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Protecting People and the Environment

NUREG-1940
Supplement 1

RASCAL 4.3: Description of Models and Methods

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RASCAL 4.3: Description of Models and Methods

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ABSTRACT

RASCAL 4.3 contains a number of new features and revision of several old features in response to the lessons learned by the U.S. Nuclear Regulatory Commission staff during its response to the events at the Fukushima Daiichi nuclear power plants following the March 11, 2011 earthquake off the coast of Japan and the tsunami that it triggered. This document is a supplement to NUREG-1940, "RASCAL 4: Description of Models and Methods," which is the technical basis for the models and methods used in the RASCAL computer code versions 4.0, 4.1 and 4.2. This supplement contains the technical basis for changes and additions to RASCAL implemented in versions RASCAL 4.3 and 4.3.1. Additionally, Appendix A provides errata information for NUREG-1940.

RASCAL 4.3 adds a long-term station blackout source term (LTSBO) model based on the results of the State-of-the-Art Reactor Consequences Analyses (SOARCA) study. It extends the RASCAL domain to 100 miles, increases the allowed calculation duration to 96 hours, and it adds a child thyroid dose calculation. RASCAL 4.3 includes new features to acquire meteorological data from the Internet, to calculate custom radionuclide inventories for reactor cores and spent fuel, to display a radionuclide activity balance within the power plant components and atmosphere, and to sort and display the radionuclides released to the atmosphere by importance to dose pathways. RASCAL 4.3 also adds the capability to import, merge, and export source terms. Improvements include revision of the pressure-hole size method of calculating the leak rate from containment and revision of the calculation of spent fuel source terms.

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ABBREVIATIONS

ABWR	advanced boiling-water reactor
ADAMS	Agencywide Documents Access and Management System
AP1000	advanced passive reactor
ATD	atmospheric transport, dispersion, and deposition
Bq	becquerel
BWR	boiling-water reactor
CED	committed equivalent dose
CEDE	committed effective dose equivalent
CF	conversion factor
Ci	curie
CSV	comma separated values
d	days
DCD	design control document
DRD	direct reading dosimeter
ED	effective dose
EFPD	effective full-power day
EPA	U.S. Environmental Protection Agency
EPR	Evolutionary Power Reactor
ESBWR	economic simplified boiling-water reactor
FDA	U.S. Food and Drug Administration
FGR	Federal Guidance Report
FSAR	final safety analysis report
ft^3	cubic feet
gal	gallon
gal/d	gallons per day
gal/h	gallon per hour
GSD	geometric standard deviation
GSM	geometric standard mean
GWd	gigawatt day
h	hours
HF	hydrogen fluoride
ICRP	International Commission on Radiological Protection
in^2	square inch
KI	potassium iodide
km	kilometer
LBLOCA	large-break loss-of-coolant accident
LOCA	loss-of-coolant accident
LTSBO	long-term station blackout
m	meters
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
mi	miles
min	minutes

mph	miles per hour
MWd	megawatt-day(s)
MWt	megawatt thermal
MTU	metric ton of uranium
NDFD	National Digital Forecast Database
NLCD	National Land-Cover Dataset
NRC	U.S. Nuclear Regulatory Commission
NWS	National Weather Service
ORIGEN_ARP	Oak Ridge Isotope Generator_Automatic Rapid Processing
PAGs	protective action guidelines
psig	pounds per square inch gauge
PWR	pressurized-water reactor
RASCAL	Radiological Assessment System for Consequence Analysis
RCS	reactor coolant system
SBGTS	standby gas treatment system
SFP	spent fuel pool
SOARCA	State-of-the-Art Reactor Consequence Analyses
STDose	source term to dose
Sv	sievert
TED	total effective dose
TEDE	total effective dose equivalent
UF_6	uranium hexafluoride
UO_2F_2	uranyl fluoride
US-APWR	US-advanced pressurized-water reactor
XML	extensible markup language
yr	years

1.0 INTRODUCTION

The Radiological Assessment System for Consequence Analysis (RASCAL) version 4.3 contains a number of new features and revision of several old features in response to the lessons learned by the U.S. Nuclear Regulatory Commission (NRC) staff during its response to the events at the Fukushima Daiichi nuclear power plants following the March 11, 2011 earthquake off the coast of Japan and the tsunami that it triggered. This document is a technical supplement to NUREG-1940, "RASCAL 4: Description of Models and Methods." It is one of a set of documents that describe RASCAL [1]. NUREG-1940 (referred to as the RASCAL technical document) presents the technical bases for the RASCAL computation codes described as RASCAL 4.2.

This supplement describes the changes from RASCAL 4.2 to RASCAL 4.3.1 and provides the technical bases for those changes. This supplement has nine sections including an introduction to the changes in RASCAL 4.3 and 4.3.1 and one appendix of errata information for NUREG-1940. The second section describes the changes and additions to the source term calculations (Sections 1, 2, and 3 of NUREG-1940). The third section describes the changes to the atmospheric transport, dispersion, and dose calculations (Section 4 of NUREG-1940). Section four describes the new meteorological data acquisition module. Section five describes a new utility tool to create reactor core, coolant, and spent fuel inventories based on specific reactor operating practices and most recent outage information. Section six describes a new feature to display the radionuclide activity at several locations in a power plant and in the atmosphere and activity transfers between locations as a function of time during an accident. Section seven describes a new calculation which evaluates the importance to dose pathways of radionuclides released to the atmosphere. Section eight describes the new source term import option and the new Source Term Merge/Export tool. Finally, section nine includes a list of references.

2.0 SOURCE TERM CALCULATIONS

This section describes the changes to the RASCAL source term module implemented in RASCAL 4.3 and RASCAL 4.3.1. The most significant change is the addition of a long-term station blackout model (LTSBO). The model, which delays the onset of releases for 48-plus hours after the loss of power, is described in Section 2.1. Section 2.2 describes changes to the loss-of-coolant (LOCA) model. Section 2.3 describes changes to the option to specify a source term based on an estimate of core damage. Section 2.4 describes changes made to the treatment of coolant releases. Most notably, the changes to coolant system releases result in increased early releases of noble gases and decreased iodine releases for some release pathways. Section 2.5 describes revisions to the model used to estimate releases based on hole size and containment pressure. RASCAL 4.3 now decreases the containment pressure and the leak rate to the atmosphere as a function of time. Section 2.6 describes changes to the method for estimating source terms from containment radiation monitor readings. Sections 2.7 and 2.8 provide additional information about the source term methods using measured effluent releases. Section 2.9 contains the discussion of the changes to the calculation for the spent fuel source terms. The changes improve the estimates of the at-risk radionuclide inventory in these accidents.

Several additions to RASCAL 4.3 related to source terms are described in later sections. Section 5 describes a new feature that can be used to estimate custom radionuclide inventories in reactor cores, in reactor cooling systems, and in spent fuel. Section 6 describes a new feature that provides users with estimates of activity of selected radionuclides in several locations within the plant and the activity released to the atmosphere as a function of time. Finally, Section 8 describes changes related to the import and export of source terms from RASCAL. There is a new feature that allows the import of atmospheric source terms. Also described is a new feature which allows the merging of source terms from multiple reactors at a single site into a file that can be used by RASCAL to evaluate potential consequences of an event involving multiple reactors at the site.

2.1 Long-Term Station Blackout (LTSBO)

One of the first lessons learned in the NRC response to the Fukushima Daiichi accident was that RASCAL did not have a readily apparent method for calculating reactor source terms for LTSBO events. The RASCAL source term was based on the core damage progression timing set forth in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” [2]. Although LTSBO events were considered in preparation of NUREG-1465, the release timing set forth in Table 3.6 of NUREG-1465 is associated with large-break loss-of-coolant accidents (LBLOCAs). LTSBO accidents have a much longer time frame than LOCAs because loss of coolant occurs over a period of hours rather than seconds or minutes. A LTSBO release type option was added to RASCAL 4.3 to facilitate source term calculations for this type of accident. This LTSBO option was updated in RASCAL 4.3.1 by revising the associated in-core release fractions for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs).

The LTSBO implementation in RASCAL is based on the LTSBO accident progression as described in the following reports:

- NUREG-1935, Part 1, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Report” [3],
- NUREG-1935, Part 2, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Report” [4],
- NUREG/CR-7110, Volume 1, Revision 1, “State-of-the-Art Reactor Consequence Analysis Project, Volume 1: Peach Bottom Integrated Analysis” [5], and
- NUREG/CR-7110, Volume 2, Revision 1, “State-of-the-Art Reactor Consequence Analysis Project Volume 2: Surrey Bottom Integrated Analysis” [6].

The basic scenario for the LTSBO accident is initiated by an external event that results in a total loss of offsite power. The reactor shuts down as expected, but the diesel generators fail to start following the earthquakes, so there is no onsite AC power. Reactor cooling is maintained for a period of several hours using systems powered by DC power. Ultimately, the onsite batteries are depleted; cooling is lost; and reactor coolant system (RCS) coolant gradually boils off. Core damage and releases from the core begin when the core is uncovered. The delay between depletion of the batteries and the beginning of the releases from the core is several hours.

The releases from the reactor begin once the core starts to uncover. SOARCA terminology does not identify specific release phases as gap activity release, early in-vessel release, ex-vessel release or late in-vessel release. Rather, the SOARCA reports present release fractions for radionuclide classes as a function of time. Figure 2-1 shows the SOARCA release fractions as implemented in RASCAL for PWRs for 11 radionuclide groups. The initial releases are primarily noble gases. The noble gases are followed by the other radionuclides. Figure 2-2 shows the SOARCA release fractions as implemented in RASCAL for BWRs for the same nine radionuclide groups. Only 9 of the 11 groups were used in the SOARCA analyses and those 9 groups are not exactly the same as those in NUREG-1465. The groups not used are the uranium (U) and antimony (Sb) groups. Additionally, the group fuel release fractions do not account for decay or ingrowth.

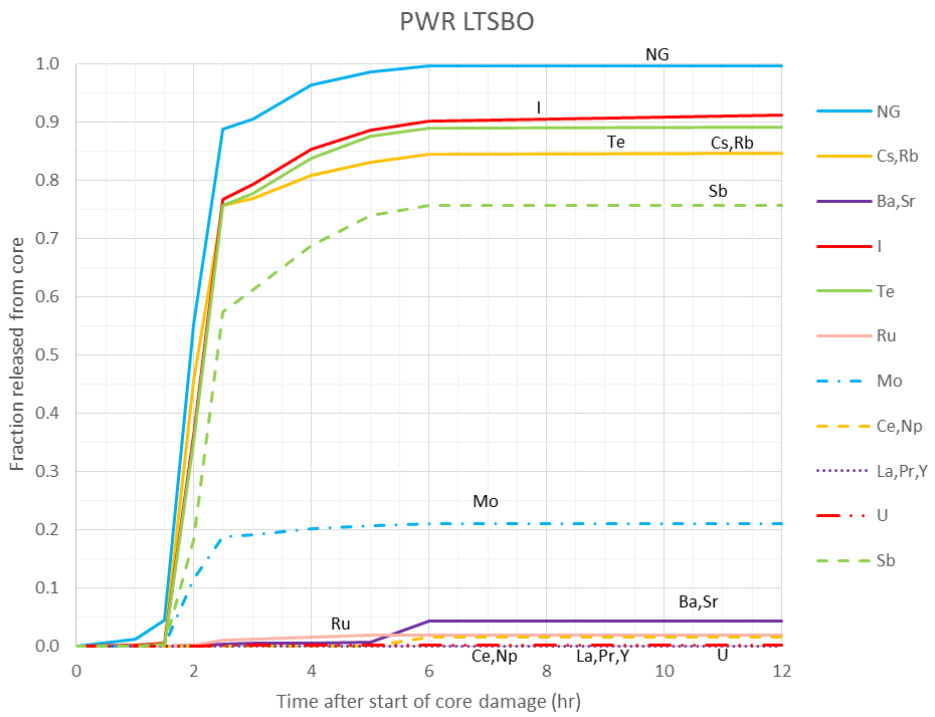


Figure 2-1 RASCAL 4.3 radionuclide release timing for a PWR LTSBO accident

Time zero in Figures 2-1 and 2-2 is the beginning of release, not the time of the initiating event. As indicated above, the time from the LTSBO initiating event (initiation of uncovering the core) to the beginning of release from the reactor may be several hours. Table 2-1 gives the default delay times RASCAL 4.3 assumes for battery depletion and the beginning of release from the reactor. The time for the start of the release in Table 2-1 includes the 8-hour heat up time for PWRs and the 6-hour heat up time for BWRs from the SOARCA analyses. However, as shown by the Fukushima Daiichi reactor operators' actions, various emergency measures can extend these delays.

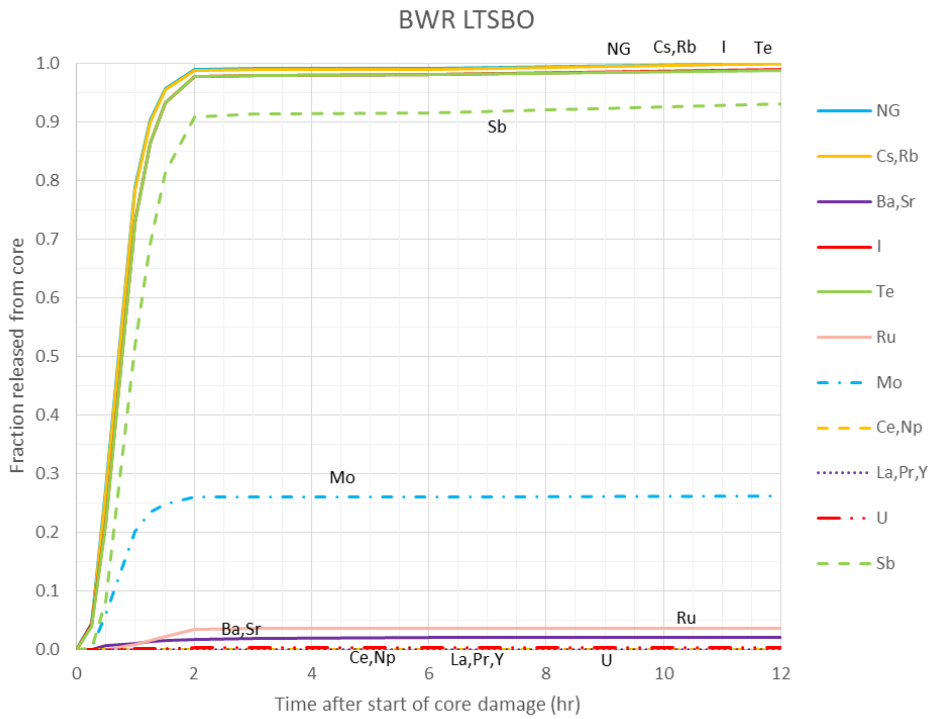


Figure 2-2 RASCAL 4.3 radionuclide release timing for a BWR LTSBO accident

Table 2-1 RASCAL 4.3 Default times after LTSBO initiating event to battery depletion and the start of release from the reactor

	PWR	BWR
Battery Depletion	4 h	4 h
Start of Release	12 h	10 h

2.1.1 LTSBO and LOCA Source Term Differences

There are several differences between the LTSBO and LOCA source terms. The difference in time delay between the initiating event and the start of release to the atmosphere is the most obvious. It has been discussed above. The increased delay for the LTSBO provides time for decay and ingrowth of radionuclides. The release fractions and release timing are also different (see Section 2.2 below).

Figures 2-3 and 2-4 compare the release timing for LTSBOs with the timing for LOCAs for the first 12 hours (h) after the beginning of release from the reactor for PWRs and BWRs, respectively. These figures clearly show significant differences in the release rates from the reactor core for important radionuclide groups such as the iodine, cesium, and tellurium groups.

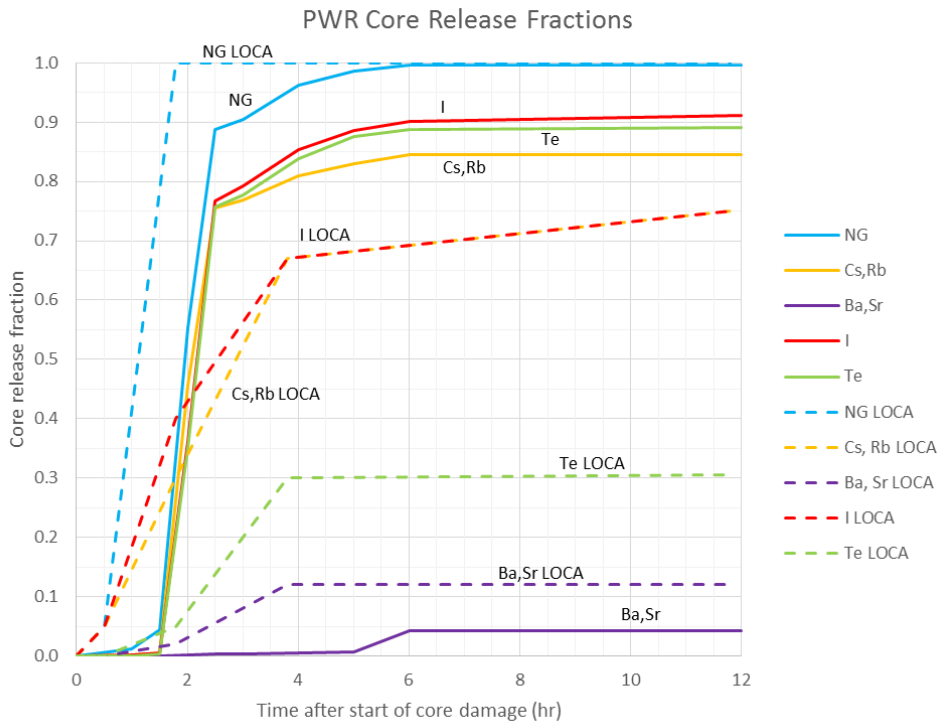


Figure 2-3 Comparison of release timing for LTSBO and LOCA events for PWRs

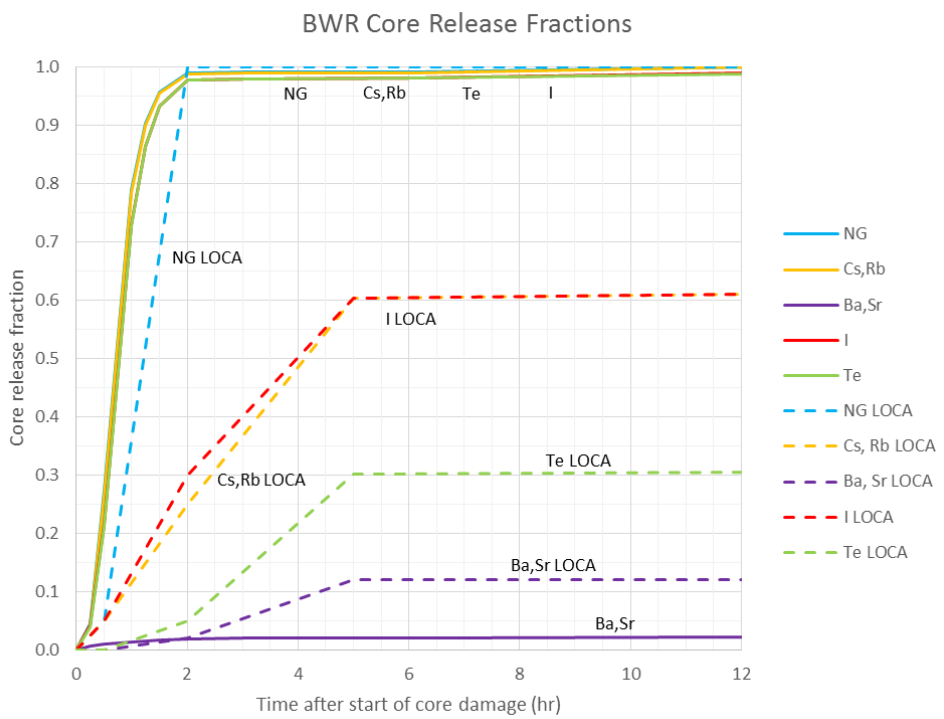


Figure 2-4 Comparison of release timing for LTSBO and LOCA events for BWRs

Figures 2-5 and 2-6 compare the activity released to the atmosphere and doses from LOCA and LTSBO accidents based on RASCAL 4.3.1 calculations for a 3586 megawatts thermal (MWt) PWR reactor and a 3515 MWt BWR reactor with a Mark II containment. The PWR accident release path (Figure 2-5) is an unfiltered ground level (10 m) release via the containment leakage pathway without depletion by sprays or filters. The BWR accident release path (Figure 2-6) is from the dry well to the atmosphere via the standby gas treatment system (SBGTS) with active filters. [*Note: It is unlikely that the SBGTS would be available during a LTSBO and in this instance the releases of iodine and cesium would increase significantly*.] Activities and doses are based on the first 8-hour releases. The radionuclides identified in these figures provide an indication of which radionuclides are more prevalent in LTSBOs (those above the diagonal line) and which are more prevalent in LOCAs (those below the line). Some of the extreme differences (e.g., krypton-87 (Kr-87), iodine-134 (I-134), barium-139 (Ba-139), lanthanum-142 (La-142), and cesium-138 (Cs-138)) are caused by the delay in the start of the LTSBO core damage. Other notable differences (e.g., strontium-89 (Sr-89), Sr-90) are related to differences in core release fractions. The differences in core release fractions may well be due to the differences in time frames of NUREG-1465 and SOARCA analyses.

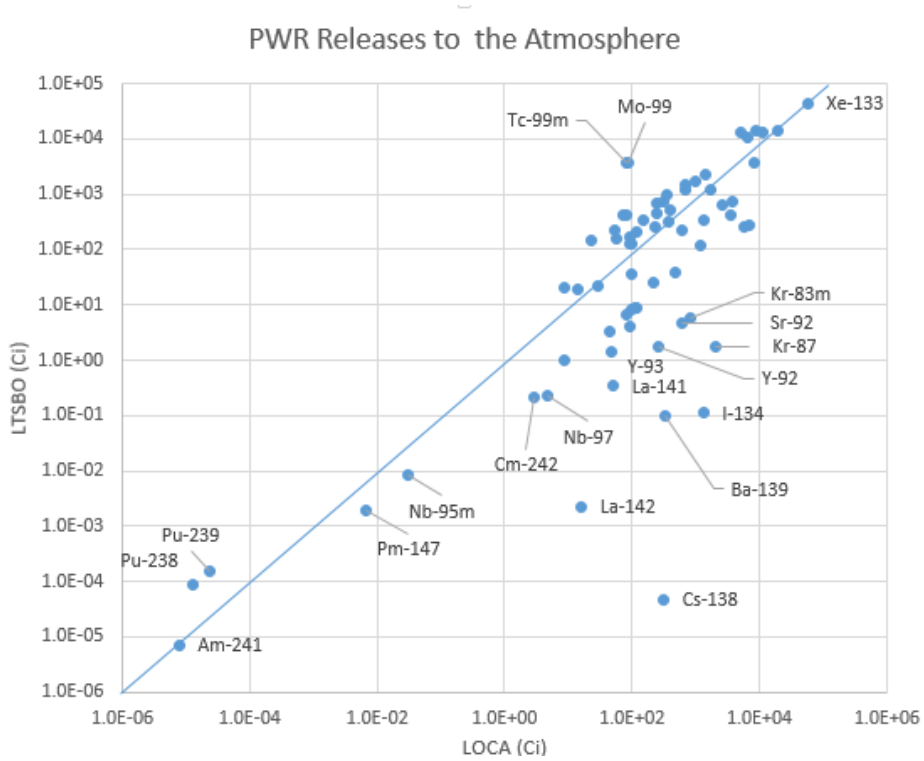


Figure 2-5 Comparison of activity released to the atmosphere in the first 8 h after beginning of release for PWR LTSBO and LOCA events

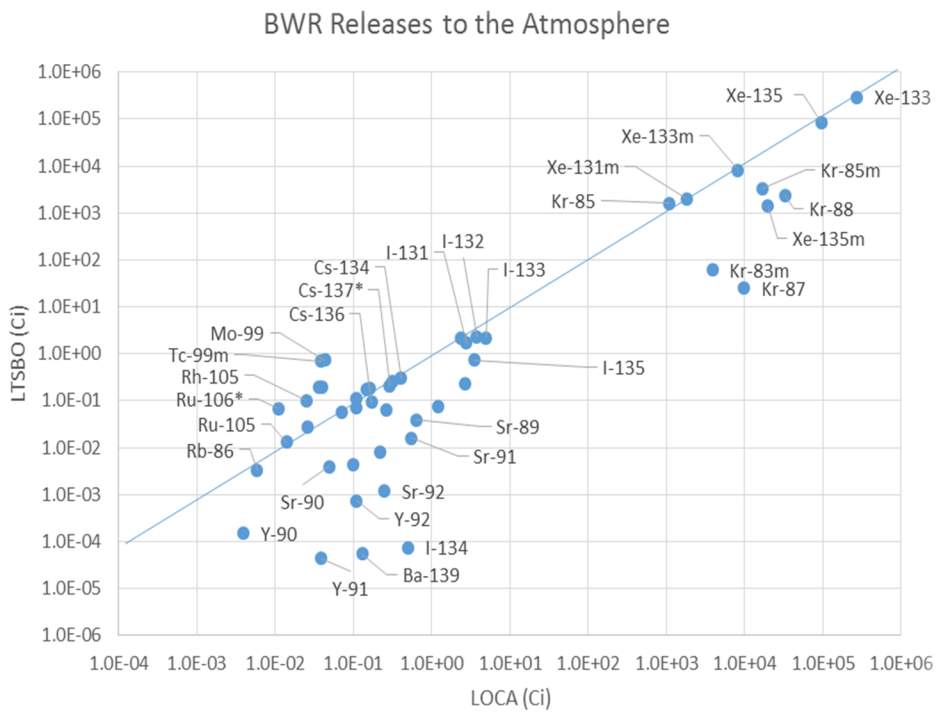


Figure 2-6 Comparison of activity released to the atmosphere in the first 8 h after beginning of release for BWR LTSBO and LOCA events

The activity releases to the atmosphere in these events are summarized and compared in Tables 2-2 and 2-3. The tables show that the LTSBO events have smaller noble gas releases and larger releases of iodines and other particles than the LOCA events. The reduction in noble gas releases for LTSBOs is directly related to the delay between the time of reactor shutdown and the beginning of core damage. Most of the krypton activity is associated with Kr-87 and Kr-88 which have half-lives of less than 3 hours, and about half of the xenon (Xe) activity is associated with Xe-135m and Xe-138, which have half-lives of about 15 minutes. The releases of iodines and other particles in PWR LTSBO accidents are larger than the releases in LOCAs because the LTSBO core release fractions for the iodine and cesium groups are larger than the corresponding LOCA core release fractions.

Table 2-2 Activity released to the atmosphere in the PWR events

Radionuclide	LOCA		LTSBO		LTSBO/LOCA Ratio
	Activity (Ci)	Percent (%)	Activity (Ci)	Percent (%)	
Noble Gas	9.4E+04	60.3	6.2E+04	44.9	0.66
Iodine	3.6E+04	23.1	4.1E+04	29.7	1.14
Other	2.6E+04	16.6	3.5E+04	25.4	1.36
Total	1.6E+05		1.4E+05		0.88

The BWR event releases more activity to the atmosphere than the PWR event, but because the BWR pathway passes through the wet well and is filtered in the SBGTS, almost all of the released activity is from the noble gases. A very small fraction of the released activity is from the iodines or other particles.

Table 2-3 Activity released to the atmosphere in the BWR events

Radionuclide	LOCA		LTSBO		LTSBO/LOCA Ratio
	Activity (Ci)	Percent (%)	Activity (Ci)	Percent (%)	
Noble Gas	4.6E+05	100.0	3.9E+05	100.0	0.85
Iodine	1.6E+01	0.0	7.0E+00	0.0	0.44
Other	1.2E+01	0.0	6.1E+00	0.0	0.51
Total	4.6E+05		3.9E+05		0.85

Table 2-4 compares the consequences of these events using RASCAL computational results extracted from the Maximum Dose Values table for a receptor at a distance of 2 miles (mi). The doses were calculated using RASCAL standard meteorology (4 miles per hour (mph) wind, dose (D) stability, no precipitation) and the International Commission on Radiological Protection Publication 60 and 72 (ICRP 60/72) dose factors. All doses for the PWR LTSBO event are larger than those for the LOCA event. The BWR LTSBO event thyroid doses are also larger than the LOCA event thyroid doses. However, the BWR LTSBO event total effective dose (ED) and cloud shine doses are lower than those for the LOCA event. Factors that contribute to these results are:

- The wet well and SBGTS filter significantly reduce the release of iodine radionuclides and other particles.

- The elevated release keeps the plume centerline above ground level.

Consequently, the iodines and other particles do not contribute significantly to inhalation doses including the thyroid doses.

Table 2-4 Comparison of RASCAL dose estimates for LTSBO and LOCA events

Dose	PWR			BWR		
	LOCA	LTSBO	Ratio	LOCA	LTSBO	Ratio
Total ED[a]	6.0E-01	8.7E-01	1.45	2.0E-03	5.5E-03	0.28
Thyroid CED[a]	4.5E+00	7.0E+00	1.55	2.1E-03	1.3E-03	0.57
Child Thyroid CED[a]	9.1E+00	1.3E+01	1.43	4.0E-03	2.2E-03	0.55
Inhalation CED[a]	4.7E-01	6.7E-01	1.43	1.4E-03	***	
Cloud shine[a]	1.3E-02	1.2E-02	0.92	1.8E-02	5.2E-03	0.29
4-day Ground shine[a]	1.1E-01	2.0E-01	1.82	***	***	

[a] rem (roentgen equivalent man)
*** <1.0E-03 rem

2.1.2 Calibration of BWR Mark I Suppression Pool Reduction Factors

Insights gained from ongoing benchmarking studies using the LTSBO feature in RASCAL 4.3 showed general agreement with the SOARCA analysis for a BWR. Noble gas releases to the atmosphere comport with the SOARCA analysis, but the halogen and particulate releases are significantly underestimated. The differences in the halogen and particulate releases to the atmosphere can be attributed to modeling differences between RASCAL 4.3 and the SOARCA BWR analysis in the treatment of depletion mechanisms within the suppression pool and late revaporization from the reactor pressure vessel directly into the drywell. Late revaporization accounts for the bypass of the suppression pool after lower head failure leading to containment failure (liner melt through in a Mark I BWR containment).

RASCAL is a fast-running emergency response code that models fission product behavior in containment differently than the SOARCA analysis for a LTSBO accident at a Mark I BWR. Below is a summary of the major differences between RASCAL and the SOARCA analysis for the LTSBO accident sequence:

- RASCAL models holdup of radioactive material in the containment, but it does not delay radioactive material movement between the fuel and containment or between containment and the atmosphere. Releases through the suppression pool are modeled as releases from the containment at the containment leak rate with no holdup in the suppression pool.

- RASCAL models the suppression pool reduction mechanism as a single filter at the point of release to the atmosphere, rather than a time-varying depletion mechanism in containment. Table 2-5 presents the RASCAL 4.3 suppression pool reduction factors (saturated and subcooled) which are applied to three categories of fission products. The technical basis documents for these reduction factors are NUREG-1228, "Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents," [7] and NUREG/CR-3727, "Fission Product Removal in Engineered Safety Feature (ESF) Systems" [8].

- RASCAL 4.3.1 does not allow the user to switch between the suppression pool and drywell during a single calculation. Therefore, the suppression pool release path cannot be bypassed once it is selected by the user. In comparison, the SOARCA analyses used multiple release paths to model fission product releases to the atmosphere.

- RASCAL models fission product deposition in containment as a time-dependent function but does not account for the revaporization of deposited radionuclides in the reactor pressure vessel in the LTSBO sequence. Note that the "LOCA (NUREG-1465)" source term option in RASCAL 4.3 incorporates a "late in vessel" release which accounts for revaporization and other processes.

Given the variability of suppression pool reduction factors, a series of RASCAL calculations were performed to determine the possibility of achieving better agreement with the SOARCA results for halogens and particulates using RASCAL v4.3.1 models and modified suppression pool reduction factors. Table 2-5 presents the RASCAL 4.3.1 suppression pool factors results of these calculations.

Table 2-5 RASCAL 4.3 and 4.3.1 BWR Mark I suppression pool reduction factors

	Noble Gases	Halogens	Other Particles
RASCAL 4.3 Saturated	1.0	0.05	0.05
RASCAL 4.3.1 Saturated	1.0	0.5	0.15
RASCAL 4.3 Subcooled	1.0	0.01	0.01
RASCAL 4.3.1 Subcooled	1.0	0.01	0.01

RASCAL 4.3.1 incorporates saturated suppression pool reduction factors of 1.0 for noble gases, 0.5 for halogens, and 0.15 for other particles for the LTSBO scenario at Mark I BWRs only, to account for existing differences between RASCAL and SOARCA in modelling fission product behavior in the suppression pool and drywell at BWRs with Mark I containments.

These modified RASCAL 4.3.1 suppression pool reduction factors for the LTSBO feature account for additional releases from the revaporization of fission products which bypass the

suppression pool. This represents an overall “effective” reduction factor for BWRs with a Mark I containment, and is not necessarily applicable to other BWR designs. Additionally, these modified reduction factors should not be used for the “LOCA (NUREG-1465)” source term option in RASCAL 4.3.1, other computer codes, or cited as references for suppression pool reduction factors.

2.2 Loss of Coolant Accident (LOCA) Core Release Fractions

NUREG-1465 defines reactor release fractions for BWR and PWR LOCAs for 4 release phases: gap activity release, early in-vessel release, ex-vessel release, and late in-vessel release. RASCAL 4.3 and earlier versions of the code use the release fractions for the gap release, the early in-vessel release, and the ex-vessel release phases to define the release of core activity into the containment. These previous versions of the RASCAL computer code do not include the late in-vessel release phase. However, RASCAL 4.3.1 includes the activity released from the late in-vessel phase in LOCA source terms.

The durations of each phase including the late in-vessel phase are defined in NUREG-1465 Table 3.6. The late in-vessel release phase commences simultaneously with the occurrence of the ex-vessel release phase and extends several hours after the end of the ex-vessel phase release. Table 3.8, of NUREG-1465, classifies radionuclides into eight radionuclide groups based on expected release characteristics. Core release fractions to containment for each group by release phase for BWRs and PWRs are listed in Tables 3.12 and 3.13 of NUREG-1465.

The only radionuclide groups that have releases during the late in-vessel release phase are the halogens (iodine and bromine group), the alkali metals (cesium and rubidium group), and the tellurium group (tellurium, antimony, and selenium). In RASCAL 4.3.1 the core release fractions for these groups for the ex-vessel phase have been increased to include the portion of the late in-vessel release phase that occurs during the time of the ex-vessel release phase, and the late in-vessel release fractions have been decreased to account for the activity released during the ex-vessel release phase. Tables 2-6 and 2-7 present the core release fractions used by RASCAL 4.3.1 for the eight NUREG-1465 groups for PWRs and BWRs respectively. These tables include the revised release fractions for the ex-vessel and late in-vessel phases.

