NUREG-0016
REVISION 1

CALCULATION OF RELEASES OF RADIOACTIVE MATERIALS IN GASEOUS LIQUID EFFLUENTS FROM BOILING WATER REACTORS (BWR-GALE CODE)

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U.S. Nuclear Regulatory Commission

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FOREWORD

The calculational procedures described in this NUREG report reflect current NRC staff practice. Therefore, the methods described herein will be used in the evaluation of applications for construction permits and operating licenses docketed after January 1, 1979, until this NUREG is revised as a result of additional staff review.

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CHAPTER 1. BWR-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-0016 by incorporating more recent operating data now available and also by incorporating the results of a number of in-plant measurements programs at operating BWRs.

The BWR-GALE (Boiling Water Reactor Gaseous and Liquid Effluents) Code is a computerized mathematical model for calculating the release of radioactive material in gaseous and liquid effluents from boiling water reactors (BWRs). The calculations are based on data generated from operating reactors, field tests, laboratory tests, and plant-specific design considerations incorporated to reduce the quantity of radioactive materials that may be released to the environment.

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 calculations performed by the BWR-GALE Code are based on (1) standardized coolant activities derived from American Nuclear Society (ANS) 18.1 Working Group recommendations (Ref. 1), (2) 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 environs, and (4) information received on the operation of nuclear power plants.

In a BWR, water is converted to steam by heat from the fuel elements in the reactor. The steam expands through a turbine and then is condensed and returned to the reactor. The principal mechanisms that affect the concentrations of radioactive materials in the reactor 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 by the reactor coolant cleanup system, (4) radioactivity removed by the condensate demineralizers, (5) radioactivity removed through the steam-jet air ejectors, and (6) radioactivity removed due to reactor coolant leakage. These mechanisms are described briefly in the following paragraphs.

Fission products enter the coolant as a result of defects in the fuel cladding and from the tramp uranium on the cladding surfaces, while corrosion products are activated in the reactor core. These impurities must be continuously removed from the reactor coolant to prevent damage to the fuel elements and other reactor components. The removal is accomplished in two ways: (1) after passing through the turbine, the condensed steam is processed through the condensate cleanup system (e.g., demineralizers) and returned to the reactor for reuse and (2) a side stream of reactor coolant is continuously withdrawn, processed through the reactor water cleanup system (demineralizers), and returned to the reactor vessel. Both cleanup systems remove particulates and ionic impurities from the reactor coolant. The materials collected by the demineralizers are removed periodically by chemical regeneration or by replacement of resins. The liquid wastes are processed in the liquid waste treatment system, and the spent ion exchange resins are transferred to the solid waste treatment system and prepared for offsite shipment.

Radioactive gases are removed from the condensing steam in the main condenser by the steam-jet air ejectors. This source of gaseous waste is treated principally by delaying the release to permit radioactive decay. Treatment methods include holdup lines, long-term holdup using charcoal delay systems, and cryogenic distillation.

Additional radioactive material is released with the exhaust from the turbine gland sealing system when a sidestream of primary steam flows through the turbine gland seal. The steam is condensed and returned to the condenser hotwell for reuse in the reactor. However, noble gases, activation gases, radioactive particulates, and radioiodine that remain in the gaseous phase must be vented. The treatment provided this source of gaseous waste is normally a two-minute holdup line that permits decay of the short-lived noble and activation gases before they are released to the environment. Clean steam (nonradioactive steam) may be used in place of primary steam to eliminate the turbine gland seal as a potential activity release point.

Following plant shutdowns, mechanical vacuum pumps are used to reestablish the main condenser vacuum. In addition, the mechanical vacuum pumps may be used during plant shutdowns to maintain a slight condenser vacuum and thereby prevent outleakage of radioactive gases from the main condenser. If required to meet the design objectives of Appendix I, the effluent from the mechanical vacuum pump effluent could be processed through charcoal adsorbers for removal of radioiodine prior to release to the environment.

In addition to the above release points, the BWR-GALE Code considers ventilation system releases from the turbine, containment, auxiliary (including the spent fuel pool area), and radwaste buildings due to leakage from contaminated systems. Such leakage from systems containing main steam or reactor coolant may have an appreciable effect on the radioactive source term. Leakage may occur through valve stems, pump seals, and flanged connections. The amount of airborne radioactive material released is a function of reactor coolant temperature, pressure, and activity at the point where the leak occurs. Included with the leaking steam or coolant are noble gases, iodine, and particulates that are released directly to the building atmosphere. In some cases, leakage may be reduced by special design features such as vacuum leakoff drains or "clean" steam on the valve bonnets in addition to normal precautions such as backseating valves and using all-welded systems. Leakage can also be reduced by the use of closed leakoff drains and by increased maintenance.

Liquid waste sources include liquid streams used to sluice (transfer), backwash, regenerate, and rinse demineralizer resins; laundry waste water; personnel shower wastes; laboratory drain wastes; decontamination wastes; and water collected in equipment drains and floor drains.

This chapter provides a step-by-step explanation of the BWR-GALE Code and a description of the parameters that have been built into the Code for use with all BWR source term calculations. These parameters, which apply generically to all BWRs, have been incorporated into the Code to eliminate the need for their entry on input data cards. This chapter also describes the entries 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 BWR-GALE Code and explains the bases for each parameter. Chapter 3 contains a sample data input sheet and a FORTRAN listing of the BWR-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 due to 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 Stream: Liquids that contain relatively high concentrations of decontamination wastes or chemical compounds other than detergents. These liquids originate primarily from resin regenerants and laboratory waste.

Carryover Factor: Ratio of I-131 concentration in the condenser hotwell to its concentration in the reactor vessel. This value is used to express the partition coefficient between the steam and water phases in the reactor.

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 and normally have a low radioactivity content.

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

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

where

P_i is the ith power level, in MWt;

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

T_i is the time of operation at power level 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: Gaseous waste containing radioactive materials resulting from the operation of a nuclear power reactor.

High-Purity Waste Stream: Liquids, normally of low conductivity, consisting primarily of liquid waste collected from building equipment drains, valve and pump seal leakoffs, demineralizer backwash, ultrasonic resin cleaning, and resin transfer. These liquids are normally reused as primary coolant makeup water after processing.

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

Low-Purity Waste Stream: Liquids, normally of high conductivity and not of primary coolant quality, collected from building sumps, uncollected valve and pump seal leakoffs, miscellaneous vents, and floor drains.

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

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

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

Reactor Coolant: The fluid circulated through the reactor to remove heat. In a BWR, the fluid is allowed to boil in the reactor vessel to generate steam and power the turbine. The reactor coolant activity is considered to be constant over a range of power levels, coolant and cleanup flows, and reactor coolant volumes. The radionuclide distributions and concentrations for the reactor coolant and main steam are based on the values given in American National Standard, ANSI N237, Source Term Specification, (Ref. 1) but have been adjusted to plants with pumped forward heater drains. In addition, radioiodine and noble gas concentrations are based on a recent compilation of available operating data. Therefore, the concentration values in NUREG-0016, Rev. 1 differ slightly from the ANSI N237 values. Provisions are made in the BWR-GALE Code, in accordance with the recommendations of the standard, for adjusting reactor coolant concentrations should the plant be designed to parameters that are outside the ranges considered in the standard. The ANSI N237 radionuclide concentrations used are also 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. Cesium and Rubidium
4. Water activation products
5. Tritium

6. Other nuclides (as listed in Table 2-2 of Chapter 2 of this document)

Regenerant Solutions Waste Stream: Liquids containing regeneration chemical compounds that originate from regeneration of the condensate demineralizer resins.

Source Term: The calculated annual 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. Normal operation includes routine outages for maintenance and scheduled refuelings.

Tramp Uranium: The uranium present on the exterior of the cladding of a fuel rod and core support structure surfaces.

1.3 GASEOUS SOURCE TERMS

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

1. Main condenser offgas system,

2. Turbine gland sealing system,

3. Mechanical vacuum pumps, and

4. Ventilation exhaust air from the containment, auxiliary, radwaste, and turbine buildings, and the spent fuel pool area

The releases of radioactive materials in gaseous effluents are based on measurements made at operating BWRs. The radioactive particulate and noble gas release rates are specified in the BWR-GALE Code and are modified only as needed to reflect treatment processes. Gaseous releases for building ventilation exhaust systems and the main condenser offgas system are based on the average of actual measurements. Radioiodine releases are related to the iodine-131 reactor water concentrations for the BWR being evaluated.

Chapter 2 provides iodine and particulate decontamination factors for removal equipment and parameters for calculating holdup times for noble gases and for calculating tritium releases.

1.4 LIQUID SOURCE TERMS

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

1. Processed liquid wastes from the high-purity waste system,

2. Processed liquid wastes from the low-purity waste system,

3. Processed liquid wastes from the chemical waste system,

4. Processed liquid regenerant wastes, and

5. Detergent wastes.

The radioactivity input to the liquid radwaste treatment system is based on flow rates of the liquid waste streams and their radioactivity levels, expressed as a fraction of the primary reactor coolant activity (PCA). The primary coolant activity (PCA) is based on the recommendations of American National Standard (ANSI N237) Source Term Specification, (Ref. 1), with the changes as noted in Section 1.2 under the Reactor 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 BWRs 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 radioactivity content of the demineralizer regenerant solution is obtained by considering that all of the activity that is collected by the condensate demineralizers 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 having an onsite laundry, a standard detergent source term, adjusted for the treatment provided, is added to the adjusted source term.

1.5 INSTRUCTIONS FOR COMPLETING BWR-GALE CODE INPUT DATA CARDS

1.5.1 PARAMETERS INCLUDED IN THE BWR-GALE CODE

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

1.5.1.1 Plant Capacity Factor

0.80 (292 effective full power days per year)

1.5.1.2 Radionuclide Concentrations in the Reactor Coolant and Main Steam

See Chapter 2, Tables 2-2 through 2-5 of this document.

1.5.1.3 Noble Gas, Radioiodine, and Particulate Releases From Building Ventilation Systems Prior to Treatment

See Tables 1-1 and 1-2. For a discussion of the normalization techniques see Section 2.2.4.

1.5.1.4 Radioiodine Input Rate to Main Condenser Offgas System

6 Ci/yr per reactor downstream of main condenser air ejectors.

1.5.1.5 Main Condenser Vacuum Pump Release

Xe-133 -- 1300 Ci/yr

Xe-135 -- 500 Ci/yr

I-131 -- See Table 1-3

1.5.1.6 Charcoal Delay Systems

For a charcoal delay system used to treat the offgases from the main condenser air ejector, the BWR-GALE Code calculates the holdup times for Kr and Xe. Iodine releases from charcoal delay systems are negligible due to the large quantities of charcoal used in the system. The holdup times for noble gases are calculated by the Code using the following equation and the data entered on Cards 29-32.

$$T = 43.1 \frac{MK}{P}$$

where

K is the dynamic adsorption coefficient, in cm^3/gm (see chart on page 2-35);

M is the mass of charcoal, in 10^3 lbs

T is the holdup time, in hr, and

P is the thermal power level (MWt) entered in Card 2.

TABLE 1-1
GASEOUS RELEASES FROM VENTILATION SYSTEMS PRIOR TO TREATMENT
(in Ci/yr per Reactor)

NUCLIDE	CONTAINMENT BUILDING	AUXILIARY BUILDING	TURBINE BUILDING	RADWASTE BUILDING
Kr-83m	**	**	**	**
Kr-85m	1	3	25	**
Kr-85	**	**	**	**
Kr-87	**	2	61	**
Kr-88	1	3	91	**
Kr-89	**	2	580	29
Xe-131m	**	**	**	**
Xe-133m	**	**	**	**
Xe-133	27	83	150	220
Xe-135m	15	45	400	530
Xe-135	33	94	330	280
Xe-137	45	135	1000	83
Xe-138	2	6	1000	2
Cr-51*	0.0002	0.0009	0.0009	0.0007
Mn-54	0.0004	0.001	0.0006	0.004
Fe-59	0.00009	0.0003	0.0001	0.0003
Co-58	0.0001	0.0002	0.001	0.0002
Co-60	0.001	0.004	0.001	0.007
Zn-65	0.001	0.004	0.006	0.0003
Sr-89	0.00003	0.00002	0.006	NA
Sr-90	0.000003	0.000007	0.00002	NA
Zr-95	0.0003	0.0007	0.00004	0.0008
Nb-95	0.001	0.009	0.000006	0.000004
Mo-99	0.006	0.06	0.002	0.000003
Ru-103	0.0002	0.004	0.00005	0.000001
Ag-110m	0.0000004	0.000002	NA	NA
Sb-124	0.00002	0.00003	0.0001	0.00007
Cs-134	0.0007	0.004	0.0002	0.0024
Cs-136	0.0001	0.0004	0.0001	NA
Cs-137	0.001	0.005	0.001	0.004
Ba-140	0.002	0.02	0.010	0.000004
Ce-141	0.0002	0.0007	0.010	0.000007

*Particulate release rates are prior to filtration.

**Less than 1 Ci/yr per reactor.

NA Not Analyzed; analysis for the isotope was not performed.

TABLE 1-2

RADIOIODINE RELEASES FROM BUILDING VENTILATION SYSTEMS PRIOR TO TREATMENT (Ci/yr/μCi/gm)

	Containment Bldg**	Auxiliary Bldg**	Turbine Bldg***	Radwaste Bldg**
Annual Normalized* Iodine Release Rate				
Power Operation	1.2	11.1	3.8×10^3	4.6
Refueling/Maintainence Outages	4.7	0.5	4.1×10^2	1.4

*The normalized release rate, expressed in grams of water during the 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 radiodine between the water phase in the leakage and the gas phase where it is measured. For the turbine building the effective leak rate also includes the carryover for radioiodine from reactor water to steam in the reactor vessel.

**To obtain the actual iodine release from these bldgs in Ci/yr, multiply the normalized release by the coolant concentration in μCi/gm.

***To obtain the actual iodine release from the turbine building in Ci/yr, multiply the normalized release by the coolant concentration in μCi/gm and by the iodine carryover from Table 2-4.

TABLE 1-3

RADIOIODINE RELEASES FROM MECHANICAL VACUUM PUMP (Ci/yr/μCi/gm)

	Annual Normalized* Iodine Release Rate**
Short-term outages	4.9×10^2
Refueling/Maintenance Outages	1.1×10^3

*The normalized release rate, expressed in grams of water during the modes of operation, represents the effective leak rate for radioiodine. It is the combination of the reactor water leakage rate, the partitioning of the radioiodine between the water phase in the leakage and the gas phase where it is measured and the carryover for radioiodine from reactor water to steam in the reactor vessel.

**To obtain the actual iodine release from the mechanical vacuum pump in Ci/yr, multiply the normalized release by the coolant concentration in μCi/gm and by the iodine carryover from Table 2-4.

1.5.1.7 Cryogenic Distillation System

For a cryogenic distillation system, the BWR-GALE Code uses a partition coefficient of 0.0001 for Xe and I and 0.00025 for Kr to calculate Xe, I, and Kr removal during separation by distillation. The Xe, I, and Kr separated by distillation are considered to be released following 90-day holdup. The calculated releases are the sum of the noble gases and radioiodine released from the overheads during distillation without holdup and the noble gases and iodine released following 90-day holdup.

1.5.1.8 Decontamination Factors for Condensate Demineralizers

Demineralizer	Anions	Cs, Rb	Other Nuclides
Deep bed	10	2	10
Powdex	10	2	10

1.5.1.9 Detergent Wastes

The radionuclides listed in Table 2-28 of Chapter 2 are assumed to be released unless treatment is provided or laundry is not processed on site.

1.5.1.10 Tritium Releases

Total tritium release equals 0.03 Ci/yr per MWt. The quantity of tritium released through the liquid pathway is 50% of the total quantity calculated to be available for release, and 50% is calculated to be released in gaseous effluents. Of that released in gaseous effluents, half is released from the turbine building ventilation system and half is released from the containment building ventilation system.

1.5.1.11 Argon-41 Releases

The argon-41 input to the main condenser offgas treatment system is 40 μCi/sec. The dynamic adsorption coefficients for argon-41 in charcoal delay beds are 6.4 cm^3/gm and 16 cm^3/gm for ambient and chilled temperature systems, respectively. The argon-41 release from purging or venting of the drywell is 15 Ci/yr.

1.5.1.12 Regeneration of Condensate Demineralizers

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

1. Liquid radioactivity flow to the demineralizer is based on the radioactivity of the main steam and the fraction of radioactivity which does not bypass the condensate demineralizers in the pumped foward flow.

2. All radionuclides removed from the condensate by the demineralizers are removed from the demineralizer resins during chemical regeneration. The regenerant waste radioactivity is adjusted for radionuclide decay during operation of the demineralizers.

1.5.1.13 Adjustment to Liquid Radwaste Source Terms for Anticipated Operational Occurrences

1. The calculated source term is increased by 0.1 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 alternative 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. 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 not available.

1.5.2 PARAMETERS REQUIRED FOR THE BWR-GALE CODE

The parameters described in the following sections must be entered on input data cards. Complete the cards designated 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.

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

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

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: Total Steam Flow Rate (SAR/ER)

Enter in spaces 73-80 the total steam flow rate from the reactor (in 10^6 lbs/hr).

1.5.2.4 Card 4: Mass of Coolant in Reactor Vessel (SAR/ER)

Enter in spaces 73-80 the mass of water in the reactor vessel and recirculation lines (in 10^6 lbs).

1.5.2.5 Card 5: Cleanup Demineralizer Flow (SAR/ER)

Enter in spaces 73-80 the reactor coolant flow rate (in 10^6 lbs/hr) through the reactor coolant cleanup system demineralizers.

1.5.2.6 Card 6: Condensate Demineralizer Regeneration Time

For deep-bed condensate demineralizers, use a 3.5-day regeneration frequency. If ultrasonic resin cleaning is used, assume 8-day regeneration frequency. Multiply the frequency by the total number of demineralizers and enter the calculated number of days in spaces 73-80. For filter/demineralizers (Powdex), enter 0.0 in spaces 73-80.

1.5.2.7 Card 7: Fraction of Feedwater Through Condensate Demineralizer (SAR/ER)

Enter in spaces 73-80 the fraction of feedwater processed through the condensate demineralizers.

1.5.2.8 Cards 8-19: Liquid Radwaste Treatment System Input Parameters

Four liquid radwaste inlet streams are considered in the BWR-GALE Code (see Section 1.5.2.22 for detergent wastes):

1. High-Purity Waste, Cards 8-10
2. Low-Purity Waste, Cards 11-13
3. Chemical Waste, Cards 14-16
4. Regenerant Solutions Waste, Cards 17-19

Three input data cards are used to define the major parameters for each of the four waste streams. Essentially the same information is needed on the three input data cards used for each of the four streams. The instructions given in this section are applicable to all four waste streams, with the following exception: the inlet waste activity is not entered on Card 17 for the regenerant solutions wastes for systems using regenerable condensate demineralizers since that activity is calculated by the Code.

The entries required on the first card (8, 11, and 14) for the High-Purity, Low-Purity, and Chemical Waste Systems, respectively, are outlined below and described in more detail in Section 1.5.2.8.1.

1. Enter in spaces 18-41 the name of the waste inlet stream (e.g., high-purity wastes).

2. Enter in spaces 42-49 the flow rate (in gal/day) of the inlet stream.

3. Enter in spaces 57-61 the activity of the inlet stream expressed as a fraction of the primary coolant activity (PCA).

On the first card for the Regenerant Solutions Waste System (i.e., Card 17), enter in spaces 73-80 the flow rate of the regenerant solutions waste inlet stream. For the calculation of liquid effluents for regeneration of demineralizers other than the condensate demineralizers, see Appendix A.

The second card (9, 12, 15, and 18) for each waste stream contains the overall system decontamination factors for three categories of radionuclides, as follows:

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

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 (10, 13, 16, and 19) for each waste stream:

1. Enter in spaces 29-33 the waste collection time (in days) prior to processing.

2. Enter in spaces 48-53 the sum of the waste processing and discharge time (in days).

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

The following sections explain in more detail the use of the parameters in this document and the information given in the SAR/ER to make the data entries in Cards 8-19 listed above.

1.5.2.8.1 Liquid Waste Flow Rates and Activities (Cards 8, 11, 14, and 17)

Calculate flow rates and activities to complete the first card for each liquid radwaste inlet stream by using the waste volumes and activities given in Table 1-4. To the input flow rates and activity given in the table, add expected flows and activities more specific to the plant design as given in the SAR/ER. The inlet streams should be combined to form the four principal waste streams (high-purity, low-purity, chemical wastes, and regenerant wastes) considered in this document. Calculate the primary coolant activity (PCA) of each of the four principal inlet streams (except for the regenerant waste as indicated above) by determining the weighted average activity of the composite stream entering the waste collection tanks. For example, if inlet streams A, B, and C enter the low-purity waste collector tank at average rates and PCA as listed below:

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

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

$$\frac{(1000\ \text{gal/day})(0.01\ \text{PCA}) + (2000\ \text{gal/day})(0.1\ \text{PCA}) + (500\ \text{gal/day})(1.0\ \text{PCA})}{(1000\ \text{gal/day} + 2000\ \text{gal/day} + 500\ \text{gal/day})} = 0.2\text{PCA}$$

The entries on Card 11 for this example would then be: spaces 18-41, "Low-Purity Waste"; spaces 42-49, "3500"; spaces 57-61, "0.2."

TABLE 1-4
BWR LIQUID WASTES

SOURCE	EXPECTED DAILY AVERAGE INPUT FLOW RATE (in gal/day): DEEP BED PLANT WITH ULTRASONIC RESIN CLEANER	DEEP BED PLANT WITHOUT ULTRASONIC RESIN CLEANER OR A FILTER/DEMINERALIZER PLANT	FRACTION OF THE PRIMARY COOLANT ACTIVITY (PCA)
Equipment Drains			
Drywell	3,400	3,400	1.00
Containment, auxiliary building, and fuel pool	3,700	3,700	0.1
Radwaste building	1,100	1,100	0.1
Turbine building	3,000	3,000	0.001
Ultrasonic resin cleaner	15,000	-	0.05
Resin rinse*	2,500	5,000	0.002
Floor Drains			
Drywell	700	700	0.001
Containment, auxiliary building, and fuel handling	2,000	2,000	0.001
Radwaste building	1,000	1,000	0.001
Turbine building	2,000	2,000	0.001
Other Sources			
Cleanup phase separator decant	640	640	0.002
Laundry drains	1,000	1,000	-
Lab drains	500	500	0.02
Regenerants*	1,700	3,400	**
Condensate demineralizer backwash†	-	8,100	2×10^{-6}
Chemical lab waste	100	100	0.02

*Deep-bed condensate demineralizers only.

**Calculated by BWR-GALE Code.

†Filter/demineralizer (Powdex) condensate demineralizers only.

The input flows and activities are entered in units of gal/day and fraction of PCA, respectively.

1.5.2.8.2 Decontamination Factors for Equipment Used in the Liquid Radwaste Treatment System (Cards 9, 12, 15, and 18)

The system decontamination factors (DFs) should be entered in the second card for each liquid radwaste inlet stream. The DFs represent the expected equipment performance averaged over the life of the plant. The following factors are to be considered in calculating overall decontamination factors for the various systems.

1. DFs are categorized by one of the following types of radionuclides:

   a. Anions

   b. Cs, Rb

   c. Other nuclides

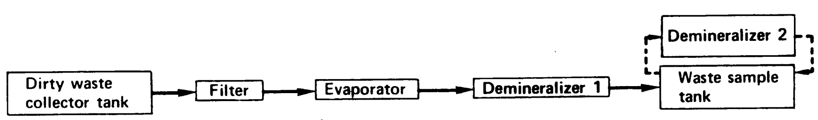
Note: A DF of 1 is assumed by the BWR-GALE Code for tritium. Dissolved noble gases and water activation products are not considered in the liquid code.

2. The system DF for each inlet stream is the product of the individual equipment DFs 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 1-5 shows the decontamination factors to be used for BWR liquid waste treatment systems.

The following example illustrates the calculation of the decontamination factor for a low-purity waste treatment system: Assume that low-purity 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 schematically as:



Extracting from Table 1-5 gives the following values for the example:

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

These values were obtained as follows:

- A DF of 1.0 was applied to all nuclides for the filter.

- A DF of 10^3 for anions and 10^4 for Cs, Rb, and other nuclides was applied for the radwaste evaporator.

- A DF of 10 was applied for anions, Cs, Rb, and other nuclides for the evaporator condensate polishing demineralizer.

- A DF of 1 was applied to the second demineralizer since this demineralizer's use is optional and it is not used for normal operations.

- The product of the DFs was obtained by combining the first four columns for each radionuclide.

TABLE 1-5
DECONTAMINATION FACTORS FOR BWR LIQUID WASTE TREATMENT SYSTEMS

TREATMENT SYSTEM	DECONTAMINATION FACTOR		
Demineralizers	Anion	Cs, Rb	Other Nuclides
Mixed-bed			
Reactor Coolant Cleanup	10	2	10
Condensate (deep bed)	10	2	10
High-purity waste	10^2(10)*	10(10)	10^2(10)
Low-Purity Waste	10^2(10)	2(10)	10^2(10)
Cation bed (any system)	1(1)	10(10)	10^2(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 Anions		Anions
Miscellaneous	10^4		10^3
Detergent wastes	10^2		10^2
Reverse Osmosis	All Nuclides		
Laundry wastes	30		
Other liquid wastes	10		
Filters	DF of 1.0 for all nuclides		

*For an evaporator polishing demineralizer or for the second demineralizer in series, the DF is given in parentheses.

Thus in Card 9 the following would be entered: in spaces 21-28, "10,000"; in spaces 34-41, "100,000"; and in spaces 47-54, "100,000."

1.5.2.8.3 Collection Time for Liquid Wastes (Cards 10, 13, 16, 19 -- Spaces 29-33)

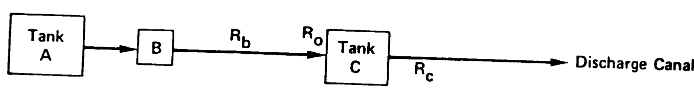
Collection time prior to processing is based on the input flow calculated above. Where redundant tanks are provided, assume the collection tank will be processed when filled to 80% capacity. If only one tank is provided, assume the tank will be processed when filled to 40% capacity. For example, if flow 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:

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

Then, for this example, "16" should be entered in spaces 29-33 on Card 13.

1.5.2.8.4 Processing and Discharge Time (Cards 10, 13, 16, 19 -- Spaces 48-53)

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



where

- A is the capacity of the initial tank in the flow scheme, in gal;
- B is the limiting process based on equipment flow capacity, dimensionless;
- C is the capacity of the final tank in the 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 the Tank C discharge pump, in gal/day; and
- R_o is the rate of flow of additional wastes inputs to Tank C, in gal/day.

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

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

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 processing and discharge by determining whether

$$0.8C > T_p(R_b + R_o) \text{ for redundant tanks, or } 0.4C > T_p(R_b + R_o) \text{ for a single tank}$$

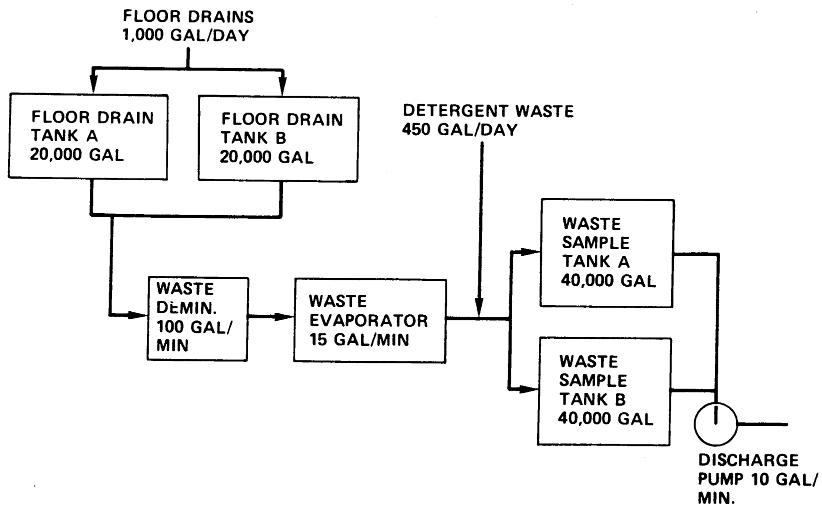
If so, then

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

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 10, 13, 16, and 19).

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 would be calculated as follows:

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

$$\text{Discharge Time } (T_d) = \frac{(0.8)(40{,}000 \text{ gal})}{(10 \text{ gal/min})(1440 \text{ min/day})} = 2 \text{ days}$$

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 1.7 days for processing and discharge. The input on spaces 48-53 to the Code is 1.7 days for processing and discharge time.

1.5.2.8.5 Fraction of Wastes Discharged (Cards 10, 13, 16, and 19 -- Spaces 72-77)

The percent of the wastes discharged after processing may vary between 1% and 100% 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 1% discharge for high-purity wastes and 10% discharge for other wastes is used when the radwaste system is designed for maximum waste recycle, the system capacity is sufficient to process wastes for reuse during equipment downtime and anticipated operational occurrences, and a discharge route is provided.

The BWR-GALE Code calculates the release of radioactive materials in liquid waste from the four inlet streams after processing. Releases included in each stream are:

1. High-Purity Waste - Combined releases from equipment drains and sumps.

2. Low-Purity Waste - Combined releases from floor drains and sumps.

3. Chemical Waste - Combined releases from laboratory and decontamination wastes and from demineralizer regenerant solutions according to the design of the condensate demineralizer system. If a filter/demineralizer (Powdex) system is used, the laboratory and decontamination wastes are combined with the low-purity waste or solidified in the solid waste system.

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

1.5.2.9 Card 20: <u>Gland Seal Steam Flow</u>

Enter in spaces 73-80 of Card 20 the steam flow (in 10^3 lbs/hr) to the turbine gland seal, as follows:

1. If main steam is used for the sealing steam, enter a flow rate 0.001 times the main steam flow entered previously on Card 3.

2. If clean (nonradioactive) steam from an auxiliary boiler is used for sealing steam, enter 0.0 in spaces 73-80.

1.5.2.10 Card 21: Gland Seal Holdup Time (SAR/ER)

Enter in spaces 73-80 the design holdup time (in hr) for gases vented from the gland seal condenser.

1.5.2.11 Card 22: Holdup Time for Condenser Air Ejector Offgas (SAR/ER)

Enter in spaces 73-80 the design holdup time (in hr) for offgases from the main condenser air ejector prior to being processed through the offgas treatment system, e.g., a 10-minute holdup time prior to cryogenic distillation.

1.5.2.12 Card 23: Containment Building Releases

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 43-46 for radioiodine corresponding to the depth of charcoal as indicated in Table 1-6.

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 53-56.

3. If no treatment is provided for the ventilation exhaust air to remove radioiodine or if the charcoal adsorbers do not satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter 0.0 in spaces 43-46; if no treatment is provided 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 53-56.

1.5.2.13 Card 24: Turbine Building Releases

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 43-46 for radioiodine corresponding to the depth of charcoal as indicated in Table 1-6.

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 53-56.

3. If no treatment is provided for the ventilation exhaust air to remove radioiodine or if the charcoal adsorbers do not satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter 0.0 in spaces 43-46; if no treatment is provided 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 53-56.

1.5.2.14 Card 25: Fraction of Radioiodine Released from Turbine Gland Seal Condenser Vent

1. If, prior to release, the offgases from the turbine gland seal condenser vent 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 1-6.

2. If the offgases are released from the turbine gland seal condenser without treatment, if clean steam is used, or if charcoal adsorbers provided do not satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter 0.0 in spaces 73-80.

TABLE 1-6

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 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).

1.5.2.15 Card 26 Fraction of Radioiodine Released from the Condenser Air Ejector Offgas Treatment System

1. If, prior to release, the offgases 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 1-6.

2. If the offgas is released without treatment or through charcoal adsorbers that do not satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter 0.0 in spaces 73-80.

3. Enter a 1. in spaces 73-80 if the offgas is processed through a charcoal delay system.

4. If the offgas is processed through a cryogenic distillation system (removal of iodine by the cryogenic distillation system is built into the Code -see Card 29), enter 0.0 in spaces 73-80.

1.5.2.16 Card 27: Auxiliary Building Releases

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 43-46 for radioiodine corresponding to the depth of charcoal as indicated in Table 1-6.

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 53-56.

3. If no treatment is provided for the ventilation exhaust air to remove radioiodine or if the charcoal adsorbers do not satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter 0.0 in spaces 43-46; if no treatment is provided 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 53-56.

1.5.2.17 Card 28: Radwaste Building Releases

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 43-46 for radioiodine corresponding to the depth of charcoal as indicated in Table 1-6.

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 53-56.

3. If no treatment is provided for the ventilation exhaust air to remove radioiodine or if the charcoal adsorbers do not satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter 0.0 in spaces 43-46; if no treatment is provided 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 53-56.

1.5.2.18 Card 29: Condenser Air Ejector Offgas Treatment System (SAR/ER)

1. Enter 1 in space 80 if a charcoal delay system is used to treat the offgas from the condenser air ejector.

2. Enter 2 in space 80 if the offgas from the condenser air ejector is processed by a cryogenic distillation system.

3. Enter a zero in space 80 if the offgas is not treated either through a charcoal delay system or by cryogenic distillation.

Note: Enter 0.0 on Cards 30, 31, and 32 if a charcoal delay system is not used to treat the offgases from the condenser air ejector.

1.5.2.19 Card 30: Dynamic Adsorption Coefficient for Krypton

Enter in spaces 73-80 the dynamic adsorption coefficient for Kr based on the system design and the dynamic adsorption coefficients noted below.

	DYNAMIC ADSORPTION COEFFICIENT (cm^3/gm)			
	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.

1.5.2.20 Card 31: Dynamic Adsorption Coefficient for Xenon

Enter in spaces 73-80 the dynamic adsorption coefficient for Xe based on the system design and dynamic adsorption coefficients noted below.

	DYNAMIC ADSORPTION COEFFICIENT (cm^3/gm)			
	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
Xe	330.	440.	1160.	2410.

1.5.2.21 Card 32: Mass of Charcoal in Charcoal Delay System (SAR/ER)

Enter in spaces 73-80 the mass of charcoal (in 10^3 lbs) used in the charcoal delay system.

1.5.2.22 Card 33: Detergent Waste

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 decontamination factor in spaces 73-80. The parameters in Chapter 2 are used in determining the DF for the treatment applied to detergent waste.

CHAPTER 2. PRINCIPAL PARAMETERS USED IN BWR 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 BWR-GALE Code are 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 BWR-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 BWRs 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 REACTOR COOLANT

2.2.3.1 Parameter

As used in the BWR-GALE Code, Table 2-2 lists the expected radionuclide concentrations in the reactor coolant and steam for BWRs with design parameters within the ranges listed in Table 2-3. Should any design parameter be outside the ranges in Table 2-3, the BWR-GALE Code adjusts the concentrations in Table 2-2, using the factors in Tables 2-4 and 2-5. Figure 2-1 shows the graphical relationship of the design parameters.

