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Prepared for the U.S. Department of Energy
under Contract DE-AC05-76RL01830

Sensitivity Analysis of Hardwired Parameters in GALE Codes

KJ Geelhood
MR Mitchell
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December 2008

Pacific Northwest
NATIONAL LABORATORY

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Summary

The U.S. Nuclear Regulatory Commission asked Pacific Northwest National Laboratory to provide a data-gathering plan for updating the hardwired data tables and parameters of the Gaseous and Liquid Effluents (GALE) codes to reflect current nuclear reactor performance. This would enable the GALE codes to make more accurate predictions about the normal radioactive release source term applicable to currently operating reactors and to the cohort of reactors planned for construction in the next few years. A sensitivity analysis was conducted to define the importance of hardwired parameters in terms of each parameter's effect on the emission rate of the nuclides that are most important in computing potential exposures. The results of this study were used to compile a list of parameters that should be updated based on the sensitivity of these parameters to outputs of interest.

Section 1 of this report provides an introduction. Section 2 contains the results of the perturbation-based sensitivity analysis for the liquid-effluent and gaseous-effluent portions of the pressurized water reactor GALE codes. Section 3 contains the sensitivity analysis for the liquid-effluent and gaseous-effluent portions of the boiling water reactor GALE codes. Section 4 provides the conclusions of the study.

Acronyms

AOO	anticipated operational occurrence
BWR	boiling water reactor
DF	decontamination factor
GALE	Gaseous and Liquid Effluents
HEPA	high-efficiency particulate air (filter)
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PNNL	Pacific Northwest National Laboratory
PWR	pressurized water reactor
PWRGE	pressurized water reactor gaseous effluent
PWRLE	pressurized water reactor liquid effluent
REIRS	Radiation Exposure Information and Reporting System
XP1	primary coolant
XP2	secondary coolant

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Tables

1.0 Introduction

The Gaseous and Liquid Effluents (GALE) series of codes consists of four codes that calculate the gaseous and liquid effluent releases from pressurized water reactors (PWRs) and boiling water reactors (BWRs). The GALE computer code uses a combination of input data and hardwired parameters to calculate the source term of radionuclides generated by a nuclear power plant during routine operation. Parameters that vary from plant to plant are treated as "inputs"; GALE asks the operator for input values on each run. Hardwired parameters are plant characteristics that are assumed to be the same for all reactors.

The hardwired data tables and parameters were based on reactor surveys, reports, and standards dating from 1970–1999. Many improvements to reactor operations, fuel design, and fuel handling have been made in the last 10 to 40 years. The plant capacity factor is a good example of a hardwired parameter that has changed significantly since the last update of the GALE code—then an 80% capacity factor was typical, and now 90% is closer to the national average. At the time that the GALE codes were last updated, these parameters were considered to be typical for current reactor operation.

To make the GALE predictions applicable to modern reactor operation, the U.S. Nuclear Regulatory Commission (NRC) has requested that Pacific Northwest National Laboratory (PNNL) update the hardwired data tables and parameters to reflect current reactor performance. A sensitivity analysis was conducted to define the importance of hardwired parameters in terms of each parameter's effect on the emission rate of the nuclides that are most important in computing potential exposures.

Each GALE code outputs the release rate (curies per year) for a large number of different nuclides. A few of these nuclides dominate the potential exposures of the public. To prioritize the updates to the hardwired parameters, PNNL requested that NRC provide a list for each of the four codes containing nuclides of greatest interest from an exposure standpoint. NRC staff provided a list of nuclides for each code that are of greatest importance for defining potential exposures. PNNL then performed a parameter perturbation-based sensitivity analysis on each of the identified hardwired parameters to determine the relative importance of each of the parameters. The standard input deck provided for each GALE code was used to perform this sensitivity study. Although the results of the sensitivity study may be slightly different for different inputs, the results using the standard input decks are fairly representative of the overall code behavior.

The results of this study were used to compile a list of parameters that should be updated based on the sensitivity of these parameters to outputs of interest. This list will be used to create a data-gathering plan to update the GALE codes.

2.0 PWR GALE

This section contains the results of the perturbation-based sensitivity analysis for the following PWR GALE codes: PWR liquid effluents (PWRLE) and PWR gaseous effluents (PWRGE). Section 2.1 lists the parameters identified as hardwired data within the PWR GALE codes. These parameters are identified as those that are used in the PWRGE code, those that are used in the PWRLE code, and those that are provided to the codes through input. Section 2.2 contains the results of the sensitivity analysis performed on PWRGE where the relevant parameters identified in Section 2.1 were varied to determine the effects of changing these parameters on the outputs of interest. Section 2.3 contains the results of the sensitivity analysis performed on PWRLE where the relevant parameters identified in Section 2.1 were varied to determine the effects of changing these parameters on the outputs of interest. A sensitivity analysis was not performed for those parameters that were identified as input values as they are not intrinsic to the codes. Section 2.4 contains a table showing what parameters in the PWR GALE codes should be updated. This table also lists the current basis and vintage of each parameter and the location of new information that could be used to update each parameter.

2.1 Parameters Identified

Table 2.1 shows a list of parameters that have been identifies as plant-operation data needs for the PWR GALE codes. There are two PWR GALE codes. PWRGE calculates the gaseous effluents from a PWR and PWRLE calculates the liquid effluents for a PWR. Not all of these parameters are applicable to both codes, so Table 2.1 identifies which code each parameter applies to. In addition, some of these parameters are not hardwired into the code, but rather are controlled though input variables. These parameters are identified in Table 2.1 and were not included in the sensitivity analysis because they do not need to be updated in the code.