Table 2-6 RASCAL 4.3.1 PWR LOCA release phase duration and core release fractions

	Cladding	Core Melt	Ex-Vessel	Late In-Vessel
Duration (h)	0.5	1.3	2.0	8.0
Core Release Fraction				
Noble Gases (Xe, Kr)	0.05	0.95	0.00	0.00
Halogens (I, Br)	0.05	0.35	0.27	0.08
Alkali Metals (Cs, Rb)	0.05	0.25	0.37	0.08
Tellurium Group (Te, Sb, Se)	0.00	0.05	0.251	0.004
Barium/Strontium Group (Ba, Sr)	0.00	0.02	0.10	0.00
Noble Metals (Ru, Rh, Mo, Tc, Co)	0.00	0.0025	0.0025	0.00
Lanthanides (La, Zr, Cm, Am)	0.00	0.0002	0.005	0.00
Cerium Group (Ce, Pu, Np)	0.00	0.0005	0.005	0.00

Table 2-7 RASCAL 4.3.1 BWR LOCA release phase duration and core release fractions

	Cladding	Core Melt	Ex-Vessel	Late In-Vessel
Duration (h)	0.5	1.5	3.0	7.0
Core Release Fraction				
Noble Gases (Xe, Kr)	0.05	0.95	0.00	0.00
Halogens (I, Br)	0.05	0.25	0.303	0.007
Alkali Metals (Cs, Rb)	0.05	0.20	0.353	0.007
Tellurium Group (Te, Sb, Se)	0.00	0.05	0.2515	0.0035
Barium/Strontium Group (Ba, Sr)	0.00	0.020	0.10	0.00
Noble Metals (Ru, Rh, Mo, Tc, Co)	0.00	0.0025	0.0025	0.00
Lanthanides (La, Zr, Cm, Am)	0.00	0.0002	0.005	0.00
Cerium Group (Ce, Pu, Np)	0.00	0.0005	0.005	0.00

Note that in RASCAL 4.3.1, the only groups that have late in-vessel release fractions are the halogens (iodine and bromine), the alkali metals (cesium and rubidium), and the tellurium group (tellurium, antimony, and selenium). Core release fractions for the other groups are the same in RASCAL 4.3.1 as they were in previous versions of the RASCAL computer code.

2.3 Core Damage State

The "core damage state" option from RASCAL 3.0.5 has been moved in RASCAL 4.3. It is now an option for both the LTSBO and LOCA source terms. In this current version of the RASCAL computer code, the "percent damage" option is essentially equivalent to selecting the "core recovered" option and specifying a time that the core is recovered with the exception that the "percent damage" option does not require that the user specify a time for the core recovery. When the "core damage state" option is selected, RASCAL calculates the duration that the core is uncovered and adds the core uncovered duration to the time core is initially uncovered to determine when to recover the core.

The relationships among core damage phase, percent damage, and effective time of core recovery for LTSBOs and LOCAs for both BWRs and PWRs are based on event progression descriptions in NUREG/CR-7110 Volume 1, Revision 1; NUREG-7110 Volume 2, Revision 1; and NUREG-1465. Tables 2-8 through 2-11 give the relationships used by RASCAL 4.3.1 in estimating effective core recovery times.

Table 2-8 LTSBO – BWR damage state and effective core recovery times

Core Damage Phase	Percent Damage (%)	Core Recovery Time (min)[a]
Cladding Failure	0 to 100	0 to 30[b]
Core Melt	0 to 100	30 to 120[b]
Ex-Vessel (Vessel Melt)	NA	360[c]

a. Times are minutes after the start of core damage. The RASCAL interface will convert the relative core recovery time to date and time.
b. Linear interpolation of time as function of percent is appropriate.
c. Ex-vessel release is evaluated at a single specific time.

Table 2-9 LTSBO – PWR damage state and effective core recovery times

Core Damage Phase	Percent Damage (%)	Core Recovery Time (min)[a]
Cladding Failure	0 to 100	0 to 90[b]
Core Melt	0 to 100	90 to 300[c]
Ex-Vessel (Vessel Melt)	NA	360[d]

a. Times are minutes after the start of core damage. The RASCAL interface will convert the relative core recovery time to date and time.
b. Linear interpolation of time as function of percent is appropriate.
c. Linear interpolation of time as function of percent should be done in two segments. Use segment 1 for 0 to 75 percent with corresponding times of 90 to 240 minutes, and use segment 2 for 75 percent to 100 percent with corresponding times of 240 to 300 minutes.
d. Ex-vessel release is evaluated at a single specific time.

Table 2-10 LOCA – BWR damage state effective core recovery times

Core Damage Phase	Percent Damage (%)	Core Recovery Time (min)[a]
Cladding Failure	0 to 100	0 to 30[b]
Core Melt	0 to 100	30 to 120[b]
Ex-Vessel (Vessel Melt)	NA	360[c]

a. Times are minutes after the start of core damage. The RASCAL interface will convert the relative core recovery time to date and time.
b. Linear interpolation of time as function of percent is appropriate.
c. Ex-vessel release is evaluated at a single specific time.

Table 2-11 LOCA – PWR damage state effective core recovery times

Core Damage Phase	Percent Damage (%)	Core Recovery Time (min)[a]
Cladding Failure	0 to 100	0 to 30[b]
Core Melt	0 to 100	30 to 105[b]
Ex-Vessel (Vessel Melt)	NA	240[c]

a. Times are minutes after the start of core damage. The RASCAL interface will convert the relative core recovery time to date and time.
b. Linear interpolation of time as function of percent is appropriate.
c. Ex-vessel release is evaluated at a single specific time.

Note that no option is provided for percent damage for the ex-vessel release phase. The source term for this phase is based on damage that occurs prior to a fixed time in the middle of the phase duration.

2.4 Coolant Releases

In RASCAL 4.2 and earlier versions, activity entering the RCS was mixed in cooling system water and released to the environment as the coolant was released. Depending on the rate of transfer of activity from the reactor to the RCS and coolant release rate from the RCS, the activity in the RCS increased and decreased as the event progressed. Isotopes of noble gases and other elements were treated the same.

In RASCAL 4.3, noble gases entering the RCS of a BWR are immediately vented to the environment; they do not accumulate in the RCS. This change makes RASCAL consistent with BWR designs in which non-condensable gases are stripped from the steam by the air ejectors. Other radionuclides are accumulated in the RCS as in previous versions of RASCAL. Noble gases entering the primary system of a PWR are accumulated as are other radionuclides, but noble gases leaking from the primary system to the secondary system are vented to the environment immediately through the air ejectors.

These changes in the treatment of noble gases alter the timing of noble gas releases to the environment for those reactor accident release paths that involve the RCS. Noble gases are released earlier in the accident sequence in RASCAL 4.3 than they were in previous versions of RASCAL. Also, because the noble gases aren't accumulated in either the BWR or PWR RCS, the total noble gas release in RASCAL 4.3 is larger than it is in RASCAL 4.2. Figure 2-7 illustrates the differences in noble gas release for a PWR LOCA with bypass release calculated using RASCAL default parameters.

Noble gases are significant contributors to the cloud shine dose. As a result, the change in the treatment of noble gas releases is reflected in cloud shine doses. Figure 2-8 shows the cumulative cloud shine dose at 1 mile (mi) as a function of time for a PWR LOCA Bypass accident for RASCAL 4.2 and RASCAL 4.3. The left panel shows the RASCAL 4.2 dose, and the right panel shows the RASCAL 4.3 dose.

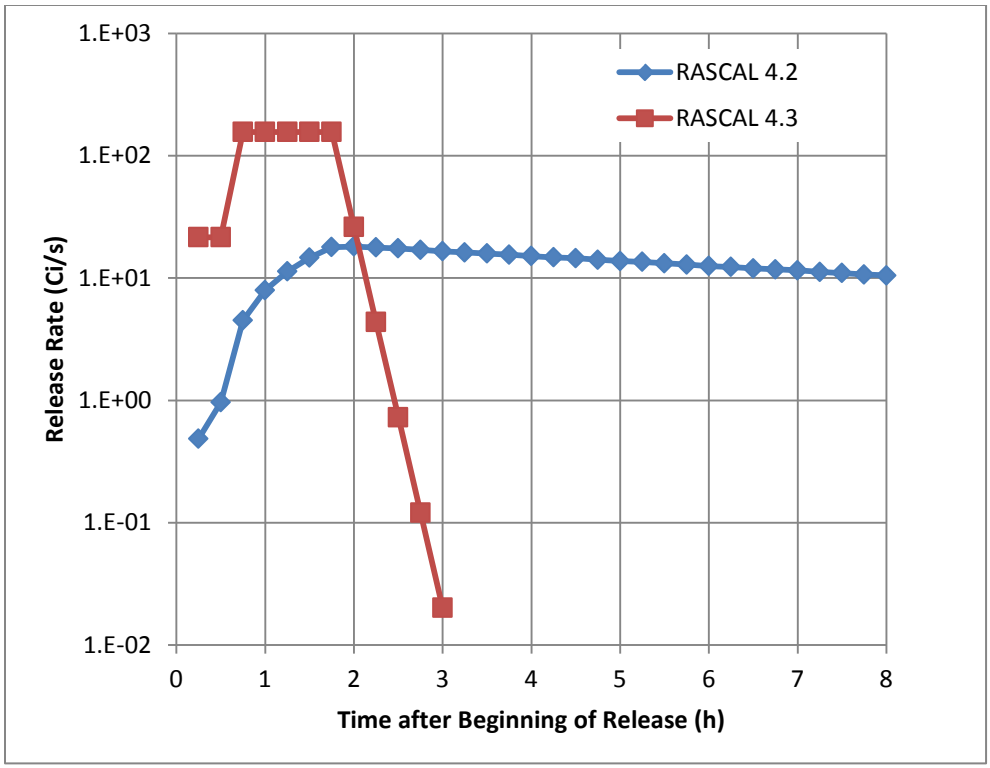


Figure 2-7 Kr-85 release rate comparison for a PWR LOCA with a bypass release

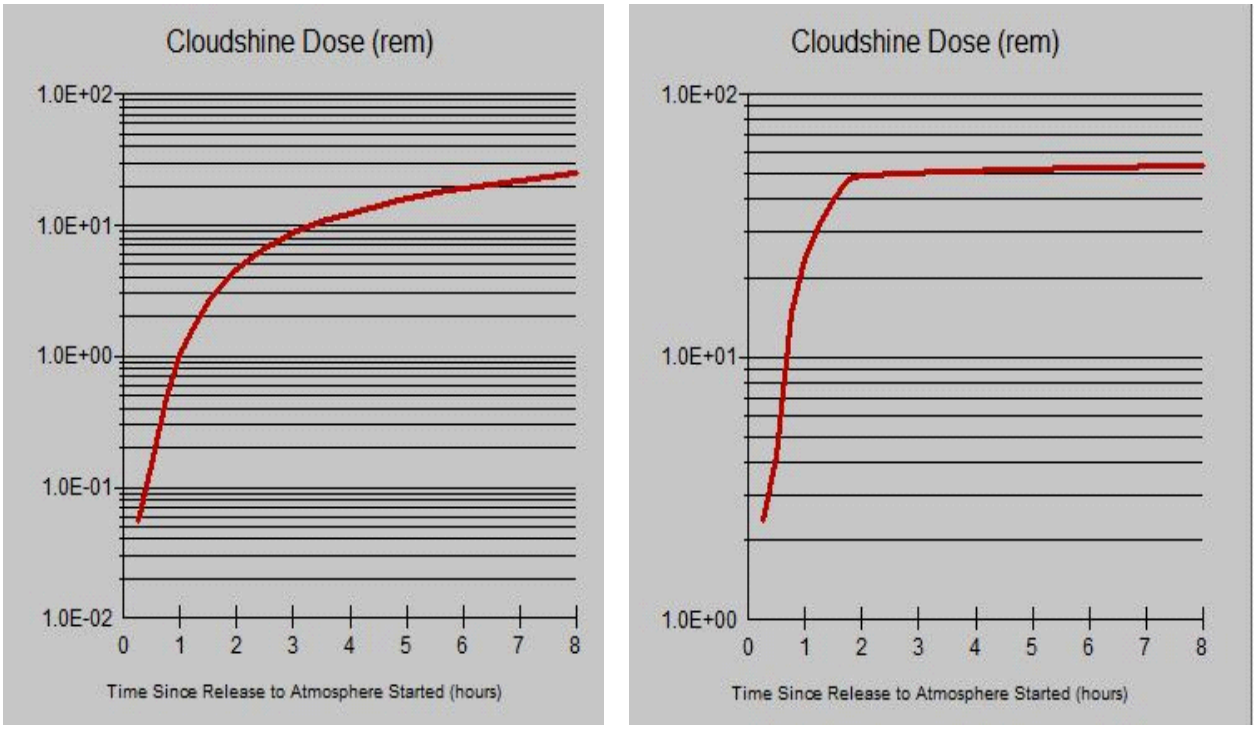


Figure 2-8 Comparison of RASCAL 4.2 (left) and RASCAL 4.3 (right) cloud shine dose accumulation

In RASCAL 4.2 there was no decay or ingrowth of activity within the RCS. The treatment of activity entering the RCS has been revised in RASCAL 4.3 to include decay and daughter ingrowth. This change results in a reduction of release of short-lived radionuclides such as I-132, I-133, I-134, and I-135 to the environment compared with previous versions of RASCAL. Table 2-12 shows the changes in iodine activity released in the environment for a LOCA/bypass accident that is associated with the addition of decay and ingrowth in the RCS. The total iodine release in RASCAL 4.3 is reduced to about 75 percent of the RASCAL 4.2 total iodine release.

Table 2-12 Comparison of RASCAL 4.2 and RASCAL 4.3 iodine releases for a PWR LOCA/bypass accident

Radionuclide	Half Life (h)	Dose Coefficients (Sv/Bq)	RASCAL 4.2 (Ci)	RASCAL 4.3 (Ci)	Fraction
I-131	193.0	8.89E-09	1.10E+05	1.09E+05	0.991
I-132	2.3	1.03E-10	1.74E+05	1.17E+05	0.673
I-133	20.8	1.58E-09	2.13E+05	1.94E+05	0.912
I-134	0.9	3.55E-11	9.78E+04	2.36E+04	0.241
I-135	6.6	3.32E-10	1.84E+05	1.39E+05	0.757
Total			7.77E+05	5.82E+05	0.749

There are significant reductions in the releases of I-134 and I-132, while the release of I-131 is not reduced significantly. Note that the I-131 and I-132 activities in Table 2-6 include ingrowth from decay of tellurium parents.

In contrast to the effect of increasing the release of noble gas on the cloud shine dose, the decrease in iodine releases from RASCAL 4.2 to RASCAL 4.3 has relatively little effect. The thyroid committed effective dose equivalent (CEDE) at 1 mi calculated by RASCAL 4.2 for a PWR LOCA/Bypass accident is 141 roentgen equivalent man (rem). The thyroid CEDE at 1 mi calculated by RASCAL 4.3 for the same accident is 136 rem. Even though the total iodine release decreased by 25 percent, the thyroid CEDE decreased by less than 4 percent. The relative insensitivity of the thyroid CEDE to reduction in total iodine released is explained by the fact that most of the reduction in the iodine release is associated with radionuclides having relatively small dose coefficients. The release of I-131, which has the largest iodine dose coefficients, is essentially unchanged from RASCAL 4.2 to RASCAL 4.3.

Daughter ingrowth in the RCS does increase the release of some radionuclides; for example, the Xe-135m and Xe-135 daughters of I-135. The immediate release of noble gases from the steam generator in RASCAL 4.3 increases the noble gas released to the environment by about a factor of 2.1. The ingrowth of Xe-135m and Xe-135 from decay of I-135 in the primary and secondary systems increases the releases of the radionuclides by approximately an additional 35 percent and 10 percent, respectively. Noble gases originating in the secondary system from ingrowth are immediately vented to the environment.

2.5 Pressure-Hole Size Release Model

Section 1.5.2 of NUREG-1940 describes the RASCAL option for estimating containment release rates based on containment pressure and hole size. As noted at the end of the section, RASCAL 4.2 does not calculate the change in containment pressure as gases leave the containment; it is up to the user to enter changing containment pressure as the consequence assessment proceeds. The algorithm implementing the option to estimate release rates based on containment pressure and hole size was revised in RASCAL 4.3. The code adjusts the containment pressure to account for the gases leaving the containment. This section describes the assumptions and methods RASCAL 4.3 uses in calculating the release rate. Additionally, RASCAL 4.3.1 corrects a coding error in the conversion of pressures in the containment pressure and hole size method in RASCAL 4.3.

As a starting point for calculating releases from containment pressure and hole size, RASCAL 4.3 assumes that the containment pressure prior to the initiating event is equal to atmospheric pressure of 14.7 pounds per square inch absolute (psia), with a temperature of 68 degrees Fahrenheit and that the containment atmosphere is essentially dry air. An initial hole size and containment pressure should be entered for the time of beginning of release. If the hole size is 0.01 square centimeters (cm^2) (0.0015 square inches (in^2)) or less, RASCAL 4.3.1 assumes that the containment leak rate is equal to the containment design leak rate. If the hole size is 0.01 cm^2 or larger, RASCAL uses equation 1-12 from NUREG-1940 to calculate the mass flow from the containment to the environment. Note that this equation assumes that the area of the hole is small compared to the surface area of the containment and it represents steady-state leakage of a non-compressible fluid from a pipe. Conceptually this is not an appropriate representation of the containment leak rate under pressure; however, it does provide initial leak rate estimates that are not significantly different than those that would be obtained from a more representative containment model.

When RASCAL 4.3 determines that the release rate is to be calculated by the containment pressure and hole size method, it checks to see if the increase in containment temperature is sufficient to account for the increase in containment pressure. If it is, RASCAL 4.3 assumes that the containment atmosphere consists of essentially dry air, and uses the density of dry air in the calculation of containment leak rate. If the increase in temperature is not sufficient to account for the increase in containment pressure, RASCAL 4.3 assumes that the increase in pressure is the result of a major leak in the RCS. In this case, RASCAL 4.3 calculates the containment atmosphere density assuming that the water in the RCS has been drained to containment and has evaporated. RASCAL 4.3 also adds the activity in the RCS to the activity in containment.

RASCAL 4.3.1 calculates the initial leak rate based on containment size, hole size, and containment pressure. Table 2-13 lists reactor and containment parameters of four representative PWR reactor containment designs. The equivalent design hole is a hole that would give the design leak rate at the design pressure assuming the RASCAL 4.3.1 pressure hole model. Figure 2-9 shows the initial containment leak rates as a function of effective hole size for the representative PWR containment designs, assuming that the containment is initially at design pressure. The equivalent design holes in Table 2-13 may be estimated directly from

Figure 2-9. Containment design parameters, with the exception of the equivalent design hole size (Table 2-13), are included for each reactor in the United States RASCAL facility database file. The containment design parameters are listed under the reactor parameters portion of the user interface case summary. Note that the term effective hole size is used because the hole in the model is assumed to be circular in shape, but the actual hole shape during an event may not be circular. Additionally, the leak rate may be smaller than the design leak rate if the pressure is less than the design pressure.

Table 2-13 Containment parameters for four representative PWR containment designs

PWR	Containment Type	Rated Power (MWt)	Containment Volume (ft^3)	Design Pressure (psig)	Design Leak Rate (%/day)	Equivalent Design Hole (cm^2)
Braidwood Unit 1	Dry Ambient	3586	2.90E+06	61	0.1	0.1
Watts Bar Unit 1	Ice Condenser	3459	9.20E+05	15	0.5	0.5
Prairie Island Unit 1	Dry Ambient	1677	1.30E+06	41	0.5	0.3
Beaver Valley Unit 1	Subatmospheric	2900	1.80E+06	54	0.1	0.07

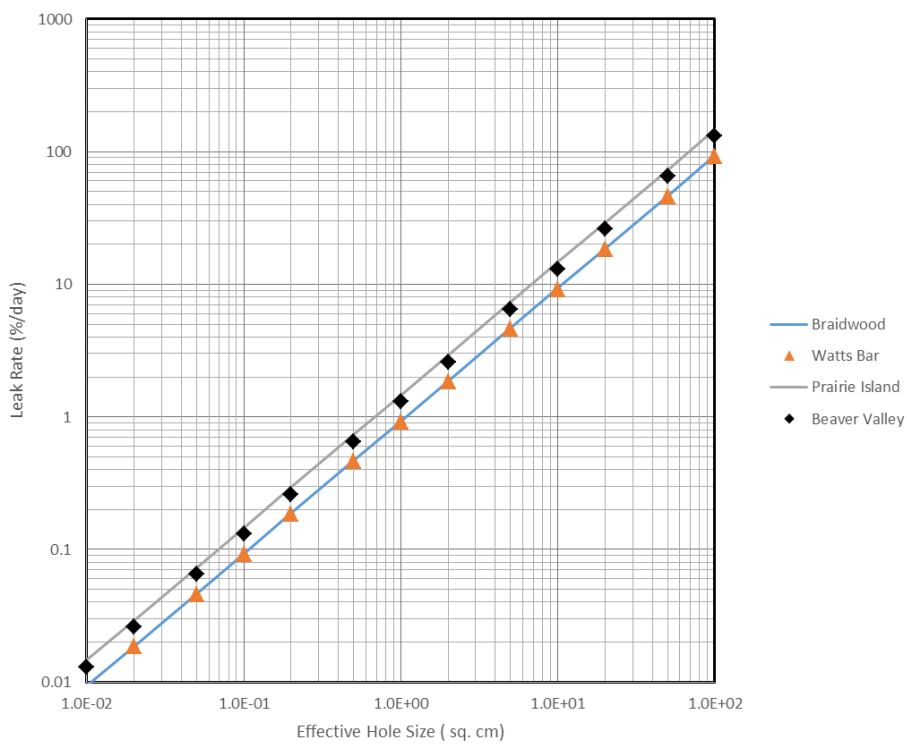


Figure 2-9 Initial containment leak rates of representative PWR containment designs as a function of effective hole size

As the containment leaks to the environment RASCAL 4.3.1 updates the containment pressure and leak rate to account for the decrease in mass of the containment atmosphere. Figure 2-10 shows the decrease in containment pressure for each of the four representative reactors. In each case the initial pressure is the containment design pressure.

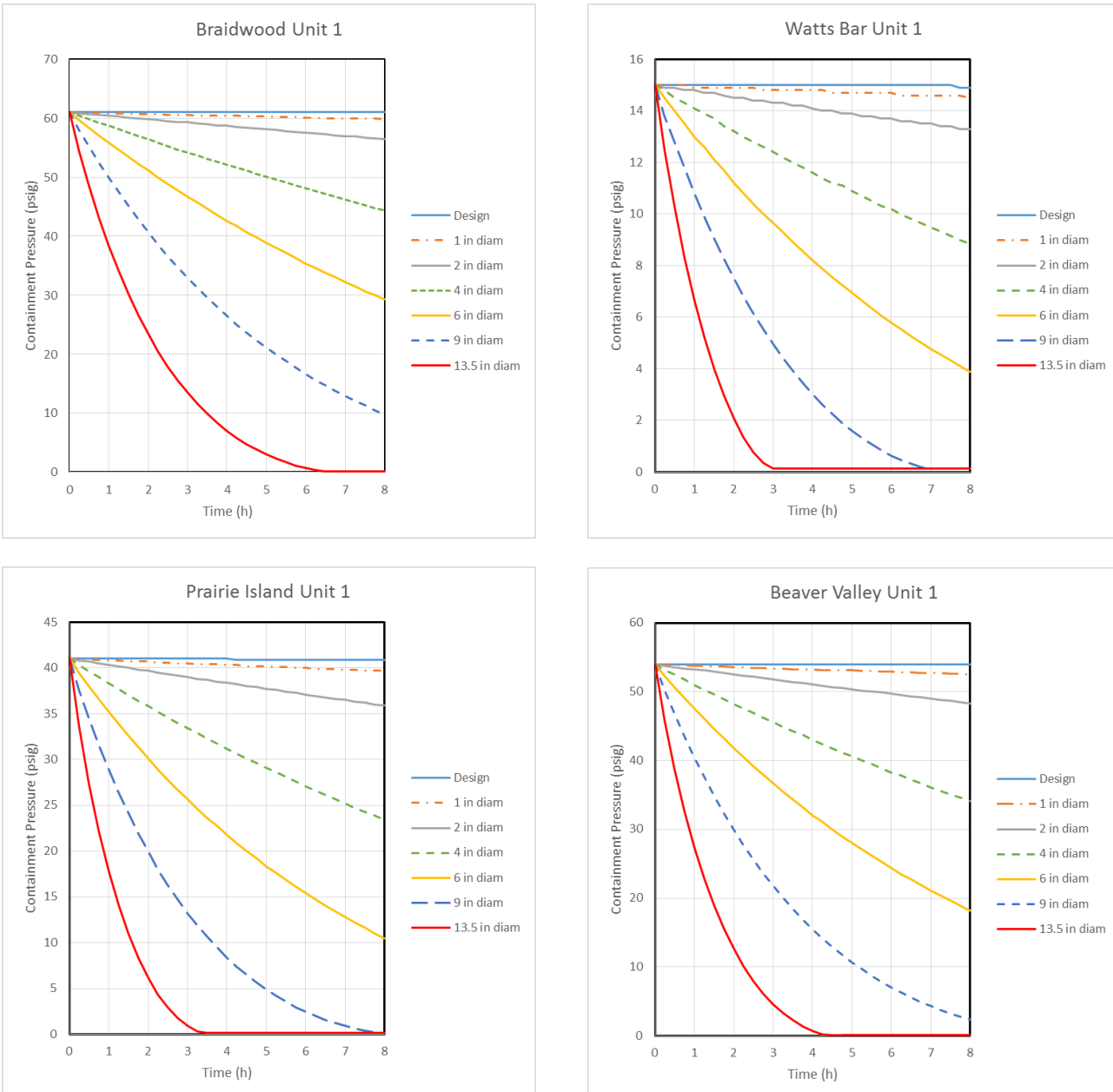


Figure 2-10 Pressure decrease with time for various effective hole sizes for representative PWR containment designs

Table 2-14 lists reactor and containment parameters of three representative BWR reactor containment designs. Figure 2-11 shows the initial containment leak rates as a function of effective hole size for the representative BWRs containment designs, assuming that the

containment is initially at design pressure. Figure 2-12 shows the decrease in containment pressure for each of the four representative reactors. In each case the initial pressure is the containment design pressure.

Table 2-14 Containment parameters for three representative BWR containment designs

BWR	Containment Type	Rated Power (MWt)	Containment Volume (ft^3)	Design Pressure (psig)	Design Leak Rate (%/day)	Equivalent Design Hole (cm^2)
Quad Cities Unit 1	Mark I	2957	2.70E+05	56	0.5	0.1
LaSalle Unit 1	Mark II	3546	3.74E+05	45	0.5	0.15
Perry Unit 1	Mark III	3758	1.08E+06	15	0.2	0.2

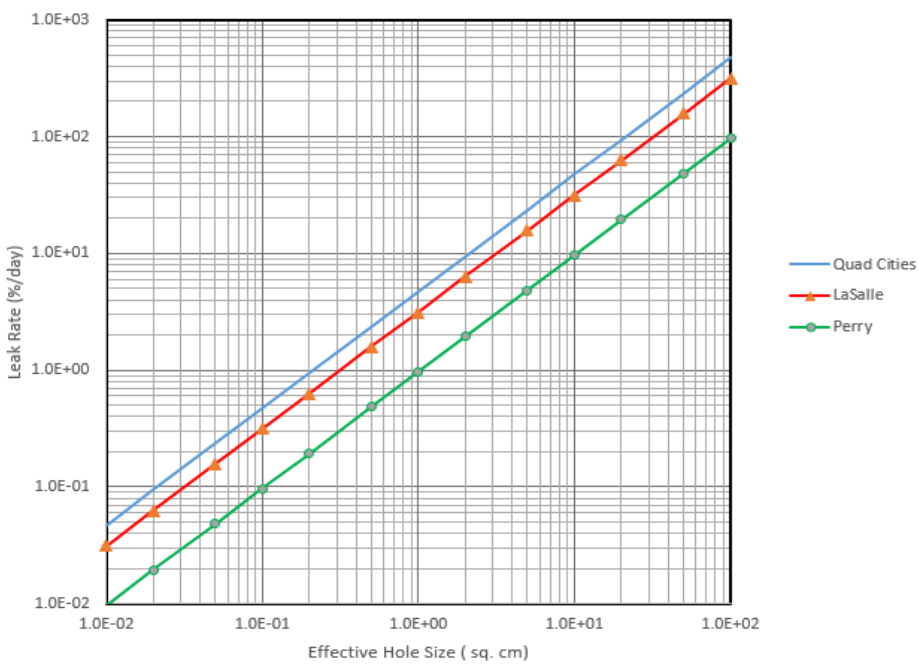


Figure 2-11 Pressure decrease with time for various effective hole sizes for representative BWR containment designs

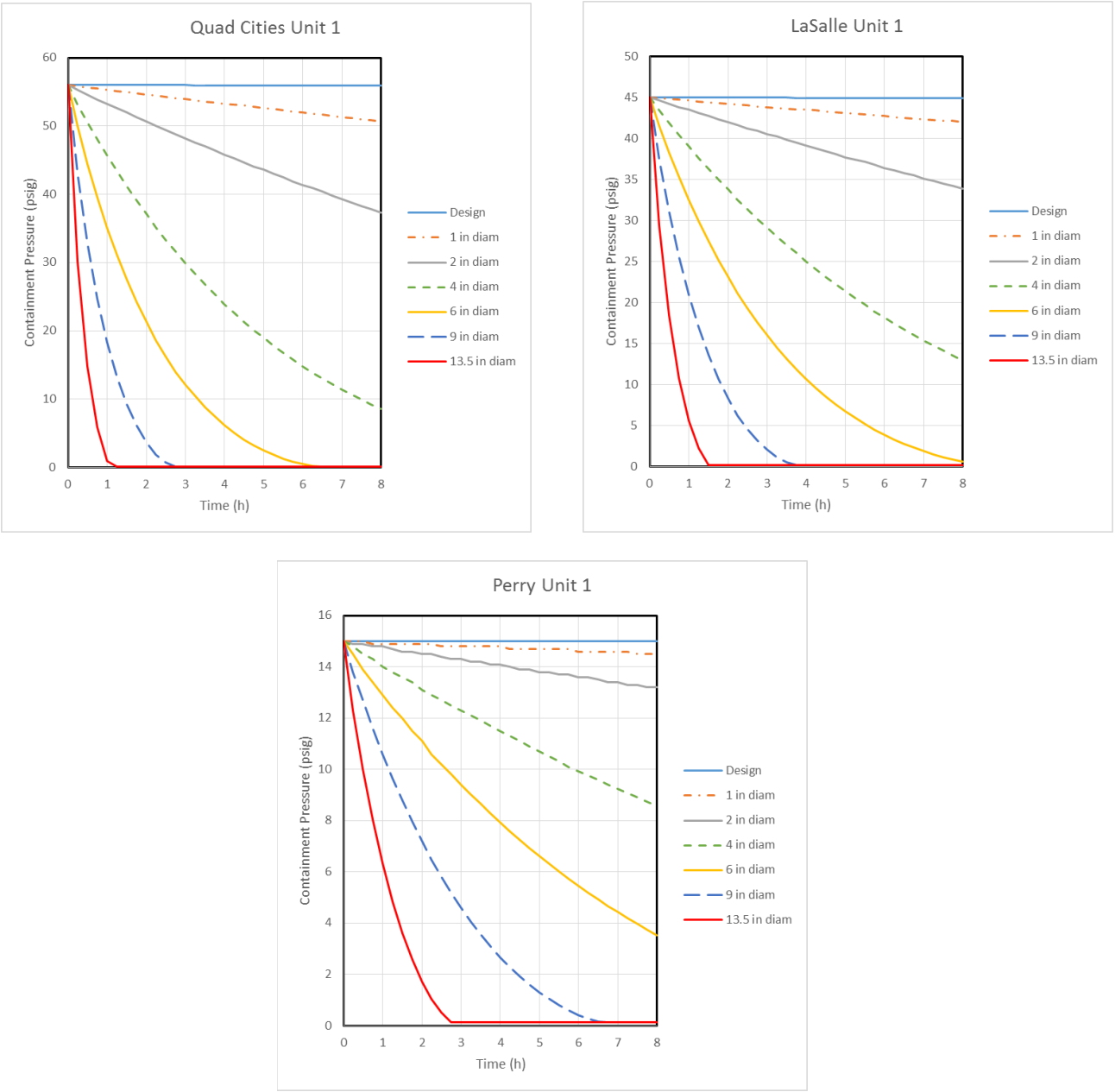


Figure 2-12 Pressure decrease with time for various effective hole sizes for representative BWR containment designs

In reality, the pressure should decrease to atmospheric pressure. However, RASCAL assumes that the containment pressure will only decrease to a pressure that is one percent above atmospheric pressure. This 1 percent residual pressure differential is used to provide leakage to the environment through the open hole. If containment integrity is reestablished, the leak rate may be reset to the original leak rate by resetting the hole size.

In calculating the release through a hole, RASCAL updates the containment pressure and leak rate every 15 minutes using the leak rate at the beginning of the 15 minute period. This procedure causes an overestimation of the release rate in the early release periods if the leak

rates are high. If the release period is long enough, the overestimation of release rates will be followed by period of underestimated release rates that will ultimately compensate for the overestimated early releases. The magnitude of the overestimation of early release rates is a function of the hole size.

At this point, it should be noted that the RASCAL 4.3 approach to estimating releases via the containment pressure-hole size release path does not consider all of the factors that may alter the release rate through the hole. For example, the approach does not consider any decrease in pressure that may be associated with condensation of the reactor coolant or a decrease in temperature of the containment atmosphere. These factors are likely to reduce the containment atmosphere pressure and the release rate. Similarly, RASCAL 4.3 does not consider any potential heating of the containment atmosphere by the reactor. Heating of the containment atmosphere could increase the containment pressure and release rate.

RASCAL does not include any of the physics that would be required to predict increases in containment pressure as an event progresses. It is up to the user to enter containment pressures as they increase. It is also important to enter containment pressures as the pressures decrease because estimates of hole size are likely to be crude at best. RASCAL reinitializes its pressure calculation each time a containment pressure is entered.

If the leak rate from containment is small (design leak rate or less), containment pressures may be entered less frequently than when the leak rate is large. If there isn't containment damage or venting, it is likely to be sufficient to enter the containment pressure every hour or two. When there is venting or containment damage, pressures should be entered more frequently.

2.6 Containment Radiation Monitor Source Term

Section 1.2.4 of NUREG-1940 describes the RASCAL option for estimating source terms using containment radiation monitor readings. The containment radiation monitor source term calculation in RASCAL 4.3 is based on Method A.4 of NUREG/BR-0150, "Response Technical Manual (RTM)-96" [9]. Figures A-5 through A-12 of NUREG/BR-0150 show estimates of containment monitor readings for core damage states ranging from 1 to 100 percent for in-vessel core melt, gap releases, and coolant releases for PWR and BWR reactors for 1 and 24 hours after reactor shutdown. Figures 1-1 through 1-5 in NUREG-1940 are based on these figures in NUREG/BR-0150.

The RASCAL implementation of the method described in NUREG/BR-0150 scales the containment monitor reading entered by the user to account for the difference between the 3,000 MWt reactor power used to produce the figure and the actual reactor power. Equation 1-4 of NUREG-1940 is used to calculate this scaled containment monitor value. This method of scaling the containment monitor reading based upon reactor power is not part of Method A.4 of NUREG/BR-0150. This scaling of the containment radiation monitor reading to reactor power has been deleted from RASCAL 4.3.1. The effect of deleting this scaling of the containment radiation monitor reading to reactor power has resulted in a source term that is more sensitive to differences in reactor designs in RASCAL 4.3.1.

RASCAL 4.3.1 also includes a revision to the procedure for interpolation/extrapolation of the damage ranges as a function of the time after reactor shutdown. Although the interpolation is not explicitly discussed in NUREG/BR-0150, it is based on the data in Figures A-5 through A-12 of NUREG/BR-0150.

The RASCAL 4.3 interpolation/extrapolation procedure is as follows:

- For times between shutdown and the containment radiation monitor reading of zero to 1 hour, the containment radiation monitor reading corresponding to 1-percent damage is equal to the monitor reading shown in NUREG/BR-0150 figures for 1 hour.
- For times greater than 24 hours, the containment radiation monitor reading corresponding to 1-percent damage is equal to the containment radiation monitor reading shown in NUREG/BR-0150 figures for 24 hours.
- For times between 1 and 24 hours, the containment radiation monitor reading corresponding to 1-percent damage is estimated by linear interpolation between the monitor readings shown in NUREG/BR-0150 figures for 1 hour and 24 hours.

The solid blue line in Figure 2-13 shows the RASCAL 4.3 estimates of the 1-percent damage containment radiation monitor reading for a PWR reactor with cladding failure.