TABLE 2-1

PLANT CAPACITY FACTORS AT OPERATING BWRs[a]

FACILITY[b]	DATE OF COMMERICAL OPERATION[c]	1972	1973	1974	1975	1976	1977
Oyster Creek	12/69	77	65	66	58	70	58[e]
Nine Mile Point-1	12/69	62	68	63	60	81	57[f]
Millstone-1	03/71	55	34[d]	63	68	66	84
Monticello	06/71	74	68	57	61	84	75
Dresden-3	11/71	67	54	47[e]	33[f]	60	76
Dresden-2	06/72		74	51	44[f]	66	54
Vermont Yankee	11/72		44[e]	59	81	73	80
Pilgrim-1	12/72		72	34[f]	46[e]	43[f]	47[f]
Quad Cities-1	02/73			51[f]	65	52[f]	55[e]
Quad Cities-2	03/73			68	40[f]	66	67
Cooper	07/74				60	57	70
Peach Bottom-2	07/74				57	61	45[f]
Peach Bottom-3	12/74				59	67	54[f]
Duane Arnold	02/75					55	67
Fitzpatrick	07/75					59	55[f]
Brunswick-2	11/75					37[g]	35[g]
Hatch-1	12/75					65	57
AVERAGE		67	67	61	63	66	70

[a]From Semi-Annual Operating Reports for each facility, as submitted by respective licensees.

[b]Big Rock Point, Dresden 1, Humboldt Bay, and Lacrosse are not included since they are small reactors (< 700 MWt) and are not considered to be typical of modern-day reactors. Browns Ferry 1, 2 are not included since they were not operating due to fire.

[c]Plant capacity factors listed begin with the first full year of commercial operation.

[d]Not included due to extended maintenance outage to replace feedwater sparger.

[e]Not included due to extended operation at reduced power.

[f]Not included due to extended refueling outage.

[g]Not included due to extended maintenance outage to correct power monitor tube vibrations.

TABLE 2-2
RADIONUCLIDE CONCENTRATIONS
IN BOILING WATER REACTOR COOLANT AND MAIN STEAM*
(in μCi/gm)

ISOTOPE	REACTOR COOLANT	REACTOR STEAM
Noble Gases		
Kr-83m		9.1(-4)**
Kr-85m		1.6(-3)
Kr-85		5.0(-6)
Kr-87		5.5(-3)
Kr-88		5.5(-3)
Kr-89		3.4(-2)
Kr-90		7.5(-2)
Kr-91		9.1(-2)
Kr-92		9.1(-2)
Kr-93		2.4(-2)
Kr-94		5.9(-3)
Kr-95		5.5(-4)
Kr-97		3.6(-6)
Xe-131m		3.9(-6)
Xe-133m		7.5(-5)
Xe-133		2.1(-3)
Xe-135m		7.0(-3)
Xe-135		6.0(-3)
Xe-137		3.9(-2)
Xe-138		2.3(-2)
Xe-139		7.5(-2)
Xe-140		8.0(-2)
Xe-141		6.5(-2)
Xe-142		1.9(-2)
Xe-143		3.2(-3)
Xe-144		1.5(-4)
Halogens		
Br-83	6(-3)	9(-5)***
Br-84	7(-3)	1(-4)
Br-85	3(-3)	5(-5)
I-131	3.7(-3)	6(-5)
I-132	6(-2)	9(-4)
I-133	5(-2)	8(-4)
I-134	1(-1)	2(-3)
I-135	5(-2)	8(-4)
Cesium and Rubidium		
Rb-89	5(-3)	5(-6)
Cs-134	3(-5)	3(-8)
Cs-136	2(-5)	2(-8)
Cs-137	8(-5)	8(-8)
Cs-138	1(-2)	1(-5)

*The reactor coolant concentration is specified at the nozzle where reactor water leaves the reactor vessel. Similarly, the reactor steam concentration is specified at time 0 at the nozzle.

**1.1(-3) = 1.1×10^{-3}.

***Halogen concentrations listed in reactor steam are based on a carryover of 0.015. For a carryover of 0.004 the halogen reactor steam concentrations would be reduced proportionately.

TABLE 2-2 (Continued)

ISOTOPE	REACTOR COOLANT	REACTOR STEAM
Water Activation Products		
N-13	5(-2)	7(-3)
N-16	6(+1)	5(+1)
N-17	9(-3)	2(-2)
O-19	7(-1)	2(-1)
F-18	4(-3)	4(-3)
Tritium*		
H-3	1(-2)	1(-2)
Other Nuclides		
Na-24	1(-2)	1(-5)
P-32	2(-4)	2(-7)
Cr-51	6(-3)	6(-6)
Mn-54	7(-5)	7(-8)
Mn-56	5(-2)	5(-5)
Fe-55	1(-3)	1(-6)
Fe-59	3(-5)	3(-8)
Co-58	2(-4)	2(-7)
Co-60	4(-4)	4(-7)
Ni-63	1(-6)	1(-9)
Ni-65	3(-4)	3(-7)
Cu-64	3(-2)	3(-5)
Zn-65	2(-4)	2(-7)
Zn-69	2(-3)	2(-6)
Sr-89	1(-4)	1(-7)
Sr-90	7(-6)	7(-9)
Sr-91	4(-3)	4(-6)
Sr-92	1(-2)	1(-5)
Y-91	4(-5)	4(-8)
Y-92	6(-3)	6(-6)
Y-93	4(-3)	4(-6)
Zr-95	8(-6)	8(-9)
Zr-97	6(-6)	6(-9)
Nb-95	8(-6)	8(-9)
Nb-98	4(-3)	4(-6)
Mo-99	2(-3)	2(-6)
Tc-99m	2(-2)	2(-5)
Tc-101	9(-2)	9(-5)
Tc-104	8(-2)	8(-5)
Ru-103	2(-5)	2(-8)
Ru-105	2(-3)	2(-6)
Ru-106	3(-6)	3(-9)
Ag-110m	1(-6)	1(-9)
Te-129m	4(-5)	4(-8)
Te-131m	1(-4)	1(-7)

*Measured values increased to account for liquid recycle.

TABLE 2-2 (Continued)

ISOTOPES	REACTOR COOLANT	REACTOR STEAM
Te-132	1(-5)	1(-8)
Ba139	1(-2)	1(-5)
Ba140	4(-4)	4(-7)
Ba-141	1(-2)	1(-5)
Ba-142	6(-3)	6(-6)
La-142	5(-3)	5(-6)
Ce-141	3(-5)	3(-8)
Ce-143	3(-5)	3(-8)
Ce-144	3(-6)	3(-9)
Pr-143	4(-5)	4(-8)
Nd-147	3(-6)	3(-9)
W-187	3(-4)	3(-7)
Np-239	8(-3)	8(-6)

TABLE 2-3
PARAMETERS USED TO DESCRIBE THE REFERENCE BOILING WATER REACTOR

PARAMETER	SYMBOL	UNITS	NOMINAL VALUE	RANGE MAXIMUM	RANGE MINIMUM
Thermal power	P	MWt	3400	3800	3000
Weight of water in the reactor vessel	WP	lb	3.8(5)*	4.2(5)	3.4(5)
Cleanup demineralizer flow rate	FA	lb/hr	1.3(5)	1.5(5)	1.1(5)
Steam flow rate	FS	lb/hr	1.5(7)	1.7(7)	1.3(7)
Ratio of condensate demineralizer flow rate to steam flow rate	NC**	-	0.75	0.99	0.5

*3.8(5) = 3.8 x 10^5

**For a BWR that is within the range indicated above, i.e. a BWR with pumped forward feedwater heater drains, the value for NC used in the BWR-GALE Code is 0.18 for iodine and 0.01 for Cs, Rb and other nuclides, as discussed on page 2-11. For a BWR that has a ratio of condensate demineralizer flow rate to steam flow rate equal to 1.0, i.e., full flow condensate demineralizers, a value of NC=1.0 is used in the BWR-GALE Code.

TABLE 2-4
VALUES USED IN DETERMINING ADJUSTMENT FACTORS FOR BOILING WATER REACTORS

SYMBOL	DESCRIPTION	NOBLE GASES	HALOGENS	Cs, Rb	WATER ACTIVATION PRODUCTS	TRITIUM	OTHER NUCLIDES
NA	Fraction of material removed in the reactor water cleanup system	0.0	0.9	0.5	0.0	0.0	0.9*
NB	Fraction of material removed by the condensate demineralizers	0.0	0.9	0.5	0.0	0.0	0.9*
NS	Ratio of concentration in reactor steam to the concentration in reactor water	**	0.015†††	0.001	***	1.0	0.001
R	Removal rate from the reactor water (hr^{-1}).	**	0.40	0.17	***	††	0.31

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

** All noble gases released from the core are transported rapidly out of the reactor water to the reactor steam and are stripped from the system in the main condenser. Therefore the concentration in the reactor water is negligible and the steam concentration is approximately equivalent to the ratio of the release rate and the steam flow rate.

*** Water activation products exhibit varying chemical and physical properties in reactor coolan which are not well defined. However, most are stripped off as gases. They are not effectively removed by the demineralizers of the systems, but their concentrations are controlled by decay.

† These values of R apply to the reference BWR whose parameters are given in Table 2-3 and have been used in developing Table 2-5. For BWRs not included in Table 2-3, the appropriate value for R is determined by the BWR-GALE Code using the following equation:

$$R = \frac{FA \cdot NA + NC \cdot FS \cdot NS \cdot NB}{WP} \text{ for halogens, Cs, Rb, and other nuclides}$$

where the symbols are defined in this table, Table 2-3 and Figure 2-1. The values for R for noble gases and water activation products are not used in the adjustment factors of Table 2-5.

†† The tritium concentrations in the reactor water and the steam are expected to be equal. They are controlled by loss of water from the main coolant system by evaporation or leakage.

††† The value of 0.015 is used for BWRs which have Deep Bed Condensate Treatment. A value of 0.015 is also used for BWRs with Powdex Condensate Treatment and stainless steel condenser tubing. For BWRs which have Powdex Condensate Treatment systems and copper condenser tubing, a value of 0.004 should be used.

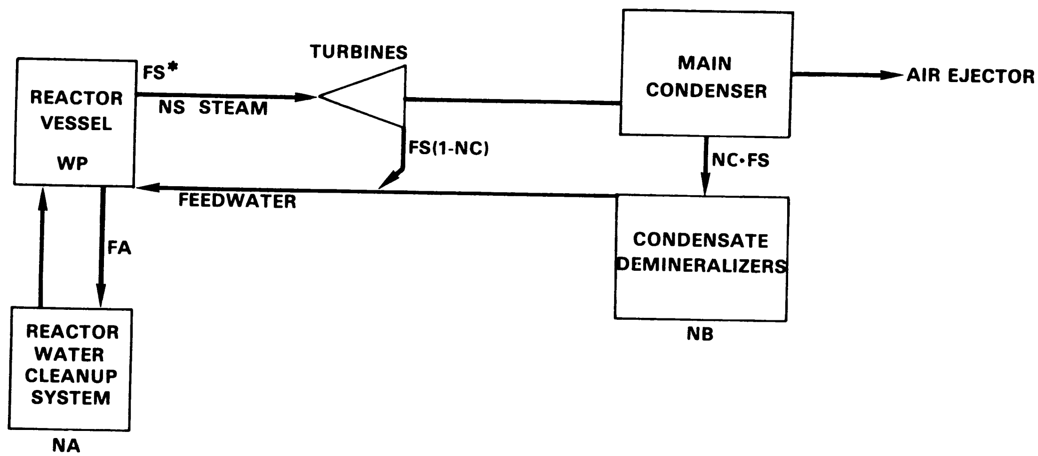
TABLE 2-5
ADJUSTMENT FACTORS FOR BOILING WATER REACTORS

NUCLIDES	REACTOR COOLANT	REACTOR STEAM
Noble gases*	1.0	1.0
Halogens**	$\frac{P}{WP}(110\ \frac{lb}{MWt})\frac{0.40 + \lambda}{R + \lambda}$	$\frac{P}{WP}(110\ \frac{lb}{MWt})\frac{0.40 + \lambda}{R + \lambda}$
Cs, Rb	$\frac{P}{WP}(110\ \frac{lb}{MWt})\frac{0.17 + \lambda}{R + \lambda}$	$\frac{P}{WP}(110\ \frac{lb}{MWt})\frac{0.17 + \lambda}{R + \lambda}$
Water activation products	1.0	1.0
Tritium***	1.0	1.0
Other nuclides	$\frac{P}{WP}(110\ \frac{lb}{MWt})\frac{0.31 + \lambda}{R + \lambda}$	$\frac{P}{WP}(110\ \frac{lb}{MWt})\frac{0.31 + \lambda}{R + \lambda}$

*Assumes that the ratio of power to steam flow is essentially the same for all BWRs.

**λ is the isotope's decay constant (hr^{-1}).

***The tritium concentrations in the reactor coolant and the steam are expected to be equal. They are controlled by loss of water from the main coolant system by evaporation or leakage. The concentration is therefore given by the ratio of the appearance rate in the coolant, which is about 100 Ci/yr, and the total loss from the system.



* SYMBOLS ARE DEFINED IN TABLES 2-3 AND 2-4

FIGURE 2-1
REMOVAL PATHS FOR THE REFERENCE
BOILING WATER REACTOR

2.2.3.2 Bases

The radionuclide concentrations, adjustment factors, and procedures for effecting adjustments are based on the values and methods in American National Standard ANSI N237, Source Term Specification, (Ref. 1) with the changes noted in Section 1.2 under the Reactor Coolant definitions. The values in Table 2-2 provide a set of typical radionuclide concentrations in the reactor coolant and steam for reactor designs within the parameters specified in Table 2-3. The values in Table 2-2 were those determined to be representative of radionuclide concentrations in a BWR over its lifetime based on the currently available data and models (Refs. 3 and 4). It is recognized that some systems will have design parameters that are outside the ranges specified in Table 2-3. For that reason a means of adjusting the concentrations to the actual design parameters has been provided in the BWR-GALE Code based on factors presented in Tables 2-4 and 2-5. The adjustment factors in Tables 2-4 and 2-5 are based on the following expression:

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

where

C is the specific activity, in μCi/gm;

k is a conversion factor, 454 gm/lbs;

R is the removal rate of the isotope from the system due to demineralization, leakage, etc., in hr^{-1};

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

λ is the decay constant, in hr^{-1}.

The following sample calculations illustrate the method by which the BWR-GALE Code will adjust the radionuclide concentrations in Table 2-2. As indicated in Table 2-5, adjustment factors will be calculated only for halogens, Cs, Rb, and other nuclides.

As an example, the sample case parameters shown below compare with the range of values in Table 2-3 as follows:

Parameter	Sample Case Value	Range Values
Thermal power level (MWt)	3758	3000-3800
Water weight in vessel (lbs)	4.9×10^5	$3.4 \times 10^5 - 4.2 \times 10^5$
Cleanup demineralizer flow (lbs/hr)	1.5×10^5	$1.1 \times 10^5 - 1.5 \times 10^5$
Steam flow rate (lbs/hr)	15.4×10^6	$13.0 \times 10^6 - 17.0 \times 10^6$
Condensate demineralizer flow fraction	0.75	0.5 - 0.99

Since in this example one of the parameters (water weight in vessel) is outside the range, adjusted values of the three types of radionuclide concentrations are calculated using the actual value of each parameter, as follows:

1. Halogens (I-131 is used as an example) -- Using the equation for halogens in Table 2-5, the adjustment factor A is calculated as follows:

$$A = \frac{P}{WP} (110) \frac{0.40 + \lambda}{R + \lambda} \qquad (2\text{-}1)$$

where the terms in the equation are as defined in Tables 2-3 and 2-4.

In calculating A, the variable R is calculated first, using the equation given in Table 2-4:

$$R = \frac{FA\ NA + NC\ FS\ NS\ NB}{WP} \qquad (2\text{-}2)$$

where the terms in the equation are as defined in Tables 2-3 and 2-4.

Using the sample case parameters given above, the halogen parameters given in Table 2-4, and the pumped forward parameter given in Table 2-3, and substituting in Equation (2-2) gives

$$R = \frac{1.5 \times 10^5 \times 0.9 + 0.18 \times 15.4 \times 10^6 \times 0.015 \times 0.9}{4.9 \times 10^5} = 0.35$$

Then, using this value of R in Equation (2-1):

$$A = \frac{3758}{4.9 \times 10^5}(110)\frac{0.40 + 3.6 \times 10^{-3}}{0.35 + 3.6 \times 10^{-3}} = 0.96$$

The adjusted I-131 concentration

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

= $0.96 \times 4 \times 10^{-3}$ μCi/g = 3.8×10^{-3} μCi/gm

2. Cs, Rb (Cs-137 is used as an example) -- Using the equation for Cs and Rb in Table 2-5, the adjustment factor A is calculated as follows:

$$A = \frac{P}{WP}(110)\frac{0.17 + \lambda}{R + \lambda} \qquad (2\text{-}3)$$

where the terms in the equation are as defined in Tables 2-3 and 2-4.

In calculating A, the variable R is calculated first, using Equation (2-2). The Cs and Rb parameters given in Table 2-4, the pumped forward parameter given in Table 2-3, and the sample case parameters are used in the equation.

$$R = \frac{1.5 \times 10^5 \times 0.5 + 0.01 \times 15.4 \times 10^6 \times 0.001 \times 0.5}{4.9 \times 10^5} = 0.15$$

Then, using this value of R in Equation (2-3) above:

$$A = \left(\frac{3758}{4.9 \times 10^5}\right)(110)\left(\frac{0.17 + 2.6 \times 10^{-6}}{0.15 + 2.6 \times 10^{-6}}\right) = 0.96$$

The adjusted Cs-137 concentration

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

= $0.96 \times 8 \times 10^{-5}$ μCi/g = 7.6×10^{-5} μCi/gm

3. Other Nuclides (Na-24 is used as an example) -- Using the equation for other nuclides in Table 2-5, the adjustment factor A is calculated as follows:

$$A = \frac{P}{WP}(110)\frac{0.31 + \lambda}{R + \lambda} \qquad (2\text{-}4)$$

where the terms in the equation are as defined in Tables 2-3 and 2-4.

In calculating A, the variable R is calculated first, using Equation (2-2). The other nuclide parameters given in Table 2-4, the pumped forward parameter given in Table 2-3, and the sample case parameters are used in the equation:

$$R = \frac{1.5 \times 10^5 \times 0.9 + 0.01 \times 15.4 \times 10^6 \times 0.001 \times 0.9}{4.9 \times 10^5} = 0.28$$

Then, using this value of R in Equation (2-4):

$$A = \left(\frac{3758}{4.9 \times 10^5}\right) (110) \left(\frac{0.31 + 4.62 \times 10^{-2}}{0.28 + 4.62 \times 10^{-2}}\right) = 0.92$$

The adjusted concentration of Na-24

= (adjustment factor) x (standard Na-24 concentration)

= $0.92 \times 1 \times 10^{-2}$ µCi/gm = 9.2×10^{-3} µCi/gm

The noble gas concentrations in Table 2-2 are based on an offgas release rate of 50,000 µCi/sec measured at 30-minute decay. The value of 50,000 µCi/sec can be determined as discussed below. Recent data supplied by General Electric (Refs. 5,6,7) shows that improved (7 x 7R and 8 x 8) fuel which is in the process of being installed in both new and reload cores has shown considerable improvement over the 7 x 7 fuel which was previously used and still present in some BWRs.

The improvements in the fuel are primarily due to a reduction in zircaloy hydriding and pellet cladding interaction (PCI). However, there is not extensive experience of the improved fuel over complete burnup cycles to date. At the higher burnups, the probability for PCI failure increases even though the power generation rate in the fuel is diminished. This results because at the higher burnups the fuel-to-cladding gap closes due to fission product swelling of the fuel and because the cladding loses its ductility from neutron damage and hydrogen pickup (Ref. 8).

For these reasons, there is not sufficient justification to use the low values of noble gas release rates experienced at reactors having cores loaded with 100% of improved fuel. However, a review was performed of BWR operating experience with noble gas release rates for the period 1975-1977 (Refs. 9, 10) which include the time period during which the improved fuel was introduced. In order to account for the potential effects of increased releases at high burnups and also to account for the fact that the 7 x 7 fuel will be phased out (Ref. 6), only the experience at those reactors whose cores are loaded with greater than 50% of improved fuel was considered. A summary of these data is contained in Table 2-6. Based on this review, the noble gas release rate that will be used as an interim measure until more improved fuel experience is obtained is 50,000 µCi/sec at 30 minutes decay and normalized to 3400 MWt.

A carryover factor of 0.015 is used to calculate the halogen concentrations in the main steam in Table 2-2 for BWRs which have deep bed condensate treatment, or for BWRs with powdex filter/demineralizer condensate treatment and having stainless steel condenser tubing. For BWRs with powdex filter/demineralizer condensate treatment systems and copper condenser tubing a carryover factor of 0.004 is used to calculate the halogen concentrations in the main steam. This carryover factor is derived from data taken at operating reactors (Refs. 3, 4, 5, 11 and 12) which are listed in Table 2-7. The average of the data in Table 2-7 is 0.015 and 0.004 for halogen (iodine) carryover, respectively, for the two types of BWRs listed in that table.

The nominal value of the ratio of the condensate demineralizer flow rate to the steam flow rate is 0.75. 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 75% flowing to the low pressure turbines and the main condenser, and 25% 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 and Cs, Rb, and Other Nuclides of Table 2-2 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 to the feedwater system. Based on data provided in Ref. 13 and 14 for Brunswick and Point Beach, the ratios used in the BWR-GALE Code are 82% bypass of condensate demineralizers for iodine and 99% bypass of condensate demineralizer for Cs, Rb, and Other Nuclides of Table 2-2. Since the remainder of the nuclides listed in Table 2-2 are not removed in the condensate demineralizers, we have not considered the magnitude of bypass for those nuclides.

TABLE 2-6
SUMMARY OF NOBLE GAS RELEASE
RATES FOR OPERATING BWRs*
(μCi/sec)

Facility	1977	1976	1975
Cooper	290	31	-
Dresden 3	215,000	-	-
Duane Arnold	NA	1,300	-
Fitzpatrick	1,190	-	-
Hatch 1	1,300	-	-
Millstone 1	NA	152,000	-
Monticello	91,000	137,000	-
Nine Mile Pt 1	60,000	64,000	-
Oyster Creek	113,000	79,000	-
Peach Bottom 2	12,000	10,000	-
Peach Bottom 3	6,900	-	-
Vermont Yankee	NA	11,000	12,000
	55,000	56,000	12,000

*Data in this table are based on measured noble gas release rates in references 9 and 10 and were adjusted to 30 minutes decay and to 3400 MWt.

NA - Data not available.

TABLE 2-7
REACTOR VESSEL HALOGEN CARRYOVER FACTORS
(PARTITION COEFFICIENTS) OBSERVED AT OPERATING BWRs

BWRs which have Deep Bed Condensate Treatment; BWRs which have Powdex Systems with Stainless Steel Condenser Tubing			BWRs which have Powdex Treatment Systems with Copper Condenser Tubing		
Plant	Partition Coefficient	Ref	Plant	Partition Coefficient	Ref
Oyster Creek	0.023	3, 4, 11			
Dresden 2	0.017	3, 12	Monticello	0.004	3
Dresden 3	0.021	3, 12	Browns Ferry 1	0.005	5
Millstone 1	0.012	3, 4	Browns Ferry 2	0.0023	5
Nine Mile Point 1	0.02	4	Browns Ferry 3	0.003	5
Quad Cities 1	0.013	3	Duane Arnold	0.004	5
Cooper	0.012	5	Hatch 1	0.0035	5
Fitzpatrick	0.018	5	Peach Bottom 2	0.004	5
Pilgrim 1	0.0082	5	Peach Bottom 3	0.0044	5
AVERAGE	0.015		Vermont Yankee	0.004	5
			AVERAGE	0.004	

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 GASEOUS RELEASES FROM BUILDING VENTILATION SYSTEMS

2.2.4.1 Parameter

The noble gas and radioactive particulate releases from ventilation systems for facilities with the BWR/6, Mark III containment design, prior to treatment, are shown in Table 2-12.

The iodine releases from ventilation systems for facilities with the BWR/6, Mark III containment design, prior to treatment, are calculated by the BWR-GALE Code using the data in Table 1-2, Tables 2-2 through 2-5, and 2-8 through 2-10.

2.2.4.2 Bases

The iodine-131 releases from building ventilation systems are based on measurements made at a number of operating reactors. These measurements were made during routine operation and during plant shutdowns. Extensive work on identifying sources of radioiodine at BWRs has been done by C. Pelletier, et al (Ref. 15) for the Electric Power Research Institute (EPRI), at three operating BWRs, Monticello, Vermont Yankee and Oyster Creek, and for the U.S. Nuclear Regulatory Commission at one operating BWR, Pilgrim (Ref. 16).

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

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

where

R_N = normalized release rate of reactor water containing iodine-131, gm/sec

R_A = absolute (measured) iodine-131 release rate, μCi/sec

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

The normalized reactor water release rate, expressed in gm/sec, 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 iodine releases are directly related to the partition coefficient for radioiodine from reactor water to steam, in addition to being directly related to the reactor water 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 reactor water containing iodine-131, gm/sec

R_A = absolute (measured) iodine-131 release rate, μCi/sec

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

PC = measured partition coefficient from the reactor water to reactor steam.

The normalized release rate is used to estimate the releases from BWRs since this expression for release rate is least variable with time and least variable from plant to plant for comparable time periods (Ref. 15). For this reason, it is useful in the determination of releases from BWRs.

Data on the normalized release rates from the three reactors used in the EPRI NP-495 study and the reactor in the NRC study are given for normal operation and shutdown periods in Tables 2-8, 2-9, and 2-10 for the turbine building, the reactor building and the radwaste building, respectively.

Also given in Tables 2-8, 2-9 and 2-10 are normalized values of the iodine release data based on References 3 and 5. The data and the plants included in these references are listed in Tables 2-10 through 2-15 of NUREG-0016, dated April 1976. These data are presented as one data point since the measurements used were of short duration compared to the lengthy measurements carried out in the EPRI NP-495 and the NRC study. Also given in Tables 2-8, 2-9, 2-10 are normalized values of iodine release data from Browns Ferry during 1977 (Ref. 5).

The data in Tables 2-8 through 2-10 are expressed as the total normalized release during power operation of 300 days and the total normalized release during extended shutdown of 65 days. Since the reactors used in the EPRI NP-495 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.

In order to obtain the releases in curies/yr from the reactor building and radwaste building of a particular BWR, the normalized release data in Tables 2-9 and 2-10, respectively, are multiplied in the BWR-GALE Code by the iodine reactor water concentration for that particular BWR using the following expression:

$$R_{BWR} = R_N^1 \times C_{BWR}$$

where

R_{BWR} = calculated annual release for particular BWR, Ci/yr

R_N^1 = normalized annual release of reactor water containing iodine-131 from Tables 2-9 and 2-10, Ci/yr/μCi/gm

C_{BWR} = calculated reactor water concentration for particular BWR, μCi/gm

To obtain the release in curies/yr from the turbine building of a particular BWR, the normalized release data in Table 2-8 are multiplied in the BWR-GALE Code by the iodine reactor water concentration and the iodine carryover from the reactor water to reactor steam for that particular BWR using the following expression:

$$R_{BWR} = R_N^1 \times C_{BWR} \times PC_{BWR}$$

where

R_{BWR} = calculated annual release for particular BWR, Ci/yr

R_N^1 = normalized annual release of reactor water from Table 2-8, Ci/yr/μCi/gm

C_{BWR} = calculated reactor water concentration for particular BWR, μCi/gm

PC_{BWR} = calculated carryover from the reactor water to reactor steam for the particular BWR (See Section 2.2.3.2 and Table 2-4)

The value for the iodine carryover for the reactor water to reactor steam can be determined for the particular BWR from Table 2-4.

To obtain the releases during extended shutdown, multiply the normalized release rates for the extended shutdown period by the same reactor water concentration as for power operation. Use of this reactor water concentration is acceptable since the normalization technique of EPRI-NP-495 based the extended shutdown normalized release rate on the reactor water concentrations prior to shutdown.

The value for the iodine-131 reactor water concentration can be determined as discussed below. Recent data supplied by General Electric (Refs. 5, 6, 7) shows that improved (7 x 7R and 8 x 8) fuel which is in the process of being installed in both new and reload cores has shown considerable improvement over the 7 x 7 fuel which was previously used and still present in some BWRs.

The improvements in the fuel are primarily due to a reduction in zircaloy hydriding and pellet cladding interaction (PCI). However, there is not extensive experience of the improved fuel over complete burnup cycles to date. At the higher burnups, the probability for PCI failure increases even though the power generation rate in the fuel is diminished. This results because at the higher burnups the fuel-to-cladding gap closes due to fission product swelling of the fuel and because the cladding loses its ductility from neutron damage and hydrogen pickup (Ref. 8).

For these reasons, there is not sufficient justification to use the low values of iodine-131 reactor water concentration experienced at reactors having cores loaded with 100% of improved fuel. However, a review was performed of BWR operating experience (Ref. 5) with iodine-131 reactor water concentrations for the period 1975-1977 which includes the time period during which the improved fuel was introduced. In order to account for the potential effects of increased releases at high burnups and also to account for the fact that the 7 x 7 fuel will be phased out (Ref. 6), only the experience at those reactors whose cores are loaded with greater than 50% of improved fuel was considered. A summary of these data is contained in Table 2-11. Based on this review, the iodine-131 reactor water concentration that will be used as an interim measure until more improved fuel experience is obtained is 0.0037 μCi/gm.

The reactor building releases reported in References 13 and 14 are based on reactors with a BWR Mark I containment design. Equipment such as the reactor water cleanup (RWCU) pumps, the residual-heat removal system, and emergency core cooling systems have been placed in an auxiliary building in the BWR/6, Mark III containment design concept. Based on data gathered in Reference 15, the RWCU pumps are the major souce of leakage in the reactor building. As a result of these measurements, the releases from the Mark III auxiliary building ventilation system are determined to be 90% of Mark I reactor building release, and releases from the Mark III containment building ventilation are determined to be 10% of Mark I releases during power operation. During shutdown, 90% of the releases are determined to be from the Mark III containment building ventilation system and 10% from the auxiliary building ventilation system. For the turbine building, based on data gathered in Ref. 15, 85% of the releases are determined to come from the ventilation system serving the main condenser area during power operation. The remainder of the releases come from miscellaneous areas such as the steam jet air ejector room, the turbine operating floor, the feedwater pump room, and the mechanical vacuum pump room. During the shutdown since there is potential for iodine release during maintenance of the turbines, the release from the ventilation system serving the main condenser area is approximately 50% of the total and the remainder of the releases come from the miscellaneous areas.

For the radwaste building, based on data gathered in Ref. 15, 10% of the releases are determined to come from the solid waste handling area and 90% of the releases are determined to come from the liquid waste handling area.

Within the building ventilation systems, charcoal adsorbers may be added on individual equipment cells and appropriate credit taken for iodine removal if the fraction of total iodine being assigned to that particular equipment cell is in accordance with Ref. 15.

Iodine released from BWR building ventilation systems appear in one of the following chemical forms: particulate, elemental, hypoiodious acid (HOI) and organic. Based on data in References 15 and 16 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 BWR BUILDING VENTILATION SYSTEMS

	CONTAINMENT	AUXILIARY	TURBINE	RADWASTE
Particulate	0.11	0.2	0.2	0.002
Elemental	0.32	0.48	0.50	0.28
HOI	0.38	0.24	0.22	0.25
Organic	0.19	0.09	0.08	0.47

TABLE 2-8
ANNUAL IODINE NORMALIZED RELEASES*
FROM TURBINE BUILDING VENTILATION SYSTEMS

NORMAL OPERATION

Data Source	Normalized Release (Ci/yr/μCi/gm)
Monticello (Ref. 15)	3.1×10^3
Oyster Creek (Refs. 15, 16)	6.0×10^3
Vermont Yankee (Ref. 15)	0.35×10^3
Pilgrim (Ref. 16)	8.5×10^3
Browns Ferry (Ref. 5)	1.3×10^3
References 3 and 5***	3.3×10^3
AVERAGE	3.8×10^3

EXTENDED SHUTDOWN

Data Source	Normalized Release (Ci/yr/μCi/gm)
Monticello (Ref. 15)	1.7×10^2
Oyster Creek (Ref. 15)	3.5×10^2**
Vermont Yankee (Ref. 15)	0.63×10^2
Browns Ferry (Ref. 5)	1.3×10^2
References 3 and 5***	1.4×10^3
AVERAGE	4.1×10^2**

*The normalized release rate, expressed in grams of water during the modes of operation, represents the effective leak rate for radioiodine. It is the combination of the reactor water leakage rate into the buildings, the partitioning of the radiodine between the water phase in the leakage and the gas phase where it is measured and the partition coefficient for radioiodine from reactor water to steam in the reactor vessel.

**Oyster Creek data in this table does not include effect of use of reheater protection system exhaust since the system design of this component is not typical of current BWRs (Nine Mile Point, Unit No. 1 is the only other BWR with this design). If a BWR uses this design, the additional release is 8.7×10^2 Ci/yr/μCi/gm during the shutdown period (Ref. 15) and should be included in the total turbine building shutdown release.

***The data and the plants included in these references are listed in Tables 2-10 through 2-15 of NUREG-0016, dated April 1976.

TABLE 2-9
ANNUAL IODINE NORMALIZED RELEASES*
FROM REACTOR BUILDING VENTILATION SYSTEMS

NORMAL OPERATION

Data Source	Normalized Releases (Ci/yr/μCi/gm)
Monticello (Ref. 15)	11
Pilgrim (Ref. 16)	13
Brown Ferry (Ref. 5)	4.2
References 3 and 5***	21
	12.3**

EXTENDED SHUTDOWN

Data Source	Normalized Releases (Ci/yr/μCi/gm)
Monticello (Ref. 15)	0.47
Oyster Creek (Ref. 15)	1.3
Vermont Yankee (Ref. 15)	3.2
Browns Ferry (Ref. 5)	1.4
References 3 and 5***	20
	5.2

*The normalized release rate, expressed in grams of water during the modes of operation, 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 radiodine between the water phase in the leakage and the gas phase where it is measured.

**Oyster Creek and Vermont Yankee data are not included here since Monticello leakage is considered to be more typical of similar problems at other BWRs where the RWCU pump in upstream of the RWCU demineralizers.

***The data and the plants included in these references are listed in Tables 2-10 through 2-15 of NUREG-0016, dated April 1976.

TABLE 2-10
ANNUAL IODINE NORMALIZED RELEASES*
FROM RADWASTE BUILDING VENTILATION SYSTEMS

NORMAL OPERATION

Data Source	Normalized Release (Ci/yr/μCi/gm)
Monticello (Ref. 15)	0.72
Oyster Creek (Refs. 15, 16)	6.8
Vermont Yankee (Ref. 15)	1.0
Pilgrim (Ref. 16)	12
Browns Ferry (Ref. 5)	2.0
References 3 and 5**	5.3
AVERAGE	4.6

EXTENDED SHUTDOWN

Data Source	Normalized Release (Ci/yr/μCi/gm)
Monticello (Ref. 15)	0.02
Oyster Creek (Ref. 15)	1.4
Vermont Yankee (Ref. 15)	0.4
Browns Ferry (Ref. 5)	0.6
References 3 and 5**	4.4
AVERAGE	1.4

*The normalized release rate, expressed in grams of water during the modes of operation, 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 radiodine between the water phase in the leakage and the gas phase where it is measured.

**The data and the plants included in these references are listed in Tables 2-10 through 2-15 of NUREG-0016, dated April 1976.

