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

F.P. Cardile, Editor
R.R. Bellamy, Editor

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

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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 back-seating 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.,
Effective Full Power Days = \( \frac{\text{Integrated Thermal Power}}{\text{Licensed Power Level}} = \frac{\sum P_i T_i}{P_{\text{total}}} \)

where

- \( P_i \) is the \( i \)th power level, in MWe;
- \( P_{\text{total}} \) is the licensed power level, in MWe; 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-GAEE 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.

1-4
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 \( \text{cm}^3/\text{gm} \) (see chart on page 2-35);
- \( M \) is the mass of charcoal, in \( 10^3 \text{ lbs} \);
- \( T \) is the holdup time, in hr, and
- \( P \) is the thermal power level (MWt) entered in Card 2.
<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>CONTAINMENT BUILDING</th>
<th>AUXILIARY BUILDING</th>
<th>TURBINE BUILDING</th>
<th>RADWASTE BUILDING</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kr-83m</td>
<td>**</td>
<td>**</td>
<td>**</td>
<td>**</td>
</tr>
<tr>
<td>Kr-85m</td>
<td>1</td>
<td>3</td>
<td>25</td>
<td>**</td>
</tr>
<tr>
<td>Kr-85</td>
<td>**</td>
<td>**</td>
<td>**</td>
<td>**</td>
</tr>
<tr>
<td>Kr-87</td>
<td>**</td>
<td>2</td>
<td>61</td>
<td>**</td>
</tr>
<tr>
<td>Kr-88</td>
<td>1</td>
<td>3</td>
<td>91</td>
<td>**</td>
</tr>
<tr>
<td>Kr-89</td>
<td>**</td>
<td>2</td>
<td>580</td>
<td>29</td>
</tr>
<tr>
<td>Xe-131m</td>
<td>**</td>
<td>**</td>
<td>**</td>
<td>**</td>
</tr>
<tr>
<td>Xe-133m</td>
<td>**</td>
<td>**</td>
<td>**</td>
<td>**</td>
</tr>
<tr>
<td>Xe-133</td>
<td>27</td>
<td>83</td>
<td>150</td>
<td>220</td>
</tr>
<tr>
<td>Xe-135m</td>
<td>15</td>
<td>45</td>
<td>400</td>
<td>530</td>
</tr>
<tr>
<td>Xe-135</td>
<td>33</td>
<td>94</td>
<td>330</td>
<td>280</td>
</tr>
<tr>
<td>Xe-137</td>
<td>45</td>
<td>135</td>
<td>1000</td>
<td>83</td>
</tr>
<tr>
<td>Xe-138</td>
<td>2</td>
<td>6</td>
<td>1000</td>
<td>2</td>
</tr>
<tr>
<td>Cr-51*</td>
<td>0.0002</td>
<td>0.0009</td>
<td>0.0009</td>
<td>0.0007</td>
</tr>
<tr>
<td>Mn-54</td>
<td>0.0004</td>
<td>0.001</td>
<td>0.0006</td>
<td>0.004</td>
</tr>
<tr>
<td>Fe-59</td>
<td>0.00009</td>
<td>0.0003</td>
<td>0.0001</td>
<td>0.0003</td>
</tr>
<tr>
<td>Co-58</td>
<td>0.0001</td>
<td>0.0002</td>
<td>0.001</td>
<td>0.0002</td>
</tr>
<tr>
<td>Co-60</td>
<td>0.001</td>
<td>0.004</td>
<td>0.001</td>
<td>0.007</td>
</tr>
<tr>
<td>Zn-65</td>
<td>0.001</td>
<td>0.004</td>
<td>0.006</td>
<td>0.0003</td>
</tr>
<tr>
<td>Sr-89</td>
<td>0.00003</td>
<td>0.00002</td>
<td>0.006</td>
<td>NA</td>
</tr>
<tr>
<td>Sr-90</td>
<td>0.000003</td>
<td>0.000007</td>
<td>0.00002</td>
<td>NA</td>
</tr>
<tr>
<td>Zr-95</td>
<td>0.0003</td>
<td>0.0007</td>
<td>0.00004</td>
<td>0.0008</td>
</tr>
<tr>
<td>Nb-95</td>
<td>0.001</td>
<td>0.009</td>
<td>0.000006</td>
<td>0.000004</td>
</tr>
<tr>
<td>Mo-99</td>
<td>0.006</td>
<td>0.06</td>
<td>0.002</td>
<td>0.000003</td>
</tr>
<tr>
<td>Ru-103</td>
<td>0.0002</td>
<td>0.004</td>
<td>0.00005</td>
<td>0.000001</td>
</tr>
<tr>
<td>Ag-110m</td>
<td>0.0000004</td>
<td>0.000002</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>Sb-124</td>
<td>0.000002</td>
<td>0.000003</td>
<td>0.0001</td>
<td>0.000007</td>
</tr>
<tr>
<td>Cs-134</td>
<td>0.00007</td>
<td>0.004</td>
<td>0.0002</td>
<td>0.0024</td>
</tr>
<tr>
<td>Cs-136</td>
<td>0.0001</td>
<td>0.0004</td>
<td>0.0001</td>
<td>NA</td>
</tr>
<tr>
<td>Cs-137</td>
<td>0.001</td>
<td>0.005</td>
<td>0.001</td>
<td>0.004</td>
</tr>
<tr>
<td>Ba-140</td>
<td>0.002</td>
<td>0.02</td>
<td>0.010</td>
<td>0.000004</td>
</tr>
<tr>
<td>Ce-141</td>
<td>0.00002</td>
<td>0.00007</td>
<td>0.010</td>
<td>0.000007</td>
</tr>
</tbody>
</table>

*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)

<table>
<thead>
<tr>
<th></th>
<th>Containment Bldg**</th>
<th>Auxiliary Bldg**</th>
<th>Turbine Bldg***</th>
<th>Radwaste Bldg**</th>
</tr>
</thead>
<tbody>
<tr>
<td>Power Operation</td>
<td>1.2</td>
<td>11.1</td>
<td>3.8 x 10^3</td>
<td>4.6</td>
</tr>
<tr>
<td>Refueling/Maintainence Outages</td>
<td>4.7</td>
<td>0.5</td>
<td>4.1 x 10^2</td>
<td>1.4</td>
</tr>
</tbody>
</table>

*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 radiiodine 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)

<table>
<thead>
<tr>
<th></th>
<th>Annual Normalized* Iodine Release Rate**</th>
</tr>
</thead>
<tbody>
<tr>
<td>Short-term outages</td>
<td>4.9 x 10^2</td>
</tr>
<tr>
<td>Refueling/Maintenance Outages</td>
<td>1.1 x 10^3</td>
</tr>
</tbody>
</table>

*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 radiiodine 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 radiiodine 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

<table>
<thead>
<tr>
<th>Demineralizer</th>
<th>Anions</th>
<th>Cs, Rb</th>
<th>Other Nuclides</th>
</tr>
</thead>
<tbody>
<tr>
<td>Deep bed</td>
<td>10</td>
<td>2</td>
<td>10</td>
</tr>
<tr>
<td>Powdex</td>
<td>10</td>
<td>2</td>
<td>10</td>
</tr>
</tbody>
</table>

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 cm³/gm and 16 cm³/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 forward 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:

<table>
<thead>
<tr>
<th>Stream</th>
<th>Flow Rate (gal/day)</th>
<th>Activity (PCA)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Stream A</td>
<td>1,000</td>
<td>0.01</td>
</tr>
<tr>
<td>Stream B</td>
<td>2,000</td>
<td>0.1</td>
</tr>
<tr>
<td>Stream C</td>
<td>500</td>
<td>1.0</td>
</tr>
</tbody>
</table>

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>
<thead>
<tr>
<th>SOURCE</th>
<th>DEEP BED PLANT</th>
<th>DEEP BED PLANT</th>
<th>DEEP BED PLANT</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>WITH ULTRASONIC RESIN CLEANER</td>
<td>WITHOUT ULTRASONIC RESIN CLEANER</td>
<td>OR A FILTER/DEMINERALIZER PLANT</td>
</tr>
<tr>
<td>Equipment Drains</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Drywell</td>
<td>3,400</td>
<td>3,400</td>
<td>1.00</td>
</tr>
<tr>
<td>Containment, auxiliary building, and fuel pool</td>
<td>3,700</td>
<td>3,700</td>
<td>0.1</td>
</tr>
<tr>
<td>Radwaste building</td>
<td>1,100</td>
<td>1,100</td>
<td>0.1</td>
</tr>
<tr>
<td>Turbine building</td>
<td>3,000</td>
<td>3,000</td>
<td>0.001</td>
</tr>
<tr>
<td>Ultrasonic resin cleaner</td>
<td>15,000</td>
<td>-</td>
<td>0.05</td>
</tr>
<tr>
<td>Resin rinse*</td>
<td>2,500</td>
<td>5,000</td>
<td>0.002</td>
</tr>
<tr>
<td>Floor Drains</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Drywell</td>
<td>700</td>
<td>700</td>
<td>0.001</td>
</tr>
<tr>
<td>Containment, auxiliary building, and fuel handling</td>
<td>2,000</td>
<td>2,000</td>
<td>0.001</td>
</tr>
<tr>
<td>Radwaste building</td>
<td>1,000</td>
<td>1,000</td>
<td>0.001</td>
</tr>
<tr>
<td>Turbine building</td>
<td>2,000</td>
<td>2,000</td>
<td>0.001</td>
</tr>
<tr>
<td>Other Sources</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cleanup phase separator decant</td>
<td>640</td>
<td>640</td>
<td>0.002</td>
</tr>
<tr>
<td>Laundry drains</td>
<td>1,000</td>
<td>1,000</td>
<td>-</td>
</tr>
<tr>
<td>Lab drains</td>
<td>500</td>
<td>500</td>
<td>0.02</td>
</tr>
<tr>
<td>Regenerants*</td>
<td>1,700</td>
<td>3,400</td>
<td>**</td>
</tr>
<tr>
<td>Condensate demineralizer backwash†</td>
<td>-</td>
<td>8,100</td>
<td>$2 \times 10^{-6}$</td>
</tr>
<tr>
<td>Chemical lab waste</td>
<td>100</td>
<td>100</td>
<td>0.02</td>
</tr>
</tbody>
</table>

*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:

<table>
<thead>
<tr>
<th></th>
<th>Filter</th>
<th>Evaporator</th>
<th>Demineralizer 1</th>
<th>Demineralizer 2</th>
<th>Product</th>
</tr>
</thead>
<tbody>
<tr>
<td>Anions</td>
<td>1</td>
<td>$10^3$</td>
<td>10</td>
<td>1</td>
<td>$10^4$</td>
</tr>
<tr>
<td>Cs, Rb</td>
<td>1</td>
<td>$10^4$</td>
<td>10</td>
<td>1</td>
<td>$10^5$</td>
</tr>
<tr>
<td>Other Nuclides</td>
<td>1</td>
<td>$10^4$</td>
<td>10</td>
<td>1</td>
<td>$10^5$</td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th>TREATMENT SYSTEM</th>
<th>DECONTAMINATION FACTOR</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Anion</td>
</tr>
<tr>
<td>Demineralizers</td>
<td></td>
</tr>
<tr>
<td>Mixed-bed</td>
<td></td>
</tr>
<tr>
<td>Reactor Coolant Cleanup</td>
<td>10</td>
</tr>
<tr>
<td>Condensate (deep bed)</td>
<td>10</td>
</tr>
<tr>
<td>High-purity waste</td>
<td>(10^2(10)^*)</td>
</tr>
<tr>
<td>Low-Purity Waste</td>
<td>(10^2(10))</td>
</tr>
<tr>
<td>Cation bed (any system)</td>
<td>1(1)</td>
</tr>
<tr>
<td>Anion bed (any system)</td>
<td>(10^2(10))</td>
</tr>
<tr>
<td>Powdex (any system)</td>
<td>(10(10))</td>
</tr>
<tr>
<td>Evaporators</td>
<td>All Nuclides Except Anions</td>
</tr>
<tr>
<td>Miscellaneous</td>
<td>(10^4)</td>
</tr>
<tr>
<td>Detergent wastes</td>
<td>(10^2)</td>
</tr>
<tr>
<td>Reverse Osmosis</td>
<td>All Nuclides</td>
</tr>
<tr>
<td>Laundry wastes</td>
<td>30</td>
</tr>
<tr>
<td>Other liquid wastes</td>
<td>10</td>
</tr>
<tr>
<td>Filters</td>
<td>DF of 1.0 for all nuclides</td>
</tr>
</tbody>
</table>

*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:

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

![Diagram of liquid waste processing and discharge]

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:

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

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

\[
T_d = \frac{0.8C}{R_c} \quad \text{for redundant tanks, or} \quad T_d = \frac{0.4C}{R_c} \quad \text{for a single tank}
\]

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

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

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) < \( T_p(R_b + R_o) \), \( T_p \) 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:

```
FLOOR DRAINS
1,000 GAL/DAY

FLOOR DRAIN TANK A
20,000 GAL

FLOOR DRAIN TANK B
20,000 GAL

DETERGENT WASTE
450 GAL/DAY

WASTE DTMIN.
100 GAL/MIN

WASTE EVAPORATOR
15 GAL/MIN

WASTE SAMPLE TANK A
40,000 GAL

WASTE SAMPLE TANK B
40,000 GAL

DISCHARGE PUMP 10 GAL/MIN.
```

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, \(0.8C/(R_b + R_o) = 1.45 \text{ days}\), 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: Gland Seal Steam Flow

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

```
1-15
```
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**

<table>
<thead>
<tr>
<th>Activated Carbon&lt;sup&gt;a&lt;/sup&gt; Bed Depth</th>
<th>Removal Efficiencies&lt;sup&gt;b&lt;/sup&gt; for Radiiodine %</th>
</tr>
</thead>
<tbody>
<tr>
<td>2 inches. Air filtration system</td>
<td>90.</td>
</tr>
<tr>
<td>designed to operate inside reactor</td>
<td></td>
</tr>
<tr>
<td>containment</td>
<td></td>
</tr>
<tr>
<td>2 inches. Air filtration system</td>
<td>70.</td>
</tr>
<tr>
<td>designed to operate outside the</td>
<td></td>
</tr>
<tr>
<td>reactor containment and relative</td>
<td></td>
</tr>
<tr>
<td>humidity is controlled at 70%</td>
<td></td>
</tr>
<tr>
<td>4 inches. Air filtration system</td>
<td>90.</td>
</tr>
<tr>
<td>designed to operate outside the</td>
<td></td>
</tr>
<tr>
<td>reactor containment and relative</td>
<td></td>
</tr>
<tr>
<td>humidity is controlled at 70%</td>
<td></td>
</tr>
<tr>
<td>6 inches. Air filtration system</td>
<td>99.</td>
</tr>
<tr>
<td>designed to operate outside the</td>
<td></td>
</tr>
<tr>
<td>reactor containment and relative</td>
<td></td>
</tr>
<tr>
<td>humidity is controlled to 70%</td>
<td></td>
</tr>
</tbody>
</table>

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

<sup>b</sup>The removal efficiencies assigned HEPA filters for particulate removal and charcoal adsorbers for radiiodine 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 offgases 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 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 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 charcoal adsorbers do not satisfy the guidelines of Regulatory Guide 1.140 (Ref. 2), enter 0.0 in spaces 43-46; if treatment is provided to remove particulates or if 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.

<table>
<thead>
<tr>
<th>OPERATING 77°F</th>
<th>OPERATING 77°F</th>
<th>OPERATING 77°F</th>
<th>OPERATING 0°F</th>
</tr>
</thead>
<tbody>
<tr>
<td>DEW POINT 45°F</td>
<td>DEW POINT 0°F</td>
<td>DEW POINT -40°F</td>
<td>DEW POINT -20°F</td>
</tr>
<tr>
<td>Kr</td>
<td>18.5</td>
<td>25.</td>
<td>70.</td>
</tr>
</tbody>
</table>

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.

<table>
<thead>
<tr>
<th>OPERATING 77°F</th>
<th>OPERATING 77°F</th>
<th>OPERATING 77°F</th>
<th>OPERATING 0°F</th>
</tr>
</thead>
<tbody>
<tr>
<td>DEW POINT 45°F</td>
<td>DEW POINT 0°F</td>
<td>DEW POINT -40°F</td>
<td>DEW POINT -20°F</td>
</tr>
<tr>
<td>Xe</td>
<td>330.</td>
<td>440.</td>
<td>1160.</td>
</tr>
</tbody>
</table>

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.
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

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Oyster Creek</td>
<td>12/69</td>
<td>77</td>
<td>65</td>
<td>66</td>
<td>58</td>
<td>70</td>
<td>58e</td>
</tr>
<tr>
<td>Nine Mile Point-1</td>
<td>12/69</td>
<td>62</td>
<td>68</td>
<td>63</td>
<td>60</td>
<td>81</td>
<td>57f</td>
</tr>
<tr>
<td>Millstone-1</td>
<td>03/71</td>
<td>55</td>
<td>34d</td>
<td>63</td>
<td>68</td>
<td>66</td>
<td>84</td>
</tr>
<tr>
<td>Monticello</td>
<td>06/71</td>
<td>74</td>
<td>68</td>
<td>57</td>
<td>61</td>
<td>84</td>
<td>75</td>
</tr>
<tr>
<td>Dresden-3</td>
<td>11/71</td>
<td>67</td>
<td>54</td>
<td>47e</td>
<td>33f</td>
<td>60</td>
<td>76</td>
</tr>
<tr>
<td>Dresden-2</td>
<td>06/72</td>
<td>74</td>
<td>51</td>
<td>44f</td>
<td>66</td>
<td>51f</td>
<td></td>
</tr>
<tr>
<td>Vermont Yankee</td>
<td>11/72</td>
<td>44e</td>
<td>59</td>
<td>81</td>
<td>73</td>
<td>80e</td>
<td></td>
</tr>
<tr>
<td>Pilgrim-1</td>
<td>12/72</td>
<td>72</td>
<td>34f</td>
<td>46e</td>
<td>43f</td>
<td>47f</td>
<td></td>
</tr>
<tr>
<td>Quad Cities-1</td>
<td>02/73</td>
<td>51f</td>
<td>65</td>
<td>52f</td>
<td>55e</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Quad Cities-2</td>
<td>03/73</td>
<td>68</td>
<td>40f</td>
<td>66</td>
<td>67</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cooper</td>
<td>07/74</td>
<td>60</td>
<td>57</td>
<td>70</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Peach Bottom-2</td>
<td>07/74</td>
<td>57</td>
<td>61</td>
<td>45f</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Peach Bottom-3</td>
<td>12/74</td>
<td>59</td>
<td>67</td>
<td>54f</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Duane Arnold</td>
<td>02/75</td>
<td>55</td>
<td>67</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fitzpatrick</td>
<td>07/75</td>
<td>59</td>
<td>55f</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Brunswick-2</td>
<td>11/75</td>
<td>37g</td>
<td>35g</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Hatch-1</td>
<td>12/75</td>
<td>----</td>
<td>----</td>
<td>----</td>
<td>65</td>
<td>57g</td>
<td></td>
</tr>
</tbody>
</table>

AVERAGE 67 67 61 63 66 70

aFrom Semi-Annual Operating Reports for each facility, as submitted by respective licensees.
bBig 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.
cPlant capacity factors listed begin with the first full year of commercial operation.
dNot included due to extended maintenance outage to replace feedwater sparger.
eNot included due to extended operation at reduced power.
fNot included due to extended refueling outage.
gNot 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 $\mu$Ci/gm)

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

*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 x 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)

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

*Measured values increased to account for liquid recycle.
### TABLE 2-2 (Continued)

<table>
<thead>
<tr>
<th>ISOTOPES</th>
<th>REACTOR COOLANT</th>
<th>REACTOR STEAM</th>
</tr>
</thead>
<tbody>
<tr>
<td>Te-132</td>
<td>1(-5)</td>
<td>1(-8)</td>
</tr>
<tr>
<td>Ba-139</td>
<td>1(-2)</td>
<td>1(-5)</td>
</tr>
<tr>
<td>Ba-140</td>
<td>4(-4)</td>
<td>4(-7)</td>
</tr>
<tr>
<td>Ba-141</td>
<td>1(-2)</td>
<td>1(-5)</td>
</tr>
<tr>
<td>Ba-142</td>
<td>6(-3)</td>
<td>6(-6)</td>
</tr>
<tr>
<td>La-142</td>
<td>5(-3)</td>
<td>5(-6)</td>
</tr>
<tr>
<td>Ce-141</td>
<td>3(-5)</td>
<td>3(-8)</td>
</tr>
<tr>
<td>Ce-143</td>
<td>3(-5)</td>
<td>3(-8)</td>
</tr>
<tr>
<td>Ce-144</td>
<td>3(-6)</td>
<td>3(-9)</td>
</tr>
<tr>
<td>Pr-143</td>
<td>4(-5)</td>
<td>4(-8)</td>
</tr>
<tr>
<td>Nd-147</td>
<td>3(-6)</td>
<td>3(-9)</td>
</tr>
<tr>
<td>W-187</td>
<td>3(-4)</td>
<td>3(-7)</td>
</tr>
<tr>
<td>Np-239</td>
<td>8(-3)</td>
<td>8(-6)</td>
</tr>
</tbody>
</table>

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

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

*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

<table>
<thead>
<tr>
<th>SYMBOL</th>
<th>DESCRIPTION</th>
<th>NOBLE GASES</th>
<th>HALOGENS</th>
<th>Cs, Rb</th>
<th>WATER ACTIVATION PRODUCTS</th>
<th>TRITIUM</th>
<th>OTHER NUCLIDES</th>
</tr>
</thead>
<tbody>
<tr>
<td>NA</td>
<td>Fraction of material removed in the reactor water cleanup system</td>
<td>0.0</td>
<td>0.9</td>
<td>0.5</td>
<td>0.0</td>
<td>0.0</td>
<td>0.9*</td>
</tr>
<tr>
<td>NB</td>
<td>Fraction of material removed by the condensate demineralizers</td>
<td>0.0</td>
<td>0.9</td>
<td>0.5</td>
<td>0.0</td>
<td>0.0</td>
<td>0.9*</td>
</tr>
<tr>
<td>NS</td>
<td>Ratio of concentration in reactor steam to the concentration in reactor water</td>
<td>**</td>
<td>0.015†††</td>
<td>0.001***</td>
<td>1.0</td>
<td>0.001</td>
<td></td>
</tr>
<tr>
<td>R</td>
<td>Removal rate from the reactor water (hr⁻¹)</td>
<td>**</td>
<td>0.40</td>
<td>0.17</td>
<td>***</td>
<td>††</td>
<td>0.31</td>
</tr>
</tbody>
</table>

* 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 coolant 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:

\[ R = \frac{FA \cdot NA + NC \cdot FS \cdot NS \cdot NB}{WP} \]

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>
<thead>
<tr>
<th>NUCLIDES</th>
<th>REACTOR COOLANT</th>
<th>REACTOR STEAM</th>
</tr>
</thead>
<tbody>
<tr>
<td>Noble gases*</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Halogens**</td>
<td>$\frac{P_{(110) \text{ lb} \cdot 0.40 + \lambda}}{\text{MWe} \cdot R + \lambda}$</td>
<td>$\frac{P_{(110) \text{ lb} \cdot 0.40 + \lambda}}{\text{MWe} \cdot R + \lambda}$</td>
</tr>
<tr>
<td>Cs, Rb</td>
<td>$\frac{P_{(110) \text{ lb} \cdot 0.17 + \lambda}}{\text{MWe} \cdot R + \lambda}$</td>
<td>$\frac{P_{(110) \text{ lb} \cdot 0.17 + \lambda}}{\text{MWe} \cdot R + \lambda}$</td>
</tr>
<tr>
<td>Water activation products</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Tritium***</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Other nuclides</td>
<td>$\frac{P_{(110) \text{ lb} \cdot 0.31 + \lambda}}{\text{MWe} \cdot R + \lambda}$</td>
<td>$\frac{P_{(110) \text{ lb} \cdot 0.31 + \lambda}}{\text{MWe} \cdot R + \lambda}$</td>
</tr>
</tbody>
</table>

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

**$\lambda$ 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 \( \mu \text{Ci/gm} \);
- \( k \) is a conversion factor, 454 gm/lbs;
- \( R \) is the removal rate of the isotope from the system due to demineralization, leakage, etc., in \( \text{hr}^{-1} \);
- \( s \) is the rate of release to and/or production of the isotope in the system, in \( \mu \text{Ci/hr} \);
- \( w \) is the fluid weight, in lb; and
- \( \lambda \) is the decay constant, in \( \text{hr}^{-1} \).

