

# **Calculation of Release of Radioactive Materials in Gaseous and Liquid Effluents from Boiling-Water Reactors**

GALE-BWR 3.2 Code

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# **Calculation of Release of Radioactive Materials in Gaseous and Liquid Effluents from Boiling-Water Reactors**

GALE-BWR 3.2 Code

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## ABSTRACT

This report is documentation for the release of the GALE-BWR 3.2 (Gaseous and Liquid Effluents—Boiling-Water Reactor) code. The GALE-BWR 3.2 code is a computerized mathematical model for calculating the releases of radioactive material in gaseous and liquid effluents (i.e., the gaseous and liquid source terms) from nuclear power plants (NPPs). The code is a tool that can be used to determine conformance with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion ‘As Low As Is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents.”

For domestic and international applicants submitting a new reactor application for the review and approval of the U.S. Nuclear Regulatory Commission (NRC), the staff considers use of the following to be an acceptable method for demonstrating compliance with 10 CFR Part 50, Appendix I:

- GALE86 code
- DC/COL-ISG-05, “Interim Staff Guidance on the Use of the GALE86 Code for Calculation of Routine Radioactive Releases in Gaseous and Liquid Effluents from Boiling-Water Reactors and Pressurized-Water Reactors To Support Design Certification and Combined License Applications,” issued July 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081710299)
- Regulatory Guide (RG) 1.112, Revision 1, “Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors,” issued March 2007 (ADAMS Accession No. ML070320241)

When the source term or reactor design parameters differ from those given in the GALE86 code, DC/COL-ISG-05, and RG 1.112, Revision 1, they should be described with sufficient detail, and the basis of the alternate method, model parameters, and assumptions should be provided, to allow the NRC to conduct an independent evaluation.

The purpose of the release of this version of the GALE-BWR code is to comprehensively verify applicability of the current methodology described in NUREG-0016, Revision 1, (ADAMS Accession No. ML091910213), “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code),” issued January 1979, to both the current U.S. BWR facilities and proposed NPP designs. Additionally, the GALE-BWR 3.2 code includes a graphical user interface (GUI) to facilitate easier use of the code.

This report documents the revised GALE-BWR 3.2 code, which is in a self-contained executable and includes installation instructions and a description of the various screen shots of the GALE-BWR 3.2 code GUI.



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## ABBREVIATIONS

ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
Ar	argon
BWR	boiling-water reactor
BWRGE	boiling-water reactor gaseous effluent model
BWRLE	boiling-water reactor liquid effluent model
C	carbon
CFR	<i>Code of Federal Regulations</i>
Ci	curie
cm <sup>3</sup> /g	cubic centimeters per gram
CMMP	configuration management and maintenance plan
Co	cobalt
Cs	cesium
d	day
DCD	design certification document
DF	decontamination factor
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
°F	Fahrenheit
Fortran	Formula Translation (formerly FORTRAN)
FSAR	final safety analysis report
ft <sup>3</sup>	cubic feet
ft <sup>3</sup> /d	cubic feet per day
ft <sup>3</sup> /m	cubic feet per minute
ft <sup>3</sup> /yr	cubic feet per year
gal	gallons
gal/d	gallons per day
gal/min	gallons per minute
GALE	Gaseous and Liquid Effluents computer code
GALE-BWR 3.2	<u>G</u> aseous and <u>L</u> iquid <u>E</u> ffluents-Boiling-Water Reactor computer code version 3.2
GUI	graphical user interface
GWd	gigawatt day
h	hour
H-3	tritium
HEPA	high-efficiency particulate air (filter)
HOI	hypoiodous acid
I	iodine
in.	inch
ISG	interim staff guidance
kg	kilogram
kg/h	kilograms per hour
Kr	krypton
lb	pounds
lb/h	pounds per hour
L/min	liters per minute
LWR	light water reactor

m <sup>3</sup>	cubic meter
m <sup>3</sup> /min	cubic meters per minute
MCOTS	main condenser offgas treatment system
MTU	metric ton of uranium
MWd	megawatt day
MWdt	megawatt day thermal
MWe	megawatt electric
MWt	megawatt thermal
μCi/cm <sup>3</sup>	microcuries per cubic centimeter
μCi/g	microcuries per gram
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NUREG	U.S. Nuclear Regulatory Commission technical report designation
ORNL	Oak Ridge National Laboratory
P	phosphorus
PC	partition coefficient
PCA	primary coolant activity
PNNL	Pacific Northwest National Laboratory
PWR	pressurized-water reactor
Rb	rubidium
RG	regulatory guide
SI	International System of Units
SMR	small modular reactor
SQAP	software quality assurance plan
Sr	strontium
TVA	Tennessee Valley Authority
URC	ultrasonic resin cleaner
Xe	xenon
Zn	zinc



## 1.0 INTRODUCTION

In promulgating Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion ‘As Low As Is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents,” to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities” [Ref. 1], the U.S. Nuclear Regulatory Commission (NRC) indicated its desire for improving the calculation models used by the staff to determine conformance with the requirements of the regulation. To conform to the requirements of Appendix I to 10 CFR Part 50, the NRC developed the GALE-BWR (Gaseous and Liquid Effluents—Boiling-Water Reactor) code as a computerized mathematical model for calculating the releases of radioactive material in gaseous and liquid effluents from boiling-water reactors (BWRs). This original NUREG-0016, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling-Water Reactors (BWR-GALE Code),” issued April 1976 [Ref. 2], and its Revision 1, issued January 1979 [Ref. 3], documented this version of the GALE-BWR code, referred to as GALE-BWR 86 (GALE86).

The NRC staff considers the following as an acceptable method for demonstrating compliance with 10 CFR Part 50, Appendix I:

- GALE86 code
- DC/COL-ISG-05, “Interim Staff Guidance on the Use of the GALE86 Code for Calculation of Routine Radioactive Releases in Gaseous and Liquid Effluents from Boiling-Water Reactors and Pressurized-Water Reactors to Support Design Certification and Combined License Applications,” issued July 2008 [Ref. 4]
- Regulatory Guide (RG) 1.112, Revision 1, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors,” issued March 2007 [Ref. 5]

When the source term or reactor design parameters differ from those given in the GALE86 code, DC/COL-ISG-05, and RG 1.112, Revision 1, they should be described with sufficient detail, and the basis of the alternate method, model parameters, and assumptions should be provided, to allow the staff to conduct an independent evaluation. Additionally, the licensing guidance of RG 1.112, Revision 1, and NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” issued March 2007 [Ref. 6], reference the GALE86 code.

This report describes the GALE-BWR 3.2 code and the work Pacific Northwest National Laboratory (PNNL) performed on the GALE86 code. The GALE-BWR 3.2 code is an update to the GALE86 code, and it maintains direct traceability back to the GALE86 code. The GALE-BWR 3.2 code is available for download from the NRC’s Radiation Protection Computer Code Analysis and Maintenance Program (RAMP) Web site (<https://ramp.nrc-gateway.gov/>).

### 1.1 GALE-BWR 3.2 Code Update

The GALE-BWR 3.2 code version updates the GALE86 version. The GALE86 version of the code based on NUREG-0016, Revision 1, contained two modules, the BWR liquid effluent (BWRLE86) and the BWR gaseous effluent (BWRGE86), based upon the reactor coolant

source term (tables and adjustment equations) described in American National Standards Institute/American Nuclear Society (ANSI/ANS-18.1-1984), "Radioactive Source Term for Normal Operation of Light Water Reactors" [Ref. 7].

The GALE-BWR 3.2 code version maintains the input/output functionality of previous versions, including the Fortran card-based format input file. A new graphical user interface (GUI) function allowing for direct creation of the input file was used to simplify the data input process.

### **1.1.1 Graphical User Interface**

The GALE-BWR 3.2 code was designed and developed to be a flexible tool that provides the same functionality as the GALE86 code with additional features available to the user. In the original version of the GALE86 code, the user was required to enter data using the Fortran card-based format. A Visual Basic GUI, which was developed for the GALE-BWR 3.2 code, allows for easier input of plant-specific parameters and is described in Section 3.0. Additionally, with the development of the GALE-BWR 3.2 GUI, other features were now easier to add to the original GALE86 code. These additional features include the options to select the specific reactor coolant source term and the option to modify some of the GALE86 modeling parameters using the BWR fixed parameter text file.

In GALE-BWR 3.2, the values for the reactor coolant source term and GALE-BWR fixed modeling parameters are defaulted to the ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors" [Ref. 8] source term values and the NUREG-0016, Revision 1, GALE86 code fixed modeling parameters. These GALE-PWR 3.2 default values are consistent with the guidance in DC/COL-ISG-05; RG 1.112, Revision 1; and NUREG-0800. Additionally, because the updates to the GALE-BWR 3.2 source codes did not involve changes in the GALE86 model formulations, the revised source code had the same functionality as the previous version with differences in the outputs reflecting only reactor coolant source term and the GALE-BWR fixed modeling parameter options available to the user in GALE-BWR 3.2.

### **1.1.2 Reactor Coolant Source Term (ANS-18.1 Version) Options**

As stated above, GALE-BWR 3.2 gives the user the option of selecting the reactor coolant source term (ANS-18.1 Version) desired for the GALE-BWR 3.2 calculations. The GALE-BWR 3.2 code provides three options for the reactor coolant source terms, including the default option of ANSI/ANS-18.1-1999, which is consistent with the guidance in DC/COL-ISG-05; RG 1.112, Revision 1; and NUREG-0800. Section 4.2.1 describes the parameters used by GALE-BWR 3.2 when the default reactor coolant source term option is selected.

The other reactor coolant source term options available in the GALE-BWR 3.2 code are ANSI/ANS-18.1-1984 and ANSI/ANS-18.1-2016, "Radioactive Source Term for Normal Operation of Light Water Reactors," [Ref. 9]. Section 4.2.2 describes the parameters used by GALE-BWR 3.2 when the ANSI/ANS-18.1-1984 reactor coolant source term option is selected. Section 4.2.3 describes the parameters used by GALE-BWR 3.2 when the ANSI/ANS-18.1-2016 reactor coolant source term option is selected. When the source term or reactor design parameters differ from those given in the GALE86 code, DC/COL-ISG-05, and RG 1.112, Revision 1, they should be described with sufficient detail, and the basis of the alternate method, model parameters, and assumptions should be provided, to allow the NRC to conduct an independent evaluation.

### **1.1.3 GALE-BWR Fixed Modeling Parameters (GALE Version) Options**

The GALE-BWR 3.2 includes the fixed modeling parameters from the GALE86 code, which is described in NUREG-0016, Revision 1, and is consistent with the guidance in DC/COL-ISG-05 and RG 1.112, Revision 1. Section 4.1.1 describes the parameters used by GALE-BWR 3.2 with the GALE86 fixed modeling parameters.

The GALE-BWR 3.2 code also comes with a BWR fixed parameter text file, which allows the entry of user-defined values for certain GALE-BWR fixed modeling parameters by means of a text file. Section 4.3 describes the use of the BWRfixed-parameter.txt file. Finally, the GALE-BWR 3.2 code output files have been modified to identify any changes made to the default GALE-BWR 3.2 inputs (GALE86 and ANSI/ANS-18.1).

## **1.2 Software Quality Assurance and Configuration Management Plans**

In addition to the updates described above, PNNL developed a GALE software quality assurance plan (SQAP), which is documented in PNNL-24249, "Software Quality Assurance Plan: Support for the Gaseous and Liquid Effluent (GALE) Computer Code Project," issued April 2015 [Ref. 10], and a GALE code configuration management and maintenance plan (CMMP), documented in PNNL-24250, "Support for the Gaseous and Liquid Effluent (GALE) Computer Code Project: Configuration Management & Maintenance Plan," issued April 2015 [Ref. 11]. The NRC has defined three levels of software, per NUREG/BR-0167, "Software Quality Assurance Program and Guidelines," issued February 1993 [Ref. 12]:

- (1) Level 1—technical application software used in a safety decision by the NRC
- (2) Level 2—technical or nontechnical application software not used in a safety decision by the NRC
- (3) Level 3—technical or nontechnical application software not used in a safety decision and having local or limited use by the NRC

The quality assurance documents discussed above are written to conform to the Level 2 requirements. Code development on GALE-BWR 3.2 has proceeded under SQAP and code CMMP. PNNL-28818, "GALE-3.2 Verification Report," issued June 2019 [Ref. 13], describes the work done to verify that the new coding was implemented properly.



## 2.0 DEFINITIONS AND TERMS

This section describes the GALE-BWR 3.2 code source computations that estimate airborne radionuclide emission rates and waterborne radionuclide effluent rates. It also contains guidance for defining these input parameters based on specific plant characteristics.

### 2.1 Code Source Computation

The average quantity of radioactive material released to the environment from a nuclear power reactor during normal operation, including AOOs, is called the "source term." The calculations performed by the GALE-BWR 3.2 code using the default values are based on (1) ANSI/ANS-18.1-1999 tables and adjustment factors, (2) the release and transport mechanisms that result in the appearance of radioactive material in liquid and gaseous waste streams, (3) plant-specific design features used to reduce the quantities of radioactive materials ultimately released to the environment, and (4) NUREG-0016, Revision 1, information received on the operation of nuclear power plants (NPPs).

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 because of reactor coolant leakage. The following paragraphs briefly describe these mechanisms.

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 side stream 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 involves holdup,

which permits decay of the short-lived noble and activation gases before they are released to the environment. Clean (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 necessary to meet the design objectives of Appendix I to 10 CFR Part 50, the effluent from the mechanical vacuum pump could be processed through charcoal adsorbers for removal of radioiodine before release to the environment.

In addition to the above release points, GALE-BWR 3.2 considers ventilation system releases from the turbine, containment, auxiliary (including the spent fuel pool area), and radwaste buildings caused by 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, radioiodine, and particulates that are released directly to the building atmosphere. In some cases, leakage may be reduced by special design features, such as vacuum leadoff drains or “clean” steam on the valve bonnets, in addition to normal precautions such as back-seating valves and using all-welded systems. The use of closed leadoff drains and increased maintenance can also reduce leakage.

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.

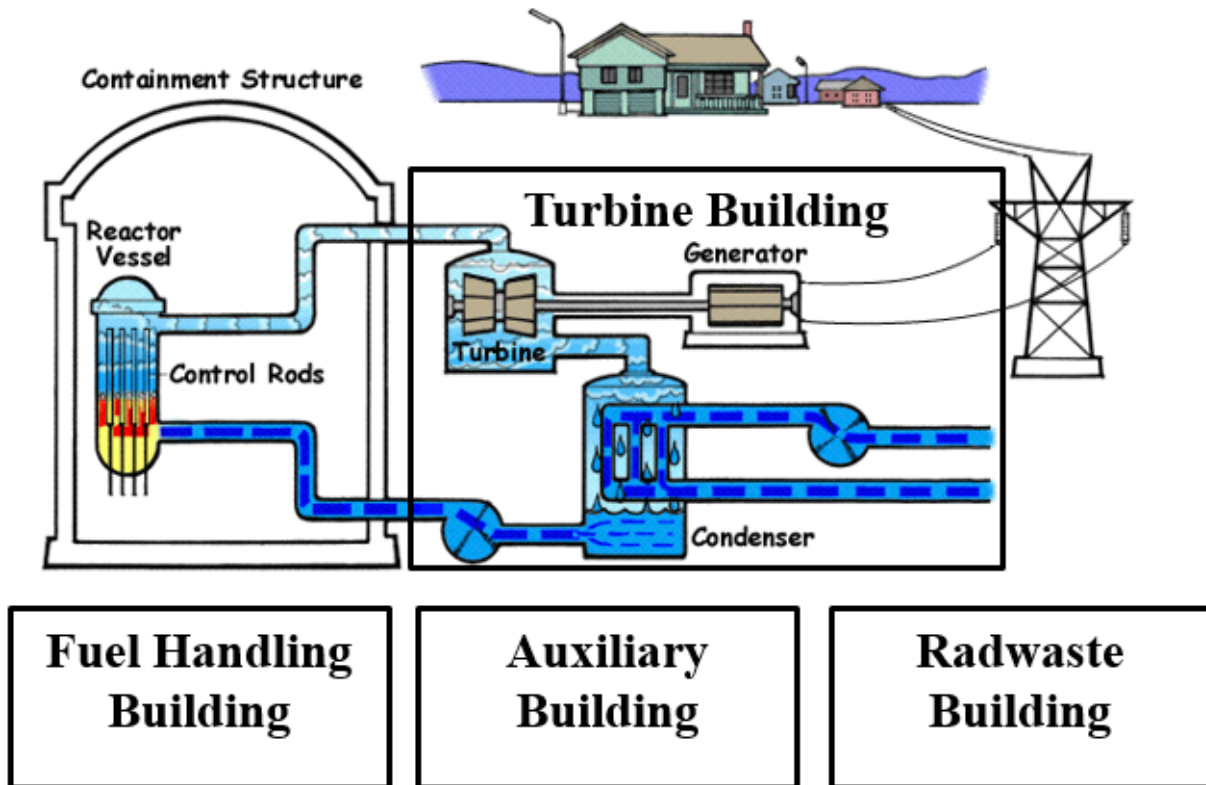
This document describes both the plant-specific input parameters and code-defined parameters that are used in the BWR source term calculations. The plant-specific input parameters are values that are used in the models that depend on the design and operations of the facility. The document explains the input data required and acceptable means for obtaining these data. Section 4 addresses the required source term parameters developed for use with the GALE-BWR 3.2 code and explain the basis for each parameter, documentation of the GALE-BWR 3.2 code models (i.e., computer programs), and information needed to generate source terms that an applicant is required to submit with an application. Data input and output files are provided for base-case runs of the liquid and gaseous effluent computer programs.

## **2.2 Definitions**

### **2.2.1 Plant Structures**

The GALE-BWR 3.2 code release rates are based on the sum of estimated annual emissions in different structural components of the plant. Figure 2-1 is a schematic diagram showing the containment structure (reactor), containment building, turbine building, auxiliary building, radwaste building, and fuel handling building. This building nomenclature represents the typical names of the buildings in most current BWRs; however, different BWR designs may use different terminology and different building names (i.e., compound building, fuel building, and reactor building). It may be necessary to compare the functions of the building in a specific design to those described in this section, especially since the outputs will be tailored to the

buildings described below. As noted in the following discussion, these structures have various sources of radionuclide releases during both operational and shutdown activities.



**Figure 2-1 Principal structures at an operating BWR nuclear power generation station**

The main buildings are described below.

Containment or Drywell Building: A building that houses the reactor and the related cooling system that contains highly radioactive fluids. The building is of steel construction and, in some designs, is surrounded by a concrete structure that is designed for much lower pressure. The area between the steel and concrete building is called the annulus.

Fuel Handling Building: A building separate from the containment that is used to store spent fuel assemblies in steel racks in a large deep storage pool. Casks for shipping or onsite dry storage of spent fuel assemblies will be loaded (or unloaded) from this pool. A new fuel storage area is provided for receipt of new assemblies and storage before they go into the containment and subsequently into the reactor during refueling.

Auxiliary Building: A building separate from the containment that houses much of the support equipment that may contain radioactive liquids and gases. Emergency equipment is also normally located in this building.

Turbine Building: A building that houses the turbine, generator, condenser, and condensate and feedwater systems. Some BWRs in the United States have a structure without the traditional roof and walls.

**Radwaste Building:** A building that houses various systems to process liquid, solid, and gaseous radioactive wastes generated by the plant. In some BWR designs, the turbine building and radwaste building may house shared components of the offgas-recombiner system.

### 2.2.2 Terminology

The following definitions apply to terms used in this report:

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

**Anticipated Operational Occurrence (AOO):** Unplanned releases of radioactive materials from miscellaneous actions, such as equipment failure, operator error, and administrative error, that are not of consequence to be considered an accident. Note that, in the context of the GALE-BWR 3.2 code, this definition is different than the one in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50.

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

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

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

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

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

**Effective Full-Power Days:** The number of days a plant would have to operate at 100-percent licensed power to produce the integrated thermal power output during a calendar year; that is:

$$\text{Effective Full Power Days} = \frac{\text{Integrated Thermal Power}}{\text{Licensed Power Level}} = \frac{\sum P_i T_i}{P_{\text{total}}} \quad (2-1)$$

where  $P_i$  = the  $i^{\text{th}}$  power level, in megawatt thermal (MWt);  
 $P_{\text{total}}$  = the licensed power level, in MWt; and  
 $T_i$  = the time of operation at power level  $P_i$ , in days.

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

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



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

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

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

**Partition Coefficient (PC):** The ratio of the concentration of a radionuclide in the gas phase to the concentration of a radionuclide 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 radionuclides of fluorine, chlorine, bromine, and radioiodine. The radioactive radionuclides of radioiodine are the key radionuclides considered in dose calculations.

**Radioactive Noble Gases:** The GALE-BWR 3.2 code focuses on the radioactive radionuclides of argon, krypton, and xenon, which are characterized by their chemical inactivity. These are the noble gases that are generated by a nuclear reactor or those that are precursors to the production of specific radionuclides. The radioactive radionuclides of argon, krypton, and xenon are the key elements 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 default values for the reactor coolant and main steam are based on ANSI/ANS-18.1-1999, but the GALE-BWR 3.2 code provides additional options to allow the user to select ANSI/ANS-18.1-1984 or ANSI/ANS-18.1-2016. In accordance with the recommendations of the standard, the GALE-BWR 3.2 code provides for adjusting reactor coolant concentrations, should the plant be designed to parameters that are outside the ranges considered in the standard. The radionuclides are divided into the following categories:

- noble gases
- halogens (bromine, radioiodine)
- cesium and rubidium
- water activation products
- tritium
- other radionuclides (as listed in GALE-BWR 3.2 code-liquid effluent model output files)

**Regenerate Solutions Waste:** The liquid solution from ion exchange columns or condensate polishers that are part of the demineralizer system.

**Source Term:** The calculated average quantity of radioactive material released to the environment from a nuclear power reactor during normal operation, including AOOs. 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 refueling.

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

### **2.3 Gaseous Source Terms**

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

- main condenser offgas system
- turbine gland sealing system
- mechanical vacuum pumps
- gases and vapors from the venting of tanks and components
- 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. Gaseous effluent releases from BWRs may be released either from a single stack for the entire plant or via multiple stacks and vents by specific buildings, depending on the design. The radioactive particulate and noble gas release rates are specified in the GALE-BWR 3.2 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.

Later sections provide radioiodine and particulate DFs for removal equipment and parameters for calculating holdup times for noble gases and for calculating tritium, argon-41, and carbon-14 releases.

### **2.4 Liquid Source Terms**

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

- processed liquid wastes from the high-purity waste system
- processed liquid wastes from the low-purity waste system
- processed liquid waste wastes from the chemical waste system
- processed liquid regenerant wastes
- detergent wastes

The radioactivity input to the liquid radwaste treatment system is based on the flow rates of the liquid waste streams and their radioactivity levels, expressed as a fraction of the primary coolant activity (PCA). The default values for the PCAs are based on the recommendations of ANSI/ANS-18.1-1999; however, these values can also come from the ANSI/ANS-18.1 version (1984, 1999, or 2016) selected by the user.

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

- decay during collection and processing
- removal by the proposed treatment systems (e.g., filtration, ion exchange, evaporation, reverse osmosis, plateout)

For BWRs using a deep-bed condensate demineralizer, the inventory of radionuclides collected on the demineralizer resins is calculated by considering the flow rate of condensate at main steam activity that is processed through the demineralizers and radionuclide removal using the DFs given in Sections 4.1.1 and 4.1.2. The activity on the condensate demineralizer resins will also include the steam generator blowdown activity if the blowdown is recycled to the condensate demineralizers. The radioactivity content of the demineralizer regenerant solution is obtained by considering that all radioactivity is removed from the resins at the interval dictated by the regeneration frequency.

Sections 4.1.1 and 4.1.2 contain the methods for calculating collection and processing times and the DFs for radwaste treatment equipment. The liquid radioactive source terms are adjusted to compensate for equipment downtime and AOOs.

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

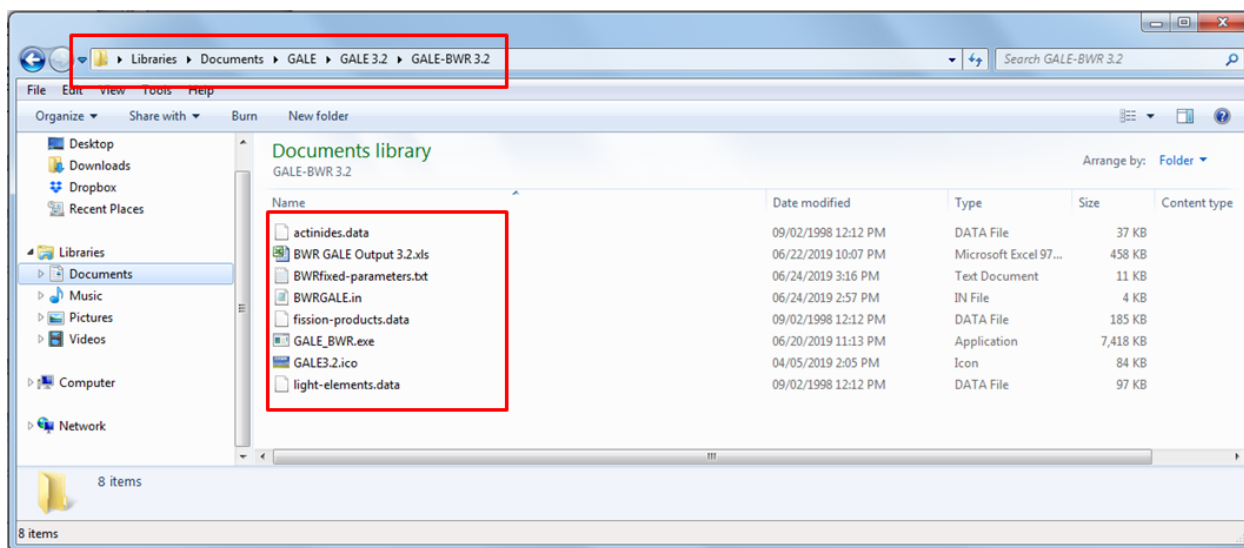


## 3.0 INSTALLATION AND USE

This section describes the GALE-BWR 3.2 executable and data files required for operation. Additionally, this section contains information on how to install the code, how to run the code executable file, and how to view the code output files.

### 3.1 Code Installation

GALE-BWR 3.2 is a Fortran-based computer code used to calculate the effluents released from a BWR during normal operations. GALE-BWR 3.2 can be installed on any computer running Windows operating systems 7.0, 8.0, and 10.0. To install the code, the user should copy the three data files (i.e., `actinides.data`, `fission-products.data`, and `light-elements.data`), the existing BWR input file, the BWR fixed parameter text file, the BWR output Excel file, and the executable file to the computer and set up a working directory, as shown in Figure 3-1.



**Figure 3-1** Windows Explorer directory with the GALE-BWR 3.2 files

The names and descriptions of the GALE-BWR 3.2 files to be copied to the working directory are given below:

- `actinides.data`—This data file is required to calculate actinide radionuclides. The actinides have atomic numbers from 89 to 103 and represent the heavy elements in the fuel. Because of radioactive decay of the actinides, the data file contains radionuclides with atomic numbers from 81 to 99.
- `fission-products.data`—This data file is required to calculate fission products. These radionuclides have atomic numbers from 1 to 68 and represent the radionuclides created during the fission of the actinide fuel.
- `light-elements.data`—Similar to the fission product file, this data file is required to calculate other elements in the reactor. This can include impurities in the fuel and other materials associated with the reactor. The atomic numbers range from 1 to 84 in this data file.

- BWRGALE.in—This is sample input for gaseous and liquid effluents from BWRs. This file, which comes with the GALE-BWR 3.2 code download, is a sample BWR input that provides the user with default values.
- BWRfixed-parameters.txt—This text file allows the user to change selected fixed modeling GALE-BWR parameters. This file can be opened and edited in any text editor program (e.g., NotePad) by the user. The text file should start with “**\$user**” and end with “**\$end**” and comments can be included after any “!” The default values in this text file are set to ANSI/ANS-18.1-1984 and the GALE86 fixed modeling parameters in NUREG-0016, Revision 1.
- BWR GALE Output 3.2.xls—This is an Excel file to read and display GALE-BWR 3.2 output from BWRs.
- GALE\_BWR.exe—This is the executable file that opens up the GALE-BWR 3.2 GUI. This program allows for the generation of plant-specific inputs and the execution of the code.
- GALE3.2.ico—This file provides the user with an icon to attach to a desktop shortcut. The user can create a desktop icon for GALE-BWR 3.2 by right clicking on the GALE\_BWR.exe file and selecting the option “Create shortcut.” The shortcut can then be placed on the desktop, and the icon file can be added by right clicking the desktop shortcut and selecting the properties option on the dropdown menu. Select the “Change Icon” button in the properties window and navigate to the GALE3.2.ico file shown in Figure 3-1.

**Note:** It is important to include all of the data files (.data extensions listed in Section 3.1) in the same directory as the input file. Otherwise, the code will give the user an error message stating that it cannot locate the .data files and will terminate the run.

## **3.2 Code Execution and Use**

To execute the GALE-BWR 3.2 code, either double click on the executable file, GALE\_BWR.exe, or the desktop shortcut icon and the Introductory Screen for code will open as shown in Figure 3-2.

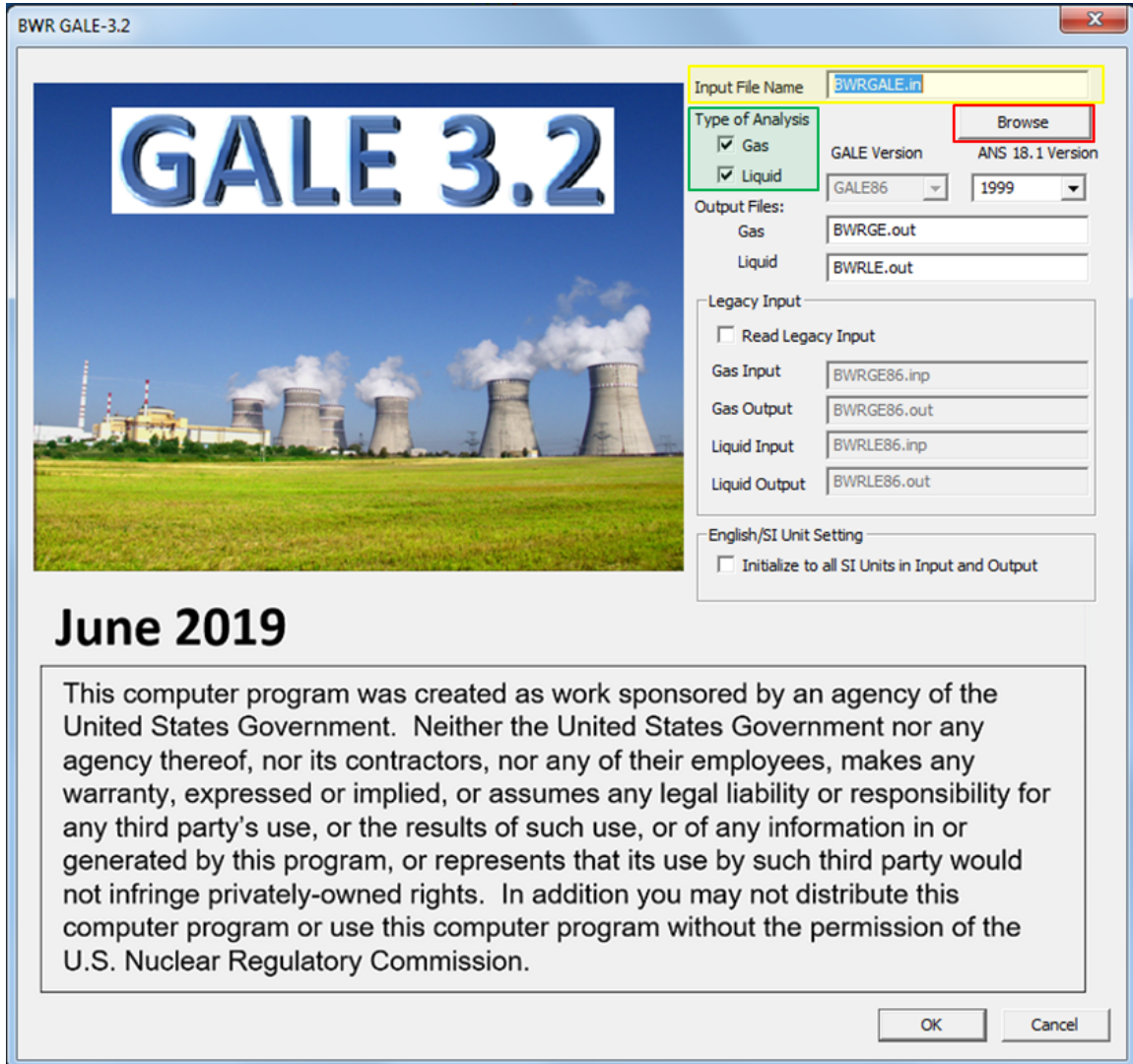
### **3.2.1 Introductory GUI Inputs**

The Introductory Screen, Figure 3-2, is the GUI screen from which the user selects the name of the input file to be used; the type of analysis to be performed (gaseous, liquid, or both); the GALE version and ANS-18.1 version options to be used; the name of the gaseous and liquid output files; and whether to use legacy inputs in the run.

#### **3.2.1.1 *Input File Name Field***

On the Introductory Screen, the user should specify the name of the input file to be used for the run. GALE-BWR 3.2 comes with a sample case, the BWRGALE.in file, which appears as the default value for Input File Name. Sections 3.2.2 through 3.2.4 give the plant-specific parameter values for this file. The user can also browse for an existing input file that has previously been created by selecting the “Browse” button in Figure 3-2. This button opens Windows Explorer to the default working directory, which contains the GALE-BWR 3.2 executable file (Figure 3-1), but the user can navigate to the directory that contains the input file of interest. If the existing

GALE-BWR 3.2 input file is not located in the default working directory noted in Figure 3-1, then all of the data files (.data file extensions listed in Section 3.1) should be in the same directory as the input file. Otherwise, the code will give the user an error message stating that it cannot locate the data files and will terminate the run.



**Figure 3-2**    **Introductory Screen**

### 3.2.1.2 *Type of Analysis Section*

This section of the Introductory Screen allows the user the option of selecting the Type of Analysis to be performed by choosing either the “Gas,” “Liquid,” or both radio buttons. The default settings, as shown in Figure 3-2, are that both radio buttons are checked.

### 3.2.1.3 GALE Version and ANS-18.1 Version Options

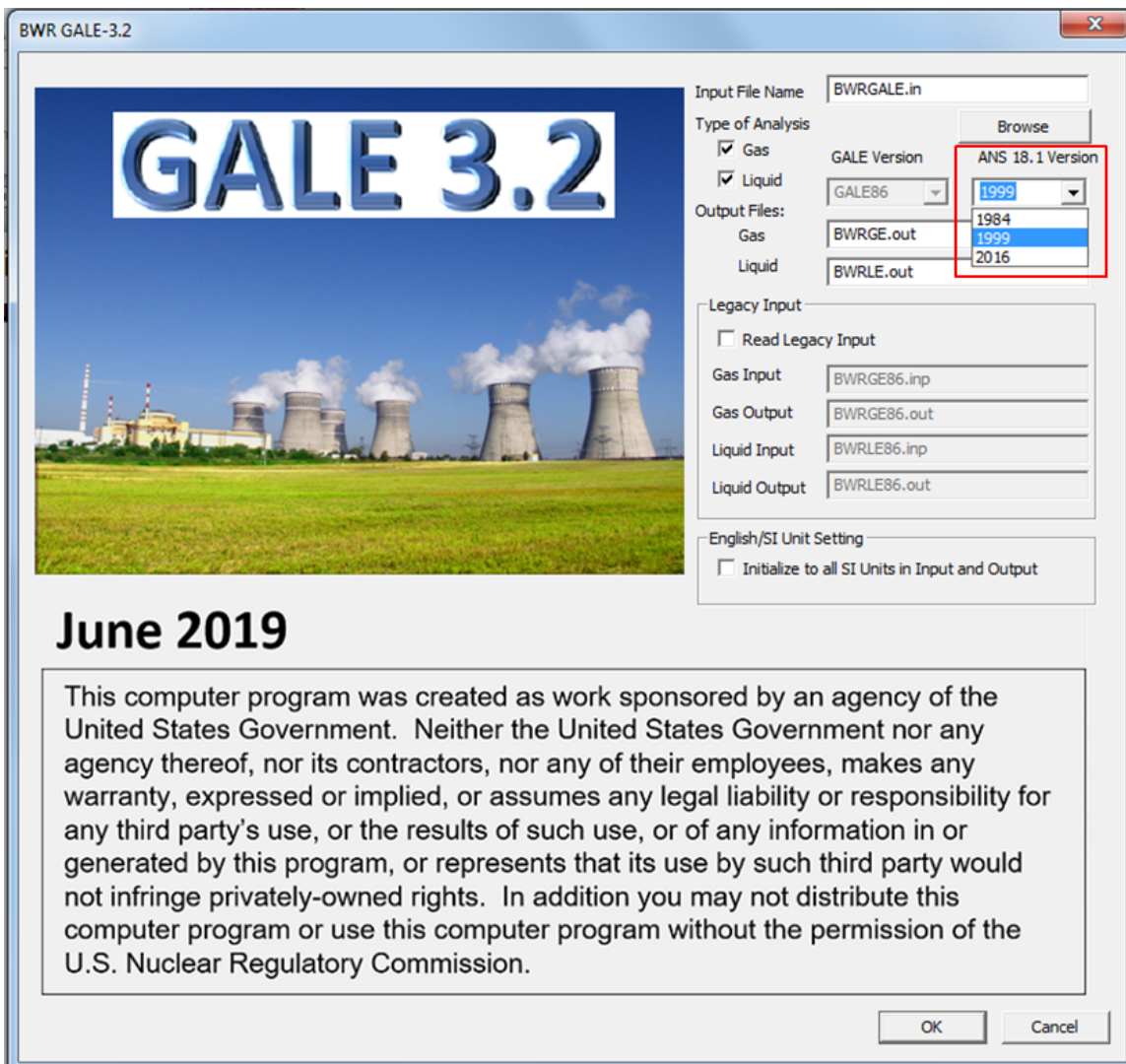
Figures 3-3 and 3-4 show the GALE version options and the ANS-18.1 version options available to the user. The user also has the option of manually changing the certain fixed modeling parameters through the use of the BWRfixed-parameters.txt file listed in Figure 3-1 and discussed in Section 4.3. Section 4.0 discusses in detail the GALE version option (GALE86), the ANS-18.1 version options, and the BWRfixed-parameters.txt file in greater detail.

**Note:** It is important for the user to recognize that the default values for the GALE version option and the ANS-18.1 version option are the **GALE86** code and **ANSI/ANS-18.1-1999**, which are consistent with the guidance in DC/COL-ISG-05; RG 1.112, Revision 1; and NUREG-0800.



Figure 3-3 Introductory Screen—GALE version options

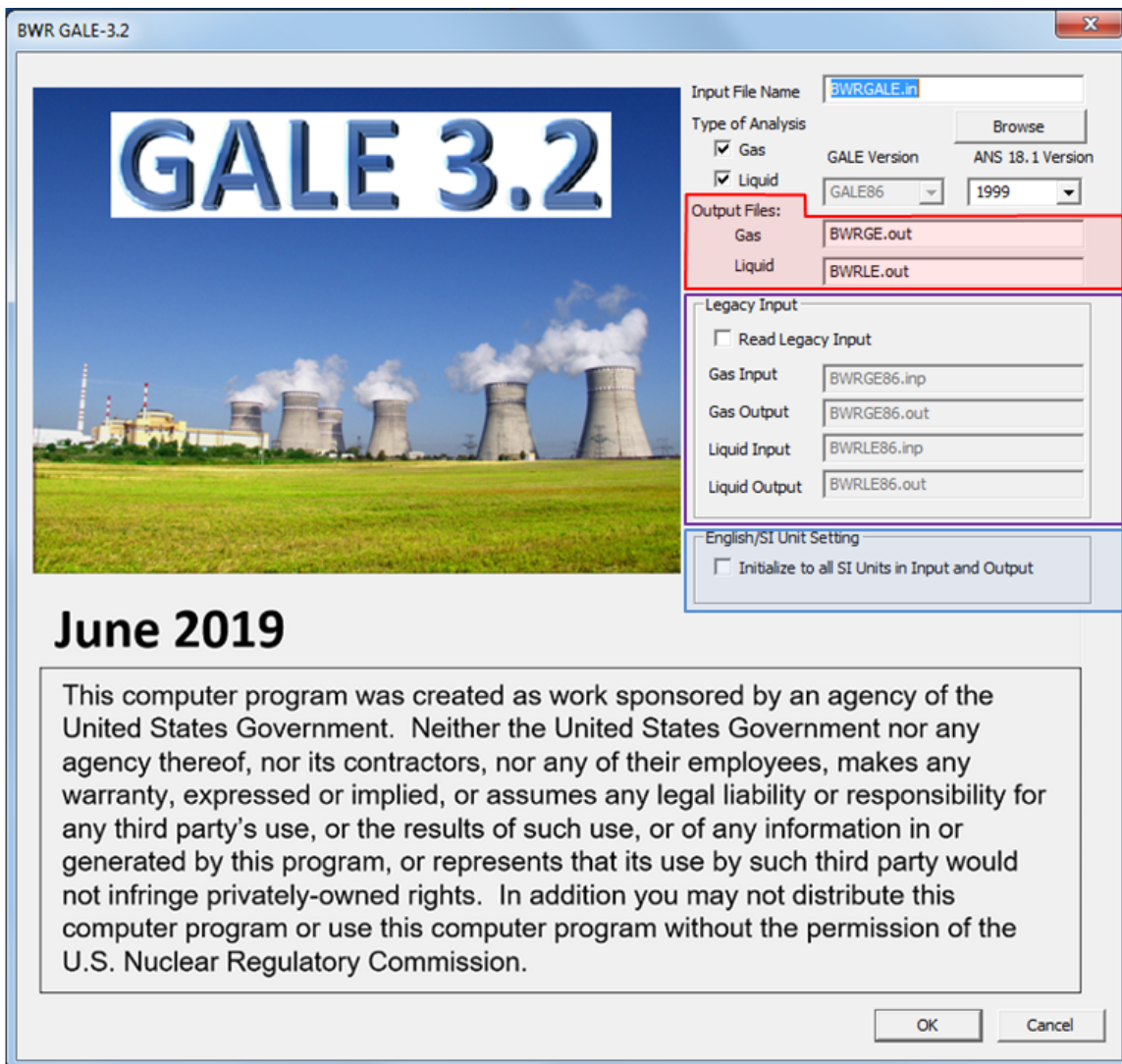




**Figure 3-4 Introductory Screen—ANS-18.1 version options**

The default ANSI/ANS-18.1-1999 reactor coolant source term option is consistent with the guidance in DC/COL-ISG-05; RG 1.112, Revision 1; and NUREG-0800.

The Introductory Screen contains three additional sections that give the user options to control the Output files, Legacy Input, and English/SI Unit Setting for the code, as shown in Figure 3-5.



**Figure 3-5 Introductory Screen—file output/input and unit options**

#### 3.2.1.4 *Output Files Section*

The Output Files section provides the user with the option of choosing the file names for the output files. The default names for output files are BWRGE.out and BWRLE.out, which will be automatically written to the working directory described in Section 3.1 as shown in Figure 3-1.

#### 3.2.1.5 *Legacy Input Section*

The Legacy Input section allows the user to read in legacy GALE-BWR input files (e.g., GALE86 input files) by selecting the “Read Legacy File” radio button. When the “Read Legacy File” radio button is selected, the four input file fields in this section become active and used. As shown in Figure 3-5, the default setting for the “Read Legacy File” radio button is unchecked with the four input file fields “greyed out” and prepopulated with the BWRGE86.inp, BWRGE86.out, BWRLE86.inp, and BWRLE86.out in the respective fields. The user should note that, when

using this option, the legacy files should be located in the working directory described in Section 3.1 and shown in Figure 3-1.

### 3.2.1.6 English/SI Unit Setting Section

This section provides the user with the option of changing to the input and output units used in the code. When the user selects the “Initialize to all SI Units in Input and Output” radio button, all of the GALE-BWR inputs and outputs will be in the International System of Units (SI). As shown in Figure 3-5, the default setting for the “Initialize to all SI Units in Input and Output” radio button is unchecked.

Once the input and output files, the GALE version, and the ANS-18.1 version are specified, the user should click the “OK” button to enter the plant-specific parameters for GALE-BWR 3.2 inputs. These input plant-specific parameters are entered via three GUI input screens: (1) General Reactor Parameters GUI, (2) Liquid GUI Inputs (Liquid Radwaste Treatment System Input Screen), and (3) Gaseous GUI Inputs (Gaseous Radwaste Treatment System Input Screen).

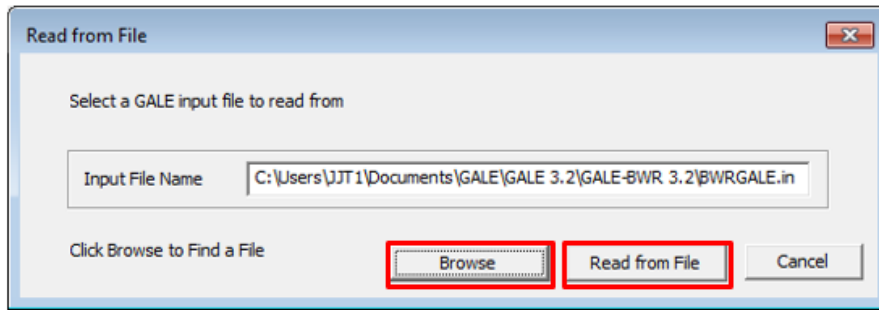
### 3.2.2 General Reactor Parameters GUI Inputs

This section identifies the plant-specific parameters required and the acceptable ranges of those parameters, if applicable, to be entered in the General Reactor Parameters GUI shown in Figure 3-6. Users are able to read values in from previous input files entered on the Introductory Screen, as well as use the values identified in the sample case in the BWRGALE.in file discussed above. If the user has previously specified an existing input file, the user should click the “Read from File” button to open the Read from File Screen shown in Figure 3-7.