TABLE 2-11
SUMMARY OF IODINE-131 REACTOR WATER CONCENTRATIONS IN BWR's*
(μCi/Kg)

Facility	1977	1976	1975
Browns Ferry 1	0.9	-	-
Browns Ferry 2	1.5	-	-
Browns Ferry 3	0.14	-	-
Brunswick 1	0.02	-	-
Brunswick 2	3.1	0.93	0.007
Cooper	0.072	0.09	0.013
Dresden 3	17.6	12.6	-
Duane Arnold	0.042	0.09	0.0023
Fitzpatrick	0.24	0.29	-
Hatch 1	0.9	0.11	-
Millstone Pt. 1	8.9	5.6	7.1
Monticello	5.9	9.0	8.7
Nine Mile Pt. 1	9.4	5.9	-
Oyster Creek	8.4	5.3	4.8
Peach Bottom 2	7.3	16.	0.045
Peach Bottom 3	1.1	0.83	0.063
Quad Cities 1	3.4	-	-
Vermont Yankee	0.38	0.51	0.78
	3.8	4.4	2.4

*Data in these tables are based on measured iodine-131 coolant concentrations in Ref. 5 and have been adjusted to the NSSS parameters listed in Table 2-3 of this report. These adjustments were made by considering the individual plant parameters and the nominal plant parameters (Table 2-3) and adjusting the actual coolant concentration using the equations in Table 2-5 of this report.

TABLE 2-12
GASEOUS AND PARTICULATE RELEASES FROM
BUILDING VENTILATION SYSTEMS
(in Ci/yr per Reactor)

NUCLIDE	CONTAINMENT BUILDING	AUXILIARY BUILDING	TURBINE BUILDING	RADWASTE BUILDING
Kr83m	**	**	**	**
Kr-85m	1	3	25	**
Kr-85	**	**	**	**
Kr-87	**	2	61	**
Kr-88	1	3	91	**
Kr-89	**	2	580	29
Xe-131m	**	**	**	**
Xe-133m	**	**	**	**
Xe-133	27	83	150	220
Xe-135m	15	45	400	530
Xe-135	33	94	330	280
Xe-137	45	135	1000	83
Xe-138	2	6	1000	2
Cr-51*	0.0002	0.0009	0.0009	0.0007
Mn-54	0.0004	0.001	0.0006	0.004
Fe-59	0.00009	0.0003	0.0001	0.0003
Co-58	0.0001	0.0002	0.001	0.0002
Co-60	0.001	0.004	0.001	0.007
Zn-65	0.001	0.004	0.006	0.0003
Sr-89	0.00003	0.00002	0.006	NA
Sr-90	0.000003	0.000007	0.00002	NA
Zr-95	0.0003	0.0007	0.00004	0.0008
Nb-95	0.001	0.009	0.000006	0.000004
Mo-99	0.006	0.06	0.002	0.000003
Ru-103	0.0002	0.004	0.00005	0.000001
Ag-110	0.0000004	0.000002	NA	NA
Sb-124	0.00002	0.00003	0.0001	0.00007
Cs-134	0.0007	0.004	0.0002	0.0024
Cs-136	0.0001	0.0004	0.0001	NA
Cs-137	0.001	0.005	0.001	0.004
Ba-140	0.002	0.02	0.010	0.000004
Ce-141	0.0002	0.0007	0.010	0.000007

*Particulate release rates are prior to filtration.

**Less than 1 Ci/yr per reactor.

NA Not Analyzed; analysis for the isotope was not performed.

TABLE 2-13

RELEASE RATES OF NOBLE GASES FROM THE REACTOR BUILDING VENTILATION SYSTEM (μCi/sec)

NUCLIDE	MILLSTONE-1	OYSTER CREEK	OYSTER CREEK	MONTICELLO	AVERAGE
Kr-85m	0.26	ND	ND	0.20	0.12
Kr-87	0.24	ND	ND	0.10	0.085
Kr-88	0.38	0.02	ND	0.20	0.15
Kr-89	ND	ND	ND	0.38	0.095
Xe-133	0.52	15	ND	2.0	4.4
Xe-135m	3.6	ND	2.5	3.5	2.4
Xe-135	3.0	2.1	14	1.8	5.2
Xe-137	ND	ND	ND	30	7.5
Xe-138	0.44	0.3	ND	0.4	0.29

ND - Not Detected. For averaging purposes a value of zero was assumed.

TABLE 2-14

RELEASE RATES OF NOBLE GASES FROM THE TURBINE BUILDING VENTILATION SYSTEM (μCi/sec)

NUCLIDE	MILLSTONE-1	OYSTER CREEK	OYSTER CREEK	MONTICELLO	NINE MILE PT 1	AVERAGE
Kr-85m	2.7	ND	2.3	0.10	0.097	1.0
Kr-87	5.3	ND	6.2	0.15	0.53	2.4
Kr-88	8.2	5.2	4.2	0.065	0.21	3.6
Kr-89	ND	ND	70	42	4.5	23
Xe-133	7.4	13	ND	5.0	3.5	5.8
Xe-135m	29	12	26	8.2	2.5	16
Xe-135	25	25	7.4	6.8	2.3	13
Xe-137	ND	ND	115	86	ND	40
Xe-138	63	26	97	11	4.3	40

ND - Not Detected. For averaging purposes a value of zero was assumed.

TABLE 2-15

RELEASE RATES OF NOBLE GASES FROM THE RADWASTE BUILDING VENTILATION SYSTEM (μCi/sec)

NUCLIDE	MILLSTONE-1	OYSTER CREEK	MONTICELLO	AVERAGE
Kr-89	ND	ND	3.0	1.0
Xe-133m	ND	ND	5.3	1.8
Xe-133	0.25	0.56	26	8.9
Xe-135m	ND	4	59	21
Xe-135	2.0	1.5	20	7.8
Xe-137	ND	ND	10	3.3
Xe-138	ND	ND	0.2	0.067

ND - Not Detected. For averaging purposes a value of zero was assumed.

TABLE 2-16

PARTICULATE RELEASE RATES FROM REACTOR BUILDING VENTILATION SYSTEM, NORMAL OPERATION (10^{-6} μCi/sec)

NUCLIDE	QUAD CITIES 1	QUAD CITIES 2	VERMONT YANKEE	OYSTER CREEK	MONTICELLO	AVERAGE
Co-60	210	0.9	37	610	140	200
Co-58	20	0.4	5.5	31	ND	11
Cr-51	140	0.5	13	39	ND	38
Mn-54	19	0.1	14	210	24	53
Fe-59	NA	NA	5	33	4.2	14
Zn-65	23	0.1	46	6.4	750	160
Sr-89	NA	NA	NA	6.8	NA	6.8
Sr-90	NA	NA	NA	0.3	NA	0.3
Zr-95	1.6	ND	1.5	0.5	115	24
Nb-95	2.7	ND	7.4	0.3	2200	440
Mo-99	NA	NA	4.4	140	7300	2500
Ru-103	NA	NA	ND	2.8	65	23
Ag-110m	0.2	ND	NA	NA	NA	0.1
Sb-124	NA	NA	ND	2.4	ND	0.8
Cs-134	48	0.1	12	16	760	170
Cs-136	2.3	ND	6.8	7.1	79	19
Cs-137	44	0.5	37	31	990	220
Ba-140	ND	ND	16	76	2600	540
Ce-141	NA	ND	ND	3.9	120	31

NA - Not Analyzed.
ND - Not Detected. For averaging purposes a value of zero was assumed.

The noble gas release rates for building ventilation systems are the average of measurements made at Oyster Creek (Ref 17), Millstone Unit No. 1 (Ref. 18), Monticello (Ref. 15), and Nine Mile Point (Ref. 19). These data are given in Tables 2-13 through 2-15 and are based on the fuel handling area being in the containment building. The noble gas release rates are divided between the containment and auxiliary buildings to reflect the BWR/6, Mark III design, in a manner similar to that for the iodine-131 release.

For the Mark III design during shutdown, 90% of the releases are assumed to be from the containment building and 10% from the auxiliary building ventilation system since the releases from the fuel handling area are considered to be the major source. For the BWR/6 Mark-III containment system design, the fuel building releases are considered to be part of the containment building releases.

The radioactive particulate release rates for building ventilation systems are the average of measurements made at Vermont Yankee, Oyster Creek, Dresden 2 & 3, Quad Cities 1 & 2, Monticello, and Nine Mile Point (Refs. 3, 5, 15 and 19). These data are given in Tables 2-16 through 2-21.

The calculated annual average rates given above are based on an 80% plant capacity factor, i.e., 80% normal operation at 100% power and 20% plant downtime. The releases for normal operation are weighted to account for the operating and shutdown modes. The particulate releases for the reactor building are divided between the containment and auxiliary buildings to reflect the BWR/6, Mark III containment design in a manner similar to that for the iodine-131 releases.

2.2.5 IODINE INPUT TO THE MAIN CONDENSER OFFGAS TREATMENT SYSTEM

2.2.5.1 Parameter

The iodine-131 input to the main condenser offgas treatment system, downstream of the air ejectors, is 6 Ci/yr.

2.2.5.2 Bases

Table 2-22 lists the measured iodine-131 releases and integrated thermal power outputs for BWRs with thermal ratings exceeding 1000 MWt, with more than one year of plant operation and without main condenser offgas treatment. The average ratio of the iodine-131 release in Ci/yr to the integrated thermal power in MWd for the years 1972 through 1976 is approximately 6.3×10^{-6} Ci/MWd per year. Based on a power rating of 3400 MWt and an 80% plant capacity factor, the iodine-131 release from the main condenser air ejector is approximately 6 Ci/yr.

2.2.6 TURBINE GLAND SEALING SYSTEM EXHAUST

2.2.6.1 Parameter

If main steam is used, the annual radioiodine releases from the gland seal condenser exhaust are:

I-131 = 8.1×10^{-1} Ci/yr per μCi/gm of I-131 in the reactor coolant.
I-133 = 2.2×10^{-1} Ci/yr per μCi/gm of I-133 in the reactor coolant.

If the clean steam is supplied to the gland seal, the radioiodine source term is negligible (less than 10^{-4} Ci/yr). If sealing steam is supplied from a low-activity source, i.e., steam produced from demineralized condensate, consider the release to be zero.

2.2.6.2 Bases

Radioiodine measurements have been reported (Ref. 15) for two operating facilities that use main steam in the turbine gland seal system. The sample location necessitated including any radioiodines released from the mechanical vacuum pump during sampling. Table 2-23 summarizes this available data for radioiodines released from the gland seal condenser exhaust when the mechanical vacuum pump was not in operation or infrequently used. The radioiodine release rates are dependent on the radioiodine concentration in the reactor coolant and carryover in the reactor.

It is assumed that there is no radioiodine source term when clean steam (non-radioactive steam from an auxiliary steam supply system) is used for the gland seal. Because of noble gas removal in the main steam condenser, radioiodine removal by the condensate demineralizers, and partitioning in the boiler, steam produced from demineralized condensate is considered to be clean steam. Data in Tables 2-24 and 2-25 show the release of radioactive particulates from the turbine gland seal to be negligible (less than 10^{-5} Ci/year).

TABLE 2-17

PARTICULATE RELEASE RATES FROM REACTOR BUILDING VENTILATION SYSTEM, REFUELING SHUTDOWN

(10^{-6} μCi/sec)

NUCLIDE	QUAD CITIES 1	QUAD CITIES 2	VERMONT YANKEE	OYSTER CREEK	MONTICELLO	AVERAGE
Co-60	16	0.88	250	330	1.70	120
Co-58	2.3	0.35	41	19	ND	13
Cr-51	8.9	0.50	63	28	ND	20
Mn-54	2.6	0.061	20	140	47	42
Fe-59	ND	ND	21	5.0	3.4	6.0
Zn-65	58	0.11	770	1.4	73	180
Sr-89	NA	NA	NA	2.0	NA	2.0
Sr-90	NA	NA	NA	0.36	NA	0.36
Zr-95	0.31	ND	78	0.24	ND	16
Nb-95	0.40	0.021	ND	0.41	160	32
Mo-99	NA	NA	ND	13	4.4	5.8
Ru-103	NA	NA	NA	1.3	36	19
Ag-110m	0.11	NA	NA	NA	NA	0.11
Sb-124	ND	ND	NA	7.0	ND	1.8
Cs-134	6.2	0.14	82	13	170	54
Cs-136	1.0	NA	20	ND	21	11
Cs-137	14	0.54	240	23	200	95
Ba-140	ND	ND	14	1.1	200	43
Ce-141	ND	ND	NA	7.5	45	13

NA - Not Analyzed.
ND - Not Detected. For averaging purposes a value of zero was assumed.

TABLE 2-18

PARTICULATE RELEASE RATES FROM TURBINE BUILDING VENTILATION SYSTEM, NORMAL OPERATION

(10^{-6} μCi/sec)

NUCLIDE	OYSTER CREEK	MONTICELLO	VERMONT YANKEE	DRESDEN 2	DRESDEN 3	AVERAGE
Co-60	61	15	4.1	4.5	6.0	18
Co-58	12	ND	2.0	ND	48	12
Cr-51	440	ND	NA	ND	160	150
Mn-54	30	7.5	1.9	ND	5	8.9
Fe-59	5.8	ND	ND	ND	ND	1.2
Zn-65	1.7	23	7.8	ND	ND	6.5
Sr-89	610	NA	NA	48	3.6	220
Sr-90	1.3	NA	NA	0.3	0.25	0.6
Zr-95	0.59	ND	ND	ND	4.0	0.92
Nb-95	0.33	ND	NA	ND	ND	0.1
Mo-99	91	150	NA	ND	ND	61
Ru-103	1.7	ND	ND	ND	ND	0.34
Sb-124	4.6	ND	ND	ND	ND	0.92
Cs-134	18	23	2.7	ND	3.0	9.3
Cs-136	1.1	16	ND	ND	ND	3.4
Cs-137	57	100	5.1	1.8	10	35
Ba-140	1400	16	83	120	65	340
Ce-141	29	1600	ND	5.5	5	328

NA - Not Analyzed.
ND - Not Detected. For averaging purposes a value of zero was assumed.

TABLE 2-19

PARTICULATE RELEASE RATES FROM TURBINE BUILDING VENTILATION SYSTEM, REFUELING SHUTDOWN

(10^{-6} μCi/sec)

NUCLIDE	OYSTER CREEK	VERMONT YANKEE	MONTICELLO	AVERAGE
Co-60	290	2.5	5.8	100
Co-58	16	1	NA	8.5
Cr-51	51	ND	NA	26
Mn-54	110	NA	0.30	57
Fe-59	31	ND	ND	10
Zn-65	11	NA	10	10
Sr-89	2.5	NA	NA	2.5
Sr-90	0.25	NA	NA	0.25
Zr-95	0.06	ND	ND	0.02
Nb-95	0.40	ND	ND	0.13
Mo-99	125	NA	9.7	67
Ru-103	5.2	NA	ND	2.6
Sb-124	9.5	ND	NA	4.8
Cs-134	19	1.9	1.3	7.4
Cs-136	ND	ND	4.1	1.4
Cs-137	39	3.4	5.8	16
Ba-140	8.2	110	49	56
Ce-141	17	ND	9.1	8.7

NA - Not Analyzed.
ND - Not Detected. For averaging purposes a value of zero was assumed.

TABLE 2-20

PARTICULATE RELEASE RATE FROM RADWASTE BUILDING VENTILATION SYSTEM, NORMAL OPERATION

(10^{-6} μCi/sec)

NUCLIDE	VERMONT YANKEE	OYSTER CREEK	AVERAGE
Co-60	6.0	580	290
Co-58	1.0	16	8
Cr-51	3.0	48	26
Mn-54	1.0	330	170
Fe-59	ND	26	13
Zn-65	1.0	21	11
Sr-89	NA	NA	NA
Sr-90	NA	NA	NA
Zr-95	ND	63	31
Mo-99	2.0	ND	1.0
Sb-124	ND	5.4	2.7
Cs-134	1.2	190	96
Cs-136	ND	ND	0
Cs-137	2.0	290	150
Ba-140	0.3	ND	0.15
Ce-141	ND	6.3	3.2

NA - Not Analyzed
ND - Not Detected. For averaging purposes a value of zero was assumed.

TABLE 2-21

PARTICULATE RELEASE RATE FROM RADWASTE BUILDING VENTILATION SYSTEM, REFUELING SHUTDOWN

(10^{-6} μCi/sec)

NUCLIDE	MONTICELLO	AVERAGE
Co-60	1.3	1.3
Co-58	0.21	0.21
Cr-51	ND	0
Mn-54	0.40	0.40
Fe-59	3.2	3.2
Zn-65	5.1	5.1
Sr-89	NA	NA
Sr-90	NA	NA
Nb-95	6.0	6.0
Mo-99	1.0	1.0
Ru-103	1.0	1.0
Ru-103	1.0	1.0
Sb-124	ND	0
Cs-134	1.0	1.0
Cs-136	ND	0
Cs-137	2.2	2.2
Ba-140	ND	0
Ce-141	1.2	1.2
Ce-144	4.0	4.0

NA - Not Analyzed.
ND - Not Detected. For averaging purposes a value of zero was assumed.

TABLE 2-22

RADIOIODINE-131 RELEASES FROM THE MAIN CONDENSER AIR EJECTORS[a]

FACILITY	1972			1973			1974		
	RADIO-IODINE RELEASE (Ci/yr)	INTEGRATED THERMAL POWER (10^6 MWd)	$\frac{Ci/yr}{10^6 MWd}$	RADIO-IODINE RELEASE (Ci/yr)	INTEGRATED THERMAL POWER (10^6 MWd)	$\frac{Ci/yr}{10^6 MWd}$	RADIO-IODINE RELEASE (Ci/yr)	INTEGRATED THERMAL POWER (10^6 MWd)	$\frac{Ci/yr}{10^6 MWd}$
Oyster Creek	6.3	0.542	11.6	6.7	0.453	14.8	3.3	0.46	7.2
Nine Mile Point 1	0.89	0.417	2.1	1.9	0.457	4.2	0.7	0.43	1.6
Millstone 1	1.2	0.404	3.0	0.15	0.248	0.6	0.29	0.47	0.6
Dresden 2/3[b]	5.1	1.05	4.9	9.8	1.18	8.3	4.0	0.91	4.4
Monticello	0.58	0.454	1.3	1.2	0.413	2.9	5.7	0.34	16.8
Pilgrim 1			c	0.46	0.523	0.9	1.4	0.25	5.6
Quad Cities 1/2[b]			c	5.5	1.32	4.2	8.2	1.09	7.5
Average			4.6			5.1			6.2

Combined average for 1972 through 1976 = $6.3 \times 10^{-6} \frac{Ci/yr}{MWd}$

[a]Data from semiannual operating for 1972 through 1976 for facilities listed.
[b]Two-unit plants with a single stack.
[c]Not included in 1972 average because plants had not achieved a full year of operation.
[d]Augmented offgas system put in operation October 1975.
[e]Augmented offgas system put in operation late 1976.
[f]Augmented offgas system put in operation May 1975.
[g]Augmented offgas system put into operation 1977.
[h]Augmented offgas system put into operation late 1974.

TABLE 2-22 (continued)

RADIOIODINE-131 RELEASES FROM THE MAIN CONDENSER AIR EJECTORS

FACILITY	1975 RADIOIODINE RELEASE (Ci/yr)	1975 INTEGRATED THERMAL POWER (10^6 MWd)	1975 Ci/yr / 10^6 MWd	1976 RADIOIODINE RELEASE (Ci/yr)	1976 INTEGRATED THERMAL POWER (10^6 MWd)	1976 Ci/yr / 10^6 MWd
Oyster Creek	5.5	0.41	13.4	6.2	0.49	12.7
Nine Mile Point 1	2.1	0.4	5.3	2.1	0.55	3.8[g]
Millstone 1	9.8	0.5	19.6	2.7	0.48	5.6
Dresden 2/3[b]	0.75	0.71	1.06	1.9	1.14	1.7[e]
Monticello	3.5	0.37	9.5	d		
Pilgrim 1	h					
Quad Cities 1/2[b]	f					
Average			9.8			6.0

TABLE 2-23

RADIOIODINE RELEASE RATE FROM GLAND SEAL CONDENSER EXHAUST
FOR SYSTEMS USING MAIN STEAM FOR THE SEALING SYSTEM AT 7000 lbs/hr.

Nuclide	Facility	Sample Period	Days	Measured Gland Seal I-131 Release (μCi/sec)	Measured Reactor Water I-131 Concentration (μCi/gm)	Iodine-131 Release Ci/yr per Ci/gm for 292 days/year
I-131	Vermont Yankee	6/18/74 to 6/19/74	1	3.9(-4)	2.5(-2)	3.9(5)
		6/20/74 to 6/21/74	1	4.2(-4)	2.5(-2)	4.2(5)
		9/13/74 to 9/14/74	1	4.7(-4)	3.8(-2)	3.1(5)
		10/10/74 to 10/11/74	1	4.5(-4)	3.5(-2)	3.2(5)
		3/5/75 to 3/8/75	3	2.1(-5)	8.8(-4)	6.0(5)
		10/7/75 to 10/21/74	14	1.4(-5)	7.7(-4)	4.6(5)
	Oyster Creek	6/16/75 to 6/30/75	14	7.8(-5)	1.8(-3)	1.1(6)
		6/30/75 to 7/17/75	17	6.8(-5)	1.7(-3)	1.0(6)
	Weighted Average According to Sample Days					8.1(5)

Nuclide	Facility	Sample Period	Days	Measured Gland Seal I-133 Releases (μCi/sec)	Measured Reactor Water I-133 Concentration (μCi/gm)	Parameter (Ci/yr per Ci/gm) for 292 days/year
I-133	Vermont Yankeee	6/18/74 to 6/19/74	1	1.8(-4)	5.6(-2)	8.1(4)
		6/20/74 to 6/21/74	1	1.9(-4)	5.6(-2)	8.6(4)
		9/13/74 to 9/14/74	1	2.2(-4)	1.1(-2)	5.0(5)
		10/10/74 to 10/11/74	1	1.7(-4)	1.0(-2)	4.3(5)
	Oyster Creek	6/16/75 to 6/30/75	14	2.3(-4)	2.6(-2)	2.2(5)
		6/30/75 to 7/17/75	17	2.0(-4)	2.4(-2)	2.1(5)
	Weighted Average According to Sample Days					2.2(5)

TABLE 2-24

PARTICULATE RELEASE RATE FROM VERMONT YANKEE MECHANICAL VACUUM PUMP AND GLAND EXHAUST CONDENSER VENT, SHORT-TERM SHUTDOWN

(10^{-6} μCi/sec)

NUCLIDE	RELEASE RATE
Cr-51	0.9
Co-60	0.5
Zn-65	0.3
Cs-134	0.8
Cs-136	1.1
Cs-137	3.3
Ba-140	2.1

TABLE 2-25

PARTICULATE RELEASE RATE FROM VERMONT YANKEE MECHANICAL VACUUM PUMP AND GLAND EXHAUST CONDENSER VENT, REFUELING SHUTDOWN

(10^{-6} μCi/sec)

NUCLIDE	RELEASE RATE
Cs-134	0.45
Cs-136	0.13
Cs-137	1.0
Ba-140	1.6

TABLE 2-26

NORMALIZED IODINE RELEASES FROM
MECHANICAL VACUUM PUMP
(≤ 80 HRS)

Plant	Normalized Release (Ci/yr/μCi/gm)
Monticello	8.3(2)
Vermont Yankee	1.5(2)
AVERAGE	4.9(2)

NORMALIZED IODINE RELEASES FROM
MECHANICAL VACUUM PUMP DURING REFUELING/MAINTENANCE
OUTAGES (≥ 80 HRS)

Plant	Normalized Release (Ci/yr/μCi/gm)
Monticello	1.5(3)
Vermont Yankee	6.0(2)
AVERAGE	1.1(3)

1/ Assume (4) short-term outages per year.

2.2.7 MAIN CONDENSER MECHANICAL VACUUM PUMP

2.2.7.1 Parameter

Xe-133	1300 Ci/yr per reactor
Xe-135	500 Ci/yr per reactor

The iodine releases from the Main Condenser Mechanical Vacuum Pump are calculated by the BWR-GALE Code using the data in Tables 2-2, 2-4 and 2-26.

2.2.7.2 Bases

The release rates for Xe-133 and Xe-135 were derived from Dresden 1 and 2 operating data and adjusted to 50,000 μCi/sec (Ref. 5). These data indicate that approximately 300 Ci of Xe-133 and 120 Ci of Xe-135 were released with the mechanical vacuum pump effluent when the main condenser vacuum pumps were used to establish main condenser vacuum following a plant shutdown. At the point in the fuel cycle where the data were taken, the reactor was operating at an offgas rate of approximately 60,000 μCi/sec. The annual release estimates for noble gases assumes four short-term shutdowns per year and one refueling/maintenance outage.

The release rates for iodine-131 are based on measurements made at operating reactors (Ref. 15). Investigations for the Electric Power Research Institute (EPRI) at three operating Boiling Water Reactors (BWRs), Monticello, Vermont Yankee, and Oyster Creek, have shown that iodine releases from the mechanical vacuum pump are at their highest levels for the first 80 hours after shutdown. In accordance with the EPRI study, the releases from the mechanical vacuum pump can be as much as a factor of 100 greater than releases measured during the pre-shutdown period. The normalized iodine-131 releases in Table 2-26 are based on data from Monticello and Vermont Yankee.

The annual iodine-131 release estimates assume four short term shutdowns per year and one refueling/maintenance outage per year.

To calculate releases from the mechanical vacuum pump, a normalized release rate is used. The normalized release rate is calculated by the BWR-GALE Code using the following expression:

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

where

R_N = normalized release rate of reactor water containing iodine-131, (gm/sec)

R_A = absolute (measured) iodine-131 release rate, (μCi/sec)

C_{RW} = measured reactor water iodine-131 concentration, (μCi/gm)

PC = measured partition coefficient from reactor water to reactor steam.

To calculate the release in Ci/yr from the mechanical vacuum pump of a particular BWR, the normalized release data in Table 2-26 are multiplied by the iodine reactor water concentration and the iodine carryover from reactor water to reactor steam for the particular BWR using the following expression:

$$R_{MVP} = R'_N \times C_{BWR} \times PC_{BWR}$$

where:

R'_N = normalized release rate of reactor water containing iodine-131, (Ci/yr/μCi/gm)

R_{MVP} = calculated annual iodine release, (Ci/yr) from the mechanical vacuum pump

C_{BWR} = reactor water concentration for a particular BWR, (μCi/gm)

PC_{BWR} = calculated carryover for particular BWR from Table 2-4.

Iodine released during the operation of the Mechanical Vacuum Pump at BWRs appear in one of the following chemical forms: particulate, elemental, hypoiodious acid (HOI), and organic. Based on data in Reference 15, the fraction of the iodine appearing in each of the chemical forms for the Mechanical Vacuum Pump is given below:

Fraction of Iodine Appearing In Each Chemical Form From BWR Mechanical Vacuum Pump

	Time < 80 hrs[1/]	Time > 80 hrs[2/]
Particulate	0.004	0.01
Elemental	0.009	0.06
HOI	0.023	0.21
Organic	0.97	0.72

[1/] Average of samples taken within the first 80 hrs after shutdown.

[2/] Average of samples taken after the initial 80 hrs of a refueling maintenance outage.

Data in Tables 2-24 and 2-25 show the release of radioactive particulates from the mechanical vacuum pump to be negligible.

2.2.8 AIR INLEAKAGE TO THE MAIN CONDENSER

2.2.8.1 Parameter

0.0062 ft^3/min air inleakage to the main condenser per MWt of design reactor power with a minimum of 5 ft^3/min.

2.2.8.2 Bases

Air inleakage occurs in the main condensers of all power reactors. In a BWR, the amount of holdup time calculated for a charcoal bed offgas delay system is inversely proportional to the amount of air inleakage to the main condenser.

Operational data for inleakage vary widely. At Oyster Creek and at Dresden Unit No. 2, air inleakage measurements during early phases of operation indicated leakage rates from 4 ft^3/min to 250 ft^3/min. (Refs. 21 and 22). Subsequent measurements at Dresden Unit 2 (Ref. 20), showed an air inleakage of 4.4 ft^3/min during operation at 1600 MWt. Air inleakage measurements reported for six TVA fossil plants, representing more than 50 years of cumulative experience, indicate leakage rates ranging from 4 to 25 ft^3/min per condenser shell and a statistical mean inleakage rate of 6.7 ft^3/min per condenser shell (Ref. 23). Measurements made in 1976 and 1977, at Quad Cities Units Nos. 1 and 2 (Ref. 24), showed average flow rates of 9.6 ft^3/min for Unit No. 1 and 25 ft^3/min for Unit No. 2; measurements ranged from 6 ft^3/min to 55 ft^3/min and power level for both units during the test period was 2511 MWt.

The parameter for air inleakage was developed assuming that air inleakage is proportional to the reactor design thermal power level. Available data, which were considered to represent long-term operational results, were converted by extrapolation to the common base of a 3400 MWt BWR with a 3 shell condenser. The use of data from Dresden Unit No. 2, Quad Cities Unit Nos. 1 and 2, and TVA fossil plants resulted in an average of 21 ft^3/min main condenser air inleakage for a plant with a design thermal power level of 3400 MWt. This is approximately equivalent to 0.0062 ft^3/min inleakage for each MWt of design thermal power. For BWRs of less than 800 MWt design thermal power level, a minimum condenser air inleakage of 5 ft^3/min should be used, independent of reactor design thermal power level.

CONDENSER AIR INLEAKAGE

Plant	Power Level	Reported Data	Extrapolated to 3400 MWt/ 3 shell
Dresden 2	1600 MWt	4.4 ft^3/min	9.4
TVA Fossil Plants	700 MWe (average of 6)	6.7 ft^3/min/shell	28.7
Quad Cities 1	2511 MWt	9.6 ft^3/min	13
Quad Cities 2	2511 MWt	25 ft^3/min	34
		AVERAGE	21 cfm

2.2.9 HOLDUP TIMES FOR CHARCOAL DELAY SYSTEMS

2.2.9.1 Parameter

$T = 43.1\ MK/P$

where

K is the dynamic adsorption coefficient, in cm^3/gm (see chart below);

M is the mass of charcoal adsorber, in 10^3 lbs;

T is the holdup time, in hours; and

P is the thermal power level (MWt) entered in Card 2

Dynamic adsorption coefficients (in cm^3/gm) are as follows:

	OPERATING 77°F DEW POINT 45°F	OPERATING 77°F DEW POINT -40°F	OPERATING 77°F DEW POINT 0°F	OPERATING 0°F DEW POINT -20°F
Kr	18.5	70	25	105
Xe	330	1160	440	2410

2.2.9.2 Bases

Charcoal delay systems are evaluated using the above equation and dynamic adsorption coefficients. T = MK/flow rate is a standard equation for the calculation of delay times in charcoal adsorption systems (Ref. 25). The dynamic adsorption coefficients (K values) for Xe and Kr are dependent on operating temperature and moisture content (Ref. 26 and 27) 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 (Refs. 28 and 30) and reports concerning charcoal adsorption systems (Refs. 26-28, 31-33).

The factors influencing the selection of K values are

1. Operational data from KRB (K_{Kr} = 20-30, K_{Xe} = 260-430) (Ref. 28) and from KWL (K_{Kr} = 30, K_{Xe} = 500) (Ref. 29), and from Vermont Yankee (Ref. 31).

2. The effect of temperature on the dynamic adsorption coefficients, indicated in Figure 2-2 (Ref. 26).

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

4. The variation in K values between researchers and between the types of charcoal used in these systems (Refs. 26, 34, and 35). 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 2-2 are used.

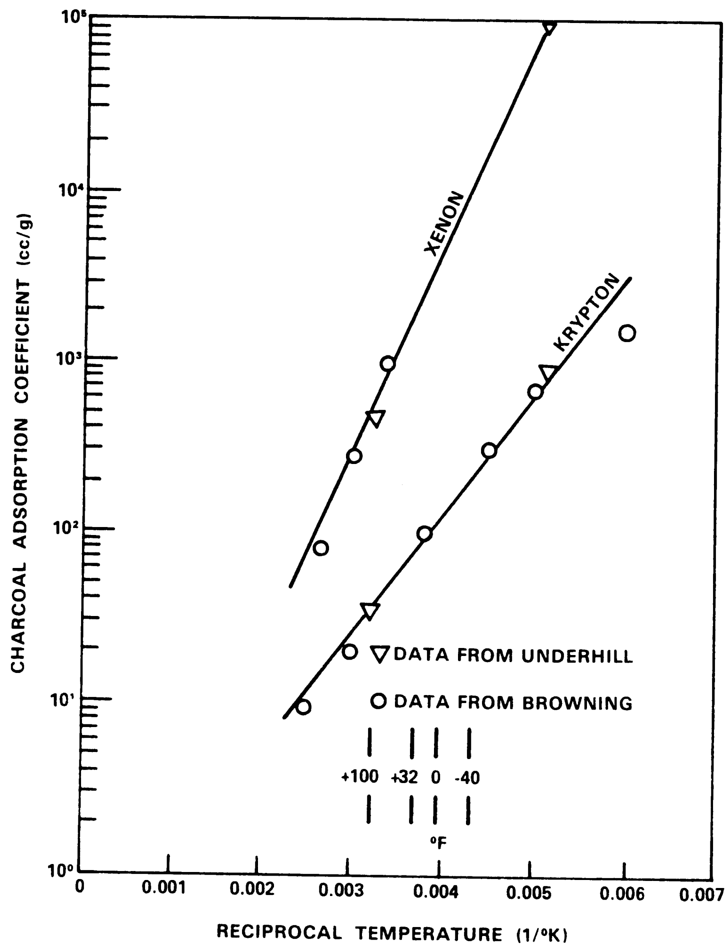


FIGURE 2-2
KRYPTON AND XENON K VALUES AS A FUNCTION OF RECIPROCAL TEMPERATURE

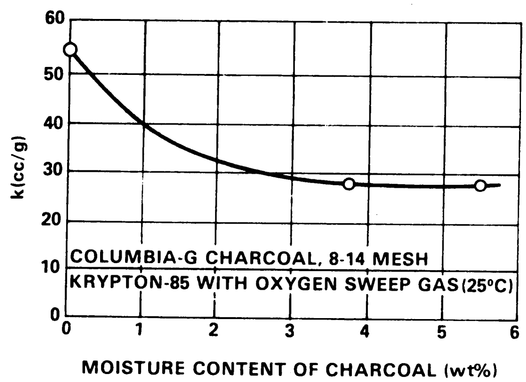


FIGURE 2-3
EFFECT OF MOISTURE CONTENT ON THE DYNAMIC ADSORPTION COEFFICIENT

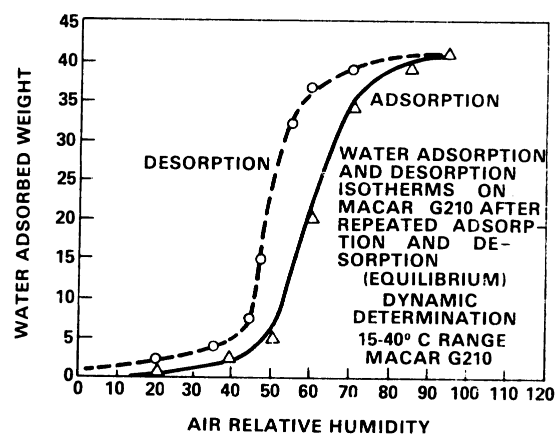


FIGURE 2-4
CHARCOAL MOISTURE AS A FUNCTION OF RELATIVE HUMIDITY

The coefficient 43.1 adjusts the units and was calculated as follows:

$$T(hr) = \frac{M(10^3 \text{ lbs})\ K(cm^3/gm)(454\ gm/lb)(3.53 \times 10^{-5}\ ft^3/cm^3)}{(0.0062\ ft^3/min/MWt)\ (60\ min/hr) P\ (MWt)}$$

$$T = 43.1\ \frac{MK}{P}$$

2.2.10 DECONTAMINATION FACTORS FOR CRYOGENIC DISTILLATION

2.2.10.1 Parameter

NUCLIDES	DECONTAMINATION FACTOR
I, Xe	1×10^4
Kr	4×10^3

The holdup times are calculated on the basis of gas residence time in the system prior to release.