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:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Sample Case Value</th>
<th>Range Values</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power level (MWt)</td>
<td>3758</td>
<td>3000-3800</td>
</tr>
<tr>
<td>Water weight in vessel (lbs)</td>
<td>4.9 x 10^5</td>
<td>3.4 x 10^5 - 4.2 x 10^5</td>
</tr>
<tr>
<td>Cleanup demineralizer flow (lbs/hr)</td>
<td>1.5 x 10^5</td>
<td>1.1 x 10^5 - 1.5 x 10^5</td>
</tr>
<tr>
<td>Steam flow rate (lbs/hr)</td>
<td>15.4 x 10^6</td>
<td>13.0 x 10^6 - 17.0 x 10^6</td>
</tr>
<tr>
<td>Condensate demineralizer flow fraction</td>
<td>0.75</td>
<td>0.5 - 0.99</td>
</tr>
</tbody>
</table>

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:

\[ A = \frac{P}{\Delta P} \left( \frac{0.40 + \lambda}{R + \lambda} \right) \]  \hspace{1cm} (2-1)

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:

\[
R = \frac{FA \cdot NA + NC \cdot FS \cdot NS \cdot NB}{WP}
\]  \hspace{1cm} (2-2)

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

\[
= (\text{adjustment factor}) \times (\text{standard I-131 concentration})
\]

\[
= 0.96 \times 4 \times 10^{-3} \ \mu\text{Ci/g} = 3.8 \times 10^{-3} \ \mu\text{Ci/gm}
\]

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:

\[
A = \frac{P_{\text{cs}}(110)\left(0.17 + \frac{\lambda}{4.9 \times 10^5}\right)}{R + \lambda}
\]  \hspace{1cm} (2-3)

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:

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

The adjusted Cs-137 concentration

\[
= (\text{adjustment factor}) \times (\text{standard Cs-137 concentration})
\]

\[
= 0.96 \times 8 \times 10^{-5} \ \mu\text{Ci/g} = 7.6 \times 10^{-5} \ \mu\text{Ci/gm}
\]

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:

\[
A = \frac{P_{\text{na}}(110)\left(0.31 + \frac{\lambda}{4.9 \times 10^5}\right)}{R + \lambda}
\]  \hspace{1cm} (2-4)

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:

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

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

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

The adjusted concentration of Na-24

$$= (\text{adjustment factor}) \times (\text{standard Na-24 concentration})$$

$$= 0.92 \times 1 \times 10^{-2} \mu\text{Ci/gm} = 9.2 \times 10^{-3} \mu\text{Ci/gm}$$

The noble gas concentrations in Table 2-2 are based on an offgas release rate of 50,000 $\mu$Ci/sec measured at 30-minute decay. The value of 50,000 $\mu$Ci/sec can be determined as discussed below. Recent data supplied by General Electric (Refs. 5, 6, 7) shows that improved (7x7R and 8x8) fuel which is in the process of being installed in both new and reload cores has shown considerable improvement over the 7x7 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 7x7 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 $\mu$Ci/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 these nuclides.
<table>
<thead>
<tr>
<th>Facility</th>
<th>1977</th>
<th>1976</th>
<th>1975</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cooper</td>
<td>290</td>
<td>31</td>
<td>-</td>
</tr>
<tr>
<td>Dresden 3</td>
<td>215,000</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Duane Arnold</td>
<td>NA</td>
<td>1,300</td>
<td>-</td>
</tr>
<tr>
<td>Fitzpatrick</td>
<td>1,190</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Hatch 1</td>
<td>1,300</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Millstone 1</td>
<td>NA</td>
<td>152,000</td>
<td>-</td>
</tr>
<tr>
<td>Monticello</td>
<td>91,000</td>
<td>137,000</td>
<td>-</td>
</tr>
<tr>
<td>Nine Mile Pt 1</td>
<td>60,000</td>
<td>64,000</td>
<td>-</td>
</tr>
<tr>
<td>Oyster Creek</td>
<td>113,000</td>
<td>79,000</td>
<td>-</td>
</tr>
<tr>
<td>Peach Bottom 2</td>
<td>12,000</td>
<td>10,000</td>
<td>-</td>
</tr>
<tr>
<td>Peach Bottom 3</td>
<td>6,900</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Vermont Yankee</td>
<td>NA</td>
<td>11,000</td>
<td>12,000</td>
</tr>
<tr>
<td>Total</td>
<td>55,000</td>
<td>56,000</td>
<td>12,000</td>
</tr>
</tbody>
</table>

*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*

<table>
<thead>
<tr>
<th>Plant</th>
<th>Partition Coefficient</th>
<th>Ref</th>
<th>Plant</th>
<th>Partition Coefficient</th>
<th>Ref</th>
</tr>
</thead>
<tbody>
<tr>
<td>Oyster Creek</td>
<td>0.023</td>
<td>3, 4, 11</td>
<td>Monticello</td>
<td>0.004</td>
<td>3</td>
</tr>
<tr>
<td>Dresden 2</td>
<td>0.017</td>
<td>3, 12</td>
<td>Browns Ferry 1</td>
<td>0.005</td>
<td>5</td>
</tr>
<tr>
<td>Dresden 3</td>
<td>0.021</td>
<td>3, 12</td>
<td>Browns Ferry 2</td>
<td>0.0023</td>
<td>5</td>
</tr>
<tr>
<td>Millstone 1</td>
<td>0.012</td>
<td>3, 4</td>
<td>Browns Ferry 3</td>
<td>0.003</td>
<td>5</td>
</tr>
<tr>
<td>Nine Mile Point 1</td>
<td>0.02</td>
<td>4</td>
<td>Duane Arnold</td>
<td>0.004</td>
<td>5</td>
</tr>
<tr>
<td>Quad Cities 1</td>
<td>0.013</td>
<td>3</td>
<td>Hatch 1</td>
<td>0.0035</td>
<td>5</td>
</tr>
<tr>
<td>Cooper</td>
<td>0.012</td>
<td>5</td>
<td>Peach Bottom 2</td>
<td>0.004</td>
<td>5</td>
</tr>
<tr>
<td>Fitzpatrick</td>
<td>0.018</td>
<td>5</td>
<td>Peach Bottom 3</td>
<td>0.0044</td>
<td>5</td>
</tr>
<tr>
<td>Pilgrim 1</td>
<td>0.0082</td>
<td>5</td>
<td>Vermont Yankee</td>
<td>0.004</td>
<td>5</td>
</tr>
<tr>
<td><strong>AVERAGE</strong></td>
<td>0.015</td>
<td></td>
<td><strong>AVERAGE</strong></td>
<td>0.004</td>
<td></td>
</tr>
</tbody>
</table>

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
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:

\[ R_{BWR} = R^I_N \times C_{BWR} \]

where

- \( R_{BWR} \) = calculated annual release for particular BWR, Ci/yr
- \( R^I_N \) = 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:

\[ R_{BWR} = R^I_N \times C_{BWR} \times P_{C_{BWR}} \]

where

- \( R_{BWR} \) = calculated annual release for particular BWR, Ci/yr
- \( R^I_N \) = 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
- \( P_{C_{BWR}} \) = 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.

2-15
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 source 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, hypoiodous 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:

<table>
<thead>
<tr>
<th>FRACTION OF IODINE APPEARING IN EACH CHEMICAL FORM</th>
<th>CONTAINMENT</th>
<th>AUXILIARY</th>
<th>TURBINE</th>
<th>RADWASTE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Particulate</td>
<td>0.11</td>
<td>0.2</td>
<td>0.2</td>
<td>0.002</td>
</tr>
<tr>
<td>Elemental</td>
<td>0.32</td>
<td>0.48</td>
<td>0.50</td>
<td>0.28</td>
</tr>
<tr>
<td>HOI</td>
<td>0.38</td>
<td>0.24</td>
<td>0.22</td>
<td>0.25</td>
</tr>
<tr>
<td>Organic</td>
<td>0.19</td>
<td>0.09</td>
<td>0.08</td>
<td>0.47</td>
</tr>
</tbody>
</table>
### TABLE 2-8
ANNUAL IODINE NORMALIZED RELEASES*
FROM TURBINE BUILDING VENTILATION SYSTEMS

**NORMAL OPERATION**

<table>
<thead>
<tr>
<th>Data Source</th>
<th>Normalized Release (Ci/yr/μCi/gm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Monticello (Ref. 15)</td>
<td>3.1 x 10^3</td>
</tr>
<tr>
<td>Oyster Creek (Refs. 15, 16)</td>
<td>6.0 x 10^3</td>
</tr>
<tr>
<td>Vermont Yankee (Ref. 15)</td>
<td>0.35 x 10^3</td>
</tr>
<tr>
<td>Pilgrim (Ref. 16)</td>
<td>8.5 x 10^3</td>
</tr>
<tr>
<td>Browns Ferry (Ref. 5)</td>
<td>1.3 x 10^3</td>
</tr>
<tr>
<td>References 3 and 5***</td>
<td>3.3 x 10^3</td>
</tr>
<tr>
<td><strong>AVERAGE</strong></td>
<td>3.8 x 10^3</td>
</tr>
</tbody>
</table>

**EXTENDED SHUTDOWN**

<table>
<thead>
<tr>
<th>Data Source</th>
<th>Normalized Release (Ci/yr/μCi/gm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Monticello (Ref. 15)</td>
<td>1.7 x 10^2</td>
</tr>
<tr>
<td>Oyster Creek (Ref. 15)</td>
<td>3.5 x 10^2**</td>
</tr>
<tr>
<td>Vermont Yankee (Ref. 15)</td>
<td>0.63 x 10^2</td>
</tr>
<tr>
<td>Browns Ferry (Ref. 5)</td>
<td>1.3 x 10^2</td>
</tr>
<tr>
<td>References 3 and 5***</td>
<td>1.4 x 10^3</td>
</tr>
<tr>
<td><strong>AVERAGE</strong></td>
<td>4.1 x 10^2**</td>
</tr>
</tbody>
</table>

---

*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 radiiodine between the water phase in the leakage and the gas phase where it is measured and the partition coefficient for radiiodine 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 x 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**

<table>
<thead>
<tr>
<th>Data Source</th>
<th>Normalized Releases (Ci/yr/µCi/gm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Monticello (Ref. 15)</td>
<td>11</td>
</tr>
<tr>
<td>Pilgrim (Ref. 16)</td>
<td>13</td>
</tr>
<tr>
<td>Brown Ferry (Ref. 5)</td>
<td>4.2</td>
</tr>
<tr>
<td>References 3 and 5***</td>
<td>21</td>
</tr>
<tr>
<td></td>
<td><strong>12.3</strong></td>
</tr>
</tbody>
</table>

**EXTENDED SHUTDOWN**

<table>
<thead>
<tr>
<th>Data Source</th>
<th>Normalized Releases (Ci/yr/µCi/gm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Monticello (Ref. 15)</td>
<td>0.47</td>
</tr>
<tr>
<td>Oyster Creek (Ref. 15)</td>
<td>1.3</td>
</tr>
<tr>
<td>Vermont Yankee (Ref. 15)</td>
<td>3.2</td>
</tr>
<tr>
<td>Browns Ferry (Ref. 5)</td>
<td>1.4</td>
</tr>
<tr>
<td>References 3 and 5***</td>
<td><strong>20</strong></td>
</tr>
<tr>
<td></td>
<td>5.2</td>
</tr>
</tbody>
</table>

*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 is in the 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

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

EXTENDED SHUTDOWN

<table>
<thead>
<tr>
<th>Data Source</th>
<th>Normalized Release (Ci/yr/μCi/gm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Monticello (Ref. 15)</td>
<td>0.02</td>
</tr>
<tr>
<td>Oyster Creek (Ref. 15)</td>
<td>1.4</td>
</tr>
<tr>
<td>Vermont Yankee (Ref. 15)</td>
<td>0.4</td>
</tr>
<tr>
<td>Browns Ferry (Ref. 5)</td>
<td>0.6</td>
</tr>
<tr>
<td>References 3 and 5**</td>
<td>4.4</td>
</tr>
<tr>
<td><strong>AVERAGE</strong></td>
<td>1.4</td>
</tr>
</tbody>
</table>

*The normalized release rate, expressed in grams of water during the modes of operation, represents the effective leak rate for radiiodine. It is the combination of the reactor water leakage rate into the buildings, and the partitioning of the radiiodine 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)

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

*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>
<thead>
<tr>
<th>NUCLIDE</th>
<th>CONTAINMENT BUILDING</th>
<th>AUXILIARY BUILDING</th>
<th>TURBINE BUILDING</th>
<th>RADWASTE BUILDING</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kr-83m</td>
<td>**</td>
<td>**</td>
<td>**</td>
<td>**</td>
</tr>
<tr>
<td>Kr-85m</td>
<td>1</td>
<td>3</td>
<td>25</td>
<td>**</td>
</tr>
<tr>
<td>Kr-85</td>
<td>**</td>
<td>**</td>
<td>**</td>
<td>**</td>
</tr>
<tr>
<td>Kr-87</td>
<td>**</td>
<td>2</td>
<td>61</td>
<td>**</td>
</tr>
<tr>
<td>Kr-88</td>
<td>1</td>
<td>3</td>
<td>91</td>
<td>**</td>
</tr>
<tr>
<td>Kr-89</td>
<td>**</td>
<td>2</td>
<td>580</td>
<td>29</td>
</tr>
<tr>
<td>Xe-131m</td>
<td>**</td>
<td>**</td>
<td>**</td>
<td>**</td>
</tr>
<tr>
<td>Xe-133m</td>
<td>**</td>
<td>**</td>
<td>**</td>
<td>**</td>
</tr>
<tr>
<td>Xe-133</td>
<td>27</td>
<td>83</td>
<td>150</td>
<td>220</td>
</tr>
<tr>
<td>Xe-135m</td>
<td>15</td>
<td>45</td>
<td>400</td>
<td>530</td>
</tr>
<tr>
<td>Xe-135</td>
<td>33</td>
<td>94</td>
<td>330</td>
<td>280</td>
</tr>
<tr>
<td>Xe-137</td>
<td>45</td>
<td>135</td>
<td>1000</td>
<td>83</td>
</tr>
<tr>
<td>Xe-138</td>
<td>2</td>
<td>6</td>
<td>1000</td>
<td>2</td>
</tr>
<tr>
<td>Cr-51*</td>
<td>0.0002</td>
<td>0.0009</td>
<td>0.0009</td>
<td>0.0007</td>
</tr>
<tr>
<td>Mn-54</td>
<td>0.0004</td>
<td>0.001</td>
<td>0.0006</td>
<td>0.004</td>
</tr>
<tr>
<td>Fe-59</td>
<td>0.00009</td>
<td>0.0003</td>
<td>0.0001</td>
<td>0.0003</td>
</tr>
<tr>
<td>Co-58</td>
<td>0.0001</td>
<td>0.0002</td>
<td>0.001</td>
<td>0.0002</td>
</tr>
<tr>
<td>Co-60</td>
<td>0.001</td>
<td>0.004</td>
<td>0.001</td>
<td>0.007</td>
</tr>
<tr>
<td>Zn-65</td>
<td>0.001</td>
<td>0.004</td>
<td>0.006</td>
<td>0.003</td>
</tr>
<tr>
<td>Sr-89</td>
<td>0.00003</td>
<td>0.00002</td>
<td>0.006</td>
<td>NA</td>
</tr>
<tr>
<td>Sr-90</td>
<td>0.000003</td>
<td>0.000007</td>
<td>0.00002</td>
<td>NA</td>
</tr>
<tr>
<td>Zr-95</td>
<td>0.0003</td>
<td>0.0007</td>
<td>0.00004</td>
<td>0.0008</td>
</tr>
<tr>
<td>Nb-95</td>
<td>0.001</td>
<td>0.009</td>
<td>0.000006</td>
<td>0.000004</td>
</tr>
<tr>
<td>Mo-99</td>
<td>0.006</td>
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<td>0.002</td>
<td>0.000003</td>
</tr>
<tr>
<td>Ru-103</td>
<td>0.0002</td>
<td>0.004</td>
<td>0.00005</td>
<td>0.000001</td>
</tr>
<tr>
<td>Ag-110</td>
<td>0.0000004</td>
<td>0.000002</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>Sb-124</td>
<td>0.00002</td>
<td>0.00003</td>
<td>0.0001</td>
<td>0.00007</td>
</tr>
<tr>
<td>Cs-134</td>
<td>0.0007</td>
<td>0.004</td>
<td>0.0002</td>
<td>0.0024</td>
</tr>
<tr>
<td>Cs-136</td>
<td>0.0001</td>
<td>0.0004</td>
<td>0.0001</td>
<td>NA</td>
</tr>
<tr>
<td>Cs-137</td>
<td>0.001</td>
<td>0.005</td>
<td>0.001</td>
<td>0.004</td>
</tr>
<tr>
<td>Ba-140</td>
<td>0.002</td>
<td>0.02</td>
<td>0.010</td>
<td>0.000004</td>
</tr>
<tr>
<td>Ce-141</td>
<td>0.0002</td>
<td>0.0007</td>
<td>0.010</td>
<td>0.000007</td>
</tr>
</tbody>
</table>

*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)

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

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)

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

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
($\mu$Ci/sec)

<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>MILLSTONE-1</th>
<th>OYSTER CREEK</th>
<th>MONTICELLO</th>
<th>AVERAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kr-89</td>
<td>ND</td>
<td>ND</td>
<td>3.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Xe-133m</td>
<td>ND</td>
<td>ND</td>
<td>5.3</td>
<td>1.8</td>
</tr>
<tr>
<td>Xe-133</td>
<td>0.25</td>
<td>0.56</td>
<td>26</td>
<td>8.9</td>
</tr>
<tr>
<td>Xe-135m</td>
<td>ND</td>
<td>4</td>
<td>59</td>
<td>21</td>
</tr>
<tr>
<td>Xe-135</td>
<td>2.0</td>
<td>1.5</td>
<td>20</td>
<td>7.8</td>
</tr>
<tr>
<td>Xe-137</td>
<td>ND</td>
<td>ND</td>
<td>10</td>
<td>3.3</td>
</tr>
<tr>
<td>Xe-138</td>
<td>ND</td>
<td>ND</td>
<td>0.2</td>
<td>ND</td>
</tr>
</tbody>
</table>

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
($10^{-6}$Ci/sec)

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

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 $6.3 \times 10^{-6}$ 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 x $10^{-1}$ Ci/yr per $\mu$Ci/gm of $I$-131 in the reactor coolant.
- $I$-133 = 2.2 x $10^{-1}$ Ci/yr per $\mu$Ci/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 radiiodine 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>
<thead>
<tr>
<th>NUCLIDE</th>
<th>QUAD CITIES 1</th>
<th>QUAD CITIES 2</th>
<th>VERMONT YANKEE</th>
<th>OYSTER CREEK</th>
<th>MONTICELLO</th>
<th>AVERAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Co-60</td>
<td>16</td>
<td>0.88</td>
<td>250</td>
<td>330</td>
<td>1.70</td>
<td>120</td>
</tr>
<tr>
<td>Co-58</td>
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<td>0.35</td>
<td>41</td>
<td>19</td>
<td>ND</td>
<td>13</td>
</tr>
<tr>
<td>Cr-51</td>
<td>8.9</td>
<td>0.50</td>
<td>63</td>
<td>28</td>
<td>ND</td>
<td>20</td>
</tr>
<tr>
<td>Mn-54</td>
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<td>0.061</td>
<td>20</td>
<td>140</td>
<td>47</td>
<td>42</td>
</tr>
<tr>
<td>Fe-59</td>
<td>ND</td>
<td>ND</td>
<td>21</td>
<td>5.0</td>
<td>3.4</td>
<td>6.0</td>
</tr>
<tr>
<td>Zn-65</td>
<td>58</td>
<td>0.11</td>
<td>770</td>
<td>1.4</td>
<td>73</td>
<td>180</td>
</tr>
<tr>
<td>Sr-89</td>
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<td>NA</td>
<td>NA</td>
<td>2.0</td>
<td>NA</td>
<td>2.0</td>
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<td>Sr-90</td>
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<td>NA</td>
<td>0.36</td>
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<tr>
<td>Zr-95</td>
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<td>ND</td>
<td>78</td>
<td>0.24</td>
<td>ND</td>
<td>16</td>
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<td>Nb-95</td>
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<td>0.021</td>
<td>ND</td>
<td>0.41</td>
<td>160</td>
<td>32</td>
</tr>
<tr>
<td>Mo-99</td>
<td>NA</td>
<td>NA</td>
<td>ND</td>
<td>13</td>
<td>4.4</td>
<td>5.8</td>
</tr>
<tr>
<td>Ru-103</td>
<td>NA</td>
<td>NA</td>
<td>NA</td>
<td>1.3</td>
<td>36</td>
<td>19</td>
</tr>
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<td>ND</td>
<td>7.0</td>
<td>ND</td>
<td>1.8</td>
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<tr>
<td>Cs-124</td>
<td>ND</td>
<td>ND</td>
<td>NA</td>
<td>ND</td>
<td>ND</td>
<td></td>
</tr>
<tr>
<td>Cs-134</td>
<td>6.2</td>
<td>0.14</td>
<td>82</td>
<td>13</td>
<td>170</td>
<td>54</td>
</tr>
<tr>
<td>Cs-136</td>
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<td>NA</td>
<td>20</td>
<td>ND</td>
<td>21</td>
<td>11</td>
</tr>
<tr>
<td>Cs-137</td>
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<td>0.54</td>
<td>240</td>
<td>23</td>
<td>200</td>
<td>95</td>
</tr>
<tr>
<td>Ba-140</td>
<td>ND</td>
<td>ND</td>
<td>14</td>
<td>1.1</td>
<td>200</td>
<td>43</td>
</tr>
<tr>
<td>Ce-141</td>
<td>ND</td>
<td>ND</td>
<td>7.5</td>
<td>45</td>
<td>13</td>
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</table>

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

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

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)

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

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)

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

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

$(10^{-6} \mu Ci/sec)$

<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>MONTICELLO</th>
<th>AVERAGE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Co-60</td>
<td>1.3</td>
<td>1.3</td>
</tr>
<tr>
<td>Co-58</td>
<td>0.21</td>
<td>0.21</td>
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<tr>
<td>Cr-51</td>
<td>ND</td>
<td>0</td>
</tr>
<tr>
<td>Mn-54</td>
<td>0.40</td>
<td>0.40</td>
</tr>
<tr>
<td>Fe-59</td>
<td>3.2</td>
<td>3.2</td>
</tr>
<tr>
<td>Zn-65</td>
<td>5.1</td>
<td>5.1</td>
</tr>
<tr>
<td>Sr-89</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>Sr-90</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>Nb-95</td>
<td>6.0</td>
<td>6.0</td>
</tr>
<tr>
<td>Mo-99</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Ru-103</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Ru-103</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Sb-124</td>
<td>ND</td>
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<tr>
<td>Cs-134</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>Cs-136</td>
<td>ND</td>
<td>0</td>
</tr>
<tr>
<td>Cs-137</td>
<td>2.2</td>
<td>2.2</td>
</tr>
<tr>
<td>Ba-140</td>
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NA - Not Analyzed.
ND - Not Detected. For averaging purposes a value of zero was assumed.
<table>
<thead>
<tr>
<th>FACILITY</th>
<th>1972 RADIOIODINE RELEASE (Ci/yr)</th>
<th>INTEGRATED THERMAL POWER (10^6 MWD)</th>
<th>1973 RADIOIODINE RELEASE (Ci/yr)</th>
<th>INTEGRATED THERMAL POWER (10^6 MWD)</th>
<th>1974 RADIOIODINE RELEASE (Ci/yr)</th>
<th>INTEGRATED THERMAL POWER (10^6 MWD)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Oyster Creek</td>
<td>6.3</td>
<td>0.542</td>
<td>6.7</td>
<td>0.453</td>
<td>3.3</td>
<td>0.46</td>
</tr>
<tr>
<td>Nine Mile Point 1</td>
<td>0.89</td>
<td>0.417</td>
<td>1.9</td>
<td>0.457</td>
<td>0.7</td>
<td>0.43</td>
</tr>
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<td>Millstone 1</td>
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<td>0.15</td>
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<td>0.47</td>
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<td>1.05</td>
<td>9.8</td>
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<td>4.0</td>
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<td>0.413</td>
<td>5.7</td>
<td>0.34</td>
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<td>Pilgrim 1</td>
<td>c</td>
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<td>0.46</td>
<td>0.523</td>
<td>1.4</td>
<td>0.25</td>
</tr>
<tr>
<td>Quad Cities 1/2b</td>
<td>c</td>
<td>5.5</td>
<td>1.32</td>
<td>4.2</td>
<td>8.2</td>
<td>1.09</td>
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<tr>
<td><strong>Average</strong></td>
<td><strong>4.6</strong></td>
<td><strong>1.32</strong></td>
<td><strong>4.2</strong></td>
<td><strong>1.32</strong></td>
<td><strong>6.2</strong></td>
<td></td>
</tr>
</tbody>
</table>

Combined average for 1972 through 1976 = 6.3 x 10^-6 Ci/yr

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

<table>
<thead>
<tr>
<th>FACILITY</th>
<th>1975</th>
<th></th>
<th></th>
<th>1976</th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>RADIOIODINE RELEASE (Ci/yr)</td>
<td>INTEGRATED THERMAL POWER (10^6 Mwd)</td>
<td>Ci/yr 10^6 Mwd</td>
<td>RADIOIODINE RELEASE (Ci/yr)</td>
<td>INTEGRATED THERMAL POWER (10^6 Mwd)</td>
<td>Ci/yr 10^6 Mwd</td>
</tr>
<tr>
<td>Oyster Creek</td>
<td>5.5</td>
<td>0.41</td>
<td>13.4</td>
<td>6.2</td>
<td>0.49</td>
<td>12.7</td>
</tr>
<tr>
<td>Nine Mile Point 1</td>
<td>2.1</td>
<td>0.4</td>
<td>5.3</td>
<td>2.1</td>
<td>0.55</td>
<td>3.8(g)</td>
</tr>
<tr>
<td>Millstone 1</td>
<td>9.8</td>
<td>0.5</td>
<td>19.6</td>
<td>2.7</td>
<td>0.48</td>
<td>5.6</td>
</tr>
<tr>
<td>Dresden 2/3(b)</td>
<td>0.75</td>
<td>0.71</td>
<td>1.06</td>
<td>1.9</td>
<td>1.14</td>
<td>1.7(e)</td>
</tr>
<tr>
<td>Monticello</td>
<td>3.5</td>
<td>0.37</td>
<td>9.5</td>
<td>d</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pilgrim 1</td>
<td>h</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Quad Cities 1/2(b)</td>
<td>f</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Average</td>
<td></td>
<td></td>
<td>9.8</td>
<td></td>
<td></td>
<td>6.0</td>
</tr>
</tbody>
</table>
TABLE 2-23
RADIOIODINE RELEASE RATE FROM GLAND SEAL CONDENSER EXHAUST
FOR SYSTEMS USING MAIN STEAM FOR THE SEALING SYSTEM AT 7000 lbs/hr.