Table 2.1. Parameters Identified as Plant Operation Data Needs for PWR GALE Code

Parameter	Applicability		Comment	Section for Sensitivity Analysis
	PWRGE	PWRLE		
Plant capacity	Yes	Yes		2.2.1, 2.3.1
Nuclides in primary and secondary coolant	Yes	Yes	Secondary coolant concentrations calculated assuming primary to secondary leakage of 70 lbs/day	2.2.2, 2.3.2
Primary to secondary coolant leakage	NA	NA	Covered under previous item	
Radionuclide releases from various ventilation systems before treatment	Yes	Not used		2.2.3
Iodine release from main condenser exhaust	Yes	Not used		2.2.4

Table 2.1 (Contd)

Parameter	Applicability		Comment	Section for Sensitivity Analysis
	PWRGE	PWRLE		
Duration of containment air cleanup	Yes	Not used		2.2.5
% of containment air treated	Yes	Not used		2.2.5
Containment purge frequency	Yes	Not used		2.2.5
Radionuclide removal efficiencies for charcoal absorbers and high-efficiency particulate air (HEPA) filters	Input	Not Used	Input on Cards 35, 36, 38, 39, and 40	
Dynamic adsorption coefficients for charcoal	Input	Not Used	Holdup times are input to the code and are a function of mass of charcoal, dynamic adsorption, and system flow rate Cards 31 and 32	
Flow rates into liquid radwaste systems	Not Used	Input	Provided as inputs Cards 12-29	
Radionuclides in untreated detergent waste	Not Used	Yes		2.3.3
Demineralizer regeneration cycle	Not Used	Input	Condensate demineralizer regeneration time is input to the code Card 10	
Tritium releases	Yes	Yes		2.2.6
Decontamination factors for demineralizers, evaporators, filters, reverse osmosis units	Not Used	Yes		2.3.5
Frequency and extent of unplanned liquid releases	Not Used	Yes		2.3.6
Carbon-14 release	Yes	Not Used		2.2.7
Argon-41 release	Yes	Not Used		2.2.8

2.2 PWRGE Sensitivity

The nuclides that have the greatest impact on PWR gaseous effluents dose are H-3, C-14, Cs-137, Sr-90, Co-60, Kr-88, Xe-133, and I-131. The following sensitivity studies will identify the impact of changes to hardwired parameters on these nuclides.

Table 2.2 shows the nominal values of nuclide release rates for the current, standard GALE case.

Table 2.2. Nuclide Release Rates from PWRGE for Standard Case (curies/year)

H-3	C-14	Cs-137	Sr-90	Co-60	Kr-88	Xe-133	I-131
1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}

2.2.1 Plant Capacity Factor

A plant capacity factor of 80% is assumed in GALE. The modern operation of plants has allowed for significantly higher capacity factors. If a plant capacity factor of 100% is used, the concentrations of the selected nuclides do not change except for Xe-133, which increases to 65 Ci. This demonstrates that the Xe-133 release rate scales directly with the plant capacity factor. The results of this study are shown in Table 2.3.

Table 2.3. Sensitivity to Increasing the Plant Capacity Factor

		Release Rate, Ci/year							
Bias	**Value**	**H-3**	**C-14**	**Cs-137**	**Sr-90**	**Co-60**	**Kr-88**	**Xe-133**	**I-131**
Nominal	0.8	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
Upper	1.0	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	65	8.3×10^{-4}

2.2.2 Nuclides in Primary and Secondary Coolant

Concentrations of various nuclides in the primary and secondary coolant are hardwired into the PWRGE code. To investigate sensitivity, all nuclide concentrations were reduced by a factor of 10. It is not expected that the nuclide concentrations will be increased because of increased fuel reliability. This had the result of decreasing the release rate of Kr-88 and Xe-133 by the same factor of 10. This demonstrates that the concentrations of Kr-88 and Xe-133 in the primary coolant are directly proportional to the Kr-88 and Xe-133 release rates. The concentrations in the secondary coolant were also reduced by a factor of 10. This had no impact on the release rate for the selected nuclides. This demonstrates that the concentration of Kr-88 and Xe-133 in the secondary coolant does not impact the release rates for the selected nuclides. The results of this study are shown in Table 2.4.

Table 2.4. Sensitivity to Decreasing Concentration of Various Nuclides in the Primary Coolant (XP1) and the Secondary Coolant (XP2)

		Release Rate, Ci/year							
Bias	**Value**	**H-3**	**C-14**	**Cs-137**	**Sr-90**	**Co-60**	**Kr-88**	**Xe-133**	**I-131**
Nominal		1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
Primary down	XP1/10	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	0.0	5.0	$< 1\times10^{-4}$
Secondary down	XP2/10	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}

2.2.3 Radionuclide Releases from Various Ventilation Systems Before Treatment

Release rates from the ventilation systems of various buildings are hardwired into the PWRGE code. For noble gas, it is assumed that 3% of the primary coolant concentration is released per day. Release from the auxiliary building is based on 160 lb/day primary coolant leakage. Release from the turbine building is based on 1700 lb/hr steam leakage. Changing the fraction released was found to have a major impact on the release rate of Kr-88 and Xe-133. Changing the other parameters had no impact on the release rate of the selected nuclides. This is shown in Table 2.5.

For the iodine and the particulates, hardwired data statements within the code are used to input the release rates from various buildings. These rates are found to have a major impact on the release rates for Cs-137, Sr-90, Co-60, and I-131. The results of these sensitivity studies are shown in Table 2.6 and Table 2.7 for iodine and particulates, respectively.

