holdup time during processing, since Tank C may be discharged before Tank A has been completely processed. In this case, the T_p value should be entered in spaces 48-53 of the third card.

For example, for the following input waste stream:



Decay time during processing and discharge is calculated as follows:

$$\text{Process Time } (T_p) = \frac{(0.8)(20{,}000 \text{ gal})}{(15 \text{ gal/min})(1440 \text{ min/day})} = 0.7 \text{ day}$$

$$\text{Discharge Time } (T_d) = \frac{(0.8)(40{,}000 \text{ gal})}{(10 \text{ gal/min})(1440 \text{ min/day})} = 2.2 \text{ days}$$

Then, checking for decay credit, $0.8C/(R_b + R_o) = 1.45$ days, which is greater than T_p; therefore, credit is taken for $(T_p + 0.5T_d)$ or 1.8 days for processing and discharge. The input in spaces 48-53 to the Code is 1.8 days for processing and discharge time.

1.5.2.12.5 Fraction of Wastes Discharged (Cards 14, 17, 20, 23, 26, and 29 -- Spaces 72-77)

The percent of the wastes discharged after processing may vary between 10% and 100%, except as noted below, based on the capability of the system to process liquid waste during equipment downtime, waste volume surges, tritium control requirements, and tank surge capacity. A minimum value of 10% discharge for the liquid radioactive waste treatment system is used when the system is designed for maximum waste recycle, when the system capacity is sufficient to process wastes for reuse during equipment downtime and anticipated operational occurrences, and when a discharge route is provided. For steam generator blowdown treatment systems, less than 10% discharge should be considered on a case-by-base basis, depending on system capacity.

The PWR-GALE Code calculates the release of radioactive materials in liquid waste from the following systems after processing. The quantity released is shown on the printout.

1. Boron Recovery System - Combined releases from both shim bleed and equipment drains.

2. Miscellaneous Liquid Waste System - Combined releases from both clean and dirty waste subsystems.

3. Secondary Waste System - Releases from steam generator blowdown system, regenerant wastes from demineralizer regenerations, or both according to the plant design.

4. Turbine Building Floor Drain System - Releases of liquid from the turbine building floor drain system are calculated assuming no treatment prior to release. Straight decay time of 6 hours is built into the code.

5. Detergent Waste System - Combined releases from laundry operations, equipment decontamination solutions, and personnel decontamination showers.

1.5.2.13 Card 30: Letdown System

1. Enter 0 in space 80 if there is not continuous gas stripping of the full letdown flow. (This sets Y = 0.0.)

2. Enter 1 in space 80 if there is continuous degassification of the full letdown flow to the gaseous radwaste system via a gas stripper. (This sets Y = 1.0.)

3. Enter 2 in space 80 if there is continuous purging of the volume control tank. (This sets Y = 0.25.)

The total amount of fission gases routed to the gaseous radwaste system from several systems in the plant (e.g., volume control tank, shim bleed gas stripper, equipment drain tanks, cover gas) is calculated in the PWR-GALE Code. (For definition of "Y", see Tables 2-4 & 2-5.)

1.5.2.14 Cards 31-33: Holdup Time for Fission Gases Stripped from Primary Coolant

The holdup time for gases stripped from the primary coolant is hand calculated because of the multiplicity of holdup system designs. The calculations are based on the following parameters:

1. Pressurized Storage Tanks

   a. One storage tank is held in reserve for back-to-back shutdowns, one tank is in the process of filling, and the remainder are used for storage. The PWR-GALE Code will calculate the effective holdup time for filling and add it to the holdup time for storage.

   b. Calculations are based on the waste gas input flow rate to the pressurized storage tanks, and a storage tank pressure 70% of the design value.

   c. If the calculated holdup time exceeds 90 days, assume the remaining gases are released after 90 days.

The holdup time (T_h) and fill time (T_f) are calculated as follows:

$$T_f = \frac{PV}{F}$$

$$T_h = \frac{PV(n-2)}{F}$$

where

n is the number of tanks;

n-2 is the correction to subtract the tank being filled and the tank held in reserve;

P is the storage pressure, in atmospheres (dimensionless in this particular calculation);

T_f is the time required to fill one tank, in days;

T_h is the holdup time, in days;

V is the volume of each tank, in ft^3 (STP); and

F is the waste gas flow rate to pressurized storage tanks. This flow rate should be supplied by the applicant for the specific type of waste gas system design. In the absence of specific data supplied by the applicant, we will use the data given in Section 2.2.12.1, in which the average value for the PWR's listed in Table 2-24 is 170 ft^3 /day (STP) per reactor for PWR's without recombiners; and for PWR's with recombiners, the average value for the PWR's listed in Table 2-25 is 30 ft^3/day (STP) per reactor.

Enter on Card 31 the holdup time, in days, for Xe in spaces 73-80.

Enter on Card 32 the holdup time, in days, for Kr in spaces 73-80.

Enter on Card 33 the fill time, in days, in spaces 73-80.

2. Charcoal Delay Systems

Charcoal delay system holdup times are based on the following equation:

$$T = 0.011\ MK/F$$

where

F is the system flow rate, in ft^3/min; (see 1.5.2.14.1.c, above)

K is the dynamic adsorption coefficient, in cm^3/g;

M is the mass of charcoal adsorber, in thousands of pounds; and

T is the holdup time, in days.

The dynamic absorption coefficient, K, for Xe and Kr and based on the system design noted below.

DYNAMIC ABSORPTION COEFFICIENT, K (cm^3/g)

	Operating 77°F Dew Point 45°F	Operating 77°F Dew Point 0°F	Operating 77°F Dew Point -40°	Operating 0°F Dew Point -20°
Kr	18.5	25	70	105
Xe	330.0	440	1160	2410

Enter on Card 31 the holdup time, in days, for Xe in spaces 73-80.

Enter on Card 32 the holdup time, in days, for Kr in spaces 73-80.

Leave Card 33 blank.

3. Cover Gas Recycle System

For this system or other systems designed to hold gases indefinitely, the calculations are based on a 90-day holdup time.

Enter on Card 31 the holdup time (90 days) for Xe in spaces 73-80.

Enter on Card 32 the holdup time (90 days) for Kr in spaces 73-80.

Enter on Card 33 the fill time (0 days) in spaces 73-80.

1.5.2.15 Card 34: Waste Gas System Particulate Releases

Card 34 identifies the treatment provided for particulate removal from the waste gas system effluent.

1. If ventilation exhaust air is treated through HEPA filters which satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter a removal efficiency of 99. for particulates in spaces 39-41.

2. If no treatment is provided for the ventilation exhaust air to remove particulates or if the HEPA filters do not satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter 0.0 in spaces 39-41.

1.5.2.16 Cards 35 and 36: Fuel Handling and Auxiliary Buildings Releases

Cards 35 and 36 indicate the fractions of airborne iodine and radioactive particulates released from the fuel handling and auxiliary buildings, respectively.

TABLE 1-5

ASSIGNED REMOVAL EFFICIENCIES FOR CHARCOAL ADSORBERS FOR RADIOIODINE REMOVAL

Activated Carbon[a] Bed Depth	Removal Efficiencies[b] for Radioiodine %
2 inches. Air filtration system designed to operate inside reactor containment	90.
2 inches. Air filtration system designed to operate outside the reactor containment and relative humidity is controlled at 70%	70.
4 inches. Air filtration system designed to operate outside the reactor containment and relative humidity is controlled at 70%	90.
6 inches. Air filtration system designed to operate outside the reactor containment and relative humidity is controlled to 70%	99.

[a] Multiple beds, e.g., two 2-inch beds in series, should be treated as a single bed of aggregate depth of 4 inches.

[b] The removal efficiencies assigned to HEPA filters for particulate removal and charcoal adsorbers for radioiodine removal are based on the design, testing, and maintenance criteria recommended in Regulatory Guide 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 2).

1. If ventilation exhaust air is treated through charcoal adsorbers which satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter the appropriate removal efficiency in spaces 47-49 for radioiodine corresponding to the depth of charcoal as indicated in Table 1-5.

2. If ventilation exhaust air is treated through HEPA filters which satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter a removal efficiency of 99. for particulates in spaces 56-58.

3. If no treatment is provided for the ventilation exhaust air to remove radioiodine, enter 0.0 in spaces 47-49; if no treatment is provided to remove particulates, enter 0.0 in spaces 56-58.

1.5.2.17 Card 37: Containment Free Volume (SAR/ER)

Enter the containment volume (in 10^6 ft^3) in spaces 73-80.

1.5.2.18 Card 38: Containment Internal Cleanup System (SAR/ER)

1. If the containment internal cleanup system uses charcoal adsorbers which satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter the appropriate removal efficiency in spaces 47-49 for radioiodine corresponding to the depth of charcoal as indicated in Table 1-5.

2. If the containment internal cleanup system uses HEPA filters which satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter a removal efficiency of 99. for particulates in spaces 56-58.

3. If there is no containment internal cleanup system, enter 0.0 in spaces 47-49 and in spaces 56-58.

4. Enter the flow rate (in 10^3 ft^3/min) through the internal cleanup system in spaces 73-80.

The airborne concentration calculations are based on the following parameters:

a. A primary coolant leakage rate corresponding to the normalized release rate given in Table 1-1.

b. A continuous normal ventilation flow rate as specified by the applicant.

c. Operation of the cleanup system for 16 hours prior to purging.

d. A DF for the charcoal adsorber corresponding to the values in Table 1-5, a DF of 100 for the HEPA filters, and a mixing efficiency of 70%. The mixing efficiency is an effective removal efficiency which takes into account the effects of incomplete mixing in the containment.

e. Continuous leakage of primary coolant during the operation of the internal cleanup system.

1.5.2.19 Card 39: Containment Building Iodine Releases - During Large Volume Purge System Operation

Card 39 indicates the fraction of airborne iodine and radioactive particulates released during purging of the containment building with the large volume containment purge system.

Note: Treatment referred to below does not include the internal recirculation system.

1. If ventilation exhaust air is treated through charcoal adsorbers which satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter the appropriate removal efficiency in spaces 47-49 for radioiodine corresponding to the depth of charcoal as indicated in Table 1-5.

2. If ventilation exhaust air is treated through HEPA filters which satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter a removal efficiency of 99. for particulates in spaces 56-58.

3. If no treatment is provided for the ventilation exhaust air to remove radioiodine, enter 0.0 in spaces 47-49; if no treatment is provided to remove particulates, enter 0.0 in spaces 56-58.

4. Enter the number of purges per year during power operations in spaces 78-80. (Note: The 2 purges at shutdown are stored in the PWR GALE Code and need not be entered on card 39.)

1.5.2.20 Card 40: Containment Building Iodine Releases - Low Volume Purge During Power Operation

Card 40 indicates the fraction of airborne iodine in the containment atmosphere that is released during the low volume purge of the containment building while the reactor is at power.

Note: Treatment referred to below does not include the internal recirculation system.

1. If ventilation exhaust air is treated through charcoal adsorbers which satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter the appropriate removal efficiency in spaces 47-49 for radioiodine corresponding to the depth of charcoal as indicated in Table 1-5.

2. If ventilation exhaust air is treated through HEPA filters which satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter a removal efficiency of 99. for particulates in spaces 56-58.

3. If no treatment is provided for the ventilation exhaust air to remove radioiodine, enter 0.0 in spaces 47-49; if no treatment is provided to remove particulates, enter 0.0 in spaces 56-58.

4. Enter the continuous containment purge rate (ft^3/min) in spaces 73-80.

1.5.2.21 Card 41: Steam Generator Blowdown Tank Vent

1. Enter 0.0 in spaces 73-80 if the gases from the blowdown flash tank are vented through a condenser prior to release.

2. Enter 0.0 in spaces 73-80 if the blowdown flash tank is vented to the main condenser air ejector.

3. Enter 0.0 in spaces 73-80 for a once-through steam generator system.

4. For older plants which still use flash tanks which vent directly to the atmosphere an iodine partition factor of 0.05 is used.

1.5.2.22 Card 42: Percentage of Iodine Removed by the Condenser Air Ejector Offgas Treatment System

1. If, prior to release, the offgases from the condenser air ejector are processed through charcoal adsorbers which satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter the removal efficiency in spaces 73-80 for radioiodine corresponding to the depth of charcoal as indicated in Table 1-5.

2. If the offgases are released from the condenser air ejector without treatment, enter 0.0 in spaces 73-80.

1.5.2.23 Card 43: Detergent Wastes

1. If the plant does not have an onsite laundry, enter 0.0 in spaces 73-80.

2. If the plant has an onsite laundry and detergent wastes are released without treatment, enter 1.0 in spaces 73-80.

3. If detergent wastes are treated prior to discharge, enter the fraction of radionuclides remaining after treatment (1/DF) in spaces 73-80. The parameters in Chapter 2 of this document should be used in determining the DF for the treatment applied to detergent wastes.

CHAPTER 2. PRINCIPAL PARAMETERS USED IN PWR SOURCE TERM CALCULATIONS AND THEIR BASES

2.1 INTRODUCTION

The principal parameters used in source term calculations have been compiled to standardize the calculation of radioactive source terms.

The following sections describe parameters used in the evaluation of radwaste treatment systems. The parameters have been derived from reactor operating experience where data were available. Where operating data were inconclusive or not available, information was drawn from laboratory and field tests and from engineering judgment. The bases for the source term parameters explain the reasons for choosing the numerical values listed. A list of references used in developing the parameters is also included.

The parameters in the PWR-GALE Code will be updated periodically and published in revisions to this NUREG as additional operating data become available. The source term parameters used are believed to provide a realistic assessment of reactor and radwaste system operation.

2.2 PRINCIPAL PARAMETERS AND THEIR BASES

2.2.1 THERMAL POWER LEVEL

2.2.1.1 Parameter

The maximum thermal power level (MWt) evaluated for safety considerations in the Safety Analysis Report.

2.2.1.2 Bases

The power level used in the source term PWR-GALE Code is the maximum power level evaluated for safety considerations in the Safety Analysis Report. Using this value, the evaluation of the radwaste management systems need not be repeated when the applicant applies for a stretch power license at a later date. Past experience indicates that most utilities request approval to operate at maximum power soon after reaching commercial operation.

2.2.2 PLANT CAPACITY FACTOR

2.2.2.1 Parameter

A plant capacity factor of 80% is used, i.e., 292 effective full power days.

2.2.2.2 Bases

The source term calculations are based on a plant capacity factor of 80% averaged over the 30-year operating life of the plant, i.e., the plant operates at 100% power 80% of the time. The plant capacity factors experienced at PWR's are listed in Table 2-1 for the period 1972 through 1977.

The average plant capacity factors shown in Table 2-1 indicate that the 80% factor assumed is higher than the average factors experienced. However, it is expected that the major maintenance problems and extended refueling outages that have contributed to the lower plant capacity factors will be overcome and that the plants will achieve the 80% capacity factor when averaged over 30 years of operation.

2.2.3 RADIONUCLIDE CONCENTRATIONS IN THE PRIMARY AND SECONDARY COOLANT

2.2.3.1 Parameter

As used in the PWR-GALE Code, Tables 2-2 and 2-3 list the expected radionuclide concentrations in the reactor coolant and steam for PWR's with design parameters within the ranges listed in Tables 2-4 and 2-5. Should any design parameter be outside the range in Tables 2-4 and 2-5, the PWR-GALE Code adjusts the concentrations in Tables 2-2 and 2-3, using the factors in Tables 2-6, 2-7, and 2-8. Figures 2-1 and 2-2 show the graphical relationship of the design parameters.

2.2.3.2 Bases

The radionuclide concentrations, adjustment factors, and procedure for effecting adjustments are based on the values and methods in American National Standard ANSI N237, Source Term Specification, (Ref. 1) but have been updated based on a recent compilation of available operating data concerning primary coolant concentrations, steam generator tube leakage, and secondary side radionuclide behavior. Therefore, the concentration values in NUREG-0017, Rev. 1 differ from the ANSI N237 values.

The values in Tables 2-2 and 2-3 provide a set of typical radionuclide concentrations in the primary and secondary systems for reactor designs within the parameters specified in Tables 2-4 and 2-5. The values in Tables 2-2 and 2-3 are those determined to be representative of radionuclide concentrations in a PWR over its lifetime based on the currently available data and models. The secondary coolant concentrations given in Tables 2-2 and 2-3 are calculated by using the reference parameters given in Table 2-6 and the equations given in Tables 2-7 and 2-8. It is recognized that some systems will have design parameters that are outside the ranges specified in Tables 2-4 and 2-5. For that reason, a means of adjusting the concentrations to the actual design parameters has been provided in Tables 2-6 through 2-8. The adjustment factors in Table 2-6 through 2-8 are based on the following expression;

TABLE 2-1

PLANT CAPACITY FACTORS AT OPERATING PWR's[a]

FACILITY[b]	Date of Commercial Operation[c]	1972	1973	1974	1975	1976	1977
Haddam Neck	1/68	86	48[d]	89	84	81	82
San Onofre 1	1/68	72	60	83	85	66	62
R. E. Ginna	7/70	58	81	50[d]	73	52	83
Point Beach 1	12/70	69	67	76	70	78	85
H. B. Robinson 2	3/71	78	65	81	71	82	74
Palisades	12/71	61	40[e]	d	46[e]	50[d]	78
Point Beach 2	10/72		72	77	88	86	82
Turkey Point 3	12/72		55	61	76	75	78
Surry 1	12/72		51	50[d]	60	67	78
Maine Yankee	12/72		17[f]	54	69	91	77
Surry 2	5/73			40[d]	76	51[e]	65
Oconee 1	7/73			54	71	54	54
Indian Point 2	8/73			51	68	31[d]	73
Turkey Point 4	9/73			71	68	64	62
Fort Calhoun	9/73			61	54	57	76
Prairie Island 1	12/73			36[e]	83	73	83
Zion 1	12/73			49[g]	68	55	58
Kewaunee	6/74				75	75	77
Three Mile Island 1	9/74				79	63	79
Oconee 2	9/74				68	58	53[d]
Zion 2	9/74				68	54	71
Oconee 3	12/74				69	64	71
Arkansas 1	12/74				69	54	73
Prairie Island 2	12/74				73	69	87
Rancho Seco	4/75					28[g]	75
Calvert Cliffs 1	5/75					88	65
Cook 1	8/75					75	54[f]
Millstone 2	12/75					68	63
Trojan	5/76						71
Indian Point 3	8/76						72
Beaver Valley 1	10/76						44[e]
St. Lucie 1	12/76						78
AVERAGE		71	64	69	72	69	74

[a] From monthly Operating Units Status Reports.

[b] Indian Point 1 and Yankee Rowe are not included since they are small reactors [< 700 MW(t)].

[c] Plant capacity factors listed are for the first full year of commercial operation. Therefore, this list does not include the following plants which began commercial operation in 1977 and 1978: Calvert Cliffs 2, Cook 2, Crystal River 3, Davis Besse 1, Farley 1, Salem 1, North Anna 1, and Three Mile Island 2.

[d] Not included due to extended outage for refueling/amintenance.

[e] Not included due to extended maintenance/repair to the secondary system.

[f] Not included due to extended operation at reudced power.

[g] Not included due to extended maintenance outage to repair generator.

TABLE 2-2

NUMERICAL VALUES - CONCENTRATIONS IN PRINCIPAL FLUID STREAMS OF THE REFERENCE PWR WITH U-TUBE STEAM GENERATORS

(μCi/g)

Isotope	Reactor Coolant**	Secondary Coolant* Water***	Secondary Coolant* Steam††
Noble Gases			
Kr-85m	1.6(-1)†††		3.4(-8)
Kr-85	4.3(-1)		8.9(-8)
Kr-87	1.5(-1)		3.0(-8)
Kr-88	2.8(-1)		5.9(-8)
Xe-131m	7.3(-1)		1.5(-7)
Xe-133m	7.0(-2)		1.5(-8)
Xe-133	2.6(0)		5.4(-7)
Xe-135m	1.3(-1)		2.7(-8)
Xe-135	8.5(-1)		1.8(-7)
Xe-137	3.4(-2)		7.1(-9)
Xe-138	1.2(-1)		2.5(-8)
Halogens			
Br-84	1.6(-2)	7.5(-8)	7.5(-10)
I-131	4.5(-2)	1.8(-6)	1.8(-8)
I-132	2.1(-1)	3.1(-6)	3.1(-8)
I-133	1.4(-1)	4.8(-6)	4.8(-8)
I-134	3.4(-1)	2.4(-6)	2.4(-8)
I-135	2.6(-1)	6.6(-6)	6.6(-8)
Cs, Rb			
Rb-88	1.9(-1)	5.3(-7)	2.6(-9)
Cs-134	7.1(-3)	3.3(-7)	1.7(-9)
Cs-136	8.7(-4)	4.0(-8)	2.0(-10)
Cs-137	9.4(-3)	4.4(-7)	2.2(-9)
Water Activation Products			
N-16	4.0(+1)	1.0(-6)	1.0(-7)
Tritium			
H-3	1.0(0)	1.0(-3)	1.0(-3)
Other Nuclides			
Na-24	4.7(-2)	1.5(-6)	7.5(-9)
Cr-51	3.1(-3)	1.3(-7)	6.3(-10)
Mn-54	1.6(-3)	6.5(-8)	3.3(-10)

TABLE 2-2 (continued)

Isotope	Reactor Coolant**	Secondary Coolant* Water***	Secondary Coolant* Steam††
Fe-55	1.2(-3)	4.9(-8)	2.5(-10)
Fe-59	3.0(-4)	1.2(-8)	6.1(-11)
Co-58	4.6(-3)	1.9(-7)	9.4(-10)
Co-60	5.3(-4)	2.2(-8)	1.1(-10)
Zn-65	5.1(-4)	2.1(-8)	1.0(-10)
Sr-89	1.4(-4)	5.7(-9)	2.9(-11)
Sr-90	1.2(-5)	4.9(-10)	2.5(-12)
Sr-91	9.6(-4)	2.8(-8)	1.4(-10)
Y-91m	4.6(-4)	3.2(-9)	1.6(-11)
Y-91	5.2(-6)	2.1(-10)	1.1(-12)
Y-93	4.2(-3)	1.2(-7)	6.1(-10)
Zr-95	3.9(-4)	1.6(-8)	7.9(-11)
Nb-95	2.8(-4)	1.1(-8)	5.7(-11)
Mo-99	6.4(-3)	2.5(-7)	1.2(-9)
Tc-99m	4.7(-3)	1.1(-7)	5.7(-10)
Ru-103	7.5(-3)	3.1(-7)	1.6(-9)
Ru-106	9.0(-2)	3.7(-6)	1.8(-8)
Ag-110m	1.3(-3)	5.3(-8)	2.7(-10)
Te-129m	1.9(-4)	7.8(-9)	3.9(-11)
Te-129	2.4(-2)	2.2(-7)	1.1(-9)
Te-131m	1.5(-3)	5.4(-8)	2.7(-10)
Te-131	7.7(-3)	2.9(-8)	1.5(-10)
Te-132	1.7(-3)	6.6(-8)	3.3(-10)
Ba-140	1.3(-2)	5.2(-7)	2.6(-9)
La-140	2.5(-2)	9.3(-7)	4.6(-9)
Ce-141	1.5(-4)	6.1(-9)	3.1(-11)
Ce-143	2.8(-3)	1.0(-7)	5.1(-10)
Ce-144	3.9(-3)	1.6(-7)	8.2(-10)
W-187	2.5(-3)	8.7(-8)	4.4(-10)
Np-239	2.2(-3)	8.4(-8)	4.2(-10)

* Based on a primary-to-secondary leak of 75 lb/day.

** The concentrations given are for reactor coolant entering the letdown line. These concentrations are obtained from Tables 2-9 and 2-10. N-16 and H-3 concentrations are obtained from Reference 1.

*** The concentrations given are for water in a steam generator.

†† The concentrations given are for steam leaving a steam generator.

††† 1.6(-1) = 1.6×10^{-1}.

TABLE 2-3

NUMERICAL VALUES - CONCENTRATIONS IN PRINCIPAL FLUID STREAMS OF THE REFERENCE PWR WITH ONCE-THROUGH STEAM GENERATORS

(μCi/g)

Isotope	Reactor Coolant*	Secondary Coolant**
Noble Gases		
Kr-85m	1.6(-1)	3.4(-8)
Kr-85	4.3(-1)	8.9(-8)
Kr-87	1.5(-1)	3.0(-8)
Kr-88	2.8(-1)	5.9(-8)
Xe-131m	7.3(-1)	1.5(-7)
Xe-133m	7.0(-2)	1.5(-8)
Xe-133	2.6(0)	5.4(-7)
Xe-135m	1.3(-1)	2.7(-8)
Xe-135	8.5(-1)	1.8(-7)
Xe-137	3.4(-2)	7.1(-9)
Xe-138	1.2(-1)	2.5(-8)
Halogens		
Br-84	1.6(-2)	1.8(-8)
I-131	4.5(-2)	5.2(-8)
I-132	2.1(-1)	2.4(-7)
I-133	1.4(-1)	1.6(-7)
I-134	3.4(-1)	3.8(-7)
I-135	2.6(-1)	3.0(-7)
Cs, Rb		
Rb-88	1.9(-1)	6.0(-7)
Cs-134	7.1(-3)	3.0(-8)
Cs-136	8.7(-4)	3.6(-9)
Cs-137	9.4(-3)	3.9(-8)
Water Activation Products		
N-16	4.0(+1)	1.0(-6)
Tritium		
H-3	1.0(0)	1.0(-3)
Other Nuclides		
Na-24	4.7(-2)	1.0(-7)
Cr-51	3.1(-3)	6.9(-9)
Mn-54	1.6(-3)	3.6(-9)
Fe-55	1.2(-3)	2.7(-9)
Fe-59	3.0(-4)	6.7(-10)
Co-58	4.6(-3)	1.0(-8)
Co-60	5.3(-4)	1.2(-9)

TABLE 2-3 (continued)

Isotope	Reactor Coolant*	Secondary Coolant**
Zn-65	5.1(-4)	1.1(-9)
Sr-89	1.4(-4)	3.1(-10)
Sr-90	1.2(-5)	2.7(-11)
Sr-91	9.6(-4)	2.1(-9)
Y-91m	4.6(-4)	9.7(-10)
Y-91	5.2(-6)	1.2(-11)
Y-93	4.2(-3)	9.3(-9)
Zr-95	3.9(-4)	8.7(-10)
Nb-95	2.8(-4)	6.2(-10)
Mo-99	6.4(-3)	1.4(-8)
Tc-99m	4.7(-3)	1.0(-8)
Ru-103	7.5(-3)	1.7(-8)
Ru-106	9.0(-2)	2.0(-7)
Ag-110m	1.3(-3)	2.9(-9)
Te-129m	1.9(-4)	4.2(-10)
Te-129	2.4(-2)	5.1(-8)
Te-131m	1.5(-3)	3.3(-9)
Te-131	7.7(-3)	1.5(-8)
Te-132	1.7(-3)	3.8(-9)
Ba-140	1.3(-2)	2.9(-8)
La-140	2.5(-2)	5.6(-8)
Ce-141	1.5(-4)	3.3(-10)
Ce-143	2.8(-3)	6.2(-9)
Ce-144	3.9(-3)	8.7(-9)
W-187	2.5(-3)	5.6(-9)
Np-239	2.2(-3)	4.9(-9)

* The concentrations given are reactor coolant entering the letdown line. These concentrations are obtained from Tables 2-9 and 2-10. N-16 and H-3 concentrations are obtained from Reference 1.

** Based on primary-to-secondary leakage of 75 lb/day. The concentrations given are for steam leaving a steam generator.

TABLE 2-4

PARAMETERS USED TO DESCRIBE THE REFERENCE PRESSURIZED WATER REACTOR WITH U-TUBE STEAM GENERATORS

Parameter	Symbol	Units	Nominal Value	Range Maximum	Range Minimum
Thermal Power	P	MWt	3,400	3,800	3,000
Steam flow rate	FS	lb/hr	1.5(7)	1.7(7)	1.3(7)
Weight of water in reactor coolant system	WP	lb	5.5(5)	6.0(5)	5.0(5)
Weight of water in all steam generators	WS	lb	4.5(5)	5.0(5)	4.0(5)
Reactor coolant letdown flow (purification)	FD	lb/hr	3.7(4)	4.2(4)	3.2(4)
Reactor coolant letdown flow (yearly average for boron control)	FB	lb/hr	500	1,000	250
Steam generator blowdown flow (total)	FBD	lb/hr	75,000	100,000	50,000
Fraction of radioactivity in blowdown stream that is not returned to the secondary coolant system	NBD	--	1.0*	1.0	0.9
Flow through the purification system cation demineralizer	FA	lb/hr	3,700	7,500	0.0
Ratio of condensate demineralizer flow rate to the total steam flow rate	NC	--	0.0**	0.01	0.0
Ratio of the total amount of noble gases routed to gaseous radwaste from the purification system to the total amount of noble gases routed from the primary coolant system to the purification system (not including the boron recovery system)	Y	--	0.0	0.01	0.0

* This value is based on a nominal case of blowdown through blowdown demineralizers back to the main condenser (no condensate demineralizers). Value taken from blowdown demineralizer DF's in Section 2.2.18. Value for cesium and rubidium is 0.9.

** This value is based on a nominal case of no condensate demineralizers. For a U-tube steam generator PWR with full flow condensate demineralizers, a value of NC = 1.0 is used by the PWR-GALE Code. For a U-tube steam generator PWR with condensate demineralizers and pumped forward feedwater heater drains, the value for NC used by the PWR-GALE Code is 0.2 for iodine, and 0.1 for Cs, Rb and other nuclides as discussed on page 2-20.

TABLE 2-5

PARAMETERS USED TO DESCRIBE THE REFERENCE PRESSURIZED WATER REACTOR WITH ONCE-THROUGH STEAM GENERATORS

Parameter	Symbol	Units	Nominal Value	Range	
				Maximum	Minimum
Thermal Power	P	MWt	3,400	3,800	3,000
Steam flow rate	FS	lb/hr	1.5(7)	1.7(7)	1.3(7)
Weight of water in reactor coolant system	WP	lb	5.5(5)	6.0(5)	5.0(5)
Weight of water in all steam generators	WS	lb	1.0(5)	*	*
Reactor coolant letdown flow (purification)	FD	lb/hr	3.7(4)	4.2(4)	3.2(4)
Reactor coolant letdown flow (yearly average for boron control)	FB	lb/hr	500	1,000	250
Flow through the purification system cation demineralizer	FA	lb/hr	3,700	7,500	0.0
Ratio of condensate demineralizer flow rate to the total steam flow rate	NC	--	0.65**	0.75	0.55
Ratio of the total amount of noble gases routed to gaseous radwaste from the purification system to the total amount routed from the primary coolant system to the purification system (not including the boron recovery system)	Y	--	0.0	0.01	0.0

* The secondary coolant inventory is not of importance in a once-through steam generator plant because decay is not an important removal mechanism for most of the isotopes.

** For a PWR that is within the range indicated above, i.e., a PWR with pumped forward feedwater heater drains, the value for NC used by the PWR-GALE Code is 0.2 for iodine and 0.1 for Cs, Rb and other nuclides, as discussed on page 2-20. For a PWR that has full flow condensate demineralizer, a value of NC = 1.0 is used by the PWR-GALE Code.

TABLE 2-6

VALUES USED IN DETERMINING ADJUSTMENT FACTORS FOR PRESSURIZED WATER REACTORS

Symbol	Description	Element Class: Noble Gases	Halogens	Cs, Rb	Water Activation Products	H-3	Other Nuclides
NA	Fraction of material removed in passing through the cation demineralizer	0.0	0.0	0.9	0.0	0.0	0.9*
NB	Fraction of material removed in passing through the purification demineralizer	0.0	0.99	0.5	0.0	0.0	0.98
R	Removal rate reactor coolant (Hr^{-1})**	0.0009	0.067	0.037	0.0	***	0.066
NS	Ratio of concentration in steam to that in water in the steam generator						
	U-tube steam generator	†	0.01	0.005	††	1.0	0.005
	Once-through steam generator	†	1.0	1.0	1.0	1.0	1.0
NX	Fraction of activity removed in passing through the condensate demineralizers	0.0	0.9	0.5	0.0	0.0	0.9
r	Removal rate secondary coolant (Hr^{-1})†††						
	U-tube steam generator	†	0.17	0.15	††	***	0.17
	Once-through steam generator	†	27	7.5	††	***	14
FL	Primary-to-secondary leakage (lb/day)	75	75	75	75	75	75

* These represent effective removal terms and include mechanisms such as plateout. Plateout would be applicable to nuclides such as Mo and corrosion products.

TABLE 2-6 (continued)

** These values of R apply to the reference PWR's whose parameters are given in Tables 2-4 and 2-5 and have been used in developing Tables 2-7 and 2-8. For PWR's not included in Tables 2-4 and 2-5, the appropriate value for R may be determined by the following equations.

$$R = \frac{FB + (FD - FB)Y}{WP} \quad \text{for noble gases}$$

$$R = \frac{(FD)(NB) + (1 - NB)(FB + (FA)(NA))}{WP} \quad \text{for halogens, Cs, Rb, and other nuclides}$$

*** The concentration of tritium is a function of (1) the inventory of tritiated liquids in the plant, (2) the rate of production of tritium due to activation in the reactor coolant as well as releases from the fuel, and (3) the extent to which tritiated water is recycled or discharged from the plant. The tritium concentrations given in Tables 2-2 and 2-3 are representative of PWR's with a moderate amount of tritium recycle and can be used to calculate source terms in accordance with Regulatory Guide 1.112, "Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors."

† Noble gases are rapidly transported out of the water in the steam generator and swept out of the vessel in the steam; therefore, the concentration in the water is negligible and the concentration in the steam is approximately equal to the ratio of the release rate to the steam generator and the steam flow rate. These noble gases are removed from the system at the main condenser.

†† Water activation products exhibit varying chemical and physical properties in reactor coolants that are not well defined. Most are not effectively removed by the demineralizers, but their concentrations are controlled by decay.

††† These values of r apply to the reference PWR's whose parameters are given in Tables 2-4 and 2-5 and have been used in developing Tables 2-7 and 2-8. For PWR's not included in Tables 2-4 and 2-5, the appropriate value for r may be determined by the following equation:

$$r = \frac{(FBD)(NBD) + (NS)(FS)(NC)(NX)}{WS} \quad \text{for halogens, Cs, Rb, and other nuclides}$$

TABLE 2-7

ADJUSTMENT FACTORS FOR PWR's WITH U-TUBE STEAM GENERATORS

	Adjustment Factors		
		Secondary Coolant	
Element Class	Reactor Water (f)*	Water	Steam
Noble gases	$\frac{162P}{WP}\ \frac{0.0009 + \lambda}{R + \lambda}$**	--	$\frac{1.5 \times 10^7}{FS}\ f$
Halogens	$\frac{162P}{WP}\ \frac{0.067 + \lambda}{R + \lambda}$	$\frac{4.5 \times 10^5}{WS}\ \frac{0.17 + \lambda}{r + \lambda}\ f$	$\frac{4.5 \times 10^5}{WS}\ \frac{0.17 + \lambda}{r + \lambda}\ f$
Cs, Rb	$\frac{162P}{WP}\ \frac{0.037 + \lambda}{R + \lambda}$	$\frac{4.5 \times 10^5}{WS}\ \frac{0.15 + \lambda}{r + \lambda}\ f$	$\frac{4.5 \times 10^5}{WS}\ \frac{0.15 + \lambda}{r + \lambda}\ f$
Water activation products	1.0	$\frac{4.5 \times 10^5}{WS}$	$\frac{4.5 \times 10^5}{WS}$
Tritium	***	***	***
Other nuclides	$\frac{162P}{WP}\ \frac{0.066 + \lambda}{R + \lambda}$	$\frac{4.5 \times 10^5}{WS}\ \frac{0.17 + \lambda}{r + \lambda}\ f$	$\frac{4.5 \times 10^5}{WS}\ \frac{0.17 + \lambda}{r + \lambda}\ f$

* f is the reactor water adjustment factor and is used in the secondary coolant adjustment factors.

** λ is the isotopic decay constant (hr^{-1}).

*** The concentration of tritium is a function of (1) the inventory of tritiated liquids in the plant, (2) the rate of production of tritium due to activation in the reactor coolant as well as releases from the fuel, and (3) the extent to which tritiated water is recycled or discharged from the plant. The tritium concentrations given in Tables 2-2 and 2-3 are representative of PWR's with a moderate amount of tritium recycle and can be used to calculate source terms in accordance with Regulatory Guide 1.112.

TABLE 2-8

ADJUSTMENT FACTORS FOR PWR's WITH ONCE-THROUGH STEAM GENERATORS

Nuclide	Adjustment Factors: Reactor Water (f)*	Adjustment Factors: Secondary Coolant
Noble gases	$\frac{162P}{WP} \frac{0.0009 + \lambda}{R + \lambda}$	$\frac{1.5 \times 10^7}{FS} f$
Halogens	$\frac{162P}{WP} \frac{0.067 + \lambda}{R + \lambda}$	$\frac{10^5}{WS} (\frac{27 + \lambda}{r + \lambda}) f$
Cs, Rb	$\frac{162P}{WP} \frac{0.037 + \lambda}{R + \lambda}$	$\frac{10^5}{WS} (\frac{7.5 + \lambda}{r + \lambda}) f$
Water activation products	1.0	$\frac{1.0 \times 10^5}{WS}$
Tritium	**	**
Other nuclides	$\frac{162P}{WP} \frac{0.066 + \lambda}{R + \lambda}$	$\frac{10^5}{WS} (\frac{14 + \lambda}{r + \lambda}) f$

* f is the reactor water adjustment factor and is used in the secondary coolant adjustment factors.

** The concentration of tritium is a function of (1) the inventory of tritiated liquids in the plant, (2) the rate of production of tritium due to activation in the reactor coolant as well as releases from the fuel, and (3) the extent to which tritiated water is recycled or discharged from the plant. The tritium concentrations given in Tables 2-2 and 2-3 are representative of PWR's with a moderate amount of tritium recycle and can be used to calculate source terms in accordance with the Regulatory Guide 1.112.

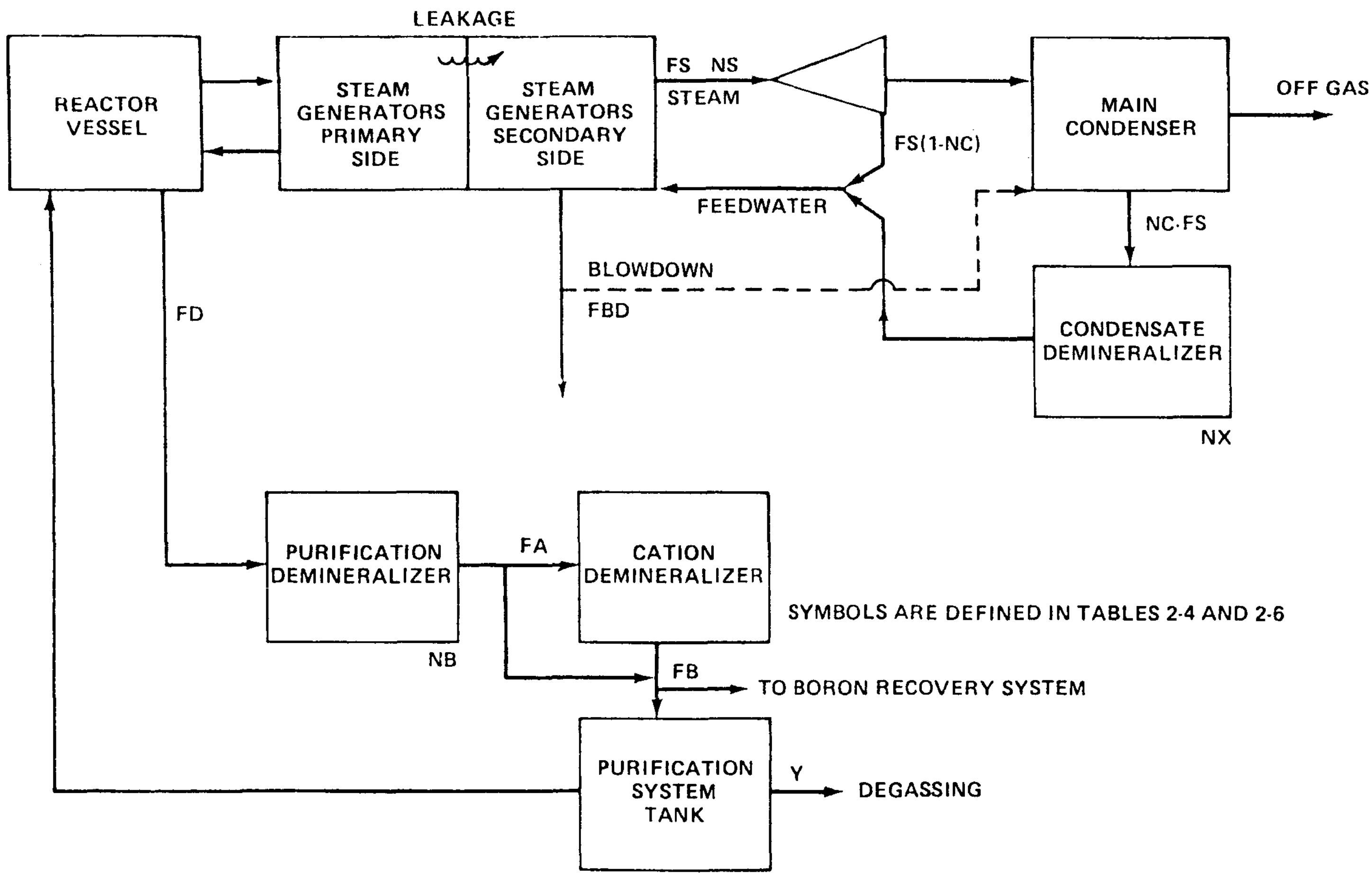


FIGURE 2-1 REMOVAL PATHS FOR PRESSURIZED WATER REACTOR WITH U-TUBE STEAM GENERATORS

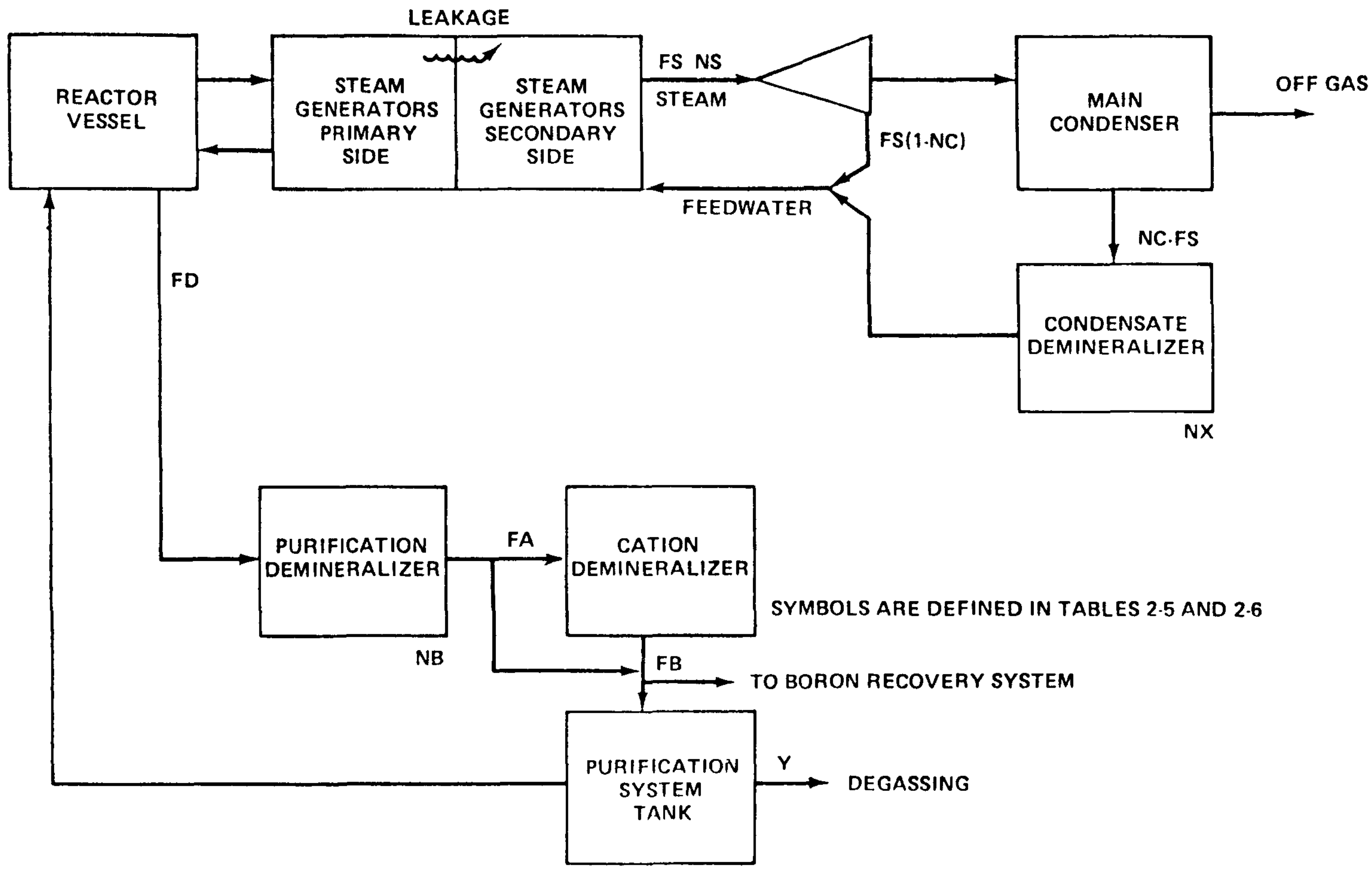


FIGURE 2-2 REMOVAL PATHS FOR PRESSURIZED WATER REACTOR WITH ONCE-THROUGH STEAM GENERATORS

$$C = \frac{s}{w(\lambda + R)K}$$

where

C is the specific activity (in μCi/g)

K is a conversion factor, 454 g/lb

R is the removal rate of the isotope from the system due to demineralization, leakage, etc. (hr^{-1}). (If considering secondary coolant R = r).

s is the rate of release to and/or production of the isotope in the system (in μCi/hr)

w is the fluid weight (in lb), and

is the decay constant (hr^{-1}).

The following sample calculations illustrate the method by which the PWR-GALE Code will adjust the radionuclide concentrations in Tables 2-2 and 2-3. As indicated in Tables 2-7 and 2-8, adjustment factors will be calculated for noble gases, halogens, Cs, Rb, and other nuclides.

As an example, the sample case parameters shown below compare with the range of values in Table 2-4 as follows.