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

Weighted Average According to Sample Days
8.1(5)

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

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  
\[(10^{-6} \mu\text{Ci/sec})\]

<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>RELEASE RATE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cr-51</td>
<td>0.9</td>
</tr>
<tr>
<td>Co-60</td>
<td>0.5</td>
</tr>
<tr>
<td>Zn-65</td>
<td>0.3</td>
</tr>
<tr>
<td>Cs-134</td>
<td>0.8</td>
</tr>
<tr>
<td>Cs-136</td>
<td>1.1</td>
</tr>
<tr>
<td>Cs-137</td>
<td>3.3</td>
</tr>
<tr>
<td>Ba-140</td>
<td>2.1</td>
</tr>
</tbody>
</table>

### TABLE 2-25

PARTICULATE RELEASE RATE FROM VERMONT YANKEE MECHANICAL VACUUM PUMP AND GLAND EXHAUST CONDENSER VENT, REFUELING SHUTDOWN  
\[(10^{-6} \mu\text{Ci/sec})\]

<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>RELEASE RATE</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cs-134</td>
<td>0.45</td>
</tr>
<tr>
<td>Cs-136</td>
<td>0.13</td>
</tr>
<tr>
<td>Cs-137</td>
<td>1.0</td>
</tr>
<tr>
<td>Ba-140</td>
<td>1.6</td>
</tr>
</tbody>
</table>
### TABLE 2-26
NORMALIZED IODINE RELEASES FROM MECHANICAL VACUUM PUMP
(≤ 80 HRS)

<table>
<thead>
<tr>
<th>Plant</th>
<th>Normalized Release (Ci/yr/μCi/gm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Monticello</td>
<td>8.3(2)</td>
</tr>
<tr>
<td>Vermont Yankee</td>
<td>1.5(2)</td>
</tr>
<tr>
<td><strong>AVERAGE</strong></td>
<td><strong>4.9(2)</strong></td>
</tr>
</tbody>
</table>

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

<table>
<thead>
<tr>
<th>Plant</th>
<th>Normalized Release (Ci/yr/μCi/gm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Monticello</td>
<td>1.5(3)</td>
</tr>
<tr>
<td>Vermont Yankee</td>
<td>6.0(2)</td>
</tr>
<tr>
<td><strong>AVERAGE</strong></td>
<td><strong>1.1(3)</strong></td>
</tr>
</tbody>
</table>

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 assume 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, hypoiodous 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:

<table>
<thead>
<tr>
<th>Fraction of Iodine Appearing In Each Chemical Form From BWR Mechanical Vacuum Pump</th>
<th>Time &lt; 80 hrs¹</th>
<th>Time &gt; 80 hrs²</th>
</tr>
</thead>
<tbody>
<tr>
<td>Particulate</td>
<td>0.004</td>
<td>0.01</td>
</tr>
<tr>
<td>Elemental</td>
<td>0.009</td>
<td>0.06</td>
</tr>
<tr>
<td>HOI</td>
<td>0.023</td>
<td>0.21</td>
</tr>
<tr>
<td>Organic</td>
<td>0.97</td>
<td>0.72</td>
</tr>
</tbody>
</table>

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 ft³/min air inleakage to the main condenser per MWt of design reactor power with a minimum of 5 ft³/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 ft³/min to 250 ft³/min. (Refs. 21 and 22). Subsequent measurements at Dresden Unit 2 (Ref. 20), showed an air inleakage of 4.4 ft³/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 ft³/min per condenser shell and a statistical mean inleakage rate of 6.7 ft³/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 ft³/min for Unit No. 1 and 25 ft³/min for Unit No. 2; measurements ranged from 6 ft³/min to 55 ft³/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 ft³/min main condenser air inleakage for a plant with a design thermal power level of 3400 MWt. This is approximately equivalent to 0.0062 ft³/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 ft³/min should be used, independent of reactor design thermal power level.
**CONDENSER AIR INLEAKAGE**

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

2.2.9 **HOLDUP TIMES FOR CHARCOAL DELAY SYSTEMS**

2.2.9.1 **Parameter**

\[ T = \frac{43.1 \text{MK/P}}{} \]

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:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Operating 77°F Dew Point 45°F</th>
<th>Operating 77°F Dew Point -40°F</th>
<th>Operating 77°F Dew Point 0°F</th>
<th>Operating 77°F Dew Point -20°F</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kr</td>
<td>18.5</td>
<td>70</td>
<td>25</td>
<td>105</td>
</tr>
<tr>
<td>Xe</td>
<td>330</td>
<td>1160</td>
<td>440</td>
<td>2410</td>
</tr>
</tbody>
</table>

2.2.9.2 **Bases**

Charcoal delay systems are evaluated using the above equation and dynamic adsorption coefficients. \( T = \frac{MK}{flow \ rate} \) 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:

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

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

2.2.10 DECONTAMINATION FACTORS FOR CRYOGENIC DISTILLATION

2.2.10.1 Parameter

<table>
<thead>
<tr>
<th>NUCLIDES</th>
<th>DECONTAMINATION FACTOR</th>
</tr>
</thead>
<tbody>
<tr>
<td>I, Xe</td>
<td>$1 \times 10^4$</td>
</tr>
<tr>
<td>Kr</td>
<td>$4 \times 10^3$</td>
</tr>
</tbody>
</table>

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 $4 \times 10^3$ 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 = $4 \times 10^3$) 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:

<table>
<thead>
<tr>
<th>ACTIVATED CARBON BED DEPTH$^a$</th>
<th>ASSIGNED ACTIVATED CARBON REMOVAL EFFICIENCIES FOR RADIOIODINE</th>
</tr>
</thead>
<tbody>
<tr>
<td>2 inches. Air filtration system designed to operate inside primary containment.</td>
<td>90%</td>
</tr>
<tr>
<td>2 inches. Air filtration system designed to operate outside the primary containment and relative humidity is controlled to 70%.</td>
<td>70%</td>
</tr>
<tr>
<td>4 inches. Air filtration system designed to operate outside the primary containment and relative humidity is controlled to 70%.</td>
<td>90%</td>
</tr>
<tr>
<td>6 inches. Air filtration system designed to operate outside the primary containment and relative humidity is controlled to 70%.</td>
<td>99%</td>
</tr>
</tbody>
</table>

$^a$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


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

<table>
<thead>
<tr>
<th>SOURCE</th>
<th>DEEP BED PLANT WITH ULTRASONIC RESIN CLEANER</th>
<th>DEEP BED PLANT WITHOUT ULTRASONIC RESIN CLEANER OR A FILTER/DEMINERALIZER PLANT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Equipment Drains</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Drywell</td>
<td>3,400</td>
<td>3,400</td>
</tr>
<tr>
<td>Containment, auxiliary building, and fuel pool</td>
<td>3,700</td>
<td>3,700</td>
</tr>
<tr>
<td>Radwaste building</td>
<td>1,100</td>
<td>1,100</td>
</tr>
<tr>
<td>Turbine building</td>
<td>3,000</td>
<td>3,000</td>
</tr>
<tr>
<td>Ultrasonic resin cleaner</td>
<td>15,000</td>
<td>-</td>
</tr>
<tr>
<td>Resin rinse*</td>
<td>2,500</td>
<td>5,000</td>
</tr>
<tr>
<td>Floor Drains</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Drywell</td>
<td>700</td>
<td>700</td>
</tr>
<tr>
<td>Containment, auxiliary building, and fuel handling</td>
<td>2,000</td>
<td>2,000</td>
</tr>
<tr>
<td>Radwaste building</td>
<td>1,000</td>
<td>1,000</td>
</tr>
<tr>
<td>Turbine building</td>
<td>2,000</td>
<td>2,000</td>
</tr>
<tr>
<td>Other Sources</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cleanup phase separator decant</td>
<td>640</td>
<td>640</td>
</tr>
<tr>
<td>Laundry drains</td>
<td>1,000</td>
<td>1,000</td>
</tr>
<tr>
<td>Lab drains</td>
<td>500</td>
<td>500</td>
</tr>
<tr>
<td>Regenerants*</td>
<td>1,700</td>
<td>3,400</td>
</tr>
<tr>
<td>Condensate demineralizer backwash</td>
<td>-</td>
<td>8,100</td>
</tr>
<tr>
<td>Chemical lab waste</td>
<td>100</td>
<td>100</td>
</tr>
</tbody>
</table>

*Deep-bed condensate demineralizers only.
**Calculated by BWR-GALE Code.
\[ \text{Filter/demineralizer (Powdex) condensate demineralizers only.} \]
TABLE 2-28
CALCULATED ANNUAL RELEASE OF RADIOACTIVE MATERIALS
IN UNTREATED DETERGENT WASTE FOR A BWR

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

TOTAL 0.09 Ci
<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>C-14</td>
<td>1.2(-2)</td>
<td>NA^e</td>
<td>2(-2)</td>
<td>4.2(-2)</td>
</tr>
<tr>
<td>P-32</td>
<td>1.5(-2)</td>
<td>NA</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>Cr-51</td>
<td>2.3(-1)</td>
<td>NA</td>
<td>1.3</td>
<td>NA</td>
</tr>
<tr>
<td>Mn-54</td>
<td>1.3</td>
<td>1.1(-1)</td>
<td>1.3(-1)</td>
<td>2.3(-2)</td>
</tr>
<tr>
<td>Fe-55</td>
<td>3.5(-1)</td>
<td>NA</td>
<td>1.9</td>
<td>1.6(-1)</td>
</tr>
<tr>
<td>Fe-59</td>
<td>2.9(-2)</td>
<td>NA</td>
<td>2.6(-1)</td>
<td>NA</td>
</tr>
<tr>
<td>Co-57</td>
<td>7.5(-3)</td>
<td>NA</td>
<td>1.7(-2)</td>
<td>NA</td>
</tr>
<tr>
<td>Co-58</td>
<td>3.5(-1)</td>
<td>4.1(-1)</td>
<td>2.3</td>
<td>1(-1)</td>
</tr>
<tr>
<td>Co-60</td>
<td>3.8</td>
<td>9(-1)</td>
<td>9.1(-1)</td>
<td>4(-2)</td>
</tr>
<tr>
<td>Ni-63</td>
<td>NA</td>
<td>NA</td>
<td>3.5(-1)</td>
<td>7.1(-2)</td>
</tr>
<tr>
<td>Sr-89</td>
<td>2.1(-2)</td>
<td>NA</td>
<td>7(-3)</td>
<td>1.4(-3)</td>
</tr>
<tr>
<td>Sr-90</td>
<td>2.5(-3)</td>
<td>NA</td>
<td>7.4(-3)</td>
<td>NA</td>
</tr>
<tr>
<td>Y-91</td>
<td>NA</td>
<td>NA</td>
<td>7.4(-2)</td>
<td>NA</td>
</tr>
<tr>
<td>Zr-95</td>
<td>8.3(-2)</td>
<td>1.4(-1)</td>
<td>1.4(-1)</td>
<td>NA</td>
</tr>
<tr>
<td>Nb-95</td>
<td>1.6(-2)</td>
<td>2(-1)</td>
<td>2.7(-1)</td>
<td>NA</td>
</tr>
<tr>
<td>Mo-99</td>
<td>NA</td>
<td>5(-3)</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>Ru-103</td>
<td>1.3(-2)</td>
<td>1.4(-2)</td>
<td>5.2(-2)</td>
<td>NA</td>
</tr>
<tr>
<td>Ru-106</td>
<td>NA</td>
<td>2.5(-1)</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>Ag-110m</td>
<td>NA</td>
<td>5(-2)</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>Sb-124</td>
<td>6.1(-2)</td>
<td>NA</td>
<td>4.7(-2)</td>
<td>NA</td>
</tr>
<tr>
<td>I-131</td>
<td>4.3(-1)</td>
<td>6(-2)</td>
<td>1.7(-1)</td>
<td>1.6(-2)</td>
</tr>
<tr>
<td>Cs-134</td>
<td>1.7(-1)</td>
<td>1.4</td>
<td>1.4</td>
<td>1.0</td>
</tr>
<tr>
<td>Cs-136</td>
<td>NA</td>
<td>NA</td>
<td>4.7(-2)</td>
<td>NA</td>
</tr>
<tr>
<td>Cs-137</td>
<td>2.9(-1)</td>
<td>2.5</td>
<td>2.0</td>
<td>1.2</td>
</tr>
<tr>
<td>Ba-140</td>
<td>7.6(-2)</td>
<td>NA</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>Ce-141</td>
<td>3.3(-2)</td>
<td>1(-3)</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>Ce-144</td>
<td>7.3(-2)</td>
<td>5.3(-1)</td>
<td>NA</td>
<td>NA</td>
</tr>
</tbody>
</table>

**Total** | 7.4 | 6.6 | 11.3 | 2.7

---


^dNUREG/CR-0140, "In-Plant Source Term Measurements at Ft. Calhoun Station, Unit 1," August 1978.

^eNA, 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 elsewhere (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.

<table>
<thead>
<tr>
<th>DEMINERALIZER TYPE</th>
<th>ANION</th>
<th>Cs, Rb</th>
<th>OTHER NUCLIDES</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mixed Bed (H⁺-OH⁻)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Reactor Coolant</td>
<td>10</td>
<td>2</td>
<td>10</td>
</tr>
<tr>
<td>Condensate</td>
<td>10</td>
<td>2</td>
<td>10</td>
</tr>
<tr>
<td>Clean waste</td>
<td>10⁰(10)</td>
<td>10(10)</td>
<td>10²(10)</td>
</tr>
<tr>
<td>Dirty waste (floor drains)</td>
<td>10²(10)</td>
<td>2(10)</td>
<td>10²(10)</td>
</tr>
<tr>
<td>Cation Bed (H⁺)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Dirty waste</td>
<td>1(1)</td>
<td>10(10)</td>
<td>10²(10)</td>
</tr>
<tr>
<td>Powdex (any system)</td>
<td>10(10)</td>
<td>2(10)</td>
<td>10(10)</td>
</tr>
</tbody>
</table>

*For an evaporator polishing demineralizer or for the second demineralizer in series, the DF is given in the parentheses.
<table>
<thead>
<tr>
<th>REACTOR NAME</th>
<th>POWER STARTUP</th>
<th>NUCLEAR THERMAL OUTPUT 10^6 MWT</th>
<th>GASEOUS TRITIUM RELEASED (Ci/yr)</th>
<th>LIQUID TRITIUM RELEASED (Ci/yr-MWt at 80% capacity)</th>
<th>TOTAL TRITIUM RELEASED (Ci/yr-MWt at 80% capacity)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Oyster Creek</td>
<td>1930 1969</td>
<td>0.54 0.45 0.46 0.41 0.49 0.41</td>
<td>0.8 0.4 0.4 2.8 1.1 0.7</td>
<td>62 36.6 14.1 18 38 3.4</td>
<td>0.034 0.024 0.009 0.015 0.023 0.003</td>
</tr>
<tr>
<td>Nine Mile Point 1</td>
<td>1850 1969</td>
<td>0.42 0.46 0.44 0.40 0.55 0.38</td>
<td>18 26.8 ** 20 19 33</td>
<td>28 46.5 18.7 28 2.5 0.5</td>
<td>0.032 0.047 0.012 0.035 0.011 0.026</td>
</tr>
<tr>
<td>Dresden 2/3</td>
<td>2577 1970/71</td>
<td>1.04 1.18 0.91 0.70 1.15 1.20</td>
<td>31 10 11 220 170 330</td>
<td>26 26 22.6 54 20 0.1</td>
<td>0.016 0.009 0.011 0.11 0.048 0.081</td>
</tr>
<tr>
<td>Millstone 1</td>
<td>2011 1970</td>
<td>0.40 0.25 0.47 0.50 0.48 0.62</td>
<td>4.2 1.7 2.8 17 29 33</td>
<td>21 3.7 24.1 80 20 7.5</td>
<td>0.018 0.006 0.017 0.057 0.030 0.019</td>
</tr>
<tr>
<td>Monticello</td>
<td>1670 1970</td>
<td>0.46 0.41 0.37 0.37 0.51 0.46</td>
<td>12 ** ** 66 77 139</td>
<td>*** *** *** ** 0 0 0</td>
<td>0.008 - - 0.052 0.044 0.088</td>
</tr>
<tr>
<td>Vermont Yankee</td>
<td>1593 1972</td>
<td>0.06 0.18 0.34 0.47 0.42 0.46</td>
<td>+ 1.0 0.9 7.1 14 28</td>
<td>+ 0.2 *** 0 1.6 0.1</td>
<td>- 0.002 0.001 0.004 0.011 0.018</td>
</tr>
<tr>
<td>Quad Cities 1/2</td>
<td>2511 1971/72</td>
<td>0.52 1.32 1.09 0.96 1.08 1.12</td>
<td>4.7 34++ 29 280 300 40</td>
<td>4.7 24.5 34 54 24 19</td>
<td>0.005 0.013 0.017 0.10 0.088 0.016</td>
</tr>
<tr>
<td>Pilgrim 1</td>
<td>1998 1972</td>
<td>0.11 0.53 0.25 0.34 0.32 0.34</td>
<td>** 14 8 74 37 61</td>
<td>4.2 0.4 10.4 18 47 33</td>
<td>0.011 0.008 0.22 0.079 0.077 0.080</td>
</tr>
<tr>
<td>Peach Bottom 2/3</td>
<td>3293 1973/74</td>
<td>1.39 1.55 1.19</td>
<td>0.3 27 260</td>
<td>31 74 71</td>
<td>0.007 0.019 0.081</td>
</tr>
<tr>
<td>Browns Ferry 1-3</td>
<td>3293 73/74/76</td>
<td>0.36 0.37 2.24</td>
<td>5.1 0.6 23</td>
<td>10 4.0 22</td>
<td>0.012 0.004 0.006</td>
</tr>
<tr>
<td>Cooper</td>
<td>2381 1974</td>
<td>0.52 0.49 0.60</td>
<td>43 67 50</td>
<td>8.3 3.9 3.8</td>
<td>0.029 0.045 0.029</td>
</tr>
<tr>
<td>Hatch 1</td>
<td>2436 1974</td>
<td>0.41 0.57 0.51</td>
<td>1.8 1.4 1.2</td>
<td>6.1 9.0 14</td>
<td>0.006 0.005 0.009</td>
</tr>
<tr>
<td>Fitzpatrick</td>
<td>2436 1974</td>
<td>0.28 0.53 0.49</td>
<td>** 15 9.5</td>
<td>** 4.2 28</td>
<td>- 0.011 0.007</td>
</tr>
<tr>
<td>Duane Arnold</td>
<td>1558 1974</td>
<td>0.31 0.33 0.39</td>
<td>19 16 15</td>
<td>0.3 0.3 0.2</td>
<td>0.018 0.014 0.011</td>
</tr>
<tr>
<td>Brunswick 1/2</td>
<td>2436 1975/76</td>
<td>0.20 0.33 0.66</td>
<td>2.0 22 19</td>
<td>3.2 5.9 7.4</td>
<td>0.008 0.025 0.012</td>
</tr>
<tr>
<td>WEIGHTED AVERAGE+++</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0.021 0.016 0.015 0.043 0.035 0.032</td>
</tr>
</tbody>
</table>

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

*Average weighted by thermal nuclear output
**Insufficient Data
*Prior to first refueling
### TABLE 2-32

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

<table>
<thead>
<tr>
<th>PLANT</th>
<th>REACTOR BUILDING</th>
<th>TURBINE BUILDING</th>
<th>MISCELLANEOUS</th>
<th>TOTAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>Monticello</td>
<td>68</td>
<td>29</td>
<td>3</td>
<td>100</td>
</tr>
<tr>
<td>Vermont Yankee</td>
<td>35</td>
<td>53</td>
<td>12</td>
<td>100</td>
</tr>
<tr>
<td>Oyster Creek</td>
<td>13</td>
<td>79</td>
<td>8</td>
<td>100</td>
</tr>
<tr>
<td>AVERAGE</td>
<td>39</td>
<td>54</td>
<td>7</td>
<td>100</td>
</tr>
</tbody>
</table>
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

<table>
<thead>
<tr>
<th>Parameter</th>
<th>ALL NUCLIDES</th>
<th>ANIONS</th>
</tr>
</thead>
<tbody>
<tr>
<td>Miscellaneous radwaste evaporator</td>
<td>$10^4$</td>
<td>$10^3$</td>
</tr>
<tr>
<td>Separate evaporator for detergent wastes</td>
<td>$10^2$</td>
<td>$10^2$</td>
</tr>
</tbody>
</table>

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.