Parameter	Unit	Range Button
Name of Reactor		Range
Thermal Power Level	MW(th)	Range
Total Steam Flow	million lb/hr	Range
Mass of Water in Reactor Vessel	million lb	Range
Cleanup Demineralizer Flow	million lb/hr	Range
Condensate demineralizer regeneration time	days	Range
Fraction of feedwater through condensate demineralizers	fraction	Range

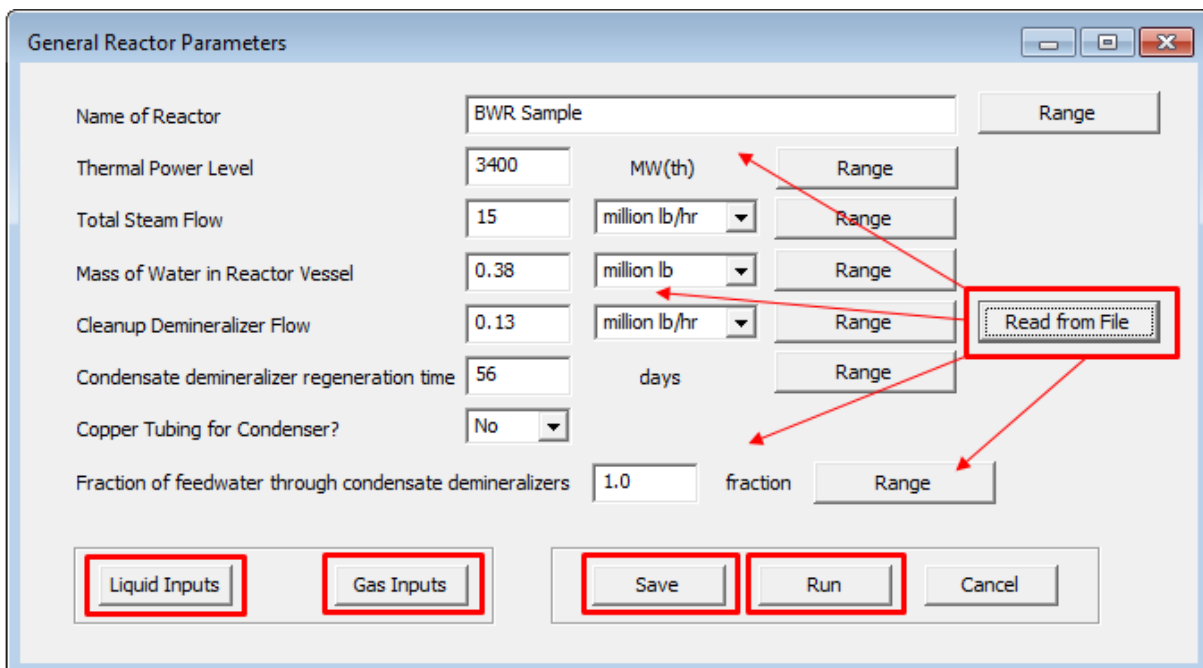
Buttons: Liquid Inputs, Gas Inputs, Save, Run, Cancel, Read from File

Figure 3-6 General Reactor Parameters Screen



**Figure 3-7 Read from File Screen**

Select the “Browse” button to open a Windows Explorer Screen and navigate to the location of the input file in the working directory described in Section 3.1 and shown in Figure 3-1. Select the BWRGALE.in file on the Windows Explorer Screen and then select the “Read from File” button in Figure 3-7 to populate the input field boxes with values from the BWRGALE.in file. These values may be changed and will be updated in the input file once it is run. Figure 3-8 show the General Reactor Parameters Screen after the “Read from File” button has been selected with the values from the BWRGALE.in file. The General Reactors Parameter Screen also contains two option buttons regarding the execution of the GALE-BWR 3.2 code. As shown in Figure 3-8, the user has the option of either saving any changes made to the inputs on this screen by selecting the “Save” button or executing the GALE calculations by selecting the “Run” button. The user should note that selecting the “Run” button without first selecting the “Save” button will result in the code executing without saving the changes made to the inputs on this screen. Therefore, if the user wishes to save any changes made to the inputs, the “Save” button should be selected before selecting the “Run” button.



**Figure 3-8 General Reactor Parameters Screen with default values from the BWRGALE.in file**

Figure 3-9 shows the basic structure of a BWR. This figure should be used in reference to the items described in the following sections on the general reactor parameters.

### 3.2.2.1 Name of Reactor

Enter the name of the reactor for which the analysis is performed. Selecting the “Range” button next to this input field will open the Name of Reactor Allowable Range Screen shown in Figure 3-10. This input field is a text field with a character limit of 120. The default value from the BWRGALE.in file is “**Sample BWR.**”

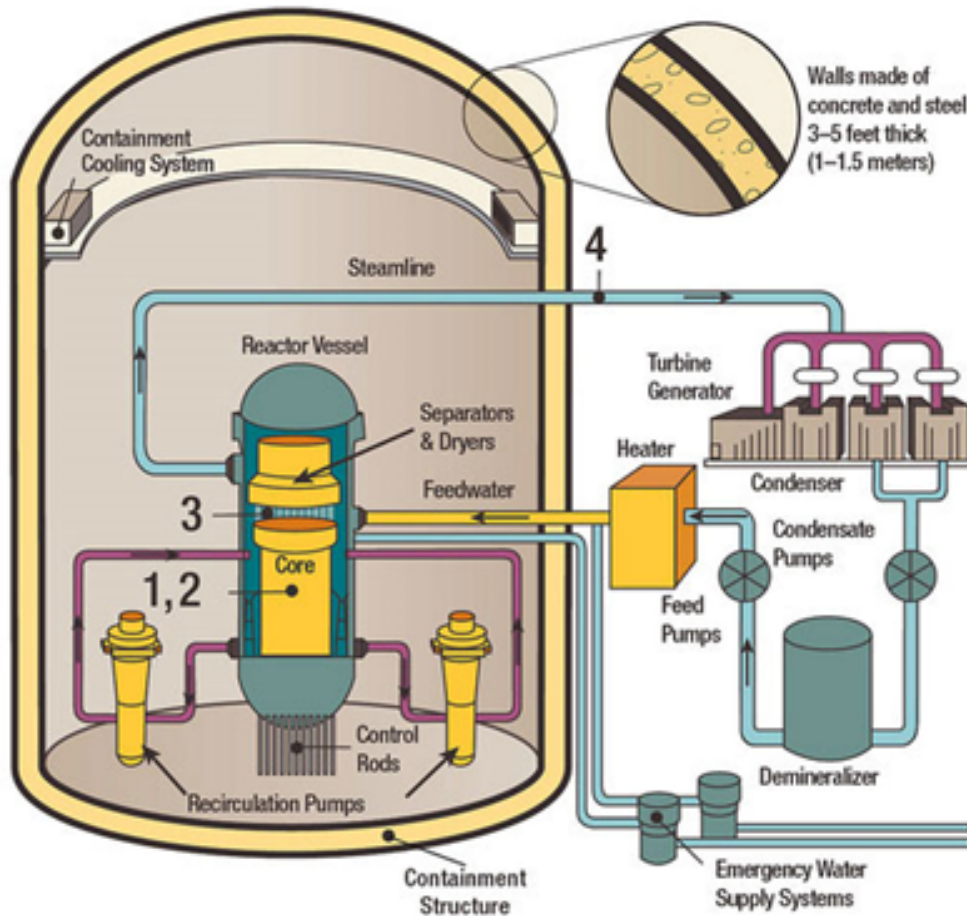


Figure 3-9 Basic structure of a BWR

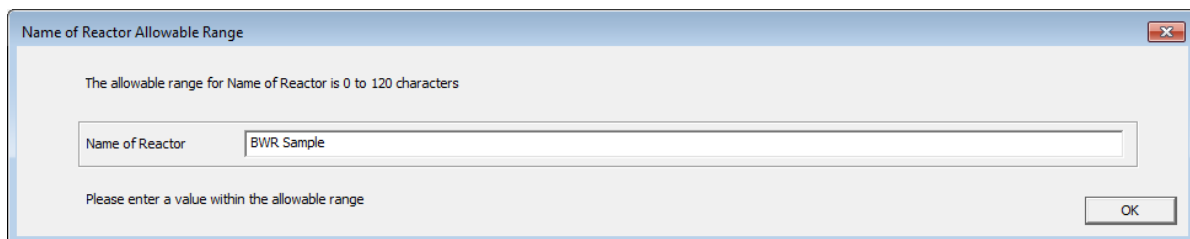
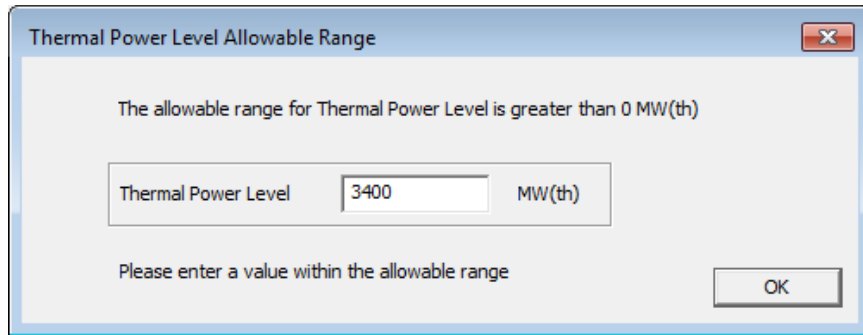


Figure 3-10 Name of Reactor Allowable Range Screen



### 3.2.2.2 Thermal Power Level

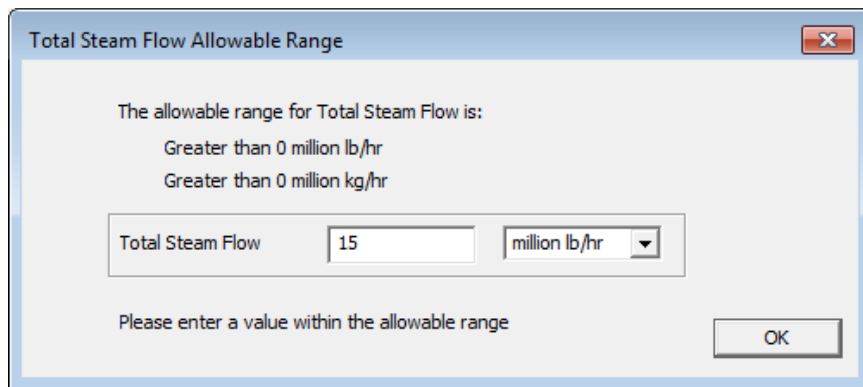
Enter the maximum thermal power level (MWt) evaluated for safety considerations for the plant from the Design Certification Document (DCD), Final Safety Analysis Report (FSAR), or other plant design documentation. Selecting the “Range” button next to this input field will open the Thermal Power Level Allowable Range Screen shown in Figure 3-11. The default value from the BWRGALE.in file is “3400” MWt, and the allowable range for values in this field is greater than 0 MWt.



**Figure 3-11 Thermal Power Level Allowable Range Screen**

### 3.2.2.3 Total Steam Flow

In a BWR, water boils in the reactor core, and steam exits the top of the reactor vessel. Enter the total steam flow rate from the reactor in either units of millions of pounds per hour (million lb/h) or millions of kilograms per hour (million kg/h). Selecting the “Range” button next to this input field will open the Total Steam Flow Allowable Range Screen shown in Figure 3-12. The default value from the BWRGALE.in file is “15” million lb/h, and the allowable range for values in this field is greater than either 0 million lb/h or 0 million kg/h.

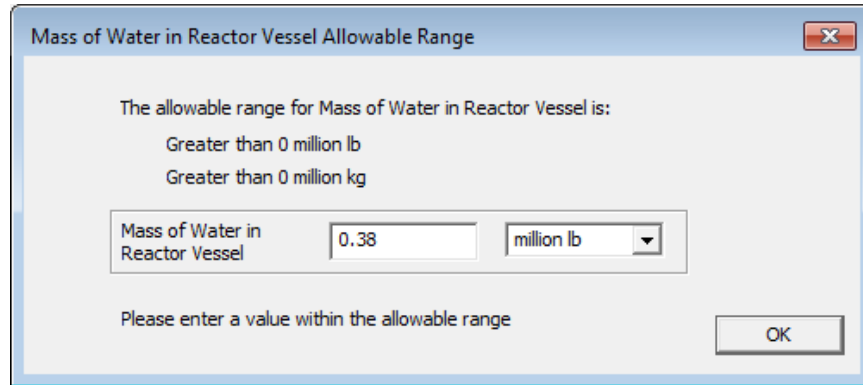


**Figure 3-12 Total Steam Flow Allowable Range Screen**

### 3.2.2.4 Mass of Water in Reactor Vessel

Figure 3-9 shows the reactor vessel. A fixed quantity of water exists as liquid and vapor in the reactor vessel and the recirculation lines. Enter the mass of water in the reactor vessel and recirculation lines in units of either millions of pounds (million lb) or millions of kilograms (million kg). Selecting the “Range” button next to this input field will open the Mass of Water in

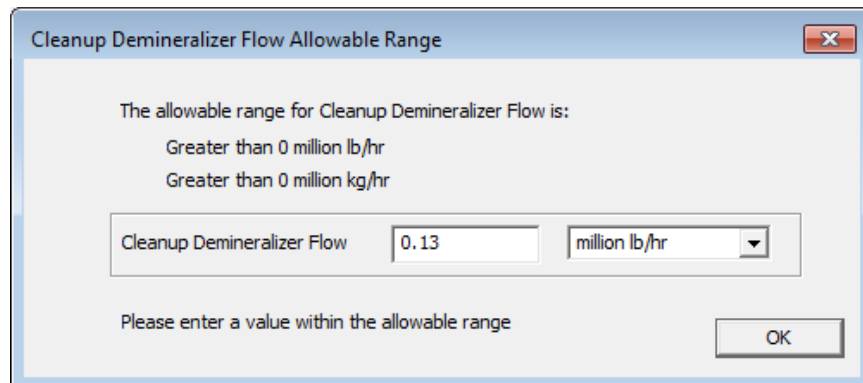
Reactor Vessel Allowable Range Screen shown in Figure 3-13. The default value from the BWRGALE.in file is “0.38” million lb, and the allowable range for values in this field is greater than either 0 million lb or 0 million kg.



**Figure 3-13 Mass of Water in Reactor Vessel Allowable Range Screen**

#### 3.2.2.5 Cleanup Demineralizer Flow

The cleanup demineralizer takes all or some fraction of the water that condenses through the turbine and condenser and cleans dissolved minerals from the water to keep the reactor water clean. Enter the reactor coolant flow rate through the reactor coolant cleanup system demineralizers in either units of million lb/h or million kg/h. Selecting the “Range” button next to this input field will open the Cleanup Demineralizer Flow Allowable Range Screen shown in Figure 3-14. The default value from the BWRGALE.in file is “0.13” million lb/h, and the allowable range for values in this field is greater than either 0 million lb/h or 0 million kg/h.



**Figure 3-14 Cleanup Demineralizer Flow Allowable Range Screen**

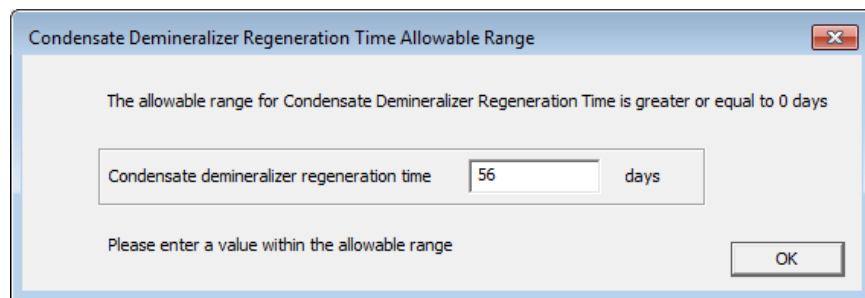
#### 3.2.2.6 Condensate Demineralizer Regeneration Time

The condensate demineralizer may be regenerated from time to time to remove the material taken from the reactor water. Enter the condensate demineralizer regeneration time in units of days. Selecting the “Range” button next to this input field will open the Condensate

Demineralizer Regeneration Time Allowable Range Screen shown in Figure 3-15. If this value is not given, use the following sample values:

- For deep-bed condensate demineralizers, use a 3.5-day regeneration frequency.
- If URC is used, assume an 8-day regeneration frequency.
- For filter/demineralizers (Powdex), enter 0.0.
- Multiply the frequency by the total number of demineralizers and enter the calculated number of days.

The default value from the BWRGALE.in file is “56” days, and the allowable range for values in this field is greater than or equal to 0 days.



**Figure 3-15 Condensate Demineralizer Regeneration Time Allowable Range Screen**

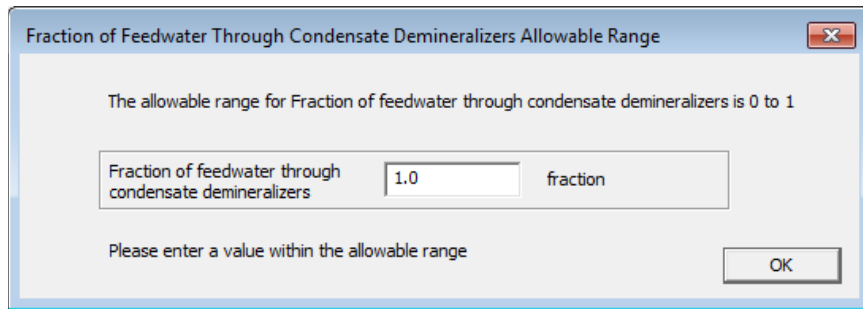
### 3.2.2.7 Copper Tubing for Condenser

The condenser, shown in Figure 3.8, is used to condense the steam that passes through the turbine back to liquid water. The material the condenser is made of can affect the radionuclide content in the primary coolant. Select “Yes” if copper tubing is used in the condenser. The default value from the BWRGALE.in file is “No.”

### 3.2.2.8 Fraction of Feedwater through Condensate Demineralizer

Enter the fraction of feedwater processed through the condensate demineralizers. 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 “1.0” fraction should be input. Selecting the “Range” button next to this input field will open the Fraction of Feedwater through Condensate Demineralizers Allowable Range Screen shown in Figure 3-16. The default value from the BWRGALE.in file is “1.0” fraction, and the allowable range for values in this field is between 0 and 1.0.

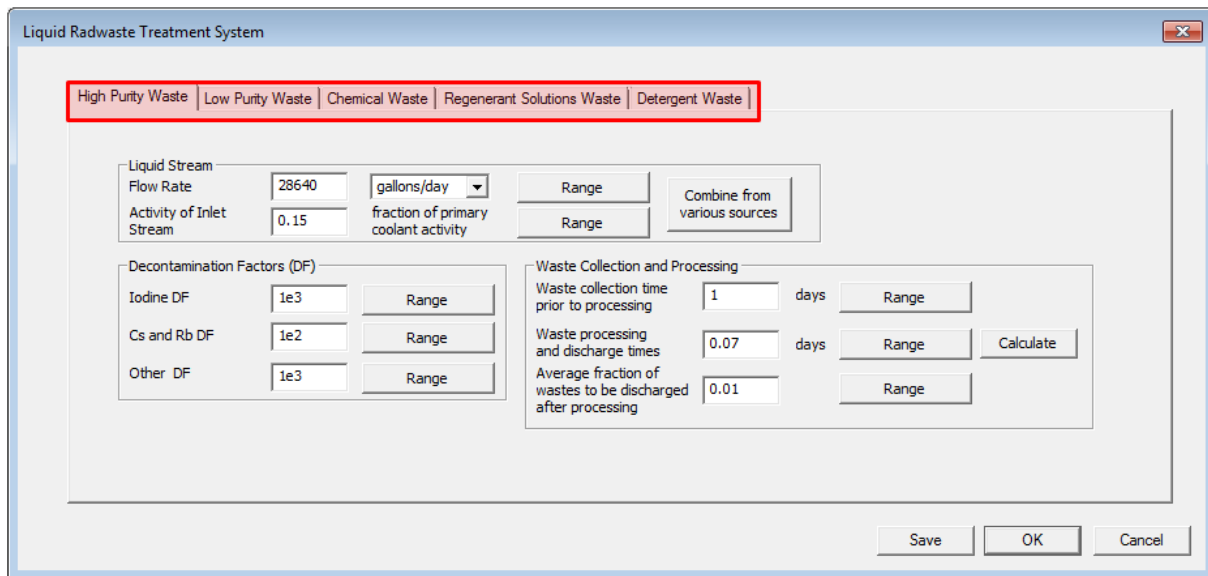




**Figure 3-16 Fraction of Feedwater Through Condensate Demineralizers Allowable Range Screen**

### 3.2.3 Liquid GUI Inputs (Liquid Radwaste Treatment System Input Screen)

The “Liquid Inputs” button on the General Reactor Parameters Screen (Figure 3-8) allows the user to open the Liquid Radwaste Treatment System Screen as shown in Figure 3-17. The Liquid Radwaste Treatment System Screen contains five tabs, as shown in Figure 3-17, for the input of the liquid radwaste inlet streams that are considered in the code. Table 3-1 lists five liquid radwaste inlet stream tabs and the potential sources for each stream.



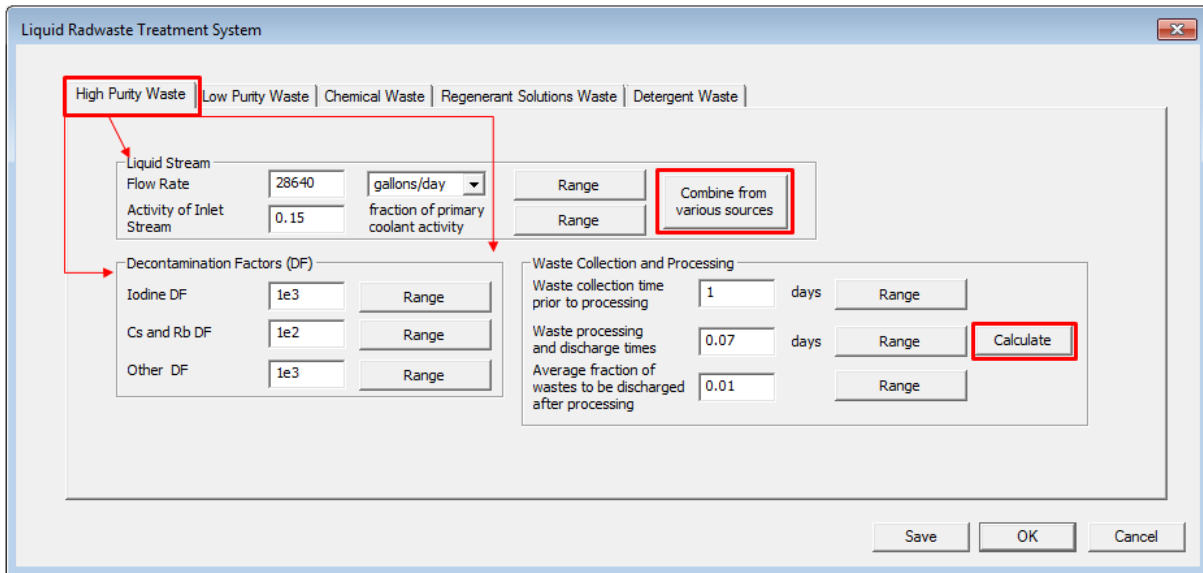
**Figure 3-17 Liquid Radwaste Treatment System Screen**

**Table 3-1 Five BWR liquid radwaste inlet stream tabs and their sources**

Liquid Radwaste Inlet Stream Tab	Potential Source of Liquid Radwaste Stream
High-Purity Waste	Equipment drains from: Drywell Reactor Building Turbine Building Radwaste Building Auxiliary Building Fuel Pool Building  URC overheads Resin backwash Transfer water Filter backwash Phase separator decant liquid Radwaste evaporator condensate
Low-Purity Waste	Floor drains from: Drywell Reactor Building Turbine Building Radwaste Building Auxiliary Building Fuel Pool Building  Uncollected valve and pump seal leakoffs Water resulting from dewatering of slurry wastes
Chemical Waste	Laboratory drains Nondetergent chemical decontamination wastes
Regenerant Solutions Waste	Regenerant solution from ion exchange columns (condensate polishers)
Detergent Waste	Sites that have onsite laundry facilities

**3.2.3.1 High Purity Waste Tab**

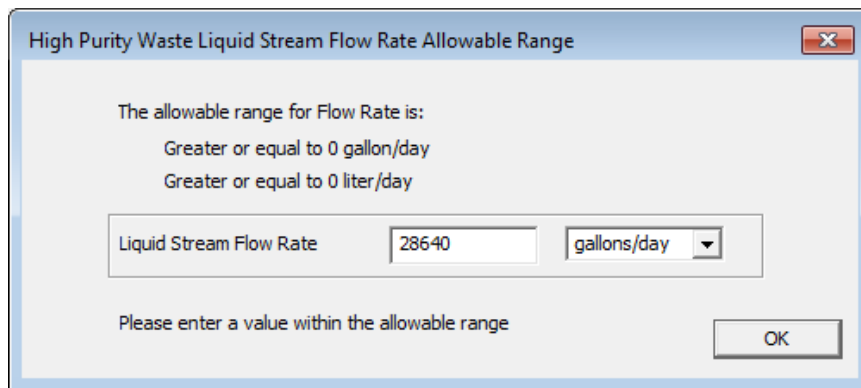
As defined in Section 2.2.2, high-purity waste is liquids, normally of low electrical conductivity, consisting primarily of liquid waste collected from building equipment drains, valve and pump seal leakoffs, demineralizer backwash, URC, and resin transfer. These liquids are normally reused as primary coolant makeup water after processing. Figure 3-18 shows the Liquid Radwaste Treatment System Screen selected to the High Purity Waste Tab with the default values from the BWRGALE.in file. The inputs are described below.



**Figure 3-18 Liquid Radwaste Treatment System Screen (High Purity Waste Tab)**

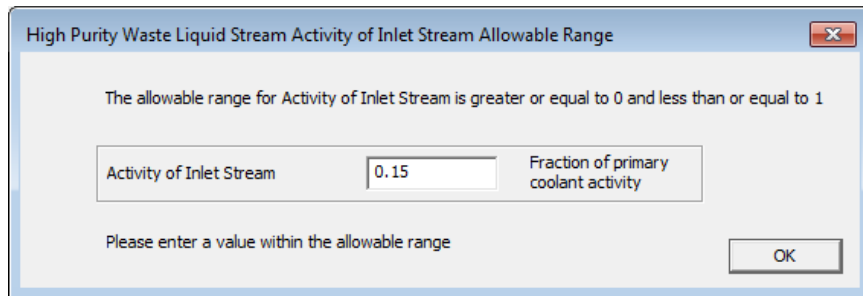
Liquid Stream Field

Enter the Liquid Stream Flow Rate in either units of gallons per day (gal/d) or liters per day (L/d). Selecting the “Range” button next to this input field will open the High Purity Waste Liquid Stream Flow Rate Allowable Range Screen shown in Figure 3-19. The default value from the BWRGALE.in file is “**28640**” gal/d, and the allowable range for values in this field is greater than or equal to either 0 gal/d or 0 L/d.



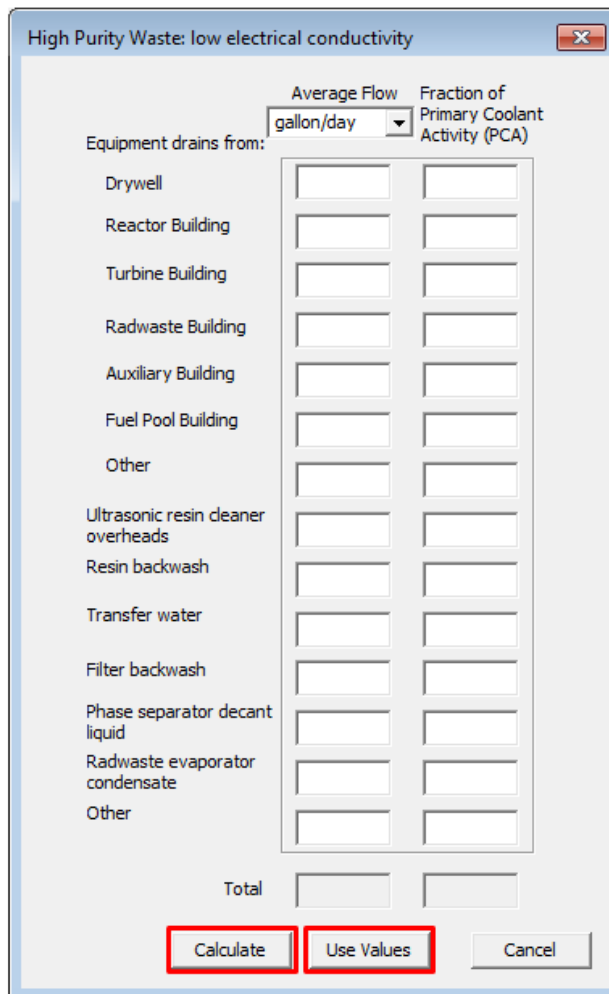
**Figure 3-19 High Purity Waste Liquid Stream Flow Rate Allowable Range Screen**

Enter the Activity of Inlet Stream in units of fraction of PCA. Selecting the “Range” button next to this input field will open the High Purity Waste Liquid Stream Activity of Inlet Stream Allowable Range Screen shown in Figure 3-20. The default value from the BWRGALE.in file is “**0.15**,” and the allowable range for values in this field are between 0.0 and 1.0 fraction of PCA. Table 4-29 gives sample values of typical flow rates and PCAs for various reactor drains and other liquid streams.



**Figure 3-20 High Purity Waste Liquid Stream Activity of Inlet Stream Allowable Range Screen**

In addition, because there are multiple sources of liquid waste for the different waste types, the user can access a separate calculation screen by selecting the “Combine from various sources” button (Figure 3-18). Figure 3-21 shows the High Purity Waste: Low Electrical Conductivity Screen, where specific flow rates and activity fractions can be input to calculate the total liquid stream for the various waste types.



**Figure 3-21 High Purity Waste: Low Electrical Conductivity Screen**

After inputting the values in the top section, the user should press the “Calculate” button to calculate the values in the bottom section. GALE-BWR 3.2 calculates the input PCAs based on the weighted average activity of the composite stream entering the waste collection tanks using Equation (3-1):

$$PCA = \frac{(R_1)(PCA_1) + (R_2)(PCA_2) + (R_3)(PCA_3)}{(R_1 + R_2 + R_3)} \quad (3-1)$$

where  $R_1$  = the flow rate of the first input stream (gal/d);  
 $PCA_1$  = the PCA for the first input stream (unit less);  
 $R_2$  = the flow rate of the second input stream (gal/d);  
 $PCA_2$  = the PCA for the second input stream (unit less);  
 $R_3$  = the flow rate of the third input stream (gal/d); and  
 $PCA_3$  = the PCA for the third input stream (unit less).

For example, if the inlet stream 1, 2, and 3 flow rates and PCAs are as listed below—

Stream 1	1.0E+03 gal/d at 1.0E-02 PCA
Stream 2	2.0E+03 gal/d at 1.0E-01 PCA
Stream 3	5.0E+02 gal/d at 1.0E+00 PCA

then the composite 1, 2, 3 activity would be calculated as follows:

$$PCA = \frac{(1.0E+03 \text{ gal/d})(1.0E-02) + (2.0E+03 \text{ gal/d})(1.0E-01) + (5.0E+02 \text{ gal/d})(1.0E+00)}{(1.0E+03 \text{ gal/d} + 2.0E+03 \text{ gal/d} + 1.0E+03 \text{ gal/d})} = 2.0E-01$$

The calculated composite PCA of inlet stream is 2.0E-01 and the total average flow rate is 3.5E+03 gal/d. To use these values in the input, the user should click on “Use Values” button.

### Decontamination Factors Field

Enter the DFs for high-purity waste for the three categories of radionuclides shown in Figure 3-18 (Iodine DF, Cs and Rb [cesium and rubidium] DF, and Other DF). Selecting the “Range” buttons next to these input fields will open the appropriate allowable range screen shown in Figure 3-22. The default values from the BWRGALE.in file are “1e3” for Iodine DF, “1e2” for Cs and Rb DF, and “1e3” for Other DF, and the allowable range for values in these fields is greater than or equal to 1.0. Table 3-2 provides values of typical DFs for BWR liquid waste treatment systems.

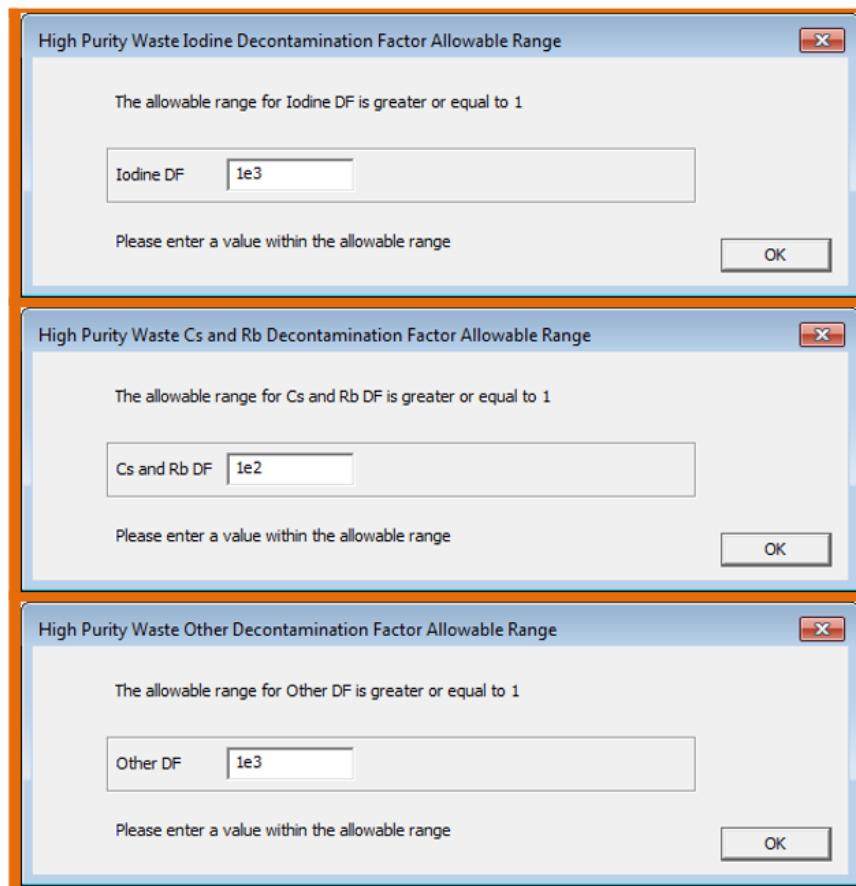


Figure 3-22 High Purity Waste Decontamination Factor Allowable Range Screens

**Table 3-2 Summary of typical Decontamination Factors for BWR liquid waste treatment streams**

Demineralizers <sup>a</sup> <i>Mixed-Bed</i>	Anion <sup>b</sup> (e.g., Iodine)	Cesium & Rubidium <sup>b</sup>	Other Radionuclides <sup>b</sup>
Reactor coolant cleanup	1.0E+01	2.0E+00	1.0E+01
Condensate (deep-bed)	1.0E+01	1.0E+00	1.0E+01
High-purity waste	1.0E+02 (1.0E+01)	1.0E+01 (1.0E+01)	1.0E+02 (1.0E+01)
Low-purity waste	1.0E+02 (1.0E+01)	2.0E+01 (1.0E+01)	1.0E+02 (1.0E+01)
Cation bed (any system)	1.0E+00 (1.0E+00)	1.0E+01 (1.0E+01)	1.0E+02 (1.0E+01)
Anion bed (any system)	1.0E+02 (1.0E+01)	1.0E+00 (1.0E+00)	1.0E+00 (1.0E+00)
Powdex (any system)	1.0E+01 (1.0E+01)	2.0E+01 (1.0E+01)	1.0E+02 (1.0E+01)
Evaporators <sup>d</sup>	All Radionuclides Except Anions		Anions
Miscellaneous	1.0E+04		1.0E+03
Detergent wastes	1.0E+02		1.0E+02
Reverse Osmosis <sup>d</sup>	All Radionuclides		
Laundry wastes	3.0E+01		
Other liquid wastes	1.0E+01		
Filters <sup>e</sup>	DF of 1.0 for all radionuclides		

<sup>a</sup> These values are from Table 1-5 of NUREG-0016, Revision 1, and Table 4-35.

<sup>b</sup> For an evaporator polishing demineralizer or for the second demineralizer in series, the DF is given in parentheses.

<sup>c</sup> These values are from Table 1-5 of NUREG-0016, Revision 1, and Table 4-36.

<sup>d</sup> These values are from Table 1-5 of NUREG-0016, Revision 1, and Section 4.1.1.17.

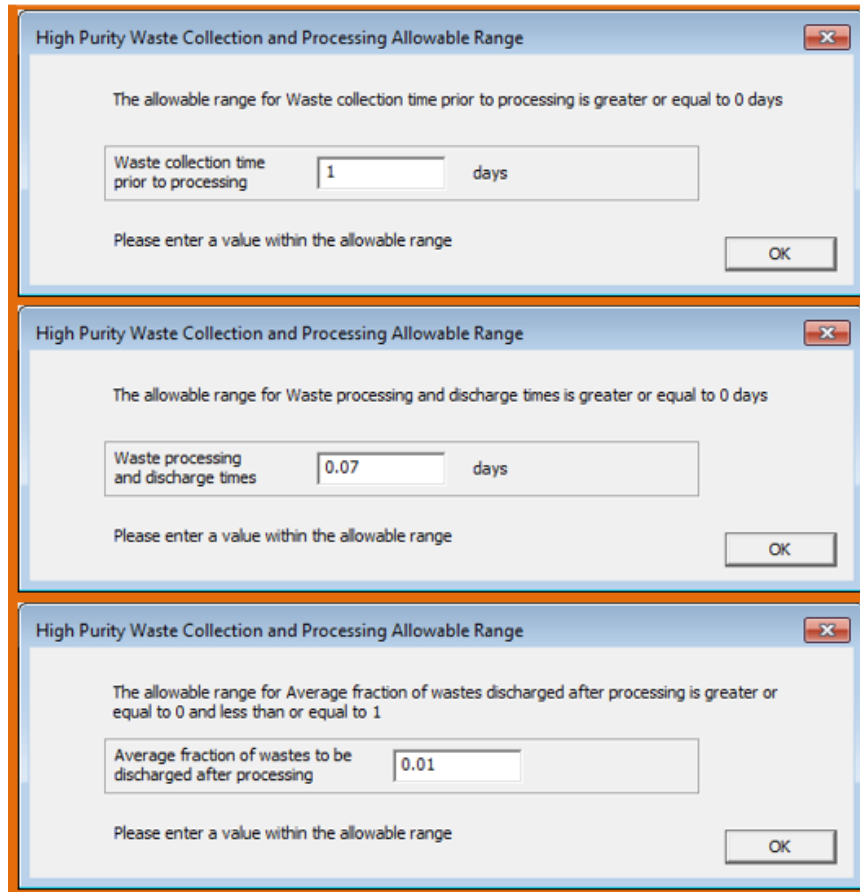
<sup>e</sup> This value is from Table 1-5 of NUREG-0016, Revision 1, and Section 4.1.1.18.

**Waste Collection and Processing Field**

Enter the Waste Collection and Processing parameters for the High Purity Waste Tab:

- Enter the waste collection time in days before processing. Selecting the “Range” button next to this input field will open the appropriate allowable range screen shown in Figure 3-23. The default value from the BWRGALE.in file is “1” day, and the allowable range for values in this field is greater than or equal to 0.0 days.
- Enter the waste processing and discharge time in days. Selecting the “Range” button next to this input field will open the appropriate allowable range screen shown in Figure 3-23. The default value from the BWRGALE.in file is “0.07” days, and the allowable range for values in this field is greater than or equal to 0.0 days.
- Enter the average fraction of wastes to be discharged after processing. Selecting the “Range” button next to this input field will open the appropriate allowable range screen

shown in Figure 3-23. The default value from the BWRGALE.in file is “0.01,” and the allowable range for values in this field are between 0.0 and 1.0.



**Figure 3-23 High Purity Waste Collection and Processing Allowable Range Screens**

In addition to inputting the specific parameters, the information for Waste Collection and Processing can be calculated using the “Calculate” button (Figure 3-18). Figure 3-24 shows the calculation screen for the input of information for waste collection time, processing, and discharge time. Users input the specific information for the various waste streams and then can calculate the final values that are used by the code.

Selecting the “Range” buttons next to each of these input fields in Figure 3-24 will open the appropriate allowable range screen shown in Figure 3-25. The appropriate units for these input fields are either units of volume (i.e., gal or L) or rates (i.e., gal/d or L/d). The allowable range for values in this field is greater than either 0 gal or 0 L (volumes) and either 0 gal/d or 0 L/d (rates). The following sections describe how these calculations are performed.



Waste collection time, and processing and discharge time calculation

Volume of collection tank  gallons

Rate into collection tank  gallons/day

Are there redundant tanks?

Limiting equipment flow capacity of cleanup process  gallons/day

Volume of final tank following cleanup  gallons

Rate of addition waste into final tank  gallons/day

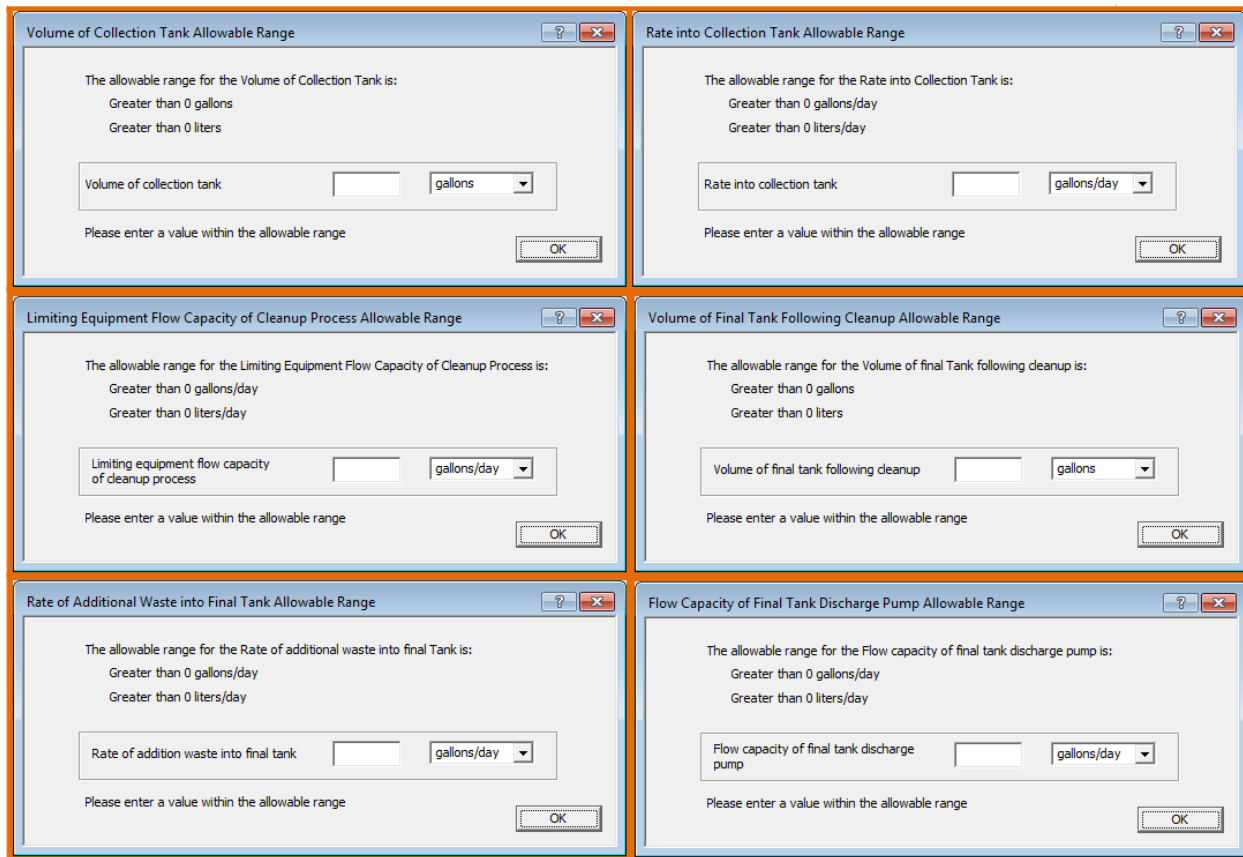
Flow capacity of final tank discharge pump  gallons/day

Waste collection time prior to processing  days

Waste processing and discharge time  days

The diagram illustrates a three-stage process flow. It begins with a 'Collection Tank' receiving 'Rate into collection tank'. The output of the Collection Tank flows into a 'Limiting Process' (represented by an oval). From the Limiting Process, the flow goes to a 'Final Tank'. An additional input, 'Rate of additional waste into final tank', also enters the Final Tank. Below the Limiting Process, the text 'Limiting equipment flow capacity' is shown. Below the Final Tank, the text 'Flow capacity of final discharge pump' is shown. The final output of the Final Tank is indicated by an arrow pointing to the right.