Table 2.5. Sensitivity to Decreasing Noble Gas Release Parameters from Building Ventilation Systems

		Release Rate, Ci/year							
Bias	**Value**	**H-3**	**C-14**	**Cs-137**	**Sr-90**	**Co-60**	**Kr-88**	**Xe-133**	**I-131**
Nominal	3% per day Aux 160 lb/day Steam 1700 lb/hr	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
	1% per day	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	1.0	17	8.0×10^{-4}
	Aux 50 per day	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
	Steam 500 lb/hr	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}

Table 2.6. Sensitivity to Decreasing Iodine Release Parameters from Building Ventilation Systems

	Release Rate, Ci/year							
Value	**H-3**	**C-14**	**Cs-137**	**Sr-90**	**Co-60**	**Kr-88**	**Xe-133**	**I-131**
Nominal	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
Containment/10	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	1.0	17	7.5×10^{-4}
Auxiliary/10	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	6.7×10^{-4}
Turbine/10	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
Aux(outage)/10	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	3.5×10^{-4}
Turbine(outage)/10	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}

Table 2.7. Sensitivity to Decreasing Particulate Release Parameters from Building Ventilation Systems

	Release Rate, Ci/year							
Value	**H-3**	**C-14**	**Cs-137**	**Sr-90**	**Co-60**	**Kr-88**	**Xe-133**	**I-131**
Nominal	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
Containment/10	1100	7.3	4.0×10^{-5}	1.6×10^{-5}	9.0×10^{-5}	4.0	52	8.0×10^{-4}
Auxiliary/10	1100	7.3	8.3×10^{-5}	6.0×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
Fuel Pool/10	1100	7.3	6.6×10^{-5}	5.6×10^{-5}	3.9×10^{-5}	4.0	52	8.0×10^{-4}
Waste Gas System/10	1100	7.3	8.9×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}

2.2.4 Iodine Release from Main Condenser Exhaust

The iodine release from the main condenser exhaust is hardwired in PWRGE at 1.7×10^{-3} Ci/yr per μCi/g in the primary coolant. Biasing this value up and down by a factor of 10 had no impact on the release rates for the selected nuclides. This parameter only impacts the release rate of I-131 and I-133, and for the sample case, the fraction of I-131 released from the main condenser exhaust is not a significant fraction of the total I-131 released.

2.2.5 Duration of Containment Air Cleanup, Percent of Containment Air Treated, and Containment Purge Frequency

The containment air cleanup system is modeled using several parameters. PWRGE assumes that the containment building is purged two times each year. Additional purges can be specified in the input file. PWRGE assumes that the cleanup system will operate for 16 hours before purging, and will provide a decontamination factor (DF) of 100 for iodine. Finally, PWRGE assumes a mixing efficiency of 70%.

Table 2.8 shows the results of biasing these parameters. Increasing the number of purges to five times per year has no impact on the outputs of interest. Decreasing the number of purges to one time per year slightly increased the I-131 concentration (~1%). Changing the operation time before purging between 8 and 24 hours had no impact on the outputs of interest. Changing the mixing efficiency between 50% and 100% has no impact on the outputs of interest. Changing the DF from 10 to 1000 has no impact on the outputs of interest.

Table 2.8. Sensitivity to Changing Parameters Related to the Air Cleanup System

Bias	Value	Release Rate, Ci/year							
		H-3	C-14	Cs-137	Sr-90	Co-60	Kr-88	Xe-133	I-131
Nominal	2 purges/year 16 hours DF=100 Mix=70%	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
Number of purges	5 purges/year	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
	1 purge/year	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.1×10^{-4}
Operation time prior to purge	24 hours	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
	8 hours	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
Mixing efficiency	Mix=100%	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
	Mix=50%	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
Decontamination factor	DF=10	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
	DF=1000	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}

2.2.6 Tritium Releases

The total tritium release in PWRGE (and PWRLE) is hardwired to 0.4 Ci/yr/MW. Of this amount, up to 90% of the total tritium released is released through liquid pathways at a rate of 1.0 μCi/mL of liquid waste released. PWRLE allows up to 90% of the total tritium to be released in liquids. If enough liquid waste is entered as an input to account for more than 90% of the total tritium release, then PWRLE caps the liquid release fraction at 90% and assigns the remaining 10% of the tritium to PWRGE. Table 2.9 shows the results of biasing the assumed tritium release up and down. Biasing the tritium release only affects the H-3 release rate. This also shows that when the tritium release is biased up, a greater fraction of the tritium is released via gaseous pathways.

Table 2.9. Sensitivity to Changing the Tritium Release

Bias	Value	Release Rate, Ci/year							
		H-3	C-14	Cs-137	Sr-90	Co-60	Kr-88	Xe-133	I-131
Nominal	0.4 Ci/yr/MW	1100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
Upper	1.0 Ci/yr/MW	3100	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}
Lower	0.2 Ci/yr/MW	400	7.3	9.0×10^{-5}	6.3×10^{-5}	1.1×10^{-4}	4.0	52	8.0×10^{-4}

2.2.7 Carbon-14 Release

The carbon-14 release rate is fixed in the code as 7.3 Ci/year. This value was arrived at based on surveys of 10 reactors between 1975 and 1975, so it can be considered empirical. Carbon-14 release rate does not impact the release rate of any other nuclides. PWRGE does not do any calculations to arrive at the C-14 release rate, but simply uses a hardwired value for C-14.

2.2.8 Argon-41 Release

The argon-41 release rate is fixed in the code as 34 Ci/year. This value does not impact the release rate of any other nuclides. PWRGE does not do any calculations to arrive at the Ar-41 release rate, but simply takes the hardwired value for Ar-41. Ar-41 is not one of the nuclides identified as important to NRC for the purposes of this study.

2.2.9 Summary of Results

The following parameters that are hardwired in the PWRGE code have been identified to have a significant impact on the calculated concentration of one or more of the nuclides that were identified as significant outputs for the code.

- Plant capacity factor
- Nuclides in the primary coolant
- Radionuclide releases from various ventilation systems before treatment
- Tritium release
- Carbon-14 release.

2.3 PWRLE Sensitivity

The nuclides that have the greatest impact on the dose for PWR liquid effluents are H-3, Cs-137, Ru-106, Co-60, and I-131. The following sensitivity studies will identify the impact of changes to hardwired parameters on these nuclides.