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

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

<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>FEED ACTIVITY ($\mu$Ci/cm²)</th>
<th>PRODUCT ACTIVITY ($\mu$Ci/cm²)</th>
<th>MEMBRANE DF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ce-144</td>
<td>$2.68 \times 10^{-4}$</td>
<td>$&lt;2.2 \times 10^{-7}$</td>
<td>1200</td>
</tr>
<tr>
<td>Co-58</td>
<td>$8.55 \times 10^{-5}$</td>
<td>$&lt;3.4 \times 10^{-8}$</td>
<td>2500</td>
</tr>
<tr>
<td>Ru-103</td>
<td>$5.83 \times 10^{-5}$</td>
<td>$&lt;5.5 \times 10^{-8}$</td>
<td>1100</td>
</tr>
<tr>
<td>Cs-137</td>
<td>$4.09 \times 10^{-4}$</td>
<td>$6.6 \times 10^{-6}$</td>
<td>60</td>
</tr>
<tr>
<td>Cs-134</td>
<td>$2.02 \times 10^{-4}$</td>
<td>$3.2 \times 10^{-8}$</td>
<td>60</td>
</tr>
<tr>
<td>Nb-95</td>
<td>$5.35 \times 10^{-5}$</td>
<td>$&lt;5.3 \times 10^{-8}$</td>
<td>1000</td>
</tr>
<tr>
<td>Zr-95</td>
<td>$2.36 \times 10^{-5}$</td>
<td>$&lt;3.7 \times 10^{-8}$</td>
<td>640</td>
</tr>
<tr>
<td>Mn-54</td>
<td>$8.82 \times 10^{-5}$</td>
<td>$&lt;3.4 \times 10^{-8}$</td>
<td>2600</td>
</tr>
<tr>
<td>Co-60</td>
<td>$9.62 \times 10^{-4}$</td>
<td>$&lt;8.1 \times 10^{-8}$</td>
<td>12,000</td>
</tr>
<tr>
<td>Total isotopic</td>
<td>$2.15 \times 10^{-3}$</td>
<td>$9.8 \times 10^{-6}$</td>
<td>220</td>
</tr>
<tr>
<td>Gross $\beta$</td>
<td>$1.63 \times 10^{-3}$</td>
<td>$1.86 \times 10^{-5}$</td>
<td>88</td>
</tr>
</tbody>
</table>

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

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

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

<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>FEED ACTIVITY (µCi/ml)</th>
<th>PRODUCT ACTIVITY (µCi/ml)</th>
<th>MEMBRANE DF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Co-60</td>
<td>2.5 x 10^-4</td>
<td>5 x 10^-7</td>
<td>500</td>
</tr>
<tr>
<td>Mo-99</td>
<td>3.8 x 10^-2</td>
<td>1 x 10^-3</td>
<td>40</td>
</tr>
<tr>
<td>I-131, 132, 133, 134, 135</td>
<td>1.2 x 10^-1</td>
<td>4 x 10^-3</td>
<td>30</td>
</tr>
<tr>
<td>Cs-134, 137</td>
<td>4.3 x 10^-2</td>
<td>2 x 10^-4</td>
<td>200</td>
</tr>
</tbody>
</table>
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:

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

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*

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

*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:

\[ Q = N_o \sigma_o \rho m t p s (\text{Ci/yr}) \]

where

- \( m \) is the 3.9 x 10^4 kg, mass of water in reactor core;
- \( N_o \) is the 1.3 x 10^{22} \text{ atoms O-17/kg natural water};
- \( p \) is the 0.80, plant capacity factor;
- \( s \) is the 1.03 x 10^{-22} \text{ Ci/atom, specific activity for C-14};
- \( t \) is the 3.15 x 10^7 sec/yr, maximum irradiation time per year;
- \( \sigma_o \) is the 2.4 x 10^{-25} \text{ cm}^2, thermal neutron cross section for 0-17; and
- \( \phi \) is the 3 x 10^{13} \text{ neutrons/cm}^2-\text{sec}, average thermal neutron flux.

Based on the above parameters, \( Q = 9.5 \text{ Ci/yr} \).

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 x 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 \text{ uCi/sec}. The dynamic adsorption coefficients for argon-41 in charcoal delay beds of a MCOTS are 6.4 \text{ cm}^2/gm for an ambient temperature charcoal system and 16 \text{ cm}^2/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 \text{ uCi/sec} to 37 \text{ uCi/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 \text{ uCi/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>
<thead>
<tr>
<th>PLANT</th>
<th>Argon-41 Release</th>
</tr>
</thead>
<tbody>
<tr>
<td>Browns Ferry 1</td>
<td>38</td>
</tr>
<tr>
<td>Browns Ferry 2</td>
<td>17</td>
</tr>
<tr>
<td>&quot;</td>
<td>12</td>
</tr>
<tr>
<td>Browns Ferry 3</td>
<td>7.1</td>
</tr>
<tr>
<td>&quot;</td>
<td>5.8</td>
</tr>
<tr>
<td>&quot;</td>
<td>7.4</td>
</tr>
<tr>
<td>&quot;</td>
<td>34</td>
</tr>
<tr>
<td>&quot;</td>
<td>32</td>
</tr>
<tr>
<td>&quot;</td>
<td>19</td>
</tr>
<tr>
<td>Hatch 1</td>
<td>12</td>
</tr>
<tr>
<td>&quot;</td>
<td>16</td>
</tr>
<tr>
<td>Fitzpatrick</td>
<td>36</td>
</tr>
</tbody>
</table>

*Data in this table are based on measured argon-41 release rates in reference 49 and were adjusted to 3400 MWe.*
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:
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.
<table>
<thead>
<tr>
<th>CARD</th>
<th>NAME</th>
<th>NAME OF REACTOR ( )</th>
<th>TYPE = BWR</th>
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<tr>
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<td>PWR</td>
<td>THERMAL POWER LEVEL (MEGA WATTS)</td>
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<tr>
<td></td>
<td>GTO</td>
<td>TOTAL STEAM FLOW (MILLION LBS/HR)</td>
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<tr>
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<td>WLIQ</td>
<td>MASS OF WATER IN REACTOR VESSEL (MILLION LBS)</td>
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<td>GDE</td>
<td>CLEAN-UP DEMINERALIZER FLOW (MILLION LBS/HR)</td>
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<td>REGENT</td>
<td>CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS)</td>
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<td></td>
<td>FF/COM</td>
<td>FRACTION FEEDWATER THROUGH CONDENSATE DEMIN</td>
<td>( )</td>
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<td></td>
<td>HIGH PURITY WASTE INPUT ( )</td>
<td>GPD AT ( )</td>
<td>PCA</td>
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<tr>
<td></td>
<td>DF/C</td>
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<td>DF/C ( )</td>
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<td>COLLECTION( )</td>
<td>DAYS PROCESS ( )</td>
<td>DAYS FRACT DISCH ( )</td>
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<td>PCA</td>
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<td>COLLECTION( )</td>
<td>DF/C ( )</td>
<td>DF/C ( )</td>
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<tr>
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<td>CHEMICAL WASTE INPUT ( )</td>
<td>GPD AT ( )</td>
<td>PCA</td>
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<td>COLLECTION( )</td>
<td>DF/C ( )</td>
<td>DF/C ( )</td>
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<td>REGENERANT SOLN'S WASTE INPUT - GPD</td>
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<td>COLLECTION( )</td>
<td>DF/C ( )</td>
<td>DF/C ( )</td>
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<tr>
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<td>GGS</td>
<td>GLAND SEAL STEAM FLOW (THOUSAND LBS/HR)</td>
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<td>TIM</td>
<td>GLAND SEAL HOLD UP TIME (HOURS)</td>
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<td></td>
<td>AIR</td>
<td>EJECTOR OFF/GAS HOLD UP TIME (HOURS)</td>
<td>( )</td>
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<td>CARD 21</td>
<td>CONTAINMENT BLDG CHARCOAL? ( ) HEPA? ( )</td>
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<tr>
<td>CARD 22</td>
<td>TURBINE BLDG CHARCOAL? ( ) HEPA? ( )</td>
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<tr>
<td>CARD 23</td>
<td>GLAND SEAL VENT IODINE PF ( )</td>
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<tr>
<td>CARD 24</td>
<td>AUXILIARY BLDG CHARCOAL? ( ) HEPA? ( )</td>
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<tr>
<td>CARD 25</td>
<td>RADIOACTIVE BLDG CHARCOAL? ( ) HEPA? ( )</td>
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<tr>
<td>CARD 26</td>
<td>CHARCOAL DECAY SYSTEM Q=NO, 1=YES, 2=CRYOGENIC DISTILL ( )</td>
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<tr>
<td>CARD 27</td>
<td>Krypton Dynamic Adsorption Coefficient (cm^3/gm) ( )</td>
<td></td>
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<tr>
<td>CARD 28</td>
<td>Xenon Dynamic Adsorption Coefficient (cm^3/gm) ( )</td>
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<tr>
<td>CARD 29</td>
<td>Mass Dynamic Adsorption Coefficient (thousand lbs) ( )</td>
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<tr>
<td>CARD 30</td>
<td>Detergent Waste Decon Factor ( )</td>
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</table>
NAME OF REACTOR SAMPLE BWR CASE 1
THERMAL POWER LEVEL (MEGAWATTS) 3400.
TOTAL STEAM FLOW (MILLION LBS/HR) 15.
MASS OF WATER IN REACTOR VESSEL (MILLION LBS) 0.38
CLEAN-UP DEMINERALIZER FLOW (MILLION LBS/HR) 0.13
CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS) 56.
FRACTION FEED WATER THROUGH CONDENSATE DEMIN 1.00
HIGH PURITY WASTE INPUT 28640. GPD AT .15 PCA
INPUT 28640, 1.0E02DF0 = 1.0003
INPUT 5700. 1.0E04DF0 = 1.0004
INPUT 600. 1.0E04DF0 = 1.0005
COLLECTION 1.0 DAYS PROCESS 0.07 DAYS FRACT DISCH 0.01
COLLECTION 3.10 DAYS PROCESS 0.6 DAYS FRACT DISCH 1.0
CHEMICAL WASTE INPUT 600. GPD AT .02 PCA
INPUT 600. 1.0E04DF0 = 1.0004
COLLECTION 3.1 DAYS PROCESS 0.60 DAYS FRACT DISCH 1.0
REGENERATION SOLUTNS INPUT GPD 1700.
INPUT 1.0E04DF0 = 1.0005
INPUT 9.4 DAYS PROCESS 0.44 DAYS FRACT DISCH 1.0
GLAND SEAL STEAM FLOW (THOUSAND LBS/HR) 0.0
GLAND SEAL HOLDUP TIME (HOURS) 0.0
AIR EJECTOR OFFGAS HOLDUP TIME (HOURS) .167
CONTAINMENT BLDG. CHARCOAL 90.0 HEPA99.0
TURBINE BLDG. CHARCOAL 00.0 HEPA99.0
CHARCOAL DELAY SYSTEM 0=NO, 1=YES, 2=CRYOGENIC DISTILL
KRYPTON DYNAMIC ADSORPTION COEFFICIENT (CM3/GM) 105.0
XENON DYNAMIC ADSORPTION COEFFICIENT (CM3/GM) 2410.0
MASS OF CHARCOAL (THOUSAND LBS) 48.
DETERGENT WASTE DECONTAMINATION FACTOR 1.0
FLOW CHART FOR BWR LIQUID CODE

START

1

READ CARD 1 REACTOR NAME TYPE

READ CARD 2 POWER

READ CARD 3 TOTAL STEAM FLOW

READ CARD 4 MASS OF WATER IN REACTOR VESSEL

READ CARD 5 CLEAN-UP DEM. FLOW

PRINT TOTAL STEAM FLOW

PRINT MASS OF WATER REACTOR VESSEL

READ CARD 6 CONDENSATE DEM. REGENERATION TIME

PRINT CONDENSATE DEM. REGENERATION TIME

READ CARD 7 FRACTION FEED WATER THRU CONDENSATE DEM.

PRINT FRACTION FEED WATER THRU CONDENSATE DEM.

READ CARD 8 HIGH PURITY WASTE INPUT AND ACTIVITY

READ CARD 9 DF IODINE DF CS-RB DF ALL OTHERS

READ CARD 10 COLLECTION TIME PROCESS TIME FRACTION DISCH.

PRINT INPUT FROM CARDS 14, 15, AND 16

READ CARD 11 LOW PURITY WASTE INPUT AND ACTIVITY

READ CARD 12 DF IODINE DF CS-RB DF ALL OTHERS

READ CARD 13 COLLECTION TIME PROCESS TIME FRACTION DISCH.

PRINT INPUT FROM CARDS 11, 12, AND 13

READ CARD 14 CHEMICAL WASTE INPUT AND ACTIVITY

READ CARD 15 DF IODINE DF CS-RB DF ALL OTHERS

READ CARD 16 COLLECTION TIME PROCESS TIME FRACTION DISCH.

PRINT INPUT FROM CARDS 14, 15, AND 16

READ CARD 17 REGENERANT WASTE INPUT

READ CARD 18 DF IODINE DF CS-RB DF ALL OTHERS

READ CARD 19 COLLECTION TIME PROCESS TIME FRACTION DISCH.

READ CARD 18 CHEMICAL WASTE INPUT AND ACTIVITY

READ CARD 19 REGENERANT WASTE INPUT

READ CARD 19 COLLECTION TIME PROCESS TIME FRACTION DISCH.

3-6
FLOWCHART FOR BWR GAS CODE

START

DATA FROM TABLES
2-10, 2-26
2-11, 2-27
2-12, 2-14

DECAY CONSTANTS AND REACTOR COOLANT CONC.

READ PLANT DATA

PRINT PLANT DATA

READ LIQUID WASTE INPUT PARAMETERS

PRINT LIQUID WASTE INPUT PARAMETERS

READ GASEOUS WASTE INPUT PARAMETERS

PRINT GASEOUS WASTE INPUT PARAMETERS

HEPA1 = 1.0
FIL1 = 1.0
HEPA2 = 1.0
FIL2 = 1.0

HEPA5 = 1.0
FIL5 = 1.0
HEPA6 = 1.0
FIL6 = 1.0

READ CARD 23
WURD, CBCH CBHEPA

1

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

READ CARD 25 GLAND SEAL IODINE PF

PRINT GLAND SEAL IODINE PF

READ CARD 26 AIR EJECTOR OFF GAS IODINE PF

---

DETERMINE CONTAINMENT BLDG. HEPA DF

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

---

DETERMINE THE EXISTANCE OF A CHARCOAL DELAY SYS.

READ 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.

---

3-9
UNIT CONVERSION: GTO, GGS, GDE, WLJQ, WSTE

EVALUATE BWR DESIGN PARAMETERS TABLE 2-3 PARAMETER 2-2.3.1

EVALUATE IODINE RELEASES BLDG. VENT. SYS. TABLE 2-10 THRU 2-12 PARAMETER 2.2.4.1

EVALUATE IODINE RELEASES CONDENSER OFF GAS SYS. PARAMETER 2.2.5.1

EVALUATE IODINE RELEASES GLAND SEALING SYS. EXHAUST PARAMETER 2.2.6

EVALUATE IODINE RELEASES MECH. VACUUM PUMP PARAMETER 2.2.7

PRINT CALCULATED IODINE RELEASES

EVALUATE TRITIUM RELEASES GASEOUS PATH PARAMETER 2.2.15

PRINT CALCULATED TRITIUM RELEASES

EVALUATE CARBON-14 RELEASES PARAMETER 2.2.22

PRINT CARBON-14 RELEASES

EVALUATION OF NOBLE GASES RELEASED SEE NOTE 4

PRINT CALCULATED NOBLE GAS RELEASES

EVALUATION PARTICULATES RELEASED SEE NOTE 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
### Liquid Waste Inputs

<table>
<thead>
<tr>
<th>Stream</th>
<th>Flow Rate (GAL/DAY)</th>
<th>Fraction</th>
<th>Fraction Discharged (GAL/DAY)</th>
<th>Collection Time (DAYS)</th>
<th>Decay Time (DAYS)</th>
<th>Decontamination Factors</th>
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<tr>
<td>High Purity Waste</td>
<td>2.86E+04</td>
<td>0.150</td>
<td>0.010</td>
<td>1.000</td>
<td>0.070</td>
<td>1.00E+03 1.00E+02 1.00E+03</td>
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<tr>
<td>Low Purity Waste</td>
<td>5.70E+03</td>
<td>0.150</td>
<td>1.000</td>
<td>3.100</td>
<td>0.600</td>
<td>1.00E+03 1.00E+04 1.00E+04</td>
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<tr>
<td>Chemical Waste</td>
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<td>Regen Sol</td>
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<td>9.400</td>
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<td>1.00E+04 1.00E+05 1.00E+05</td>
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### Gaseous Waste Inputs

- **Glue Seal Steam Flow (THOUSAND LBS/HR)**: 0.0
- **Glue 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 Fraction**: 1.00000
- **Particulate Release Fraction**: 1.00000
- **Glue 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
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<tr>
<th>NUCLIDE</th>
<th>Half-Life (DAYS)</th>
<th>Concentration (MICRO CI/ML)</th>
<th>Annual Releases to Discharge Canal (CI/YR)</th>
<th>Adjusted Total Wastes (CI/YR)</th>
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<tr>
<td></td>
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<td>HIGH PURITY</td>
<td>LOW PURITY</td>
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<tr>
<td></td>
<td></td>
<td>(CURES)</td>
<td>(CURES)</td>
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<td>COOLANT</td>
<td>CHEMICAL</td>
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<td>(CURES)</td>
<td>(CURES)</td>
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<td>TOTAL</td>
<td>TOTAL LWS</td>
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<td>NA 24</td>
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<td>Co 58</td>
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</table>

**Fission Products**

|              |                  |                             |                              |                              |
|              |                  | COOLANT                      | CHEMICAL                                   |                              |
|              |                  | (CURES)                      | (CURES)                                    |                              |
|              |                  | TOTAL                       | TOTAL LWS                                  |                              |
|              |                  |                             |                                             |                              |
| Br 83       | 1.00E+01         | 3.45E-03                    | 0.00002                                   | 0.00007                      |
| Sr 89       | 5.20E+01         | 9.07E-05                    | 0.00001                                   | 0.00020                      |
| Sr 90       | 1.03E+04         | 6.35E-06                    | 0.00000                                   | 0.00002                      |
| Y 90        | 2.67E+00         | 0.0                         | 0.0                                       | 0.00002                      |
| Sr 91       | 4.03E-01         | 3.69E-03                    | 0.00009                                   | 0.00012                      |
| Y 91 M      | 3.47E-02         | 0.0                         | 0.0                                       | 0.00007                      |
| Y 91        | 5.88E+01         | 3.63E-05                    | 0.00000                                   | 0.00012                      |
| Sr 92       | 1.13E-01         | 9.47E-03                    | 0.00006                                   | 0.00012                      |
| Y 92        | 1.67E-01         | 5.64E-03                    | 0.00012                                   | 0.00012                      |
| Y 93        | 4.25E-01         | 3.69E-03                    | 0.00010                                   | 0.00012                      |
| Zr 95       | 6.50E+01         | 7.25E-06                    | 0.00000                                   | 0.00000                      |
| Nb 95       | 3.50E+01         | 7.26E-06                    | 0.00000                                   | 0.00000                      |
| Nb 98       | 5.74E+01         | 3.89E-03                    | 0.00015                                   | 0.00015                      |
| Mo 99       | 2.79E+00         | 1.82E-03                    | 0.00009                                   | 0.00022                      |
| Tc 99 M     | 2.50E-01         | 1.86E-02                    | 0.00037                                   | 0.00053                      |
| Ru 103      | 3.96E+01         | 1.81E-05                    | 0.00000                                   | 0.00053                      |
| Rh 103 M    | 3.96E-02         | 0.0                         | 0.0                                       | 0.00004                      |
| Ru 105      | 1.55E-01         | 1.87E-03                    | 0.00000                                   | 0.00001                      |
| Rh 105 M    | 5.21E-04         | 0.0                         | 0.0                                       | 0.00002                      |
| Rh 105      | 1.50E+00         | 0.0                         | 0.0                                       | 0.00007                      |
| Ru 106      | 3.67E+02         | 2.72E-06                    | 0.00000                                   | 0.00000                      |
| Ag 110      | 2.35E+02         | 9.07E-07                    | 0.00000                                   | 0.00000                      |
| Te 129      | 3.40E+01         | 3.63E-05                    | 0.00001                                   | 0.00001                      |
| Te 129      | 4.79E-02         | 0.0                         | 0.0                                       | 0.00001                      |
| Te 131 M    | 1.25E+00         | 9.13E-05                    | 0.00000                                   | 0.00003                      |
| I 131       | 8.05E+00         | 1.60E-03                    | 0.00009                                   | 0.00009                      |
| I 132       | 9.85E-02         | 3.48E-02                    | 0.00017                                   | 0.00017                      |
| I 133       | 8.75E-01         | 2.26E-02                    | 0.00088                                   | 0.00088                      |
| I 134       | 3.67E-02         | 7.05E-02                    | 0.00006                                   | 0.00006                      |
| Cs 134      | 7.49E+02         | 2.72E-05                    | 0.00002                                   | 0.00004                      |
| Cs 135      | 2.79E-01         | 2.46E-02                    | 0.00045                                   | 0.00045                      |
| Cs 136      | 1.30E+01         | 7.26E-03                    | 0.00004                                   | 0.00004                      |
## Liquid Effluents (Continued)

### SAMPLE BWR CASE 1

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<thead>
<tr>
<th>Nuclide</th>
<th>Half-Life (DAYS)</th>
<th>Coolant (MICRO CI/ML)</th>
<th>High Purity (CURIES)</th>
<th>Low Purity (CURIES)</th>
<th>Chemical (CURIES)</th>
<th>Total LWS (CURIES)</th>
<th>Adjusted Total (CURIES)</th>
<th>Detergent Wastes (CURIES)</th>
<th>Total Wastes (CURIES)</th>
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<td>CS137</td>
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### Tritium Release

<table>
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<th>Tritium Release</th>
<th>26 CURIES PER YEAR</th>
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### SAMPLE BWR CASE 1

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<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
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<td>THERMAL POWER LEVEL (MEGAWATTS)</td>
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<tr>
<td>PLANT CAPACITY FACTOR</td>
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<tr>
<td>TOTAL STEAM FLOW (MILLION LBS/HR)</td>
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<tr>
<td>MASS OF WATER IN REACTOR VESSEL (MILLION LBS)</td>
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<tr>
<td>CLEAN-UP DEMINERALIZER FLOW (MILLION LBS/HR)</td>
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<tr>
<td>CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS)</td>
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<td>FRACTION FEED WATER THROUGH CONDENSATE DEMIN</td>
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<tr>
<td>REACTOR VESSEL HALOGEN CARRYOVER FACTOR</td>
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#### LIQUID WASTE INPUTS

<table>
<thead>
<tr>
<th>Stream</th>
<th>Flow Rate (GAL/DAY)</th>
<th>Fraction of PCA Discharged</th>
<th>Collection Time (DAYS)</th>
<th>Decay Time (DAYS)</th>
<th>Decontamination Factors</th>
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<tr>
<td>High Purity Waste</td>
<td>2.86E+04</td>
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<td>Low Purity Waste</td>
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<td>1.00E+03 1.00E+04 1.00E+04</td>
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<tr>
<td>Chemical Waste</td>
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<td>1.00E+03 1.00E+04 1.00E+04</td>
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<td>Regenerant SolS</td>
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<td>9.400</td>
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#### GASEOUS WASTE INPUTS

<p>| | | | | | |</p>
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<tbody>
<tr>
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<td>(THOUSAND LBS/HR)</td>
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<tr>
<td>Gland Seal Holdup</td>
<td>(HOURS)</td>
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<tr>
<td>Air Ejector Offgas</td>
<td>(HOURS)</td>
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<tr>
<td>Containment Bldg. Iodine Release Fraction</td>
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<tr>
<td>Particulate Release Fraction</td>
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<tr>
<td>Turbine Bldg. Iodine Release Fraction</td>
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</tr>
<tr>
<td>Particulate Release Fraction</td>
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<tr>
<td>Gland Seal Vent. Iodine PF</td>
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<td>Auxiliary Bldg. Iodine Release Fraction</td>
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<tr>
<td>Particulate Release Fraction</td>
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<td>Radwaste Bldg. Iodine Release Fraction</td>
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</table>

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

#### GASEOUS RELEASE RATE
(CURIES PER YEAR)

<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>COOLANT CONC. (MICROCURIES/G)</th>
<th>CONTAINMENT BLDG.</th>
<th>TURBINE BLDG.</th>
<th>AUXILIARY BLDG.</th>
<th>RADWASTE BLDG.</th>
<th>GLAND SEAL</th>
<th>AIR EJECTOR</th>
<th>MECH VAC PUMP</th>
<th>TOTAL</th>
</tr>
</thead>
<tbody>
<tr>
<td>I-131</td>
<td>1.785E-03</td>
<td>1.1E-03</td>
<td>1.1E-01</td>
<td>2.1E-02</td>
<td>1.1E-02</td>
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<td>0.0</td>
<td>8.2E-02</td>
<td>2.3E-01</td>
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<tr>
<td>I-133</td>
<td>2.499E-02</td>
<td>1.5E-02</td>
<td>1.6E+00</td>
<td>2.9E-01</td>
<td>1.5E-01</td>
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<td>0.0</td>
<td>9.2E-01</td>
<td>3.0E+00</td>
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**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

<table>
<thead>
<tr>
<th>NUCLIDE</th>
<th>COOLANT CONC. (MICROCURIES/G)</th>
<th>CONTAINMENT BLDG.</th>
<th>TURBINE BLDG.</th>
<th>AUXILIARY BLDG.</th>
<th>RADWASTE BLDG.</th>
<th>GLAND SEAL</th>
<th>AIR EJECTOR</th>
<th>MECH VAC PUMP</th>
<th>TOTAL</th>
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<td>AR-41</td>
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<tr>
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TOTAL NOBLE GASES: 7.4E+03

0.0 appearing in the table indicates release is less than 1.0 Ci/yr for noble gas.
### Airborne Particulate Release Rate
(Curies per Year)

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</tbody>
</table>
GALE CODE FOR CALCULATING LIQUID EFFLUENTS FROM LWRS. MODIFIED OCT. 1978 TO IMPLEMENT APPENDIX I TO 10 CFR PART 50. REACTOR WATER CONCENTRATIONS CALCULATED USING METHODS OF DRAFT STANDARD ANSI 237 ' RADIOACTIVE MATERIALS IN PRINCIPAL FLUID STREAMS OF LIGHT WATER COOLED NUCLEAR POWER PLANTS’ DRFAST DATED MAY 20, 1974 MODIFIED EDITION OF ORIGEN PROGRAM TO COMPUTE EFFLUENTS FROM BWR AND PWR RADIATE SYSTEMS

LOGICAL DISCHG,POWRIT INTEGER*2 LOC,NON0,KD REAL*4 LETDWN.NOGEN REALX4 LETDWA REAL KKR,KXE,KNO,KMASS INTEGER*2 NAME(3) COMMON/MATRIX/A(2500),LOC(2500),NON0(800),KD(800) COMMON/FLEX/FLUX(10),MMN,MOUT,INDEX,QXN,AXN,ERR,NOBLND,MZERO COMMON/PROCSS/MPR0S,PRATE(8),N0PR0S(8),NZPROS(8,20),PR(800) COMMON/EQ/XZERO(800),XZH(800),XTEMP(800),XNEM(10,800) COMMON/CONC/PCONC(800),DWCONC(800),CWCONC(800),CMCONC(800),SCON(800),RINV(800) COMMON/COOL/REACTR,PW1,TYPE,PCVOL,LETDWN,NDGAS,SCVOL,STMVOL,N0GEN COMMON/NDGAS/REACTR,NDGAS(7),NZMBDM(26),NZCBDM(26) COMMON/FIX/TFIX(WITH THEIR CONCENTRATIONS IN UCI/GM) ARE ALSO PRESENT IN THE PRIMARY COOLANT BUT ARE NOT CONSIDERED SIGNIFICANT IN EFFLUENT CALCULATIONS:N-13,.05;N-16,60;N-17,.009;O-19,.7;F-18,.004.