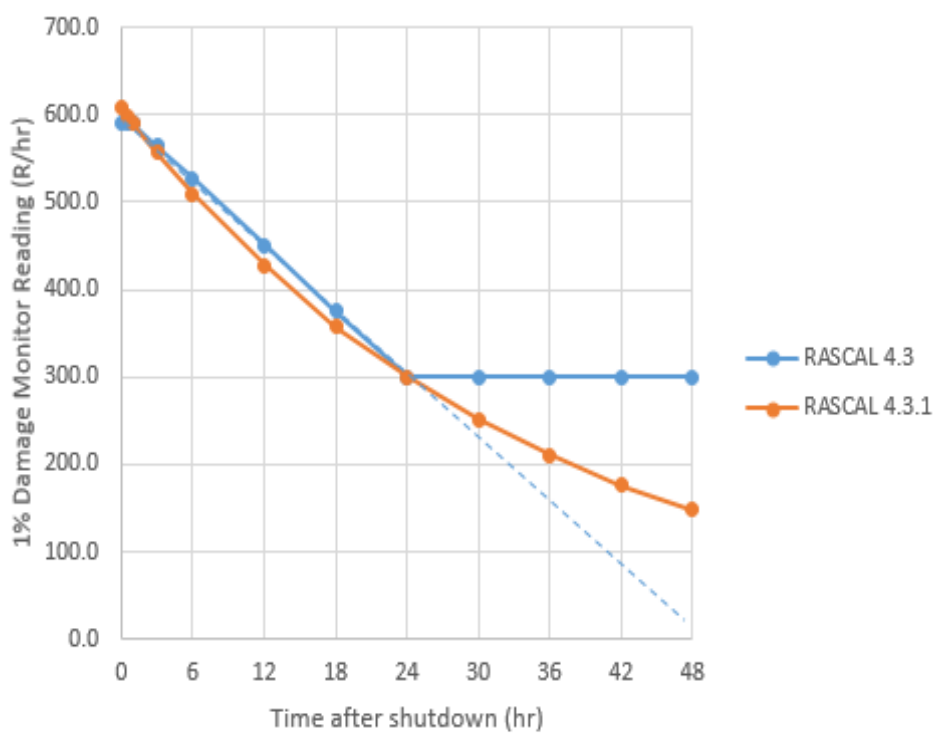


Figure 2-13 RASCAL estimates of 1-percent damage containment radiation monitor readings

As indicated in Figure 2-13, the RASCAL 4.3 interpolation/extrapolation procedure provides reasonable estimates of the containment radiation monitor readings associated with 1-percent damage for times between shutdown and 24 hours. However as indicated in Figure 2-13, for times longer than 24 hours, the RASCAL 4.3 interpolation/extrapolation procedure is likely to overestimate the 1-percent damage containment radiation monitor reading. This overestimate occurs because the activity associated with one percent damage will continue to decrease because of radioactive decay.

RASCAL 4.3 calculates the percentage of damage associated with the actual containment radiation monitor reading by dividing actual containment radiation monitor reading by the estimated monitor reading for 1-percent core damage. Consequently, an overestimate of the 1-percent damage containment radiation monitor reading will result in an underestimate of the reactor damage and ultimately potential radiation doses. Additionally, if this procedure were revised to include timeframes greater than 24 hours, the extrapolation of the linear decrease (Figure 2-13 dashed blue line) leads to a zero monitor reading at about 48 hours and unreal negative values for times greater than 48 hours. Therefore, the RASCAL 4.3 interpolation/extrapolation procedure needed to be revised to adequately address scenarios in which the time between reactor shutdown and damage may be significantly longer than 24 hours (e.g., Fukushima Units 2 and 3).

In RASCAL 4.3.1, an exponential decay function was fitted to the 1 hour and 24 hour 1-percent damage containment monitor readings as an alternative to the RASCAL 4.3 linear interpolation/extrapolation procedure. In this RASCAL 4.3.1 method, the 1 hour and 24 hour 1-percent damage containment radiation monitor readings would be used to estimate the parameters a and b in Equation 2-1:

$$1\%\,MR = ae^{-bt} \tag{2-1}$$

where

1% MR	= one percent damage containment monitor reading entered in (R/h)
t	= time between reactor shutdown and the monitor reading (h), and
a and *b*	= derived from the 1 hour and 24 hour 1-percent damage containment radiation monitor readings included with Method A-4 of NUREG/BR-0150 (listed in Table 2-15).

Table 2-15 Containment radiation monitor reading interpolation/extrapolation parameter values

Reactor Type	Path	Condition	a	b
PWR	no spray	Normal Coolant	1.11E-02	1.00E-01
PWR	no spray	Spiked Coolant	5.48E-01	9.22E-02
PWR	no spray	Cladding Failure	6.08E+02	2.94E-02
PWR	no spray	Core Melt	5.86E+03	4.48E-02
PWR	spray	Normal Coolant	6.52E-03	8.25E-02
PWR	spray	Spiked Coolant	2.21E-02	1.00E-01
PWR	spray	Cladding Failure	1.41E+02	8.14E-02
PWR	spray	Core Melt	2.51E+03	8.86E-02
BWR Mark I&II	dry well, no spray	Normal Coolant	1.11E-03	1.00E-01
BWR Mark I&II	dry well, no spray	Spiked Coolant	1.00E-01	1.05E-01
BWR Mark I&II	dry well, no spray	Cladding Failure	1.16E+03	5.65E-02
BWR Mark I&II	dry well, no spray	Core Melt	1.05E+04	5.23E-02
BWR Mark I&II	dry well with spray	Normal Coolant	1.05E-04	5.23E-02
BWR Mark I&II	dry well with spray	Spiked Coolant	3.37E-03	1.18E-01
BWR Mark I&II	dry well with spray	Cladding Failure	2.74E+02	9.22E-02
BWR Mark I&II	dry well with spray	Core Melt	4.72E+03	9.36E-02
BWR Mark I&II	wet well	Normal Coolant	1.00E-05	0.00E+00
BWR Mark I&II	wet well	Spiked Coolant	1.00E-03	0.00E+00
BWR Mark I&II	wet well	Cladding Failure	1.09E+02	9.97E-02
BWR Mark I&II	wet well	Core Melt	1.98E+03	9.55E-02

Reactor Type	Path	Condition	a	b
BWR Mark III	dry well, no spray	Normal Coolant	4.56E-04	1.30E-01
BWR Mark III	dry well, no spray	Spiked Coolant	1.57E-02	1.15E-01
BWR Mark III	dry well, no spray	Cladding Failure	2.17E+02	1.33E-01
BWR Mark III	dry well, no spray	Core Melt	1.86E+03	3.53E-02
BWR Mark III	dry well with spray	Normal Coolant	3.18E-05	5.75E-02
BWR Mark III	dry well with spray	Spiked Coolant	4.38E-04	9.04E-02
BWR Mark III	dry well with spray	Cladding Failure	4.40E+01	7.10E-02
BWR Mark III	dry well with spray	Core Melt	7.62E+02	8.46E-02
BWR Mark III	wet well	Normal Coolant	1.05E-04	5.23E-02
BWR Mark III	wet well	Spiked Coolant	1.04E-03	3.98E-02
BWR Mark III	wet well	Cladding Failure	2.63E+02	9.04E-02
BWR Mark III	wet well	Core Melt	4.85E+03	7.51E-02

The RASCAL 4.3.1 curve, the orange line in Figure 2-13, demonstrates the predicted 1-percent damage containment radiation monitor readings for cladding failure in a PWR. This procedure provides 1-percent damage containment radiation monitor damage estimates that are about the same as the RASCAL 4.3 one percent damage containment radiation monitor damage estimates for timeframes up to 24 hours. For timeframes greater than 24 hours, this procedure provides 1-percent damage containment radiation monitor damage readings that continue to decrease, but do not become zero. Additionally, as indicated in Figure 2-13, the 1-percent damage containment radiation monitor readings at 48 hours is about twice as large as the RASCAL 4.3 method. Finally, if the RASCAL 4.3.1 method is extended to estimate the 1-percent damage containment radiation monitor readings at 96 hours, the value increases by about a factor of 8 over the RASCAL 4.3 method.

It is important to note that containment radiation monitor readings may be used to track increasing reactor damage by entering a sequence of containment radiation monitor readings. However, RASCAL will not reduce the damage estimate if one or more of the containment radiation monitor readings indicate that damage is not as great as indicated by a previous containment radiation monitor reading. RASCAL always uses the highest damage estimate when damage is estimated from containment radiation monitor readings.

2.7 Effluent Release by Mixtures Source Term

There have not been any changes to the effluent release by mixtures source terms in RASCAL 4.3 or RASCAL 4.3.1. However, there have been a number of questions related to these methods that indicate that additional discussion is warranted.

When using this option in RASCAL to calculate a source term the user assumes full responsibility for specifying the monitored mixture source term. RASCAL assumes that the monitor mixture data are the independent measurements of noble gases, iodine concentrations, and particles. RASCAL assumes that the monitored mixture particles consist of cesium iodide (CsI) and, given this assumption, the activity in particles would be more than 99.5 percent if the monitor reading were taken in the first hour after reactor shutdown. Section 1.2.8 of NUREG-1940 discusses the particle radionuclide composition. RASCAL adds the iodine in the particles to the iodine release rate specified by the user to get a total iodine release for use in determining potential consequences in the environment. If the monitors that give the estimates of iodine and particle release rates are in the same release path, it is likely that the iodine in the particles contributes to both the iodine release rate and the particle release rate. As a result, these monitor readings are independent and the total iodine release would be likely to be overestimated by as much as a factor of two.

When the iodine and particle release rates are from a monitor or monitors in the same release path, it is appropriate to use one release rate or the other. If near-term consequences (e.g., doses from plume passage) are of most interest, the larger of the release rates should be used. This is because the iodine release is likely to be more important than the cesium release. If long-term consequences (e.g., future land use) are of most interest, the particle release rate should be used because the cesium releases are likely to be more important. When in doubt of which reading to use, the user should refer to the importance output described in Section 7 of this supplement to determine which monitor reading to use.

2.8 Effluent Release Rate and Concentrations by Radionuclide Source Terms

There have not been any changes to the calculation of effluent release rate or concentration release rate methods of calculating source terms in RASCAL 4.3 or RASCAL 4.3.1. However, there have been a number of questions related to these methods that indicate that additional discussion is warranted.

When either of these two methods is selected, the user assumes full responsibility specifying the release rate. RASCAL neither evaluates the source term to determine whether it is reasonable or not, nor does the code adjust the source term for radioactive decay or daughter ingrowth, although decay and ingrowth are included in calculations once radionuclides are released to the atmosphere. RASCAL does not adjust the source term for depletion in the release path; therefore there is no reduction for deposition or filters.

If the source term entered by the user includes radionuclides that have daughters, which are not included in the source term entered by the user, RASCAL will add the daughter to the list of

radionuclides released. The release rates for daughters added by RASCAL will be zero, but RASCAL will grow in the daughters in the environment. As a result, the daughters will be included in the importance calculation.

2.9 Spent Fuel and Fuel Cycle Facility Accidents

RASCAL 4.3 features an increase in the modularization of the computer code to enhance the maintainability of the code. This increase includes the separation of the calculation of the spent fuel pool (SFP) and fuel cycle facility source terms from the calculation of the reactor source terms. The RASCAL 4.3 code package now has three source term modules; one for calculating reactor source terms, one for calculating spent fuel source terms, and one for calculating fuel-cycle facility source terms.

There are two components to the RASCAL 4.3 spent fuel accident source term calculation. The first is the calculation of the at-risk radionuclide inventory, and the second is the calculation of the portion of the at-risk radionuclide inventory that is released to the environment. The changes to the spent fuel accident source term calculation in RASCAL 4.3 are in the determination of the at-risk radionuclide inventory in the spent fuel. There have not been any significant changes in the calculation of the spent fuel or fuel cycle facility source terms related to release paths.

RASCAL 4.3 has two methods of estimating reactor fuel radionuclide inventories. The default method, based on average reactor power and end-of-life fuel burnup is described in Section 1.1.1 of NUREG-1940. A new method, referred to as the custom fuel option, was added in RASCAL 4.3. This method is described in detail in Section 5 of this supplement. The method tracks fuel batches through the reactor. Radionuclides build into the fuel in each as the batch is exposed and decay during reactor outages. When an outage occurs, each batch of fuel in the reactor has its own radionuclide inventory. The reactor core inventory is the sum of the inventories of the individual batches. The inventory in the assemblies in a batch of fully custom spent fuel is similar to the inventory in the assemblies calculated from the default inventory. The inventories of short-lived radionuclides (e.g., I-131) in assemblies of custom fuel at the end of its first or second cycle are similar to the inventories in assemblies using the default method because they are primarily functions of reactor power. In contrast, the assembly inventories of long-lived radionuclides (e.g., Cs-137) are significantly lower when the custom fuel option is used than they are when the assembly inventory is calculated using the default method. The difference in assembly inventories for long-lived radionuclides between the methods can be a factor of six or more.

RASCAL 4.3 revised the calculation of the at-risk inventory for fuel damaged underwater to include consideration of the fraction of fuel rods damaged in damaged fuel assemblies. RASCAL 4.3.1 corrects an error in the calculation of spent fuel inventories used when the spent fuel is uncovered. Also, the code has an enforced delay after reactor shutdown before spent fuel damage can occur. The delay is 1 day for BWRs and 3 days for PWRs. These delays account for the time from reactor shutdown until fuel is relocated into the SFP. RASAL 4.3 included this time delay for the spent fuel damaged underwater option. RASCAL 4.3.1 adds this feature for the spent fuel uncovered option.

2.9.1 Drained Spent Fuel Pool (SFP)

The at-risk radionuclide inventory when the fuel in the SFP is uncovered is a function of the number, age, and burnup of the fuel assemblies in the pool at the time the pool is drained. RASCAL 4.3 calculates the at-risk radionuclide inventory starting with the fuel from the most recent refueling outage. If the SFP is drained during a refueling outage, there may be a full core of recently exposed fuel in the pool. If there is, then RASCAL 4.3 calculates the radionuclide inventory for each batch of fuel in the core offload. Otherwise, RASCAL 4.3 only calculates the inventory for the oldest batch of fuel in the reactor at the last refueling. The radionuclide inventories of fuel in the SFP from previous refueling are calculated assuming that one batch of fuel is removed from the reactor at each refueling and the time between refueling is equal to the plant refueling cycle.

There may also be fuel in the SFP from other reactors. This is likely to be the case when reactors share a common SFP. RASCAL 4.3 includes an option to add the inventory of radionuclides in spent fuel from a second reactor to the SFP at-risk radionuclide inventory in the event of a drained pool. The basic assumptions for the spent fuel from the other reactor are as follows:

1. The other reactor is the same type as the first reactor and is operated in the same manner (e.g., both reactors are PWRs, have the same fuel management, and have the same spent fuel burnup).

2. The spent fuel from the other reactor has full burnup.

3. The ages of the spent fuel batches increase from the youngest to oldest in increments of the fuel cycle length.

Under these assumptions, the radionuclide inventory in a batch of fuel at the other reactor at the time of shutdown will be the same as the inventory of full cycle fuel from the first reactor.

RASCAL 4.3 has two methods for calculating the at-risk inventories for uncovered SFP events. These two methods are the default fuel inventory and the custom fuel inventory method, which are both described in the paragraphs below.

<u>Default Fuel Inventory</u> - The following procedure is used by RASCAL 4.3 to calculate the at-risk inventory in a SFP for uncovered fuel events when the default core inventory method is selected.

1. The inventory of a batch of fuel is calculated from the core inventory by dividing the inventory of each radionuclide by the number of batches of fuel in a full core. RASCAL 4.3 assumes that there are 3 batches of fuel in the core when using the default fuel inventory.

2. If there is fuel from another reactor in the SFP, an initial batch inventory for the other reactor fuel is defined to be the same as the batch inventory calculated in step 1.

3. The reactor batch inventory defined in step 1 is decayed to account for the time delay between reactor shutdown and the time of the event.

4. The decayed reactor inventory is added to the SFP inventory. If there has been a full core offload at the time of the event, the inventory added to the SFP is increased to include all batches from the core. All batches are assumed to have the same inventory.

5. If there are batches of fuel in the SFP from the previous reactor refueling outages, the batch inventory in step 3 is decayed to account for the refueling interval (typically 18 to 24 months).

6. Add the decayed inventory to the SFP inventory.

7. Repeat steps 5 and 6 for all batches of fuel from previous refueling outages of the reactor.

8. If there is fuel from another reactor in the SFP, decay the batch inventory from step 2 to account for the time between the event and the most recent refueling of the other reactor.

9. Add the decayed inventory for the other reactor fuel batch to the SFP inventory.

10. If there are additional batches of fuel from the other reactor, decay the batch inventory for the other reactor to account for the refueling interval of the other reactor. The refueling interval for the other reactor is assumed to be the same as the refueling interval for the primary reactor.

11. Add the decayed inventory for the other reactor fuel batch to the SFP inventory.

12. Repeat steps 10 and 11 until inventories for all of the other reactor fuel batches have been added to the SFP inventory.

When steps 1 through 12 have been completed, the RASCAL 4.3 calculated default fuel inventory is the SFP at-risk inventory for the event.

Custom Fuel Inventory - A reactor inventory base file must exist before the custom fuel inventory option can be selected. RASCAL 4.3 contains an additional tool to “Create Reactor Inventory Base File” on the RASCAL 4.3 main menu screen. This tool is used to enter the information on reactor operations and refueling that is needed for calculating custom fuel inventories.

The method for calculating the at-risk inventory for fuel uncovered events when the custom fuel option is selected is similar to the method used with default fuel inventories. The differences are primarily in the definition of the batch inventories for the most recent offload from the primary reactor. The following procedure is used by RASCAL 4.3 to calculate the at-risk inventory in a SFP for uncovered fuel events when the custom inventory option is selected:

1. The batch inventories for batches currently offloaded from the reactor are read from an inventory file (sfpool_inventory.tmp) created for the event. The file contains batch inventories for all fuel batches in core at the time of shutdown. All activities are in becquerel (Bq). If the full core offload option is selected, all batches of fuel are offloaded. If the full core offload option is not selected, only the batch of fully spent fuel is offloaded. The radionuclide activities in the offloaded batches are converted to curies (Ci) and added to the SFP activity. Note that the inventory for each batch of fuel will be different.

2. If there is fuel from previous refueling outages of the reactor in the SFP, an initial batch inventory for the previous fuel batches is defined to be the same as the inventory for the fully spent fuel most recently offloaded from the primary reactor.

3. If there is fuel from another reactor in the SFP, the activity inventory for a batch of fuel from the other reactor is defined to be the same as the inventory for the fully spent fuel from the primary reactor.

4. The activity in the SFP is decayed to account for the time between reactor shutdown and the event.

5. The previous fuel batch inventory from step 2 is decayed to account for the time between the next to last refueling and the time of the event and added to the SFP inventory.

6. The previous fuel batch inventory from step 5 is decayed to account for the refueling cycle time and added to the SFP inventory.

7. Step 6 is repeated until the inventories for all previous batches of fuel have been added to the SFP.

8. Inventories for batches of fuel from the other reactor are added to the SFP in the same manner used in the default fuel inventory calculation.

When steps one through eight have been completed, the RASCAL 4.3 calculated custom fuel inventory is the SFP at-risk inventory for the event.

It is <u>important</u> for the user to understand that, the start time from when the fuel is uncovered to the time that runaway cladding oxidation (a.k.a., a zirconium fire) (if one occurs) is highly dependent on the age of the fuel, its storage configuration relative to other 'hot' fuel, the size and location of any leakage in the SFP, etc. RASCAL users should select parameter values for these calculations in consultation with subject matter experts. Additionally, RASCAL users should realize that several RASCAL cases may need to be run in order to produce a reasonable range of possible outcomes.

Under very challenging conditions (e.g., co-located, recently offloaded fuel with a large leak), the time can be on the order of a couple of hours. While, under other conditions, the time can be very long (e.g., days) or even infinite (in the case of the fuel being air cooled). Some sample times can be taken from the SFP study documented in the enclosure to SECY-13-0112,

"Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark 1 Boiling-Water Reactor" [10]. Specifically, Table 27 presents Methods for Estimation of Leakages and Consequences of Releases (MELCOR)-derived timings for a BWR Mark 1 SFP with fuel in a 1x4 pattern, for several different combinations of time since discharge, leakage size, and mitigation effects. The delta-times from the start of fuel becoming uncovered to the time the hottest assemblies experience a gap release (a precursor to potential runaway cladding oxidation) range from 3 to 22 hours. Table 50 of that study provides additional times for more limiting conditions associated with hot fuel being co-located, but the same general range of timings applies.

In the unlikely event that a runaway cladding oxidation occurs, RASCAL 4.3 assumes that if the runaway cladding oxidation occurs; radioactive materials may be released from the fuel using the release fractions in Section 5 (Table 5-1). The model assumes that clad burning will not start earlier than 2 hours after the fuel is uncovered and that it takes 24 hours for the releases specified in Section 5 (Table 5.1) for clad burning to occur. Thus, 1/96 of the total release fraction listed in Section 5 (Table 5-1) will occur during each 15-minute time step. The release from the fuel terminates when the fuel is recovered or is otherwise cooled by sprays or steam cooling. Additionally, there is no reduction factor other than radioactive decay, which results in a conservative estimate of accident consequences.

2.9.2 Damaged Fuel

The at-risk radionuclide inventory in a damaged spent fuel accident is limited to the radionuclide inventories in the damaged fuel assemblies. RASCAL 4.3 calculates the at-risk radionuclide inventory in the damaged fuel assemblies based on the burnup of the damaged fuel assemblies and the time since the irradiation of the fuel. Similar to the SFP uncovered event, RASCAL 4.3 has two methods for calculating the at-risk inventories for the damaged spent fuel accidents. These two methods are the default fuel inventory and the custom fuel inventory method, which are both described in the paragraphs below.

In each method, the default time since exposure is based on technical specifications for each reactor type related to the minimum time between shutdown and removing the reactor head. The actual time between reactor shutdown and fuel damage is likely to be longer than the minimum and may be significantly longer if the damage occurs during reracking of the fuel or movement of fuel assemblies to dry cask storage.

<u>Default Fuel Inventory</u> - The following procedure is used by RASCAL 4.3 to calculate the at-risk inventory in fuel-damaged underwater events when the default core inventory method is selected. This method is based on extremely limited data; therefore, this method provides uncertain input to consequence assessments.

1. RASCAL calculates the inventory for a fuel assembly from, the normalized core inventory in Table 1-1 of NUREG-1940 adjusted for burnup as described in NUREG-1940 Section 1.1.1, the average reactor power, and the number of fuel assemblies in core. The burnup, average reactor power and the number of assemblies in core are obtained from the RASCAL facility database. RASCAL

users may adjust the reactor power and burnup when the plant is selected in the user interface.

2. The assembly inventory is multiplied by the number of damaged assemblies and the fraction of rods damaged (new in RASCAL 4.3.1) to obtain the at-risk inventory for the damaged fuel event. The number of damaged assemblies and fraction of rod damaged are specified by users in the Source Term Pool Storage-Damaged Assembly Underwater data entry screen in the user interface.

3. The at-risk inventory is adjusted for delay between reactor shutdown and fuel damage.

This procedure assumes that all damaged fuel assemblies have the same inventory, both initially and after decay and the inventory is an end-of-cycle, core average inventory. These assumptions can lead to significant errors in the at-risk inventory if the fuel assemblies are from different fuel batches within the same offload, or they are from batches off loaded from different refueling outages. There can also be a significant error if damage only occurs to a single fuel assembly and the assembly is from a fuel batch that has not been exposed for three full cycles. In this latter case, the at-risk inventory of long-lived isotopes such as cesium-137 (Cs-137) and strontium-90 (Sr-90) can be overestimated by a factor of five or more.

Custom Fuel Inventory - A reactor inventory base file must exist before the custom fuel inventory option can be selected. The following procedure is used to calculate the at-risk inventory in fuel-damaged underwater events with custom fuel inventories. The method is based on limited data; therefore, this method provides uncertain input to consequence assessments.

1. RASCAL reads a batch inventory from an inventory file (sfpool_inventory.tmp) created for the event and converts the inventory from Bq to Ci. The file contains batch inventories for all batches in the reactor at shutdown. The inventory that is read is determined by the “number of days fuel assemblies were exposed in the reactor” which is entered by the user in the Source Term Pool Storage-Damaged Assembly Underwater data entry screen in the user interface.

2. An assembly inventory is created by dividing the batch inventory by the number of assemblies in a batch (number of assemblies in core/number of batches in core).

3. An initial at-risk inventory is calculated from the assembly inventory by multiplying the assembly inventory by the number of assemblies damaged and the fraction of rods damaged.

4. The initial at-risk inventory is adjusted to account for decay between the time of reactor shutdown and time of the fuel damage event.

This procedure assumes that all damaged fuel assemblies have the same inventory, both initially and after decay. It does not assume that the assemblies have a core average inventory

or that the assemblies have been exposed for more than one cycle. These assumptions can lead to significant errors in the at-risk inventory if the fuel assemblies are from different fuel batches within the same offload, or they are from batches offloaded from different refueling outages.

2.10 Uranium Hexafluoride (UF_6) Source Terms

There have not been any changes to the calculation of uranium hexafluoride (UF_6) source terms in RASCAL 4.3 or RASCAL 4.3.1. However, there have been questions related to these calculation methods that indicate that additional discussion is warranted.

Specifically, the questions pertain to several cases in which the UF_6 release rate inside a building was varied without what was considered an appropriate variation in the RASCAL results. The RASCAL results did not change as expected because the total UF_6 inventory specified by the RASCAL user was of a sufficiently small amount that the UF_6 release to the building terminated well before the release from the building (to the environment) terminated. All of the values in the Maximum Dose Values table for UF_6 releases are values based on integration over the full calculation period. These maximum values in the table reflect the total release from the building (to the environment) not the UF_6 release rate into the building. In these situations the total UF_6 releases to the building and the total uranyl fluoride (UO_2F_2) and hydrogen fluoride (HF) releases from the building are the same for all practical purposes. Conversely, if the total inventory specified by the RASCAL user had been much larger or much smaller, the RASCAL results would have changed as expected.

3.0 TRANSPORT, DISPERSION, AND DOSE CALCULATIONS

This section describes the changes to the atmospheric transport and dispersion calculations and the dose calculations implemented in RASCAL 4.3. RASCAL 4.3 adds a fourth model domain for puff (far field) dispersion and deposition calculations. RASCAL 4.3.1 adds two new output files, and corrects an error in calculation of depletion in the plume (near field) deposition calculations.

3.1 Transport and Dispersion Calculations

The general features of the atmospheric transport and dispersion calculations and dose calculations have not changed in RASCAL 4.3. However, in response to staff requests arising from the Fukushima Daiichi accident, RASCAL 4.3 has added a fourth Cartesian computational grid which increases the RASCAL 4.3 domain from a 50-mile (mi) (80 kilometer (km)) radius to a 100-mi (160 km) radius. The 100-mi grid has 5-mi (8 km) spacing between computational nodes. The default close-in polar grid associated with the 100-mi Cartesian grid extends to 10 mi just as it does with the 50-mi Cartesian grid.

In conjunction with the addition of the 100-mi grid, RASCAL 4.3 includes updated topographic and surface roughness data files for all grids. The data in the topographic data files, which are used in modification of wind fields, were extracted from U.S. Geological Survey digital elevation data. The terrain elevation values in the file are spatial averages of elevations at nine locations within 1 mi of the nominal grid location. The data in the surface roughness, which are used in the modification of wind fields and in the transport and dispersion calculations, are based on land use data extracted from the National Land Cover Data set. The surface roughness lengths were estimated from land use data. The surface roughness estimates are also 9-point spatial averages centered on the node position with about 100 ft spacing between points. The surface roughness length for overwater locations have been set to a value of 0.001 meters (m). The topographic and surface roughness files are described more fully in Section 4.

3.1.1 Model Grid Dose Consistency

Figure 3-1 shows the consistency in dose calculations for the eight model grids (4 polar and 4 Cartesian). The calculations were made with a cardinal wind direction (90 degrees, 180 degrees, 270 degrees, or 360 degrees), a 4 mi/h wind speed and D class stability. A cardinal wind direction should be used when testing model calculations with a constant wind direction because the centerline of the plume passes directly over computational nodes in these directions. When an off-cardinal wind direction is used, the plume centerline does not necessarily pass directly over the computational nodes on the Cartesian grid. In fact, for most wind directions, the plume centerline passes between Cartesian grid nodes. Near the release point where the plume is narrow, the grid spacing is large compared to the plume width. As a result, the doses shown on the Maximum Dose Values table and in the Detailed Results for the Cartesian grid may be significantly lower than centerline doses. These points are illustrated in Figure 3-2.

The only difference in the RASCAL 4.3 input values for the calculations leading to Figures 3-1 and 3-2 is the wind direction. A cardinal wind direction was used for the calculations for Figure 3-1 and a wind direction 10 degrees off a cardinal direction was used for the calculations for Figure 3-2. Note that the largest deviation in doses is for doses calculated on the Cartesian grid at distances less than the minimum distance shown for the Cartesian grid in the Maximum Dose Values table. Note also that the doses are correct for the nodes but the nodes are not on the plume centerline.

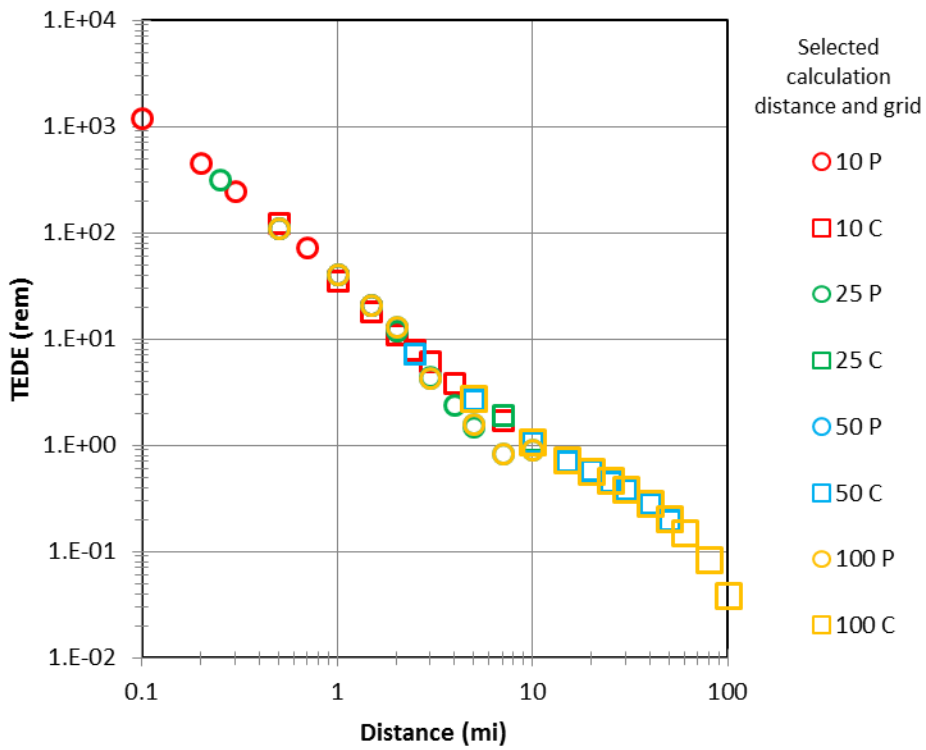


Figure 3-1 Dose consistency among the eight RASCAL 4.3 computational grids

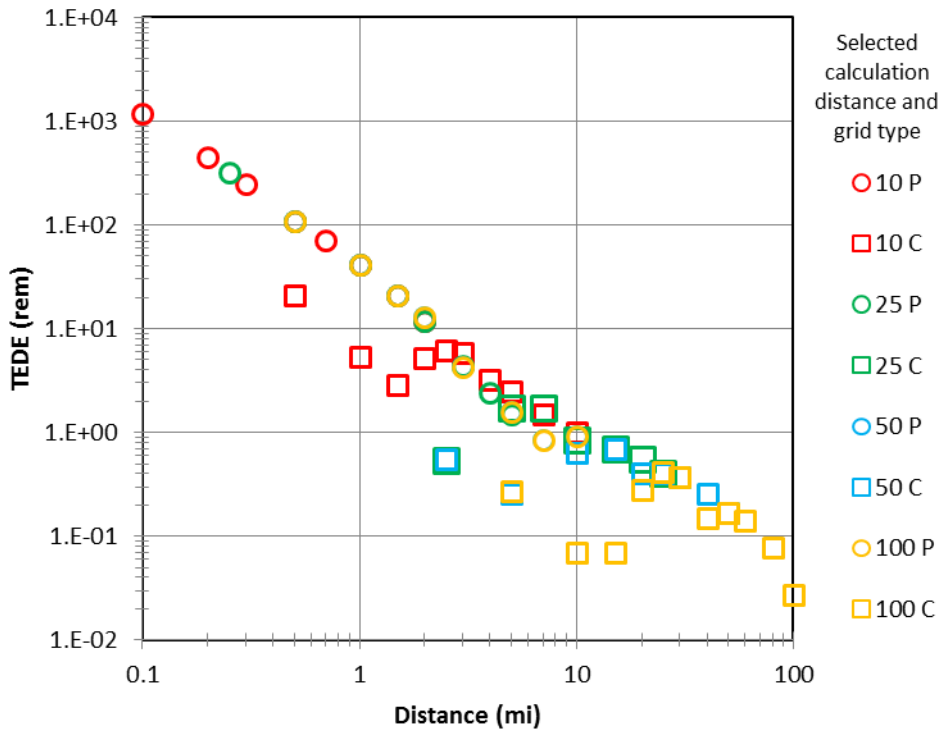


Figure 3-2 Maximum doses as a function of distance for a 10° off-cardinal wind direction

Figure 3-3 illustrates plume centerlines relative to Cartesian grid node locations for nodes at nominal distances of 1 and 10 mi on the 10-mi grid.

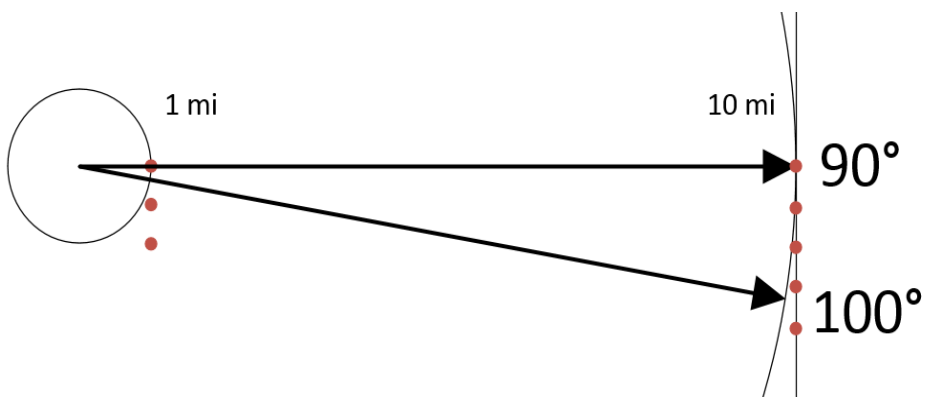


Figure 3-3 Cartesian node locations relative to plume centerlines

With a wind direction of 270 degrees (west wind), a plume centerline would pass directly over nodes at both nominal distances at 90 degrees (east). With a wind direction of 280 degrees, the plume centerline would pass between nodes at both distances.

Table 3-1 lists the actual direction and distance from the release point for the nodes shown in Figure 3-3. At the 1-mi distance, there is a 10-degree separation between the plume centerline

(100 degrees) and the closest computational node; at the 10-mi distance the separation is only 1.3 degrees. As a result, the dose calculated for the node closest to the plume centerline at 1 mi is about a factor of 20 lower than the centerline dose, while the dose calculated for the node closest to the plume centerline at 10 mi is only about 15 percent lower that the centerline dose.