2.2.10.2 Bases

A DF of 10^4 for iodine and xenon and a DF of 4×10^3 for krypton are used for a cryogenic distillation system. The values are based on data submitted in Amendment 11 to the PSAR for the Hope Creek Nuclear Generating Station, Units 1 and 2 (Ref. 36), which were derived from a proprietary report (Ref. 37) of Air Products and Chemical, Inc. The PSAR states that a maximum of 0.025% Kr (DF = 4×10^3) and 0.01% Xe (DF = 10^4) will escape from the system. These decontamination factors are considered reasonable.

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, removal efficiencies for all forms of radioiodine are as follows:

ACTIVATED CARBON BED DEPTH[a]	ASSIGNED ACTIVATED CARBON 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%
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%

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

2.2.11.2 Bases

The removal efficiencies assigned 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 LIQUID WASTE INPUTS

2.2.12.1 Parameter

The flow rates listed in Table 2-27 are used as inputs to the liquid radwaste treatment system. Flows that cannot be standardized are added to those listed in Table 2-27 to fit an individual application. Disposition of liquid streams to the appropriate collection tanks is based on the applicant's intended method of processing.

2.2.12.2 Bases

The liquid waste inputs are based on the values proposed by the ANS 55.3 Working Group in American National Standard, "Boiling Water Reactor Liquid Radioactive Waste Processing System," ANSI N197-1976 (Ref. 38). Activity inputs are based on the reactor coolant concentrations given in Parameter 2.2.3. The values given are those that were judged to be representative for a typical BWR design.

2.2.13 CHEMICAL WASTES FROM REGENERATION OF CONDENSATE DEMINERALIZERS

2.2.13.1 Parameter

1. Liquid flows to demineralizer at main steam activity.

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

3. Use a regeneration cycle of 3.5 days times the number of demineralizers. (For systems using ultrasonic resin cleaning, use 8 days times the number of demineralizers.

2.2.13.2 Bases

Operating data from Dresden 2 and 3 indicate that one condensate demineralizer regeneration occurs every 3.5 days (Ref. 39) when ultrasonic cleaning is not used.

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. If ultrasonic resin cleaning is used to remove insolubles between regenerations, operating data from Dresden 2 and 3 indicates that one condensate demineralizer regeneration occurs every 7.1 days (Ref. 40 and 41), from Pilgrim 1 at 8.2 days (Ref. 42) and from Nine Mile Point 1 at 10 days (Ref. 41).

2.2.14 DETERGENT WASTE

2.2.14.1 Parameter

For plants with an onsite laundry, use the radionuclide distribution given in Table 2-28 for untreated detergent wastes. The quantities shown in Table 2-28 are added to the adjusted liquid source term. They are reduced for any treatment provided using the appropriate decontamination factors.

2.2.14.2 Bases

In the evaluation of liquid radwaste treatment systems, it is assumed that detergent wastes (laundry drains, personnel and equipment decontamination drains, and cask cleaning drains) will total approximately 1000 gal/day per reactor. The radionuclide distribution given in Table 2-28 is based on data given in Table 2-29.

TABLE 2-27

BWR LIQUID WASTES

SOURCE	EXPECTED DAILY AVERAGE INPUT FLOW RATE (in gal/day): DEEP BED PLANT WITH ULTRASONIC RESIN CLEANER	EXPECTED DAILY AVERAGE INPUT FLOW RATE (in gal/day): DEEP BED PLANT WITHOUT ULTRASONIC RESIN CLEANER OR A FILTER/DEMINERALIZER PLANT	FRACTION OF THE PRIMARY COOLANT ACTIVITY (PCA)
Equipment Drains			
Drywell	3,400	3,400	1.00
Containment, auxiliary building, and fuel pool	3,700	3,700	0.1
Radwaste building	1,100	1,100	0.1
Turbine building	3,000	3,000	0.001
Ultrasonic resin cleaner	15,000	-	0.05
Resin rinse*	2,500	5,000	0.002
Floor Drains			
Drywell	700	700	0.001
Containment, auxiliary building, and fuel handling	2,000	2,000	0.001
Radwaste building	1,000	1,000	0.001
Turbine building	2,000	2,000	0.001
Other Sources			
Cleanup phase separator decant	640	640	0.002
Laundry drains	1,000	1,000	-
Lab drains	500	500	0.02
Regenerants*	1,700	3,400	**
Condensate demineralizer backwash.	-	8,100	2×10^{-6}
Chemical lab waste	100	100	0.02

*Deep-bed condensate demineralizers only.

**Calculated by BWR-GALE Code.

†Filter/demineralizer (Powdex) condensate demineralizers only.

TABLE 2-28

CALCULATED ANNUAL RELEASE OF RADIOACTIVE MATERIALS IN UNTREATED DETERGENT WASTE FOR A BWR

NUCLIDE		Ci/yr
C-14		2.8(-4)
P-32		1.7(-4)
Cr-51		9.1(-3)
Mn-54		4.6(-3)
Fe-55		9.6(-3)
Fe-59		1.6(-4)
Co-57		1.3(-4)
Co-56		9.3(-3)
Co-60		1.6(-2)
Ni-63		2.5(-4)
Sr-89		1.1(-4)
Sr-90		5.8(-5)
Y-91		1.6(-4)
Zr-95		1.5(-3)
Nb-95		1.8(-3)
Mo-99		6(-5)
Ru-103		3.1(-4)
Ru-106		3(-4)
Ag-110m		6(-4)
Sb-124		6.5(-4)
I-131		2(-3)
Cs-134		1.1(-2)
Cs-136		5.6(-4)
Cs-137		1.6(-2)
Ba-140		9.1(-4)
Ce-141		2(-4)
Ce-144		3.5(-3)
	TOTAL	0.09 Ci

TABLE 2-29

RADIONUCLIDE DISTRIBUTION OF DETERGENT WASTE
(MILLICURIES/MONTH)

NUCLIDE	Oyster Creek[a] (1971-1973)	Ginna[b] (1972-1973)	Zion[c] (1977)	Fort Calhoun[d] (1977)
C-14	1.2(-2)	NA[e]	2(-2)	4.2(-2)
P-32	1.5(-2)	NA	NA	NA
Cr-51	2.3(-1)	NA	1.3	NA
Mn-54	1.3	1.1(-1)	1.3(-1)	2.2(-2)
Fe-55	3.5(-1)	NA	1.9	1.6(-1)
Fe-59	2.9(-2)	NA	2.6(-1)	NA
Co-57	7.5(-3)	NA	1.7(-2)	NA
Co-58	3.5(-1)	4.1(-1)	2.3	1(-1)
Co-60	3.8	9(-1)	9.1(-1)	4(-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	7.4(-3)	NA
Y-91	NA	NA	1.4(-2)	NA
Zr-95	8.3(-2)	1.4(-1)	1.4(-1)	NA
Nb-95	1.6(-2)	2(-1)	2.7(-1)	NA
Mo-99	NA	5(-3)	NA	NA
Ru-103	1.3(-2)	1.4(-2)	5.2(-2)	NA
Ru-106	NA	2.5(-1)	NA	NA)
Ag-110m	NA	5(-2)	NA	NA
Sb-124	6.1(-2)	NA	4.7(-2)	NA
I-131	4.3(-1)	6(-2)	1.7(-1)	1.6(-2)
Cs-134	1.7(-1)	1.4	1.4	1.0
Cs-136	NA	NA	4.7(-2)	NA
Cs-137	2.9(-1)	2.5	2.0	1.2
Ba-140	7.6(-2)	NA	NA	NA
Ce-141	3.3(-2)	1(-3)	NA	NA
Ce-144	7.3(-2)	5.3(-1)	NA	NA
TOTAL	7.4	6.6	11.3	2.7

[a] U.S.E.P.A., EPA-520/5-76-003, "Radiological Surveillance Studies at the Oyster Creek BWR Nuclear Generating Station," June 1976.

[b] Westinghouse Corporation WCAP-8253, "Source Term Data for Westinghouse Pressurized Water Reactors," May 1974.

[c] EG&G Idaho, Inc. and Allied Chemical Corp., Idaho National Engineering Laboratory, "Draft Report, In-Plant-Measurements at Zion Station," 1976.

[d] NUREG/CR-0140, "In-Plant Source Term Measurements at Ft. Calhoun Station, Unit 1," August 1978.

[e] NA, Radionuclides were not analyzed.

2.2.15 TRITIUM RELEASES

2.2.15.1 Parameter

The total tritium release through liquid and gaseous pathways is 0.03 Ci/yr per MWt. The quantity of tritium released through the liquid pathway is approximately 50% of the total quantity of tritium calculated to be available for release. The remainder of the tritium produced is assumed to be released as a gas from building ventilation exhaust systems. 50% of the tritium in gaseous effluents is released from the turbine building ventilation system and the remaining 50% of the tritium in gaseous effluents is released from the containment building ventilation system. For "zero liquid release" plants, assign all of the tritium calculated to be available for release to building ventilation exhaust systems.

2.2.15.2 Bases

Table 2-30 lists the measured liquid and gaseous tritium releases from BWRs for 1972 through 1977. Based on the total tritium release for each facility, the integrated thermal power produced during the year, and a plant capacity factor of 80%, the total annual release is approximately 0.03 Ci/MWt through the combined liquid and gaseous pathways.

The tritium can be released either in liquid wastes or as a gas with ventilation effluents, the relative amounts being dependent on liquid recycle practices. Table 2-31 lists the percentage of total tritium which is released in liquid effluents (based on the data in Table 2-30). The weighted average* indicates that approximately 50% of the tritium available for release is released in liquid effluents.

Tritium in gaseous effluents is released largely through building ventilation exhaust systems. Based on measurements taken in 1974 and 1975 of tritium release rates in building ventilation systems at Monticello, Vermont Yankee, and Oyster Creek (Ref.15), Table 2-32 provides the distribution of tritium released from various sources within the plant. Based on data in Table 2-32, approximately 50% of the tritium in gaseous effluents is released through the turbine building ventilation systems. Assuming that miscellaneous sources (radwaste building ventilation, fuel pool area) are released via the reactor building vent, the remaining 50% of the tritium in gaseous effluents is released through the reactor building ventilation system. Although it is recognized that tritium should be released via the gaseous pathway from the fuel handling area, data is available only from operating reactors (Mark I containments) where the spent fuel pool area is inside containment. It is not possible with the present data base to identify what fraction of the tritium from the reactor building is associated with the spent fuel pool area. Accordingly, until sufficient data is available, tritium releases from the spent fuel pool area will be considered to be released from the containment building, even if the spent fuel pool is located elswhere (BWR/6 Mark III's).

2.2.16 DECONTAMINATION FACTORS FOR DEMINERALIZERS

2.2.16.1 Parameter

The following are the expected decontamination factor (DFs) for demineralizers used on process or radwaste streams.

DEMINERALIZER TYPE	DECONTAMINATION FACTORS* ANION	Cs, Rb	OTHER NUCLIDES
Mixed Bed (H^+ OH^-)			
Reactor Coolant	10	2	10
Condensate	10	2	10
Clean waste	10^2(10)	10(10)	10^2(10)
Dirty waste (floor drains)	10^2(10)	2(10)	10^2(10)
Cation Bed (H^+)			
Dirty waste	1(1)	10(10)	10^2(10)
Powdex (any system)	10(10)	2(10)	10(10)

*For an evaporator polishing demineralizer or for the second demineralizer in series, the DF is given in the parentheses.

TABLE 2-30

TRITIUM RELEASE DATA FROM OPERATING BWRS*

REACTOR NAME	POWER MWT	STARTUP DATE	NUCLEAR THERMAL OUTPUT 10^6 MWDt						TRITIUM RELEASED (Ci/Yr) GASEOUS						TRITIUM RELEASED (Ci/Yr) LIQUID						TOTAL TRITIUM RELEASED (Ci/yr-MWt at 80% capacity)					
			1972	1973	1974	1975	1976	1977	1972	1973	1974	1975	1976	1977#	1972	1973	1974	1975	1976	1977#	1972	1973	1974	1975	1976	1977
Oyster Creek	1930	1969	0.54	0.45	0.46	0.41	0.49	0.41	0.8	0.4	0.4	2.8	1.1	0.7	62	36.6	14.1	18	38	3.4	0.034	0.024	0.009	0.015	0.023	0.003
Nine Mile Point 1	1850	1969	0.42	0.46	0.44	0.40	0.55	0.38	18	26.8	**	20	19	33	28	46.5	18.7	28	2.5	0.5	0.032	0.047	0.012	0.035	0.011	0.026
Dresden 2/3	2577	1970/71	1.04	1.18	0.91	0.70	1.15	1.20	31	10	11	220	170	330	26	26	22.6	54	20	0.1	0.016	0.009	0.011	0.11	0.048	0.081
Millstone 1	2011	1970	0.40	0.25	0.47	0.50	0.48	0.62	4.2	1.7	2.8	17	29	33	21	3.7	24.1	80	20	7.5	0.018	0.006	0.017	0.057	0.030	0.019
Monticello	1670	1970	0.46	0.41	0.37	0.37	0.51	0.46	12	**	**	66	77	139	***	***	***	0	0	0	0.008	-	-	0.052	0.044	0.088
Vermont Yankee	1593	1972	0.06	0.18	0.34	0.47	0.42	0.46	†	1.0	0.9	7.1	14	28	†	0.2	***	0	1.6	0.1	-	0.002	0.001	0.004	0.011	0.018
Quad Cities 1/2	2511	1971/72	0.52	1.32	1.09	0.96	1.08	1.12	4.7	34††	29	280	300	40	4.7	24.5	34	54	24	19	0.005	0.013	0.017	0.10	0.088	0.016
Pilgrim 1	1998	1972	0.11	0.53	0.25	0.34	0.32	0.34	**	14	8	74	37	61	4.2	0.4	10.4	18	47	33	0.011	0.008	0.22	0.079	0.077	0.080
Peach Bottom 2/3	3293	1973/74				1.39	1.55	1.19				0.3	27	260				31	74	71				0.007	0.019	0.081
Browns Ferry 1-3	3293	73/74/76				0.36	0.37	2.24				5.1	0.6	23				10	4.0	22				0.012	0.004	0.006
Cooper	2381	1974				0.52	0.49	0.60				43	67	50				8.3	8.3	9.0				0.029	0.045	0.029
Hatch 1	2436	1974				0.41	0.57	0.51				1.8	1.4	1.2				6.1	9.0	14				0.006	0.005	0.009
Fitzpatrick	2436	1974				0.28	0.53	0.49				**	15	9.5				**	4.2	28				-	0.011	0.007
Duane Arnold	1658	1974				0.31	0.33	0.39				19	16	15				0.3	0.3	0.2				0.018	0.014	0.011
Brunswick 1/2	2436	1975/76				0.20	0.33	0.66				2.0	22	19				3.2	5.9	7.4				0.008	0.025	0.012
WEIGHTED AVERAGE†††																					0.021	0.016	0.015	0.043	0.035	0.032

*Data from semiannual reports of reactor listed.

**No reported data.

***No measurement made.

†Prior to first refueling.

††Measured only during the July-December 1973 period.

†††Average weighted by nuclear thermal output.

#Data for first half of 1977 have been extrapolated to the end of 1977 for Oyster Creek, Nine Mile Point-1, Millstone-1, Monticello, Browns Ferry 1, 2 and 3, Hatch-1, Fitzpatrick and Brunswick 1 and 2.

TABLE 2-31

TRITIUM RELEASE DATA FROM OPERATING BWR'S
PERCENT OF TOTAL TRITIUM RELEASED IN LIQUID EFFLUENTS

REACTOR	1972	1973	1974	1975	1976	1977
Oyster Creek	98.7	91.5	97.2	86.5	97.2	69.4
Nine Mile Point 1	60.9	63.4	**	58.3	11.6	1.6
Dresden 2/3	45.6	72.2	67.3	31.0	10.5	0
Millstone 1	83.3	68.5	89.6	82.5	40.8	18.6
Monticello	**	**	**	0	0	0
Vermont Yankee		16.7	**	0	10.3	0.4
Quad Cities 1/2	50.0	41.9	54.0	16.2	7.4	32.5
Pilgrim 1	**	2.8	56.8	19.6	56.0	35.1
Peach Bottom 2/3				99.0	73.3	21.5
Browns Ferry 1/2/3				66.2	87.3	47.7
Cooper				16.2	11.0	15.3
Hatch 1				77.2	86.5	92.1
Fitzpatrick				**	21.9	22.5
Duane Arnold				1.5	2.0	1.4
Brunswick 1/2				61.5	21.1	27.8
*Weighted Average	63.4	53.1	69.5	51.4	36.2	28.2

*Average weighted by thermal nuclear output

**Insufficient Data

•Prior to first refueling

TABLE 2-32

DISTRIBUTION OF TRITIUM RELEASE IN GASEOUS EFFLUENTS (Ref. 15)

PLANT	SOURCE OF GASEOUS TRITIUM RELEASE (% OF TOTAL) REACTOR BUILDING	TURBINE BUILDING	MISCELLANEOUS	TOTAL
Monticello	68	29	3	100
Vermont Yankee	35	53	12	100
Oyster Creek	13	79	8	100
AVERAGE	39	54	7	100

2.2.16.2 Bases

The DFs for demineralizers used in the evaluation of liquid waste treatment systems are derived from the findings of a generic review in the nuclear industry by ORNL (Ref. 41). This reference contains operating and theoretical data that provides a basis for the numerical values assigned. The information contained in this report was projected to obtain a performance value expected over an extended period of operation. It was also considered that attempts to extend the service life of the resin will reduce the DFs below those expected under controlled operating conditions.

The following operating conditions were factored into the evaluation of demineralizer performance:

1. In general, the DF for waste treatment systems will vary with the quality of the water to be treated, increasing with increasing activity. Normally, when two demineralizers are used in series, the first demineralizer will have a higher DF than the second. However, the data in Reference 41 indicate that Cs and Rb will be more strongly exchanged in the second demineralizer in series than in the first, since the concentration of preferentially exchanged competing nuclides is reduced.

2. As indicated in Reference 41, 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 area provided by demineralizer resin lead to 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 normally undetectable in the effluent streams.

2.2.17 DECONTAMINATION FACTORS FOR EVAPORATORS

2.2.17.1 Parameter

	ALL NUCLIDES EXCEPT ANIONS	ANIONS
Miscellaneous radwaste evaporator	10^4	10^3
Separate evaporator for detergent wastes	10^2	10^2

2.2.17.2 Bases

The decontamination factors for evaporators are derived from the findings of a generic review by ORNL of evaporators used in the nuclear industry (Ref. 43). The principal conclusions reached in the report are:

1. Decontamination factors of 10^4 can be expected for nonvolatile radioactive nuclides in a single-stage evaporator.
2. Decontamination factors for iodine are a factor of 10 less than the DFs for nonvolatile nuclides (10^3).
3. Decontamination factors for wastes containing detergents that tend to foam are a factor of 10 to 100 lower than DFs expected for nonfoaming wastes.

These conclusions have been extended to take into account the following factors:

1. For nonvolatile nuclides in a nonfoaming solution, a DF of 10^4 is used.
2. For iodine in a nonfoaming solution, a DF of 10^3 is used.
3. If an evaporator is used for detergent wastes, the DF for the evaporator is reduced to 100 to reflect carryover due to foaming, which will reduce the DF.

2.2.18 DECONTAMINATION FACTORS FOR REVERSE OSMOSIS

2.2.18.1 Parameter

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

2.2.18.2 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. For source term calculations, the system DF 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 (percent recovery).

Tables 2-33 through 2-35 give membrane DFs derived from operating data at Point Beach and Ginna (Refs. 45 and 46) and laboratory data on simulated radwaste liquids (Ref. 47). 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.

$$DF_s = \frac{0.95}{1 - (1 - .95)^{1/100}} = 30$$

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 and 2-35, the membrane DF for these isotopes is lower than the average membrane DF used in the evaluation for laundry waste.

2.2.19 DECONTAMINATION FACTORS FOR LIQUID RADWASTE FILTERS

2.2.19.1 Parameter

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

2.2.19.2 Bases

Reference 44 contains the findings of a generic review by ORNL of liquid radwaste filters used in the nuclear industry. Due to the various filter types and filter media employed, reported decontamination factors vary widely, with no discernible trend. The principal conclusion reached in the ORNL report is that no credit should be assigned liquid radwaste filters (DF of 1) until a larger data base is obtained.

2.2.20 ADJUSTMENT TO LIQUID RADWASTE SOURCE TERMS FOR ANTICIPATED OPERATIONAL OCCURRENCES

2.2.20.1 Parameter

1. Increase the calculated source term by 0.1 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 holdup 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:

TABLE 2-33

REVERSE OSMOSIS DECONTAMINATION FACTORS, GINNA STATION

NUCLIDE	FEED ACTIVITY ($\mu Ci/cm^3$)	PRODUCT ACTIVITY ($\mu Ci/cm^3$)	MEMBRANE DF
Ce-144	2.68×10^{-4}	$<2.2 \times 10^{-7}$	1200
Co-58	8.55×10^{-5}	$<3.4 \times 10^{-8}$	2500
Ru-103	5.83×10^{-5}	$<5.5 \times 10^{-8}$	1100
Cs-137	4.09×10^{-4}	6.6×10^{-6}	60
Cs-134	2.02×10^{-4}	3.2×10^{-6}	60
Nb-95	5.35×10^{-5}	$<5.3 \times 10^{-8}$	1000
Zr-95	2.36×10^{-5}	$<3.7 \times 10^{-8}$	640
Mn-54	8.82×10^{-5}	$<3.4 \times 10^{-8}$	2600
Co-60	9.62×10^{-4}	$<8.1 \times 10^{-8}$	12,000
Total isotopic	2.15×10^{-3}	9.8×10^{-6}	220
Gross β	1.63×10^{-3}	1.86×10^{-5}	88
Average*			200

*The average DF is calculated from the average of the reciprocals of the isotopic DFs.

TABLE 2-34
REVERSE OSMOSIS DECONTAMINATION FACTORS, POINT BEACH

DATE	TIME	FEED ACTIVITY (μCi/ml)	PRODUCT ACTIVITY (μCi/ml)	MEMBRANE DF
6/14/71	0840	1.1×10^{-5}	6.8×10^{-7}	16
	1225	6.3×10^{-5}	4.2×10^{-7}	150
	1530	8.8×10^{-5}	3.2×10^{-7}	280
6/15/71	1030	2.7×10^{-4}	3.1×10^{-6}	87
	1315	1.0×10^{-4}	1.7×10^{-6}	59
	1440	1.3×10^{-4}	1.1×10^{-7}	1200
	1510	1.6×10^{-4}	1.1×10^{-7}	1500
	1530	1.8×10^{-4}	5.7×10^{-7}	320

TABLE 2-35
EXPECTED REVERSE OSMOSIS DECONTAMINATION FACTORS FOR SPECIFIC NUCLIDES

NUCLIDE	FEED ACTIVITY (μCi/ml)	PRODUCT ACTIVITY (μCi/ml)	MEMBRANE DF
Co-60	2.5×10^{-4}	5×10^{-7}	500
Mo-99	3.8×10^{-2}	1×10^{-3}	40
I-131, 132, 133, 134, 135	1.2×10^{-1}	4×10^{-3}	30
Cs-134, 137	4.3×10^{-2}	2×10^{-4}	200

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 unavailable.

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:

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

2.2.20.2 Bases

Reactor operating data over an 8 year period, January 1970 through December 1977, representing 127 reactor years of operation were evaluated to determine the frequency and extent of unplanned liquid releases. During the period evaluated, 50 unplanned liquid releases occurred; 28 due to operator errors, 13 due to component failures, 5 due to inadequate procedures or failure to follow procedures, and the remainder (4) due to miscellaneous causes such as design errors. Table 2-36 summarizes the findings of this evaluation. Based on the data provided in Table 2-36 it is estimated that 0.1 Ci/yr/reactor will be discharged in unplanned releases in liquid effluents. Tritium releases for BWR anticipated operational occurrences were less than 1% of the total normal operational release value, and was, therefore, judged to be negligible.

The availability for 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.

2.2.21 GUIDELINES FOR ROUNDING OFF NUMERICAL VALUES

In calculating the estimated annual release of radioactive materials in liquid and gaseous wastes, round off all numerical values to two significant figures.

TABLE 2-36

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

UNPLANNED LIQUID RELEASES	
Total number (unplanned releases)	50
Fraction due to personnel error	0.56
Fraction due to component failure	0.26
Fraction due to inadequate procedures or failure to follow procedures	0.10
Fraction due to other causes	0.08
Approximate activity (Ci)	10.62
Fraction of cumulative occurrences per reactor year (plants reporting releases >5 gals of liquid waste/reactor year)	0.15
Fraction of cumulative occurrences per reactor year (plants reporting activity released >0.01 mCi/reactor year	0.27
Activity per release (Ci/release)	0.30
Activity released per reactor year (Ci/reactor/year)	0.10
Volume of release per reactor year (gal/reactor year)	1.66×10^4

*Values in this table are based on reported values in 1970-1977 Licensee Event Reports

2.2.22 CARBON-14 RELEASES

2.2.22.1 Parameter

The estimated annual quantity of carbon-14 released from a boiling water reactor is 9.5 Ci/yr. It is assumed that the carbon-14 reacts with oxygen in the reactor water and behaves like a noble gas fission product; thus all carbon-14 produced will be released through the main condenser offgas system.

2.2.22.2 Bases

The principal source of carbon-14 is the thermal neutron reaction with oxygen-17 in the reactor coolant. The production rate of carbon-14 from oxygen-17 is given by the equation:

$$Q = N_o \sigma_o \phi m t p s \text{ (Ci/yr)}$$

where

m is the 3.9×10^4 kg, mass of water in reactor core;

N_o is the 1.3×10^{22} atoms 0-17/kg natural water;

p is the 0.80, plant capacity factor;

s is the 1.03×10^{-22} Ci/atom, specific activity for C-14;

t is the 3.15×10^7 sec/yr, maximum irradiation time per year;

σ_o is the 2.4×10^{-25} cm^2, thermal neutron cross section for 0-17; and

ϕ is the 3×10^{13} neutrons/cm^2-sec, average thermal neutron flux.

Based on the above parameters, Q = 9.5 Ci/yr.

Carbon-14 can also be produced by neutron activation of nitrogen-14 dissolved in the reactor coolant and present in air in the drywell. These sources contribute a small fraction of a curie per year to the annual production of carbon-14 due to the low concentration of nitrogen-14 in the reactor coolant (less than 1 ppm by weight), and the low neutron flux in the drywell (approximately 4×10^8 neutrons/cm^2-sec).

The annual release of 9.5 Ci of carbon-14 is in good agreement with measurements at Nine Mile Point 1 reported by Kunz et al. (Ref. 48), who found that 8 curies per year of carbon-14 were released, principally in the form of CO_2.

2.2.23 ARGON-41 RELEASES

The argon-41 input to the main condenser offgas treatment system (MCOTS) downstream of the air ejectors, is 40 μCi/sec. The dynamic adsorption coefficients for argon-41 in charcoal delay beds of a MCOTS are 6.4 cm^3/gm for an ambient temperature charcoal system and 16 cm^3/gm for a chilled charcoal system. The holdup time for argon-41 in a charcoal delay system is determined using the equation in Section 2.2.9.1.

The argon-41 release from the purging or venting of the drywell is 15 Ci/yr.

2.2.23.2 Bases

Argon-41 is formed by neutron activation of stable naturally occurring argon-40. This reaction may occur with argon-40 present in the reactor coolant and also with argon-40 in the drywell air surrounding the reactor vessel.

Argon-40 will enter the reactor coolant as a part of air inleakage at or downstream of the main condenser. Argon-41 produced by activation of the argon-40 in the reactor vessel will be transported to the main condenser offgas treatment system (MCOTS). Data in reference 49 and summarized in Table 2-37 indicates that the argon-41 input to the MCOTS during the measurements ranged from 5.6 μCi/sec to 37 μCi/sec. Due to the limited duration of these measurements, the mean value of the data is not used. Instead, in these evaluations a release rate of 40 μCi/sec is considered to be a value that is not likely to be exceeded, on the average, over the 30 year life of the plant.

Argon-41 will be held up in charcoal delay beds of the MCOTS in the same manner as discussed for xenon and krypton in Section 2.2.9.2. Values of the dynamic adsorption coefficient are based on data contained in references 49, 50 and 51 for ambient and chilled temperature systems. Holdup times for argon-41 are determined using these k values and the delay equation in Section 2.2.9.1.

Argon-41 release from the drywell are based on data in reference 49 concerning the neutron flux in the drywell and on an assumed drywell purging frequency of 24 purges per year.

TABLE 2-37

SUMMARY OF ARGON-41 RELEASES
TO THE MAIN CONDENSER OFFGAS TREATMENT SYSTEM*
(μCi/sec)

PLANT	Argon-41 Release
Browns Ferry 1	38
Browns Ferry 2	17
"	12
Browns Ferry 3	7.1
"	5.8
"	7.4
"	34
"	32
"	19
Hatch 1	12
"	16
Fitzpatrick	36

*Data in this table are based on measured argon-41 release rates in reference 49 and were adjusted to 3400 MWt.

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

3.1 INTRODUCTION

This chapter contains additional information for using the BWR-GALE Code. Chapter 1 of this report described the entries required to be entered on input data cards, and Section 3.2 of this chapter contains sample input data sheets and flow charts 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 cards for a sample problem and the resultant output for that sample problem. Section 3.4 contains a discussion of the nuclear data library used and a FORTRAN listing of the BWR-GALE Code.

3.2 INPUT DATA SHEETS

The following pages (3-3 through 3-11) show (1) the form in which data should be entered on input data sheets and (2) a sample completed sheet and flow sheets for both the liquid and gas codes.

3.3 SAMPLE PROBLEM - INPUT AND OUTPUT

The following pages (3-12 through 3-18) show printouts of the input and output for a sample problem using the BWR-GALE Code.

3.4 LISTING OF BWR-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.
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)
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

The remainder of this chapter (pages 3-19 through 3-58) provides the program listing for the BWR-GALE Code.

