RASCAL 4.3 Technical Supplement (Draft)

Draft Manuscript: October 2013

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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 Dai-ichi 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 *RASCAL 4.3: Description of Models and Methods* (NUREG-1940). It contains the technical basis for changes and additions to RASCAL implemented in RASCAL 4.3.

RASCAL 4.3 adds a long-term station blackout source term model based on the results of the State-of-the-Art Reactor Consequences Analyses (SOARCA) study. It extends the RASCAL domain to 100 miles, and it adds a child thyroid dose calculation. RASCAL 4.3 includes new utility programs to acquire meteorological data from the internet, calculate custom radionuclide inventories for reactor cores and spent fuel, display a nuclide activity balance within the power plant components and atmosphere, and to sort and display the nuclides 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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Acknowledgements

The improvements to the RASCAL computer code would not have been possible without the guidance, feedback and help of all the users of the code. Specific contributors to this report include all NRC Headquarters and Regional Performance Monitoring Team members, State Emergency Response Personnel and Louis Brandon of NRC's Office of Nuclear Security and Incident Response. We would like to thank the NRC staff and RASCAL users who have provided comments on their experiences with RASCAL and suggestions for RASCAL improvement. The software continues to grow and improve because of your efforts and ideas.

Abbreviations

ABWR	Advanced Boiling-Water Reactor
AP1000	Advanced Passive Reactor
Bq	becquerel
BWR	Boiling-Water Reactor
Ci	curie
CEDE	committed effective dose equivalent
CF	conversion factor
CSV	comma separated values
DCD	Design Control Document
DCF	dose conversion factor
EPA	Environmental Protection Agency
EPR	Evolutionary Power Reactor
ESBWR	Economic Simplified Boiling-Water Reactor
FDA	U.S. Food and Drug Administration
FMDose	Field Measurement to Dose
FSAR	Final Safety Analysis Report
GTOPO30	Global Digital Elevation Model
gsd	geometric standard deviation
GWd	Gigawatt day
in^2	inches squared
KI	potassium iodide
LOCA	loss of coolant accident
LTSBO	long term station blackout
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
Mi	miles
MWd	Megawatt day
MWt	Megawatts thermal
MTU	Metric Ton of Uranium
NOAAMP	National Oceanic and Atmospheric Administration Meteorological Processor
NLCD	National Land Cover Dataset
NRC	U.S. Nuclear Regulatory Commission
NWS	National Weather Service
psia	pounds per square inch absolute
psig	pounds per square inch gauge
PWR	Pressurized Water Reactor
RASCAL	Radiological Assessment System for Consequence Analysis
RCS	reactor coolant system
SOARCA	State-of-the-Art- Reactor Consequence Analyses
STDose	Source Term to Dose
TEDE	total effective dose equivalent
US-APWR	US Advanced Pressurized-Water Reactor
XML	extensible markup language

Introduction

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 Dai-ichi 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 RASCAL 4.3: Description of Models and Methods (NUREG-1940). It is one of a set of documents that describe RASCAL. *RASCAL 4: Description of Models and Methods* (NUREG-1940) (referred to as the RASCAL technical document) which presents the technical bases for the RASCAL computation codes describes RASCAL 4.2.

This supplement describes the changes from RASCAL 4.2 to RASCAL 4.3 and provides the technical bases for those changes. The *RASCAL 4.3 User's Guide* provides an introduction to the user interface and data required to run RASCAL. Finally, the *RASCAL 4.3 Workbook* provides a systematic approach to learning how to use RASCAL through a set of problems.

This Supplement has 10 chapters. The first chapter describes the changes and additions to the source term calculations (Chapters 1, 2, and 3 of NUREG-1940). The second chapter describes the changes to the atmospheric transport, dispersion, and dose calculations (Chapter 4 of NUREG-1940). Chapters 3 through 6 describe new utility programs that are included as part of RASCAL 4.3. Chapter 3 describes the new meteorological data acquisition module. Chapter 4 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. Chapter 5 describes a new utility to display the activity and activity transfers at several locations in a power plant and in the atmosphere as a function of time during an accident. Chapter 6 describes new utility tool to evaluate the importance to dose pathways of nuclides released to the atmosphere. Chapter 7 describes the new source term import option and the new Source Term Merge/Export tool.