Table 3-1 Cartesian grid node positions at nominal distance of 1 and 10 miles

Direction	Distance (mi)	Direction	Distance
090°	1.0	090°	10.0
116.6°	1.12	092.9°	10.01
135°	1.41	095.7°	10.05
		098.5°	10.11
		101.3°	10.20

As mentioned above, the doses calculated for the nodes are correct; they are just not on the plume centerline. When real meteorological data are used and the wind direction varies in time and space, the footprint of the plume is spread out and the centerline/off-centerline distinction becomes less significant. Ultimately, over a period of hours, the doses calculated for the Cartesian grid give a better picture of the plume footprint than the doses on the polar (close-in) grid. The limitations associated with the node spacing can be minimized by using the appropriate grid. For example, for doses at distances of about 10 mi, use the 10-mi grid or the 25-mi grid. Do not use the doses from the 100-mi grid. The doses calculated for the 100-mi grid are correct, but they do not provide the optimum spatial resolution in doses near 10 mi.

3.1.2 Plume Characterization Output Files

The two additional output files created by RASCAL 4.3.1 are Plume_Char.tmp and Puff_Char.tmp. These files contain intermediate computational results related to the plume. The information may be used to check model dispersion and dose calculations, or it may be used to track plume positions with time.

Plume_Char.tmp contains nine lines for each time step. The first of these lines gives the time (hours (h)) since the beginning of release, the release height (m), the wind direction (10 degrees) and wind speed (meters per second (m/s)), stability class, mixing layer depth (m), precipitation class, turbulence parameters (m/s), deposition velocities (m/s), and washout coefficients (1/h). These parameters apply to all arcs on the polar grid. The next eight lines give the plume characteristics. There is one line for each arc. Each line has the following information:

- arc number

- distance (m)
- plume height (m)
- horizontal and vertical dispersion parameters (m)
- vertical reflection term
- fraction of iodine remaining in the plume
- fraction of other depositing material remaining in the plume

In RASCAL, puff positions are tracked on a 21 node by 21 node Cartesian grid. The Puff_Char.tmp file starts with a line that gives the node spacing for the domain in use, and the release point location. In RASCAL 4.3.1, the release location is always 11.0. The puff information is presented in blocks by time. The first line of each block is the elapsed time in hours since the beginning of the release to the atmosphere. Blocks of data are written at 15 minute intervals. The remaining lines in each block give the puff information starting with the most recently released puff and continuing in order of increasing puff age. The order of puff information is:

- puff number (the youngest puff has the largest number)
- puff age (h)
- puff location (x, y) (Puffs are tracked after they leave the domain and when the puff gets far enough out of the domain, the puff is deleted. The remaining puffs are not renumbered as puffs are deleted)
- plume height (m)
- horizontal and vertical dispersion parameters (m)
- vertical reflection term (Note that this term is not the same term used for plumes and it is a factor of two less that the plume vertical reflection term)
- deposition velocity for iodine (m/s)
- deposition velocity for other depositing material (m/s)
- washout coefficient for iodine (1/h)
- washout coefficient for other depositing material (1/h)
- fraction of iodine remaining in the puff
- fraction of other depositing material remaining in the puff

3.1.3 Plume Depletion Calculation Correction

The new plume characteristics file revealed a variability in plume depletion fractions that was not readily apparent in the normal plume model output. This evaluation of the variability led to the discovery of coding error in calculation of plume depletion associated with dry deposition. The error resulted in small overestimates of dry deposition and underestimates of material remaining in the plume. Figure 3-4 shows an example of the fraction of iodine remaining in the plume model in RASCAL 4.3 (original code – red curve) and the revised model in RASCAL 4.3.1 (revised code – blue curve). In the RASCAL 4.3 code, approximately 89 percent of the iodine would remain in the plume at 10 km (6.2); 11 percent of the iodine would have deposited by 10 km. In the RASCAL 4.3.1 revised code, about 92 percent of the iodine remains in the plume at 10 km; only about 8 percent of the iodine would deposit by 10 km.

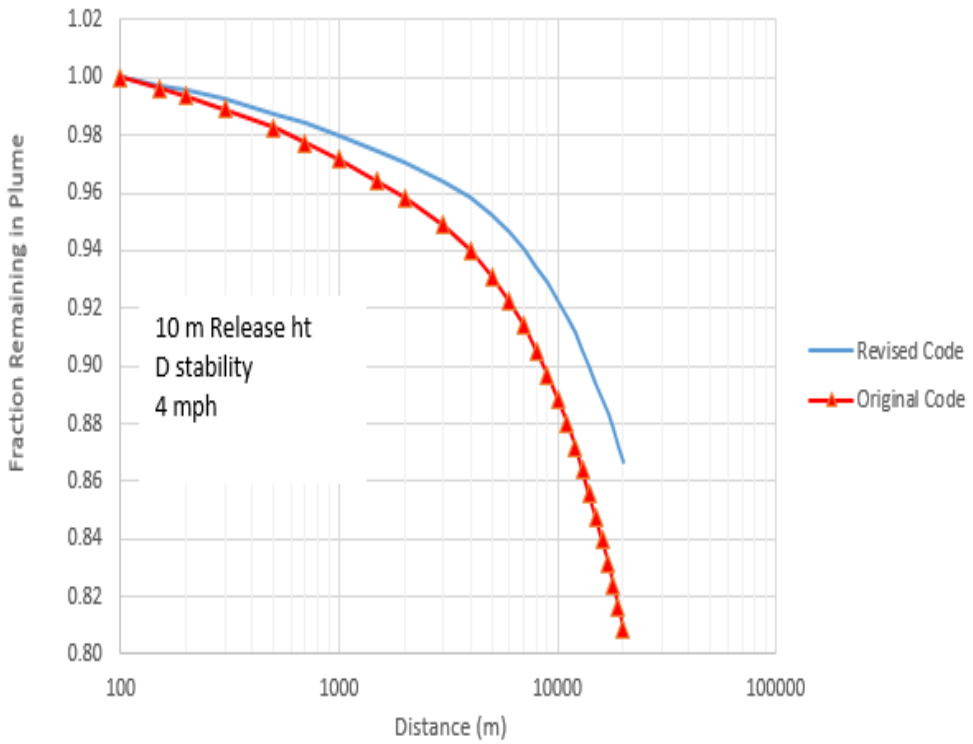


Figure 3-4 Iodine depletion due to dry deposition in RASCAL 4.3 versus RASCAL 4.3.1

3.2 Child Thyroid Dose Calculations

The U.S. Environmental Protection Agency (EPA) has released a draft Protective Action Guide (PAG) manual for interim use and public comment [11]. Section 2.3 of the draft PAG manual considers the protective actions during the early phases of a radiological incident. Specifically, it deals with the administration of potassium iodide (KI). Table of 2-1 of the PAG manual suggests administration of KI when the projected dose to the child thyroid from exposure to iodine exceeds 5 rem. Calculation of the child thyroid dose has been added to RASCAL 4.3 to allow evaluation of recommendations for administration of KI. The draft PAG manual also

discusses U.S. Food and Drug Administration (FDA) recommendations for administration of KI to adults [12] and [13]. The child thyroid doses are calculated when the Federal Guidance Report Number 13 (FGR 13) [14] International Commission on Radiological Protection (ICRP) 72 dose coefficients are selected. The child thyroid dose is not calculated when FGR 11 [15] ICRP-26/30 dose coefficients are selected.

RASCAL 4.3 child thyroid dose coefficients are based on the dose coefficients for a 1-year old child in file FGR13INH.HDB included with FGR13PAK [16]. For consistent treatment of iodine the child dose coefficients in RASCAL 4.3 are weighted averages assuming that of the iodine in the atmosphere 25 percent is associated with particles, 30 percent is associated with reactive gases (e.g., iodine (I_2)), and 45 percent is associated with nonreactive gases (e.g., methyl iodide (CH_3I)).

The child thyroid doses calculated by RASCAL 4.3 in the Source Term to Dose (STDose) and Field Measurement to Dose modules are based only on exposure to iodine isotopes for consistency with the EPA PAG definition. In contrast, the adult thyroid doses calculated by RASCAL are based on exposure to all radionuclides.

As mentioned above, FDA recommendations for administration of KI are based on age. The only factors in the thyroid dose calculation that are functions of age are the breathing rate and dose coefficients. Table 3-2 lists the light-activity breathing rates and the composite thyroid dose coefficients for the iodine isotopes that are typically associated with reactor accidents.

Table 3-2 Breathing rates and dose coefficients for use in estimating thyroid doses from iodine isotopes inhaled during plume passage

Age	Breathing Rate (m/s)	Dose Coefficients (Sv/Bq)				
		I-131	I-132	I-133	I-134	I-135
100 d	5.28E-05	2.52E-06	3.40E-08	6.78E-07	7.02E-09	1.39E-07
1 y	9.72E-05	2.47E-06	3.03E-08	6.12E-07	6.24E-09	1.24E-07
5 y	1.58E-04	1.40E-06	1.57E-08	3.18E-07	3.23E-09	6.43E-08
10 y	3.11E-04	7.10E-07	7.02E-09	1.42E-07	1.44E-09	2.88E-08
15 y	3.83E-04	4.59E-07	4.53E-09	9.09E-08	9.30E-10	1.85E-08
20 y	4.17E-04	2.93E-07	2.87E-09	5.70E-08	5.90E-10	1.17E-08

The values in Table 3-2 may be used to scale the RASCAL 4.3 dose estimates for a 1-year old child to estimates of thyroid doses from iodine for children of other ages and for adults (20 yr). Figure 3-5, which is based on the data in Table 3-2, shows appropriate scaling factors. According to the FDA, there is generally little value in use of KI for adults over the age of 40.

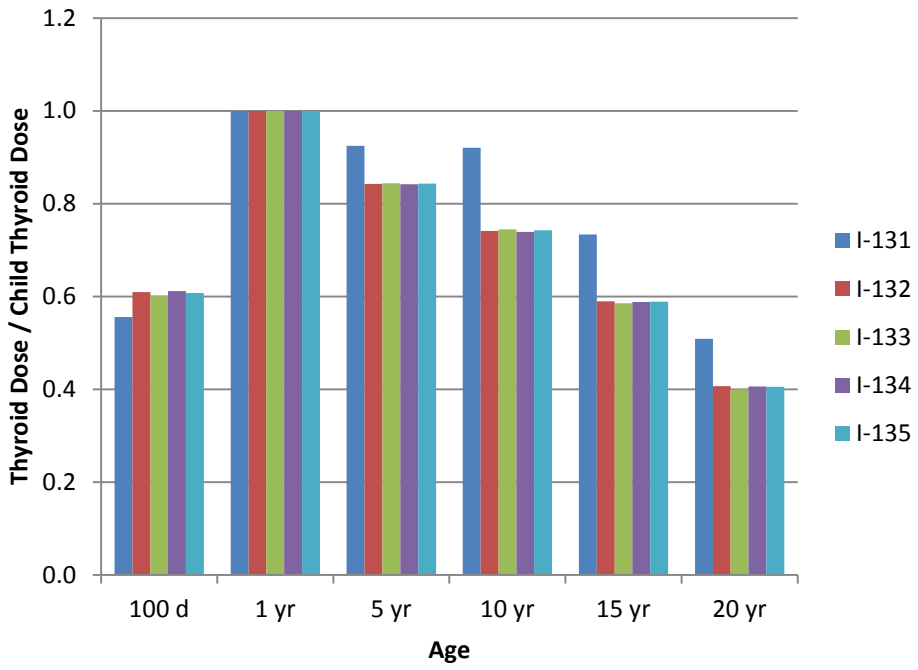


Figure 3-5 Scaling factors (thyroid dose/child thyroid dose) for adjusting RASCAL 4.3 child

3.3 Ingestion Dose Coefficients

Ingestion dose coefficients have been added to the radionuclide database in RASCAL 4.3. The database includes ingestion dose coefficients from both FGR No 11 and FGR No.13. The FGR 13 ingestion dose coefficients are for adults. These dose coefficients are not used in any dose calculations in RASCAL 4.3. They are used to establish relative importance of radionuclides in the source term to potential doses via the food pathway. The importance model and calculation are discussed in Section 7.

3.4 Field Team Dose Estimates

During the course of an event, field teams may be exposed to the plume for short durations. RASCAL 4.2 included a calculation that; (1) estimated external and total doses for field teams prior to exposure, and (2) provided conversion factors that may be used to estimate total dose from the readings of direct reading dosimeters after exposure.

The dose estimates and conversion factors are found in the detailed results portion of the RASCAL output when either the External Gamma Exposure Rate or the External Gamma + Beta Exposure Rate result type is selected. Generally, the dose estimates and conversion factors should be obtained from the Cartesian grid to ensure that they appropriately account for the plume transit time. To view the dose and conversion factor estimates, place the cursor over the field team location on a dose footprint and left click to display the dose rate versus time plot

for the location (Figure 3-6). The doses and conversion factors are found in the table on the right side of the screen.

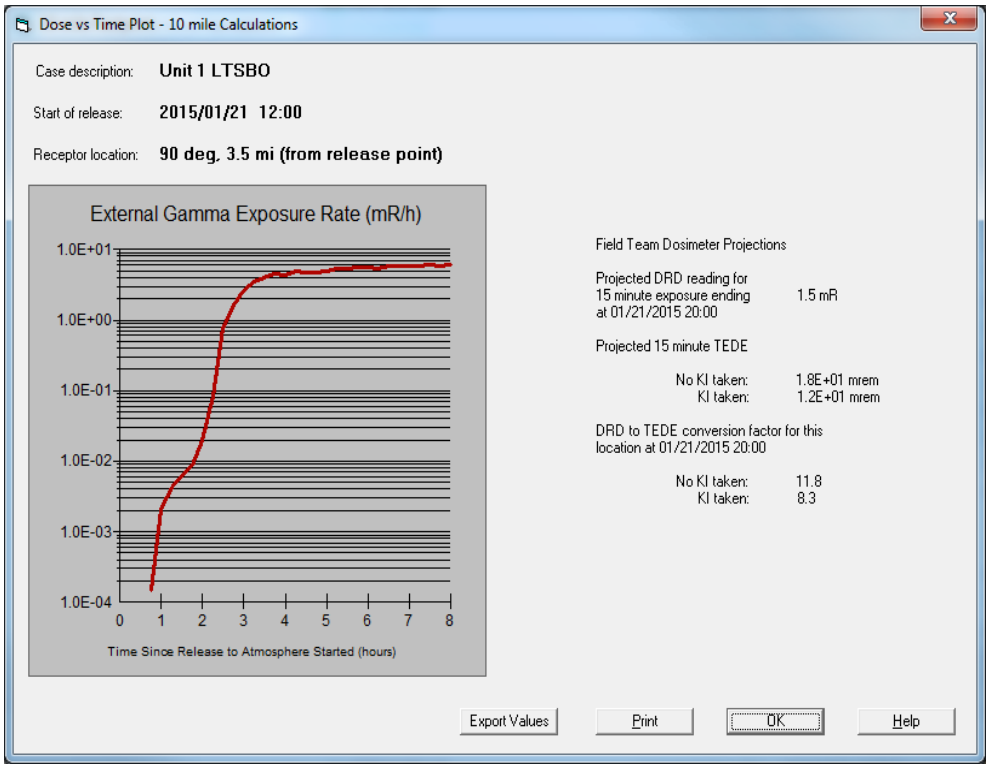


Figure 3-6 RASCAL detailed results for dose and conversion factor estimates

3.4.1 Projected Direct Reading Dosimeter (DRD) Reading and Doses

The estimated direct reading dosimeter (DRD) reading for a field team at the time and location selected is based on the gamma or gamma + beta exposure rates assuming a 15-minute exposure period. The calculation of these exposure rates is discussed in Section 4 of NUREG-1940.

The estimated total external and internal dose at the time and location selected for a field team that has not taken KI is the sum of the ground shine, cloud shine, and inhalation CEDE assuming a 15 minute exposure period. Calculation of these dose rates are described in Sections 4.9.1, 4.9.2 and 4.9.3 of NUREG-1940, “RASCAL 4: Description of Models and Methods.”

The estimated total dose for a field team that has taken KI is lower than the estimated total because KI effectively blocks iodine absorption by the thyroid. RASCAL 4.3 calculated the correction to the total dose to account for the effect of KI as

$$\Delta D_{KI} = 0.9 \times 0.03 \times D_{thy} \tag{3-1}$$

where

ΔD_{KI} = dose reduction due to administration of KI (mrem),
0.9 = effectiveness of KI in reducing the thyroid dose,
0.03 = thyroid weighting factor in calculation of the inhalation CEDE, and
D_{thy} = thyroid dose (mrem).

Calculation of the thyroid dose is discussed in Section 4.9.1 of NUREG-1940. The total dose estimate for field teams that have taken KI is then calculated by subtracting the KI dose reduction from the dose estimate for field teams that did not take KI.

As stated above, the projected DRD readings and doses are for a 15-minute exposure period. If the expected duration of the field team exposure is less than or greater than 15 minutes, the projected DRD reading and doses may be adjusted using the ratio of the expected exposure duration to 15 minutes. However, if the expected exposure duration is significantly longer than 15 minutes, this adjustment procedure should not be used. Instead, the DRD readings and doses should be estimated by adding the projected DRD readings and doses for successive 15-minute periods within the exposure period.

3.4.2 Post Exposure Dose Estimates

The projected DRD readings and doses discussed above are all dependent on the accuracy of the source term, release path, meteorological conditions, plume location and transit time, and field team exposure time and location. There is a reasonable likelihood that actual DRD readings and doses will differ from the projected values. As soon as DRD readings are available they should be used instead of the projected values. RASCAL 4.3 estimates conversion factors (CFs) that may be used to estimate total dose commitment (external dose + internal dose) from DRD readings. The CF is found at the bottom of the table that presents the projected DRD readings and doses described above.

There are two CFs, one for field teams that have not taken KI and the other for field teams that have taken KI. For field teams that have not taken KI, the CF is calculated as

$$CF = \frac{D_{gs} + D_{cs} + D_{inh}}{DRD_p} \tag{3-2}$$

where

CF = CF for field teams that have not taken KI (mrem/mR),
D_{gs} = projected ground shine dose (mrem),
D_{cs} = projected cloud shine dose (mrem),
D_{inh} = projected inhalation CEDE (mrem), and
DRD_p = projected DRD reading (mR).

Given the actual DRD reading and the CF, the total dose may be estimated by the user as:

$$D = CF \times DRD_a \tag{3-3}$$

where

D = the estimated external + internal dose (mrem)
DRD_a = the actual dosimeter reading (mR)

The dose calculation for field teams that have taken KI is similar to the calculation for teams that have not taken KI. The first step is to calculate a CF. In this case, RASCAL 4.3 calculates the conversion factor as

$$CF_{KI} = \frac{D_{gs} + D_{cs} + D_{inh} - \Delta D_{KI}}{DRD_p} \tag{3-4}$$

where

CF_{KI} = CF for field teams that have taken KI (mrem/mR), and
ΔD_{KI} = dose reduction associated with KI (mrem).

Given the actual DRD reading and this CF, the total dose may be estimated by the user as

$$D_{KI} = CF_{KI} \times DRD_a \tag{3-5}$$

where

D_{KI} = estimated external + internal dose (mrem) for those who have taken KI.

Doses calculated in this manner will be consistent with the DRD reading. However, the doses may still be in error because the CF is based on the radionuclide mix assumed in calculating the projected DRD readings and doses.

Table 3-3 shows an example of the variation in the CFs as a function of time after the beginning of release and distance from release point at the center of the plume. The example is based on a LOCA with containment bypass. The variations in CFs seen in the table are the result of variation in source term radionuclide mix with time, the transit time from the release point to the receptor, and the changes in radionuclide mix in the plume caused by decay and deposition between the release point and the receptor. The variation of the conversion factors across the plume at any given time and distance is small compared to the variations with time and distance. There is no need to correct the doses estimated from actual DRD readings for exposure time unless the exposure time significantly exceeds 15 minutes. If the exposure time significantly exceeds 15 minutes, consideration should be given to calculating conversion factors for successive 15 minute intervals during the exposure period and using an average conversion factor to estimate doses. It may be appropriate to weight the interval conversion factors by the projected DRD for the interval in the averaging process.

Table 3-3 Conversion factor variation with time and distance from the release point

Conversion Factors (CF)			
Time after LOCA (h)	**Distance (mi)**	**Without KI**	**With KI**
2	1.0	1.0	1.0
	2.0	1.0	1.0
	3.0	1.0	1.0
	5.0	1.0	1.0
	10.0	1.1	1.1
3	1.0	1.6	1.3
	2.0	1.5	1.3
	3.0	1.3	1.2
	5.0	1.1	1.0
	10.0	1.1	1.0
4	1.0	2.1	1.7
	2.0	2.2	1.8
	3.0	2.4	1.8
	5.0	2.3	1.7
	10.0	1.1	1.0
8	1.0	2.2	1.7
	2.0	2.6	2.0
	3.0	3.2	2.4
	5.0	4.3	3.1
	10.0	5.3	3.7

4.0 METEOROLOGICAL DATA ACQUISITION AND PROCESSING

A new utility to download meteorology data from the Internet, called MetFetch, has been developed to allow RASCAL 4.3 users to process and download model-ready observations (*.obs) and forecasts (*.fcs) from the Internet for direct use in a RASCAL 4.3 calculation. This section describes the MetFetch program, the meteorological data available to download, and the topographic data that are used in RASCAL 4.3 to process the meteorological data.

4.1 Meteorological Data

Observations and forecasts are available for approximately 2500 station locations and include primary National Weather Service (NWS) and Federal Aviation Administration (FAA) stations located throughout the United States and its territories. Observations are obtained through the NWS Western Regional Headquarters Web server; forecasts are obtained through the NWS National Digital Forecast Database (NDFD) [17]. Since MetFetch downloads observations and forecasts from remote servers, an Internet connection is required to use this utility.

4.2 Topographic Data

RASCAL gridded terrain files (*.top) have been derived for each site in the RASCAL facility database using elevation data from a Global Digital Elevation Model (GTOPO30) file [18]. A RASCAL terrain *.top file contains four terrain grids with 22 points in the X and Y directions, at grid spacings of 10.0, 5.0, 2.5, and 1.0 mi, respectively.

GTOPO30 is a digital elevation model (DEM) covering the full extent of the globe, with a native horizontal resolution of 30-arc seconds (approximately 1 kilometer); these data were developed by the U.S. Geological Survey's Earth Resources Observation and Science (EROS) Data Center in 1996 using a variety of data sources. Figure 4-1 is a raster image of the GTOPO30 file used to derive site terrain files for RASCAL. Darker shades of gray represent lower terrain and lighter shades of gray represent higher terrain. Oceans are colored white.

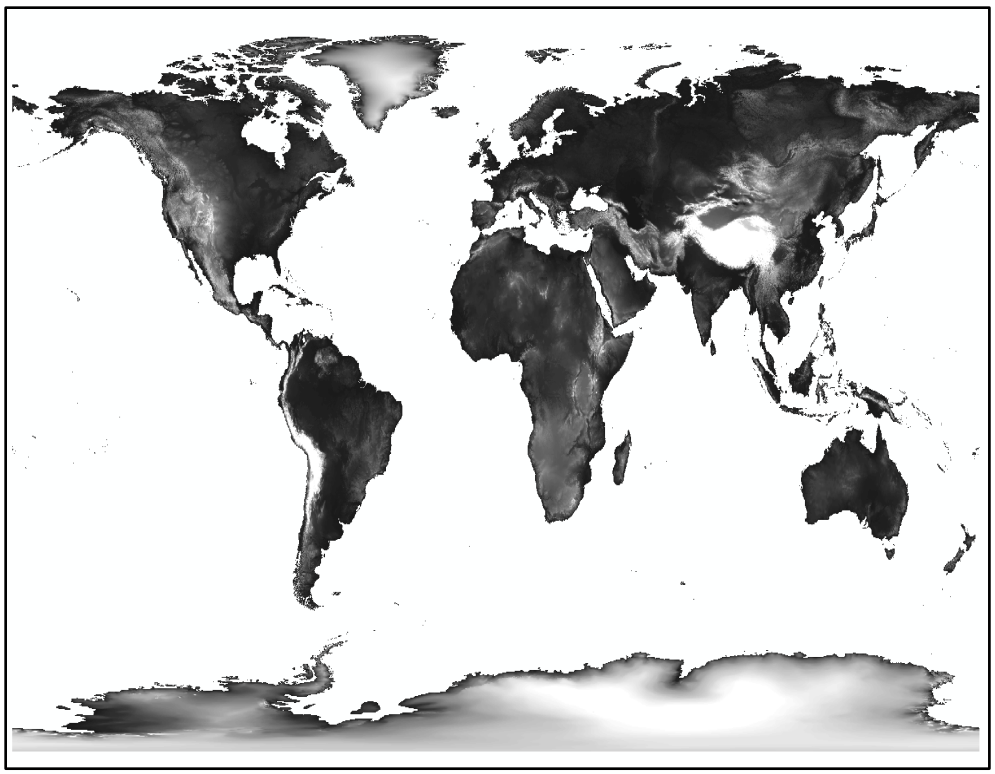


Figure 4-1 Raster image of the GTOPO30 file used to derive site terrain elevation files (*.top) for RASCAL 4.3

4.3 Surface Roughness Data

RASCAL gridded surface roughness files (*.gz0) have been derived for each site in the RASCAL facility database using land use/land cover data from the National Land Cover Dataset 1992 (NLCD1992) file [19]. A RASCAL roughness *.gz0 file contains four surface roughness grids with 21 points in the X and Y directions, at grid spacings of 10.0, 5.0, 2.5, and 1.0 mi, respectively.

The NLCD1992 file utilizes a 21-class land cover classification scheme that has been applied to the continental United States at a spatial resolution of 30 m. NLCD92 is based primarily on the classification of Landsat Thematic Mapper circa 1990's satellite data. Other ancillary data sources used to generate these data included topography, census, and agricultural statistics, soil characteristics, and other types of land cover and wetland maps.

A lookup table from the EPA AERSURFACE User's Guide [20] is then used to convert the 21 land-cover codes to corresponding surface roughness values. Figure 4-2 is a raster image of the NLCD1992 file used to derive site surface roughness values for RASCAL. Each color represents an NLCD class code and has a corresponding surface roughness value.

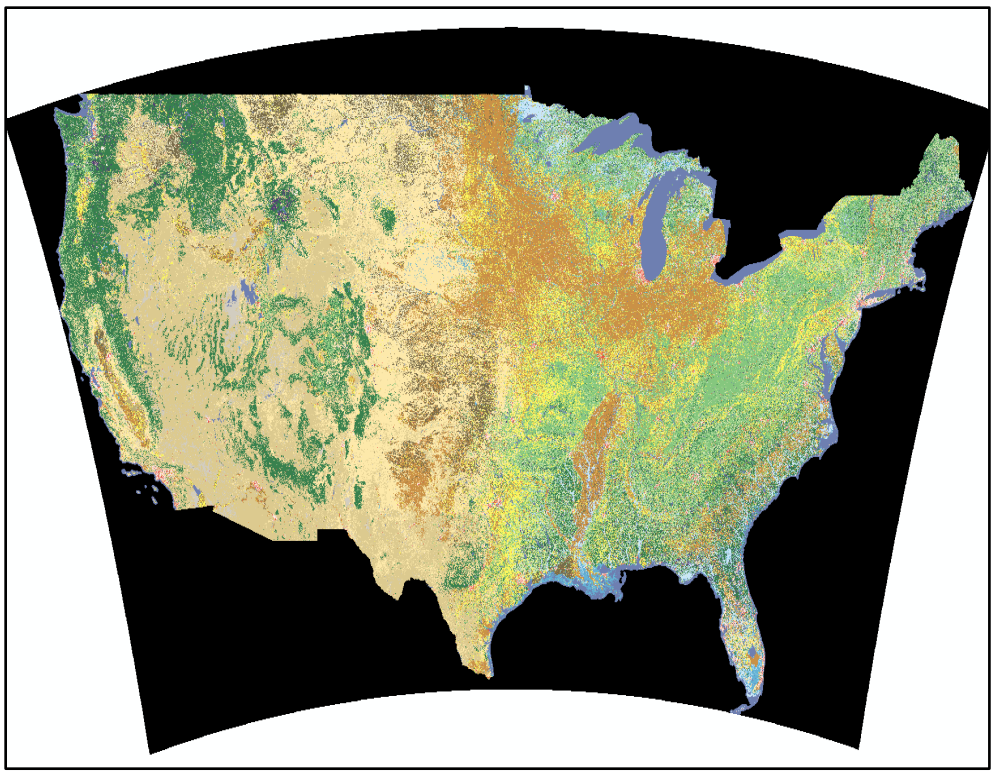


Figure 4-2 Raster image of the NLCD1992 file used to derive site surface roughness files (*.gz0) for RASCAL 4.3

5.0 UPDATED REACTOR CORE AND COOLANT INVENTORY MODELS

RASCAL 4.2 and earlier versions of RASCAL have used in their source term calculations reactor core inventories based on an assumed power level of 3,479 MWt and fuel burnup of 38,585 megawatt days per metric ton of uranium (MWd/MTU). Similarly, earlier versions of RASCAL used tabled values of reactor coolant radionuclide concentrations which were scaled up and down based on reactor power.

RASCAL 4.3 uses these same assumptions for its default reactor radionuclide inventories. However, RASCAL 4.3 also has an option to estimate reactor core and coolant radionuclide inventories at the time of an accident using reactor specific information on reactor operations prior to the accident. This option is implemented in a new feature that is bundled with RASCAL 4.3 and may be called from the RASCAL 4.3 user interface. This section describes the updated inventory models and utility code implementing them. Sections 5.1 through 5.4 deal with the reactor core inventory; Section 5.5 deals with the reactor coolant inventory. Section 5.6 addresses the input to and output from the code, Rx_Inventory_S, which implements the updated RASCAL 4.3 inventory model.

5.1 Reactor Core Activity Model

Production rate P'_i of a radionuclide *i* [(Bq/MWt)/d] is a function of reactor type (BWR or PWR) and is independent of burnup. This is a reasonably good assumption for most radionuclides. Production of the actinides increases with burnup, so RASCAL assumes a production rate that is an average for peak rod burnup of 42 Gigawatt day/metric ton of uranium (GWd/MTU) for a BWR and 62 GWd/MTU for a PWR.

Given constant reactor power, the rate of change of activity of a radionuclide as a function of time is given by

$$\frac{dA_i(t)}{dt} = P'_i - \lambda_i A_{i(t)} \tag{5-1}$$

where

A_i(t)	= total core activity of radionuclide i in Bq/MWt at time t
P'_i	= production rate of radionuclide i in (Bq/MWt)/d in the core
λ_i	= decay constant for radionuclide i

Solving this equation, the activity at time *t* is given by:

$$A_i\,(t) = \frac{P'_i}{\lambda_i} \times [1.0 - \exp(-\lambda_i t)] \tag{5-2}$$

Routine reactor operation involves alternating periods when the reactor is at power and when it is shutdown for refueling or maintenance. While the reactor is shutdown, the activity in the core decreases due to radioactive decay. During refueling a portion of the irradiated fuel, typically one-third, is replaced by fresh fuel which has no fission product activity. The fission product activity in the remaining fuel decays during the refueling outage, so that there is less activity

when the reactor restarts than there was when it shutdown. As a result, the activity of each radionuclide at restart cannot be calculated directly from Equation (5-2). The activity remaining at the time of restart is calculated as:

$$A_i(t_{rs}) = A_i(t_{sd})\exp[-\lambda_i(t_{rs} - t_{sd})] \tag{5-3}$$

where

t_{rs} = time of restart,
t_{sd} = time of shutdown.

To calculate activity $A_i(t_{rs})$ after restart, it is necessary to first determine an effective reactor operation exposure time that would result in $A_i(t_{rs})$. The effective reactor operation time, t_{effi}, is calculated as:

$$t_{effi} = -\frac{1}{\lambda_i}\ln\left[1 - \frac{A_i(t_{rs})\lambda_i}{P_i}\right] \tag{5-4}$$

Finally, the activity during the second and subsequent fuel cycles is calculated using Equation (5-2) replacing t, the time since the start of initial exposure, with;

$$t_i = t_{effi} + (t - t_{rs}) \tag{5-5}$$

Note that after the initial exposure period, the time used for the calculation of activity becomes a function of the radionuclide.

Figure 5-1 shows the activity for selected radionuclides in a fuel batch as a function of time from initial exposure to final discharge for a typical reactor with a 540 d fuel cycle assuming a 30 d refueling outage during which one-third of the fuel in the core is replaced. Table 5-1 gives the production and decay parameters for the selected radionuclides. The production parameters are core-average numbers. For individual fuel batches, the parameters must be divided by the number of cycles that fuel remains in the core.

The iodine-131 (I-131) and I-133 activities in Figure 5-1 clearly show the refueling outages. These radionuclides also show that the activity level for short-lived radionuclides is reasonably constant throughout the fuel life. The activity decreases significantly during refueling outages, but quickly returns to the equilibrium level when the reactor restarts. In contrast, the activities of long-lived radionuclides such as cesium-137 (Cs-137) and plutonium (Pu-240) continue to increase throughout the fuel life. The activity does not decrease significantly during refueling outages and does not reach equilibrium. Assuming three-cycle operation for a typical reactor, Figure 5-2 shows the variation in core activity during a cycle for a mature reactor core (core with fuel in first, second, and third cycles). For short-lived radionuclides, the activity increases from zero to equilibrium, while for long-lived radionuclides, the activity level increases by a factor of two from beginning of cycle to end of cycle. About one half of the core activity of long-lived isotopes is in the spent fuel at the end of the cycle.