Parameter (U-tube steam generator PWR)	Value	Range
Thermal power level, MWt	3800	3000 - 3800
Steam flow rate, lb/hr	17×10^6	$13 \times 10^6 - 17 \times 10^6$
Mass of reactor coolant, lb	5.5×10^5	$5.0 \times 10^5 - 6.0 \times 10^5$
Water weight in all steam generators, lb	4.4×10^5	$4.0 \times 10^5 - 5.0 \times 10^5$
Reactor coolant letdown, lb/hr	4.9×10^4	$3.2 \times 10^4 - 4.2 \times 10^4$
Cation demineralizer flow, lb/hr	4.9×10^3	$0 - 7.5 \times 10^3$
Shim bleed rate - yearly average, lb/hr	650	250 - 1000
Steam generator blowdown flow, lb/hr	60,000	50,000 - 100,000
Fraction of blowdown activity not returned to secondary system	0.99	0.9 - 1.0
Cation demineralizer flow, lb/hr	4900	0.0 - 7500
Condensate demineralizer flow fraction	0.0	0.0 - 0.01
Y (see definition in Table 2-4 and page 1-26)		

Since in this example the parameter for reactor coolant letdown rate (4.9×10^4 lb/hr) is outside the range specified in Table 2-4 ($3.2 - 4.2 \times 10^4$ lb/hr), and the sample case employs continuous purging of the volume control tank, the primary coolant activity is recalculated using the actual design value for all parameters employing the methods described below.

1. Noble Gases (Xe-133 is used as an example)

Using the equation for noble gases in Table 2-7, the adjustment factor, f, is calculated as follows:

$$f = \frac{162P}{WP} \; \frac{0.0009 + \lambda}{R + \lambda} \qquad (1)$$

where the terms in the equations are defined in Tables 2-4 and 2-6.

In calculating f, the variable R is calculated first by using the equation given in Table 2-6 for noble gases

$$R = \frac{FB + (FD - FB)(Y)}{WP} \qquad (2)$$

where the terms of the equation are as defined in Tables 2-4 and 2-6.

Use the sample case parameters given above and the noble gas parameters given in Table 2-6 and substitute in Equation (2) above.

$$R = \frac{650 + (4.9 \times 10^4 - 650) \times 0.25}{5.5 \times 10^5} = 0.023$$

Use the value of R in Equation (1) above.

$$f = \frac{162 \times 3800}{5.5 \times 10^5} \; \frac{0.0009 + 5.5 \times 10^{-3}}{0.023 + 5.5 \times 10^{-3}} = 0.25$$

The adjusted Xe-133 primary coolant concentration

= (adjustment factor) x (standard Xe-133 concentration)

= 0.25×2.6 µCi/g = 0.65 µCi/g

2. Halogens (I-131 is used as an example)

Using the equation for halogens in Table 2-7, the adjustment factor, f, is calculated as follows:

$$f = \frac{162P}{WP} \frac{0.067 + \lambda}{R + \lambda} \qquad (3)$$

where the terms in the equations are defined in Tables 2-4 and 2-6.

In calculating f, the variable R is calculated first by using the equation given in Table 2-6.

$$R = \frac{(FD)(NB) + (1 - NB)(FB + (FA)(NA))}{WP} \qquad (4)$$

where the terms in the equation are as defined in Tables 2-4 and 2-6.

Use the sample case parameters given above and the halogen parameters given in Table 2-6 and substitute in Equation (4) above.

$$R = \frac{(4.9 \times 10^4 \times 0.99) + (1 - 0.99)(650 + (4900)(0.0))}{5.5 \times 10^5} = 0.088$$

Use the value of R in Equation (3) above.

$$f = \frac{162(3800)}{5.5 \times 10^5} \frac{0.067 + 3.6 \times 10^{-3}}{0.088 + 3.6 \times 10^{-3}} = 0.86$$

The adjusted I-131 concentration

= (adjustment factor) x (standard I-131 concentration)

= 0.86 x 0.045 μCi/g = 0.039 μCi/g

3. Cs, Rb (Cs-137 is used as an example)

Using the equation for Cs and Rb in Table 2-7, the adjustment factor, f, is calculated as follows:

$$f = \frac{162P}{WP} \frac{0.037 + \lambda}{R + \lambda} \qquad (5)$$

where the terms in the equation are as defined in Tables 2-4 and 2-6.

In calculating f, the variable R is calculated first by using Equation (4) above. The Cs and Rb parameters given in Table 2-6 and the sample case parameters given in Table 2-9 are used in the equation.

$$R = \frac{(4.9 \times 10^4 \times 0.5) + (0.5)(650 + (4900)(0.9))}{5.5 \times 10^5} = 0.05$$

Use the value of R in Equation (5) above.

$$f = \frac{162(3800)}{5.5 \times 10^5} \frac{0.037 + 2.6 \times 10^{-6}}{0.05 + 2.6 \times 10^{-6}} = 0.83$$

The adjusted Cs-137 concentration

= (adjustment factor) x (standard Cs-137 concentration)

$= 0.83 \times 9.4 \times 10^{-3}\ \mu Ci/g = 7.8 \times 10^{-3}\ \mu Ci/g$

4. Other Nuclides (Te-132 is used as an example)

Using the equation for other nuclides in Table 2-7, the adjustment factor, f, is calculated as follows:

$$f = \frac{162P}{WP} \frac{0.066 + \lambda}{R + \lambda} \quad (6)$$

where the terms in the equation are as defined in Tables 2-4 and 2-6.

In calculating f, the variable R is calculated first by using Equation (4) above. The parameters for other nuclides given in Table 2-6 and the sample case parameters given in Table 2-9 are used in the equation.

$$R = \frac{(4.9 \times 10^4)\ (0.98) + (1 - 0.98)(650 + (4900)(0.9))}{5.5 \times 10^5} = 0.087$$

Use the value of R in equation (6) above.

$$f = \frac{162\ (3800)}{5.5 \times 10^5} \frac{(0.066 + 8.9 \times 10^{-3})}{0.087 + 8.9 \times 10^{-3}} = 0.87$$

The adjusted concentration of Te-132

= (adjustment factor) x (standard Te-132 concentration)

$= 0.87 \times 1.7 \times 10^{-3}\ \mu Ci/g = 1.5 \times 10^{-3}\ \mu Ci/g$

A similar method is used in the PWR-GALE Code to adjust secondary coolant concentrations for reactors with parameters outside the ranges specified in Tables 2-4 and 2-5.

The radionuclide primary coolant concentrations in Tables 2-2 and 2-3 are based on data submitted by utilities with operating PWR's (Ref. 3). The data are also based on measurements taken by the NRC at Ft. Calhoun (Ref. 4), Zion 1 and 2 (Ref. 5), Turkey Point 3 and 4 (Ref. 6), Rancho Seco (Ref. 43), and Prairie Island 1 and 2 (Ref. 42); by EPRI (Ref. 7) at Three Mile Island 1 and Calvert Cliffs; and by measurements at various other PWR's (Ref. 8, 9, and 39).

These data are summarized in Table 2-9 and Table 2-10 indicating the average value of the nuclide concentration for each plant, the years over which the data was obtained, and the total number of years of data for each nuclide.

The secondary coolant concentrations are based on the primary coolant concentrations as obtained above, on 75 lb/day primary-to-secondary leakage in the steam generators, on appropriate steam generator carryover factors, on the appropriate main steam flow, steam generator blowdown flow and fraction of a blowdown flow returned to the secondary coolant, as defined in the plant design, and on the fraction of the nuclides in the main steam which return to the steam generators.

The secondary coolant concentrations are based on 75 lb/day primary-to-secondary leakage. The primary-to-secondary leakage rate experience for 79 years of experience at operating PWR's is given in Table 2-11. The average primary-to-secondary leakage rate in Table 2-11 is 75 lb/day. Westinghouse estimates that the data in Table 2-11 are accurate within ± 25% (Ref. 8, 39).

For plants using recirculating U-tube steam generators, carryover due to mechanical entrainment is based on 0.5% moisture in the steam. Table 2-12 provides measured values for moisture carryover at five operating PWR's that use recirculating U-tube steam generators. Based on data from Turkey Point 3 and 4 (Ref. 6) a value of 1% iodine carryover with the steam is used in our evaluations. For once-through steam generators, it is assumed that 100% of both nonvolatile and volatile species is carried over with the steam since this type of steam generator has no liquid reservoir and 100% of the feed is converted to steam.

For PWR's that use condensate demineralizers in the secondary system, the nominal value of the ratio of the condensate demineralizer flow rate to the total steam flow rate is 0.65. This indicates that the nominal case is a design which utilizes a pumped forward model, that is, one in which the reactor steam flow is split with 65% flowing to the low pressure turbines and the main condenser, and 35% pumped forward to the feedwater. The fraction pumped forward to the feedwater does not undergo any treatment in the condensate demineralizers. We have determined that the iodine, Cs, Rb, and "Other Nuclides" of Table 2-2 and Table 2-3 preferentially go with the "pumped forward" fraction. The reason for this is that these nuclides show a tendency to go with the condensed steam in the moisture separator-reheater drains and with the extraction steam lines from the high pressure turbines to the feedwater system. Based on data provided in Ref. 6, 7, 12 and 13 for Turkey Point, Point Beach and Brunswick, the percentages used in the PWR-GALE Code for the amount of activity which is pumped forward and which bypasses the condensate demineralizers is 80% for iodine and 90% for Cs, Rb, and "Other Nuclides" of Table 2-2 and Table 2-3. Since the remainder of the nuclides listed in Tables 2-2 and 2-3 are not removed in the condensate demineralizers, we have not considered the magnitude of bypass for those nuclides.

TABLE 2-9

SUMMARY OF I-131 AND I-133 PRIMARY COOLANT CONCENTRATIONS IN PWR'S*
(μCi/g)

Isotope	H.B. Robinson 2 (1973-1978)**	Arkansas 1 (1976)	D.C. Cook 1 (1976-1978)	Trojan (1977-1978)	Palisades (1972-1976)	Point Beach 1/2 (1972-1979)	R.E. Ginna (1971-1978)
I-131	3.1E-03	7.3E-03	7.8E-03	1.3E-02	1.2E-02	7.7E-02	2.2E-01
I-133	1.2E-02	***	1.8E-02	1.5E-02	1.6E-02	3.6E-01	6.9E-01

Isotope	Fort Calhoun 1 (1976-1977)	Zion 1/2 (1975-1978)	Turkey Point 3/4 (1974-1978)	Three Mile Island 1 (1975-1977)	Calvert Cliffs 1 (1976)	Beaver Valley 1 (1977-1978)
I-131	1.8E-01	2.3E-02	2.1E-02	2.1E-02	3.6E-02	1.8E-03
I-133	1.6E-01	6.9E-02	6.9E-02	***	***	5.5E-03

Isotope	Indian Point 2/3 (1975-1978)	Kewaunee (1975-1978)	Prairie Island 1/2 (1975-1981)	Surry 1/2 (1973-1978)	J.M. Farley 1 (1978)	Yankee Rowe (1975)	Rancho Seco (1978)
I-131	2.5E-02	5.3E-03	8.1E-03	2.1E-02	5.4E-04	4.9E-03	1.3E-02
I-133	5.1E-02	1.3E-02	1.0E-02	3.2E-02	1.9E-03	2.4E-02	4.7E-02

* Data in this table are based on I-131 and I-133 primary coolant concentrations in Ref. 3 through 9, 42 and 43, and have been adjusted to the NSS parameters listed in Table 2-4 of this report. These adjustments were made using the individual plant parameters and the nominal plant parameters (Table 2-4) and adjusting the actual coolant concentration using the equations in Table 2-7 of this report.

** Data in this table were gathered during the indicated inclusive dates. It does not necessarily imply that data were available during each of the years covered by the period, nor does it mean that the number of data points should be the same for each radionuclide.

*** No value reported.

TABLE 2-10

SUMMARY OF RADIONUCLIDE PRIMARY COOLANT CONCENTRATIONS IN PWR'S*
(μCi/g)

Isotope	H.B. Robinson 2 (1973-1978)**	Arkansas 1 (1976)	D.C. Cook 1 (1976-1978)	Trojan (1977-1978)	Palisades (1972-1976)	Point Beach 1/2 (1972-1979)	R.E. Ginna (1971-1978)
Kr-85m	1.8E-02	8.0E-03	3.4E-02	2.2E-02	4.0E-01	2.2E-01	2.4E-01
Kr-85	***	5.2E-03	***	3.5E+00	8.3E-04	2.9E-02	***
Kr-87	1.7E-02	6.7E-03	4.7E-02	4.2E-02	4.6E-01	1.1E-01	3.8E-01
Kr-88	2.3E-02	1.3E-02	5.7E-02	4.0E-02	7.5E-01	3.0E-01	6.3E-01
Xe-131m	***	***	***	***	4.4E+00	9.4E-01	***
Xe-133m	1.6E-03	3.3E-03	1.5E-02	2.7E-03	1.9E-01	6.5E-02	***
Xe-133	2.3E-01	2.1E-01	5.8E-01	5.7E-01	4.9E+00	2.8E+00	6.0E+00
Xe-135m	2.1E-02	***	***	***	1.0E-02	1.4E-01	1.5E-01
Xe-135	7.8E-02	2.7E-02	1.9E-01	1.2E-01	1.1E+00	1.1E+00	2.2E+00
Xe-137	***	***	***	***	***	***	***
Xe-138	***	***	***	6.7E-02	2.2E-03	1.7E-01	***
Br-84	***	***	***	***	***	***	***
I-132	1.5E-02	***	***	1.9E-02	7.1E-03	3.6E-01	7.3E-01
I-134	3.2E-02	***	2.4E-02	2.2E-02	1.0E-02	6.2 E-01	1.2E+00
I-135	1.9E-02	***	2.0E-02	1.7E-02	9.2E-03	5.7E-01	6.6E-01
Rb-88	***	***	***	3.5E-02	2.6E-02	1.7E-01	3.7E-01
Cs-134	1.9E-03	5.6E-04	2.7E-03	6.0E-04	1.7E-04	1.4E-02	1.1E-02
Cs-136	3.1E-04	***	5.2E-03	7.2E-04	4.6E-05	2.2E-03	***
Cs-137	2.3E-03	1.5E-03	4.9E-03	1.3E-03	2.6E-04	1.1E-02	3.1E-02
N-16	***	***	***	***	***	***	***
H-3	***	6.3E-02	2.1E-01	***	7.5E-02	5.0E-01	6.7E-01
Na-24	1.3E-01	8.7E-02	1.2E-02	1.3E-02	5.6E-03	7.6E-02	***
Cr-51	3.5E-04	3.2E-03	***	***	9.1E-03	***	1.1E-04
Mn-54	3.4E-04	7.6E-04	8.3E-03	9.7E-04	1.1E-04	2.8E-03	2.5E-05
Fe-55	***	***	***	***	***	***	***
Fe-59	1.4E-05	1.6E-03	***	***	1.6E-04	***	2.6E-05
Co-58	1.3E-03	7.0E-03	1.4E-02	2.2E-03	3.4E-03	9.6E-03	7.6E-04
Co-60	3.5E-04	6.4E-04	4.5E-03	3.4E-05	1.1E-04	2.3E-04	1.6E-04
Zn-65	1.7E-05	***	***	***	7.0E-05	***	***
Sr-89	2.3E-05	***	***	***	***	2.4E-04	***
Sr-90	5.2E-06	***	***	***	1.1E-04	***	***

TABLE 2-10 (continued)

SUMMARY OF RADIONUCLIDE PRIMARY COOLANT CONCENTRATIONS IN PWR'S*
(μCi/g)

Isotope	H.B. Robinson 2 (1973-1978)**	Arkansas 1 (1976)	D.C. Cook 1 (1976-1978)	Trojan (1977-1978)	Palisades (1972-1976)	Point Beach 1/2 (1972-1979)	R.E. Ginna (1971-1978)
Sr-91	4.9E-04	***	***	***	1.1E-04	***	***
Y-91m	***	***	***	***	***	***	***
Y-91	***	***	***	***	***	***	***
Y-93	***	***	***	***	***	***	***
Zr-95	1.3E-05	3.4E-04	4.5E-03	***	1.0E-04	***	1.5E-03
Nb-95	1.3E-05	3.1E-04	2.4E-03	***	7.6E-05	3.6E-04	8.1E-05
Mo-99	***	7.2E-05	***	***	5.7E-04	3.8E-02	4.1E-04
Tc-99m	***	***	***	***	7.3E-04	2.5E-02	***
Ru-103	***	***	***	***	***	***	1.2E-03
Ru-106	***	***	***	***	***	***	***
Ag-110m	***	***	***	***	***	8.8E-03	***
Te-129m	***	***	***	***	***	***	***
Le-129	***	***	***	***	***	***	***
Te-131m	***	***	***	***	***	***	***
Te-131	***	***	***	***	***	***	***
Te-132	***	1.3E-03	***	***	6.6E-05	8.8E-03	***
Ba-140	2.0E-04	***	***	***	6.2E-06	1.6E-01	5.9E-05
La-140	9.2E-05	***	***	***	3.0E-05	5.2E-01	***
Ce-141	***	***	***	***	***	***	***
Ce-143	***	***	***	***	***	***	***
Ce-144	2.6E-04	1.4E-03	***	***	***	4.5E-02	***
W-187	3.4E-04	***	***	***	5.8E-04	***	***
Np-239	***	***	***	***	***	***	2.0E-03

TABLE 2-10 (continued)

SUMMARY OF RADIONUCLIDE PRIMARY COOLANT CONCENTRATIONS IN PWR'S*
(μCi/g)

Isotope	Fort Calhoun 1 (1976-1977)	Zion 1/2 (1975-1978)	Turkey Point 3/4 (1974-1978)	Indian Pt 2/3 (1975-1978)	Yankee Rowe (1975)	Calvert Cliffs 1 (1976)	Three Mile Island 1 (1975-1977)	Prairie Island 1/2 (1981)	Rancho Seco (1979)
Kr-85m	1.9E-01	***	7.8E-02	3.4E-02	5.7E-03	***	***	4.9E-04	5.5E-02
Kr-85	3.4E-02	***	***	***	***	***	***	3.3E-04	2.2E-01
Kr-87	1.9E-01	***	9.0E-02	***	7.6E-03	***	***	1.1E-03	5.9E-02
Kr-88	3.2E-01	***	1.3E-01	7.3E-02	1.9E-02	***	***	1.1E-03	9.9E-02
Xe-131m	6.8E-02	***	1.2E-03	***	***	***	***	4.2E-05	3.5E-03
Xe-133m	1.6E-01	***	9.1E-03	***	***	***	***	7.2E-05	4.5E-02
Xe-133	6.7E+00	***	8.8E-01	8.3E-01	2.1E-01	***	***	2.2E-03	1.5E+00
Xe-135m	9.5E-02	***	1.7E-01	1.0E-01	***	***	***	1.4E-03	6.0E-01
Xe-135	9.3E-01	***	5.1E-01	1.9E-01	3.0E-02	***	***	3.6E-03	4.6E-01
Xe-137	***	***	3.4E-02	***	***	***	***	***	***†
Xe-138	1.8E-01	***	7.6E-02	***	***	***	***	2.9E-03	1.7E-01
Br-84	***	***	1.1E-02	***	***	***	***	1.0E-03	5.5E-02
I-132	7.1E-02	9.6E-02	9.3E-02	***	1.8E-02	***	***	5.1E-03	5.3E-02
I-134	3.8E-02	1.3E-01	1.5E-01	***	***	***	***	9.0E-03	8.3E-02
I-135	7.4E-02	1.1E-01	8.6E-02	***	***	***	***	5.8E-03	6.0E-02
Rb-88	5.0E-01	2.3E-01	1.0E-01	***	***	***	***	5.7E-03	1.5E-01
Cs-134	1.8E-02	9.4E-03	1.8E-03	1.9E-02	***	***	***	2.2E-05	7.7E-03
Cs-136	1.7E-03	1.2E-03	1.1E-04	***	***	***	***	3.2E-06	1.9E-04
Cs-137	2.0E-02	1.2E-02	3.1E-03	2.4E-02	***	***	***	6.7E-05	9.4E-03
N-16	***	***	***	***	***	***	***	***	***
H-3	1.3E-01	1.5E-01	***	***	***	4.4E-02	1.2E-01	2.9E-01	2.5E-01
Na-24	8.8E-03	1.0E-01	1.0E-02	3.6E-03	***	***	***	9.0E-03	1.4E-02
Cr-51	1.5E-02	2.1E-03	3.4E-04	***	1.7E-03	***	***	3.0E-05	6.4E-03
Mn-54	4.4E-03	2.2E-03	3.9E-05	1.5E-02	1.1E-04	***	***	1.0E-05	6.8E-04
Fe-55	6.5E-04	1.6E-04	***	***	***	***	***	2.1E-05	9.1E-03
Fe-59	5.2E-04	6.2E-04	2.3E-04	***	6.9E-04	***	***	1.5E-06	5.2E-04
Co-58	1.4E-02	4.6E-03	6.7E-04	3.6E-03	5.8E-04	***	***	8.0E-05	2.4E-02
Co-60	1.0E-03	7.8E-04	1.2E-04	3.1E-03	4.7E-04	***	***	1.6E-05	9.2E-04
Zn-65	2.6E-03	2.4E-03	1.6E-05	***	***	***	***	1.7E-06	2.2E-05
Sr-89	6.8E-04	7.7E-05	6.8E-07	***	***	***	***	6.6E-06	***
Sr-90	4.2E-06	3.4E-06	1.6E-06	***	***	***	***	5.4E-08	***

TABLE 2-10 (continued)

SUMMARY OF RADIONUCLIDE PRIMARY COOLANT CONCENTRATIONS IN PWR'S*
(μCi/g)

Isotope	Fort Calhoun 1 (1976-1977)	Zion 1/2 (1975-1978)	Turkey Point 3/4 (1974-1978)	Indian Pt 2/3 (1975-1978)	Yankee Rowe (1975)	Calvert Cliffs 1 (1976)	Three Mile Island 1 (1975-1977)	Prairie Island 1/2 (1981)	Rancho Seco (1979)
Sr-91	***	3.8E-03	3.7E-04	***	***	***	***	3.3E-05	7.2E-04
Y-91m	***	8.7E-04	***	***	***	***	***	5.0E-05	***
Y-91	5.0E-06	4.4E-06	***	***	***	***	***	4.3E-07	1.8E-05
Y-93	***	7.9E-03	2.2E-03	***	***	***	***	2.1E-04	2.6E-03
Zr-95	1.5E-03	4.2E-04	4.5E-05	***	2.8E-04	***	***	4.6E-06	2.9E-04
Nb-95	1.3E-03	2.2E-04	3.8E-05	***	2.4E-04	***	***	3.9E-06	4.6E-04
Mo-99	5.7E-03	3.5E-03	8.1E-04	***	5.0E-03	***	***	1.3E-04	1.7E-03
Tc-99m	4.1E-04	***	2.7E-06	***	4.9E-03	***	***	***	***
Ru-103	5.4E-02	1.8E-04	2.1E-05	***	***	***	***	6.7E-07	7.0E-05
Ru-106	***	9.0E-02	***	***	***	***	***	***	***
Ag-110m	2.2E-04	3.1E-03	1.1E-05	***	***	***	***	3.7E-06	9.7E-05
Te-129m	2.1E-04	3.8E-04	1.9E-04	***	***	***	***	2.0E-06	***
Te-129	***	***	2.4E-02	***	***	***	***	***	***
Te-131m	***	2.1E-03	3.7E-04	***	***	***	***	***	***
Te-131	***	***	7.9E-03	***	***	***	***	***	7.4E-03
Te-132	***	1.8E-04	4.0E-05	***	***	***	***	1.2E-06	3.1E-05
Ba-140	1.1E-03	1.0E-03	1.1E-04	***	***	***	***	1.9E-05	2.5E-04
La-140	4.2E-04	1.8E-03	1.3E-04	***	***	***	***	1.4E-05	1.1E-04
Ce-141	4.3E-04	1.1E-04	1.7E-05	***	***	***	***	***	4.6E-05
Ce-143	8.2E-03	4.6E-04	4.9E-05	***	***	***	***	1.9E-05	***
Ce-144	***	1.4E-04	1.2E-05	***	2.6E-05	***	***	5.4E-06	4.6E-04
W-187	1.4E-02	3.1E-03	3.0E-04	***	***	***	***	1.1E-04	2.9E-03
Np-239	1.2E-02	9.3E-04	1.0E-04	***	***	***	***	3.7E-06	7.6E-04

* See Footnote of Table 2-9.

** See Footnote of Table 2-9.

*** See Footnote of Table 2-9.

† Data unreliable.

TABLE 2-11

MONTHLY AVERAGE* PRIMARY/SECONDARY LEAKAGE (REF. 8, 39)
(gal/day at 70°F; density = 8.3 lb/gal)

1970

Plant	J	F	M	A	M	J	J	A	S	O	N	D
San Onofre	4	4	4	4	3	9	11	8	14	S**	S	0
Connecticut Yankee	0	10	0	S	0	0	20	10	20	0	0	0
R. E. Ginna							0	0	0	0	0	0
Point Beach 1												0

1971

Plant	J	F	M	A	M	J	J	A	S	O	N	D
San Onofre	0	0	0	0	0	0	0	0	0	0	0	0
Connecticut Yankee	0	30	15	0	0	10	20	20	15	40	40	40
R. E. Ginna	0	0	S	S	0	0	0	0	0	0	0	0
H. B. Robinson			S	S	S	S	S	S	0	50	55	20
Point Beach 1	0	0	0	10	90	100	53	30	20	20	20	20

1972

Plant	J	F	M	A	M	J	J	A	S	O	N	D
San Onofre	S	0	0	0	0	0	22	0	10	30	4	31
Connecticut Yankee	40	40	40	40	40	S	0	0	0	0	0	0
R. E. Ginna	0	0	0	S	S	0	0	0	0	S	0	0
H. B. Robinson	60	60	60	60	3	0	0	0	0	0	0	0
Point Beach 1	40	50	55	55	55	55	55	55	55	S	S	S
Point Beach 2										0	0	0
Surry 1												0
Turkey Point 3												0

* Leakage values listed begin with the first year of commercial operation.

** Shutdown not included in average.

NA - Not Available.

TABLE 2-11 (continued)

MONTHLY AVERAGE* PRIMARY/SECONDARY LEAKAGE
(gal/day at 70°F; density = 8.3 lb/gal)

1973

Plant	J	F	M	A	M	J	J	A	S	O	N	D
San Onofre	3	3	0	0	0	0	0	0	0	0	S	S
Connecticut Yankee	0	0	0	0	10	S	0	S	S	S	S	0
R. E. Ginna	0	0	0	0	0	0	0	0	0	0	0	0
H. B. Robinson	6	6	6	S	0	0	1	1	1	1	7	5
Point Beach 1	S	S	0	0	0	0	0	0	0	0	0	0
Point Beach 2	0	0	0	0	0	0	0	0	0	0	0	0
Surry 1	0	0	0	0	0	0	0	0	0	0	0	0
Turkey Point 3	0	0	0	0	0	0	0	0	0	0	0	0
Surry 2					0	0	0	0	0	0	0	0
Turkey Point 4									0	0	0	0

1974

Plant	J	F	M	A	M	J	J	A	S	O	N	D
San Onofre	0	44	60	60	0	0	0	0	0	2	2	2
Connecticut Yankee	0	0	0	S	0	0	0	0	0	0	0	0
R. E. Ginna	S	S	S	0	0	0	0	0	0	0	0	0
H. B. Robinson 2	2	10	112	98	NA	19	2	1	1	1	1	1
Point Beach 1	0	0	0	S	0	0	0	0	0	0	0	0
Point Beach 2	0	0	0	0	0	0	0	0	0	0	S	S
Surry 1	S	S	0	0	0	115	55	115	115	4	S	S
Turkey Point 3	0	0	0	0	0	0	0	NA	NA	S	S	S
Surry 2	0	0	0	0	S	38	0	0	0	S	S	S
Turkey Point 4	S	0	0	0	0	0	0	22	0	0	0	0
Zion 1	S	S	S	0	0	0	S	S	0	0	0	0
Zion 2									0	0	0	0
Indian Point 2								0	0	0	0	0
Prairie Island 1							0	0	0	0	0	0

TABLE 2-11 (continued)

MONTHLY AVERAGE* PRIMARY/SECONDARY LEAKAGE
(gal/day at 70°F; density = 8.3 lb/gal)

Plant	1975 J	F	M	A	M	J	J	A	S	O	N	D
San Onofre	2	2	2	2	3	5	0	0	0	0	0	0
Connecticut Yankee	0	0	0	0	0	S	0	0	0	0	0	0
R. E. Ginna	0	0	3	S	0	0	0	0	0	0	0	0
H. B. Robinson 2	1	1	1	3	1	5	3	2	0	0	S	7
Point Beach 1	0	61	S	0	1	2	2	2	1	2	S	S
Point Beach 2	0	0	0	0	0	0	0	1	0	0	0	0
Surry 1	S	0	0	0	0	0	0	0	125	S	S	26
Turkey Point 3	0	0	0	0	0	0	0	0	0	0	S	S
Surry 2	0	0	0	0	S	0	0	0	0	0	0	0
Turkey Point 4	0	0	0	S	S	S	7	20	79	0	0	50
Zion 1	0	0	S	0	0	S	0	0	S	0	0	0
Zion 2	0	S	0	0	0	S	0	0	S	0	0	0
Indian Point 2	0	102	S	0	0	0	0	0	0	S	0	0
Prairie Island 1	0	0	0	0	0	0	0	0	0	0	0	0
Prairie Island 2	0	0	0	0	0	0	0	0	0	0	0	0
Cook 1									0	0	0	0

TABLE 2-11 (continued)

MONTHLY AVERAGE* PRIMARY/SECONDARY LEAKAGE
(gal/day at 70°F; density = 8.3 lb/gal)

Plant	1976											
	J	F	M	A	M	J	J	A	S	O	N	D
San Onofre	0	0	0	0	0	0	46	0	0	S	S	S
Connecticut Yankee	0	0	0	0	S	S	0	0	0	S	0	0
R. E. Ginna	0	S	S	14	0	0	0	S	0	S	0	0
H. B. Robinson 2	2	1	1	1	2	1	2	2	2	6	S	S
Point Beach 2	32	200	5	29	10	12	13	21	23	25	25	25
Surry 1	0	0	28	86	NA	19	39	14	33	1	S	S
Turkey Point 3	12	6	14	0	11	19	0	12	1	S	S	S
Surry 2	95	31	10	0	S	0	0	0	6	S	S	200
Turkey Point 4	62	0	0	S	S	S	0	0	80	42	S	0
Zion 1	0	0	S	S	S	S	0	0	0	S	0	0
Zion 2	S	S	0	S	S	0	0	0	0	S	0	0
Indian Point 2	0	0	0	S	S	S	S	S	S	139	S	S
Prairie Island 1	0	0	S	S	0	0	0	0	0	0	0	0
Prairie Island 2	S	0	0	0	0	0	0	0	0	S	S	S
Cook 1	0	0	0	S	S	0	0	0	0	0	0	0
Trojan					0	S	S	S	0	S	S	0
Indian Point 3								0	S	0	0	0
Point Beach 1	0	0	3	3	3	2	3	3	3	S	S	0

TABLE 2-11 (continued)

MONTHLY AVERAGE* PRIMARY/SECONDARY LEAKAGE
(gal/day at 70°F; density = 8.3 lb/gal)

	1977											
Plant	J	F	M	A	M	J	J	A	S	O	N	D
San Onofre	S	S	S	0	0	1	2	2	S	0	2	1
Connecticut	0	0	0	0	0	0	0	0	0	S	S	0
R. E. Ginna	0	0	0	S	S	0	0	0	0	0	0	0
H. B. Robinson 2	1	1	0	1	1	0	0	1	0	6	41	52
Point Beach 1	4	5	3	6	3	5	5	5	4	S	8	7
Point Beach 2	25	35	33	S	0	0	0	0	0	0	0	0
Surry 1	S	77	144	53	0	0	0	26	58	58	21	0
Turkey Point 3	0	0	0	0	0	0	0	28	72	72	56	S
Surry 2	548	360	S	0	NA	18	10	8	4	0	14	0
Turkey Point 4	23	29	71	96	7	S	S	0	0	4	0	0
Zion 1	0	0	0	0	0	0	0	0	S	S	S	0
Zion 2	S	S	S	0	0	0	0	0	0	0	0	0
Indian Point 2	0	0	0	S	0	0	S	0	0	0	0	0
Prairie Island 1	0	0	0	S	0	0	0	0	0	0	0	0
Prairie Island 2	0	0	0	0	0	0	0	0	0	1	S	S
Cook 1	S	S	0	0	0	0	0	0	0	0	0	0
Trojan	0	0	0	0	S	S	0	0	0	0	0	0
Indian Point 3	0	0	0	0	0	0	0	0	0	S	S	S
Beaver Valley 1				0	0	S	0	0	S	S	0	0
Salem 1							0	0	0	S	S	0
Farley 1												0

TABLE 2-11 (continued)

MONTHLY AVERAGE* PRIMARY/SECONDARY LEAKAGE
(gal/day at 70°F; density = 8.3 lb/gal)

1978

Plant	J	F	M	A	M	J	Average,* gal/day
San Onofre	1	1	1	S	1	1	4.6
Connecticut Yankee	0	0	0	0	0	0	5.7
R. E. Ginna	4	0	0	S	S	0	0.27
H. B. Robinson 2	441	S	S	18	88	190	21
Point Beach 1	20	7	7	7	120	7	15
Point Beach 2	0	0	0	S	0	0	7.9
Surry 1	0	0	0	0	S	S	22
Turkey Point 3	S	0	0	0	0	0	5.5
Surry 2	0	46	278	0	0	0	32.
Turkey Point 4	36	193	0	0	0	0	17.
Zion 1	0	0	0	0	0	0	0
Zion 2	0	S	S	S	S	0	0
Indian Point 2	0	S	S	S	S	0	7.8
Prairie Island 1	0	0	0	S	0	0	0
Prairie Island 2	0	0	0	0	0	0	0.03
Cook 1	0	0	0	S	S	S	0
Trojan	2	2	2	S	S	S	0.38
Indian Point 3	0	0	0	0	0	S	0
Beaver Valley 1	0	0	0	0	S	S	0
Salem 1	0	0	0	S	S	S	0
Farley 1	0	0	0	0	0	0	0
Operation Weighted Average							9

* Average daily value for each reactor is obtained by the sum of the total monthly leakage rates divided by the total number of days in operation.

TABLE 2-12

MOISTURE CARRYOVERS IN RECIRCULATING U-TUBE STEAM GENERATORS*

Facility	Percent Carryovers	Reference
Palisades	0.08	10, 11
Kansai	0.05	10, 11
Point Beach	0.2	8, 12
Turkey Point 3	0.6	6
Turkey Point 4	1.6	6
Average	0.5	

* Measurement based on Na concentration.

The category "Other Nuclides" includes Mo, Y, and Tc which are generally present in colloidal suspensions or as "crud." Although the actual removal mechanism for Y, Mo, and Tc is expected to be plateout or filtration, the quantitative effect of removal is expected to be commensurate with the removal of ionic impurities by ion exchange (within the accuracy of the calculations) and consequently plateout of these nuclides is included in the parameters for ion exchange.

2.2.4 IODINE RELEASES FROM BUILDING VENTILATION SYSTEMS

2.2.4.1 Parameter

The iodine releases from building ventilation systems prior to treatment are calculated by the PWR-GALE Code using the data in Tables 1-1, Tables 2-2 through 2-8 and 2-13 through 2-16.

2.2.4.2 Bases

The iodine-131 releases from building ventilation systems are based on measurements made at a number of operating reactors. The measurements were made during routine plant operation and during plant shutdowns. Work on identifying sources of radioiodine at PWR's has been conducted by C. Pelletier, et al. (Ref. 7) for the Electric Power Research Institute (EPRI), at three operating PWR's; Ginna, Calvert Cliffs 1, and Three Mile Island 1. Measurements have also been made by EG&G Idaho, Inc., Allied Chemical Corp., Idaho National Engineering Laboratory, for the U. S. Nuclear Regulatory Commission at Fort Calhoun (Ref. 4), Zion 1 and 2 (Ref. 5), Turkey Point 3 and 4 (Ref. 6), Prairie Island (Ref.42), and Rancho Seco (Ref. 43).

These measurements indicate that iodine-131 building vent releases are directly related to the reactor coolant iodine-131 concentration. As a result, the releases of iodine are expressed as "normalized" releases, that is, the absolute measured release rate in Ci/yr is divided by the reactor coolant concentration in μCi/g to give a "normalized" release rate of iodine-131 in Ci/yr/μCi/g as shown in the following equation:

$$R_N = \frac{R_A}{C_{RW}}$$

where

R_N = normalized release rate of iodine-131, Ci/yr/μCi/g

R_A = absolute (measured) iodine-131 release rate, Ci/yr

C_{RW} = measured reactor water iodine-131 concentration, μCi/g

TABLE 2-13

ANNUAL IODINE NORMALIZED RELEASES
FROM CONTAINMENT VENTILATION SYSTEMS†

NORMAL OPERATION LEAK RATE*

Data Source	Normalized Release/Unit 10^{-3} %/day
Ft. Calhoun (Ref. 4)	0.0014
Three Mile Island 1 (Ref. 7)	2.5
Turkey Point 3/4 (Ref. 6)	0.9
Main Yankee (Ref. 16)	0.1
Ginna (Ref. 19)	0.064
Yankee Rowe (Ref. 14, 16)	1.0
Prairie Island 1/2 (Ref. 42)	0.005
Rancho Seco (Ref. 43)	2.56
Average	0.80

RELEASE FOR EXTENDED OUTAGES**

Data Source	Normalized Release/Unit (Ci/yr/μCi/g)
Three Mile Island 1 (Ref. 7)	0.44
Calvert Cliffs 1 (Ref. 7)	0.19
Average	0.32

* The normalized release rate, expressed in %/day of leakage of primary coolant inventory of iodine, represents the effective leak rate for radioiodine. It is the combination of the reactor water leakage rate into the buildings, and the partitioning of the radioiodine between the water phase in the leakage and the gas phase where it is measured.

** The normalized release rate, expressed in Ci/yr/μCi/g, represents the effective leak rate for radioiodine. It is the combination of the reactor water iodine leakage rate into the buildings, and the partitioning of the radioiodine between the water phase in the leakage and the gas phase where it is measured.

† These results were obtained using ^{131}I data. The normalized release rates are applicable to both ^{131}I and ^{133}I.

TABLE 2-14

ANNUAL IODINE NORMALIZED RELEASES*
FROM AUXILIARY BUILDING VENTILATION SYSTEMS†

NORMAL OPERATION

Data Source	Normalized Release/Unit (Ci/yr/μCi/g)
Zion 1/2 (Ref. 5)	1.0
Fort Calhoun (Ref. 4)	0.12
Ginna (Ref. 7)	0.032
Calvert Cliffs 1 (Ref. 7)	0.57
Three Mile Island 1 (Ref. 7)	0.034
Turkey Point 3/4 (Ref. 6)	1.85
Prairie Island 1/2 (Ref. 42)	0.013
Rancho Seco (Ref. 43)	0.97
Average	0.68

SHUTDOWN

Data Source	Normalized Release/Unit (Ci/yr/μCi/g)
Ginna (Ref. 7)	0.08
Calvert Cliffs 1 (Ref. 7)	0.016
Three Mile Island 1 (Ref. 7)	0.14
Turkey Point 3/4 (Ref. 6)	6.8
Rancho Seco (Ref. 43)	1.14
Average	2.50

* The normalized release rate, expressed in Ci/yr/μCi/g during different modes of operation, represents the effective leak rate for radioiodine. It is the combination of the reactor water iodine leakage rate into the buildings and the partitioning of the radioiodine between the water phase in the leakage and the gas phase where it is measured.

† These results were obtained using ^{131}I data. The normalized release rates are applicable to both ^{131}I and ^{133}I.

TABLE 2-15

ANNUAL IODINE NORMALIZED RELEASES*

FROM REFUELING AREA VENTILATION SYSTEMS†

NORMAL OPERATION

Data Source	Normalized Release/Unit (Ci/yr/μCi/g)
Ginna (Ref. 7)	0.008
Calvert Cliffs 1 (Ref. 7)	0.049
Three Mile Island 1 (Ref. 7)	0.0012
Turkey Point 3 (Ref. 6)	0.16
Prairie Island 1/2 (Ref. 42)	0.019
Rancho Seco (Ref. 43)	0.01
Average	0.038

SHUTDOWN

Data Source	Normalized Release/Unit (Ci/yr/μCi/g)
Ginna (Ref. 7)	0.014
Calvert Cliffs 1 (Ref. 7)	0.039
Three Mile Island 1 (Ref. 7)	0.06
Turkey Point 3 (Ref. 6)	0.05
Rancho Seco (Ref. 43)	0.30
Average	0.093

* The normalized release rate, expressed in Ci/yr/μCi/g during different modes of operation, represents the effective leak rate for radioiodine. It is the combination of the reactor water iodine leakage rate into the building, and the partitioning of the radioiodine between the water phase in the leakage and the gas phase where it is measured.

† These results were obtained using ^{131}I data. The normalized release rates are applicable to both ^{131}I and ^{133}I.

TABLE 2-16*

ANNUAL IODINE NORMALIZED RELEASES**

FROM TURBINE BUILDING VENTILATION SYSTEMS†

NORMAL OPERATION

Data Source	Normalized Release/Unit (Ci/yr/μCi/g)
Monticello	3.1×10^3
Oyster Creek	6.0×10^3
Vermont Yankee	0.35×10^3
Pilgrim	8.5×10^3
Browns Ferry	1.3×10^3
References 3, 5 of Ref. 15	3.3×10^3
Average	3.8×10^3

EXTENDED SHUTDOWN

Data Source	Normalized Release/Unit (Ci/yr/μCi/g)
Monticello	1.7×10^2
Oyster Creek	3.5×10^2
Vermont Yankee	0.63×10^2
Browns Ferry	1.3×10^2
References 3, 5 of Ref. 15	1.4×10^3
Average	4.2×10^2

* The data in this table are taken from Table 2-8, NUREG-0016, Revision 1, January 1979 (Ref. 15).

** The normalized release rate, expressed in Ci/yr/μCi/g during different modes of operation represents the effective leak rate for radioiodine. It is a function of iodine leak rate via steam and the partition coefficient for radioiodine from reactor water to steam in the reactor vessel.

† These results were obtained using ^{131}I data. The normalized release rates are applicable to both ^{131}I and ^{133}I.

The normalized reactor water release rate, expressed in Ci/yr/μCi/g represents an effective leak rate for reactor water containing iodine. It is the combination of the water leakage rate into the building and the effect of iodine partitioning between the water phase in the systems leakage and the vapor phase in the building atmosphere.

For the turbine building, the secondary coolant iodine releases are directly related to the secondary coolant iodine-131 concentration. Therefore, for the turbine building, the normalized iodine release, R_N, is determined using the following expression:

$$R_N = \frac{R_A}{C_{RW} \times PC}$$

where

R_N = normalized release rate of secondary coolant water containing iodine-131, Ci/yr/μCi/g

R_A = absolute (measured) iodine-131 release rate, Ci/yr

C_{RW} = measured secondary coolant iodine-131 concentration, μCi/g

PC = measured iodine partition coefficient from secondary coolant water to steam.

The normalized release rate is used to estimate the release from PWR's since this expression for release rate is least variable with time for a given mode of operation. For this reason, it is useful in the determination of releases from PWR's.

Data on normalized release rates from the three reactors used in the EPRI study and the five reactors used in the NRC sponsored study are given for normal operation and shutdown periods in Tables 2-13 through 2-15, for the containment building, auxiliary building and refueling area, respectively. Also given in Table 2-13 is the normalized value of the iodine release data discussed in NUREG-0017, April 1976 (Ref. 14). For Table 2-16, it was considered that since the basic design and operation of PWR and BWR power generation equipment which is housed in the turbine building is essentially identical, the turbine building leakage rates from PWR's and BWR's should be similar. Therefore, for the PWR turbine building normalized iodine release rate, the values for BWR's given in Table 2-15 of NUREG-0016, Revision 1 (Ref. 15) have been used and reproduced here as Table 2-16 of this report.

The data in Tables 2-14 through 2-16 are expressed as total normalized releases during power operation of 300 days and the total normalized releases during shutdowns of 65 days. Since the reactors used in the EPRI study and the NRC study experienced several intermittent shutdowns of short duration during the power operation measurement period, the iodine releases during these short duration outages are included under power operation.

Since the releases from the containment building are dependent on the method of containment purging (see Section 2.2.9, Containment Purging Frequency), the releases in Table 2-13 are expressed in terms of a leak rate (in %/day of primary coolant inventory). In addition, the release from the containment building during extended outages is expressed as a total normalized release as discussed above for other buildings.

In order to obtain the releases in curies/year from the auxiliary building and the refueling area of a particular PWR, the normalized release data in Tables 2-14 and 2-15, respectively, are multiplied in the PWR-GALE Code by the iodine concentrations in the reactor coolant for that particular PWR using the following expression:

$$R_{PWRi} = R_N \times C_{PWRi}$$

where

R_{PWRi} = calculated annual release rate for particular PWR for iodine isotope i, Ci/yr

R_N = normalized annual release rate of iodine from Tables 2-14 and 2-15, Ci/yr/μCi/g

C_{PWRi} = calculated reactor water concentration for particular PWR for iodine isotope i, μCi/g

To obtain the release in curies/year from the turbine building of a particular PWR, the normalized release data in Table 2-16 are multiplied in the PWR-GALE Code by the iodine concentration in the secondary coolant water and the iodine partition coefficient from the water to steam in the steam generator for that particular PWR using the following expression:

$$R_{PWRi} = R_N \times SC_{PWRi} \times PC_{PWR}$$

where

R_{PWRi} = calculated annual release rate for particular PWR for iodine isotope i, Ci/yr

R_N = normalized annual release rate of iodine from Table 2-16, Ci/yr/μCi/g

SC_{PWRi} = calculated secondary coolant concentration for particular PWR for iodine isotope i, $\mu Ci/g$

PC_{PWR} = partition coefficient from the secondary coolant water to steam for the particular PWR (see Table 2-6)

In order to obtain the releases in curies/year from the containment building of a particular PWR, the normalized leak rates in Table 2-13, are multplied in the PWR-GALE Code by the iodine concentration in the reactor coolant for that particular PWR, and then this leak rate is considered along with the containment purging method for that particular PWR.

To obtain the releases during shutdown, multiply the normalized release rates for the shutdown period by the same reactor coolant concentration as for power operations. Use of this reactor coolant concentration is acceptable since the normalization technique based the shutdown normalized release rate on the reactor coolant concentrations prior to shutdown.

Iodine released from PWR building ventilation systems appear in one of the following chemical forms: particulate, elemental, hypoiodious acid (HOI) and organic. Based on data in References 4, 5, 6, 7, 42 and 43, the fraction of the iodine appearing in each of the chemical forms for each building ventilation system is given below:

FRACTION OF IODINE APPEARING IN EACH CHEMICAL FORM
FROM PWR BUILDING VENTILATION SYSTEMS

	Containment	Auxiliary	Turbine	Fuel Handling
Particulate	0.09	0.04	*	0.01
Elemental	0.21	0.21	0.78	0.17
HOI	0.21	0.22	*	0.57
Organic	0.49	0.53	*	0.25

* No data on breakdown of other species.