**Figure 3-24 Waste Collection Time, and Processing and Discharge Time Calculation Screen**



**Figure 3-25 Waste Collection Time and Processing and Discharge Time Calculation Allowable Range Screens**

Collection Time for Liquid Wastes

Collection time before processing is based on the input flow calculated above. Where redundant tanks are provided, assume the collection tank to be filled to 80-percent design capacity. If only one tank is provided, assume the tank to be filled to 40-percent design capacity. For example, for the liquid waste processing system shown in Figure 3-26; if flow from a 1,000-gal/day floor drain is collected in two 20,000-gallon tanks before processing, collection time would be calculated using Equation (3-2):

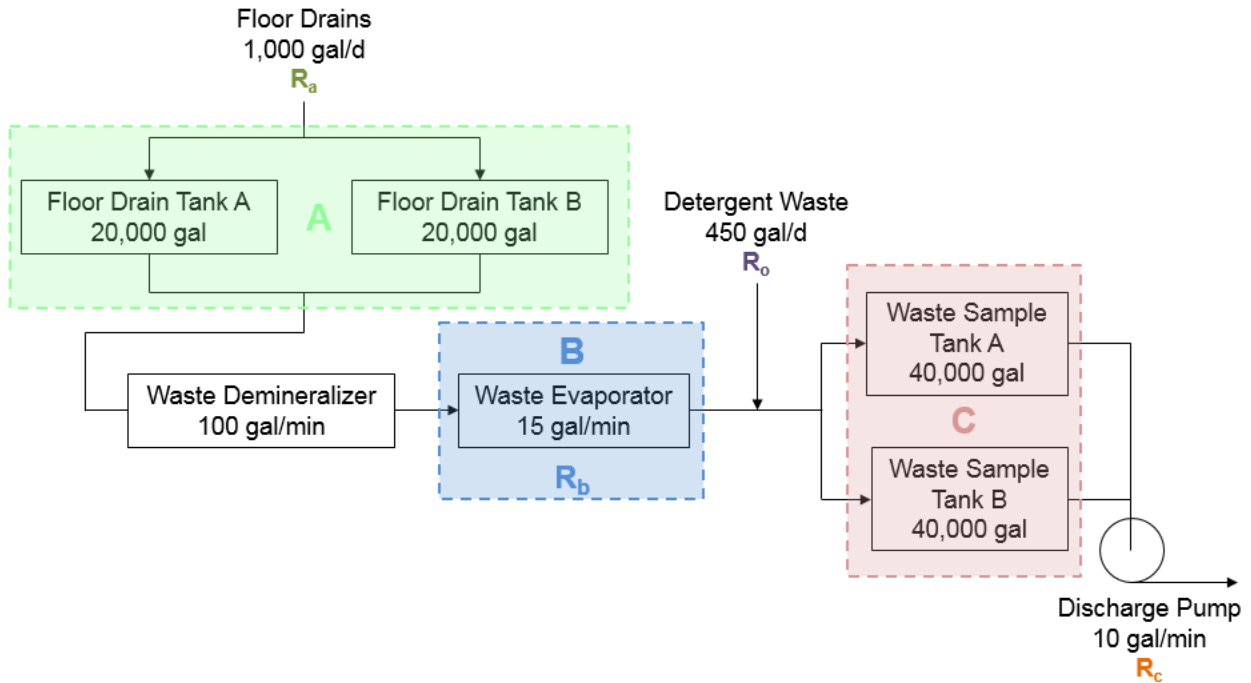
$$T_c = \frac{(8.0E-01) (A)}{R_a} = \frac{(8.0E-01) (2.0E+04 \text{ gal})}{(1.0E+03 \text{ gal/d})} = 16 \text{ days} \quad (3-2)$$

where  $T_c$  = collection time in days;  
 8.0E-01 = the percent design capacity for redundant tanks (4.0E-01 for single tanks);  
 A = the capacity of initial tank in liquid waste flow stream (gal); and  
 $R_a$  = flow rate into the initial collection tank (gal/d).

Additionally, the following are other variables in Figure 3-26:

B = the limiting process based on equipment flow capacity, dimensionless;  
 C = the capacity of final tank in flow scheme before discharge (gal);  
 $R_b$  = the equipment flow capacity of the limiting process (B) (gal/d);

$R_c$  = the flow capacity of the final tank (C) discharge pump (gal/d); and  
 $R_o$  = the rate of flow of additional waste inputs to the final tank (C) (gal/d).



**Figure 3-26 Example of liquid waste processing system flow stream scheme**

Then, for example, 16 days would be calculated for the waste collection time prior to processing block (Figure 3-24) and the user should click on the “Use Values” button (Figure 3-24). This will transfer the value of 16 days to waste collection time prior to processing field in Figure 3-18.

### Processing and Discharge Time

Decay during processing and discharge of liquid wastes is also shown graphically in Figure 3-26. The process time ( $T_p$ ), credited for decay, is calculated using Equation (3-3) for redundant tanks and Equation (3-4) for a single tank.

$$T_p = \frac{(8.0E-01) (A)}{R_b} \quad (3-3)$$

$$T_p = \frac{(4.0E-01) (A)}{R_b} \quad (3-4)$$

The discharge time ( $T_d$ ), 50 percent credited for decay, is calculated using Equation (3-5) for redundant tanks and Equation (3-6) for a single tank.

$$T_d = \frac{(8.0E-01) (C)}{R_c} \quad (3-5)$$

$$T_d = \frac{(4.0E-01) (C)}{R_c} \quad (3-6)$$

After performing the above two calculations, use Equation (3-7) for redundant tanks and Equation (3-8) for a single tank to calculate whether credit may be taken for decay during discharge.

$$(8.0E-01) (C) > T_p (R_b + R_o) \quad (3-7)$$

$$(4.0E-01) (C) > T_p (R_b + R_o) \quad (3-8)$$

If so, then use Equation (3-9) where “Decay” is the new processing and discharge time to be entered for each input stream for the input variables listed above.

$$\text{Decay} = T_p + (5.0E-01)(T_d) \quad (3-9)$$

If, however, the results of Equations 3-7 and 3-8 are  $\leq T_p(R_b + R_o)$ ,  $T_p$  is used for the holdup time, Tank C may be discharged before Tank A has been completely processed. In this case, the  $T_p$  value should be entered.

For example, using Equations 3-3 and 3-5 and the liquid waste flow scheme in Figure 3-26, the following process time ( $T_p$ ) and discharge time ( $T_d$ ) can be calculated:

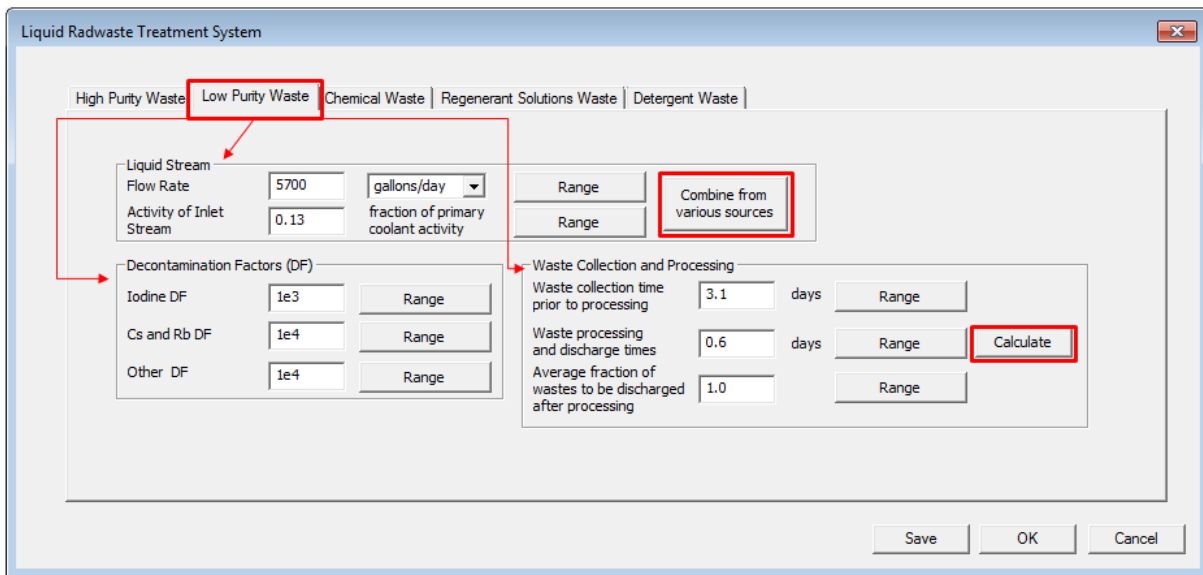
$$T_p = \frac{(8.0E-01) (2.0E+04 \text{ gal})}{(1.5E+01 \text{ gal/min}) (1.44E+03 \text{ min/d})} = 0.7 \text{ days}$$

$$T_d = \frac{(8.0E-01) (4.0E+04 \text{ gal})}{(1.0E+01 \text{ gal/min}) (1.44E+03 \text{ min/d})} = 2.2 \text{ days}$$

Then, checking for decay credit,  $8.0E-01(C)/(R_b + R_o) = 1.45$  days, which is greater than  $T_p$ ; therefore, credit is taken for  $[T_p + 5.0E-01(T_d)]$  or 2.2 days for processing and discharge. The input for the applicable parameter is 2.2 days for processing and discharge time. To use these values in the input, the user should click on the “Use Values” button (Figure 3-24).

### 3.2.3.2 Low Purity Waste Tab

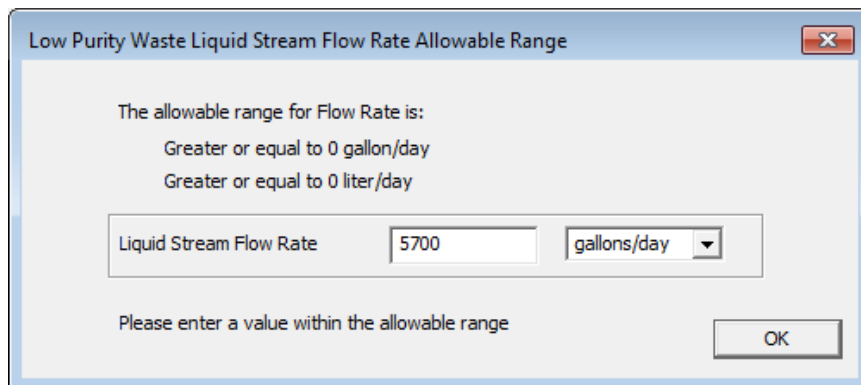
As defined in Section 2.2.2, low-purity waste consists of liquids, normally of high electrical conductivity and not of primary coolant quality, collected from building sumps, uncollected valve and pump seal leakoffs, miscellaneous vents, and floor drains. Figure 3-27 shows the Liquid Radwaste Treatment System Screen selected to the Low Purity Waste Tab with the default values from the BWRGALE.in file. The inputs are described below.



**Figure 3-27 Liquid Radwaste Treatment System Screen (Low Purity Waste Tab)**

Liquid Stream Field

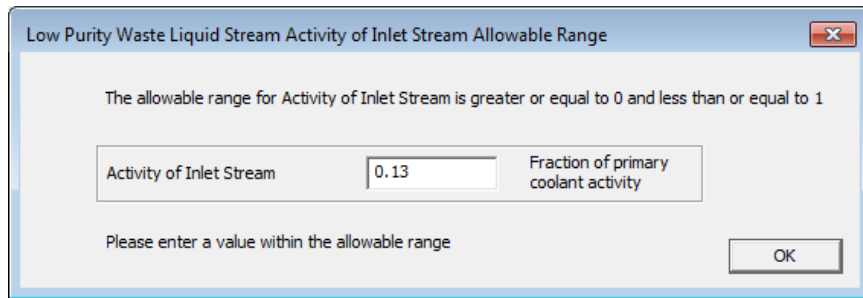
Enter the Liquid Stream Flow Rate in either units of gal/d or L/d. Selecting the “Range” button next to this input field will open the Low Purity Waste Liquid Stream Flow Rate Allowable Range Screen shown in Figure 3-28.



**Figure 3-28 Low Purity Waste Liquid Stream Flow Rate Allowable Range Screen**

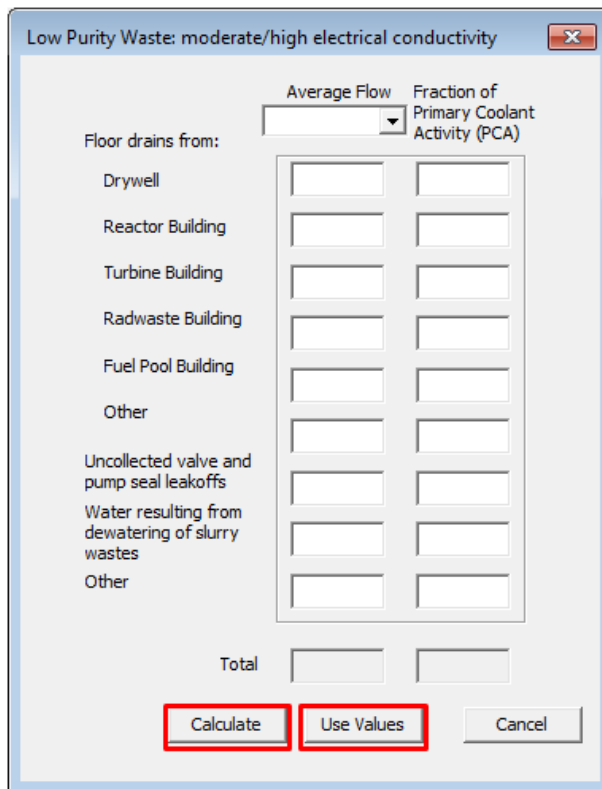
The default value from the BWRGALE.in file is “5700” gal/d, and the allowable range for values in this field is greater than or equal to either 0 gal/d or 0 L/d.

Enter the Activity of Inlet Stream in units of fraction of PCA. Selecting the “Range” button next to this input field will open the Low Purity Waste Liquid Stream Activity of Inlet Stream Allowable Range Screen shown in Figure 3-29. The default value from the BWRGALE.in file is “0.13,” and the allowable range for values in this field are between 0.0 and 1.0 fraction of PCA. Table 4-29 gives sample values of typical flow rates and PCAs for various reactor drains and other liquid streams.



**Figure 3-29 Low Purity Waste Liquid Stream Activity of Inlet Stream Allowable Range Screen**

In addition, because there are multiple sources of liquid waste for the different waste types, the user can access a separate calculation screen by selecting the “Combine from various sources” button (Figure 3-27). Figure 3-30 shows the Low Purity Waste: Moderate/High Electrical Conductivity Screen, where specific flow rates and activity fractions can be input to calculate the total liquid stream for the various waste types. After inputting the values in the top section, the user should press the “Calculate” button to calculate the values in the bottom section. To use these values in the input, the user should click on the “Use Values” button. Section 3.2.3.1 explains the equation used to combine multiple liquid streams to calculate an effective flow rate and PCA.



**Figure 3-30 Low Purity Waste: Moderate/High Electrical Conductivity Screen**

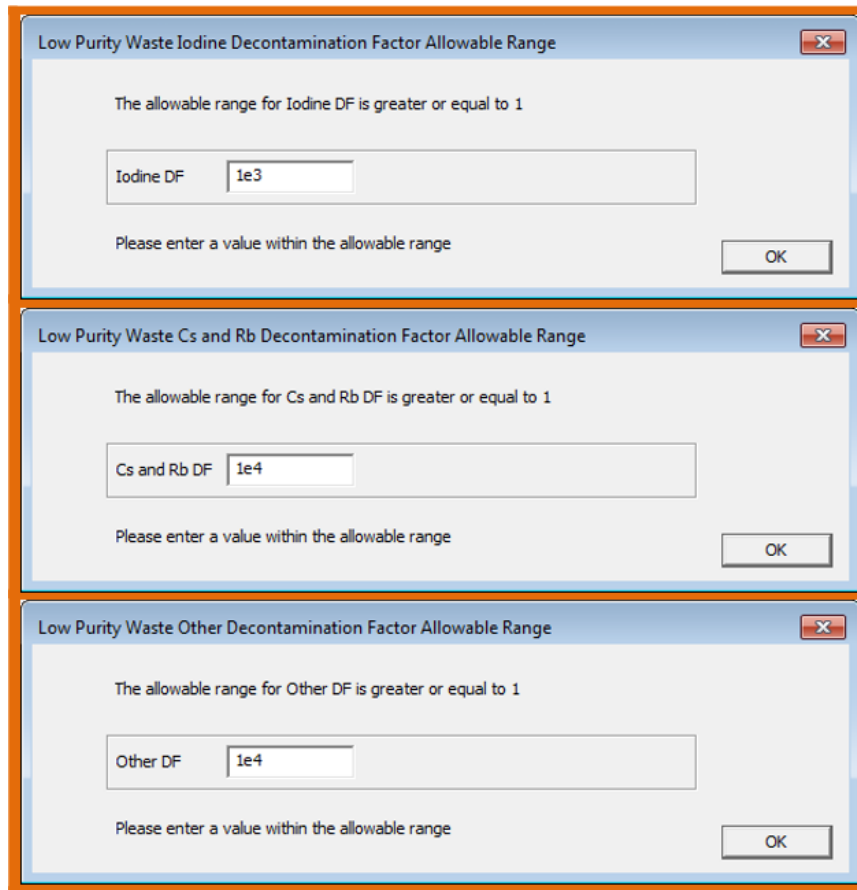
### Decontamination Factors Field

Enter the DFs for low-purity waste for the three categories of radionuclides shown in Figure 3-27 (Iodine DF, Cs and Rb DF, and Other DF). Selecting the “Range” buttons next to these input fields will open the appropriate allowable range screen shown in Figure 3-31. The default values from the BWRGALE.in file are “**1e3**” for Iodine DF, “**1e4**” for Cs and Rb DF, and “**1e4**” for Other DF, and the allowable range for values in these fields is greater than or equal to 1.0. Table 3-2 provides values of typical DFs for BWR liquid waste treatment systems.

### Waste Collection and Processing Field

Enter the waste collection and processing parameters for the Low Purity Waste Tab:

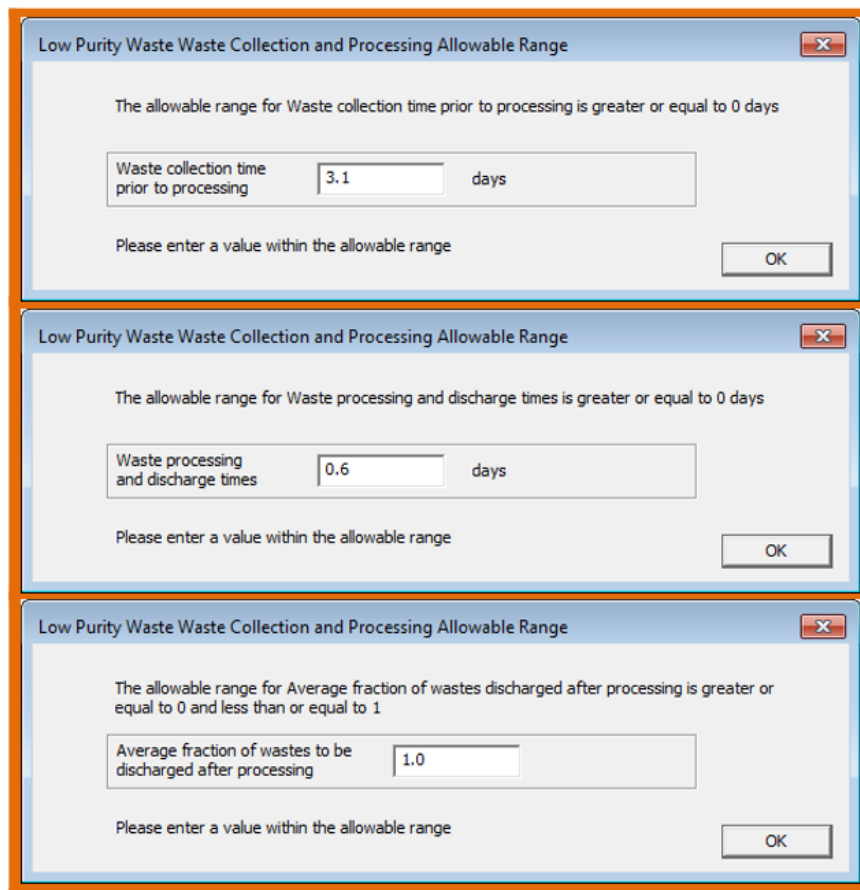
- Enter the waste collection time in days prior to processing. Selecting the “Range” button next to this input field will open the appropriate allowable range screen shown in Figure 3-32. The default value from the BWRGALE.in file is “**3.1**” days, and the allowable range for values in this field is greater than or equal to 0.0 days.
- Enter the waste processing and discharge time in days. Selecting the “Range” button next to this input field will open the appropriate allowable range screen shown in Figure 3-32. The default value from the BWRGALE.in file is “**0.6**” days, and the allowable range for values in this field is greater than or equal to 0.0 days.



**Figure 3-31 Low Purity Waste Decontamination Factor Allowable Range Screens**

- Enter the average fraction of wastes to be discharged after processing. Selecting the “Range” button next to this input field will open the appropriate allowable range screen shown in Figure 3-32. The default value from the BWRGALE.in file is “1.0,” and the allowable range for values in this field are between 0.0 and 1.0.

In addition to inputting the specific parameters, the information for waste collection and processing can be calculated using the “Calculate” button (Figure 3-27). Figure 3-24 shows the calculation screen for the input of information for waste collection time and processing and discharge time. Users input the specific information for the various waste streams and then can calculate the final values used by the code. After inputting the values in the top section, the user should press the “Calculate” button to calculate the values in the bottom section. To use these values in the input, the user should click on the “Use Values” button (Figure 3-24). Section 3.2.3.1 explains the equations used to determine these three parameters.



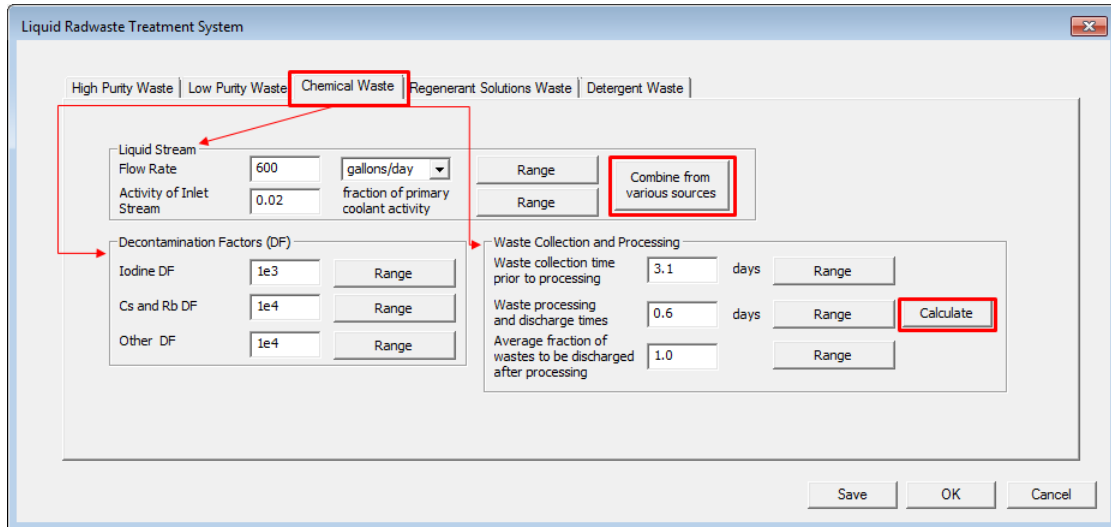
**Figure 3-32 Low Purity Waste Collection and Processing Allowable Range Screens**

### 3.2.3.3 Chemical Waste Tab

In Section 2.2.2, the “chemical waste stream” is defined as liquids, normally of relatively high concentrations of decontaminants, regenerants, or chemical compounds other than detergents. These liquids originate primarily from resin regenerants and laboratory wastes with a high electrical conductivity, and sources include combined releases from laboratory and decontamination wastes and from demineralizer regenerant solutions, according to the design



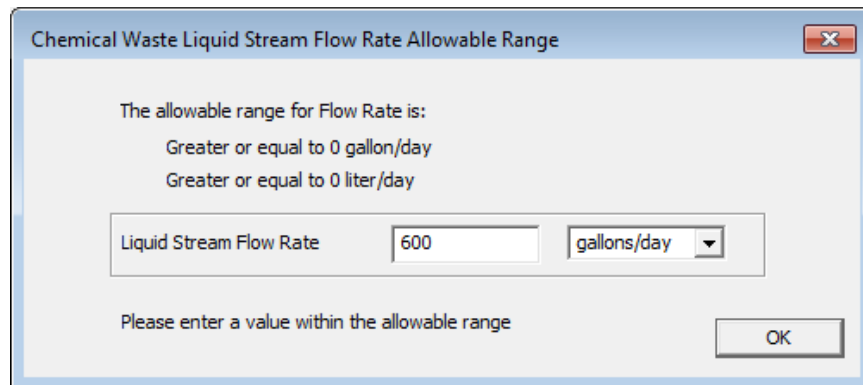
of the condensate demineralizer system. If a filter/demineralizer (Powdex) system is used, the laboratory and decontamination wastes are combined with the low-purity waste or solidified in the solid waste system. Figure 3-33 shows the Liquid Radwaste Treatment System Screen selected to the Clean Waste Tab with the default values from the BWRGALE.in file. The inputs are described below.



**Figure 3-33 Liquid Radwaste Treatment System Screen (Chemical Waste Tab)**

Liquid Stream Field

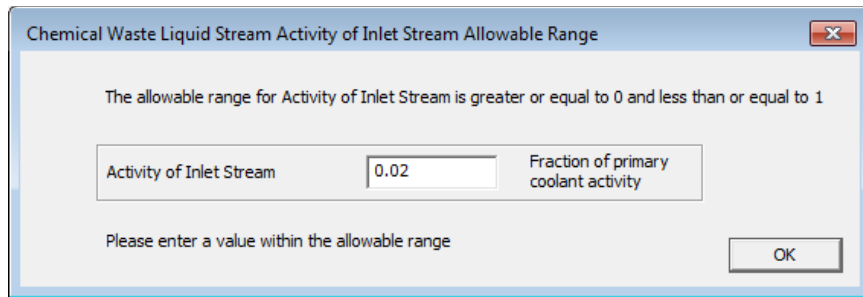
Enter the Liquid Stream Flow Rate in either units of gal/d or L/d. Selecting the “Range” button next to this input field will open the Chemical Waste Liquid Stream Flow Rate Allowable Range Screen shown in Figure 3-34. The default value from the BWRGALE.in file is “600” gal/d, and the allowable range for values in this field is greater than or equal to either 0 gal/d or 0 L/d.



**Figure 3-34 Chemical Waste Liquid Stream Flow Rate Allowable Range Screen**

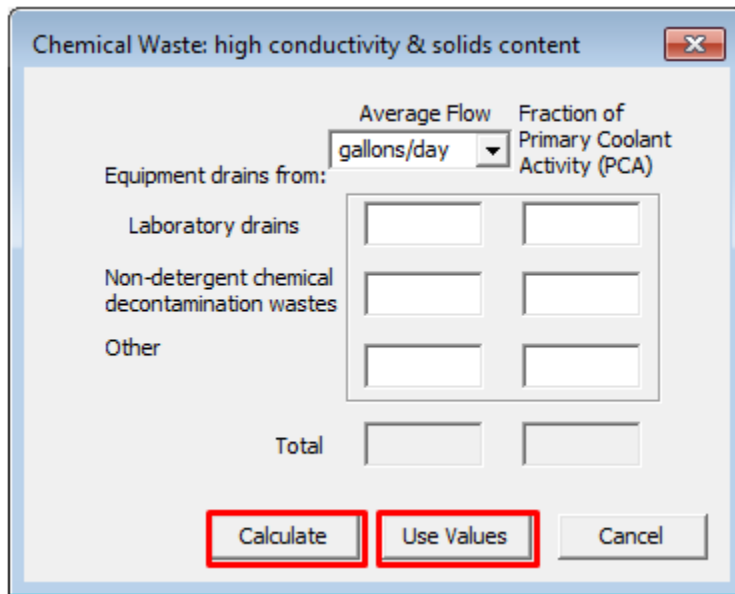
Enter the Activity of Inlet Stream in units of fraction of PCA. Selecting the “Range” button next to this input field will open the Chemical Waste Liquid Stream Activity of Inlet Stream Allowable Range Screen shown in Figure 3-35. The default value from the BWRGALE.in file is “0.02,” and

the allowable range for values in this field are between 0.0 and 1.0 fraction of PCA. Table 4-29 gives sample values of typical flow rates and PCAs for various reactor drains and other liquid streams.



**Figure 3-35 Chemical Waste Liquid Stream Activity of Inlet Stream Allowable Range Screen**

In addition, because there are multiple sources of liquid waste for the different waste types, the user can access a separate calculation screen by selecting the “Combine from various sources” button. Figure 3-36 shows the Chemical Waste: High Conductivity & Solids Content Screen where specific flow rates and activity fractions can be input to calculate the total liquid stream for the various waste types. After inputting the values in the top section, the user should press the “Calculate” button to calculate the values in the bottom section. To use these values in the input, the user should click on the “Use Values” button (Figure 3-36). Section 3.2.3.1 explains the equation used to combine multiple liquid streams to calculate an effective flow rate and PCA.

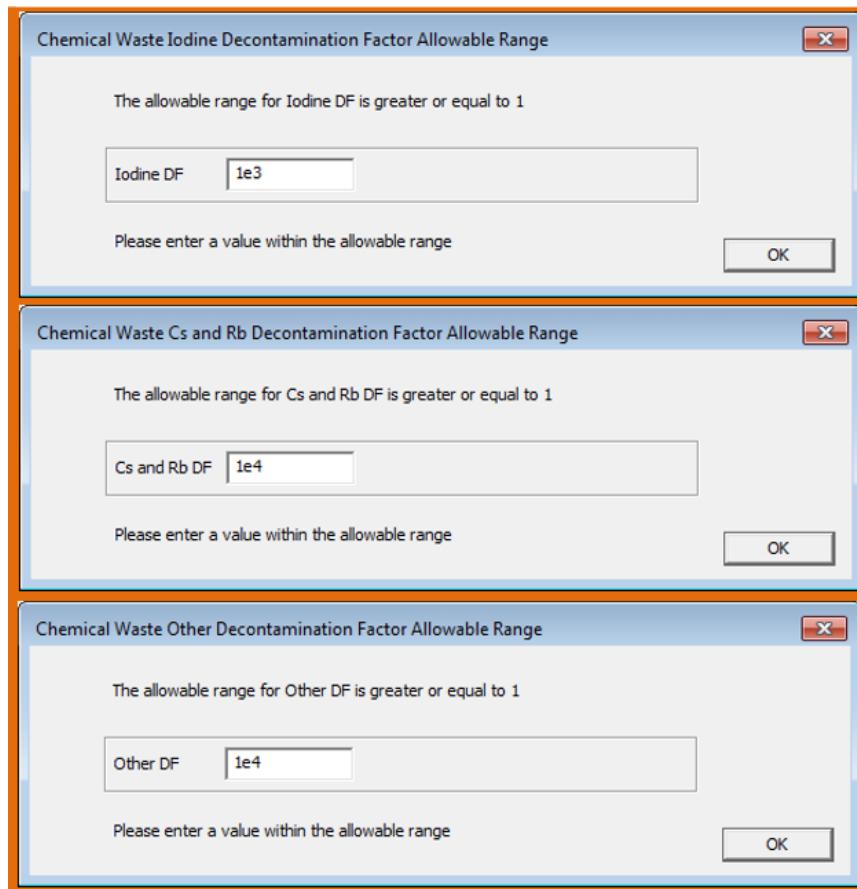


**Figure 3-36 Chemical Waste: High Conductivity & Solids Content Screen**

Decontamination Factors Field

Enter the DFs for clean waste for the three categories of radionuclides shown in Figure 3-33 (Iodine DF, Cs and Rb DF, and Other DF). Selecting the “Range” buttons next to these input fields will open the appropriate allowable range screen shown in Figure 3-37. The default

values from the BWRGALE.in file are “1e3” for Iodine DF, “1e4” for Cs and Rb DF, and “1e4” for Other DF, and the allowable range for values in these fields is greater than or equal to 1.0. Table 3-2 provides values of typical DFs for BWR liquid waste treatment systems.



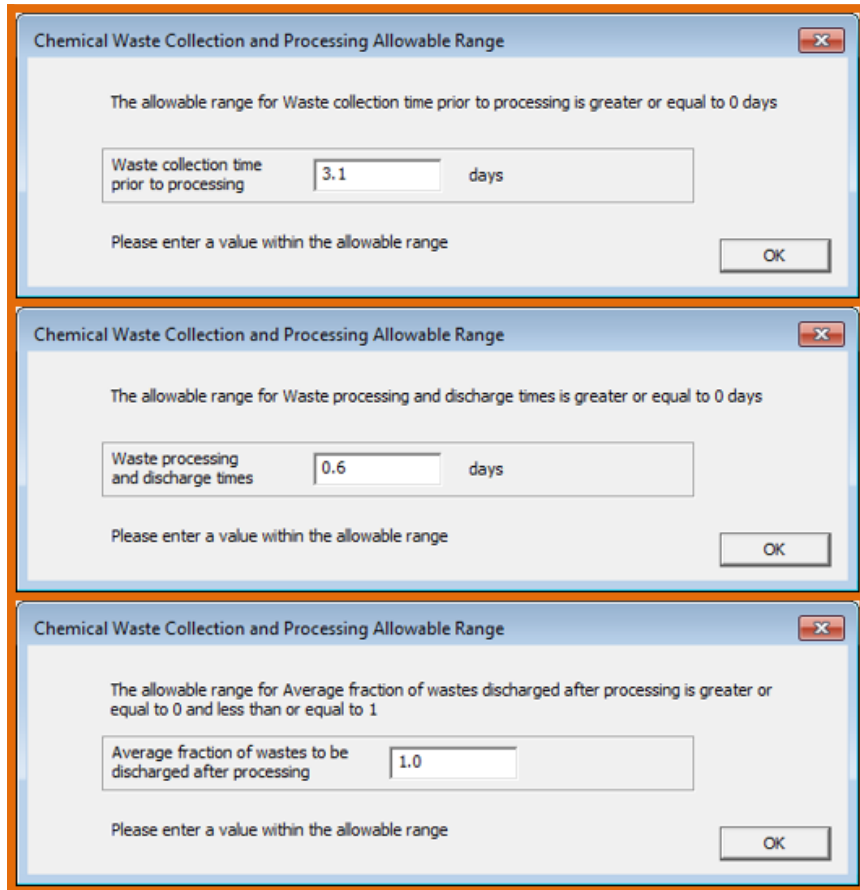
**Figure 3-37 Chemical Waste Decontamination Factor Allowable Range Screens**

Waste Collection and Processing Field

Enter the waste collection and processing parameters for the Chemical Waste Tab:

- Enter the waste collection time in days prior to processing. Selecting the “Range” button next to this input field will open the appropriate allowable range screen shown in Figure 3-38. The default value from the BWRGALE.in file is “3.1” days, and the allowable range for values in this field is greater than or equal to 0.0 days.
- Enter the waste processing and discharge time in days. Selecting the “Range” button next to this input field will open the appropriate allowable range screen shown in Figure 3-38. The default value from the BWRGALE.in file is “0.6” days, and the allowable range for values in this field is greater than or equal to 0.0 days.
- Enter the average fraction of wastes to be discharged after processing. Selecting the “Range” button next to this input field will open the appropriate allowable range screen

shown in Figure 3-38. The default value from the BWRGALE.in file is “1.0,” and the allowable range for values in this field are between 0.0 and 1.0.

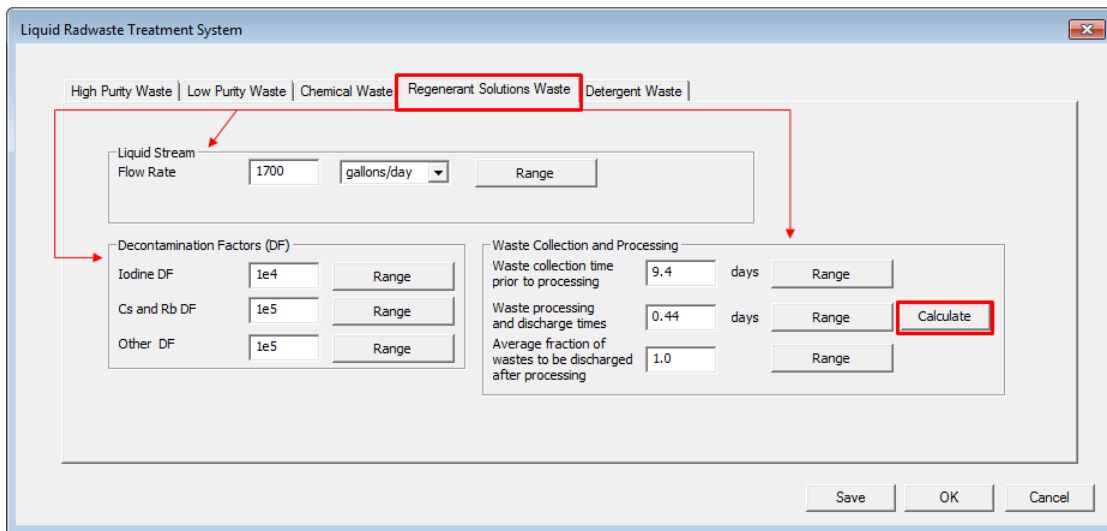


**Figure 3-38 Chemical Waste Collection and Processing Allowable Range Screens**

In addition to inputting the specific parameters, the information for waste collection and processing can be calculated using the “Calculate” button (Figure 3-33). Figure 3-24 shows the calculation screen for the input of information for waste collection time and processing and discharge time. Users input the specific information for the various waste streams and then can calculate the final values used by the code. After inputting the values in the top section, the user should press the “Calculate” button to calculate the values in the bottom section. To use these values in the input, the user should click on the “Use Values” button (Figure 3-24). Section 3.2.3.1 explains the equations used to determine these three parameters.

#### 3.2.3.4 *Regenerant Solutions Waste Tab*

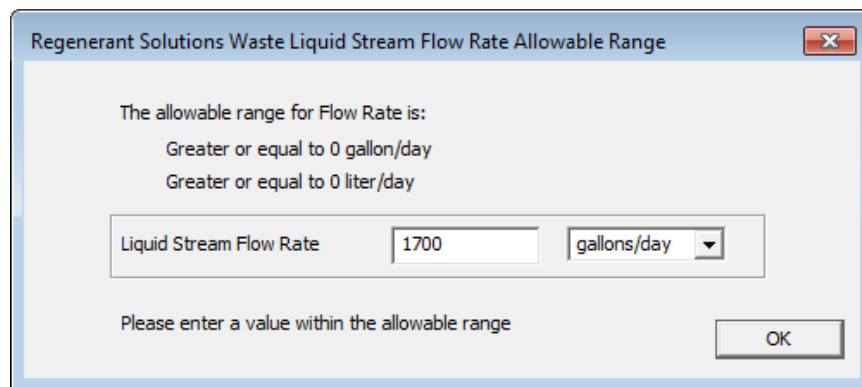
In Section 2.2.2, “regenerant solutions waste” is defined as a liquid solution from ion exchange columns or condensate polishers that are part of the demineralizer system. Figure 3-39 shows the Liquid Radwaste Treatment System Screen selected to the Regenerant Solutions Waste Tab with the default values from the BWRGALE.in file. The inputs are described below.



**Figure 3-39 Liquid Radwaste Treatment System Screen (Regenerant Solutions Waste Tab)**

Liquid Stream Field

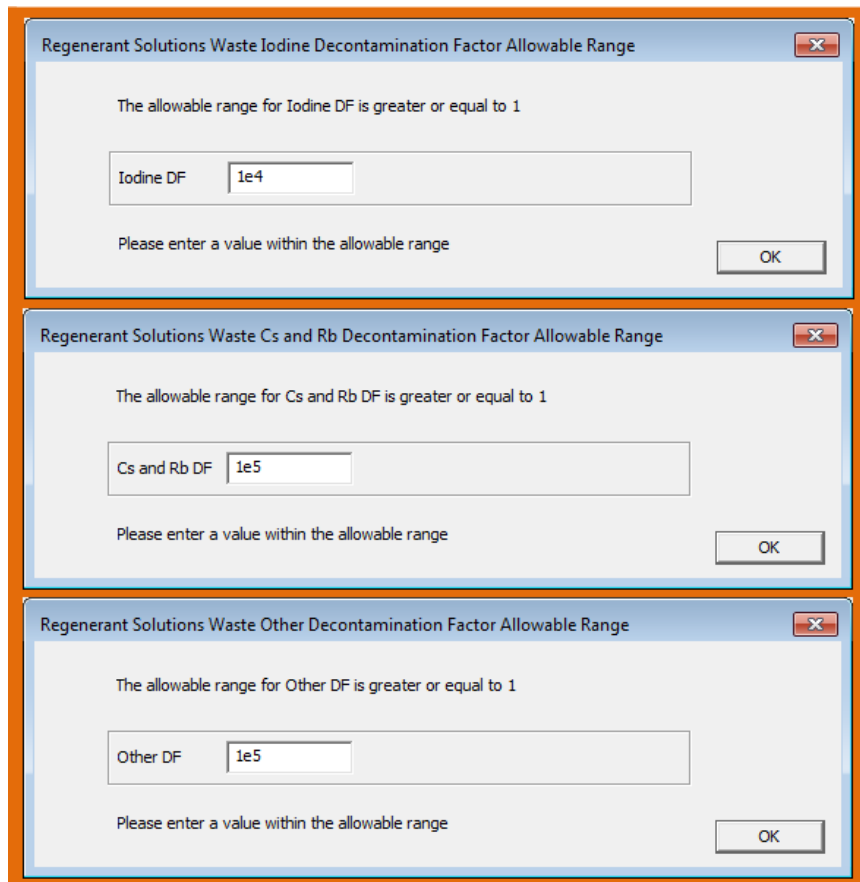
Enter the Liquid Stream Flow Rate in either units of gal/d or L/d. Selecting the “Range” button next to this input field will open the Regenerant Solutions Waste Liquid Stream Flow Rate Allowable Range Screen shown in Figure 3-40. The default value from the BWRGALE.in file is “1700” gal/d, and the allowable range for values in this field is greater than or equal to either 0 gal/d or 0 L/d.



**Figure 3-40 Regenerant Solutions Waste Liquid Stream Flow Rate Allowable Range Screen**

Decontamination Factors Field

Enter the DFs for regenerant waste for the three categories of radionuclides shown in Figure 3-39 (Iodine DF, Cs and Rb DF, and Other DF). Selecting the “Range” buttons next to these input fields will open the appropriate allowable range screen shown in Figure 3-41. The default values from the BWRGALE.in file are “1e4” for Iodine DF, “1e5” for Cs and Rb DF, and “1e5” for Other DF, and the allowable range for values in these fields is greater than or equal to 1.0. Table 3-2 provides the values of typical DFs for BWR liquid waste treatment systems.

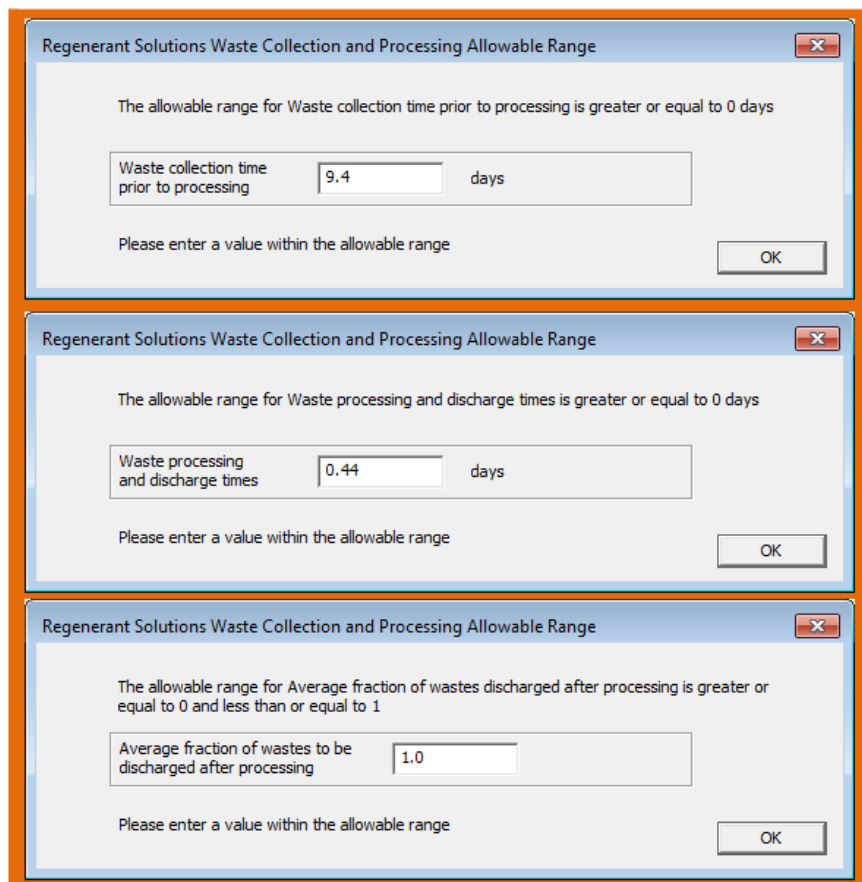


**Figure 3-41 Regenerant Solutions Decontamination Factor Allowable Range Screens**

Waste Collection and Processing Field

Enter the waste collection and processing parameters for the Regenerant Solutions Waste Tab:

- Enter the waste collection time in days prior to processing. Selecting the “Range” button next to this input field will open the appropriate allowable range screen shown in Figure 3-42. The default value from the BWRGALE.in file is “**9.4**” days, and the allowable range for values in this field is greater than or equal to 0.0 days.
- Enter the waste processing and discharge time in days. Selecting the “Range” button next to this input field will open the appropriate allowable range screen shown in Figure 3-42. The default value from the BWRGALE.in file is “**0.44**” days, and the allowable range for values in this field is greater than or equal to 0.0 days.
- Enter the average fraction of wastes to be discharged after processing. Selecting the “Range” button next to this input field will open the appropriate allowable range screen shown in Figure 3-42. The default value from the BWRGALE.in file is “**1.0**,” and the allowable range for values in this field are between 0.0 and 1.0.



**Figure 3-42 Regenerant Solutions Waste Collection and Processing Allowable Range Screens**

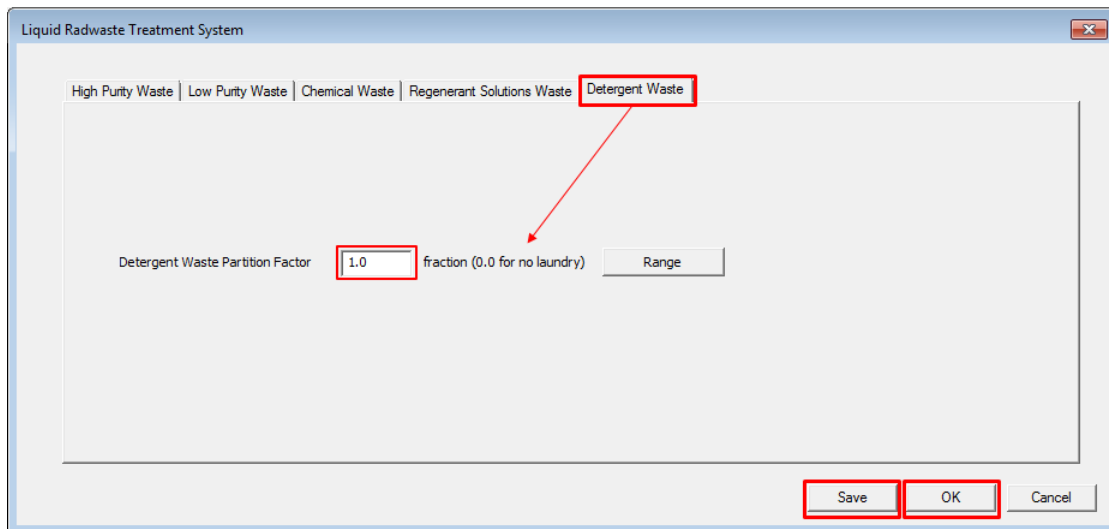
In addition to inputting the specific parameters, the user can calculate the information for waste collection and processing by using the “Calculate” button (Figure 3-39). Figure 3-24 shows the calculation screen for the input of information for waste collection time and processing and discharge time. Users input the specific information for the various waste streams and then can calculate the final values used by the code. After inputting the values in the top section, the user should press the “Calculate” button to calculate the values in the bottom section. To use these values in the input, the user should click on the “Use Values” button (Figure 3-24). Section 3.2.3.1 explains the equations used to determine these three parameters.