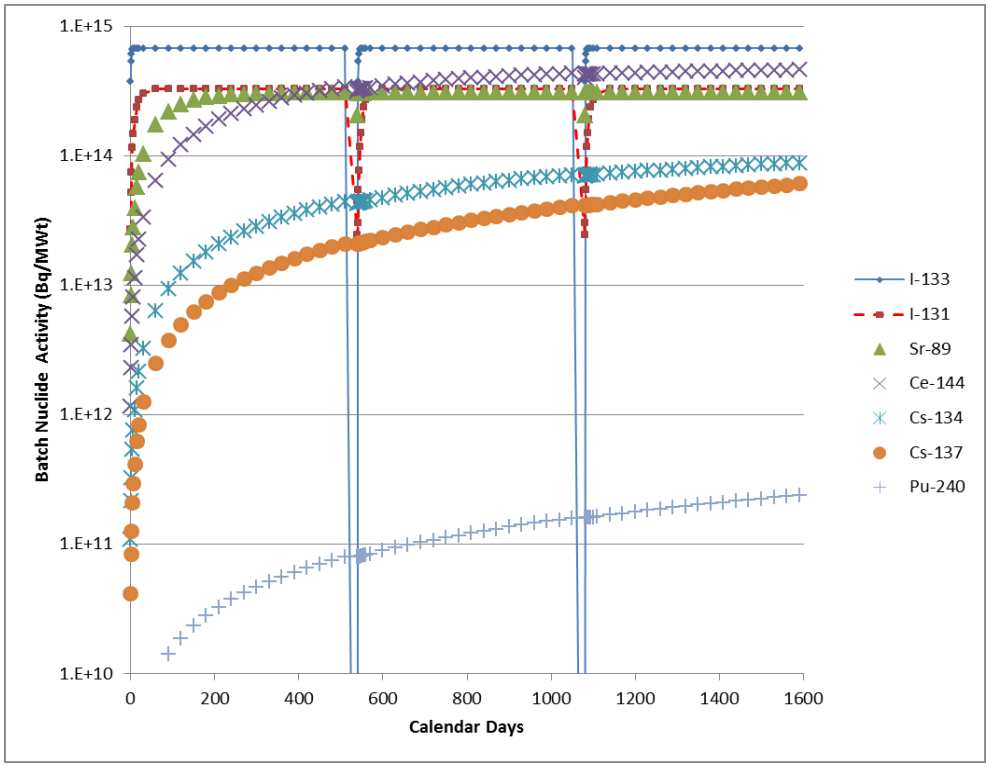


Figure 5-1 **Radionuclide activity as a function of exposure time in fuel batch**

Table 5-1 **Production and decay parameters for selected radionuclides**

Radionuclide	Half Life (d)	Decay Constant (1/d)	Production Rate (Bq/MWt)/d
I-133	8.67E-01	7.99E-01	1.62E+15
I-131	8.04E+00	8.62E-02	8.46E+13
Sr-89	5.05E+01	1.37E-02	1.26E+13
Ce-144	2.84E+02	2.44E-03	3.46E+12
Cs-134	7.53E+02	9.21E-04	3.25E+11
Cs-137	1.10E+04	6.30E-05	1.25E+11
Pu-240	2.39E+06	2.90E-07	4.67E+08

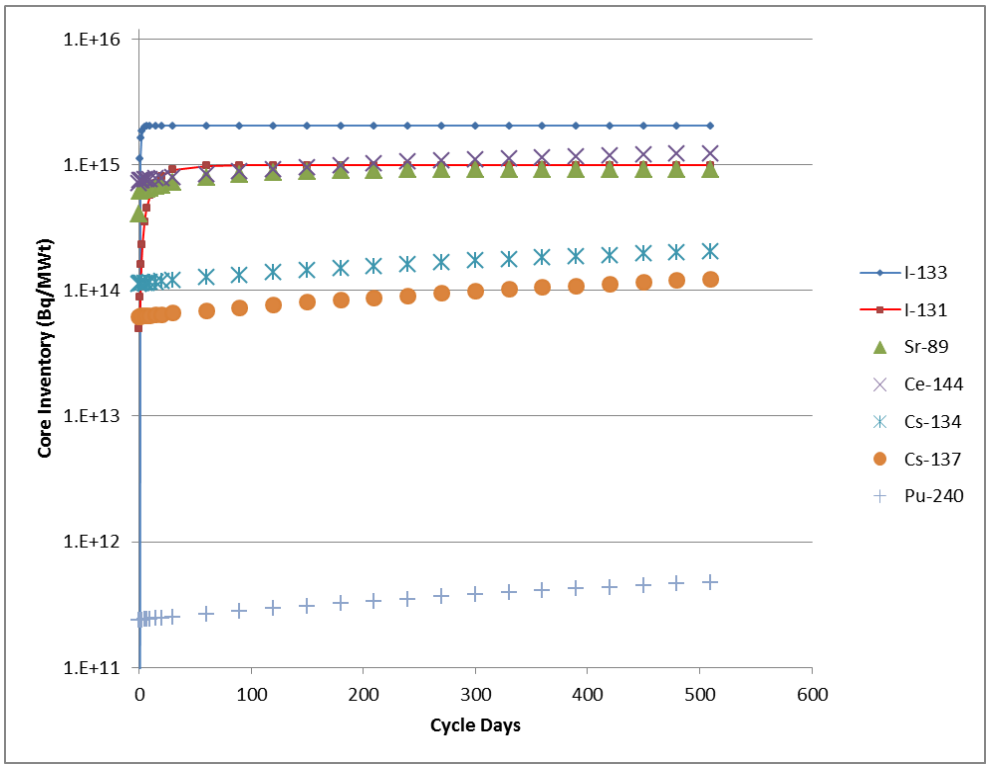


Figure 5-2 Core activity as a function of time during a cycle

5.2 Model Parameter Estimates

The updated core inventory model has a set of parameters that relate to reactor operation, and another set of parameters that are radionuclide specific. The reactor related parameters are the number of fuel cycles, the length of fuel cycle, the average duration in days of a refueling outage, the date of the beginning of the last cycle, the length of the last refueling outage, the amount of uranium in core (MTU) and the reactor power (MWt). Information on reactor operations and fuel is generally available in Chapters 4, 7 and 15 of reactor final safety analysis reports (FSARs).

The radionuclide specific parameters are the half-life and production rate of each radionuclide. NUREG/CR-6703, "Environmental Effects of Extending Fuel Burnup Above 60 GWd/MTU" [21], lists the most important radionuclides in terms of dose. The half-lives of these radionuclides are listed in DCFPAK 3.02. Appendix E of NUREG/CR-6703 lists core-average activity for important radionuclides for PWRs and BWRs for peak rod burnups ranging from 22 GWd/MTU to 75 GWd/MTU based on a series of Oak Ridge Isotope Generator_Automatic Rapid Processing (ORIGEN_ARP) calculations. The radionuclide production constant values can be calculated from the data in NUREG/CR-6703 Appendix E by solving Eq (2) for P'_i given reactor power of 28.3 MWt/MTU for PWRs and 22.22 MWt/MTU for BWRs used in the ORIGEN_ARP calculations. Table 5-2 lists the radionuclide specific parameter values for the updated model.

Table 5-2 Radionuclide specific core inventory parameters for the updated inventory model

Radionuclide	Half-life (d)	Decay Constant (1/d)	RASCAL 4.3 Model Production	
			BWR (Bq/MWt)/d	PWR (Bq/MWt)/d
H-3	4.51E+03	1.54E-04	5.770E+08	5.811E+08
Co-58	7.08E+01	9.79E-03	1.500E+11	1.401E+11
Co-60	1.92E+03	3.61E-04	9.716E+09	1.045E+10
Kr-85m	1.87E-01	3.71E+00	8.925E+14	8.291E+14
Kr-85	3.91E+03	1.77E-04	1.146E+10	1.110E+10
Kr-87	5.30E-02	1.31E+01	6.298E+15	5.846E+15
Kr-88	1.18E-01	5.87E+00	3.913E+15	3.622E+15
Rb-86	1.87E+01	3.71E-02	8.925E+10	9.817E+10
Sr-89	5.05E+01	1.37E-02	1.263E+13	1.169E+13
Sr-90	1.06E+04	6.54E-05	9.395E+10	9.053E+10
Sr-91	3.96E-01	1.75E+00	2.052E+15	1.908E+15
Sr-92	1.13E-01	6.13E+00	7.674E+15	7.229E+15
Y-90	2.67E+00	2.60E-01	4.796E+13	4.224E+13
Y-91	5.85E+01	1.18E-02	1.418E+13	1.336E+13
Y-92	1.48E-01	4.68E+00	5.902E+15	7.108E+11
Y-93	4.21E-01	1.65E+00	1.586E+15	1.513E+15
Zr-95	6.40E+01	1.08E-02	1.835E+13	1.789E+13
Zr-97	7.04E-01	9.85E-01	1.639E+15	1.606E+15
Nb-95	3.52E+01	1.97E-02	3.363E+13	3.284E+13
Mo-99	2.75E+00	2.52E-01	4.663E+14	4.596E+14
Tc-99m	2.51E-01	2.76E+00	4.518E+15	4.498E+15
Ru-103	3.93E+01	1.76E-02	2.774E+13	2.873E+13
Ru-105	1.85E-01	3.75E+00	4.047E+15	4.336E+15
Ru-106	3.68E+02	1.88E-03	1.331E+12	1.446E+12
Rh-105	1.47E+00	4.72E-01	4.860E+14	5.132E+14
Sb-125	1.01E+03	6.86E-04	1.321E+10	1.408E+09
Sb-127	3.85E+00	1.80E-01	1.527E+13	1.597E+13
Te-127m	1.09E+02	6.36E-03	9.172E+10	9.562E+10
Te-127	3.90E-01	1.78E+00	1.492E+14	1.561E+14
Te-129m	3.36E+01	2.06E-02	1.253E+12	1.276E+12

Radionuclide	Half-life (d)	Decay Constant (1/d)	RASCAL 4.3 Model Production	
			BWR (Bq/MWt)/d	PWR (Bq/MWt)/d
Te-129	4.83E-02	1.44E+01	4.301E+15	4.371E+15
Te-131m	1.25E+00	5.55E-01	1.077E+14	1.102E+14
Te-132	3.26E+00	2.13E-01	2.995E+14	2.975E+14
I-131	8.04E+00	8.62E-02	8.458E+13	8.484E+13
I-132	9.58E-02	7.24E+00	1.037E+16	1.035E+16
I-133	8.67E-01	7.99E-01	1.617E+15	1.599E+15
I-134	3.65E-02	1.90E+01	4.239E+16	4.184E+16
I-135	2.75E-01	2.52E+00	4.866E+15	4.827E+15
Xe-133	5.25E+00	1.32E-01	2.638E+14	2.659E+14
Xe-135	3.79E-01	1.83E+00	1.486E+15	1.228E+15
Cs-134	7.53E+02	9.21E-04	3.254E+11	3.592E+11
Cs-136	1.31E+01	5.29E-02	4.929E+12	4.908E+12
Cs-137	1.10E+04	6.30E-05	1.249E+11	1.247E+11
Ba-139	5.74E-02	1.21E+01	2.152E+16	2.108E+16
Ba-140	1.27E+01	5.46E-02	9.727E+13	9.595E+13
La-140	1.68E+00	4.13E-01	7.734E+14	7.552E+14
La-141	1.64E-01	4.23E+00	6.867E+15	6.713E+15
La-142	6.42E-02	1.08E+01	1.713E+16	1.673E+16
Ce-141	3.25E+01	2.13E-02	3.489E+13	3.444E+13
Ce-143	1.38E+00	5.02E-01	7.607E+14	7.383E+14
Ce-144	2.84E+02	2.44E-03	3.464E+12	3.406E+12
Pr-143	1.36E+01	5.10E-02	7.535E+13	7.366E+13
Nd-147	1.10E+01	6.30E-02	4.169E+13	4.130E+13
Np-239	2.36E+00	2.94E-01	5.598E+15	6.030E+15
Pu-238	3.20E+04	2.17E-05	4.020E+09	4.517E+09
Pu-239	8.78E+06	7.89E-08	2.991E+08	2.816E+08
Pu-240	2.39E+06	2.90E-07	4.670E+08	4.372E+08
Pu-241	5.26E+03	1.32E-04	1.440E+11	1.419E+11
Am-241	1.58E+05	4.39E-06	2.343E+08	1.946E+08
Am-242	6.68E-01	1.04E+00	1.604E+14	1.577E+14
Cm-242	1.63E+02	4.25E-03	4.336E+11	4.315E+11
Cm-244	6.61E+03	1.05E-04	3.689E+09	6.095E+09

5.3 Comparison with RASCAL 4.2

RASCAL 4.2 core inventory estimates are based on the result of an ORIGEN_ARP calculation made in December 2003. Full details of the ORIGEN_ARP calculations are not available. However, notes accompanying the inventory data indicate that the calculations were made for reactor power of 3,479 MWt with a burnup of 38,585 MWd /MTU and 17,850 MWd/assembly for a reactor with 193 fuel assemblies. These values give a reactor fuel load of about 89.3 MTU and a 0.463 MTU/assembly. Although the reactor design is not specified in the notes accompanying the ORIGEN_ARP results, the fuel characteristics listed above are consistent with a late generation Westinghouse PWR (possibly Sequoyah Unit 1 or 2).

Table 1-1 of NUREG-1940 lists the core inventory assumed in RASCAL 4.2 normalized to a reactor power of 3,479 MWt. The radionuclide activity used in RASCAL 4.2 is the normalized activity multiplied by the thermal power of the reactor being modeled. An additional adjustment is made for activities of radionuclides having half-lives longer than 1 year. The activities of the long-lived radionuclides are scaled up by the ratio of the core-average fuel burnup (MWd/MTU) to 38,585 MWd/MTU.

Figure 5-3 shows the ratio of the normalized RASCAL 4.3 core activity to normalized RASCAL 4.2 core activity as a function of radionuclide half-life for the radionuclides common to both models. Ratios larger than 1.0 indicate that the RASCAL 4.3 model predicts larger inventory values as compared to RASCAL 4.2. For most radionuclides, the normalized activity calculated with the RASCAL 4.3 model is nearly the same as the activity in RASCAL 4.2. Radionuclides with significantly increased activity are identified in the Figure 5-3. The RASCAL 4.3 model includes eight radionuclides not included in RASCAL 4.2. Those radionuclides are cobalt-58 (Co-58), Co-60, plutonium-238 (Pu-238), Pu-239, Pu-240, americium-241 (Am-241), Am-242, and curium-244 (Cm-244). There are also four radionuclides that were included in the RASCAL 4.2 inventory that are not included in the RASCAL 4.3 model inventory. Those radionuclides are krypton-83m (Kr-83m), antimony-129 (Sb-129), xenon-133m (Xe-133m), and Xe-138. A fifth radionuclide, Xe-135m, is not included in the RASCAL 4.3 inventory production model; however, it may be included in equilibrium with I-135 which is included in the model.

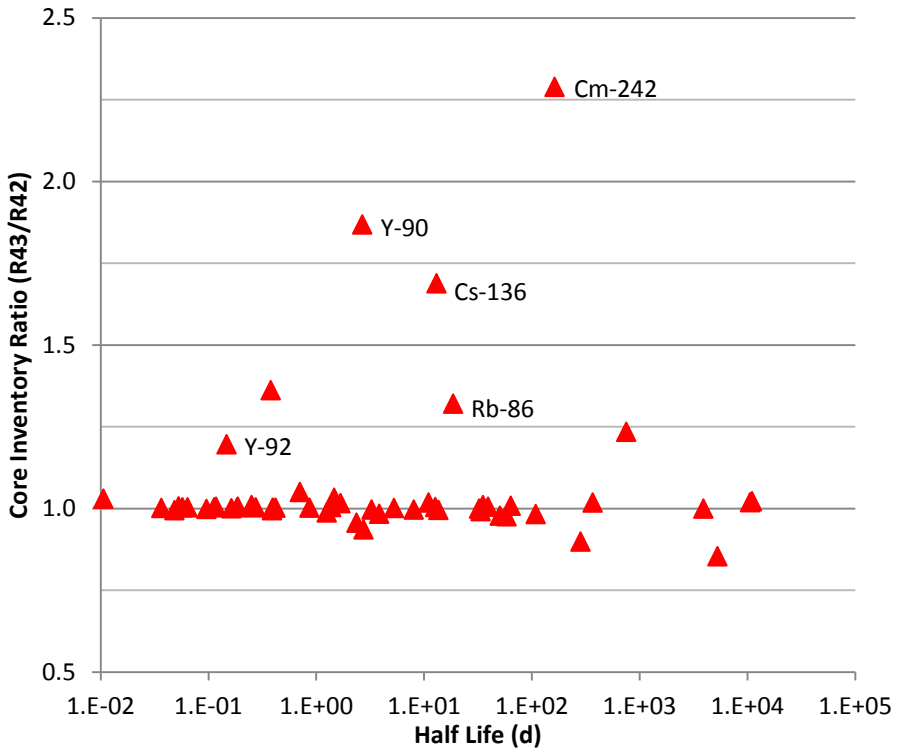


Figure 5-3 Comparison of updated model radionuclide inventory with the RASCAL 4.2 inventory as a function of radionuclide half-life

The RASCAL 4.3 custom fuel inventory utility provides a convenient method of obtaining the core inventory for a specific time in the fuel cycle. The inventory is calculated from the date of last startup and the time in question and the duration of the last outage. Figure 5-4 shows the variation in activity from early cycle (1 month after startup) to the end of cycle (18 months after startup) as a function of radionuclide half-life. Time in cycle is important for radionuclides with half-lives of about 50 days and longer. For some important radionuclides (e.g., Cs-134 and Cs–137), the activity in the core can increase by almost a factor of two during the cycle. The two radionuclides that show significantly larger increases than expected given their half-lives are praseodymium-144 (Pr-144) (half-life = 0.012 d) and tellurium-125m (Te-125m) (half-life = 58 d). These radionuclides are short-lived daughters of longer lived parents (cerium-144 (Ce-144) and Sb-125).

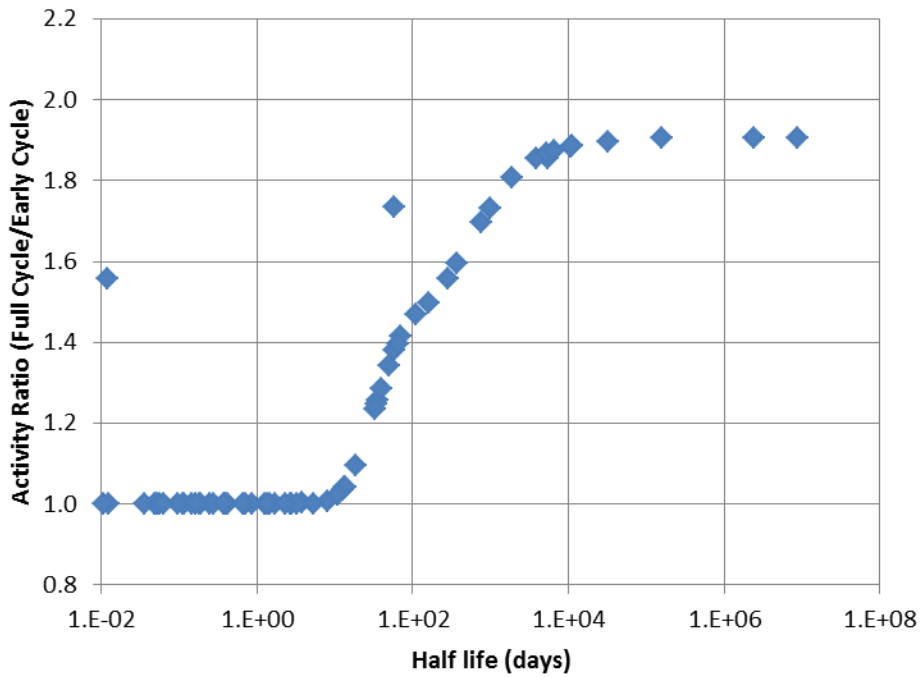


Figure 5-4 Increase in core inventory during the fuel cycle as a function of radionuclide half-life

By default, RASCAL 4.3, RASCAL 4.2 and earlier versions use end-of-cycle inventories. However, because they adjust the inventory of long-lived radionuclides for burnup, they can approximate early-cycle inventories using the average burnup value for the fuel in the reactor on the user interface screen used to select the plant location. The average burnup at the beginning of a cycle is approximately one-half of the burnup at the end of cycle (for three cycle fuel management). For four-cycle fuel management, the burnup at the beginning of cycle is about 60 percent of the end-of-cycle burnup. Figure 5-5 shows the approximate increase in activity during the fuel cycle as a function of half-life that would be estimated by RASCAL 4.2 and RASCAL 4.3 from the default inventory using the average burnup. Comparing Figures 5-4 and 5-5, it is clear that the approximate method based on adjustment of the default end-of-cycle inventory gives reasonable results for most radionuclides. However, it does not properly account for the increase in Pr-144 and Te-125m. The choice between using a custom inventory or a burnup adjusted end-of-cycle inventory may be reasonably made based on the availability of information at the time of an event, refueling dates, etc., or average core burnup.

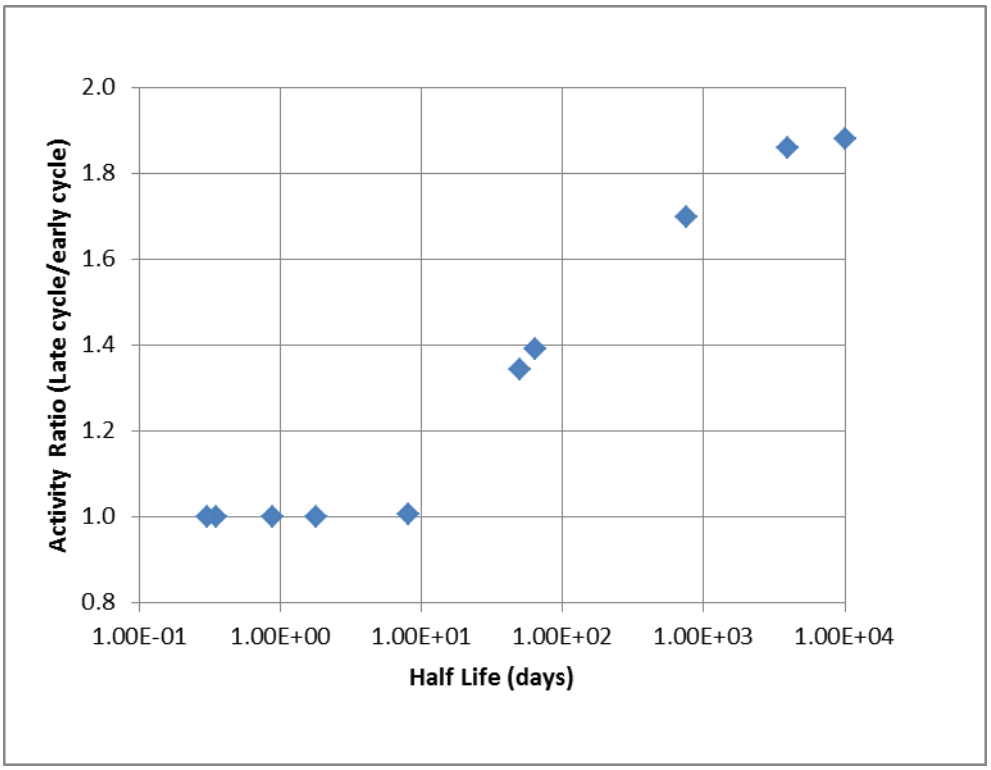


Figure 5-5 Radionuclide activity increase during the fuel cycle as a function of half-life estimated from end-of-cycle inventory and burnup

5.4 Model Validation

The updated core inventory model has been used to estimate core inventories for five current generation reactor types and four new generation reactor types. In each case, the reactor specific parameters have been selected to match the conditions used to estimate the core inventories in the current generation reactor FSARs and the new reactor design control documents (DCDs). Table 5-3 lists the reactor specific model parameters for each reactor type modeled. For each type, the fuel load and reactor power were matched to the data in the FSAR or DCD [22, 23, 24, and 25], when available. When the data were not available, data were obtained from the reactor characteristics in NUREG/CR-6703. Except for the Evolutionary Power Reactor (EPR), the number of cycles and cycle exposure (effective full power days (EFPD)) were adjusted until the updated core inventory model gave fuel burnup consistent with the burnup used for the FSAR or DCD core inventory. The DCD core inventory for the EPR is a composite based on a set of burnups ranging from 5 to 62 GWD/MTU. The core inventory in the EPR DCD is the maximum activity for each radionuclide from the set of burnups.

Figures 5-6 and 5-7 compare the updated core inventory model estimates of core activity with core activity estimates included in FSARs and DCDs. Core estimates for six types of current generation U.S. reactors are compared in Figure 5-6. These comparisons indicate that the updated core inventory model provides reasonable core inventory estimates for the current fleet of U.S. reactors. Similarly, the comparisons of the updated model core inventory estimates with

DCD core inventories for new generation reactors shown in Figure 5-7 indicate that the model also provides reasonable core inventory estimates for new reactors.

Table 5-3 Reactor specific characteristics used to test the updated core inventory model

Reactor Type	Number of Cycles	Cycle Exposure (EFPD)	Fuel load (MTU)	Power (MWt)	Core Ave Burnup (GWd/MTU)	Max Burnup (GWd/MTU)
W 2-loop	3	460	46.1	1783	35.59	53.40
W 4-loop	3	509	104.3	4100	40.00	59.99
CE	3	535	95.5	3507	39.65	60.00
B&W	3	480	2568	2619	28.47	42.71
GE 3	3	670	150	3016	24.20	32.2
GE 4	3	487	104.3	3499	23.82	35.01
AP1000	3	506	84.8	3468	41.36	62.00
ESBWR	3	639	160	4590	35.00	51.03
US-APWR	3	730	138.7	4540	34.76	54.99
EPR	3	517	128.9	4590	41.30	61.98

Assuming that the FSAR and DCD core inventories are the true inventories, the ratio of model inventory estimates to the FSAR and DCD values should be approximately log-normal distributed. If the model is unbiased, the geometric mean and median ratios should be 1.00. The geometric standard deviation (GSD) characterizes the scatter of the estimates. A GSD of 1.0 would indicate a perfect model. Otherwise, about 68 percent of the ratios would fall in the range between 1/GSD and GSD, and about 95.5 percent of the ratios would fall in the range between 1/(2×GSD) and 2×GSD.

Tables 5-4 and 5-5 present model statistics for each reactor type and average statistics for the nine reactor types. Based on these estimates, it is reasonable to conclude that the updated core inventory model is unbiased with a GSD of about 1.3. About 70 percent of the ratios would be expected to fall within the range 0.8 to 1.3 and about 95 percent of the values would be in the range of 0.4 to 2.6. Table 5-5 presents the ratios for 6 radionuclides of interest. There is considerably more scatter at this level of detail. Since the updated model provides consistent inventory estimates given the input, the scatter may, in large part, be due to the modeling assumptions that were made in estimating the core inventories for the FSARs and DCDs. The variation in ratios for curium-242 (Cm-242) is a case in point. Estimates of the core inventory of actinides have increased in recent years. Thus, the variation in ratios for Cm-242m inventory estimates may be associated with the age of the FSAR/DCD core inventory estimate.

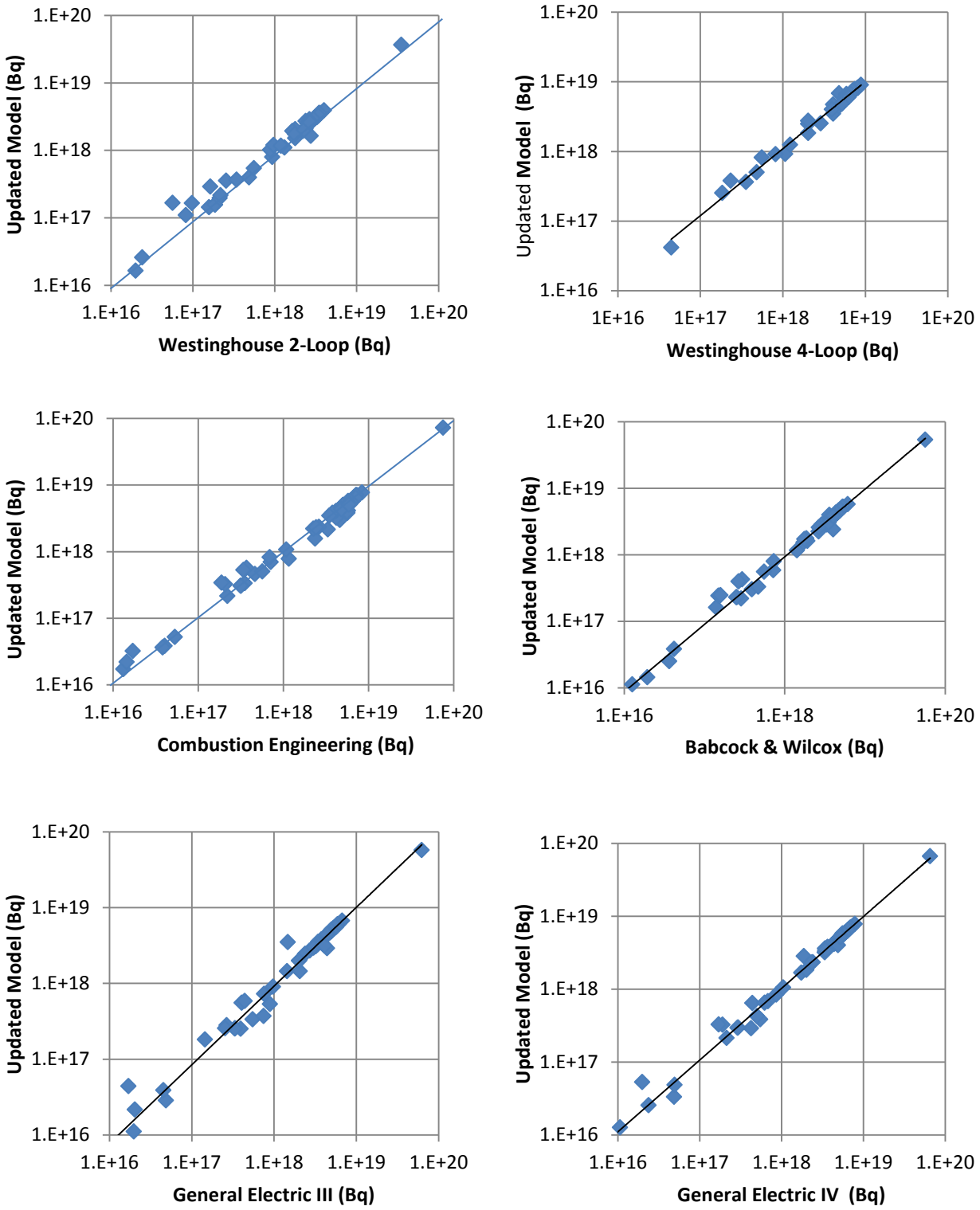


Figure 5-6 Comparison of the updated core inventory estimates with core inventory estimates for current generation reactors

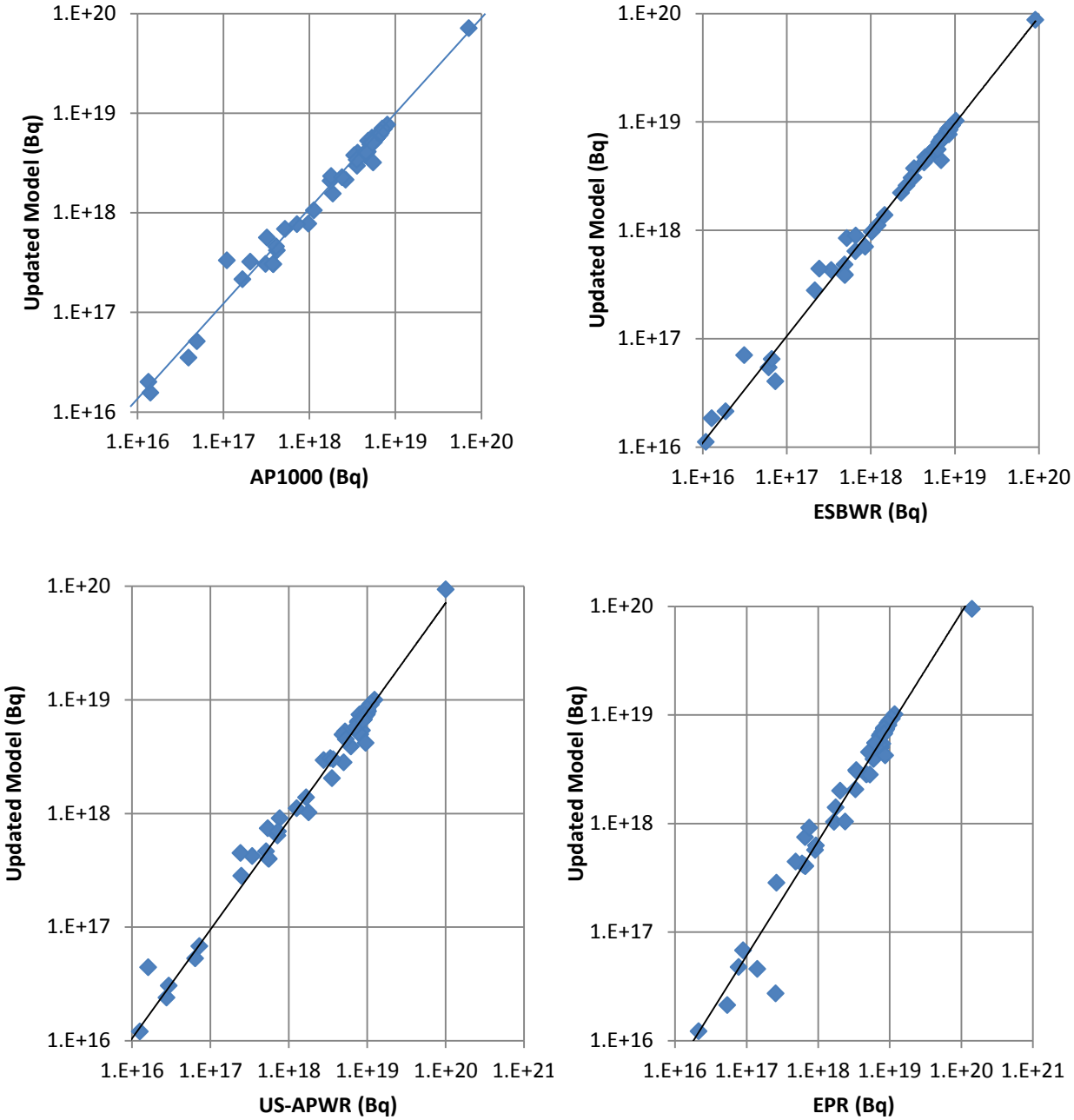


Figure 5-7 Comparison of updated core inventory estimates with core inventory estimates for new reactors

Table 5-4 Statistics of the ratios of the updated core inventory model estimates to the FSAR and DCD inventory estimates

Reactor Type	n	median	GSM	GSD
W 2-loop	57	0.990	0.172	1.287
W 4-loop	33	0.997	0.179	1.170
CE	62	0.919	0.918	1.411
B&W	57	0.965	0.994	1.186
GE 3	59	0.989	0.920	1.351
GE 4	59	0.995	1.028	1.264
AP1000	57	0.947	1.007	1.271
ESBWR	59	0.980	1.006	1.231
US-APWR	58	0.825	0.849	1.317
EPR	58	0.797	0.712	1.435
Average		0.940	0.954	1.292
Standard Deviation		0.073	0.108	0.088

Table 5-5 Statistics of the ratios of the updated core inventory model estimates to the FSAR and DCD inventory estimates for radionuclides of interest

Reactor Type	I-131	Sr-89	Sr-90	Cs-134	Cs-137	Cm-242
W 2-loop	0.994	0.852	1.180	1.249	1.203	3.038
W 4-loop	1.004	0.838	0.997	1.102	1.027	--
CE	0.995	0.647	0.610	0.889	1.094	1.687
B&W	0.970	0.839	1.077	1.229	1.094	1.639
GE 3	0.979	1.004	0.640	0.596	0.611	1.004
GE 4	0.997	0.950	0.682	1.05	0.700	1.004
AP1000	0.957	0.786	0.992	1.069	1.005	2.923
ESBWR	0.961	0.975	0.988	0.866	0.872	1.800
US-APWR	0.822	0.626	0.900	0.805	0.781	1.849
EPR	0.861	0.656	0.660	0.425	0.614	0.914
Average	0.954	0.817	0.873	0.928	0.854	1.863
St. Dev.	0.062	0.138	0.207	0.267	0.217	0.727

5.5 Cooling System Model

The starting point for the updated coolant inventory model is a variant of Equation 5-1, which was the starting point for the core inventory model. It is:

$$\frac{dC_i(t)}{dt} = \frac{S_i'(t)}{M} - (\lambda_{i+} + R_i) \times C_i(t) \tag{5-6}$$

where

C_i(t) = coolant concentration of radionuclide i (Bq/kg) at time t,
S'_i(t) = appearance rate of radionuclide i in (Bq/h) in the coolant,
M = total mass of coolant (kg),
λ_i = decay constant (h^{-1}) for radionuclide i,
R_i = combined removal rate (h^{-1}) for radionuclide i for all of the processes other than decay that remove radionuclides from the coolant.

The appearance rate of activity in the reactor coolant should be related to the core activity inventory. As seen in the previous section, except for short-lived radionuclides at reactor startup, the core inventory of a radionuclide tends to be nearly constant for periods of a few

days. The removal rate for radionuclides in a reactor coolant system is typically of the order of 1 per hour. As a result, it is reasonable to assume that under normal operating conditions, the radionuclide activity in coolant is in equilibrium with the core activity and dCi (t)/dt = 0.0. Thus, the activity concentration in the primary coolant can be calculated as:

$$C_i(t) = \frac{S_i'(t)}{[M \times (\lambda_i + R_i)]} \tag{5-7}$$

The following develops this model for BWR and PWR reactors. The development is based on the procedures for estimating reactor coolant activity set forth in NUREG-0016 [26], NUREG-0017 [27], and ANSI/ANS-18.1-1999 [28]. These documents provide coolant inventories for three reference reactors: a BWR, a PWR with U-tube steam generators, and a PWR with once-through steam generators. They also provide details related to the calculation of the removal rates for each reactor type. Unfortunately, the documents do not provide any information related to estimating the radionuclide appearance rates.

Assuming that as a practical matter it is not realistic to try to characterize the cause of variations in appearance rate as either variations in fraction of fuel defects or variations in escape coefficients, the RASCAL 4.3 coolant inventory model assumes:

$$S_i'(t) = A_i(t) \times Es_i \tag{5-8}$$

where

$A_i(t)$ = core inventory of radionuclide i (Bq),
Es_i = escape rate coefficient.