2.2.5 RADIOACTIVE PARTICULATES RELEASED IN GASEOUS EFFLUENTS

2.2.5.1 Parameter

Use the radioactive particulate release rates in gaseous effluents given in Table 2-17.

TABLE 2-17

PARTICULATE RELEASE RATE FOR GASEOUS EFFLUENTS*
(Ci/yr)/unit

Nuclide	Containment	Auxiliary Building	Fuel Pool Area	Waste Gas System
Cr-51	9.2(-3)	3.2(-4)	1.8(-4)	1.4(-5)
Mn-54	5.3(-3)	7.8(-5)	3.0(-4)	2.1(-6)
Co-57	8.2(-4)	NA	NA	NA
Co-58	2.5(-2)	1.9(-3)	2.1(-2)	8.7(-6)
Co-60	2.6(-3)	5.1(-4)	8.2(-3)	1.4(-5)
Fe-59	2.7(-3)	5.0(-5)	NA	1.8(-6)
Sr-89**	1.3(-2)	7.5(-4)	2.1(-3)	4.4(-5)
Sr-90**	5.2(-3)	2.9(-4)	8.0(-4)	1.7(-5)
Zr-95	NA	1.0(-3)	3.6(-6)	4.8(-6)
Nb-95	1.8(-3)	3.0(-5)	2.4(-3)	3.7(-6)
Ru-103	1.6(-3)	2.3(-5)	3.8(-5)	3.2(-6)
Ru-106	NA	6.0(-6)	6.9(-5)	2.7(-6)
Sb-125	NA	3.9(-6)	5.7(-5)	NA
Cs-134	2.5(-3)	5.4(-4)	1.7(-3)	3.3(-5)
Cs-136	3.2(-3)	4.8(-5)	NA	5.3(-6)
Cs-137	5.5(-3)	7.2(-4)	2.7(-3)	7.7(-5)
Ba-140	NA	4.0(-4)	NA	2.3(-5)
Ce-141	1.3(-3)	2.6(-5)	4.4(-7)	2.2(-6)

* Particulate release rates are prior to filtration.

NA - No release observed from this source. Release assumed to be less than 1% of total.

** Data not available from Ref. 4, 5, 6 or 7, therefore Sr-89 and Sr-90 data were extracted from Semi-annual Effluent Release Reports. Release from each area above calculated by use of percent released from each area from Ref. 4, 5, 6 and 7 data.

2.2.5.2 Bases

Tables 2-18 through 2-21 list measured particulate releases at 12 operating reactors (Ref. 4, 5, 6, 7, 42, and 43). The average annual release rates for each nuclide released from four sources within the plant have been calculated based on the data in Tables 2-18 through 2-21. The measurements shown in Tables 2-18 through 2-21 were taken upstream of HEPA filters on streams on which HEPA filters are located. Based on the data in Tables 2-18 through 2-21, 63% of the releases came from the containment, 5% from the auxiliary building, 31% from the fuel pool area, and less than 1% from the waste gas processing system.

2.2.6 NOBLE GAS RELEASES FROM BUILDING VENTILATION SYSTEMS

2.2.6.1 Parameter

The noble gas releases from building ventilation systems are based on a daily leak rate of 3% of the noble gas inventory in the primary coolant released to the containment atmosphere; on a 160 lb/day primary coolant leak to the auxiliary building; and on a 1700 lb/hr steam leak rate in the turbine building.

2.2.6.2 Bases

The containment building leakage rate is derived from xenon-133 measurements in the containment atmosphere at Ginna and Maine Yankee (Ref. 17). The xenon-133 concentrations in the containment atmospheres at steady state were approximately 5×10^{-3} μCi/cc for Main Yankee and 7×10^{-3} μCi/cc for Ginna. The containment volumes at these facilities are approximately 1.8×10^{6} ft^3 for Maine Yankee and 1×10^{6} ft^3 for Ginna. Based on these values, the total microcuries of xenon-133 in the containment building atmosphere are

Maine Yankee

$$(5 \times 10^{-3}\ \mu\text{Ci/cc})(1.8 \times 10^{6}\ \text{ft}^3)(2.83 \times 10^{4}\ \text{cc/ft}^3) = 2.5 \times 10^{8}\ \mu\text{Ci Xe-133}$$

Ginna

$$(7 \times 10^{-3}\ \mu\text{Ci/cc})(1 \times 10^{6}\ \text{ft}^3)(2.83 \times 10^{4}\ \text{cc/ft}^3) = 2.0 \times 10^{8}\ \mu\text{Ci Xe-133}$$

Based on the half-life of xenon-133 (5.3d) and the assumption of a constant leakage rate to containment, the daily leakage rate of xenon-133 to the containment for the two plants is

Main Yankee

$$\frac{2.5 \times 10^{8}\ \mu\text{Ci}}{(5.3\ \text{day}/0.693)} = 3.3 \times 10^{7}\ \mu\text{Ci/day Xe-133 leakage}$$

TABLE 2-18

MEASURED RELEASES UPSTREAM OF HEPA FILTERS - CONTAINMENT
(Ci/yr)

Nuclide	Three Mile Island 1 (Ref. 7)	Fort Calhoun (Ref. 4)	Zion 1 & 2 (Ref. 5)	Turkey Point 3 & 4 (Ref. 6)	Calvert Cliffs 1 (Ref. 7)	Ginna (Ref. 7)	Prairie Island 1 & 2 (Ref. 42)	Rancho Seco (Ref. 43)	Average (Ci/yr)/unit
Cr-51	5.5(-2)	ND	ND	ND	NA	NA	NA	NA	9.2(-3)
Mn-54	2.1(-2)	1.4(-8)	3.9(-6)	NA	NA	NA	NA	NA	5.3(-3)
Co-57	4.9(-3)	ND	ND	ND	NA	NA	NA	NA	8.2(-4)
Co-58	2.2(-1)	5.6(-8)	1.5(-5)	3.2(-6)	NA	NA	6.6(-8)	2.5(-3)	2.5(-2)
Co-60	2.3(-2)	3.8(-8)	1.2(-5)	3.0(-5)	NA	NA	1.4(-7)	3.3(-4)	2.6(-3)
Fe-59	1.6(-2)	ND	ND	ND	NA	NA	NA	NA	2.7(-3)
Zr-95	NA	NA	NA	NA	NA	NA	NA	NA	NA
Nb-95	1.1(-2)	ND	ND	ND	NA	NA	NA	NA	1.8(-3)
Ru-103	9.5(-3)	ND	ND	ND	NA	NA	NA	NA	1.6(-3)
Ru-106	NA	NA	NA	NA	NA	NA	NA	NA	NA
Sb-125	NA	NA	NA	NA	NA	NA	NA	NA	NA
Cs-134	2.1(-2)	3.2(-6)	2.3(-4)	7.7(-5)	NA	NA	3.2(-8)	1.5(-3)	2.5(-3)
Cs-136	1.9(-2)	ND	ND	ND	NA	NA	NA	NA	3.2(-3)
Cs-137	4.4(-2)	4.1(-6)	3.2(-4)	1.9(-4)	NA	NA	6.6(-8)	5.0(-3)	5.5(-3)
Ba-140	NA	NA	NA	NA	NA	NA	NA	NA	NA
Ce-141	8.0(-3)	ND	ND	ND	NA	NA	NA	NA	1.3(-3)

ND = Not detected. For averaging purposes, a value of zero was assumed.

NA = Not analyzed (or no measurement taken); plants not included in averaging.

TABLE 2-19

MEASURED RELEASES UPSTREAM OF HEPA FILTERS - AUXILIARY BUILDING
(Ci/yr)

Nuclide	Three Mile Island 1 (Ref. 7)	Fort Calhoun (Ref. 4)	Zion 1 & 2 (Ref. 5)	Turkey Point 3 & 4 (Ref. 6)	Calvert Cliffs 1 (Ref. 7)	Ginna (Ref. 7)	Prairie Island 1 & 2 (Ref. 42)	Rancho Seco (Ref. 43)	Average (Ci/yr)/unit
Cr-51	1.4(-3)	ND	NA	ND	NA	1.9(-4)	NA	NA	3.2(-4)
Mn-54	1.1(-4)	NA	NA	6.3(-5)	3.0(-4)	6.7(-5)	2.7(-6)	NA	7.8(-5)
Co-57	NA	NA	NA	NA	NA	NA	NA	NA	NA
Co-58	1.1(-3)	2.0(-3)	NA	1.1(-3)	4.8(-4)	6.3(-4)	4.0(-5)	1.2(-2)	1.9(-3)
Co-60	2.0(-4)	2.7(-4)	NA	6.0(-4)	2.0(-3)	7.7(-4)	4.5(-5)	7.3(-4)	5.1(-4)
Fe-59	2.3(-4)	ND	ND	ND	NA	1.9(-5)	NA	NA	5.0(-5)
Zr-95	2.7(-4)	ND	ND	ND	7.9(-3)	4.1(-5)	5.7(-6)	NA	1.0(-3)
Nb-95	1.4(-4)	ND	ND	ND	NA	6.0(-5)	1.0(-5)	NA	3.0(-5)
Ru-103	9.1(-5)	ND	NA	ND	NA	6.9(-5)	2.7(-6)	NA	2.3(-5)
Ru-106	NA	ND	NA	ND	NA	2.4(-5)	NA	NA	6.0(-6)
Sb-125	NA	NA	NA	NA	NA	NA	7.7(-6)	NA	3.9(-6)
Cs-134	8.0(-5)	1.6(-3)	NA	7.9(-4)	2.0(-3)	3.4(-4)	1.5(-6)	5.2(-5)	5.4(-4)
Cs-136	NA	ND	NA	ND	NA	1.9(-4)	NA	NA	4.8(-5)
Cs-137	2.0(-4)	1.8(-3)	NA	1.4(-3)	1.9(-3)	1.1(-3)	9.4(-6)	8.0(-5)	7.2(-4)
Ba-140	NA	ND	ND	ND	NA	1.6(-3)	NA	NA	4.0(-4)
Ce-141	1.5(-4)	ND	NA	ND	NA	2.8(-5)	1.5(-6)	NA	2.6(-5)

ND = Not detected. For averaging purposes, a value of zero was assumed.

NA = Not analyzed (or no measurement taken); plants not included in averaging.

Measurements were made downstream of the auxiliary building HEPA filter. Due to uncertainty in the DF's of the HEPA filter, the data is not considered.

TABLE 2-20

MEASURED RELEASES UPSTREAM OF HEPA FILTERS - FUEL POOL AREA
(Ci/yr)/unit

Nuclide	Three Mile Island 1 (Ref. 7)	Fort Calhoun (Ref. 4)	Zion 1 & 2 (Ref. 5)	Turkey Point 3 & 4 (Ref. 6)	Calvert Cliffs 1 & 2 (Ref. 7)	Ginna (Ref. 7)	Prairie Island 1 & 2 (Ref. 42)	Rancho Seco (Ref. 43)	Average
Cr-51	1.8(-4)	NA	NA	NA	NA	NA	NA	NA	1.8(-4)
Mn-54	1.0(-5)	NA	NA	NA	1.2(-3)	NA	2.6(-6)	NA	2.4(-4)
Co-57	NA	NA	NA	NA	NA	NA	NA	NA	NA
Co-58	8.5(-5)	NA	NA	NA	1.1(-2)	NA	8.8(-6)	6.7(-5)	1.8(-3)
Co-60	4.4(-5)	NA	NA	NA	5.0(-3)	NA	6.9(-6)	7.6(-6)	8.4(-3)
Fe-59	NA	NA	NA	NA	NA	NA	NA	NA	NA
Zr-95	NA	NA	NA	NA	NA	NA	7.2(-6)	NA	3.6(-6)
Nb-95	3.0(-5)	NA	NA	NA	9.5(-3)	NA	1.7(-5)	NA	1.9(-3)
Ru-103	9.8(-5)	NA	NA	NA	NA	NA	1.7(-5)	NA	3.8(-5)
Ru-106	6.9(-5)	NA	NA	NA	NA	NA	NA	NA	6.9(-5)
Sb-125	1.7(-4)	NA	NA	NA	NA	NA	ND	NA	5.7(-5)
Cs-134	9.0(-6)	NA	NA	NA	2.2(-3)	NA	9.8(-7)	9.6(-7)	3.7(-4)
Cs-136	NA	NA	NA	NA	NA	NA	NA	NA	NA
Cs-137	2.4(-5)	NA	NA	NA	5.6(-3)	NA	4.1(-6)	7.4(-7)	9.4(-4)
Ba-140	NA	NA	NA	NA	NA	NA	NA	NA	NA
Ce-141	NA	NA	NA	NA	NA	NA	8.8(-7)	NA	4.4(-7)

ND = Not detected. For averaging purposes, a value of zero was assumed.

NA = Not analyzed (or no measurement taken); plants not included in averaging.

TABLE 2-21

MEASURED RELEASES UPSTREAM OF HEPA FILTERS - WASTE GAS SYSTEM
(Ci/yr)

Nuclide	Three Mile Island 1 (Ref. 7)	Fort Calhoun (Ref. 4)	Zion 1 & 2 (Ref. 5)	Turkey Point 3 & 4 (Ref. 6)	Calvert Cliffs 1 (Ref. 7)	Ginna (Ref. 7)	Prairie Island 1 & 2 (Ref. 42)	Rancho Seco (Ref. 43)	Average (Ci/yr)/unit
Cr-51	8.4(-5)	ND	ND	ND	NA	NA	NA	NA	1.4(-5)
Mn-54	1.1(-5)	ND	4.0(-6)	ND	NA	NA	NA	8.4(-9)	2.1(-6)
Co-57	NA	NA	NA	NA	NA	NA	NA	NA	NA
Co-58	4.5(-5)	3.8(-6)	1.1(-5)	8.8(-7)	NA	NA	NA	5.1(-8)	8.7(-6)
Co-60	8.0(-5)	NA	3.2(-6)	2.9(-7)	NA	NA	NA	5.9(-8)	1.4(-5)
Fe-59	7.2(-6)	1.8(-6)	1.9(-6)	ND	NA	NA	NA	NA	1.8(-6)
Zr-95	1.9(-5)	ND	ND	NA	NA	NA	NA	NA	4.8(-6)
Nb-95	2.2(-5)	ND	ND	ND	NA	NA	NA	NA	3.7(-6)
Ru-103	1.9(-5)	ND	ND	ND	NA	NA	NA	NA	3.2(-6)
Ru-106	1.6(-5)	ND	ND	ND	NA	NA	NA	NA	2.7(-6)
Sb-125	NA	NA	NA	NA	NA	NA	NA	NA	NA
Cs-134	1.2(-4)	1.2(-6)	1.1(-4)	3.8(-8)	NA	NA	NA	1.1(-8)	3.3(-5)
Cs-136	3.2(-5)	ND	ND	ND	NA	NA	NA	NA	5.3(-6)
Cs-137	3.5(-4)	NA	1.1(-4)	8.8(-8)	NA	NA	NA	2.6(-8)	7.7(-5)
Ba-140	1.4(-4)	ND	ND	ND	NA	NA	NA	NA	2.3(-5)
Ce-141	1.3(-5)	ND	ND	ND	NA	NA	NA	NA	2.2(-6)

ND = Not detected. For averaging purposes, a value of zero was assumed.

NA = Not analyzed (or no measurement taken); plants not included in averaging.

Ginna

$$\frac{2 \times 10^8 \ \mu Ci}{(5.3 \ day/0.693)} = 2.6 \times 10^7 \ \mu Ci/day \text{ Xe-133 leakage}$$

Based on the xenon-133 concentration during power operation (Ref. 29) and the masses of primary coolant of the two plants, the fraction of the xenon-133 inventory in the containment released per day is

Maine Yankee

$$\frac{3.3 \times 10^7 \ \mu Ci/day}{(10^{\dagger} \ \mu Ci/cc \times 28{,}300 \ cc/ft^3 \times 11{,}000 \ ft^3)} = 0.01/day = 1\%/day$$

Ginna

$$\frac{2.6 \times 10^7 \ \mu Ci/day}{(30^{\dagger} \ \mu Ci/cc \times 28{,}300 \times 6{,}234 \ ft^3)} = 0.005/day - 0.5\%/day$$

Reference 16 also contains data for the xenon-133 concentration in the containment atmosphere and the primary coolant at Yankee Rowe for the periods August-October 1971, December 1971 - January 1972 and August-November 1973. These periods encompass several shutdowns and a wide variety of operating conditions, and during these periods the xenon concentration in the containment and in the primary coolant varied by two orders of magnitude. The percent of xenon-133 inventory in the coolant released to the containment atmosphere varied from approximately 0.05%/day to 0.5%/day. Also from Ref. 43, this percent was determined to be 10.4 for Rancho Seco.

On the basis of these data, we consider that 3%/day of the noble gas inventory in the primary coolant is released to the containment atmosphere.

In the auxiliary building, the source term calculation is based on an assumed primary coolant leakage rate of 160 lb/day (20 gal/day). In the absence of available data, this value is based on engineering judgment and is consistent with values proposed in Environmental Reports.

† The reactor coolant concentrations for Xe-133 are measured values during 12/73 - 6/74 for Main Yankee and September and October of 1971 for Ginna (Ref. 16).

In the turbine building, it is assumed that steam will leak to the turbine building atmosphere at a rate of 1700 lb/hr. The leakage is considered to be from many sources, each too small to be detected individually, but which, taken collectively, total 1700 lb/hr. The most significant leakage pathway is considered to be leakage through valve stem packings.

2.2.7 STEAM GENERATOR BLOWDOWN FLASH TANK VENT

2.2.7.1 Parameter

1. Pressurized water reactors, with U-tube steam generators, that are currently under design, either direct their blowdown through a heat exchanger to cool the blowdown or, if a flash tank is used, vent the flash tank to a flash tank vent condenser or the main condenser. For these plants, iodine releases by this path are negligible and a partition factor of zero is used for the steam generator blowdown flash tank vent.

2. For older plants which still utilize flash tanks which vent directly to the atmosphere an iodine partition factor of 0.05 is used.

2.2.7.2 Bases

Approximately one-third of the blowdown stream flashes to steam in the flash tank, provided there is a heat balance between steam generator operating conditions (550°F, 1000 psia) and the blowdown flash tank conditions (240°F, sat.). Although the iodine species in the blowdown stream will be predominantly nonvolatile (volatile species are degassed in the steam generator), significant iodine removal will occur because of entrainment by the flashing steam. A steam quality of 85% is considered in the evaluation. For currently designed PWR's which have provisions to prevent flashing (cooling blowdown below 212°F) or to condense the steam leaving the flash tank, the entrainment losses will be negligible, i.e., a partition factor of zero.

2.2.8 IODINE RELEASES FROM MAIN CONDENSER AIR EJECTOR EXHAUST

2.2.8.1 Parameter

The iodine releases from the main condenser air ejector exhaust prior to treatment are calculated by the PWR-GALE Code using the data in Tables 2-2 through 2-8, and in Table 2-22.

2.2.8.2 Bases

The iodine releases from the main condenser air ejector exhaust are based on secondary side measurements made by EPRI at Point Beach 2, (Ref. 7), by EG&G Idaho, Inc., for the NRC, at Turkey Point 3 and 4 (Ref. 6), and by Westinghouse at Point Beach 1 (Ref. 12) and Haddam Neck (Ref. 38).

TABLE 2-22

ANNUAL IODINE NORMALIZED RELEASES FROM MAIN CONDENSER AIR EJECTOR EXHAUST*

Data Source	Normalized Release (Ci/yr/μCi/g)
Turkey Point 3/4 (Ref. 6)	3.5 (+3)
Point Peach 1/2 (Ref. 7, 12)	6.1 (+2)
Haddam Neck (Ref. 38)	3.0 (+1)
Average	1.7 (+3)

* The normalized release rate represents the effective release rate for radioiodine. It is the combination of the steam flow to the main condenser, the partitioning of radioiodine between the main condenser and the air ejector exhaust where it is measured and the partition coefficient for radioiodine from water to steam in the steam generator.

In a manner similar to the discussion of normalized releases for building ventilation releases in Section 2.2.4, the main condenser air ejector exhaust iodine releases are directly related to the secondary coolant iodine-131 concentration. Therefore, for the air ejector exhaust, the normalized iodine release, R_N, is determined using the following expression:

$$R_N = \frac{R_A}{C_{RW} \times PC}$$

where

R_N = normalized effective release rate of iodine-131, Ci/yr/μCi/g

R_A = measured (absolute) iodine-131 release rate, Ci/yr

C_{RW} = measured secondary coolant iodine-131 concentration, μCi/g

PC = measured iodine partition coefficient from secondary coolant water to steam in the steam generator.

Data on normalized release rates from the main condenser air ejector exhaust are given in Table 2-22. To obtain the release in curies/year from the air ejector exhaust of a particular PWR, the normalized release data in Table 2-22 are multiplied in the PWR-GALE Code by the iodine concentration in the secondary coolant water and the iodine partition coefficient from the water to steam for that particular PWR using the following expression:

$$R_{PWRi} = R_N \times C_{PWRi} \times PC_{PWR}$$

where

R_{PWRi} = calculated annual release for particular PWR for iodine isotope i, Ci/yr

R_N = normalized annual release rate of iodine from Table 2-22, Ci/yr/μCi/g

C_{PWRi} = calculated secondary coolant concentration for particular PWR for iodine isotope i, μCi/g

PC_{PWR} = Iodine partition coefficient from water to steam in the steam generator for the particular PWR (see Table 2-6)

As discussed in references 6 and 7, most of the iodine in the secondary system is not available for release to the main condenser air ejector exhaust due to iodine bypassing the condenser hotwell in the moisture separator/reheater drains and extraction steam, and possibly due to iodine plating out in the moisture separator/reheater, turbine and main condenser. As a result, the iodine release from the main condenser air ejector exhaust is small compared to the building ventilation releases.

2.2.9 CONTAINMENT PURGE FREQUENCY

2.2.9.1 Parameter

For those plants equipped with small diameter purge lines (diameter of about 8 inches or less), releases are based on continuous ventilation during power operation and on 2 purges per year at cold shutdown with the large containment purge lines. The continuous ventilation rate used in the evaluation is based on the applicant's design.

For older plants (those under review for operating licenses or those for which the construction permit SER was issued prior to July 1, 1975) not equipped with small diameter purge lines, releases are based on 2 purges per year at cold shutdown and 22 purges per year during power operation. The 22 purges consider the effect of use of large containment purge lines and of separate vent lines, if any. If, for a specific plant, there is filtration on the large purge lines but not on the vent lines, an additional GALE Code run will be made to account for the effect of the vent.

Operating experience and special design features (for example, little or no air operated equipment in the containment) to reduce the frequency of containment purging will be considered on a case-by-case basis.

2.2.9.2 Bases

It is assumed that the containment building is purged twice a year for refueling and maintenance. The two purges are considered for cold shutdowns for annual fuel loading and planned maintenance. In addition, experience at operating reactors (Table 2-23) has indicated a need to purge or vent the containment frequently during full power operation and hot standby to control the containment pressure, temperature, humidity, and airborne activity levels (Ref. 17). For the above reasons, new plant designs are to include the capability to purge the containment continuously through small-diameter purge lines (about 8 inches in diameter) and only use the large containment purge lines at cold shutdowns and refueling outages (Ref. 18). On this basis, source term calculations for new plants should consider a continuous ventilation rate based on the applicant's containment design, along with the two cold shutdown purges per year with the large containment purge lines, unless special provisions are made to eliminate or reduce the need for continuous ventilation flow.

TABLE 2-23

PWR CONTAINMENT PURGING AND VENTING EXPERIENCE (REF. 17)

Yankee Rowe

Purge and vent only after cooldown following shutdown
Reasons: Routinely pressurize containment for leak detection system checks and bring activity down
Duration: 2 to 6 hours

Maine Yankee

Purge once per quarter
Reason: Bring activity down
Duration: 2 to 3 days each quarter

Indian Point 2

Vent 2 times each day
Reason: Pressure balance control
Duration: Approximately 1 to 2 hours
Purge once every 2 weeks (duration not stated)

Three Mile Island 1

Purge approximately once per week during operation, always purge prior to shutdown
Reason: Improve temperature and humidity conditions
Duration: Approximately 48 hours

Connecticut Yankee

Purge - Cannot purge during operation, only during shutdown
Reason: Primarily to remove activity
Duration: 1 to 2 days

San Onofre 1

Purge each cooldown approximately 4 times per year, no purging during power operation
Purge for at least 24 hours, ventilate during entire shutdown period

Oconee 1

Continuous purge from startup through 7/1/74
Purged twice since 7/1/74, once on 7/8/74 for several days and again on 8/22/74 for 1 to 2 days
Reason: Reduction of gaseous activity for maintenance, etc.

TABLE 2-23 (continued)

PWR CONTAINMENT PURGING AND VENTING EXPERIENCE (REF. 17)

Oconee 2

Continuous purge since startup, lowest purge rate approximately 20,000 ft^3/min

Reason: Reduction of gaseous activity for maintenance, etc.

Robinson 2

Purge approximately 20 times per year for 2 minutes each purge for testing of purge valves. In addition, purge approximately 10 times per year for an average of 100 hours each purge for personnel comfort reasons.

Vent about 75 times per year for about four hours each. Venting occurs to control containment pressure and to bring containment pressure to zero gauge prior to purging as noted above.

Turkey Point 3

For period 1/1/74 to 7/1/74

Total purges	14
Total time	502 hours*
Maximum duration (1 purge)	253 hours
Minimum duration (1 purge)	3 hours

Infrequent purges or vents of 10 minutes for pressure control.

Turkey Point 4

For period 1/1/74 to 7/1/74

Total purges	5
Total time	984 hours*
Maximum time (1 purge)	742 hours
Minimum time (1 purge)	5 hours

Surry 1 and 2

Containment operates at negative pressure. Discharge from vacuum pumps through filters to stack. During cold shutdown, there is continuous purging of containment.

Prairie Island

Frequency: Once per week for about 8 hours

Reason: To relieve pressure buildup due to instrument air leakage to containment

TABLE 2-23 (continued)

PWR CONTAINMENT PURGING AND VENTING EXPERIENCE (REF. 17)

Kewaunee

Frequency: 5 times in 60 days usually for less than 1 hour, longer if for personnel entry.
Reason: Pressure control. During the 60-day period, purging occurred for personnel entry.

Point Beach

Continuous venting through a monitoring line at about 10 ft^3/min flow. Gas filtered on way to stack.

Palidades

One per week for about 10 minutes duration (planned upon resumption of power operation)
Reason: To control pressure buildups

Zion

Venting for pressure buildup about twice per week depending on outside temperature. Ranges from twice per day to once every two weeks.
Purges to control environment range from once per day to once every two weeks.
Duration: 3/4 hour on venting; 3-4 hours on purging.

Fort Calhoun**

For periods from 1/1/76 to 6/31/76 and 5/5/77 to 12/31/77.
Average of 65 purges per year with an average duration of about 20 hours.

Millstone 2**

For period from 7/1/75 through 12/31/77.
About 45 purges per year with an average duration of about 9 hours.

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* Generally, long purges occur during plant outages while at cold shutdown conditions.

** Data for these plants was obtained from the Semi-annual Release Reports for the plants for the period indicated.

For older plants (those under review for operating licenses or those for which the construction permit SER was issued prior to July 1, 1975) (Ref. 18) not equipped with small diameter purge lines, frequent periodic purges or vents will be used to control the above parameters (Ref. 18). A frequency of 22 purges per year during power operation is considered representative of plant operating experience for the combined effects of purging and venting.

2.2.10 CONTAINMENT INTERNAL CLEANUP SYSTEM

2.2.10.1 Parameter

Assume the internal cleanup system will operate for 16 hours prior to purging, that it provides a DF for radioiodine removal on charcoal adsorbers corresponding to the values in Table 1-5, and a DF of 100 for particulate removal on HEPA filters and that there is a containment air mixing efficiency of 70%.

2.2.10.2 Bases

Internal cleanup systems may be used to reduce airborne iodine concentrations in the containment air prior to purging. Such systems normally recirculate containment air through HEPA filters and charcoal adsorbers to effect iodine and particulare removal. For source term calculations, it is assumed that the cleanup systems are operated for 16 hours prior to purging. It is considered that charcoal adsorbers provide a DF for iodine corresponding to the values in Table 1-5, that HEPA filters provide a DF of 100 for particulates, and that the containment air mixing efficiency is 70%. The system operation time of 16 hours considers that two shifts will elapse following a decision to enter the containment. The time period of two shifts is a reasonable amount of time for pre-entry preparations.

A 70% mixing efficiency, based on data from the Ginna Station containment building atmosphere test conducted in 1971 (Ref. 19), is used in evaluations. Data from Reference 19 are

Parameter	Symbol	Value
Length of test run	T	6 hours
Initial iodine activity	A_0	1.2×10^{-8} μCi/cc
Final iodine activity	A	1.2×10^{-9} μCi/cc
Containment volume	V	10^6 ft^3
Internal recirculation system flow rate	F	6.1×10^5 ft^3/hr

The efficiency of iodine removal, E, can be estimated from

$$\frac{A_o}{A} = \exp\left(\frac{FET}{V}\right)$$

Substituting Ginna data into the equation

$$\frac{1 \times 10^{-8}}{1 \times 10^{-9}} = \exp[(6.1 \times 10^5)(E)(6)/(10^6)]$$

$$10 = \exp(3.7E), \text{ therefore } E = 0.63.$$

The iodine removal efficiency E is a function of filter efficiency, E_a, and mixing efficiency, E_m.

$$E = E_a E_m = 0.63$$

In calculating E_m we used the assumed DF of 10 for charcoal derived from Table 1-5, (90% removal). Using E_a equal to 0.9, E_m is calculated to be 70%.

$$E_m = E/E_a = 0.63/0.9 = 0.7$$

2.2.11 RADIOIODINE REMOVAL EFFICIENCIES FOR CHARCOAL ADSORBERS AND PARTICULATE REMOVAL EFFICIENCIES FOR HEPA FILTERS

2.2.11.1 Parameter

Use a removal efficiency of 99% for particulate removal by HEPA filtration. For charcoal adsorbers, which satisfy the guideline of Reg. Guide 1.140 (Rev. 2), removal efficiencies for all forms of radioiodine are as follows:

Activated Carbon Bed Depth[a]	Removal Efficiencies For Radioiodine(%)
2 inches. Air filtration system designed to operate inside primary containment.	90
2 inches. Air filtration system designed to operate outside the primary containment and relative humidity is controlled to 70%.	70

[a] Multiple beds, e.g., two 2-inch beds in series, should be treated as single bed of aggregate depth of 4 inches.

Activated Carbon Bed Depth[a]	Removal Efficiencies For Radioiodine(%)
4 inches. Air filtration system designed to operate outside the primary containment and relative humidity is controlled to 70%.	90
6 inches. Air filtration system designed to operate outside the primary containment and relative humidity is controlled to 70%.	99

2.2.11.2 Bases

The removal efficiencies assigned to HEPA filters for particulate removal and charcoal adsorbers for radioiodine removal are based on the design, testing and maintenance criteria recommended in Regulatory Guide 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 2).

2.2.12 WASTE GAS SYSTEM INPUT FLOW TO PRESSURIZED STORAGE TANKS

2.2.12.1 Parameter

The input flow rate to the pressurized storage tanks is variable depending on the system design as can be seen from Table 2-24 and 2-25. Therefore each applicant should supply the value of F, the waste gas system input flow to the pressurized storage tanks. If detailed design information is not available, the data given in Tables 2-24 and 2-25 may be used. These data show that the average waste gas input flow is 170 ft^3/day (STP) per reactor for PWR's without recombiners and 30 ft^3/day (STP) per reactor for PWR's with recombiners.

2.2.12.2 Bases

As can be seen from Tables 2-24 and 2-25 there is variation among PWR system designs for the waste gas system input flow.

A review of the waste gas processing systems proposed for a number of PWR's as given in the respective PSAR's and FSAR's has yielded the design flow rates shown in Tables 2-24 and 2-25. Table 2-24 indicates that for reactors designed without recombiners to treat the gas prior to holdup in pressurized storage tanks, the average expected flow is approximately 170 ft^3/day (STP) per reactor. Table 2-25 indicates that for reactors designed with recombiners to remove hydrogen prior to holdup in pressurized storage tanks, the average expected flow is approximately 30 ft^3/day (STP) per reactor.

TABLE 2-24

WASTE GAS SYSTEM INPUT FLOW TO PRESSURIZED STORAGE TANKS FOR PWR's WITHOUT RECOMBINERS

Reactor	Net Flow per Reactor ft^3/day (STP)
San Onofre 2/3	57
Waterford 3	171
Pilgrim 2	69
St. Lucie 1/2	139
Millstone 2	49
Arkansas 1/2	68
Byron 1/2	173
Sequoyah 1/2	173
Marble Hill 1/2	173
Diablo Canyon 1/2	343
Trojan	225
Oconee 1/2/3	180
Davis Besse 1	144
Bellefonte 1/2	163

Average Net Flow for PWR's without recombiners = 170 ft^3/day (STP) per reactor

TABLE 2-25

WASTE GAS SYSTEM INPUT FLOW TO PRESSURIZED STORAGE TANKS
FOR PWR's WITH RECOMBINERS

Reactor	Net Flow per Reactor ft^3/day (STP)
WPPSS 1	96
Farley 1/2	3
McGuire 1/2	18

Average Net Flow for PWR's with recombiners = 30 ft^3/day (STP) per reactor

* Net flow rate is determined downstream of any recombiner (which is assumed 100% effective in removing hydrogen).

2.2.13 HOLDUP TIMES FOR CHARCOAL DELAY SYSTEMS

2.2.13.1 Parameter

$$T = 0.011\ MK/F$$

where

- T is the holdup time, in days; and
- K is the dynamic adsorption coefficient, in cm^3/g, (see chart below);
- M is the mass of charcoal adsorber, in 10^3 lbs;
- F is the system flow rate, in ft^3/min;
- 0.011 is the factor to convert from $(10^3\ lb.\ cm^3/g)/(ft^3/min)$ to days.

Dynamic adsorption coefficients, K, (in cm^3/g) are as follows:

	Operating 77°F Dew Point 45°F	Operating 77°F Dew Point 0°F	Operating 77°F Dew Point -40°F	Operating 0°F Dew Point -20°F
Kr	18.5	25	70	105
Xe	330	440	1160	2410

2.2.13.2 Bases

Charcoal delay systems are evaluated using the above equation and dynamic adsorption coefficients. T = MK/F is a standard equation for the calculation of delay times in charcoal adsorption systems (Ref. 20). The dynamic adsorption coefficients (K values) for Xe and Kr are dependent on operating temperature and moisture content (Ref. 21 and 22) in the charcoal, as indicated by the values in the above parameter. The K values represent a composite of data from operating reactor charcoal delay systems (Ref. 23 and 24) and reports concerning charcoal adsorption systems (Ref. 20-22, 24-27).

The factors influencing the selection of K values are:

1. Operational data from KRB (Ref. 23) and from KWL (Ref. 24), and from Vermont Yankee (Ref. 28).

2. The effect of temperature on the dynamic adsorption coefficients, indicated in Figure 2-3 (Ref. 21).

3. The effect of moisture on the dynamic adsorption coefficients, shown in Figure 2-4. The affinity of charcoal for moisture, shown in Figure 2-5.

4. The variation in K values between researchers and between the types of charcoal used in these systems (Refs. 21 and 27). Because of the variation in K values based on different types of charcoal and the data reported, average values taken from KRB and KWL data shown in Figure 2-3 are used.

The coefficient 0.011 adjusts the units and was calculated as follows:

$$T(\text{days}) = \frac{M(10^3 \text{ lbs})\ K(\text{cm}^3/\text{g})(454 \text{ g/lb})\ 3.53 \times 10^{-5} \text{ ft}^3/\text{cm}^3)}{F(\text{ft}^3/\text{min})(1440 \text{ min/day})}$$

$$T = 0.011 \frac{MK}{F}$$

2.2.14 LIQUID WASTE INPUTS

2.2.14.1 Parameter

The flow rates listed in Table 2-26 are used as inputs to the liquid radwaste treatment system. Flows that cannot be standardized are added to those listed in Table 2-26 to fit an individual application, e.g., shim bleed and equipment leaks to the reactor coolant drain tank. Disposition of liquid streams to the appropriate collection tanks is based on the applicant's proposed method of processing.

2.2.14.2 Bases

The flow rates used represent average values for a plant operating at steady-state conditions. The values are derived from values proposed by the ANS 55.6 Working Group in proposed American National Standard, "Liquid Radioactive Waste Processing System for Light Water Reactor Plants," (Ref. 29) from operating and design data, and from information furnished by applicants in response to source term questions. Data from Zion (Ref. 5) indicate that the values for fraction of primary coolant activity given in Table 2-26 provide reasonable estimates of plant operating experience.

2.2.15 DETERGENT WASTE

2.2.15.1 Parameters

For plants with an onsite laundry, use the radionuclide distribution given in Table 2-27 for untreated detergent wastes. The quantities shown

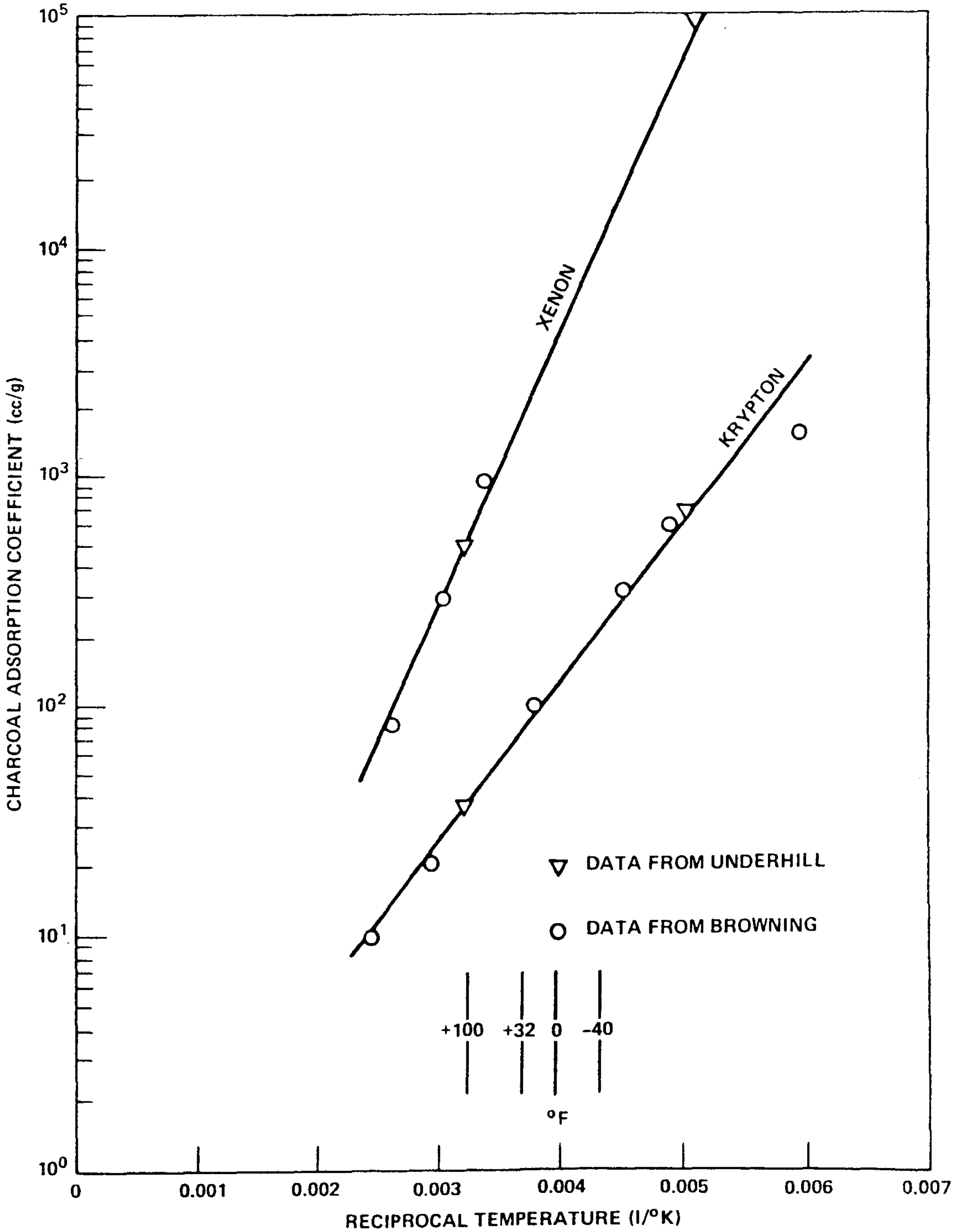

FIGURE 2-3 KRYPTON AND XENON K VALUES AS A FUNCTION OF RECIPROCAL TEMPERATURE

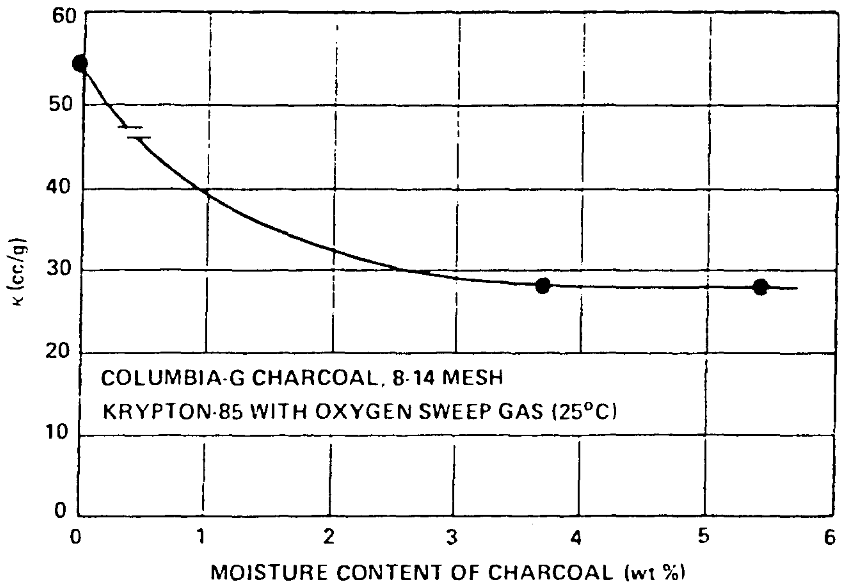

FIGURE 2-4 EFFECT OF MOISTURE CONTENT ON THE DYNAMIC ADSORPTION COEFFICIENT

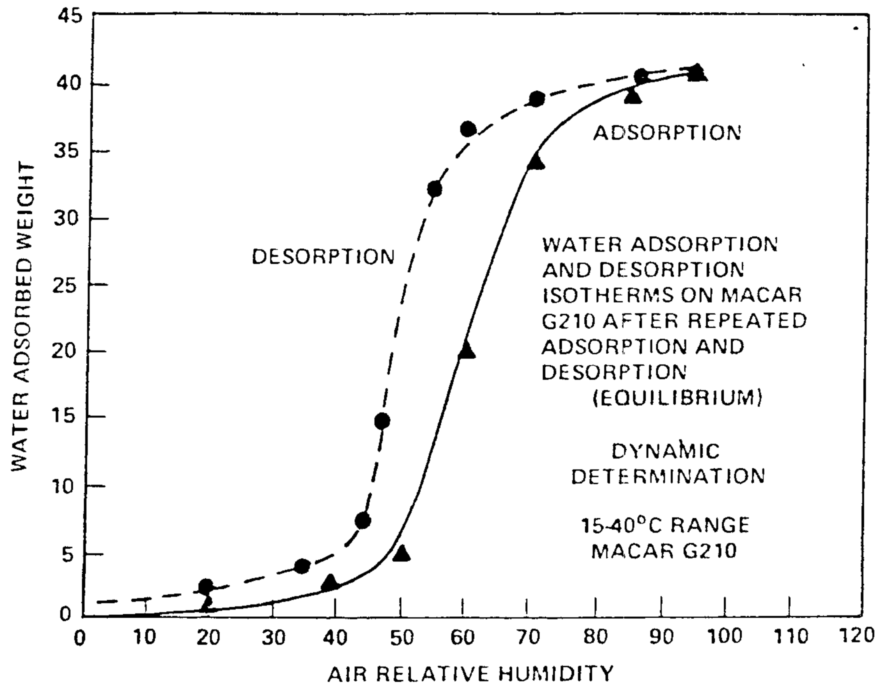

FIGURE 2-5 CHARCOAL MOISTURE AS A FUNCTION OF RELATIVE HUMIDITY

TABLE 2-26

PWR LIQUID WASTES

SOURCE	EXPECTED DAILY AVERAGE INPUT FLOW RATE (in Gal/day) — Type of treatment of blowdown recycled to secondary system (U-tube steam generator plants) or type of treatment of condensate (once-through steam generator plants): Deep-bed cond. demineralizers with ultrasonic resin cleaner	Deep-bed cond. demineralizers without ultrasonic resin cleaner	Filter-demineralizer	Plant with blowdown treatment. Product not recycled to condenser or secondary coolant system	FRACTION OF PRIMARY COOLANT ACTIVITY (PCA)
1. REACTOR CONTAINMENT					
a. Primary coolant pump seal leakage	20	20	20	20	0.1
b. Primary coolant leakage, miscellaneous sources	10	10	10	10	1.67*
c. Primary coolant equipment	500	500	500	500	0.001
2. PRIMARY COOLANT SYSTEMS (OUTSIDE OF CONTAINMENT)					
a. Primary coolant system equipment drains	80	80	80	80	1.0
b. Spent fuel pit liner drains	700	700	700	700	0.001
c. Primary coolant sampling system drains	200	200	200	200	0.05
d. Auxiliary building floor drains	200	200	200	200	0.1

TABLE 2-26 (Continued)

3. <u>SECONDARY COOLANT SYSTEMS</u>					
a. Secondary coolant sampling system drains	1400	1400	1400	1400	10^{-4}
b. Condensate demineralizer rinse and transfer solutions	3000	12000	-	-	10^{-8}
c. Condensate demineralizer regenerant solutions	850	3400	-	-	Calculated in GALE Code
d. Ultrasonic resin cleaner solutions	15000	-	-	-	10^{-6}
e. Condensate filter-demineralizer backwash	-	-	8100	-	2×10^{-6}
f. Steam generator blowdown	-	-	-	Plant dependent††	Plant dependent††
g. Turbine building floor drains	7200	7200	7200	7200	Calculated in GALE Code
4. <u>DETERGENT AND DECONTAMINATION SYSTEMS</u>					
a. On-site laundry facility	300	300	300	300	**
b. Hot showers	Negligible	Negligible	Negligible	Negligible	-
c. Hand wash sink drains	200	200	200	200	**
d. Equipment and area decontamination	40	40	40	40	**
TOTALS	29,700	26,300	19,000	10,000	

** GALE Code uses release data given in Table 2-27 to calculate releases from this source.

†† Input parameter

* About 40% of the leakage flashes, resulting in PCA fraction of the leakage greater than 1.0.