Table 2.10 shows the nominal values of nuclide release rates for the standard GALE case.

Table 2.10. Nuclide Release Rates from PWRLE for Standard Case (curies/year)

H-3	Cs-137	Ru-106	Co-60	I-131
280	0.039	0.032	0.014	0.0079

2.3.1 Plant Capacity Factor

A plant capacity factor of 80% is assumed in GALE. The modern operation of plants has allowed for significantly higher capacity factors. If a plant capacity factor of 100% is used, the concentrations of the selected nuclides do not change. Although the plant capacity factor is hardwired into PWRLE, it is not used in any calculations.

2.3.2 Nuclides in Primary and Secondary Coolant

Concentrations of various nuclides in the primary and secondary coolant are hardwired into the PWRLE code. The concentrations in the primary coolant were all biased down by a factor of 10. This had the result of decreasing the release rate of Cs-137 and I-131 and increasing the release rate of Ru-106. The concentrations in the secondary coolant were all biased down by a factor of 10. This had the result of decreasing the release rate of Ru-106 and increasing the release rate of Cs-137 and I-131. These results are shown in Table 2.11. The decreases are due to decreases in coolant concentrations. The increases are due to an adjustment that is made to the final release rates for unexpected released. This adjustment is made in proportion to the amount of each nuclide, so if several nuclides are decreased, then the others will be increased. This will be discussed in more detail in Section 2.3.6.

Table 2.11. Sensitivity to Decreasing Concentration of Various Nuclides in the Primary Coolant and the Secondary Coolant

		Release Rate, Ci/year				
Bias	**Value**	**H-3**	**Cs-137**	**Ru-106**	**Co-60**	**I-131**
Nominal		280	0.039	0.032	0.014	0.0079
	Primary/10	280	0.025	0.040	0.014	0.0048
	Secondary/10	280	0.043	0.029	0.014	0.0088

2.3.3 Radionuclides in Untreated Detergent Waste

PWRLE allows the user to specify if there is an onsite laundry or if laundry is done offsite. If onsite laundry is done, PWRLE has a hardwired list of nuclides and release rates that are added to the liquid effluents from other sources. If the detergent wastes are treated, the user can specify the DF that should be applied.

The sample case for PWRLE assumes onsite laundry with no treatment of detergent wastes. To determine the impact of the assumed detergent releases on the selected nuclides, the release rate for each nuclide was reduced by a factor of 10. Table 2.12 shows the result of this study.

It can be seen in this table that for the sample case, the detergent wastes form a significant fraction of the total release rate. In particular, the Co-60 release is almost entirely from detergent wastes. This explains why other parameters do not significantly impact the Co-60 concentration.

Table 2.12. Sensitivity to Decreasing Concentration of Various Nuclides in the Detergent Waste

		Release Rate, Ci/year				
Bias	**Value**	**H-3**	**Cs-137**	**Ru-106**	**Co-60**	**I-131**
Nominal		280	0.039	0.032	0.014	0.0079
	Detergent/10	280	0.025	0.024	0.0015	0.0065

2.3.4 Tritium Releases

The total tritium release in PWRLE is hardwired to 0.4 Ci/yr/MW. Of this amount, liquids released are assumed to have a concentration of 1.0 μCi/mL. PWRLE allows up to 90% of the total tritium to be released in liquids. If enough liquid waste is entered as an input to account for more than 90% of the total tritium release, then PWRLE caps the liquid release fraction at 90% and assigns the remaining 10% of the tritium to PWRGE. The flow rates from various systems are provided to PWRLE though input. The remainder is released through gaseous pathways. Table 2.13 shows the results of biasing the assumed tritium release up and down. Biasing the tritium release does not impact the H-3 released by the liquid pathway because PWRLE assumes that 1.0 μCi/mL is released through liquid pathways. If the tritium concentration in the primary coolant (1.0 μCi/mL) were to change, it would have a direct impact on the tritium release rate.

Table 2.13. Sensitivity to Changing the Tritium Release

		Release Rate, Ci/year				
Bias	**Value**	**H-3**	**Cs-137**	**Ru-106**	**Co-60**	**I-131**
Nominal	0.4 Ci/yr/MW	280	0.039	0.032	0.014	0.0079
	1.0 Ci/yr/MW	280	0.039	0.032	0.014	0.0079
	0.2 Ci/yr/MW /10	280	0.039	0.032	0.014	0.0079

2.3.5 DFs for Demineralizers, Evaporators, Filters, Reverse Osmosis Units

Many of the DFs that are used in PWRLE are provided in the code input. DFs for cleanup of shim bleed, equipment drain waste, clean waste, dirty waste, blowdown waste, and regenerant wastes are all input to the code. DFs are hardwired in PWRLE for the condensate demineralizer and the primary coolant purification demineralizer.

Table 2.14 shows the results of increasing the DFs for the condensate demineralizer. This has no impact on the release rate of the selected nuclides.

Table 2.15 shows the results of increasing the DFs for the primary coolant demineralizer. The only DF that has an impact is increasing the DF for Cs and Rb from 2 to 20. This reduces the release rate of Cs-137. In this case, the release rates of Ru-106 and I-131 are slightly increased because of the unexpected release discussed in Section 2.3.6. Co-60 release rates are not impacted because this release is primarily from detergent wastes.