### 3.2.3.5 Detergent Waste Tab

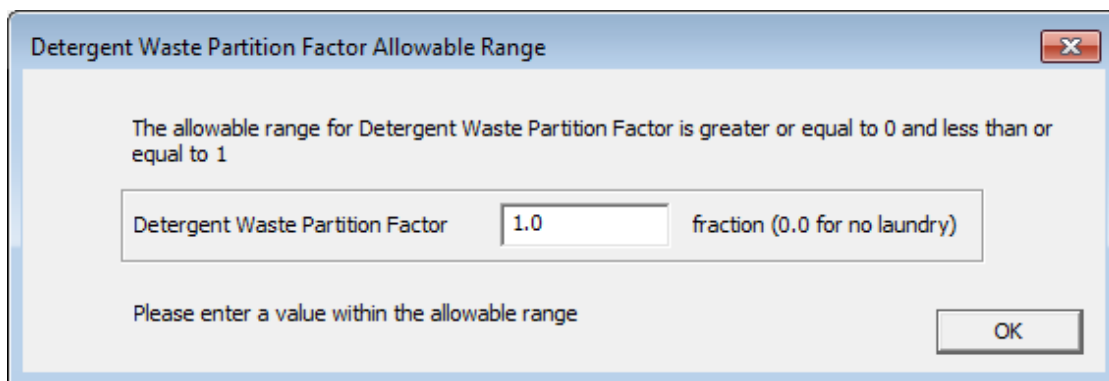
Figure 3-43 shows the Liquid Radwaste Treatment System Screen selected to the Detergent Waste Tab.

#### Detergent Waste Partition Factor

Enter the Detergent Waste Partition Factor in this field. Selecting the “Range” button next to this input field will open the Detergent Waste Partition Factor Allowable Range Screen shown in Figure 3-44. The default value from the BWRGALE.in file is “1.0,” and the allowable range for values in this field is either greater than or equal to 0 and less than or equal to 1.0.



**Figure 3-43 Liquid Radwaste Treatment System Screen (Detergent Waste Tab)**



**Figure 3-44 Detergent Waste Partition Factor Allowable Range Screen**

If the plant does not have an onsite laundry, enter “0.0.” If the plant has an onsite laundry and detergent wastes are released without treatment, enter “1.0.” If detergent wastes are treated before discharge, enter the fraction of radionuclides remaining after treatment (1/DF). The radionuclides added to the liquid release caused by the presence of an onsite laundry and any cleanup systems in place for the laundry are calculated according to Table 4-30 in Section 4.1.1.13.

If detergent wastes are treated before discharge, enter the fraction of radionuclides remaining after treatment (1/DF). The radionuclides added to the liquid release caused by the presence of an onsite laundry and any cleanup systems in place for the laundry are calculated according to Table 4-30 in Section 4.1.1.13.

After the user has finished entering the Liquid Radwaste Treatment System Screen parameters, select the “Save” button to save any changes made to the liquid radwaste input parameters. Then select the “OK” button to close the Liquid Radwaste Treatment System Screen (Figure 3-43) and return to the General Reactor Parameters Screen (Figure 3-8). Note that selecting the “OK” button without first selecting the “Save” button will result in the code returning to the General Reactor Parameters Screen without saving any changes made to the liquid



radwaste input parameters. Therefore, if the user wishes to save any changes made to the inputs, the “Save” button should be selected before selecting the “OK” button.

### 3.2.4 Gaseous GUI Inputs (Gaseous Radwaste Treatment System Input Screen)

The “Gas Inputs” button on the General Reactor Parameters Screen (Figure 3-8) allows the user to open the Gaseous Radwaste Treatment System Screen as shown in Figure 3-45. The Gaseous Radwaste Treatment System Screen includes sections for the Containment Building, Auxiliary Building, Radwaste Building, and Turbine Building, which includes the Gland Seal and Air Ejector Offgas system information and the Charcoal Adsorber Removal Efficiency Information.

**Figure 3-45 Gaseous Radwaste Treatment System Screen**

#### 3.2.4.1 Containment Building Section

Section 2.2.1 describes the BWR containment or drywell building, while Figure 3-46 shows the inputs for the fields in the Containment Building section as described below:

- **Charcoal Adsorbers**—If the ventilation exhaust air from the containment building is treated through charcoal adsorbers that satisfy the guidelines of NUREG-0016, Revision 1, and RG 1.140, Revision 3, Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Light-Water-Cooled Nuclear Power Plants,” issued June 2001 [Ref. 14], select “**Yes**,” and enter the percent removal efficiency for

iodine corresponding to the depth of charcoal. The “Reg. Guide 1.140 Efficiency” and “NUREG-0016 Efficiency” buttons provide these reference values, as shown in Figures 3-47 and 3-48. If “No” is selected, the code uses a removal efficiency of 0.0 percent. Selecting the “Range” button next to this input field will open the Containment Building Charcoal Adsorber Removal Efficiency Allowable Range Screen shown in Figure 3-49. Figure 3-49 also contains the “Reg. Guide 1.140 Efficiency” and “NUREG-0016 Efficiency” buttons, providing the user with access to Figures 3-47 and 3-48. The default value from the BWRGALE.in file is “Yes” and a removal efficiency of “90 %.” The allowable range for values in the removal efficiency field is greater than or equal to 0.0 and less than or equal to 100.0 percent.

**Figure 3-46 Gaseous Radwaste Treatment System Screen with Containment Building section options**

- High-Efficiency Particulate Air (HEPA) filters—If ventilation exhaust air from the containment building is treated through HEPA filters that satisfy the guidelines of RG 1.140, Revision 3, select “Yes.” If “Yes” is selected, the code uses a removal efficiency of 99 percent for particulates. Select “No” if there is no treatment provided to remove particulates. If the HEPA filters do not satisfy the guidelines of RG 1.140, Revision 3, the code uses a removal efficiency of 0.0 percent. The default value from the BWRGALE.in file is “Yes.”

Table 1 taken from Regulatory Guide 1.140, Revision 3

Note: See Regulatory Guide 1.140, Revision 3 for additional details

**Table 1. Inservice Adsorber Laboratory Tests for Activated Carbon**

Activated Carbon Total Bed Depth	Maximum Assigned Activated Carbon Decontamination Efficiencies		Methyl Iodide Penetration Acceptance Criterion for Representative Sample
2 inches	Elemental iodine	95%	Penetration $\leq 5\%$ when tested in accordance with ASTM D3803-1991
	Organic iodide	95%	
4 inches or greater (in-series beds are treated as a single bed of aggregate depth)	Elemental iodine	99%	Penetration $\leq 1\%$ when tested in accordance with ASTM D3803-1991
	Organic iodide	99%	
Activated carbon in delay tanks or beds	Elemental iodine	99%	Penetration $\leq 1\%$ when tested in accordance with ASTM D3803-1991
	Organic iodide	99%	

**Table 1 Notes:**

- (1) See Appendix I to ASME N509-2002 for the definition of a representative sample.
- (2) Credited decontamination efficiencies (a portion of which includes bypass leakage) are based on a 0.25-second residence time per 5.1 centimeter (2-inch bed depth).
- (3) The activated carbon, when new, should meet the specifications of Regulatory Position 4.i of this guide. Table 1 provides acceptable decontamination efficiencies and methyl iodide test penetrations of used activated carbon samples for laboratory testing. Testing should be performed at the frequencies specified in Regulatory Position 7.c of this guide. Testing should be performed in accordance with American Society for Testing and Materials (ASTM) D3803-1991 (Reaffirmed 2014, "Standard Test Methods for Nuclear-Grade Activated Carbon" (Ref. 18), with an entering air temperature of 30 degrees Celsius (86 degrees Fahrenheit) and a relative humidity of 95 percent (or 70 percent with humidity control). Humidity control can be provided by heaters, cooling coils, or an analysis that demonstrates that the air entering the installed adsorber section would be maintained less than or equal to a 70-percent relative humidity level.
- (4) Organic iodide and elemental iodine are the forms of iodine that are expected to be absorbed by activated carbon. Organic iodide is more difficult for activated carbon to adsorb than elemental iodine. Therefore, the laboratory test to determine the performance of the activated carbon adsorber is based on organic iodide. Methyl iodide is the organic form of iodine that is used in the laboratory test.
- (5) For power cycle waste offgas systems relying on activated carbon delay tanks or beds, the inservice adsorber laboratory tests should confirm that the proper types of activated carbon (nuclear grade and defined mesh size) are tested for representative batches of activated carbon over the entire design inventory of the waste offgas treatment system, given the design capacity and number of delay tanks or beds.

OK

**Figure 3-47 RG 1.140, Revision 3 Charcoal Efficiency Information Screen**

NUREG-0016, Revision 1 Charcoal Efficiency Information

Table 1-6 taken from NUREG-0016, Revision 1

Note: Recommended for use with GALE86

**TABLE 1-6**  
**ASSIGNED REMOVAL EFFICIENCIES FOR CHARCOAL ADSORBERS**  
**FOR RADIOIODINE REMOVAL**

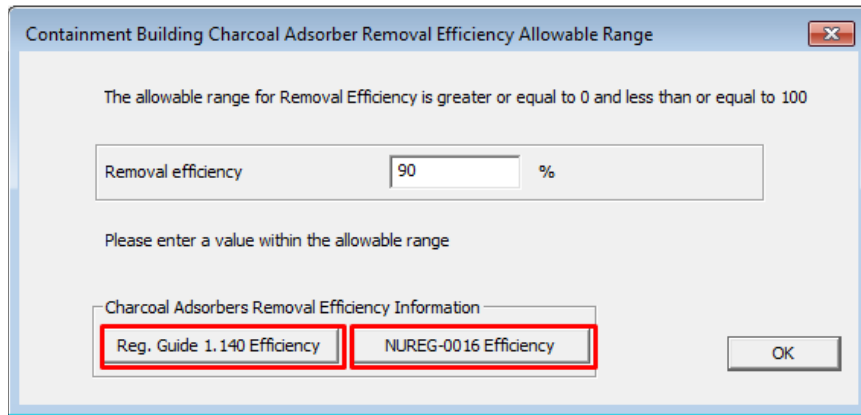
<u>Activated Carbon<sup>a</sup></u> <u>Bed Depth</u>	<u>Removal Efficiencies<sup>b</sup></u> <u>for Radioiodine %</u>
2 inches. Air filtration system designed to operate inside reactor containment	90.
2 inches. Air filtration system designed to operate outside the reactor containment and relative humidity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

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

OK

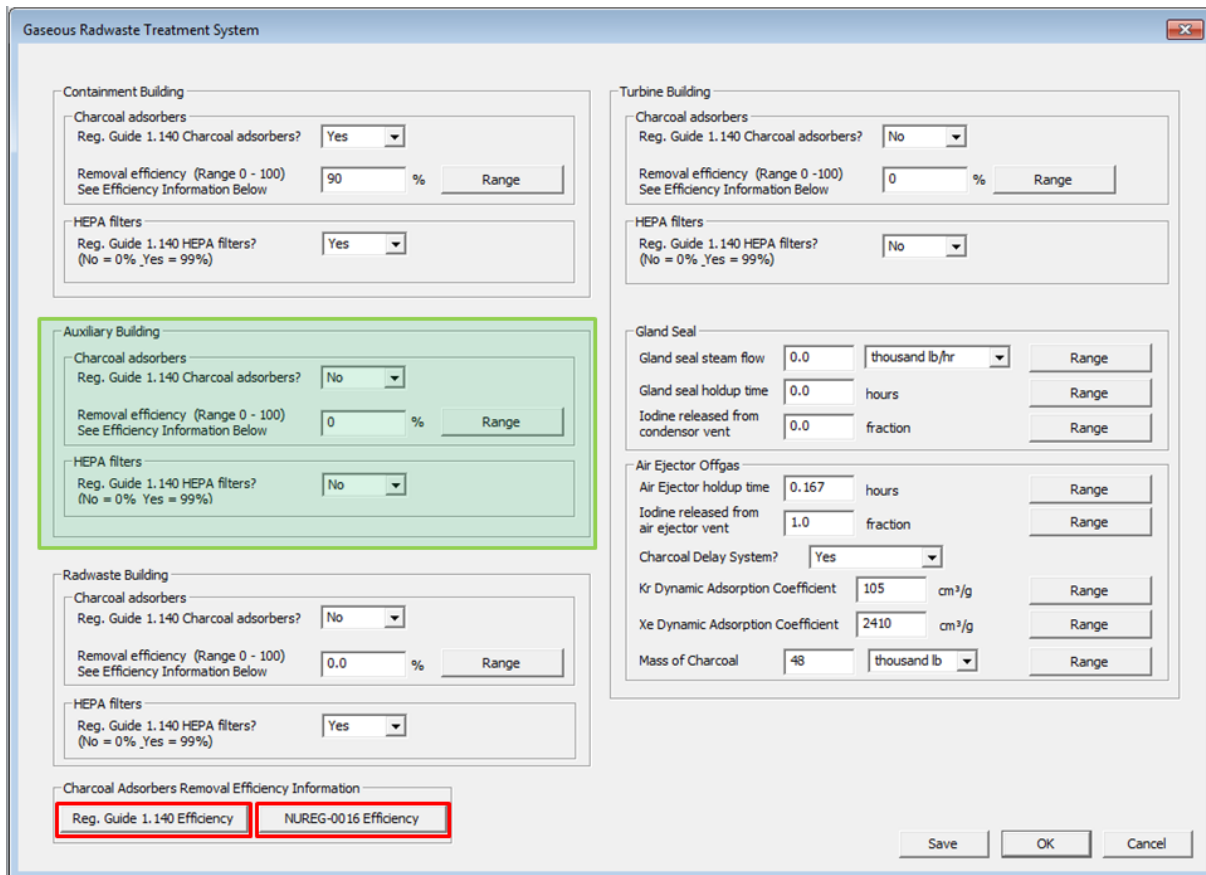
**Figure 3-48 NUREG-0016, Revision 1 Charcoal Efficiency Information Screen**



**Figure 3-49 Containment Building Charcoal Adsorber Removal Efficiency Allowable Range Screen**

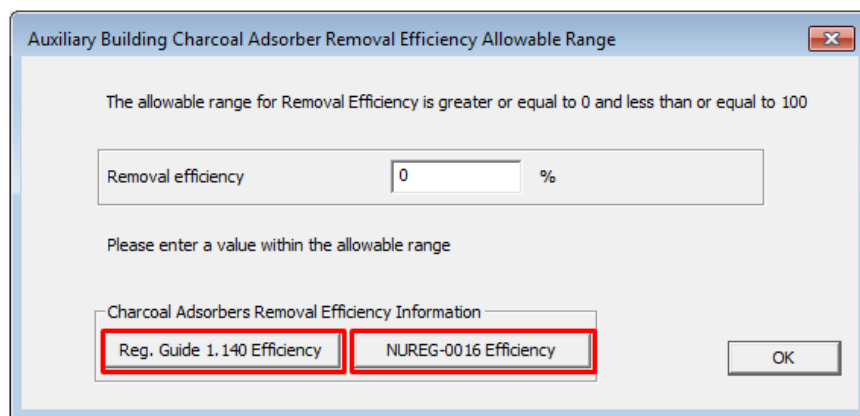
### 3.2.4.2 Auxiliary Building Section

Section 2.2.1 describes the BWR auxiliary building, while Figure 3-50 shows the inputs for the fields in the Auxiliary Building section as described below.



**Figure 3-50 Gaseous Radwaste Treatment System Screen with Auxiliary Building section options**

- Charcoal adsorbers**—If the ventilation exhaust air from the auxiliary building is treated through charcoal adsorbers that satisfy the guidelines of NUREG-0016, Revision 1, and RG 1.140, Revision 3, select “**Yes**,” and enter the percent removal efficiency for iodine corresponding to the depth of charcoal. The “Reg. Guide 1.140 Efficiency” and “NUREG-0016 Efficiency” buttons provide these reference values, as shown in Figures 3-47 and 3-48. If “**No**” is selected, the code uses a removal efficiency of 0.0 percent. Selecting the “Range” button next to this input field will open the Containment Building Charcoal Adsorber Removal Efficiency Allowable Range Screen shown in Figure 3-51. Figure 3-51 also contains the “Reg. Guide 1.140 Efficiency” and “NUREG-0016 Efficiency” buttons, providing the user with access to Figures 3-47 and 3-48. The default value from the BWRGALE.in file is “**No**” and a removal efficiency of “**0 %**.” The allowable range for values in the removal efficiency field is greater than or equal to 0.0 and less than or equal to 100.0 percent.



**Figure 3-51 Auxiliary Building Charcoal Adsorber Removal Efficiency Allowable Range Screen**

- HEPA filters**—If ventilation exhaust air from the auxiliary building is treated through HEPA filters that satisfy the guidelines of RG 1.140, Revision 3, select “**Yes**.” If “**Yes**” is selected, the code uses a removal efficiency of 99 percent for particulates. Select “**No**” if there is no treatment provided to remove particulates. If the HEPA filters do not satisfy the guidelines of RG 1.140, Revision 3, the code uses a removal efficiency of 0.0 percent. The default value from the BWRGALE.in file is “**No**.”

### 3.2.4.3 Radwaste Building Field

Section 2.2.1 describes the BWR radwaste building, while Figure 3-52 shows the inputs for the fields in the Radwaste Building section as described below:

- Charcoal adsorbers**—If the ventilation exhaust air from the radwaste building is treated through charcoal adsorbers that satisfy the guidelines of NUREG-0016, Revision 1, and RG 1.140, Revision 3, select “**Yes**,” and enter the percent removal efficiency for iodine corresponding to the depth of charcoal. The “Reg. Guide 1.140 Efficiency” and “NUREG-0016 Efficiency” buttons provide the user with these reference values, as shown in Figures 3-47 and 3-48. If “**No**” is selected, the code uses a removal efficiency of 0.0 percent. Selecting the “Range” button next to this input field will open the Radwaste Building Charcoal Adsorber Removal Efficiency Allowable Range Screen



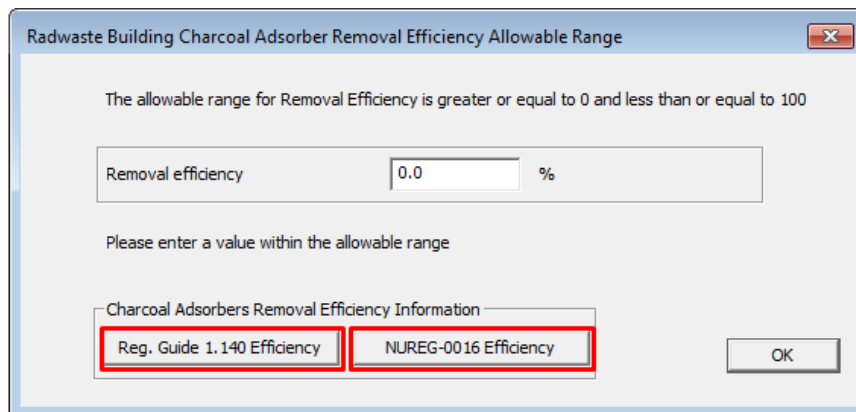
shown in Figure 3-53. Figure 3-53 also contains the “Reg. Guide 1.140 Efficiency” and “NUREG-0016 Efficiency” buttons, providing the user with access to Figures 3-47 and 3-48. The default value from the BWRGALE.in file is “No” and a removal efficiency of “0.0 %.” The allowable range for values in the removal efficiency field is greater than or equal to 0.0 and less than or equal to 100.0 percent.

The screenshot shows the 'Gaseous Radwaste Treatment System' interface. It is divided into several sections:

- Containment Building:** Charcoal adsorbers (Yes), Removal efficiency (90%), HEPA filters (Yes).
- Auxiliary Building:** Charcoal adsorbers (No), Removal efficiency (0%), HEPA filters (No).
- Radwaste Building (highlighted in yellow):** Charcoal adsorbers (No), Removal efficiency (0.0%), HEPA filters (Yes). Below this section are two buttons: 'Reg. Guide 1.140 Efficiency' and 'NUREG-0016 Efficiency', both highlighted with red boxes.
- Turbine Building:** Charcoal adsorbers (No), Removal efficiency (0%), HEPA filters (No). Gland Seal: Gland seal steam flow (0.0 thousand lb/hr), Gland seal holdup time (0.0 hours), Iodine released from condenser vent (0.0 fraction). Air Ejector Offgas: Air Ejector holdup time (0.167 hours), Iodine released from air ejector vent (1.0 fraction), Charcoal Delay System? (Yes), Kr Dynamic Adsorption Coefficient (105 cm<sup>3</sup>/g), Xe Dynamic Adsorption Coefficient (2410 cm<sup>3</sup>/g), Mass of Charcoal (48 thousand lb).

At the bottom right, there are 'Save', 'OK', and 'Cancel' buttons.

**Figure 3-52 Gaseous Radwaste Treatment System Screen with Radwaste Building section options**



**Figure 3-53 Radwaste Building Charcoal Adsorber Removal Efficiency Allowable Range Screen**

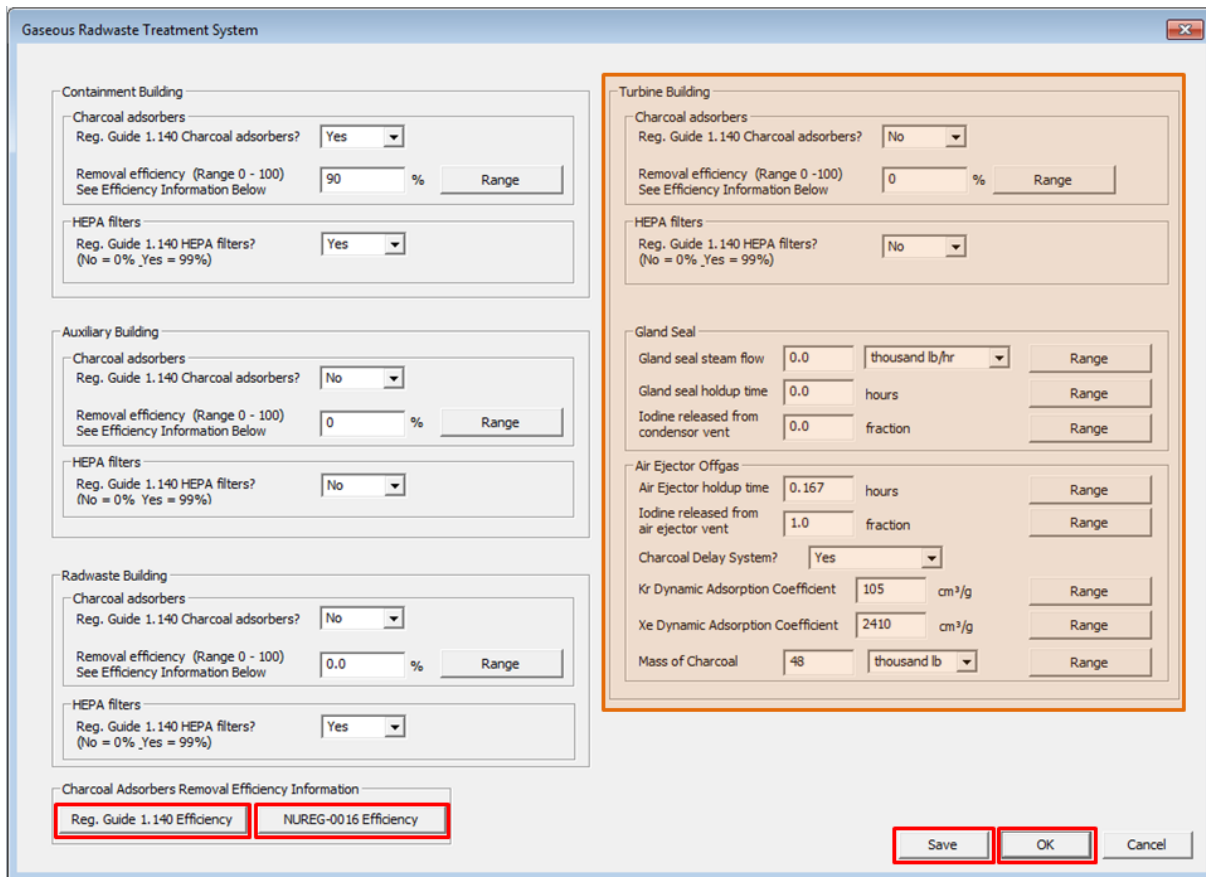
- HEPA filters**—If ventilation exhaust air from the radwaste building is treated through HEPA filters that satisfy the guidelines of RG 1.140, Revision 3, select “**Yes**.” If “**Yes**” is selected, the code uses a removal efficiency of 99 percent for particulates. Select “**No**” if there is no treatment provided to remove particulates. If the HEPA filters do not satisfy the guidelines of RG 1.140, Revision 3, the code uses a removal efficiency of 0.0 percent. The default value from the BWRGALE.in file is “**Yes**.”

#### 3.2.4.4 Turbine Building Field

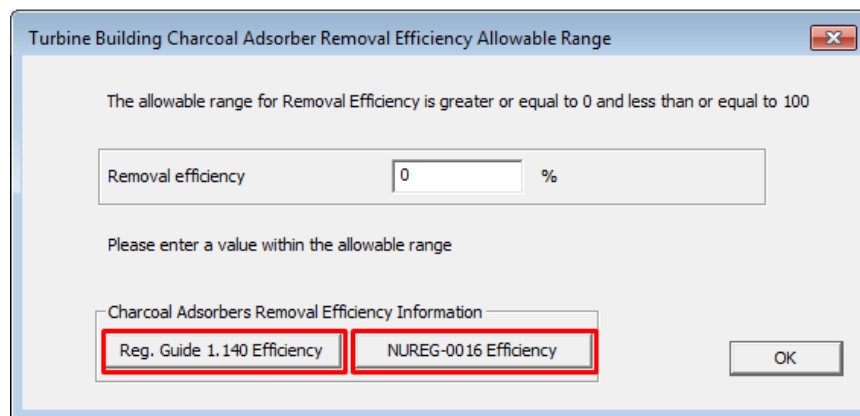
Section 2.2.1 describes the BWR turbine building, while Figure 3-54 shows the inputs for the fields in the Turbine Building section as described below:

- Charcoal adsorbers**—If the ventilation exhaust air from the turbine building is treated through charcoal adsorbers that satisfy the guidelines of NUREG-0016, Revision 1, and RG 1.140, Revision 3, select “**Yes**” and enter the percent removal efficiency for iodine corresponding to the depth of charcoal. The “Reg. Guide 1.140 Efficiency” and “NUREG-0016 Efficiency” buttons provide these reference values, as shown in Figures 3-47 and 3-48. If “**No**” is selected, the code uses a removal efficiency of 0.0 percent. Selecting the “Range” button next to this input field will open the Turbine Building Charcoal Adsorber Removal Efficiency Allowable Range Screen shown in Figure 3-55. Figure 3-55 also contains the “Reg. Guide 1.140 Efficiency” and “NUREG-0016 Efficiency” buttons, providing the user with access to Figures 3-47 and 3-48. The default value from the BWRGALE.in file is “**No**,” and the removal efficiency is “**0 %**.” The allowable range for values in the removal efficiency field is greater than or equal to 0.0 and less than or equal to 100.0 percent.
- HEPA filters**—If ventilation exhaust air from the turbine building is treated through HEPA filters that satisfy the guidelines of RG 1.140, Revision 3, select “**Yes**.” If “**Yes**” is selected, the code uses a removal efficiency of 99 percent for particulates. Select “**No**” if there is no treatment provided to remove particulates; if the HEPA filters do not satisfy the guidelines of RG 1.140, Revision 3, the code uses a removal efficiency of 0.0 percent. The default value from the BWRGALE.in file is “**No**.”





**Figure 3-54 Gaseous Radwaste Treatment System Screen with turbine building section options**

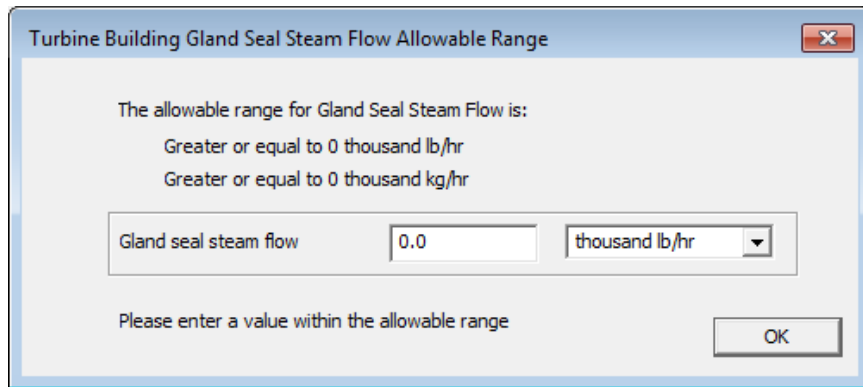


**Figure 3-55 Turbine Building Charcoal Adsorber Removal Efficiency Allowable Range Screen**

- Gland Seal—The user should enter the gland seal steam flow, the design holdup time for gases vented from the gland seal condenser, and the fraction of radioiodine released from the turbine gland seal condenser vent.

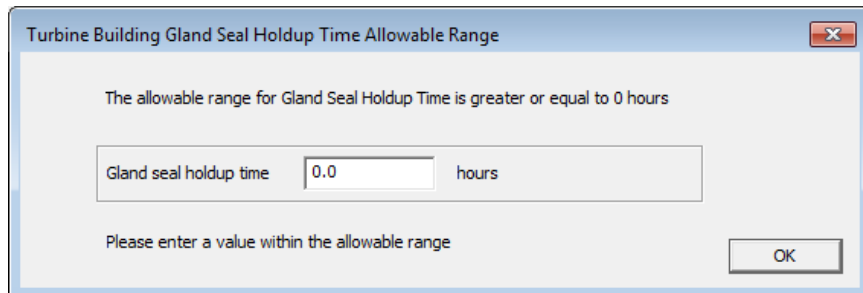
- Enter the gland seal steam flow in either units of thousands of pounds per hour (thousand lb/h) or thousands of kilograms per hour (thousand kg/h) to the turbine gland seal, as follows:
  - (1) If main steam is used for the sealing steam, enter a flow rate 0.001 times the main steam flow entered previously for Total Steam Flow, as described in Section 3.3.1.3.
  - (2) If clean (nonradioactive) steam from an auxiliary boiler is used for sealing steam, enter 0.0.

Selecting the “Range” button next to this input field will open the Turbine Building Gland Seal Steam Flow Allowable Range Screen shown in Figure 3-56. The default value from the BWRGALE.in file is “0.0” thousand lb/h, and the allowable range for values in this field is greater than or equal to either 0 thousand lb/h or 0 thousand kg/h.



**Figure 3-56 Turbine Building Gland Seal Steam Flow Allowable Range Screen**

- Enter the gland seal holdup time in hours for gases vented from the gland seal condenser. Selecting the “Range” button next to this input field will open the Turbine Building Gland Seal Holdup Time Allowable Range Screen shown in Figure 3-57. The default value from the BWRGALE.in file is “0.0” h, and the allowable range for values in this field is greater than or equal to 0.0 h.

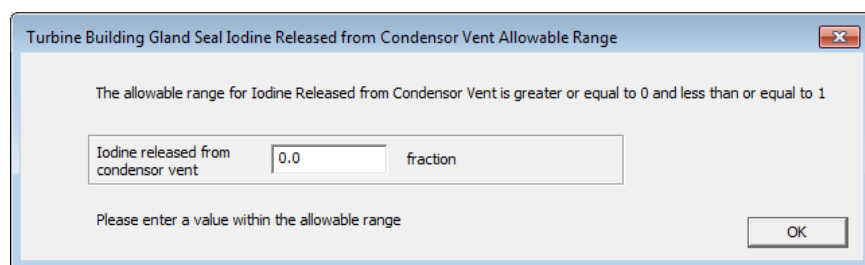


**Figure 3-57 Turbine Building Gland Seal Holdup Time Allowable Range Screen**

- Enter the fraction of the iodine released from the condenser vent, as follows:

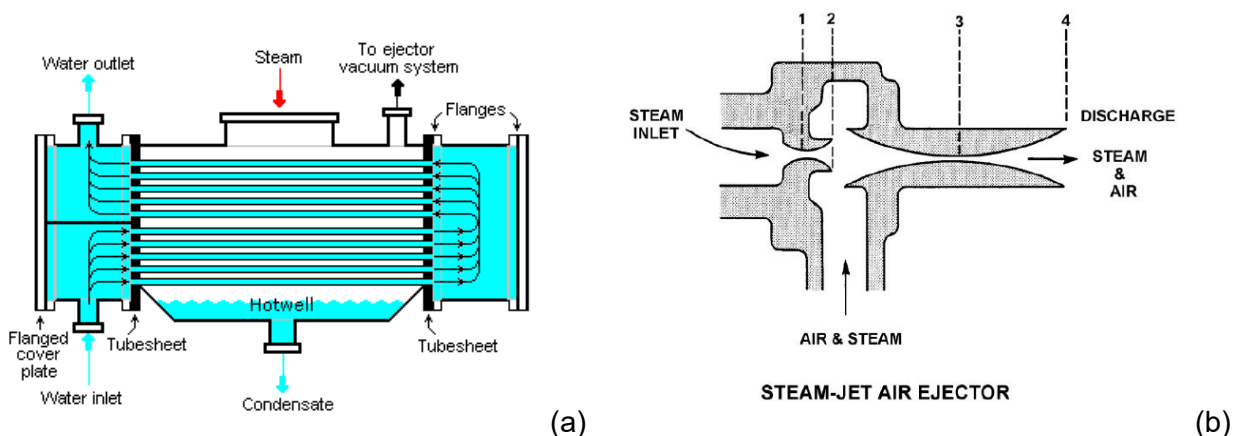
- (1) If before release, the offgases from the turbine gland seal condenser vent are processed through charcoal adsorbers that satisfy the guidelines of RG 1.140, Revision 3, enter the removal efficiency as a fraction for radioiodine corresponding to the depth of charcoal, as indicated in Figure 3-47.
- (2) If the offgases are released from the turbine gland seal condenser without treatment, if clean steam is used, or if the charcoal adsorbers provided do not satisfy the guidelines of RG 1.140, Revision 3, enter “0.0.”

Selecting the “Range” button next to this input field will open the Turbine Building Gland Seal Iodine Released from Condenser Vent Allowable Range Screen shown in Figure 3-58. The default value from the BWRGALE.in file is “0.0” fraction, and the allowable range for values in this field are between 0.0 and 1.0.



**Figure 3-58 Turbine Building Gland Seal Iodine Released from Condenser Vent Allowable Range Screen**

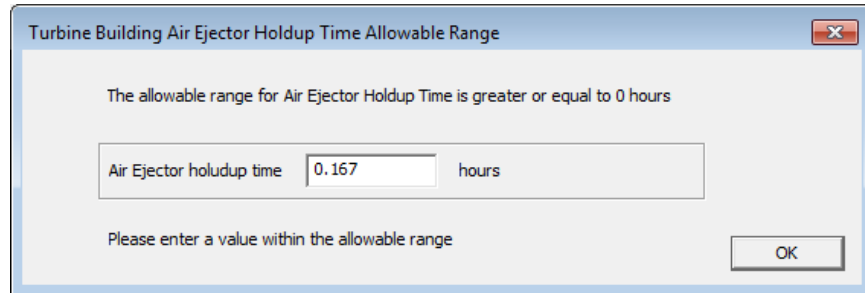
- Air Ejector Offgas—In the turbine condenser, which condenses the steam that passes through the turbine, an air ejector is used to remove air from the condenser by passing steam through a series of nozzles to create a vacuum that removes air from the condenser. Figure 3-59 is a diagram of this system showing the turbine condenser and air ejector.



**Figure 3-59 Diagram of turbine condenser (a) and steam-jet air ejector (b)**

- Enter the air ejector holdup time in hours for offgases from the main condenser air ejector before being processed through the offgas treatment system (e.g., a

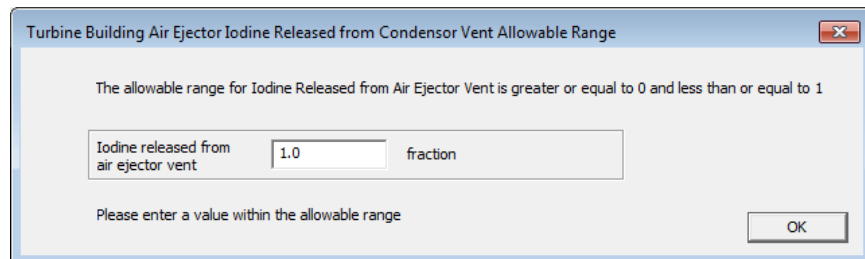
10-minute holdup time before cryogenic distillation). Selecting the “Range” button next to this input field will open the Turbine Building Air Ejector Holdup Time Allowable Range Screen shown in Figure 3-60. The default value from the BWRGALE.in file is “0.167” hours, and the allowable range for values in this field is greater than or equal to 0.0 hours.



**Figure 3-60 Turbine Building Air Ejector Holdup Time Allowable Range Screen**

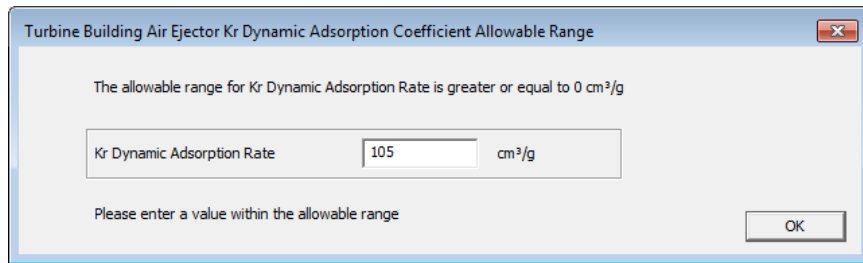
- Enter the fraction of the iodine released from the air ejector vent, as follows:
  - (1) If before release, the offgases from the air ejector vent are processed through charcoal adsorbers that satisfy the guidelines of RG 1.140, Revision 3, enter the removal efficiency as a fraction for radioiodine corresponding to the depth of charcoal, as indicated in Figure 3-47.
  - (2) If the offgas is released without treatment or through charcoal adsorbers that do not satisfy the guidelines of RG 1.140, Revision 3, enter “0.0.”
  - (3) If the offgas is processed through a charcoal delay system, enter 1.0 for the fractional removal efficiency.
  - (4) If the offgas is processed through a cryogenic distillation system (removal of radioiodine by the cryogenic distillation system is built into the code), enter 0.0 for the fractional removal efficiency.

Selecting the “Range” button next to this input field will open the Turbine Building Air Ejector Iodine Released from Condenser Vent Allowable Range Screen shown in Figure 3-61. The default value from the BWRGALE.in file is “1.0” fraction, and the allowable range for values in this field are between 0.0 and 1.0.



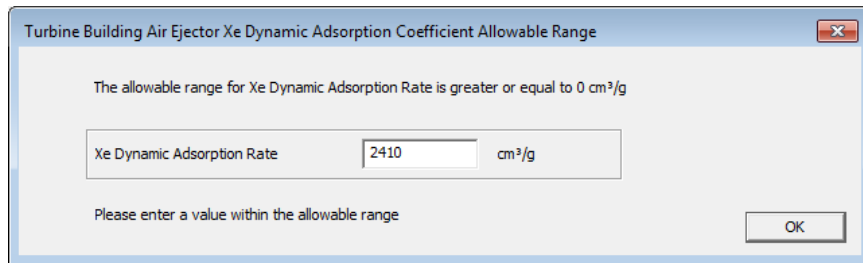
**Figure 3-61 Turbine Building Air Ejector Iodine Released from Condenser Vent Allowable Range Screen**

- Enter the status of treatment of the air ejector vent offgas through a charcoal delay system. There are three options to select from in the dropdown menu: “Yes,” “No,” and “Cryogenic distill.” The default value from the BWRGALE.in file is “**Yes**.”
- Enter the krypton (Kr) dynamic adsorption coefficient in units of cubic centimeters per gram ( $\text{cm}^3/\text{g}$ ) if a charcoal delay system is used to treat the offgases from the air ejector vent. Enter the dynamic adsorption coefficient for krypton based on the system design and the dynamic adsorption coefficients, as noted in Table 4-26. If a charcoal delay system is not used, enter “0.0.” Selecting the “Range” button next to this input field will open the Turbine Building Air Ejector Kr Dynamic Adsorption Coefficient Allowable Range Screen shown in Figure 3-62. The default value from the BWRGALE.in file is “**105**”  $\text{cm}^3/\text{g}$ , and the allowable range for values in this field is greater than or equal to  $0.0 \text{ cm}^3/\text{g}$ .



**Figure 3-62 Turbine Building Air Ejector Kr Dynamic Adsorption Coefficient Allowable Range Screen**

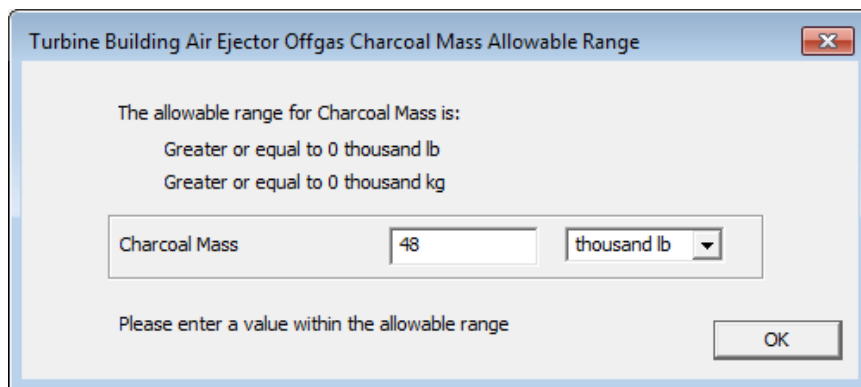
- Enter the xenon (Xe) dynamic adsorption coefficient in units of cubic centimeters per gram ( $\text{cm}^3/\text{g}$ ) if a charcoal delay system is used to treat the offgases from the air ejector vent. Enter the dynamic adsorption coefficient for Xe based on the system design and the dynamic adsorption coefficients, as noted in Table 4-26. If a charcoal delay system is not used, enter “0.0.” Selecting the “Range” button next to this input field will open the Turbine Building Air Ejector Xe Dynamic Adsorption Coefficient Allowable Range Screen shown in Figure 3-63. The default value from the BWRGALE.in file is “**2410**”  $\text{cm}^3/\text{g}$ , and the allowable range for values in this field is greater than or equal to  $0.0 \text{ cm}^3/\text{g}$ .



**Figure 3-63 Turbine Building Air Ejector Xe Dynamic Adsorption Coefficient Allowable Range Screen**

- Enter the mass of the charcoal in either units of thousands of pounds (thousand lb) or thousands of kilograms (thousand kg) if a charcoal delay system

is used to treat the offgases from the air ejector vent. If a charcoal delay system is not used, enter "0.0." Selecting the "Range" button next to this input field will open the Turbine Building Air Ejector Offgas Charcoal Mass Allowable Range Screen shown in Figure 3-64. The default value from the BWRGALE.in file is "48" thousand lb, and the allowable range for values in this field is greater than or equal to either 0 thousand lb or 0 thousand kg.



**Figure 3-64 Turbine Building Air Ejector Offgas Charcoal Mass Allowable Range Screen**

Once the user has finished entering the Gaseous Radwaste Treatment System Screen parameters, select the "Save" button to save any changes made to the gaseous radwaste input parameters. Then select the "OK" button to close the Liquid Radwaste Treatment System Screen (Figure 3-54) and return to the General Reactor Parameters Screen (Figure 3-8). The user should note that selecting the "OK" button without first selecting the "Save" button will result in the code returning to the General Reactor Parameters Screen without saving any changes made to the gaseous radwaste input parameters. Therefore to save any changes made to the inputs, the user should select the "Save" button before selecting the "OK" button.

Upon returning to the General Reactor Parameters Screen (Figure 3-8), the user can either choose to save the GALE-BWR 3.2 inputs or execute the code by selecting either the "Save" or "Run" button, respectively. If the user chose the "Save" button after completing the liquid and gaseous radwaste inputs, there is no need to select the "Save" button on the General Reactor Parameters Screen (Figure 3-8), as all input changes previously made have been saved. If the "Save" button was not chosen, then it is recommended that the user select the "Save" button on the General Reactor Parameters Screen (Figure 3-8). To run the GALE-BWR 3.2 code, the user should press the "Run" button in the General Reactor Parameters Screen (Figure 3-8).

When the code has completed the calculations the GALE-BWR 3.2 dialog box opens to inform the user that the run is complete and that results are available, as shown in Figure 3-65. When the user presses the "Exit" button, the code will open Windows Explorer in the directory containing the input and output files, as shown in Figure 3-66. Note that the code will save the BWRGALE.in and the two output files to the working directory set by the user in Section 3.1. The two output files generated by the code are BWRGE.out for the gaseous effluent output and BWRLE.out for the liquid effluent output file.