TABLE 2-27

CALCULATED ANNUAL RELEASE OF RADIOACTIVE MATERIALS IN UNTREATED DETERGENT WASTE

Nuclide		Ci/yr/reactor
P-32		1.8(-4)
Cr-51		4.7(-3)
Mn-54		3.8(-3)
Fe-55		7.2(-3)
Fe-59		2.2(-3)
Co-58		7.9(-3)
Co-60		1.4(-2)
Ni-63		1.7(-3)
Sr-89		8.8(-5)
Sr-90		1.3(-5)
Y-91		8.4(-5)
Zr-95		1.1(-3)
Nb-95		1.9(-3)
Mo-99		6(-5)
Ru-103		2.9(-4)
Ru-106		8.9(-3)
Ag-110m		1.2(-3)
Sb-124		4.3(-4)
I-131		1.6(-3)
Cs-134		1.1(-2)
Cs-136		3.7(-4)
Cs-137		1.6(-2)
Ba-140		9.1(-4)
Ce-141		2.3(-4)
Ce-144		3.9(-3)
	TOTAL	0.09 Ci

in Table 2-27 should be added to the adjusted liquid source term. Detergent waste releases should be reduced, using appropriate decontamination factors from this report if treatment is provided.

2.2.15.2 Bases

In the evaluation of liquid radwaste treatment systems, it is assumed that detergent wastes (laundry and personnel drains) will have the radionuclide distribution given in Table 2-27. The radionuclide distribution is based on measurements at four nuclear power plants, which are given in Table 2-28.

2.2.16 CHEMICAL WASTES FROM REGENERATION OF CONDENSATE DEMINERALIZERS

2.2.16.1 Parameter

1. Liquid flows to demineralizer at main steam activity.

2. All nuclides removed from the secondary coolant by the demineralizers are removed from the resins during regeneration.

3. Use a regeneration cycle of 1.2 days times the number of demineralizers for deep bed condensate system without ultrasonic resin cleaner (URC); for systems using URC, use a regeneration cycle of 8 days times the number of demineralizers.

2.2.16.2 Bases

Operating data (Ref. 30, 31) from Arkansas Nuclear One-Unit 1 indicate that one condensate demineralizer (without URC) is chemically regenerated every 1.2 days. The 8-day period for systems using URC is from Reference 29.

All material exchanged or filtered out by the resins between regenerations is contained in the regenerant waste streams, therefore, each regeneration will have approximately the same effectiveness (i.e., each regeneration removes all material collected since the previous regeneration, leaving a constant quantity of material on the resins after regeneration). Regeneration cycles are normally controlled by particulate buildup on resin beds, resulting in high pressure drops across the bed.

2.2.17 TRITIUM RELEASES

2.2.17.1 Parameter

The tritium releases through the combined liquid and vapor pathways are 0.4 Ci/yr per MWt. The quantity of tritium released through the liquid pathway is based on the calculated volume of liquid released, excluding secondary system wastes, with a primary coolant tritium concentration of 1.0 μCi/ml up to a maximum of 90% of the total quantity of tritium calculated to be available for release. It is assumed that the remainder of the tritium produced is released as a gas from building

TABLE 2-28

RADIONUCLIDE DISTRIBUTION OF DETERGENT WASTE
(millicuries/month)

Nuclide	Oyster Creek (1971-1973) (Ref. 41)	Ginna (1972-1973) (Ref. 8)	Zion* (1977) (Ref. 5)	Fort Calhoun (1977) (Ref. 4)
P-32	1.5(-2)	NA	NA	NA
Cr-51	2.3(-1)	NA	9.4(-1)	NA
Mn-54	1.3	1.2(-1)	1.6(-1)	1.9(-2)
Fe-55	3.5(-1)	NA	1.9	1.6(-1)
Fe-59	2.9(-1)	NA	2.6(-1)	NA
Co-58	3.5(-1)	4.1(-1)	2.4	1.5(-1)
Co-60	3.8	9(-1)	9.8(-1)	3(-2)
Ni-63	NA	NA	3.5(-1)	7.1(-2)
Sr-89	2.1(-2)	NA	7(-3)	1.4(-3)
Sr-90	2.5(-3)	NA	7.6(-4)	NA
Y-91	NA	NA	1.4(-2)	NA
Zr-95	8.3(-2)	1.6(-1)	1.4(-1)	NA
Nb-95	1.6(-1)	2(-1)	2.7(-1)	NA
Mo-99	NA	5(-3)	NA	NA
Ru-103	1.3(-2)	3.2(-2)	5.2(-2)	NA
Ru-106	NA	7.4(-1)	NA	NA
Ag-110m	NA	1(-1)	NA	NA
Sb-124	6.1(-2)	NA	4.7(-2)	NA
I-131	4.3(-1)	5.5(-2)	1.7(-1)	1.7(-2)
Cs-134	1.7(-1)	1.4	1.5	1.4
Cs-136	NA	NA	6.2(-2)	NA
Cs-137	2.9(-1)	2.5	2.1	1.7
Ba-140	7.6(-2)	NA	NA	NA
Ce-141	3.3(-2)	5(-3)	NA	NA
Ce-144	7.3(-2)	5.8(-1)	NA	NA
TOTAL	7.7	7.2	11.4	3.5

Note: NA = radionuclides were not analyzed.

* For two units.

ventilation exhaust systems. About eighty percent of the tritium in the gaseous effluents is released from the auxiliary building ventilation system, including the refueling area, and the remaining 20% of the tritium in gaseous effluents is released from the containment building ventilation system.

2.2.17.2 Bases

The release rate of 0.4 Ci/yr/MWt is based on a review of the tritium release rates at a number of PWR's and on data from specific measurements of tritium inventory and tritium releases at the Ginna plant (Ref. 8). The measurements at Ginna were made during the first two core cycles during which the reactor operated 605 effective full power days. The observed tritium buildup during this period was 1410 Ci. For the same period, 910,000 MWd of thermal power were generated. Using these data, considering an 80% plant capacity factor and considering tritium decay, the annual average tritium release is

$$\frac{1410 \text{ Ci}}{910{,}000 \text{ MWd}} (0.8)(365 \text{ days/yr})\ e^{-0.693(1)/12.3} = 0.43 \text{ Ci/yr per MWt}$$

Table 2-29 gives the reported liquid and gaseous tritium releases for 1972-1978 for thirty-five operating PWR's that use zircaloy clad fuel and started commercial operation before 1978. Table 2-29 shows these data expressed as the average release rate from the plants as a function of the number of years of operation of each plant. The tritium release rate from a PWR should reach a steady state value after a few years as a result of leakages from the plant. Table 2-30 illustrates the fact that the tritium release rate is approaching a steady state value of approximately 0.4 Ci/yr/MWt which is the value obtained from the Ginna measurements. At steady state, the release rate from a plant is approximately equal to the amount entering the primary coolant since only about 5% per year of the plant tritium inventory will decay. Based on the data from Ginna and the data in Table 2-30 we will use a release rate of 0.4 Ci/yr/MWt, which considers both liquid and vapor pathways.

The amount of tritium released via the liquid pathway is calculated from the volume of primary coolant that is released in the nonrecyclable waste streams for the boron recovery, clean waste, and dirty waste systems. The concentration of tritium in wastes originating from primary coolant is assumed to be 1 μCi/ml, consistent with the N237 source term. Tritium in liquid that leaks into, or is used as makeup to, the secondary system is considered to be released in liquid effluents through the turbine building floor drain discharge. The parameters for primary coolant activity prior to processing are used to calculate the tritium concentration in the waste streams.

Data in Table 2-31 indicate that tritium released in liquid effluents can make up a large fraction of the total tritium produced. Therefore we have considered that the tritium calculated to be released in liquid effluents is up to a maximum of 90% of the total quantity of tritium calculated to be available for release.

TABLE 2-29

TRITIUM RELEASE DATA FROM OPERATING PWR's WITH ZIRCALOY-CLAD FUELS*

Reactor Name	Power per unit MWt	Startup Date	Nuclear Thermal Output per unit 10^6 MWDt 1972	1973	1974	1975	1976	1977
R. E. Ginna	1520	1969	0.32	0.45	0.28	0.40	0.29	0.46
H. B. Robinson	2200	1970	0.62	0.51	0.39	0.57	0.66	0.59
Point Beach 1/2	1518	1970/72	0.42	0.77	0.43	0.87	0.91	0.93
Palisades	2530	1971	0.24	0.27	0.02	0.37	0.40	0.72
Maine Yankee	2440	1972			0.48	0.61	0.81	0.69
Indian Point 2/3	2758	1973/76			0.48	0.69	0.56	1.46
Surry 1/2	2441	1972/73			0.80	1.21	1.05	1.27
Turkey Point 3/4	2200	1972/73			1.08	1.16	1.12	1.13
Oconee 1/2/3	2568	1973/74/74			0.51	1.95	1.65	1.67
Zion 1/2	3250	1973/73				1.37	1.29	1.53
Fort Calhoun	1420	1973				0.28	0.30	0.39
Prairie Island 1/2	1650	1973/74				0.94	0.86	1.03
Kewaunee	1650	1974				0.45	0.45	0.46
Three Mile Island 1	2535	1974				0.73	0.58	0.73
Rancho Seco	2772	1974				0.17	0.29	0.75
Arkansas 1	2568	1974				0.64	0.50	0.68
Calvert Cliffs 1/2	2700	1974/76				0.58	0.84	1.24
Cook 1	3250	1975					0.90	0.64
Millstone 2	2560	1975					0.63	0.59
Trojan	3411	1975					0.31	0.88
St. Lucie 1	2560	1976						0.73
Beaver Valley 1	2652	1976						0.42
Salem 1	3338	1976						0.28

* Data from semiannual reports of reactors listed.

TABLE 2-29 (continued)

TRITIUM RELEASE DATA FROM OPERATING PWR's WITH ZIRCALOY-CLAD FUELS*

Reactor Name	Power per unit MWt	Startup Date	Tritium Released (Ci/Yr) Per Site Gaseous 1972	1973	1974	1975	1976	1977
R. E. Ginna	1520	1969	0.01	1.1	0.36	5.8	23.6	50
H. B. Robinson	2200	1970	1.0	4.0	52.0	193	158	61
Point Beach 1/2	1518	1970/72	8.0	25.0	43.0	177	395	194
Palisades	2530	1971	5.0	0.3	**	**	**	2.2
Maine Yankee	2440	1972			7.5	4.7	3.7	2.1
Indian Point 2/3	2758	1973/76			20.0	24.5	23.7	12.4
Surry 1/2	2441	1972/73			60.0	32	372	879
Turkey Point 3/4	2200	1972/73			9.2	3.5	5.2	3.9
Oconee 1/2/3	2568	1973/74/74			0.75	1600	502	62.6
Zion 1/2	3250	1973/73				**	**	**
Fort Calhoun	1420	1973				2.4	2.5	3.0
Prairie Island 1/2	1650	1973/74				10.1	33.1	88
Kewaunee	1650	1974				37.3	0.70	3.75
Three Mile Island 1	2535	1974				40.3	717	129
Rancho Seco	2772	1974				7.73	9.1	20.7
Arkansas 1	2568	1974				0.52	6.7	190
Calvert Cliffs 1/2	2700	1974/76				1.23	41	117
Cook 1	3250	1975					0.11	0.20
Millstone 2	2560	1975					21.3	47
Trojan	3411	1975					1.5	2.9
St. Lucie 1	2560	1976						320
Beaver Valley 1	2652	1976						213
Salem 1	3338	1976						51

* Data from semiannual reports of reactors listed.

** No reported data.

TABLE 2-29 (continued)

TRITIUM RELEASE DATA FROM OPERATING PWR's WITH ZIRCALOY-CLAD FUELS*

Reactor Name	Power per unit MWt	Startup Date	Tritium Released (Ci/Yr) Per Site Liquid					
			1972	1973	1974	1975	1976	1977
R. E. Ginna	1520	1969	120	286	195	261	242	119
H. B. Robinson	2200	1970	410	431	475	624	980	685
Point Beach 1/2	1518	1970/72	560	556	832	886	694	1000
Palisades	2530	1971	210	185	8.3	41.3	9.6	56
Maine Yankee	2440	1972			219	177	368	153
Indian Point 2/3	2758	1973/76			48	366	332	371
Surry 1/2	2441	1972/73			246	442	782	408
Turkey Point 3/4	2200	1972/73			580	793	771	924
Oconee 1/2/3	2568	1973/74/74			124	3550	2192	1918
Zion 1/2	3250	1973/73				39.4†	1.1†	727†
Fort Calhoun	1420	1973				111	122	157
Prairie Island 1/2	1650	1973/74				763	1925	1349
Kewaunee	1650	1974				277	213	295
Three Mile Island 1	2535	1974				463	189	192
Rancho Seco	2772	1974				132	0.0††	0.09††
Arkansas 1	2568	1974				460	212	245
Calvert Cliffs 1/2	2700	1974/76				263	274	575
Cook 1	3250	1975					192	285
Millstone 2	2560	1975					277	211
Trojan	3411	1975					36	311
St. Lucie 1	2560	1976						242
Beaver Valley 1	2652	1976						108
Salem 1	3338	1976						296

* Data from semiannual reports of reactors listed.

† No radioactive liquid wastes were discharged from Unit 2 during the entire year. Note: For 1975, there were no radioactive liquid wastes discharged from Unit 1 during the last 6 months.

†† Rancho Seco is designed to be a zero or very low liquid release plant.

TABLE 2-29 (continued)

TRITIUM RELEASE DATA FROM OPERATING PWR's WITH ZIRCALOY-CLAD FUELS*

Reactor Name	Power per unit MWt	Startup Date	Total Tritium Released Per Unit (Ci/yr.-Mwt at 80% capactiy) 1972	1973	1974	1975	1976	1977
R. E. Ginna	1520	1969	0.11	0.19	0.20	0.19	0.27	0.11
H. B. Robinson	2200	1970	0.19	0.25	0.39	0.42	0.50	0.37
Point Beach 1/2	1518	1970/72	0.39	0.22	0.59	0.36	0.35	0.37
Palisades	2530	1971	0.26	0.20	-	-	-	0.02
Maine Yankee	2440	1972			0.14	0.09	0.13	0.07
Indian Point 2/3	2758	1973/76			0.04	0.17	0.19	0.08
Surry 1/2	2441	1972/73			0.11	0.11	0.32	0.30
Turkey Point 3/4	2200	1972/73			0.16	0.20	0.20	0.24
Oconee 1/2/3	2568	1973/74/74			0.07	0.79	0.48	0.35
Zion 1/2	3250	1973/73				-	-	-
Fort Calhoun	1420	1973				0.12	0.12	0.12
Prairie Island 1/2	1650	1973/74				0.24	0.66	0.41
Kewaunee	1650	1974				0.20	0.14	0.19
Three Mile Island 1	2535	1974				0.20	0.46	0.13
Rancho Seco	2772	1974				0.24	0.01	0.01
Arkansas 1	2568	1974				0.21	0.13	0.19
Calvert Cliffs 1/2	2700	1974/76				0.13	0.11	0.16
Cook 1	3250	1975					0.06	0.13
Millstone 2	2560	1975					0.14	0.13
Trojan	3411	1975					0.04	0.10
St. Lucie 1	2560	1976						0.22
Beaver Valley 1	2652	1976						0.22
Salem 1	3338	1976						0.36

* Data from semiannual reports of reactors listed.

TABLE 2-30

TRITIUM RELEASE RATE FROM OPERATING PWR's
AS A FUNCTION OF NUMBER OF YEARS OF OPERATION
(Ci/yr.-MWt per unit at 80% capacity)

	1	2	3	4	5	6	7
Ginna	0.11	0.19	0.20	0.19	0.27	0.11	0.17
Robinson	0.19	0.25	0.39	0.42	0.50	0.37	-
Pt. Beach 1/2	0.39	0.22	0.59	0.36	0.35	0.37	0.51
Maine Yankee	0.14	0.09	0.13	0.07	0.18	-	-
Indian Pt. 2/3	0.04	0.17	0.19	0.08	-		
Surry 1/2	0.11	0.11	0.32	0.30	-		
Turkey Pt. 3/4	0.16	0.20	0.20	0.24	0.20		
Oconee 1/2/3	0.07	0.77	0.48	0.35	0.19		
Ft. Calhoun	0.12	0.12	0.12	0.13	-		
Prairie Is. 1/2	0.24	0.66	0.41	0.25			
Kewaunee	0.20	0.14	0.19	0.20	-		
TMI 1	0.20	0.46	0.13	0.17	-		
Arkansas 1	0.21	0.13	0.19	-	-		
Calvert Cliffs 1/2	0.13	0.11	0.16	-	-		
Cook	0.06	0.13	0.31	-	-		
Millstone	0.14	0.13	-	-	-		
Trojan	0.04	0.10	-	-	-		
St. Lucie	0.22	-	-	-	-		
Beaver Valley	0.22	0.51	-	-	-		
Salem	0.36	0.41	-	-	-		
Average	0.16	0.29	0.30	0.25	0.26	0.31	0.40

TABLE 2-31

TRITIUM RELEASE DATA FROM OPERATING PWR's

PERCENT OF TOTAL TRITIUM RELEASED IN LIQUID EFFLUENTS

Reactor	1972	1973	1974	1975	1976	1977
R. E. Ginna	100.0	99.6	99.8	97.8	91.1	70.4
H. B. Robinson	99.8	99.1	90.1	76.4	86.1	91.8
Point Beach 1/2	98.6	95.7	95.1	83.3	63.7	83.8
Palisades	97.7	99.8	**	**	**	96.2
Maine Yankee			96.7	97.4	99.0	98.6
Indian Point 2/3			70.6	93.7	93.3	96.8
Surry 1/2			80.4	93.2	67.8	31.7
Turkey Point 3/4			98.4	99.6	99.3	99.6
Oconee 1/3			99.4	68.9	81.4	96.8
Zion 1/2				**	**	**
Fort Calhoun				97.9	98.0	98.1
Prairie Island 1/2				98.7	98.3	93.9
Kewaunee				88.1	99.7	98.7
Three Mile Island 1				92.0	20.9	59.8
Rancho Seco				94.5	0.0††	0.43††
Arkansas 1				99.9	96.9	56.3
Calvert Cliffs 1/2				99.5	87.0	83.1
Cook 1					100.0	100.0
Millstone 2					92.9	81.8
Trojan					96.0	99.1
St. Lucie 1						43.1
Beaver Valley 1						33.6
Salem 1						85.3
Weighted Average*	99.2	98.0	91.1	89.5	83.5	78.5

* Average weighted by nuclear thermal output per unit.

** No reported data.

†† Rancho Seco is designed to be a zero or very low liquid release plant.

The difference between the tritium calculated to be available for release from the primary coolant and the tritium calculated to be released in liquid effluents is considered to be released as a vapor through building ventilation exhaust systems. Based on measurements taken in 1975 through 1977 at Ginna, Calvert Cliffs and Three Mile Island (Ref. 7) and in 1976 and 1977 at Zion 1/2 (Ref. 5), and in 1977 at Turkey Point (Ref. 6), in 1978-79 at Rancho Seco (Ref .43), and in 1980-1981 at Prairie Island 1/2 (Ref. 42), Table 2-32 provides the distribution of tritium released from various sources within the plant. Based on data in Table 2-32, approximately 32% of tritium in gaseous effluents is released from the auxiliary building, 50% from the refueling area, and 18% from the containment. Since the refueling area in a PWR generally vents to the same release point as the auxiliary building, we have included these two releases together in our parameter.

2.2.18 DECONTAMINATION FACTORS FOR DEMINERALIZERS

2.2.18.1 Parameter

	Anion	Cs, Rb	Other Nuclides
Mixed bed purification system ($LiBO_3$)	100	2	50
Boron recycle system	10	2	10
Evaporator condensate (H^+OH^-)	5	1	10
Radwaste (H^+ OH^-)	10^2(10)	2(10)	10^2(10)
Steam Generator Blowdown	10^2(10)	10(10)	10^2(10)
Cation bed (H^+) (any system)	1(1)	10(10)	10(10)
Anion bed (OH^-) (any system)	10^2(10)	1(1)	1(1)
Powdex (any system)	10(10)	2(10)	10(10)

Note: For two demineralizers in series, the DF for the second demineralizer is given in parentheses.

The following operating conditions were considered for the evaluation of demineralizer performance:

1. The DF is dependent upon the inlet radioactivity and ion concentrations and bed volume ion exchange capacity. For demineralizer performance within the same range of controlled operating conditions, the DF increases with inlet radioactivity concentration and decreases with bed volume throughout.

TABLE 2-32*

DISTRIBUTION OF TRITIUM RELEASE IN GASEOUS EFFLUENTS

Source of Gaseous Tritium Release (% of Total)

Plant	Auxiliary Building	Refueling Area	Containment Building
Ginna (Ref. 7)	31	69	NM
Calvert Cliffs 1 (Ref. 7)	38	46	16
Three Mile Is. 1 (Ref. 7)	5	43	52
Zion 1/2 (Ref. 5)	79	WA	21
Turkey Point 3/4 (Ref. 6)	75	17	8
Rancho Seco (Ref. 43)	92	WA	8
Prairie Island 1/2 (Ref. 43)	7.2	91.8	1.0
Average	32	50	18

NM - Not measured.

WA - Release from refueling area combined with auxiliary building release.

* The following method is used to determine the ^{3}H release in this table.

Containment Building operation average % of total release

(16 + 52 + 21 + 8+8 + 1)% ÷ (6) = 17.7% = 18%

Then the Refueling Area for Ginna is reduced by

18%, i.e., (69-18)% = 51%

Now the operation average % of the total release for the Refueling Area is

(51 + 46 + 43 + 17 + 91.8)% ÷ (5) = 50%

Then use (79-50)% = 29% and (92-50)% = 42% into Zion and Rancho Seco auxiliary building's data, respectively, to calculate the operational average of Auxiliary Building release which is equal to

(31 + 38 + 5 + 29 + 75 + 42 + 7.2)% ÷ (7) = 32%

2. When two demineralizers are used in series, the first demineralizer will have a higher DF than the second. However, the data in Reference 32 indicate that Cs and Rb will be more strongly exchanged in the second demineralizer in series than the first as the concentration of preferentially exchanged competing nuclides is reduced.

3. As indicated in Reference 32, compounds of Y, Mo, and Tc form colloidal particles that tend to plate out on solid surfaces. Mechanisms such as plateout on the relatively large surface areas provided by demineralizer resin beds result in removal of these nuclides to the degree stated above. An analysis of effluent release data indicates that these nuclides, although present in the primary coolant, are not found in the effluent streams.

2.2.18.2 Bases

The decontamination factors (DF's) for purification, radwaste, and evaporator condensate demineralizers are based on (1) source term measurements made at Fort Calhoun, Zion, Turkey Point, Prairie Island, and Rancho Seco stations by In-Plant Source Term Measurement Program (Refs. 4, 5, 6, 42, and 43); (2) the findings of a generic review in the nuclear industry by the Oak Ridge National Laboratory (ORNL) (Ref. 32); and (3) measurements taken at Three Mile Island 1 (Ref. 40). The DF's for the remaining demineralizers are based on ORNL findings.

The ORNL generic review contains operating and theoretical data which provides a basis for the numerical values assigned. The ORNL data were projected to obtain a performance value expected over an extended period of operation. It is considered that attempts to extend the service life of the resin will reduce the DF's below those expected under controlled operating conditions.

Average DF's for Ft. Calhoun, Zion, Turkey Point, Rancho Seco, and Prairie Island stations were obtained by dividing the average inlet radionuclide concentration of samples by that of the average outlet concentrating for each nuclide.

Based on the data in References 4, 5, 6, 32, 42, and 43, the DF used for the parameter was that considered to be representative of the data.

2.2.19 DECONTAMINATION FACTORS FOR EVAPORATORS

2.2.19.1 Parameter

	Decontamination Factors	
	All Nuclides Except Iodine	Iodine
Miscellaneous radwaste evaporators	10^3	10^2
Boric acid evaporators	10^3	10^2
Separate evaporator for detergent wastes	10^2	10^2

2.2.19.2 Bases

The decontamination factors for evaporators are based on: (1) source term measurements made at Fort Calhoun, Zion, Turkey Point, Prairie Island, and Rancho Seco stations by In-Plant Source Term Measurement Program (Ref. 4, 5, 6, 42, and 43) and (2) the findings of a generic review in the nuclear industry by the Oak Ridge National Laboratory (Ref. 33).

Average DF's for Zion, Ft. Calhoun, Turkey Point, Rancho Seco, and Prairie Island, were obtained by dividing the average inlet radioactivity of samples by the average outlet radioactivity of samples for each radionuclide.

Based on the data given in References 4, 5, 6, 33, 42, and 43, the DF used for the parameter was that considered to be the most representative of the data.

2.2.20 DECONTAMINATION FACTORS FOR LIQUID RADWASTE FILTERS

2.2.20.1 Parameter

A DF of 1 for liquid radwaste filters is assigned for all radionuclides.

2.2.20.2 Bases

Reference 34 contains findings of a generic review by ORNL of liquid radwaste filters used in the nuclear industry. Due to various filter types and filter media employed, reported values of decontamination factors vary widely, with no discernible trend. The principal conclusion reached in the ORNL report is that no credit should be assigned to liquid radwaste filters (DF of 1) until a larger data base is obtained.

Additional data from Ft. Calhoun (Ref. 4), Zion 1/2 (Ref. 5) and Turkey Point 3/4 (Ref. 6), Rancho Seco (Ref. 43), and Prairie Island 1/2 (Ref. 42) indicate that decontamination factors in liquid radwaste filters vary widely from less than 1 to greater than 50 (with a mean value of 1.3). Therefore a DF of 1 for liquid radwaste filters is used.

2.2.21 DECONTAMINATION FACTORS FOR REVERSE OSMOSIS

2.2.21.1 Parameter

Overall DF of 30 for laundry wastes and DF of 10 for other liquid radwastes.

2.2.21.2 Bases

Reverse osmosis processes are generally run as semibatch processes. The concentrated stream rejected by the membrane is recycled until a desired fraction of the batch is processed through the membrane. The ratio of the volume processed through the membrane to the inlet batch volume is the percent recovery. The DF normally specified for the process is the ratio of nuclide concentrations in the concentrated liquor stream to the concentrations in the effluent stream. This ratio is termed as the membrane DF (DF_m). For source term calculations, the system DF (DF_s) should be used. The system DF is the ratio of the nuclide concentrations in the feed stream to those in the effluent stream. The relationship between the system DF and the membrane DF is nonlinear and is a function of the percent recovery. This relationship can be expressed as follows:

$$DF_s = \frac{F}{1 - (1 - F)^{1/DF_m}}$$

where

DF_m is the membrane DF;

DF_s is the system DF; and

F is the ratio of effluent volume to inlet volume (fractional recovery).

Tables 2-33 through 2-36 give membrane DF's derived from operating data at Point Beach, Ginna and Robinson (Ref. 35) and laboratory data on simulated radwaste liquids (Ref. 36). These data indicate that the overall membrane DF is approximately 100. The percent recovery for liquid radwaste processes using reverse osmosis is expected to be approximately 95%, i.e., 5% concentrated liquor. Using these values in the above equation, the system DF is approximately 30.

$$DF_s = \frac{0.95}{1 - (1 - 0.95)^{1/100}} = 30$$

The data used were derived mainly from tests on laundry wastes. The DF for other plant wastes, e.g., floor drain wastes, is expected to be lower because of the higher concentrations of iodine and cesium isotopes. As indicated by the data in Tables 2-33, 2-35 and 2-36, the membrane DF for these isotopes is lower than the average membrane DF used in the evaluation for laundry waste.

2.2.22 GUIDELINES FOR CALCULATING LIQUID WASTE HOLDUP TIMES

The holdup times to permit radioactive decay applied to the input waste streams are calculated using the following parameters:

1. The collection time should be calculated for an 80% volume change in the tank, based on the liquid waste flow rates from the inlet sources.

TABLE 2-33

REVERSE OSMOSIS DECONTAMINATION FACTORS, GINNA STATION

Nuclide	Concentrate Activity ($\mu Ci/cm^3$)	Product Activity ($\mu Ci/cm^3$)	Membrane DF
Ce-144	2.68 (-4)	<2.2 (-7)	1200
Co-58	8.55 (-5)	<3.4 (-8)	2500
Ru-103	5.83 (-5)	<5.5 (-8)	1100
Cs-137	4.09 (-4)	6.6 (-6)	60
Cs-134	2.02 (-4)	3.2 (-6)	60
Nb-95	5.35 (-5)	<5.3 (-8)	1000
Zr-95	2.36(-5)	<3.7 (-8)	640
Mn-54	8.82 (-5)	<3.4 (-8)	2600
Co-60	9.62 (-4)	<8.1 (-8)	12,000
Total isotopic	2.15(-3)	9.8 (-6)	219
Gross beta	1.63 (-3)	1.86 (-5)	88
TOTAL	3.78(-3)	2.84(-5)	
Average			133

TABLE 2-34

REVERSE OSMOSIS DECONTAMINATION FACTORS, POINT BEACH

Date	Time	Concentrate Activity (μCi/ml)	Product Activity (μCi/ml)	Membrane DF
6/14/71	0840	1.1 (-5)	6.8 (-7)	16
	1225	6.3 (-5)	4.2 (-7)	150
	1350	6.8 (-5)	3.2 (-7)	213
6/15/71	1030	2.7 (-4)	3.1 (-6)	87
	1315	1.0 (-4)	1.7 (-6)	59
	1440	1.3 (-4)	1.1 (-7)	1200
	1510	1.6 (-4)	1.1 (-7)	1500
	1530	1.8 (-4)	5.7 (-7)	316
TOTAL		9.8 (-4)	7.0 (-6)	
	Average			140

TABLE 2-35

REVERSE OSMOSIS DECONTAMINATION FACTORS,
H. B. ROBINSON NO. 2 STATION

	Co-60	Co-58	I-131
	264	29	14
	382	---	20
	436	---	39
	107	229	26
	76	490	96
	94	131	11
Average	227	220	34

TABLE 2-36

EXPECTED REVERSE OSMOSIS DECONTAMINATION FACTORS FOR SPECIFIC NUCLIDES

Nuclide	Concentrate Activity (μCi/ml)	Product Activity (μCi/ml)	Membrane DF
Co-60	2.5 (-4)	5 (-7)	500
Mo-99	3.8 (-2)	1 (-3)	40
I-131, 132, 133, 134, 135	1.2 (-1)	4 (-3)	30
Cs-134, 137	4.3 (-2)	2 (-4)	200
TOTAL	2 (-1)	5 (-3)	
AVERAGE			40

2. The process time is the total time liquid remains in the system for processing, based on the flow rate through the limiting process step.

3. The discharge time is one-half the time required to empty the final liquid waste sample (test) tank to the environment. This value is based on the maximum rate of the discharge pumps and the nominal tank volume.

The calculated values in 1. and the total of 2. and 3. are used as inputs to the computer PWR-GALE Code.

2.2.23 ADJUSTMENT TO LIQUID RADWASTE SOURCE TERMS FOR ANTICIPATED OPERATIONAL OCCURRENCES

2.2.23.1 Parameter

1. Increase the calculated source term by 0.16 Ci/yr per reactor using the same isotopic distribution as for the calculated source term to account for anticipated operational occurrences such as operator errors that result in unplanned releases.

2. Assume evaporators to be unavailable for two consecutive days per week for maintenance. If a 2-day hold-up capacity exists in the system (including surge tanks) or an alternative evaporator is available, no adjustment is needed. If less than a 2-day capacity is available, assume the waste excess is handled as follows:

   a. High-purity or low-purity waste - Processed through an alternative system (if available) using a discharge fraction consistent with the lower purity system.

   b. Chemical Waste - Discharged to the environment to the extent holdup capacity or an alternative evaporator is available.

3. The following methods should be used for calculating holdup times and effective system DF:

   a. Holdup Capacity - If two or more holdup tanks are available, assume one tank is full (80% capacity) with the remaining tanks empty at the start of the two-day outage. If there is only one holdup tank, assume that it is 40% full at the start of the two-day outage with a usable capacity of 80%.

   b. Effective System DF - Should the reserve storage capacity be inadequate for waste holdup over a two-day evaporator outage, and should an alternate evaporator be unavailable to process the wastes from the out-of-service evaporator, the subsystem DF should be adjusted to show the effect of the evaporator outage.

For example, a DF of 10^5[†] was calculated for a radwaste demineralizer and radwaste evaporator in series. If an adjustment were required for the evaporator being out-of-service two days/week, with only one day holdup tank capacity, then the effective system DF can be calculated as follows:

1. For 6 days (7 - 2 + 1) out of 7 the system DF would be 10^5.

2. For the remaining one day, the system DF would be 10^2 (only the demineralizer DF is considered). The effective DF is:

$$DF = [(\frac{6}{7})(10^{-5}) + (\frac{1}{7})(10^{-2})]^{-1} = 7.0 \times 10^2$$

2.2.23.2 Bases

Reactor operating data over an 8 year period, January 1970 through December 1977, representing 154 reactor years of operation, were evaluated to determine the frequency and extent of unplanned liquid releases. During the period evaluated, 62 unplanned liquid releases occurred; 23 due to operator errors, 26 due to component failures, 5 due to inadequate procedures or failure to follow procedures, and the remaining 8 due to miscellaneous causes such as design errors. Table 2-37 summarizes the findings of this evaluation. Based on the data provided in Table 2-37 it is estimated that 0.16 Ci/reactor year will be discharged in unplanned releases in liquid effluents.

The availability of evaporators in waste treatment systems is expected to be in the range of 60 to 80%. Unavailability is attributed to scaling, fouling of surfaces, instrumentation failures, corrosion, and occasional upsets resulting in high carryovers requiring system cleaning. A value of two consecutive days unavailability per week was chosen as being representative of operating experience. For systems having sufficient tank capacity to collect and hold wastes during the assumed 2-day/week outage, no adjustments are required for the source term. If less capacity is available, the difference between the waste expected during two days of normal operation and the available holdup capacity is assumed to follow an alternative route for processing. Since processing through an alternative route implies mixing of wastes having different purities and different dispositions after treatment, it is assumed that the fraction of waste discharged following processing will be that normally assumed for the less pure of the two waste streams combined.

Since chemical and regenerant wastes are not amenable to processes other than evaporation, it is assumed that unless an alternative evaporation route is available, chemical and regenerant wastes in excess of the storage capacity are discharged without treatment.

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† 10^3 (Evap.) x 10^2 (demin) = 10^5 is obtained using DF's from Section 2.2.19.1.

TABLE 2-37

FREQUENCY AND EXTENT OF UNPLANNED LIQUID RADWASTE RELEASES FROM OPERATING PLANTS*

Unplanned Liquid Releases	
Total number (unplanned releases)	62
Fraction due to personnel error	0.37
Fraction due to component failure	0.42
Fraction due to inadequate procedures or failure to follow procedures	0.08
Fraction due to other causes	0.13
Approximate activity (Ci)	24.
Fraction of cumulative occurrence per reactor year (plants reporting releases <5 gals of liquid waste/reactor year)	0.16
Fraction of cumulative occurrences per reactor year (plants reporting activity released >0.01 Ci/reactor year)	0.28
Activity per release (Ci/release)	0.39
Activity released per reactor year (Ci/reactor year)	0.16
Volume of release per reactor year (gal/reactor year)	633.

* Values in this table are based on reported values in 1970-1977 Licensee Event Reports representing 154 reactor years of operation.

2.2.24 ATMOSPHERIC STEAM DUMP

2.2.24.1 Parameter

Noble gases and radioiodines released to the atmosphere from the steam dumps because of turbine trips and low-power physics tests will have a negligible effect on the calculated gaseous source term.

2.2.24.2 Bases

In the evaluation, consideration has been given to the quantity of noble gases and radioiodine released to the atmosphere from steam dumps because of low-power physics testing and turbine trips from full power. The evaluation indicates that the iodine-131 and noble gas releases will be less than 1% of the turbine building gaseous source term.

The evaluation of releases following a turbine trip from full power is based on the following parameters:

1. An average of two turbine trips annually;

2. 40% turbine bypass capacity to the main condenser;

3. Two-second rod insertion time required to scram the reactor following a turbine trip; and

4. Twelve-second cycle time to recirculate one primary coolant volume through the reactor and steam generator.

The above parameters are based on a 3400-MWt RESAR-3 reactor. Using these parameters, it is postulated that steam will continue to be produced at a full-power rate during the time the control rods are inserted and during the time required to recirculate one primary coolant volume. After this time, the turbine bypass will be adequate to handle steam generated from decay heat. The quantity of steam released

$$= (1.5 \times 10^7 \text{ lb/hr})(60\%)(14 \text{ sec})(2 \text{ trips/year})(454 \text{ g/lb})\left(\frac{\text{hr}}{3600 \text{ sec}}\right)$$

$$= 3 \times 10^7 \text{ g-steam/yr}$$

The iodine-131 concentration in the main steam for a U-tube steam generator is approximately 1.8×10^{-8} μCi/g-steam from Table 2-2.

Based on the steam release calculated above, the associated iodine-131 release is approximately 6.0×10^{-7} Ci/yr.

$$\text{I-131/yr} = \left(3.2 \times 10^7 \frac{\text{g-steam}}{\text{yr}}\right)(1.8 \times 10^{-8} \ \mu\text{Ci/g-steam})\left(10^{-6} \frac{\text{Ci}}{\mu\text{Ci}}\right)$$

$$= 5.8 \times 10^{-7} \text{ Ci/yr}$$

Releases due to low-power physics testing are calculated based on one 10-hour release of steam each year following a refueling. For a RESAR-3 reactor, low-power physics testing is conducted at 5% power. The conditions given above for power level and steady-state main steam iodine-131 activity are used. In addition, it is assumed that the reactor will be shut down for 30 days for refueling prior to low-power physics testing. The iodine-131 releases are calculated to be approximately 4.6×10^{-6} Ci/yr using the following equation:

$$\text{I-131/yr} = (1.5 \times 10^{7}\ \text{lb/hr steam})(0.05)(454\ \text{g/lb})(10\ \text{hr/yr})$$

$$(1.8 \times 10^{-8}\ \mu\text{Ci/g-steam})\ \exp\left[\frac{-(0.693)(30\ \text{days})}{(8.05\ \text{days})}\right] 10^{-6}\ \text{Ci}/\mu\text{Ci}$$

$$\text{I-131/yr} = 4.6 \times 10^{-6}\ \text{Ci/yr}$$

2.2.25 CARBON-14 RELEASES

2.2.25.1 Parameter

The annual quantity of carbon-14 released from a pressurized water reactor is 7.3 Ci/yr. It is assumed that most of the carbon-14 will form volatile compounds that will be released from the waste gas processing system and from the containment and auxiliary building atmospheres to the environment.

2.2.25.2 Bases

The annual release of 7.5 Ci of carbon-14 is based on measurements at ten operating PWR's presented in Table 2-38. Kunz et al. (Ref. 37) found that the carbon-14 reacts to form volatile compounds (principally CH_4, C_2H_6, and CO_2) that are collected in the waste gas processing system through degassing of the primary coolant and released to the environment via the plant vent. Data from Refs. 4, 5, 6, 42, and 43 also indicate carbon-14 is released from the containment and auxiliary building vent as a result of leakage of primary coolant into the containment and auxiliary building atmospheres.

As shown in Table 2-39, an average of measurements, made at Turkey Point 3 and 4, Zion 1 and 2, Fort Calhoun, Prairie Island 1 and 2, and Rancho Seco indicates that the release of carbon-14 breaks down to 22.6% from the containment building, 61.0% from the auxiliary building vents and 16.4% from the waste gas processing system. Therefore on this basis, it is assumed that 1.6 Ci/yr of carbon-14 is released from the containment building, 4.5 Ci/yr of carbon-14 is released from the auxiliary building vents and 1.2 Ci/yr of carbon-14 is released from the waste gas processing system.

TABLE 2-38

CARBON-14 RELEASE DATA FROM OPERATING PWR's

Plant*	1975	1976	1977	1978	Annual Average Ci/yr-unit
Conn. Yankee	44	40	30	70	46
Yankee Rowe	1.6	0.13	0.24	0.33	0.58

Plant**	Area	Annual Release Ci/yr-unit
Turkey Point 3/4	Aux. Bldg.	2.4
	Containment	0.075
	WGPS†	0.82
	Spent Fuel Area	0.38
	Total	3.7
Fort Calhoun	Fuel Pool and Aux. Bldg.	0.30
	WGPS	0.81
	Containment Bldg.	0.78
	Total	1.9
Zion 1/2	Cont. Bldg.	1.8
	Fuel Handling and Aux. Bldg.	1.4
	WGPS	0.062
	Total	3.3
Prairie Island 1/2	Cont. Building	0.016
	Fuel Handling and Aux. Bldg.	3.3
	WGPS	0.25
	Total	3.6
Rancho Seco	Cont. Building	0.9
	Fuel Handling and Aux. Bldg.	1.85
	WGPS	0.85
	Total	3.6
	Average	7.3

* Based on semi-annual release reports.
** Based on In-Plant Source Term Measurements.
† Waste gas processing system.

TABLE 2-39

DISTRIBUTION OF CARBON-14 RELEASED IN GASEOUS EFFLUENTS

Plant	Plant Areas: Containment	Aux. Bldg. and Fuel Handling	WGPS
Turkey Point 3/4	2%	75%	23%
Fort Calhoun	41%	16%	43%
Zion 1/2	55%	43%	2%
Rancho Seco	25%	51%	24%
Prairie Island 1/2	0.5%	92.5%	7%
Average:	22.6%	61.0%	16.4%

2.2.26 ARGON-41 RELEASES

2.2.26.1 Parameter

The annual quantity of argon-41 released from a pressurized water reactor is 34 Ci/yr. The argon-41 is released to the environment via the containment vent when the containment is vented or purged.

2.2.26.2 Bases

Argon-41 is formed by neutron activation of stable naturally occurring argon-40 in the containment air surrounding the reactor vessel. The argon-41 is released to the environment when the containment is vented or purged. Table 2-40 provides a summary of available data and gaseous argon-41 releases from operating PWR's. The information reported by the licensees is not sufficiently detailed to correlate reported argon-41 releases with plant size and plant operating parameters. However, the average argon-41 release is estimated to be 34 curies per year.

TABLE 2-40

SUMMARY OF ARGON-41 RELEASES FOR OPERATING PWR's FOR 1973-1978

(Ci/yr per reactor)

Reactor Name	Year	Release
Yankee Rowe	1974	0.85
	1975	0.93
	1976	0.3
	1977	0.49
	1978 (1/2 yr)	0.47
Haddam Neck	1973	0.044
	1977	0.08
	1978 (1/2 yr)	0.041
Ginna	1975	5.8
	1976	0.19
Point Beach 1/2	1973	17.6
	1974	16
	1975	208
	1976	31
	1977	9.2
	1978 (1/2 yr)	13.3
H. B. Robinson	1975 (1/2 yr)	16.2
	1976	15.4
	1977	23.1
	1978 (1/2 yr)	46.2
Surry	1974 (1/2 yr)	15
	1975	0.32
	1976	9.15
	1977 (1/2 yr)	16.5
D. C. Cook	1978 (1/2 yr)	19.7
Turkey Pt. 3/4	1974	26
	1975	51.3
	1976	39.4
	1977	45
Oconee 1/2/3	1974 (1/2 yr)	59.5
	1975	42
	1976	118
	1977	8.1
	1978 (1/2 yr)	19.9

TABLE 2-40 (continued)

SUMMARY OF ARGON-41 RELEASES FOR OPERATING PWR's FOR 1973-1978

(Ci/yr per reactor)

Reactor Name	Year	Release
Fort Calhoun	1975	8.2
	1976	2.2
	1977	2.3
	1978 (1/2 yr)	0.27
Palisades	1978 (1/2 yr)	0.01
Zion 1/2	1978 (1/2 yr)	24.8
Prairie Island 1/2	1975	1.3
	1976	21
	1977	31.8
	1978	13.5
Kewaunee	1976 (1/2 yr)	30
	1978 (1/2 yr)	5.9
Three Mile Island 1	1975 (1/2 yr)	50
	1976	12
	1977	66
	1978 (1/2 yr)	46.5
Calvert Cliffs	1976 (1/2 yr)	2
	1977 (1/2 yr)	3.1
Rancho Seco	1977	9.8
	1978 (1/2 yr)	1.8

* All data provided by the semiannual effluent release reports and the annual operating reports for each PWR listed.

CHAPTER 3. INPUT FORMAT, SAMPLE PROBLEM, AND FORTRAN LISTING OF THE PWR-GALE CODE

3.1 INTRODUCTION

This chapter contains additional information for using the PWR-GALE Code. Chapter 1 of this report described the entries required to be entered on input data cards. Section 3.2 of this chapter contains sample input data and an explanation of the input to orient the user in making the entries described in Chapter 1.

Section 3.3 of this chapter contains a listing of the input data for the sample problem and the resultant output. Section 3.4 contains a discussion of the nuclear data library used and a FORTRAN listing of the PWR-GALE Code.

3.2 INPUT DATA

This section contains (a) an explanation of the input used in the sample problem and (b) input coding sheets for the sample problem.

3.2.1 EXPLANATION OF THE INPUTS FOR THE SAMPLE PROBLEM

Only the inputs for the GALE code runs for the sample problem that are not obvious are explained:

Condensate demineralizer regeneration time (days)
Input - 8.4 days
Put this input in card 10 in the appropriate field allotted for this input.

Basis

The sample problem assumes eight condensate deep beds, one of which is spare in parallel with no ultrasonic resin cleaning. The regeneration time for a bed is therefore 7 x 1.2 days = 8.4 days.