RASCAL does not calculate health effects. Chapters 9 and 10 present summaries of information on exposure guidelines prepared by other agencies and on health effects of exposure to radiation and uranium, uranium hexafluoride, and hydrogen fluoride. These additional topics are presented to provide context for evaluating the environmental consequences (doses and chemical exposures) calculated by RASCAL.

1 Changes to Source Term Calculations

This chapter describes the changes to the RASCAL source term module implemented in RASCAL 4.3. The most significant change is the addition of a long-term station blackout model. The model, which delays the onset of releases for several hours after the loss of power, is described in Section 1.1. Section 1.2 describes the option to specify a source term based on an estimate of core damage. Section 1.3 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 1.4 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 1.5 discusses changes to the calculation spent fuel source terms. The changes improve the estimates of the nuclide inventory at risk in these accidents.

Several additions to RASCAL 4.3 related to source terms are described in later chapters. Chapter 4 describes a utility program that can be used to estimate custom nuclide inventories in reactor cores, in reactor cooling systems, and in spent fuel. Chapter 5 describes a utility program that provides users with estimates of activity of selected nuclides in several locations within the plant and the activity released to the atmosphere as a function of time. Finally, Chapter 7 describes a utility program that can be used to import source terms from other programs, export RASCAL source terms for use by other programs, or merge 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.

1.1 Long-Term Station Blackout (LTSBO)

One of the first lessons learned in the NRC response to the Fukushima Dai-ichi accident was that RASCAL did not have a readily apparent method for calculating reactor source terms for long-term station blackout (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,* (Soffer et al. 1995). Although long-term station blackout 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 (LOCAs). Long-term station blackout accidents have a much longer time frame than LOCAs because loss of coolant occurs over a period of hours rather than seconds or minutes. An LTSBO release type option has been added to RASCAL 4.3 to facilitate source term calculations for this type of accident.

The LTSBO implementation in RASCAL is based on the LTSBO accident progression as described in a report “Implementation of New Source Term Information for RASCAL,” (Jun et al. 2009) and NUREG-1935, *State-of-the-Art Reactor Consequence Analyses* (SOARCA) Report, (Chang et al. 2012). 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 nuclide classes as a function of time. Figure 1-1 shows the SOARCA release fractions for pressurized-water reactors (PWRs) for nine nuclide groups. The initial releases are primarily noble gases. The noble gases are followed by the other nuclides. Figure 1-2 shows the SOARCA release fractions for boiling-water reactors (BWRs) for the same nine nuclide groups. (Note: The nine groups used in the SOARCA analyses are not exactly the same as those used in NUREG-1465.)