READ NUCLEAR DATA AND CONSTRUCT TRANSITION MATRIX

CALL NUDATA(NLIBE)

DO 20 I=2,IT0T
     NON0(I)=NON0(I)+NON0(I-1)
     KD(I)=KD(I)+NON0(I-1)
     DISCHG=.FALSE.
     POWRIT=.FALSE.

DATA YES/'YES'/
61. KT=0 0000300
62. INDEX=0 0000305
63. FLXB=0.0 0000310
64. PWRH=0.0 0000315
65. BURN=0.0 0000320
66. BDTFR=1.0 0000325
67. QXN=0.001 0000330
68. AXN=-ALOG(QXN) 0000335
69. NE=ITOT 0000340
70. TCONST=86400. 0000345
71. MMN=0 0000350
72. MZERO=21 0000355
73. EDA=0.0 0000360
74. TMSC=0.0 0000365
75. TE=0.0 0000370
76. TS=0.0 0000375
77. T2=0.0 0000380
78. TSTOR2=0.0 0000385
79. DWFLZ=0.0 0000390
80. DW2=0.0 0000395
81. DWDF2=0.0 0000400
82. DO 40 J=1,800 0000405
83. PCONC(J)=0.0 0000410
84. DWCONC(J)=0.0 0000415
85. SCON(J)=0.0 0000420
86. RINV(J)=0.0 0000425
87. CWCONC(J)=0.0 0000430
88. CMCONC(J)=0.0 0000435
89. XZH(J)=0.0 0000440
90. 40 CONTINUE 0000445
91. C 0000450
92. C READ DESCRIPTION OF REACTOR AND RADWASTE TREATMENT PLANT 0000455
93. C 0000460
94. PRINT 9026 0000465
95. READ 9010,REACTR,TYPEN 0000470
96. PRINT 9010,REACTR,TYPE 0000475
97. READ 9022,WORD56,POW1 0000480
98. PRINT 9022,WORD56,POW1 0000485
99. PF=0.80 0000490
100. PRINT 9027 0000495
101. C 0000500
102. C READ DATA FOR BWR LIQUID CODE 0000505
103. C 0000510
104. 50 READ 9022,WORD56,STMFR 0000515
105. PRINT 9022,WORD56,STMFR 0000520
106. READ 9022,WORD56,PCVOL 0000525
107. PRINT 9022,WORD56,PCVOL 0000530
108. FPEF=0.001 0000535
109. HEF=0.020 0000540
110. PRINT 9030,FPEF,HEF 0000545
111. READ 9022,WORD56,LETDWN 0000550
112. PRINT 9022,WORD56,LETDWN 0000555
113. PC = 0.015 0000560
114. READ 9022,WORD56,REGENT 0000565
115. IF(REGENT.EQ.0.0) PC = 0.004 0000570
116. READ 9022,WORD56,REGENT 0000575
117. READ 9022,WORD56,FFCDM 0000580
118. PRINT 9022,WORD56,FFCDM 0000585
119. PRINT 9045 0000590
120. READ 9013,WORD18,CWLR,WORD8,CWA 0000595
READ 9014, DFICW, DFCSCW, DFCW
READ 9015, TC, TS TORDC, CMFD
PRINT 9016
PRINT 9017, WORD18, CWFLR, CWA, CWFD, TC, TS TORDC, DFICW, DFCSCW, DFCW
READ 9013, WORD18, DWFLR, WORD8, DWA
READ 9014, DFIDW, DFCSDW, DFDW
READ 9015, TD, TS TORDD, DFIDW, DFCSDW, DFDW
PRINT 9017, WORD18, CWFLR, CWA, CFWD, TD, TS TORDD, DFICW, DFCSCW, DFCW
READ 9013, WORD18, CMWFR, WORD8, CMA
READ 9014, DFICM, DFCSCM, DF CM
READ 9015, TCM, TS TORDM, CMFD
PRINT 9017, WORD18, CMWFR, CMA, CMFD, TCM, TSTORD, DFICM, DFCSCM, DF CM
READ 9037, RGWFR
READ 9014, DFIRG, DFCSRG, DFRG
READ 9015, TRG, TS TORKR, RGFD
PRINT 9038, RGWFR, RGFD, TRG, TS TORKR, DFIRG, DFCSRG, DFRG
READ DATA FOR BMR GAS CODE
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.)
READ 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
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 9020, WORD56, KMASS  
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  
98 CONTINUE  
PRINT 9048  
99 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 CALCULATE BWR PRIMARY COOLANT CONCENTRATIONS  
DO 2251 J=1, ITOT  
PCONC(J) = BCONC(J)  
P0WA = 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 - 3400.0) .GT. 400.0) 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(ABS(FFCDA - 0.99) GO TO 252  
GO TO 256  
C CALCULATE BWR ADJUSTMENT FACTORS  
252 DO 255 J=1, ITOT  
RHAL2 = (LETDWA * 0.9 + FFCDA * STMFA * PC * 0.9) / PCVOA  
IF(FFCDA .GT. 0.99) GO TO 299  
FFCDA = 0.01  
299 CONTINUE  
RCSR2 = (LETDWA * 0.5 + FFCDA * STMFA * 5E-4) / PCVOA  
RCFP2 = (LETDWA * 0.9 + FFCDA * STMFA * 9E-4) / PCVOA  
RKZ = 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. 53.0 OR NZ .EQ. 35) GO TO 253
241. IF (NZ.EQ.37.0R.NZ.EQ.55)GO TO 254
242. PCONC(J)=PCONC(J)*RK2*(0.3114+DL)/(RCFP2+DL)
243. GO TO 255
244. 253 PCONC(J)=PCONC(J)*RK2*(0.3612+DL)/(RHAL2+DL)
245. GO TO 255
246. 254 PCONC(J)=PCONC(J)*RK2*(0.1730+DL)/(RCSRB2+DL)
247. 255 CONTINUE
248. 256 PCVOL=PCVOL*1000000./62.4
249. LETDUN=LETDWN*200.
250. STMFR=STMFR*2351.
251. DO 2255 J=1,IT0T
252. IF (PCONC(J).GT.O.O)PCONC(J)=PCONC(J)/(DIS(J)*1.6283E13)
253. 2255 CONTINUE
254. IF(REGENT.GT.O.O) GO TO 257
255. CALL EFFTAB
256. GO TO 30
257. C COMPUTE REMOVAL CONSTANT FOR CONDENSATE DEMINERALIZER IN BWR
258. 257 FFCDM = FFCD
259. IF(FFCDM.EQ.O.0) GO TO 300
260. GO TO 301
261. 300 FFCDM = 0.01
262. 301 CONTINUE
263. CCBDM=0.9*STMFR*FPEF/(PCVOL*7.48*60.)*FFCDM
264. CSBDM=0.5*STMFR*FPEF/(PCVOL*7.48*60.)*FFCDM
265. IF(FFCDM.EQ.O.0) GO TO 304
266. CCBDM = CCBDM
267. CSBDI = CSBDM
268. GO TO 305
269. 304 CONTINUE
270. FFCDM = 0.1
271. 305 CONTINUE
272. CCBDI = 0.9*STMFR*FPEF/(PCVOL*7.48*60.)*FFCDM
273. CCBDI = CCBDM
274. GO TO 305
275. 300 CONTINUE
276. FFCDM = 0.1
277. 301 CONTINUE
278. CCBDI = 0.9*STMFR*FPEF/(PCVOL*7.48*60.)*FFCDM
279. CCBDI = CCBDM
280. GO TO 305
281. 258 CONTINUE
282. XZEROC(I)=0.
283. C CALCULATE INVENTORIES ON BWR CONDENSATE RESINS
284. 290 T(I)=REGENT
285. CALL SOLVE
286. DO 295 I=1,IT0T
287. 295 RINV(I)=XTMP(I)
288. CALL EFFTAB
289. GO TO 30
290. C FORMATS
291. 9000 FORMAT(20A4)
292. 9001 FORMAT(10E8.2)
293. 9002 FORMAT(6E10.3)
294. 9003 FORMAT(5(E9.2),I5)
295. 9004 FORMAT(16I5)
296. 9005 FORMAT('OMMH OR MOUT EXCEEDS DIMENSIONS')
297. 9006 FORMAT('MOUT SHOULD NOT EXCEED MMN BY MORE THAN 10')
298. 9007 FORMAT(8(E8.2,I2))
421. EDCONC(J)=0.0
422. DWCNCONC(J)=0.0
423. DWCN2(J)=0.0
424. CMCONC(J)=0.0
425. NZ=NUCL(J)/10000
426. IF(NZ.EQ.36.0R.NZ.EQ.54) GO TO 30
427. CUC0NC(J)=PC0NC(J)XCUA
428. EDCONC(J)=PC0HC(J)XEDA
429. DUC0NC(J)=PC0NC(J)XDUA
430. DUC0N2(J)=PC0NC(J)XDU2
431. CMCONC(J)=PCONC(J)XCMA
432. 30 CONTINUE
433. C
434. C CALCULATE RADIOACTIVITY AFTER COLLECTION AT A CONSTANT RATE
435. C
436. CALL COLECKTCX85400.,CUCONC,ILITE,ITOT)
437. CALL COLECT(TEX85400.,EDCONC,ILITE,ITOT)
438. CALL COLECT(TDX85400.,DUCONC,ILITE,ITOT)
439. CALL COLECT(T2X86400.,DUC0N2,ILITE,ITOT)
440. CALL COLECT(TCMX36400.,CMCONC,ILITE,ITOT)
441. 40 IF(REGENT.LE.0.0) GO TO 50
442. CALL STORAG(TRGXS6400.,RINV,ILITE,ITOT)
443. 50 DO 100 1=1,ITOT
444. NZ=NUCL(I)/10000
445. TURBDR(I)=1991.X5.XSC0N(I)
446. IF(NZ.EQ.I) GO TO 100
447. IF(NZ.EQ.35.0R.NZ.EQ.53) GO TO 60
448. IF(NZ.EQ.37.0R.NZ.EQ.55) GO TO 70
449. C
450. C CHEMICAL TREATMENT FOR OTHER CATIONS
451. C
452. CUC0NC(I)=CUCONC(I)/DFCU
453. EDCONC(I)=EDCONC(I)/DFED
454. DUCONC(I)=DUCONC(I)/DFDW
455. DUC0N2(I)=DUC0N2(I)/DFD2
456. CMCONC(I)=CMCONC(I)X(1.0-BDTFRX(1.0-CMFD/DFCM))
457. C TO TREAT PUR TURBINE BUILDING FLOOR DRAINS THROUGH DIRTY WASTE
458. C SYSTEM, DELETE C FOR COMMENT ON CARDS BELOW, UNTIL NEXT MESSAGE
459. RINV(I)=RINV(I)/DFRG
460. TURBDR(I)=1991.X5.XSC0N(I)XHEF
461. C TURBDR(I)=1991.X5.XSC0N(I)XHEF/DFDW
462. GO TO 100
463. C
464. C CHEMICAL TREATMENT FOR RB AND CS
465. C
466. CUCONC(I)=CUCONC(I)/DFCSW
467. EDCONC(I)=EDCONC(I)/DFCSW
468. DUCONC(I)=DUCONC(I)/DFCSW
469. 60 DO 100 1=1,ITOT
481. DCON2(I)=DCON2(I)/DFCSD2 0002400
482. CMCNC(I)=CMCNC(I)*(1.0-BDTRF*(1.0-CMFD/DFSCM)) 0002405
483. RIN(V(I))=RIN(V(I))/DFCSR 0002410
484. TURBDR(I)=1991.5*SCON(I)*FPEF 0002420
485. TURBDR(I)=1991.5*SCON(I)*FPEF/DFCSDW 0002425
486. CONTINUE 0002430
487. COMPUTE RADIOACTIVE DECAY DURING PROCESSING AND SAMPL 0002435
488.ING 0002440
489. CALL STORAG(TSTORC*86400.,CMCONC,ILITE,ITOT) 0002445
490. CALL STORAG(TS*86400.,EDCONC,ILITE,ITOT) 0002450
491. CALL STORAG(TSTORB*86400.,DCONC,ILITE,ITOT) 0002455
492. CALL STORAG(TSTORT*86400.,CMCONC,ILITE,ITOT) 0002460
493. CALL STORAG(TSTORR*86400.,RINV,ILITE,ITOT) 0002465
494. CALL STORAG(TSTORR*86400.,RINV,ILITE,ITOT) 0002470
495. CALL STORAG(TSTORR*86400.,RINV,ILITE,ITOT) 0002475
496. CALL STORAG(TS*6000.,TURBDR,ILITE,ITOT) 0002480
497. DO 130 I=1,ITOT 0002485
498. NZ=NUCL(I)/10000 0002490
499. ABLow=0.0 0002495
500. IF(REGENT.LT.0.001) GO TO 110 0002500
501. ABLow=RINV(I)*365.5*RGFD/REGENT 0002505
502. CONTINUE 0002510
503. 120 ABLow=ABLlow+CMWF+1.382*CMCONC(I) 0002515
504. 130 CMCONC(I)=ABLlow 0002520
505. 133 CWF=CMWF*1.382*CWF 0002525
506. DWF=DFWF*1.382 0002530
507. TLR=CMWF+DWF+1.382*(CMFD*CMWF+RGFD*RGWF) 0002535
508. TRITR=TRITR*TLR 0002540
509. IF(TRITR.GT.0.5*TRITPR)TRITR=0.5*TRITPR 0002545
510. RTRITR=RTRITR+0.5 0002550
511. RTRITR=RTRITR 0002555
512. TOTAL=0.0 0002560
513. DO 140 I=1,ITOT 0002565
514. NZ=NUCL(I)/10000 0002570
515. IF(NZ.EQ.36.0R.NZ.EQ.54) GO TO 140 0002575
516. DISI=DISI*1.6283E13 0002580
517. CMCONC(I)=DISI*(CMCONC(I)+CMWF+EDCONC(I)+EDFLR) 0002585
518. DCONC(I)=DCONC(I)+CMCONC(I)+CMCONC(I)+CMCONC(I) 0002590
519. TURBDR(I)=TURBDR(I)+DISI 0002595
520. IF(NZ(EQ.10030) GO TO 140 0002600
521. TOTAL=TOTAL+DCONC(I)+CMCONC(I)+CMCONC(I)+TURBDR(I) 0002605
522. CONTINUE 0002610
523. 140 CONTINUE 0002615
524. AOD=1.0 0002620
525. AOD=AOD+TOTAL/TOTAL 0002625
526. 150 SCNORM=0.0 0002630
527. SAPRIM=0.0 0002635
528. SSEC=0.0 0002640
529. SCWAST=0.0 0002645
530. SDWAST=0.0 0002650
531. SABLOW=0.0 0002655
532. STB=0.0 0002660
533. STOT=0.0 0002665
534. SPER=0.0 0002670
535. PAPRIM=0.0 0002675
536. PSEC=0.0 0002680
537. PCWAST=0.0 0002685
538. PDWAST=0.0 0002690
539. PABLOW=0.0 0002695
540. PB=0.0 0002700
PTOTAL = 0.0
PPER = 0.0
PNORM = 0.0
TLAUND = 0.0
TOTAL = 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.36.0R.NZ.EQ.54) GO TO 180
IF (NZ.EQ.51) GO TO 130
DISI = DIS(I) *1.6283E+13
APRIM = PC0NC(I) * DISI
ASEC = 0.0
CUASTE = CUCONC(I)
DUASTE = DUCONC(I)
ABLOU = CMCONC(I)
TB = TURBDR(I)
TOTAL = CUASTE + DUASTE + ABLOU + TB
TOTALN = TOTAL * XAOR
NUCLI = NUCL(I)
XLAUND = 0.0
TOTALG = 0.0
IF (NUCLI .NE. LAUNDY(L)) GO TO 155
XLAUND = ULAUND(L) * PFLAUN
TOTALG = TOTALN + XLAUND * PFLAUN
L = L + 1
155 CONTINUE
IF (TOTALG .LT. 1E-5) GO TO 156
ISUB = 2
IF (TOTALG .LT. 1E-5) ISUB = 1
DIV = 10** (INT(ALOG10(TOTALG)) - ISUB)
TOTALG = AINT(TOTALG/DIV) * DIV
156 CONTINUE
IF (NUCL(I).EQ.10030) TOTALN = TOTAL
IF (NZ.EQ.1) GO TO 160
SAPRIM = SAPRIM + APRIM
SSEC = SSEC + ASEC
SABLOU = SABLOU + ABLOW
SCWAST = SCWAST + CWASTE
SDWAST = SDWAST + DWASTE
STB = STB + TB
TOTAL = TOTAL + TOTALN + XLAUND * PFLAUN
SCNORM = SCNORM + TOTALN
TLAUND = TLAUND + XLAUND
CTOTAL = CTOTAL + TOTALG
160 IF (TOTALG .LT. 1E-5) GO TO 180
IF (MOD(KOUNTR, 50).NE.0) GO TO 170
PRINT 9000, REACTR
PRINT 9006
CALL NOAH(NUCL(I), NAME)
THALF = 8.0223E-5/DIS(I)
PRINT 9007, NAME, THALF, APRIM, CWASTE, DWASTE,
ABLOW, TOTAL, TOTALN, XLAUND, TLAUND, CTOTAL, SCNORM
KOUNTR = KOUNTR + 1
IF (NZ.EQ.1) GO TO 180

PAPRIM=PAPRIM+APRIM
PSEC=PSEC+ASEC
PCUAST=PCUAST+CWASTE
PDWAST=PDWAST+DWASTE
PABLW=PABLLOW+PABLLOW
PTB=PTB+TB
PTOTAL=PTOTAL+TOTAL
PNORM=PNORM+TOTALN
180 CONTINUE
PAPRIM=SAPRIM-PAPRIM
PSEC=SSEC-PSEC
PCWAST=PCUAST-PCWAST
PDWAST=PDWAST-PDUAST
PABLOW=SABLOW-PABLLOW
PTB=STB-PTB
PTOTAL=STOTAL-PTOTAL
PNORM=SCNORM-PNORM
ISUBC=2
IF (CTOTAL.GT.1.) ISU3C=1
DIV=10.X**((INT(ALOG10(CTOTAL))-ISUBC))
CTOTAL = AINKCTOTAL/DIV + 0.5)XDIV
IF (PNORM.LT. 1E-5) GO TO 184
DIV=10.X**((INT(ALOG10(PNORM))-2))
PNORMT = AINKPNORM/DIV + 0.5)XDIV
184 PNORMT=PNORM
PRINT 9008, PAPRIM,PCUAST,PDUAST,PABLOU,PTOTAL,PNORM,PNORMT
PRINT 9009, SAPRIM,SCUAST,SDUAST,SABLOU,STOTAL,SCNORM,TLAUND
195 PRINT 9012, ITRITR
9000 FORMAT(1H1,20X,7A4,' LIQUID EFFLUENTS (CONTINUED)')
9001 FORMAT(1H1,20X,7A4, ' LIQUID EFFLUENTS')
9003 FORMAT(1X,A2,13.1,2X,1PE9.2,2(2X,E9.2,2X),5(1X,0PF9.5,1X),2(1X,0P)
9004 1 F9.5,1X,0PF9.5)
9006 FORMAT(1H0,17X,'CONCENTRATION', ANNUAL RELEASES TO DISCHARGE CAN
9007 1AL/18X. TOTAL /' ',' ADJUSTED DETERGENT TOTAL
9008 2 '/' NUCLIDE HALF-LIFE COOLANT HIGH PURITY LOW PURITY CHEMICA
9009 3L TOTAL LWS TOTAL WASTES /'10X,'(DAYS) (MICRO C
9010 4I/ML)',3(' (CURIES) '), ' (CURIES) (CI/YR) (CI/YR)
9011 FORMAT(1H12X,'TRITIUM RELEASE',12X, 13,' CURIES PER YEAR')
9012 FORMAT('CORROSION AND ACTIVATION PRODUCTS')
9013 FORMAT('FOSSION PRODUCTS')
9014 FORMAT('TRITIUM RELEASE',12X,I3,' CURIES PER YEAR')
9015 RETURN
END

BLOCK DATA
COMMON /CONB/BCONC(800)
DATA BCONC/2X0.0,0.0 1,33X0.0, 0003000
2E-2,3E-2,4E-3,15X0.0,3E-4,0.0,7E-5,0.0,5E-5,2.3X0.0,0.0,1E-2,13X0.0
2E-4,5X0.0,6E-6,4X0.0,7E-6,5X0.0,6E-5,5X0.0,4E-5,6X0.0,6E-5,4X0.0
4E-5,5X0.0,3E-4,19X0.0,0.5E-3,1E-3,4.3X0.0,0.7E-6,5X0.0,6E-5,5X0.0,4E-5,3X0.0
5X0.0,6E-5,5X0.0,4E-5,6X0.0,6E-5,5X0.0,4E-5,3X0.0
6X0.0,6E-5,5X0.0,4E-5,6X0.0,6E-5,5X0.0,4E-5,3X0.0
SUBROUTINE SOLVE

COMMON/EQ/XZERO(800), XZH(800), XTEMP(800), XNEW(10,800), B(800), D(800)

COMMON/FLEX/FLUX(10), MMN, MOUT, INDEX, QXN, AXN, ERR, NOBLND, MZERO

COMMON/PROCESS/MPROS, RATE(8), NOPROS(8), N2PROS(8,20), PR(800)

COMMON/FLUXN/T(20), POWER(10), TOCAP(800), FISS(100), DIS(800), ILITE,

I ACT, IFP, ITOT, NON, INPT

COMMON/OUT/NUCL(800), TITLE(20), Q(800), FG(800), CUTOFF(7),

1, POS, BURNUP, FLUXB, MSTAR, ALPHAN(100), SPONFC(100), ABUND(500),

2, BASIS(10), TCONST, TUNIT

DO 10 I=1, ITOT

D(I)=-DIS(I)