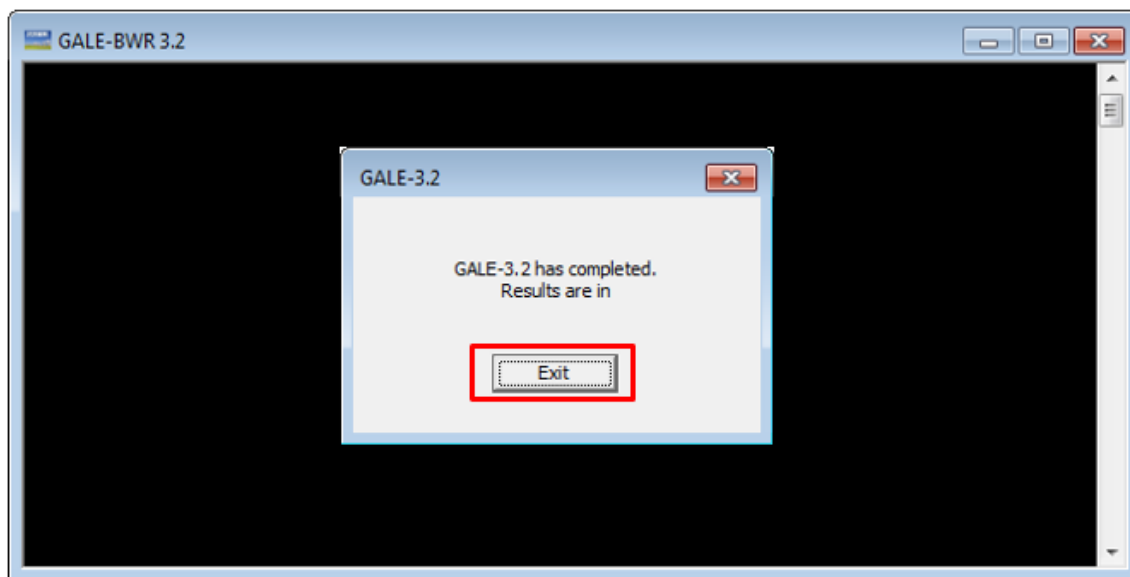


Figure 3-65 GALE code completion dialog box

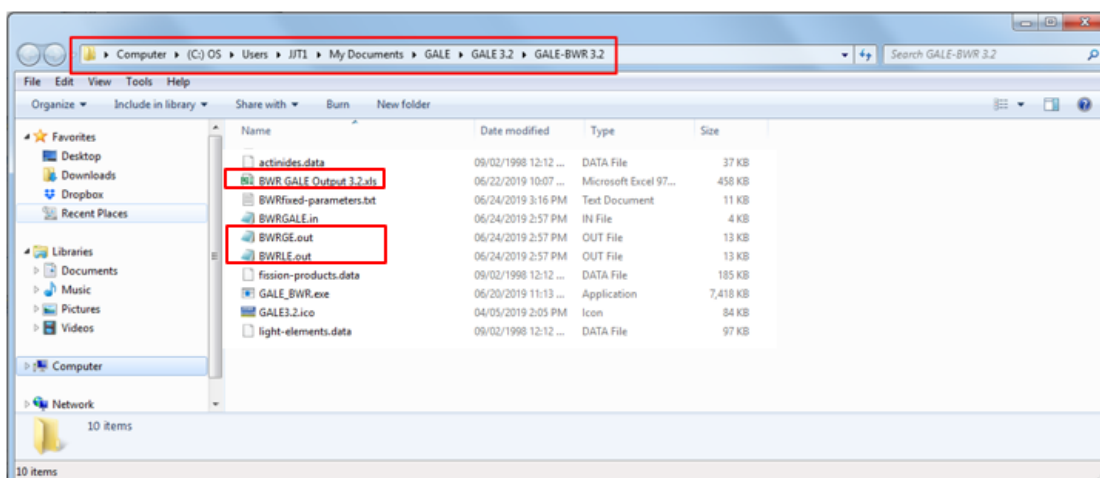


Figure 3-66 Windows Explorer directory with GALE-BWR 3.2 input and output files

### 3.3 Viewing the Code Outputs

The code's two output files, BWRGE.out and BWRLE.out, are ASCII text files and may be viewed using any text editor, such as Notepad, as shown in Figures 3-67 and 3-68. Both output files display the fixed model parameters that the user selected (GALE Version) and the source term (ANS-18.1 Version) at the very top of the output files. Under these data, the output files also list any modifications that the user requested to the GALE fixed parameter file, followed by the echo if the input parameters from the General Reactor Parameter, Liquid Radwaste Treatment System Input, and Gaseous Radwaste Treatment System Input Screens. Lastly, the output files display the liquid and gaseous effluent releases.

Alternatively, the user may choose an Excel file to view the code outputs. To use this file, open the Excel file, BWR GALE Output 3.2.xls, in the working directory shown in Figure 3-66.

Figure 3-69 displays the BWR GALE Output 3.2.xls file and the first tab, specifying the output file names along with the respective locations of these files in the working directory. The user also has the option of browsing for these files by selecting the “Browse” button. Selection of the “Read GE Data” and “Read LE Data” buttons will cause the Excel file to read the BWRGE and BWRLE output files listed in the cells above them to the spreadsheet. Once this is done, the BWRGE and BWRLE outputs and some plots are available on the other tabs of the BWR GALE Output 3.2.xls file for easier visualization or for copying into other programs or spreadsheets.



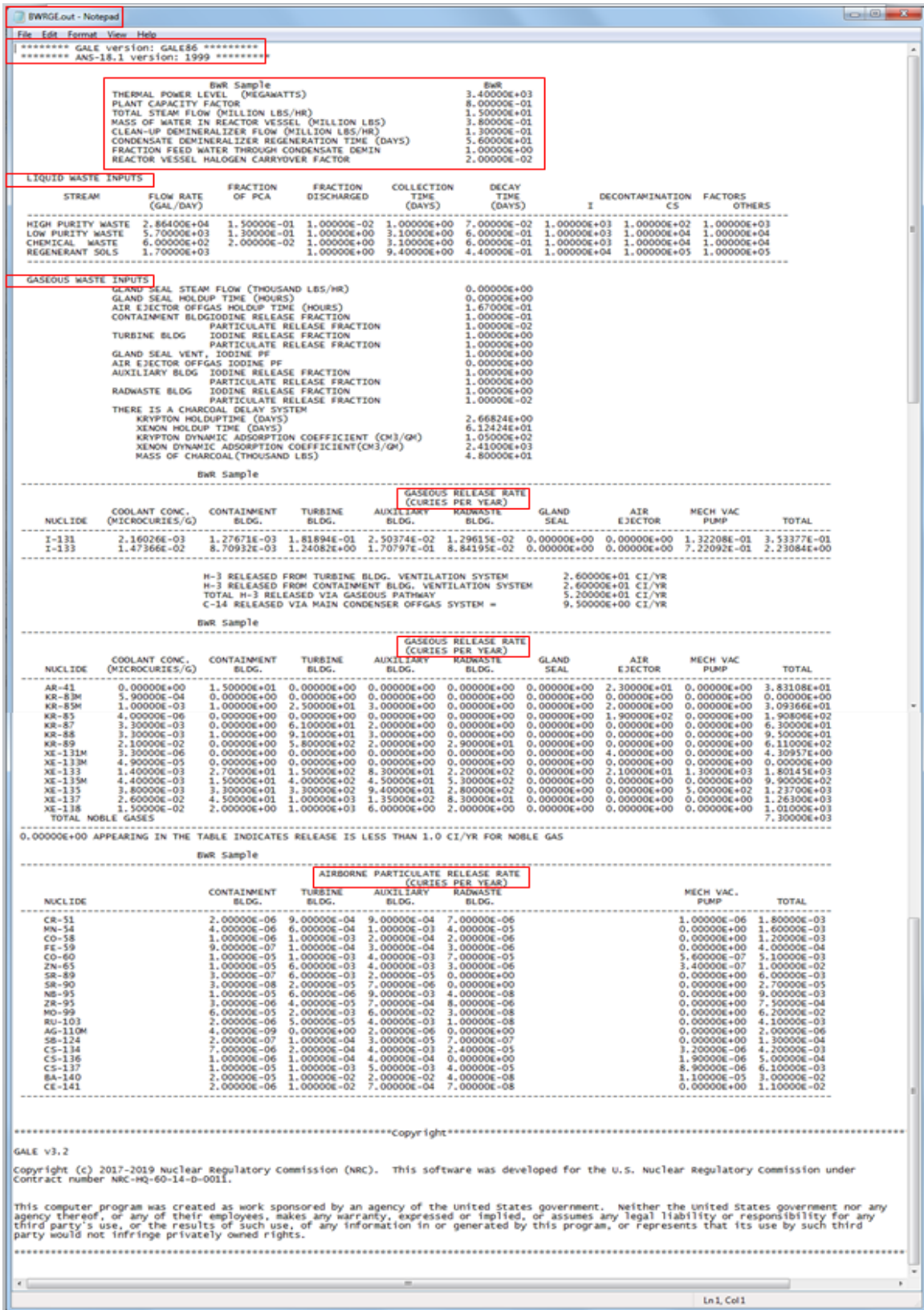


Figure 3-67 BWRGE.out file

BWRLE.out - Notepad

\*\*\*\*\* GALE v3.2 \*\*\*\*\*  
\*\*\*\*\* ANS-18.1 version: 1999 \*\*\*\*\*

BWR Sample		BWR	
THERMAL POWER LEVEL (MEGAWATTS)		1.00000E+03	
PLANT CAPACITY FACTOR		8.00000E-01	
TOTAL STEAM FLOW (MILLION LBS/HR)		1.50000E+01	
MASS OF WATER IN REACTOR VESSEL (MILLION LBS)		1.80000E+01	
CLEAN-UP DEMINERALIZER FLOW (MILLION LBS/HR)		1.30000E-01	
CONDENSATE DEMINERALIZER REGENERATION TIME (DAYS)		5.60000E+01	
FISSION PRODUCT CARRY-OVER FRACTION		1.00000E-01	
HALOGEN CARRY-OVER FRACTION		2.00000E-02	
FRACTION FEED WATER THROUGH CONDENSATE DEMIN		1.00000E+00	

LIQUID WASTE INPUTS

STREAM	FLOW RATE (GAL/DAY)	FRACTION OF PCA	FRACTION DISCHARGED	COLLECTION TIME (DAYS)	DECAY TIME (DAYS)	DECONTAMINATION FACTORS	
						I	CS
HIGH PURITY WASTE	2.84000E+04	1.50000E-01	1.00000E-02	1.00000E+00	7.00000E-02	1.00000E+03	1.00000E+02
LOW PURITY WASTE	5.70000E+01	1.30000E-01	1.00000E+00	1.10000E+00	8.00000E-01	1.00000E+03	1.00000E+04
CHEMICAL WASTE	6.00000E+02	2.00000E-02	1.00000E+00	1.10000E+00	8.00000E-01	1.00000E+03	1.00000E+04
REGENERANT SOLS	1.70000E+01	1.00000E+00	1.00000E+00	9.40000E+00	4.40000E-01	1.00000E+04	1.00000E+05

GASEOUS WASTE INPUTS

CONCENTRATION IN PRIMARY COOLANT (MICRO-CI/ML)		LIQUID EFFLUENTS ANNUAL RELEASES TO DISCHARGE CANAL		TOTAL LWS (CURIES)	ADJUSTED TOTAL (CI/YR)	DETERGENT WASTES (CI/YR)	TOTAL (CI/YR)
CORROSION AND ACTIVATION PRODUCTS							
NA 24	6.25000E-03	7.00000E-05	3.00000E-05	0.00000E+00	1.00000E-04	4.20000E-04	0.00000E+00
P 32	1.42998E+01	4.00000E-05	0.00000E+00	0.00000E+00	1.00000E-05	4.00000E-05	1.80000E-04
CR 51	2.77997E+01	2.97000E-03	1.70000E-04	2.90000E-04	5.40000E-04	1.01000E-03	4.45000E-03
HN 54	3.02996E+02	0.00000E-05	0.00000E+00	0.00000E+00	1.00000E-05	2.00000E-05	9.00000E-05
MN 56	1.07500E-01	2.48600E-02	1.50000E-04	0.00000E+00	1.50000E-04	6.60000E-04	0.00000E+00
FE 55	9.49624E+02	9.90000E-04	6.00000E-05	1.00000E-04	4.20000E-04	5.80000E-04	2.55000E-03
FE 59	4.49994E+01	3.00000E-05	0.00000E+00	0.00000E+00	1.00000E-05	5.00000E-05	2.20000E-03
CO 58	7.12991E+01	1.00000E-04	1.00000E-05	1.00000E-05	5.00000E-05	2.00000E-04	9.90000E-03
CO 60	1.92116E+03	2.00000E-04	1.00000E-05	2.00000E-05	8.00000E-05	5.10000E-04	1.40000E-02
NI 63	3.16021E+04	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	1.70000E-03
CU 64	5.33131E-01	9.70000E-03	9.00000E-05	1.00000E-04	5.00000E-04	1.00000E-04	5.50000E-04
ZN 65	2.44997E+02	1.00000E-03	6.00000E-05	1.00000E-04	3.90000E-04	5.50000E-04	2.43000E-03
ZN 69M	5.75000E-01	1.98000E-03	6.00000E-05	3.00000E-05	0.00000E+00	9.00000E-05	3.90000E-04
NI 67	9.95813E-01	3.00000E-04	1.00000E-05	1.00000E-05	0.00000E+00	2.00000E-05	9.00000E-05
NP239	2.34997E+00	7.92000E-03	4.00000E-04	4.50000E-04	2.00000E-05	8.60000E-04	3.82000E-03
FISSION PRODUCTS							
BR 83	1.00417E-01	5.92000E-03	3.00000E-05	0.00000E+00	0.00000E+00	4.00000E-05	1.60000E-04
SR 89	5.19994E+01	1.00000E-04	1.00000E-05	1.00000E-05	4.00000E-05	1.90000E-04	9.00000E-05
SR 90	1.00000E+04	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	2.00000E-05	1.00000E-05
SR 91	4.02937E-01	3.97000E-03	1.00000E-04	3.00000E-05	0.00000E+00	1.30000E-04	5.60000E-04
Y 91	5.87993E+01	4.00000E-05	0.00000E+00	1.00000E-05	2.00000E-05	3.00000E-05	1.50000E-04
SR 92	1.12917E-01	9.84000E-05	0.00000E+00	0.00000E+00	6.00000E-05	2.80000E-04	0.00000E+00
Y 92	1.47083E-01	5.96000E-03	1.40000E-04	1.00000E-05	0.00000E+00	1.50000E-04	6.80000E-04
ZR 93	4.25000E-01	3.97000E-03	1.00000E-04	3.00000E-05	0.00000E+00	1.30000E-04	5.90000E-04
ZR 95	6.49992E+01	1.00000E-05	0.00000E+00	0.00000E+00	0.00000E+00	2.00000E-05	1.10000E-03
NB 95	3.49998E+01	1.00000E-05	0.00000E+00	0.00000E+00	0.00000E+00	2.00000E-05	1.90000E-03
NB 98	3.54179E-02	3.99000E-03	0.00000E+00	0.00000E+00	0.00000E+00	1.00000E-05	1.00000E-05
MO 99	2.79167E+00	1.98000E-03	1.00000E-04	1.20000E-04	1.00000E-05	2.30000E-04	1.00000E-03
TC 99M	2.50000E-01	1.99000E-03	1.00000E-04	1.20000E-04	1.00000E-05	2.20000E-04	1.00000E-03
RU103	3.95995E+01	2.00000E-05	0.00000E+00	0.00000E+00	0.00000E+00	3.00000E-05	2.90000E-04
RU105	1.85000E-01	1.99000E-03	2.00000E-05	0.00000E+00	0.00000E+00	3.00000E-05	1.10000E-04
RU106	3.66995E+02	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	1.00000E-05	8.90000E-03
AG110M	2.52997E+02	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	1.20000E-03
TE112M	3.39998E+01	4.00000E-05	0.00000E+00	0.00000E+00	1.00000E-05	1.00000E-05	0.00000E+00
TE113M	1.25000E+00	1.00000E-04	0.00000E+00	0.00000E+00	0.00000E+00	3.00000E-05	0.00000E+00
I131	8.04990E+00	2.16000E-03	1.20000E-04	1.85000E-03	1.64300E-02	8.14100E-02	1.60000E-03
I132	9.58131E-02	1.05900E-02	1.20000E-04	1.00000E-05	0.00000E+00	1.20000E-04	5.40000E-04
I133	8.75000E-01	1.47400E-02	5.00000E-04	3.49000E-03	7.00000E-05	1.30000E-03	8.28000E-02
I134	3.66679E-02	4.23600E-02	4.00000E-05	0.00000E+00	0.00000E+00	4.00000E-05	1.60000E-04
CS134	7.48742E+02	3.00000E-05	2.00000E-05	0.00000E+00	1.00000E-05	2.00000E-05	1.10000E-02
I135	2.79167E-01	2.16400E-02	4.00000E-04	6.50000E-04	1.00000E-05	1.06000E-03	4.70000E-03
CS136	1.29998E+01	2.00000E-05	1.00000E-05	0.00000E+00	0.00000E+00	6.00000E-05	3.70000E-04
CS137	1.09572E+04	7.00000E-05	4.00000E-05	1.00000E-05	2.00000E-05	7.00000E-05	3.00000E-04
CS138	2.26198E-02	9.86000E-03	2.00000E-05	0.00000E+00	0.00000E+00	1.00000E-04	0.00000E+00
BA139	5.75714E-02	9.96000E-03	2.00000E-05	0.00000E+00	0.00000E+00	9.00000E-05	0.00000E+00
BA140	1.27998E+01	4.00000E-04	2.00000E-05	4.00000E-05	3.00000E-05	9.00000E-05	4.00000E-04
CL141	3.29996E+01	3.00000E-05	0.00000E+00	0.00000E+00	1.00000E-05	5.00000E-05	2.80000E-04
LA142	6.38911E-02	4.98000E-03	1.00000E-05	0.00000E+00	0.00000E+00	7.00000E-05	0.00000E+00
CL143	1.37500E+00	3.00000E-05	0.00000E+00	0.00000E+00	0.00000E+00	1.00000E-05	0.00000E+00
PR143	1.36998E+01	4.00000E-05	0.00000E+00	0.00000E+00	0.00000E+00	4.00000E-05	0.00000E+00
CL144	2.83998E+02	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	1.00000E-05	3.90000E-03
ALL OTHERS	2.01000E-01	2.30000E-04	9.00000E-05	7.00000E-05	3.90000E-04	1.72000E-03	0.00000E+00
TOTAL (EXCEPT TRITIUM)	4.14590E+01	3.39000E-03	7.58000E-03	1.82400E-02	2.92100E-02	1.29210E-01	8.93300E-02

TRITIUM RELEASE 5.10000E+01 CURIES PER YEAR

NOTE: 0.00000E+00 INDICATES THAT THE VALUE IS LESS THAN 1.00000E-5.

ALL OTHERS REFERS TO: ZN 65, ZN 69, U235, PU239, BR 84, BR 85, BR 89, Y 90, Y 91M, ZR 93, NB 93M, NB 99M, ZR 97, NB 97M, NB 97, TC 101, RH103M, TC104, RH105M, RH105, RH106, AG110, TE129, I129, TE131, TE132, CS135, BA137M, LA140, BA141, LA141, BA142, PR144, ND147, PR147

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GALE v3.2

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Ln1, Col1

Figure 3-68 BWRLE.out file

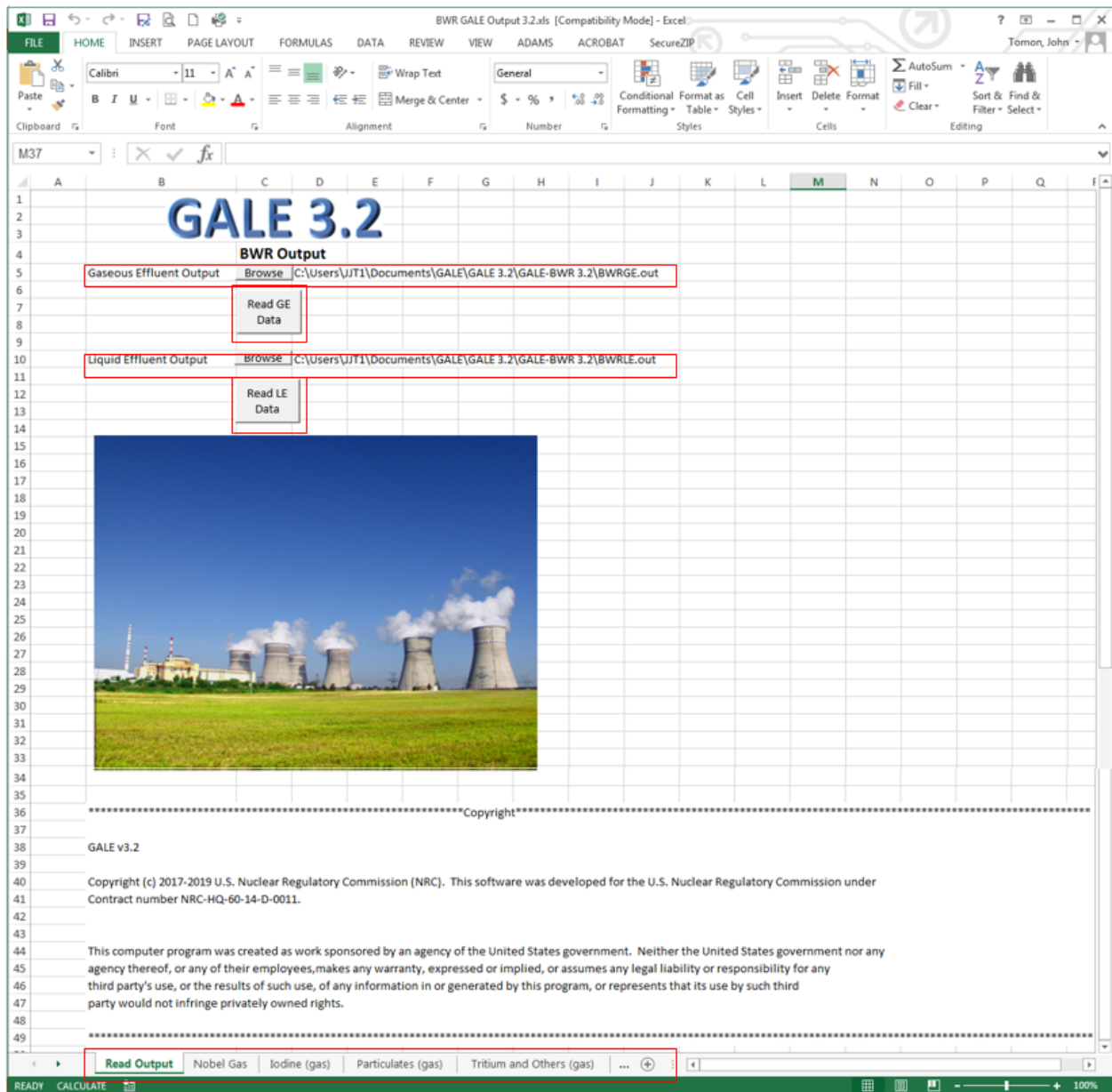


Figure 3-69 BWR GALE Output 3.2.xls file



## 4.0 MODELING PARAMETERS

This section discusses the GALE version option and the reactor coolant source term (ANS-18.1 Version) options available in GALE-BWR 3.2. In the code, the options for the ANS-18.1 version and GALE version are defaulted to ANSI/ANS-18.1-1999 source term and GALE86 reactor design parameters (NUREG-0016, Revision 1), respectively. These GALE-BWR 3.2 default values are consistent with the guidance in references DC/COL-ISG-05; RG 1.112, Revision 1; and NUREG-0800. Note that, when the source term or reactor design parameters differ from those given in ANSI/ANS-18.1-1999 and the GALE86 code, DC/COL-ISG-05, and RG 1.112, Revision 1, they should be described with sufficient detail, and the basis of the alternate method, model parameters, and assumptions should be provided, to allow the NRC to conduct an independent evaluation. Section 4.1 discusses the GALE version option available, the default GALE86 modeling parameters, which is shown in Figure 3-3. Section 4.2 discusses the reactor coolant source term, ANS-18.1 version options, and radionuclide concentrations in the primary and secondary coolant options available to the user (either the default 1999, 1984, or the 2016 options, which are shown in Figure 3-4). Finally, Section 4.3 discusses the user option to modify the certain GALE-BWR fixed modeling parameters by means of the BWRfixed-parameters.txt file.

### 4.1 GALE Version (Fixed Model Parameters)

#### 4.1.1 GALE86 Fixed Modeling Parameters

##### 4.1.1.1 *Thermal Power Level*

###### Parameter

This is the maximum thermal power level (in MWt) evaluated for safety considerations in the DCD or FSAR.

###### Bases

The power level used in the source term in the code is the maximum power level evaluated for safety considerations in the DCD or FSAR. 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. Experience shows that most utilities request approval to operate at maximum power soon after reaching commercial operation.

##### 4.1.1.2 *Plant Capacity Factor*

###### Parameter

A plant capacity factor of 80 percent is used (i.e., 292 effective full-power days).

###### Bases

The source term calculations are based on a plant capacity factor of 80 percent averaged over the 30-year operating life of the plant (i.e., the plant operates at 100-percent power 80 percent of the time). Table 4-1 lists the plant capacity factors experienced at BWRs for the period 1972 through 1977.



The average plant capacity factors shown in Table 4-1 indicate that the 80-percent 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-percent capacity factor when averaged over 30 years of operation.

**Table 4-1 Plant capacity factors at operating BWRs**

Facility <sup>a</sup>	Date of Commercial Operation <sup>c</sup>	Percent Capacity Factor <sup>b</sup>					
		1972	1973	1974	1975	1976	1977
Oyster Creek	12/69	77	65	66	58	70	58 <sup>e</sup>
Nine Mile Point-1	12/69	62	68	63	60	81	57 <sup>f</sup>
Millstone-1	03/71	55	34 <sup>d</sup>	63	68	66	84
Monticello	06/71	74	68	57	61	84	75
Dresden-3	11/71	67	54	47 <sup>e</sup>	33 <sup>f</sup>	60	76
Dresden-2	06/72		74	51	44 <sup>f</sup>	66	54
Vermont Yankee	11/72		44 <sup>e</sup>	59	81	73	80
Pilgrim-1	12/72		72	34 <sup>f</sup>	46 <sup>e</sup>	43 <sup>f</sup>	47 <sup>f</sup>
Quad Cities-1	02/73			51 <sup>f</sup>	65	52 <sup>f</sup>	55 <sup>e</sup>
Quad Cities-2	03/73			68	40 <sup>f</sup>	66	67
Cooper	07/74				60	57	70
Peach Bottom-2	07/74				57	61	45 <sup>f</sup>
Peach Bottom-3	12/74				59	67	54 <sup>f</sup>
Duane Arnold	02/75					55	67
FitzPatrick	07/75					59	55 <sup>f</sup>
Brunswick-2	11/75					37 <sup>g</sup>	35 <sup>g</sup>
Hatch-1	12/75					65	57
Average		67	67	61	63	66	70

<sup>a</sup> These values are from monthly Operating Units Status Reports and Table 2-1 of NUREG-0016, Revision 1.

<sup>b</sup> Big Rock Point, Dresden 1, Humboldt Bay, and Lacrosse are not included because they are small reactors (<700 MWt) and are not considered to be typical of modern-day reactors. Browns Ferry 1 and 2 are not included because they were not operating as a result of fire.

<sup>c</sup> Plant capacity factors listed are for the first full year of commercial operation.

<sup>d</sup> Not included because of extended maintenance outage to replace feedwater sparger.

<sup>e</sup> Not included because of extended operation at reduced power.

<sup>f</sup> Not included because of extended refueling outage.

<sup>g</sup> Not included because of extended maintenance outage to correct power monitor tube vibrations.

#### 4.1.1.3 Gaseous Releases from Building Ventilation Systems

##### Parameter

Table 4-2 shows the noble gas and radioactive particulate releases from ventilation systems for facilities with the BWR/6, Mark III, containment design, before treatment.

**Table 4-2 Gaseous and particulate releases from building ventilation systems**

Nuclide <sup>a</sup>	Containment Building (Ci/yr/unit)	Auxiliary Building (Ci/yr/unit)	Turbine Building (Ci/yr/unit)	Radwaste Building (Ci/yr/unit)
Kr-83m	b	b		b
Kr-85m	1.0E+00	3.0E+00	2.5E+01	b
Kr-85	b	b	b	b
Kr-87	b	2.0E+00	6.1E+01	
Kr-88	1.0E+00	3.0E+00	9.1E+01	b
Kr-89	b	2.0E+00	5.8E+02	2.9E+01
Xe-131m	b	b	b	b
Xe-133m	b	b	b	b
Xe-133	2.7E+01	8.3E+01	1.5E+02	2.2E+02
Xe-135m	1.5E+01	4.5E+01	4.0E+02	5.3E+02
Xe-135	3.3E+01	9.4E+01	3.3E+02	2.8E+02
Xe-137	4.5E+01	1.35E+02	1.0E+03	8.3E+01
Xe-138	2.0E+00	6.0E+00	1.0E+03	2.0E+00
Cr-51 <sup>c</sup>	2.0E-04	9.0E-04	9.0E-04	7.0E-04
Mn-54	4.0E-04	1.0E-03	6.0E-04	4.0E-03
Fe-59	9.0E-05	3.0E-04	1.0E-04	3.0E-04
Co-58	1.0E-04	2.0E-04	1.0E-03	2.0E-04
Co-60	1.0E-03	4.0E-03	1.0E-03	7.0E-03
Zn-65	1.0E-03	4.0E-03	6.0E-03	3.0E-04
Sr-89	3.0E-05	2.0E-05	6.0E-03	NA
Sr-90	3.0E-06	7.0E-06	2.0E-05	NA
Zr-95	3.0E-04	7.0E-04	4.0E-05	8.0E-04
Nb-95	1.0E-03	9.0E-03	6.0E-06	4.0E-06
MP-99	6.0E-03	6.0E-02	2.0E-02	3.0E-06
Ru-103	2.0E-04	4.0E-03	5.0E-04	1.0E-06
Ag-110	4.0E-07	20.E-06	NA	NA
Sb-124	2.0E-05	3.0E-05	1.0E-04	7.0E-05
Cs-134	7.0E-04	4.0E-03	2.0E-04	2.4E-03
Cs-136	1.0E-04	4.0E-04	1.0E-04	NA
Cs-137	1.0E-03	5.0E-03	1.0E-03	4.0E-03
Ba-140	2.0E-03	2.0E-02	1.0E-02	4.0E-06
Ce-141	2.0E-04	7.0E-04	1.0E-02	7.0E-06

<sup>a</sup> The values in this table come from Table 2-12 of NUREG-0016, Revision 1.

<sup>b</sup> Less than 1 Ci/yr per reactor.

<sup>c</sup> Particulate release rates are before filtration.

N/A = Not analyzed; analysis for the radionuclide was not performed.

The iodine releases from ventilation systems for facilities with the BWR/6, Mark III, containment design, before treatment, are calculated by the code using the data in Table 4-3, Tables 4-42 through 4-45, and Tables 4-4 through 4-6.

**Table 4-3 Radioiodine releases from building ventilation systems before treatment**

Annual Normalized Radioiodine Release Rate <sup>a</sup>	Containment Building (Ci/yr/ $\mu$ Ci/g)	Auxiliary Building <sup>b</sup> (Ci/yr/ $\mu$ Ci/g)	Turbine Building <sup>c</sup> (Ci/yr/ $\mu$ Ci/g)	Radwaste Building <sup>b</sup> (Ci/yr/ $\mu$ Ci/g)
Power Operation	1.2E+00	1.11E+01	3.8E+03	4.6E+00
Refueling/Maintenance Outages	4.7E+00	5.0E-01	4.1E+02	1.4E+00

- <sup>a</sup> The values in this table come from Table 1-2 of NUREG-0016, Revision 1. 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 within the building and the partitioning of the radioiodine between the water phase in the leakage and the gas phase where it is measured. For the turbine building, the effective leak rate also includes the carryover for radioiodine from reactor water to steam in the reactor vessel.
- <sup>b</sup> To obtain the actual radioiodine release from these buildings in Ci/yr, multiply the normalized release by the radioiodine coolant concentration in  $\mu$ Ci/g.
- <sup>c</sup> To obtain the actual radioiodine release from the turbine building in Ci/yr, multiply the normalized release by the coolant concentration in  $\mu$ Ci/g and by the radioiodine carryover from Table 4-44.

### Bases

The iodine-131 releases from building ventilation systems are based on measurements made at a number of operating reactors. The measurements were made during routine plant operation and during plant shutdowns. Work on identifying sources of radioiodine at BWRs has been conducted by the Electric Power Research Institute (EPRI) at three operating BWRs (Monticello Nuclear Generating Plant, Vermont Yankee Nuclear Power Station, and Oyster Creek Nuclear Generating Station), described in EPRI NP-495, "Sources of Radioiodine at Boiling Water Reactors," issued February 1978 [Ref. 15], and at one operating BWR (Pilgrim Nuclear Power Station) for an NRC-sponsored study (NUREG/CR-0395, "Evaluation of Radioiodine Measurements at Pilgrim Nuclear Power Plant," issued October 1978 [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 radioiodine are expressed as "normalized" releases; that is, the absolute measured release rate in  $\mu$ Ci/sec is divided by the measured reactor water concentration in  $\mu$ Ci/g to give a normalized release rate of reactor water containing iodine-131 in g/sec, as shown in Equation (4-1):

$$R_N = \frac{R_A}{C_{RW}} \quad (4-1)$$

where  $R_N$  = the normalized release rate of reactor water containing iodine-131 (g/sec);  
 $R_A$  = absolute (measured) iodine-131 release rate (Ci/sec); and  
 $C_{RW}$  = the measured reactor water iodine-131 concentration ( $\mu$ Ci/g).

The normalized reactor water release rate, expressed in g/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 radioiodine 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 PC 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 radioiodine release,  $R_N$ , is determined using Equation (4-2):

$$R_N = \frac{R_A}{C_{RW} \times PC} \quad (4-2)$$

where  $R_N$  = normalized release rate of secondary coolant water containing iodine-131 (g/sec);  
 $R_A$  = absolute (measured) iodine-131 release rate (Ci/sec);  
 $C_{RW}$  = measured secondary coolant iodine-131 concentration ( $\mu$ Ci/g); and  
 $PC$  = measured PC from the reactor water to reactor steam.

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

Tables 4-4 through 4-6 give data on normalized release rates from the three reactors used in the EPRI study (Reference 15) and the reactor used in the NRC-sponsored study (Reference 16) for normal operation and shutdown periods for the turbine building, reactor building, and radwaste building, respectively.

Tables 4-4 through 4-6 also give the normalized value of the iodine release data discussed in NUREG-0016, Revision 1. Tables 2-10 through 2-15 of NUREG-0016 list the data and the plants included in these references. These data are presented as one data point because the measurements used were of short duration compared to the lengthy measurements carried out in EPRI NP-495 and NUREG/CR-0395.

The data in Tables 4-4 through 4-6 are expressed as total normalized releases during power operation of 300 days and the total normalized releases during shutdowns of 65 days. Because the reactors used in the EPRI study (Reference 15) and the NRC study (Reference 16) experienced several intermittent shutdowns of short duration during the power operation measurement period, the radioiodine releases during these short-duration outages are included under power operation.

**Table 4-4 Annual iodine normalized releases from turbine building ventilation systems**

Normal Operation <sup>a</sup>	
Data Source	Normalized Releases <sup>b</sup> (Ci/yr/ $\mu$ Ci/g)
Monticello	3.1E+03
Oyster Creek	6.0E+03
Vermont Yankee	3.5E+02
Pilgrim	8.5E+03
Browns Ferry	1.3E+03
NRC <sup>c</sup>	3.3E+03
Average	3.8E+03
Extended Shutdown <sup>a</sup>	
Data Source	Normalized Releases <sup>b</sup> (Ci/yr/ $\mu$ Ci/g)
Monticello	1.7E+02
Oyster Creek	3.5E+02 <sup>d</sup>
Vermont Yankee	6.3E+01
Browns Ferry	1.3E+02
NRC <sup>c</sup>	1.4E+02
Average	4.1E+02

<sup>a</sup> These values are from Table 2-8 of NUREG-0016, Revision 1.

<sup>b</sup> 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 radioiodine between the water phase in the leakage and the gas phase where it is measured, and the PC for radioiodine from reactor water to steam in the reactor vessel.

<sup>c</sup> Tables 2-10 through 2-15 of NUREG-0016 list the data and the plants included in these references.

<sup>d</sup> Oyster Creek data in this table do not include the effect of use of reheater protection system exhaust because the system design of this component is not typical of current BWRs (Nine Mile Point, Unit 1, is the only other BWR with this design). If a BWR uses this design, the additional release is 8.7E+02 Ci/yr/ $\mu$ Ci/g during the shutdown period and should be included in the total turbine building shutdown release.

**Table 4-5 Annual iodine normalized releases from reactor building ventilation systems**

Normal Operation <sup>a</sup>	
Data Source	Normalized Releases <sup>b</sup> (Ci/yr/ $\mu$ Ci/g)
Monticello	1.1E+01
Pilgrim	1.3E+01
Browns Ferry	4.2E+00
NRC <sup>c</sup>	2.1E+01
Average	1.23E+01 <sup>d</sup>
Extended Shutdown <sup>a</sup>	
Data Source	Normalized Releases <sup>b</sup> (Ci/yr/ $\mu$ Ci/g)
Monticello	4.7E-01
Oyster Creek	1.3E+00
Vermont Yankee	3.2E+00
Browns Ferry	1.4E+00
NRC <sup>c</sup>	2.0E+01
Average	5.2E+00

<sup>a</sup> These values are from Table 2-9 of NUREG-0016, Revision 1.

<sup>b</sup> 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 radioiodine between the water phase and the gas phase where it is measured.

<sup>c</sup> Tables 2-10 through 2-15 of NUREG-0016 list the data and the plants included in these references.

<sup>d</sup> Oyster Creek and Vermont Yankee data are not included here because Monticello leakage is considered to be more typical of similar problems at other BWRs, where the reactor water cleanup pump is upstream of the reactor water cleanup demineralizers.

**Table 4-6 Annual iodine normalized releases from radwaste building ventilation systems**

Normal Operation <sup>a</sup>	
Data Source	Normalized Releases <sup>b</sup> (Ci/yr/ $\mu$ Ci/g)
Monticello	7.2E-01
Oyster Creek	6.8E+00
Vermont Yankee	1.0E+00
Pilgrim	1.2E+01
Browns Ferry	2.0E+00
NRC <sup>c</sup>	5.3E+00
Average	4.6E+00
Extended Shutdown <sup>a</sup>	
Data Source	Normalized Releases <sup>b</sup> (Ci/yr/ $\mu$ Ci/g)
Monticello	2.0E-02
Oyster Creek	1.4E+00
Vermont Yankee	4.0E-01
Browns Ferry	6.0E-01
NRC <sup>c</sup>	4.4E+00
Average	1.4E+00

<sup>a</sup> These values are from Table 2-10 of NUREG-0016, Revision 1.

<sup>b</sup> 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 radioiodine between the water phase in the leakage and the gas phase where it is measured.

<sup>c</sup> Tables 2-10 through 2-15 of NUREG-0016 list the data and the plants included in these references.

To obtain the releases in Ci/year from the reactor building and radwaste building of a particular BWR, the normalized release data in Tables 4-5 and 4-6, respectively, are multiplied in the code by the iodine reactor water concentration for that particular BWR using Equation (4-3):

$$R_{BWR} = R_N^I \times C_{BWR} \quad (4-3)$$

where  $R_{BWR}$  = calculated annual release rate for particular BWR (Ci/yr);  
 $R_N^I$  = normalized annual release rate of reactor water containing iodine-131 from Tables 4-5 and 4-6 (Ci/yr/ $\mu$ Ci/g); and  
 $C_{BWR}$  = calculated reactor water concentration for particular BWR ( $\mu$ Ci/g).

To obtain the release in Ci/yr from the turbine building of a particular BWR, the normalized release data in Table 4-4 are multiplied in the code by the iodine reactor water concentration and the iodine carryover from the reactor water to reactor steam for that particular BWR using Equation (4-4):

$$R_{BWR} = R_N^I \times C_{BWR} \times PC_{BWR} \quad (4-4)$$

where  $R_{BWR}$  = calculated annual release rate for particular BWR (Ci/yr);  
 $R_N^I$  = normalized annual release rate of reactor water from Table 4-4 (Ci/yr/ $\mu$ Ci/g);  
 $C_{BWR}$  = calculated reactor water concentration for particular BWR ( $\mu$ Ci/g); and  
 $PC_{BWR}$  = calculated carryover from the reactor water to reactor steam for the particular BWR (see Section 4.2.1 and Table 4-44).

The value for the iodine carryover from the reactor water to reactor steam can be determined for the particular BWR from Table 4-44.

To obtain the releases during shutdown, multiply the normalized release rates for the extended shutdown period by the same reactor water concentration as for power operations. Use of this reactor water concentration is acceptable as the normalization technique of the EPRI study (Reference 15) based the extended shutdown normalized release rate on the reactor water concentrations before shutdown.

The value for the iodine-131 reactor water concentration can be determined as discussed below. As noted in NUREG-0016, Revision 1, the 7 x 7R and 8 x 8 fuel, which has replaced the 7 x 7, has shown considerable improvement over the 7 x 7 fuel previously used in some BWRs.

The improvements in the fuel are primarily the result of a reduction in Zircaloy hydriding and pellet cladding interaction. However, there is not extensive experience with the improved fuel over complete burnup cycles to date. At the higher burnups, the probability for pellet cladding interaction 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 as the result of fission product swelling of the fuel and the cladding losing its ductility from neutron damage and hydrogen pickup.

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 percent of improved fuel. However, a review was performed of BWR operating experience (Reference 3) with iodine-131 reactor water concentrations for the period 1975–1977, which includes the time period during which the improved fuel was introduced. 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, only the experience at those reactors whose cores are loaded with greater than 50 percent of improved fuel was considered. Table 4-7 contains a summary of these data. Based on this review, the iodine-131 reactor water concentration that will be used as an interim measure until more experience with improved fuel is obtained is 3.7E-03  $\mu\text{Ci/g}$ .

**Table 4-7 Summary of iodine-131 in BWRs**

Facility <sup>a</sup>	Iodine-131 Concentration ( $\mu\text{Ci/kg}$ )		
	1977	1976	1975
Browns Ferry 1	9.0E-01		
Browns Ferry 2	1.5E-00		
Browns Ferry 3	1.4E-01		
Brunswick 1	2.0E-02		
Brunswick 2	3.1E+00	9.3E-01	7.0E-03
Cooper	7.2E-02	9.0E-01	1.3E-01
Dresden 3	1.76E+01	1.26E+01	
Duane Arnold	4.2E-02	9.0E-02	2.3E-03
FitzPatrick	2.4E-01	2.9E-01	
Hatch 1	9.0E-01	1.1E-01	
Millstone 1	8.9E+00	5.6E+00	7.1E+00
Monticello	5.9E+00	9.0E+00	8.7E+00
Nine Mile Point 1	9.4E+00	5.9E+00	
Oyster Creek	8.4E+00	5.3E+00	4.8E+00
Peach Bottom 2	7.3E+00	1.6E=01	4.5E-02
Peach Bottom 3	1.1E+00	8.3E-01	6.3E-02
Quad Cities 1	3.4E+00		
Vermont Yankee	3.8E-01	5.1E-01	7.8E-01
	3.8E+00	4.4E+00	2.4E+00

<sup>a</sup> These values are from Table 2-11 of NUREG-0016, Revision 1.

The reactor building releases reported in NUREG-0016, Revision 1 are based on reactors with a BWR Mark I containment design. Equipment such as the reactor water cleanup 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 from NUREG-0016, Revision 1, the reactor water cleanup 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 percent of Mark I reactor building releases, and releases from the Mark III containment building ventilation are determined to be 10 percent of Mark I releases during power operation. During shutdown, 90 percent of the releases are determined to be from the Mark III containment building ventilation system and 10 percent from the auxiliary building ventilation system. For the turbine building, 85 percent of the releases are determined to come from the ventilation system serving the main condenser area during power operation (Reference 3). The remaining 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, because 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 percent of the total and the remaining releases come from the miscellaneous areas.

For the radwaste building, 10 percent of the releases are determined to come from the solid waste handling area, and 90 percent of the releases are determined to come from the liquid waste handling area (Reference 3).

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 EPRI NP-495.

Iodine released from BWR building ventilation systems appears in one of the following chemical forms: particulate, elemental, hypoiodous acid (HOI), and organic. Table 4-8 shows the fraction of iodine appearing in each of the chemical forms for each building ventilation system.

**Table 4-8 Fraction of radioiodine appearing in each chemical form from BWR building ventilation systems**

Iodine Type <sup>a</sup>	Containment	Auxiliary	Turbine	Radwaste
Particulate	1.1E-01	2.0E-01	2.0E-01	2.0E-03
Elemental	3.2E-01	4.8E-01	5.0E-01	2.8E-01
Hypoiodous acid (HOI)	3.8E-01	2.4E-01	2.2E-01	2.5E-01
Organic	1.9E-01	9.0E-02	8.0E-02	4.7E-01

<sup>a</sup> The data in this table are taken from Section 2.2.4.2 of NUREG-0016, Revision 1.

The noble gas release rates summarized in NUREG-0016, Revision 1, for building ventilation systems are based on the average of measurements made at Oyster Creek; Millstone Power Station, Unit 1; Monticello; and Nine Mile Point Nuclear Station. These data are given in Tables 4-9 through 4-11 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 percent of the releases are assumed to be from the containment building and 10 percent from the auxiliary building ventilation system because 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.

**Table 4-9 Release rates of noble gases from the reactor building ventilation system**

Nuclide <sup>a</sup>	Noble Gases Release Rate ( $\mu\text{Ci/s}$ )				
	Millstone-1	Oyster Creek	Oyster Creek	Monticello	Average
Kr-85m	2.6E-01	ND	ND	2.0E-01	1.2E-01
Kr-87	2.4E-01	ND	ND	1.0E-01	8.5E-02
Kr-88	3.84E-01	2.0E-02	ND	2.0E-01	1.5E-01
Kr-89	ND	ND	ND	3.8E-01	9.5E-02
Xe-133	5.2E-01	1.5E+01	ND	2.0E+00	4.4E+00
Xe-135m	3.6E+00	ND	2.5E+00	3.5E+00	2.4E+00
Xe-135	3.0E+00	2.1E+00	1.4E+01	1.8E+00	5.2E+00
Xe-137	ND	ND	ND	3.0E+01	7.5E+00
Xe-138	4.4E-01	3.0E-01	ND	4.0E-01	2.9E-01

<sup>a</sup> These values are from Table 2-13 of NUREG-0016, Revision 1.