The Rx_Inventory_S code calculates $A_i(t)$, and default values of Es_i for various reactor types are included in data statements in the code. For daughter radionuclides, the appearance rate $S'_i(t)$ has an additional term to represent ingrowth in the reactor cooling system. This ingrowth term is shown in Equation 5-9:

$$S_i'(t) = A_i(t) \times Es_i + A_{cj}(t) \times \lambda_j \times BF_j \tag{5-9}$$

where

$A_{cj}(t)$ = activity of the parent in the coolant,
λ_j = decay constant for the parent,
BF_j = fraction of disintegrations of the parent j that yield daughter i.

There is also activity in the coolant that does not come from the reactor core. There are corrosion products that are not a function of fuel defects. ANSI/ANS-18.1 and new reactor DCDs list and provide estimates of coolant concentrations for chromium-58 (Cr-58), Cr-60, zirconium-95 (Zr-95) and Zr-97 and a number of other activation products. The only activation products in the reactor coolant system (RCS) other than the isotopes of Cr and Zr that might

contribute significantly to radiological doses are sodium-24 (Na-24), manganese-54 (Mn-54), Mn-56, copper-64 (Cu-64) and silver-110m (Ag-110m). Appearance rates for these radionuclides are included in the inventory calculation module as appropriate for each reactor type.

For the purpose of estimating coolant activity, a BWR reactor coolant system has two flow paths. The first path is through the reactor water cleanup system, and the second path is the steam flow path through the turbine. Thus, the removal rate for most radionuclides for a BWR reactor is given by Equation 5-10 (these variable symbols are used in ANSI/ANS-18.1):

$$R_i = \frac{(FA \times NA_i) + (FS \times NS_i \times NC_i \times NB_i)}{WP} \tag{5-10}$$

where

R_i	= removal rate of the radionuclide i from the RCS (h^{-1}),
FA	= flow through the purification system cation demineralizer (kg/h),
NA_i	= fraction of radionuclide i removed in the RCS cleanup system,
FS	= steam flow rate (kg/h),
NS_i	= ratio of concentration radionuclide i in reactor steam to the concentration in reactor water,
NC_i	= ratio of condensate demineralizer flow rate to the total steam flow rate,
NB_i	= fraction of radionuclide i removed by the condensate demineralizers,
WP	= weight of water in the RCS (kg).

For the 3.4E+03 MWt reference reactor in ANSI/ANS-18.1, the mass of coolant in the reactor coolant system is 1.7E+05 kg, the reactor cleanup system flow is 5.76E+04 kg/h, the steam flow is 6.84E+06 kg/h, and 100 percent of the steam flow passes through the condensate demineralizer.

PWRs are divided into two classes based on steam generator design, U-tube steam generators and once-through steam generators. However, the same basic equations apply to both classes; the difference between the classes is captured in the model parameter values. There are two flow paths for water passing through the reactor, a path through the steam generator (primary side) and a path though the reactor cleanup system. The steam driving the turbine is in a separate path. This path includes the steam generator (secondary side), the turbines, the condenser, and a condensate cleanup system. Activity in the primary coolant (primary side water) comes directly from the reactor. It is removed by the reactor water cleanup system. Activity in the secondary side steam and condensate comes from primary to secondary leakage in the steam generator. It is removed by the condensate cleanup system. Equation 5-7 applies separately to the primary and secondary side. Equation 5-9 defines the appearance rate on the primary side. A similar equation is used to define the appearance rate on the secondary side.

$$s_{i(t)} = C_i(t) \times L \times C_j(t) \times \lambda_j \times BF_j \tag{5-11}$$

Where

s_i(t) = secondary side appearance rate (Bq/h),
C_i(t) = concentration in the primary coolant (Bq/g),
L = primary to secondary leak rate (gal/h),
C_j(t) = concentration of the parent in the secondary coolant (Bq/g),
λ_j = decay constant of the parent (h^{-1}),
BF_j = fraction of disintegrations of parent j that yield daughter i.

ANSI/ANS-18.1 secondary coolant inventories are based on a primary to secondary leak rate of 1.4E+00 kg/h (75 lbs/d). This value is used to estimate realistic secondary side inventories in the Advanced Passive Reactor (AP1000), EPR and US-Advanced Pressurized-Water Reactor (US-APWR) DCDs. Design-basis secondary side coolant inventories for these reactors are based on significantly larger primary to secondary leak rates (e.g., 4.9E+05 gal/h for the AP1000).

The ANSI/ANS-18.1 PWR model for removal of activity in primary side coolant consists of a purification demineralizer followed by a cation demineralizer. Treated water is stored in a purification system tank where it may be degassed. A portion of the flow may be diverted to the boron recovery system. Using the nomenclature of ANSI/ANS-18.1, the removal rate for primary side coolant noble gas activity is:

$$R = \frac{[FB + (FD - FB) \times Y]}{WP} \tag{5-12}$$

where

R = removal rate (h^{-1}),
FB = RCS flow to the boron recovery system (kg/h),
FD = flow through the purification demineralizer (kg/h),
Y = fraction of noble gases removed by degassing,
WP = mass of the primary coolant (kg).

The removal rate is for other radionuclides, except H_3, is:

$$R = \frac{\left[(FD \times NB_i) + (1 - NB_i) \times \left(FB + (FA \times NA_i)\right)\right]}{WP} \tag{5-13}$$

where

R = removal rate (h^{-1}),
FD = flow through the purification demineralizer (kg/h),
NB_i = fraction of activity of radionuclide i removed in the purification demineralizer,
FB = RCS flow to the boron recovery system (kg/h),
FA = flow through the cation demineralizer (kg/h),
NA_i = fraction of activity of radionuclide i removed in the cation demineralizer.

The ANSI/ANS-18.1 model does not include a removal mechanism for H_3 other than decay.

On the secondary side, noble gases are assumed to transfer to the steam immediately and to be effectively removed by the condenser offgas system. Therefore, the concentration of noble gases in the condensate is assumed to be 0.0 and the concentration in steam is calculated as:

$$C_i(t) = \frac{s_i(\mathrm{t})}{FS} \tag{5-14}$$

where

C_i(t)	= concentration of radionuclide i at time (t) (μCi/g),
s_i(t)	= rate of introduction of radionuclide i at time (t) (Ci/s),
FS	= steam flow (kg/h).

Equation 5-15 is used to calculate the secondary side concentrations of H_3 and other nuclides. For these radionuclides, the secondary side removal rate is calculated as:

$$R_i = \frac{[(FBD \times NBD) + (NS_i \times FS \times NC \times NX_i)]}{WS} \tag{5-15}$$

where

FBD	= steam generator blowdown flow (kg/h),
NBD	= fraction of blowdown removed from secondary side,
NS_i	= ratio radionuclide i concentration in steam to that steam generator,
FS	= steam flow (kg/h),
NC	= fraction of steam flow passing through condensate demineralizer,
NX_i	= fraction of radionuclide i removed by condensate demineralizer.

The Rx_Inventory_S code includes default values for R and NS for various reactor types in data statements. The default values are based on the ANSI/ANS-18.1 model. They were calculated for each reactor type using the RCS parameters given in reference documents. The flows and cleanup fractions for current-generation light water reactors are based on the BWR and PWR reference reactors in ANSI/ANS-18.1. The flows and cleanup fractions for new reactor designs are based on reactor characteristics as set forth in Section 11.1 of the reactor DCD. Where the DCD includes design basis and realistic characteristics, the default values of R are based on the realistic values.

Default escape coefficients have also been calculated for each reactor type using the published primary coolant concentrations and RCS parameters. These coefficients, which are applied to individual radionuclides in the core, have units of fraction of a core activity per hour. They are included in data statements in the Rx_Inventory_S code.

5.6 Inventory Utility: Rx_Inventory_S

The inventory model Rx_Inventory_S uses two input files. One of the input files contains the radionuclide decay and production data. It is named Coreinventory.dat. The other input file contains user inputs related to the reactor and cooling system. It is named Reactor.tmp. This

file has two parts. The first section (first seven lines) provides information about the reactor: its power, how it is operated, and the date of the last refueling outage. This information is used to calculate the core inventory. The second section provides information on the cooling and steam systems. There is one line in this section for BWRs and two lines for PWRs. Table 5-6 lists the format content of each of the lines in the Reactor.tmp file.

5.6.1 Core Inventory Input

The first line of the Reactor.tmp file is used to identify the information in the file. This line is copied to the output files. The next six lines contain the data used to run the core inventory model. Each of the remaining lines starts with a key word or phrase that identifies the information in the record. The keyword is followed by the input.

The second line identifies the reactor type. The core inventory model recognizes seven reactor types and selects the PWR or BWR radionuclide production parameters based on reactor type. The reactor types are: BWR, Advanced Boiling-Water Reactor (ABWR), Economic Simplified Boiling-Water Reactor (ESBWR), PWR, US-APWR, EPR, and AP1000.

The third line lists the reactor power (MWt) and core uranium load (MTU). The characteristics of U.S. reactors, including rated power and fuel load, are listed in Appendix A of NUREG/CR-6703. However, power uprates since 1997 may have resulted in changes to characteristics of these reactors. The results of power uprates are listed on the NRC Web site at: http://www.nrc.gov/reactors/operating/licensing/power-uprates/status-power-apps.html.

The fourth line in this section lists the fuel management practices for the reactor the first entry is a key to typical management styles. Table 5-7 gives the number of cycles that fuel remains in the core and the cycle duration represented by each key number. The second entry in this line is the cycle of the oldest batch of fuel in the core. In general, it should be equal to the number of cycles. This entry may be less than the number of cycles when the reactor is just starting up. If the entry is greater than the number of cycles, the model will assume that it is equal to the number of cycles. The final entry is the length in days of a typical refueling outage. Figure 5-8 shows the distribution of the duration of refueling outages in days for the 60 U.S. reactors that refueled in 2011.

Table 5-6 Reactor.tmp file format and contents

Line	Format	Variables and Keywords	Notes
1	a80	header	
2	a20, 1x, a6	reactor type	BWR, ABWR, ESBWR, PWR, AP1000, US-APWR, or EPR
3	a20, 2f10.0	reactor power (MWt), total uranium (MTU)	
4	a20, 3i5	fuel management type, cycle, normal refueling outage (d)	
5	a20, 4i5	month, day, year, duration	End of last refueling outage, outage duration (d)
6	a20, 3i5	month, day, year	Date of initiating event
7	a20, 1x, a10	status	Status at time of initiating event: normal operation, shutdown, low power
BWR			
8	a20, 4f9.0	total coolant mass (kg), steam flow (kg/h), condensate cleanup flow fraction	WP, FS, NC
PWR			
8	a20, 5f10.0	primary coolant mass (kg)	WP
9	a20, 6f9.0	steam generator type, primary to secondary leak rate, total secondary coolant mass (kg), steam flow (kg/h)	SG_type (‘U-Tube’ or ‘Once-Through’), L, WS, FS

Table 5-7 Fuel management characteristics

Key	Number of Cycles	Cycle Duration (d)
1	2	730
2	3	547
3	3	730
4	4	365
5	3	365

Outage durations ranged from 25 to 183 d with a median duration of 41 d and a mean duration of 55 d. The longer duration outages generally indicate major reactor maintenance activities took place in addition to refueling.

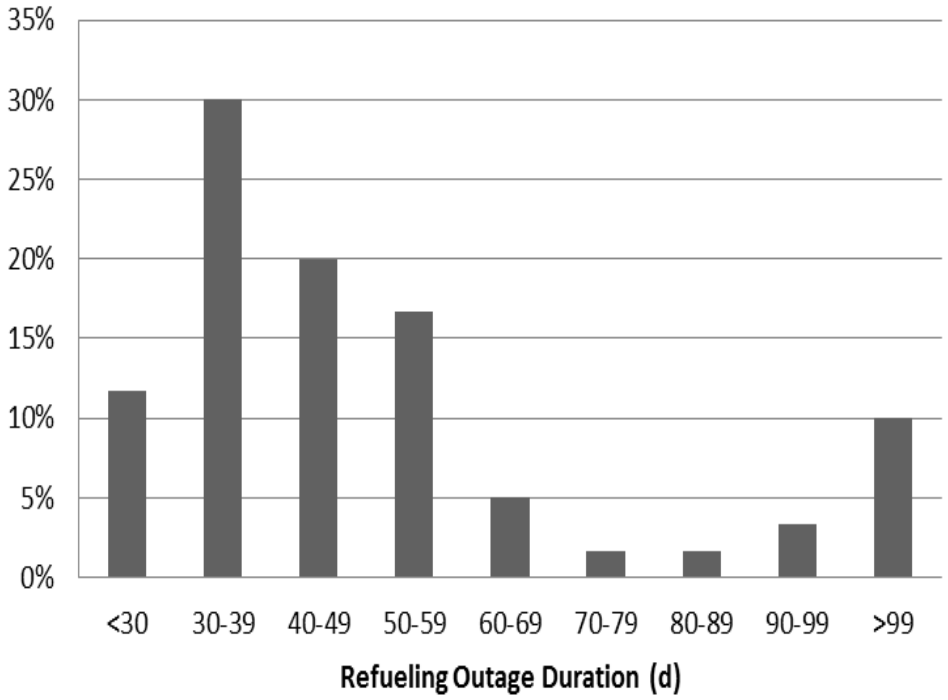


Figure 5-8 Distribution of refueling outage duration in 2011

The last three lines in this section are related to the most recent fuel cycle. The fifth line in the first section contains the date that the reactor restarted after the last refueling outage and the duration of the outage. The date that U.S reactor restarted after the last refueling outage can be found by searching the daily status reports. These reports are found at: http://www.nrc.gov/reading-rm/doc-collections/event-status/reactor-status/.

Recent status reports only list the status of the reactor as percent of rated power. Status reports more than 28-days old include additional information such as the reason for an outage.

The sixth line contains the date for which the core inventory is to be calculated. Typically this line would contain the date of the event.

The last line of this section is the status of the reactor at the time of the event. The status is not used in the current model. However, future model development will include source term estimates for low-power events and events that occur when the reactor is shutdown.

5.6.2 Cooling and Steam System Input

The second section of the Reactor.tmp file contains information related to the reactor cooling system. The contents of this section are determined by reactor type (BWR or PWR). Table 5-6

lists the input for the two basic reactor types. The variable names in this section of the input file are the same as the names used in ANSI/ANS-18.1. For BWRs (current generation, ABWR, and ESBWR) the input is the mass of water in the reactor cooling system, the steam flow and the fraction of the condensate that returns through a cleanup system. This information is input in a single line in the file.

The mass of water in the coolant system is included in the RASCAL 4.3 facility database. The steam flow may be obtained from the plant FSAR or DCD. If it isn't readily available, the steam flow may be estimated from reactor thermal and electrical power. Figure 5-9 shows the relationship between design-basis steam flow and design-basis thermal power for current generation and new nuclear power plant designs.

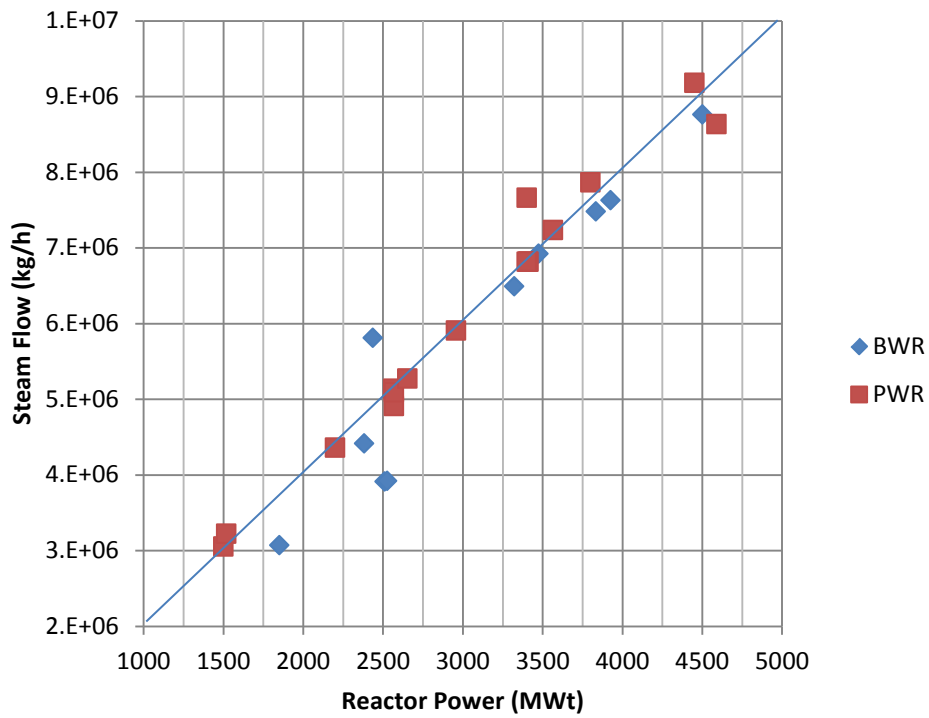


Figure 5-9 Steam flow as function of reactor power

To a very good approximation, the steam flow may be estimated from the thermal power using:

$$FS = \frac{2000\ (kg/h)}{MWt} \times P \tag{5-16}$$

where

FS = steam flow (kg/h),
P = reactor power (MWt).

This relationship is based on the design values of power and steam flow; however, RASCAL 4.3 also uses the relationship to estimate steam flow at lower power levels when the reactor is operating at less than design power.

The ANSI/ANS-18.1 gives a nominal value of 1.0 for the ratio of condensate demineralizer flow to steam flow (NC) for BWR reactors. It also has a note that the reference plant is assumed to be a “non-pumped forward drained plant,” and states that for a “pumped forward plant” the value of NC should be consistent with the plant design values. The value of NC for most of the BWR plants reviewed was 1.0. However, three plants had NC values in the range of 0.65 to 0.67. Lower values of NC result in larger coolant inventories. If the specific BWR reactor being modeled does not have a condensate cleanup system, the condensate cleanup fraction should be set to zero.

For PWRs (current generation, AP1000, US-APWR, or EPR), the cooling system information is included in two lines in the file. The first line contains the mass of water on the primary cooling side. The second line lists the steam generator type (U-tube or once-through), the primary to secondary leak rate, the mass of water in the secondary side, and the steam flow. Most PWRs use U-tube steam generators. Babcock and Wilcox (B&W) reactors are the exception; they use once-through steam generators (B&W reactors in the United States are: Crystal River Unit 3; Oconee Units 1, 2, and 3; Arkansas Unit 1; Three Mile Island Unit 1; and Davis-Besse Unit 1).

The RASCAL 4.3 facility database includes the primary system water mass and the steam generator type for U.S. reactors. The primary to secondary leak rate is a design value that should be available from the plant FSAR or DCD. A primary to secondary leak rate of 75 lb/d (1.4E+00 kg/h) is assumed in ANSI/ANS-18.1. The AP1000 DCD assumes a primary to secondary leak rate of 500 gal/d (approximately 4000 lb/d) for design-basis calculations and a primary to secondary leak rate of 75 lb/d (9 gal/d) as a realistic value. Given no better information, the primary to secondary leak rate might be estimated as the difference between the primary makeup flow and the primary letdown flow. The mass of the secondary side water should be available from design documents such as the FSAR and DCD. If it is not readily available from a design document, the secondary water mass may be estimated from the reactor design power. Figure 5-10 shows the relationship between secondary water mass and reactor power for current generation and new power plant designs.

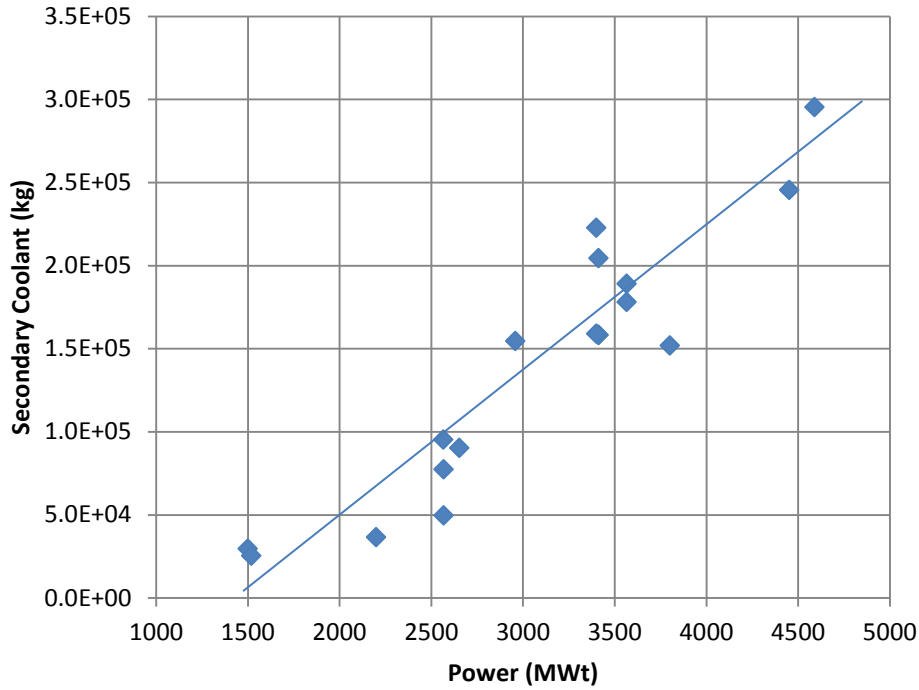


Figure 5-10 Secondary water mass as a function of reactor power for current generation and new PWR reactor designs

When necessary, RASCAL 4.3 estimates the secondary water mass for PWR using:

$$WS = 85.7 \times P - 1.24\mathrm{E} + 05 \tag{5-17}$$

where

WS = secondary water mass (kg),
P = reactor design power (MWt).

RASCAL 4.3 estimates the steam flow on the secondary side using Equation 5-16.

5.6.3 Rx_Inventory_S Model Output

The Rx_Inventory_S program generates four output files. Three of these files provided inventories for the core, the coolant, and spent fuel pool (SFP). The remaining file is a debug file that contains intermediate computational results. This section describes the inventory files.

5.6.3.1 Core Inventory

The core inventory file (core_inventory.tmp) is intended for use by the reactor accident source term module. Figure 5-11 presents a portion of a core inventory file. The first line states that the file is a reactor core inventory. The next seven lines echo the user input used to calculate the inventory. These lines are followed by the year and day of the year for which the core

inventory was calculated, the core-average burnup of the fuel, and the number of radionuclides in the inventory. There are other radionuclides in the inventory, but the radionuclides in this file contribute most to potential offsite consequences. The remainder of the file is a listing of the activity (Bq) of the radionuclide inventory.

```
Reactor core inventory estimates
ANSI/ANS-18.1 Reference BWR
Reactor Type         BWR
Reactor Parameters       3400.0     140.3
Fuel Management         2    3   60
Last Startup Date       3   24 2010   60
Shutdown Date           5   28 2011
Reactor Status      Operating
Exposure date,  year=   2011  day=   148
Core average burnup, GWd/MTU     2.314E+01
Total radionuclides =     67
 Nuclide       Bq
H-3           1.65E+15
Co-58         5.16E+16
Co-60         2.48E+16
Kr-85m        8.19E+17
Kr-85         3.23E+16
Kr-87         1.64E+18
Kr-88         2.26E+18
Rb-86         8.19E+15
Rb-88         2.26E+18
Sr-89         3.12E+18
Sr-90         2.82E+17
Sr-91         3.99E+18
Sr-92         4.25E+18
```

Figure 5-11 Sample core_inventory.tmp file

5.6.3.2 Coolant Inventory

Cooling system inventory estimates are stored in the file rcs_inventory.tmp. This file lists the inventories that would be used to estimate source terms for releases of coolant to the environment that do not involve damage to the reactor core, e.g., steam generator tube ruptures. Again, the first line of the file describes the contents of the file and the following three lines for a PWR (two lines for a BWR) list the reactor parameter inputs provided by the user. The next four lines list the RCS parameters that vary by radionuclide. These parameters, which are included in Rx_Inventory_S data statements, are listed by the MELCOR radionuclide classes plus a class for tritium. The MELCOR classes are listed in Table 5-8.

Table 5-8 MELCOR radionuclide classes

MELCOR Class Number	Radionuclide Class
1	noble gases (Xe)
2	cesium (Cs)
3	barium (Ba)
4	iodine (I)
5	tellurium (Te)
6	ruthenium (Ru)
7	molybdenum (Mo)
8	cerium (Ce)
9	lanthanum (La)
10	uranium (U)
11	cadmium (Cd)
12	tin (Sn)
16	cesium iodide (CsI)

Many of the parameter values in each of these lines are the same because the parameter values in the reference documents are generally only given for four radionuclide classes, noble gases, I, Cs and rubidium (Rb), and other radionuclides. The escape rates are in fraction of core activity per hour, the primary and secondary cleanup values are removal rates per hour, and carry over is the fraction of activity in the secondary water that carries over into the steam. The remainder of the file presents the RCS inventory estimates. Figure 5-12 presents a portion of a PWR rcs_inventory.tmp file. Each radionuclide is followed by its estimated concentrations (Bq/g) in the primary coolant, the secondary water, and in the steam.

```
Cooling system inventory estimates
ANSI Reference PWR
RCS Primary Param         2.50E+05
RCS Secondary Param   U-Tube            1.40E+00    2.05E+05    6.84E+06
Escape Rates  7.00E-08  2.00E-10  5.00E-10  1.00E-08  5.00E-10  5.00E-10
5.00E-10  5.00E-10  5.00E-10  5.00E-10  5.00E-10  0.00E+00
Primary Cleanup  9.20E-04  3.71E-02  6.61E-02  6.66E-02  6.61E-02  6.61E-02
6.61E-02  6.61E-02  6.61E-02  6.61E-02  6.61E-02  0.00E+00
Carry Over  1.00E+00  5.00E-03  5.00E-03  1.00E-02  5.00E-03  5.00E-03  5.00E-
03  5.00E-03  5.00E-03  5.00E-03  5.00E-03  1.00E+00
Secondary Cleanup  0.00E+00  1.67E-01  1.67E-01  1.67E-01  1.67E-01  1.67E-01
1.67E-01  1.67E-01  1.67E-01  1.67E-01  1.67E-01  0.00E+00
Coolant inventory concentrations in Bq/g
           Primary Secondary     Steam
H-3       3.70E+04  3.95E+04  3.95E+04
Co-58     2.89E+01  1.18E-03  1.18E-05
Co-60     1.76E+01  7.18E-04  7.18E-06
Kr-85m    1.37E+03  0.00E+00  2.81E-04
Kr-85     1.04E+04  0.00E+00  2.13E-03
Kr-87     7.80E+02  0.00E+00  1.60E-04
Kr-88     2.39E+03  0.00E+00  4.89E-04
Rb-86     1.86E-01  7.55E-06  3.78E-08
Rb-88     7.09E-01  1.94E-06  9.70E-09
Sr-89     8.68E+01  3.54E-03  1.77E-05
Sr-90     9.10E+00  3.72E-04  1.86E-06
Sr-91     5.33E+01  1.52E-03  7.59E-06
Sr-92     2.49E+01  4.03E-04  2.01E-06
```

Figure 5-12 Sample PWR rcs_inventory.tmp file

The BWR rcs_inventory.tmp file is similar to the PWR rcs_inventory.tmp file with minor exceptions. There is only one line of RCS parameter values in the user input; the secondary water cleanup removal rates are replaced by removal rates for a condensate cleanup system; and radionuclide concentrations are only provided for water and steam. Figure 5-13 presents a portion of a BWR rcs_inventory.tmp file.

```
Cooling system inventory estimates
ANSI/ANS-18.1 Reference BWR
RCS Parameters        1.70E+05  6.84E+06  1.00E+00
Escape Rates  3.00E-07  1.00E-10  2.00E-10  2.00E-08  2.00E-10  2.00E-10
2.00E-10  2.00E-10  2.00E-10  2.00E-10  2.00E-10  0.00E+00
Primary Cleanup  0.00E+00  1.71E-01  3.08E-01  3.08E-01  3.08E-01  3.08E-01
3.08E-01  3.08E-01  3.08E-01  3.08E-01  3.08E-01  0.00E+00
Carry Over  1.00E+00  2.00E-02  1.00E-03  2.00E-02  1.00E-03  1.00E-03
1.00E-03  1.00E-03  1.00E-03  1.00E-03  1.00E-03  1.00E+00
Condensate Cleanup  0.00E+00  1.97E-02  3.55E-02  7.11E-01  3.55E-02  3.55E-
02  3.55E-02  3.55E-02  3.55E-02  3.55E-02  3.55E-02  0.00E+00
Coolant inventory concentrations in Bq/g
             Water     Steam
H-3       3.70E+02  3.70E+02
Co-58     5.96E+00  1.19E-01
Co-60     2.86E+00  5.72E-02
Kr-85m    0.00E+00  3.59E+01
Kr-85     0.00E+00  1.42E+00
Kr-87     0.00E+00  7.18E+01
Kr-88     0.00E+00  9.93E+01
Rb-86     2.50E-02  5.01E-04
Rb-88     5.29E-01  1.06E-02
Sr-89     1.07E+01  1.07E-02
Sr-90     9.66E-01  9.66E-04
Sr-91     1.13E+01  1.13E-02
```

Figure 5-13 Sample BWR rcs_inventory.tmp file

5.6.3.3 Spent Fuel Pool (SFP) Inventory

Figure 5-14 presents a portion of the third output file for the spent-fuel inventory estimates. It contains the inventory for each batch of fuel in the reactor at the time of shutdown. The first line of the file indicates the file content, and the second line is the header provided by the user. The third line gives the date for which the inventories have been calculated. Generally this should be the date that the reactor was shutdown. The next three lines provide batch specific information. The first of these lines lists the number of full-power exposure days for the fuel in each batch. The second lists the metric tons of uranium in each batch, and the third line lists the burnup of each batch. The remainder of the file contains inventory information.

```
spent fuel batch inventory estimates
ANSI/ANS-18.1 Reference BWR
Exposure date,  year=   2011  day=   148
Days                487         974        1404
MTUs          4.677E+01   4.677E+01   4.677E+01
GWd/MTU       1.180E+01   2.360E+01   3.402E+01
Nuclide           Bq          Bq          Bq
H-3            2.72E+14    5.57E+14    8.18E+14
Co-58          1.71E+16    1.72E+16    1.72E+16
Co-60          4.39E+15    8.51E+15    1.19E+16
Kr-85m         2.73E+17    2.73E+17    2.73E+17
Kr-85          5.38E+15    1.09E+16    1.60E+16
Kr-87          5.46E+17    5.46E+17    5.46E+17
Kr-88          7.55E+17    7.55E+17    7.55E+17
Rb-86          2.73E+15    2.73E+15    2.73E+15
Rb-88          0.00E+00    0.00E+00    0.00E+00
```

Figure 5-14 Sample sfpool_inventory.tmp file

6.0 ACTIVITY BALANCE

The RASCAL 4.3 source term calculations track radionuclide activity in the fuel, as it passes through the plant, and as it enters the environment. The details of these calculations have been available in files used to track program execution and troubleshoot problems. However, these files have not been available to the RASCAL users in previous versions of the code. RASCAL 4.3 makes this information available to users by creating an activity balance file and displaying the contents through the user interface.

6.1 Activity Balance File Description

The activity balance file tracks activity for selected radionuclides and radionuclide groups from the reactor fuel (Rx Core) and reactor coolant system (RCS) through various pathways to the environment. The activity balance file lists, for each time step, the activity in five locations [e.g., Rx Core, RCS, steam generators (SG), Containment (Cont), and environment (Env)] and its movement through nine pathways. The locations and transfer pathways are listed in Table 6-1. In the activity balance file terminology, the containment refers to the entire release pathway between the reactor fuel and the free atmosphere.

Table 6-1 Locations and transfer pathways for balance activity summaries

Col	Name	Activity Balance File Locations and Transfer Pathways
1	Rx Core	Activity in the reactor fuel
2	Rx-RCS	Activity transferred from the reactor fuel to primary coolant
3	Rx-Cont	Activity transferred from the reactor fuel to containment
4	Rx-Env	Activity transferred from the reactor fuel to the environment
5	RCS	Activity in the primary coolant
6	RCS-Cont	Activity transferred from primary coolant to containment
7	RCS-SG	Activity transferred from primary to secondary coolant (steam generators)
8	RCS-Env	Activity transferred from primary coolant to the environment
9	SG	Activity in the secondary coolant (steam generators)
10	SG-Cont	Activity transferred from secondary coolant (steam generators) to containment
11	SG-Env	Activity transferred from secondary coolant (steam generators) to the environment
12	Cont	Activity in containment (Dry well for BWR)
13	Cont-Env	Activity transferred from containment to the environment
14	Env	Total activity released to the environment (cumulative from start of release)

Although depletion processes and decay are included in RASCAL 4.3, the activity removed from consideration by these processes is not tracked in the activity balance file. However, the activity listed in the activity balance file for the Rx Core, RCS, SG, and Cont does reflect the reduction in activity by depletion and decay. The activity released to the Env reflects the reduction in activity prior to release to the environment, but it does not reflect decay or ingrowth in the environment. Depletion of activity within the plant may be estimated from the initial activity in the fuel, the activity remaining in the fuel, the activity in the release path, and the activity released to the free atmosphere.

The activity balance file data are particularly useful in developing an understanding of how the RASCAL source term model works. For LTSBO and LOCA events, the RASCAL source term model may be represented as three boxes—reactor, containment, and atmosphere—as shown in Figure 6-1. The reactor box includes calculation of the initial reactor fuel inventory, decay of radionuclides in the fuel, and release of radionuclides from the fuel. The containment box may include a portion of the reactor vessel and various pipes and ducts in addition to the containment building. In RASCAL, the containment building is characterized by a single volume (free volume of the containment building for PWRs, and drywell plus wetwell volume for BWRs). When radionuclides are released from the fuel, they are assumed to become uniformly mixed within the containment instantaneously.

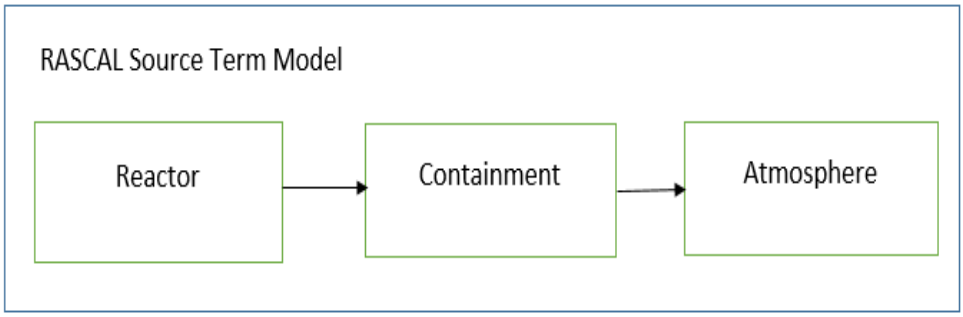


Figure 6-1 Simplified RASCAL LTSBO and LOCA source term model

RASCAL has three mechanisms for depleting activity between the reactor fuel and the free atmosphere. The radionuclide activity is modified by decay and ingrowth between the time of reactor shutdown and ultimate release to the atmosphere. While radionuclides are in containment, they are assumed to gradually deposit on surfaces as a function of time. The radionuclides may also be washed out by sprays, if sprays are activated. Finally, radionuclides may be removed from the containment atmosphere by the action of filters as the containment atmosphere is released to the atmosphere outside (environment) of containment. In RASCAL BWR wetwell suppression pools are modeled as filters. These reduction mechanisms are functions of flow out of the containment. The details of the depletion mechanisms are found in sections 1.3, 1.4, and 1.6 in NUREG-1940.

6.2 Generation of the Activity Balance File by RASCAL

RASCAL 4.3.1 creates an activity balance file for source terms types based on reactor conditions. These source terms types include LTSBO, LOCA, Coolant Release, Containment Radiation Monitor, Coolant Sample and Containment Air Sample. A file is created for source terms based on monitored release pathways, but the file is empty. Table 6-2 lists the activity locations and transfers tracked in the activity balance file for each source type and release path.