Table 2.14. Sensitivity to Changing DFs for Condensate Demineralizer

		Release Rate, Ci/year				
Bias	**Value**	**H-3**	**Cs-137**	**Ru-106**	**Co-60**	**I-131**
Nominal	DF Anion = 10 DF Cs, Rb=2	280	0.039	0.032	0.014	0.0079
	DF Anion = 20	280	0.039	0.032	0.014	0.0079
	DF Cs, Rb = 10	280	0.039	0.032	0.014	0.0079

Table 2.15. Sensitivity to Changing DFs for Primary Coolant Demineralizer

		Release Rate, Ci/year				
Bias	**Value**	**H-3**	**Cs-137**	**Ru-106**	**Co-60**	**I-131**
Nominal	DF Br, I=100 DF Cs, Rb=2 DF other=50	280	0.039	0.032	0.014	0.0079
	DF Br, I=200	280	0.039	0.032	0.014	0.0079
	DF Cs, Rb=20	280	0.032	0.035	0.014	0.0087
	DF other=100	280	0.039	0.032	0.014	0.0079

2.3.6 Frequency and Extent of Unplanned Liquid Releases

PWRLE assumes that an additional 0.16 Ci/yr beyond the calculated release will be released via liquid pathways and uses the total calculated isotopic distribution without the detergent wastes to partition this release. H-3 is not increased by this adjustment. As discussed in other sections, this is the reason that if the release rate of several nuclides is decreased, then the release rate of the other nuclides will increase because the calculated isotopic distribution changes.

The adjustment to the total release rate for anticipated operational occurrences (AOOs) was biased between 0.0 and 0.5 Ci/year. The impact of these biases is shown in Table 2.16. This table shows that changing the release for AOO has a significant impact on the Cs-137, Ru-106, and I-131 release rate. It does not impact the H-3 release because different mechanisms are used to control that release as discussed in Section 2.3.4. This does not impact the Co-60 release because a majority of the Co-60 release in this case comes from detergent waste, and the detergent waste is not used in the isotopic distribution used to determine the AOO release.

Table 2.16. Sensitivity to Changing Additional Release Rate for Anticipated Operational Occurrences

		Release Rate, Ci/year				
Bias	**Value**	**H-3**	**Cs-137**	**Ru-106**	**Co-60**	**I-131**
Nominal	0.16 Ci/yr	280	0.039	0.032	0.014	0.0079
Upper	0.5 Ci/yr	280	0.087	0.079	0.014	0.021
Lower	0.0 Ci/yr	280	0.017	0.0095	0.014	0.0018

2.3.7 Summary of Results

The following parameters that are hardwired in the PWRLE code have been identified to have a significant impact on the calculated concentration of one or more of the nuclides that were identified as significant outputs for the code.

- Nuclides in the primary and secondary coolant
- Radionuclides in untreated detergent waste
- Tritium release
- DF for Cs and Rb in the primary coolant demineralizer
- Frequency and extent of unplanned liquid releases

2.4 Recommended Updates for PWR GALE Codes

Based on the sensitivity analysis performed on the PWR GALE codes, certain hardwired parameters have been identified as having a significant impact on the relevant output of these codes. Table 2.17 lists these parameters along with the basis and vintage of each of these parameters and the location of possible newer information.

Table 2.17. Hardwired Parameters in PWR GALE Codes that Have a Significant Impact on the Relevant Output

Parameter	Current Basis	New Information
Plant Capacity Factor	Survey of 32 reactors from 1972-1977	Institute of Nuclear Power Operations database
Nuclides in the primary and secondary coolant	ANS-18.1 Standard published in 1999	American Nuclear Society (ANS)-18.1 committee and Electric Power Research Institute data
Radionuclide releases from various ventilation systems before treatment	Iodine: Survey of 5 to 8 reactors from 1978-1980 Particulates: Survey of 12 reactors from 1978-1980 Noble Gas: Measurements from 2 reactors in 1974	Reactor survey campaign
Radionuclides in untreated detergent waste	Survey of 4 reactors from 1971-1977	Very few plants do onsite laundry. Survey those that do.
Tritium Release	Survey of 23 reactors from 1972-1977	Radiation Exposure Information and Reporting System (REIRS) report, 1.21 report, and RadBench
DF for Cs and Rb in the primary coolant demineralizer	Survey of 6 reactors from 1978-1981 and Oak Ridge National Laboratory generic review in 1978	Reactor survey campaign
Carbon-14 release	Survey of 10 reactors from 1975-1978	Possible information in 1.21 Report
Frequency and extent of unplanned liquid releases	154 reactor years of data from 1970-1977 with 62 unplanned liquid releases	1.21 Report

3.0 BWR GALE

This section contains the sensitivity analysis for the following BWR GALE codes: BWR liquid effluents (BWRLE) and BWR gaseous effluents (BWRGE). Section 3.1 lists the parameters identified as hardwired data within the BWR GALE codes. These parameters are identified as those that are used in the BWRGE code, those that are used in the BWRLE code, and those that are provided to the codes through input. Section 3.2 contains the results of the sensitivity analysis performed on BWRGE where the relevant parameters identified in Section 3.1 were varied to determine the effects of changing these parameters on the outputs of interest. Section 3.3 contains the results of the sensitivity analysis performed on BWRLE where the relevant parameters identified in Section 3.1 were varied to determine the effects of changing these parameters on the outputs of interest. No sensitivity analysis was performed for those parameters that were identified as input values as they are not intrinsic to the codes. Section 3.4 contains a table showing what parameters in the BWR GALE codes should be updated. This table also lists the current basis and vintage of each parameter and the location of new information that could be used to update each parameter.

3.1 Parameters Identified

Table 3.1 shows a list of parameters that have been identifies as plant operation data needs for the BWR GALE codes. There are two BWR GALE codes. BWRGE calculates the gaseous effluents from a BWR, and BWRLE calculates the liquid effluents for a BWR. Not all of these parameters apply both codes, so Table 3.1 identifies which code each parameter applies to. In addition, some of these parameters are not hardwired into the code, but rather are controlled though input variables. These parameters are identified in Table 3.1 and were not included in the sensitivity analysis because they do not need to be updated in the code.