NRC FORM 53A
(12-75)

U.S. NUCLEAR REGULATORY COMMISSION
ADP TRANSCRIPTION SHEET

JOB TITLE BWR-GALE Code

OFFICE

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DATE December 1978

PREPARED BY

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CARD  1   NAME   NAME OF REACTOR (                         )        TYPE=    BWR
CARD  2   POWTH  THERMAL POWER LEVEL (MEGAWATTS)                        (      )
CARD  3   GTO    TOTAL STEAM FLOW (MILLION LBSH/HR)                     (      )
CARD  4   WLIC   MASS OF WATER IN REACTOR VESSEL (MILLION LBS)          (      )
CARD  5   GDE    CLEAN-UP DEMINERALIZER FLOW (MILLION LBS HR)           (      )
CARD  6   REGENT CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS)      (      )
CARD  7   FFCDM  FRACTION FEEDWATER THROUGH CONDENSATE DEMIN            (      )
CARD  8          HIGH PURITY WASTE INPUT (      )GPD AT (   )PCA
CARD  9          DFI(       )DFCS (      )DFO (      )
CARD 10          COLLECTION(     )DAYS PROCESS (     )DAYS FRACT DISCH (    )
CARD 11          LOW PURITY WASTE INPUT(      )GPD AT (    )PCA
CARD 12          DFI(       )DFCS (      )DFO (      )
CARD 13          COLLECTION(     )DAYS PROCESS (     )DAYS FRACT DISCH (    )
CARD 14          CHEMICAL WASTE INPUT    (      ) GPD AT(    ) PCA
CARD 15          DFI(       )DFCS (      )DFO (      )
CARD 16          COLLECTION(    )DAYS PROCESS(    )DAYS FRACT DISCH     (    )
CARD 17          REGENERANT SOLTNS WASTE INPUT-GPD                      (      )
CARD 18          DFI(       )DFCS (      )DFO (      )
CARD 19          COLLECTION(   )DAYS PROCESS  (     )DAYS FRACT DISCH  (    )
CARD 20   GGS    GLAND SEAL STEAM FLOW (THOUSAND LBS/HR)                (      )
CARD 21   TIM 3  GLAND SEAL HOLD UP TIME (HOURS)                        (      )
CARD 22   TIM 4  AIR EJECTOR OFFGAS HOLDUP TIME (HOURS)                 (      )
```

NRC FORM 53A
(12 75)

U.S. NUCLEAR REGULATORY COMMISSION

ADP TRANSCRIPTION SHEET

JOB TITLE: BWR-GALE Code

OFFICE:

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PREPARED BY:

```
CARD 23          CONTAINMENT BLDG  CHARCOAL?(   )HEPA?(   )
CARD 24          TURBINE BLDG      CHARCOAL?(   )HEPA?(   )
CARD 25 FIL3     GLAND SEAL VENT IODINE PF
CARD 26 FIL4     AIR EJECTOR OFFGAS IODINE PF                         (      )
CARD 27          AUXILIARY BLDG    CHARCOAL?(   ) HEPA?(   )          (      )
CARD 28          RADWASTE BLDG     CHARCOAL?(   ) HEPA?(   )
CARD 29 KCHAR    CHARCOAL DECAY SYSTEM 0=NO,1=YES,2=CRYGENIC DISTILL       ()
CARD 30 KKR      KRYPTON DYNAMIC ADSORPTION COEFFICIENT (CM3/GM)      (      )
CARD 31 KXE      XENON DYNAMIC ADSORPTION COEFFICIENT (CM3/GM)        (      )
CARD 32 KMASS    MASS OF CHARCOAL (THOUSAND LBS)                      (      )
CARD 33 PFLAUN   DETERGENT WASTE DECON FACTOR                         (      )
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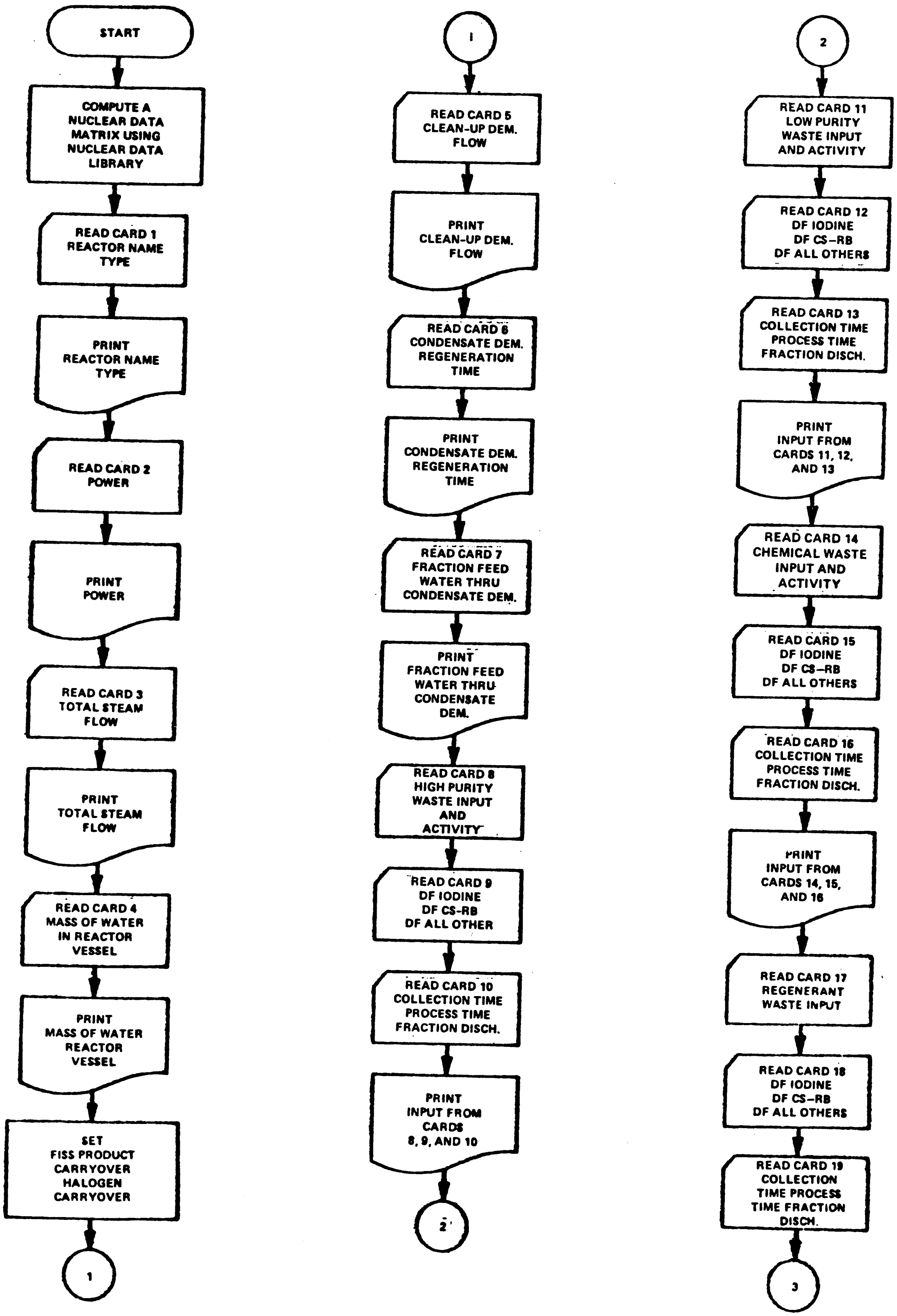
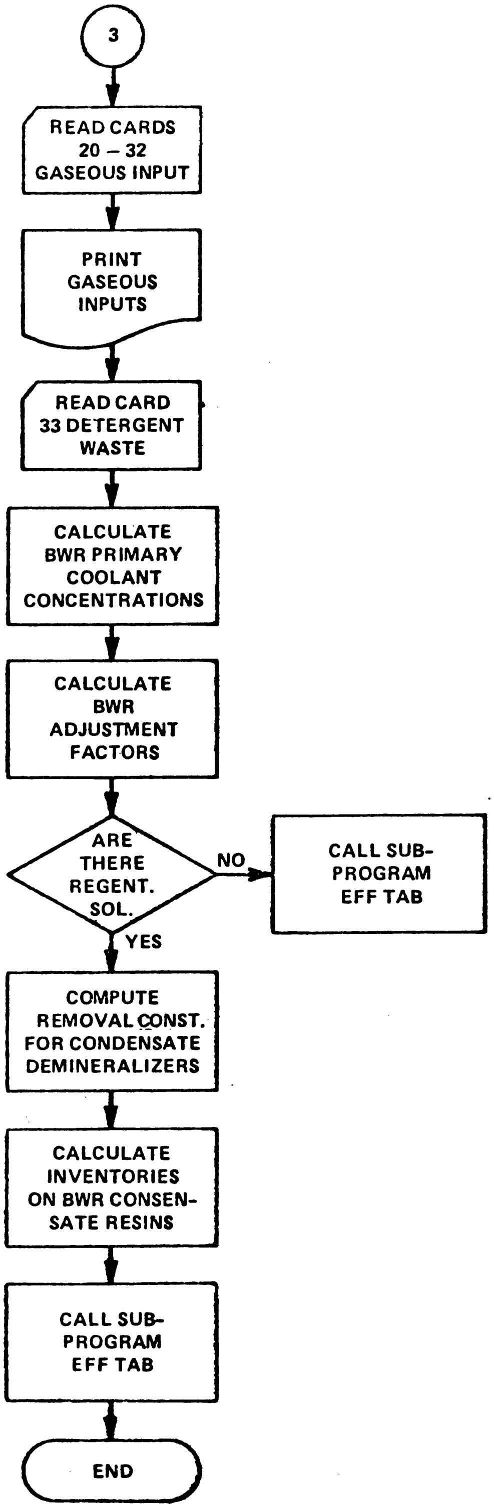
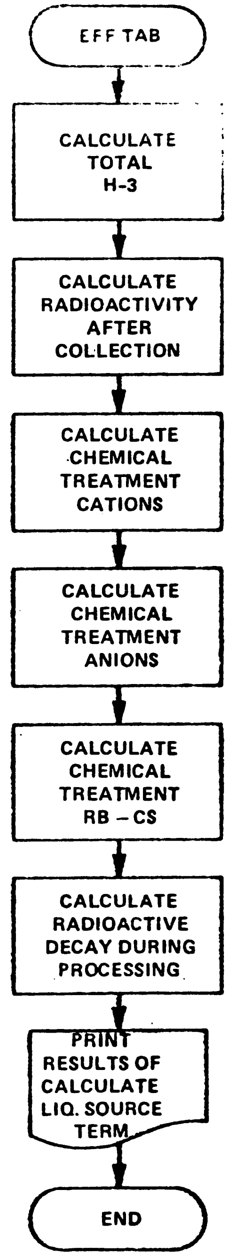
```
CARD  1  NAME   NAME OF REACTOR   SAMPLE BWR CASE 1                 TYPE =   BWR
CARD  2  POWTH  THERMAL POWER LEVEL (MEGAWATTS)                            3400.
CARD  3  GTO    TOTAL STEAM FLOW (MILLION LBS/HR)                          15.
CARD  4  WL1Q   MASS OF WATER IN REACTOR VESSEL (MILLION LBS)              0.38
CARD  5  GDE    CLEAN-UP DEMINERALIZER FLOW (MILLION LBS/HR)               0.13
CARD  6  REGENT CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS)          56.
CARD  7  FFCDM  FRACTION FEED WATER THROUGH CONDENSATE DEMIN               1.00
CARD  8         HIGH PURITY WASTE  INPUT  28640.  GPD AT .15  PCA
CARD  9         DFI=  1.0E03DFCS=  1.0E02DFO =  1.0E03
CARD 10         COLLECTION  1.0   DAYS PROCESS  0.07  DAYS FRACT DISCH   0.01
CARD 11         LOW PURITY WASTE   INPUT  5700.   GPD AT .13
CARD 12         DFI=  1.0E03DFCS=  1.0E04DFO =  1.0E04
CARD 13         COLLECTION 3.10   DAYS PROCESS  0.6   DAYS FRACT DISCH   1.0
CARD 14         CHEMICAL  WASTE   INPUT   600.    GPD AT .02  PCA
CARD 15         DFI=  1.0E03DFCS=  1.0E04DFO =  1.0E04
CARD 16         COLLECTION 3.1    DAYS PROCESS  .60   DAYS FRACT DISCH   1.0
CARD 17         REGENERATION SOLTNS INPUT  GPD                              1700.
CARD 18         DFI=  1.0E04DFCS=  1.0E05DFO =  1.0E05
CARD 19         COLLECTION  9.4   DAYS PROCESS  .44   DAYS FRACT DISCH   1.0
CARD 20  GGS    GLAND SEAL STEAM FLOW (THOUSAND LBS/HR)                    0.0
CARD 21  T1M3   GLAND SEAL HOLDUP TIME (HOURS)                             0.0
CARD 22  TIM4   AIR EJECTOR OFFGAS HOLDUP TIME (HOURS)                     .167
CARD 23         CONTAINMENT BLDG.CHARCOAL 90.0 HEPA?99.0
CARD 24         TURBINE BLDG.    CHARCOAL 00.0 HEPA?00.0
CARD 25  FIL3   GLAND SEAL VENT, IODINE PF                                 00.0
CARD 26  FIL4   AIR EJECTOR OFFGAS IODINE PF                              1.0
CARD 27         AUXILIARY BLDG.  CHARCOAL 00.0 HEPA?00.0
CARD 28         RADWASTE BLDG.   CHARCOAL 00.0 HEPA?99.0
CARD 29  KCHAR  CHARCOAL DELAY SYSTEM 0=NO,1=YES,2=CRYOGENIC DISTILL           1
CARD 30  KKR    KRYPTON DYNAMIC ADSORPTION COEFFICIENT (CM3/GM)            105.0
CARD 31  KXE    XENON DYNAMIC ADSORPTION COEFFICIENT (CM3/GM)             2410.0
CARD 32  KMASS  MASS OF CHARCOAL (THOUSAND LBS)                            48.
CARD 33  PFLAUN DETERGENT WASTE DECONTAMINATION FACTOR                     1.0
```

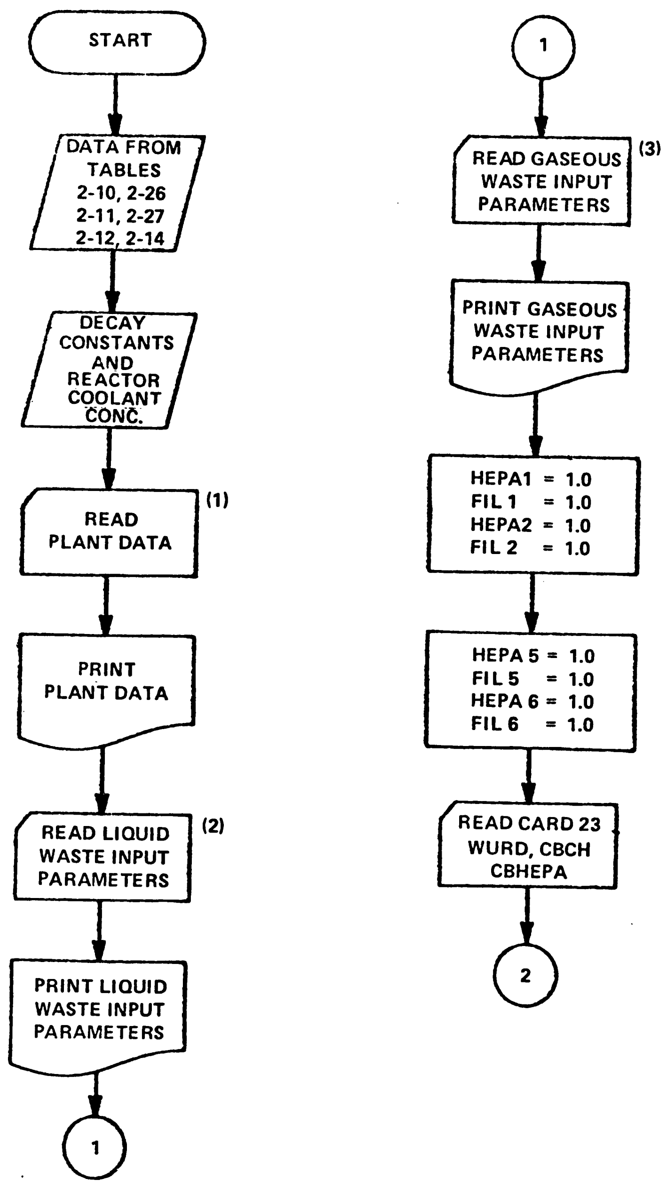
FLOW CHART FOR BWR LIQUID CODE



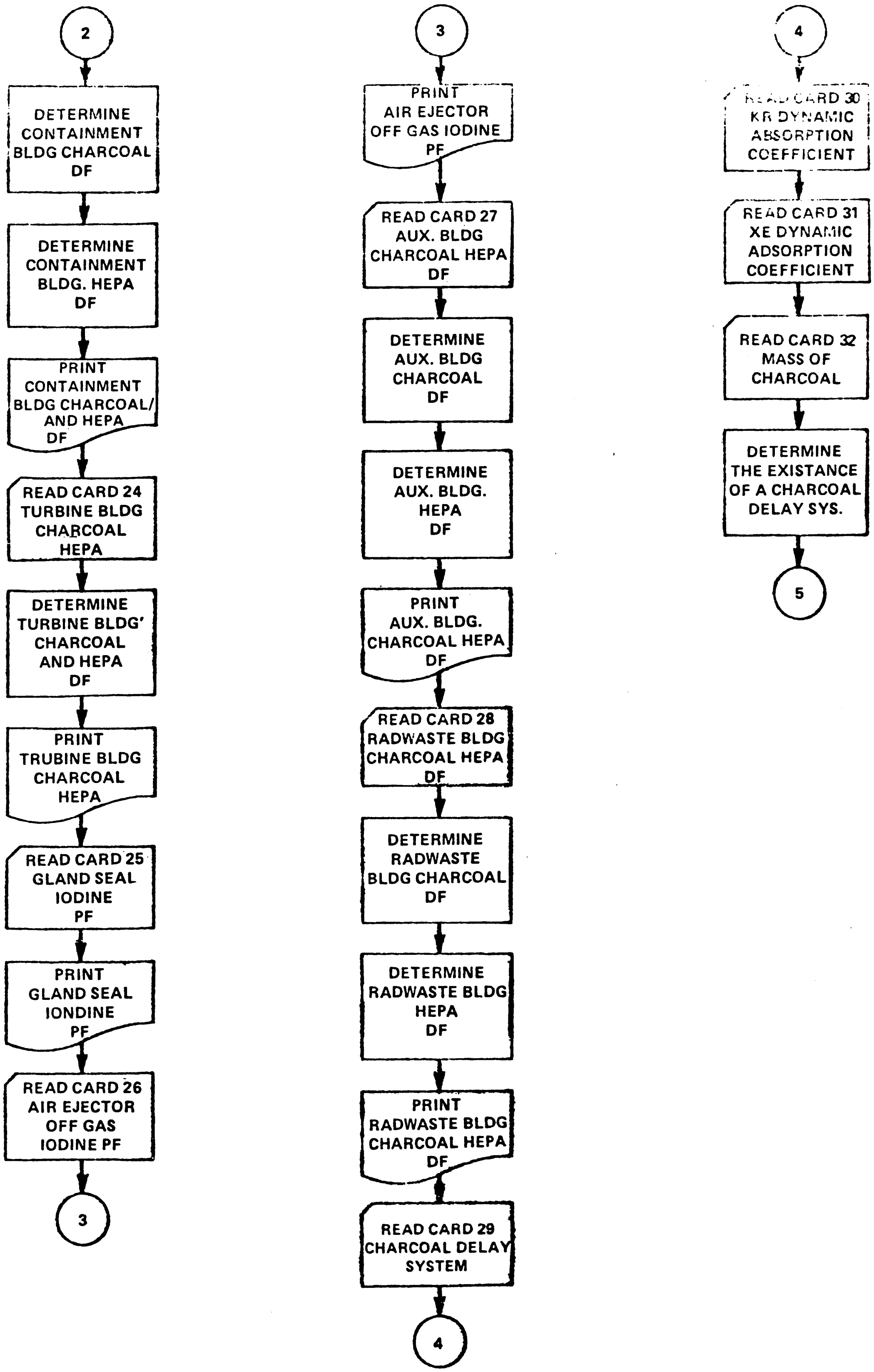

3
READ CARDS 20 – 32 GASEOUS INPUT
PRINT GASEOUS INPUTS
READ CARD 33 DETERGENT WASTE
CALCULATE BWR PRIMARY COOLANT CONCENTRATIONS
CALCULATE BWR ADJUSTMENT FACTORS
ARE THERE REGENT. SOL.
NO
CALL SUB-PROGRAM EFF TAB
YES
COMPUTE REMOVAL CONST. FOR CONDENSATE DEMINERALIZERS
CALCULATE INVENTORIES ON BWR CONSEN-SATE RESINS
CALL SUB-PROGRAM EFF TAB
END


EFF TAB
CALCULATE TOTAL H-3
CALCULATE RADIOACTIVITY AFTER COLLECTION
CALCULATE CHEMICAL TREATMENT CATIONS
CALCULATE CHEMICAL TREATMENT ANIONS
CALCULATE CHEMICAL TREATMENT RB – CS
CALCULATE RADIOACTIVE DECAY DURING PROCESSING
PRINT RESULTS OF CALCULATE LIQ. SOURCE TERM
END

FLOWCHART FOR BWR GAS CODE



(1) SEE SAMPLE INPUT CARDS 1 THRU 7.
(2) SEE SAMPLE INPUT CARDS 8 THRU 19.
(3) SEE SAMPLE INPUT CARDS 20 THRU 22.


2
DETERMINE CONTAINMENT BLDG CHARCOAL DF
DETERMINE CONTAINMENT BLDG. HEPA DF
PRINT CONTAINMENT BLDG CHARCOAL/ AND HEPA DF
READ CARD 24 TURBINE BLDG CHARCOAL HEPA
DETERMINE TURBINE BLDG' CHARCOAL AND HEPA DF
PRINT TRUBINE BLDG CHARCOAL HEPA
READ CARD 25 GLAND SEAL IODINE PF
PRINT GLAND SEAL IONDINE PF
READ CARD 26 AIR EJECTOR OFF GAS IODINE PF
3
3
PRINT AIR EJECTOR OFF GAS IODINE PF
READ CARD 27 AUX. BLDG CHARCOAL HEPA DF
DETERMINE AUX. BLDG CHARCOAL DF
DETERMINE AUX. BLDG. HEPA DF
PRINT AUX. BLDG. CHARCOAL HEPA DF
READ CARD 28 RADWASTE BLDG CHARCOAL HEPA DF
DETERMINE RADWASTE BLDG CHARCOAL DF
DETERMINE RADWASTE BLDG HEPA DF
PRINT RADWASTE BLDG CHARCOAL HEPA DF
READ CARD 29 CHARCOAL DELAY SYSTEM
4
4
CARD 30 KR DYNAMIC ABSORPTION COEFFICIENT
READ CARD 31 XE DYNAMIC ADSORPTION COEFFICIENT
READ CARD 32 MASS OF CHARCOAL
DETERMINE THE EXISTANCE OF A CHARCOAL DELAY SYS.
5

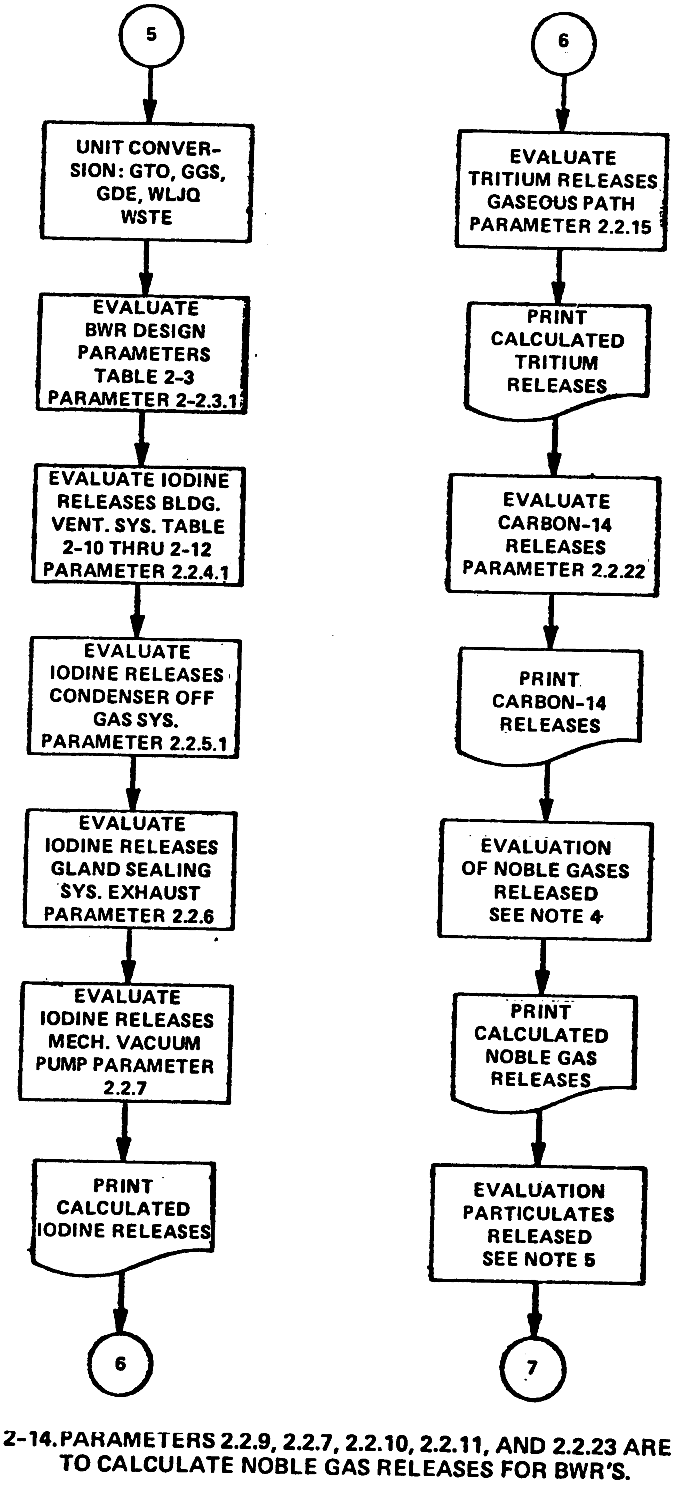


TABLE 2-14. PARAMETERS 2.2.9, 2.2.7, 2.2.10, 2.2.11, AND 2.2.23 ARE USED TO CALCULATE NOBLE GAS RELEASES FOR BWR'S.

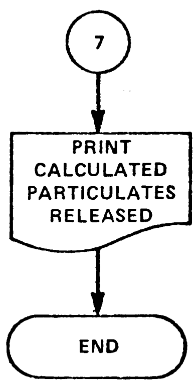


TABLE 2-14, PARAMETER 2.2.11 ARE USED TO
CALCULATE PARTICULATE RELEASES

SAMPLE BWR CASE 1

THERMAL POWER LEVEL (MEGAWATTS)	3400.00000
PLANT CAPACITY FACTOR	0.80
TOTAL STEAM FLOW (MILLION LBS/HR)	15.00000
MASS OF WATER IN REACTOR VESSEL (MILLION LBS)	0.38000
FISSION PRODUCT CARRY-OVER FRACTION	0.0010
HALOGEN CARRY-OVER FRACTION	0.0200
CLEAN-UP DEMINERALIZER FLOW (MILLION LBS/HR)	0.13000
CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS)	56.00000
FRACTION FEED WATER THROUGH CONDENSATE DEMIN	1.00000

LIQUID WASTE INPUTS

STREAM	FLOW RATE (GAL/DAY)	FRACTION OF PCA	FRACTION DISCHARGED	COLLECTION TIME (DAYS)	DECAY TIME (DAYS)	DECONTAMINATION FACTORS I	CS	OTHERS
HIGH PURITY WASTE	2.86E+04	0.150	0.010	1.000	0.070	1.00E+03	1.00E+02	1.00E+03
LOW PURITY WASTE	5.70E+03	0.130	1.000	3.100	0.600	1.00E+03	1.00E+04	1.00E+04
CHEMICAL WASTE	6.00E+02	0.020	1.000	3.100	0.600	1.00E+03	1.00E+04	1.00E+04
REGENERANT SOLS	1.70E+03		1.000	9.400	0.440	1.00E+04	1.00E+05	1.00E+05

GASEOUS WASTE INPUTS

GLAND SEAL STEAM FLOW (THOUSAND LBS/HR)		0.0
GLAND SEAL HOLDUP TIME (HOURS)		0.0
AIR EJECTOR OFFGAS HOLDUP TIME (HOURS)		0.16700
CONTAINMENT BLDG	IODINE RELEASE FRACTION	0.10000
	PARTICULATE RELEASE FRACTION	0.01000
TURBINE BLDG.	IODINE RELEASE FRACTIOM	1.00000
	PARTICULATE RELEASE FRACTION	1.00000
GLAND SEAL VENT, IODINE PF		1.00000
AIR EJECTOR OFFGAS IODINE PF		0.0
AUXILIARY BLDG.	IODINE RELEASE FRACTION	1.00000
	PARTICULATE RELEASE FRACTION	1.00000
RADWASTE BLDG.	IODINE RELEASE FRACTION	1.00000
	PARTICULATE RELEASE FRACTION	0.01000
THERE IS A CHARCOAL DELAY SYSTEM		
KRYPTON HOLDUP TIME (DAYS)		2.66823
XENON HOLDUP TIME (DAYS)		61.24234
KRYPTON DYNAMIC ADSORPTION COEFFICIENT (CM3/GM)		105.00000
XENON DYNAMIC ADSORPTION COEFFICIENT(CM3/GM)		2410.00000
MASS OF CHARCOAL(THOUSAND LBS)		48.00000

SAMPLE BWR CASE 1 LIQUID EFFLUENTS

NUCLIDE	HALF-LIFE (DAYS)	CONCENTRATION IN PRIMARY COOLANT (MICRO CI/ML)	ANNUAL RELEASES TO DISCHARGE CANAL: HIGH PURITY (CURIES)	LOW PURITY (CURIES)	CHEMICAL (CURIES)	TOTAL LWS (CURIES)	ADJUSTED TOTAL (CI/YR)	DETERGENT WASTES (CI/YR)	TOTAL (CI/YR)
CORROSION AND ACTIVATION PRODUCTS									
NA 24	6.25E-01	9.18E-03	0.00030	0.00014	0.00000	0.00044	0.00155	0.0	0.00160
P 32	1.43E+01	1.81E-04	0.00001	0.00002	0.00003	0.00006	0.00020	0.0	0.00020
CR 51	2.78E+01	5.44E-03	0.00032	0.00053	0.00177	0.00262	0.00918	0.0	0.00920
MN 54	3.03E+02	6.35E-05	0.00000	0.00001	0.00004	0.00006	0.00019	0.00100	0.00120
MN 56	1.07E-01	4.74E-02	0.00028	0.00001	0.00000	0.00028	0.00099	0.0	0.00099
FE 55	9.50E+02	9.07E-04	0.00005	0.00009	0.00068	0.00082	0.00289	0.0	0.00290
FE 59	4.50E+01	2.72E-05	0.00000	0.00000	0.00001	0.00002	0.00006	0.0	0.00006
CO 58	7.13E+01	1.81E-04	0.00001	0.00002	0.00010	0.00013	0.00044	0.00400	0.00440
CO 60	1.92E+03	3.63E-04	0.00002	0.00004	0.00028	0.00033	0.00117	0.00870	0.00990
CU 64	5.33E-01	2.76E-02	0.00084	0.00032	0.00001	0.00116	0.00406	0.0	0.00410
ZN 65	2.45E+02	1.81E-04	0.00001	0.00002	0.00013	0.00015	0.00054	0.0	0.00054
ZN 69M	5.75E-01	1.84E-03	0.00006	0.00002	0.00000	0.00008	0.00029	0.0	0.00029
ZN 69	3.96E-02	0.0	0.00006	0.00003	0.00000	0.00009	0.00031	0.0	0.00031
ZR 95	6.50E+01	0.0	0.0	0.0	0.0	0.0	0.0	0.00140	0.00140
NB 95	3.50E+01	0.0	0.0	0.0	0.0	0.0	0.0	0.00200	0.00200
W187	9.96E-01	2.74E-04	0.00001	0.00001	0.00000	0.00002	0.00007	0.0	0.00007
NP239	2.35E+00	6.37E-03	0.00032	0.00036	0.00002	0.00070	0.00246	0.0	0.00250
FISSION PRODUCTS									
BR 83	1.00E-01	3.45E-03	0.00002	0.00000	0.00000	0.00002	0.00007	0.0	0.00007
SR 89	5.20E+01	9.07E-05	0.00001	0.00001	0.00004	0.00006	0.00020	0.0	0.00020
SR 90	1.03E+04	6.35E-06	0.00000	0.00000	0.00000	0.00001	0.00002	0.0	0.00002
Y 90	2.67E+00	0.0	0.00000	0.00000	0.00000	0.00001	0.00002	0.0	0.00002
SR 91	4.03E-01	3.69E-03	0.00009	0.00003	0.00000	0.00012	0.00042	0.0	0.00042
Y 91M	3.47E-02	0.0	0.00006	0.00002	0.00000	0.00007	0.00026	0.0	0.00026
Y 91	5.88E+01	3.63E-05	0.00000	0.00001	0.00003	0.00004	0.00014	0.0	0.00014
SR 92	1.13E-01	9.47E-03	0.00006	0.00000	0.00000	0.00006	0.00021	0.0	0.00021
Y 92	1.47E-01	5.64E-03	0.00014	0.00001	0.00000	0.00015	0.00051	0.0	0.00051
Y 93	4.25E-01	3.69E-03	0.00010	0.00003	0.00000	0.00012	0.00044	0.0	0.00044
ZR 95	6.50E+01	7.25E-06	0.00000	0.00000	0.00000	0.00000	0.00002	0.0	0.00002
NB 95	3.50E+01	7.26E-06	0.00000	0.00000	0.00000	0.00001	0.00002	0.0	0.00002
NB 98	3.54E-02	3.89E-03	0.00000	0.00000	0.00000	0.00000	0.00001	0.0	0.00001
MO 99	2.79E+00	1.82E-03	0.00009	0.00011	0.00001	0.00022	0.00076	0.0	0.00076
TC 99M	2.50E-01	1.86E-02	0.00037	0.00015	0.00001	0.00053	0.00185	0.0	0.00180
RU103	3.96E+01	1.81E-05	0.00000	0.00000	0.00001	0.00001	0.00004	0.00014	0.00018
RH103M	3.96E-02	0.0	0.00000	0.00000	0.00001	0.00001	0.00004	0.0	0.00004
RU105	1.85E-01	1.87E-03	0.00002	0.00000	0.00000	0.00002	0.00008	0.0	0.00008
RH105M	5.21E-04	0.0	0.00002	0.00000	0.00000	0.00002	0.00008	0.0	0.00008
RH105	1.50E+00	0.0	0.00001	0.00001	0.00000	0.00002	0.00007	0.0	0.00007
RU106	3.67E+02	2.72E-06	0.00000	0.00000	0.00000	0.00000	0.00001	0.00240	0.00240
AG110M	2.53E+02	9.07E-07	0.00000	0.00000	0.00000	0.00000	0.00000	0.00044	0.00044
TE129M	3.40E+01	3.63E-05	0.00000	0.00000	0.00001	0.00002	0.00007	0.0	0.00007
TE129	4.79E-02	0.0	0.00000	0.00000	0.00001	0.00001	0.00004	0.0	0.00004
TE131M	1.25E+00	9.13E-05	0.00000	0.00000	0.00000	0.00001	0.00003	0.0	0.00002
I131	8.05E+00	1.60E-03	0.00009	0.00137	0.02165	0.02311	0.08113	0.00006	0.08100
I132	9.58E-02	3.48E-02	0.00017	0.00002	0.00000	0.00019	0.00068	0.0	0.00068
I133	8.75E-01	2.26E-02	0.00088	0.00535	0.00012	0.00634	0.02226	0.0	0.02200
I134	3.67E-02	7.05E-02	0.00006	0.00000	0.00000	0.00006	0.00021	0.0	0.00021
CS134	7.49E+02	2.72E-05	0.00002	0.00000	0.00001	0.00003	0.00011	0.01300	0.01300
I135	2.79E-01	2.46E-02	0.00045	0.00074	0.00001	0.00120	0.00422	0.0	0.00420
CS136	1.30E+01	7.26E-05	0.00004	0.00001	0.00001	0.00005	0.00019	0.0	0.00019

SAMPLE BWR CASE 1 LIQUID EFFLUENTS (CONTINUED)

NUCLIDE	HALF-LIFE (DAYS)	CONCENTRATION IN PRIMARY COOLANT (MICRO CI/ML)	ANNUAL RELEASES TO DISCHARGE CANAL: HIGH PURITY (CURIES)	LOW PURITY (CURIES)	CHEMICAL (CURIES)	TOTAL LWS (CURIES)	ADJUSTED TOTAL (CI/YR)	DETERGENT WASTES (CI/YR)	TOTAL (CI/YR)
CS137	1.10E+04	1.81E-05	0.00001	0.00000	0.00001	0.00002	0.00007	0.02400	0.02400
BA137M	1.77E-03	0.0	0.00001	0.00000	0.00001	0.00002	0.00007	0.0	0.00007
CS138	2.24E-02	9.88E-03	0.00002	0.00000	0.00000	0.00002	0.00008	0.0	0.00008
BA139	5.76E-02	9.62E-03	0.00002	0.00000	0.00000	0.00002	0.00007	0.0	0.00007
BA140	1.28E+01	3.63E-04	0.00002	0.00003	0.00005	0.00011	0.00037	0.0	0.00037
LA140	1.67E+00	0.0	0.00000	0.00002	0.00006	0.00008	0.00029	0.0	0.00029
LA141	1.62E-01	0.0	0.00001	0.00000	0.00000	0.00001	0.00003	0.0	0.00003
CE141	3.24E+01	2.72E-05	0.00000	0.00000	0.00001	0.00002	0.00006	0.0	0.00006
LA142	6.39E-02	4.80E-03	0.00001	0.00000	0.00000	0.00001	0.00005	0.0	0.00005
PR143	1.37E+01	3.63E-05	0.00000	0.00000	0.00001	0.00001	0.00004	0.0	0.00004
CE144	2.84E+02	2.72E-06	0.00000	0.00000	0.00000	0.00000	0.00001	0.00520	0.00520
ALL OTHERS		1.98E-01	0.00001	0.00000	0.00001	0.00002	0.00006	0.0	0.00006
TOTAL (EXCEPT TRITIUM)		5.29E-01	0.00513	0.00953	0.02517	0.03983	0.13983	0.06234	0.20000

TRITIUM RELEASE 26 CURIES PER YEAR

SAMPLE BWR CASE 1

	BWR
THERMAL POWER LEVEL (MEGAWATTS)	3400.00000
PLANT CAPACITY FACTOR	0.80
TOTAL STEAM FLOW (MILLION LBS/HR)	15.00000
MASS OF WATER IN REACTOR VESSEL (MILLION LBS)	0.38000
CLEAN-UP DEMINERALIZER FLOW (MILLION LBS/HR)	0.13000
CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS)	56.00000
FRACTION FEED WATER THROUGH CONDENSATE DEMIN	1.00000
REACTOR VESSEL HALOGEN CARRYOVER FACTOR	0.01500

LIQUID WASTE INPUTS

STREAM	FLOW RATE (GAL/DAY)	FRACTION OF PCA	FRACTION DISCHARGED	COLLECTION TIME (DAYS)	DECAY TIME (DAYS)	DECONTAMINATION FACTORS I	CS	OTHERS
HIGH PURITY WASTE	2.86E+04	0.150	0.010	1.000	0.070	1.00E+03	1.00E+02	1.00E+03
LOW PURITY WASTE	5.70E+03	0.130	1.000	3.100	0.600	1.00E+03	1.00E+04	1.00E+04
CHEMICAL WASTE	6.00E+02	0.020	1.000	3.100	0.600	1.00E+03	1.00E+04	1.00E+04
REGENERANT SOLS	1.70E+03		1.000	9.400	0.440	1.00E+04	1.00E+05	1.00E+05