10 XTEMP(I)=0.0

DELT = K1)XTC0NST

CALL DECAY( 1,DELT, IT0T)

CALL TERM(DELT,1,ILITE,ITOT)

CALL EQUIL(1, ITOT)

DO 30 I=1,ITOT

XTEMP(I)=XNEU(1,1)

30 RETURN

END

SUBROUTINE TERM(T,M,ILITE,ITOT)

LOGICAL LONG

INTEGER LOC, NON0, KD

INTEGER LCP(2500)

INTEGER NONP(800)

INTEGER NQ, NQU, NQUEUE

REAL B, TE, BATM

REAL CIMN(800), CSUM(800), CIMO(800)

DIMENSION AP(2500), CIMB(800), CIMO(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

C TERM ADDS ONE TERM TO EACH ELEMENT OF THE SOLUTION VECTOR

CSUM(J) IS THE CURRENT APPROXIMATION TO XNEW(M, J)

CIMO(J) IS THE VECTOR CONTAINING THE LAST TERM ADDED TO EACH

ELEMENT OF CSUM(J)

CIMN(J) IS THE VECTOR CONTAINING 1/TON TIMES THE NEU TERM TO BE

ADDED TO CSUM(J)

CIMN(J) IS GENERATED FROM CIMO(J) BY A RECURSION RELATION:

CIMN(J) = SUM OVER L OF (AP(J,L) * CIMO(L))

AP(I,J) IS THE REDUCED TRANSITION MATRIX FOR THE LONG-LIVED

NUCLIDES

LOGICAL L

INTEGER LOC, NON0, KD

INTEGER LCP(2500)

INTEGER NONP(800)

INTEGER NQ, NQU, NQUEUE

REAL B, TE, BATM

REAL CIMN(800), CSUM(800), CIMO(800)

DIMENSION AP(2500), CIMB(800), CIMO(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

C TERM ADDS ONE TERM TO EACH ELEMENT OF THE SOLUTION VECTOR

CSUM(J) IS THE CURRENT APPROXIMATION TO XNEW(M, J)

CIMO(J) IS THE VECTOR CONTAINING THE LAST TERM ADDED TO EACH

ELEMENT OF CSUM(J)

CIMN(J) IS THE VECTOR CONTAINING 1/TON TIMES THE NEU TERM TO BE

ADDED TO CSUM(J)

CIMN(J) IS GENERATED FROM CIMO(J) BY A RECURSION RELATION:

CIMN(J) = SUM OVER L OF (AP(J,L) * CIMO(L))

AP(I,J) IS THE REDUCED TRANSITION MATRIX FOR THE LONG-LIVED

NUCLIDES

LOGICAL L

INTEGER LOC, NON0, KD

INTEGER LCP(2500)

INTEGER NONP(800)

INTEGER NQ, NQU, NQUEUE

REAL B, TE, BATM

REAL CIMN(800), CSUM(800), CIMO(800)

DIMENSION AP(2500), CIMB(800), CIMO(800)

NUL=0

NN=0

C FIRST CONSTRUCT REDUCED TRANSITION MATRIX FOR LONG-LIVED ISOTOPES
NUM=NON0(L)
IF(M.GT.MMN.OR.M.EQ.MZERO) NUM=KD(L)
CIMB(L)=B(L)
IF(NUM.LE.NUL) GO TO 210
NS=N+1
N=NUL
NL=NUM-NUL
DO 200 N1=1,NL
J=LOC(N)
DJ=-D(J)

C THIS IS A TEST TO SEE IF ONE OF THE ASSYMPTOTIC SOLUTIONS APPLIES
IF(.NOT.LONG(J)) GO TO 10
NN=NN+1
AP(NN)=A(N)
LDCP(NN)=J
GO TO 200

C GOING BACK UP THE CHAIN TO FIND A PARENT WHICH IS NOT IN

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=NON0(J)
IF(M.GT.MMN.OR.M.EQ.MZERO) NUX=KD(J)
NUF=0
IF(J.GT.1) NUF=NON0(J-1)
NX=NUX-NUF
IF(NX.LT.1) GO TO 190
K=NUF
DO 180 KK=1,NX
J1=LOC(K)
DJ=-D(J1)
JP=J
IF(J1.EQ.NQ(KP)) GO TO 180
KP=Q0(KP)
IF(KP.NE.0) GO TO 30
AKDJQ=QUEA(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=J
40 DD(I)=-D(KP)
DXP(I)=XP(KP)
KP=Q0(KP)
IF(KP.EQ.0) GO TO 50
I=I+1
IF(I.LE.IOO) GO TO 40

C IF QUEUE OF SHORT-LIVED NUCLIDES EXCEEDS 100 ISOTOPES, TERMINATE
PRINT 9000, M,L,J,J,AKDJQ
900 FORMAT(1TOO LONG A QUEUE HAS BEEN FORMED IN TERM/*,415,E12.5)  
GO TO 190  
IM=I-1  
DO 110 I=2,IM  
DL=DD(I)  
XPL=DXP(I)  
BATE=0.DO  
I=1-1  
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/PK-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  
PROD=(XPJ-XPL)/PROD  
PRO1=XPJ/DKR  
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.OE-4) PROD=PRO1  
S=S1  
PI=P1*PI  
IF(ABS(PI).GT.1.E25) GO TO 100  
CONTINUE  
BATE=BATE+PROD/PI  
CONTINUE  
IF SUMMATION IS NEGATIVE, SET EQUAL TO ZERO AND PRINT MESSAGE  
IF(BATE.LT.O.DO) PRINT 9001, L,IM,BATE,BATM  
9001 FORMAT(1,L15,1P2E12.5)  
IF(BATE.LT.O.DO) BATE=O.DO  
BATM=BATM+BATE  
CONTINUE  
DRA=AKDQ*DX*(TRM-BATM)/TRM  
GO TO 130  
DRA=AKDQ*AMAX1(DRB,0.0)*DJ  
IF(NS.GT.NN) GO TO 150  
DO 140 LJ=NS,NN  
IF(LCP(LJ).NE.J1) GO TO 140  
AP(LJ)=AP(LJ)+DRA  
GO TO 180  
CONTINUE  
NN=NN+1  
AP(NN)=DRA  
LCP(NN)=J1  
GO TO 180  
IF(AKDQ.LE.1.0E-06) GO TO 180  
IF(NSAVE.GE.50) GO TO 180  
NSAVE=NSAVE+1  
NQUEUE(NSAVE)=J1
CO 841. QUEUE(NSAVE)=AKDJQ 0004200
842. NQU(NSAVE)=J 0004205
843. QUB(NSAVE)=DRB-1./(DJXT) 0004210
844. 180 CONTINUE 0004215
845. 190 IF(NSAVE.LE.O) GO TO 200 0004220
846. J=HQUEUE(NSAVE) 0004225
847. QUE=QUEUE(NSAVE) 0004230
848. NQ(J)=NQU(NSAVE) 0004235
849. DRB=QUB(NSAVE) 0004240
850. CIMB(L)=CIMB(L)+QUEXB(J)*AMAX1(DRB,0.0) 0004245
851. NSAVE=NSAVE-1 0004250
852. GO TO 20 0004255
853. 200 CONTINUE 0004260
854. 210 NUL=NONP(L) 0004265
855. NONP(L)=NN 0004270
856. 220 CONTINUE 0004275
857. C FIND NORM OF MATRIX AND ESTIMATE ERROR AS DESCRIBED IN LAPIUS 0004280
858. C AND LUUS, OPTIMAL CONTROL OF ENGINEERING PROCESSES BLAISDELL 1967 0004285
859. C FIND THE MINIMUM OF THE MAXIMUM ROW SUM AND THE MAXIMUM COLUMN SUM 0004290
860. ASUM =0.0 0004295
861. ASUMJ=0.0 0004300
862. NUL=1 0004305
863. DO 250 I=1,ITOT 0004310
864. IF(.NOT.LONG(I)) GO TO 250 0004315
865. DI=-D(I)*XT 0004320
866. AJ=DI 0004325
867. NUM=NONP(I) 0004330
868. IF(NUL.GT.NUM) GO TO 240 0004335
869. DO 230 N=NUL,NUM 0004340
870. AJ=AJ+APCN) 0004345
871. 230 CONTINUE 0004350
872. AI=DI+DI 0004355
873. IF(AI.GT.ASUM) ASUM =AI 0004360
874. IF(AJ.GT.ASUMJ) ASUMJ =AJ 0004365
875. NUL=NONP(I)+1 0004370
876. IF(ASUMJ.LT.ASUM) ASUM=ASUMJ 0004375
877. C USE ASUM TO DECIDE HOW MANY TERMS ARE REQUIRED AND ESTIMATE ERROR 0004380
878. NLARGE=3.5*ASUM +5. 0004385
879. XLARGE=NLARGE 0004390
880. ERR1=EXP(ASUM )*(ASUM+2.1828/XLARGE)**NLARGE/SQRK6.2332XXLARGE) 0004395
881. 9002 FORMAT('OMAXIMUM ERROR GT 0.001, =F10.6,', TRACE = 'F10.4, 0004400
882. 1 ' NLARGE = '16) 0004405
883. C NEXT GENERATE MATRIX EXPONENTIAL SOLUTION 0004410
884. DO 260 I=1,ITOT 0004415
885. CSUM(I)=XTEMP(I) 0004420
886. CIMN(I)=XTEMP(I) 0004425
887. 260 CONTINUE 0004430
888. ERR3=0.001*XERR 0004435
889. DO 310 NT=1,NLARGE 0004440
890. DO 270 I=1,ITOT 0004445
891. CIMO(I)=CIMN(I) 0004450
892. 270 CONTINUE 0004455
893. TON=T/NT 0004460
894. NUL=1 0004465
895. DO 300 I=1,ITOT 0004470
896. IF(.NOT.LONG(I)) GO TO 300 0004475
897. NUM=NONP(I) 0004480
898. CIMNI=0.0 0004485
899. IF(NUL.GT.1) CIMNI=CIMB(I) 0004490
900. IF(NUL.GT.NUM) GO TO 290 0004495
SUBROUTINE DECAY(M,T,ITOT)

C DECAY TREATS SHORT-LIVED ISOTOPES AT BEGINNING OF CHAINS USING
C BATEMAN EQUATIONS

LOGICAL X 1 LONG
REAL BATE
INTEGER LOC,NON0,KD
INTEGER 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),B(800),D(800)
COMMON/MATRIX/A(2500),LOC(2500),NON0(800),KD(800)
COMMON/TERMD/DD(100),DXP(100),QUEUE(50),NQU(50),NQUEUE(50),NQ(800)

DO 10 I=1,ITOT
XP(I)=0.0
LONG(I)=.FALSE.
XP1=0.0
DT=D(I)*T
IF(DT.LT.-50.) GO TO 10
IF(ABS(DT).LE.AXN) LONG(I)=.TRUE.
XP1=EXP(DT)
1 XP(I)=XP1

NUL=1
DO 160 L=1,ITOT
XTEN=0.0
DL=-D(L)
NUM=NON0(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
1 XTEM=XTEM+XTEMP(J)*A(N)*(XP(J)-XP(I))/DL-DJ
NQ(L)=0
NQ(J)=L
NSAVE=0
20 NUX=NON0(J)
IF(M.GT.MMN.OR.M.EQ.MZERO) NUX=KD(J)

RETURN
END
ORNL-TM-361

NUMF=1
IF(J.GT.1) NUMF=NUMF(J-1)+1
IF(NUMF.GT.NUX) GO TO 130
DO 120 K=NUMF,NUX
J1=LOC(K)
IF(LONG(J1)) GO TO 120
KP=J
120 CONTINUE

30 IF(J1.EQ.NQ(KP)) GO TO 120
KP=NQ(KP)
40 DD(I)=-D(KP)
DXP(I)=XP(KP)
KP=NQ(KP)
50 BATE=0.DO
110 NSAVE=NSAVE+1
GO TO 100

DD(I)=-D(KP)
DXP(I)=XP(KP)
KP=NQ(KP)
60 PROD=(XPJ-XPL)/PROD
PR01=XPJ/DKR
70 PI=1.0
S1=2./(DKT)
DO 90 JK=1,I1
S=1.0-DK/DD(JK)
IF(ABS(S).GT.1.E-4) GO TO 80
90 CONTINUE
BATE=BATE+PROD/PI
80 CONTINUE

USE THIS FORM FOR TWO NEARLY EQUAL HALF-LIVES

9000 FORMAT('I1',415,E12.5)
GO TO 130
50 BATE=0.DO
110 NSAVE=NSAVE+1
GO TO 100

CO
1021. NQUEUE(NSAVE)=J1
1022. QUEUE(NSAVE)=AKDJQ
1023. NQU(NSAVE)=J
1024. CONTINUE
1025. IF(NSAVE.LE.0) GO TO 140
1026. J=NQUEUE(NSAVE)
1027. QUE=QUEUE(NSAVE)
1028. NQ(J)=NQU(NSAVE)
1029. NSAVE=NSAVE-1
1030. GO TO 20
1031. CONTINUE
1032. IF(LONG(L)) XPAR(L)=XTEM/XP(L)
1033. NUL=NON0(L)+1
1034. IF(.NOT.LONG(L)) XNEW(M,L)=XTEM+XTEMP(L)*XP(L)
1035. CONTINUE
1036. DO 170 1=1,ITOT
1037. IF(.NOT.LONG(I)) XTEMP(I)=XTEM(I)
1038. XPAR(I)=AMAX1(XNEW(M,I)-XTEMP(I),0.0)
1039. 170 CONTINUE
1040. RETURN
1041. END
1042. SUBROUTINE EQUILCM,ITOT
1043. C
1044. C EQUIL PUTS SHORT-LIVED DAUGHTERS IN EQUILIBRIUM WITH PARENTS
1045. C EQUIL USES GAUSS-SEIDEL ITERATION TO GENERATE STEADY STATE
1046. C CONCENTRATIONS
1047. C
1048. LOGICAL I:LONG
1049. INTEGER LOC, NON0, KD
1050. COMMON/EQ/XZERO(800), XZH(800), XTEMP(800), XNEW(10,800),
1051. J(800), D(800)
1052. COMMON/MATRIX/A(2500), LOC(2500), NON0(800), KD(800)
1053. COMMON/FLEX/FLUX(10), MMN, MOUT, INDEX, QXN, AXN, ERR, NOBLND, MZERO
1054. COMMON/SERIES/ XP(800), XPAR(S00), LONG(S00)
1055. DO 10 I=1, ITOT
1056. XPAR(I)=0.0
1057. IF(.NOT.LONG(I)) GO TO 10
1058. XPAR(I)=XTEMP(I)+XP(I)
1059. XTEMP(I)=AMAX1(XNEW(M,I)-XTEMP(I),0.0)
1060. 10 CONTINUE
1061. ITER=1
1062. 20 N=0
1063. BIG=0.0
1064. DO 60 I=1, ITOT
1065. NUM=NON0(I)-N
1066. DI=D(I)
1067. IF(LONG(I)) GO TO 50
1068. XPAR(I)=0.0
1069. IF(.NOT.LONG(I)) GO TO 10
1070. XPAR(I)=AMAX1(XNEW(M,I)-XTEMP(I),0.0)
1071. 50 CONTINUE
1072. N=N+1
1073. J=LOC(N)
1074. DJ=D(J)
1075. XJ=XPAR(J)
1076. IF(LONG(J)) XJ=XJ+XTEMP(J)/(1.0-DJ/DI)
1077. XPAR= XPAR+A(N)*XJ
1078. 30 CONTINUE
1079. XPAR= XPAR/5
1080. IF(XPAR.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=NON0(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)) XNEUCM(I)=XNEU(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 LOC,NON0,KD
INTEGER ELE(99),STA(2)
INTEGER KAP(800),MMAX(800)
INTEGER NAME(3)
DIMENSION COEFF(7,800),NPROD(7,800),CAPT(6),
NUCAL(6),NSORS(5), YIELD(5,500),TYLD(5)
DIMENSION Y(5)
DIMENSION SKIP(20)
DIMENSION MSRS(20)
C
COMMON/ELE,STA
COMMON/FLEX/FLUX(10),MMN,MOUT,INDEX,QXN,AXN,ERR,NOBLND,MZERO
COMMON/EQ/XZERO(80),XZH(800),XTEMP(800),XNEW(10,800),
B(800),D(800)
COMMON/MPC/MPCTAB,AMPC(800),WMPC(800)
COMMON/FLUXV(I(20),POWER(10),TOCAP(800),FISS(100),DIS(800),ILITE,
IAC.T,IFP,ITOT,NON,INPT
COMMON/OUT/NUCL(800),TITLE(20),Q(800),FG(800),CUTOFF(7),
P0W.BURNUP.FLUXB,MSR0,ALPHAN(100),SPONF(100),ABUND(500),
BASIS(10),TCOND,TUNIT
COMMON/MATRIX/A(2500),LOC(2500),NON0(800),KD(800)
COMMON/EQUIVALENCE (XZERO(1),KAP(1)),XZERO(401),MMAX(1)),
EQUIVALENCE (A1,DLAM)
DATA NUCAL/-20030,-10000,10,11,-10,-9/
DATA DATA MSRS/922330,922350,902320,922380,942390,922330,922350,942410,
922380,942390,922330,942410,922350,942390,922330,922350,942390,
922380,942390,922350,942400,922380,942390,922330,922350,942390,
922380,922350,922380,942390/
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
C IGWC=0
C IGWC=1
C NLIBE=-NLIBE
C PRINT 9000
9000 FORMAT('1D,18,A18,NLIBE'),WILL READ TAPE GENERATED BY CASDAR')
1141. 10 N1=4-MLIBE
1142. 20 READ 9001, THERM,RES,FAST,ERR,NMO,NDAY,NYR
1143. 30 PRINT 9005, NMO,NDAY,NYR
1144. 40 PRINT 9006
1145. 50 PRINT 9007
1146. 60 PRINT 9008
1147. 70 PRINT 9009
1148. 80 PRINT 9010
1149. 90 PRINT 9013
1150. 100 PRINT 9014
1151. C
1152. C THERM = RATIO OF THERMAL FLUX TO TOTAL FLUX
1153. C RES = RATIO OF RESONANCE FLUX TO TOTAL FLUX
1154. C FAST = RATIO OF FAST FLUX TO TOTAL FLUX
1155. C ERR = TRUNCATION ERROR LIMIT
1156. C READ DATA FOR LIGHT ELEMENTS
1157. C
1158. K=5*(MLIBE-1)
1159. DO 30 K1=1,5
1160. K2=K+K1
1161. 30 NSORS(K1)=MSRS(K2)
1162. PRINT 9018, THERM,RES,FAST,(NSORS(K),K=1,5),MLIBE
1163. I=0
1164. NUTAPE=0
1165. 40 I=I+1
1166. 50 READ(8,9034,END=260)NUCL(I),DLAM,IFB1,FP1,FP1,FT,FA,FSF,
1167. Q(I),FQ(I),ABUND(I),WMPCCI(I),AMPCCI(I)
1168. IF(IFGC.GT.0) GO TO 70
1169. DO 60 N=1,MLIBE
1170. 60 READ(8,9035) SIGTH,FNG1,FNA,FNP,RITH,FINA,FINP,SIGMEV,FN2N1,FFNA,
1171. FFNP,IT
1172. 70 DO 80 N=1,MLIBE
1173. 80 READ(8,9040) SIGTH,FNG1,FNA,FNP,RITH,FINA,FINP,SIGMEV,FN2N1,FFNA,
1174. FFNP,IT
1175. 90 IF(N1.EQ.0) GO TO 110
1176. DO 100 N=1,N1
1177. 100 READ(8,9036) SKIP
1178. 110 IF(IT.EQ.0) GO TO 50
1179. 120 M=0
1180. CALL HALF(A1,IF)
1181. NUCL=NUCL(I)
1182. IF(NUCL.EQ.0) GO TO 260
1183. CALL NGAH(NUCL,NAMES)
1184. IF(MOD(I-1,50).EQ.0) PRINT 9012, TITLE(N1).N=1,18
1185. IF(MOD(I-1,50).EQ.0) PRINT 9016
1186. SIGTH=THERM*SIGTH
1187. RITH=RES*RITH
1188. SIGMEV=FAST*SIGMEV
1189. SIGMA=SIGTH+FNA+RITH*FINA+SIGMEV*FFNA
1190. SIGM=SIGTH+FINP+RITH*FINP+SIGMEV*FFNP
1191. FNG=1.0-FNA-FFNA
1192. IF(FNG.LT.1.0E-4) FNG=0.
1193. FING=1.0-FINP-FFNP
1194. IF(FING.LT.1.0E-4) FING=0.
1195. FN2N=1.0-FFNA-FFNP
1196. IF(FN2N.LT.1.0E-4) FN2N=0.
1197. SIGNG=SIGTH+FNG*RITH*FING
1198. SIGNG=SIGMEV*FN2N
1199. SIGE=SIGNG*FINP+RITH*FINP
1200. SIGMA=SIGE+FINP+RITH*FINP+SIGMEV*FFNP
1201. PRINT 9033, NAME, DLAM,FB1,FP,FP1,FT,FA,SIGNG, 0006000
1202. 1  FNG1,SIGN2N,FN2N1,SIGNA,SIGNP,Q(I),FG(I),ABUND(I) 0006005
1203. C TEST RADIOACTIVITY 0006010
1204. C 140 IF(A1.LE.ERR) GO TO 180 0006020
1205. ABETA=1.0 0006025
1206. C TEST POSITRON EMISSION 0006030
1207. C IF(FP .LT. ERR) GO TO 150 0006040
1208. M=M+1 0006045
1209. C COEFF(M,I)=FP+1 0006050
1210. NPROD(M,I)=NUCLI-10000 0006055
1211. ABETA=ABETA-FP 0006060
1212. C TEST POSITRON EMISSION TO EXCITED STATE OF PRODUCT NUCLIDE 0006065
1213. C IF(FP1 .LT. ERR) GO TO 150 0006070
1214. M=M+1 0006075
1215. C COEFF(M-1,I)=COEFF(M-1,I)-COEFF(M,I) 0006080
1216. C TEST ISOMERIC TRANSITION 0006085
1217. C IF(FT .LT.ERR) GO TO 160 0006090
1218. M=N+1 0006095
1219. C COEFF(M,I)=FT*A1 0006100
1220. NPROD(M,I)=NUCLI 0006105
1221. ABETA=ABETA-FT 0006110
1222. C TEST ALPHA EMISSION 0006115
1223. C IF(FA .LT. ERR) GO TO 170 0006120
1224. M=N+1 0006125
1225. C COEFF(M,I)=FA*A1 0006130
1226. NPROD(M,I)=NUCLI-20040 0006135
1227. ABETA=ABETA-FA 0006140
1228. C TEST NEGATRON EMISSION 0006145
1229. C IF(ABETA .LT.1.E-4) GO TO 180 0006150
1230. M=1+1 0006155
1231. C COEFF(M,I)=ABETA*A1 0006160
1232. NPROD(M,I)=NUCLI+10000 0006165
1233. ABETA=ABETA-FA 0006170
1234. C TEST NEGATRON EMISSION TO EXCITED STATE OF PRODUCT NUCLIDE 0006175
1235. C IF(FB1 .LT. ERR) GO TO 180 0006180
1236. M=M+1 0006185
1237. C COEFF(M,I)=FB1*COEFF(M-1,I) 0006190
1238. NPROD(M,I)=NPROD(M-1,I)+1 0006195
1239. C COEFF(M-1,I)=COEFF(M-1,I)-COEFF(M,I) 0006200
1240. COMPUTE NEUTRON CAPTURE CROSS SECTIONS IN THREE REGIONS 0006205
1241. KAP(I)=M 0006210
1242. C 180 KAP(I)=M 0006215
1243. 0006220
1244. 0006225
1245. 0006230
1246. 0006235
1247. 0006240
1248. 0006245
1249. 0006250
1250. 0006255
1251. 0006260
1252. 0006265
1253. 0006270
1254. 0006275
1255. 0006280
1256. 0006285
1257. 0006290
1258. 0006295
DO 190 KI = 1, 6
190   CAPT(KI) = 0.0
CAPT(1) = SIGNA
CAPT(2) = SIGNP
CAPT(4) = SIGN*FRING1
CAPT(3) = SIGN - CAPT(4)
CAPT(6) = SIGN2N*FRING1
CAPT(5) = SIGN - CAPT(6)
TOCAP(I) = 0.0
C TOTAL NEUTRON CROSS SECTION FOR NUCLIDE(I)
DO 220 K = 1, 6
220   CAPKI = CAPT(K)
   IF(CAPKI .LT. ERR) GO TO 220
   M = M + 1
   NPR0D(M, I) = NUCLI + NUCL(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)
   NPR0D(M, I) = 20040
   210 IF(K .NE. 2) GO TO 220
   M = M + 1
   COEFF(M, I) = COEFF(M-1, I)
   NPR0D(M, I) = 10010
   220 CONTINUE
   IF(MOD(NUCLI, 10).EQ.0) GO TO 250
   DO 240 K = 1, M
   240   NPR0D(K, I) = NPR0D(K-1, I)
   250   MMAX(I) = M
   IF(M .GT. 7) PRINT 9039, M
   DIS(I) = A1
   GO TO 40
   260 ILITE = I - 1
   IACT = 0
   C READ DATA ON ACTINIDES
   C
   270 READ(8, 9034, END=450) NUCL(I), DLAM, IU, FB1, FP1, FT, FA, FSF, IQ(I), FG(I), DUMMY, WMPC(I), AMPC(I)
   100 DO 280 N = 1, NLIBE
   280   READ(8, 9037) SIGNR, RINGF, RING1, SIGF, RIF, SIGFF, SIG2N, FN2N1, SIG3N, IT
   290 CONTINUE
   IF(N1 .EQ. 0) GO TO 250
   DO 290 N = 1, N1
   290   SKIP
   300 IFIT .EQ. 0) GO TO 270
   310 M = 0
   NUCLI = NUCL(I)
   IF(NUCLI .EQ. 0) GO TO 450
   DO 320 K = 1, 5
   320   IF(NUCLI .EQ. 0) GO TO 450
   330 IF(N1 .EQ. 0) NSORS(K) NSORS(K) = I
   311 DO 320 K = 1, 5
   320   IF(N1 .EQ. 0) NSORS(K) NSORS(K) = I
   330 CONTINUE
   CALL HALF(A1, IU)
   CALL NOAH(NUCLI, NAME)
   SIGNR = THERM*SIGNR + RES*RING
   SIGF = THERM*SIGF + RES*RIF + FAST*SIGFF
   SIG2N = SIGN2N*FAST
   SIG3N = SIGN3N*FAST
   IF(MOD(IACIT, 50).EQ.0) PRINT 9012, (TITLE(N), N=1, 18)
IF(MOD(IACT,50).EQ.0) PRINT 9024
PRINT 9026, NAME, DLAM, FB1, FP, FP1, FT, FA, FSF, SIGNG, FNG1, SIGF, SIGN2N, SIGN3N, Q(I), FG(I)
IACT=IACT+1
C TEST RADIOACTIVITY
IF(CAI.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*AB1
NPROD(M,I)=NUCLI-1000
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
IF(FT .LT.ERR)GO TO 360
M=M+1
COEFF(M,I)=FT*AB1
NPROD(M,I)=NUCLI
C  ALPHA EMISSION
IF(FA .LT.ERR)GO TO 370
M=M+1
COEFF(M,I)=FA*AB1
NPROD(M,I)=NUCLI+20040
C  BETA DECAY
IFCABETA.LT. 1.E-4) GO TO 380
M=M+1
COEFF(M,I)=ABETA*AB1
NPROD(M,I)=NUCLI+20040
C  NEUTRON CAPTURE CROSS SECTIONS
KAP(I)=M
DO 390 K=1,6
CAPT(K )=0.0
CAPT(2)=SIGNG*FNG1
CAPT(1)=SIGNG-CAPT(2)
CAPT(4)=SIGN2N*FNG1
CAPT(3)=SIGN2N-CAPT(4)
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)
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(K,I)=NPROD(K,I)-1
MMAX(I)=M
IF(M.GT.7) PRINT 9039, M
SPONF(IACT)=FSFXA1*6.023E23
ALPHAN(IACT)=FA*A1^6.023E13*Q(I)*X3.65
DIS(I)=A1
I=I+1
GO TO 270
IL=0
DO 460 K=1,M
TYLD(K)=0.0
READ DATA FOR FISSION PRODUCTS
READ(8,90 34,END=6 90)NUCL(I),DLAM,IU,FB1,FP1,FT,FA,FSF,
Q(I),FG(I),DUMMY,WMPC(I),AMPC(I)
DO 480 N=1,NLIBE
READ(8,9038) SIGNG,RING,FNG1,Y,IT
IF(NI.EQ.O) GO TO 500
DO 490 N=1,N1
READ(8,9036) SKIP
IF(IT .EQ. 0) GO TO 470
M=0
CALL HALF(A1,IU)
NUCLI=NUCL(I)
IFCNUCLI.EQ.O) GO TO 690
CALL NOAH(NUCLI,NAME)
IF(MOD(IL,50).EQ.0) PRINT 9012, (TITLE (H),N=1,18)
SIGNG= THERMXSIGNG+RES*X*RING
IF(NLIBE.EQ.3) GO TO 540
IF(MOD(IL,50).EQ.0) PRINT 9019
PRINT 9021, NAME, DLAM,FB1,FP,FTP,FT,SIGNG,
17,FNG1,Y(Q(I),FG(I)
GO TO 550
IF(MOD(IL,50).EQ.0) PRINT 9020
PRINT 9022, NAME, DLAM,FB1,FP,FTP,FT,SIGNG,FNG1,
Y2),Y(4),Y(5),Q(I),FG(I)
GO TO 550
IF(A1.LT.ERR) GO TO 600
ABETA=1.0
C POSITRON EMISSION
A3=FP
1441. IF (A3 .LT. ERR) GO TO 570
1442. ABETA = ABETA - A3
1443. AP = A3*FP1
1444. AP = A3 - AP
1445. IF (AP .LT. ERR) GO TO 560
1446. M = M + 1
1447. CDEFF(M,I) = AP*AI
1448. NPROD(M,I) = NUCLI - 10000
1449. 560 IF (AP1 .LT. ERR) GO TO 570
1450. M = M + 1
1451. CDEFF(M,I) = AP1*AI
1452. NPROD(M,I) = NUCLI - 9999
1453. C ISOMERIC TRANSITION
1454. 570 IF (FT .LT. ERR) GO TO 580
1455. M = M + 1
1456. CDEFF(M,I) = FT*AI
1457. NPROD(M,I) = NUCLI
1458. ABETA = ABETA - FT
1459. C NEUTRON EMISSION
1460. 580 IF (ABETA .LT. 1.0E-4) GO TO 600
1461. A2 = FB1
1462. AB1 = ABETA * A2
1463. AB = ABETA - AB1
1464. IF (AB .LT. 1.0E-4) GO TO 590
1465. M = M + 1
1466. CDEFF(M,I) = AB*AI
1467. NPROD(M,I) = NUCLI + 10000
1468. 590 IF (AB1 .LT. 1.0E-6) GO TO 600
1469. M = M + 1
1470. CDEFF(M,I) = AB1*AI
1471. NPROD(M,I) = NUCLI + 10001
1472. C NEUTRON CAPTURE CROSS SECTIONS FOR FISSION PRODUCTS USING THREE
1473. C REGION APPROXIMATION
1474. 600 KAP(I) = M
1475. DO 610 K = 1, 6
1476. 610 CAPT(K) = 0.0
1477. CAPT(2) = SIGNG*FGN1
1478. CAPT(1) = SIGNG - CAPT(2)
1479. TOCAP(I) = 0.0
1480. DO 620 K = 1, 2
1481. CAPKI = CAPT(K)
1482. IF (CAPKI .LT. ERR) GO TO 620
1483. TOCAP(I) = TOCAP(I) + CAPKI
1484. CDEFF(M,I) = CAPKI
1485. NPROD(M,I) = NUCLI + NUCAL(K+2)
1486. 620 CONTINUE
1487. 630 IF (MOD(NUCLI, 10) .EQ. 0) GO TO 650
1488. DO 640 K = 1, M
1489. 640 NPROD(K,I) = NPROD(K,I) - 1
1490. 650 IL = IL + 1
1491. DO 660 J = 1, 5
1492. 660 YJ = Y(J) + 0.010
1493. TYLD(J) = TYLD(J) + YJ
1494. 670 YIELD(J, IL) = YJ
1495. IF (NLII .EQ. 1 OR NLII .EQ. 4) GO TO 680
1496. YIELD(3, IL) = YJ
1497. 680
ALL DATA ON NUCLIDES HAS BEEN READ, BEGIN TO COMPUTE MATRIX COEFF