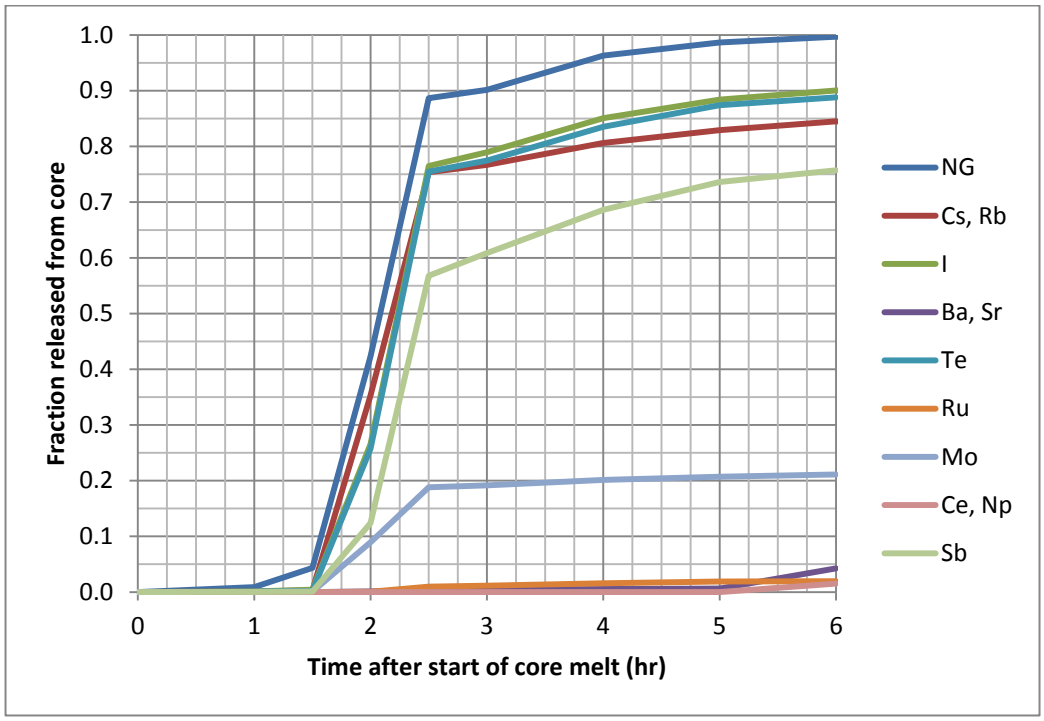


Figure 1-1 RASCAL 4.3 Nuclide Release Timing for a PWR LTSBO Accident

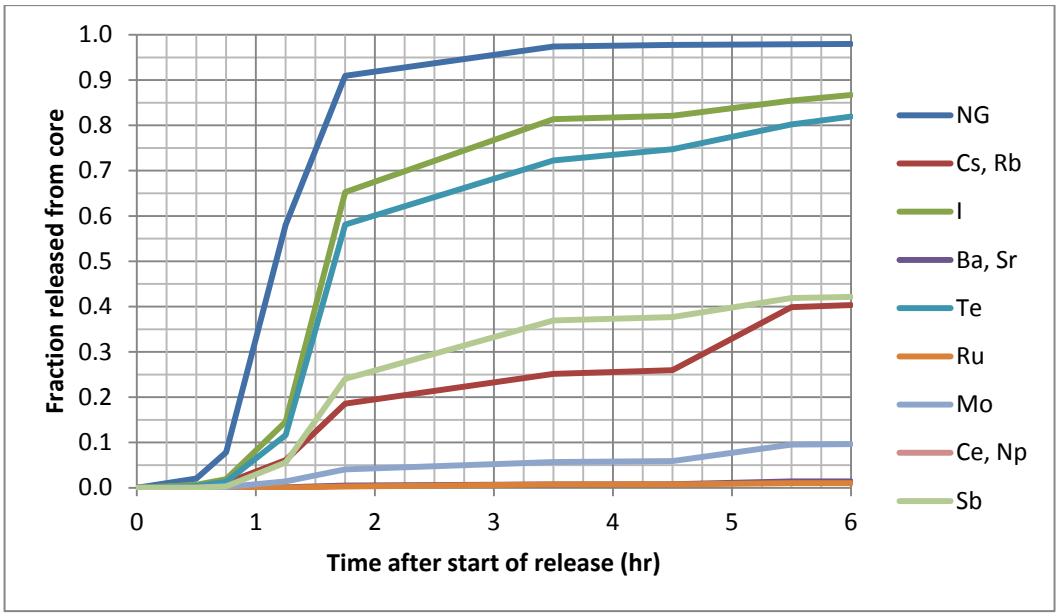


Figure 1-2 RASCAL 4.3 Nuclide Release Timing for a BWR LTSBO Accident

Time zero in Figures 1-1 and 1-2 is the beginning of release, not the time of the initiating event. As indicated above, the time from the LSBO initiating event to the beginning of release from the reactor may take several hours. Table 1-1 gives the default delay times RASCAL 4.3 assumes for battery depletion and the beginning of release from the reactor. However, as shown by the Fukushima Dai-ichi reactor operators, various emergency measures can extend these delays.

Table 1-1 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 hr	4 hr
Start of Release	12 hr	10 hr

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

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 nuclides. The release fractions and release timing are also different. Figures 1-3 and 1-4 compare the release timing for LTSBOs with the timing for LOCAs for the first 6 hours 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 and ultimate release fractions for important nuclide groups such as the iodines and the cesiums.