Table 6-2 Activity balance locations and transfer pathways for reactor accident categories

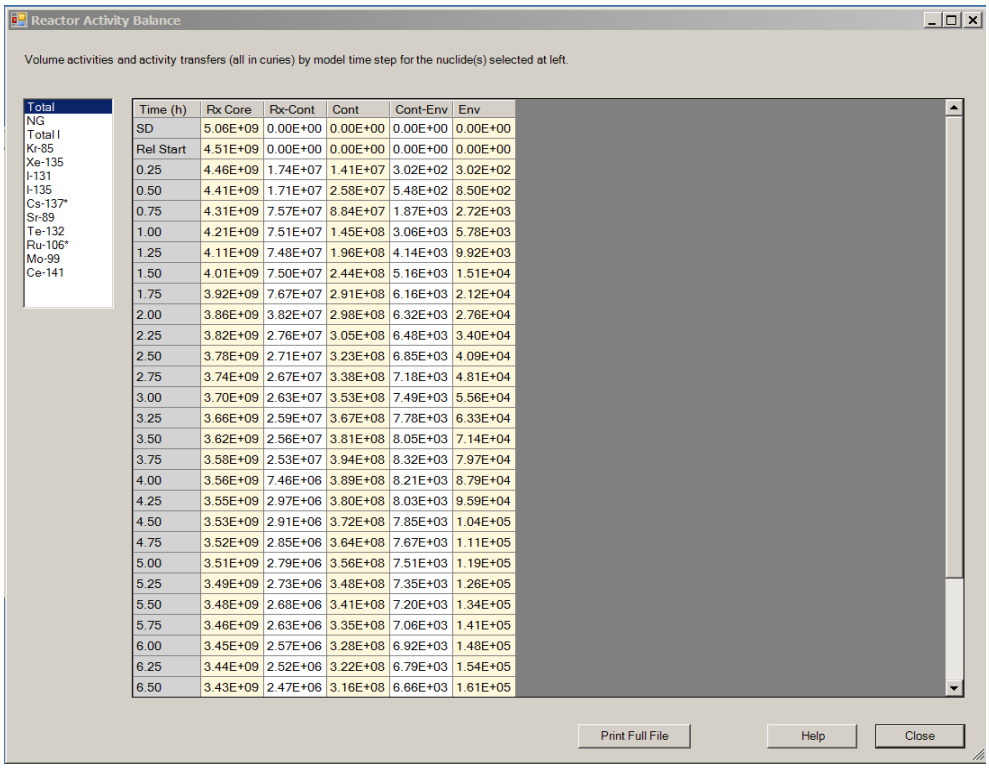
Source Term Type	Release Pathways	Activity Locations and Transfers
PWR Events		
LTSBO	Cont Leak	Rx Core, Rx-Cont, RCS, RCS-Cont, Cont, Cont-Env, Env
	SGTR	Rx Core, Rx-RCS, RCS, RCS-SG, SG, SG-Env, Env
	Bypass	Rx Core, Rx-RCS, RCS, RCS-Env, Env
LBLOCA	Cont Leak	Rx Core, Rx-Cont, RCS, RCS-Cont, Cont, Cont-Env, Env
	SGTR	Rx Core, Rx-RCS, RCS, RCS-SG, SG, SG-Env, Env
	Bypass	Rx Core, Rx-RCS,RCS, RCS-Env, Env
Coolant Release	Cont Leak	NA
	SGTR	RCS, RCS-SG, SG, SG-Env, Env
	Bypass	RCS, RCS-Env, Env
BWR Events		
LTSBO	Wetwell	Rx Core, Rx-Cont, RCS, RCS-Cont, Cont, Cont-Env, Env
	Drywell	Rx Core, Rx-Cont, Cont, Cont-Env, Env
	Bypass	Rx Core, Rx-RCS, RCS, RCS-Env, Env
LBLOCA	Wetwell	Rx Core, Rx-Cont, RCS, RCS-Cont, Cont, Cont-Env, Env
	Drywell	Rx Core, Rx-Cont, RCS, RCS-Cont, Cont, Cont-Env, Env
	Bypass	Rx Core, Rx-RCS, RCS, RCS-Env, Env
Coolant Release	Wetwell	NA
	Drywell	NA
	Bypass	RCS, RCS-Env, Env

The activity balance file includes separate tables for 12 activity groups and radionuclides. In each table, the activity for each location and transfer is presented as a function of time. The initial time is the time of reactor shutdown. The second time is the time that the release starts (t = 0), and the remaining times are at 900 second (s) intervals following the beginning of release. These features are shown in Table 6-3 and represent data from a portion of the activity balance file for a PWR LOCA as it is displayed in the user interface. The activity shown is "total activity" in curies (Ci). The first column is the time of the activity balance. The shutdown and the release start time (Rel Start) rows give the activity balance at the time of reactor shutdown and the beginning of release from the Rx Core. At these two times all of the activity is in the Rx Core. If the event involved release of activity from the RCS, the activity in the RCS would also be shown. The activities shown in the Rx Core, Cont, and Env columns are the activities at the end of the period. They account for decay and depletion. The activity in the Env column is the total activity released to the atmosphere. The activity balance file does not reflect the decay that takes place in the atmosphere. Decay in the atmosphere and on the ground is calculated in the RASCAL transport and diffusion models. The activities shown in the Rx-Cont and Cont-Env columns are the activities transferred during the time period.

The activity balance information may be accessed from the user interface using the View Balance button on the Source Term Tab. Figure 6-2 shows how the activity balance file information is presented through the RASCAL 4.3.1 user interface.

Table 6-3 Sample of the Balance Summary Available in the User Interface

Time Step Period (h)	Activity (Ci)				
	Rx Core	Rx-Cont	Cont	Cont-Env	Env
Shutdown	7.07E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Release Start Time (Rel Start)	7.07E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.25	6.97E+09	3.63E+07	2.86E+07	3.21E+02	3.21E+02
0.50	6.87E+09	3.42E+07	4.97E+07	5.51E+02	8.72E+02
0.75	6.66E+09	1.42E+08	1.61E+08	1.75E+03	2.63E+03
1.00	6.46E+09	1.34E+08	2.55E+08	2.74E+03	5.37E+03
1.25	6.27E+09	1.30E+08	3.37E+08	3.60E+03	8.97E+03



Time (h)	Rx Core	Rx-Cont	Cont	Cont-Env	Env
SD	5.06E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rel Start	4.51E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.25	4.46E+09	1.74E+07	1.41E+07	3.02E+02	3.02E+02
0.50	4.41E+09	1.71E+07	2.58E+07	5.48E+02	8.50E+02
0.75	4.31E+09	7.57E+07	8.84E+07	1.87E+03	2.72E+03
1.00	4.21E+09	7.51E+07	1.45E+08	3.06E+03	5.78E+03
1.25	4.11E+09	7.48E+07	1.96E+08	4.14E+03	9.92E+03
1.50	4.01E+09	7.50E+07	2.44E+08	5.16E+03	1.51E+04
1.75	3.92E+09	7.67E+07	2.91E+08	6.16E+03	2.12E+04
2.00	3.86E+09	3.82E+07	2.98E+08	6.32E+03	2.76E+04
2.25	3.82E+09	2.76E+07	3.05E+08	6.48E+03	3.40E+04
2.50	3.78E+09	2.71E+07	3.23E+08	6.85E+03	4.09E+04
2.75	3.74E+09	2.67E+07	3.38E+08	7.18E+03	4.81E+04
3.00	3.70E+09	2.63E+07	3.53E+08	7.49E+03	5.56E+04
3.25	3.66E+09	2.59E+07	3.67E+08	7.78E+03	6.33E+04
3.50	3.62E+09	2.56E+07	3.81E+08	8.05E+03	7.14E+04
3.75	3.58E+09	2.53E+07	3.94E+08	8.32E+03	7.97E+04
4.00	3.56E+09	7.46E+06	3.89E+08	8.21E+03	8.79E+04
4.25	3.55E+09	2.97E+06	3.80E+08	8.03E+03	9.59E+04
4.50	3.53E+09	2.91E+06	3.72E+08	7.85E+03	1.04E+05
4.75	3.52E+09	2.85E+06	3.64E+08	7.67E+03	1.11E+05
5.00	3.51E+09	2.79E+06	3.56E+08	7.51E+03	1.19E+05
5.25	3.49E+09	2.73E+06	3.48E+08	7.35E+03	1.26E+05
5.50	3.48E+09	2.68E+06	3.41E+08	7.20E+03	1.34E+05
5.75	3.46E+09	2.63E+06	3.35E+08	7.06E+03	1.41E+05
6.00	3.45E+09	2.57E+06	3.28E+08	6.92E+03	1.48E+05
6.25	3.44E+09	2.52E+06	3.22E+08	6.79E+03	1.54E+05
6.50	3.43E+09	2.47E+06	3.16E+08	6.66E+03	1.61E+05

Figure 6-2 User interface presentation of the RASCAL 4.3.1 activity balance information

6.3 Activity Balance File Example for BWR LTSBO

Figures 6-3 through 6-7 illustrate the movement of five radionuclides from a RASCAL 4.3.1 activity balance file for an unmitigated BWR LTSBO scenario described in NUREG/CR-7110 Volume 1, Revision 1, “State-of-the-Art Reactor Consequence Analysis Project, Volume 1: Peach Bottom Integrated Analysis”. The radionuclides presented in the figures are in order of increasing complexity of the source term model with respect to depletion mechanisms.

Figure 6-3 depicts the plot of the RASCAL 4.3.1 krypton-85 (Kr-85) activity balance file for the unmitigated BWR LTSBO scenario. Kr-85 is inert noble gas that does not deplete by natural deposition, sprays, or the action of filters. Additionally, given the half-life of Kr-85 and the time frame of the scenario, Kr-85 decay can be neglected. Figure 6-3 shows the Kr-85 in the fuel (blue line) from the time of the initiating event until fuel damage begins at about 9 hours. At this point, almost all of the Kr-85 activity moves from the fuel to containment (orange line) over a period of about 2 hours. During this period of time and the next few hours the Kr-85 activity slowly leaks to the atmosphere (green line). Finally, at approximately 20 hours the containment fails and most of the Kr-85 activity remaining in the containment is released to the free atmosphere.

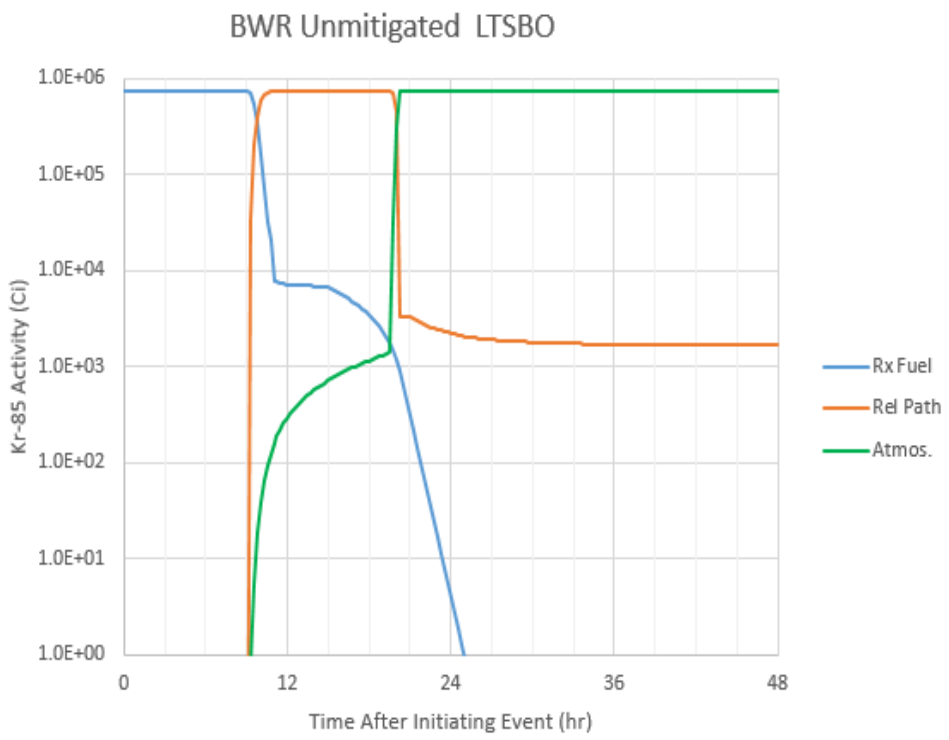


Figure 6-3 RASCAL 4.3.1 Kr-85 activity balance during a BWR Mark III LTSBO event

Figure 6-4 depicts the plot of the RASCAL 4.3.1 xenon-135 (Xe-135) activity balance file for the unmitigated BWR LTSBO scenario. Like Kr-85, Xe-135 is also a noble gas that is not depleted by natural deposition, sprays, or the action of filters. However unlike Kr-85, Xe-135 experiences significant ingrowth from iodine-135 (I-135) and decay during the BWR LTSBO event. The ingrowth and decay of Xe-135 can be seen by comparing the activity balance file plot for Xe-135 shown in Figure 6-4 with the activity balance file plot for Kr-85 shown in Figure 6-3. The ingrowth of Xe-135 is clearly seen in Figure 6-4 in the increase in Xe-135 activity in the fuel (blue line) for the period between the reactor shutdown and the beginning fuel damage (9 hours). Additionally, the continued ingrowth of the Xe-135 in the fuel is clearly indicated by its persistence at greater than 24 hours after the initiating event. The decay of Xe-135 is indicated in the release path depletion (gray line) between 9 and 20 hours. This depletion reduces the fraction of Xe-135 release to the free atmosphere to about 50 percent of the Xe-135 in the fuel at the beginning of fuel damage.

Figure 6-5 depicts the plot of the RASCAL 4.3.1 cesium-137 (Cs-137) activity balance file for the unmitigated BWR LTSBO scenario. The decay of Cs-137 can be ignored because it has a long half-life. Additionally, the BWR LTSBO scenario does not include the use of containment sprays or filtration by the SBGTS. Consequently, the only depletion mechanism applied to activity prior to release to the free atmosphere are the natural deposition of Cs-137 particles in the release path and scrubbing as the Cs-137 particles pass through the wetwell suppression pool.

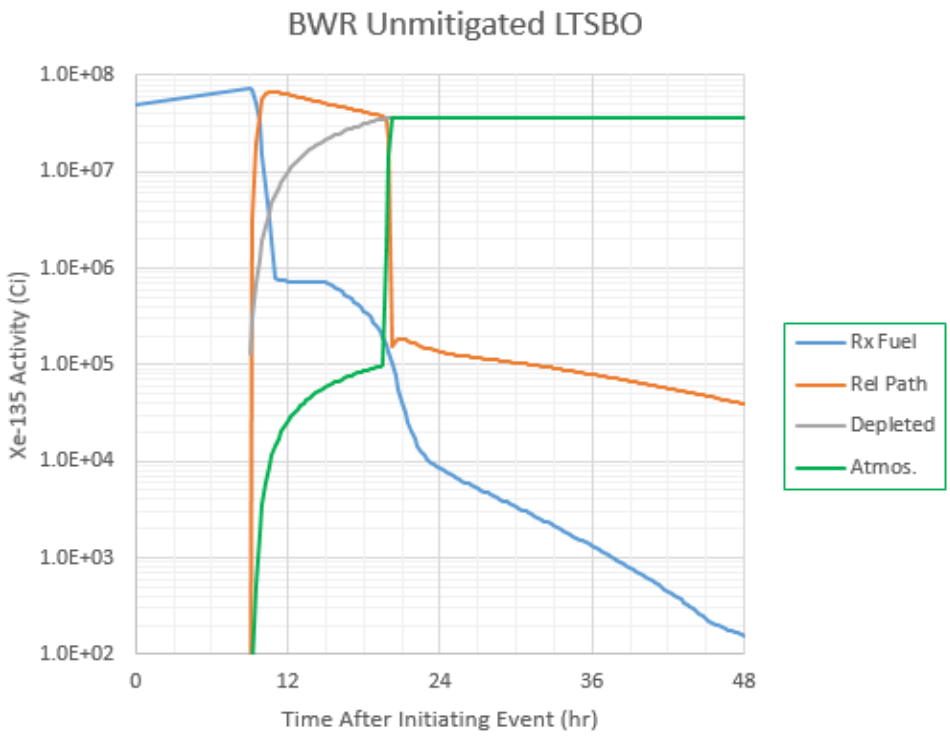


Figure 6-4 RASCAL 4.3.1 Xe-135 activity balance during a BWR Mark III LTSBO event

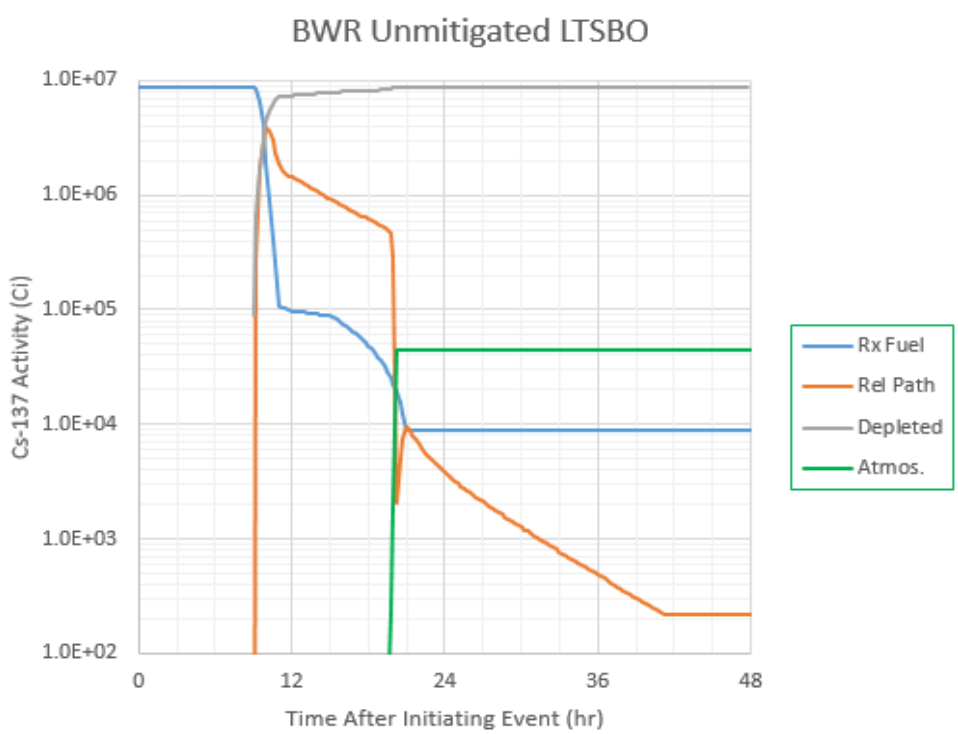


Figure 6-5 RASCAL 4.3.1 Cs-137 activity balance during a BWR Mark III LTSBO event

There is a significant difference between the Cs-137 activity balance file plot (Figure 6-5) and the activity balance file plots for Kr-85 and Xe-135 (Figure 6-3 and 6-4). Almost all of the activity released from the fuel is depleted in the release path (orange line). As a result, the activity released to the atmosphere and available for later release is a small value compared to the activity initially in the fuel (blue line) and the depleted activity (gray line). In fact, the activity available for release to the free atmosphere is the small difference between two large numbers. This fact has significant consequences related to the accuracy and precision of the estimates of activity released to the free atmosphere.

Assuming that the initial Cs-137 activity in the fuel is well-known compared to the activity depleted in the release path, a small change in the calculated value of depleted activity will result in a relatively large change in activity released to the free atmosphere. For example, changing the fraction of activity depleted from 99.9 percent of the initial activity to 99.0 percent would increase the activity released to the free atmosphere by an order of magnitude.

The practical implication of the sensitivity of activity release to the atmosphere to changes in depletion is that large differences can be expected in Cs-137 source term estimates of various models as a result of small differences in representations of depletion processes within the models. It seems unlikely that the models used to represent the depletion processes in RASCAL and other consequence assessment models warrant precision greater than 2 significant figures.

Figures 6-6 and 6-7 depict the plots of the RASCAL 4.3.1 activity balances for I-131 and I-135, respectively. The radionuclides decay, as well as deposit, in the release path. Both radionuclides show the dominance of depletion in the activity balance.

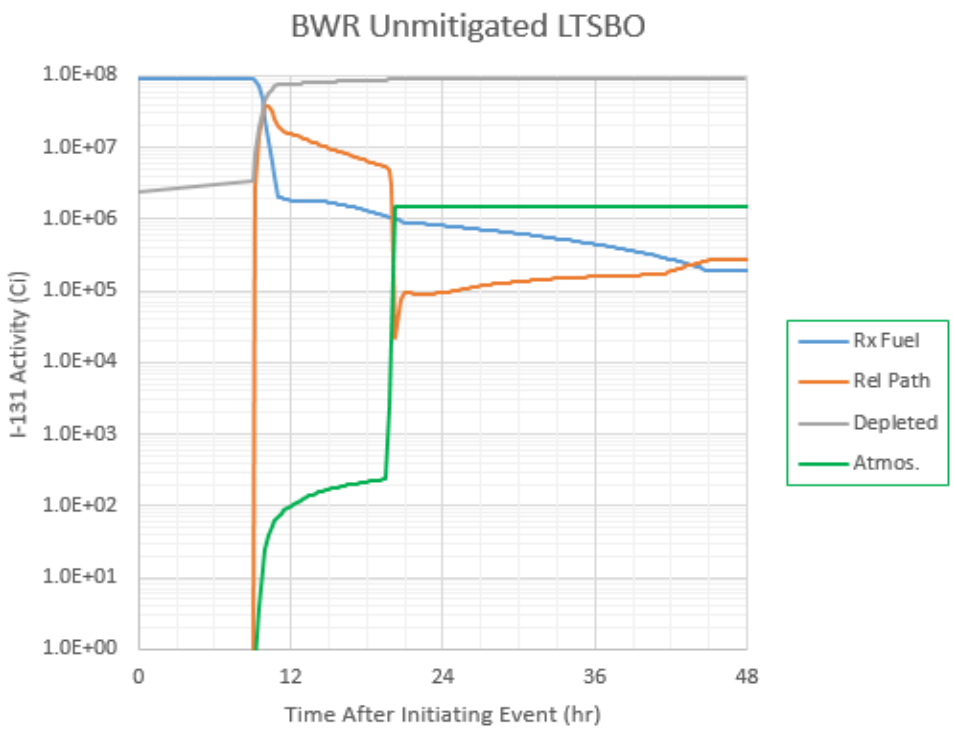


Figure 6-6 RASCAL 4.3.1 I-131 activity balance during a BWR Mark III LTSBO event

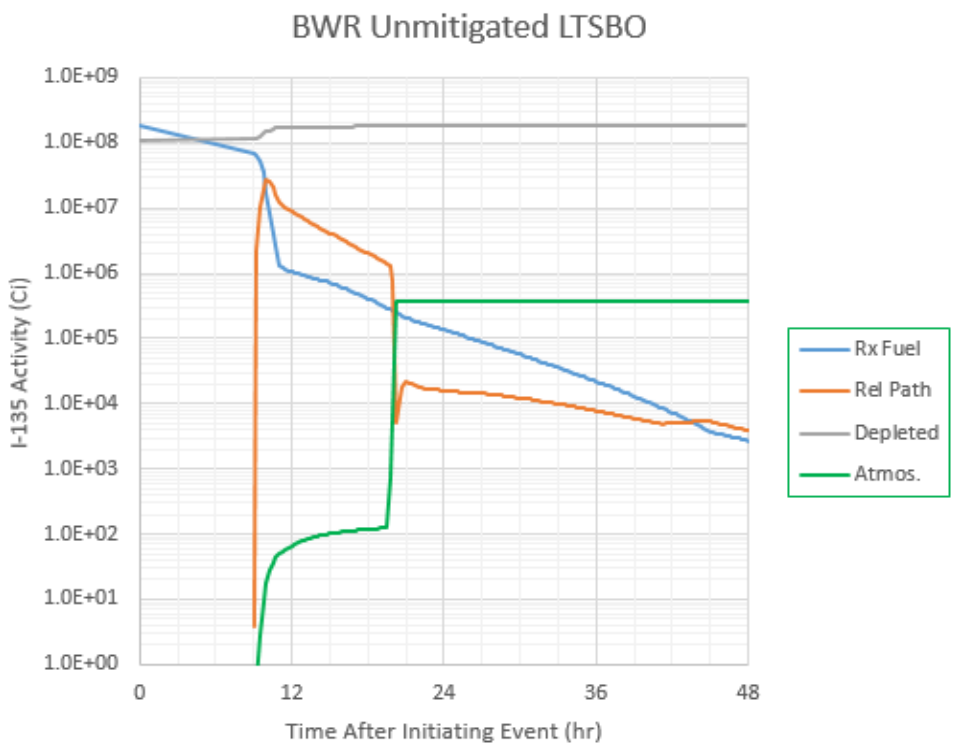


Figure 6-7 RASCAL 4.3.1 I-135 activity balance during a BWR Mark III LTSBO event

7.0 IMPORTANCE

During the emergency response to the Fukushima Daiichi accident, there were frequent requests to identify a limited number of radionuclides that were the most important contributors to doses. RASCAL 4.3 includes a feature to process the total radionuclide activity released to the environment in the course of an event and evaluate the relative importance of the radionuclides to five dose measures and rank the radionuclides in order of importance. The order of importance is a function of the time step as well as the dose calculated. The importance utility calculates importance at the end of the release and at seven additional time steps ranging from 1 day (d) to 10 years (yr). The results of the importance calculation are available through the user interface.

7.1 The Importance Model

The importance model assumes that the importance of radionuclides can be determined from pseudo dose calculations that are based on the dose calculation methods described in NUREG-1940. The pseudo doses calculated in RASCAL 4.3.1 are for the submersion (cloud shine), inhalation, ground shine, and ingestion pathways, and for total effective dose equivalent (TEDE).

The pseudo dose for a radionuclide i for cloud submersion during plume passage is defined as:

$$pD_i(t) = Q_i(t) \times DC_i \tag{7-1}$$

where

pD_i(t) = the pseudo dose from radionuclide i at time t = 0,
Q_i(t) = total activity release of radionuclide i decayed to the release time,
DC_i = submersion dose coefficient for radionuclide i.

After plume passage, the cloud submersion pseudo dose is calculated as:

$$pD_i(t) = Q_i(t) \times DC_i \times dv_i \times rs_i(t) \tag{7-2}$$

where

dv_i = deposition velocity for radionuclide i, (0.0 m/s for noble gases, 0.003 m/s for iodines, and 0.001 m/s for other radionuclides),
rs_i(t) = resuspension factor for time t after the end of release (NUREG-1940 Section 7).

Similar equations define the pseudo doses for the inhalation, ground shine, and ingestion pathways. The pseudo doses for the inhalation pathway during plume passage are calculated by:

$$pD_i(t) = Q_i(t) \times vb \times DC_i \tag{7-3}$$

where

vb = breathing rate (m/s),
DC_i = inhalation dose coefficient for radionuclide i.

After plume passage the inhalation pseudo dose is calculated as:

$$pD_i(t) = Q_i(t) \times dv_i \times rs_i(t) \times vb \times DC_i \tag{7-4}$$

The pseudo doses for the ground shine pathway are calculated by:

$$pD_i(t) = Q_i(t) \times DC_i \times dv_i \times wx(t) \tag{7-5}$$

where

DC_i = ground shine dose coefficient for radionuclide i,
wx(t) = weathering factor for ground.

RASCAL 4.3 calculates a pseudo TEDE dose potential. It is the sum of the pseudo doses for the submersion, inhalation, and ground shine pathways.

Finally, the ingestion pathway pseudo dose is calculated as:

$$pD_i(t) = Q_i(t) \times DC_i \times dv_i \times wxlv(t) \tag{7-6}$$

where

DC_i = ingestion dose coefficient for radionuclide i,
wxlv(t) = weathering factor for leafy vegetables.

The importance of radionuclide i to each of the dose pathways at time t is calculated as:

$$I_i(t) = \frac{pD_i(t)}{\sum_i pD_i(t)} \tag{7-7}$$

where

$I_i(t)$ = importance of radionuclide i and the summation over all radionuclides releases.

The sum of pseudo doses of all radionuclides is referred to as the dose potential for the pathway.

Note that RASCAL 4.3 uses the dose coefficient for the submersion, inhalation, and ground shine pathways that were selected in the user interface, either FGR 11 and FGR 12 [29] dose coefficients or FGR 13 dose coefficients. The FGR 13 dose coefficients are for an adult.

The importance of each radionuclide in the source term includes the contributions to the pseudo dose from all daughters. When a parent radionuclide and a daughter radionuclide are both explicitly included in the source term, they are treated separately. The importance of the parent will include contributions from daughter activity that grows in after release. It will not include contributions from the explicit daughter activity that is included in the source term. By the same token, the importance of the explicit daughter will only be based on the activity i in the source term and the activity of any additional daughters that grow in after release. It will not include any contribution of the daughter that grows in from the explicit parent.

The ingestion pathway is generally not treated in RASCAL 4.3 because it is of secondary importance in emergency response. Ingestion pathway doses can be minimized by monitoring and interdiction. However, there may be some delay between plume passage and the start of effective monitoring and interdiction. The ingestion pathway importance calculation in RASCAL 4.3 is meant to provide a quick, if gross, indication of those radionuclides that might contribute most to ingestion doses.

The ingestion importance calculation assumes that the likely early ingestion pathway is via consumption of leafy vegetables. Activity is deposited on the vegetables during plume passage and is removed from the leaves by various environmental processes until the vegetables are harvested. These processes are represented by weathering function in Equation 7-8. The weathering function is defined as:

$$wxlv(t) = exp^{-\ln(2)\times\frac{t}{14}} \tag{7-8}$$

where

14 = effective half-life of the weathering process, which is assumed to end after 183 d (6 months).

There are many factors that are important in calculating ingestion pathway doses that are not included in RASCAL 4.3. Among these factors are the variation of interception fractions of various types of leafy vegetables and the removal of activity from leaf surfaces by handling, washing, and food preparation. Age is another important factor that is not considered in RASCAL importance calculations. The dose coefficient and the consumption are both functions of age. Neither the variation of the dose coefficient with age nor the variation of consumption with age is treated in the calculation of the pseudo doses used to determine relative importance. As a result, caution should be used in the interpretation and extrapolation of the ingestion importance numbers.

7.2 Importance Results

The importance calculation results are presented in two parts. The first part establishes a context for interpretation of the radionuclide specific importance values. This context is important because all importance values are normalized to give a total importance of 1.00. The contextual information in the first part of the results shows the change in TEDE dose potential with time and the change with time of the relative contributions of the submersion, inhalation, and ground shine pathways to TEDE. Figure 7-1 shows an example of the contextual portion of the importance results file.

```
Relative Importance of TEDE
Days           0.         1.         7.        30.       183.       365.      1825.      3650.
rTEDE     1.0E+00    5.0E-07    2.9E-07    1.1E-07    6.1E-08    4.9E-08    1.5E-08    7.5E-09

Relative Importance of Pathways to TEDE
Days           0.         1.         7.        30.       183.       365.      1825.      3650.
cs        3.4E-02    4.2E-04    2.8E-04    5.7E-05    3.0E-07    2.4E-07    9.2E-08    8.8E-08
inh       9.7E-01    3.9E-02    3.9E-02    1.4E-02    8.9E-05    7.4E-05    5.1E-05    7.5E-05
gs        8.5E-07    9.6E-01    9.6E-01    9.9E-01    1.0E+00    1.0E+00    1.0E+00    1.0E+00
```

Figure 7-1 Example of the importance file contextual portion

The rTEDE row clearly shows that the dose potential during plume passage (t = 0) is orders of magnitude larger than the potential following plume passage. Similarly, the cloud shine (cs), inhalation (inh), and ground shine (gs) rows show that inhalation is the dominant pathway during plume passage and that ground shine is the dominant pathway following plume passage.

The second portion of the importance results file shows the importance of specific radionuclides. Figure 7-2 shows a portion of the activity balance file that presents the importance of specific radionuclides for a PWR LOCA event.

```
Cumulative Importance to dose at t =       1. days
        Cloudshine        Inhalation        Groundshine       Ingestion         TEDE
  1  I-132       0.21  Sr-90       0.23  I-132       0.21  I-131       0.30  I-132       0.20    1
  2  La-140      0.41  Pu-241      0.36  La-140      0.39  Cs-134      0.51  La-140      0.38    2
  3  Cs-134      0.54  I-131       0.47  Cs-134      0.52  Te-132      0.61  Cs-134      0.50    3
  4  I-133       0.61  Sr-89       0.56  I-133       0.59  Cs-137*     0.71  I-133       0.58    4
  5  Te-132      0.68  Cm-242      0.65  Te-132      0.67  Ba-140      0.78  Te-132      0.65    5

Cumulative Importance to dose at t =       7. days
        Cloudshine        Inhalation        Groundshine       Ingestion         TEDE
  1  La-140      0.44  Sr-90       0.27  La-140      0.41  Cs-134      0.29  La-140      0.40    1
  2  Cs-134      0.65  Pu-241      0.41  Cs-134      0.62  I-131       0.54  Cs-134      0.60    2
  3  Cs-136      0.73  Sr-89       0.52  Cs-136      0.70  Cs-137*     0.68  Cs-136      0.68    3
  4  I-131       0.79  Cm-242      0.61  I-131       0.76  Ba-140      0.75  I-131       0.74    4
  5  Cs-137*     0.84  Cs-134      0.70  Cs-137*     0.82  Sr-90       0.81  Cs-137*     0.79    5

Cumulative Importance to dose at t =      30. days
        Cloudshine        Inhalation        Groundshine       Ingestion         TEDE
  1  Cs-134      0.56  Sr-90       0.32  Cs-134      0.54  Cs-134      0.47  Cs-134      0.53    1
  2  Cs-137*     0.70  Pu-241      0.50  Cs-137*     0.68  Cs-137*     0.69  Cs-137*     0.68    2
  3  La-140      0.81  Cs-134      0.60  La-140      0.78  Sr-90       0.79  La-140      0.77    3
  4  Cs-136      0.87  Cm-242      0.71  Cs-136      0.84  I-131       0.85  Cs-136      0.83    4
  5  Ba-140      0.90  Sr-89       0.79  Ba-140      0.87  Sr-89       0.90  Ba-140      0.86    5

Cumulative Importance to dose at t =     183. days
        Cloudshine        Inhalation        Groundshine       Ingestion         TEDE
  1  Cs-134      0.76  Sr-90       0.41  Cs-134      0.75  Cs-134      0.53  Cs-134      0.75    1
  2  Cs-137*     0.98  Pu-241      0.62  Cs-137*     0.97  Cs-137*     0.83  Cs-137*     0.97    2
  3  Zr-95       0.99  Cs-134      0.74  Y-90        0.98  Sr-90       0.96  Y-90        0.98    3
  4  Nb-95       0.99  Cm-242      0.81  Zr-95       0.98  Ce-144*     0.97  Zr-95       0.98    4
  5  Ru-106*     1.00  Ce-144*     0.88  Sr-89       0.99  Y-90        0.98  Sr-89       0.99    5
```

Figure 7-2 Example of the importance file activity balance portion

The example only shows the five most important radionuclides for four time periods. The file lists the importance of all radionuclides (typically 70 to 80) for eight time periods. The time periods not shown in the figure are 0 d (end of model run), 3.65E+02 d (1 yr), 1.825E+03 d (5 yr), and 3.65E+03 d (10 yr). For each time period and pathway, the radionuclides are listed in order of decreasing importance. The number following the radionuclide is the fraction of dose potential contributed by that radionuclide and the radionuclides that are above it in the list.

The importance lists in Figure 7-2 illustrate two features that are common to most importance lists. The first is that the most important radionuclides are different for different dose pathways and the most important radionuclides change with increasing time after the release. The second is that the number of radionuclides that contribute significantly to dose potential decreases with increasing time. These points are clearly illustrated in the ground shine importance lists by comparing the lists for 1 and 183 d. The two most important radionuclides for ground shine at day 1 contribute about 39 percent of the dose potential. Neither radionuclide is in the top 5 list for 183 d. Further, the top 2 radionuclides at day 183 contribute about 97 percent of the dose potential.