GASEOUS WASTE INPUTS

GLAND SEAL STEAM FLOW (THOUSAND LBS/HR)	0.0
GLAND SEAL HOLDUP TIME (HOURS)	0.0
AIR EJECTOR OFFGAS HOLDUP TIME (HOURS)	0.16700
CONTAINMENT BLDGIODINE RELEASE FRACTION	0.10000
PARTICULATE RELEASE FRACTION	0.01000
TURBINE BLDG. IODINE RELEASE FRACTION	1.00000
PARTICULATE RELEASE FRACTION	1.00000
GLAND SEAL VENT, IODINE PF	1.00000
AIR EJECTOR OFFGAS IODINE PF	0.0
AUXILIARY BLDG. IODINE RELEASE FRACTION	1.00000
PARTICULATE RELEASE FRACTION	1.00000
RADWASTE BLDG. IODINE RELEASE FRACTION	1.00000
PARTICULATE RELEASE FRACTION	0.01000
THERE IS A CHARCOAL DELAY SYSTEM	
KRYPTON HOLDUP TIME (DAYS)	2.66823
XENON HOLDUP TIME (DAYS)	61.24234
KRYPTON DYNAMIC ADSORPTION COEFFICIENT (CM3/GM)	105.00000
XENON DYNAMIC ADSORPTION COEFFICIENT(CM3/GM)	2410.00000
MASS OF CHARCOAL(THOUSAND LBS)	48.00000

SAMPLE BWR CASE 1

NUCLIDE	COOLANT CONC. (MICROCURIES/G)	GASEOUS RELEASE RATE (CURIES PER YEAR) CONTAINMENT BLDG.	TURBINE BLDG.	AUXILIARY BLDG.	RADWASTE BLDG.	GLAND SEAL	AIR EJECTOR	MECH VAC PUMP	TOTAL
I-131	1.785E-03	1.1E-03	1.1E-01	2.1E-02	1.1E-02	0.0	0.0	8.2E-02	2.3E-01
I-133	2.499E-02	1.5E-02	1.6E+00	2.9E-01	1.5E-01	0.0	0.0	9.2E-01	3.0E+00

H-3 RELEASED FROM TURBINE BLDG. VENTILATION SYSTEM 2.6E+01

H-3 RELEASED FROM CONTAINMENT BLDG. VENTILATION SYSTEM 2.6E+01

TOTAL H-3 RELEASED VIA GASEOUS PATHWAY 5.2E+01

C-14 RELEASED VIA MAIN CONDENSER OFFGAS SYSTEM = 9.5 CI/YR

SAMPLE BWR CASE 1

GASEOUS RELEASE RATE (CURIES PER YEAR)

NUCLIDE	COOLANT CONC. (MICROCURIES/G)	CONTAINMENT BLDG.	TURBINE BLDG.	AUXILIARY BLDG.	RADWASTE BLDG.	GLAND SEAL	AIR EJECTOR	MECH VAC PUMP	TOTAL
AR-41	0.0	1.5E+01	0.0	0.0	0.0	0.0	2.3E+01	0.0	3.8E+01
KR-83M	9.100E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
KR-85M	1.600E-03	1.0E+00	2.5E+01	3.0E+00	0.0	0.0	3.0E+00	0.0	3.2E+01
KR-85	5.000E-06	0.0	0.0	0.0	0.0	0.0	2.4E+02	0.0	2.4E+02
KR-87	5.500E-03	0.0	6.1E+01	2.0E+00	0.0	0.0	0.0	0.0	6.3E+01
KR-88	5.500E-03	1.0E+00	9.1E+01	3.0E+00	0.0	0.0	0.0	0.0	9.5E+01
KR-89	3.400E-02	0.0	5.8E+02	2.0E+00	2.9E+01	0.0	0.0	0.0	6.1E+02
XE-131M	3.900E-06	0.0	0.0	0.0	0.0	0.0	5.0E+00	0.0	5.1E+00
XE-133M	7.500E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
XE-133	2.100E-03	2.7E+01	1.5E+02	8.3E+01	2.2E+02	0.0	3.2E+01	1.3E+03	1.8E+03
XE-135M	7.000E-03	1.5E+01	4.0E+02	4.5E+01	5.3E+02	0.0	0.0	0.0	9.9E+02
XE-135	6.000E-03	3.3E+01	3.3E+02	9.4E+01	2.8E+02	0.0	0.0	5.0E+02	1.2E+03
XE-137	3.900E-02	4.5E+01	1.0E+03	1.4E+02	8.3E+01	0.0	0.0	0.0	1.3E+03
XE-138	2.300E-02	2.0E+00	1.0E+03	6.0E+00	2.0E+00	0.0	0.0	0.0	1.0E+03
TOTAL NOBLE GASES									7.4E+03

0.0 APPEARING IN THE TABLE INDICATES RELEASE IS LESS THAN 1.0 CI/YR FOR NOBLE GAS

SAMPLE BWR CASE 1

AIRBORNE PARTICULATE RELEASE RATE

(CURIES PER YEAR)

NUCLIDE	CONTAINMENT BLDG.	TURBINE BLDG.	AUXILIARY BLDG.	RADWASTE BLDG.	MECH VAC. PUMP	TOTAL
CR-51	2.0E-06	9.0E-04	9.0E-04	7.0E-06	1.0E-06	1.8E-03
MN-54	4.0E-06	6.0E-04	1.0E-03	4.0E-05	0.0	1.6E-03
CO-58	1.0E-06	1.0E-03	2.0E-04	2.0E-06	0.0	1.2E-03
FE-59	9.0E-07	1.0E-04	3.0E-04	3.0E-06	0.0	4.0E-04
CO-60	1.0E-05	1.0E-03	4.0E-03	7.0E-05	5.6E-07	5.1E-03
ZN-65	1.0E-05	6.0E-03	4.0E-03	3.0E-06	3.4E-07	1.0E-02
SR-89	3.0E-07	6.0E-03	2.0E-05	0.0	0.0	6.0E-03
SR-90	3.0E-08	2.0E-05	7.0E-06	0.0	0.0	2.7E-05
NB-95	1.0E-05	6.0E-06	9.0E-03	4.0E-08	0.0	9.0E-03
ZR-95	3.0E-06	4.0E-05	7.0E-04	8.0E-06	0.0	7.5E-04
MO-99	6.0E-05	2.0E-03	6.0E-02	3.0E-08	0.0	6.2E-02
RU-103	2.0E-06	5.0E-05	4.0E-03	1.0E-08	0.0	4.1E-03
AG-110M	4.0E-09	0.0	2.0E-06	0.0	0.0	2.0E-06
SB-124	2.0E-07	1.0E-04	3.0E-05	7.0E-07	0.0	1.3E-04
CS-134	7.0E-06	2.0E-04	4.0E-03	2.4E-05	3.2E-06	4.2E-03
CS-136	1.0E-06	1.0E-04	4.0E-04	0.0	1.9E-06	5.0E-04
CS-137	1.0E-05	1.0E-03	5.0E-03	4.0E-05	8.9E-06	6.1E-03
BA-140	2.0E-05	1.0E-02	2.0E-02	4.0E-08	1.1E-05	3.0E-02
CE-141	2.0E-06	1.0E-02	7.0E-04	7.0E-08	0.0	1.1E-02

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C       GALE CODE FOR CALCULATING LIQUID  EFFLUENTS FROM LWRS.  MODIFIED
C      OCT. 1978 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
       LOGICAL DISCHG,POWRIT
       INTEGER*2 LOC,NONO,KD
       REAL*4 LETDWN,NOGEN
       REAL*4 LETDWA
       REAL KKR,KXE,KNO,KMASS
       INTEGER*2 NAME(3)
       COMMON/MATRIX/A(2500),LOC(2500),NONO(800),KD(800)
       COMMON/FLEX/FLUX(10),MMN,MOUT,INDEX,QXN,AXN,ERR,NOBLND,MZERO
       COMMON/PROCSS/  MPROS,PRATE(8),NOPROS(8),NZPROS(8,20),PR(800)
       COMMON/EQ/XZERO(800), XZH(800),XTEMP(800),XNEW(10,800),
      1     B(800),D(800)
       COMMON/FLUXN/T(20),POWER(10),TOCAP(800),FISS(100),DIS(800),ILITE,
      1 IACT,IFP,ITOT,NON,INPT
       COMMON/OUT/NUCL(800),TITLE(20),Q(800),FG(800),CUTOFF(7),
      1 POW,BURNUP,FLUXB,MSTAR,ALPHAN(100),SPONF(100),ABUND(500),
      2 BASIS(10),TCONST,TUNIT
       COMMON/CONC/PCONC(800),DWCONC(800),CWCONC(800),CMCONC(800),
      1     SCON(800),RINV(800)
       COMMON/COOL/REACTR,POW1,TYPE,PCVOL,LETDWN,NDGAS,SCVOL,STMVOL,NOGEN
      1            ,NPURGE,NVCSPG,SBLDR,GASDLA,STLKR,BLWDWN,EJCTR,WLKRAT,
      2             GASLKR,CONDLR,DFMBDM,DFCBDM,NZMBDM,NZCBDM,NE,PF,
      3             STMFR,FPEF,HEF,FFCDM,DFMBCS,CBFLR,DFI,DFCS,DF,
      4             EDFLR,DFIED,DFCSED,DFED,DWFLR,DFIDW,DFCSDW,DFDW,DILUT
      5            ,SBA,EDA,DWA,CWFLR,CMWFR,CWA,CMA,DFCM,DFICM,DFCSCM,
      6             DFCW,DFICW,DFCSCW,MHOLD,MHOLDE,MHOLDW,MHOLCM,MHOLDC
      7            ,BDTFR,DFIBD,DFCSBD,DFBD,MHOLDB,REGENT,
      8              SBFD,DFMO,DFY,EDFD,DFMOED,DFYED,DWFD,DFMODW,DFYDW,
      9              DFMOBD,DFYBD,CWFD,DFMOCW,DFYCW,CMFD,DFMOCM,DFYCM
      A    ,TS,TE,TD,TB,TC,TCM,TSTORC,TSTORD,TSTORB,TMSC,DWFL2,DW2,DWF2,
      B   T2,TSTOR2,DFID2,DFCSD2,DFMOD2,DFYD2,DFD2,PFLAUN
       COMMON/APCOOL/RGWFR,DFIRG,DFCSRG,DFRG,TRG,TSTORR,RGFD
       COMMON/BDTES/RFNRT
       COMMON /CONB/BCONC(800)
       DIMENSION WORD13(4), WORD15(4),WORD18(5),WORD23(6),WORD28(7),
      1           WORD33(9),WORD10(3),WORD8(2),WORD40(10)
       DIMENSION REACTR(7),NZMBDM(26),NZCBDM(26)
       DIMENSION FACT(10),XCOMP(20),INUCL(20)
       DIMENSION WORD56(14)
       DATA YES/'YES'/
C      FOLLOWINFISOTOPES(WITH THEIR CONCENTRATIONS IN UCI/GM) ARE
C      ALSO PRESENT IN THE PRIMARY COOLANT BUT ARE NOT CONSIDERED
C      SIGNIFICANT IN EFFLUENT CALCULATIONS:N-13,.05;N-16,60;N-17,.009;
C      O-19,.7;F-18,.004.
C      READ NUCLEAR DATA AND CONSTRUCT TRANSITION MATRIX
C
   10  CALL NUDATA(NLIBE)
C
       DO 20 I=2,ITOT
       NONO(I)=NONO(I)+NONO(I-1)
   20  KD(I)=KD(I)+NONO(I-1)
   30  DISCHG=.FALSE.
       POWRIT=.FALSE.
```

```
      KT=0
      INDEX=0
      FLXB=0.0
      PWRH=0.0
      BURN=0.0
      BDTFR=1.0
      QXN=0.001
      AXN=-ALOG(QXN)
      NE=ITOT
      TCONST=86400.
      MMN=0
      MZERO=21
      EDA=0.0
      TMSC=0.0
      TE=0.0
      TS=0.0
      T2=0.0
      TSTOR2=0.0
      DWFL2=0.0
      DW2=0.0
      DWF2=0.0
      DO 40 J=1,800
      PCONC(J)=0.0
      DWCONC(J)=0.0
      SCON(J)=0.0
      RINV(J)=0.0
      CWCONC(J)=0.0
      CMCONC(J)=0.0
      XZH(J)=0.0
   40 CONTINUE
C
C     READ DESCRIPTION OF REACTOR AND RADWASTE TREATMENT PLANT
C
      PRINT 9026
      READ 9010,REACTR,TYPE
      PRINT 9010,REACTR,TYPE
      READ 9022,WORD56,POW1
      PRINT 9022,WORD56,POW1
      PF=0.80
      PRINT 9027
C
C     READ DATA FOR BWR LIQUID CODE
C
   50 READ 9022,WORD56,STMFR
      PRINT 9022,WORD56,STMFR
      READ 9022,WORD56,PCVOL
      PRINT 9022,WORD56,PCVOL
      FPEF=0.001
      HEF=0.020
      PRINT 9030,FPEF,HEF
      READ 9022,WORD56,LETDWN
      PRINT 9022,WORD56,LETDWN
      PC = 0.015
      READ 9022,WORD56,REGENT
      IF(REGENT.EQ.0.0) PC = 0.004
      PRINT 9022,WORD56,REGENT
      READ 9022,WORD56,FFCDM
      PRINT 9022,WORD56,FFCDM
      PRINT 9045
      READ 9013,WORD18,CWFLR,WORD8,CWA
```

```
      READ 9014,DFICW,DFCSCW,DFCW
      READ 9015,TC,TSTORC,CWFD
      PRINT 9016
      PRINT 9017,WORD18,CWFLR,CWA,CWFD,TC,TSTORC,DFICW,DFCSCW,DFCW
      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,CMWFR,WORD8,CMA
      READ 9014,DFICM,DFCSCM,DFCM
      READ 9015,TCM,TSTORB,CMFD
      PRINT 9017,WORD18,CMWFR,CMA,CMFD,TCM,TSTORB,DFICM,DFCSCM,DFCM
      READ 9037,RGWFR
      READ 9014,DFIRG,DFCSRG,DFRG
      READ 9015,TRG,TSTORR,RGFD
      PRINT 9038,RGWFR,RGFD,TRG,TSTORR,DFIRG,DFCSRG,DFRG
C
C     READ DATA FOR BWR GAS CODE
C
      PRINT 9046
      READ 9022,WORD56,GGS
      PRINT 9022,WORD56,GGS
      READ 9022,WORD56,TIM3
      PRINT 9022,WORD56,TIM3
      READ 9022,WORD56,TIM4
      PRINT 9022,WORD56,TIM4
      HEPA1=1.0
      FIL1=1.0
      HEPA2=1.0
      FIL2=1.0
      HEPA5=1.0
      FIL5=1.0
      HEPA6=1.0
      FIL6=1.0
      FILGS = 1.0
      FILEJ = 1.0
      READ 9060,WORD15,CBCH,CBHEPA
      IF(CBCH.GT.0.0)FIL1 = (1.0 - CBCH/100.)
      IF(CBHEPA.GT.0.0)HEPA1 = (1.0 - CBHEPA/100.)
      PRINT 9061,WORD15,FIL1,HEPA1
      READ 9062,WORD15,TBCH,TBHEPA
      IF(TBCH.GT.0.0)FIL2 = (1.0 - TBCH/100.)
      IF(TBHEPA.GT.0.0)HEPA2 = (1.0 - TBHEPA/100.)
      PRINT 9063,WORD15,FIL2,HEPA2
      READ 9022,WORD56,FIL3
      IF(FIL3.GT.0.0) FILGS = (1.0 - FIL3/100.)
      PRINT 9022,WORD56,FILGS
      READ 9022,WORD56,FIL4
      IF(FIL4.EQ.1.0) FILEJ = (1.0 - FIL4)
      IF(FIL4.GT.1.0) FILEJ = (1.0 - FIL4/100.)
      IF(FIL4.EQ.0.0) FILEJ = 1.0
      PRINT 9022,WORD56,FILEJ
      READ 9060,WORD15,AXCH,AXHEPA
      IF(AXCH.GT.0.0)FIL5 = (1.0 - AXCH/100.)
      IF(AXHEPA.GT.0.0)HEPA5 = (1.0 - AXHEPA/100.)
      PRINT 9061,WORD15,FIL5,HEPA5
      READ 9060,WORD15,RWCH,RWHEPA
      IF(RWCH.GT.0.0)FIL6 = (1.0 - RWCH/100.)
      IF(RWHEPA.GT.0.0)HEPA6 = (1.0 - RWHEPA/100.)
      PRINT 9061,WORD15,FIL6,HEPA6
```