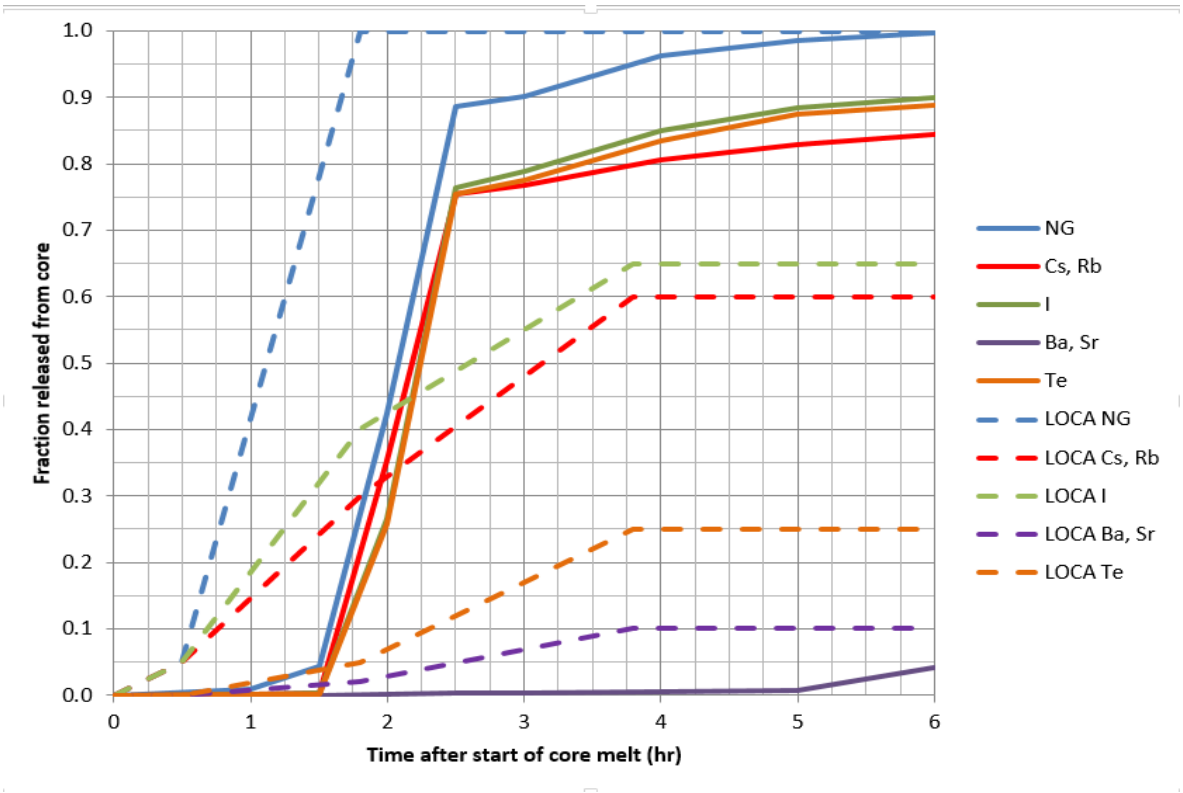


Figure 1-3 Comparison of Release Timing for LTSBO and LOCA Events for PWRs

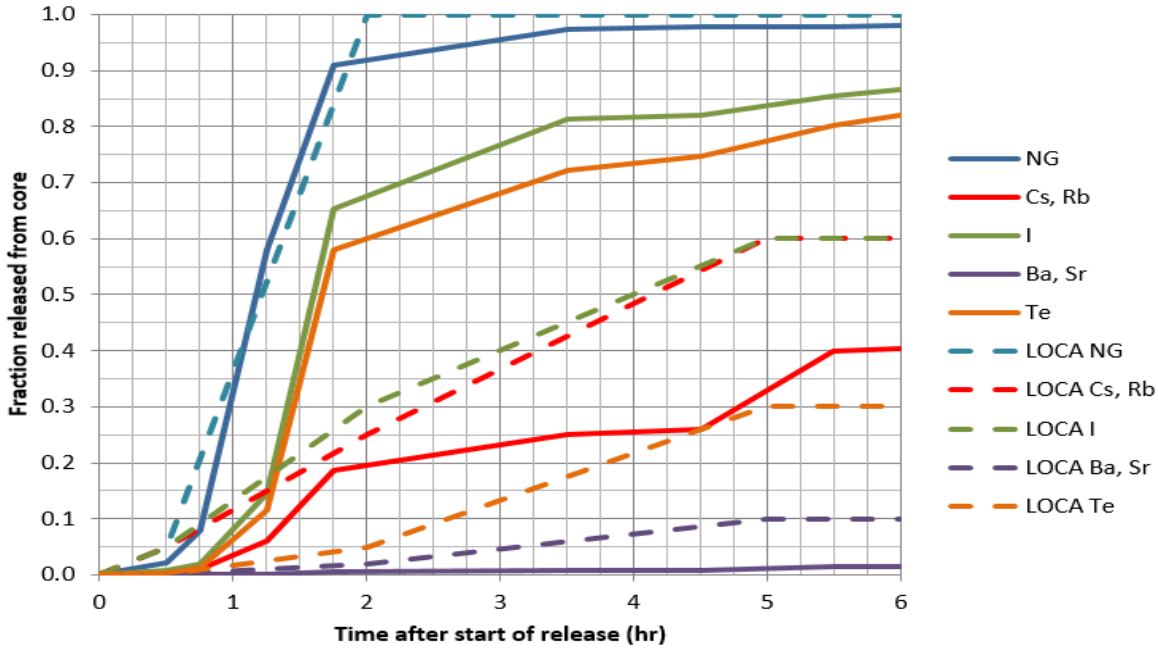


Figure 1-4 Comparison of Release Timing for LTSBO and LOCA Events for BWRs

Figures 1-5 and 1-6 compare the activity released to the environment for individual nuclides. Figure 1-5 compares activities released at 10 m in the first 8 hours from PWR accidents via the containment leakage pathway without depletion by sprays or filters. Figure 1-6 compares activities in the first 8 hours of an elevated release for a well-well pathway through