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



**Table 4-10 Release rates of noble gases from the turbine building ventilation system**

Nuclide <sup>a</sup>	Noble Gases Release Rate ( $\mu\text{Ci/s}$ )				
	Millstone-1	Oyster Creek	Oyster Creek	Nine Mile Point 1	Average
Kr-85m	2.7E+00	ND	2.3E+00	1.0E-01	9.7E-02
Kr-87	5.3E+00	ND	6.2E+00	1.5E-01	5.3E-01
Kr-88	8.2E+00	5.2E+00	4.2E+00	6.5E-02	2.1E-01
Kr-89	ND	ND	7.0E+01	4.2E+01	4.5E+00
Xe-133	7.4E+00	1.3E+01	ND	5.0E+00	3.5E+00
Xe-135m	2.9E+01	1.2E+01	2.6E+01	8.2E+00	2.5E+0
Xe-135	2.5E+01	2.5E+01	7.4E+00	6.8E+00	2.3E+00
Xe-137	ND	ND	1.15E+02	8.6E+01	ND
Xe-138	6.3E+01	2.6E+01	9.7E+01	1.1E+01	4.3E+00

<sup>a</sup> These values are from Table 2-14 of NUREG-0016, Revision 1.  
 ND = Not detected. For averaging purposes, a value of zero was assumed.

**Table 4-11 Release rates of noble gases from the radwaste building ventilation system**

Nuclide <sup>a</sup>	Noble Gases Release Rate ( $\mu\text{Ci/s}$ )			
	Millstone-1	Oyster Creek	Monticello	Average
Kr-89	ND	ND	3.0E+00	1.0E+00
Xe-133	ND	ND	5.3E+00	1.8E+00
Xe-135m	2.5E-01	5.6E-01	2.6+01	8.9E+00
Xe-135	ND	4.0E+00	5.9E+01	2.1E+01
Xe-137	2.0E+00	1.5E+00	2.0E+01	7.8E+00
Xe-138	ND	ND	1.0E+01	3.3E+00

<sup>a</sup> These values are from Table 2-15 of NUREG-0016, Revision 1.  
 ND = Not detected. For averaging purposes, a value of zero was assumed.

The radioactive particulate release rates for building ventilation systems are the average of measurements made at Vermont Yankee; Oyster Creek; Dresden Nuclear Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2; Monticello; and Nine Mile Point (Reference 3). Tables 4-12 through 4-17 give these data.

**Table 4-12 Particulate release rates from the reactor building ventilation system, normal operation**

Nuclide <sup>a</sup>	Particulate Release Rate (1.0E-06 μCi/s)					
	Quad Cities 1	Quad Cities 2	Vermont Yankee	Oyster Creek	Monticello	Average
Co-60	2.1E+02	9.0E-01	3.7E+01	6.1E+02	1.4E+02	2.0E+02
Co-58	2.0E+01	4.0E-01	5.5E+00	3.1E+01	ND	1.1E+01
Cr-51	1.4E+02	5.0E-01	1.3E+01	3.9E+01	ND	3.8E+01
Mn-54	1.9E+01	1.0E-01	1.4E+01	2.1E+02	2.4E+01	5.3E+01
Fe-59	NA	NA	5.0E+00	3.3E+01	4.2E+00	1.4E+01
Zn-65	2.3E+01	1.0E-01	4.6E+01	6.4E+00	7.5E+02	1.6E+02
Sr-89	NA	NA	NA	6.8E+00	NA	6.8E+00
Sr-90	NA	NA	NA	3.0E-01	NA	3.0E-01
Zr-95	1.6E+00	ND	1.5E+00	5.0E-01	1.15E+02	2.2E+01
Nb-95	2.7E+00	ND	7.4E+00	3.0E-01	2.2E+03	4.4E+02
Mo-99	NA	NA	4.4E+00	1.4E+02	7.3E+03	2.5E+03
Ru-103	NA	NA	ND	2.8E+00	6.5E+01	2.3E+01
Ag-110m	2.0E-01	ND	NA	NA	NA	1.0E-01
Sb-124	NA	NA	ND	2.4E+00	ND	8.0E-01
Cs-134	4.8E+01	1.0E-01	1.2E+01	1.6E+01	7.6E+02	1.7E+02
Cs-136	2.3E+00	ND	6.8E+00	7.1E+00	7.9E+01	1.9E+01
Cs-137	4.4E+01	5.0E-01	3.7E+01	3.1E+01	9.9E+02	2.2E+02
Ba-140	ND	ND	1.6E+01	7.6E+01	2.6E+03	5.4E+02
Ce-141	NA	ND	ND	3.9E+00	1.2E+02	3.1E+01

<sup>a</sup> These values are from Table 2-16 of NUREG-0016, Revision 1.

NA = Not analyzed.

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

**Table 4-13 Particulate release rates from the reactor building ventilation system, refueling shutdown**

Nuclide <sup>a</sup>	Particulate Release Rate (1.0E-06 $\mu$ Ci/s)					
	Quad Cities 1	Quad Cities 2	Vermont Yankee	Oyster Creek	Monticello	Average
Co-60	1.6E+01	8.8E-01	2.5E+02	3.3E+02	1.70E+00	1.2E+02
Co-58	2.3E+00	3.5E-01	4.1E+01	1.9E+01	ND	1.3E+01
Cr-51	8.9E+00	5.0E-01	6.3E+01	2.8E+01	ND	2.0E+01
Mn-54	2.6E+00	6.1E-02	2.0E+01	1.4E+02	4.7E+01	4.2E+01
Fe-59	ND	ND	2.1E+01	5.0E+00	3.4E+00	6.0E+00
Zn-65	5.8E+01	1.1E-01	7.7E+02	1.4E+00	7.3E+01	1.8E+02
Sr-89	NA	NA	NA	2.0E+00	NA	2.0E+00
Sr-90	NA	NA	NA	3.6E-01	NA	3.6E-01
Zr-95	3.1E-01	ND	7.8E+01	2.4E-01	ND	1.6E+01
Nb-95	4.0E-01	2.1E-02	ND	4.1E-01	1.6E+02	3.2E+01
Mo-99	NA	NA	ND	1.3E+01	4.4E+00	5.8E+00
Ru-103	NA	NA	NA	1.3E+00	3.6E+01	1.9E+01
Ag-110m	1.1E-01	NA	NA	NA	NA	1.1E-01
Sb-124	ND	ND	NA	7.0E+00	ND	1.8E+00
Cs-134	6.2E+00	1.4E-01	8.2E+01	1.3E+01	1.7E+02	5.4E+01
Cs-136	1.0E+00	NA	2.0E+01	ND	2.1E+01	1.1E+01
Cs-137	1.4E+01	5.4E-01	2.4E+02	2.3E+01	2.0E+02	9.5E+01
Ba-140	ND	ND	1.4E+01	1.1E+00	2.0E+02	4.3E+01
Ce-141	ND	ND	NA	7.5E+00	4.5E+01	1.3E+01

<sup>a</sup> These values are from Table 2-17 of NUREG-0016, Revision 1.

NA = Not analyzed.

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

**Table 4-14 Particulate release rates from the turbine building ventilation system, normal operation**

Nuclide <sup>a</sup>	Particulate Release Rate (1.0E-06 $\mu$ Ci/s)					
	Oyster Creek	Monticello	Vermont Yankee	Dresden 2	Dresden 3	Average
Co-60	6.1E+01	1.5E+01	4.1E+00	4.5E+00	6.0E+00	1.8E+01
Co-58	1.2E+01	ND	2.0E+00	ND	4.8E+01	1.2E+01
Cr-51	4.4E+02	ND	NA	ND	1.6E+02	1.5E+02
Mn-54	3.0E+01	7.5E+00	1.9E+00	ND	5.0E+00	8.9E+00
Fe-59	5.8E+00	ND	ND	ND	ND	1.2E+00
Zn-65	1.7E+00	2.3E+01	7.8E+00	ND	ND	6.5E+00
Sr-89	6.1E+02	NA	NA	4.8E+01	3.6E+00	2.2E+02
Sr-90	1.3E+00	NA	NA	3.0E-01	2.5E-01	6.0E-01
Zr-95	5.9E-01	ND	ND	ND	4.0E+00	9.2E-01
Nb-95	3.3E-01	ND	NA	ND	ND	1.0E-01
Mo-99	9.1E+01	1.5E+02	NA	ND	ND	6.1E+01
Ru-103	1.7E+00	ND	ND	ND	ND	3.4E-01
Sb-124	4.6E+00	ND	ND	ND	ND	9.2E-01
Cs-134	1.8E+01	2.3E+01	2.7E+00	ND	3.0E+00	9.3E+00
Cs-136	1.1E+00	1.6E+01	ND	ND	ND	3.4E+00
Cs-137	5.7E+01	1.0E+02	5.1E+00	1.8E+00	1.0E+01	3.5E+01
Ba-140	1.4E+03	1.6E+01	8.3E+01	1.2E+02	6.5E+01	3.4E+02
Ce-141	2.9E+01	1.6E+03	ND	5.5E+00	5.0E+00	3.28E+02

<sup>a</sup> These values are from Table 2-18 of NUREG-0016, Revision 1.

NA = Not analyzed.

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

**Table 4-15 Particulate release rates from the turbine building ventilation system, refueling shutdown**

Nuclide <sup>a</sup>	Particulate Release Rate (1.0E-06 $\mu$ Ci/s)			
	Oyster Creek	Vermont Yankee	Monticello	Average
Co-60	2.9E+02	2.5E+00	5.8E+00	1.0E+02
Co-58	1.6E+01	1.0E+00	NA	8.5E+00
Cr-51	5.1E+01	ND	NA	2.6E+01
Mn-54	1.1E+02	NA	3.0E-01	5.7E+01
Fe-59	3.1E+01	ND	ND	1.0E+01
Zn-65	1.1E+01	NA	1.0E+01	1.0E+01
Sr-89	2.5E+00	NA	NA	2.5E+00
Sr-90	2.5E-01	NA	NA	2.5E-01
Zr-95	6.0E-02	ND	ND	2.0E-02
Nb-95	4.0E-01	ND	ND	1.3E-01
Mo-99	1.25E+02	NA	9.7E+00	6.7E+01
Ru-103	5.2E+00	NA	ND	2.6E+00
Sb-124	9.5E+00	ND	NA	4.8E+00
Cs-134	1.9E+01	1.9E+00	1.3E+00	7.4E+00
Cs-136	ND	ND	4.1E+00	1.4E+00
Cs-137	3.9E+01	3.4E+00	5.8E+00	1.6E+01
Ba-140	8.2E+00	1.1E+02	4.9E+01	5.6E+01
Ce-141	1.7E+01	ND	9.1E+00	8.7E+00

<sup>a</sup> These values are from Table 2-19 of NUREG-0016, Revision 1.

NA = Not analyzed.

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

**Table 4-16 Particulate release rates from the radwaste building ventilation system, normal operation**

Nuclide <sup>a</sup>	Particulate Release Rate (1.0E-06 $\mu$ Ci/s)		
	Vermont Yankee	Oyster Creek	Average
Co-60	6.0E+00	5.8E+02	2.9E+02
Co-58	1.0E+00	1.6E+01	8.0E+00
Cr-51	3.0E+00	4.8E+01	2.6E+01
Mn-54	1.0E+00	3.3E+02	1.7E+02
Fe-59	ND	2.6E+01	1.3E+01
Zn-65	1.0E+00	2.1+E01	1.1E+01
Sr-89	NA	NA	NA
Sr-90	NA	NA	NA
Zr-95	ND	6.3E+01	3.1E+01
Mo-99	2.0E+00	ND	1.0E+00
Sb-124	ND	5.4E+00	2.7E+00
Cs-134	1.2E+00	1.9E+02	9.6E+01
Cs-136	ND	ND	0.0E+00
Cs-137	2.0E+00	2.9E+02	1.5E+02
Ba-140	3.0E-01	ND	1.5E-01
Ce-141	ND	6.3E+00	3.2E+00

<sup>a</sup> These values are from Table 2-20 of NUREG-0016, Revision 1.

NA = Not analyzed.

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

**Table 4-17 Particulate release rates from the radwaste building ventilation system, refueling shutdown**

Nuclide <sup>a</sup>	Particulate Release Rate (1.0E-06 μCi/s)	
	Monticello	Average
Co-60	1.3E+00	1.3E+00
Co-58	2.1E-01	2.1E-01
Cr-51	ND	0.0E+00
Mn-54	4.0E-01	4.0E-01
Fe-59	3.2E+00	3.2E+00
Zn-65	5.1E+00	5.1E+00
Sr-89	NA	NA
Sr-90	NA	NA
Nb-95	6.0E+00	6.0E+00
Mo-99	1.0E+00	1.0E+00
Ru-103	1.0E+00	1.0E+00
Sb-124	ND	0.0E+00
Cs-134	1.0E+00	1.0E+00
Cs-136	ND	0.0E+00
Cs-137	2.2E+00	2.2E+00
Ba-140	ND	0.0E+00
Ce-141	1.2E+00	1.2E+00
Ce-144	4.0E+00	4.0E+00

<sup>a</sup> These values are from Table 2-21 of NUREG-0016, Revision 1.

NA = Not analyzed.

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

The calculated annual average rates given above are based on an 80-percent plant capacity factor (i.e., 80-percent normal operation at 100-percent power and 20-percent 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.

#### 4.1.1.4 Iodine Input to the Main Condenser Offgas Treatment System

##### Parameter

The iodine-131 input to the main condenser offgas treatment system (MCOTS), downstream of the air ejectors, is 6.0E+00 Ci/yr.

##### Basis

Table 4-18 lists the measured iodine-131 releases and integrated thermal power outputs for BWRs with thermal ratings exceeding 1.0E+03 MWt, with more than 1 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 6.3E-06 Ci/MWd per year. Based on a power rating of 3.4E+03 MWt and an 80-percent plant capacity factor, the iodine-131 release from the main condenser air ejector is approximately 6.0E+00 Ci/yr.

**Table 4-18 Radioiodine-131 releases from the main condenser air ejectors**

Facility <sup>a</sup>	1972			1973		
	Radioiodine Release (Ci/yr)	Integrated Thermal Power (1.0E+06 MWd)	Ci/yr 1.0E+06 MWd	Radioiodine Release (Ci/yr)	Integrated Thermal Power (1.0E+06 MWd)	Ci/yr 1.0E+06 MWd
Oyster Creek	6.3E+00	5.42E-01	1.16E+01	6.7E+00	4.53E-01	1.48E+01
Nine Mile Point 1	8.9E-01	4.17E-01	2.1E+00	1.9E+00	4.57E-01	4.2E+00
Millstone 1	1.2E+00	4.04E-01	3.0E+00	1.5E-01	2.48E-01	6.0E-01
Dresden Units 2 & 3 <sup>b</sup>	5.1E+00	1.05E+00	4.9E+00	9.8E+00	1.18E+00	8.3E+00
Monticello	5.8E-01	4.54E-01	1.3E+00	1.2E+00	4.13E-01	2.9E+00
Pilgrim 1			c	4.6E-01	5.23E-01	9.0E-01
Quad Cities Units 1 & 2 <sup>b</sup>			c	5.5E+00	1.32E+00	4.2E+00
	Average		4.6E+00	Average		5.1E+00

<sup>a</sup> These values are from Table 2-22 of NUREG-0016, Revision 1, and data from semiannual operating reports for 1972 through 1976 for facilities listed.

<sup>b</sup> Two-unit plants with a single stack

<sup>c</sup> Not included in 1972 average because plants had not been in operation for a full year

<sup>d</sup> Augmented offgas system put in operation October 1975

<sup>e</sup> Augmented offgas system put in operation late 1976

<sup>f</sup> Augmented offgas system put in operation May 1975

<sup>g</sup> Augmented offgas system put in operation 1977

<sup>h</sup> Augmented offgas system put in operation late 1974



**Table 4-18 Radioiodine-131 releases from the main condenser air ejectors (cont.)**

Facility <sup>a</sup>	1974			1975		
	Radioiodine Release (Ci/yr)	Integrated Thermal Power (1.0E+06 MWd)	Ci/yr 1.0E+06 MWd	Radioiodine Release (Ci/yr)	Integrated Thermal Power (1.0E+06 MWd)	Ci/yr 1.0E+06 MWd
Oyster Creek	3.3E+00	4.6E-01	7.2E+00	5.5E+00	4.1E-01	1.34E+01
Nine Mile Point 1	7.0E-01	4.3E-01	1.6E+00	2.1E+00	4.0E-01	5.3E+00
Millstone 1	2.9E-01	4.7E-01	6.0E-01	9.8E+00	5.0E-01	1.96E+01
Dresden Units 2 & 3 <sup>b</sup>	4.0E+00	9.1E-01	4.4E+00	7.5E-01	7.1E-01	1.06E+00
Monticello	5.7E+00	3.4E-01	1.68E+01	3.5E+00	3.7E-01	9.5E+00
Pilgrim 1	1.4E+00	2.5E-01	5.6E+00	h		
Quad Cities Units 1 & 2 <sup>b</sup>	8.2E+00	1.09E+00	7.5E+00	f		
	Average		6.2E+00	Average		9.8E+00

1976			
Facility <sup>a</sup>	Radioiodine Release (Ci/yr)	Integrated Thermal Power (1.0E+06 MWd)	Ci/yr 1.0E+06 MWd
Oyster Creek	6.2E+00	4.9E-01	1.27E+01
Nine Mile Point 1	2.1E+00	5.5E-014	3.8E+00 <sup>g</sup>
Millstone 1	2.7E+00	4.8E-01	5.6E+00
Dresden Units 2 & 3 <sup>b</sup>	1.9E+00	1.14E+00	1.7E+00 <sup>e</sup>
Monticello	d		
Pilgrim 1			
Quad Cities Units 1 & 2 <sup>b</sup>			
		Average	6.0E+00

<sup>a</sup> These values are from Table 2-22 of NUREG-0016, Revision 1, and data from semiannual operating reports for 1972 through 1976 for facilities listed.

<sup>b</sup> Two-unit plants with a single stack

<sup>c</sup> Not included in 1972 average because plants had not been in operation for a full year

<sup>d</sup> Augmented offgas system put in operation October 1975

<sup>e</sup> Augmented offgas system put in operation late 1976

<sup>f</sup> Augmented offgas system put in operation May 1975

<sup>g</sup> Augmented offgas system put in operation 1977

<sup>h</sup> Augmented offgas system put in operation late 1974

#### 4.1.1.5 Turbine Gland Sealing System Exhaust

##### Parameter

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

iodine-131 = 8.1E-01 Ci/yr per  $\mu\text{Ci/gm}$  of iodine-131 in the reactor coolant.

iodine-133 = 2.2E-01 Ci/yr per  $\mu\text{Ci/gm}$  of iodine-133 in the reactor coolant.

If the clean steam is supplied to the gland seal, the radioiodine source term is negligible (less than 1.0E-04 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.

##### Basis

Radioiodine measurements have been reported (Reference 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. Tables 4-19 and 4-20 summarize the available data for radioiodines released from the gland seal condenser exhaust when the mechanical vacuum pump was not in operation or infrequently used. The radioiodine release rates are dependent on the radioiodine concentration in the reactor coolant and carryover in the reactor.

**Table 4-19 Iodine-131 release rates from gland seal condenser exhaust for systems using main steam for the sealing system**

Using Main Steam for the Sealing System (at 7.0E+03 lb/h)					
Facility <sup>a</sup>	Sample Period	Days	Measured Gland Seal Iodine-131 Release ( $\mu\text{Ci/s}$ )	Measured Reactor Water Iodine-131 Concentration ( $\mu\text{Ci/g}$ )	Iodine-131 Release Ci/yr per Ci/g for 292 days/year
Vermont Yankee	06/18/74 to 06/19/74	1	3.9E-04	2.5E-02	3.9E+05
	06/20/74 to 06/21/74	1	4.2E-04	2.5E-02	4.2E+05
	09/13/74 to 09/14/74	1	4.7E-04	3.8E-02	3.1E+05
	10/10/74 to 10/11/74	1	4.5E-04	3.5E-02	3.2E+05
	03/05/75 to 03/08/75	3	2.1E-05	8.8E-04	6.0E+05
	10/07/75 to 10/21/75	14	1.4E-05	7.7E-04	4.6E+05
Oyster Creek	06/16/75 to 06/30/75	14	7.8E-05	1.8E-03	1.1E+06
	06/30/75 to 07/17/75	17	6.8E-05	1.7E-03	1.0E+06
Weighted Average According to Sample Days					8.1E+05

<sup>a</sup> These values are from Table 2-23 of NUREG-0016, Revision 1.

**Table 4-20 Iodine-133 release rates from gland seal condenser exhaust for systems using main steam for the sealing system**

Using Main Steam for the Sealing System (at 7.0E+03 lb/h)					
Facility <sup>a</sup>	Sample Period	Days	Measured Gland Seal Iodine-133 Release (μCi/s)	Measured Reactor Water Iodine-133 Concentration (μCi/g)	Iodine-133 Release Ci/yr per Ci/g for 292 days/year
Vermont Yankee	06/18/74 to 06/19/74	1	1.8E-04	5.6E-02	8.1E+04
	06/20/74 to 06/21/74	1	1.9E-04	5.6E-02	8.6E+04
	09/13/74 to 09/14/74	1	2.2E-04	1.1E-02	5.0E+05
	10/10/74 to 10/11/74	1	1.7E-04	1.0E-02	4.3E+05
Oyster Creek	06/16/75 to 06/30/75	14	2.3E-04	2.6E-03	2.2E+05
	06/30/75 to 07/17/75	17	2.0E-04	2.0E-03	2.1E+05
Weighted Average According to Sample Days					2.2E+05

<sup>a</sup> These values are from Table 2-23 of NUREG-0016, Revision 1.

It is assumed that there is no radioiodine source term when clean steam (nonradioactive 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 4-21 and 4-22 show the release of radioactive particulates from the turbine gland seal to be negligible (less than 1.0E-05 Ci/yr).

**Table 4-21 Particulate release rates from Vermont Yankee mechanical vacuum pump and gland exhaust condenser vent, refueling shutdown**

Nuclide <sup>a</sup>	Particulate Release Rate (1.0E-06 μCi/s)
	Average
Cs-134	4.5E-01
Cs-136	1.3E-01
Cs-137	1.0E+00
Ba-140	1.6E+00

<sup>a</sup> These values are from Table 2-25 of NUREG-0016, Revision 1.

**Table 4-22 Particulate release rates from Vermont Yankee mechanical vacuum pump and gland exhaust condenser vent, short-term shutdown**

Nuclide <sup>a</sup>	Particulate Release Rate (1.0E-06 $\mu$ Ci/s)
	Average
Cr-51	9.0E-01
Co-60	5.0E-01
Zn-65	3.0E-01
Cs-134	8.0E-01
Cs-136	1.1E+00
Cs-137	3.3E+00
Ba-140	2.1E+00

<sup>a</sup> These values are from Table 2-24 of NUREG-0016, Revision 1.

4.1.1.6 *Main Condenser Mechanical Vacuum Pump*

Parameter

Xenon-133	1.3E+03 Ci/yr per reactor
Xenon-135	5.0E+02 Ci/yr per reactor

The radioiodine releases from the main condenser mechanical vacuum pump are calculated by the code using the data in Tables 4-42, 4-44, and 4-23.

Basis

The release rates for xenon-133 and xenon-135 were derived from Dresden 1 and 2 operating data and adjusted to 5.0E+04  $\mu$ Ci/s (Reference 3). These data indicate that approximately 3.0E+02 Ci of xenon-133 and 1.2E+02 Ci of xenon-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 6.0E+04  $\mu$ Ci/s. The annual release estimates for noble gases assume four short-term shutdowns per year and one refueling or maintenance outage.

**Table 4-23 Normalized iodine releases from the mechanical vacuum pump**

Facility <sup>a</sup>	Normalized Release Rate ( $\leq 80$ hours) (Ci/yr/ $\mu$ Ci/g)
Monticello	8.3E+02
Vermont Yankee	1.5E+02
Average	4.9E+02

Facility <sup>a</sup>	Normalized Release Rate during Refueling/Maintenance Outages ( $\leq 80$ hours) (Ci/yr/ $\mu$ Ci/g)
Monticello	1.5E+03
Vermont Yankee	6.0E+02
Average	1.1E+03

<sup>a</sup> These values are from Table 2-26 of NUREG-0016, Revision 1. Assume four short-term outages per year.

The release rates for iodine-131 are based on measurements made at operating reactors (Reference 15). Investigations at three of the operating BWRs (Monticello, Vermont Yankee, and Oyster Creek) have shown that radioiodine releases from the mechanical vacuum pump are at their highest levels for the first 80 hours after shutdown. In accordance with the study, the releases from the mechanical vacuum pump can be as much as a factor of 1.0E+02 greater than releases measured during the preshutdown period. The normalized iodine-131 releases in Table 4-23 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 or 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 code using Equation (4-5):

$$R_N = \frac{R_A}{C_{RW} \times PC} \quad (4-5)$$

where  $R_N$  = the normalized release rate of reactor water containing iodine-131 (g/s);  
 $R_A$  = absolute (measured) iodine-131 release rate (Ci/s);  
 $C_{RW}$  = the measured reactor water iodine-131 concentration ( $\mu$ Ci/g); and  
 $PC$  = the measured PC 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 4-23 are multiplied by the radioiodine reactor water concentration and the radioiodine carryover from reactor water to reactor steam for the particular BWR using Equation (4-6):

$$R_{MVP} = R_N^1 \times C_{BWR} \times PC_{BWR} \quad (4-6)$$

where  $R_{MVP}$  = the calculated annual radioiodine release (Ci/yr) from the mechanical vacuum pump;  
 $R_N^I$  = the normalized release rate of reactor water containing iodine-131 (Ci/yr/ $\mu$ Ci/g);  
 $C_{BWR}$  = the reactor water concentration for a particular BWR ( $\mu$ Ci/g); and  
 $PC_{BWR}$  = the calculated carryover for a particular BWR from Table 4-44.

Radioiodine released during the operation of the mechanical vacuum pump at BWRs appears in one of the following chemical forms: particulate, elemental, HOI, and organic. Based on data in EPRI NP-495, Table 4-24 gives the fraction of the radioiodine appearing in each of the chemical forms for the mechanical vacuum pump.

**Table 4-24 Fraction of radioiodine appearing in each chemical form from BWR mechanical vacuum pump**

Chemical Form <sup>a</sup>	Time $\leq$ 80 h <sup>b</sup>	Time $>$ 80 h <sup>c</sup>
Particulate	4.0E-03	1.0E-02
Elemental	9.0E-03	6.0E-02
HOI	2.3E-02	2.1E-01
Organic	9.7E-01	7.2E-01

- <sup>a</sup> These values are from Section 2.2.7.2 of NUREG-0016, Revision 1.  
<sup>b</sup> Average of samples taken within the first 80 hours after shutdown  
<sup>c</sup> Average of samples taken after the initial 80 hours of a refueling maintenance outage

Data in Tables 4-21 and 4-22 show the release of radioactive particulates from the mechanical vacuum pump to be negligible.

#### 4.1.1.7 Air Leakage to the Main Condenser

##### Parameter

The air leakage into the main condenser per MWt of design reactor power, with a minimum of 5.0E+00 ft<sup>3</sup>/min, is 6.2E-03 ft<sup>3</sup>/min.

##### Basis

Air leakage 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 leakage into the main condenser.

Operational data for air leakage into the main condenser vary widely. At Oyster Creek and at Dresden Unit 2, air inleakage measurements during early phases of operation indicated leakage rates from 4.0E+00 ft<sup>3</sup>/min to 2.5E+02 ft<sup>3</sup>/min (Reference 3). As shown in Table 4-25, subsequent measurements at Dresden Unit 2 (Reference 3), showed air inleakage of 4.4E+00 ft<sup>3</sup>/min during operation at 1.6E+03 MWt. Air inleakage measurements reported for six TVA fossil plants, representing more than 50 years of cumulative experience, indicate leakage

rates ranging from 4.0E+00 to 2.5E+01 ft<sup>3</sup>/min per condenser shell and a statistical mean leakage rate of 6.7E+00 ft<sup>3</sup>/min per condenser shell (Reference 3). Measurements made in 1976 and 1977 at Quad Cities Units 1 and 2 (Reference 3) showed average flow rates of 9.6E+00 ft<sup>3</sup>/min for Unit 1 and 2.5E+01 ft<sup>3</sup>/min for Unit 2; measurements ranged from 6.0E+00 ft<sup>3</sup>/min to 5.5E+01 ft<sup>3</sup>/min, and power level for both units during the test period was 2.511E+03 MWt.

The parameter for air leakage was developed assuming that air leakage 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 3.4E+03 MWt BWR with a three-shell condenser, as shown in Table 4-25. The use of data from Dresden Unit 2, Quad Cities Units 1 and 2, and TVA fossil plants resulted in an average of 2.1E+01 ft<sup>3</sup>/min main condenser air leakage for a plant with a design thermal power level of 3.4E+03 MWt. This is approximately equivalent to 6.2E-03 ft<sup>3</sup>/min leakage for each MWt of design thermal power. For BWRs of less than 8.0E+02 MWt design thermal power level, a minimum condenser air leakage of 5.0E+00 ft<sup>3</sup>/min should be used, independent of reactor design thermal power level.

**Table 4-25 Condenser air leakage**

Plant <sup>a</sup>	Power Level (MWt)	Air Leakage (ft <sup>3</sup> /min)	Extrapolated to 3.4E+03 MWt/3 Shell (ft <sup>3</sup> /min)
Dresden 2	1.6E+03	4.4E+00	9.4E+00
TVA Fossil Plants (average of 6)	7.0E+02 <sup>b</sup>	6.7E+00 <sup>b</sup>	2.87E+01
Quad Cities 1	2.51E+03	9.6E+00	1.3E+01
Quad Cities 2	2.51E+03	2.5E+01	3.4E+01
Average			2.1E+01

<sup>a</sup> These values are from Section 2.2.8.2 of NUREG-0016, Revision 1.

<sup>b</sup> The TVA results are an average of six fossil plants with power levels reported in units of MWe and reported air leakage data in units of ft<sup>3</sup>/min per condenser shell.

#### 4.1.1.8 Holdup Times for Charcoal Delay Systems

##### Parameter

Holdup times for charcoal delay systems are calculated using Equation (4-7):

$$T = (4.31E+01) \left( \frac{M(K)}{P} \right) \quad (4-7)$$

where T = the holdup time, in hours;  
P = the thermal power level, in MWt;  
K = the dynamic adsorption coefficient, in cm<sup>3</sup>/g;  
M = the mass of charcoal adsorber, in 1.0E+03 lb; and  
4.31E+01 = the factor to convert from (1.0E+03 lb cm<sup>3</sup>/g)/(ft<sup>3</sup>/min/MWt) to hours, Equation (4-8).

$$T(h) = \frac{M (1.0E+03 \text{ lb}) K (\text{cm}^3/\text{g}) (4.54E+02 \text{ g/lb}) (3.53E-05 \text{ ft}^3/\text{cm}^3)}{P (\text{MWt}) \left( \frac{6.2E-03 (\text{ft}^3/\text{min})}{\text{MWt}} \right) (6.0E+01 \text{ min/h})} = 4.31E+01 \left( \frac{M(K)}{P} \right) \quad (4-8)$$

Table 4-26 shows dynamic adsorption coefficients, K (in cm<sup>3</sup>/g).

**Table 4-26 Dynamic adsorption coefficient for xenon and krypton**

Nuclide <sup>a</sup>	Dynamic Adsorption Coefficient (cm <sup>3</sup> /g)			
	Operating 77 °F Dew Point 45 °F	Operating 77 °F Dew Point 0 °F	Operating 77 °F Dew Point -40 °F	Operating 0 °F Dew Point -20 °F
Kr	1.85E+01	2.5E+01	7.0E+01	1.05E+02
Xe	3.3E+02	4.4E+02	1.16E+03	2.41E+03

<sup>a</sup> The data in this table are taken from Section 2.2.9.1 of NUREG-0016, Revision 1.

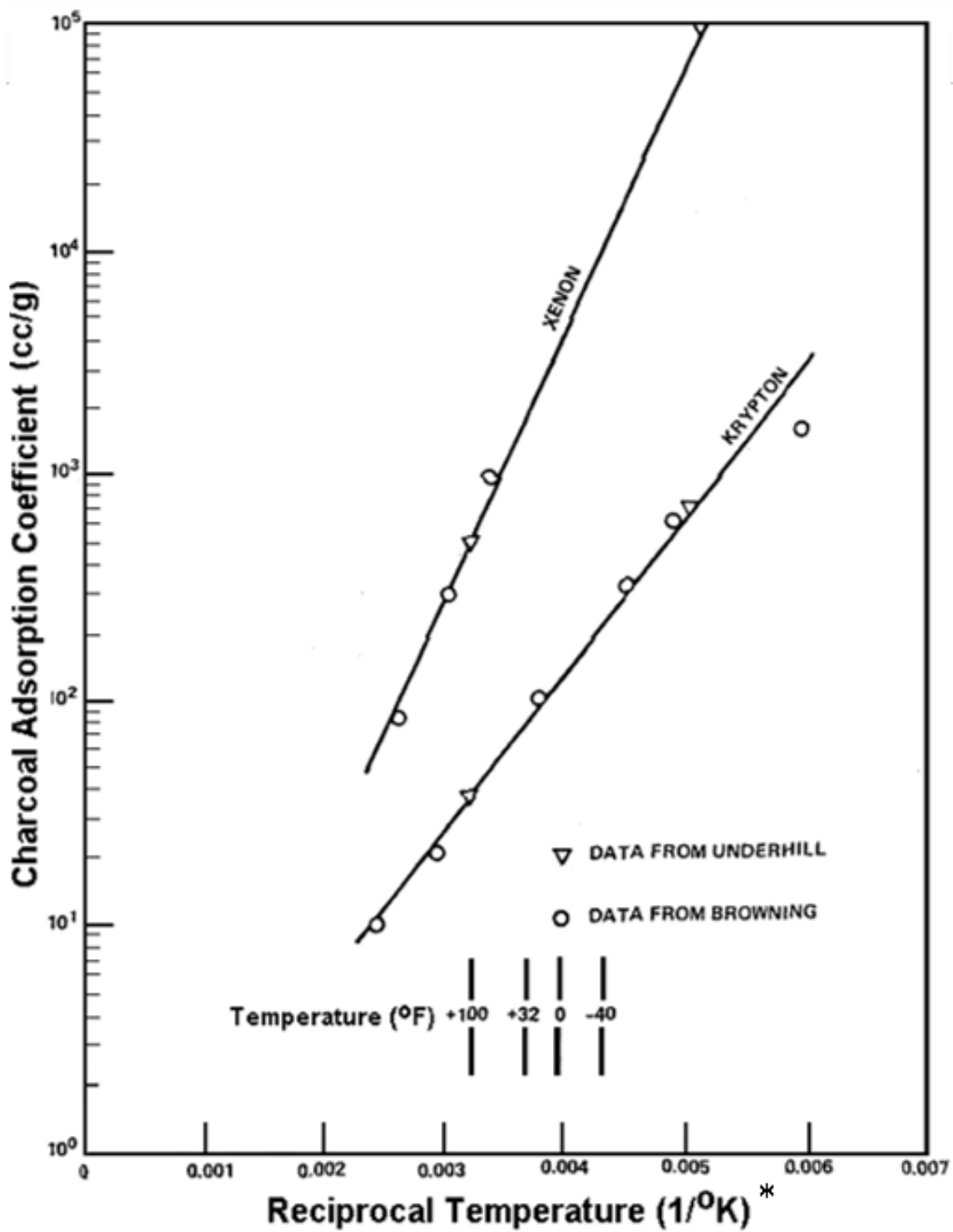
**Basis**

Charcoal delay systems are evaluated using Equation (4-7) and dynamic adsorption coefficients. Equation (4-7) is a standard equation for the calculation of delay times in charcoal adsorption systems (Reference 3). The dynamic adsorption coefficients (K) for xenon and krypton are dependent on operating temperature and moisture content (Reference 3) in the charcoal, as indicated by the values in Table 4-26. The K values represent a composite of data from operating reactor charcoal delay systems (Reference 3) and reports concerning charcoal adsorption systems (Reference 3).

Four factors influence the selection of K values:

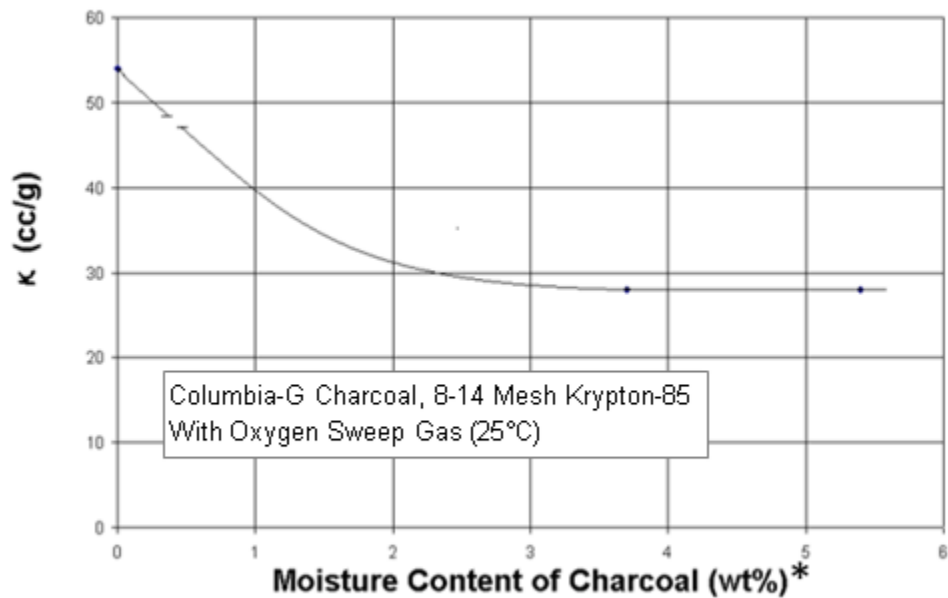
- (1) operational data from Gundremmingen NPP, Lingen NPP, and Vermont Yankee (Reference 3)
- (2) the effect of temperature on the dynamic adsorption coefficients, indicated in Figure 4-1 (Reference 3)
- (3) the effect of moisture on the dynamic adsorption coefficients, shown in Figure 4-2 and the affinity of charcoal for moisture, shown in Figure 4-3
- (4) the variation in K values between researchers and between the types of charcoal used in these systems (Reference 3), noting that, because of the variation in K values based on different types of charcoal and the data reported, average values taken from Gundremmingen NPP (Reference 3) and Lingen NPP (Reference 3) data shown in Figure 4-1 are used





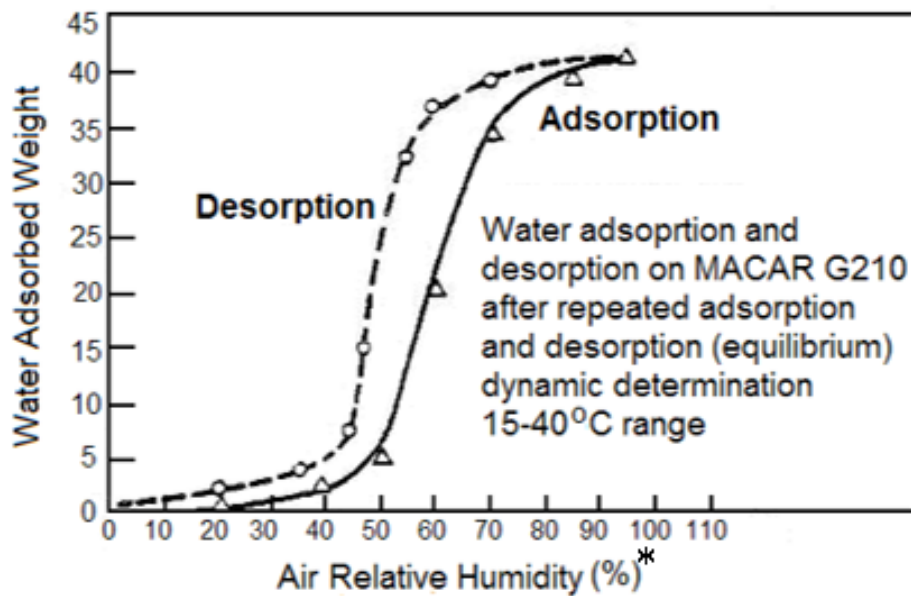
\* Figure 2-2 of NUREG-0016, Revision 1.

Figure 4-1 Krypton and xenon K values as a function of reciprocal temperature



\* Figure 2-3 of NUREG-0016, Revision 1.

**Figure 4-2** Effect of moisture content on the dynamic adsorption coefficient



\* Figure 2-4 of NUREG-0016, Revision 1.

**Figure 4-3** Charcoal moisture as a function of relative humidity

#### 4.1.1.9 Decontamination Factors for Cryogenic Distillation

##### Parameter

Table 4-27 lists the DFs for the cryogenic distillation system that are assumed by the code. The holdup times are calculated on the basis of gas residence time in the system before release.

**Table 4-27 Decontamination factors for cryogenic distillation**

Nuclides <sup>a</sup>	DFs
Iodine and Xenon	1.0E+04
Krypton	4.0E+03

<sup>a</sup> The data in this table are taken from Section 2.2.10.1 of NUREG-0016, Revision 1.

##### Basis

A DF of 1.0E+04 for radioiodine and xenon and a DF of 4.0E+03 for krypton are used for a cryogenic distillation system. As stated in NUREG-0016, Revision 1, the values are based on data submitted in Amendment 11 to the FSAR for the Hope Creek Generating Station, Units 1 and 2, which were derived from a proprietary report of Air Products and Chemical, Inc. The FSAR states that a maximum of 0.025-percent krypton (DF = 4.0E+03) and 1.0E-02-percent xenon (DF = 1.0E+04) will escape from the system. These DFs are considered reasonable.

#### 4.1.1.10 Radioiodine Removal Efficiencies for Charcoal Adsorbers and Particulate Removal Efficiencies for HEPA Filters

##### Parameter

Use a removal efficiency of 99 percent for particulate removal by HEPA filtration. For charcoal adsorbers, removal efficiencies for all forms of radioiodine are shown in Table 4-28 from Table 1-5 of NUREG-0016, Revision 1 and Table 2 of RG 1.140, Revision 1, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," issued October 1979 [Ref. 17].

##### Basis

The removal efficiencies assigned to HEPA filters for particulate removal and charcoal adsorbers for radioiodine removal are based on the design, testing, and maintenance criteria recommended in RG 1.140, Revision 1.

**Table 4-28 Removal efficiencies for charcoal adsorbers for iodine removal**

Iodine Removal Efficiencies for Charcoal Adsorbers <sup>a</sup>		Percent Removal Efficiencies for Radioiodine (Percent) <sup>c</sup>
Activated Carbon Bed Depth <sup>b</sup>		
2 inches	Air filtration system designed to operate inside primary containment.	90
2 inches	Air filtration system designed to operate outside the primary containment and relative humidity is controlled at 70 percent.	70
4 inches	Air filtration system designed to operate outside the primary containment and relative humidity is controlled at 70 percent.	90
6 inches	Air filtration system designed to operate outside the primary containment and relative humidity is controlled to 70 percent.	99

<sup>a</sup> These values are from Table 1-6 and Section 2.2.11.1 of NUREG-0016, Revision 1, and Table 2 of RG 1.140, Revision 1.

<sup>b</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>c</sup> The removal efficiencies assigned to HEPA filters for particulate removal and charcoal adsorbers for radioiodine removal are based on the design, testing, and maintenance criteria recommended in RG 1.140, Revision 1.

#### 4.1.1.11 Liquid Waste Inputs

##### Parameter

The BWR liquid waste flow rates listed in Table 4-29 are used as inputs to the liquid radwaste treatment system. Flows that cannot be standardized are added to those listed in Table 4-29 to fit an individual application. Disposition of liquid streams to the appropriate collection tanks is based on the applicant's proposed method of processing.

##### Basis

The liquid waste inputs are based on the values proposed by ANSI/ANS-55.3-1976, "Boiling-Water Reactor Liquid Radioactive Waste Processing System" (ANSI N197-1976) [Ref. 18]. Activity inputs are based on the reactor coolant concentrations given in Section A.2.3. The values given are those that were judged to be representative for a typical BWR design.

**Table 4-29 BWR liquid wastes**

Source <sup>a</sup>	Expected Daily Average Input Flow Rate		Fraction of the PCA
	Deep-Bed Plant with URC (gal/d)	Deep-Bed Plant without URC or a Filter/Demineralizer Plant (gal/d)	
<b>Equipment Drains</b>			
Drywell	3.4E+03	3.4E+03	1.0E+00
Containment, auxiliary building, and fuel pool	3.7E+03	3.7E+03	1.0E-01
Radwaste building	1.1E+03	1.1E+03	1.0E-01
Turbine building	3.0E+03	3.0E+03	1.0E-03
URC	1.5E+04		5.0E-02
Resin rinse <sup>b</sup>	2.5E+03	5.0E+03	2.0E-03
<b>Floor Drains</b>			
Drywell	7.0E+02	7.0E+02	1.0E-03
Containment, auxiliary building, and fuel handling	2.0E+03	2.0E+03	1.0E-03
Radwaste building	1.0E+03	1.0E+03	1.0E-03
Turbine building	2.0E+03	2.0E+03	1.0E-03
<b>Other Sources</b>			
Cleanup phase separator decant	6.4E+02	6.4E+02	2.0E-03
Laundry drains	1.0E+03	1.0E+03	
Lab drains	5.0E+02	5.0E+02	2.0E-02
Regenerants <sup>b</sup>	1.7E+03	3.4E+03	c
Condensate demineralizer backwash <sup>d</sup>		8.1E+03	2.0E-06
Chemical lab waste	1.0E+02	1.0E+02	2.0E-02

<sup>a</sup> These values are from Table 2-27 of NUREG-0016, Revision 1.

<sup>b</sup> Deep-bed condensate demineralizers only.

<sup>c</sup> Calculated by code.

<sup>d</sup> Filter/demineralizer (Powdex) condensate demineralizers only.

#### 4.1.1.12 *Chemical Wastes from Regeneration of Condensate Demineralizers*

##### Parameter

The code uses three assumptions:

- (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 URC, use 8 days times the number of demineralizers.