Table 3.1. Parameters Identified as Plant Operation Data Needs for BWR GALE Code

Parameter	Applicability BWRGE	BWRLE	Comment	Section for Sensitivity Analysis
Plant capacity	Yes	Yes		3.2.1, 3.3.1
Nuclides in primary coolant	Yes	Yes		3.2.2, 3.3.2
Radionuclide releases from various ventilation systems before treatment	Yes	Not used		3.2.3
Air Inleakage to Main Condenser	Yes	Not used		3.2.4
Radionuclide removal efficiencies for charcoal absorbers and HEPA filters	Input	Not Used	Input on Cards 24, 25, 28, and 29	
Dynamic adsorption coefficients for charcoal	Input	Not Used	Input on cards 31 and 32	
Flow rates into liquid radwaste systems	Not Used	Input	Provided as inputs Cards 9-20	
Radionuclides in untreated detergent waste	Not Used	Yes		3.3.3
Demineralizer regeneration cycle	Not Used	Input	Condensate demineralizer regeneration time is input to the code Card 6	
Tritium releases	Yes	Yes		3.2.5, 3.3.4
DFs for demineralizers, evaporators, filters, reverse osmosis units	Not Used	Yes		3.3.5
Frequency and extent of unplanned liquid releases	Not Used	Yes		3.3.6
Carbon-14 release	Yes	Not Used		3.2.6
Argon-41 release	Yes	Not Used		3.2.7

3.2 BWRGE Sensitivity

The nuclides that have the greatest impact on offsite dose for BWR gaseous effluents are H-3, C-14, Cs-137, I-131, Kr-89, and Xe-138. The following sensitivity studies identify the impact of changes to hardwired parameters on these nuclides.

Table 3.2 shows the nominal values of nuclide release rates for the standard GALE case.

Table 3.2. Nuclide Release Rates from BWRGE for Standard Case (curies/year)

H-3	C-14	Cs-137	I-131	Kr-89	Xe-138
52	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}

3.2.1 Plant Capacity Factor

A plant capacity factor of 80% is assumed in GALE. The modern operation of plants has allowed for significantly higher capacity factors. If a plant capacity factor of 100% is used, the concentrations of the selected nuclides do not change.

3.2.2 Nuclides in Primary Coolant

Concentrations of iodine isotopes in the primary coolant water and noble gas isotopes in the primary coolant steam are hardwired in BWRGE using data statements. The iodine concentrations were biased down by a factor of 10. This lowered the I-131 release rate by a factor of 37%. The noble gas concentrations were biased down by a factor of 10. This had no impact on any of the outputs. The release rates of Kr-89 and Xe-138 are dominated by a different mechanism as will be demonstrated later. Table 3.3 shows the results of this sensitivity study.

Table 3.3. Sensitivity of Changes to Primary Coolant Nuclide Concentrations

		Release Rate, Ci/year					
Bias	**Value**	**H-3**	**C-14**	**Cs-137**	**I-131**	**Kr-89**	**Xe-138**
Nominal		52	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}
	Iodine/10	52	9.5	6.1×10^{-3}	2.2×10^{-1}	6.1×10^{2}	1.0×10^{3}
	Noble Gas/10	52	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}

3.2.3 Radionuclide Releases from Various Ventilation Systems Before Treatment

Release rates from the ventilation systems of various buildings are hardwired into the BWRGE code. For the noble gas, iodine, and the particulates, data statements are used to input the release rates from various buildings. These rates are found to have a significant impact on the release rates for Cs-137, I-131, Kr-89, and Xe-138. For I-131, a majority of the release comes from the turbine and vacuum pump during normal operation. For Cs-137, a majority of the release comes from the auxiliary and turbine buildings. For Kr-89, a majority of the release comes from the turbine and radwaste buildings. For Xe-138, a majority of the release comes from the turbine. The results of these sensitivity studies are shown in Table 3.4 for iodine and Table 3.5 for noble gas and particulates.

Table 3.4. Sensitivity to Decreasing Iodine Release Parameters from Building Ventilation Systems

		Release Rate, Ci/year					
Bias	**Value**	**H-3**	**C-14**	**Cs-137**	**I-131**	**Kr-89**	**Xe-138**
Nominal		52	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}
Normal	Reactor/10	52	9.5	6.1×10^{-3}	3.3×10^{-1}	6.1×10^{2}	1.0×10^{3}
	Turbine/10	52	9.5	6.1×10^{-3}	2.1×10^{-1}	6.1×10^{2}	1.0×10^{3}
	Radwaste/10	52	9.5	6.1×10^{-3}	3.4×10^{-1}	6.1×10^{2}	1.0×10^{3}
	Vacuum Pump/10	52	9.5	6.1×10^{-3}	2.8×10^{-1}	6.1×10^{2}	1.0×10^{3}
Shutdown	Reactor/10	52	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}
	Turbine/10	52	9.5	6.1×10^{-3}	3.4×10^{-1}	6.1×10^{2}	1.0×10^{3}
	Radwaste/10	52	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}
	Vacuum Pump/10	52	9.5	6.1×10^{-3}	3.1×10^{-1}	6.1×10^{2}	1.0×10^{3}

Table 3.5. Sensitivity to Decreasing Noble Gas and Particulate Release Parameters from Building Ventilation Systems

		Release Rate, Ci/year					
Bias	**Value**	**H-3**	**C-14**	**Cs-137**	**I-131**	**Kr-89**	**Xe-138**
Nominal		52	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}
	Containment/10	52	9.5	6.0×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}
	Auxiliary/10	52	9.5	1.6×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}
	Turbine /10	52	9.5	5.2×10^{-3}	3.5×10^{-1}	8.9×10^{1}	1.0×10^{2}
	Radwaste/10	52	9.5	6.0×10^{-3}	3.5×10^{-1}	5.8×10^{2}	1.0×10^{3}