```
      READ 9021,KCHAR
      READ 9022,WORD56,KKR
      READ 9022,WORD56,KXE
      READ 9022,WORD56,KMASS
      READ 9020,WORD56,PFLAUN
      IF(KCHAR.EQ.0) GO TO 54
      IF(KCHAR.EQ.1) GO TO 55
      PRINT 9023
      GO TO 56
   54 PRINT 9024
      GO TO 56
   55 CHTI1 = 1.8 * (KMASS * KKR)/POW1
      CHTI2 = 1.8 * (KMASS * KXE)/POW1
      PRINT 9025,CHTI1,CHTI2,KKR,KXE,KMASS
   56 CONTINUE
      IF(PFLAUN.EQ.0.0) GO TO 99
      GO TO 98
   99 CONTINUE
      PRINT 9048
   98 CONTINUE
      PRINT 9026
   57 DO 58 I=1,ITOT
      B(I)=0.0
   58 CONTINUE
      DFIED=1.
      DFCSED=1.
      DFED=1.
      DFID2=1.
      DFCSD2=1.
      DFD2=1.
C
C     CALCULATE BWR PRIMARY COOLANT CONCENTRATIONS
      DO 2251 I=1,ITOT
 2251 PCONC(I)=BCONC(I)
      POWA=POW1
      PCVOA=PCVOL*1E6
      LETDWA=LETDWN*1E6
      STMFA=STMFR*1E6
      FFCDA=FFCDM
C     CHECK TO SEE IF PLANT PARAMETERS ARE WITHIN SPECIFIED RANGES
      IF(FFCDA.LT.0.99) FFCDA=0.18
      IF(ABS(POWA-3400).GT.400.1)GO TO 252
      IF(ABS(PCVOA-3.8E5).GT.0.4001E5)GO TO 252
      IF(ABS(LETDWA-1.3E5).GT.0.2001E5)GO TO 252
      IF(ABS(STMFA-1.5E7).GT.0.2001E7)GO TO 252
      IF(FFCDA.GT.0.99) GO TO 252
      GO TO 256
C     CALCULATE BWR ADJUSTMENT FACTORS
  252 RHAL2=(LETDWA*0.9+FFCDA*STMFA*PC*0.9)/PCVOA
      IF(FFCDA.GT.0.99) GO TO 299
      FFCDA = 0.01
  299 CONTINUE
      RCSRB2=(LETDWA*0.5+FFCDA*STMFA*5E-4)/PCVOA
      RCFP2=(LETDWA*0.9+FFCDA*STMFA*9E-4)/PCVOA
      RK2=111.76*POWA/PCVOA
      DO 255 J=1,ITOT
      IF(PCONC(J).EQ.0.0) GO TO 255
      NZ=NUCL(J)/10000
      DL=DIS(J)*3600
      IF (NZ.EQ.53.OR.NZ.EQ.35)GO TO 253
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```
      IF (NZ.EQ.37.OR.NZ.EQ.55)GO TO 254
      PCONC(J)=PCONC(J)*RK2*(0.3114+DL)/(RCFP2+DL)
      GO TO 255
  253 PCONC(J)=PCONC(J)*RK2*(0.3612+DL)/(RHAL2+DL)
      GO TO 255
  254 PCONC(J)=PCONC(J)*RK2*(0.1730+DL)/(RCSRB2+DL)
  255 CONTINUE
  256 PCVOL=PCVOL*1000000./62.4
      LETDWN=LETDWN*2000.
      STMFR=STMFR*2851.
      DO 2255 J=1,ITOT
      IF (PCONC(J).GT.0.0)PCONC(J)=PCONC(J)/(DIS(J)*1.6283E13)
 2255 CONTINUE
      IF(REGENT.GT.0.0) GO TO 257
      CALL EFFTAB
      GO TO 30
C
C     COMPUTE REMOVAL CONSTANT FOR CONDENSATE DEMINERALIZER IN BWR
  257 FFCDM = FFCDA
      IF(FFCDM.EQ.0.1) GO TO 300
      GO TO 301
  300 FFCDM = 0.01
  301 CONTINUE
      CCBDM=0.9*STMFR*FPEF/(PCVOL*7.48*60.)*FFCDM
      CSBDM=0.5*STMFR*FPEF/(PCVOL*7.48*60.)*FFCDM
      IF(FFCDM.EQ.0.01) GO TO 304
      CCBDI = CCBDM
      GO TO 305
  304 CONTINUE
      FFCDM = 0.1
      CCBDI = 0.9*STMFR*FPEF/(PCVOL*7.48*60.)*FFCDM
  305 CONTINUE
      DO 258 I=1,ITOT
      NZ=NUCL(I)/10000
      PR(I)=CCBDM
      IF(NZ.EQ.53.OR.NZ.EQ.35)PR(I)=CCBDI*HEF/FPEF
      IF(NZ.EQ.37.OR.NZ.EQ.55)PR(I)=CSBDM
      XZHJ=PCONC(I)*PR(I)*PCVOL*0.02832
      B(I)=XZHJ
      XZH(I)=XZHJ*86400.
  258 CONTINUE
      XZERO(I)=0.
C     CALCULATE INVENTORIES ON BWR CONDENSATE RESINS
 290  T(1)=REGENT
      CALL SOLVE
      DO 295 I=1,ITOT
  295 RINV(I)=XTEMP(I)
      CALL EFFTAB
      GO TO 30
C
C     FORMATS                       FORMATS                       FORMATS
C
 9000 FORMAT(20A4)
 9001 FORMAT(10E8.2)
 9002 FORMAT(8E10.3)
 9003 FORMAT(5(I6,E9.2),I5)
 9004 FORMAT(16I5)
 9005 FORMAT('0MMN OR MOUT EXCEEDS DIMENSIONS')
 9006 FORMAT('0MOUT SHOULD NOT EXCEED MMN BY MORE THAN 10'/)
 9007 FORMAT(8(E8.2,I2))
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      9008 FORMAT(20I4)
      9009 FORMAT(10A4,F7.0,A3,F10.3)
      9010 FORMAT(32X,8A4,12X,A4)
      9012 FORMAT(16X,14A4,F8.4)
      9013 FORMAT(15X,4A4,A2,8X,F8.0,1X,A4,A2,F5.3)
      9014 FORMAT(20X,F8.0,2(5X,F8.0))
      9015 FORMAT(27X,F6.2,14X,F6.2,18X,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(0PF8.3,2X),3(1PE9.2,1X))
      9020 FORMAT(16X,14A4,F8.4)
      9021 FORMAT(79X,I1)
      9022 FORMAT(16X,13A4,A2,F10.5)
      9023 FORMAT(16X,'THERE IS A CRYOGENIC DISTILLATION COLUMN'/20X,'IODINE
          1AND XENON DECONTAMINATION FACTOR',T71,'10000.'/20X,'KRYPTON DECONT
          2AMINATION FACTOR',T72,'4000.'/20X,'KRYPTON AND XENON HOLDUP TIME (
          3DAYS)',T74,'90.')
      9024 FORMAT(16X,'THERE IS NO CHARCOAL DELAY SYSTEM')
      9025 FORMAT(16X,'THERE IS A CHARCOAL DELAY SYSTEM'/20X,'KRYPTON HOLDUP
          1TIME (DAYS)',T72,F9.5/20X,'XENON HOLDUP TIME (DAYS)',T72,F9.5/
          220X,'KRYPTON DYNAMIC ADSORPTION COEFFICIENT (CM3/GM)',T72,F9.5/
          320X,'XENON DYNAMIC ADSORPTION COEFFICIENT(CM3/GM)',T71,F10.5/20X,
          4'MASS OF CHARCOAL(THOUSAND LBS)',T72,F9.5)
      9026 FORMAT(1H1)
      9027 FORMAT(16X,'PLANT CAPACITY FACTOR',T75,'0.80')
      9028 FORMAT(16X,'PERCENT FUEL WITH CLADDING DEFECTS',T75,F6.4)
      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)
      9032 FORMAT(2X,7A4,F7.0,3A4,A1,F5.3,3A4,A3,I2,2A4)
      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      0.0
          10.0      1.00E 00  1.00E 00  1.00E 00')
      9036 FORMAT(16X,'THERE IS NOT A CONDENSATE DEMINERALIZER')
      9037 FORMAT(72X,F8.2)
      9038 FORMAT(2X,'REGENERANT SOLS    ',1PE9.2,14X,0PF5.3,2X,2(F8.3,2X),
          13(1PE9.2,1X))
      9039 FORMAT(16X,'THERE IS A CRYOGENIC OFFGAS SYSTEM')
      9040 FORMAT(16X,'THERE IS NO CRYOGENIC OFFGAS SYSTEM')
      9041 FORMAT(16X,'PRIMARY TO SECONDARY LEAK RATE (LBS/DAY)',T73,'100.')
      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)
      9054 FORMAT(16X,'FLOW RATE THROUGH GAS STRIPPER (GPM)',20X,F8.4)
      9055 FORMAT(36X,F8.4,35X,I1)
      9056 FORMAT(15X,4A4,A2,8X,F8.0)
      9060 FORMAT(16X,4A4,10X,F4.0,6X,F4.0)
      9061 FORMAT (16X,4A4,'IODINE RELEASE FRACTION',16X,F10.5/32X,'PARTICULA
          1TE RELEASE FRACTION',10X,F10.5)
      9062 FORMAT(16X,4A4,10X,F4.0,6X,F4.0)
      9063 FORMAT(16X,4A4,'IODINE RELEASE FRACTIOM',16X,F10.5/32X,'PARTICULAT
          1E RELEASE FRACTION',10X,F10.5)
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 9064 FORMAT(16X,'STEAM LEAK TO TURBINE BLDG (LBS/HR)',19X,F10.5)
 9065 FORMAT(16X,4A4,6X,A3)
 9066 FORMAT(16X,4A4,4X,'PARTICULATE RELEASE FRACTION',6X,F10.5)
 9067 FORMAT(16X,5A4,10X,A3,6X,A3)
 9068 FORMAT(16X,5A4,'IODINE RELEASE FRACTION',11X,F10.5/36X,'PARTICULAT
     1E RELEASE FRACTION',6X,F10.5)
 9069 FORMAT(16X,5A4,9X,F8.2,9X,A3,6X,A3)
 9070 FORMAT(16X,5A4,'RATE(CFM)',27X,F8.2/16X,5A4,'IODINE RELEASE FRACTI
     1ON',11X,F10.5/36X,'PARTICULATE RELEASE FRACTION',6X,F10.5)
 9071 FORMAT(16X,5A4,10X,A3,6X,A3,19X,F3.0)
 9072 FORMAT (16X,'FREQUENCY OF CNTMT BLDG HIGH VOL PURGE (TIMES/YR)',
     1T74,F6.0)
 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
      SUBROUTINE EFFTAB
      INTEGER*2 NAME(3)
      INTEGER*2 LOC,NONO,KD
      REAL*4 LETDWN,NOGEN
      REAL*4 LETDWA
      DIMENSION NZMBDM(26),NZCBDM(26),REACTR(7)
      DIMENSION TURBDR(800),DWCON2(800),EDCONC(800)
      DIMENSION  LAUNDY(12),WLAUND(12)
      COMMON/EQ/XZERO(800), XZH(800),XTEMP(800),XNEW(10,800),
     1      B(800),D(800)
      COMMON/FLUXN/T(20),POWER(10),TOCAP(800),FISS(100),DIS(800),ILITE,
     1 IACT,IFP,ITOT,NON,INPT
      COMMON/OUT/NUCL(800),TITLE(20),Q(800),FG(800),CUTOFF(7),
     1 POW,BURNUP,FLUXB,MSTAR,ALPHAN(100),SPONF(100),ABUND(500),
     2 BASIS(10),TCONST,TUNIT
      COMMON/COOL/REACTR,POW1,TYPE,PCVOL,LETDWN,NDGAS,SCVOL,STMVOL,NOGEN
     1            ,NPURGE,HVCSPG,SBLDR,GASDLA,STLKR,BLWDWN,EJCTR,WLKRAT,
     2            GASLKR,CONDLR,DFMBDM,DFCBDM,NZMBDM,NZCBDM,NE,PF,
     3            STMFR,FPEF,HEF,FFCDM,DFMBCS,CBFLR,DFI,DFCS,DF,
     4            EDFLR,DFIED,DFCSED,DFED,DWFLR,DFIDW,DFCSDW,DFDW,DILUT
     5           ,SBA,EDA,DWA,CWFLR,CMWFR,CWA,CMA,DFCM,DFICM,DFCSCM,
     6            DFCW,DFICW,DFCSCW,MHOLD,MHOLDE,MHOLDW,MHOLCM,MHOLDC
     7           ,BDTFR,DFIBD,DFCSBD,DFBD,MHOLDB,REGENT,
     8             SBFD,DFMO,DFY,EDFD,DFMOED,DFYED,DWFD,DFMODW,DFYDW,
     9             DFMOBD,DFYBD,CWFD,DFMOCW,DFYCW,CMFD,DFMOCM,DFYCM
     A    ,TS,TE,TD,TB,TC,TCM,TSTORC,TSTORD,TSTORB,TMSC,DWFL2,DW2,DWF2,
     B  T2,TSTOR2,DFID2,DFCSD2,DFMOD2,DFYD2,DFD2,PFLAUN
      COMMON/APCOOL/RGWFR,DFIRG,DFCSRG,DFRG,TRG,TSTORR,RGFD
      COMMON/BDTES/RFNRT
      COMMON/FLEX/FLUX(10),MMN,MOUT,INDEX,QXN,AXN,ERR,NOBLND,MZERO
      COMMON/CONC/PCONC(800),DWCONC(800),CWCONC(800),CMCONC(800),
     1     SCON(800),RINV(800)
      COMMON/MPC/MPCTAB,AMPC(800),WMPC(800)
      EQUIVALENCE  (XZERO(1),TURBDR(1)),(XZH(1),DWCON2(1))
      DATA LAUNDY/250540,270580,270600,400950,410950,441030,441060,
     1             471101,531310,551340,551370,581440/
      DATA WLAUND/0.001,0.004,0.0087,0.0014,0.002,0.00014,0.0024,
     1            0.00044,0.000062,0.013,0.024,0.0052/
      DO 10 I=1,ITOT
   10 D(I)=-DIS(I)
   15 H3BWR = 0.03 * POW1
      TRITPR = H3BWR * 0.5
      TRITCO=.01
   20 DO 30 J=1,ITOT
      CWCONC(J)=0.0
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      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)=PCONC(J)*CMA
   30 CONTINUE
C
C     CALCULATE RADIOACTIVITY AFTER COLLECTION AT A CONSTANT RATE
C
      CALL COLECT(TC*86400.,CWCONC,ILITE,ITOT)
      CALL COLECT(TE*86400.,EDCONC,ILITE,ITOT)
      CALL COLECT(TD*86400.,DWCONC,ILITE,ITOT)
      CALL COLECT(T2*86400.,DWCON2,ILITE,ITOT)
      CALL COLECT(TCM*86400.,CMCONC,ILITE,ITOT)
   40 IF(REGENT.LE.0.0) GO TO 50
      CALL STORAG(TRG*86400.,RINV,ILITE,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     TO TREAT PWR TURBINE BUILDING FLOOR DRAINS THROUGH DIRTY WASTE
C     SYSTEM, DELETE C FOR COMMENT ON CARDS BELOW, UNTIL NEXT MESSAGE
      RINV(I)=RINV(I)/DFRG
      TURBDR(I)=1991.*5.*SCON(I)*FPEF
C     TURBDR(I)=1991.*5.*SCON(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.*SCON(I)*HEF
C     TURBDR(I)=1991.*5.*SCON(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
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      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.*SCON(I)*FPEF
C     TURBDR(I)=1991.*5.*SCON(I)*FPEF/DFCSDW
  100 CONTINUE
C
C     COMPUTE RADIOACTIVE DECAY DURING PROCESSING AND SAMPLING
C
      CALL STORAG(TSTORC*86400.,CWCONC,ILITE,ITOT)
      CALL STORAG(TS  *86400.,EDCONC,ILITE,ITOT)
      CALL STORAG(TSTORD*86400.,DWCONC,ILITE,ITOT)
      CALL STORAG(TSTOR2*86400.,DWCON2,ILITE,ITOT)
      CALL STORAG(TSTORB*86400.,CMCONC,ILITE,ITOT)
      CALL STORAG(TSTORR*86400.,RINV,ILITE,ITOT)
      CALL STORAG(21600.,TURBDR,ILITE,ITOT)
      DO   130 I=1,ITOT
      NZ=NUCL(I)/10000
      ABLOW=0.0
      IF(REGENT.LT.0.001) GO TO 110
      ABLOW=RINV(I)*365.5*RGFD/REGENT
  110 CONTINUE
  120 ABLOW=ABLOW+CMWFR*1.382*CMCONC(I)
  130 CMCONC(I)=ABLOW
  133 CWFR=CWFLR*1.382*CWFD
      DWFR=DWFLR*DWFD*1.382
      TLR=(CWFR+DWFR)+1.382*(CMFD*CMWFR+RGFD*RGWFR)
      TRITRL=TRITCO*TLR
      IF(TRITRL.GT.0.5*TRITPR)TRITRL=0.5*TRITPR
      RTRITR=TRITRL+0.5
      ITRITR=RTRITR
  136 TOTAL=0.0
      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
      CWCONC(I)=DISI*(CWCONC(I)*CWFR+EDCONC(I)*EDFLR)
      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=0.1
      AOR=(AOI+TOTAL)/TOTAL
  150 SCNORM=0.0
      SAPRIM=0.0
      SSEC=0.0
      SCWAST=0.0
      SDWAST=0.0
      SABLOW=0.0
      STB=0.0
      STOTAL =0.0
      SPER=0.0
      PAPRIM=0.0
      PSEC=0.0
      PCWAST=0.0
      PDWAST=0.0
      PABLOW=0.0
      PTB=0.0
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      PTOTAL =0.0
      PPER=0.0
      PNORM=0.0
      TLAUND=0.0
      CTOTAL=0.0
      PRINT 9001,    REACTR
      PRINT 9006
      PRINT 9010
      KOUNTR=1
      I1=ILITE+IACT+1
      L=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=0.0
      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
      TOTALG=TOTALN
      IF(NUCLI.NE.LAUNDY(L))  GO TO 155
      XLAUND=WLAUND(L)*PFLAUN
      TOTALG=TOTALN+WLAUND(L)*PFLAUN
      L=L+1
  155 CONTINUE
      IF (TOTALG.LT.1E-5) GO TO 156
      ISUB=2
      IF (TOTALG.GT.1.)ISUB=1
      DIV=10.**(INT(ALOG10(TOTALG))-ISUB)
      TOTALG=AINT(TOTALG/DIV+0.5)*DIV
  156 CONTINUE
      IF(NUCL(I).EQ.10030)  TOTALN=TOTAL
      IF(NZ.EQ.1) GO TO 160
      SAPRIM=SAPRIM+APRIM
      SSEC=SSEC+ASEC
      SABLOW=SABLOW+ABLOW
      SCWAST=SCWAST+CWASTE
      SDWAST=SDWAST+DWASTE
      STB=STB+TB
      STOTAL=STOTAL+TOTAL
      SCNORM=SCNORM+TOTALN
      TLAUND=TLAUND+XLAUND
      CTOTAL=CTOTAL+TOTALG
  160 IF(TOTALG.LT.1.E-5)  GO TO180
      IF(MOD(KOUNTR,50).NE.0) GO TO 170
      PRINT 9000,    REACTR
      PRINT 9006
  170 CALL NOAH(NUCL(I),NAME)
      THALF=8.0225E-6/DIS(I)
      PRINT 9007,    NAME,THALF,APRIM,CWASTE,DWASTE,
     1         ABLOW,TOTAL,TOTALN,XLAUND,TOTALG
      KOUNTR=KOUNTR+1
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      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.1E-5) GO TO 184
        DIV=10.**(INT(ALOG10(PNORM))-2)
      PNORMT=AINT(PNORM/DIV+0.5)*DIV
  184 PNORMT=PNORM
      PRINT 9008,    PAPRIM,PCWAST,PDWAST,PABLOW,PTOTAL,PNORM,PNORMT
      PRINT 9009,     SAPRIM,SCWAST,SDWAST,SABLOW,STOTAL,SCNORM,TLAUND,
     1                CTOTAL
  195 PRINT 9012,    ITRITR
 9000 FORMAT(1H1,20X,7A4,' LIQUID EFFLUENTS (CONTINUED)')
 9001 FORMAT(1H1,20X,7A4,' LIQUID EFFLUENTS')
 9003 FORMAT(1X,A2,I3,A1,2X,1PE9.2,2(2X,E9.2,2X),5(1X,0PF9.5,1X),2(1X,0P
     1  F9.5,1X),0PF10.5)
 9006 FORMAT(1H0,17X,'CONCENTRATION      ANNUAL RELEASES TO DISCHARGE CAN
     1AL'/18X,' IN PRIMARY  ',43('-'),'  ADJUSTED    DETERGENT    TOTAL
     2 '/' NUCLIDE HALF-LIFE  COOLANT    HIGH PURITY LOW PURITY  CHEMICA
     3L  TOTAL LWS    TOTAL      WASTES          '/10X,'(DAYS)  (MICRO C
     4I/ML)',3('  (CURIES) '), ' (CURIES)    (CI/YR)    (CI/YR)    (CI/Y
     5R)')
 9007 FORMAT(1X,A2,I3,A1,2X,1PE9.2,  2X,E9.2,2X ,4(1X,0PF9.5,1X),2(1X,0P
     1F9.5,1X),1X,0PF9.5)
 9008 FORMAT(1X,'ALL OTHERS',9X,1PE9.2,2X,4(1X,0PF9.5,1X),1X,0PF9.5,4X,
     1       '0.0',6X,0PF9.5)
 9009 FORMAT(' TOTAL'/,' (EXCEPT TRITIUM)   ',1PE9.2,2X,4(1X,0PF9.5,1X),
     1       2(1X,0PF9.5,1X),1X,0PF9.5)
 9010 FORMAT(' CORROSION AND ACTIVATION PRODUCTS')
 9011 FORMAT('0FISSION PRODUCTS')
 9012 FORMAT (1H0,1X,'TRITIUM RELEASE',12X,I3,'   CURIES PER YEAR')
      RETURN
      END
      BLOCK DATA
      COMMON /CONB/BCONC(800)
      DATA BCONC/2*0.0,0.01,33*0.0,
     1     1E-2,13*0.0,2E-4,53*0.0,6E-3,4*0.0,7E-5,0.0,5E-2,3*0.0,
     21E-3,3*0.0,3E-5,0.0,2E-4,2*0.0,4E-4,7*0.0,1E-6,0.0,3E-4,2*0.0,3E-2
     3,4*0.0,2E-4,3*0.0,2E-3,98*0.0,3E-4,64*0.0,7E-3,63*0.0,6E-3,4*0.0,7
     4E-3,2*0.0,3E-3,19*0.0,5E-3,1E-4,3*0.0,7E-6,5*0.0,4E-3,0.0,4E-5,3*0
     5.0,1E-2,6E-3,4*0.0,4E-3,11*0.0,8E-6,0.0,8E-6,6*0.0,6E-6,5*0.0,4E-3
     6,2*0.0,2E-3,2E-2,8*0.0,9E-2,7*0.0,2E-5,3*0.0,8E-2,6*0.0,2E-3,4*0.0
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     7,3E-6,21*0.,1E-6,104*0.0,4E-5,13*0.,1E-4,0.,3.7E-3,5*0.,1E-5,6E-
     82,4*0.0,5E-2,5*0.0,1E-1,2*0.0,3E-5,2*0.0,5E-2,8*0.0,8E-5,3*0.0,2E-
     95,4*0.0,1E-2,4*0.0,1E-2,3*0.0,4E-4,4*0.0,1E-2,0.0,3E-5,3*0.0,6E-3,
    115E-3,7*0.0,3E-5,4E-5,2*0.0,3E-6,10*0.0,3E-6,81*0.0/
      END
      SUBROUTINE SOLVE
      COMMON/EQ/XZERO(800), XZH(800),XTEMP(800),XNEW(10,800),
     1      B(800),D(800)
      COMMON/FLEX/FLUX(10),MMN,MOUT,INDEX,QXN,AXN,ERR,NOBLND,MZERO
      COMMON/PROCSS/  MPROS,PRATE(8),NOPROS(8),NZPROS(8,20),PR(800)
      COMMON/FLUXN/T(20),POWER(10),TOCAP(800),FISS(100),DIS(800),ILITE,
     1 IACT,IFP,ITOT,NON,INPT
      COMMON/OUT/NUCL(800),TITLE(20),Q(800),FG(800),CUTOFF(7),
     1 POW,BURNUP,FLUXB,MSTAR,ALPHAN(100),SPONF(100),ABUND(500),
     2 BASIS(10),TCONST,TUNIT
      DO 10 I=1,ITOT
      D(I)=-DIS(I)
   10 XTEMP(I)=0.0
      DELT=T(1)*TCONST
      CALL DECAY(1,DELT,ITOT)
      CALL TERM(DELT,1,ILITE,ITOT)
      CALL EQUIL(1,ITOT)
      DO 30 I=1,ITOT
   30 XTEMP(I)=XNEW(1,I)
      RETURN
      END
      SUBROUTINE TERM(T,M,ILITE,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
      LOGICAL*1 LONG
      INTEGER*2 LOC,NONO,KD
      INTEGER*2 LOCP(2500)
      INTEGER*2 NONP(800)
      INTEGER*2 NQ,NQU,NQUEUE
      REAL*8 BATE,BATM
      REAL*8 CIMN(800),CSUM(800),CIMNI
      DIMENSION AP(2500),CIMB(800),CIMO(800)
      DIMENSION QUB(50)
      COMMON/SERIES/ XP(800),XPAR(800),LONG(800)
      COMMON/FLEX/FLUX(10),MMN,MOUT,INDEX,QXN,AXN,ERR,NOBLND,MZERO
      COMMON/EQ/XZERO(800),XZH(800),XTEMP(800),XNEW(10,800),
     1      B(800),D(800)
      COMMON/MATRIX/A(2500),LOC(2500),NONO(800),KD(800)
      COMMON/DEBUGG/AP
      COMMON/TERMD/DD(100),DXP(100),QUEUE(50),NQU(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
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      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
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 9000 FORMAT('1TOO LONG A QUEUE HAS BEEN FORMED IN TERM',4I5,E12.5)
      GO TO 190
   50 BATM=0.D0
      IM=I-1
      DO 110 I=2,IM
      DL=DD(I)
      XPL=DXP(I)
      BATE=0.D0
      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.D0)  PRINT 9001,     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.D0)  BATE=0.D0
      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=CIMB(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(DABS(CIMNI).LT.ERR3)  CIMNI=0.D0
      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
      SUBROUTINE DECAY(M,T,ITOT)
C     DECAY TREATS SHORT-LIVED ISOTOPES AT BEGINNING OF CHAINS USING
C     BATEMAN EQUATIONS
      LOGICAL*1 LONG
      REAL*8 BATE
      INTEGER*2 LOC,NONO,KD
      INTEGER*2 NQ,NQU,NQUEUE
      COMMON/DEBUGG/AP(2500)
      COMMON/SERIES/ XP(800),XPAR(800),LONG(800)
      COMMON/FLEX/FLUX(10),MMN,MOUT,INDEX,QXN,AXN,ERR,NOBLND,MZERO
      COMMON/EQ/XZERO(800),XZH(800),XTEMP(800),XNEW(10,800),
     1      B(800),D(800)
      COMMON/MATRIX/A(2500),LOC(2500),NONO(800),KD(800)
      COMMON/TERMD/DD(100),DXP(100),QUEUE(50),NQU(50),NQUEUE(50),NQ(800)
      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.D0
      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.D0)  PRINT 9001,     L,I,BATE,XTEM,XTEMP(J1),AKDJQ
 9001 FORMAT(' L,I,BATE,XTEM,XTEMP(J1),AKDJQ = ',2I5,1P4E12.5)
      IF(BATE.LT.0.D0)  BATE=0.D0
      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
      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
      LOGICAL*1 LONG
      INTEGER*2 LOC,NONO,KD
      COMMON/EQ/XZERO(800),XZH(800),XTEMP(800),XNEW(10,800),
     1      B(800),D(800)
      COMMON/MATRIX/A(2500),LOC(2500),NONO(800),KD(800)
      COMMON/FLEX/FLUX(10),MMN,MOUT,INDEX,QXN,AXN,ERR,NOBLND,MZERO
      COMMON/SERIES/ XP(800),XPAR(800),LONG(800)
      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
      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
      INTEGER*2 LOC,NONO,KD
      INTEGER*2 ELE(99),STA(2)
      INTEGER*2 KAP(800),MMAX(800)
      INTEGER*2 NAME(3)
      DIMENSION COEFF(7,800),NPROD(7,800),CAPT(6),
     1 NUCAL(6),NSORS(5),           YIELD(5,500),TYLD(5)
      DIMENSION Y(5)
      DIMENSION SKIP(20)
      DIMENSION MSRS(20)
      COMMON/LABEL/ELE,STA
      COMMON/FLEX/FLUX(10),MMN,MOUT,INDEX,QXN,AXN,ERR,NOBLND,MZERO
      COMMON/EQ/XZERO(800),XZH(800),XTEMP(800),XNEW(10,800),
     1      B(800),D(800)
      COMMON/MPC/MPCTAB,AMPC(800),WMPC(800)
      COMMON/FLUXN/T(20),POWER(10),TOCAP(800),FISS(100),DIS(800),ILITE,
     1 IACT,IFP,ITOT,NON,INPT
      COMMON/OUT/NUCL(800),TITLE(20),Q(800),FG(800),CUTOFF(7),
     1 POW,BURNUP,FLUXB,MSTAR,ALPHAN(100),SPONF(100),ABUND(500),
     2 BASIS(10),TCONST,TUNIT
      COMMON/MATRIX/A(2500),LOC(2500),NONO(800),KD(800)
      EQUIVALENCE (XZERO(1),KAP(1)),(XZERO(401),MMAX(1)),
     1            ( XZH(1),COEFF(1,1)),(XNEW(1,401),NPROD(1,1))
      EQUIVALENCE (A1,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,END=260)NUCL(I),DLAM,IU,FB1,FP,FP1,FT,FA,FSF,
     1Q(I),FG(I),ABUND(I),WMPC(I),AMPC(I)
      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,END=450)NUCL(I),DLAM,IU,FB1,FP,FP1,FT,FA,FSF,
     1Q(I),FG(I),DUMMY,WMPC(I),AMPC(I)
      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))  NSORS(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),FG(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,END=690)NUCL(I),DLAM,IU,FB1,FP,FP1,FT,FA,FSF,
         1Q(I),FG(I),DUMMY,WMPC(I),AMPC(I)
          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 540
  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 9039, 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
      IO=ILITE+1
      I1=ILITE+IACT
      DO 810 I=IO,I1
      NUCLI=NUCL(I)
      DO 780 J=IO,I1
      MAX=KAP(J)
      IF(MAX.LT.1)  GO TO 780
      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)
      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 9041,     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)  GO TO 900
      IM=ILITE+IACT
      I0=IM+1
      IF(ITOT.LT.I0)  GO TO 900
      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 9041,     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 9041,     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
      IL=I-IM
      IF(YIELD(K,IL).LT.ERR)  GO TO 870
      NON=NON+1
      IF(NON.GT.2500)  PRINT 9041,      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)
 9002 FORMAT(I6,F5.3,I1,5F3.3,E5.2,F3.3,             13X,E5.2,F3.3,2E5.2
     1        ,F4.3,F3.3,F6.4)
 9003 FORMAT(I6,F5.3,I1,3X,4F3.3,2E5.2,F3.3,5E5.2,F4.3,F3.3)
 9004 FORMAT(I6,F5.3,I1,5F3.3,2E5.2,F3.3,4E5.2,F3.3,F4.3,F3.3,2E5.2)
 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.
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     1SIGNP = SIGTH * FNP.  FNA, FNP = FRACTION THERMAL N-ALPHA, N-PROTO 0008400
     2N.',/,' RITH, RING, RIF, RINA, RINP = RESONANCE INTEGRAL FOR ABSOR 0008405
     3PTION, N-GAMMA, FISSION, N-ALPHA, N-PROTON.',/,'   RING = RITH * ( 0008410
     41 - FINA - FINP). RINA = RITH * FINA. RINP = RITH * FINP.  FINA, F 0008415
     5INP = FRACTION RESONANCE N-ALPHA, N-PROTON.',/,' SIGMEV, SIGFF, SI 0008420
     6GN2N, SIGNAF, SIGNPF = FAST CROSS SECTIONS (BARNS) FOR ABSORPTION, 0008425
     7FISSION, N-2N, N-ALPHA, N-PROTON.',/,'   SIGN2N = SIGMEV * (1 - FF 0008430
     8NA - FFNP).  SIGNAF = SIGMEV * FFNA.  SIGNPF = SIGMEV * FFNP.  FFN 0008435
     9A, FFNP = FRACTION FAST N-ALPHA, N-P.')                            0008440
9007 FORMAT(' Y23, Y25, Y02, Y28, Y49 = FISSION YIELD (PERCENT) FROM 23 0008445
     13-U, 235-U, 232-TH, 238-U, 239- PU.',/,' Q = HEAT PER DISINTEGRATI 0008450
     2ON.  FG = FRACTION OF HEAT IN GAMMAS OF ENERGY GREATER THAN 0.2 ME 0008455
     3V.',/,'0 EFFECTIVE CROSS SECTIONS FOR A VOLUME AVERAGED THERMAL (L 0008460
     4T 0.876 EV) FLUX ARE AS FOLLOWS.',/,'    N-GAMMA   -  SIGNG * THERM 0008465
     5 + RING * RES.',/,'     FISSION   -   SIGF * THERM + RIF * RES + SIGF 0008470
     6F * FAST.',10X,'THERM = 1/V CORRECTION FOR THERMAL SPECTRUM AND TE 0008475
     7MPERATURE.',/,'    N-2N      -  SIGN2N * FAST.',36X,'RES   = RATIO  0008480
     8OF RESONANCE FLUX PER LETHARGY UNIT TO THERMAL FLUX.')             0008485
9008 FORMAT('    N-ALPHA   -   SIGNA * THERM + RINA * RES + SIGNAF * FAST 0008490
     1.',7X,'FAST = 1.45 *  RATIO OF FAST (GT 1.0 MEV) TO THERMAL FLUX ' 0008495
     2/'    N-PROTON -  SIGNP * THERM + RINP * RES + SIGNPF * FAST.')    0008500
9009 FORMAT(1H0,59X,'REFERENCES',/,'    HALF LIVES, DECAY SCHEMES, AND  0008505
     1THERMAL POWER',/,' C M LEDERER, J M HOLLANDER, AND I PERLMAN ''TAB 0008510
     2LE OF ISOTOPES - SIXTH EDITION'' JOHN WILEY AND SONS, INC (1967)', 0008515
     3/,' B S DZHELEPOV AND L K PEKER ''DECAY SCHEMES OF RADIOACTIVE NUC 0008520
     4LEI'' PERGAMMON PRESS (1961)',/,' D T GOLDMAN AND JAMES R ROSSER ' 0008525
     5'CHART OF THE NUCLIDES'' NINTH EDITION GENERAL ELECTRIC CO (JULY  0008530
     61966)',/,' E D ARNOLD ''PROGRAM SPECTRA'' APPENDIX A OF ORNL-3576  0008535
     7(APRIL 1964)')                                                     0008540
9010 FORMAT('    CROSS SECTIONS AND FLUX SPECTRA',/,' B E PRINCE ''NEUT 0008545
     1RON REACTION RATES IN THE MSRE SPECTRUM'' ORNL-4119, PP 79-83 (JUL 0008550
     2Y 1967)',/,' B E PRINCE ''NEUTRON ENERGY SPECTRA IN MSRE AND MSBR' 0008555
     3' ORNL-4191, PP 50-58 (DEC 1967)',/,' M D GOLDBERG ET AL ''NEUTRON 0008560
     4 CROSS SECTIONS'' BNL-325, SECOND ED, SUPP NO 2 (MAY 1964 - AUG 19 0008565
     566) ALSO EARLIER EDITIONS',/,' H T KERR, UNPUBLISHED ERC COMPILATI 0008570
     6ON (FEB 1968)',/,' M K DRAKE ''A COMPILATION OF RESONANCE INTEGRAL 0008575
     7S'' NUCLEONICS, VOL 24, NO 8, PP 108-111 (AUG 1966)',/,' BNWL STAF 0008580
     8F ''INVESTIGATION OF N-2N CROSS SECTIONS'' BNWC-98, PP 44-98 (JUNE 0008585
     9 1965)')                                                           0008590
9011 FORMAT(18A4,I3)                                                     0008595
9012 FORMAT(1H1,20X,18A4)                                                0008600
9013 FORMAT(' H ALTER AND C E WEBER ''PRODUCTION OF H AND HE IN METALS  0008605
     1DURING REACTOR IRRADIATION'' J NUCL MATLS, VOL 16, PP 68-73 (1965) 0008610
     2',/,' L L BENNETT ''RECOMMENDED FISSION PRODUCT CHAINS FOR USE IN  0008615
     3REACTOR EVALUATION STUDIES'' ORNL-TM-1658 (SEPT 1966)')            0008620
9014 FORMAT('    FISSION PRODUCT YIELDS',/,' M E MEEK AND B F RIDER, '' 0008625
     1SUMMARY OF FISSION PRODUCT YIELDS FOR U-235, U-238, PU-239, AND PU 0008630
     2-241 AT THERMAL, FISSION SPECTRUM AND'/'     14 MEV NEUTRON ENERGI 0008635
     3ES'' APED-5398-A(REV.),(OCT. 1968)'/' S KATCOFF '' FISSION PRODUCT 0008640
     4YIELDS FROM NEUTRON INDUCED FISSION'' NUCLEONICS, VOL 18, NO 11,   0008645
     5(NOV 1960)'/' N D DUDEY '' REVIEW OF LOW-MASS ATOM PRODUCTION IN F 0008650
     6AST REACTORS'' ANL-7434,(APRIL 1968) ')                            0008655
9015 FORMAT(1H0,20X,'LIGHT ELEMENTS, MATERIALS OF CONSTRUCTION, AND ACT 0008660
     1IVATION PRODUCTS  ',/,'0 NUCL    DLAM    FB1   FP    FP1   FT    F 0008665
     2A    SIGTH   FNG1   FNA   FNP   RITH    FINA  FINP    SIGMEV  FN2N1 0008670
     3  FFNA  FFNP    Q    FG')                                          0008675
9016 FORMAT(1H0,20X,'LIGHT ELEMENTS, MATERIALS OF CONSTRUCTION, AND ACT 0008680
     1IVATION PRODUCTS  ',/,'0 NUCL    DLAM    FB1   FP    FP1   FT    F 0008685
     2A    SIGNG   FNG1   SIGN2N   FN2N1   SIGNA   SIGNP    Q    FG ABU 0008690
     3NDANCE')                                                           0008695
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 9017 FORMAT(1H ,A2,I3,A1,1PE9.2,0P5F6.3,1PE9.2,0P3F6.3,1PE9.2,0P2F6.3,
     11PE9.2,0P4F6.3,0PF5.2)
 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     FB1    FP
     1 FP1    FT      SIGNG ',          ' FNG1     Y23        Y25       Y02        Y
     228       Y49        Q      FG')
 9020 FORMAT(1H0,36X,'FISSION PRODUCTS',/,'0 NUCL     DLAM     FB1    FP
     1 FP1    FT      SIGNG    FNG1     Y25      Y28      Y49      Q      FG')
 9021 FORMAT(1H ,A2,I3,A1,1PE9.2,0P4F6.3, 1PE9.2,0PF6.3,1P5E9.2,
     10P2F6.3)
 9022 FORMAT(1H ,A2,I3,A1,1PE9.2,0P4F6.3,1PE9.2,0PF6.3,1P3E9.2,0P2F6.3)
 9023 FORMAT(1H0,32X, 'ACTINIDES AND THEIR DAUGHTERS',//
     1'  NUCL     DLAM     FB1    FP      FP1    FT    FA   FSF E+6   SIGNG      R
     2ING   FNG1     SIGF      RIF      SIGFF     SIGN2N     SIGN3N    Q   FG')
 9024 FORMAT(1H0,32X, 'ACTINIDES AND THEIR DAUGHTERS',//
     1'  NUCL     DLAM     FB1    FP      FP1    FT    FA     FSF E+6   SIGNG    F
     2NG21     SIGF     SIGN2N     SIGN3N     Q     FG')
 9025 FORMAT(1H ,A2,I3,A1,1PE9.2,0P5F6.3,6PF6.1,1P2E9.2,0PF6.3,1P5E9.2,
     1         0PF6.3,F5.2)
 9026 FORMAT(1H ,A2,I3,A1,1PE9.2,0P5F6.3,6PF9.1,1PE9.2,0PF6.3,1P3E9.2,
     1         0PF7.3,F5.2)
 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(64H0SUM OF YIELDS OF ALL FISSION PRODUCTS
     1           ,1P5E9.2)
 9031 FORMAT(5I10)
 9032 FORMAT(I6,F5.3,I1,5F3.3,E5.2,3F3.3,E5.2,2F3.3,E5.2,3F3.3,F4.3,F3.3
     1,F6.4)
 9033 FORMAT(1H ,A2,I3,A1,1PE9.2,0P5F6.3,1PE9.2,0PF6.3,1PE9.2,0PF6.3,
     1         1P2E9.2,0P2F6.3,F7.3)
 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,F4.3,F9.2,I1)
 9038 FORMAT(7X,2F9.2,F5.3,5F9.2, 4X,I1)
 9039 FORMAT('0  WARNING, MOUT OF RANGE IN NUDATA, =' I5)
 9040 FORMAT(   7X,F9.2,3F8.6,F4.2,2F3.1,F9.2,3F5.3,5X,I1)
 9041 FORMAT('0 NON HAS EXCEEDED 2500, EQUAL TO '2I6)
      END
      SUBROUTINE COLECT(TMB,CWASTE,ILITE,ITOT)
      COMMON/EQ/XZERO(800), XZH(800),XTEMP(800),XNEW(10,800),
     1        B(800),D(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,1,0,ITOT)
      CALL EQUIL(1,ITOT)
      DO 20 I=1,ITOT
   20 CWASTE(I)=XNEW(1,I)/TMB
      RETURN
      END
      SUBROUTINE STORAG(TMB,CWASTE,ILITE,ITOT)
```

```
      COMMON/EQ/XZERO(800), XZH(800),XTEMP(800),XNEW(10,800),
     1       B(800),D(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,ILITE,ITOT)
      CALL EQUIL(1,ITOT)
      DO 20 I=1,ITOT
   20 CWASTE(I)=XNEW(1,I)
      RETURN
      END
C     PROGRAM BLOCK DATA
      BLOCK DATA
      INTEGER*2 ELE(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',
     4'TE',' 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
      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
      SUBROUTINE NOAH(NUCLI,NAME)
C     SUBROUTINE NOAH CONVERTS SIX DIGIT IDENTIFIER TO ALPHAMERIC SYMBOL
      INTEGER*2 NAME(3)
      INTEGER*2 ELE(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
```

```
C        GALE CODE FOR CALCULATING GASEOUS EFFLUENTS FROM LWRS.  MODIFIED
C       AUG. 1978 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
        COMMON MSIG,NSIG
        DIMENSION NAME(8)
        DIMENSION DECON(14),DECOH(14)
        DIMENSION EX3(14),EX4(14),EXC(14),EJT(14)
        DIMENSION PCBB(19),PAXBB(19),PTBB(19),PRWBB(19)
        DIMENSION CBB(14),AXBB(14),TBB(14),RADBB(14)
        DIMENSION PCBL(19),PTBL(19),PAXBL(19),PRWBL(19),PTOTB(19)
        DIMENSION WURD(4),WORD(14)
        DIMENSION PVPL(19),VPR(14),PVPR(19)
        DIMENSION XB(14),X1(14)
        DIMENSION CBL(14),AXBL(14),TBL(14),RADBL(14),SGL(14)
        DIMENSION TOT(14)
        DIMENSION EJTI(2),EXCI(2)
        DIMENSION TOTI(2),VPRI(2)
        DIMENSION RNR(2),RNRS(2),RBWRR(2),RBWRRS(2),RADBLI(2)
        DIMENSION RNT(2),RNTS(2),RBWRT(2),RBWRTS(2),TBLI(2)
        DIMENSION DECONI(2),DECOHI(2)
        DIMENSION EX3I(2),EX4I(2),CBWR1(2)
        DIMENSION RBWR(2),RN(2),CBWR(2),RBWRS(2),RNS(2)
        DIMENSION AUXMK1(2),AUXBLN(2),CBLN(2),AUXM1S(2),CBLS(2),AUXBLS(2)
        DIMENSION CBLI(2),AUXLI(2),SGLI(2)
        DIMENSION WARD(5)
        DIMENSION RNMVP(2),RNMVPS(2)
        DIMENSION RMVP(2),RMVPS(2)
        REAL KKR,KXE,KNO,KMASS,KAR
        REAL*8 NUCLID(14)/'  AR-41 ',' KR-83M ',' KR-85M ','  KR-85 ',' K
       1R-87 ','  KR-88 ','  KR-89 ',' XE-131M',' XE-133M',' XE-133 ',' XE
       2-135M',' XE-135 ',' XE-137 ',' XE-138 '/
        REAL*8 BPART(19)/'  CR-51 ','  MN-54 ','  CO-58 ','  FE-59 ','  CO
       1-60 ','  ZN-65 ','  SR-89 ','  SR-90 ','  NB-95 ','  ZR-95 ','  MO
       2-99 ',' RU-103 ',' AG-110M',' SB-124 ',' CS-134 ',' CS-136 ',' CS-
       3137 ',' BA-140 ',' CE-141 '/
        REAL*8 HAL(2)/'  I-131 ','  I-133 '/
        DATA PCBB/.2,.4,.1,.09,1.,1.,.03,.003,1.,.3,6.,.2,.0004,.02,.7,.1,
       11.,2.,.2/
        DATA PAXBB/.9,1.,.2,.3,4.,4.,.02,.007,9.,.7,60.,4.,.002,.03,4.,.4,
       15,20.,.7/
        DATA PTBB/.9,.6,1.,.1,1.,6.,6.,.02,.006,.04,2.,.05,0.,.1,.2,.1,1.,
       110.,10./
        DATA PRWBB/.7,4.,.2,.3,7.,.3,0.,0.,.004,.8,.003,.001,0.,.07,2.4,0.
       1,4.,.004,.007/
        DATA CBB/15.0,0.0,1.0,2*0.0,1.0,3*0.0,27.0,15.0,33.0,45.0,2.0/
        DATA AXBB/2*0.0,3.0,0.0,2.0,3.0,2.0,2*0.0,83.0,45.0,94.0,135.0,6.0
       1/
        DATA TBB/2*0.0,25.0,0.0,61.0,91.0,580.0,2*0.0,150.0,400.0,330.0,10
       200.0,1000.0/
        DATA RADBB/6*0.0,29.0,2*0.0,220.0,530.0,280.0,83.0,2.0/
        DATA PVPR/.001,3*0.,.00056,.00034,8*0.,.0032,.0019,.0089,.011,1*0.
       1/
        DATA XB/0.0E-0,9.1E-3,1.6E-3,5.0E-6,5.5E-3,5.5E-3,3.4E-2,3.9E-6,7.
       15E-5,2.1E-3,7.0E-3,6.0E-3,3.9E-2,2.3E-2/
        DATA DECON/1.052E-04,1.035E-04,4.375E-05,2.040E-09,1.520E-04,6.880
       1E-05,3.632E-03,6.800E-07,3.548E-06,1.520E-06,7.357E-04,2.090E-05,3
       2.024E-03,6.800E-04/
        DATA VPR/9*0.0,1300.0,0.0,500.0,2*0.0/
```

```
C
C  DECONI DEFINES THE DECAY CONSTANT FOR IODINES
C
      DATA DECONI/9.970E-07,9.170E-06/
C
C  CBWR DEFINES THE REACTOR WATER CONCENTRATIONS FOR IODINES
C
      DATA CBWR/3.7E-3,5.0E-2/
C
C  RN DEFINES THE NORMALIZED ANNUAL RELEASES DURING POWER OPERATIONS FOR
C  THE REACTOR BLDG.
C
C
      DATA RN/12.3,12.3/
C
C   RNS DEFINES THE NORMALIZED ANNUAL RELEASES DURING SHUTDOWNS FOR THE
C   REACTOR BLDG.
C
      DATA RNS/5.2,5.2/
C
C  RNT DEFINES THE NORMALIZED ANNUAL RELEASES DURING POWER OPERATION FOR
C   THE TURBINE BLDG.
C
      DATA RNT/3.8E3,3.8E3/
C
C   RNTS DEFINES THE NORMALIZED ANNUAL RELEASES DURING SHUTDOWNS FOR
C   THE TURBINE BLDG.
C
      DATA RNTS/4.1E2,4.1E2/
C
C  RNR DEFINES THE NORMALIZED RELEASES FROM THE RADWASTE BLDG. DURING
C   POWER OPERATIONS
C
      DATA RNR/4.6,4.6/
C
C    RNRS DEFINES THE NORMALIZED RELEASES FROM THE RADWASTE BLDG.
C    DURING SHUTDOWNS
C
      DATA RNRS/1.4,1.4/
C
C    RNMVP DEFINES NORMALIZED RELEASES FROM THE MECHANICAL VACUUM PUMP
      DATA RNMVP/4.9E2,4.9E2/
      DATA RNMVPS/1.1E3,1.1E3/
   49 FORMAT(16X,'REACTOR VESSEL HALOGEN CARRYOVER FACTOR',15X,F10.5)
   51 FORMAT(16X,'PLANT CAPACITY FACTOR',T74,'0.80')
   52 FORMAT(32X,8A4,12X,A4)
   53 FORMAT(16X,13A4,A2,F10.5)
   55 FORMAT(15X,4A4,A2,8X,F8.0,7X,F5.3)
   56 FORMAT(20X,F8.0,2(5X,F8.0))
   57 FORMAT(27X,F6.2,14X,F6.2,18X,F6.2)
   58 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')
   59 FORMAT(2X,4A4,A2,1PE9.2,1X,4(0PF8.3,2X),3(1PE9.2,1X))
   60 FORMAT(79X,I1)
   61 FORMAT(16X,'THERE IS A CRYOGENIC DISTILLATION COLUMN'/20X,'IODINE
     1AND XENON DECONTAMINATION FACTOR',T70,'10000.'/20X,'KRYPTON DECONT
     2AMINATION FACTOR',T71,'4000.'/20X,'KRYPTON AND  XENON HOLDUP TIME
     3(DAYS)',T73,'90.')
```

```
   62 FORMAT(16X,'THERE IS NO CHARCOAL DELAY SYSTEM')
   63 FORMAT(16X,'THERE IS A CHARCOAL DELAY SYSTEM'/20X,'KRYPTON HOLDUP
     1TIME (DAYS)',T72,F9.5/20X,'XENON HOLDUP TIME (DAYS)',T72,F9.5/
     220X,'KRYPTON DYNAMIC ADSORPTION COEFFICIENT (CM3/GM)',T72,F9.5/
     320X,'XENON DYNAMIC ADSORPTION COEFFICIENT(CM3/GM)',T71,F10.5/20X,
     4'MASS OF CHARCOAL(THOUSAND LBS)',T72,F9.5)
   68 FORMAT(70X,F10.5)
   77 FORMAT(16X,'FRACTION IODINE BYPASSING CONDENSATE DEMINERALIZER',
     17X,T72,F9.5)
   71 FORMAT(2X,'REGENERANT SOLS    ',1PE9.2,14X,0PF5.3,2X,2(F8.3,2X),
     13(1PE9.2,1X))
  903 FORMAT(16X,16A4)
  904 FORMAT (/,'  LIQUID WASTE INPUTS')
  905 FORMAT (/,'  GASEOUS WASTE INPUTS')
  906 FORMAT(1H1)
  907 FORMAT('0',15X,'THERE IS NOT AN ON-SITE LAUNDRY')
  214 FORMAT(1H0,131H----------------------------------------------------------
     1-------------------------------------------------------------------
     2-------------)
  232 FORMAT(1H0,5X,'TOTAL NOBLE GASES', 99X,1PE7.1)
  230 FORMAT(1H0,4X,A8,5X,1PE9.3,8X,1PE7.1,1X,6(4X,1PE7.1,1X),7X,1PE7.1)
  231 FORMAT(1H0,'0.0 APPEARING IN THE TABLE INDICATES RELEASE IS LESS
     1THAN 1.0 CI/YR FOR NOBLE GAS')
  245 FORMAT (1H0,49X,'AIRBORNE PARTICULATE RELEASE RATE'/1H0,63X,'(CURI
     1ES PER YEAR)'/1H0,32X,'CONTAINMENT',3X,'TURBINE',6X,'AUXILIARY',4X
     2,'RADWASTE   MECH VAC.'/1H,4X,'NUCLIDE',24X,'BLDG.',7X,'BLDG.',7X,
     3'BLDG.',8X,'BLDG.',6X,'PUMP',10X,'TOTAL')
  235 FORMAT(1H0,63X,'GASEOUS RELEASE RATE'/1H0,63X,'(CURIES PER YEAR)'/
     11H0,16X,'COOLANT CONC.',3X,'CONTAINMENT',3X,'TURBINE',4X,'AUXILIAR
     2Y',3X,'RADWASTE',5X,'GLAND',7X,'AIR',6X,'MECH VAC'/1H,4X,'NUCLIDE'
     3,3X,'(MICROCURIES/G)',6X,'BLDG.',7X,'BLDG.',7X,'BLDG.',6X,'BLDG.',
     48X,'SEAL',5X,'EJECTOR',7X,'PUMP',11X,'TOTAL')
  925 FORMAT(16X,4A4,10X,F4.0,6X,F4.0)
  927 FORMAT(16X,4A4,10X,F4.0,6X,F4.0)
  926 FORMAT (16X,4A4,'IODINE RELEASE FRACTION',16X,F10.5/32X,'PARTICULA
     1TE RELEASE FRACTION',10X,F10.5)
  928 FORMAT(16X,4A4,'IODINE RELEASE FRACTION',16X,F10.5/32X,'PARTICULAT
     1E RELEASE FRACTION',10X,F10.5)
  920 FORMAT (1H0,4X,A8,22X,5(1PE7.1,5X),3X,1PE7.1)
  960 FORMAT(1H0,30X,'H-3 RELEASED FROM TURBINE BLDG. VENTILATION SYSTEM
     1',5X,1PE7.1)
  962 FORMAT(1H0,30X,'H-3 RELEASED FROM CONTAINMENT BLDG. VENTILATION SY
     1STEM',5X,1PE7.1)
  963 FORMAT(1H0,30X,'C-14 RELEASED VIA MAIN CONDENSER OFFGAS SYSTEM =
     19.5 CI/YR')
  964 FORMAT(1H0,30X,'TOTAL H-3 RELEASED VIA GASEOUS PATHWAY',5X,1PE7.1)
    2 FORMAT(I2)
C
C     GENERIC PLANT INFORMATION
C
      READ(1,2) N
      ICOUNT = 0
   80 ICOUNT = ICOUNT + 1
      IF(ICOUNT.EQ.N) GO TO 1003
      READ(1,52)NAME,TYPE
      WRITE(6,906)
      WRITE(6,52)NAME,TYPE
      READ(1,53)WORD,POWTH
      WRITE(6,53)WORD,POWTH
      OPFRA=0.80
```