##### Basis

Operating data from Dresden Units 2 and 3 (Reference 3) indicate that one condensate demineralizer regeneration occurs every 3.5 days when a URC 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 a URC is used to remove insolubles between regenerations, operating data from Dresden Units 2 and 3 indicate that one condensate demineralizer regeneration occurs every 7.1 days (Reference 3), from Pilgrim 1 at 8.2 days (Reference 3), and from Nine Mile Point 1 at 10 days (Reference 3).

#### 4.1.1.13 *Detergent Waste*

##### Parameter

For plants with an onsite laundry, use the radionuclide distribution given in Table 4-30 for untreated detergent wastes from NUREG-0017, Revision 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized-Water Reactors (GALE-PWR code)," issued April 1985 [Ref. 19]. The quantities shown in Table 4-30 should be added to the adjusted liquid source term. Detergent waste releases should be reduced, using appropriate DFs from this report if treatment is provided.

##### Basis

In the evaluation of liquid radwaste treatment systems, it is assumed that detergent wastes (laundry and personnel drains) will have the radionuclide distribution given in Table 4-30. The radionuclide distribution is based on measurements at four nuclear power plants, which are given in Table 4-31 from NUREG-0017, Revision 1.

**Table 4-30 Calculated annual release of radioactive materials in untreated detergent waste**

Radionuclide <sup>a</sup>	Ci/yr/reactor
P-32	1.8E-04
Cr-51	4.7E-03
Mn-54	3.8E-03
Fe-55	7.2E-03
Fe-59	2.2E-03
Co-58	7.9E-03
Co-60	1.4E-02
Ni-63	1.7E-03
Sr-89	8.8E-05
Sr-90	1.3E-05
Y-91	8.4E-05
Zr-95	1.1E-03
Nb-95	1.9E-03
Mo-99	6.0E-05
Ru-103	2.9E-04
Ru-106	8.9E-03
Ag-110m	1.2E-03
Sb-124	4.3E-04
I-131	1.6E-03
Cs-134	1.1E-02
Cs-136	3.7E-04
Cs-137	1.6E-02
Ba-140	9.1E-04
Ce-141	2.3E-04
Ce-144	3.9E-03
<b>Total</b>	<b>9.0E-02 Ci</b>

<sup>a</sup> The data in this table are taken from Table 2-28 of NUREG-0017, Revision 1.

**Table 4-31 Radionuclide distribution of detergent waste**

Radionuclide <sup>a</sup>	Oyster Creek (1971–1973) ( $\mu\text{Ci}/\text{month}$ )	Ginna (1972–1973) ( $\mu\text{Ci}/\text{month}$ )	Zion <sup>b</sup> (1977) ( $\mu\text{Ci}/\text{month}$ )	Fort Calhoun (1977) ( $\mu\text{Ci}/\text{month}$ )
P-32	1.5E-02	NA	NA	NA
Cr-51	2.3E-01	NA	9.4E-01	NA
Mn-54	1.3E+00	1.2E-01	1.6E-01	1.9E-02
Fe-55	3.5E-01	NA	1.9E+00	1.6E-01
Fe-59	2.9E-01	NA	2.6E-01	NA
Co-58	3.5E-01	4.1E-01	2.4E+00	1.5E-01
Co-60	3.8E+00	9.0E-01	9.8E-01	3.0E-02
Ni-63	NA	NA	3.5E-01	7.1E-02
Sr-89	2.1E-02	NA	7.0E-03	1.4E-03
Sr-90	2.5E-03	NA	7.6E-04	NA
Y-91	NA	NA	1.4E-02	NA
Zr-95	8.3E-02	1.6E-01	1.4E-01	NA
Nb-95	1.6E-01	2.0E-01	2.7E-01	NA
Mo-99	NA	5.0E-03	NA	NA
Ru-103	1.3E-02	3.2E-02	5.2E-02	NA
Ru-106	NA	7.4E-01	NA	NA
Ag-110m	NA	1.0E-01	NA	NA
Sb-124	6.1E-02	NA	4.7E-02	NA
I-131	4.3E-01	5.5E-02	1.7E-01	1.7E-02
Cs-134	1.7E-01	1.4	1.5E+00	1.4
Cs-136	NA	NA	6.2E-02	NA
Cs-137	2.9E-01	2.5E+00	2.1E+00	1.7E+00
Ba-140	7.6E-02	NA	NA	NA
Ce-141	3.3E-02	5.0E-03	NA	NA
Ce-144	7.3E-02	5.8E-01	NA	NA
<b>Total</b>	<b>7.7E+00</b>	<b>7.2E+00</b>	<b>1.14E+01</b>	<b>3.5E+00</b>

<sup>a</sup> The data in this table are taken from Table 2-28 of NUREG-0017, Revision 1.

<sup>b</sup> For two units

NA = radionuclide was not analyzed



#### 4.1.1.14 Tritium Releases

##### 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 percent 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. Of the tritium in gaseous effluents, 50 percent is released from the turbine building ventilation system, and the remaining 50 percent 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.

##### Basis

Table 4-32 lists the annual 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 percent, the total annual release is approximately 3.0E-02 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 4-33 lists the percentage of total tritium that is released in liquid effluents (based on the data in Table 4-32). The weighted average indicates that approximately 50 percent 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 (Reference 15), Table 4-34 provides the distribution of tritium released from various sources within the plant. Based on data in Table 4-34, approximately 50 percent 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 percent 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 are available only from operating reactors (Mark I containments), where the spent fuel pool area is inside containment. It is not possible with the present database to identify what fraction of the tritium from the reactor building is associated with the spent fuel pool area. Accordingly, until sufficient data are 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).

**Table 4-32 Annual tritium release data from operating BWRs**

Reactor <sup>a</sup>	Power per Unit (MWt)	Startup Date (year)	1972	1973	1974	1975	1976	1977 <sup>b</sup>
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.54	0.45	0.46	0.41	0.49	0.41
			Gaseous Tritium Released (Ci/yr)					
Oyster Creek	1930	1969	0.8	0.4	0.4	2.8	1.1	0.7
			Liquid Tritium Released (Ci/yr)					
			62	36.6	14.1	18	38	3.4
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.034	0.047	0.009	0.015	0.023	0.003
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.42	0.46	0.44	0.40	0.55	0.38
			Gaseous Tritium Released (Ci/yr)					
Nine Mile Point 1	1850	1969	18	26.8	c	20	19	33
			Liquid Tritium Released (Ci/yr)					
			28	46.5	18.7	28	2.5	0.5
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.032	0.047	0.012	0.035	0.011	0.026
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.42	0.42	0.42	0.42	0.42	0.42
			Gaseous Tritium Released (Ci/yr)					
Dresden 2 & 3	2577	1970/71	18	18	18	18	18	18
			Liquid Tritium Released (Ci/yr)					
			28	28	28	28	28	28
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.032	0.032	0.032	0.032	0.032	0.032

<sup>a</sup> These values are from Table 2-30 of NUREG-0016, Revision 1, and semiannual reports of reactors listed.

<sup>b</sup> 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.

<sup>c</sup> No reported data.

<sup>d</sup> No measurement made.

<sup>e</sup> Before first refueling.

<sup>f</sup> Measured only during the July–December 1973 period.

<sup>g</sup> Average weighted nuclear thermal output.

**Table 4-32 Annual tritium release data from operating BWRs (cont.)**

Reactor <sup>a</sup>	Power per Unit (MWt)	Startup Date (year)	1972	1973	1974	1975	1976	1977 <sup>b</sup>
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.40	0.25	0.47	0.50	0.48	0.62
			Gaseous Tritium Released (Ci/yr)					
Millstone 1	2011	1970	4.2	1.7	2.8	17	29	33
			Liquid Tritium Released (Ci/yr)					
			21	3.7	24.1	80	20	7.5
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.018	0.006	0.017	0.057	0.030	0.019
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.46	0.41	0.37	0.37	0.51	0.46
			Gaseous Tritium Released (Ci/yr)					
Monticello	1670	1970	12	c	c	66	77	139
			Liquid Tritium Released (Ci/yr)					
			d	d	d	0	0	0
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.008			0.052	0.044	0.088
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.06	0.18	0.34	0.47	0.42	0.46
			Gaseous Tritium Released (Ci/yr)					
Vermont Yankee	1953	1972	e	1.0	0.9	7.1	14	28
			Liquid Tritium Released (Ci/yr)					
			e	0.2	d	0	1.6	0.1
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.002	0.001	0.004	0.011	0.018	

<sup>a</sup> These values are from Table 2-30 of NUREG-0016, Revision 1, and semiannual reports of reactors listed.

<sup>b</sup> 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.

<sup>c</sup> No reported data.

<sup>d</sup> No measurement made.

<sup>e</sup> Before first refueling.

<sup>f</sup> Measured only during the July–December 1973 period.

<sup>g</sup> Average weighted nuclear thermal output.

**Table 4-32 Annual tritium release data from operating BWRs (cont.)**

Reactor <sup>a</sup>	Power per Unit (MWt)	Startup Date (year)	1972	1973	1974	1975	1976	1977 <sup>b</sup>
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.52	1.32	1.09	0.96	1.08	1.12
			Gaseous Tritium Released (Ci/yr)					
Quad Cities 1 & 2	2511	1971/72	4.7	34 <sup>f</sup>	29	280	300	40
			Liquid Tritium Released (Ci/yr)					
			4.7	24.5	34	54	24	19
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.005	0.013	0.017	0.10	0.088	0.016
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.11	0.53	0.25	0.34	0.32	0.34
			Gaseous Tritium Released (Ci/yr)					
Pilgrim 1	1998	1972	c	14	8	74	37	61
			Liquid Tritium Released (Ci/yr)					
			4.2	0.4	10.4	18	47	33
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.011	0.008	0.22	0.079	0.077	0.080
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
						1.39	1.55	1.19
			Gaseous Tritium Released (Ci/yr)					
Peach Bottom 2 & 3	3293	1973/74				0.3	27	260
			Liquid Tritium Released (Ci/yr)					
						31	74	71
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
						0.007	0.019	0.081

<sup>a</sup> These values are from Table 2-30 of NUREG-0016, Revision 1, and semiannual reports of reactors listed.

<sup>b</sup> 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.

<sup>c</sup> No reported data.

<sup>d</sup> No measurement made.

<sup>e</sup> Before first refueling.

<sup>f</sup> Measured only during the July–December 1973 period.

<sup>g</sup> Average weighted nuclear thermal output.

**Table 4-32 Annual tritium release data from operating BWRs (cont.)**

Reactor <sup>a</sup>	Power per Unit (MWt)	Startup Date (year)	1972	1973	1974	1975	1976	1977 <sup>b</sup>
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.36    0.37    2.24					
			Gaseous Tritium Released (Ci/yr)					
Browns Ferry 1, 2, & 3	3293	1973/74 & 76	5.1    0.6    23					
			Liquid Tritium Released (Ci/yr)					
			10    4.0    22					
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.012    0.004    0.006					
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.52    0.49    0.60					
			Gaseous Tritium Released (Ci/yr)					
Cooper	2381	1974	43    67    50					
			Liquid Tritium Released (Ci/yr)					
			8.3    8.3    9.0					
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.029    0.045    0.029					
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.41    0.57    0.51					
			Gaseous Tritium Released (Ci/yr)					
Hatch 1	2436	1974	1.8    1.4    1.2					
			Liquid Tritium Released (Ci/yr)					
			6.1    9.0    14					
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.006    0.005    0.009					

<sup>a</sup> These values are from Table 2-30 of NUREG-0016, Revision 1, and semiannual reports of reactors listed.

<sup>b</sup> 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.

<sup>c</sup> No reported data.

<sup>d</sup> No measurement made.

<sup>e</sup> Before first refueling.

<sup>f</sup> Measured only during the July–December 1973 period.

<sup>g</sup> Average weighted nuclear thermal output.

**Table 4-32 Annual tritium release data from operating BWRs (cont.)**

Reactor <sup>a</sup>	Power per Unit (MWt)	Startup Date (year)	1972	1973	1974	1975	1976	1977 <sup>b</sup>
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.28    0.53    0.49					
			Gaseous Tritium Released (Ci/yr)					
FitzPatrick	2436	1974	c    15    9.5					
			Liquid Tritium Released (Ci/yr)					
			c    4.2    28					
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.011    0.007					
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.31    0.33    0.39					
			Gaseous Tritium Released (Ci/yr)					
Duane Arnold	1658	1974	19    16    15					
			Liquid Tritium Released (Ci/yr)					
			0.3    0.3    0.2					
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.018    0.014    0.011					
<p><sup>a</sup> These values are from Table 2-30 of NUREG-0016, Revision 1, and semiannual reports of reactors listed.</p> <p><sup>b</sup> 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.</p> <p><sup>c</sup> No reported data.</p> <p><sup>d</sup> No measurement made.</p> <p><sup>e</sup> Before first refueling.</p> <p><sup>f</sup> Measured only during the July–December 1973 period.</p> <p><sup>g</sup> Average weighted nuclear thermal output.</p>								

**Table 4-32 Annual tritium release data from operating BWRs (cont.)**

Reactor <sup>a</sup>	Power per Unit (MWt)	Startup Date (year)	1972	1973	1974	1975	1976	1977 <sup>b</sup>
			Nuclear Thermal Output/Unit (1.0E+06 MWdt)					
			0.20    0.33    0.66					
			Gaseous Tritium Released (Ci/yr)					
			2.0    22    19					
Brunswick 1 & 2	2436	1975/76	Liquid Tritium Released (Ci/yr)					
			3.2    5.9    7.4					
			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.008    0.025    0.012					
Weighted Average <sup>g</sup>			Total Tritium Released (Ci/yr-MWt at 80-percent capacity)					
			0.021	0.016	0.015	0.043	0.035	0.032

<sup>a</sup> These values are from Table 2-30 of NUREG-0016, Revision 1, and semiannual reports of reactors listed.

<sup>b</sup> 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.

<sup>c</sup> No reported data.

<sup>d</sup> No measurement made.

<sup>e</sup> Before first refueling.

<sup>f</sup> Measured only during the July–December 1973 period.

<sup>g</sup> Average weighted nuclear thermal output.

**Table 4-33 Annual percentage tritium liquid effluent release rates from operating BWRs**

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

<sup>a</sup> These values are from Table 2-31 of NUREG-0016, Revision 1, and semiannual reports of reactors listed.

<sup>b</sup> Insufficient data. Before first refueling.

<sup>c</sup> Average weighted by thermal output.

**Table 4-34 Distribution of tritium release gaseous effluents**

Plant <sup>a</sup>	Source of Gaseous Tritium Release (Percent of Total)			
	Reactor Building	Turbine Building	Miscellaneous	Total
Monticello	68	29	3	100
Vermont Yankee	35	53	12	100
Oyster Creek	13	79	8	100
Average	39	54	7	100

<sup>a</sup> These values are from Table 2-32 of NUREG-0016, Revision 1.



#### 4.1.1.15 Decontamination Factors for Demineralizers

##### Parameter

Table 4-35 lists DFs for demineralizers used on process or radwaste streams by the code when the user selects the default (GALE86) fixed modeling parameters.

**Table 4-35 Decontamination factors for demineralizers**

Demineralizer Type <sup>a</sup>	Anion <sup>b</sup>	Cesium & Rubidium <sup>b</sup>	Other Radionuclides <sup>b</sup>
<b>Mixed bed (H<sup>+</sup>, OH<sup>-</sup>)</b>			
Reactor Coolant Cleanup	1.0E+01	2.0E+00	1.0E+01
Condensate (Deep Bed)	1.0E+01	1.0E+00	1.0E+01
Clean Waste (High-Purity Waste)	1.0E+02 (1.0E+01)	1.0E+01 (1.0E+01)	1.0E+02 (1.0E+01)
Dirty Waste (Low-Purity Waste)	1.0E+02 (1.0E+01)	2.0E+01 (1.0E+01)	1.0E+02 (1.0E+01)
<b>Cation bed (H<sup>+</sup>)</b>			
Dirty Waste (Any System)	1.0E+00 (1.0E+00)	1.0E+01 (1.0E+01)	1.0E+02 (1.0E+01)
Powdex (Any System)	1.0E+01 (1.0E+01)	2.0E+01 (1.0E+01)	1.0E+02 (1.0E+01)

<sup>a</sup> These values are from Section 2.2.16.1 of NUREG-0016, Revision 1.

<sup>b</sup> For two demineralizers in series, the DF for the second demineralizer is given in parentheses.

##### Basis

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 (NUREG/CR-0143, "The Use of Ion Exchange to Treat Radioactive Liquids in Light-Water-Cooled Nuclear Power Plants," issued August 1978 [Ref. 20]). This reference contains operating and theoretical data that provide 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:

- 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 NUREG/CR-0143 indicate that cesium and rubidium will be more strongly

exchanged in the second demineralizer in series than in the first, as the concentration of preferentially exchanged competing nuclides is reduced.

- As indicated in NUREG/CR-0143, 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 stream.

#### 4.1.1.16 Decontamination Factors for Evaporators

##### Parameter

Table 4-36 lists the DFs for evaporators used by the code when the user selects the default (GALE86) fixed modeling parameters.

**Table 4-36 Decontamination factors for evaporators**

Evaporator <sup>a</sup>	All Nuclides Except Anions	Anions
Miscellaneous radwaste evaporators	1.0E+04	1.0E+03
Separate evaporator for detergent wastes	1.0E+02	1.0E+02

<sup>a</sup> These values are from Section 2.2.17.1 of NUREG-0016, Revision 1.

##### Basis

The DFs for evaporators are derived from the findings of a generic review of evaporators used in the nuclear industry (NUREG/CR-0142, "The Use of Evaporation to Treat Radioactive Liquids in Light-Water-Cooled Nuclear Power Plants," issued September 1978 [Ref. 21]. The principal conclusions reached in the report are the following:

- DFs of 1.0E+04 can be expected for nonvolatile radioactive nuclides in a single-stage evaporator.
- DFs for iodine are a factor of 1.0E+02 less than the DFs for nonvolatile nuclides (1.0E+03).
- DFs for wastes containing detergents that tend to foam are a factor of 1.0E+01 to 1.0E+02 lower than DFs expected for nonfoaming wastes.

These conclusions have been extended to consider the following factors:

- For nonvolatile nuclides in a nonfoaming solution, a DF of 1.0E+04 is used.
- For iodine in a nonfoaming solution, a DF of 1.0E+03 is used.
- If an evaporator is used for detergent wastes, the DF for the evaporator is reduced to 1.0E+02 to reflect carryover as a result of foaming, which will reduce the DF.

#### 4.1.1.17 Decontamination Factors for Reverse Osmosis

##### Parameter

An overall DF of 3.0E+02 for laundry wastes and a DF of 1.0E+02 for other liquid radwastes are used.

##### Basis

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 liquid stream to the concentrations in the effluent stream. This ratio is termed "the membrane DF ( $DF_m$ ).” For source term calculations, the system DF ( $DF_s$ ) should be used. The  $DF_s$  is the ratio of the nuclide concentrations in the feed stream to those in the effluent stream. The relationship between the  $DF_s$  and the  $DF_m$  is nonlinear and is a function of the percent recovery. This relationship can be expressed as follows in Equation (4-9):

$$DF_s = \frac{F}{1 - (1 - F)^{1/DF_m}} \quad (4-9)$$

where  $DF_m$  = the membrane DF;  
 $DF_s$  = the system DF; and  
 $F$  = the ratio of effluent volume to inlet volume (fractional recovery).

Tables 4-37 through 4-39 give membrane DFs derived from operating data at Point Beach, Ginna, and the H.B. Robinson Steam Electric Plant (NUREG/CR-0724, "A Study of Reverse Osmosis Applicability to Light Water Reactor Radwaste Processing," issued November 1978 [Ref. 22]) and laboratory data on simulated radwaste liquids (Reference 3). These data indicate that the overall membrane DF is approximately 1.0E+02. The percent recovery for liquid radwaste processes using reverse osmosis is expected to be approximately 95 percent (i.e., 5-percent concentrated liquid). Using these values in Equation (4-9), the system DF is approximately 3.0E+01, as in Equation (4-10).

$$DF_s = \frac{0.95}{1 - (1 - 0.95)^{1/100}} = 3.0E+01 \quad (4-10)$$

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 4-37 through 4-39, the membrane DF for these isotopes is lower than the average membrane DF used in the evaluation for laundry waste.

**Table 4-37 Reverse osmosis decontamination factors for Ginna Station**

Nuclide <sup>a</sup>	Concentrate Activity (μCi/cm <sup>3</sup> )	Product Activity (μCi/cm <sup>3</sup> )	Membrane DF
Ce-144	2.68E-04	<2.2E-07	1.2E+03
Co-58	8.55E-05	<3.4E-08	2.5E+03
Ru-103	5.83E-05	<5.5E-08	1.1E+03
Cs-137	4.09E-04	6.6E-06	6.0E+01
Cs-134	2.02E-04	3.2E-06	6.0E+01
Nb-95	5.35E-05	<5.3E-08	1.0E+03
Zr-95	2.36E-05	<3.7E-08	6.4E+02
Mn-54	8.82E-05	<3.4E-08	2.6E+03
Co-60	9.63E-04	<8.1E-08	1.2E+04
Total Isotopic	2.15E-03	9.8E-06	2.2E+02
Gross Beta	1.63E-03	1.86E-05	8.8E+01
		Average	2.0E+02

<sup>a</sup> These values are from Table 2-33 of NUREG-0016, Revision 1.

**Table 4-38 Expected reverse osmosis decontamination factors for specific nuclides**

Nuclide <sup>a</sup>	Concentrate Activity (μCi/ml)	Product Activity (μCi/ml)	Membrane DF
Co-60	2.5E-04	5.0E-07	5.0E+02
Mo-99	3.8E-02	1.0E-03	4.0E+01
I-131, I-132, I-133, I-134, & I-135	1.2E-01	4.0E-03	3.0E+01
Cs-134 & Cs-137	4.3E-02	2.0E-04	2.0E+02

<sup>a</sup> These values are from Table 2-35 of NUREG-0016, Revision 1.

**Table 4-39 Reverse osmosis decontamination factors for Point Beach**

Date <sup>a</sup>	Time	Concentrate Activity (μCi/ml)	Product Activity (μCi/ml)	Membrane DF
6/14/71	0840	1.1E-05	6.8E-07	1.6E+01
	1225	6.3E-05	4.2E-07	1.5E+02
	1350	6.8E-05	3.2E-07	2.8E+02
6/15/71	1030	2.7E-04	3.1E-06	8.7E+01
	1315	1.0E-04	1.7E-06	5.9E+01
	1440	1.3E-04	1.1E-07	1.2E+03
	1510	1.6E-04	1.1E-07	1.5E+03
	1530	1.8E-04	5.7E-07	3.2E+02

<sup>a</sup> These values are from Table 2-34 of NUREG-0016, Revision 1.

#### 4.1.1.18 Decontamination Factors for Liquid Radwaste Filters

##### Parameter

A DF of 1.0E+00 for liquid radwaste filters is assigned for all radionuclides.

##### Basis

A generic review by Oak Ridge National Laboratory presents the findings on liquid radwaste filters used in the nuclear industry (NUREG/CR-0141, "The Use of Filtration to Treat Radioactive Liquids in Light-Water-Cooled Nuclear Power Plants," issued September 1978 [Ref. 23]). Because of the various filter types and filter media employed, reported values of DFs vary widely, with no discernible trend. The principal conclusion reached in the report is that no credit should be assigned to liquid radwaste filters (DF of 1) until a larger database is obtained.

Additional data from Fort Calhoun Station, Zion, Turkey Point Nuclear Generating, Rancho Seco Nuclear Generating Station, and Prairie Island Nuclear Generating Plant (References 24 through 28) indicate that DFs in liquid radwaste filters vary widely from less than 1.0E+00 to greater than 5.0E+02 (with a mean value of 1.3E+00). Therefore, a DF of 1.0E+00 for liquid radwaste filters is used.

#### 4.1.1.19 Adjustment to Liquid Radwaste Source Terms for Anticipated Operational Occurrences

##### Parameter

- Increase the calculated source term by 1.0E-01 Ci/yr per reactor, using the same isotopic distribution as for the calculated source term to account for AOOs, such as operator errors that result in unplanned releases.

- Assume evaporators to be unavailable for 2 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:
  - High-purity or low-purity waste—Processed through an alternative system (if available) using a discharge fraction consistent with the lower purity system.
  - Chemical waste—Discharged to the environment to the extent holdup capacity or an alternative evaporator is available.
- The following methods should be used for calculating holdup times and effective system DF:
  - Holdup capacity—If two or more holdup tanks are available, assume one tank is full (80-percent capacity) with the remaining tanks empty at the start of the 2-day outage. If there is only one holdup tank, assume that it is 40-percent full at the start of the 2-day outage with a usable capacity of 80 percent.
  - Effective system DF—Should the reserve storage capacity be inadequate for waste holdup over a 2-day evaporator outage, and if an alternate evaporator is 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 1.0E+05 was calculated for a radwaste demineralizer (1.0E+02 from Table 4-27) and radwaste evaporator (1.0E+03 from Table 4-28) in series. If an adjustment were required for the evaporator being out-of-service 2 days/week, with only a 1-day holdup tank capacity, then the effective system DF can be calculated as follows:

- For 6 days (7 – 2 + 1) out of 7, the system DF would be 1.0E+05.
- For the remaining 1 day, the system DF would be 1.0E+02 (only the demineralizer DF is considered). The effective DF is calculated as shown in Equation (4-11):

$$DF = \left[ \left( \frac{6}{7} \right) (1.0E+05) + \left( \frac{1}{7} \right) (1.0E+02) \right]^{-1} = 7.0E+02 \quad (4-11)$$

### Basis

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 from operator errors, 13 because of component failures, 5 because of inadequate procedures or failure to follow procedures, and the remaining 4 from miscellaneous causes such as design errors. Table 4-40 summarizes the findings of this evaluation. Based on the data in Table 4-40, it is estimated that 1.0E-01Ci/reactor-year will be discharged in unplanned releases in liquid effluents. Tritium releases for BWR AOOs were less than 1 percent of the total normal operational release value, and were, therefore, judged to be negligible.

The availability of evaporators in waste treatment systems is expected to be in the range of 60 to 80 percent. Unavailability is attributed to scaling, fouling of surfaces, instrumentation failures, corrosion, and occasional upsets resulting in high carryovers requiring system cleaning.

A value of 2 consecutive days of 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 per week outage, no adjustments are required for the source term. If less capacity is available, the difference between the waste expected during 2 days of normal operation and the available holdup capacity is assumed to follow an alternative route for processing. Because processing through an alternative route implies the 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.

As 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.

**Table 4-40 Frequency and extent of unplanned liquid radwaste releases from operating plants**

Unplanned Liquid Releases <sup>a</sup>	
Total number (unplanned releases)	5.0E+01
Fraction because of personnel error	5.6E-01
Fraction because of component failure	2.6E-01
Fraction because of inadequate procedures or failure to follow procedures	1.0E-01
Fraction because of other causes	8.0E-02
Approximate activity (Ci)	1.062E+01
Fraction of cumulative occurrence per reactor-year (plants reporting releases >5 gal of liquid waste/reactor-year)	1.5E-01
Fraction of cumulative occurrences per reactor-year (plants reporting activity released >1.0E-02 Ci/reactor-year)	2.7E-01
Activity per release (Ci/release)	3.0E-01
Activity released per reactor-year (Ci/reactor-year)	1.0E-01
Volume of release per reactor-year (gal/reactor-year)	1.66E+04

<sup>a</sup> The data in this table are taken from Table 2-36 of NUREG-0016, Revision 1. Values in this table are based on reported values in 1970–1977 licensee event reports.

#### 4.1.1.20 Carbon-14 Releases

##### Parameter

The annual quantity of carbon-14 released from a BWR is 9.5E+00 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.

### Basis

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 Equation (4-12):

$$Q = N_o \sigma_o \phi m t p s \quad (4-12)$$

where

- Q = the production rate of carbon-14 (Ci/yr);
- N<sub>o</sub> = the number of atoms of oxygen-17 per kg of natural water (1.3E+22 atoms/kg);
- σ<sub>o</sub> = the thermal neutron cross section for oxygen-17 (2.4E-25 cm<sup>2</sup>);
- φ = the average thermal neutron flux (3.0E+13 neutrons/cm<sup>2</sup>-s);
- m = the mass of water in the reactor core (3.9E+04 kg);
- t = the maximum irradiation time per year (3.15E+07 s/yr);
- p = the plant capacity factor in percent (0.80); and
- s = the specific activity for carbon-14 (1.03E-22 Ci/atom).

Based on the above parameters, Q = 9.5E+00 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 Ci/yr to the annual production of carbon-14, caused by the low concentration of nitrogen-14 in the reactor coolant (less than 1 part per million by weight), and the low neutron flux in the drywell (approximately 4.0E+08 neutrons/cm<sup>2</sup>-sec).

The annual release of 9.5E+00 Ci of carbon-14 is in good agreement with reported measurements at Nine Mile Point 1 (Reference 3), which found that 8.0E+00 Ci/yr of carbon-14 were released, principally in the form of carbon dioxide.

#### 4.1.1.21 Argon-41 Releases

##### Parameter

The argon-41 input to the MCOTS downstream of the air ejectors is 4.0E+01 μCi/sec. The dynamic adsorption coefficients for argon-41 in charcoal delay beds of an MCOTS are 6.4E+00 cm/g for an ambient temperature charcoal system and 1.6E+01 cm/g for a chilled charcoal system. The holdup time for argon-41 in a charcoal delay system is determined using Equation (4-7) in Section 4.1.1.8.

The argon-41 release from the purging or venting of the drywell is 1.5E+01 Ci/yr.

##### Basis

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 MCOTS. Data from NUREG-0016, Revision 1, summarized in Table 4-41, indicate that the argon-41 input to the MCOTS during the measurements ranged from 5.6E+00 μCi/s to 3.7E+01 μCi/s. Because of the limited duration of these measurements, the mean



value of the data is not used. Instead, in these evaluations, a release rate of 4.0E+01  $\mu\text{Ci/s}$  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 4.1.1.8. Values of the dynamic adsorption coefficient are based on data for ambient and chilled temperature systems (Reference 3). Holdup times for argon-41 are determined using these k values and the delay Equation (4-7) in Section 4.1.1.8.

Argon-41 releases from the drywell are based on data concerning the neutron flux in the drywell (Reference 3) and on an assumed drywell purging frequency of 24 purges per year.

**Table 4-41 Summary of argon-41 releases to the MCOTS**

Plant <sup>a</sup>	Argon-41 Release <sup>b</sup> ( $\mu\text{Ci/s}$ )
Browns Ferry 1	3.8E+01
Browns Ferry 2	1.7E+01
Browns Ferry 3	1.2E+01
	7.1E+00
	5.8E+00
	7.4E+00
	3.4E+01
	3.2E+01
Hatch 1	1.9E+01
	1.2E+01
FitzPatrick	1.6E+01
	3.6E+01

<sup>a</sup> These values are from Table 2-37 of NUREG-0016, Revision 1.

<sup>b</sup> Data in this table are based on measured argon-41 release rates and were adjusted to 3.4E+03 MWt.

#### 4.1.1.22 Guidelines for Rounding off Numerical Values

The estimated annual release of radioactive materials in liquid and gaseous wastes is deemed at best to be no more accurate than two significant figures. The revised output listings file shows all estimated release rates in exponential form with two significant figures. Supplementary listings provide additional precision for use in special applications such as sensitivity studies.

## **4.2 ANS-18.1 Version (Source Term Options)**

Regulatory source terms are deeply embedded in the regulatory policy and practices of the NRC where the current licensing process has evolved over the past 50 years. It is based upon the concept of defense-in-depth in which power plant design, operation, siting, and emergency planning are the cornerstones of safety. This approach encourages nuclear plant designers to incorporate several lines of defense to maintain the effectiveness of physical barriers between radiation sources and materials from workers, members of the public and environment in operational states and, for some barriers, in accident conditions. The various regulatory source terms are used, in part, to establish and confirm the design basis of the nuclear facility and items important to safety; ensuring that the plant design meets the safety and numerical radiological criteria set forth in regulation and subsequent guidance.

Specific examples of the various regulatory source terms can be found in NUREG-0800 which provides information on the NRC staff's regulatory guides. In the past, the normal operating source terms have been calculated using the techniques described in the various versions of the ANSI/ANS-18.1 standard. The purpose of this standard is to provide a set of typical radionuclide concentrations for estimating the radioactivity in the principal fluid streams of a LWR. The referenced coolant concentrations are based on collected data and calculations reported by various sources which include the NRC, EPRI, and utilities. The values in this standard are considered as representative coolant concentrations in a LWR over its lifetime based upon the currently available data. It is important for current operating reactor licensees and new reactor applicants to apply a best estimate reactor coolant radiological source term for their design of plant-specific purification and radioactive waste systems and to estimate the expected release of radioactivity via various effluent streams.

Each subsequent update to the ANSI/ANS-18.1 standard reflects improvements in industry fuel performance, burnup, power uprates, treatment technologies and practices, and reduction of occurrences and severity of fuel defects. Therefore, the selection of an ANSI/ANS-18.1 version (1984, 1999, or 2016) should reflect the current design and operating conditions of the given facility/design. This is so because it has been demonstrated that previous versions of ANSI/ANS-18.1 could be non-conservative for some dosimetrically important radionuclides for facilities which have undergone significant power uprates (125 percent plus rated thermal power) and received approval to burn their cores much hotter (e.g., from 35 GWd/MTU up to 55-62 GWd/MTU) than when originally licensed. For example, in reviewing the offsite dose analyses for the various power uprate license amendment requests, a historical assumption has been that radionuclide inventories increase linearly with reactor power. The two factors mentioned above, among others, challenge this traditional assumption. The ANSI/ANS-18.1-2016 source term is, in general, lower than the ANSI/ANS-18.1-1999 source term, but there are 18 radionuclides in ANSI/ANS-18.1-2016 with higher reactor water concentrations having ratios (new/old) that exceed 10. Therefore, use of the ANSI/ANS-18.1-2016 standard may not be appropriate for a SMR designed to run at low reactor fuel burnup levels. In fact, it may be best for these reactor designs to apply the previous standards which were developed at a time before many of the LWRs were approved for power uprates and increased reactor fuel burnup levels.

In GALE-BWR 3.2, the user can select from three options for the reactor coolant source term, radionuclide concentrations in the primary and secondary coolant, as shown in Figure 3-4. Section 4.2.1 describes the reactor coolant source term values for the default option, "ANS-18.1 Version-1999", which corresponds to the values in ANSI/ANS-18.1-1999 and is consistent with

the guidance in DC/COL-ISG-05; RG 1.112, Revision 1; and NUREG-0800. Section 4.2.2 describes the reactor coolant source term values for the “ANS 18.1 Version–1984” option, which corresponds to the values in ANSI/ANS-18.1-1984 and is consistent with the guidance in DC/COL-ISG-05. Section 4.2.3 describes the reactor coolant source term values for the “ANS 18.1 Version–2016” option, which corresponds to the values in ANSI/ANS-18.1-2016.

#### **4.2.1 ANSI/ANS-18.1-1999 Source Term Parameters**

##### Parameter

As used in the GALE-BWR 3.2 code, the default values for the reactor coolant source term, ANS-18.1 version, are from ANSI/ANS-18.1-1999. Table 4-42 lists the expected radionuclide concentrations in the reactor coolant and steam for BWRs with design parameters within the ranges listed in Table 4-43. If any design parameter is outside the range in Table 4-43, the code adjusts the concentrations in Table 4-42, using the factors in Tables 4-44 and 4-45. Figure 4-4 shows the relationship of the design parameters.

**Table 4-42 Radionuclide concentrations in BWR coolant and main steam – ANSI/ANS-18.1-1999**

Radionuclide <sup>a</sup>	Reactor Water <sup>b</sup> ( $\mu\text{Ci/g}$ )	Reactor Steam <sup>c</sup> ( $\mu\text{Ci/g}$ )
<b>Noble Gases</b>		
Kr-83m		5.9E-04
Kr-85m		1.0E-03
Kr-85		4.0E-06
Kr-87		3.3E-03
Kr-88		3.3E-03
Kr-89		2.1E-02
Xe-131m		3.3E-06
Xe-133m		4.9E-05
Xe-133		1.4E-03
Xe-135m		4.4E-03
Xe-135		3.8E-03
Xe-137		2.6E-02
Xe-138		1.5E-02
<b>Halogens</b>		
I-131	2.2E-03	4.4E-05
I-132	2.2E-02	4.4E-04
I-133	1.5E-02	3.0E-04
I-134	4.3E-02	8.6E-04
I-135	2.2E-02	4.4E-04
<b>Cesium &amp; Rubidium</b>		
Rb-89	5.0E-03	5.0E-06
Cs-134	3.0E-05	3.0E-08
Cs-136	2.0E-05	2.0E-08
Cs-137, Ba-137m <sup>d</sup>	8.0E-05	8.0E-08
Cs-138	1.0E-02	1.0E-05
<b>Water Activation Products</b>		
N-16	6.0E+01	5.0E+01
<b>Tritium</b>		
H-3	1.0E-02	1.0E-02
<b>Other Radionuclides</b>		
Na-24	2.0E-03	2.0E-06
P-32	4.0E-05	4.0E-08
Cr-51	3.0E-03	3.0E-06

<sup>a</sup> These concentrations are from Table 5 of ANSI/ANS-18.1-1999.

<sup>b</sup> The reactor water concentration is specified at the nozzle where reactor water leaves the reactor vessel.

<sup>c</sup> The reactor steam concentration is specified at time = 0, and only the noble gases are used in GALE-BWR 3.2. The other isotopes are not calculated nor printed to the output for reactor steam.

<sup>d</sup> These nuclides are in secular equilibrium; other radionuclide concentrations given are those of the parent.

**Table 4-42 Radionuclide concentrations in BWR coolant and main steam – ANSI/ANS-18.1-1999 (cont.)**

Radionuclide <sup>a</sup>	Reactor Water <sup>b</sup> ( $\mu\text{Ci/g}$ )	Reactor Steam <sup>c</sup> ( $\mu\text{Ci/g}$ )
Mn-54	3.5E-05	3.5E-08
Mn-56	2.5E-02	2.5E-05
Fe-55	1.0E-03	1.0E-06
Fe-59	3.0E-05	3.0E-08
Co-58	1.0E-04	1.0E-07
Co-60	2.0E-04	2.0E-07
Ni-63	1.0E-06	1.0E-09
Cu-64	3.0E-03	3.0E-06
Zn-65	1.0E-04	1.0E-07
Sr-89	1.0E-04	1.0E-07
Sr-90, Y-90 <sup>d</sup>	7.0E-06	7.0E-09
Sr-91	4.0E-03	4.0E-06
Sr-92	1.0E-02	1.0E-05
Y-91	4.0E-05	4.0E-08
Y-92	6.0E-03	6.0E-06
Y-93	4.0E-03	4.0E-06
Zr-95, Nb-95 <sup>d</sup>	8.0E-06	8.0E-09
Mo-99, Tc-99m <sup>d</sup>	2.0E-03	2.0E-06
Ru-103, Ru-103m <sup>d</sup>	2.0E-05	2.0E-08
Ru-106, Rh-106 <sup>d</sup>	3.0E-06	3.0E-09
Ag-110m	1.0E-06	1.0E-09
Te-129m	4.0E-05	4.0E-08
Te-131m	1.0E-04	1.0E-07
Te-132	1.0E-05	1.0E-08
Ba-140, La-140 <sup>d</sup>	4.0E-04	4.0E-07
Ce-141	3.0E-05	3.0E-08
Ce-144, Pr-144 <sup>d</sup>	3.0E-06	3.0E-09
W-187	3.0E-04	3.0E-07
Np-239	8.0E-03	8.0E-06

<sup>a</sup> These concentrations are from Table 5 of ANSI/ANS-18.1-1999.

<sup>b</sup> The reactor water concentration is specified at the nozzle where reactor water leaves the reactor vessel.

<sup>c</sup> The reactor steam concentration is specified at time = 0, and only the noble gases are used in GALE-BWR 3.2. The other isotopes are not calculated nor printed to the output for reactor steam.

<sup>d</sup> These nuclides are in secular equilibrium; other radionuclide concentrations given are those of the parent.

**Table 4-43 Parameters used to describe the reference BWR**

Parameter <sup>a</sup>	Symbol	Units	Nominal Value	Range <sup>b</sup>	
				Maximum	Minimum
Thermal power	P	MWt	3.4E+03	3.8E+03	3.0E+03
Weight of water in the reactor vessel	WP	lb	3.8E+05	4.2E+05	3.4E+05
Cleanup demineralizer flow rate	FA	lb/h	1.3E+05	1.5E+05	1.1E+05
Steam flow rate	FS	lb/h	1.5E+07	1.7E+07	1.3E+07
Ratio of condensate demineralizer flow rate to steam flow rate	NC <sup>c</sup>	--	1.0E+00	9.9E-0100	0.5E-01

<sup>a</sup> These values are from Table 1 of ANSI/ANS-18.1-1999, -1984 and -2016.

<sup>b</sup> These values are from Table 2-3 of NUREG-0016, Revision 1.

<sup>c</sup> The referenced plant is assumed to be a nonpumped-forward drained plant. 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 GALE-BWR 3.2 code is 1.8E-01 for radioiodine and 1.0E-02 for cesium, rubidium, and other radionuclides. For a BWR that has a ratio of condensate demineralizer flow rate to steam flow rate equal to 1.0E+00 (i.e., full-flow condensate demineralizers), a value of NC = 1.0E+00 is used in the GALE-BWR 3.2 code, which is consistent with the values established in NUREG-0016, Revision 1.

**Table 4-44 Values used in determining adjustment factors for BWRs – ANSI/ANS-18.1-1999**

Symbol	Description	Element Class <sup>a</sup>					
		Noble Gases	Halogens	Cesium & Rubidium	Water Activation Products	H-3	Other Radionuclides
NA	Fraction of material removed in the reactor water cleanup system	0.0E+00	9.0E-01	5.0E-01	b	0.0E+00	9.0E-01 <sup>c</sup>
NB	Fraction of material removed by the condensate demineralizer.	0.0E+00	9.0E-01	5.0E-01	b	0.0E+00	9.0E-01
NS	Ratio of the concentration in reactor steam to the concentration in reactor water	d	2.0E-02 <sup>e</sup>	1.0E-03	b	1.0E+00	1.0E-03
R <sub>n</sub>	Removal rate from the reactor water (h <sup>-1</sup> ) <sup>f</sup>	d	1.0E+01	1.7E-01	b	g	3.4E-01 <sup>c</sup>

<sup>a</sup> These values are from Table 8 of ANSI/ANS-18.1-1999.

<sup>b</sup> In reactor coolant, water activation products exhibit varying chemical and physical properties that are not well defined. However, most are stripped off as gases, which are not effectively removed by the demineralizers of the systems; their concentrations are controlled by decay. The activity concentrations in reactor water and steam are subject to change when hydrogen water chemistry is employed (see Table 4-50 for adjustment factors).

<sup>c</sup> These represent effective removal terms and include other mechanisms, such as plateout. Plateout would be applicable to radionuclides such as molybdenum and corrosion products.

<sup>d</sup> Noble gases released from the core are rapidly transported 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.

<sup>e</sup> The value of 2.0E-02 is used for BWRs that have deep-bed condensate treatment. A value of 1.5E-02 is also used for BWRs. For BWRs that have Powdex Condensate Treatment systems and copper condenser tubing, a value of 5.0E-03 should be used.

<sup>f</sup> These values of R<sub>n</sub> apply to the reference BWRs whose parameters are given in Table 4-43 and have been used in developing Table 4-50. For BWRs not included in Table 4-43, the appropriate value for R<sub>n</sub> may be determined by the following equation:

$$R_n = \frac{(FA)(NA) + (NC)(FS)(NS)(NB)}{WP} \quad \text{For halogens, CS, Rb, and other nuclides}$$

where the symbols are defined in this table, Tables 4-43 and 4-49, and Figure 4-4. The values for R<sub>n</sub> for noble gases and water activation products are not used in the adjustment factors of Table 4-50.

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

**Table 4-45 Adjustment factors for BWRs – ANSI/ANS-18.1-1999**

Element Class <sup>a</sup>	Reactor Coolant	Reactor Steam
Noble gases <sup>b</sup>	1.0E+00	1.0E+00
Halogens	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{1.0E+00 + \lambda}{R_n + \lambda}\right)$	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{1.0E+00 + \lambda}{R_n + \lambda}\right)$
Cesium & Rubidium	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{1.7E-01 + \lambda}{R_n + \lambda}\right)$	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{1.7E-01 + \lambda}{R_n + \lambda}\right)$
Water activation products <sup>c</sup>	1.0E+00	1.0E+00
Tritium <sup>d</sup>	1.0E+00	1.0E+00
Other radionuclides	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{3.4E-01 + \lambda}{R_n + \lambda}\right)$	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{3.4E-01 + \lambda}{R_n + \lambda}\right)$
Zn-65 <sup>e</sup>	1.0E+01	1.0E+01

<sup>a</sup> These values are from Table 10 of ANSI/ANS-18.1-1999.