The liquid waste inputs are based on assuming the following:

A. Waste Generation Rates and Effective PCA Fractions

Waste Type	Gal/day	PCA Fraction
Shim Bleed	1440	Code applies the CVCS DFs internally
Equipment Drains		
Pump seal leakage	300	1.0
Pump seal leakage (Table 2-26)	20	0.1

A. Waste Generation Rates and Effective PCA Fractions (Cont'd)

Waste Type	Gal/day	PCA Fraction
Other primary coolant leakage from miscellaneous sources inside the containment (Table 2-26)	10	1.67
Total equipment drain wastes	330	0.97 effective
Clean Wastes		
Primary coolant equipment drains (Outside containment)	80	1.0
Spent fuel pit liner drains	700	0.001
Primary coolant sampling system drains (segregated from secondary coolant samples)	200	0.05
Total Clean Wastes	980	0.093 effective
Dirty Wastes		
Primary coolant equipment Reactor containment cooling system	500	0.001
Auxiliary building floor drains	200	0.1
Secondary coolant sampling system drains	1400	0.0001
Total Dirty Wastes	2100	0.01 effective
Regenerant Wastes	3400	Code internally calculates the buildup on the beds
Condensate demineralizer rinse and transfer solution (secondary system wastes)	12000	10^{-8}

B. Available Equipment for Liquid Wastes Processing

Equipment	Number	Capacity (Each)
Recycle holup tank (To collect shim bleed and equipment drains)	2	50,000 gal
Clean waste holdup tank	2	7,000 gal
Dirty waste holdup tank	2	10,000 gal
Regenerant solution receiving tank	2	20,000 gal

B. Available Equipment for Liquid Wastes Processing (Cont'd)

Equipment	Number	Capacity (Each)
Resin and transfer solution receiving tank (To collect secondary system condensate demineralizer resin and transfer solution)	2	20,000 gal
Clean waste monitor tank (For processed shim bleed, equipment drains and clean wastes)	1	10,000 gal
Dirty waste monitor tank	2	10,000 gal
Secondary waste monitor tank (For processed regenerant wastes and secondary system condensate demineralizer resin and transfer solution)	2	10,000 gal
Recycle feed demineralizer (To process shim bleed and equipment drains and located upstream of the recycle holdup tank)	1	50 GPM
Recycle evaporator condensate demineralizer	1	50 GPM
Evaporator condensate demineralizer A (For clean wastes)	1	50 GPM
Evaporator condensate demineralizer B (For dirty wastes)	1	50 GPM
Secondary waste evaporator condensate demineralizer (To process regenerant wastes)	1	50 GPM
Secondary waste demineralizer (To process secondary system condensate demineralizer resin and transfer solution)	1	50 GPM
Steam generator blowdown demineralizer (To process steam generator blowdown)	2 in series	300 GPM
Recycle evaporator (For processing shim bleed and equipment drains)	1	30 GPM
Radwaste evaporator (For processing dirty wastes and clean wastes)	1	30 GPM
Secondary waste evaporator (For processing regenerate wastes)	1	30 GPM

C. Additional Notes about Liquid Wastes

1. The above list includes only the processing equipment assumed for generating the liquid waste inputs for running the GALE code. For example, it does not consider such equipment as filters, evaporator condensate tank, reactor makeup water storage tank, etc.

2. Except the condensate deep bed demineralizers in the secondary system, all other demineralizers are assumed to be mixed bed and non-regenerative.

3. The processed steam generator blowdown is assumed to be totally returned to the secondary system. It is also assumed that the steam generator blowdown is 75,000 pounds/hr (~150 GMP).

4. Secondary system condensate demineralizer rinse and transfer solution waste has not been included as input for the sample problem GALE code run for the following reasons:

   a. This waste is assumed to be collected in a collection system dedicated for this waste in the sample problem.

   b. Even if 100 percent of this waste is released without treatment, the release from this stream is expected to be $\leq$ 0.15 percent of the total liquid effluent release. If, however, this waste is processed by the secondary waste demineralizer listed above, the release from this stream is expected to be $\leq$ 0.012 percent of the total liquid effluent release. Furthermore, it is likely that this waste will be processed and a major fraction of this processed waste will be recycled to the condensate storage system for eventual reuse in the secondary plant.

Note that if assumption a is not satisfied in any design, then the inputs for this waste should be properly integrated with the appropriate subsystem inputs (for example, the dirty waste subsystem) and the effective inputs for the combined waste system should be included for the GALE code run for that design.

5. The detergent wastes are assumed to be released without any prior treatment.

6. All the liquid waste subsystems included in the GALE code run for the sample problem have at least a two-day holdup capacity for holding up the wastes prior to processing them.

7. In view of what has been stated above, no additional run need be made to evaluate the liquid effluent releases; also no adjustments need be made to waste subsystem DFs for possible equipment downtime.

D. The gaseous waste inputs to the GALE code run for the sample problem are based on assuming the following:

   1. There is neither continuous degassification of the full letdown flow to the gaseous radwaste system via a gas stripper nor continuous purging of the volume control tank.

   2. Fill time and holdup time for gases stripped from the primary system are based on the following:

Number of pressurized storage tanks - 4
Volume of each tank at STP - 650 CF
Design pressure for each tank - 150 psig
No recombiners

3. Containment has small diameter (8 inches) purge line and the low volume containment purge rate is 1000 CFM.

4. Containment has no internal cleanup (kidney) system.

5. Number of high volume containment purges during power operation - 0.

6. Fuel, auxiliary and containment buildings have HEPA filters and four inch charcoal adsorbers on their exhaust lines and these filter units satisfy the guidelines of Regulatory Guide 1.140. Containment building has these filters both on the low and high volume purge exhaust lines. Waste gas system has HEPA filters on its exhaust line which satisfies the guidelines of Regulatory Guide 1.140. The iodine releases via the main condenser air ejector removal system are assumed to be released without any treatment prior to their releases.

7. Steam generator blowdown flash tank exhaust is not vented directly to the atmosphere.

3.2.2 INPUT CODING SHEETS

Figure 3-1 shows the input coding sheets used for the sample problem.

3.3 SAMPLE PROBLEM - INPUT AND OUTPUT

Figure 3-2 shows printouts of the input and output for a sample problem using the PWR-GALE Code.

3.4 LISTING OF PWR-GALE CODE

3.4.1 NUCLEAR DATA LIBRARY

Calculation of the releases of radioactive materials in liquid effluents using the GALE Code requires a library of nuclear data available from the Division of ADP Support, USNRC (301) 492-7713. For convenience, the tape consists of five files, written in card image form. The contents of the five files are:

1. File 1: A FORTRAN listing of the liquid effluent code.

2. File 2: Nuclear data library for corrosion and activation products for use with the liquid effluent code.

FORTRAN CODING FORM

CODER	DATE	ADDRESS	PHONE
PROBLEM TITLE PWR-GALE CODE		PROGRAM NO. / CHG. NO.	SHEET OF

0 = ZERO, Ø = ALPHA O; 1 = ONE, I = ALPHA I; 2 = TWO, Ƶ = ALPHA Z

```
STATEMENT
NUMBER   6 7 ............................................................72 IDENTIFICATION (73-80)
CARD      1 NAME    NAME OF REACTOR  (                               )      TYPE   = PWR
CARD      2 POWTH   THERMAL POWER LEVEL (MEGAWATTS)                               (        )
CARD      3 PCVOL   MASS OF PRIMARY COOLANT (THOUSAND LBS)                        (        )
CARD      4 LETDWN  PRIMARY SYSTEM LETDOWN RATE (GPM)                             (        )
CARD      5 CBFLR   LETDOWN CATION DEMINERALIZER FLOW (GPM)                       (        )
CARD      6 NOGEN   NUMBER OF STEAM GENERATORS                                    (        )
CARD      7 TOSTFL  TOTAL STEAM FLOW (MILLION LBS/HR)                             (        )
CARD      8 WLI     MASS OF LIQUID IN EACH STEAM GENERATOR (THOUSAND LBS)         (        )
CARD      9 BLWDWN  BLOWDOWN-THOUS LB/HR(       )BLOWDOWN TREATMENT INPUT 0,1     ,OR2    ()
CARD     10 REGENT  CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS)             (        )
CARD     11 FFCOM   CONDENSATE DEMINERALIZER FLOW FRACTION                        (        )
CARD     12         SHIM BLEED RATE           (       ) GPD
CARD     13         DFI=(       )DFCS=(       )DFO =(       )
CARD     14         COLLECTION (     ) DAYS PROCESS (     ) DAYS FRACT DISCH (     )
CARD     15         EQUIPMENT DRAINS INPUT    (       ) GPD (     ) PCA
CARD     16         DFI=(       )DFCS=(       )DFO =(       )
CARD     17         COLLECTION (     ) DAYS PROCESS (     ) DAYS FRACT DISCH (     )
CARD     18         CLEAN WASTE INPUT         (       ) GPD (     ) PCA
CARD     19         DFI=(       )DFCS=(       )DFO =(       )
CARD     20         COLLECTION (     ) DAYS PROCESS (     ) DAYS FRACT DISCH (     )
CARD     21         DIRTY WASTE INPUT         (       ) GPD (     ) PCA
CARD     22         DFI=(       )DFCS=(       )DFO =(       )
CARD     23         COLLECTION (     ) DAYS PROCESS (     ). DAYS FRACT DISCH (     )
```

FORTRAN CODING FORM

CODER	DATE	ADDRESS	PHONE
PROBLEM TITLE PWR-GALE CODE	PROGRAM NO.	CHG. NO.	SHEET OF

0 = ZERO, Ø = ALPHA O; 1 = ONE, I = ALPHA I; 2 = TWO, Ƶ = ALPHA Z

STATEMENT NUMBER (1–5)	6	7–72	IDENTIFICATION (73–80)
CARD	2	4 BLOWDOWN FRACTION PROCESSED	( )
CARD	2	5 DFI=( ) DFCS=( ) DFO =( )	
CARD	2	6 COLLECTION ( ) DAYS PROCESS ( ) DAYS FRACT DISCH (	)
CARD	2	7 REGENERANT FLOW RATE (GPD)	( )
CARD	2	8 DFI=( ) DFCS=( ) DFO =( )	
CARD	2	9 COLLECTION ( ) DAYS PROCESS ( ) DAYS FRACT DISCH (	)
CARD	3	0 IS THERE CONTINUOUS STRIPPING OF FULL LETDOWN FLOW? 0,1,	OR2 ()
CARD	3	1 TAU1 HOLDUP TIME FOR XENON (DAYS)	( )
CARD	3	2 TAU2 HOLDUP TIME FOR KRYPTON (DAYS)	( )
CARD	3	3 TAU3 FILL TIME OF DECAY TANKS FOR THE GAS STRIPPER (DAYS)	( )
CARD	3	4 GAS WASTE SYSTEM HEPA?( )	
CARD	3	5 FUEL HANDLG BLDG CHARCOAL?( ) HEPA?( )	
CARD	3	6 AUXILIARY BLDG CHARCOAL?( ) HEPA?( )	
CARD	3	7 CONVOL CONTAINMENT VOLUME (MILLION FT3)	( )
CARD	3	8 CONTMT ATM CLEANUP CHARCOAL?( ) HEPA?( ) RATE(1000CFM)	( )
CARD	3	9 CNTMT-HIGH VOL PURGE CHARCOAL?( ) HEPA?( ) NUMBER/YEAR	( )
CARD	4	0 CNTMT-LOW VOL PURGE CHARCOAL?( ) HEPA?( ) RATE (CFM)	( )
CARD	4	1 FVN FRACTION IODINE RELEASED FROM BLOWDOWN TANK VENT	( )
CARD	4	2 FEJ PERCENT OF IODINE REMOVED FROM AIR EJECTOR RELEASE	( )
CARD	4	3 PFLAUN DETERGENT WASTE PF	( )

```
CARD  1 NAME   NAME OF REACTOR  SAMPLE PWR REV 1                     TYPE =    PWR
CARD  2 POWTH   THERMAL POWER LEVEL (MEGAWATTS)                              3400.
CARD  3 PCVOL   MASS OF PRIMARY COOLANT (THOUSAND LBS)                       550.
CARD  4 LETDWN  PRIMARY SYSTEM LETDOWN RATE (GPM)                           75.
CARD  5 CBFLR   LETDOWN CATION DEMINERALIZER FLOW (GPM)                     7.5
CARD  6 NOGEN   NUMBER OF STEAM GENERATORS                                     4.
CARD  7 TOSTFL  TOTAL STEAM FLOW (MILLION LBS/HR)                           15.
CARD  8 WLI     MASS OF LIQUID IN EACH STEAM GENERATOR (THOUSAND LBS)      112.5
CARD  9 BLWDWN  BLOWDOWN-THOUS LB/HR 75.0  BLOWDOWN TREATMENT-INPUT 0,1, OR 2 0
CARD 10 REGENT  CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS)          8.4
CARD 11 FFCDM   CONDENSATE DEMINERALIZER FLOW FRACTION                      0.65
CARD 12         SHIM BLEED RATE           1440.    GPD
CARD 13         DFI=  5.0E03DFCS=  2.0E03DFO =  1.0E05
CARD 14         COLLECTION  22.6  DAYS PROCESS  0.93  DAYS FRACT DISCH  .1
CARD 15         EQUIPMENT DRAINS INPUT    330.0   GPD AT 0.97  PCA
CARD 16         DFI=  5.0E03DFCS=  2.0E03DFO =  1.0E05
CARD 17         COLLECTION  22.6  DAYS PROCESS  0.93  DAYS FRACT DISCH  .1
CARD 18         CLEAN WASTE INPUT         980.    GPD AT .093  PCA
CARD 19         DFI=  5.0E02DFCS=  1.0E03DFO =  1.0E04
CARD 20         COLLECTION  5.7   DAYS PROCESS  0.13  DAYS FRACT DISCH   0.1
CARD 21         DIRTY WASTES              2100. GPD AT 0.01   PCA
CARD 22         DFI=  5.0E02DFCS=  1.0E03DFO =  1.0E04
CARD 23         COLLECTION  3.8   DAYS PROCESS  0.19  DAYS FRACT DISCH   1.0
CARD 24         BLOWDOWN   FRACTION PROCESSED                                1.
CARD 25         DFI= 1.00E03DFCS= 1.00E02DFO = 1.00E03
CARD 26         COLLECTION  0.0   DAYS PROCESS 0.0    DAYS FRACT DISCH  0.0
CARD 27         REGENERANT FLOW RATE (GPD)                                  3400.
CARD 28         DFI=  5.0E02DFCS=  1.0E03DFO =  1.0E04
CARD 29         COLLECTION  4.7   DAYS PROCESS  0.37  DAYS FRACT DISCH   0.1
CARD 30         IS THERE CONTINUOUS STRIPPING OF FULL LETDOWN FLOW? 0,1,OR 2    0
CARD 31 TAU1    HOLDUP TIME FOR XENON (DAYS)                                60.
CARD 32 TAU2    HOLDUP TIME FOR KRYPTON (DAYS)                              60.
CARD 33 TAU3    FILL TIME OF DECAY TANKS FOR THE GAS STRIPPER (DAYS)        30.
CARD 34         GAS WASTE SYSTEM HEPA?99.
CARD 35         FUEL HANDLG BLDG     CHARCOAL?90. HEPA?99.
CARD 36         AUXILIARY BLDG       CHARCOAL?90. HEPA?99.
CARD 37 CONVOL  CONTAINMENT VOLUME (MILLION FT3)                            2.45
CARD 38         CNTMT ATM. CLEANUP   CHARCOAL?0.0 HEPA?0.0 RATE(1000CFM)
CARD 39         CNTMT-HIGH VOL PURGE CHARCOAL?90. HEPA?99.  NUMBER/YEAR       0.0
CARD 40         CNTMT LOW VOL PURGE  CHARCOAL?90. HEPA?99. RATE (CFM)      1000.
CARD 41 FVN     FRACTION IODINE RELEASED FROM BLOWDOWN TANK VENT           0.0
CARD 42 FEJ     PERCENT OF IODINE REMOVED FROM AIR EJECTOR RELEASE          0.0
CARD 43 PFLAUN  DETERGENT WASTE PF                                          1.
```

SAMPLE PWR REV 1	PWR
THERMAL POWER LEVEL (MEGAWATTS)	3400.0000
PLANT CAPACITY FACTOR	0.8000
MASS OF PRIMARY COOLANT (THOUSAND LBS)	550.0000
PRIMARY SYSTEM LETDOWN RATE (GPM)	75.0000
LETDOWN CATION DEMINERALIZER FLOW (GPM)	7.5000
NUMBER OF STEAM GENERATORS	4.0000
TOTAL STEAM FLOW (MILLION LBS/HR)	15.0000
MASS OF LIQUID IN EACH STEAM GENERATOR (THOUSAND LBS)	112.5000
MASS OF WATER IN STEAM GENERATORS (THOUSAND LBS)	450.0000
BLOWDOWN RATE (THOUSAND LBS/HR)	75.0000
PRIMARY TO SECONDARY LEAK RATE (LBS/DAY)	75.0000
CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS)	8.4000
FISSION PRODUCT CARRY-OVER FRACTION	.0050
HALOGEN CARRY-OVER FRACTION	.0100
CONDENSATE DEMINERALIZER FLOW FRACTION	.6500

LIQUID WASTE INPUTS

STREAM	FLOW RATE (GAL/DAY)	FRACTION OF PCA	FRACTION DISCHARGED	COLLECTION TIME (DAYS)	DECAY TIME (DAYS)	DECONTAMINATION FACTORS I	CS	OTHERS
SHIM BLEED RATE	1.44E+03	1.0000	.1000	22.6000	.9300	5.00E+03	2.00E+03	1.00E+05
EQUIPMENT DRAINS	3.30E+02	.9700	.1000	22.6000	.9300	5.00E+03	2.00E+03	1.00E+05
CLEAN WASTE INPUT	9.80E+02	.0930	.1000	9.7000	.1300	5.00E+02	1.00E+03	1.00E+04
DIRTY WASTES	2.10E+03	.0100	1.0000	3.8000	.1900	5.00E+02	1.00E+03	1.00E+04
BLOWDOWN	2.16E+05		0.000	0.000	0.000	1.00E+03	1.00E+02	1.00E+03
UNTREATED BLOWDOWN	0.		1.000	0.000	0.000	1.00E+00	1.00E+00	1.00E+00
REGENERANT SOLS	3.40E+03		.100	4.700	.370	5.00E+02	1.00E+03	1.00E+04

GASEOUS WASTE INPUTS

THERE IS NOT CONTINOUS STRIPPING OF FULL LETDOWN FLOW

HOLDUP TIME FOR XENON (DAYS)		60.0000
HOLDUP TIME FOR KRYPTON (DAYS)		60.0000
FILL TIME OF DECAY TANKS FOR THE GAS STRIPPER (DAYS)		30.0000
GAS WASTE SYSTEM	PARTICULATE RELEASE FRACTION	.0100
AUXILIARY BLDG	IODINE RELEASE FRACTION	.1000
	PARTICULATE RELEASE FRACTION	.0100
CONTAINMENT VOLUME (MILLION FT3)		2.4500
FREQUENCY OF CNTMT BLDG HIGH VOL PURGE (TIMES/YR)		2.0000
CNTMT-HIGH VOL PURGE	IODINE RELEASE FRACTION	.1000
	PARTICULATE RELEASE FRACTION	.0100
CNTMT LOW VOL PURGE RATE(CFM)		1000.0000
CNTMT LOW VOL PURGE	IODINE RELEASE FRACTION	.1000
	PARTICULATE RELEASE FRACTION	.0100
STEAM LEAK TO TURBINE BLDG (LBS/HR)		1700.0000
FRACTION IODINE RELEASED FROM BLOWDOWN TANK VENT		0.0000
PERCENT OF IODINE REMOVED FROM AIR EJECTOR RELEASE		1.0000

FIGURE 3-2
PRINTOUT OF INPUT AND OUTPUT FOR THE SAMPLE PROBLEM

SAMPLE PWR REV 1 LIQUID EFFLUENTS

NUCLIDE	HALF-LIFE (DAYS)	COOLANT CONCENTRATIONS PRIMARY (MICRO CI/ML)	COOLANT CONCENTRATIONS SECONDARY (MICRO CI/ML)	ANNUAL RELEASES TO DISCHARGE CANAL BORON RS (CURIES)	MISC. WASTES (CURIES)	SECONDARY (CURIES)	TURB BLDG (CURIES)	TOTAL LWS (CURIES)	ADJUSTED TOTAL (CI/YR)	DETERGENT WASTES (CI/YR)	TOTAL (CI/YR)
CORROSION AND ACTIVATION PRODUCTS											
NA 24	6.25E-01	4.70E-02	1.42E-06	.00000	.00003	.00000	.00005	.00009	.00114	0.00000	.00110
P 32	1.43E+01	0.	0.	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	.00018	.00018
CR 51	2.78E+01	3.10E-03	1.22E-07	.00000	.00001	.00000	.00001	.00002	.00025	.00470	.00500
MN 54	3.03E+02	1.60E-03	6.08E-08	.00000	.00001	.00000	.00000	.00001	.00014	.00380	.00390
FE 55	9.50E+02	1.20E-03	4.59E-08	.00000	.00000	.00000	.00000	.00001	.00010	.00720	.00730
FE 59	4.50E+01	3.00E-04	1.12E-08	.00000	.00000	.00000	.00000	.00000	.00002	.00220	.00220
CO 58	7.13E+01	4.60E-03	1.78E-07	.00000	.00002	.00000	.00001	.00003	.00038	.00790	.00830
CO 60	1.92E+03	5.30E-04	2.06E-08	.00000	.00000	.00000	.00000	.00000	.00005	.01400	.01400
NI 63	3.36E+04	0.	0.	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	.00170	.00170
ZN 65	2.45E+02	5.10E-04	1.97E-08	.00000	.00000	.00000	.00000	.00000	.00004	0.00000	.00004
W187	9.96E-01	2.50E-03	8.22E-08	.00000	.00000	.00000	.00000	.00001	.00008	0.00000	.00008
NP239	2.35E+00	2.70E-03	7.90E-08	.00000	.00000	.00000	.00000	.00001	.00011	0.00000	.00011
FISSION PRODUCTS											
SR 89	5.20E+01	1.40E-04	5.34E-09	.00000	.00000	.00000	.00000	.00000	.00001	.00009	.00010
SR 90	1.03E+04	1.20E-05	4.59E-10	.00000	.00000	.00000	.00000	.00000	.00000	.00001	.00001
SR 91	4.03E-01	9.60E-04	2.67E-08	.00000	.00000	.00000	.00000	.00000	.00002	0.00000	.00002
Y 91M	3.47E-02	4.60E-04	3.16E-09	.00000	.00000	.00000	.00000	.00000	.00001	0.00000	.00001
Y 91	5.88E+01	5.20E-06	1.97E-10	.00000	.00000	.00000	.00000	.00000	.00000	.00008	.00009
Y 93	4.25E-01	4.20E-03	1.14E-07	.00000	.00000	.00000	.00000	.00001	.00007	0.00000	.00007
ZR 95	6.50E+01	3.90E-04	1.50E-08	.00000	.00000	.00000	.00000	.00000	.00003	.00110	.00110
NB 95	3.50E+01	2.80E-04	1.03E-08	.00000	.00000	.00000	.00000	.00000	.00002	.00190	.00190
MO 99	2.79E+00	6.40E-03	2.35E-07	.00000	.00002	.00000	.00001	.00003	.00035	.00006	.00041
TC 99M	2.50E-01	4.70E-03	1.06E-07	.00000	.00002	.00000	.00001	.00002	.00030	0.00000	.00030
RU103	3.96E+01	7.50E-03	2.90E-07	.00000	.00003	.00000	.00001	.00005	.00062	.00029	.00091
RH103M	3.96E-02	0.	0.	.00000	.00003	.00000	.00001	.00005	.00061	0.00000	.00061
RU106	3.67E+02	9.00E-02	3.46E-06	.00004	.00037	.00000	.00017	.00059	.00766	.00890	.01700
RH106	3.47E-04	0.	0.	.00004	.00037	.00000	.00017	.00059	.00766	0.00000	.00770
AG110M	2.53E+02	1.30E-03	4.96E-08	.00000	.00001	.00000	.00000	.00001	.00011	.00120	.00130
AG110	2.82E-04	0.	0.	.00000	.00000	.00000	.00000	.00000	.00001	0.00000	.00001
SB124	6.00E+01	0.	0.	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	.00043	.00043
TE129M	3.40E+01	1.90E-04	7.30E-09	.00000	.00000	.00000	.00000	.00000	.00002	0.00000	.00002
TE129	4.79E-02	2.40E-02	2.17E-07	.00000	.00000	.00000	.00000	.00000	.00002	0.00000	.00002
TE131M	1.25E+00	1.50E-03	5.09E-08	.00000	.00000	.00000	.00000	.00000	.00006	0.00000	.00006
I131	8.05E+00	4.50E-02	1.36E-06	.00017	.00309	.00001	.00013	.00340	.04429	.00160	.04600
TE132	3.25E+00	1.70E-03	6.20E-08	.00000	.00000	.00000	.00000	.00001	.00010	0.00000	.00010
I132	9.58E-02	2.10E-01	2.77E-06	.00000	.00019	.00000	.00005	.00024	.00313	0.00000	.00310
I133	8.75E-01	1.40E-01	3.75E-06	.00003	.00290	.00000	.00031	.00325	.04226	0.00000	.04200
I134	3.67E-02	3.40E-01	2.27E-06	.00000	.00001	0.00000	.00000	.00002	.00021	0.00000	.00021
CS134	7.49E+02	7.10E-03	2.84E-07	.00050	.00029	.00000	.00001	.00081	.01060	.01100	.02200
I135	2.79E-01	2.60E-01	5.47E-06	.00000	.00133	.00000	.00029	.00163	.02122	0.00000	.02100
CS136	1.30E+01	8.70E-04	3.45E-08	.00003	.00003	.00000	.00000	.00007	.00089	.00037	.00130
CS137	1.10E+04	9.40E-03	3.79E-07	.00068	.00039	.00000	.00002	.00109	.01413	.01600	.03000
BA137M	1.77E-03	0.	0.	.00063	.00037	.00000	.00002	.00102	.01322	0.00000	.01300
BA140	1.28E+01	1.30E-02	4.87E-07	.00000	.00005	.00000	.00002	.00008	.00098	.00091	.00190
LA140	1.68E+00	2.50E-02	8.76E-07	.00000	.00007	.00000	.00004	.00012	.00155	0.00000	.00160
CE141	3.24E+01	1.50E-04	5.71E-09	.00000	.00000	.00000	.00000	.00000	.00001	.00023	.00024
CE143	1.38E+00	2.80E-03	9.43E-08	.00000	.00000	.00000	.00000	.00001	.00011	0.00000	.00011
PR143	1.37E+01	0.	0.	.00000	.00000	.00000	.00000	.00000	.00001	0.00000	.00001
CE144	2.84E+02	3.90E-03	1.50E-07	.00000	.00002	.00000	.00001	.00003	.00033	.00390	.00420
PR144	1.20E-02	0.	0.	.00000	.00002	.00000	.00001	.00003	.00033	0.00000	.00033
ALL OTHERS		2.14E-01	6.26E-07	.00000	.00000	.00000	.00000	.00000	.00001	0.00000	.00001
TOTAL (EXCEPT TRITIUM)		1.48E+00	2.52E-05	.00216	.00972	.00001	.00141	.01331	.17331	.08975	.26000

TRITIUM RELEASE 280 CURIES PER YEAR

NOTE: .00000 INDICATES THAT THE VALUE IS LESS THAN 1.0E-5.

SAMPLE PWR REV 1	PWR
THERMAL POWER LEVEL (MEGAWATTS)	3400.00000
PLANT CAPACITY FACTOR	0.80
MASS OF PRIMARY COOLANT (THOUSAND LBS)	550.00000
PRIMARY SYSTEM LETDOWN RATE (GPM)	75.00000
LETDOWN CATION DEMINERALIZER FLOW (GPM)	7.50000
NUMBER OF STEAM GENERATORS	4.00000
TOTAL STEAM FLOW (MILLION LBS/HR)	15.00000
MASS OF LIQUID IN EACH STEAM GENERATOR (THOUSAND LBS)	112.50000
BLOWDOWN RATE (THOUSAND LBS/HR)	75.00000
CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS)	8.40000
CONDENSATE DEMINERALIZER FLOW FRACTION	.65000

LIQUID WASTE INPUTS

STREAM	FLOW RATE (GAL/DAY)	FRACTION OF PCA	FRACTION DISCHARGED	COLLECTION TIME (DAYS)	DECAY TIME (DAYS)	DECONTAMINATION FACTORS I	CS	OTHERS
SHIM BLEED RATE	1.44E+04	1.0000	.1000	22.6000	.9300	5.00E+03	2.00E+0[illegible]	1.00E+05
EQUIPMENT DRAINS	3.30E+02	.9700	.1000	22.6000	.9300	5.00E+03	2.00E+0[illegible]	1.00E+05
CLEAN WASTE INPUT	9.80E+04	.0930	.1000	5.7000	.1300	5.00E+02	1.00E+0[illegible]	1.00E+04
DIRTY WASTES	2.10E+03	.0100	1.0000	3.8000	.1900	5.00E+02	1.00E+0[illegible]	1.00E+04
BLOWDOWN	2.16E+05		0.000	0.000	0.000	1.00E+03	1.00E+0[illegible]	1.00E+03
UNTREATED BLOWDOWN	0.		1.000	0.000	0.000	1.00E+00	1.00E+0[illegible]	1.00E+00
REGENERANT SOLS	3.40E+04		.100	4.700	.370	5.00E+02	1.00E+0[illegible]	1.00E+04

GASEOUS WASTE INPUTS

THERE IS NOT CONTINUOUS STRIPPING OF FULL LETDWN FLOW		
FLOW RATE THROUGH GAS STRIPPER (GPM)		1.22917
HOLDUP TIME FOR XENON (DAYS)		60.00000
HOLDUP TIME FOR KRYPTON (DAYS)		60.00000
FILL TIME OF DECAY TANKS FOR THE GAS STRIPPER (DAYS)		30.00000
PRIMARY COOLANT LEAK TO AUXILIARY BLDG (LB/DAY)		160.00000
GAS WASTE SYSTEM	PARTICULATE RELEASE FRACTION	.01000
FUEL HANDLG BLDG	IODINE RELEASE FRACTION	.10000
	PARTICULATE RELEASE FRACTION	.01000
AUXILIARY BLDG	IODINE RELEASE FRACTION	.10000
	PARTICULATE RELEASE FRACTION	.01000
CONTAINMENT VOLUME (MILLION FT3)		2.45000
FREQUENCY OF PRIMARY COOLANT DEGASSING (TIMES/YR)		2.00000
PRIMARY TO SECONDARY LEAK RATE (LB/DAY)		75.00000
THERE IS NOT A KIDNEY FILTER		
FRACTION IODINE BYPASSING CONDENSATE DEMINERALIZER		.35000
IODINE PARTITION FACTOR (GAS/LIQUID) IN STEAM GENERATOR		.01000
FREQUENCY OF CNTMT BLDG HIGH VOL PURGE(TIMES/YR)		2.00000
CNTMT-HIGH VOL PURGE	IODINE RELEASE FRACTION	.10000
	PARTICULATE RELEASE FRACTION	.01000
CNTMT LOW VOL PURGE RATE(CFM)		1000.00000
CNTMT LOW VOL PURGE	IODINE RELEASE FRACTION	.10000
	PARTICULATE RELEASE FRACTION	.01000
STEAM LEAK TO TURBINE BLDG (LBS/HR)		1700.00000
FRACTION IODINE RELEASED FROM BLOWDOWN TANK VENT		0.00000
PERCENT OF IODINE REMOVED FROM AIR EJECTOR RELEASE		1.00000

SAMPLE PWR REV 1

GASEOUS RELEASE RATE - CURIES PER YEAR

	PRIMARY COOLANT (MICROCI/GM)	SECONDARY COOLANT (MICROCI/GM)	BUILDING VENTILATION				BLOWDOWN VENT OFFGAS	AIR EJECTOR EXHAUST	TOTAL
			FUEL HANDLG	REACTOR	AUXILIARY	TURBINE			
I-131	4.500E-02	1.397E-06	5.9E-04	3.7E-03	1.4E-02	0.	0.	0.	1.8E-02
I-133	1.400E-01	3.777E-06	1.9E-03	7.9E-03	4.5E-02	1.0E-04	0.	0.	5.5E-02

TOTAL H-3 RELEASED VIA GASEOUS PATHWAY = 1100 CI/YR

C-14 RELEASED VIA GASEOUS PATHWAY = 7.3 CI/YR

AR-41 RELEASED VIA CONTAINMENT VENT = 34 CI/YR

SAMPLE PWR REV 1

GASEOUS RELEASE RATE - CURIES PER YEAR

	PRIMARY COOLANT (MICROCI/GM)	SECONDARY COOLANT (MICROCI/GM)	GAS STRIPPING SHUTDOWN	GAS STRIPPING CONTINUOUS	BUILDING VENTILATION REACTOR	BUILDING VENTILATION AUXILIARY	BUILDING VENTILATION TURBINE	BLOWDOWN VENT OFFGAS	AIR EJECTOR EXHAUST	TOTAL
KR-85M	1.600E-01	3.400E-08	0.	0.	4.8E+01	3.0E+00	0.	0.	2.0E+00	5.3E+01
KR-85	4.300E-01	8.900E-08	2.1E+02	8.3E+02	9.6E+02	9.0E+00	0.	0.	4.0E+00	2.0E+03
KR-87	1.500E-01	3.000E-08	0.	0.	1.4E+01	3.0E+00	0.	0.	1.0E+00	1.8E+01
KR-88	2.800E-01	5.900E-08	0.	0.	5.6E+01	6.0E+00	0.	0.	3.0E+00	6.5E+01
XE-131M	7.300E-01	1.500E-07	1.1E+01	2.0E+01	1.5E+03	1.5E+01	0.	0.	7.0E+00	1.6E+03
XE-133M	7.000E-02	1.500E-08	0.	0.	1.0E+02	1.0E+00	0.	0.	0.	1.0E+02
XE-133	2.600E+00	5.400E-07	0.	0.	4.7E+03	5.5E+01	0.	0.	2.6E+01	4.8E+03
XE-135M	1.300E-01	2.700E-08	0.	0.	3.0E+00	3.0E+00	0.	0.	1.0E+00	7.0E+00
XE-135	8.500E-01	1.800E-07	0.	0.	4.6E+02	1.8E+01	0.	0.	9.0E+00	4.9E+02
XE-137	3.400E-02	7.100E-09	0.	0.	0.	0.	0.	0.	0.	0.
XE-138	1.200E-01	2.500E-08	0.	0.	2.0E+00	3.0E+00	0.	0.	1.0E+00	6.0E+00
TOTAL NOBLE GASES										9.1E+03

0.0 APPEARING IN THE TABLE INDICATES RELEASE IS LESS THAN 1.0 CI/YR FOR NOBLE GAS, 0.0001 CI/YR FOR I

AIRBORNE PARTICULATE RELEASE RATE-CURIES PER YEAR

NUCLIDE	WASTE GAS SYSTEM	BUILDING VENTILATION REACTOR	AUXILIARY	FUEL HANDLG	TOTAL
CR-51	1.4E-07	9.2E-05	3.2E-06	1.8E-06	9.7E-05
MN-54	2.1E-08	5.3E-05	7.8E-07	3.0E-06	5.7E-05
CO-57	0.	8.2E-06	0.	0.	8.2E-06
CO-58	8.7E-08	2.5E-04	1.9E-05	2.1E-04	4.8E-04
CO-60	1.4E-07	2.6E-05	5.1E-06	8.2E-05	1.1E-04
FE-59	1.8E-08	2.7E-05	5.0E-07	0.	2.8E-05
SR-89	4.4E-07	1.3E-04	7.5E-06	2.1E-05	1.6E-04
SR-90	1.7E-07	5.2E-05	2.9E-06	8.0E-06	6.3E-05
ZR-95	4.8E-08	0.	1.0E-05	3.6E-08	1.0E-05
NB-95	3.7E-08	1.8E-05	3.0E-07	2.4E-05	4.2E-05
RU-103	3.2E-08	1.6E-05	2.3E-07	3.8E-07	1.7E-05
RU-106	2.7E-08	0.	6.0E-08	6.9E-07	7.8E-07
SB-125	0.	0.	3.9E-08	5.7E-07	6.1E-07
CS-134	3.3E-07	2.5E-05	5.4E-06	1.7E-05	4.8E-05
CS-136	5.3E-08	3.2E-05	4.8E-07	0.	3.3E-05
CS-137	7.7E-07	5.5E-05	7.2E-06	2.7E-05	9.0E-05
BA-140	2.3E-07	0.	4.0E-06	0.	4.2E-06
CE-141	2.2E-09	1.3E-05	2.6E-07	4.4E-09	1.3E-05

3. File 3: Nuclear data library for fuel materials and their transmutation products for use with the liquid effluent code.

4. File 4: Nuclear data library for fission products for use with the liquid effluent code.

5. File 5: A FORTRAN listing of the gaseous effluent code.

The tape is written in the following format:

DCB = (RECFM = FB, LRECL = 80, BLKSIZE = 3200)

Use of the tape requires two data cards in addition to those described in Chapter 1 containing the plant parameters. For a low enrichment uranium-235 oxide-fueled light water reactor, these cards should always contain the following data:

Card	Column	Input Data
1	1-72	Title
1	75	The value 2
2	1-10	The value 0.632
2	11-20	The value 0.333
2	21-30	The value 2.0
2	31-40	The value 1.0E-25
2	41-46	The date (month, day, year) of the calculation
2	48	The value 1
2	50	The value 0
2	52	The value 0

A description of the information contained in the nuclear data library can be found in the report ORNL-4628, "ORIGEN - The ORNL Isotope Generation and Depletion Code," dated May 1973.

3.4.2 FORTRAN PROGRAM LISTING

Figure 3-3 and 3-4 provides the program listings for the PWR-GALE Code gaseous and liquid determinations.

FIGURE 3-3
PROGRAM LISTING FOR GASEOUS DETERMINATION

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*DECK PGALEGS
C      GALE CODE FOR CALCULATING GASEOUS EFFLUENTS FROM PWRS.  MODIFIED
C     AUG. 1979 TO IMPLEMENT APPENDIX I TO 10 CFR 50.  REACTOR
C     WATER CONCENTRATIONS CALCULATED USING METHODS OF DRAFT STANDARD
C     ANS 237 "RADIOACTIVE MATERIALS IN PRINCIPAL FLUID STREAMS OF
C     LIGHT WATER COOLED NUCLEAR POWER PLANTS"  DRAFT DATED MAY 20, 1974
C
C     THE FOLLOWING FIRST STATEMENT IS SPECIFIC FOR THE CDC USERS.
C     FOR THE IBM USERS, DELETE THIS STATEMENT.
C
      PROGRAM PGALEGS(INPUT,OUTPUT,TAPE5=INPUT,TAPE6=OUTPUT)
      REAL NUCLID(13)
      REAL PPART(18)
      DIMENSION ACONT(13),CBCP(13),CBSP(13),ASHIMC(13),ASHIMS(13)
      DIMENSION CBPP(13)
      DIMENSION ASHIM(13),CVNCP(13),CONCS(13),DECON(13),FHBL(13)
      DIMENSION DECOH(13),NAME(8),EJT(13),TBL(13),CPL(13),AXBL(13)
      DIMENSION BVOG(13),TOT(13),X2(13),X3(13),X4(13),X5(13)
      DIMENSION CTPRO(13),XP1(13),XP2(11),WORD(14),WARD(5),WURD(4)
      DIMENSION PCBL(18),PAXBL(18),PCBP(18),PAXBP(18),PGWS(18),PTOTP(18)
      DIMENSION PGWL(18),PCBCP(18),PCBSP(18),PRCONT(18)
      DIMENSION PFHBL(18),PFHBP(18)
      DIMENSION RNAX(2),RNAXS(2),RNFH(2),RNFHS(2),RNT(2),RNTS(2),RNS(2)
      DATA NUCLID/" KR-85M"," KR-85"," KR-87"," KR-88"," XE-131M",
     1" XE-133M"," XE-133"," XE-135M"," XE-135"," XE-137",
     2" XE-138"," I-131"," I-133"/
      DATA PPART/" CR-51"," MN-54"," CO-57"," CO-58"," CO-60"," FE-59",
     1" SR-89"," SR-90"," ZR-95"," NB-95"," RU-103"," RU-106"," SB-125",
     2" CS-134"," CS-136"," CS-137"," BA-140"," CE-141"/
C
C     XP1 AND XP2 ARE THE PRIMARY COOLANT AND SECONDARY COOLANT
C     CONCENTRATIONS, RESPECTIVELY (MICROCI/GM).
C
      DATA XP1/1.6E-1,4.3E-1,1.5E-1,2.8E-1,7.3E-1,7.0E-2,2.6E+0,1.3E-1.8
     1.5E-1,3.4E-2,1.2E-1,4.5E-2,1.4E-1/
      DATA XP2/3.4E-8,8.9E-8,3.0E-8,5.9E-8,1.5E-7,1.5E-8,5.4E-7,2.7E-8.1
     1.8E-7,7.1E-9,2.5E-8/
C
C     DECAY CONSTANTS FOR THE CORRESPONDING NUCLID (1/SEC).
C
      DATA DECON/4.38E-5,2.03E-9,1.52E-4,6.88E-5,6.80E-7,3.55E-6,1.52E-6
     1,7.41E-4,2.09E-5.2.96E-3,8.14E-4,9.97E-7,9.17E-6/
C
C     NORMALIZED IODINE ANNUAL RELEASE (CI/YR/MICROCI/GM).
C
      DATA RNS/0.32,0.32/
      DATA RNAX/0.68,0.68/
      DATA RNAXS/2.5,2.5/
      DATA RNFH/0.038,0.038/
      DATA RNFHS/0.093.0.093/
      DATA RNT/3.8E3,3.8E3/
      DATA RNTS/4.2E2,4.2E2/
C
C     PARTICULATE ANNUAL RELEASE RATE (CI/YR)
C
      DATA PCBP/9.2E-3,5.3E-3,8.2E-4,2.5E-2,2.6E-3,2.7E-3,1.3E-2,5.2E-3,
     10.0E+0,1.8E-3,1.6E-3,0.0E+0,0.0E+0,2.5E-3,3.2E-3,5.5E-3,0.0E+0,1.3
     2E-3/
      DATA PAXBP/3.2E-4,7.8E-5,0.0E+0,1.9E-3,5.1E-4,5.0E-5,7.5E-4,2.9E-4
     1,1.0E-3,3.0E-5,2.3E-5,6.0E-6,3.9E-6,5.4E-4,4.9E-5,7.2E-4,4.0E-4,2.
     26E-5/
      DATA PFHBP/1.8E-4,3.0E-4,0.0E+0,2.1E-2,8.2E-3,0.0E+0,2.1E-3,8.0E-4
```

```
1,3.6E-6,2.4E-3,3.8E-5,6.9E-5,5.7E-5,1.7E-3,0.0E+0,2.7E-3,0.0E+0,4.
24E-7/
 DATA PGWS/1.4E-5,2.1E-6,0.0E+0,8.7E-6,1.4E-5,1.8E-6,4.4E-5,1.7E-5,
14.8E-6,3.7E-6,3.2E-6,2.7E-6,0.0E+0,3.3E-5,5.3E-6,7.7E-5,2.3E-5,2.2
2E-6/
C
C      BUILT-IN PARAMETERS
C
       OPFRA=0.80
       AUXLR=160.
       EM=2.0
       GENL=75.
       CLFNG=0.03
       CLFI=8.0E-6
       PURTIM=16.
       TBLK=1700.
C
       AFPTEG=0.0
       READ(5,1000)NAME,TYPE
       WRITE(6,1440)
       WRITE(6,1000)NAME,TYPE
       READ(5,1010)WORD,POWTH
       WRITE(6,1010)WORD,POWTH
       WRITE(6,1020)
       READ(5,1010)WORD,PRIVOL
       WRITE(6,1010)WORD,PRIVOL
       READ(5,1010)WORD,DEMIFL
       WRITE(6,1010)WORD,DEMIFL
       READ(5,1010)WORD,CBFLR
       WRITE(6,1010)WORD,CBFLR
       READ(5,1010)WORD,GEN
       WRITE(6,1010)WORD,GEN
       READ(5,1010)WORD,TOSTFL
       WRITE(6,1010)WORD,TOSTFL
       READ(5,1010)WORD,WLI
       WRITE(6,1010)WORD,WLI
       WLI=GEN*WLI
       READ(5,1040)TBD,KFNRT
       WRITE(6,1050)TBD
       IF(KFNRT.EQ.0)FNRTSC=0.99
       IF(KFNRT.EQ.1) FNRTSC=0.9
       IF (KFNRT.EQ.2) FNRTSC=1.0
       READ(5,1010)WORD,REGENT
       WRITE(6,1010)WORD,REGENT
C
C      READ DATA FOR LIQUID INFORMATION
C
       READ(5,1010)WORD,FFCDM
       WRITE(6,1010)WORD,FFCDM
       READ(5,1060)WARD,SBLDR
       CWA=1.0
       READ(5,1070)DFICW,DFCSCW,DFCW
       READ(5,1080)TC,TSTORC,CWFD
       WRITE(6,1090)
       WRITE(6,1100)
       WRITE(6,1110)WARD,SBLDR,CWA,CWFD,TC,TSTORC,DFICW,DFCSCW,DFCW
       READ(5,1120)WARD,EDFLR,EDA
       READ(5,1070)DFIED,DFCSED,DFED
       READ(5,1080)TE,TS,EDFD
       WRITE(6,1110)WARD,EDFLR,EDA,EDFD,TE,TS,DFIED,DFCSED,DFED
       READ(5,1120)WARD,DWFLR,DWA
       READ(5,1070)DFIDW,DFCSDW,DFDW
```

```
      READ(5,1080)TD,TSTORD,DWFD
      WRITE(6,1110)WARD,DWFLR,DWA,DWFD,TD,TSTORD,DFIDW,DFCSDW,DFDW
      READ(5,1120)WARD,DWFL2,DW2
      READ(5,1070)DFID2,DFCSD2,DFD2
      READ(5,1080)T2,TSTOR2,DWF2
      WRITE(6,1110)WARD,DWFL2,DW2,DWF2,T2,TSTOR2,DFID2,DFCSD2,DFD2
      READ(5,1130)BDTFR
      READ(5,1070)DFICM,DFCSCM,DFCM
      READ(5,1080)TCM,TSTORB,CMFD
      READ(5,1130)RGWFR
      READ(5,1070)DFIRG,DFCSRG,DFRG
      READ(5,1080)TRG,TSTORR,RGFD
      IF(TBD.EQ.0.0) GO TO 30
      BDFR=TBD*1.0E3*BDTFR/0.3476                                      005070
      WRITE(6,1140)BDFR,CMFD,TCM,TSTORB,DFICM,DFCSCM,DFCM
      BDFR=TBD*1.0E3*ABS(1.-BDTFR)/0.3476                              005090
      WRITE(6,1150)BDFR
      IF(FFCDM.EQ.0.0) GO TO 50
   30 IF(REGENT.EQ.0.0) GO TO 40
      WRITE(6,1160)RGWFR,RGFD,TRG,TSTORR,DFIRG,DFCSRG,DFRG
      GO TO 50
   40 RGWFR=0.0
      WRITE(6,1160)RGWFR,RGFD,TRG,TSTORR,DFIRG,DFCSRG,DFRG
C                                                                      005190
C     READ DATA FOR GAS INFORMATION
C                                                                      005210
   50 WRITE(6,1170)
      READ(5,1180)KGTRWT
      IF(KGTRWT.EQ.0) GO TO 70
      GTRW=(DEMIFL-SBLDR/1440.)/DEMIFL                                 005250
      IF(KGTRWT.EQ.2) GO TO 60
      WRITE(6,1190)
      GO TO 80
   60 GTRW=0.25*GTRW
      WRITE(6,1200)
      GO TO 80
   70 GTRW=0.0
      WRITE(6,1210)
   80 SRB=GTRW*DEMIFL+(SBLDR+EDFLR)/1440.
      WRITE(6,1220)SRB
      READ(5,1010)WORD,TAU1
      WRITE(6,1010)WORD,TAU1
      READ(5,1010)WORD,TAU2
      WRITE(6,1010)WORD,TAU2
      READ(5,1010)WORD,TAU3
      WRITE(6,1010)WORD,TAU3
      WRITE(6,1230)
      GWPRF=1.0
      AXIRF=1.0
      AXPRF=1.0
      CHIRF=1.0
      CHPRF=1.0
      CLIRF=1.0
      CLPRF=1.0
      FHIRF=1.0
      FHPRF=1.0
      CAIRF=1.0
      CAPRF=1.0
      READ(5,1250)WORD,GWHRE
      IF(GWHRE.GT.0.0)GWPRF=1.0-GWHRE/100.
      WRITE(6,1260)WORD,GWPRF
      READ(5,1270)WARD,FHCHRE,FOHRE
      IF(FHCHRE.GT.0.0)FHIRF=1.0-FHCHRE/100.
```