the standby gas treatment system with active filters. The nuclides identified in these figures provide an indication of which nuclides are more prevalent in LTSBOs (those above the diagonal line) and which are more prevalent in LOCAs.

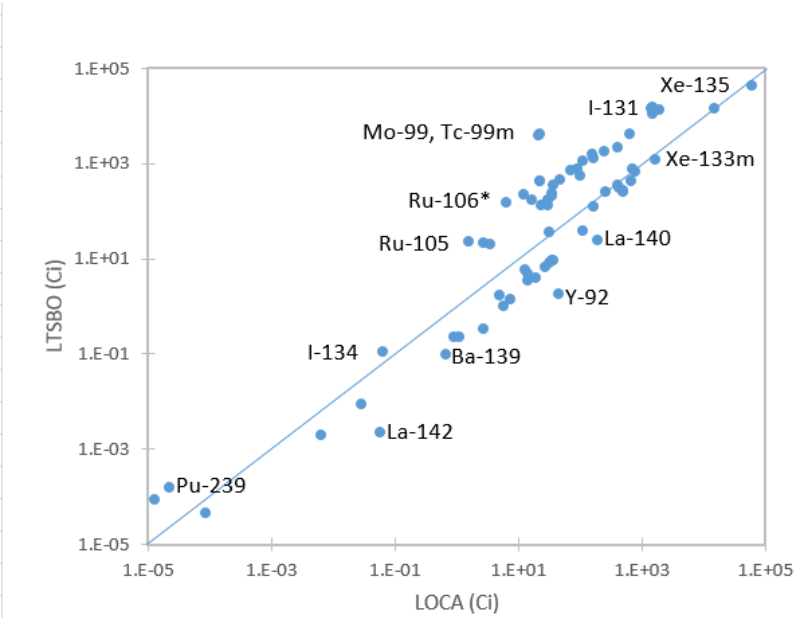


Figure 1-5 Comparison of Activity Released to the Atmosphere in the First 8 Hours after Beginning of Release for PWR LTSBO and LOCA Events

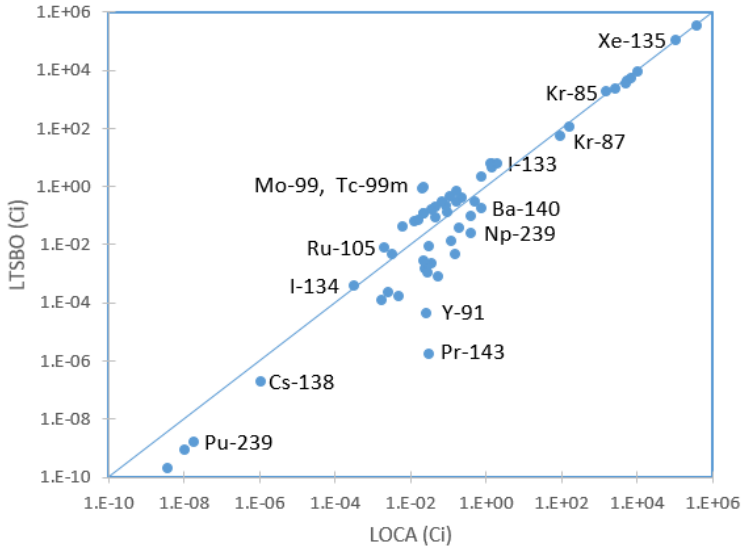


Figure 1-6 Comparison of Activity Released to the Atmosphere in the First 8 Hours after Beginning of Release for BWR LTSBO and LOCA Events