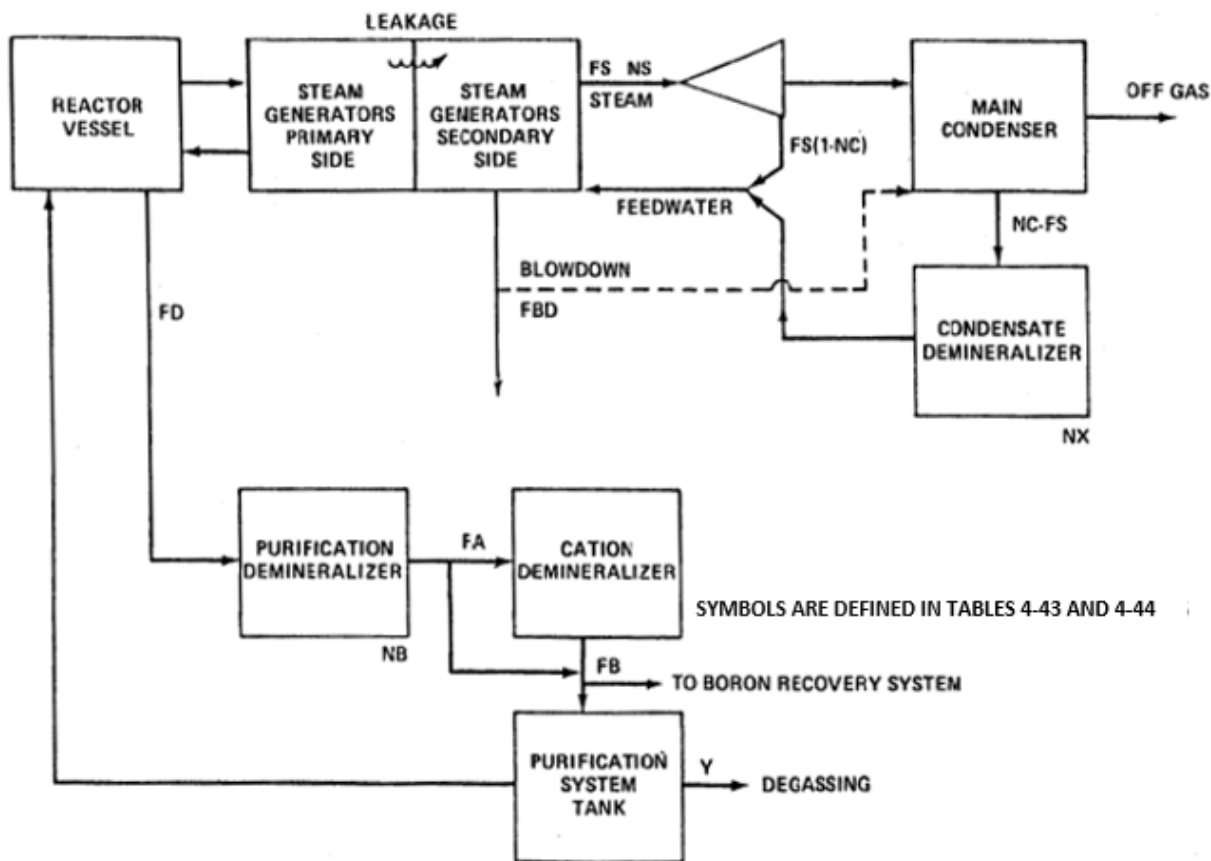
<sup>b</sup> This assumes that the ratio of power to steam flow is essentially the same for all BWRs.

<sup>c</sup> This assumes the ratio of coolant mass to power level is approximately constant.

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

<sup>e</sup> Adjustment factors are for zinc addition plants using natural zinc. Use of depleted zinc would result in a lower adjustment factor, and the decrease is a function of the reduction of zinc-64.





**Figure 4-4 Removal paths for the reference BWR**

Bases

The radionuclide concentrations, adjustment factors, and procedure for effecting adjustments are based on the values and methods in ANSI/ANS-18.1-1999. The values in Table 4-42 provide a set of typical radionuclide concentrations in the reactor coolant and steam for reactor designs within the parameters specified in Table 4-43. The values in Table 4-42 are those determined to be representative of radionuclide concentrations in a BWR over its lifetime, based on the ANSI/ANS-18.1-1999 data and models. Some systems will have design parameters that are outside the ranges specified in Table 4-43. For that reason, Tables 4-44 and 4-45 provide a means of adjusting the concentrations to the actual design parameters. The adjustment factors in Tables 4-42 through 4-45 are based on Equation (4-14):

$$C = \frac{s}{w(\lambda+R)K} \tag{4-14}$$

- where C = the specific activity in  $\mu\text{Ci/g}$ ;
- K = a conversion factor,  $4.54\text{E}+02 \text{ gal/lb}$ ;
- R = the removal rate of the radionuclide from the system because of demineralization, leakage, etc. ( $\text{h}^{-1}$ );
- s = the rate of release to and/or production of the radionuclide in the system in  $\mu\text{Ci/h}$ ;

w = the fluid weight in lb; and  
 $\lambda$  = the decay constant ( $h^{-1}$ ).

The following sample calculations illustrate the method by which the GALE-BWR 3.2 code will adjust the radionuclide concentrations in Table 4-42. As indicated in Table 4-45, adjustment factors will be calculated for noble gases, halogens, cesium, rubidium, and other radionuclides.

As an example, the sample case parameter shown below compares with the range of values shown in Table 4-43, as follows in Table 4-46.

**Table 4-46 Sample case parameters versus the range of values shown in Table 4-43**

Parameter	Value	Range
Thermal power level, MWt	3.758E+03	3.0E+03–3.8E+03
Water weight in vessel, lb	4.9E+05	3.4E+05–4.2E+05
Cleanup demineralizer flow, lb/h	1.5E+05	1.1E+05–1.5E+05
Steam flow rate, lb/h	1.54E+07	1.3E+07–1.7E+07
Condensate demineralizer flow fraction	7.5E-01	5.0E-01–9.90E-01

Because one of the parameters (water weight in vessel) in this example 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) Halogen Coolant Activity (iodine-131 is used as an example.)

Using Equation (4-15) from Table 4-45 for the noble gases, the adjustment factor,  $f$ , is calculated as follows:

$$f = \left( \frac{P}{WP} \right) \left( 1.1E+02 \frac{\text{lb}}{\text{MWt}} \right) \left( \frac{8.6E-01 + \lambda}{R_n + \lambda} \right) \quad (4-15)$$

where the terms of the equation are as defined in Tables 4-43 and 4-44.

In calculating  $f$ , the variable  $R_n$  is calculated first by using Equation (4-16) given in Table 4-44 for halogens:

$$R_n = \frac{(FA)(NA) + (NC)(FS)(NS)(NB)}{WP} \quad (4-16)$$

Use the sample case parameters given above in Table 4-46 and the halogen parameters given in Table 4-44 and substitute into Equation (4-16) above.

$$R_n = \frac{(1.5E+05)(9.0E-01) + (1.8E-01)(1.54E+07)(1.5E-02)(9.0E-01)}{4.9E+05} = 3.8E-01$$

Apply Equation (4-21) above using the computed value of  $R_n$ :

$$f = \left( \frac{3.758E+03}{4.9E+05} \right) \left( 1.1E+02 \frac{\text{lb}}{\text{MWt}} \right) \left( \frac{8.6E-01 + 3.6E-03}{3.8E-01 + 3.6E-03} \right) = 1.899E+00 = 1.9E+00$$

The adjusted iodine-131 primary coolant concentration is computed as:

$$\begin{aligned}
 &= (\text{adjustment factor}) \times (\text{standard iodine-131 concentration}) \\
 &= 1.9\text{E}+00 \times 2.2\text{E}-03 \text{ }\mu\text{Ci/g} = 4.2\text{E}-03 \text{ }\mu\text{Ci/g}
 \end{aligned}$$

(2) Cesium and rubidium coolant activity (cesium-137 is used as an example.)

Using Equation (4-17) from Table 4-45 for the cesium and rubidium, the adjustment factor,  $f$ , is calculated as follows:

$$f = \left(\frac{P}{WP}\right) \left(1.1\text{E}+02 \frac{\text{lb}}{\text{MWt}}\right) \left(\frac{1.7\text{E}-01 + \lambda}{R_n + \lambda}\right) \quad (4-17)$$

where the terms in the equation are defined in Tables 4-43 and 4-44.

In calculating  $f$ , the variable  $R_n$  is calculated first by using Equation (4-16) given in Table 4-44. Use the sample case parameters given above in Table 4-46 and the halogen parameters given in Table 4-44 and substitute into Equation (4-16) above:

$$R_n = \frac{(1.5\text{E}+05)(9.0\text{E}-01) + (1.0\text{E}-02)(1.54\text{E}+07)(1.0\text{E}-03)(9.0\text{E}-01)}{4.9\text{E}+05} = 1.5\text{E}-01$$

Use the value of  $R_n$  in Equation (4-17) above.

$$f = \left(\frac{3.758\text{E}+03}{4.9\text{E}+05}\right) \left(1.1\text{E}+02 \frac{\text{lb}}{\text{MWt}}\right) \left(\frac{1.7\text{E}-01 + 2.6\text{E}-06}{1.5\text{E}-01 + 2.6\text{E}-06}\right) = 9.6\text{E}-01$$

The adjusted cesium-137 concentration is computed as:

$$\begin{aligned}
 &= (\text{adjustment factor}) \times (\text{standard cesium-137 concentration}) \\
 &= 9.6\text{E}-01 \times 8.0\text{E}-05 \text{ }\mu\text{Ci/g} = 7.68\text{E}-05 \text{ }\mu\text{Ci/g}
 \end{aligned}$$

(3) Other Nuclides (sodium-24 is used as an example)

Using Equation (4-25) from Table 4-45 for other nuclides, the adjustment factor,  $f$ , is calculated as follows:

$$f = \left(\frac{P}{WP}\right) \left(1.1\text{E}+02 \frac{\text{lb}}{\text{MWt}}\right) \left(\frac{3.4\text{E}-01 + \lambda}{R_n + \lambda}\right) \quad (4-18)$$

where the terms in Equation (4-18) are as defined in Tables 4-43 and 4-44.

In calculating  $f$ , the variable  $R_n$  is calculated first by using Equation (4-16) above. The other nuclide parameters given in Table 4-44 and the sample case parameters given in Table 4-46 are used in Equation (4-16) above.

$$R_n = \frac{(1.5\text{E}+05)(9.0\text{E}-01) + (1.0\text{E}-02)(1.54\text{E}+07)(1.0\text{E}-03)(5.0\text{E}-01)}{4.9\text{E}+05} = 2.8\text{E}-01$$

Use the value of  $R_n$  in Equation (4-18) above.

$$f = \left(\frac{3.758\text{E}+03}{4.9\text{E}+05}\right) \left(1.1\text{E}+02 \frac{\text{lb}}{\text{MWt}}\right) \left(\frac{3.4\text{E}-01 + 4.62\text{E}-02}{2.8\text{E}-01 + 4.62\text{E}-02}\right) = 9.9\text{E}-01 = 1.0\text{E}+00$$

The adjusted sodium-24 concentration is computed as:

$$= (\text{adjusted factor}) \times (\text{standard sodium-24 concentration})$$

$$= 1.0\text{E}+00 \times 2.0\text{E}-03 \text{ }\mu\text{Ci/g} = 2.0\text{E}-03 \text{ }\mu\text{Ci/g}$$

A carryover factor of 2.0E-02 is used to calculate the halogen concentrations in the main steam in Table 4-42 (Table 8 ANSI/ANS-18.1-1999) for BWRs that 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 5.0E-03 is used to calculate the halogen concentrations in the main steam. These values were also used for halogen (iodine) carryover, respectively, for the two types of BWRs, as noted in Table 8 of ANSI/ANS-18.1-2016 (Table 4-52).

The carryover factors from the Table 8 ANSI/ANS-18.1-1984 (Table 4-49) are derived from data taken at operating reactors, which are listed in Table 4-47. The average of the data in Table 4-47 is 1.5E-02 and 4.0E-03 for halogen (iodine) carryover, respectively, for the two types of BWRs listed in the table.

**Table 4-47 Reactor vessel halogen carryover factors at operating BWRs**

BWRs with Deep Bed Condensate and Powdex Treatment Systems with Stainless Steel Condenser Tubing <sup>a</sup>		BWRs with Powdex Treatment Systems and Copper Condenser Tubing <sup>a</sup>	
Facility	PC	Facility	PC
Oyster Creek	2.3E-02	Monticello	4.0E-03
Dresden Unit 2	1.7E-02	Browns Ferry Unit 1	5.0E-03
Dresden Unit 3	2.1E-02	Browns Ferry Unit 2	2.3E-03
Millstone Unit 1	1.2E-02	Browns Ferry Unit 3	3.0E-03
Nine Mile Point Unit 1	2.0E-02	Duane Arnold	4.0E-03
Quad Cities Unit 1	1.3E-02	Hatch Unit 1	3.5E-03
Cooper	1.2E-02	Peach Bottom Unit 2	4.0E-03
FitzPatrick	1.8E-02	Peach Bottom Unit 3	4.4E-03
Pilgrim	8.2E-03	Vermont Yankee	4.0E-03
<b>Average</b>	<b>1.5E-02</b>	<b>Average</b>	<b>4.0E-03</b>

<sup>a</sup> These values are from Table 2-7 of NUREG-0016, Revision 1.

The nominal value of the ratio of the condensate demineralizer flow rate to the steam flow rate is 7.5E-01. This indicates that the nominal case is a design that uses a pumped-forward model; that is, one in which the reactor steam flow is split, with 75 percent flowing to the low-pressure turbines and the main condenser, and 25 percent pumped forward to the feedwater. The fraction pumped forward to the feedwater does not undergo any treatment in the condensate

demineralizers. The radioiodine and cesium, rubidium, and other radionuclides of Table 4-42 preferentially go with the “pumped-forward” fraction. The reason for this is that these radionuclides show a tendency to go with the condensed steam in the moisture separator-reheater drains to the feedwater system. Based on data for Brunswick Steam Electric Plant and Point Beach Nuclear Plant (Reference 3), the ratios used in the GALE-BWR 3.2 code are 82-percent bypass of condensate demineralizers for radioiodine and 99-percent bypass of condensate demineralizer for cesium, rubidium, and other radionuclides of Table 4-42. Because the remaining radionuclides listed in Table 4-42 are not removed in the condensate demineralizers, these remaining radionuclides have not considered the magnitude of bypass for those radionuclides.

The category “Other Nuclides” includes molybdenum, yttrium, and technetium, which are generally present in colloidal suspensions or as “crud.” Although the actual removal mechanism for yttrium, molybdenum, and technetium 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). Consequently, the parameters for ion exchange include plateout of these nuclides.

## 4.2.2 ANSI/ANS-18.1-1984 Source Term Parameters

### Parameter

As mentioned in Section 4.2, in GALE-BWR 3.2, the user can also select the “ANS 18.1 Version–1984” for the reactor coolant source term, radionuclide concentrations in the primary and secondary coolant, as shown in Figure 3-4. The reactor coolant source term values for the “ANS 18.1 Version–1984” option correspond to the values in ANSI/ANS-18.1-1984 and are also consistent with the guidance in DC/COL-ISG-05; RG 1.112, Revision 1; and NUREG-0800. Table 4-48 lists the expected radionuclide concentrations in the reactor coolant and steam for BWRs from ANSI/ANS-18.1-1984. Figure 4-4 shows the relationship of the design parameters.

### Bases

The values in Table 4-48 provide a set of typical radionuclide concentrations in the reactor coolant and steam for reactor designs within the parameters specified in Table 4-43. The reactor designs within the parameters specified in Table 4-43 are the same for ANSI/ANS-18.1-1999 and ANSI/ANS-18.1-1984. The values in Table 4-48 are those determined to be representative of radionuclide concentrations in a BWR over its lifetime, based on the ANSI/ANS-18.1-1984 data and models. It is recognized that some systems will have design parameters that are outside the ranges specified in Table 4-43. For that reason, the concentrations are adjusted to the actual design parameters by a method like that used for the adjustments discussed in Section 4.2.1.2, except for using the ANSI/ANS-18.1-1984 values in Tables 4-49 and 4-50.

**Table 4-48 Radionuclide concentrations in BWR coolant and main steam – ANSI/ANS-18.1-1984**

Radionuclide <sup>a</sup>	Reactor Water <sup>b</sup> ( $\mu\text{Ci/g}$ )	Reactor Steam <sup>c</sup> ( $\mu\text{Ci/g}$ )
<b>Noble Gases</b>		
Kr-83m		5.9E-04
Kr-85m		1.0E-03
Kr-85		4.0E-06
Kr-87		3.3E-03
Kr-88		3.3E-03
Kr-89		2.1E-02
Xe-131m		3.3E-06
Xe-133m		4.9E-05
Xe-133		1.4E-03
Xe-135m		4.4E-03
Xe-135		3.8E-03
Xe-137		2.6E-02
Xe-138		1.5E-02
<b>Halogens</b>		
I-131	2.2E-03	3.3E-05
I-132	2.2E-02	3.3E-04
I-133	1.5E-02	2.3E-04
I-134	4.3E-02	6.5E-04
I-135	2.2E-02	3.3E-04
<b>Cesium &amp; Rubidium</b>		
Rb-89	5.0E-03	5.0E-06
Cs-134	3.0E-05	3.0E-08
Cs-136	2.0E-05	2.0E-08
Cs-137, Ba-137m <sup>d</sup>	8.0E-05	8.0E-08
Cs-138	1.0E-02	1.0E-05
<b>Water Activation Products</b>		
N-16	6.0E+01	5.0E+01
<b>Tritium</b>		
H-3	1.0E-02	1.0E-02
<b>Other Radionuclides</b>		
Na-24	1.0E-02	1.0E-05
P-32	2.0E-04	2.0E-07
Cr-51	6.0E-03	6.0E-06

<sup>a</sup> These concentrations are from Table 5 of ANSI/ANS-18.1-1984.

<sup>b</sup> The reactor water concentration is specified at the nozzle where reactor water leaves the reactor vessel.

<sup>c</sup> The reactor steam concentration is specified at time = 0, and only the noble gases are used in GALE-BWR 3.2. The other isotopes are not calculated nor printed to the output for reactor steam.

<sup>d</sup> These nuclides are in secular equilibrium; other radionuclide concentrations given are those of the parent.

**Table 4-48 Radionuclide concentrations in BWR coolant and main steam – ANSI/ANS-18.1-1984 (cont.)**

Radionuclide <sup>a</sup>	Reactor Water <sup>b</sup> ( $\mu\text{Ci/g}$ )	Reactor Steam <sup>c</sup> ( $\mu\text{Ci/g}$ )
Mn-54	7.0E-05	7.0E-08
Mn-56	5.0E-02	5.0E-05
Fe-55	1.0E-03	1.0E-06
Fe-59	3.0E-05	3.0E-08
Co-58	2.0E-04	2.0E-07
Co-60	4.0E-04	4.0E-07
Ni-63	1.0E-06	1.0E-09
Cu-64	3.0E-02	3.0E-05
Zn-65	2.0E-04	2.0E-07
Sr-89	1.0E-04	1.0E-07
Sr-90, Y-90 <sup>d</sup>	7.0E-06	7.0E-09
Sr-91	4.0E-03	4.0E-06
Sr-92	1.0E-02	1.0E-05
Y-91	4.0E-05	4.0E-08
Y-92	6.0E-03	6.0E-06
Y-93	4.0E-03	4.0E-06
Zr-95, Nb-95 <sup>d</sup>	8.0E-06	8.0E-09
Mo-99, Tc-99m <sup>d</sup>	2.0E-03	2.0E-06
Ru-103, Ru-103m <sup>d</sup>	2.0E-05	2.0E-08
Ru-106, Rh-106 <sup>d</sup>	3.0E-06	3.0E-09
Ag-110m	1.0E-06	1.0E-09
Te-129m	4.0E-05	4.0E-08
Te-131m	1.0E-04	1.0E-07
Te-132	1.0E-05	1.0E-08
Ba-140, La-140 <sup>d</sup>	4.0E-04	4.0E-07
Ce-141	3.0E-05	3.0E-08
Ce-144, Pr-144 <sup>d</sup>	3.0E-06	3.0E-09
W-187	3.0E-04	3.0E-07
Np-239	8.0E-03	8.0E-06

<sup>a</sup> These concentrations are from Table 5 of ANSI/ANS-18.1-1984.

<sup>b</sup> The reactor water concentration is specified at the nozzle where reactor water leaves the reactor vessel.

<sup>c</sup> The reactor steam concentration is specified at time = 0, and only the noble gases are used in GALE-BWR 3.2. The other isotopes are not calculated nor printed to the output for reactor steam.

<sup>d</sup> These nuclides are in secular equilibrium; other radionuclide concentrations given are those of the parent.



**Table 4-49 Values used in determining adjustment factors for BWRs – ANSI/ANS-18.1-1984**

Symbol	Description	Element Class <sup>a</sup>					
		Noble Gases	Halogens	Cesium & Rubidium	Water Activation Products	H-3	Other Radionuclides
NA	Fraction of material removed in the reactor water cleanup system	0.0E+00	9.0E-01	5.0E-01	0.0E+00	0.0E+00	9.0E-01 <sup>b</sup>
NB	Fraction of material removed by the condensate demineralizer.	0.0E+00	9.0E-01	5.0E-01	0.0E+00 <sup>c</sup>	0.0E+00	9.0E-01
NS	Ratio of the concentration in reactor steam to the concentration in reactor water	d	1.5E-02 <sup>e</sup>	1.0E-03	c	1.0E+00	1.0E-03
R <sub>n</sub>	Removal rate from the reactor water (h <sup>-1</sup> ) <sup>f</sup>	d	8.6E-01	1.7E-01	c	<sup>g</sup>	3.4E-01 <sup>b</sup>

<sup>a</sup> These values are from Table 8 of ANSI/ANS-18.1-1984.

<sup>b</sup> These represent effective removal terms and include other mechanisms, such as plateout. Plateout would be applicable to radionuclides such as molybdenum and corrosion products.

<sup>c</sup> Water activation products in reactor coolant exhibit varying chemical and physical properties that are not well defined. However, most are stripped off as gases, which are not effectively removed by the demineralizers of the systems; their concentrations are controlled by decay.

<sup>d</sup> Noble gases released from the core are rapidly transported 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.

<sup>e</sup> The value of 1.5E-02 is used for BWRs that have deep-bed condensate treatment. A value of 1.5E-02 is also used for BWRs with Powdex Condensate Treatment and stainless steel condenser tubing. For BWRs that have Powdex Condensate Treatment systems and copper condenser tubing, a value of 4.0E-03 should be used.

<sup>f</sup> These values of R<sub>n</sub> apply to the reference BWRs whose parameters are given in Table 4-43 and have been used in developing Table 4-45. For BWRs not included in Table 4-43, the appropriate value for R<sub>n</sub> may be determined by the following equation:

$$R_n = \frac{(FA)(NA) + (NC)(FS)(NS)(NB)}{WP} \text{ For halogens, CS, Rb, and other nuclides}$$

where the symbols are defined in this table, Table 4-43, and Figure 4-4. The values for R<sub>n</sub> for noble gases and water activation products are not used in the adjustment factors of Table 4-45.

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

**Table 4-50 Adjustment factors for BWRs – ANSI/ANS-18.1-1984**

Element Class	Reactor Coolant <sup>a</sup>	Reactor Steam <sup>a</sup>
Noble gases <sup>b</sup>	1.0E+00	1.0E+00
Halogens	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{8.6E-01 + \lambda}{R_n + \lambda}\right)$	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{8.6E-01 + \lambda}{R_n + \lambda}\right)$
Cesium & Rubidium	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{1.7E-01 + \lambda}{R_n + \lambda}\right)$	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{1.7E-01 + \lambda}{R_n + \lambda}\right)$
Water activation products <sup>c</sup>	1.0E+00	1.0E+00
Tritium <sup>d</sup>	1.0E+00	1.0E+00
Other radionuclides	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{3.4E-01 + \lambda}{R_n + \lambda}\right)$	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{3.4E-01 + \lambda}{R_n + \lambda}\right)$

<sup>a</sup> These values are from Table 10 of ANSI/ANS-18.1-1984.

<sup>b</sup> This assumes that the ratio of power to steam flow is essentially the same for all BWRs.

<sup>c</sup> This assumes the ratio of coolant mass to power level is approximately constant.

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

### 4.2.3 ANSI/ANS-18.1-2016 Source Term Parameters

#### Parameter

As mentioned in Section 4.2, in GALE-BWR 3.2, the user can also select the “ANS-18.1 Version–2016” for the reactor coolant source term, radionuclide concentrations in the primary and secondary coolant, as shown in Figure 3-4. Table 4-51 lists the expected radionuclide concentrations in the reactor coolant and steam for BWRs from ANSI/ANS-18.1-2016. Figure 4-4 shows the relationship of the design parameters.

#### Bases

The values in Table 4-51 provide a set of typical radionuclide concentrations in the primary and secondary systems for reactor designs within the parameters specified in Table 4-43. The reactor designs within the parameters specified in Table 4-43 are the same for ANSI/ANS-18.1-1984 and ANSI/ANS-18.1-2016. The values in Table 4-51 are those determined to be representative of radionuclide concentrations in a BWR over its lifetime, based on the ANSI/ANS-18.1-2016 data and models. Some systems will have design parameters that are outside the ranges specified in Table 4-43. For that reason, the concentrations are adjusted to the actual design parameters by a method like that used for the adjustments discussed in Section 4.2.1.2, except for using the ANSI/ANS-18.1-2016 values provided in Tables 4-52 and 4-53.

**Table 4-51 Radionuclide concentrations in BWR coolant and main steam – ANSI/ANS-18.1-2016**

Radionuclide <sup>a</sup>	Reactor Water <sup>b</sup> ( $\mu\text{Ci/g}$ )	Reactor Steam <sup>c</sup> ( $\mu\text{Ci/g}$ )
<b>Noble Gases</b>		
Kr-83m		5.9E-04
Kr-85m		1.6E-05
Kr-85		1.3E-04
Kr-87		1.1E-04
Kr-88		6.2E-05
Kr-89		9.3E-03
Xe-131m		3.3E-06
Xe-133m		2.0E-06
Xe-133		3.0E-05
Xe-135m		5.4E-04
Xe-135		3.0E-04
Xe-137		9.9E-04
Xe-138		2.8E-03
<b>Halogens</b>		
I-131	8.3E-05	1.7E-06
I-132	1.2E-03	2.3E-05
I-133	6.5E-04	1.3E-05
I-134	4.5E-03	9.0E-05
I-135	1.4E-03	2.7E-05
<b>Cesium &amp; Rubidium</b>		
Rb-89	5.0E-03	5.0E-06
Cs-134	3.0E-05	3.0E-08
Cs-136	2.4E-05	2.4E-08
Cs-137, Ba-137m <sup>d</sup>	4.6E-05	4.6E-08
Cs-138	5.2E-03	5.2E-06
<b>Water Activation Products</b>		
N-16	6.0E+01	5.0E+01
<b>Tritium</b>		
H-3	1.4E-02	1.4E-02
<b>Other Radionuclides</b>		
Na-24	1.2E-03	1.2E-06
P-32	4.0E-05	4.0E-08
Cr-51	9.7E-04	9.7E-07

<sup>a</sup> These concentrations are from Table 5 of ANSI/ANS-18.1-2016.

<sup>b</sup> The reactor water concentration is specified at the nozzle where reactor water leaves the reactor vessel.

<sup>c</sup> The reactor steam concentration is specified at time = 0, and only the noble gases are used in GALE-BWR 3.2. The other isotopes are not calculated nor printed to the output for reactor steam.

<sup>d</sup> These nuclides are in secular equilibrium; other radionuclide concentrations given are those of the parent.

**Table 4-51 Radionuclide concentrations in BWR coolant and main steam – ANSI/ANS-18.1-2016 (cont.)**

Radionuclide <sup>a</sup>	Reactor Water <sup>b</sup> ( $\mu\text{Ci/g}$ )	Reactor Steam <sup>c</sup> ( $\mu\text{Ci/g}$ )
Mn-54	4.8E-04	4.8E-07
Mn-56	3.3E-03	3.3E-07
Fe-55	1.0E-03	1.0E-06
Fe-59	2.6E-04	2.6E-07
Co-58	2.3E-04	2.3E-07
Co-60	4.4E-04	4.4E-07
Ni-63	1.0E-06	1.0E-09
Cu-64	5.9E-03	5.9E-06
Zn-65	2.1E-04	2.1E-07
Sr-89	3.0E-06	3.0E-09
Sr-90, Y-90 <sup>d</sup>	1.5E-07	1.5E-10
Sr-91	2.4E-03	2.4E-06
Sr-92	7.0E-03	7.0E-06
Y-91	4.0E-05	4.0E-08
Y-92	2.0E-03	2.0E-06
Y-93	1.7E-04	1.7E-07
Zr-95, Nb-95 <sup>d</sup>	8.3E-05	8.3E-08
Mo-99, Tc-99m <sup>d</sup>	4.1E-04	4.1E-07
Ru-103, Ru-103m <sup>d</sup>	2.0E-05	2.0E-08
Ru-106, Rh-106 <sup>d</sup>	3.0E-06	3.0E-09
Ag-110m	1.0E-06	1.0E-09
Te-129m	4.0E-05	4.0E-08
Te-131m	5.8E-05	5.8E-08
Te-132	1.0E-05	1.0E-08
Ba-140, La-140 <sup>d</sup>	4.2E-04	4.2E-07
Ce-141	2.0E-05	2.0E-08
Ce-144, Pr-144 <sup>d</sup>	3.0E-06	3.0E-09
W-187	2.6E-04	2.6E-07
Np-239	3.4E-04	3.4E-07

<sup>a</sup> These concentrations are from Table 5 of ANSI/ANS-18.1-2016.

<sup>b</sup> The reactor water concentration is specified at the nozzle where reactor water leaves the reactor vessel.

<sup>c</sup> The reactor steam concentration is specified at time = 0, and only the noble gases are used in GALE-BWR 3.2. The other isotopes are not calculated nor printed to the output for reactor steam.

<sup>d</sup> These nuclides are in secular equilibrium; other radionuclide concentrations given are those of the parent.

**Table 4-52 Values used in determining adjustment factors for BWRs – ANSI/ANS-18.1-2016**

Symbol <sup>a</sup>	Description	Element Class					
		Noble Gases	Halogens	Cesium & Rubidium	Water Activation Products	H-3	Other Radionuclides
NA	Fraction of material removed in the reactor water cleanup system	0.0E+00	9.0E-01	5.0E-01	b	0.0E+00	9.0E-01 <sup>c</sup>
NB	Fraction of material removed by the condensate demineralizer	0.0E+00	9.0E-01	5.0E-01	b	0.0E+00	9.0E-01
NS	Ratio of the concentration in reactor steam to the concentration in reactor water	d	2.0E-02 <sup>e</sup>	1.0E-03	b	1.0E+00	1.0E-03
R <sub>n</sub>	Removal rate from the reactor water (h <sup>-1</sup> ) <sup>f</sup>	d	1.0E+01	1.7E-01	b	g	3.4E-01 <sup>c</sup>

<sup>a</sup> These values are from Table 8 of ANSI/ANS-18.1-2016.

<sup>b</sup> In reactor coolant, water activation products exhibit varying chemical and physical properties that are not well defined. However, most are stripped off as gases, which are not effectively removed by the demineralizers of the systems; their concentrations are controlled by decay. The activity concentrations in reactor water and steam are subject to change when hydrogen water chemistry is employed (see Table 4-53 for adjustment factors).

<sup>c</sup> These represent effective removal terms and include other mechanisms, such as plateout. Plateout would be applicable to radionuclides such as molybdenum and corrosion products.

<sup>d</sup> Noble gases released from the core are rapidly transported 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.

<sup>e</sup> The value of 2.0E-02 is used for BWRs that have deep-bed condensate treatment. A value of 1.5E-02 is also used for BWRs. For BWRs that have Powdex Condensate Treatment systems and copper condenser tubing, a value of 5.0E-03 should be used.

<sup>f</sup> These values of R<sub>n</sub> apply to the reference BWRs whose parameters are given in Table 4-43 and have been used in developing Table 4-53. For BWRs not included in Table 4-43, the appropriate value for R<sub>n</sub> may be determined by the following equation:

$$R_n = \frac{(FA)(NA) + (NC)(FS)(NS)(NB)}{WP} \text{ for halogens, CS, Rb, and other nuclides}$$

where the symbols are defined in this table, Table 4-43 and 4-52, and Figure 4-4. The values for R<sub>n</sub> for noble gases and water activation products are not used in the adjustment factors of Table 4-53.

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

**Table 4-53 Adjustment factors for BWRs – ANSI/ANS-18.1-2016**

Element Class	Reactor Coolant <sup>a</sup>	Reactor Steam <sup>a</sup>
Noble gases <sup>b</sup>	1.0E+00	1.0E+00
Halogens	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{1.0E+00 + \lambda}{R_n + \lambda}\right)$	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{1.0E+00 + \lambda}{R_n + \lambda}\right)$
Cesium & Rubidium	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{1.7E-01 + \lambda}{R_n + \lambda}\right)$	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{1.7E-01 + \lambda}{R_n + \lambda}\right)$
Water activation products <sup>c</sup>	1.0E+00	1.0E+00
Tritium <sup>d</sup>	1.0E+00	1.0E+00
Other radionuclides	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{3.4E-01 + \lambda}{R_n + \lambda}\right)$	$\left(\frac{P}{WP}\right)\left(1.1E+02\frac{\text{lb}}{\text{MWt}}\right)\left(\frac{3.4E-01 + \lambda}{R_n + \lambda}\right)$
Zn-65 <sup>e</sup>	1.0E+01	1.0E+01

<sup>a</sup> These values are from Table 10 of ANSI/ANS-18.1-2016.

<sup>b</sup> This assumes that the ratio of power to steam flow is essentially the same for all BWRs.

<sup>c</sup> This assumes the ratio of coolant mass to power level is approximately constant.

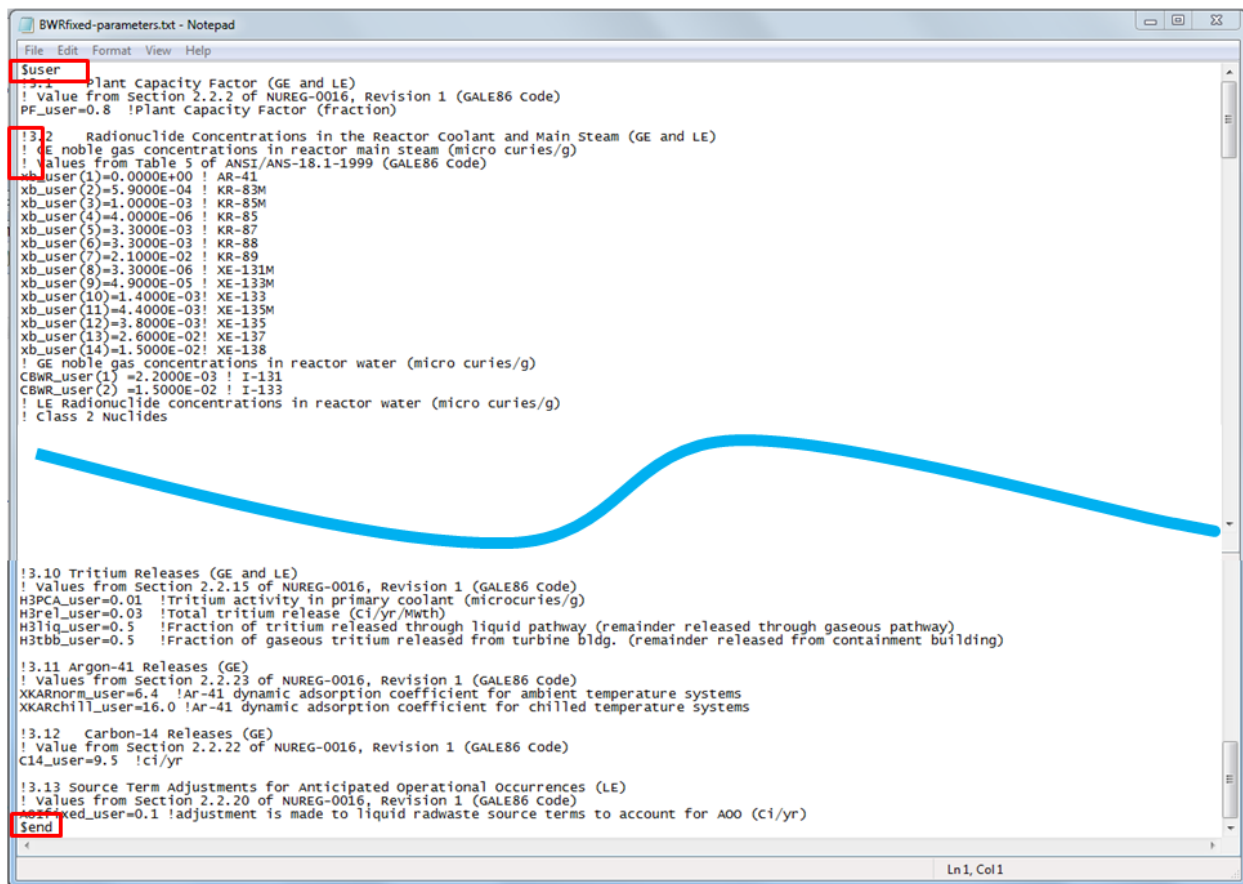
<sup>d</sup> The tritium concentrations in the reactor water and steam are expected to be equal. They are controlled by loss of water from the main coolant system by evaporation or leakage.

<sup>e</sup> Adjustment factors are for zinc addition plants using natural zinc. Use of depleted zinc would result in a lower adjustment factor, and the decrease is a function of the reduction of zinc-64.

### 4.3 BWRfixed-parameters.txt file

This section discusses the user option to modify the certain GALE-BWR fixed modeling parameters by means of the BWRfixed-parameters.txt file. The GALE-BWR 3.2 code zip file contains a sample BWRfixed-parameters.txt file (Figure 3-1), which allows the user to define 10 of the fixed modeling parameters from the default GALE86 code and ANSI/ANS-18.1-1999 values in the code.

Figure 4-5 shows the sample BWRfixed-parameters.txt file, which can be opened and edited in any text editor program (e.g., NotePad) by the user. The text file should start with “\$user” and end with “\$end” and comments can be included after any “!” Each line entry in the sample BWRfixed-parameters.txt file (Figure 4-5) contains a comment line, which indicates the reference source for the values used in the line entry in the sample BWRfixed-parameters.txt file.



```
BWRfixed-parameters.txt - Notepad
File Edit Format View Help
$user
!3.1 Plant Capacity Factor (GE and LE)
! Value from Section 2.2.2 of NUREG-0016, Revision 1 (GALE86 Code)
PF_user=0.8 !Plant Capacity Factor (fraction)
!3.2 Radionuclide Concentrations in the Reactor coolant and Main Steam (GE and LE)
! GE noble gas concentrations in reactor main steam (micro curies/g)
! Values from Table 5 of ANSI/ANS-18.1-1999 (GALE86 Code)
xb_user(1)=0.0000E+00 ! AR-41
xb_user(2)=5.9000E-04 ! KR-83M
xb_user(3)=1.0000E-03 ! KR-85M
xb_user(4)=4.0000E-06 ! KR-85
xb_user(5)=3.3000E-03 ! KR-87
xb_user(6)=3.3000E-03 ! KR-88
xb_user(7)=2.1000E-02 ! KR-89
xb_user(8)=3.3000E-06 ! XE-131M
xb_user(9)=4.9000E-05 ! XE-133M
xb_user(10)=1.4000E-03 ! XE-133
xb_user(11)=4.4000E-03 ! XE-135M
xb_user(12)=3.8000E-03 ! XE-135
xb_user(13)=2.6000E-02 ! XE-137
xb_user(14)=1.5000E-02 ! XE-138
! GE noble gas concentrations in reactor water (micro curies/g)
CBWR_user(1)=2.2000E-03 ! I-131
CBWR_user(2)=1.5000E-02 ! I-133
! LE Radionuclide concentrations in reactor water (micro curies/g)
! Class 2 Nuclides
!3.10 Tritium Releases (GE and LE)
! Values from Section 2.2.15 of NUREG-0016, Revision 1 (GALE86 Code)
H3PCA_user=0.01 !Tritium activity in primary coolant (microcuries/g)
H3rel_user=0.03 !Total tritium release (Ci/yr/Mwth)
H3liq_user=0.5 !Fraction of tritium released through liquid pathway (remainder released through gaseous pathway)
H3tbb_user=0.5 !Fraction of gaseous tritium released from turbine bldg. (remainder released from containment building)
!3.11 Argon-41 Releases (GE)
! Values from Section 2.2.23 of NUREG-0016, Revision 1 (GALE86 Code)
XKARNorm_user=6.4 !Ar-41 dynamic adsorption coefficient for ambient temperature systems
XKARChill_user=16.0 !Ar-41 dynamic adsorption coefficient for chilled temperature systems
!3.12 Carbon-14 Releases (GE)
! Value from Section 2.2.22 of NUREG-0016, Revision 1 (GALE86 Code)
C14_user=9.5 !Ci/yr
!3.13 Source Term Adjustments for Anticipated Operational Occurrences (LE)
! Values from Section 2.2.20 of NUREG-0016, Revision 1 (GALE86 Code)
BWRfixed_user=0.1 !adjustment is made to liquid radwaste source terms to account for AOO (Ci/yr)
$end
Ln1, Col1
```

Figure 4-5 BWRfixed-parameters.txt file

Table 4-54 lists the fixed modeling parameters that can be defined using the BWRfixed-parameters.txt file. Figure 4-6 shows an example of a modified BWRfixed-parameters.txt file.



**Table 4-54 BWRfixed-parameters.txt file modifiable parameters**

Code Line	Parameter	Location in GALE-BWR Code	Default Values <sup>a</sup>		
3.1	Plant Capacity Factor (fraction)	GE & LE	8.0E-01		
3.2	Radionuclide Concentration in Reactor Coolant and Main Steam (μCi/g)	GE	Noble gases in reactor main steam	See Table 4-42	
		GE	Noble gases in reactor water	See Table 4-42	
		LE	Halogens, cesium & rubidium, and other nuclides in reactor water	See Table 4-42	
3.3	Noble Gas, Radioiodine, and Particulate Releases from Building Ventilation Systems Prior to Treatment	GE	Noble gas releases (Ci/yr)		See Table 4-2
			Particulate releases (μCi/yr)		See Table 4-2
			Radioiodine releases (Ci/yr/μCi/g)	Reactor building	See Table 4-5
				Turbine building	See Table 4-4
				Radwaste building	See Table 4-6
3.4	Radioiodine Input Rate to Main Condenser Offgas System (Ci/yr)	GE	6.0E+00		
3.5	Main Condenser Vacuum Pump Release	GE	Xenon-133 (Ci/yr)		1.3E+03
			Xenon-135 (Ci/yr)		5.0E+02
			Iodine-131 normal operations (Ci/yr/μCi/g)		4.9E+02
			Iodine-131 shutdown (Ci/yr/μCi/g)		1.1E+03
3.7	Cryogenic Distillation System	GE	PC for xenon and iodine		1.0E-04
			PC for krypton		2.5E-04
			Holdup time (d)		9.0E+01
3.9	Annual Releases in Untreated Detergent Waste (Ci/yr)	LE	See Table 4-30		
<sup>a</sup> The default values are from Section 4.1.1 of NUREG 0016, Revision 1 and Section 4.2.1 of PNNL-24250.					

**Table 4-54 BWRfixed-parameters.txt file modifiable parameters (cont.)**

Code Line	Parameter	Location in GALE-BWR Code	Default Values <sup>a</sup>	
3.10	Tritium Releases	GE & LE	Tritium activity in primary coolant ( $\mu\text{Ci/g}$ )	1.0E-02
			Total tritium release (Ci/yr/MWt)	3.0E-02
			Maximum fraction of tritium released through liquid pathway	5.0E-01
			Fraction of gaseous tritium released from turbine building	5.0E-01
3.11	Argon-41 Release (Ci/yr)	GE	Dynamic adsorption coefficient for ambient temperature systems	6.4E+00
			Dynamic adsorption coefficient for chilled temperature systems	1.6E+01
3.12	Carbon-14 Releases (Ci/yr)	GE	9.5E+00	
3.13	Source Term Adjustments for AOOs (Ci/yr)	LE	1.0E-01	

<sup>a</sup> The default values are from Section 4.1.1 of NUREG 0016, Revision 1 and Section 4.2.1 of PNNL-24250.

```

$user
! Plant Capacity Factor (GE and LE)
! Value from Section 2.2.2 of NUREG-0016, Revision 1 (GALE86 Code)
PF_user=0.9 Plant Capacity Factor (fraction)

!3.3 Noble Gas, Radioiodine, and Particulate Releases from Building Ventilation Systems Prior to Treatment (GE)
!Radioiodine release from Ventilation Systems Prior to Treatment (in Ci/yr/microcurie/g)
!(Assuming 90% capacity factor, will be adjusted based on actual capacity factor)
! Values from Section 2.2.4 (Tables 2-8 through 2-10) of NUREG-0016, Revision 1 (GALE86 Code)
! Reactor Building Turbine Building Radwaste Building
RN_user(1)=1.3800E+01, RNT_user(1)=3.8000E+03, RNR_user(1)=4.6000E+00 !I-131 during operation
RN_user(2)=1.3800E+01, RNT_user(2)=3.8000E+03, RNR_user(2)=4.6000E+00 !I-133 during operation
RNS_user(1)=2.6000E+00, RNTS_user(1)=4.1000E+02, RNRS_user(1)=1.4000E+00 !I-131 during shutdown
RNS_user(2)=2.6000E+00, RNTS_user(2)=4.1000E+02, RNRS_user(2)=1.4000E+00 !I-133 during shutdown

!3.5 Main Condenser Vacuum Pump Release (GE)
! Values from Section 2.2.7 and Table 2-26 of NUREG-0016, Revision 1 (GALE86 Code)
RNMVP_user(1)=550.0 Ci/yr/microcurie/g I-131(normal operations assuming 80% capacity, will be adjusted for actual capacity factor)
RNMVP_user(2)=550.0 Ci/yr/microcurie/g I-133(normal operations assuming 80% capacity, will be adjusted for actual capacity factor)
RNMVPS_user(1)=550.0 Ci/yr/microcurie/g I-131(shutdown assuming 80% capacity, will be adjusted for actual capacity factor)
RNMVPS_user(2)=550.0 Ci/yr/microcurie/g I-133(shutdown assuming 80% capacity, will be adjusted for actual capacity factor)

!3.13 Source Term Adjustments for Anticipated Operational Occurrences (LE)
! Values from Section 2.2.20 of NUREG-0016, Revision 1 (GALE86 Code)
AOIfixed_user=0.014 adjustment is made to liquid radwaste source terms to account for AOO (Ci/yr)

$end
    
```

**Figure 4-6 Example of a modified BWRfixed-parameters.txt file**

## 5.0 REFERENCES

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**BIBLIOGRAPHIC DATA SHEET**

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John Tomon, NRC Project Manager

11. ABSTRACT (200 words or less)

This report is documentation for the release of the GALE-BWR 3.2 (Gaseous and Liquid Effluents—Boiling-Water Reactor) code. The GALE-BWR 3.2 code is a computerized mathematical model for calculating the releases of radioactive material in gaseous and liquid effluents (i.e., the gaseous and liquid source terms) from nuclear power plants (NPPs). The code is a tool that can be used to determine conformance with the requirements of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

The purpose of the release of this version of the GALE-BWR code is to comprehensively verify applicability of the current methodology described in NUREG 0016, Revision 1, (ADAMS Accession No. ML091910213), "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR GALE Code)," issued January 1979, to both the current U.S. BWR facilities and proposed NPP designs. Additionally, the GALE-BWR 3.2 code includes a graphical user interface (GUI) to facilitate easier use of the code.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

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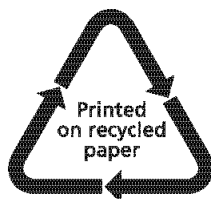
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