FIND PRODUCT NUCLIDES FOR REACTIONS OF LIGHT ELEMENTS

NON ZERO MATRIX ELEMENTS FOR THE ACTINIDES
1561. IF(NUCLI.NE.NPROD(M,J)) GO TO 770
1562. NON0(I)=NON0(I)+1
1563. NON=NON+1
1564. IF(NON.GT.2500) PRINT 9041, NON,NUCL(I)
1565. A(NON)=COEFF(M,J)
1566. JT=J
1567. LOC(NON)=JT
1568. 770 CONTINUE
1569. 780 CONTINUE
1570. KD(I)=NON0(I)
1571. DO 800 J=I0,I1
1572. M1=KAPCJ)+1
1573. M2=MMAXCJ)
1574. IF(M2.LT.M1) GO TO 800
1575. DO 790 M=M1,M2
1576. IFCNUCLI.NE.NPROD(M,J)) GO TO 790
1577. NON0(I)=NON0(I)+1
1578. NON=NON+1
1579. IF(NON.GT.2500) PRINT 9041, NON,NUCL(I)
1580. A(NON)=COEFF(M,J)
1581. JT=J
1582. LOC(NON)=JT
1583. 790 CONTINUE
1584. 800 CONTINUE
1585. 810 CONTINUE
1586. C
1587. MATRIX ELEMENTS FOR FISSION PRODUCTS
1588. C
1589. 820 IF(IFP.LT.1) GO TO 900
1590. IM=ILITE+IACT
1591. I0=IM+1
1592. IF(ITOT.LT.I0) GO TO 900
1593. DO 880 I=I0,ITOT
1594. NUCLI=NUCL(I)
1595. I2=MAX0(I0,I-10)
1596. I3=MIN0CITOT,I+10)
1597. DO 840 J=I2,I3
1598. KMAX=KAPCJ)
1599. IF(KMAX.LT.1) GO TO 840
1600. DO 830 M=1,KMAX
1601. IF(NUCLI.NE.NPROD(M,J)) GO TO 830
1602. NON0(I)=NON0(I)+1
1603. NON=NON+1
1604. IF(NON.GT.2500) PRINT 9041, NON,NUCL(I)
1605. A(NON)=COEFF(M,J)
1606. JT=J
1607. LOC(NON)=JT
1608. 830 CONTINUE
1609. 840 CONTINUE
1610. KD(I)=NON0(I)
1611. DO 860 J=12,13
1612. K1=KAPCJ)+1
1613. KMAX=MMAXCJ)
1614. IF(KMAX.LT.K1) GO TO 860
1615. DO 850 M=K1,KMAX
1616. IF(NUCLI.NE.NPROD(M,J)) GO TO 850
1617. NON0(I)=NON0(I)+1
1618. NON=NON+1
1619. IF(NON.GT.2500) PRINT 9041, NON,NUCL(I)
1620. A(NON)=COEFF(M,J)
1621. \text{JT} = \text{J}
1622. \text{LOC(NON) = JT}
1623. **CONTINUE**
1624. **CONTINUE**
1625. IF(IACT.LT.1) GO TO 880
1626. DO 870 K = 1,5
1627. \text{IL} = \text{I} - \text{IM}
1628. IF(YIELD(K,IL).LT.ERR) GO TO 870
1629. \text{NON} = \text{NON} + 1
1630. IF(NON(I).LT.2500) PRINT 9041, NON,NUCL(I)
1631. NON0CI) = NON0CI) + 1
1632. KK = NSORSCK)
1633. LOCCNON) = KK
1634. KF = KK - ILITE
1635. ACNON) = YIELDCK,IL))<FISSCKF)
1636. 870 CONTINUE
1637. 880 CONTINUE
1638. IF(IFP.LE.0) GO TO 900
1639. IF(NLIE.NE.3) GO TO 890
1640. PRINT 9027, TYLD(2),TYLD(4),TYLD(5)
1641. GO TO 900
1642. 890 PRINT 9030, (TYLDM(I),I = 1,5)
1643. C
1644. C ALL MATRIX ELEMENTS ARE NOW COMPUTED
1645. C BEGIN TRANSIENT SOLUTION
1646. C
1647. C TEMPORARILY WRITE OUT MATRIX ELEMENTS
1648. C
1649. C
1650. **CONTINUE**
1651. PRINT 9029
1652. N = 0
1653. DO 910 I = 1,ITOT
1654. NUM = NON0CI)
1655. IF(NUM.LE.0) GO TO 910
1656. N1 = N + NUM
1657. N = N + 1
1658. PRINT 9028, I,DIS CI),TOCAPCI),CACK),LOCCK),K = N ,N 1)
1659. N = N1
1660. 910 CONTINUE
1661. RETURN
1662. 920 STOP
1663. C
1664. C FORMATS FORMATS FORMATS FORMATS
1665. C 9001 FORMAT(F10.5,612)
1666. C 9002 FORMAT(I6,F5.3,I1,5F3.3,E5.2,F3.3, 13X,E5.2,F3.3,2E5.2)
1667. C 9003 FORMAT(I6,F5.3,I1,3X,4F3.3,E5.2,F3.3,5E5.2,F4.3,F3.3)
1668. C 9004 FORMAT(I6,F5.3,I1,5F3.3,2E5.2,F3.3,4E5.2,F3.3,4E5.2)
1669. C 9005 FORMAT(I14,43X,'NUCLEAR TRANSMUTATION DATA REVISED ',12,'/',12, 12,'=',NUCLIDE = 10000 * ATOMIC NO + 10 * MASS NO + ISOM )
1670. C 9006 FORMAT(1H1,43X,'\text{1/}',12, 12,'=',\text{NUCLIDE} = 10000 * ATOMIC NO + 10 * MASS NO + ISOM )
1671. C 9007 FORMAT(14X,'\text{1/}',12, 12,'=',\text{NUCLIDE} = 10000 * ATOMIC NO + 10 * MASS NO + ISOM )
1672. C 9008 FORMAT(14X,'\text{1/}',12, 12,'=',\text{NUCLIDE} = 10000 * ATOMIC NO + 10 * MASS NO + ISOM )
1673. C 9009 FORMAT(14X,'\text{1/}',12, 12,'=',\text{NUCLIDE} = 10000 * ATOMIC NO + 10 * MASS NO + ISOM )
1674. C 9010 FORMAT(14X,'\text{1/}',12, 12,'=',\text{NUCLIDE} = 10000 * ATOMIC NO + 10 * MASS NO + ISOM )
1675. C 9011 FORMAT(14X,'\text{1/}',12, 12,'=',\text{NUCLIDE} = 10000 * ATOMIC NO + 10 * MASS NO + ISOM )
1676. C 9012 FORMAT(14X,'\text{1/}',12, 12,'=',\text{NUCLIDE} = 10000 * ATOMIC NO + 10 * MASS NO + ISOM )
1677. C 9013 FORMAT(14X,'\text{1/}',12, 12,'=',\text{NUCLIDE} = 10000 * ATOMIC NO + 10 * MASS NO + ISOM )
1678. C 9014 FORMAT(14X,'\text{1/}',12, 12,'=',\text{NUCLIDE} = 10000 * ATOMIC NO + 10 * MASS NO + ISOM )
1679. C 9015 FORMAT(14X,'\text{1/}',12, 12,'=',\text{NUCLIDE} = 10000 * ATOMIC NO + 10 * MASS NO + ISOM )
1680. C 9016 FORMAT(14X,'\text{1/}',12, 12,'=',\text{NUCLIDE} = 10000 * ATOMIC NO + 10 * MASS NO + ISOM )
1. $\text{SIGNP} = \text{SIGTH} \times \text{FNP}$. \text{FNA}, \text{FNP} = \text{FRACTION THERMAL N-ALPHA, N-PROTON}.

2. $2N \cdot \text{RITH, RING, RIF, RINA, RINP} = \text{RESONANCE INTEGRAL FOR ABSORPTION.}

3. $3PTI0N, N-\gamma, \text{FISSION, N-ALPHA, N-PROTON}$. $\text{RING} = \text{RITH} \times (1 - \text{FINA} - \text{FINP})$. $\text{FINA, FINP} = \frac{\text{FINA}}{\text{FNP}}$. $\text{FNP} = \frac{\text{FRACTION THERMAL N-ALPHA, N-PROTON}}{\text{FINA}}$.

4. $41 - \text{FINA} - \text{FINP}$. $\text{RINA} = \text{RITH} \times \text{FINA}$. $\text{RINP} = \text{RITH} \times \text{FINP}$. $\text{FINA}, \text{FINP} = \frac{\text{FINA}}{\text{FNP}}$.

5. $5NP = \text{FRACTION RESONANCE N-ALPHA, N-PROTON}$.

6. $60N2N$, $\text{SIGNF}$. $\text{SIGNF} = \text{FAST CROSS SECTIONS (BARNS)}$.

7. $7FJISSION, N-2N, N-\alpha, N-\proton$. $\text{RING} = \text{RITH} \times 1.45$.

8. $8Q = \text{HEAT PER DISINTEGRATION}$.


10. $101 - \text{FINA} - \text{FINP}$. $\text{RINA} = \text{RITH} \times \text{FINA}$. $\text{RINP} = \text{RITH} \times \text{FINP}$. $\text{FINA}, \text{FINP} = \frac{\text{FINA}}{\text{FNP}}$.


12. $120$ FORMAK$^{1H1, 20X, 18A4}$.

13. $130$ FORMAK$^{2A}$.

14. $140$ FORMAK$^{0N2N, 0N-\alpha, 0N-\proton}$.

15. $150$ FORMAK$^{2A}$.

16. $160$ FORMAK$^{1H0, 20X, 18A4}$.

17. $170$ FORMAK$^{2A}$.

18. $180$ FORMAK$^{1H0, 20X, 18A4}$.

19. $190$ FORMAK$^{2A}$.

20. $200$ FORMAK$^{1H0, 20X, 18A4}$.

21. $210$ FORMAK$^{2A}$.

22. $220$ FORMAK$^{1H0, 20X, 18A4}$.

23. $230$ FORMAK$^{2A}$.

24. $240$ FORMAK$^{1H0, 20X, 18A4}$.

25. $250$ FORMAK$^{2A}$.

26. $260$ FORMAK$^{1H0, 20X, 18A4}$.

27. $270$ FORMAK$^{2A}$.

28. $280$ FORMAK$^{1H0, 20X, 18A4}$.

29. $290$ FORMAK$^{2A}$.

30. $300$ FORMAK$^{1H0, 20X, 18A4}$.
1741. 9017 FORMAT(1H0, A2, I3, A1, 1PE9.2, 0P5F6.3, 1PE9.2, 0P3F6.3, 1PE9.2, 0P2F6.3, 0008700
1742. 11PE9.2, 0P4F6.3, 0P5F6.3, 1PE9.2, 0P2F6.3, 0008710
1743. 1/,, IX,'THERM= 'F10.5, 5X,'RES= 'F10.5, 0008715
1744. 1/,, IX,'NEUTRON SOURCE= '5(110, 5X), 5X,'NLIBE= '13) 0008720
1745. 9018 FORMAK1H0, 32X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008725
1746. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008730
1747. 28 Y49 Q FG') 0008735
1748. 9019 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008740
1749. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008745
1750. 228 Y49 Q FG') 0008750
1751. 9020 FORMAK1H0, 32X, 'ACTINIDES AND THEIR DAUGHTERS',// 0008755
1752. 1' NUCL DLAM FBI FP FPI FT FA FSF E+6 SIGNG F 0008760
1753. 2ING FNGI SIGF RIF >SIGFF SIGN2N SIGN3N Q FG') 0008765
1754. 9021 FORMAK1H0, 32X, 'ACTINIDES AND THEIR DAUGHTERS',// 0008770
1755. 1' NUCL DLAM FBI FP FPI FT FA FSF E+6 SIGNG R 0008775
1756. 2ING FNGI SIGF RIF >SIGFF SIGN2N SIGN3N Q FG') 0008780
1757. 9022 FORMAK1H0, 32X, 'ACTINIDES AND THEIR DAUGHTERS',// 0008785
1758. 1' NUCL DLAM FBI FP FPI FT FA FSF E+6 SIGNG R 0008790
1759. 2ING FNGI SIGF RIF >SIGFF SIGN2N SIGN3N Q FG') 0008795
1760. 9023 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008800
1761. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008805
1762. 28 Y49 Q FG') 0008810
1763. 9024 FORMAK1H0, 32X, 'ACTINIDES AND THEIR DAUGHTERS',// 0008815
1764. 1' NUCL DLAM FBI FP FPI FT FA FSF E+6 SIGNG R 0008820
1765. 2ING FNGI SIGF RIF >SIGFF SIGN2N SIGN3N Q FG') 0008825
1766. 9025 FORMAT(1H0, 32X, 'ACTINIDES AND THEIR DAUGHTERS',// 0008830
1767. 1' NUCL DLAM FBI FP FPI FT FA FSF E+6 SIGNG R 0008835
1768. 2ING FNGI SIGF RIF >SIGFF SIGN2N SIGN3N Q FG') 0008840
1769. 9026 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008845
1770. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008850
1771. 28 Y49 Q FG') 0008855
1772. 9027 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008860
1773. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008865
1774. 28 Y49 Q FG') 0008870
1775. 9028 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008875
1776. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008880
1777. 28 Y49 Q FG') 0008885
1778. 9029 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008890
1779. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008895
1780. 28 Y49 Q FG') 0008899
1781. 9030 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008900
1782. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008905
1783. 28 Y49 Q FG') 0008910
1784. 9031 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008915
1785. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008920
1786. 28 Y49 Q FG') 0008925
1787. 9032 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008930
1788. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008935
1789. 28 Y49 Q FG') 0008940
1790. 9033 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008945
1791. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008950
1792. 28 Y49 Q FG') 0008955
1793. 9034 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008960
1794. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008965
1795. 28 Y49 Q FG') 0008970
1796. 9035 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008975
1797. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008980
1798. 28 Y49 Q FG') 0008985
1799. 9036 FORMAT(1H0, 36X, 'FISSION PRODUCTS', /,'0 NUCL DLAM FBI FP 0008990
1800. 1 FPI FT SIGNG ', ' FNG1 Y25 Y28 Y49 Q FG') 0008995
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
BII=0.0
10 XTEMP(I)=CWASTE(I)
CALL DECAY(1,DELT,ITOT)
CALL TERMCTMB,1,ILITE,ITOT)
CALL EQUIL(1,ITOT)
DO 20 I=1,ITOT
20 CWASTE(I)=XNEW(1,1)
RETURN
END

C PROGRAM BLOCK DATA
BLOCK DATA
INTEGER X2 ELEC(99), STA(2)
COMMON/ LABEL/ ELE, STA
DATA ELE/' H','HE','LI','BE',' B',' C',' N',' F',' NE',' NA',' M',
   ' O',' F',' NE',' NA',' M',
   ' C',' O',' N',' ZN',' GA',' GE',' AS',' SE',' BR',' KR',' RB',' SR',
   ' Y',' ZR',' NB',' MO',' TC',' RU',' RH',' PD',' AO',' CD',' IN',' SN',' SB',
   ' T',' FE',' Ti',' XE',' CS',' BA',' LA',' CE',' PR',' ND',' PM',' SM',' EU',' GD','
   ' ST',' DY',' ER',' TM',' YB',' Lu',' HF',' TA',' W',' RE',' Os',' Ir',' Pb',
   ' U',' NP',' Pu',' Am',' Cm',' Bk',' Cf',' Es','
   ' STA/ ', ' M '/
DATA STA/, ' M '/
RETURN
END

SUBROUTINE HALFC(A,I)
C SUBROUTINE HALFC CONVERTS HALF-LIFE TO DECAY CONSTANT (1/SEC)
DIMENSION CC(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(CA.GT.0.0) GO TO 10
IF(CI.EQ.6) GO TO 20
A=9.99
RETURN
A=CII/A
RETURN
A=0.0
RETURN
END