3.2.4 Air Inleakage to Main Condenser

The air inleakage to the main condenser is used in BWRGE to calculate the holdup time for the charcoal delay systems based on the input dynamic adsorption coefficient, the mass of the charcoal, and the thermal power level. In this equation, the air inleakage is inversely proportional to the holdup time. The nominal value of air inleakage is 0.0062 ft^3/min. This rate was biased between 0.01 and 0.002 ft^3/min. No difference was observed in the release rate of the selected nuclides. Table 3.6 shows the results of this study:

Table 3.6. Sensitivity to Changing the Air Inleakage to Main Condenser

		Release Rate, Ci/year					
Bias	**Value**	**H-3**	**C-14**	**Cs-137**	**I-131**	**Kr-89**	**Xe-138**
Nominal	0.0062 ft^3/yr/MW	52	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}
Upper	0.01 ft^3/yr/MW	52	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}
Lower	0.002 ft^3/yr/MW	52	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}

3.2.5 Tritium Releases

The tritium release in BWRGE is hardwired to 0.03 Ci/yr/MW. Of this amount, 50% is released through liquid pathways, and 50% is released through gaseous pathways. Table 3.7 shows the results of biasing the assumed tritium release up and down. Biasing the tritium release affects only the H-3 release rate.

Table 3.7. Sensitivity to Changing the Tritium Release

		Release Rate, Ci/year					
Bias	**Value**	**H-3**	**C-14**	**Cs-137**	**I-131**	**Kr-89**	**Xe-138**
Nominal	0.03 Ci/yr/MW	52	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}
Upper	0.05 Ci/yr/MW	86	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}
Lower	0.01 Ci/yr/MW	18	9.5	6.1×10^{-3}	3.5×10^{-1}	6.1×10^{2}	1.0×10^{3}

3.2.6 Carbon-14 Release

The carbon-14 release rate is fixed in the code as 9.5 Ci/year, which is based on a calculation external to the BWRGE code proper. This value does not impact the release rate of any other nuclides. BWRGE itself does not do any calculations to arrive at the C-14 release rate, but simply uses the externally-calculated, hardwired value for C-14.

3.2.7 Argon-41 Release

BWRGE contains parameters to calculate the argon-41 release rate. This value does not impact the release rate of any other nuclides. Ar-41 is not one of the nuclides that have been identified as important nuclides for this study.

3.2.8 Summary of Results

The following parameters that are hardwired in the BWRGE code have been identified to have a significant impact on the calculated concentration of one or more of the nuclides that were identified as significant outputs for the code:

- Nuclides in the primary coolant
- Radionuclide releases from various ventilation systems before treatment
- Tritium release
- Carbon-14 release.

3.3 BWRLE Sensitivity

The nuclides that have the greatest impact on the dose for BWR liquid effluents are H-3, Cs-137, P-32, I-131, and Co-60. The following sensitivity studies will identify the impact of changes to hardwired parameters on these nuclides.

Table 3.8 shows the nominal values of nuclide release rates for the standard GALE case.

Table 3.8. Nuclide Release Rates from BWRLE for Standard Case (curies/year)

H-3	Cs-137	P-32	I-131	Co-60
51	0.016	0.00022	0.082	0.015

3.3.1 Plant Capacity Factor

A plant capacity factor of 80% is assumed in GALE. The modern operation of plants has allowed for significantly higher capacity factors. If a plant capacity factor of 100% is used, the concentrations of the selected nuclides do not change. Although the plant capacity factor is hardwired into BWRLE, it is not used in any calculations.

3.3.2 Nuclides in Primary Coolant

Concentrations of various nuclides in the liquid primary coolant are hardwired into the BWRLE code. To determine sensitivity, the concentrations in the primary coolant were all reduced by a factor of 10. This had the result of slightly decreasing the release rate of P-32, I-131, and Co-60. These results are shown in Table 3.9.

Table 3.9. Sensitivity to Decreasing Concentration of Various Nuclides in the Primary Coolant

		Release Rate, Ci/year				
Bias	**Value**	**H-3**	**Cs-137**	**P-32**	**I-131**	**Co-60**
Nominal		51	0.016	0.00022	0.082	0.015
	Primary/10	51	0.016	0.00021	0.066	0.014

3.3.3 Radionuclides in Untreated Detergent Waste

BWRLE allows the user to specify if there is an onsite laundry or if laundry is done offsite. If onsite laundry is done, BWRLE has a hardwired list of nuclides and release rates that are added to the liquid effluents from other sources. If the detergent wastes are treated, the user can specify the DF that should be applied.

The sample case for BWRLE assumes onsite laundry with no treatment of detergent wastes. To determine the impact of the assumed detergent releases on the selected nuclides, the release rate for each nuclide was reduced by a factor of 10. Table 3.10 shows the result of this study.

It can be seen in this table that for the sample case, the detergent wastes form a significant fraction of the total release rate. In particular, the Cs-137, P-32, and Co-60 release is almost entirely from detergent wastes. This explains why other parameters do not have a large impact on the Cs-137, P-32, and Co-60 concentrations. The release of I-131 was only slightly impacted because the fraction of I-131 in the detergent waste is small compared to other waste streams.

Table 3.10. Sensitivity to Decreasing Concentration of Various Nuclides in the Detergent Waste

		Release Rate, Ci/year				
Bias	**Value**	**H-3**	**Cs-137**	**P-32**	**I-131**	**Co-60**
Nominal		51	0.016	0.00022	0.082	0.015
	Detergent/10	51	0.0019	0.00006	0.081	0.0019

3.3.4 Tritium Releases

The tritium release in BWRLE is hardwired to 0.03 Ci/yr/MW. Of this amount, 50% is released through liquid pathways, and 50% is released through gaseous pathways. Table 3.11 shows the results of biasing the assumed tritium release up and down. Biasing the tritium release only affects the H-3 release rate.