```
      WRITE(6,51)
      READ(1,53)WORD,GTO
      WRITE(6,53)WORD,GTO
      READ(1,53)WORD,WLIQ
      WRITE(6,53)WORD,WLIQ
      READ(1,53)WORD,GDE
      WRITE(6,53)WORD,GDE
      PC = 0.015
      READ(1,53)WORD,REGENT
      IF(REGENT.EQ.0.0) PC = 0.004
      WRITE(6,53)WORD,REGENT
      READ(1,53)WORD,FFCDM
      WRITE(6,53)WORD,FFCDM
      WRITE(6,49) PC
C
C     READ DATA FOR BWR LIQUID CODE
C
      WRITE(6,904)
      READ(1,55)WARD,CWFLR,CWA
      READ(1,56)DFICW,DFCSCW,DFCW
      READ(1,57)TC,TSTORC,CWFD
      WRITE(6,58)
      WRITE(6,59)WARD,CWFLR,CWA,CWFD,TC,TSTORC,DFICW,DFCSCW,DFCW
      READ(1,55)WARD,DWFLR,DWA
      READ(1,56)DFIDW,DFCSDW,DFDW
      READ(1,57)TD,TSTORD,DWFD
      WRITE(6,59)WARD,DWFLR,DWA,DWFD,TD,TSTORD,DFIDW,DFCSDW,DFDW
      READ(1,55)WARD,CMWFR,CMA
      READ(1,56)DFICM,DFCSCM,DFCM
      READ(1,57)TCM,TSTORB,CMFD
      WRITE(6,59)WARD,CMWFR,CMA,CMFD,TCM,TSTORB,DFICM,DFCSCM,DFCM
      READ(1,68)RGWFR
      READ(1,56)DFIRG,DFCSRG,DFRG
      READ(1,57)TRG,TSTORR,RGFD
      WRITE(6,71)RGWFR,RGFD,TRG,TSTORR,DFIRG,DFCSRG,DFRG
C
C     READ DATA FOR BWR GAS CODE
C
      WRITE(6,905)
      READ(1,53)WORD,GGS
      WRITE(6,53)WORD,GGS
      READ(1,53)WORD,TIM3
      WRITE(6,53)WORD,TIM3
      READ(1,53)WORD,TIM4
      WRITE(6,53)WORD,TIM4
      HEPA1=1.0
      FIL1=1.0
      HEPA2=1.0
      FIL2=1.0
      HEPA5=1.0
      FIL5=1.0
      HEPA6=1.0
      FIL6=1.0
      FILGS = 1.0
      FILEJ = 1.0
      READ(1,925)WURD,CBCH,CBHEPA
      IF(CBCH.GT.0.0)FIL1 = (1.0 - CBCH/100.)
      IF(CBHEPA.GT.0.0)HEPA1 = (1.0 - CBHEPA/100.)
      WRITE(6,926)WURD,FIL1,HEPA1
      READ(1,927)WURD,TBCH,TBHEPA
```

```
      IF(TBCH.GT.0.0)FIL2 = (1.0 - TBCH/100.)
      IF(TBHEPA.GT.0.0)HEPA2 = (1.0 - TBHEPA/100.)
      WRITE(6,928) WURD,FIL2,HEPA2
      READ(1,53) WORD,FIL3
      IF(FIL3.GT.0.0) FILGS = (1.0 - FIL3/100.)
      WRITE(6,53) WORD,FILGS
      READ(1,53) WORD,FIL4
      IF(FIL4.EQ.1.0) FILEJ = (1.0 - FIL4)
      IF(FIL4.GT.1.0) FILEJ = (1.0 - FIL4/100.)
      IF(FIL4.EQ.0.0) FILEJ = 1.0
      WRITE(6,53) WORD,FILEJ
      READ(1,925) WURD,AXCH,AXHEPA
      IF(AXCH.GT.0.0)FIL5 = (1.0 - AXCH/100.)
      IF(AXHEPA.GT.0.0)HEPA5 = (1.0 - AXHEPA/100.)
      WRITE(6,926) WURD,FIL5,HEPA5
      READ(1,925) WURD,RWCH,RWHEPA
      IF(RWCH.GT.0.0)FIL6 = (1.0 - RWCH/100.)
      IF(RWHEPA.GT.0.0)HEPA6 = (1.0 - RWHEPA/100.)
      WRITE(6,926) WURD,FIL6,HEPA6
      READ(1,60)KCHAR
      READ(1,53)WORD,KKR
      READ(1,53)WORD,KXE
      READ(1,53)WORD,KMASS
      IF(KXE.LT.1161.) XKAR = 6.4
      IF(KXE.GT.1161.) XKAR = 16.0
      IF(KCHAR.EQ.0) GO TO 90
      IF(KCHAR.EQ.1) GO TO 91
      WRITE(6,61)
      CHTI1=90.
      CHTI2=90.
      CHT3=90.
      GO TO 92
   90 WRITE(6,62)
      CHTI1=0.
      CHTI2=0.
      CHT3=0.
      GO TO 92
   91 CHTI1 = 1.8 * (KMASS * KKR)/POWTH
      CHTI2 = 1.8 * (KMASS * KXE)/POWTH
      CHT3 = 1.8 * (KMASS * XKAR)/POWTH
      WRITE(6,63)CHTI1,CHTI2,KKR,KXE,KMASS
   92 CONTINUE
      READ(1,53)  WORD,PFLAUN
      IF(PFLAUN.LE.0.0)  WRITE(6,907)
C
C     CONVERSION OF UNITS
C
      GTO=GTO*1000000.
      GDE=GDE*1000000.
      GGS=GGS*1000.
      WLIQ=WLIQ*1000000.
      WSTE=WSTE*1000000.
C
      IOT=1
      IF(FFCDM.LT.0.99) FFCDM = 0.18
      IF(ABS(POWTH-3400).GT.400.1) GO TO 210
      IF(ABS(WLIQ-3.8E5).GT.0.4001E5)GO TO 210
      IF(ABS(GDE-1.3E5).GT.0.2001E5)GO TO 210
      IF(ABS(GTO-1.5E7).GT.0.2001E7)GO TO 210
      IF(FFCDM.GT.0.99) GO TO 210
```

```
      GO TO 211
  210 RHAL2=(GDE*0.9+FFCDM*GTO*PC*0.9)/WLIQ
      IOT=2
  211 CONTINUE
C
C
C  CALCULATION OF IODINE RELEASES FROM BLDG. VENTILATION SYSTEMS
C
      DO 88 I = 1,2
      CBWR1(I) = CBWR(I)
      DECOHI(I) = DECONI(I) * 3600.
      EX3I(I) = DECOHI(I) * TIM3
      IF(EX3I(I).GT.75.) EX3I(I) = 75.
      EX4I(I) = DECOHI(I) * TIM4
      IF(EX4I(I).GT.75.) EX4I(I) = 75.
      IF(IOT.EQ.1) GO TO 2002
      CBWR(I) =CBWR1(I) * (111.76 * POWTH/WLIQ) * ((0.4038+ DECOHI(I))/(
     1RHAL2 + DECOHI(I)))
C
C  CALCULATION OF IODINE RELEASES FROM RX. BLDG. DURING NORMAL OPERATIONS
C
 2002 CONTINUE
      RBWR(I) = RN(I) * CBWR(I)
      AUXBLN(I) = RBWR(I) * 0.9
      CBLN(I) = RBWR(I) * 0.1
C
C  CALCULATION OF IODINE RELEASES FROM RX. BLDG. DURING SHUTDOWNS
C
      RBWRS(I) = RNS(I) * CBWR(I)
      AUXBLS(I) = RBWRS(I) * 0.1
      CBLS(I) = RBWRS(I) * 0.9
      CBLI(I) = (CBLN(I) + CBLS(I)) * FIL1
      AUXLI(I) = (AUXBLN(I) + AUXBLS(I)) * FIL5
C
C  CALCULATION OF IODINE RELEASES FROM TURBINE BLDG.DURING OPERATION
C
      RBWRT(I) = RNT(I) * CBWR(I) * PC
      RBWRTS(I) = RNTS(I) * CBWR(I) * PC
      TBLI(I) = (RBWRT(I) + RBWRTS(I)) * FIL2
C
      RBWRR(I) = RNR(I) * CBWR(I)
      RBWRRS(I) = RNRS(I) * CBWR(I)
      RADBLI(I) = (RBWRR(I) + RBWRRS(I)) * FIL6
C
      IF(GGS.EQ.0.0) GO TO 87
      SGLI(1) = 8.1E-1 * CBWR(1) * FILGS
      SGLI(2) = 2.2E-1 * CBWR(2)* FILGS
      GO TO 99
   87 SGLI(1) = 0.0
      SGLI(2) = 0.0
C
   99 EXCI(I) = 0.0
      IF(KCHAR.EQ.2) EXCI(I) = DECOHI(I) * CHTI1 * 24.
      IF(EXCI(I).GT.75.) EXCI(I) = 75.
      IF(KCHAR.EQ.2) DFCR = 0.00010
      EJTI(1) = 6.0 * EXP(-EX4I(I))*(DFCR+EXP(-EXCI(I)))*FILEJ
      EJTI(2) = EJTI(1) * CBWR(2)/CBWR(1)
      RMVP(I) = (RNMVP(I) * CBWR(I) * PC) *  4.0
      IF(I.EQ.2) GO TO 899
      RMVPS(I) = RNMVPS(I) * CBWR(I) * PC
```

```
  899 RMVPS(2) = RNMVP(I) * CBWR(I) * PC
      VPRI(I) = RMVP(I) + RMVPS(I)
      TEST = 0.0001
      IF(SGLI(I).LE.TEST) SGLI(I) = 0.0
      IF(EJTI(I).LE.TEST) EJTI(I) = 0.0
   88 CONTINUE
      MSIG = 1
      NSIG = 2
      CALL SIGF2(SGLI)
      CALL SIGF2(EJTI)
      DO 89 I = 1,2
      TOTI(I) = AUXLI(I)+TBLI(I)+SGLI(I)+EJTI(I)+CBLI(I)+RADBLI(I)+VPRI(
     1I)
   89 CONTINUE
      WRITE(6,906)
      WRITE(6,903) NAME
      WRITE(6,214)
      WRITE(6,235)
      WRITE(6,214)
      DO 79 I = 1,2
      WRITE(6,230) HAL(I),CBWR(I),CBLI(I),TBLI(I),AUXLI(I),RADBLI(I),SGL
     1I(I),EJTI(I),VPRI(I),TOTI(I)
      CBWR(I) = CBWR1(I)
   79 CONTINUE
      WRITE(6,214)
C
C CALCULATION OF TRITIUM RELEASES
C
      TRITRP = 0.03 * POWTH
      GASH3 = TRITRP * 0.5
      TBH3 = GASH3 * 0.5
      CBH3 = GASH3 - TBH3
      TH3 = TBH3 + CBH3
      DIV = 10.**(INT(ALOG10(TBH3))-1)
      TBH = INT(TBH3/DIV+0.5)*DIV
      DIV = 10.**(INT(ALOG10(CBH3))-1)
      CBH = INT(CBH3/DIV+0.5)*DIV
      DIV = 10.**(INT(ALOG10(TH3))-1)
      TH = INT(TH3/DIV+1.0)*DIV
      WRITE(6,960) TBH
      WRITE(6,962) CBH
      WRITE(6,964) TH
      WRITE(6,963)
C
C   CALCULATION OF NOBLE GAS RELEASES
C
      WRITE(6,906)
      WRITE(6,903) NAME
      WRITE(6,214)
      WRITE(6,235)
      WRITE(6,214)
      DO 5 I = 1,14
      DECOH(I)=DECON(I)*3600.
      EX3(I)=DECOH(I)*TIM3
      IF(EX3(I).GT.75.)EX3(I)=75.
      EX4(I)=DECOH(I)*TIM4
      IF(EX4(I).GT.75.)EX4(I)=75.
      X1(I)=XB(I)
      CBL(I)=CBB(I)
      AXBL(I)=AXBB(I)
```

```
      TBL(I)=TBB(I)
      RADBL(I)=RADBB(I)
      SGL(I)=GGS*X1(I)*EXP(-EX3(I))*3.977*OPFRA
      IF(I.GT.1) GO TO 98
      EXC(I) = DECOH(I) * CHT3 * 24.
      GO TO 101
   98 EXC(I)=DECOH(I)*CHTI1*24.
      IF(I.GT.7)EXC(I)=DECOH(I)*CHTI2*24.
  101 CONTINUE
      IF(EXC(I).GT.75.)EXC(I)=75.
      DFCR=0.0
      IF(KCHAR.EQ.2)  DFCR=0.00025
      IF(KCHAR.EQ.2.AND.I.GT.7)  DFCR=0.0001
      IF(I.GT.1) GO TO 105
      EJT(I) = 1E3 * EXP(-EX4(I)) * (DFCR + EXP(-EXC(I)))
      GO TO 2001
  105 EJT(I)=GTO*X1(I)*EXP(-EX4(I))*(DFCR+EXP(-EXC(I)))*3.977*OPFRA
      GO TO 2001
 2001 TEST=1.
      IF(SGL(I).LE.TEST)      SGL(I)=0.0
      IF(EJT(I).LE.TEST)      EJT(I)=0.0
    5 CONTINUE
      DO 2003 I=1,14
      TOT(I)=AXBL(I)+TBL(I)+SGL(I)+EJT(I)+CBL(I)+RADBL(I)+VPR(I)
 2003 CONTINUE
      MSIG = 2
      NSIG = 14
      CALL SIGF2(TBL)
      CALL SIGF2(SGL)
      CALL SIGF2(EJT)
      GASTOT=0.0
      DO 2004 I=1,14
      WRITE(6,230)NUCLID(I),XB(I),CBL(I),TBL(I),AXBL(I),RADBL(I),
     1     SGL(I),EJT(I),VPR(I),TOT(I)
      GASTOT=GASTOT+TOT(I)
 2004 CONTINUE
      DIV=10.**(INT(ALOG10(GASTOT))-1)
      GASTOT=AINT(GASTOT/DIV+0.5)*DIV
      WRITE(6,232)  GASTOT
      WRITE(6,214)
      WRITE(6,231)
      WRITE(6,906)
      WRITE(6,903)NAME
      WRITE(6,214)
      WRITE(6,245)
      WRITE(6,214)
      DO 9 I = 1,19
      PCBL(I) = PCBB(I)*1E-3*HEPA1
      PTBL(I) = PTBB(I)*1E-3*HEPA2
      PAXBL(I) = PAXBB(I)*1E-3*HEPA5
      PVPL(I) = PVPR(I) * 1E-3
      PRWBL(I) = PRWBB(I)*1E-3*HEPA6
      PTOTB(I) = PCBL(I) + PTBL(I) + PAXBL(I) + PRWBL(I) + PVPL(I)
    9 CONTINUE
      MSIG = 3
      NSIG = 20
      CALL SIGF2(PTOTB)
      DO 10 I =1,19
      WRITE (6,920) BPART(I),PCBL(I),PTBL(I),PAXBL(I),PRWBL(I),PVPL(I),
     1PTOTB(I)
```

```
   10 CONTINUE
      WRITE(6,214)
      GO TO 80
 1003 CONTINUE
      STOP
      END
      SUBROUTINE SIGF2(RLPT)
      COMMON MSIG,NSIG
      DIMENSION RLPT(NSIG)
      IF(MSIG.EQ.2) GO TO 25
      IF(MSIG.EQ.3) GO TO 30
      DO 20 I=1,NSIG
C  THIS PART OF SUBROUTINE IS FOR IODINE
      ISUB=2
      IF(RLPT(I).EQ.0.) GO TO 20
      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
      GO TO 50
   25 CONTINUE
      DO 35 I = 1,NSIG
C  THIS PART OF SUBROUTINE IS FOR NOBLE GASES
      IF(RLPT(I).EQ.0.) GO TO 35
      DIV=10.**(INT(ALOG10(RLPT(I)))-1)
      IF (RLPT(I).LT.10.) DIV=1.00
      RLPT(I)=AINT(RLPT(I)/DIV+0.5)*DIV
   35 CONTINUE
      GO TO 50
   30 CONTINUE
      DO 40 I = 1,NSIG
C  THIS PART OF SUBROUTINE IS FOR PARTICULATES
      IF (RLPT(I).EQ.0.) GO TO 40
      DIV=10.**(INT(ALOG10(RLPT(I)))-2)
      RLPT(I)=AINT(RLPT(I)/DIV+0.5)*DIV
   40 CONTINUE
   50 CONTINUE
      RETURN
      END
```

CHAPTER 4. DATA NEEDED FOR NRC RADIOACTIVE SOURCE TERM CALCULATIONS FOR BOILING WATER REACTORS (BWRs)

This chapter lists the information needed to generate source terms for BWRs. The information is provided by the applicant and should be consistent with the contents of the Safety Analysis Report (SAR) and the Environmental Report (ER) of the proposed boiling water reactor. This information is the basic data required to calculate the releases of radioactive material in liquid and gaseous effluents (the source terms). All data is on a per-reactor basis.

4.1 GENERAL

1. The maximum core thermal power (MWt) evaluated for safety considerations in the SAR. (Note: All of the following responses should be adjusted to this power level.)

2. The quantity of tritium released in liquid and gaseous effluents (Ci/yr per reactor).

4.2 NUCLEAR STEAM SUPPLY SYSTEM

1. Total steam flow rate (in lb/hr).

2. Mass of reactor coolant (in lb) in the reactor vessel at full power.

4.3 REACTOR COOLANT CLEANUP SYSTEM

1. Average flow rate (in lbs/hr).

2. Demineralizer type (deep bed or powdered resin) and size of resin capacity (in ft^3).

3. Regeneration or replacement frequency.

4. Regenerant volume (in gal/event) and activity (if applicable).

4.4 CONDENSATE DEMINERALIZERS

1. Average flow rate (in lbs/hr).

2. Demineralizer type (deep bed or powdered resin).

3. Number and size (in ft^3) of resin capacity of demineralizers.

4. Regeneration or replacement frequency.

5. Indicate whether ultrasonic resin cleaning is used and waste liquid volume associated with its use.

6. Regenerant volume (in gal/event) and activity.

4.5 LIQUID WASTE PROCESSING SYSTEMS

1. For each liquid waste processing system, provide in tabular form the following information:

   a. Sources, flow rates (in gal/day), and expected activities (fraction of primary coolant activity, i.e., PCA) for all inputs to each system.

   b. Holdup times associated with collection, processing, and discharge of all liquid streams.

   c. Capacities of all tanks (in gal) and processing equipment (in 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 waste demineralizer regeneration, the time between regenerations, regenerant volumes and activities, treatment of regenerants, and fractions 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 and process flow diagrams for the liquid radwaste systems, along with all other systems influencing the source term calculations.

4.6 MAIN CONDENSER AND TURBINE GLAND SEAL AIR REMOVAL SYSTEMS

1. The main condenser tubing material of construction, i.e., stainless steel or copper.

2. The holdup time (in hr) for offgases from the main condenser air ejector prior to processing by the offgas treatment system.

3. A description and the expected performance of the gaseous waste treatment systems for the offgases from the condenser air ejector and mechanical vacuum pump. The iodine source term from the condenser.

4. The mass of charcoal (in tons) in the charcoal delay system used to treat the offgases from the main condenser air ejector, the operating and dew point temperatures of the delay system, and the dynamic adsorption coefficients for Xe and Kr.

5. A description of the cryogenic distillation system, the fraction of gases partitioned during distillation, the holdup in the system, storage following distillation, and the expected system leakage rate.

6. The steam flow (in lbs/hr) to the turbine gland seal and the source of the steam (primary or auxiliary).

7. The design holdup time (in hr) for gas vented from the gland seal condenser, the iodine partition factor for the condenser, and the fraction of radioiodine released through the system vent. A description of the treatment system used to reduce radioiodine and particulate releases from the gland seal system.

8. Piping and instrumentation diagrams and process flow diagrams for the gaseous waste treatment system, along with all other systems influencing the source term calculations.

4.7 VENTILATION AND EXHAUST SYSTEMS

For each plant building housing the main condenser evacuation system, the turbine gland seal system exhaust, or any system that contains radioactive materials, 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, HEPA filters, and mechanical devices).

3. Release rates for radioiodines, noble gases, and radioactive particulates (in Ci/yr); and the bases.

4. Release point description including height above grade, height above and location relative to adjacent structures, expected average temperature difference between gaseous effluents and ambient air, flow rate, exit velocity, and size and shape of flow orifice, whether deflectors or diffusers are used.

5. For the containment building, indicate the expected purge and venting frequencies and duration and the continuous purge rate (if used).

REFERENCES

1. American National Standards Source Term Specification, ANSI N237-1976, American National Standards Institute.

2. 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, March 1978, Office of Standards Development, U.S. Nuclear Regulatory Commission.

3. U.S. Atomic Energy Commission, Docket RM-50-2, "Effluents from Light-Water-Cooled Nuclear Power Reactors; AEC Staff Exhibit No. 22; Results of Independent Measurements of Radioactivity in Process Systems and Effluents at Boiling Water Reactors," C. A. Pelletier, May 1973.

4. General Electric Co., Licensing Topical Report NEDO-10871, "Technical Derivation of BWR 1971 Design Bases Radioactive Material Source Terms," J. M. Skarpelos and R. S. Gilbert, San Jose, Calif., March 1973.

5. General Electric Co., Licensing Topical Report NEDO-21159, "Airborne Releases from BWRs for Environmental Impact Evaluation, T. R. Marrero, March 1976 and Amendments 1 and 2.

6. General Electric Company Licensing Topical Report NEDO-21660, "Experience with BWR Fuel through December 1976," R. B. Elkins, et. al., July 1977.

7. "Current Trends in Fuel Performance," General Electric adaptation of paper presented at American Nuclear Society's Topical Meeting on Water Reactor Fuel Performance, May 9-11, 1977, St. Charles, Illinois.

8. Memorandum from Paul Check, USNRC to J. Collins, USNRC, "Performance of Improved GE Fuel," March 15, 1978.

9. Memorandum from J. Smith, General Electric Co., to J. Collins, USNRC, "Summary of BWR Data 1974-1977 Noble Gases and Iodine-131," September 27, 1978.

10. Memorandum from J. Smith, General Electric Co., to J. Collins, USNRC, "Summary of BWR Data 1974-1977 Noble Gases and Iodine-131," October 11, 1978.

11. General Electric Co., NEDM-12426, "Carryover Measurements at Oyster Creek Nuclear Power Station, March 27-28, 1973," R. S. Gilbert and H. R. Helmholz, Table 1, p. 5, advance copy pending final General Electric Co. and customer clearance, May 1973.

12. Commonwealth Edison Co., Report No. 21, "Measurements of Radioactivity in Process Systems of the Dresden Nuclear Power Station and in the Environment, Jan. - Feb. 1971," J. Golden and R. Pavlick, Table 7, p. 34, April 1973.

13. General Electric Co., Figure 5 of Draft Report "Fission Product Transport Measurements at Brunswick-2," C. Lin and H. Kenitzer, (to be published).

14. Westinghouse Electric Corp., WCAP-8215, "Steam Side Iodine Transport Study at Point Beach Unit No. 1 of Wisconsin Electric Power Company," October, 1973.

15. Electric Power Research Institute Report EPRI NP-495, "Sources of Radioiodine at Boiling Water Reactors, February 1978.

16. NUREG/CR-0395 "Evaluation of Radioiodine Measurements at Pilgrim Power Plant," Science Applications, Inc., October 1978.

17. General Electric Co., NEDE-11215, "Noble Gas and Iodine Activity in the Oyster Creek Ventilation and Offgas Systems During Power Operation," H. R. Helmholz, R. S. Gilbert, and M. E. Meek, March 1973.

18. General Electric Co., "Millstone Nuclear Power Station, Ventilation Sampling Program, July 13 - August 5, 1972," H. R. Helmolz, Draft, April 1973.

19. Letter from R. S. Gilbert, General Electric Co., to V. Benaroya and J. Collins, USAEC, Re: "Information Disclosure of Progress on Ventilation Studies (Nine Mile Point Ventilation Measurements)," May 23, 1974.

20. General Electric Licensing Topical Report NEDM-12294, "Main Condenser Air Inleakage Measurements at Dresden," J. G. Holloway, March 20, 1972.

21. General Electric Licensing Topical Report NEDE-13103, "Chemical and Radiochemical Results from the Oyster Creek Startup Test Program," C. R. Hills, May 1970.

22. General Electric Licensing Topical Report NEDE-13327-1, "Chemical and Radiochemical Results from the Dresden Nuclear Power Station No. 2 Startup Test Program," L. F. Coleman and R. A. Head, September 1971.

23. "A Statistical Analysis of Condenser Air Inleakage from Operating Fossil Steam Plants," TVA Preliminary First Draft, April 1973.

24. "Review of Operating Experience of Advanced Offgas System: Progress Report 1, Quad Cities," USNRC Project No. A5156, J. H. Keller, December 20, 1977.

25. Nuclear Containment Systems, Inc., NCS-1101, "Dynamic Adsorption Coefficient and Its Application for Krypton - Xenon Delay Bed Design," J. L. Kovach, Draft, November 1971.

26. L. R. Michaels and N. R. Horton, "Improved BWR Offgas Systems," 12th Air Cleaning Conference, San Jose, Calif., August 1972.

27. Oak Ridge National Lab., ORNL Central Files Number 59-6-47, "Removal of Fission Product Gases from Reactor Offgas Streams by Adsorption," W. E. Browning, R. E. Adams, and R. D. Ackley, June 11, 1959.

28. H. J. Schroeder, H. Queiser, and W. Reim, "Offgas Facility at the Gundremmigen Nuclear Power Plant," J. Nucl. Eng. Sci., No. 5, pp. 205-213, May 1971.

29. Letter from Kernkraftwerk Lengen GmbH to Peter Lang, North American Carbon, "Gas Delay System at KWL," Dec. 30, 1970.

30. P. S. Littlefield, S. R. Miller, H. Perhagopian, "Vermont Yankee Advanced Offgas System," Proceedings of the Thirteenth AEC Air Cleaning Conference (CONF-740807), San Francisco, CA, 12-15 August 1974.

31. General Electric Co., NEDO-20116, "Experimental and Operational Confirmation of Offgas System Design Parameters," C. W. Miller, October 1973.

32. Letter from J. L. Kovach, Nuclear Containment Systems, Inc., to V. Benaroya, USAEC, "Gas Delay System," Dec. 1, 1971.

33. D. P. Siegwarth, C. K. Neulander, R. T. Pao, and M. Siegler, "Measurement of Dynamic Adsorption Coefficients for Noble Gases on Activated Carbon," 12th Air Cleaning Conference, August 1972.

34. Letter from R. W. Scherer, Georgia Power Company, to R. S. Boyd, USAEC, "Edwin Hatch Source Term Data," Feb. 24, 1972.

35. R. D. Rivers, M. Pasha, and W. H. Buckler, "Correlation of Radioiodine Efficiencies Resulting from Standardized Test Program for Activated Carbons," 12th AEC Air Cleaning Conference, CONF-720823, Vol. 1, August 1972.

36. Hope Creek Final Safety Analysis Report, Amendment 11, p. 9.4-6a, November 1971.

37. Air Products and Chemicals, Inc., "Supplemental Information for the Offgas Treatment System, Newbold Island Nuclear Generating Station," proprietary report, Nov. 1971.

38. American National Standard Boiling Water Reactor Liquid Radioactive Waste Procesing System, ANSI N197-1976, American National Standards Institute.

39. Letter from J. S. Abel, Commonwealth Edison Company, to D. F. Ziemann, AEC, Re: "Condensate Demineralizer Regeneration Data," May 9, 1974.

40. J. H. Hollway and P. J. Hollifield, "Ultrasonic Cleaning of Deep Bed Resins for Condensate Demineralizer Systems," Proceedings of the American Power Conference, Vol. 33, 1971.

41. NUREG/CR-0143, "The Use of Ion Exchange to Treat Radioactive Liquids in Light-Water-Cooled Nuclear Power Plants," August 1978.

42. Letter from R. C. Haynes, Boston Edison Company, to Director, Directorate of Licensing, USAEC, Re: "Ultrasonic Resin Cleaner Performance Report," April 10, 1973.

43. NUREG/CR-0142, "The Use of Evaporation to Treat Radioactive Liquids in Light-Water-Cooled Nuclear Power Plants," September, 1978.

44. NUREG/CR-0141, "The Use of Filtration to Treat Radioactive Liquids in Light-Water-Cooled Nuclear Power Plants," September 1978.

45. Letter from D. Filkins, Rochester Gas & Electric to J. Markind Westinghouse Power Systems, "Isotopic Analysis of Laundry Reverse Osmosis Water," June 18, 1973.

46. Memo from L. Becker, Westinghouse, to S. Works, R. Trailer, and M. Gross, Wisconsin Electric Power Co., Re: "Results of Reverse Osmosis Unit Tested at Point Beach, Wisconsin," Aug. 2, 1971.

47. W. R. Greenway, W. J. Klein, J. Markind, and R. R. Stanna, "Treatment of Radioactive Steam Generator Blowdown," 33rd Annual Meeting of International Water Conference of the Engineers Society of Western Penn., Oct. 24-26, 1972.

48. C. Kunz et al., "C-14 Gaseous Effluents from Boiling Water Reactors," Transactions of the American Nuclear Society, 1975 Annual Meeting, June 8-13, 1975, New Orleans, La., pp. 91-92.

49. Letter from A. Levine, General Electric Co., to J. Collins, USNRC, October 10, 1978.

50. Harvard Air Cleaning Laboratory, School of Public Health, "Semiannual Progress Report, January 1-June 30, 1973," M. First, et. al., December 1973.

51. Argonne National Laboratory, ANL-8108, "Measurement of the Adsorption Coefficients of Argon on Activated Charcoal," A. Panek, C. McPheeters and H. Welborn, October, 1974.

APPENDIX A

LIQUID SOURCE TERM CALCULATIONAL PROCEDURE FOR REGENERANT WASTES FROM DEMINERALIZERS OTHER THAN CONDENSATE DEMINERALIZERS

Often in BWR radwaste systems, demineralizers other than the condensate demineralizers may undergo regeneration, for example, the radwaste demineralizer in the high purity waste system. The BWR-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 1-7 and Cards 17-33

A separate computer run for calculating the regeneration waste effluent from non-condensate demineralizers is required. Cards 1-7 should be filled out as indicated for the specific plant in Sections 1.5.2.1 through 1.5.2.7 of this report. Also Cards 17 through 33 may be left blank. (except that values of 1.0 must be entered for Card 18 entries).

2. Input to Cards 8-16

The only liquid source term data cards completed (Cards 8-16) should be the three card sets used in the input data for the stream in which the demineralizer to be regenerated is located.

a. Input Flow and Activity (Cards 8, 11, or 14)

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 BWR high purity stream, the total input flow rate and weighted activity would be 30,000 gallons per day at 0.15 PCA from Table 1-4.

Note that it is not the flow rate and activity which is due to the regenerant wastes which is entered, it is the normal flow rate and activity through the component to be regenerated which is entered.

b. Regeneration Frequency (Card 9, 12, or 15)

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
Reactor Coolant Cleanup System	180 days
Equipment Drain Wastes	25,000 gal/ft^3*
Floor Drain Wastes	2,000 gal/ft^3*

*Calculated values based on 12,000 gm $CaCo_3$ ion exchange capacity per ft^3 of resin and 5 umho/cm and 50 umho/cm average conductivity of 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 BWR-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 1.5.2.8.2 of this document.

d. Decontamination Factors (Card 10, 13 or 16)

The decontamination factors entered should consider radionuclide removal by the equipment used to process the regenerant wastes using the normal source term procedures of 1.5.2.8.2. 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 was not removed by the demineralizer.

e. Sample Case

A waste demineralizer is used to process equipment drain waste and is to be regenerated. The normal flow rate and activity for the demineralizer is 30,000 gpd at 0.15 PCA. The demineralizer resin volume is 180 ft^3. The regenerant wastes will be processed through an evaporator and discharged.

Fill in the Cards 8-10 in the following manner:

Card 8

Spaces 18-41 enter - waste demin regen
Spaces 42-49 enter - 30,000
Spaces 57-61 enter - 0.15

Card 9

The wastes will be processed through an evaporator which will provide the following DF's according to Table 1-5 of Section 1.5.2.8.2:

I - 10^3
Cs, Rb - 10^4
Others - 10^4

While in operation, referring to Table 1-5 of Section 1.5.2.8.2, demineralizer DF's were:

I - 10^2
Cs, Rb - 2
Others - 10^2

Therefore, for "I" and "Others," 99% of the activity processed through the demineralizer was removed by the resins and no adjustment is needed. Only 50% of the Cs and Rb in the waste stream was removed by the resins, however, so the DF entered for Cs should be adjusted. Thus, the DFs entered on Card 9 would be:

I - 10^3
Cs, Rb - 2×10^4
Others - 10^4

Card 10

Spaces 29-32 "Collection Time." Using the value from Table A-1 of 25,000 gal/ft^3, the regeneration frequency would be:

$$\frac{(180\ ft^3)(25{,}000\ gal/ft^3)}{(30{,}000\ gal/day)} = 150 \text{ days}$$

Enter 150 days in spaces 29-32.

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 1.5.2.8.2 of this report.

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 demineralizer 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 souce term run values. Do not use the adjusted total value from the right hand column as the source term run to which the regenerant waste run will be added has already been adjusted.

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U.S. NUCLEAR REGULATORY COMMISSION

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14. *(Leave blank)*

16. ABSTRACT *(200 words or less)*

In April of 1976, the NRC published NUREG-0016. That document presented methods for calculation of releases from BWRs based on operating data available at that time. In promulgating Appendix I to 10 CFR Part 50, the Commission indicated its desire to use the best available data for improving the calculational models. Therefore, at this time, we are updating NUREG-0016 by issuing Revision 1 which incorporates more recent operating data now available and also incorporates the results of a number of in-plant measurement programs at operating BWRs.

NUREG-0016, Revision 1, is similar to NUREG-0016 in that it provides instructions for using the BWR-GALE Code. It describes the parameters incorporated in the Code, the input data required, and a step-by-step procedure for completing the input data cards. It provides parameters for an assessment of reactor and radwaste system performance for normal operation including anticipated operational occurrences, and the bases for selecting the parameters. It also contains a Fortran IV listing of the BWR-GALE Code, a form for entering the input data, and a sample calculation.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
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19. SECURITY CLASS *(This report)*

20. SECURITY CLASS *(This page)*

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