```
      IF(FHHRE.GT.0.0)FHPRF=1.0-FHHRE/100.
      WRITE(6,1280)WARD,FHIRF,FHPRF
      READ(5,1270)WARD,AXCHRE,AXHRE
      IF(AXCHRE.GT.0.0)AXIRF=1.0-AXCHRE/100.
      IF(AXHRE.GT.0.0)AXPRF=1.0-AXHRE/100.
      WRITE(6,1280)WARD,AXIRF,AXPRF
      READ(5,1010)WORD,CONVOL
      WRITE(6,1010)WORD,CONVOL
      WRITE(6,1290)
      READ(5,1370)WARD,CACHRE,CAHRE,CFM
      IF(CACHRE.GT.0.0)CAIRF=1.0-CACHRE/100.
      IF(CAHRE.GT.0.0)CAPRF=1.0-CAHRE/100.
      IF(CFM.EQ.0.0) GO TO 90
      KID=1
      WRITE(6,1300)CFM,PURTIM
      GO TO 100
   90 KID=0
      WRITE(6,1310)
  100 IF(FFCDM.GT.0.0) GO TO 110
      WRITE(6,1320)
      GO TO 120
  110 FIBCD=1.0-FFCDM
      WRITE(6,1330)FIBCD
  120 IF(TBD.EQ.0.0) GO TO 130
      CON=0.01
      WRITE(6,1340)CON
      GO TO 140
  130 CON=1.0
      WRITE(6,1340)CON
  140 READ(5,1350)WARD,CHCHRE,CHHRE,ENP
      IF(CHCHRE.GT.0.0)CHIRF=1.0-CHCHRE/100.
      IF(CHHRE.GT.0.0)CHPRF= .0-CHHRE/100.
      EN=2.0+ENP
      WRITE(6,1360)EN
      WRITE(6,1280)WARD,CHIRF,CHPRF
      READ(5,1370)WARD.CLCHRE,CLHRE,PNOV1
      IF(CLCHRE.GT.0.0)CLIRF=1.0-CLCHRE/100.
      IF(CLHRE.GT.0.0)CLPRF=1.0-CLHRE/100.
      IF(PNOV1.LT.1.0) GO TO 150
      WRITE(6,1380)WARD,PNOV1,WARD,CLIRF,CLPRF
      GO TO 160
  150 WRITE(6,1390)
  160 WRITE(6,1400)TBLK
      READ(5,1010)WORD,FVN
      WRITE(6,1010)WORD,FVN
      READ(5,1010)WORD,FEJP
      FEJP=1.0-FEJP/100.
      WRITE(6,1010)WORD,FEJP
      READ(5,1010)WORD,PFLAUN
      IF(PFLAUN.LE.0.0) WRITE(6,1430)
C
C     CONVERSION OF UNITS
C
      TOSTFL=TOSTFL*1000000.
      WLI=WLI*1000.
      CONVOL=CONVOL*1000000.
      CFM=CFM*1000.
      TBD=TBD*1E3
      PRIVOL=PRIVOL*1E3
      DEMIFL=DEMIFL*500.53
      SBLDR=SBLDR*.3476
      EDFLR=EDFLR*.3476
```

```
      DWFLR=DWFLR*.3476
      DWFL2=DWFL2*.3476
      CBFLR=CBFLR*500.53
C
C     H3COPW IS THE PWR TRITIUM PRIMARY COOLANT CONCENTRATION IN UCI/GM
C
      H3COPW=1.0
      H3PRPW=0.4*POWTH
      TPLRPW=SBLDR*CWA*CWFD+EDFLR*EDA*EDFD+DWFLR*DWA*DWFD+DWFL2*DW2*DWF2
      H3RLPW=TPLRPW*H3COPW*3.97
      IF(H3RLPW.GT.0.9*H3PRPW)H3RLPW=0.9*H3PRPW
      H3RLG=H3PRPW-H3RLPW
      DIV=10.**(INT(ALOG10(H3RLG))-1)
      IDIV=DIV
      IH3RLG=INT(H3RLG/DIV+0.5)*IDIV
      IF(TAU3.EQ.0.)TAU3=.01
      SRB=SRB*500.53
      PE=365./TAU3
      T1=3.1557E7/EN*OPFRA
      T3=3.1557E+07/PE
      T4=TAU1*86400.
      T5=TAU2*86400.
      DO 190 I=1,13
  190 DECOH(I)=DECON(I)*3600.
      DO 200 I=1,13
  200 CONCP(I)=XP1(I)
      IF(POWTH.LT.3000..OR.POWTH.GT.3800.) GO TO 210
      IF(PRIVOL.LT.5.0E5.OR.PRIVOL.GT.6.0E5) GO TO 210
      IF(DEM1FL.LT.3.2E4.OR.DEM1FL.GT.4.2E4) GO TO 210
      IF(SBLDR.LT.250..OR.SBLDR.GT.1000.) GO TO 210
      IF(CBFLR.GT.7500.) GO TO 210
      IF(KGTRWT.GT.0) GO TO 210
      GO TO 240
  210 AFPTEG=1.0
      RNG2=(SBLDR+DEM1FL* GTRW)/PRIVOL
      RHAL2=(DEM1FL*0.99+0.91*SBLDR)/PRIVOL
      RK2G=161.76*POWTH/PRIVOL
      DO 230 I=1,13
      IF(I.GT.11) GO TO 220
      CONCP(I)=CONCP(I)*RK2G*(.0009+DECOH(I))/(RNG2+DECOH(I))
      GO TO 230
  220 CONCP(I)=CONCP(I)*RK2G*(0.067+DECOH(I))/(RHAL2+DECOH(I))
  230 CONTINUE
  240 IF(TBD.EQ.0.0) GO TO 280
C
C     PWTYPE=1.0 IS FOR PWRS WITH U-TUBE STEAM GENERATORS
C
      PWTYPE=1.0
      DO 250 I=1,11
  250 CONCS(I)=XP2(I)
      CONCS(12)=1.8E-6
      CONCS(13)=4.8E-6
      IF(AFPTEG.EQ.1.0) GO TO 300
      IF(WLI.LT.4.0E5.OR.WLI.GT.5.0E5) GO TO 300
      IF(TOSTFL.LT.1.3E7.OR.TOSTFL.GT.1.7E7) GO TO 300
      IF(TBD.LT.5.0E4.OR.TBD.GT.1.0E5) GO TO 300
      IF (FFCDM.GT.0.01) GO TO 300
      GO TO 340
C
C     PWTYPE=2.0 IS FOR PWRS WITH ONCE-THROUGH STEAM GENERATORS
C
  280 PWTYPE=2.0
      DO 290 I=1,11
```

```
 290  CONCS(I)=XP2(I)
      CONCS(12)=5.2E-8
      CONCS(13)=1.6E-7
      IF(AFPTEG.EQ.1.0) GO TO 300
      IF(TOSTFL.LT.1.3E7.OR.TOSTFL.GT.1.7E7) GO TO 300
      IF(FFCDM.LT.0.55.OR.FFCDM.GT.0.75) GO TO 300
      GO TO 340
  300 CONTINUE
      IF (FFCDM.GT.0.01.AND.FFCDM.LT.1.0) FFCDM=0.2
      RHAL3=(TBD*FNRTSC+.9*CON*TOSTFL*FFCDM)/WLI
      DO 330 I=1,13
      IF(I.GT.11) GO TO 310
      CONCS(I)=CONCS(I)*1.5E7/TOSTFL*(CONCP(I)/XP1(I))
      GO TO 330
 310  IF(PWTYPE.EQ.2.0) GO TO 320
      CONCS(I)=CONCS(I)*(4.5E5/WLI)*(0.17+DECOH(I))/(RHAL3+DECOH(I))*
     1(CONCP(I)/XP1(I))
      GO TO 330
320   CONCS(I)=CONCS(I)*(1.0E5/WLI)*(27.0+DECOH(I))/(RHAL3+DECOH(I))*
     1(CONCP(I)/XP1(I))
 330  CONTINUE
 340  PNOV=PNOV1/CONVOL*60.
C
C     THIS PART OF PROGRAM IS FOR NOBLE GASES
C
      J=0
      DO 370 I=1,13
      X2(I)=(DECON(I)+PNOV/3600.)*T1
      IF(X2(I).GT.30.) X2(I)=30.
      X3(I) = DECON(I) * T3
      IF(X3(I).GT.30.) X3(I)=30.
      X4(I) = DECON(I) * T4
      IF(X4(I).GT.30.) X4(I)=30.
      X5(I) = DECON(I) * T5
      XDK=X5(I)
      IF(X5(I).GT.30.) XDK=30.
      IF(I.GT.11) GO TO 350
      IF(I.GT.4) XDK=X4(I)
      CTPRO(I)=(CONCP(I)*PRIVOL*CLFNG)/(DECOH(I)+PNOV)*1.892E-5
      ACONT(I)=CTPRO(I)*(1.-EXP(-X2(I)))
      ASHIM(I)=(CONCP(I)*SRB)/DECOH(I)*4.54E-4*(1.-EXP(-X3(I)))
      AXBL(I)=CONCP(I)*AUXLH*.1657*OPFRA
      CBCP(I)=  EN * PNOV *  (CTPRO(I)*T1/3600.+CTPRO(I)*(EXP(-X2(I))-1.)
     1/(DECOH(I)+PNOV))
      CBSP(I)=EN*ACONT(I)
      CBL(I)=CBCP(I)+CBSP(I)
      ASHIMC(I)=PE*ASHIM(I)*EXP(-XDK)*OPFRA
      ASHIMS(I)=EM*CONCP(I)*PRIVOL*4.54E-4*EXP(-XDK)
      EJT(I)=CONCS(I)*TOSTFL*3.977*OPFRA
      TBL(I)=CONCS(I)*TBLK*3.977*OPFRA
      FHBL(I)=0.0
      BVOG(I)=0.0
      TEST=1.0
      IF(CBL(I).LT.TEST)CBL(I)=0.0
      IF(ASHIMS(I).LT.TEST)ASHIMS(I)=0.0
      IF(ASHIMC(I).LT.TEST)ASHIMC(I)=0.0
      IF(EJT(I).LT.TEST)EJT(I)=0.0
      IF(TBL(I).LT.TEST)TBL(I)=0.0
      IF(AXBL(I).LT.TEST)AXBL(I)=0.0
      GO TO 370
C
C     THIS PART OF PROGRAM IS FOR IODINE
```

```
C
 350  CTPRO(I)=(CONCP(I)*PRIVOL*CLFI)/(DECOH(I)+PNOV)*1.892E-5
      ACONT(I)=CTPRO(I)*(1.-EXP(-X2(I)))
      J=J+1
      AXBL(I)=(RNAX(J)+RNAXS(J))*CONCP(I)*AXIRF
      FHBL(I)=(RNFH(J)+RNFHS(J))*CONCP(I)*FHIRF
      ASHIMC(I)=0.0
      ASHIMS(I)=0.0
      ECBCP(I)=  EN * PNOV *  (CTPRO(I)*T1/3600.+CTPRO(I)*(EXP(-X2(I))-1.)
     1/(DECOH(I)+PNOV))*CLIRF
      CBPP(I)=ENP*ACONT(I)*CHIRF
      CBSP(I)=RNS(J)*CONCP(I)*CBIRF
      CBL(I)=CBCP(I)+CBSP(I)+CBPP(I)
      EJT(I)=1.7E3*CONCS(I)*CON*FEJP
      TBL(I)=(RNT(J)+RNTS(J))*CONCS(I)*CON
      BVOG(I)=CONCS(I)*TBD*LVN*3.977*OPFRA
      IF(KID.EQ.0) GO TO 360
      DLAK=(CFM*60.*CACHRE*0.01*0.7/CONVOL)+DECOH(I)
      EXX2=DLAK*PURTIM
      IF(EXX2.GT.30.) EXX2=30.
      EXPF=EXP(-EXX2)
      EXPC=1.-EXPF
      ELSS=CHIRF*CONCP(I)*PRIVOL*CLFI*1.892E-5/DLAK*EXPC
      CBL(I)=CBPP(I)*EXPF+ELSS*ENP+CBCP(I)*(1.-PURTIM/(8760.*OPFRA/EN))+
     12*CHIRF*0.16*CONCP(I)*PURTIM/(24.*32.5)*EXPC/(DLAK*PURTIM)
     2+(CBSP(I)-2.*CHIRF*0.16*CONCP(I)*PURTIM/(24.*32.5))
 360  TEST=0.0001
      IF(CBL(I).LT.TEST)CBL(I)=0.0
      IF(EJT(I).LT.TEST)EJT(I)=0.0
      IF(BVOG(I).LT.TEST)BVOG(I)=0.0
      IF(TBL(I).LT.TEST)TBL(I)=0.0
      IF(AXBL(I).LT.TEST)AXBL(I)=0.0
      IF(FHBL(I).LT.TEST)FHBL(I)=0.0
 370  CONTINUE
      MSIG=1
      NSIG=13
      CALL SIGF2(CBL,MSIG,NSIG)
      CALL SIGF2(ASHIMS,MSIG,NSIG)
      CALL SIGF2(ASHIMC,MSIG,NSIG)
      CALL SIGF2(EJT,MSIG,NSIG)
      CALL SIGF2(BVOG,MSIG,NSIG)
      CALL SIGF2(TBL,MSIG,NSIG)
      CALL SIGF2(AXBL,MSIG,NSIG)
      CALL SIGF2(FHBL,MSIG,NSIG)
      DO 380 I=1,13
      TOT(I)=CBL(I)+EJT(I)+TBL(I)+AXBL(I)+FHBL(I)+BVOG(I)+ASHIMC(I)+ASHI
     1MS(I)
 380  CONTINUE
      CALL SIGF2(TOT,MSIG,NSIG)
      WRITE(6,1440)
      WRITE(6,1450)NAME
      WRITE(6,1460)
      WRITE(6,1560)
      WRITE(6,1480)
      DO 385 I=12,13
      WRITE(6,1495)NUCLID(I),CONCP(I),CONCS(I),FHBL(I),
     1CBL(I),AXBL(I),TBL(I),BVOG(I),EJT(I),TOT(I)
 385  CONTINUE
      WRITE(6,1480)
      WRITE(6,1510) IH3RLG
      WRITE(6,1440)
      WRITE(6,1450)NAME
      WRITE(6,1460)
```

```
      WRITE(6,1470)
      WRITE(6,1480)
      GASTOT=0.0
      DO 390 I=1,11
      WRITE(6,1490)NUCLID(I),CONCP(I),CONCS(I),ASHIMS(I),ASHIMC(I),
     1CBL(I),AXBL(I),TBL(I),BVOW(I),EJT(I),TOT(I)
      GASTOT=GASTOT+TOT(I)
 390  CONTINUE
      DIV=10.**(INT(ALOG10(GASTOT))-1)
      GASTOT=AINT(GASTOT/DIV+0.5)*DIV
      WRITE(6,1500) GASTOT
      WRITE(6,1480)
      WRITE(6,1520)
      WRITE(6,1440)
      WRITE(6,1450)NAME
      WRITE(6,1530)
      WRITE(6,1540)
      WRITE(6,1480)
      QH=8760.*OPFRA/EN
C
C     THIS PART OF PROGRAM IS FOR THE PARTICULATES
C
      DO 430 I=1,18
      PRCONT(I)=PCBP(I)/(8760.*OPFRA)
      IF(PNOV.GT.0.0) GO TO 410
      PCBCP(I)=0.0
      PCBSP(I)=EN*PRCONT(I)*QH*CHPRF
      GO TO 420
 410  PCBCP(I)=(EN*(QH*PRCONT(I)-PRCONT(I)/PNOV*(1.-EXP(-PNOV*QH))))
     1*CLPRF
      PCBSP(I)=(EN*(PRCONT(I)/PNOV*(1.0-EXP(-PNOV*QH))))*CHPRF
 420  PCBL(I)=PCBCP(I)+PCBSP(I)
      PAXBL(I)=PAXBP(I)*AXPRF
      PFHBL(I)=PFHBP(I)*FHPRF
      PGWL(I)=PGWS(I)*GWPRF
      IF(KID.EQ.0) GO TO 430
      PDLAK=CFM*60.*CAHRE*0.01*0.7/CONVOL
      PEXX2=PDLAK*PURTIM
      IF(PEXX2.GT.30.) PEXX2=30.
      PEXPF=EXP(-PEXX2)
      PEXPC=1.-PEXPF
      PELSS=PRCONT(I)/PDLAK*PEXPC*CHPRF
      PCBL(I)=PCBSP(I)*PEXPF+PELSS*EN+PCBCP(I)*(1.-PURTIM/(8760.*OPFRA/E
     1N))
 430  CONTINUE
      MSIG=2
      NSIG=18
      CALL SIGF2(PCBL,MSIG,NSIG)
      DO 440 I=1,18
      PTOTP(I)=PCBL(I)+PAXBL(I)+PGWL(I)+PFHBL(I)
 440  CONTINUE
      CALL SIGF2(PTOTP,MSIG,NSIG)
      DO 450 I=1,18
      WRITE(6,1550)PPART(I),PGWL(I),PCBL(I),PAXBL(I),PFHBL(I),PTOTP(I)
 450  CONTINUE
      WRITE(6,1480)
      STOP
C
C   FORMATS           FORMATS           FORMATS           FORMATS
C
1000  FORMAT(32X,8A4,12X,A4)
1005  FORMAT(16X,"BLOWDOWN IS PROCESSED THROUGH CONDENSATE DEMIN")
```

```
1007  FORMAT(16X,"BLOWDOWN IS NOT PROCESSED THROUGH COND. DEMIN.")
1010  FORMAT(16X,13A4,A2,F10.5)
1020  FORMAT(16X,"PLANT CAPACITY FACTOR",T74,"0.80")
1030  FORMAT(16X,"PERCENT FUEL WITH CLADDING DEFECTS",T74,F7.5)
1040  FORMAT(36X,F8.4,35X,I1)
1050  FORMAT(16X,"BLOWDOWN RATE (THOUSAND LBS/HR)",24X,F9.5)
1060  FORMAT(15X,4A4,A2,8X,F8.0)
1070  FORMAT(20X,F8.0,2(5X,F8.0))
1080  FORMAT(27X,F6.2,14X,F9.2,18X,F6.2)
1090  FORMAT(/," LIQUID WASTE INPUTS")
1100  FORMAT   (30X,"FRACTION  FRACTION  COLLECTION  DECAY"/8X,"STREAM
     1      FLOW RATE   OF PCA DISCHARGED    TIME     TIME",5X," DECONTAM
     2INATION  FACTORS"/20X,"(GAL/DAY)",23X,"(DAYS)    (DAYS)",7X,
     3"I",8X,"CS",8X,"OTHERS")
1110  FORMAT(2X,4A4,A2,1PE9.2,1X,4(0PF8.4,2X),3(1PE9.2,1X))
1120  FORMAT(15X,4A4,A2,8X,F8.0,7X,F6.4)
1130  FORMAT(70X,F10.5)
1140  FORMAT(2X,"BLOWDOWN",10X,1PE9.2,11X,F8.3,2X,2(F8.3,2X),
     13(1PE9.2,1X))
1150  FORMAT(2X,"UNTREATED BLOWDOWN",1PE9.2,11X,"    1.000      0.000
     10.000   1.00E+00  1.00E+00  1.00E+00")
1160  FORMAT(2X,"REGENERANT SOLS   ",1PE9.2,14X,0PF5.3,2X,2(F8.3,2X),
     13(1PE9.2,1X))
1170  FORMAT(/," GASEOUS WASTE INPUTS")
1180  FORMAT(79X,I1)
1190  FORMAT(16X,"THERE IS CONTINUOUS STRIPPING OF FULL LETDOWN FLOW")
1200  FORMAT(16X,"THERE IS CONTINUOUS LOW VOL PURGE OF VOL. CONTROL TK")
1210  FORMAT(16X,"THERE IS NOT CONTINUOUS STRIPPING OF FULL LETDWN FLOW"
     1)
1220  FORMAT(16X,"FLOW RATE THROUGH GAS STRIPPER (GPM)",19X,F9.5)
1230  FORMAT(16X,"PRIMARY COOLANT LEAK TO AUXILIARY BLDG (LB/DAY)",T72,
     1"160.00000")
1250  FORMAT(16X,4A4,6X,F3.0)
1260  FORMAT(16X,4A4,4X,"PARTICULATE RELEASE FRACTION",6X,F10.5)
1270  FORMAT(16X,5A4,10X,F3.0,6X,F3.0)
1280  FORMAT(16X,5A4,"IODINE RELEASE FRACTION",11X,F10.5/36X,"PARTICULAT
     1E RELEASE FRACTION",6X,F10.5)
1290  FORMAT(16X,"FREQUENCY OF PRIMARY COOLANT DEGASSING (TIMES/YR)",T74
     1,"2.00000"/16X,"PRIMARY TO SECONDARY LEAK RATE (LB/DAY)",T72,
     2" 75.00000")
1300  FORMAT(16X,"THERE IS A KIDNEY FILTER"/20X,"CONTAINMENT ATMOSPHERE
     1CLEANUP RATE (THOUSAND CFM)",T71,F10.5/20X,"PURGE TIME OF CONTAINM
     2ENT (HOURS)",T71,F10.5)
1310  FORMAT(16X,"THERE IS NOT A KIDNEY FILTER")
1320  FORMAT(16X,"THERE IS NOT A CONDENSATE DEMINERALIZER")
1330  FORMAT(16X,"FRACTION IODINE BYPASSING CONDENSATE DEMINERALIZER",
     17X,T72,F9.5)
1340  FORMAT(16X,"IODINE PARTITION FACTOR (GAS/LIQUID) IN STEAM GENERATO
     1R  ",F7.5)
1350  FORMAT(16X,5A4,10X,F3.0,6X,F3.0,19X,F3.0)
1360  FORMAT(16X,"FREQUENCY OF CNTMT BLDG HIGH VOL PURGE(TIMES/YR)",
     1T73,F8.5)
1370  FORMAT(16X,5A4,10X,F3.0,6X,F3.0,14X,F8.2)
1380  FORMAT(16X,5A4,"RATE(CFM)",24X,F11.5/16X,5A4,"IODINE RELEASE FRACT
     1ION",11X,F10.5/36X,"PARTICULATE RELEASE FRACTION",6X,F10.5)
1390  FORMAT(16X,"THERE IS NOT A CNTMT BLDG LOW VOLUME PURGE")
1400  FORMAT(16X,"STEAM LEAK TO TURBINE BLDG (LBS/HR)",19X,F10.5)
1430  FORMAT("0",15X,"THERE IS NOT AN ON-SITE LAUNDRY")
1440  FORMAT(1H1)
1450  FORMAT(16X,8A4)
1460  FORMAT(1H0,67X,"GASEOUS RELEASE RATE - CURIES PER YEAR")
1470  FORMAT(1H0,11X,"PRIMARY",4X,"SECONDARY",7X,"GAS STRIPPING",11X,
     1"BUILDING VENTILATION"/12X,"COOLANT",5X,"COOLANT",5X,21("-"),
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     24X,30("-"),5X,"BLOWDOWN   AIR EJECTOR   TOTAL"/10X,"(MICROCI/GM)(M
     3ICROCI/GM) SHUTDOWN   CONTINUOUS    REACTOR    AUXILIARY   TURBINE
     4   VENT OFFGAS   EXHAUST")
1480  FORMAT(1H0,130("-"))
1490  FORMAT("0 ",A8,2(2X,1PE10.3),8(3X,1PE8.1,1X))
1495  FORMAT("0 ",A8,2(2X,1PE10.3),12X,7(3X,1PE8.1,1X))
1500  FORMAT(1H0,"  TOTAL NOBLE GASES",101X,1PE8.1)
1510  FORMAT(1H0,30X,"TOTAL H-3 RELEASED VIA GASEOUS PATHWAY = ",I4," CI
     1/YR"//31X,"C-14 RELEASED VIA GASEOUS PATHWAY = 7.3 CI/YR"//31X,
     2"AR-41 RELEASED VIA CONTAINMENT VENT =  34 CI/YR")
1520  FORMAT(1H0,"0.0 APPEARING IN THE TABLE INDICATES RELEASE IS LESS
     1THAN 1.0 CI/YR FOR NOBLE GAS, 0.0001 CI/YR FOR I")
1530  FORMAT(1H0,54X,"AIRBORNE PARTICULATE RELEASE RATE-CURIES PER YEAR"
     ")
1540  FORMAT(1H0,36X,"WASTE GAS",16X,"BUILDING VENTILATION"/2X,"NUCLIDE"
     1,28X,"SYSTEM",14X,"REACTOR     AUXILIARY  FUEL HANDLG",7X,"TOTAL")
1550  FORMAT(1H0,A8,28X,1PE8.1,11X,1PE8.1,4X,1PE8.1,4X,1PE8.1,10X,1PE8.1
     1)
1560  FORMAT(1H0,11X,"PRIMARY",4X,"SECONDARY",25X,"BUILDING VENTILATION"
     1/12X,"COOLANT",5X,"COOLANT",15X,44("-"),6X,"BLOWDOWN   AIR EJECTOR
     2   TOTAL"/10X,"(MICROCI/GM)(MICROCI/GM)",12X,"FUEL HANDLG   REACTO
     3R    AUXILIARY   TURBINE    VENT OFFGAS   EXHAUST")
      END
*DECK SIGF2
      SUBROUTINE SIGF2(RLPT,MSIG,NSIG)
      DIMENSION RLPT(NSIG)
      IF (MSIG.EQ.2) GO TO 30
      DO 20 I=1,NSIG
      IF (RLPT(I).EQ.0.0) GO TO 20
      IF(I.GT.11) GO TO 10
C     THIS PART OF SUBROUTINE IS FOR NOBLE GASES
      DIV=10.**(INT(ALOG10(RLPT(I)))-1)
      IF (RLPT(I).LT.10.) DIV=1.00
      RLPT(I)=AINT(RLPT(I)/DIV+0.5)*DIV
      GO TO 20
C     THIS PART OF SUBROUTINE IS FOR IODINE
   10 CONTINUE
      ISUB=2
      IF(RLPT(I).GT.1.0)ISUB=1
      DIV=10.**(INT(ALOG10(RLPT(I)))-ISUB)
      RLPT(I)=AINT(RLPT(I)/DIV+0.5)*DIV
   20 CONTINUE
   30 CONTINUE
C     THIS PART OF SUBROUTINE IS FOR PARTICULATES
      DO 50 I=1,NSIG
      IF (RLPT(I).EQ.0.) GO TO 50
      DIV=10.**(INT(ALOG10(RLPT(I)))-2)
      RLPT(I)=AINT(RLPT(I)/DIV+0.5)*DIV
   50 CONTINUE
      RETURN
      END
```

FIGURE 3-4
PROGRAM LISTING FOR LIQUID DETERMINATION

```
*DECK PGALELQ
C        GALE CODE FOR CALCULATING LIQUID EFFLUENTS FROM PWRS.  MODIFIED
C       AUG. 1979 TO IMPLEMENT APPENDIX I TO 10 CFR PART 50.  REACTOR
C       WATER CONCENTRATIONS CALCULATED USING METHODS OF DRAFT STANDARD
C       ANS 237 "RADIOACTIVE MATERIALS IN PRINCIPAL FLUID STREAMS OF
C       LIGHT WATER COOLED NUCLEAR POWER PLANTS"  DRAFT DATED MAY 20, 1974
C       MODIFIED EDITION OF ORIGEN PROGRAM TO COMPUTE EFFLUENTS FROM BWR
C       AND PWR RADWASTE SYSTEMS
C
C       STATEMENTS *PROGRAM PGALELQ* AND *LEVEL 2* ARE FOR CDC USERS.
C       FOR IBM USERS DELETE THESE STATEMENTS.
C
        PROGRAM PGALELQ (INPUT,OUTPUT,TAPE5=INPUT,TAPE6=OUTPUT,
      1  TAPE8)
        REALLETDWN,NOGEN
        REALLETDWA
        COMMON/MATRIX/A(2500),LOC(2500),NONO(800),KD(800)
        LEVEL 2,A,LOC,NONO,KD
        COMMON/CONST/MMN,ERR,MZERO
        COMMON/EQ/XTEMP(800),ANEW(10,800),B(800),D(800)
        COMMON/FLUXN/REGENT,DIS(800),ILITE,IACT,ITOT
        COMMON/OUT/NUCL(800)
        COMMON/CONC/PCONC(800),SCON(800),RINV(800)
        COMMON/COOL/REACTR,POW1,SBLDR,BLWDWN,FPEF,HEF,EDFLR,DFIED,DFCSED,
      1 DFED,DFIDW,DFCSDW,DFDW,EDA,DWA,CWA,DFCM,DFICM,DFCSCM,
      2 DFCW,DFICW,DFCSCW,BDTFR,EDFD,DWFD,CWFD,CMFD,IS,TE,TD,TC,TCM,
      3 TSTORC,TSTORD,TSTORB,DWFL2,DW2,DWF2,I2,TSTOR2,DFID2,DFCSD2,DFD2,
      4 PFLAUN,DWFLR
        COMMON/APCOOL/RGWFR,DFIRG,DFCSRG,DFRG,TRG,TSTORR,RGFD
        COMMON/BDTES/RFNRT
        COMMON/CONP/PWCONC(800),SCUTV(800),SCOT(800)
        DIMENSION WORD15(4),WORD18(5),WORD56(14),WORD0(2),REACTR(7)
C
C       READ NUCLEAR DATA AND CONSTRUCT TRANSITION MATRIX
C
        CALL NUDATA(NLIBE)
C
        DO 20 I=2,ITOT
        NONO(I)=NONO(I)+NONO(I-1)
   20   KD(I)=KD(I)+NONO(I-1)
C
C       BUILT-IN PARAMETERS
C
        PF=0.80
        TBLK=1700.
C
        MMN=0
        MZERO=21
        DO 30 J=1,800
        PCONC(J)=0.0
        SCON(J)=0.0
        RINV(J)=0.0
   30   CONTINUE
C
C       READ DESCRIPTION OF REACTOR AND RADWASTE TREATMENT PLANT
C
        PRINT 9026
        READ 9010,REACTR,TYPE
        PRINT 9010,REACTR,TYPE
        READ 9011,WORD56,POW1
        PRINT 9011,WORD56,POW1
```

```
      PRINT 9027
C
C     READ DATA FOR LIQUID INFORMATION
C
      READ 9022,WORD56,PCVOL
      PRINT 9022,WORD56,PCVOL
      READ 9012,WORD56,LETDWN
      PRINT 9012,WORD56,LETDWN
      READ 9012,WORD56,CBFLR
      PRINT 9012,WORD56,CBFLR
      READ 9011,WORD56,NOGEN
      PRINT 9011,WORD56,NOGEN
      READ 9022,WORD56,STMFR
      PRINT 9022,WORD56,STMFR
      READ 9022,WORD56,WLI
      PRINT 9022,WORD56,WLI
      SCVOL=NOGEN*WLI
      PRINT 9029,SCVOL
      READ 9055,BLWDWN,KFNRT
      PRINT 9051,BLWDWN
      RFNRT=1.0
      IF (KFNRT.EQ.2) RFNRT=0.0
      PRINT  9041
      READ 9012,WORD56,REGENT
      PRINT 9012,WORD56,REGENT
      IF(BLWDWN.EQ.0.0) GO TO 40
      FPEF=0.005
      HEF=0.01
      PRINT 9030,FPEF,HEF
      GO TO 50
   40 FPEF=1.0
      HEF=1.0
      PRINT 9030,FPEF,HEF
   50 READ 9020,WORD56,FFCDM
      PRINT 9020,WORD56,FFCDM
      IF(FFCDM.LT.0.001) GO TO 60
      DFCB=10.0
      DFCBCS=2.0
      GO TO 70
   60 DFCB=1.0
      DFCBCS=1.0
   70 READ 9056,WORD18,SBLDR
      CWA=1.0
      READ 9014,DFICW,DFCSCW,DFCW
      READ 9015,TC,TSTORC,CWFD
      PRINT 9045
      PRINT 9016
      PRINT 9017,WORD18,SBLDR,CWA,CWFD,TC,TSTORC,DFICW,DFCSCW,DFCW
      READ 9013,WORD18,EDFLR,WORD8,EDA
      READ 9014,DFIED,DFCSED,DFED
      READ 9015,TE,TS,EDFD
      PRINT 9017,WORD18,EDFLR,EDA,EDFD,TE,TS,DFIED,DFCSED,DFED
      READ 9013,WORD18,DWFLR,WORD8,DWA
      READ 9014,DFIDW,DFCSDW,DFDW
      READ 9015,TD,TSTORD,DWFD
      PRINT 9017,WORD18,DWFLR,DWA,DWFD,TD,TSTORD,DFIDW,DFCSDW,DFDW
      READ 9013,WORD18,DWFL2,WORD8,DW2
      READ 9014,DFID2,DFCSD2,DFD2
      READ 9015,T2,TSTOR2,DWF2
      PRINT 9017,WORD18,DWFL2,DW2,DWF2,T2,TSTOR2,DFID2,DFCSD2,DFD2
      READ 9037,BDTFR
      READ 9014,DFICM,DFCSCM,DFCM
      READ 9015,TCM,TSTORB,CMFD
```

```
      READ 9037,RGWFR
      READ 9014,DFIRG,DFCSRG,DFRG
      READ 9015,TRG,TSTORR,RGFD
      IF(BLWDWN.EQ.0.0) GO TO 75
      BDFR=BLWDWN*1E3*BDTFR/0.3476
      PRINT 9034,BDFR,CMFD,TCM,TSTORB,DFICM,DFCSCM,DFCM
      BDFR=BLWDWN*1.0E3*ABS(1.-BDTFR)/0.3476
      PRINT 9035,BDFR
      IF(FFCDM.EQ.0.0) GO TO 90
   75 IF(REGENT.EQ.0.0) GO TO 80
      PRINT 9038,RGWFR,RGFD,TRG,TSTORR,DFIRG,DFCSRG,DFRG
      GO TO 90
   80 RGWFR=0.0
      PRINT 9038,RGWFR,RGFD,TRG,TSTORR,DFIRG,DFCSRG,DFRG
   90 IF (KFNRT.EQ.2) GO TO 100
      FNRTSO=1.0-1.0/(DFCM*DFCB)
      FNRTSI=1.0-1.0/(DFICM*DFCB)
      FNRTSC=1.0-1.0/(DFCSCM*DFCBCS)
      GO TO 110
  100 FNRTSO=1.0
      FNRTSI=1.0
      FNRTSC=1.0
C
C     READ DATA FOR GAS INFORMATION
C
  110 PRINT 9046
      READ 9021,KGTRWT
      IF (KGTRWT.EQ.0) PRINT 9053
      IF (KGTRWT.EQ.1) PRINT 9052
      IF (KGTRWT.EQ.2) PRINT 9075
      READ 9012,WORD56,TAU1
      PRINT 9012,WORD56,TAU1
      READ 9012,WORD56,TAU2
      PRINT 9012,WORD56,TAU2
      READ 9012,WORD56,TAU3
      PRINT 9012,WORD56,TAU3
      GWPRF=1.0
      AXIRF=1.0
      AXPRF=1.0
      CHIRF=1.0
      CHPRF=1.0
      CLIRF=1.0
      CLPRF=1.0
      FHIRF=1.0
      FHPRF=1.0
      CAIRF=1.0
      CAPRF=1.0
      READ 9065,WORD15,GWHRE
      IF(GWHRE.GT.0.0) GWPRF=1.0-GWHRE/100.
      PRINT 9066,WORD15,GWPRF
      READ 9067,WORD18,FHCHRE,FHHRE
      IF(FHCHRE.GT.0.0) FHIRF=1.0-FHCHRE/100.
      IF(FHHRE.GT.0.0) FHPRF=1.0-FHHRE/100.
      READ 9067,WORD18,AXCHRE,AXHRE
      IF(AXCHRE.GT.0.0) AXIRF=1.0-AXCHRE/100.
      IF(AXHRE.GT.0.0) AXPRF=1.0-AXHRE/100.
      PRINT 9068,WORD18,AXIRF,AXPRF
      READ 9022,WORD56,CONVOL
      PRINT 9022,WORD56,CONVOL
      READ 9069,WORD18,CACHRE,CAHRE,CFM
      IF(CACHRE.GT.0.0) CAIRF=1.0-CACHRE/100.
      IF(CAHRE.GT.0.0) CAPRF=1.0-CAHRE/100.
```

```
      READ 9071,WORD18,CHCHRE,CHHRE
      IF(CHCHRE.GT.0.0) CHIRF=1.0-CHCHRE/100.
      IF(CHHRE.GT.0.0) CHPRF=1.0-CHHRE/100.
      EN=2.0
      PRINT 9072,EN
      PRINT 9068,WORD18,CHIRF,CHPRF
      READ 9069,WORD18,CLCHRE,CLHRE,PNOV1
      IF(CLCHRE.GT.0.0) CLIRF=1.0-CLCHRE/100.
      IF(CLHRE.GT.0.0) CLPRF=1.0-CLHRE/100.
      IF(PNOV1.LT.1.0) GO TO 120
      PRINT 9070,WORD18,PNOV1,WORD18,CLIRF,CLPRF
      GO TO 130
  120 PRINT 9073
  130 PRINT 9064,TBLK
      READ 9020,WORD56,FVN
      PRINT 9020,WORD56,FVN
      READ 9020,WORD56,FEJ
      FEJ=1.0-FEJ/100.
      PRINT 9020,WORD56,FEJ
      READ 9020,WORD56,PFLAUN
      IF(PFLAUN.LE.0.0)  PRINT 9048
      PRINT 9026
C
C     CONVERSION OF UNITS
C
      EDFLR=EDFLR*48.8
      DWFLR=DWFLR*48.8
      DWFL2=DWFL2*48.8
C
C     CALCULATE PRIMARY COOLANT CONCENTRATIONS
C
      AFPTES=0.0
      DO 140 I=1,ITOT
  140 PCONC(I)=PWCONC(I)
      POWA=POW1
      PCVOA=PCVOL*1E3
      LETDWA=LETDWN*500.53
      SBLDA=SBLDR*.3476
      CBFLA=CBFLR*500.53
C
C     CHECK TO SEE IF PRIMARY PLANT PARAMETERS ARE WITHIN SPECIFIED
C     RANGES
C
      IF(POWA.LT.3000..OR.POWA.GT.3800.) GO TO 150
      IF(PCVOA.LT.5.0E5.OR.PCVOA.GT.6.0E5) GO TO 150
      IF(LETDWA.LT.3.2E4.OR.LETDWA.GT.4.2E4) GO TO 150
      IF(SBLDA.LT.250..OR.SBLDA.GT.1000.) GO TO 150
      IF(CBFLA.GT.7500.) GO TO 150
      GO TO 190
C
C     CALCULATE PRIMARY COOLANT ADJUSTMENT FACTORS
C
  150 AFPTES=1.0
      RHAL2=(LETDWA*0.99+0.01*SBLDA)/PCVOA
      RCSRB2=(LETDWA*0.5+0.5*(SBLDA+CBFLA*0.9))/PCVOA
      RCFP2=(LETDWA*0.98+0.02*(SBLDA+CBFLA*0.9))/PCVOA
      RK2=161.76*POWA/PCVOA
      DO 180 J=1,ITOT
      IF(PCONC(J).EQ.0.0) GO TO 180
      NZ=NUCL(J)/10000
      DL=DIS(J)*3600.
      IF(NZ.EQ.53.OR.NZ.EQ.35) GO TO 160
      IF(NZ.EQ.37.OR.NZ.EQ.55) GO TO 170
```

```
      PCONC(J)=PCONC(J)*RK2*(0.066+DL)/(RCFP2+DL)
      GO TO 180
 160  PCONC(J)=PCONC(J)*RK2*(0.067+DL)/(RHAL2+DL)
      GO TO 180
 170  PCONC(J)=PCONC(J)*RK2*(0.037+DL)/(RCSRB2+DL)
 180  CONTINUE
 190  SBLDR=SBLDR*48.8
      PCVOL=PCVOL*1000.*0.7/62.8
C
C     CALCULATE SECONDARY COOLANT CONCENTRATIONS
C
      SCVOA=SCVOL*1E3
      BLWDWA=BLWDWN*1E3
      STMFA=STMFR*1E6
      FFCDA=FFCDM
C
C     CHECK TO SEE IF SECONDARY PLANT PARAMETERS ARE WITHIN SPECIFIED
C     RANGES
C
      IF(BLWDWN.EQ.0.0) GO TO 230
C
C     PWTYPE=1.0 IS FOR PWRS WITH U-TUBE STEAM GENERATORS
C
      PWTYPE=1.0
      DO 200 I=1,ITOT
 200  SCON(I)=SCUTV(I)
      IF(AFPTES.EQ.1.0) GO TO 250
      IF(SCVOA.LT.4.0E5.OR.SCVOA.GT.5.0E5) GO TO 250
      IF(STMFA.LT.1.3E7.OR.STMFA.GT.1.7E7) GO TO 250
      IF(BLWDWA.LT.5.0E4.OR.BLWDWA.GT.1.0E5) GO TO 250
      IF(FFCDA.GT.0.01) GO TO 250
      IF(FNRTSC.LT.0.8999) GO TO 370
      GO TO 390
C
C     PWTYPE=2.0 IS FOR PWRS WITH ONCE-THROUGH STEAM GENERATORS
C
 230  PWTYPE=2.0
      DO 240 I=1,ITOT
 240  SCON(I)=SCOT(I)
      IF(AFPTES.EQ.1.0) GO TO 250
      IF(STMFA.LT.1.3E7.OR.STMFA.GT.1.7E7) GO TO 250
      IF(FFCDA.LT.0.55.OR.FFCDA.GT.0.75) GO TO 250
      GO TO 390
C
C     CALCULATE SECONDARY COOLANT ADJUSTMENT FACTORS
C
 250  IF(FFCDA.GT.0.01.AND.FFCDA.LT.1.0) FFCDA=0.2
      RHAL3=(BLWDWA*FNRTSI+0.9*HEF*STMFA*FFCDA)/SCVOA
      IF(FFCDA.GT.0.01.AND.FFCDA.LT.1.0) FFCDA=0.1
      RCSRB3=(BLWDWA*FNRTSC+0.5*FPEF*STMFA*FFCDA)/SCVOA
      RCFP3=(BLWDWA*FNRTSO+0.9*FPEF*STMFA*FFCDA)/SCVOA
      IF(PWTYPE.EQ.2.0) GO TO 330
      RK3=4.5E5/SCVOA
      DO 320 I=1,ITOT
      IF(SCON(I).EQ.0.0) GO TO 320
      NZ=NUCL(I)/10000
      DL=DIS(I)*3600.
      IF(NZ.EQ.53.OR.NZ.EQ.35) GO TO 300
      IF(NZ.EQ.37.OR.NZ.EQ.55) GO TO 310
      SCON(I)=SCON(I)*RK3*(0.17+DL)/(RCFP3+DL)*(PCONC(I)/PWCONC(I))
      GO TO 320
 300  SCON(I)=SCON(I)*RK3*(0.17+DL)/(RHAL3+DL)*(PCONC(I)/PWCONC(I))
```

```
      GO TO 320
 310  SCON(I)=SCON(I)*RK3*(0.15+DL)/(RCSRB3+DL)*(PCONC(I)/PWCONC(I))
 320  CONTINUE
      GO TO 390
 330  RK3=1.0E5/SCVOA
      DO 360 I=1,ITOT
      IF(SCON(I).EQ.0.0) GO TO 360
      NZ=NUCL(I)/10000
      DL=DIS(I)*3600.
      IF(NZ.EQ.53.OR.NZ.EQ.35) GO TO 340
      IF(NZ.EQ.55.OR.NZ.EQ.37) GO TO 350
      SCON(I)=SCON(I)*RK3*(14.0+DL)/(RCFP3+DL)*(PCONC(I)/PWCONC(I))
      GO TO 360
 340  SCON(I)=SCON(I)*RK3*(27.0+DL)/(RHAL3+DL)*(PCONC(I)/PWCONC(I))
      GO TO 360
 350  SCON(I)=SCON(I)*RK3*(7.5+DL)/(RCSRB3+DL)*(PCONC(I)/PWCONC(I))
 360  CONTINUE
      GO TO 390
 370  RCSRB3=(BLWDWA*FNRTSC+0.5*FPEF*STMFA*FFCDA)/SCVOA
      RK3=4.5E5/SCVOA
      DO 380 I=1,ITOT
      IF(SCON(I).EQ.0.0) GO TO 380
      NZ=NUCL(I)/10000
      IF(NZ.NE.37.AND.NZ.NE.55) GO TO 380
      DL=DIS(I)*3600.
      RSC=0.15
      SCON(I)=SCON(I)*RK3*(RSC+DL)/(RCSRB3+DL)*(PCONC(I)/PWCONC(I))
 380  CONTINUE
 390  BLWDWN=BLWDWN*1E3/500.53
      SCVOL=SCVOL*1000./62.4
      STMFR=STMFR*2000.
      DO 400 I=1,ITOT
      IF(PCONC(I).EQ.0.0) GO TO 400
      PCONC(I)=PCONC(I)/(DIS(I)*1.6283E13)
      SCON(I)=SCON(I)/(DIS(I)*1.6283E13)
 400  CONTINUE
C
C     COMPUTE REMOVAL CONSTANT FOR CONDENSATE DEMINERALIZER
C
      IF(FFCDM.GT.0.01.AND.FFCDM.LT.1.0) FFCDM=0.1
      CIXRC=(0.9*BLWDWN*RFNRT/DFCM+0.9*STMFR*FPEF*FFCDM)/(SCVOL*7.48*60.
     1)
      CIXRCS=(0.5*BLWDWN*RFNRT/DFCSCM+0.5*STMFR*FPEF*FFCDM)/(SCVOL*7.48*
     1 60.)
      IF(FFCDM.GT.0.01.AND.FFCDM.LT.1.0) FFCDM=0.2
      CIXRIB=(0.9*BLWDWN*RFNRT/DFICM+0.9*STMFR*HEF*FFCDM)/(SCVOL*7.48*60
     1.)
      DO 410 I=1,ITOT
      NZ=NUCL(I)/10000
      PR=CIXRC
      IF(NZ.EQ.37.OR.NZ.EQ.55) PR=CIXRCS
      IF(NZ.EQ.53.OR.NZ.EQ.35) PR=CIXRIB
      XZHJ=SCON(I)*PR*SCVOL*0.02832
      B(I)=XZHJ
 410  CONTINUE
C
C     CALCULATE RADIOISOTOPE INVENTORIES ON CONDENSATE RESINS
C
      CALL SOLVE
      DO 420 I=1,ITOT
 420  RINV(I)=XTEMP(I)
      CALL EFFTAB
      STOP
```