The activity release to the atmosphere in these events is summarized and compared in Table 1-2 and Table 1-3. 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 BWR events release more activity to the atmosphere than the PWR event, but because the BWR pathway was filtered almost all of the released activity was noble gases. A very small fraction was iodines or other particles.

Table 1-2 Activity Released to the Atmosphere in the PWR Events

	LOCA		LTSBO		LTSBO/LOCA
	Ci	**% of Total**	**Ci**	**% of Total**	
Noble Gas	7.9×10^{4}	87.7	6.2×10^{4}	43.8	0.79
Iodines	5.5×10^{3}	6.1	4.3×10^{4}	30.4	7.8
Other	5.6×10^{3}	6.2	3.6×10^{4}	25.8	6.4
Total	9.0×10^{4}		1.4×10^{5}		1.6

Table 1-3 Activity Released to the Atmosphere in the BWR Events

	LOCA		LTSBO		LTSBO/LOCA
	Ci	**% of Total**	**Ci**	**% of Total**	**ratio**
Noble Gas	5.3×10^{5}	100.0	4.6×10^{5}	100.0	0.87
Iodines	6.0×10^{0}	0.0	1.9×10^{1}	0.0	3.2
Other	5.6×10^{0}	0.0	1.1×10^{1}	0.0	2.0
Total	5.3×10^{5}		4.6×10^{5}		0.87

Table 1-4 compares the consequences of these events using RASCAL computational results extracted from the Maximum Value Table for a receptor at a distance of 2 mi. The doses were calculated using RASCAL standard meteorology (4 mph wind, D stability, no precipitation). 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 TEDE and cloudshine doses are lower than those for the LOCA event. Two factors that contribute to this result are:

- the SBGTS filters significantly reduced the release of iodines and other particles, and
- 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 1-4 Comparison of RASCAL Dose Estimates for LTSBO and LOCA Events

	PWR			BWR		
Dose	**LOCA**	**LTSBO**	**Ratio**	**LOCA**	**LTSBO**	**Ratio**
Total EDE[a]	1.5×10^{-1}	9.0×10^{-1}	6.0	8.3×10^{-3}	7.2×10^{-3}	0.89
Thyroid EDE [a] [a]	9.7×10^{-1}	7.3×10^{0}	7.5	1.0×10^{-3}	3.1×10^{-3}	3.1
Child Thyroid EDE [a]	1.9×10^{0}	$1.4\times10^{+1}$	7.4	2.0×10^{-3}	5.9×10^{-3}	3.0
Inhalation CEDE [a]	1.2×10^{-1}	6.9×10^{-1}	5.8	***[b]	***	
Cloudshine [a]	2.6×10^{-3}	1.2×10^{-2}	4.6	7.7×10^{-3}	6.7×10^{-3}	0.87
4-day Groundshine [a]	3.0×10^{-2}	2.0×10^{-1}	6.7	***	***	

[a] rem
[b] *** less than 1.0×10^{-3}

1.2 Core Damage State

The Core Damage State option from RASCAL 3.05 has been restored in RASCAL 4.3. It is an option for both the LTSBO and LOCA source terms. 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.

For LTSBO events, the core uncovered duration required to arrive at the core damage state is estimated from release timing in the SOARCA reports. As stated above, the SOARCA reports do not associate releases with the gap, core-melt, and vessel melt through labels that were assigned to the releases in NUREG-1465. For consistency with the nomenclature in NUREG-1465 and earlier versions of RASCAL, these labels have been added to the LTSBO releases for RASCAL 4.3. Table 1-5 lists the phases and durations of LTSBO releases.

Table 1-5 RASCAL 4.3 LTSBO Release Phases and Durations (hr)

Release Stage	PWR	BWR
Cladding Failure	1.5	0.75
Core Melt	3.5	4.75
Vessel Melt Through	1.0	0.5
Total duration	6.0	6.0

For LOCA events the core uncovered duration is estimated from accident sequences in Table 3.16 of NUREG-1465. Table 1-6 reproduces the release stage duration estimates of NUEG-1465.