SUBROUTINE NOAH(NUCLI,NAME)
C SUBROUTINE NOAH CONVERTS SIX DIGIT IDENTIFIER TO ALPHAMERIC SYMBOL
INTEGER X2 NAME(3)
INTEGER X2 ELE(99), STA(2)
COMMON/ LABEL/ ELE, STA
IS=MOD(NUCLI,10)+1
NZ =NUCLI/10000
M2=NUCLI/10-NZ *1000
NAME(1)=ELE(NZ)
NAME(2)=M2
NAME(3)=STA(IS)
RETURN
END
C WATER CONCENTRATIONS CALCULATED USING METHODS OF DRAFT STANDARD
C GALE CODE FOR CALCULATING GASEOUS EFFLUENTS FROM LWRS. MODIFIED
C
1. AUG. 1978 TO IMPLEMENT APPENDIX I TO 10 CFR PART 50. REACTOR
2. WATER CONCENTRATIONS CALCULATED USING METHODS OF DRAFT STANDARD
3. ANS 237 'RADIOACTIVE MATERIALS IN PRINCIPAL FLUID STREAMS OF
4. LIGHT WATER COOLED NUCLEAR POWER PLANTS' DRAFT DATED MAY 20, 1974
C DECONI DEFINES THE DECAY CONSTANT FOR IODINES
DATA DECONI/9.970E-07, 9.170E-06/
C CBWR DEFINES THE REACTOR WATER CONCENTRATIONS FOR IODINES
DATA CBWR/3.7E-3, 3.5E-2/
C RN DEFINES THE NORMALIZED ANNUAL RELEASES DURING POWER OPERATIONS FOR THE REACTOR BLDG.
DATA RN/12.3, 12.3/
C RNS DEFINES THE NORMALIZED ANNUAL RELEASES DURING SHUTDOWNS FOR THE REACTOR BLDG.
DATA RNS/5.2, 5.2/
C RNT DEFINES THE NORMALIZED ANNUAL RELEASES DURING POWER OPERATIONS FOR THE TURBINE BLDG.
DATA RNT/3.8E3, 3.8E3/
C RNTS DEFINES THE NORMALIZED ANNUAL RELEASES DURING SHUTDOWNS FOR THE TURBINE BLDG.
DATA RNTS/4.1E2, 4.1E2/
C RNR DEFINES THE NORMALIZED RELEASES FROM THE RADWASTE BLDG. DURING POWER OPERATIONS
DATA RNR/4.6, 4.6/
C RNRS DEFINES THE NORMALIZED RELEASES FROM THE RADWASTE BLDG. DURING SHUTDOWNS
DATA RNRS/1.4, 1.4/
C RNMVP DEFINES NORMALIZED RELEASES FROM THE MECHANICAL VACUUM PUMP
DATA RNMVP/4.9E2, 4.9E2/
DATA RNMVPS/1.1E1, 1.1E1/
C RWK FORMAT(16X,"REACTOR VESSEL HALOGEN CARRYOVER FACTOR", 15X, F10.5)
C RMK FORMAT(16X,"PLANT CAPACITY FACTOR", T74, "0.80")
C RRM FORMAT(8A4, 12X, A4)
C RRS FORMAT(16X, 13A4, A2, F10.5)
C RRT FORMAT(16X, 13A4, A2, F10.5)
C RT FORMAT(20X, F8.0, 2(5X, F8.0))
C R1 FORMAT(27X, F6.2, 14X, F6.2, 18X, F6.2)
C R2 FORMAT(30X, "FRACTION COLLECTION DECAY/8X, "STREAM FLOW RATE OF PCA DISCHARGED")
C R3 FORMAT(16X, "THERE IS A CRYOGENIC DISTILLATION COLUMN")
C R4 FORMAT(16X, "AND XENON DECONTAMINATION FACTOR")
C R5 FORMAT(16X, "AND KRYPTON DECONTAMINATION FACTOR")
C R6 FORMAT(16X, "AND KRYPTON AND XENON HOLDUP TIME")
C R7 FORMAT(30X, "AND KRYPTON AND XENON HOLDUP TIME")
<table>
<thead>
<tr>
<th>Line</th>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>121.</td>
<td>FORMAT(16X,'THERE IS NO CHARCOAL DELAY SYSTEM')</td>
<td>00</td>
</tr>
<tr>
<td>122.</td>
<td>FORMAT(16X,'THERE IS A CHARCOAL DELAY SYSTEM')</td>
<td>00</td>
</tr>
<tr>
<td>123.</td>
<td>TIME (DAYS), F7.2, F9.5/20X, XENON HOLDUP TIME (DAYS), F7.2, F9.5/20X</td>
<td>00, 00</td>
</tr>
<tr>
<td>124.</td>
<td>KRYPTON DYNAMIC ADOPTION COEFFICIENT (CM^3/GM), F7.2, F9.5/20X,</td>
<td>00, 00</td>
</tr>
<tr>
<td>125.</td>
<td>XENON DYNAMIC ADOPTION COEFFICIENT (CM^3/GM), F7.2, F9.5/20X,</td>
<td>00, 00</td>
</tr>
<tr>
<td>126.</td>
<td>MASS OF CHARCOAL (THOUSAND LBS)</td>
<td>00</td>
</tr>
<tr>
<td>127.</td>
<td>TIME (DAYS), F7.2, F9.5/20X, XENON HOLDUP TIME (DAYS), F7.2, F9.5/20X</td>
<td>00, 00</td>
</tr>
<tr>
<td>128.</td>
<td>KRYPTON DYNAMIC ADOPTION COEFFICIENT (CM^3/GM), F7.2, F9.5/20X,</td>
<td>00, 00</td>
</tr>
<tr>
<td>129.</td>
<td>XENON DYNAMIC ADOPTION COEFFICIENT (CM^3/GM), F7.2, F9.5/20X,</td>
<td>00, 00</td>
</tr>
<tr>
<td>130.</td>
<td>MASS OF CHARCOAL (THOUSAND LBS)</td>
<td>00</td>
</tr>
<tr>
<td>131.</td>
<td>FRACTION IODINE BYPASSING CONDENSATE DEMINERALIZER, 17X, F7.2, F9.5</td>
<td>00, 00</td>
</tr>
<tr>
<td>133.</td>
<td>LIQUID WASTE INPUTS</td>
<td>00</td>
</tr>
<tr>
<td>134.</td>
<td>GASEOUS WASTE INPUTS</td>
<td>00</td>
</tr>
<tr>
<td>135.</td>
<td>TOTAL NOBLE GASES</td>
<td>00</td>
</tr>
<tr>
<td>136.</td>
<td>GASEOUS RELEASE RATE</td>
<td>00</td>
</tr>
<tr>
<td>137.</td>
<td>AIRBORNE PARTICULATE RELEASE RATE</td>
<td>00</td>
</tr>
<tr>
<td>138.</td>
<td>COOLANT CONC.</td>
<td>00</td>
</tr>
<tr>
<td>139.</td>
<td>COOLANT CONC.</td>
<td>00</td>
</tr>
<tr>
<td>140.</td>
<td>H-3 RELEASED FROM TURBINE BLDG. VENTILATION SYSTEM</td>
<td>00</td>
</tr>
<tr>
<td>141.</td>
<td>C-14 RELEASED VIA MAIN CONDENSER OFFGAS SYSTEM = 19.5 CI/YR</td>
<td>00</td>
</tr>
<tr>
<td>142.</td>
<td>TOTAL H-3 RELEASED VIA GASEOUS PATHWAY, 5X, 1PE7.1</td>
<td>00</td>
</tr>
</tbody>
</table>

```
352
```
181. WRITE(6,51)
182. READ(1,53)WORD,GTO
183. WRITE(6,55)WORD,GTO
184. READ(1,53)WORD,WLQ
185. WRITE(6,55)WORD,WLQ
186. READ(1,53)WORD,GDE
187. WRITE(6,55)WORD,GDE
188. PC = 0.015
189. READ(1,53)WORD,REGENT
190. IF(REGENT.EQ.O.O) PC = 0.004
191. WRITE(6,55)WORD,REGENT
192. READ(1,53)WORD,FFCDM
193. WRITE(6,55)WORD,FFCDM
194. WRITE(6,49) PC
195. C
196. C READ DATA FOR BWR LIQUID CODE
197. C
198. WRITE(6,904)
199. READ(1,55)WARD,CWFLR,CWA
200. READ(1,56)DFICW,DFSCSW,DFCW
201. READ(1,57)TC,TSTORC,CFWD
202. WRITE(6,58)
203. WRITE(6,59)WARD,CWFLR,CWA,CWFD,TC,TSTORC,DFICW,DFSCSW,DFCW
204. READ(1,55)WARD,CWFLR,CWA
205. READ(1,56)DFIDW,DFSCDW,DFDW
206. READ(1,57)TD,TSTORD,DFWD
207. WRITE(6,59)WARD,CWFLR,CWA,CWFD,TC,TSTORD,DFICW,DFSCSW,DFCW
208. READ(1,55)WARD,CWFLR,CWA
209. READ(1,56)DFICM,DFSCSM,DFCM
210. READ(1,57)TCM,TSTORB,CFMD
211. WRITE(6,59)WARD,CWFR,CMA,CMFD,TCM,TSTORB,DFICM,DFSCSM,DFCM
212. READ(1,68)RGFR
213. WRITE(6,905)
214. READ(1,55)RGWFR,RGFS,TRG,TSTORR,DFIRG,DFCSRG,DFRG
215. WRITE(6,71)RGWFR,RGFD,TRG,TSTORR,DFIRG,DFCSRG,DFRG
216. C
217. C READ DATA FOR BWR GAS CODE
218. C
219. WRITE(6,905)
220. READ(1,53)WORD,GGS
221. WRITE(6,55)WORD,GGS
222. READ(1,53)WORD,TIM3
223. WRITE(6,55)WORD,TIM3
224. READ(1,53)WORD,TIM4
225. WRITE(6,53)WORD,TIM4
226. HEPA1=1.0
227. FIL1=1.0
228. HEPA2=1.0
229. FIL2=1.0
230. HEPA5=1.0
231. FIL5=1.0
232. HEPA6=1.0
233. FIL6=1.0
234. FILGS = 1.0
235. FILEJ = 1.0
236. READ(1,925)WURD,CBCH,CBHEPA
237. IF(CBCH.GT.0.0)FIL1 = (1.0 - CBCH/100.0)
238. IF(CBHEPA.GT.0.0)HEPA1 = (1.0 - CBHEPA/100.0)
239. WRITE(6,926)WURD,HEPA1
240. READ(1,927)WURD,TBCB,TBHEPA
IF(TBCH.GT.0.0) FIL2 = (1.0 - TBCH/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)
WRITE(6,53) WORD,FIL4
READC1,53) W0RD,FIL5
IFCFIL3.GT.0.0) FILGS =
WRITEC6,53) WORD,FILGS
READC1,53) W0RD,FIL4
IFCFIL4.EQ.1.0) FILEJ =
IFCFIL4.GT.1.0) FILEJ =
IF(FIL4.EQ.0.0) FILEJ =
WRITE(6,53) WORD,FILEJ
READC1,925) WURD,AXCH,AXHEPA
IFCAXCH.GT.0.0) FIL5 = C1.0 -
IFCAXHEPA.GT.0.0) HEPA5 = C1.0 - AXHEPA/100.)
WRITEC6,926) WURD,FIL5,HEPA5
READC1,925) W0RD,RWCH,RWHEPA
IF(RWCH.GT.0.0) FIL6 = (1.0 - RWCH/100.)
IF(RWHEPA.GT.0.0) HEPA6 = C1.0 - RWHEPA/100.)
WRITE(6,926) WURD,FIL6
READC1,53) WORD,KCHAR
READC1,53) W0RD,KKR
READC1,53) W0RD,KXE
READC1,53) W0RD,KMASS
IFCKXE.LT.1161.) XKAR = 6.4
IFCKXE.GT.1161.) XKAR = 16.0
IFCKCHAR.EQ.O) GO TO 90
IFCKCHAR.EQ.I) 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
CHT2 = 1.8 * (KMASS * KXE)/POWTH
CHT3 = 1.8 * (KMASS * XKAR)/POWTH
WRITE(6,63)CHTI1,CHTI2,CHT3,KMASS
92 CONTINUE
READ(1,53) WORD,PFLAUN
IF(PFLAUN.LE.0.0) WRITE(6,907)
C CONVERSION OF UNITS
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.0.400.1E5) GO TO 210
IF(ABS(GDE-1.3E5).GT.0.200.1E5) GO TO 210
IF(ABS(GTO-1.5E7).GT.0.200.1E7) GO TO 210
IF(FFCDM.GT.0.99) GO TO 210
100001200
100001205
100001210
100001215
100001220
100001225
100001230
100001235
100001240
100001245
100001250
100001255
100001260
100001265
100001270
100001275
100001280
100001285
100001290
100001295
100001300
100001305
100001310
100001315
100001320
100001325
100001330
100001335
100001340
100001345
100001350
100001355
100001360
100001365
100001370
100001375
100001380
100001385
100001390
100001395
100001400
100001405
100001410
100001415
100001420
100001425
100001430
100001435
100001440
100001445
100001450
100001455
100001460
100001465
100001470
100001475
100001480
100001485
100001490
100001495
GO TO 211

RHAL2 = (GDEX0.9 + FFCDM*GTO*PC*0.9)/WL1Q

IOT = 2

211 CONTINUE

C

C CALCULATION OF IODINE RELEASES FROM BLDG. VENTILATION SYSTEMS

C

DO 88 I = 1, 2

CBWR(I) = CBWR(I)

DECONHI(I) = DECONHI(I) * 3600.

EX3I(I) = DECONHI(I) * TIM3

IF(EX3I(I).GT.75.) EX3I(I) = 75.

EX4I(I) = DECONHI(I) * TIM4

IF(EX4I(I).GT.75.) EX4I(I) = 75.

IF(IOT.EQ.1) GO TO 2002

CBWR(I) = CBWR(I) * (111.76 * POWTH/WL1Q) * ((0.4038 + DECONHI(I)) / (1RHAL2 + DECONHI(I)))

C

C CALCULATION OF IODINE RELEASES FROM RX. BLDG. DURING NORMAL OPERATIONS

C

2002 CONTINUE

RBWR(I) = RNI(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

RBWR(SI) = RNSI(I) * CBWR(I)

AUXBLS(I) = RBWRS(I) * 0.9

CBLSI(I) = (CBLNCI(I) + CBLSI(I)) * FIL1

AUXLIC1(I) = (AUXBLNCI(I) + AUXBLSI(I)) * FIL5

C

C CALCULATION OF IODINE RELEASES FROM TURBINE BLDG. DURING OPERATION

C

RBWRT(I) = RNTI(I) * CBWR(I) * PC

RBWRTS(I) = RNTS(I) * CBWR(I) * PC

TBLIC1(I) = (RBWRT(I) + RBWRTS(I)) * FIL2

C

RBWR(I) = RNI(I) * CBWR(I)

RBWRS(I) = RNSI(I) * CBWR(I)

RADBLI(I) = (RBWR(I) + RBWRS(I)) * FIL6

C

IF(CGGS.EQ.0.0) GO TO 87

SGLI(I) = 8.1E-1 * CBWR(I) * FILGS

SGLI(2) = 2.2E-1 * CBWR(2) * FILGS

GO TO 899

87 SGLI(I) = 0.0

SGLI(2) = 0.0

C

99 EXC1I = 0.0

IF(KCHAR.EQ.2) EXC1I = DECONHI(I) * CHTI1 * 24.

IF(EXC1I.GT.75.) EXC1I = 75.

IF(KCHAR.EQ.2) DFCR = 0.00010

EJII(I) = 6.0 * EXP(-EXC1I) * DFCR * EXP(-EXC1I) * FILEJ

EJII(2) = EJII(1) * CBWR(2) / CBWR(1)

RMVP(I) = (RMVMP(I) * CBWR(I) * PC) * 4.0

IF(I.EQ.2) GO TO 899

RMVPS(I) = RMVPS(I) * CBWR(I) * PC
361. \[ RMVPS(2) = RMVPS(I) \times CBWR(I) \times PC \]
362. \[ VPR(2) = RMVPS(I) + RMVPS(I) \]
363. \[ TEST = 0.0001 \]
364. \[ IF(SGLI(I) \leq TEST) SGLI(I) = 0.0 \]
365. \[ IF(EJTJ(I) \leq TEST) EJTJ(I) = 0.0 \]
366. \[ CONTINUE \]
367. \[ MSIG = 1 \]
368. \[ NSIG = 2 \]
369. \[ CALL SIGF2(SGLI) \]
370. \[ CALL SIGF2(EJTJ) \]
371. \[ DO 89 I = 1, 2 \]
372. \[ TOTI(I) = AUXLI(I) + TBLI(I) + SGLI(I) + EJTJ(I) + CBILI(I) + RADBLI(I) + VPRI(T) \]
373. \[ CONTINUE \]
374. \[ WRITE(6,906) \]
375. \[ WRITE(6,903) NAME \]
376. \[ WRITE(6,214) \]
377. \[ WRITE(6,235) \]
378. \[ WRITE(6,214) \]
379. \[ DO 79 I = 1, 2 \]
380. \[ WRITE(6,230) HAL(I), CBWR(I), CBILI(I), TBLI(I), AUXLI(I), RADBLI(I), SGLI(I), EJTJ(I), VPR(I), TOTI(I) \]
381. \[ CBWR(I) = CBWR(I) \]
382. \[ CONTINUE \]
383. \[ WRITE(6,214) \]
384. \[ C \]
385. \[ CALCULATION OF TRITIUM RELEASES \]
386. \[ C \]
387. \[ TRITRP = 0.03 \times P0WTH \]
388. \[ GASH3 = TRITRP \times 0.5 \]
389. \[ TBH3 = GASH3 \times 0.5 \]
390. \[ CBH3 = GASH3 \times 0.5 \]
391. \[ TH3 = TBH3 + CBH3 \]
392. \[ DIV = 10.0 \times INTCALOG10(TBH3) + 1 \]
393. \[ TBH = INT(TBH3/DIV + 0.5) \times DIV \]
394. \[ CBH = INT(CBH3/DIV + 0.5) \times DIV \]
395. \[ TH = INT(TH3/DIV + 0.5) \times DIV \]
396. \[ WRITE(6,960) TBH \]
397. \[ WRITE(6,962) CBH \]
398. \[ WRITE(6,964) TH \]
399. \[ WRITE(6,963) \]
400. \[ C \]
401. \[ CALCULATION OF NOBLE GAS RELEASES \]
402. \[ C \]
403. \[ WRITE(6,906) \]
404. \[ WRITE(6,903) NAME \]
405. \[ WRITE(6,214) \]
406. \[ WRITE(6,235) \]
407. \[ WRITE(6,214) \]
408. \[ DO 5 I = 1, 14 \]
409. \[ DECOHCI) = DECONCI) \times N3600. \]
410. \[ EX3(I) = DECOHH(I) \times TIX3 \]
411. \[ IF(EX3(I) \geq 75.) EX3(I) = 75. \]
412. \[ EX4(I) = DECOHH(I) \times TIX4 \]
413. \[ IF(EX4(I) \geq 75.) EX4(I) = 75. \]
414. \[ X5(I) = XB(I) \]
415. \[ CBILI(I) = CBILI(I) \]
416. \[ AXBILI(I) = AXBILI(I) \]
417. \[ C\]
TBL(I)=TDB(I)
RADB(I)=RADB(I)
SGL(I)=SGS(I)*EXP(-EX3(I))*3.977*OPRA
IF(I.GT.1) GO TO 98
EXCI = DECOH(I) * CHT3 * 24.
GO TO 101

98 IF(CL(I).GT.1.7) EXCI = DECOH(I) * CHT1 * 24.
IF(I.GT.7) EXCI = DECOH(I) * CHT2 * 24.

101 CONTINUE
IF(CL(I).GT.75.) EXCI = 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
EXCI = E3 * EXP(-EX4(I)) * (DFCR + EXP(-EXC(I)));
EXC(I) = DECOH(I) * CHT3 * 24.
GO TO 2001
105 EXC(I) = DECOH(I) * CHT1 * 24.

2001 TEST=1.
IF(SG(L(I)).LE.TEST) SGL(I)=0.0
IF(EJ(T(I)).LE.TEST) EJ(I)=0.0
5 CONTINUE
DO 2003 I=1,14
TOT(I)=AXBL(I)+TBL(I)+SGL(I)+EJ(I)+CBL(I)+RADB(I)+VPR(I)

2003 CONTINUE
MSIG = 2
NSIG = 14
CALL SIGF2(TBL)
CALL SIGF2(GSL)
CALL SIGF2(EJ)
GASTOT=0.0
DO 2004 I=1,14
WRITE(6,230)NUCLID(I),XB(I),CBL(I),TBL(I),AXBL(I),RADB(I),SGL(I),EJ(I),VPR(I),TOT(I)

2004 CONTINUE
DIV=10.**INT(ALOG10(GASTOT)))-1
GASTOT=AINT(GASTOT/DIV+0.5)*DIV
WRITE(6,232) GASTOT
WRITE(6,231)
WRITE(6,230)
WRITE(6,229)
WRITE(6,228)
WRITE(6,227)
WRITE(6,226)
WRITE(6,225)
WRITE(6,224)
WRITE(6,223)
WRITE(6,222)
WRITE(6,221)
WRITE(6,220)
WRITE(6,219)
WRITE(6,218)
WRITE(6,217)
WRITE(6,216)
WRITE(6,215)
WRITE(6,214)
WRITE(6,213)
WRITE(6,212)
WRITE(6,211)
WRITE(6,210)
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WRITE(6,208)
WRITE(6,207)
WRITE(6,206)
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WRITE(6,24)
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WRITE(6,22)
WRITE(6,21)
WRITE(6,20)
WRITE(6,19)
WRITE(6,18)
WRITE(6,17)
WRITE(6,16)
WRITE(6,15)
WRITE(6,14)
WRITE(6,13)
WRITE(6,12)
WRITE(6,11)
WRITE(6,10)
WRITE(6,9)
WRITE(6,8)
WRITE(6,7)
WRITE(6,6)
WRITE(6,5)
WRITE(6,4)
WRITE(6,3)
WRITE(6,2)
WRITE(6,1)
WRITE(6,0)

9 CONTINUE
MSIG = 3
NSIG = 20
CALL SIGF2(TOTB)
DO 10 I=1,19
WRITE(6,920) BPAR1(I),PCBL(I),PTBL(I),PAHBL(I),PRWBL(I),PVPL(I)
10 PTOB1(I)
10 CONTINUE
482. WRITE(6,214)
483. GO TO 80
484. 1003 CONTINUE
485. STOP
486. END
487. COMMON MSIG,NSIG
488. DIMENSION RLPTCNSIG
489. IF(MSIG.EQ.2) GO TO 25
490. IF(MSIG.EQ.3) GO TO 30
491. DO 20 I=1,NSIG
492. C THIS PART OF SUBROUTINE IS FOR IODINE
493. ISUB=2
494. IF(RLPT(I).EQ.0.) GO TO 20
495. IF(RLPT(I).GT.1.)ISUB=1
496. DIV=10.*INT(ALOG10(RLPT(I)))-ISUB
497. RLPT(I)=AINT(RLPT(I)/DIV+0.5)*DIV
498. 20 CONTINUE
499. GO TO 50
500. 25 CONTINUE
501. DO 35 I = 1,NSIG
502. C THIS PART OF SUBROUTINE IS FOR NOBLE GASES
503. IF(RLPT(I).EQ.0.) GO TO 35
504. IF(RLPT(I).LT.1.) DIV=1.0
505. RLPT(I)=AINT(RLPT(I)/DIV+0.5)*DIV
506. 35 CONTINUE
507. GO TO 50
508. 30 CONTINUE
509. DO 40 I = 1,NSIG
510. C THIS PART OF SUBROUTINE IS FOR PARTICULATES
511. IF(RLPT(I).EQ.0.) GO TO 40
512. DIV=10.*INT(ALOG10(RLPT(I)))-2
513. RLPT(I)=AINT(RLPT(I)/DIV+0.5)*DIV
514. 40 CONTINUE
515. RETURN
516. 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 radiiodine released through the system vent. A description of the treatment system used to reduce radiiodine 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 radiiodines, 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


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:

      | 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 ft³. 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 - $10^2$

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 - $2 \times 10^4$
- Others - $10^2$

**Card 10**

Spaces 29-32 "Collection Time." Using the value from Table A-1 of 25,000 gal/ft³, the regeneration frequency would be:

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

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 source 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.