```
C
C     FORMATS                          FORMATS                          FORMATS
C
 9010 FORMAT(32X,7A4,16X,A4)
 9011 FORMAT(16X,13A4,A3,F9.4)
 9012 FORMAT(16X,14A4,F8.4)
 9013 FORMAT(15X,4A4,A2,8X,F8.0,1X,A4,A2,F6.4)
 9014 FORMAT(20X,F8.0,2(5X,F8.0))
 9015 FORMAT(27X,F6.2,14X,F9.2,18X,F6.2)
 9016 FORMAT("0",30X,"FRACTION  FRACTION  COLLECTION  DECAY"/8X,"STREAM
     1     FLOW RATE   OF PCA  DISCHARGED   TIME       TIME",10X,"DECONTA
     2MINATION FACTORS"/20X,"(GAL/DAY)"23X,"(DAYS)     (DAYS)",7X,
     3"I",8X,"CS",6X,"OTHERS")
 9017 FORMAT(2X,4A4,A2,1PE9.2,1X,4(0PF8.4,2X),3(1PE9.2,1X))
 9020 FORMAT(16X,14A4,F8.4)
 9021 FORMAT(79X,I1)
 9022 FORMAT(16X,14A4,F8.4)
 9026 FORMAT(1H1)
 9027 FORMAT(16X,"PLANT CAPACITY FACTOR",T75,"0.8000")
 9029 FORMAT(16X,"MASS OF WATER IN STEAM GENERATORS (THOUSAND LBS)",T73,
     1F8.4)
 9030 FORMAT(16X,"FISSION PRODUCT CARRY-OVER FRACTION",T75,F6.4/16X,
     1"HALOGEN CARRY-OVER FRACTION",T75,F6.4)
 9034 FORMAT(2X,"BLOWDOWN",10X,1PE9.2,14X,0PF5.3,2X,2(F8.3,2X),
     13(1PE9.2,1X))
 9035 FORMAT(2X,"UNTREATED BLOWDOWN",1PE9.2,11X,"   1.000      0.000
     10.000    1.00E+00  1.00E+00  1.00E+00")
 9037 FORMAT(72X,F8.2)
 9038 FORMAT(2X,"REGENERANT  OLS   ",1PE9.2,14X,0PF5.3,2X,2(F8.3,2X),
     13(1PE9.2,1X))
 9039 FORMAT(3I6,E21.14)
 9040 FORMAT(I6,E21.14)
 9041 FORMAT(16X,"PRIMARY TO SECONDARY LEAK RATE (LBS/DAY)",T73,
     1" 75.0000")
 9045 FORMAT   (/,"0 LIQUID WASTE INPUTS")
 9046 FORMAT   (/,"0 GASEOUS WASTE INPUTS")
 9048 FORMAT("0",15X,"THERE IS NOT AN ON-SITE LAUNDRY")
 9051 FORMAT(16X,"BLOWDOWN RATE (THOUSAND LBS/HR)",25X,F8.4)
 9052 FORMAT(16X,"THERE IS CONTINUOUS STRIPPING OF FULL LETDOWN FLOW")
 9053 FORMAT (16X,"THERE IS NOT CONTINOUS STRIPPING OF FULL LETDWN FLOW"
     1)
 9055 FORMAT(36X,F8.4,35X,I1)
 9056 FORMAT(15X,4A4,A2,8X,F8.0)
 9064 FORMAT(16X,"STEAM LEAK TO TURBINE BLDG (LBS/HR)",19X,F10.4)
 9065 FORMAT(16X,4A4,6X,F3.0)
 9066 FORMAT(16X,4A4,4X,"PARTICULATE RELEASE FRACTION",6X,F10.4)
 9067 FORMAT(16X,5A4,10X,F3.0,6X,F3.0)
 9068 FORMAT(16X,5A4,"IODINE RELEASE FRACTION",11X,F10.4/36X,"PARTICULAT
     1E RELEASE FRACTION",6X,F10.4)
 9069 FORMAT(16X,5A4,10X,F3.0,6X,F3.0,14X,F8.2)
 9070 FORMAT(16X,5A4,"RATE(CFM)",25X,F10.4/16X,5A4,"IODINE RELEASE FRACT
     1ION",11X,F10.4/36X,"PARTICULATE RELEASE FRACTION",6X,F10.4)
 9071 FORMAT(16X,5A4,10X,F3.0,6X,F3.0,19X,F3.0)
 9072 FORMAT (16X,"FREQUENCY OF CNTMT BLDG HIGH VOL PURGE (TIMES/YR)",
     1T74,F7.4)
 9073 FORMAT (16X,"THERE IS NOT A CNTMT BLDG LOW VOL PURGE")
 9075 FORMAT(16X,"THERE IS CONTINUOUS LOW VOL PURGE OF VOL. CONTROL TK")
      END
*DECK EFFTAB
      SUBROUTINE EFFTAB
      DIMENSION ISOTP(3,100)
      DIMENSION REACTR(7),NAME(3),CWCONC(800),DWCONC(800),CMCONC(800)
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      DIMENSION TURBDR(800),DWCON2(800),EDCONC(800),IOTHER(100)
      COMMON/FLUXN/REGENT,DIS(800),ILITE,IACT,ITOT
      COMMON/OUT/NUCL(800)
      COMMON/COOL/REACTR,POW1,SBLDR,BLWDWN,FPEF,HEF,EDFLR,DFIED,DFCSED,
     1 DFED,DFIDW,DFCSDW,DFDW,EDA,DWA,CWA,DFCM,DFICM,DFCSCM,
     2 DFCW,DFICW,DFCSCW,BDTFR,EDFD,DWFD,CWFD,CMFD,TS,TE,TD,TC,TCM,
     3 TSTORC,TSTORD,TSTORB,DWFL2,DW2,DWF2,T2,TSTOR2,DFID2,DFCSD2,DFD2,
     4 PFLAUN,DWFLR
      COMMON/APCOOL/RGWFR,DFIRG,DFCSRG,DFRG,TRG,TSTORR,RGFD
      COMMON/BDTES/RFNRT
      COMMON/CONC/PCONC(800),SCON(800),RINV(800)
      COMMON/DET/LAUNDRY(25),WLAUND(25)
C
C     H3COPW IS THE PWR TRITIUM PRIMARY COOLANT CONCENTRATION IN
C     UCI/GM
C
      H3PRPW=0.4*POW1
      H3COPW=1.0
      DO 30 J=1,ITOT
      CWCONC(J)=0.0
      EDCONC(J)=0.0
      DWCONC(J)=0.0
      DWCON2(J)=0.0
      CMCONC(J)=0.0
      NZ=NUCL(J)/10000
      IF(NZ.EQ.36.OR.NZ.EQ.54) GO TO 30
      CWCONC(J)=PCONC(J)*CWA
      EDCONC(J)=PCONC(J)*EDA
      DWCONC(J)=PCONC(J)*DWA
      DWCON2(J)=PCONC(J)*DW2
      CMCONC(J)=SCON(J)
      DFCVCS=50.
      IF(NZ.EQ.1)DFCVCS=1.0
      IF(NZ.EQ.35.OR.NZ.EQ.53) DFCVCS=100.
      IF(NZ.EQ.37.OR.NZ.EQ.55) DFCVCS=2.
      CWCONC(J)=CWCONC(J)/DFCVCS
   30 CONTINUE
C
C     CALCULATE RADIOACTIVITY AFTER COLLECTION AT A CONSTANT RATE
C
      CALL COLLECT(TC*86400.,CWCONC,ITOT)
      CALL COLLECT(TE*86400.,EDCONC,ITOT)
      CALL COLLECT(TD*86400.,DWCONC,ITOT)
      CALL COLLECT(T2*86400.,DWCON2,ITOT)
      CALL COLLECT(TCM*86400.,CMCONC,ITOT)
      IF(REGENT.LE.0.0) GO TO 50
      CALL STORAG(TRG*86400.,RINV,ITOT)
   50 DO 100 I=1,ITOT
      NZ=NUCL(I)/10000
      TURBDR(I)=1991.*5.*SCON(I)
      IF(NZ.EQ.1) GO TO 100
      IF(NZ.EQ.35.OR.NZ.EQ.53) GO TO 60
      IF(NZ.EQ.37.OR.NZ.EQ.55) GO TO 70
C
C     CHEMICAL TREATMENT FOR OTHER CATIONS
C
      CWCONC(I)=CWCONC(I)/DFCW
      EDCONC(I)=EDCONC(I)/DFED
      DWCONC(I)=DWCONC(I)/DFDW
      DWCON2(I)=DWCON2(I)/DFD2
      CMCONC(I)=CMCONC(I)*(1.0-BDTFR*(1.0-CMFD/DFCM))
C
C     TO TREAT PWR TURBINE BUILDING FLOOR DRAINS THROUGH DIRTY WASTE
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C     SYSTEM, DELETE C FOR COMMENT ON CARDS BELOW, UNTIL NEXT MESSAGE
C
      RINV  (I)=RINV  (I)/DFRG
      TURBDR(I)=1991.*5.*SCUN(I)*FPEF
C     TURBDR(I)=1991.*5.*SCUN(I)*FPEF/DFDW
      GO TO 100
C
C     CHEMICAL TREATMENT FOR ANIONS
C
 60   CWCONC(I)=CWCONC(I)/DFICW
      EDCONC(I)=EDCONC(I)/DFIED
      DWCONC(I)=DWCONC(I)/DFIDW
      DWCON2(I)=DWCON2(I)/DFID2
      CMCONC(I)=CMCONC(I)*(1.0-BDTFR*(1.0-CMFD/DFICM))
      RINV  (I)=RINV  (I)/DFIRG
      TURBDR(I)=1991.*5.*SCUN(I)*HEF
C     TURBDR(I)=1991.*5.*SCUN(I)*HEF/DFIDW
      GO TO 100
C
C     CHEMICAL TREATMENT FOR RB AND CS
C
 70   CWCONC(I)=CWCONC(I)/DFCSCW
      EDCONC(I)=EDCONC(I)/DFCSED
      DWCONC(I)=DWCONC(I)/DFCSDW
      DWCON2(I)=DWCON2(I)/DFCSD2
      CMCONC(I)=CMCONC(I)*(1.0-BDTFR*(1.0-CMFD/DFCSCM))
      RINV  (I)=RINV  (I)/DFCSRG
      TURBDR(I)=1991.*5.*SCUN(I)*FPEF
C     TURBDR(I)=1991.*5.*SCUN(I)*FPEF/DFCSDW
 100  CONTINUE
C
C     COMPUTE RADIOACTIVE DECAY DURING PROCESSING AND SAMPLING
C
      CALL STORAG(TSTORC*86400.,CWCONC,ITOT)
      CALL STORAG(TS*86400.,EDCONC,ITOT)
      CALL STORAG(TSTORD*86400.,DWCONC,ITOT)
      CALL STORAG(TSTOR2*86400.,DWCON2,ITOT)
      CALL STORAG(TSTORB*86400.,CMCONC,ITOT)
      CALL STORAG(TSTORR*86400.,RINV,ITOT)
      CALL STORAG(21600.,TURBDR,ITOT)
      DO   130 I=1,ITOT
      ABLOW=0.0
      IF(REGENT.LT.0.001) GO TO 110
      ABLOW=RINV(I)*292.4*RUFD/REGENT
 110  ABLOW=ABLOW+BLWDWN*1991.*CMCONC(I)*(1.0-RENRT)
 130  CMCONC(I)=ABLOW
      CWFR=SBLDR*CWFD*0.02832
      EDFLR=EDFLR*EDFD*0.02832
      DWFR=DWFLR*DWFD*0.02832
      DWFR2=DWFL2*DWF2*0.02832
      TPLRPW=CWFR*CWA+EDFLR*EDA+DWFR*DWA+DWFR2*DW2
      H3RLPW=TPLRPW*H3COPW
      IF(H3RLPW.GT.0.9*H3PRPW) H3RLPW=0.9*H3PRPW
      RH3RLP=H3RLPW/10.
      INTRIM=RH3RLP
      IH3RLP=INTRIM*10
      TOTAL=0.0
      I1=ILITE+IACT+1
      DO 140 I=1,ITOT
      NZ=NUCL(I)/10000
      IF(NZ.EQ.36.OR.NZ.EQ.54) GO TO 140
      DISI=DIS(I)*1.6283E13
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      CWCONC(I)=DISI*(CWCONC(I)*CWFR+EDCONC(I)*EDFLM)
      DWCONC(I)=(DWCONC(I)*DWFR+DWCON2(I)*DWFR2)*DISI
      CMCONC(I)=CMCONC(I)*DISI
      TURBDR(I)=TURBDR(I)*DISI
      IF(NUCL(I).EQ.10030) GO TO 140
      TOTAL =TOTAL +CWCONC(I)+DWCONC(I)+CMCONC(I)+TURBDR(I)
 140  CONTINUE
      AOI=0.16
      AOR=(AOI+TOTAL)/TOTAL
      SCNORM=0.0
      SAPRIM=0.0
      SSEC=0.0
      SCWAST=0.0
      SDWAST=0.0
      SABLOW=0.0
      STB=0.0
      STOTAL =0.0
      PAPRIM=0.0
      PSEC=0.0
      PCWAST=0.0
      PDWAST=0.0
      PABLOW=0.0
      PTB=0.0
      PTOTAL =0.0
      PNORM=0.0
      TLAUND=0.0
      CTOTAL=0.0
      PRINT 9001,    REACTR
      PRINT 9002
      PRINT 9010
      KOUNTR=1
      DO 180 I=1,ITOT
      IF(I.EQ.I1) PRINT 9011
      NZ=NUCL(I)/10000
      IF(NZ.EQ.36.OR.NZ.EQ.54) GO TO 180
      IF(NZ.EQ.1) GO TO 180
      DISI=DIS(I)*1.6283E+13
      APRIM=PCONC(I)*DISI
      ASEC=SCON(I)*DISI
      CWASTE=CWCONC(I)
      DWASTE=DWCONC(I)
      ABLOW=CMCONC(I)
      TB=TURBDR(I)
      TOTAL=CWASTE+DWASTE+ABLOW+TB
      TOTALN=TOTAL*AOR
      NUCLI=NUCL(I)
      XLAUND=0.0
      IF (I.GT.155.AND.I.LT.190) GO TO 152
      IF (I.EQ.225) GO TO 152
      DO 150 L=1,25
      IF(LAUNDRY(L).EQ.NUCLI) XLAUND=WLAUND(L)*PFLAUN
150   CONTINUE
 152  CONTINUE
155   TOTALG=TOTALN+XLAUND
      IF(TOTALG.LT.0.00001) GO TO 160
      ISUB=2
      IF (TOTALG.GT.1.)ISUB=1
      DIV=10.**(INT(ALOG10(TOTALG))-ISUB)
      TOTALG=AINT(TOTALG/DIV+0.5)*DIV
160   IF(NUCL(I).EQ.10030) TOTALN=TOTAL
      IF(NZ.EQ.1) GO TO 162
      SAPRIM=SAPRIM+APRIM
      SSEC=SSEC+ASEC
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```
      SABLOW=SABLOW+ABLOW
      SCWAST=SCWAST+CWASTE
      SDWAST=SDWAST+DWASTE
      STB=STB+TB
      STOTAL=STOTAL+TOTAL
      SCNORM=SCNORM+TOTALN
      TLAUND=TLAUND+XLAUND
      CTOTAL=CTOTAL+TOTALG
  162 IF(TOTALG.LT.0.00001) GO TO 180
  168 IF(MOD(KOUNTR,50).NE.0) GO TO 170
      PRINT 9000,    REACTR
      PRINT 9002
  170 CALL NOAH(NUCL(I),NAME)
      THALF=8.0225E-6/DIS(I)
      PRINT 9003,NAME,THALF,APRIM,ASEC,CWASTE,DWASTE,ABLOW,
     1TB,TOTAL,TOTALN,XLAUND,TOTALG
      KOUNTR=KOUNTR+1
      IF(NZ.EQ.1) GO TO 180
      PAPRIM=PAPRIM+APRIM
      PSEC=PSEC+ASEC
      PCWAST=PCWAST+CWASTE
      PDWAST=PDWAST+DWASTE
      PABLOW=PABLOW+ABLOW
      PTB=PTB+TB
      PTOTAL=PTOTAL+TOTAL
      PNORM=PNORM+TOTALN
  180 CONTINUE
      PAPRIM=SAPRIM-PAPRIM
      PSEC=SSEC-PSEC
      PCWAST=SCWAST-PCWAST
      PDWAST=SDWAST-PDWAST
      PABLOW=SABLOW-PABLOW
      PTB=STB-PTB
      PTOTAL=STOTAL-PTOTAL
      PNORM=SCNORM-PNORM
      ISUBC=2
      IF (CTOTAL.GT.1.)ISUBC=1
      DIV=10.**(INT(ALOG10(CTOTAL))-ISUBC)
      CTOTAL=AINT(CTOTAL/DIV+0.5)*DIV
      IF(PNORM.LT.0.00001) GO TO 190
        DIV=10.**(INT(ALOG10(PNORM))-2)
      PNORMT=AINT(PNORM/DIV+0.5)*DIV
      GO TO 200
  190 PNORMT=PNORM
  200 PRINT 9004,    PAPRIM,PSEC,PCWAST,PDWAST,PABLOW,PTB,PTOTAL,PNORM,
     1               PNORMT
      PRINT 9005,    SAPRIM,SSEC,SCWAST,SDWAST,SABLOW,STB,STOTAL,SCNORM,
     1               TLAUND,CTOTAL
      PRINT 9012,    IH3RLP
      PRINT 9013
      RETURN
 9000 FORMAT(1H1,20X,7A4," LIQUID EFFLUENTS (CONTINUED)")
 9001 FORMAT(1H1,20X,7A4," LIQUID EFFLUENTS")
 9002 FORMAT(1H0,55X,"ANNUAL RELEASES TO DISCHARGE CANAL"/20X,"COOLANT C
     1ONCENTRATIONS",57("-"),"  ADJUSTED   DETERGENT    TOTAL "/" NUCLID
     2E HALF-LIFE    PRIMARY    SECONDARY    BORON RS MISC. WASTES SECON
     3DARY  TURB BLDG  TOTAL LWS     TOTAL       WASTES          "/10X,
     4"(DAYS)    "2("(MICRO CI/ML)"),1X,4("(CURIES)    "),"(CURIES) ",
     5"    (CI/YR)     (CI/YR)      (CI/YR)")
 9003 FORMAT(1X,A2,I3,A1,2X,1PE9.2,2(2X,E9.2,2X),0P,7(1X,F9.5,1X),F10.5)
 9004 FORMAT(1X,"ALL OTHERS",9X,1PE9.2,4X,E9.2,0P,2X,6(1X,F9.5,1X),3X,
     1 "0.00000",1X,F10.5)
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9005 FORMAT(" TOTAL"/," (EXCEPT TRITIUM)    ",1PE9.2,4X,E9.2,0P,2X,
    1 7(1X,F9.5,1X),F10.5)
9010 FORMAT(" CORROSION AND ACTIVATION PRODUCTS")
9011 FORMAT("0FISSION PRODUCTS")
9012 FORMAT (1H0,1X,"TRITIUM RELEASE",12X,I3,"   CURIES PER YEAR")
9013 FORMAT(1H0,1X,"NOTE: .00000 INDICATES THAT THE VALUE IS LESS THAN
    11.0E-5.")
9014 FORMAT(3X,10(2X,A2,I3,A1)/3X,10(2X,A2,I3,A1))
     END
*DECK BLKDAT
     BLOCK DATA BLKDAT
C
C    PWCONC CONTAINS PRIMARY COOLANT CONCENTRATIONS FOR PWRS.  SCUTV
C    AND SCOT CONTAIN SECONDARY COOLANT CONCENTRATIONS FOR PLANTS
C    WITH U-TUBE STEAM GENERATORS AND FOR PLANTS WITH ONCE-THROUGH
C    STEAM GENERATORS, RESPECTIVELY.
C
     COMMON/CONP/PWCONC(800),SCUTV(800),SCOT(800)
     COMMON/DET/LAUNDRY(25),WLAUND(25)
     DATA PWCONC/36*0,4.7E-2,67*0,3.1E-3,4*0,1.6E-3,5*0,1.2E-3,3*0,3.0E
    1-4,0.0,4.6E-3,2*0,5.3E-4,17*0,5.1E-4,102*0,2.5E-3,64*0,2.2E-3,68*0
    2,1.6E-2,18*0,1.9E-1,4*0,1.4E-4,3*0,1.2E-5,5*0,9.6E-4,4.6E-4,5.2E-6
    3,9*0,4.2E-3,11*0,3.9E-4,0.0,2.8E-4,15*0,6.4E-3,4.7E-3,16*0,7.5E-3,
    415*0,9.0E-2,0.0,20*0,1.3E-3,104*0,1.9E-4,2.4E-2,12*0,1.5E-3,7.7E-3
    5,4.5E-2,5*0,1.7E-3,2.1E-1,4*0,1.4E-1,5*0,3.4E-1,2*0,7.1E-3,2*0,2.6
    6E-1,8*0,8.7E-4,3*0,9.4E-3,13*0,1.3E-2,2.5E-2,5*0,1.5E-4,12*0,2.8E-
    73,3*0,3.9E-3,92*0/
     DATA SCUTV/36*0,1.5E-9,67*0,1.3E-7,4*0,6.5E-8,5*0,4.9E-8,3*0,1.2E-
    18,0.0,1.9E-7,2*0,2.2E-8,17*0,2.1E-8,102*0,8.7E-8,64*0,8.4E-8,68*0,
    27.5E-8,18*0,5.3E-7,4*0,5.7E-9,3*0,4.9E-10,5*0,2.8E-8,3.2E-9,2.1E-1
    30,9*0,1.2E-7,11*0,1.6E-8,0.0,1.1E-8,15*0,2.5E-7,1.1E-7,16*0,3.1E-7
    4,15*0,3.7E-6,0.0,20*0,5.3E-8,104*0,7.8E-9,2.2E-7,12*0,5.4E-8,2.9E-
    58,1.8E-6,5*0,6.6E-8,3.1E-6,4*0,4.8E-6,5*0,2.4E-6,2*0,3.3E-7,2*0,6.
    66E-6,8*0,4.0E-8,3*0,4.4E-7,13*0,5.2E-7,9.3E-7,5*0,6.1E-9,12*0,1.0E
    7-7,3*0,1.6E-7,92*0/
     DATA SCOT/36*0,1.0E-7,67*0,6.9E-9,4*0,3.6E-9,5*0,2.7E-9,3*0,6.7E-1
    10,0.0,1.0E-8,2*0,1.2E-9,17*0,1.1E-9,102*0,5.6E-9,64*0,4.9E-9,68*0,
    21.8E-8,18* ,6.0E-7,4*0,3.1E-10,3*0,2.7E-11,5*0,2.1E-9,9.7E-10,1.2E
    3-11,9*0,9.3E-9,11*0,8.7E-10,0.0,6.2E-10,15*0,1.4E-8,1.0E-8,16*0,1.
    47E-8,15*0,2.0E-7,0.0,20*0,2.9E-9,104*0,4.2E-10,5.1E-8,12*0,3.3E-9,
    51.5E-8,5.2E-8,5*0,3.8E-9,2.4E-7,4*0,1.6E-7,5*0,3.8E-7,2*0,3.0E-8,2
    6*0,3.0E-7,8*0,3.6E-9,3*0,3.9E-8,13*0,2.9E-8,5.6E-8,5*0,3.3E-10,12*
    70,6.2E-9,3*0,8.7E-9,92*0/
C
C    LAUNDRY ARE THE RADIOISOTOPES IN THE DETERGENT WASTES.
C    WLAUND ARE THE CORRESPONDING CONCENTRATIONS.
C
     DATA LAUNDRY/150320,240510,250540,260550,260590,270580,270600,2806
    130,380890,380900,390910,400950,410950,420990,441030,441060,471101,
    2511240,531310,551340,551360,551370,561400,581410,581440/
     DATA WLAUND/1.8E-4,4.7E-3,3.8E-3,7.2E-3,2.2E-3,7.9E-3,1.4E-2,1.7E-
    13,8.8E-5,1.3E-5,8.4E-5,1.1E-3,1.9E-3,6.0E-5,2.9E-4,8.9E-3,1.2E-3,4
    2.3E-4,1.6E-3,1.1E-2,3.7E-4,1.6E-2,9.1E-4,2.3E-4,3.9E-3/
     END
*DECK SOLVE
     SUBROUTINE SOLVE
     COMMON/EQ/XTEMP(800),XNEW(10,800),B(800),D(800)
     COMMON/FLUXN/REGENT,DIS(800),ILITE,IACT,ITOT
     DO 10 I=1,ITOT
     D(I)=-DIS(I)
  10 XTEMP(I)=0.0
     DELT=REGENT*86400.
     CALL DECAY(1,DELT,ITOT)
```

```
      CALL TERM(DELT,1,ITOT)
      CALL EQUIL(1,ITOT)
      DO 30 I=1,ITOT
 30   XTEMP(I)=XNEW(1,I)
      RETURN
      END
*DECK TERM
      SUBROUTINE TERM(T,M,ITOT)
C
C     TERM ADDS ONE TERM TO EACH ELEMENT OF THE SOLUTION VECTOR
C     CSUM(J) IS THE CURRENT APPROXIMATION TO XNEW(M,J)
C     CIMO(J) IS THE VECTOR CONTAINING THE LAST TERM ADDED TO EACH
C     ELEMENT OF CSUM(J)
C     CIMN(J) IS THE VECTOR CONTAINING 1/TON TIMES THE NEW TERM TO BE
C     ADDED TO  CSUM(J)
C     CIMN(J) IS GENERATED FROM CIMO(J) BY A RECURSION RELATION:
C           CIMN(J)= SUM OVER L OF (AP(J,L)*CIMO(L))
C     AP(I,J) IS THE REDUCED TRANSITION MATRIX FOR THE LONG-LIVED
C     NUCLIDES
C
      LOGICALLONG
      DIMENSION AP(2500),CIMB(800),CIMO(800),CIMN(800),CSUM(800)
      DIMENSION QUB(50),LOCP(2500),NONP(800)
      COMMON/SERIES/ XP(800),XPAR(800),LONG(800)
      COMMON/CONST/MMN,ERR,MZERO
      COMMON/EQ/XTEMP(800),XNEW(10,800),B(800),D(800)
      COMMON/MATRIX/A(2500),LOC(2500),NONO(800),KD(800)
      LEVEL 2,A,LOC,NONO,KD
      COMMON/TERMD/DD(100),DXP(100),QUEUE(50),NUU(50),NQUEUE(50),NQ(800)
      NUL=0
      NN=0
C     FIRST CONSTRUCT REDUCED TRANSITION MATRIX FOR LONG-LIVED ISOTOPES
      DO 220 L=1,ITOT
      IF(.NOT.LONG(L)) GO TO 210
      NUM=NONO(L)
      IF(M.GT.MMN.OR.M.EQ.MZERO)  NUM=KD(L)
      CIMB(L)=B(L)
      IF(NUM.LE.NUL) GO TO 210
      NS=NN+1
      N=NUL
      NL=NUM-NUL
      DO 200 N1=1,NL
      N=N+1
      J=LOC(N)
      DJ=-D(J)
C
C      THIS IS A TEST TO SEE IF ONE OF THE ASSYMPTOTIC SOLUTIONS APPLIES
C
      IF(.NOT.LONG(J)) GO TO 10
      NN=NN+1
      AP(NN)=A(N)
      LOCP(NN)=J
      GO TO 200
C
C     GOING BACK UP THE CHAIN TO FIND A PARENT WHICH IS NOT IN
C     EQUILIBRIUM
C
 10   NSAVE=0
      QUE=A(N)/DJ
      DRB=1.0
      CIMB(L)=CIMB(L)+QUE*B(J)
      NQ(L)=0
```

```
      NQ(J)=L
 20   NUX=NONO(J)
      IF(M.GT.MMN.OR.M.EQ.MZERO)  NUX=KD(J)
      NUF=0
      IF(J.GT.1)  NUF=NONO(J-1)
      NX=NUX-NUF
      IF(NX.LT.1)  GO TO 190
      K=NUF
      DO 180 KK=1,NX
      K=K +1
      J1=LOC(K)
      DJ=-D(J1)
      KP=J
 30   IF(J1.EQ.NQ(KP))  GO TO 180
      KP=NQ(KP)
      IF(KP.NE.0)  GO TO 30
      AKDJQ=QUE*A(K)/DJ
      IF(.NOT.LONG(J1)) GO TO 160
      TRM=1.0-XP(J1)
      IF(TRM.LT.1.0E-6) GO TO 120
      NQ(J1)=J
      I=1
      KP=J1
 40   DD(I)=-D(KP)
      DXP(I)=XP(KP)
      KP=NQ(KP)
      IF(KP.EQ.0) GO TO 50
      I=I+1
      IF(I.LE.100) GO TO 40
C     IF QUEUE OF SHORT-LIVED NUCLIDES EXCEEDS 100 ISOTOPES, TERMINATE
C     CHAIN AND WRITE MESSAGE
      PRINT 9000,    M,L,J1,J,AKDJQ
 9000 FORMAT("1TOO LONG A QUEUE HAS BEEN FORMED IN TERM",4I5,E12.5)
      GO TO 190
 50   BATM=0.E0
      IM=I-1
      DO 110 I=2,IM
      DL=DD(I)
      XPL=DXP(I)
      BATE=0.E0
      I1=I-1
C     D R VONDY FORM OF BATEMAN EQUATIONS -- ORNL-TM-361
      DO 100 KB=1,I1
      XPJ=DXP(KB)
      IF(XPL+XPJ.LT.ERR) GO TO 100
      DK=DD(KB)
      PROD=(DL/DK-1.0)
      DKR=PROD
      IF( ABS(PROD).GT.1.E-4) GO TO 60
C     USE THIS FORM FOR TWO NEARLY EQUAL HALF-LIVES
      PROD=T*DK*XPJ*(1.0-0.5*(DL-DJ)*T)
      GO TO 70
 60   PROD=(XPJ-XPL)/PROD
      PRO1=XPJ/DKR
 70   PI=1.0
      S1=2./(DK*T)
      DO 90 JK=1,I1
      IF(JK.EQ.KB) GO TO 90
      S=1.0-DK/DD(JK)
      IF( ABS(S).GT.1.E-4) GO TO 80
      IF(ABS(DKR).GT.1.0E-4)  PROD=PRO1
      S=S1
 80   PI=PI*S
```

```
      IF(ABS(PI).GT.1.E25)  GO TO 100
   90 CONTINUE
      BATE=BATE+PROD/PI
  100 CONTINUE
C     IF SUMMATION IS NEGATIVE, SET EQUAL TO ZERO AND PRINT MESSAGE
      IF(BATE.LT.0.E0)PRINT9001,L,IM,BATE,BATM
 9001 FORMAT("1BATE IS NEGATIVE IN TERM. THERE ARE MORE THAN TWO SHORT-L
     1IVED NUCLIDES IN A CHAIN WITH NEARLY EQUAL DIAGONAL ELEMENTS"/
     2" L,IM,BATE,BATM = ",2I5,1P2E12.5)
      IF(BATE.LT.0.E0)BATE=0.E0
      BATM=BATM+BATE
  110 CONTINUE
      DRA=AKDJQ*DJ*(TRM-BATM)/TRM
      GO TO 130
  120 DRA=AKDJQ*AMAX1(DRB,0,0)*DJ
  130 IF(NS.GT.NN) GO TO 150
      DO 140 LJ=NS,NN
      IF(LOCP(LJ).NE.J1)  GO TO 140
      AP(LJ)=AP(LJ)+DRA
      GO TO 180
  140 CONTINUE
  150 NN=NN+1
      AP(NN)=DRA
      LOCP(NN)=J1
      GO TO 180
  160 IF(AKDJQ.LE.1.0E-06)  GO TO 180
      IF(NSAVE.GE.50) GO TO 180
  170 NSAVE=NSAVE+1
      NQUEUE(NSAVE)=J1
      QUEUE(NSAVE)=AKDJQ
      NQU(NSAVE)=J
      QUB(NSAVE)=DRB-1./(DJ*T)
  180 CONTINUE
  190 IF(NSAVE.LE.0) GO TO 200
      J=NQUEUE(NSAVE)
      QUE=QUEUE(NSAVE)
      NQ(J)=NQU(NSAVE)
      DRB=QUB(NSAVE)
      CIMB(L)=CIMB(L)+QUE*B(J)*AMAX1(DRB,0.0)
      NSAVE=NSAVE-1
      GO TO 20
  200 CONTINUE
  210 NUL=NONO(L)
      NONP(L)=NN
  220 CONTINUE
C     FIND NORM OF MATRIX AND ESTIMATE ERROR AS DESCRIBED IN LAPIDUS
C     AND LUUS, OPTIMAL CONTROL OF ENGINEERING PROCESSES BLAISDELL 1967
C     FIND THE MINIMUM OF THE MAXIMUM ROW SUM AND THE MAXIMUM COLUMN SUM
      ASUM =0.0
      ASUMJ=0.0
      NUL=1
      DO 250 I=1,ITOT
      IF(.NOT.LONG(I)) GO TO 250
      DI=-D(I)*T
      AJ=DI
      NUM=NONP(I)
      IF(NUL.GT.NUM) GO TO 240
      DO 230 N=NUL,NUM
  230 AJ=AJ+AP(N)
  240 AI=DI+DI
      IF(AI.GT.ASUM )  ASUM =AI
      IF(AJ.GT.ASUMJ)  ASUMJ=AJ
```

```
 250  NUL=NONP(I)+1
      IF(ASUMJ.LT.ASUM)  ASUM=ASUMJ
C     USE ASUM TO DECIDE HOW MANY TERMS ARE REQUIRED AND ESTIMATE ERROR
      NLARGE=3.5*ASUM +5.
      XLARGE=NLARGE
      ERR1=EXP(ASUM )*(ASUM *2.71828/XLARGE)**NLARGE/SQRT(6.2832*XLARGE)
      IF(ERR1.GT.1.E-3)  PRINT 9002,      ERR1,ASUM ,NLARGE
 9002 FORMAT("0MAXIMUM ERROR GT 0.001, ="F10.6,", TRACE = "F10.4,
     1  " NLARGE = "I6)
C     NEXT GENERATE MATRIX EXPONENTIAL SOLUTION
      DO 260 I=1,ITOT
      CSUM(I)=XTEMP(I)
      CIMN(I)=XTEMP(I)
 260  CONTINUE
      ERR3=0.001*ERR
      DO 310 NT=1,NLARGE
      DO 270 I=1,ITOT
      CIMO(I)=CIMN(I)
 270  CONTINUE
      TON=T/NT
      NUL=1
      DO 300 I=1,ITOT
      IF(.NOT.LONG(I)) GO TO 300
      NUM=NONP(I)
      CIMNI=0.0
      IF(NT.EQ.1) CIMNI=CIMO(I)
      IF(NUL.GT.NUM) GO TO 290
      DO 280 N=NUL,NUM
      J=LOCP(N)
 280  CIMNI=CIMNI+AP(N)*CIMO(J)
 290  CIMNI=CIMNI+D(I)*CIMO(I)
      CIMNI=TON*CIMNI
      IF(ABS(CIMNI).LT.ERR3)CIMNI=0.E0
      CIMN(I)=CIMNI
      CSUM(I)=CSUM(I)+CIMNI
 300  NUL=NONP(I)+1
 310  CONTINUE
      DO 320 I=1,ITOT
      IF(CSUM(I).LT.ERR)  CSUM(I)=0.0
      IF(LONG(I)) XNEW(M,I)=CSUM(I)
 320  CONTINUE
      RETURN
      END
*DECK DECAY
      SUBROUTINE DECAY(M,T,ITOT)
C     DECAY TREATS SHORT-LIVED ISOTOPES AT BEGINNING OF CHAINS USING
C     BATEMAN EQUATIONS
      LOGICALLONG
      COMMON/SERIES/ XP(800),XPAR(800),LONG(800)
      COMMON/CONST/MMN,ERR,MZERO
      COMMON/EQ/XTEMP(800),XNEW(10,800),B(800),D(800)
      COMMON/MATRIX/A(2500),LOC(2500),NONO(800),KD(800)
      LEVEL 2,A,LOC,NONO,KD
      COMMON/TERMD/DD(100),DXP(100),QUEUE(50),NQU(50),NQUEUE(50),NQ(800)
      AXN=-ALOG(0.001)
      DO 10 I=1,ITOT
      XPAR(I)=0.0
      LONG(I)=.FALSE.
      XPI=0.0
      DT=D(I)*T
      IF(DT.LT.-50.)  GO TO 10
      IF(ABS(DT).LE.AXN)  LONG(I)=.TRUE.
      XPI=EXP(DT)
```

```
   10 XP(I)=XPI
      NUL=1
      DO 160 L=1,ITOT
      XTEM=0.0
      DL=-D(L)
      NUM=NONO(L)
      IF(M.GT.MMN.OR.M.EQ.MZERO)  NUM=KD(L)
      IF(NUM.LT.NUL) GO TO 150
      DO 140 N=NUL,NUM
      J=LOC(N)
      DJ=-D(J)
      IF(LONG(J)) GO TO 140
C     USE THIS FORM FOR TWO NEARLY EQUAL HALF-LIVES
      IF(ABS(DL/DJ-1.0).LE.1.0E-5)  XTEM=XTEM+XTEMP(J)*A(N)*XP(J)*T
      IF(ABS(DL/DJ-1.0).GT.1.0E-5)
     1            XTEM=XTEM+XTEMP(J)*A(N)*(XP(J)-XP(L))/(DL-DJ)
      QUE=A(N)/DJ
      NQ(L)=0
      NQ(J)=L
      NSAVE=0
   20 NUX=NONO(J)
      IF(M.GT.MMN.OR.M.EQ.MZERO)  NUX=KD(J)
      NUF=1
      IF(J.GT.1)  NUF=NONO(J-1)+1
      IF(NUF.GT.NUX)  GO TO 130
      DO 120 K=NUF,NUX
      J1=LOC(K)
      IF(LONG(J1)) GO TO 120
      KP=J
   30 IF(J1.EQ.NQ(KP))  GO TO 120
      KP=NQ(KP)
      IF(KP.NE.0)  GO TO 30
      DJ=-D(J1)
      AKDJQ=A(K)/DJ*QUE
      IF(AKDJQ.LE.1.0E-06)  GO TO 120
      NQ(J1)=J
      I=1
      KP=J1
   40 DD(I)=-D(KP)
      DXP(I)=XP(KP)
      KP=NQ(KP)
      IF(KP.EQ.0) GO TO 50
      I=I+1
      IF(I.LE.100) GO TO 40
      PRINT 9000,    M,L,J1,J,AKDJQ
 9000 FORMAT("1",4I5,E12.5)
      GO TO 130
   50 BATE=0.E0
      I1=I-1
      XPL=XP(L)
C     D R VONDY FORM OF BATEMAN EQUATIONS -- ORNL-TM-361
      DO 100 KB=1,I1
      XPJ=DXP(KB)
      IF(XPL+XPJ.LT.ERR) GO TO 100
      DK=DD(KB)
      PROD=(DL/DK-1.0)
      DKR=PROD
      IF( ABS(PROD).GT.1.E-4) GO TO 60
      PROD=T*DK*XPJ*(1.0-0.5*(DL-DJ)*T)
      GO TO 70
   60 PROD=(XPJ-XPL)/PROD
      PRO1=XPJ/DKR
```

```
 70    PI=1.0
       S1=2./(DK*T)
       DO 90 JK=1,I1
       IF(JK.EQ.KB) GO TO 90
       S=1.0-DK/DD(JK)
       IF( ABS(S).GT.1.E-4) GO TO 80
C      USE THIS FORM FOR TWO NEARLY EQUAL HALF-LIVES
       IF(ABS(DKR).GT.1.0E-4)  PROD=PRO1
       S=S1
 80    PI=PI*S
       IF(ABS(PI).GT.1.E25)  GO TO 100
 90    CONTINUE
       BATE=BATE+PROD/PI
 100   CONTINUE
       IF(BATE.LT.0.E0)PRINT9001,L,I,BATE,XTEM,XTEMP(J1),AKDJQ
 9001  FORMAT(" L,I,BATE,XTEM,XTEMP(J1),AKDJQ = ",2I5,1P4E12.5)
       IF(BATE.LT.0.E0)BATE=0.E0
       XTEM=XTEM+XTEMP(J1)*AKDJQ*BATE
       IF(NSAVE.GE.50) GO TO 120
 110   NSAVE=NSAVE+1
       NQUEUE(NSAVE)=J1
       QUEUE(NSAVE)=AKDJQ
       NQU(NSAVE)=J
 120   CONTINUE
 130   IF(NSAVE.LE.0) GO TO 140
       J=NQUEUE(NSAVE)
       QUE=QUEUE(NSAVE)
       NQ(J)=NQU(NSAVE)
       NSAVE=NSAVE-1
       GO TO 20
 140   CONTINUE
       IF(LONG(L)) XPAR(L)=XTEM/XP(L)
 150   NUL=NONO(L)+1
       IF(.NOT.LONG(L)) XNEW(M,L)=XTEM+XTEMP(L)*XP(L)
 160   CONTINUE
       DO 170 I=1,ITOT
       IF(LONG(I)) XTEMP(I)=XTEMP(I)+XPAR(I)
       IF(.NOT.LONG(I)) XTEMP(I)=0.0
 170   CONTINUE
       RETURN
       END
*DECK EQUIL
       SUBROUTINE EQUIL(M,ITOT)
C
C      EQUIL PUTS SHORT-LIVED DAUGHTERS IN EQUILIBRIUM WITH PARENTS
C       EQUIL USES GAUSS-SEIDEL ITERATION TO GENERATE STEADY STATE
C      CONCENTRATIONS
C
       LOGICALLONG
       COMMON/EQ/XTEMP(800),XNEW(10,800),B(800),D(800)
       COMMON/MATRIX/A(2500),LOC(2500),NONO(800),KD(800)
       LEVEL 2,A,LOC,NONO,KD
       COMMON/CONST/MMN,ERR,MZERO
       COMMON/SERIES/ XP(800),XPAR(800),LONG(800)
       QXN=0.001
       DO 10 I=1,ITOT
       XPAR(I)=0.0
       IF(.NOT.LONG(I)) GO TO 10
       XTEMP(I)=XTEMP(I)*XP(I)
       XPAR(I)=AMAX1(XNEW(M,I)-XTEMP(I),0.0)
 10    CONTINUE
       ITER=1
 20    N=0
```

```
      BIG=0.0
      DO 60 I=1,ITOT
      NUM=NONO(I)-N
      DI=-D(I)
      IF(LONG(I)) GO TO 50
      XNW=B(I)
      IF(M.GT.MMN.OR.M.EQ.MZERO)  NUM=KD(I)-N
      IF(NUM.EQ.0)  GO TO 31
      DO 30 K=1,NUM
      N=N+1
      J=LOC(N)
      DJ=-D(J)
      XJ=XPAR(J)
      IF(LONG(J)) XJ=XJ+XTEMP(J)/(1.0-DJ/DI)
      XNW=XNW+A(N)*XJ
   30 CONTINUE
   31 XNW=XNW/DI
      IF(XNW.LT.1.0E-50)  GO TO 40
      ARG=ABS((XNW-XPAR(I))/XNW)
      IF(ARG.GT.BIG)  BIG=ARG
   40 XPAR(I)=XNW
   50 N=NONO(I)
   60 CONTINUE
      IF(BIG.LT.QXN  )   GOTO 70
      ITER=ITER+1
      IF(ITER.LT.100)  GO TO 20
      PRINT 9000
      STOP
   70 DO 80 I=1,ITOT
      IF(.NOT.LONG(I)) XNEW(M,I)=XNEW(M,I)+XPAR(I)
   80 CONTINUE
      RETURN
 9000 FORMAT(" GAUSS SEIDEL ITERATION DID NOT CONVERGE IN EQUIL")
      END
*DECK NUDATA
      SUBROUTINE NUDATA(NLIBE)
C     NUDATA VERSION TO HANDLE THREE TYPES OF NUCLEAR DATA LIBRARIES
C     HAS POINTER, NLIBE, = 1 FOR HTGR
C                         = 2 FOR LIGHT WATER REACTOR
C                         = 3 FOR LMFBR
C                         = 4 FOR MSBR
      INTEGERELE(99),STA(2)
      DIMENSION COEFF(7,800),NPROD(7,800),CAPT(6),YIELD(5,500)
      DIMENSION Y(5),NSORS(9),TYLD(5),NUCAL(6)
      DIMENSION SKIP(20),MSRS(20),NAME(3)
      DIMENSION TOCAP(800),FISS(100),TITLE(20),Q(800),FG(800),
     1ALPHAN(100),SPONF(100),ABUND(500),KAP(800),MMAX(800)
      COMMON/LABEL/ELE,STA
      COMMON/CONST/MMN,ERR,MZERO
      COMMON/EQ/XTEMP(800),XNEW(10,800),B(800),D(800)
      COMMON/FLUXN/REGENT,DIS(800),ILITE,IACT,ITOT
      COMMON/OUT/NUCL(800)
      COMMON/MATRIX/A(2500),LOC(2500),NONO(800),KD(800)
      LEVEL 2,A,LOC,NONO,KD
      COMMON/CCOEFF/COEFF
      LEVEL 2,COEFF
      EQUIVALENCE (XNEW(1,401),NPROD(1,1))
      EQUIVALENCE (AI,DLAM)
      DATA NUCAL/-20030,-10000,10,11,-10,-9/
      DATA MSRS/922330,922350,902320,922380,942390,922330,922350,942410,
     1          922380,942390,942410,922350,942400,922380,942390,922330,
     2          922350,902320,922380,942390/
```