Table 7-1 summarizes the variation in pathway importance for representative BWRs and PWRs as a function of time after the end of release assuming a mid-cycle core melt event with containment failure. The 0 day entries include the dose potential from plume passage. Inhalation is the most important pathway during plume passage followed by cloud shine. During plume passage, ground shine is typically insignificant. After plume passage, ground shine is the dominant dose pathway, inhalation is a secondary pathway, and cloud shine is insignificant. The numbers in parentheses are the pathway fraction of the total dose potential.

Table 7-1 Dose pathway importance following a mid-cycle core melt event with containment failure

		Most Important →		Least Important
BWR Mark III LTSBO	0 days	Inhalation (0.97)	Cloud shine (0.029)	Ground shine (< 0.001)
	1 day	Ground shine (0.94)	Inhalation (0.055)	Cloud shine (< 0.001)
	7 days	Ground shine (0.94)	Inhalation (0.057)	Cloud shine (< 0.001)
	30 days	Ground shine (0.99)	Inhalation (0.01)	Cloud shine (< 0.001)
	6 months	Ground shine (1.00)	Inhalation (<0.001)	Cloud shine (< 0.001)
BWR Mark III LOCA	0 days	Inhalation (0.96)	Cloud shine (0.036)	Ground shine (< 0.001)
	1 day	Ground shine (0.94)	Inhalation (0.062)	Cloud shine (< 0.001)
	7 days	Ground shine (0.94)	Inhalation (0.058)	Cloud shine (< 0.001)
	30 days	Ground shine (0.98)	Inhalation (0.016)	Cloud shine (< 0.001)
	6 months	Ground shine (1.00)	Inhalation (<0.001)	Cloud shine (< 0.001)
PWR LTSBO	0 days	Inhalation ((0.98)	Cloud shine (0.021)	Ground shine (< 0.001)
	1 day	Ground shine (0.92)	Inhalation (0.081)	Cloud shine (< 0.001)
	7 days	Ground shine (0.91)	Inhalation (0.092)	Cloud shine (< 0.001)
	30 days	Ground shine (0.98)	Inhalation (0.022)	Cloud shine (< 0.001)
	6 months	Ground shine (1.00)	Inhalation (<0.001)	Cloud shine (< 0.001)
PWR LOCA	0 days	Inhalation (0.97)	Cloud shine (0.027)	Ground shine (< 0.001)
	1 day	Ground shine (0.94)	Inhalation (0.064)	Cloud shine (< 0.001)
	7 days	Ground shine (0.94)	Inhalation (0.059)	Cloud shine (< 0.001)
	30 days	Ground shine (0.99)	Inhalation (0.014)	Cloud shine (< 0.001)
	6 months	Ground shine (1.00)	Inhalation (<0.001)	Cloud shine (< 0.001)

Table 7-2 lists the 5 radionuclides with the greatest dose potential for the three important dose pathways following a mid-cycle core-melt accident with loss of containment at a typical BWR reactor with the Mark III containment. Table 7-3 provides the same information for a typical PWR reactor. The numbers in parentheses following the pathway are the relative pathway importance. For each pathway, the radionuclides are listed in order of decreasing importance. The number in parentheses following the radionuclide is the cumulative importance of the radionuclide plus the importance of the more important radionuclides in the pathway (i.e. the radionuclides listed above).

Table 7-2 Radionuclides most important to dose for a BWR Mark III following core melt with containment failure

Event	Time	Pathway		
BWR Mark III LTSBO	0 days	Cloud shine (0.029)	Inhalation (0.97)	Ground shine (<0.001)
		Xe-133 (0.55) I-132 (0.68) Xe-135 (0.81) I-131 (0.87) I-133 (0.92)	I-131 (0.70) I-133 (0.77) Cs.137* (0.83) Cs-134 (0.87) Te-132 (0.91)	I-132 (0.30) I-131 (0.45) I-133 (0.57) Te-132 (0.66) Cs-134 (0.74)
BWR Mark III LOCA	0 days	Cloud shine (0.036)	Inhalation (0.97)	Ground shine (<0.001)
		Xe-133 (0.56) Xe-135 (0.73) I-132 (0.84) La-140 (0.88) I-131 (0.91)	I-131 (0.31) Cs-137* (0.42) Pu-241 (0.52) Cs-134 (0.60) Sr-90 (0.68)	I-132 (0.24) La-140 (0.48) Cs-134 (0.60) Te-132 (0.67) Cs-136 (0.72)
BWR Mark III LTSBO	1 day	Cloud shine (<0.001)	Inhalation (0.056)	Ground shine (0.94)
		I-132 (0.26) I-131 (0.44) Cs-134 (0.55) Te-132 (0..64) La-140 (0.72)	I-131 (0.48) Cs-137* (0.62) Cs-134 (0.72) Ru-106* (0.79) Te-132 (0.85)	I-132 (0.26) I-131 (0.44) Cs-134 (0.55) Te-132 (0.64) La-140 (0.71)
BWR Mark III LOCA	1 day	Cloud shine (<0.001)	Inhalation (0.062)	Ground shine (0.94)
		La-140 (0.36) I-132 (0.53) Cs-134 (0.67) Cs-136 (0.73) Te-132 (0.78)	Cs-137* (0.15) Pu-241 (0.28) I-131 (0.41) Cs-134 (0.52) Sr-90 (0.62)	LA-140 (0.33) I-132 (0.51) Cs-134 (0.64) Te-132 (0.70) Cs-136 (0.76)

Table 7-3 Radionuclides most important to dose for a PWR following core melt with containment failure

Event	Time	Pathway		
PWR LTSBO	0 days	Cloud shine (0.021)	Inhalation (0.98)	Ground shine (<0.001)
		Xe-133 (0.61) I-132 (0.81) I-131 (0.85) La-140 (0.88) Cs-134 (0.91)	I-131 (0.33) Pu-241 (0.59) Cs-137* (0.68) Cs-134 (0.75) Ce-144* (0.81)	I-132 (0.32) La-140 (0.47) Cs-134 (0.59) Te-132 (0.68) I-131 (0.75)
PWR LOCA	0 days	Cloud shine (0.0271)	Inhalation (0.97)	Ground shine (<0.001)
		Xe-133 (0.71) I-132 (0.79) La-140 (0.86) I-131 (0.90) Cs-134 (0.93)	I-131 (0.35 Cs-137* (0.49 Cs-134 (0.59) Pu-241 (0.67) Cm-242 (0.74)	La-140 (0.34) Cs-134 (0.50) I-132 (0.66) Cs-136 (0.73) I-131 (0.80)
PWR LTSBO	1 day	Cloud shine (<0.001)	Inhalation (0.081)	Ground shine (0.92)
		I-132 (0.25) La-140 (0.46) Cs-134 (0.61) Te-132 (0.69) I-131 (0.76)	Pu-241 (0.35) I-131 (0.48) Cs-137* (0.60) Cs-134 (0.69) Ce-144* (0.78)	I-132 (0.25) La-140 (0.44) Cs-134 (0.58) Te-132 (0.67) I-131 (0.75)
PWR LOCA	1 day	Cloud shine (<0.001)	Inhalation (0.064)	Ground shine (0.94)
		La-140 (0.41) Cs-134 (0.58) I-132 (0.70) Cs-136 (0.77) I-131 (0.83)	Cs-137* (0.18) I-131 (0.33) Cs-134 (0.46) Pu-241 (0.58) Cm-242 (0.68)	La-140 (0.38) Cs-134 (0.56 I-132 (0.67) Cs-136 (0.75) I-131 (0.81)

Importance is particularly useful as a tool to allow users to focus on the radionuclides that are most significant to dose. As seen in Tables 7-2 and 7-3, a small number of radionuclides contribute almost all of the dose potential for each pathway. The remaining radionuclides contribute very little to dose potential. However, they may be a distraction. To illustrate this point, consider the unmitigated BWR LTSBO scenario described in NUREG/CR-7110 Volume 1, Revision 1. Figure 7-3 compares the source term calculated by RASCAL 4.3.1 for the scenario with the MELCOR source term for the 61 radionuclides that are common to the two source terms. (Note: The MELCOR source term activities have been adjusted for decay from the initiating event to the time of release.) Figure 7-4 shows the same comparison for the

17 radionuclides that contribute the most to the cloud shine dose pathway. The 44 radionuclides not shown in Figure 7-4, but which appear in Figure 7-3, combined contribute less than 0.5 percent to the total cloud shine dose pathway dose potential. The 0 days in the Figure title indicates that the activities are at the end of the model run. Figure 7-5 compares activities for the 19 radionuclides most important to the inhalation pathway dose potential. The 42 radionuclides not shown in Figure 7-5 but which appear in Figure 7-3, combined contribute less than 0.5 percent to the total inhalation pathway dose potential.

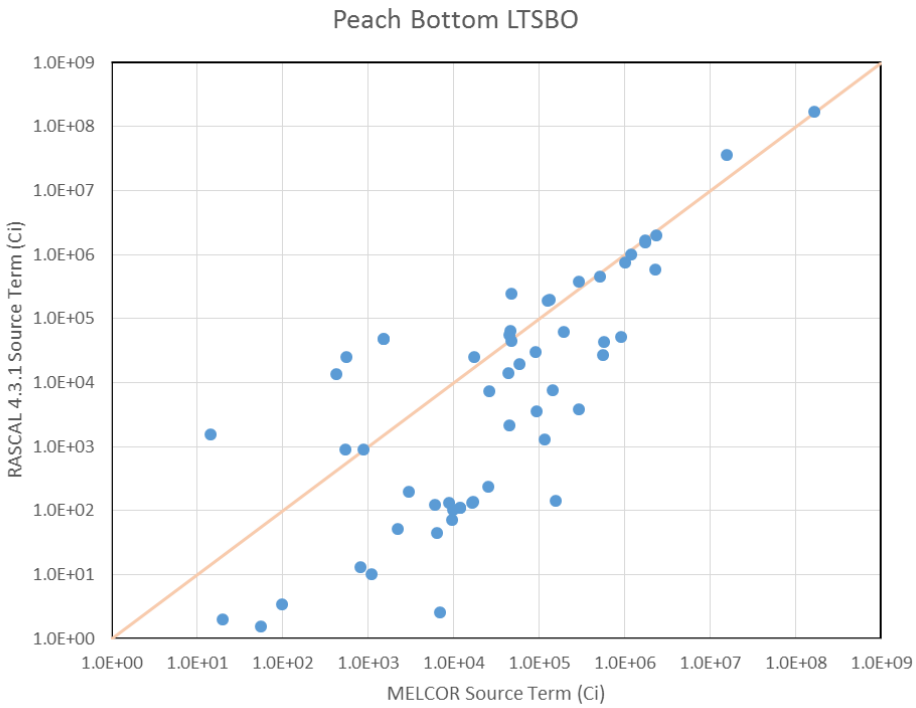


Figure 7-3 Comparison of MELCOR and RASCAL 4.3.1 source term estimates radionuclides

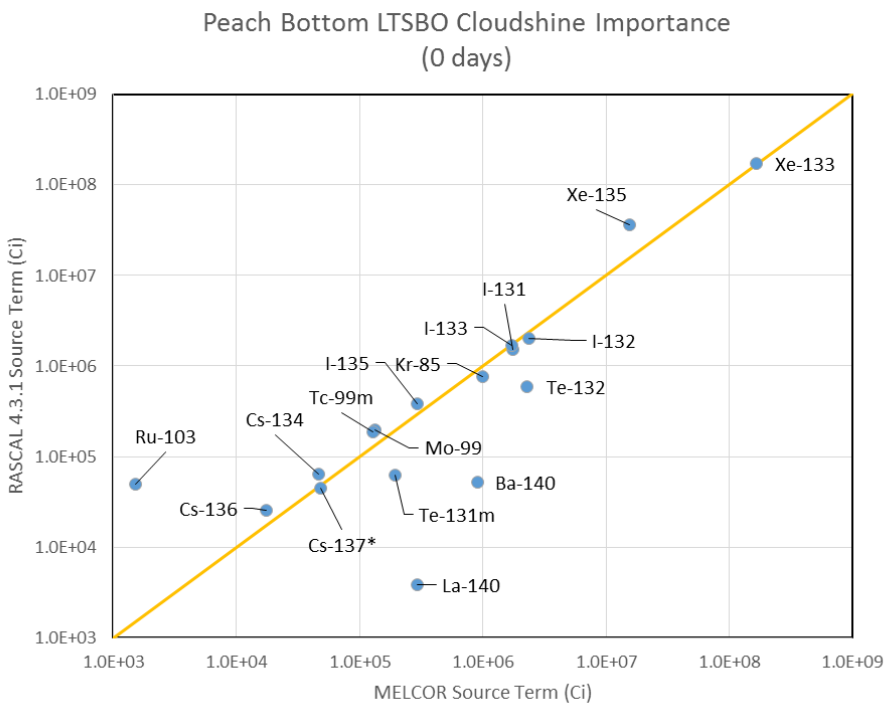


Figure 7-4 Comparison of MELCOR and RASCAL 4.3.1 source term estimates of the most important radionuclides contributing to cloud shine dose

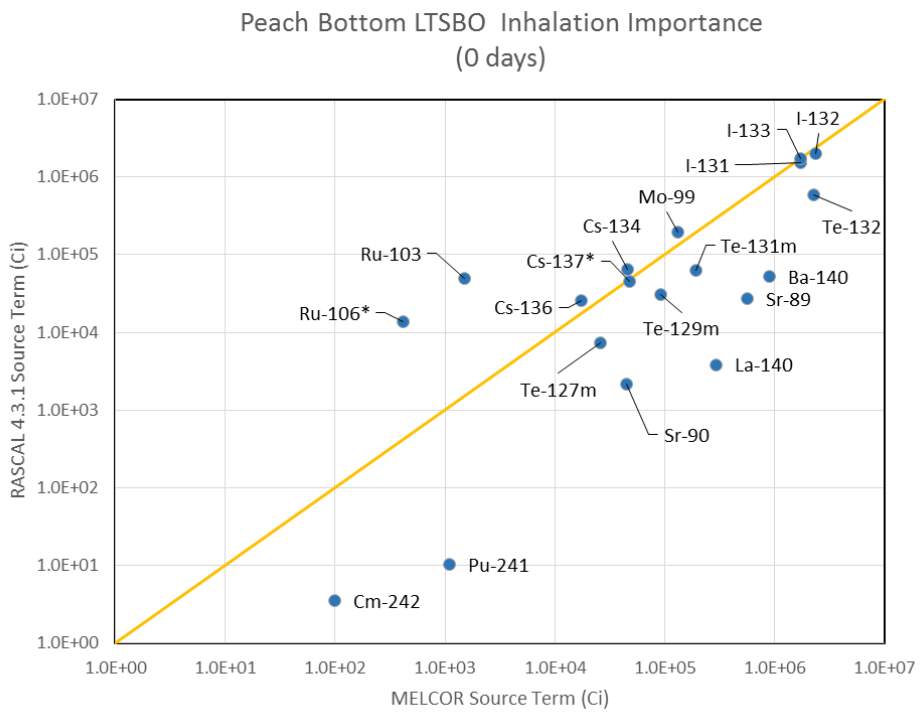


Figure 7-5 Comparison of MELCOR and RASCAL 4.3.1 source term estimates of the most important radionuclides contributing to inhalation dose

8.0 SOURCE TERM IMPORT, EXPORT AND MERGE

RASCAL 4 has always included a function to export the atmospheric source term to an external file. Early versions supported only the comma-separated values (CSV) format. RASCAL 4.3 introduces several new features related to handling the atmospheric source terms. The first feature is the option to export this source term in an extensible markup language (XML) format and to apply an importance filter. The second feature added is the capability to import an atmospheric source term to be used with the RASCAL atmospheric transport, dispersion, and deposition (ATD) models bypassing the RASCAL internal source term model. Finally, RASCAL 4.3 includes a new tool to export and merge source terms. When combined, these features and the new tool give RASCAL 4.3 the capability to provide support for multi-unit releases.

The following sections describe the formats of the CSV and XML files, the export and import processes, and the capabilities of the merge/export tool.

8.1 Source Term Import Option

RASCAL 4.3 accepts imported source terms in either CSV or XML format. RASCAL 4.3 also creates exported atmospheric source terms in these formats. The export can be done from within the Source Term to Dose (STDose) model or using the Source Term Merge/Export tool.

8.1.1 CSV File Format

The CSV format is a commonly used way to move data between programs. It is a useful format as it is in plain-text form and can be edited by a variety of applications. It is also directly supported by the Excel spreadsheet software. This allows a user to modify a source term generated by the RASCAL source term models to make adjustments for processes not modeled by RASCAL. The file can then be imported and used as the basis for a model run using the RASCAL ATD models. Figure 8-1 illustrates a sample of an exported RASCAL 4.3 atmospheric release source term file in the CSV format.

```
Creator, RASCAL v4.3.0 Source Term
File_Created, 2013/09/15  09:47
Site_Name, Arkansas - Unit 1
Release_Latitude, 35.310000
Release_Longitude, -93.231390
UTC_Offset, -6
Release_Height, 10.0 m
Case_Title, Basic 2 nuclide
Case_Runtime, 2013/09/15 09:46
Case_Desc,
Activity_Units, Ci
Other_Info, Importance filter, None
Interval,2013/09/15,2013/09/15,2013/09/15,2013/09/15,2013/09/15
Start,00:00,00:15,00:30,00:45,01:00
Cs-137,3.00E-01,3.00E-01,3.00E-01,3.00E-01,0.00E+00
I-131,1.11E+00,1.11E+00,1.11E+00,1.11E+00,0.00E+00
```

Figure 8-1 Example of a RASCAL 4.3 CSV format exported atmospheric release source term file

Table 8-1 presents a description of the lines in the exported atmospheric source term in the RASCAL 4.3 CSV file. Keywords start each line and must appear exactly as shown. The description indicates what follows the comma after the keyword. In some cases (e.g., time steps) there will be multiple items separated by commas. The table also indicates whether a line is required for import to RASCAL.

Table 8-1 Description of the lines in a RASCAL 4.3 CSV format exported atmospheric release source term file

Keyword	Description of values after keyword	Requirement for import file
Creator	Model name and version; result type	Optional
File_Created	Date and time the export file was created	Optional
Site_Name	Name of the location	Optional
Release_Latitude	Latitude in decimal degrees of the model release point.	Optional
Release_Longitude	Longitude in decimal degrees of the model release point.	Optional
UTC_Offset	Hours from UTC	Optional
Release_Height	Release height and units	Recommended
Case_Title	Title given to the modeling run	Optional
Case_Runtime	When the modeling run occurred	Optional
Activity_Units	Units for the reported activities	Recommended
Other_Info	Status of the importance filter option	Optional
Interval	The dates of each time step, separated by commas	Required
Start	The times of each time step, separated by commas. Must be exactly 15 minutes apart	Required
The above is followed by a line for each radionuclide in the release consisting of the radionuclide name, a comma, and the activity for that radionuclide for each time step (separated by commas)		Required

All the above elements are part of the exported CSV file. Only the last three are required for an import file. Note that the time step for RASCAL 4.3 atmospheric source terms is 15 minutes.

8.1.2 XML File Format

The XML format is used to create documents that are readable by both humans and computers. It is useful to create a structured file that can be validated easily. This format is included in RASCAL 4.3 as part of the effort to standardize information exchange formats. Figure 8-2 illustrates a sample of an exported atmospheric release source term file in the XML format.

```
<?xml version="1.0" encoding="utf-8"?>
<!--RASCAL atmospheric source term XML export file-->
<Atmospheric_SourceTerm>
<EventLocation Name="Arkansas - Unit 1" Latitude="35.31" Longitude="-93.23139"
  Elevation="103" UTC_Offset="-6" TimeType="Local 24 hour clock"/>
<Creator ModelName="RASCAL 4.3" CaseName="Basic 2 nuclide" ModelRunTimeStamp="2013-
  09-15T09:46:00" Description="" CreationDate="2013-09-15" CreationTime="09:47:14"
  Analyst_Name="Dose Analyst" OtherInfo="Importance filter: None"/>
<ReleasePoint Name="Release Point 1" Release_Height="10" Release_Height_Units="m"
  Start_Date="2013-09-15" Start_Time="00:00:00" End_Date="2013-09-15"
  End_Time="04:00:00" Activity_Units="Ci">
<Release_Step Nuclide_Count="2" Step_Sequence_Number="1" Start_Date="2013-09-15"
Start_Time="00:00:00">
  <Activity_Release Nuclide_Name="Cs-137" Released_Amount="3.00E-01"/>
  <Activity_Release Nuclide_Name="I-131" Released_Amount="1.11E+00"/>
</Release_Step>

…… repeats for each time step

<Release_Step Nuclide_Count="2" Step_Sequence_Number="5" Start_Date="2013-09-15"
  Start_Time="01:00:00">
  <Activity_Release Nuclide_Name="Cs-137" Released_Amount="0.00E+00"/>
  <Activity_Release Nuclide_Name="I-131" Released_Amount="0.00E+00"/>
</Release_Step>
</ReleasePoint>
</Atmospheric_SourceTerm>
```

Figure 8-2 Example of a RASCAL 4.3 XML format exported atmospheric release source term file

Table 8-2 presents a description of the lines in the exported atmospheric release source term in the RASCAL 4.3 XML file.

Table 8-2 Description of the lines in a RASCAL 4.3 XML format exported atmospheric release source term file

Line or Element	Description
Line 1	Standard XML file header
Line 2	Comment
Deposition_Sourceterm	Main element of the file; indicates that it is deposited material (not an atmospheric release)
EventLocation	Contains the following attributes describing the release point: Name Latitude (degrees; negative is South) Longitude (degrees; negative is West) Elevation (meters) UTC_Offset (hours from Greenwich) TimeType (always = Local 24 hour clock)
Creator	Contains the following attributes describing the tool creating the file: ModelName CaseName ModelRunTimeStamp (when the model was run) CreationDate (when this file was created) CreationTime (when this file was created) Analyst_Name OtherInfo (importance filter information)
Release_Point	Contains the following attributes describing the location of the release being reported; Name Release_Height Release_Height_Units Start_Date Start_Time End_Date End_Time Activity_Units
Release_Step	Repeats for 15 minute time step of the STDose time dependent source term Nuclide_Count – how many radionuclides released Step_Sequence_Number – from 1 to N, when N = number time steps Start_Date Start_Time
Activity_Release	There will be one element for each radionuclide in the release step. Nuclide_Name Released_Amount

8.2 Import Process

The atmospheric source term import option is available in the STDose model for all event types.

When an atmospheric source term is imported, RASCAL 4.3 obtains the time-dependent atmospheric source term and release height from the imported file. The RASCAL 4.3 modules that would normally calculate the source term and release information are bypassed.

Following is a list of how the import process handles inputs:

- No checking is done to make sure the site name or the geographic coordinates in the import file match the location specified in the STDose model. Any import file can be used for any site selected.
- The import file release height will be used if available. If the release height and/or units is missing or unrecognized, a release height of 10 meters (m) will be used.
- If the activity units are missing or unrecognized a default unit of curies (Ci) will be used.
- The time step must be exactly 15 minutes. The import process is aborted if a nonstandard time step is detected.
- Radionuclide names must be formatted as used by RASCAL. For example "Kr-85" [krypton-85] is made up of the following: first letter capitalized, following letters not capitalized; a dash followed by the numerals. Finally, any trailing characters such as 'm' or '*'. The import process checks the radionuclide name against the names in the RASCAL radionuclide database. It will recognize radionuclides for which RASCAL includes implicit daughters and will add the * (e.g., Cs-137 will be converted to Cs-137* for RASCAL use). Records containing unrecognized radionuclide names will be ignored but the import process will not be aborted.
- Up to 120 radionuclides can be in the import file. The import process will abort if there are 121 or more radionuclides.
- The import file can span more than 96 hours but only the first 96 hours of data will be imported. This matches the maximum 96-hour calculation duration of the STDose model.
- For the XML format, the file is validated against the schema (XSD). Failure to validate results in the import aborting.

RASCAL 4.3.1 expands the import function to build in missing daughters. For example, assume the source term was filtered on export to include only the top five contributors to dose and was then imported back into RASCAL. In RASCAL 4.3 an error might have been generated as some daughters needed by the ATD models may have been removed by the filtering process. An upgrade to RASCAL 4.3.1 detects those missing daughters and adds them when needed so that the import process can complete.

8.3 Export Process

The atmospheric source term export option is available for all STDose model event types. It is accessed from the Release vs. Time screen from the Source Term tab on the main STDose display.

The exported file can be in either CSV or XML format. Section 8.1 provides a description of the formats. There are two options that impact the formatting of the export file. First, there is an option to strip * characters from the radionuclide names. This is useful when the export file is to be used with other software (e.g., National Atmospheric Release Advisory Center) that does not indicate that implicit daughters are built-in with the asterisk. Second, there is an option to use the RASCAL 4.2 format. This will not include all the release information but is retained for backward compatibility. This last option applies to the CSV format only.

There is also an option to filter the list of nuclides in the source term to be included in the export file. The default is to include all the radionuclides in the release to the atmosphere. However, this can be a list of 70 plus radionuclides and may be problematic when imported by other codes. The export process can be configured to export only a select set of radionuclides that are important to dose. Section 6 provides a detailed discussion of the RASCAL 4.3 importance calculation. The user may specify the filter to export only a certain top number of radionuclides that contribute to dose or to export all the nuclide needed to get a percentage of the dose.

Note: The Filter by Importance to Dose option should not be used if the exported source term file is to be imported into RASCAL 4.3. The import function in RASCAL 4.3 is not able to add any missing daughters. However, filtered files will import properly into RASCAL 4.3.1.

8.4 RASCAL 4.3 Source Term Merge/Export Tool

The Source Term Merge/Export tool adds an external capability to the RASCAL 4.3 STDose model. It provides two basic capabilities. First, it can export the atmospheric source term from a saved STDose case in either the CSV or XML format. Second, it can merge atmospheric source terms from multiple STDose model runs into a single new source term that can then be exported.

8.4.1 Load Cases

The merge/export tool works with completed and saved STDose case files. The saved cases to be exported or merged must include the calculated results. Cases saved before doing the dose calculations do not contain the source term information and cannot be used.

The load operation extracts the atmospheric source term component from the case file. If a single case is loaded it can only be exported. If multiple cases are loaded then they must be merged before exporting.

8.4.2 Merge

Before the merge begins, the user needs to review these parameters and select which values to use. There is no requirement that the source terms to be merged be from the same release point. No checking is done on the loaded cases to enforce a location match.

As such, the release height and latitude/longitude cannot be merged. The RASCAL user must select from one of the values used in the loaded cases. For example, only a single release height is allowed. If the RASCAL cases used differing release heights, the user must choose the height to be used in the merged file from one of the heights in the cases to be merged. The case title, analyst name, and site name can be edited as needed. The case descriptions for the cases to be merged are concatenated and presented in an editable form. The use may modify this text to adequately describe the merged source terms.

The time steps of the merge file begin with the earliest time found in the loaded cases. If the time steps of the other cases do not align exactly on the 15-minute steps, the activity is divided up between time steps as needed. There can be time gaps between the individual case source terms where no activity is released. The merge tool will build in the needed time steps with zero activity. The combined source terms cannot exceed 7 days in total duration.

8.4.3 Export

The export function in the Merge/Export tool operates the same and has the same options as the export in the STDose mode (Section 8.3).

8.4.4 Multi-Unit

The merging of atmospheric source terms provides a means to calculate doses for releases from multiple units on a single nuclear power plant site. This technique assumes that the releases occur at the same location. In reality, the individual release points may be separated by as much as several hundred feet. This separation may cause the results near the release points to be in error, but it should not have a significant effect on the results calculated for offsite locations.

The general procedure to be followed is to run RASCAL 4.3 for each release individually and save the cases as usual. When all individual cases have been run, use the Source Term Merge/Export tool to create and export a combined source term file. Then re-run RASCAL 4.3 and import the merged combined source term file to calculate doses of the combined source term.

9.0 REFERENCES

[1] **NUREG-1940**, "RASCAL 4: Description of Models and Methods," U.S. Nuclear Regulatory Commission, Washington, DC, December 2012.

[2] **NUREG-1465**, "Accident Source Terms for Light-Water Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC, February 1995.

[3] **NUREG-1935**, **Part 1**, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," U.S. Nuclear Regulatory Commission, Washington, DC, November 2012.

[4] **NUREG-1935**, **Part 2**, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," U.S. Nuclear Regulatory Commission, Washington, DC, November 2012.

[5] **NUREG/CR-7110**, **Volume 1**, **Revision 1**, "State-of-the-Art Reactor Consequence Analysis Project Volume 1: Peach Bottom Integrated Analysis," U.S. Nuclear Regulatory Commission, Washington, DC, May 2013.

[6] **NUREG/CR-7110**, **Volume 2**, **Revision 1,** "State-of-the-Art Reactor Consequence Analysis Project Volume 2: Surry Bottom Integrated Analysis," U.S. Nuclear Regulatory Commission, Washington, DC, August 2013.

[7] **NUREG-1228**, "Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents," U.S. Nuclear Regulatory Commission, Washington, DC, October 1988.

[8] **NUREG/CR-3727**, "Fission Product Removal in Engineered Safety Feature (ESF) Systems," U.S. Nuclear Regulatory Commission, Washington, DC, April 1984 (available in Agencywide Documents Access and Management System (ADAMS) under Accession No. 8405220035).

[9] **NUREG/BR-0150**, **Volume 1**, **Revision 4**, "Response Technical Manual (RTM)-96," U.S. Nuclear Regulatory Commission, Washington, DC, March 1996.

[10] **SECY-13-0112**, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark 1 Boiling-Water Reactor," U.S. Nuclear Regulatory Commission, Washington, DC. October 9, 2013.

[11] **U.S. Environmental Protection Agency (EPA)**, "PAG Manual Protective Action Guides And Planning Guidance For Radiological Incidents," Draft for Interim Use and Public Comment, Washington, DC, March 2013.

[12] **U.S. Food and Drug Administration**, "Guidance Potassium Iodide as a Thyroid Blocking Agent in Radiation Emergencies; Availability" *Federal Register*, Vol. 66, December 11, 2001, p. 64046 (66 FR 64046).

[13] **U.S. Food and Drug Administration**, **Revision 1**, "Guidance for Industry – KI in Radiation Emergencies – Questions and Answers," Rockville, MD, December 2002.

[14] **EPA 402-R-99-001**, “Cancer Risk Coefficients for Environmental Exposure to Radionuclides,” Federal Guidance Report No. 13, U.S. Environmental Protection Agency, Washington, DC, September 1999.

[15] **EPA-520/1-88-020**, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” Federal Guidance Report No. 11, U.S. Environmental Protection Agency, Washington, DC. September 1988.

[16] **DCPAK3.02**, “Dose and Risk Coefficient File Package for FGR 13,” FGR13PAK, Oak Ridge National Laboratory, Oak Ridge, TN. 2011.

[17] **U.S. National Weather Service**, “National Digital Forecast Database (NDFD),” http://www.nws.noaa.gov/ndfd, Silver Spring, MD. November 2013.

[18] **U.S. Geological Survey**, “Global Digital Elevation Model (GTOPO30),” EROS Data Center Distributed Active Archive Center (EDC DAAC), https://lta.cr.usgs.gov/GTOPO30, Sioux Falls, SD, 1996.

[19] **Raytheon ITSS and U.S. Geological Survey**, “Completion of the 1990's National Land Cover Data Set for the Conterminous United States,” *Photogrammetric Engineering and Remote Sensing*, Vol. 67, p. 650-662, Sioux Falls, SD, 2001.

[20] **EPA-454/B-08-001**, “AERSURFACE User’s Guide,” U.S. Environmental Protection Agency, Washington, DC, January 16, 2013.

[21] **NUREG/CR-6703**, “Environmental Effects of Extending Fuel Burnup Above 60 GWd/MTU,” U.S. Nuclear Regulatory Commission, Washington, DC, January 2001.

[22] **NUREG-1793**, **Supplement 2**, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Plant Design,” U.S. Nuclear Regulatory Commission, Washington, DC, September 2011.

[23] **26A6642AH**, **Revision 9**, “ESBWR Design Control Document,” GE-Hitachi Nuclear Energy Americas LLC, Wilmington, NC, December 2010 (available in ADAMS under Accession No. ML103440215).

[24] **UAP-HF-13212**, **Revision 4**, “Submittal of US-APWR Design Control Document Revision 4 in Support of Mitsubishi Heavy Industries, Ltd.'s Application for Design Certification of the US-APWR Standard Plant Design, Arlington, VA, September 10, 2013 (available in ADAMS under Accession No. ML13262A304).

[25] **AREVA NP Inc.**, Letter from Sandra M. Sloan Director of Regulatory Affairs, AREVA NP Inc. to Document Control Desk NRC, NRC:11:085 ‘Submittal of Revision 3 of the U.S. EPR Final Safety Analysis Report for Design Certification,’ Lynchburg, VA, August 10, 2011 (available in ADAMS under Accession No. ML11230A572).

[26] **NUREG-0016**, **Revision 1**, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code),” U.S. Nuclear Regulatory Commission, Washington, DC, January 1979.

[27] **NUREG-0017**, **Revision 1**, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code),” U.S. Nuclear Regulatory Commission, Washington, DC, April 1985.

[28] **ANSI/ANS-18.1-1999**, “Radioactive Source Term for Normal Operation of Light Water Reactors,” American Nuclear Society, La Grange Park, IL, September 21, 1999.

[29] **EPA-402-R-93-081**, “External Exposure to Radionuclides in Air, Water, and Soil,” Federal Guidance Report No. 12, U.S. Environmental Protection Agency, Washington, DC, September 1993.

APPENDIX A NUREG-1940 ERRATA INFORMATION

The following appendix contains the errata information for NUREG-1940. The revised information is denoted in red (bold).

Section 4.5.2.2 Gases (page 4-20):

When the temperature falls below about **-**3ºC, the physical characteristics of snow change. As a result, there is a significant decrease in the wet deposition velocity of gases. RASCAL 4 does not calculate scavenging of gases by snow when the temperature is less than **-**3ºC (i.e., the wet deposition velocity is set to zero).

Section 6.4.3 Mixing Height, Equation (6-7) (page 6-11):

The exponential interpolation of the mixing height is:

$$H(t) = a \cdot exp\left(-\frac{bt}{24}\right), \tag{6-7}$$

where

$H(t)$	= mixing height (m) at time t
a	= H_{morn} /exp(-b)
b_i	= 24 ln(H_{aft}/H_{morn}) **/** (24 - t_{sunset})
H_{morn}	= morning mixing height (m)
H_{aft}	= afternoon mixing height (m)
t_{sunset}	= time of sunset (h)

Appendix A, Table A-4 RASCAL 4 Radionuclide Decay Chains (Page A-14):

Parent Name	Depos Type	Parent HL (days)	No. of Dau	D1 Name	D1 PI	D1 HL (days)	D1 BF	D2 Name	D2 PI	D2 HL (days)	D2 BF	D3 Name	D3 PI	D3 HL (days)	D3 BF
Sr-83	2	1.35E+00	1	Rb-83	1	8.62E+01	**1**								
Tc-93m	2	3.02E-02	**2**	Mo-93	1	1.28E+06	0.182	Tc-93	1	1.15E-01	0.818				
Os-182	2	9.17E-01	**1**	Re-182a	1	5.29E-01	1								
Po-205	2	7.50E-02	**3**	Pb-201	1	3.92E-01	0.0014	Tl-201	2	3.04E+00	1	Bi-205	1	1.53E+01	0.9986
U-230	2	2.08E+01	**3**	Th-226*	1	2.15E-02	1	Pb-210	2	8.14E+03	1	Bi-210	3	5.01E+00	1

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RASCAL 4.3 contains a number of new features and revision of several old features in response to the lessons learned by the U.S. Nuclear Regulatory Commission staff during its response to the events at the Fukushima Daiichi nuclear power plants following the March 11, 2011 earthquake off the coast of Japan and the tsunami that it triggered. This document is a supplement to NUREG-1940, "RASCAL 4: Description of Models and Methods," which is the technical basis for the models and methods used in the RASCAL computer code versions 4.0, 4.1 and 4.2. This supplement contains the technical basis for changes and additions to RASCAL implemented in versions RASCAL 4.3 and 4.3.1. Additionally, Appendix A provides errata information for NUREG-1940.

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