Table 1-6 RASCAL 4.3 LOCA Release Phases and Durations (hr)

Release Stage	PWR	BWR
Cladding Failure	0.5	0.5
Core Melt	1.3	1.5
Vessel Melt Through	2.0	3.0
Total duration	3.8	5.0

1.3 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 nuclides are accumulated RCS as before. Noble gases entering the primary system of a PWR are accumulated as are other nuclides, 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 earlier versions of RASCAL. Also, because the noble gases aren't accumulated the in BWR RCSs or PWR, the total noble gas release in RASCAL 4.3 is larger than it is in RASCAL 4.2. Figure 1-7 illustrates the differences in noble gas release for a PWR LOCA with bypass release calculated using RASCAL default parameters.

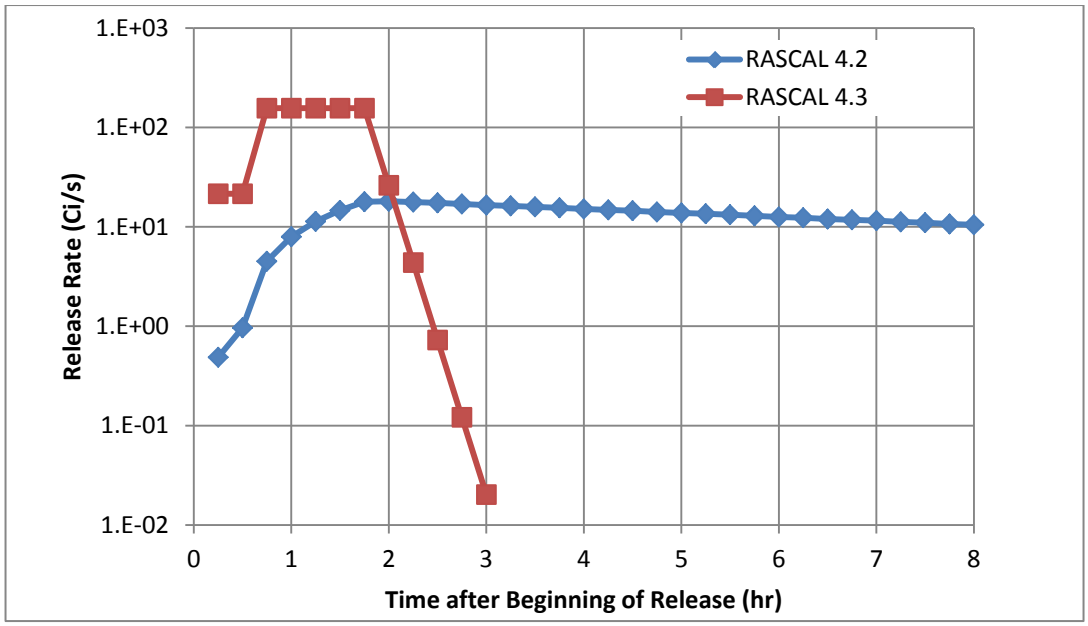


Figure 1-7 Kr-85 release rate comparison for a PWR LOCA with a bypass release

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 1-8 shows the cumulative cloud shine dose at 1 mile 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.