```
C
C     PROGRAM TO COMPUTE A MATRIX (TRANSITION MATRIX) FROM NUCLEAR DATA
C
      READ 9011,              (TITLE(I),I=1,18),NLIBE
C     IF(NLIBE.LT.0) PROGRAM WILL READ TAPE IN CASDAR FORMAT
      IGWC=0
      IF(NLIBE.GT.0)  GO TO  10
      IGWC=1
      NLIBE=-NLIBE
      PRINT 9000
 9000 FORMAT(1H0,"WILL READ TAPE GENERATED BY CASDAR")
 10   N1=4-NLIBE
 20   READ 9001,     THERM,RES,FAST,ERR,NMO,NDAY,NYR,MPCTAB,INPT,IR
      PRINT 9005,      NMO,NDAY,NYR
      PRINT 9006
      PRINT 9007
      PRINT 9008
      PRINT 9009
      PRINT 9010
      PRINT 9013
      PRINT 9014
C
C     THERM = RATIO OF THERMAL FLUX TO TOTAL FLUX
C     RES =  RATIO OF RESONANCE FLUX TO TOTAL FLUX
C     FAST = RATIO OF FAST FLUX TO TOTAL FLUX
C     ERR = TRUNCATION ERROR LIMIT
C
C     READ DATA FOR LIGHT ELEMENTS
C
      K=5*(NLIBE-1)
      DO 30 K1=1,5
      K2=K+K1
 30   NSORS(K1)=MSRS(K2)
      PRINT 9018,    THERM,RES,FAST,(NSORS(K),K=1,5),NLIBE
      I=0
      NUTAPE=0
 40   I=I+1
 50   READ(8,9034)NUCL(I),DLAM,IU,FB1,FP,FP1,FT,FA,ESF,Q(I),FG(I),ABUND(
     1I),DUMY1,DUMY2
      IF(EOF(8).NE.0)GOTO260
      IF(IGWC.GT.0)  GO TO 70
      DO 60 N=1,NLIBE
 60   READ(8,9035) SIGTH,FNG1,FNA,FNP,RITH,FINA,FINP,SIGMEV,FN2N1,FFNA,
     1             FFNP,IT
      GO TO 90
 70   DO 80 N=1,NLIBE
 80   READ(8,9040) SIGTH,FNG1,FNA,FNP,RITH,FINA,FINP,SIGMEV,FN2N1,FFNA,
     1             FFNP,IT
 90   IF(N1.EQ.0) GO TO 110
      DO 100 N=1,N1
 100  READ(8,9036)  SKIP
 110  IF(IT.EQ.0) GO TO 50
 120  M=0
      CALL HALF(A1,IU)
      NUCLI=NUCL(I)
      IF(NUCLI.EQ.0)  GO TO 260
      CALL NOAH(NUCLI,NAME)
      IF(MOD(I-1,50) .EQ. 0) PRINT 9012,    (TITLE (N),N=1,18)
      IF(MOD(I-1,50) .EQ. 0) PRINT 9016
      SIGTH=THERM*SIGTH
      RITH=RES*RITH
      SIGMEV=FAST*SIGMEV
      SIGNA=SIGTH*FNA+RITH*FINA+SIGMEV*FFNA
```

```
      SIGNP=SIGTH*FNP+RITH*FINP+SIGMEV*FFNP
      FNG=1.0-FNA-FNP
      IF(FNG.LT.1.0E-4)FNG=0.
      FING=1.0-FINA-FINP
      IF(FING.LT.1.0E-4)FING=0.
      FN2N=1.0-FFNA-FFNP
      IF(FN2N.LT.1.0E-4)FN2N=0.
      SIGNG=SIGTH*FNG+RITH*FING
      SIGN2N=SIGMEV*FN2N
  130 PRINT 9033,     NAME,              DLAM,FB1,FP,FP1,FT,FA,SIGNG,
     1                FNG1,SIGN2N,FN2N1,SIGNA,SIGNP,Q(I),FG(I),ABUND(I)
C     TEST RADIOACTIVITY
C
  140 IF(A1.LE.ERR)  GO TO 180
      ABETA=1.0
C
C     TEST POSITRON EMISSION
C
      IF(FP .LT. ERR) GO TO 150
      M=M+1
      COEFF(M,I)=FP*A1
      NPROD(M,I)=NUCLI-10000
      ABETA=ABETA-FP
C
C     TEST POSITRON EMISSION TO EXCITED STATE OF PRODUCT NUCLIDE
C
      IF(FP1 .LT. ERR) GO TO 150
      M=M+1
      COEFF(M,I)=FP1*COEFF(M-1,I)
      NPROD(M,I)=NPROD(M-1,I)+1
      COEFF(M-1,I)=COEFF(M-1,I)-COEFF(M,I)
C
C     TEST ISOMERIC TRANSITION
C
  150 IF(FT .LT.ERR) GO TO 160
      M=M+1
      COEFF(M,I)=FT*A1
      NPROD(M,I)=NUCLI
      ABETA=ABETA-FT
C
C     TEST ALPHA EMISSION
C
  160 IF(FA .LT. ERR) GO TO 170
      M=M+1
      COEFF(M,I)=FA*A1
      NPROD(M,I)=NUCLI-20040
      M=M+1
      COEFF(M,I)=COEFF(M-1,I)
      NPROD(M,I)=20040
      ABETA=ABETA-FA
C
C     TEST NEGATRON EMISSION
C
  170 IF(ABETA.LT.1.E-4) GO TO 180
      M=M+1
      COEFF(M,I)=ABETA*A1
      NPROD(M,I)=NUCLI+10000
C
C     TEST NEGATRON EMISSION TO EXCITED STATE OF PRODUCT NUCLIDE
C
      IF(FB1 .LT. ERR)GO TO 180
      M=M+1
```

```
      COEFF(M,I)=FB1*COEFF(M-1,I)
      NPROD(M,I)=NPROD(M-1,I)+1
      COEFF(M-1,I)=COEFF(M-1,I)-COEFF(M,I)
C
C     COMPUTE NEUTRON CAPTURE CROSS SECTIONS IN THREE REGIONS
C
 180  KAP(I)=M
      DO 190 KI=1,6
 190  CAPT(KI) =0.0
      CAPT(1)=SIGNA
      CAPT(2)=SIGNP
      CAPT(4)=SIGNG*FNG1
      CAPT(3)=SIGNG-CAPT(4)
      CAPT(6)=SIGN2N*FN2N1
      CAPT(5)=SIGN2N-CAPT(6)
 200  TOCAP(I)=0.0
C     TOTAL NEUTRON CROSS SECTION FOR NUCLIDE(I)
      DO 220 K=1,6
      CAPKI=CAPT(K)
      IF(CAPKI.LT.ERR)  GO TO 220
      M=M+1
      NPROD(M,I)=NUCLI+NUCAL(K)
      COEFF(M,I)=CAPKI
      TOCAP(I)=TOCAP(I)+CAPKI
      IF(K.NE.1) GO TO 210
      M=M+1
      COEFF(M,I)=COEFF(M-1,I)
      NPROD(M,I)=20040
 210  IF(K.NE.2) GO TO 220
      M=M+1
      COEFF(M,I)=COEFF(M-1,I)
      NPROD(M,I)=10010
 220  CONTINUE
 230  IF(MOD(NUCLI,  10).EQ.0) GO TO 250
      DO 240 K=1,M
 240  NPROD(K,I)=NPROD(K,I)+1
 250  MMAX(I) =M
      IF(M.GT.7)  PRINT 9039, M
      DIS(I)=A1
      GO TO 40
 260  ILITE = I-1
      IACT=0
C
C     READ DATA ON ACTINIDES
C
 270  READ(8,9034)NUCL(I),DLAM,IU,FB1,FP,FP1,FT,FA,FSF,Q(I),FG(I),DUMMY,
     1DUMY1,DUMY2
      IF(EOF(8).NE.0)GOTO450
      DO 280 N=1,NLIBE
      READ(8,9037) SIGNG,RING,FNG1,SIGF,RIF,SIGFF,SIGN2N,FN2N1,SIGN3N,IT
 280  CONTINUE
      IF(N1.EQ.0) GO TO 300
      DO 290 N=1,N1
 290  READ(8,9036)  SKIP
 300  IF(IT .EQ. 0) GO TO 270
 310  M=0
      NUCLI=NUCL(I)
      IF(NUCLI.EQ.0)  GO TO 450
      DO 320 K=1,5
      IF(NUCLI.EQ.NSORS(K))  NSURS(K)=I
 320  CONTINUE
      CALL HALF(A1,IU)
      CALL NOAH(NUCLI,NAME)
```

```
      SIGNG=THERM*SIGNG+RES*RING
      SIGF =THERM*SIGF +RES*RIF +FAST*SIGFF
      SIGN2N=SIGN2N*FAST
      SIGN3N=SIGN3N*FAST
      IF(MOD(IACT,50).EQ.0)  PRINT 9012,    (TITLE (N),N=1,18)
  330 IF(MOD(IACT,50).EQ.0) PRINT 9024
      PRINT 9026,    NAME,                DLAM,FB1,FP,FP1,FT,FA,FSF,SIGNG,
     1               FNG1,SIGF,SIGN2N,SIGN3N,Q(I),FP(I)
  340 IACT=IACT+1
C
C     TEST RADIOACTIVITY
C
      IF(A1.LT.ERR)  GO TO 380
      ABETA=1.0
C     TEST POSITRON EMISSION
      IF(FP .LT. ERR) GO TO 350
      ABETA=ABETA-FP
      M=M+1
      COEFF(M,I)=FP*A1
      NPROD(M,I)=NUCLI-10000
C     POSITRON EMISSION TO EXCITED STATE
      IF(FP1 .LT. ERR)GO TO 350
      M=M+1
      COEFF(M,I)=FP1*COEFF(M-1,I)
      NPROD(M,I)=NPROD(M-1,I)+1
      COEFF(M-1,I)=COEFF(M-1,I)-COEFF(M,I)
C     ISOMERIC TRANSITION
  350 IF(FT .LT.ERR)GO TO 360
      M=M+1
      COEFF(M,I)=FT*A1
      NPROD(M,I)=NUCLI
      ABETA=ABETA-FT
C     ALPHA EMISSION
  360 IF(FA .LT.ERR)GO TO 370
      M=M+1
      COEFF(M,I)=FA*A1
      NPROD(M,I)=NUCLI-20040
      M=M+1
      COEFF(M,I)=COEFF(M-1,I)
      NPROD(M,I)=20040
      ABETA=ABETA-FA
C     BETA DECAY
  370 IF(ABETA.LT.1.E-4) GO TO 380
      M=M+1
      COEFF(M,I)=ABETA*A1
      NPROD(M,I)=NUCLI+10000
      IF(FB1 .LT. ERR)GO TO 380
      M=M+1
      COEFF(M,I)=COEFF(M-1,I)*FB1
      COEFF(M-1,I)=COEFF(M-1,I)-COEFF(M,I)
      NPROD(M,I)=NPROD(M-1,I)+1
C
C     NEUTRON CAPTURE CROSS SECTIONS
C
  380 KAP(I)=M
      DO 390 K=1,6
  390 CAPT(K  )=0.0
      CAPT(2)=SIGNG*FNG1
      CAPT(1)=SIGNG-CAPT(2)
      CAPT(4)=SIGN2N*FN2N1
      CAPT(3)=SIGN2N-CAPT(4)
  400 FISS(IACT)=SIGF
```

```
      TOCAP(I)=0.0
      DO 410 K=1,4
      CAPKI=CAPT(K)
      IF(CAPKI.LT.ERR)  GO TO 410
      M=M+1
      TOCAP(I)=TOCAP(I)+CAPKI
      COEFF(M,I)=CAPKI
      NPROD(M,I)=NUCLI+NUCAL(K+2)
  410 CONTINUE
      TOCAP(I)=TOCAP(I)+FISS(IACT)
C     N-3N CROSS SECTION
      A17=SIGN3N
      IF(A17.LT.ERR) GO TO 420
      M=M+1
      COEFF(M   ,I)= A17
      NPROD(M   ,I)= NUCLI-20
      TOCAP(I)=TOCAP(I)+A17
  420 IF(MOD(NUCLI,10).EQ.0) GO TO 440
      DO 430 K=1,M
  430 NPROD(K,I)=NPROD(K,I)-1
  440 MMAX(I)=M
      IF(M.GT.7)  PRINT 9039, M
      SPONF(IACT)=FSF*A1*6.023E23
      ALPHAN(IACT)=FA*A1*6.023E13*Q(I)**3.65
      DIS(I)=A1
      I=I+1
      GO TO 270
  450 IL=0
      DO 460 K=1,5
  460 TYLD(K)=0.0
C
C     READ DATA FOR FISSION PRODUCTS
C
  470 READ(8,9034)NUCL(I),DLAM,IU,FB1,FP,FP1,FT,FA,FSF,Q(I),FG(I),DUMMY,
     1DUMY1,DUMY2
      IF(EOF(8).NE.0)GOTO0690
      DO 480 N=1,NLIBE
  480 READ(8,9038) SIGNG,RING,FNG1,Y,IT
      IF(N1.EQ.0) GO TO 500
      DO 490 N=1,N1
  490 READ(8,9036)  SKIP
  500 IF(IT .EQ. 0) GO TO 470
  510 M=0
      CALL HALF(A1,IU)
  520 NUCLI=NUCL(I)
      IF(NUCLI.EQ.0) GO TO 690
      CALL NOAH(NUCLI,NAME)
      IF(MOD(IL,50).EQ. 0)   PRINT 9012,    (TITLE (N),N=1,18)
      SIGNG=THERM*SIGNG+RES*RING
      IF(NLIBE.EQ.3) GO TO 3 0
  530 IF(MOD( IL           ,50).EQ.0)  PRINT 9019
      PRINT 9021,     NAME,                 DLAM,FB1,FP,FP1,FT,SIGNG,
     1      FNG1,Y,Q(I),FG(I)
      GO TO 550
  540 IF(MOD(IL,50).EQ.0) PRINT 9020
      PRINT 9022,     NAME ,                DLAM,FB1,FP,FP1,FT,SIGNG,FNG1,
     1                Y(2),Y(4),Y(5),Q(I),FG(I)
C
C     TEST RADIOACTIVITY
C
  550 IF(A1.LT.ERR)   GO TO 600
      ABETA=1.0
C     POSITRON EMISSION
```

```
      A3=FP
      IF(A3.LT.ERR)  GO TO 570
      ABETA=ABETA-A3
      AP1=A3*FP1
      AP=A3-AP1
      IF(AP.LT.ERR)  GO TO 560
      M=M+1
      COEFF(M,I)=AP*A1
      NPROD(M,I)=NUCLI-10000
  560 IF(AP1.LT.ERR)  GO TO 570
      M=M+1
      COEFF(M,I)=AP1*A1
      NPROD(M,I)=NUCLI-9999
C     ISOMERIC TRANSITION
  570 IF(FT  .LT. ERR) GO TO 580
      M=M+1
      COEFF(M,I)=FT*A1
      NPROD(M,I)=NUCLI
      ABETA=ABETA-FT
C     NEGATRON EMISSION
  580 IF(ABETA.LT.1.0E-4) GO TO 600
      A2=FB1
      AB1=ABETA*A2
      AB=ABETA-AB1
      IF(AB.LT.1.E-4) GO TO 590
      M=M+1
      COEFF(M,I)=AB*A1
      NPROD(M,I)=NUCLI+10000
  590 IF(AB1.LT.1.E-6) GO TO 600
      M=M+1
      COEFF(M,I)=AB1*A1
      NPROD(M,I)=NUCLI+10001
C
C     NEUTRON CAPTURE CROSS SECTIONS FOR FISSION PRODUCTS USING THREE
C     REGION APPROXIMATION
C
  600 KAP(I)=M
      DO 610 K=1,6
  610 CAPT(K)=0.0
      CAPT(2)=SIGNG*FNG1
      CAPT(1)=SIGNG-CAPT(2)
      TOCAP(I)=0.0
      DO 620 K=1,2
      CAPKI=CAPT(K)
      IF(CAPKI.LT.ERR)  GO TO 620
      M=M+1
      TOCAP(I)=TOCAP(I)+CAPKI
      COEFF(M,I)=CAPKI
      NPROD(M,I)=NUCLI+NUCAL(K+2)
  620 CONTINUE
  630 IF(MOD(NUCLI,10).EQ.0) GO TO 650
      DO 640 K=1,M
  640 NPROD(K,I)=NPROD(K,I)+1
  650 IL=IL+1
      DO 660 J=1,5
      YJ=Y(J)*0.010
      TYLD(J)=TYLD(J)+YJ
  660 YIELD(J,IL)=YJ
      IF(NLIBE.EQ.1.OR.NLIBE.EQ.4)  GO TO 680
  670 IF(NLIBE.EQ.3)  YIELD(1,IL)=YJ
      YIELD(3,IL)=YJ
  680 MMAX(I)=M
```

```
      IF(M.GT.7)  PRINT 9031, M
      DIS(I)=A1
      I=I+1
      GO TO 470
  690 IFP=IL
C
C     ALL DATA ON NUCLIDES HAS BEEN READ, BEGIN TO COMPUTE MATRIX COEFF
C
      ITOT=I-1
C
C     FIND PRODUCT NUCLIDES FOR REACTIONS OF LIGHT ELEMENTS
C
      NON=0
      DO 700 K=1,ITOT
  700 NONO(K)=0
      IF(ILITE.LT.1)  GO TO 760
      DO 750 I=1,ILITE
      NUCLI=NUCL(I)
      DO 720  J=1,ILITE
      KMAX=KAP(J)
      IF(KMAX.LT.1)  GO TO 720
      DO 710 M=1,KMAX
      IF(NUCLI.NE.NPROD(M,J)) GO TO 710
      NONO(I)=NONO(I)+1
      NON=NON+1
      IF(NON.GT.2500)  PRINT 9041,     NON,NUCL(I)
      A(NON)=COEFF(M,J)
      JT=J
      LOC(NON)=JT
  710 CONTINUE
  720 CONTINUE
      KD(I)=NONO(I)
      DO 740  J=1,ILITE
      K1=KAP(J)+1
      KMAX=MMAX(J)
      IF(KMAX.LT.K1)  GO TO 740
      DO 730 M=K1,KMAX
      IF(NUCLI.NE.NPROD(M,J)) GO TO 730
      NONO(I)=NONO(I)+1
      NON=NON+1
      IF(NON.GT.2500)  PRINT 9041,     NON,NUCL(I)
      A(NON)=COEFF(M,J)
      JT=J
      LOC(NON)=JT
  730 CONTINUE
  740 CONTINUE
  750 CONTINUE
C
C     NON ZERO MATRIX ELEMENTS FOR THE ACTINIDES
C
  760 IF(IACT.LT.1)  GO TO 820
      I0=ILITE+1
      I1=ILITE+IACT
      DO 810 I=I0,I1
      NUCLI=NUCL(I)
      DO 780 J=I0,I1
      MAX=KAP(J)
      IF(MAX.LT.1)  GO TO 790
      DO 770 M=1,MAX
      IF(NUCLI.NE.NPROD(M,J))  GO TO 770
      NONO(I)=NONO(I)+1
      NON=NON+1
      IF(NON.GT.2500)  PRINT 9041,     NON,NUCL(I)
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```
      A(NON)=COEFF(M,J)
      JT=J
      LOC(NON)=JT
  770 CONTINUE
  780 CONTINUE
      KD(I)=NONO(I)
      DO 800 J=I0,I1
      M1=KAP(J)+1
      M2=MMAX(J)
      IF(M2.LT.M1)  GO TO 800
      DO 790 M=M1,M2
      IF(NUCLI.NE.NPROD(M,J)) GO TO 790
      NONO(I)=NONO(I)+1
      NON=NON+1
      IF(NON.GT.2500)  PRINT 9001,      NON,NUCL(I)
      A(NON)=COEFF(M,J)
      JT=J
      LOC(NON)=JT
  790 CONTINUE
  800 CONTINUE
  810 CONTINUE
C
C     MATRIX ELEMENTS FOR FISSION PRODUCTS
C
  820 IF(IFP.LT.1) RETURN
      IM=ILITE+IACT
      I0=IM+1
      IF(ITOT.LT.I0) RETURN
      DO 880 I=I0,ITOT
      NUCLI=NUCL(I)
      I2=MAX0(I0,I-10)
      I3=MIN0(ITOT,I+10)
      DO 840 J=I2,I3
      KMAX=KAP(J)
      IF(KMAX.LT.1)  GO TO 840
      DO 830 M=1,KMAX
      IF(NUCLI.NE.NPROD(M,J)) GO TO 830
      NONO(I)=NONO(I)+1
      NON=NON+1
      IF(NON.GT.2500)  PRINT 9001,      NON,NUCL(I)
      A(NON)=COEFF(M,J)
      JT=J
      LOC(NON)=JT
  830 CONTINUE
  840 CONTINUE
      KD(I)=NONO(I)
      DO 860 J=I2,I3
      K1=KAP(J)+1
      KMAX=MMAX(J)
      IF(KMAX.LT.K1)  GO TO 860
      DO 850 M=K1,KMAX
      IF(NUCLI.NE.NPROD(M,J)) GO TO 850
      NONO(I)=NONO(I)+1
      NON=NON+1
      IF(NON.GT.2500)  PRINT 9001,      NON,NUCL(I)
      A(NON)=COEFF(M,J)
      JT=J
      LOC(NON)=JT
  850 CONTINUE
  860 CONTINUE
      IF(IACT.LT.1)  GO TO 880
      DO 870 K=1,5
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      IL=I-IM
      IF(YIELD(K,IL).LT.ERR)  GO TO 870
      NON=NON+1
      IF(NON.GT.2500)  PRINT 9061,      NON,NUCL(I)
      NONO(I)=NONO(I)+1
      KK=NSORS(K)
      LOC(NON)=KK
      KF=KK-ILITE
      A(NON)=YIELD(K,IL)*FISS(KF)
  870 CONTINUE
  880 CONTINUE
      IF(IFP.LE.0) GO TO 900
      IF(NLIBE.NE.3) GO TO 890
      PRINT 9027,     TYLD(2),TYLD(4),TYLD(5)
      GO TO 900
  890 PRINT 9030,      (TYLD(I),I=1,5)
C
C     ALL MATRIX ELEMENTS ARE NOW COMPUTED
C     BEGIN TRANSIENT SOLUTION
C
C
C     TEMPORARILY WRITE OUT MATRIX ELEMENTS
C
  900 IF(IR .EQ. 0) RETURN
      PRINT 9029
      N=0
      DO 910 I=1,ITOT
      NUM=NONO(I)
      IF(NUM.LE.0)   GO TO 910
      N1=N+NUM
      N=N+1
      PRINT 9028,     I,DIS (I),TOCAP(I),(A(K),LOC(K),K=N,N1)
      N=N1
  910 CONTINUE
      RETURN
  920 STOP
C
C     FORMATS      FORMATS      FORMATS      FORMATS
C
 9001 FORMAT(4F10.5,6I2)
 9005 FORMAT(1H1,43X,"NUCLEAR TRANSMUTATION DATA   REVISED ",I2,"/",I2,"
     1/",I2,/,"0NUCL = NUCLIDE = 10000 * ATOMIC NO + 10 * MASS NO + ISOM
     2ERIC STATE (0 OR 1)",10X,"DLAM = DECAY CONSTANT (1/SEC).",/," FB, 
     3FP, FA, FT = FRACTIONAL DECAY BY BETA, POSITRON (OR ELECTRON CAPTU
     4RE), ALPHA, INTERNAL TRANSITION.   FB = 1 - FP - FA - FT",/," FB1,
     5 FP1, FNG1, FN2N1 = FRACTION OF BETA, POSITRON, N-GAMMA, N-2N TRAN
     6SITIONS TO EXCITED STATE OF PRODUCT NUCLIDE",/," SIGTH, SIGNG, SIG
     7F, SIGNA, SIGNP = THERMAL CROSS SECTIONS (BARNS) FOR ABSORPTION, N
     8-GAMMA, FISSION, N-ALPHA, N-PROTON.")
 9006 FORMAT("   SIGNG = SIGTH * (1 - FNA -FNP).  SIGNA = SIGTH * FNA.  
     1SIGNP = SIGTH * FNP.  FNA, FNP = FRACTION THERMAL N-ALPHA, N-PROTO
     2N.",/," RITH, RING, RIF, RINA, RINP = RESONANCE INTEGRAL FOR ABSOR
     3PTION, N-GAMMA, FISSION, N-ALPHA, N-PROTON.",/,"   RING = RITH * (
     41 - FINA - FINP). RINA = RITH * FINA. RINP = RITH * FINP.  FINA, F
     5INP = FRACTION RESONANCE N-ALPHA, N-PROTON.",/," SIGMEV, SIGFF, SI
     6GN2N, SIGNAF, SIGNPF = FAST CROSS SECTIONS (BARNS) FOR ABSORPTION,
     7FISSION, N-2N, N-ALPHA, N-PROTON.",/,"   SIGN2N = SIGMEV * (1 - FF
     8NA - FFNP).  SIGNAF = SIGMEV * FFNA.  SIGNPF = SIGMEV * FFNP.  FFN
     9A, FFNP = FRACTION FAST N-ALPHA, N-P.")
 9007 FORMAT(" Y23, Y25, Y02, Y28, Y49 = FISSION YIELD (PERCENT) FROM 230
     13-U, 235-U, 232-TH, 238-U, 239- PU.",/," Q = HEAT PER DISINTEGRATI
     2ON.  FG = FRACTION OF HEAT IN GAMMAS OF ENERGY GREATER THAN 0.2 ME
     3V.",/,"0 EFFECTIVE CROSS SECTIONS FOR A VOLUME AVERAGED THERMAL (L
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     4T 0.876 EV) FLUX ARE AS FOLLOWS.",/,"    N-GAMMA  -  SIGNG * THERM
     5 + RING * RES.",/,"    FISSION  -  SIGF * THERM + RIF * RES + SIGF
     6F * FAST.",10X,"THERM = 1/V CORRECTION FOR THERMAL SPECTRUM AND TE
     7MPERATURE.",/,"    N-2N     -  SIGN2N * FAST.",36X,"RES   = RATIO
     8OF RESONANCE FLUX PER LETHARGY UNIT TO THERMAL FLUX.")
9008 FORMAT("    N-ALPHA  -  SIGNA * THERM + RINA * RES + SIGNAF * FAST
     1.",7X,"FAST = 1.45 *  RATIO OF FAST (GT 1.0 MEV) TO THERMAL FLUX "
     2/"    N-PROTON  -  SIGNP * THERM + RINP * RES + SIGNPF * FAST.")
9009 FORMAT(1H0,59X,"REFERENCES",/,"    HALF LIVES, DECAY SCHEMES, AND
     1THERMAL POWER",/," C M LEDERER, J M HOLLANDER, AND I PERLMAN ""TAB
     2LE OF ISOTOPES - SIXTH EDITION"" JOHN WILEY AND SONS, INC (1967)",
     3/," B S DZHELEPOV AND L K PEKER ""DECAY SCHEMES OF RADIOACTIVE NUC
     4LEI"" PERGAMMON PRESS (1961)",/," D T GOLDMAN AND JAMES R ROSSER "
     5"CHART OF THE NUCLIDES"" NINTH EDITION  GENERAL ELECTRIC CO (JULY
     61966)",/," E D ARNOLD ""PROGRAM SPECTRA"" APPENDIX A OF ORNL-3576
     7(APRIL 1964)")
9010 FORMAT("    CROSS SECTIONS AND FLUX SPECTRA",/," B E PRINCE ""NEUT
     1RON REACTION RATES IN THE MSRE SPECTRUM"" ORNL-4119, PP 79-83 (JUL
     2Y 1967)",/," B E PRINCE ""NEUTRON ENERGY SPECTRA IN MSRE AND MSBR"
     3" ORNL-4191, PP 50-58 (DEC 1967)",/," M D GOLDBERG ET AL ""NEUTRON
     4 CROSS SECTIONS"" BNL-325, SECOND ED, SUPP NO 2 (MAY 1964 - AUG 19
     566) ALSO EARLIER EDITIONS",/," H T KERR, UNPUBLISHED ERC COMPILATI
     6ON (FEB 1968)",/," M K DRAKE ""A COMPILATION OF RESONANCE INTEGRAL
     7S"" NUCLEONICS, VOL 24, NO 8, PP 108-111 (AUG 1966)",/," BNWL STAF
     8F ""INVESTIGATION OF N-2N CROSS SECTIONS"" BNWC-98, PP 44-98 (JUNE
     9 1965)")
9011 FORMAT(18A4,I3)
9012 FORMAT(1H1,20X,18A4)
9013 FORMAT(" H ALTER AND C E WEBER ""PRODUCTION OF H AND HE IN METALS
     1DURING REACTOR IRRADIATION"" J NUCL MATLS, VOL 16, PP 68-73 (1965)
     2",/," L L BENNETT ""RECOMMENDED FISSION PRODUCT CHAINS FOR USE IN
     3REACTOR EVALUATION STUDIES"" ORNL-TM-1658 (SEPT 1966)")
9014 FORMAT("    FISSION PRODUCT YIELDS",/," M E MEEK AND B F RIDER, ""
     1SUMMARY OF FISSION PRODUCT YIELDS FOR U-235, U-238, PU-239, AND PU
     2-241 AT THERMAL, FISSION SPECTRUM AND"/"     14 MEV NEUTRON ENERGI
     3ES"" APED-5398-A(REV.1,(OCT. 1968)"/" S KATCOFF "" FISSION PRODUCT
     4YIELDS FROM NEUTRON INDUCED FISSION"" NUCLEONICS, VOL 18, NO 11,
     5(NOV 1960)"/" N D DUDEY "" REVIEW OF LOW-MASS ATOM PRODUCTION IN F
     6AST REACTORS"" ANL-7434,(APRIL 1968) ")
9016 FORMAT(1H0,20X,"LIGHT ELEMENTS, MATERIALS OF CONSTRUCTION, AND ACT
     1IVATTION PRODUCTS  ",/,"0 NUCL      DLAM      FB1      FP      "
     2"FP1     FT      FA      SIGNG      FNG1      SIGN2N     FN2N1"
     3"    SIGNA     SIGNP       Q       FG  ABUNDANCE")
9018 FORMAT(1H0,10X,"THERM= "F10.5,5X,"RES= "F10.5,5X,"FAST= "F10.5,
     1//,1X,"NEUTRON SOURCE= "5(I10,5X),5X,"NLIBE= "I3)
9019 FORMAT(1H0,36X,"FISSION PRODUCTS",/,"0 NUCL       DLAM       "
     1 "FB1      FP      FP1      FT       SIGNG     FNG1       Y23      "
     2 "Y25        Y02        Y28        Y49        Q        FG")
9020 FORMAT(1H0,36X,"FISSION PRODUCTS",/,"0 NUCL       DLAM       FB1      "
     1 "FP      FP1      FT       SIGNG      FNG1       Y25       Y28"
     2 "       Y49       Q       FG")
9021 FORMAT(1H ,A2,I3,A1,1PE10.2,0P4F7.3,1PE10.2,0PF7.3,
     1  1P5E10.2,0P2F7.3)
9022 FORMAT(1H ,A2,I3,A1,1PE10.2,0P4F7.3,1PE10.2,
     1  0PF7.3,1P3E10.2,0P2F7.3)
9024 FORMAT(1H0,32X, "ACTINIDES AND THEIR DAUGHTERS",//
     1"  NUCL       DLAM       FB1      FP       FP1      FT"
     2"      FA      FSF E+6   SIGNG      FNG21      SIGF"
     3"     SIGN2N      SIGN3N       Q       FG")
9026 FORMAT(1H ,A2,I3,A1,1PE10.2,0P5F7.3,6PF10.1,1PE10.2,
     1  0PF7.3,1P3E10.2,0PF9.3,F6.2)
9027 FORMAT("0SUM OF YIELDS OF ALL FISSION PRODUCTS =",15X,1P3E9.2)
```

```
9028 FORMAT(I5,2X,1PE10.3,3X,E10.3,5(2X,E10.3,3X,I5)/(30X,5(2X,E10.3,
    1      3X,I5)))
9029 FORMAT("1NON-ZERO MATRIX ELEMENTS AND THEIR LOCATIONS"/
    1"  I        DIS(I)           CAP(I)          A(I,J)          J     A(I,J)
    2J       A(I,J)          J        A(I,J)           J        A(I,J)          J  ")
9030  FORMAT(63H0SUM OF YIELDS OF ALL FISSION PRODUCTS
    1         ,5(E9.2,1X))
9033  FORMAT(1H ,A2,I3,A1,1PE10.2,0P5F7.3,1PE10.2,0PF7.3,1PE10.2,
    1   0PF7.3,1P2E10.2,0P2F7.3,F8.3)
9034 FORMAT(I7,F9.3,I1,5F5.3,1PE9.2,0P2F5.3,F7.3,2E6.0)
9035 FORMAT(7X,F9.2,3F5.3,F9.2,2F5.3,F9.2,3F5.3, 5X,I1)
9036 FORMAT(20A4)
9037  FORMAT(7X,2F9.2,F5.3,4F9.2,F4.1,F9.2,I1)
9038 FORMAT(7X,2F9.2,F5.3,3F9.2, 4X,I1)
9039 FORMAT("0  WARNING, MOUT OF RANGE IN NUDATA, =" I5)
9040 FORMAT(    7X,F9.2,3F8.6,F6.2,2F3.1,F9.2,3F5.3,5X,I1)
9041 FORMAT("0 NON HAS EXCEEDED 2500, EQUAL TO "2I6)
     END
*DECK COLLECT
     SUBROUTINE COLLECT(TMB,CWASTE,ITOT)
     COMMON/EQ/XTEMP(800),XNEW(10,800),B(800),P(800)
     DIMENSION CWASTE(800)
     IF(TMB.LT.1) RETURN
     DO 10 I=1,ITOT
     B(I)=CWASTE(I)
 10  XTEMP(I)=0.0
     CALL DECAY(1,TMB,ITOT)
     CALL TERM(TMB,1,ITOT)
     CALL EQUIL(1,ITOT)
     DO 20 I=1,ITOT
 20  CWASTE(I)=XNEW(1,I)/TMB
     RETURN
     END
*DECK STORAG
     SUBROUTINE STORAG(TMB,CWASTE,ITOT)
     COMMON/EQ/XTEMP(800),XNEW(10,800),B(800),P(800)
     DIMENSION CWASTE(ITOT)
     IF(TMB.LT.1)  RETURN
     DELT=TMB
     DO 10 I=1,ITOT
     B(I)=0.0
 10  XTEMP(I)=CWASTE(I)
     CALL DECAY(1,DELT,ITOT)
     CALL TERM(TMB,1,ITOT)
     CALL EQUIL(1,ITOT)
     DO 20 I=1,ITOT
 20  CWASTE(I)=XNEW(1,I)
     RETURN
     END
*DECK BLKDAT1
C    PROGRAM BLOCK DATA
     BLOCK DATA BLKDAT1
     INTEGERELE(99),STA(2)
     COMMON/LABEL/  ELE,STA
     DATA ELE/" H"," HE"," LI"," BE"," B"," C"," N"," O"," F","NE","NA","M
    1G","AL","SI"," P"," S","CL","AR"," K","CA","SC","TI"," V","CR","MN
    2","FE","CO","NI","CU","ZN","GA","GE","AS","SE","BR","KR","RB","SR"
    3," Y","ZR","NB","MO","TC","RU","RH","PD","AG","CD","IN","SN","SB","
    4TE"," I","XE","CS","BA","LA","CE","PR","ND","PM","SM","EU","GD","
    5TB","DY","HO","ER","TM","YB","LU","HF","TA"," W","RE","OS","IR","P
    6T","AU","HG","TL","PB","BI","PO","AT","RN","FR","RA","AC","TH","PA
    7"," U","NP","PU","AM","CM","BK","CF","ES"/
     DATA STA/"  ","M "/
```

```
      END
*DECK HALF
      SUBROUTINE  HALF(A,I)
C     SUBROUTINE HALF CONVERTS HALF-LIFE TO DECAY CONSTANT (1/SEC)
      DIMENSION C(9)
      DATA C/6.9315E-01,1.1552E-02,1.9254E-04,8.0226E-06,2.1965E-08,0.0,
     1       2.1965E-11,2.1965E-14,2.1965E-17/
      IF(A.GT.0.0)  GO TO 10
      IF(I.EQ.6)  GO TO 20
      A=9.99
      RETURN
 10   A=C(I)/A
      RETURN
 20   A=0.0
      RETURN
      END
*DECK NOAH
      SUBROUTINE NOAH(NUCLI,NAME)
C     SUBROUTINE NOAH CONVERTS SIX DIGIT IDENTIFIER TO ALPHAMERIC SYMBOL
      INTEGERNAME(3)
      INTEGERELE(99),STA(2)
      COMMON/LABEL/  ELE,STA
      IS=MOD(NUCLI,10)+1
      NZ  =NUCLI/10000
      MW=NUCLI/10-NZ  *1000
      NAME(1)=ELE(NZ)
      NAME(2)=MW
      NAME(3)=STA(IS)
      RETURN
      END
```

CHAPTER 4. DATA FOR RADIOACTIVE SOURCE TERM CALCULATIONS FOR PRESSURIZED WATER REACTORS (PWR's)

This chapter lists the information needed to generate source terms for PWR's. The information is provided by the applicant and is consistent with the contents for the Safety Analysis Report (SAR) and the Environmental Report (ER) of the proposed pressurized water reactor. This information constitutes the basic data required in calculating the releases of radioactive material in liquid and gaseous effluents (the source terms). All data are on a per-reactor basis.

4.1 GENERAL

1. The maximum core thermal power (MWt) evaluated for safety considerations in the SAR.

   Note: All the information required in calculating the releases should be adjusted to this power level.

2. The quantity of tritium released in liquid and gaseous effluents (Ci/yr per reactor).

4.2 PRIMARY SYSTEM

1. The total mass (lb) of coolant in the primary system, excluding the pressurizer and primary coolant purification system, at full power.

2. The average primary system letdown rate (gal/min) to the primary coolant purification system.

3. The average flow rate (gal/min) through the primary coolant purification system cation demineralizers.

   Note: The letdown rate should include the fraction of time the cation demineralizers are in service.

4. The average shim bleed flow rate (gal/min).

4.3 SECONDARY SYSTEM

1. The number and type of steam generators and the carryover factor used in the evaluation for iodine and nonvolatiles.

2. The total steam flow rate (lb/hr) in the secondary system.

3. The mass of liquid in each steam generator (lb) at full power.

4. The primary-to-secondary system leakage rate (lb/day) used in the evaluation.

5. Description of the steam generator blowdown purification system. The average steam generator blowdown rate (lb/hr) used in the evaluation.

6. The fraction of the steam generator feedwater processed through the condensate demineralizers and the DF's used in the evaluation for the condensate demineralizer system.

7. Condensate demineralizers

   a. Average flow rate (lb/hr);

   b. Demineralizer type (deep bed or powdered resin);

   c. Number and size (ft^3) of demineralizers;

   d. Regeneration frequency;

   e. Indication whether ultrasonic resin cleaning is used and the waste liquid volume associated with its use; and

   f. Regenerant volume (gal/event) and activity.

4.4 LIQUID WASTE PROCESSING SYSTEMS

1. For each liquid waste processing system, including the shim bleed, steam generator blowdown, and detergent waste processing systems, provide in tabular form the following information:

   a. Sources, flow rates (gal/day), and expected activities (fraction of primary coolant activity) for all inputs to each system.

   b. Holdup times associated with collection, processing, and discharge of all liquid streams.

   c. Capacities of all tanks (gal) and processing equipment (gal/day) considered in calculating holdup times.

   d. Decontamination factors for each processing step.

   e. Fraction of each processing stream expected to be discharged over the life of the plant.

   f. For demineralizer regeneration, provide time between regenerations, regenerant volumes and activities, treatment of regenerants, and fraction of regenerant discharged. Include parameters used in making these determinations.

   g. Liquid source term by radionuclide in Ci/yr for normal operation, including anticipated operational occurrences.

2. Provide piping and instrumentation diagrams (P&ID's) and process flow diagrams for the liquid radwaste systems along with all other systems influencing the source term calculations.

4.5 GASEOUS WASTE PROCESSING SYSTEM

For the waste gas processing system, provide the following:

1. The method of stripping gases from the primary coolant, the volumes (ft^3/yr) of gases stripped from the primary coolant, the bases for these volumes.

2. Description of the process used to hold up gases stripped from the primary system during normal operations and reactor shutdown. If pressurized storage tanks are used, include a process flow diagram of the system indicating the capacities (ft^3), number, and design and operating storage pressures for the storage tanks.

3. Describe the normal operation of the system, e.g., number of tanks held in reserve for back-to-back shutdown, fill time for tanks. Indicate the minimum holdup time used in the evaluation and the basis for this number.

4. If HEPA filters are used downstream of the pressurized storage tanks, provide the decontamination factor used in the evaluation.

5. If a charcoal delay system is used, describe this system and indicate the minimum holdup times for each radionuclide considered in the evaluation. List all parameters, including mass of charcoal (lb), flow rate ft^3/min), operating and dew point temperatures, and the dynamic adsorption coefficients for Xe and Kr used in calculating holdup times.

6. Provide piping and instrumentation diagrams (P&ID's) and process flow diagrams for the gaseous radwaste systems along with other systems influencing the source term calculations.

4.6 VENTILATION AND EXHAUST SYSTEMS

For each building housing systems that contain radioactive materials, the steam generator blowdown system vent exhaust, gaseous waste processing system vent, and the main condenser air removal system, provide the following:

1. Provisions incorporated to reduce radioactivity releases through the ventilation or exhaust systems.

2. Decontamination factors assumed and the bases (include charcoal adsorbers, depth of charcoal beds, HEPA filters, and mechanical devices).

3. Release rates for radioiodine, noble gases, and radioactive particulates (Ci/yr), radioactive particulate size distribution, and the bases.

4. Release point description, including height above grade, height above relative location to adjacent structures, relative temperature difference between gaseous effluents and ambient air, flow rate, velocity, and size and shape of flow orifice.

5. For the containment building, the building free volume (ft^3) and a thorough description of the internal recirculation system (if provided), including the recirculation rate, charcoal bed depth, operating time assumed, and mixing efficiency. Indicated the expected purge and venting frequencies and duration and continuous purge rate (if used).

APPENDIX A

LIQUID SOURCE TERM CALCULATIONAL PROCEDURE FOR REGENERANT WASTES FROM DEMINERALIZERS OTHER THAN CONDENSATE DEMINERALIZERS

Often in PWR radwaste systems, demineralizers other than the condensate demineralizers may undergo regeneration, for example, the radwaste demineralizer in the dirty waste system. The PWR-GALE Code can calculate the liquid effluent resulting from periodic regeneration of non-condensate demineralizers by following the procedure outlined below.

1. Input to Cards 1-11 and Cards 27-42

   A separate computer run for calculating the regeneration waste effluent from non-condensate demineralizers is required. Cards 1-11 should be filled out as indicated for the specific plant in Sections 1.5.2.1 through 1.5.2.11 of this report. Also Cards 27 through 41 may be left blank (except that values of 1.0 must be entered for Card 28 entries). Card 42 should be left blank.

2. Input to Cards 12-26

   The only liquid source term data cards completed (Cards 12-26) should be the three card sets used in the input data for the stream in which the demineralizer to be regenerated is located. The remaining card sets should have a zero entered for the input flow rate.

   a. Input Flow and Activity (Card 12, 15, 18, 21 or 24)

      The input flow rate and input activity should be the average daily input flow rate and input activity processed through the demineralizer to be regenerated. For example, if the demineralizer to be regenerated is used to process a shim bleed waste stream, the total input flow rate might be 1440 gallons per day.

      Note that it is not the flow rate and activity which is due to the regenerant waste which is entered, it is the normal flow rate and activity through the component to be regenerated which is entered.

   b. Regeneration Frequency (Card 14, 17, 20, 23 or 26)

      Enter the time between regenerations in days as the "collection time." If a regeneration frequency is stated by the applicant, it may be used; otherwise the following frequency may be used:

TABLE A-1

Demineralizer Service	Regeneration Frequency
Primary Coolant Letdown	180 days
Boron Recovery System	180 days
Equipment Drain Wastes	*
Floor Drain Wastes	*
Steam Generator Blowdown	90 days

* Regeneration frequency is calculated by dividing the waste quantity (gallons) by the waste flow rate in gallons per day. The waste quantity is 25000 gal/ft^3 times the volume in ft^3 of resin for equipment drain waste and 2000 gal/ft^3 times the volume in ft^3 of resin for floor drain waste. The calculated values of 25,000 and 2,000 gal/ft^3 of resin for the waste are based on 12,000 g $CaCO_3$ ion exchange capacity per ft^3 of resin and 5 μmho/cm and 50 μmho/cm average conductivity for equipment and floor drain liquid wastes.

By inputting the normal flow rate and activity in Item a and the regeneration frequency as the collection time in Item b the PWR-GALE Code will accumulate all of the activity processed through the demineralizer during its normal operation and decay the activity as a function of the time over which it was collected.

c. Process Time and Fraction Discharged

Use the same "process time" and "fraction discharged" as indicated for the stream in which the regeneration wastes are processed as indicated in Section 1.5.2.12.4 of this document.

d. Decontamination Factors (Card 13, 16, 19, 22 or 25)

The decontamination factors entered should consider radionuclide removal by the equipment used to process the regenerant wastes using the normal source term procedures of 1.5.2.12.2. In addition, the decontamination factors entered should be used to adjust the source term for the fraction of the activity in the process stream flowing through the demineralizer during normal operation which is not removed by the demineralizer.

e. Sample Case

A demineralizer is used to process shim bleed waste and is to be regenerated. The normal flow rate for the demineralizer is 1440 gpd and the activity is calculated in the PWR-GALE Code. The regenerant wastes will be processed through an evaporator and discharged.

Fill in the Cards 12-14 in the following manner:

Card 12

Spaces 18-41 enter - shim bleed demin regen
Spaces 42-49 enter - 1440.0

Card 13

The wastes will be processed through an evaporator which will provide the following DF's according to Table 1-4 of Section 1.5.2.12.2.

I - 10^2
Cs, Rb - 10^3
Others - 10^3

While in operation, referring to Table 1-4 of Section 1.5.2.12.2 demineralizer DF's are:

I - 10
Cs, Rb - 2
Others - 10

Therefore, for "I" and "Others," 90% of the activity processed through the demineralizer is removed by the resins and no adjustment is needed. Only 50% of the Cs and Rb in the waste stream is removed by the resins, however, so the DF entered for Cs should be adjusted. Thus, the DF's entered on Card 13 would be:

I - 100.0
Cs, Rb - 2000.0
Others - 1000.0

Card 14

Spaces 29-33 "Collection Time." Using the value from Table A-1 of 180 days for the regeneration frequency

Enter 180.0 days in spaces 29-33.

Use the same "Process time" and "fraction discharged" as is indicated for the stream in which the regeneration wastes are processed as indicated in Section 1.5.2.12.4 of this report.

Note: If there is more than one stream for which non-condensate regenerant demineralizer is used, follow the same procedures explained under item A2 for the other stream or streams.

3. Components in Service

a. If the waste is processed through a component other than a regenerable demineralizer prior to processing by the regenerable demineralizer, the activity in the steam entering the demineralizer will be less than the activity entered as described above. To compensate for this difference, the DF's for the regenerant waste calculation should be adjusted in a manner similar to that described above. The product of the DF's should be used.

b. If two regenerable demineralizers are used in series, follow the procedure in a above. Adjust the DF for nuclides removed from the waste stream, by using the product of the DF's for two demineralizers in series, i.e., consider the two demineralizers as one larger demineralizer.

4. Use of Computer Calculated Result

Combine the values printed out in the individual liquid source term columns for the system in which the demineralizer is being regenerated (not the adjusted total value) with the normal liquid source term run values. Do not use the adjusted total value from the right hand column since the source term run to which the regenerant waste run will be added has already been adjusted.

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