Table 3.11. Sensitivity to Changing the Tritium Release

		Release Rate, Ci/year				
Bias	**Value**	**H-3**	**Cs-137**	**P-32**	**I-131**	**Co-60**
Nominal	0.03 Ci/yr/MW	51	0.016	0.00022	0.082	0.015
Upper	0.05 Ci/yr/MW	85	0.016	0.00022	0.082	0.015
Lower	0.01 Ci/yr/MW	17	0.016	0.00022	0.082	0.015

3.3.5 DFs for Demineralizers, Evaporators, Filters, and Reverse Osmosis Units

Many of the DFs that are used in BWRLE are provided in the code input. DFs for cleanup of high-purity waste, low-purity waste, chemical waste, and regenerant solutions waste are all input to the code. DFs are hardwired in BWRLE for the condensate demineralizer.

Table 3.12 shows the results of increasing the DFs for the condensate demineralizer. The only change that had an impact was increasing the DF for I and Br, which increased the release rate of I-131. This is because increasing the DF for the condensate demineralizers increases the rate into the chemical waste stream. The release from the chemical waste processing is significant for I-131. The slight decrease in Co-60 is due to the adjustment factor discussed in Section 3.3.6.

Table 3.12. Sensitivity to Changing DFs for Condensate Demineralizer

		Release Rate, Ci/year				
Bias	**Value**	**H-3**	**Cs-137**	**P-32**	**I-131**	**Co-60**
Nominal	DF I, Br = 10 DF Cs, Rb=2 DF Other = 10	51	0.016	0.00022	0.082	0.015
	DF I, Br = 20	51	0.016	0.00022	0.084	0.014
	DF Cs, Rb = 10	51	0.016	0.00022	0.082	0.015
	DF Other = 20	51	0.016	0.00022	0.082	0.015

3.3.6 Frequency and Extent of Unplanned Liquid Releases

BWRLE assumes that an additional 0.1 Ci/yr beyond the calculated release will be released via liquid pathways and uses the total calculated isotopic distribution without the detergent wastes to partition this release. H-3 is not increased by this adjustment. As discussed in other sections, this is the reason that if the release rate of several nuclides is decreased, then the release rate of the other nuclides will increase because the calculated isotopic distribution changes.

The adjustment to the total release rate for AOOs was biased between 0.0 and 0.5 Ci/year. The impact of these biases is shown in Table 3.13. This table shows that changing the release for AOO has a significant impact on the P-32 and I-131 release rates and a small impact on the Cs-137 and Co-60 release rates. It does not impact the H-3 release because different mechanisms are used to control that release as discussed in Section 3.3.4.

Table 3.13. Sensitivity to Changing Additional Release Rate for Anticipated Operational Occurrences

		Release Rate, Ci/year				
Bias	**Value**	**H-3**	**Cs-137**	**P-32**	**I-131**	**Co-60**
Nominal	0.1 Ci/yr	51	0.016	0.00022	0.082	0.015
Upper	0.5 Ci/yr	51	0.017	0.00035	0.33	0.016
Lower	0.0 Ci/yr	51	0.016	0.00019	0.02	0.014

3.3.7 Summary of Results

The following parameters that are hardwired in the BWRLE code have been identified to have a significant impact on the calculated concentration of one or more of the nuclides that were identified as significant outputs for the code:

- Nuclides in the primary coolant
- Radionuclides in untreated detergent waste
- Tritium release
- DF for I and Br in the condensate demineralizer
- Frequency and extent of unplanned liquid releases.

3.4 Recommended Updates for BWR GALE Codes

Based on the sensitivity analysis performed on the BWR GALE codes, certain hardwired parameters have been identified as having a significant impact on the relevant output of these codes. Table 3.14 lists these parameters along with the basis and vintage of each of these parameters and the location of possible newer information.

Table 3.14. Hardwired Parameters in PWR GALE Codes that Have a Significant Impact on the Relevant Output

Parameter	Current Basis	New Information
Nuclides in the primary and secondary coolant	ANS-18.1 Standard published in 1999	ANS-18.1 committee and Electric Power Research Institute data
Radionuclide releases from various ventilation systems before treatment	Iodine: Survey of 4 to 6 reactors from 1973-1978 Particulates: Survey of 1 to 5 reactors from 1973-1978 Noble Gas: Survey of 4 to 5 reactors from 1973-1978	Reactor survey campaign
Radionuclides in untreated detergent waste	Survey of 4 reactors from 1974-1978	Very few plants do onsite laundry. Survey those that do.
Tritium Release	Survey of 21 reactors from 1972-1977	REIRS report, 1.21 report, and RadBench
DF for I and Br in the condensate demineralizer	ORNL generic review in 1978	Reactor survey campaign
Carbon-14 release	Calculation done using standard parameters	Could change to do calculation in GALE using input values where appropriate
Frequency and extent of unplanned liquid releases	127 reactor years of data from 1970-1977 with 50 unplanned liquid releases	1.21 Report

4.0 Conclusions

A sensitivity analysis was performed on each of the four GALE codes where hardwired parameters were varied to determine their impact on the output of interest. The result of this analysis is a reduced list of parameters that should be updated in GALE based on their impact on outputs. Tables containing these parameters are shown in Sections 2.4 and 3.4 (Table 2.17 and Table 3.14, respectively). Only two parameters are expected to require surveying operating plants: radionuclide releases from various ventilation systems before treatment and the DF for the condensate demineralizer. These two parameters will need to be surveyed on both BWR and PWR plants.

This is not to say that the two parameters noted above are the only parameters that PNNL plans to investigate, only that these are the most important fixed parameters used in GALE calculations of nuclides for each code. These parameters are of greatest importance for defining potential exposures. PNNL will investigate all important parameters and in particular will study why C-14 and similar isotopes are modeled as a fixed number of Ci/year rather than calculated based on appropriate input factors.