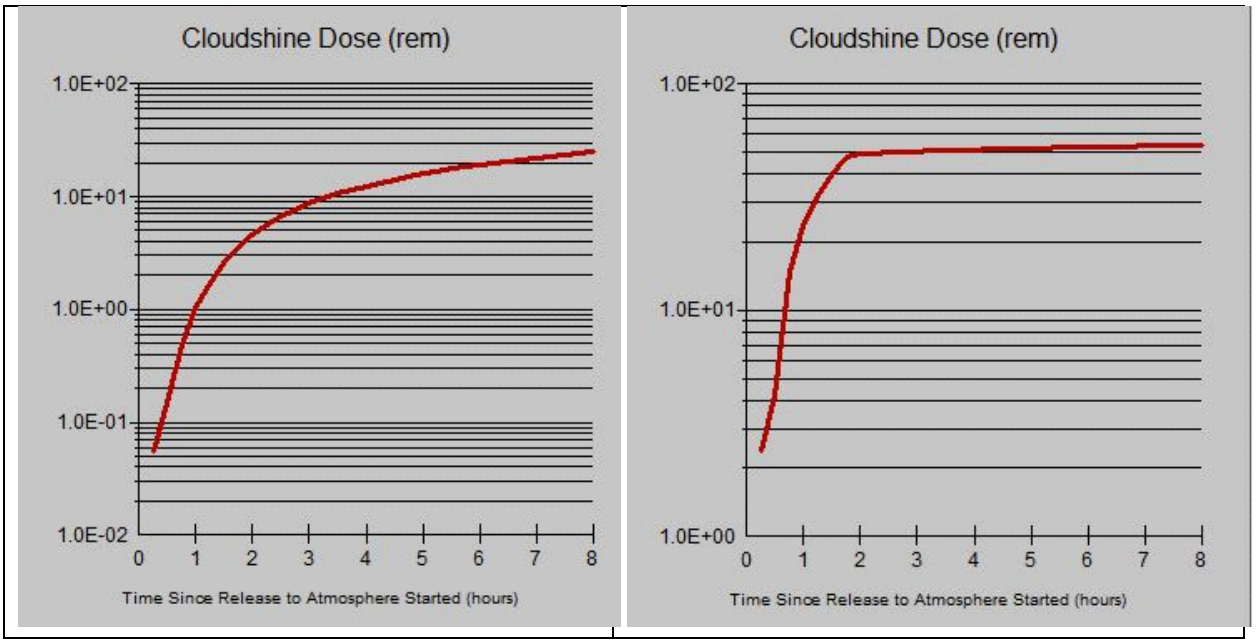


Figure 1-8 Comparison of RASCAL 4.2 (left) and RASCAL 4.3 (right) Cloud Shine Dose Accumulation

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 nuclides such as I-132, I-133, I-134, and I-135 to the environment compared with previous versions of RASCAL. Table 1-7 shows the changes in iodine activity released the environment for a LOCA/Bypass accident that are 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 1-7 Comparison of RASCAL 4.2 and RASCAL 4.3 iodine releases for a PWR LOCA/Bypass Accident

Nuclide	Half Life (hr)	DCF (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 1-7 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 CEDE at 1 mi calculated by RASCAL 4.2 for a PWR LOCA/Bypass accident is 141 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 nuclides having relatively small dose conversion factors. The release of I-131, which has the largest iodine dose conversion factor, is essentially unchanged from RASCAL 4.2 to RASCAL 4.3.

Daughter ingrowth in the RCS does increase the release of some nuclides, 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 nuclides by about an additional 35 percent and 10 percent, respectively. Noble gases originating in the secondary system from ingrowth are immediately vented to the environment.

1.4 Pressure-Hole Size Release Model

Section 1.5.2 of the RASCAL 4 technical documentation (NUREG-1940) (Ramsdell et al. 2012) describes the RASCAL option for estimating containment release rate base 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 rated based on containment pressure and hole size has been revised in RASCAL 4.3. The code now 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.

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 psia, that the temperature is 68ºF 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 less than 1 in^2, RASCAL assumes that the containment leak rate is equal to the containment design leak rate. If the hole size is larger than 1 in^2, RASCAL uses equation 1-12 from the RASCAL 4 technical documentation 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.

When RASCAL determines that the release rate is to be calculated by the containment pressure and hole size, it checks to see if the increase in containment temperature is sufficient to account for the increase in containment pressure. If it is, RASCAL 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 assumes that the increase in pressure is the result of a major leak in the RCS. In this case RASCAL calculates the containment atmosphere density assuming that the water in the RCS has been drained to containment and has evaporated. RASCAL also adds the activity in the RCS to the activity in containment.

As the containment leaks to the environment RASCAL updates the containment pressure and leak rate to account for the decrease in mass of the containment atmospheric. Figures 1-9 and 1-10 show the decrease in containment pressure and leak rate with time for various hole sizes for a PWR LOCA. The initial pressure for all cases is 61 psig. For the smaller holes, the containment pressure does not decrease to atmospheric pressure within the 8 hr simulation. It does for the largest hole. Containment design pressure for PWRs is generally in the 40 to 60 psig range. However, the containment design pressure for BWRs is considerably lower, about 15 psig. Specific containment design pressures for reactors are provided in the RASCAL 4.3 Case Summary. Note that the larger hole sizes may be larger than can be considered to fall within the small hole assumption of the equation used to calculate flow out of the containment.

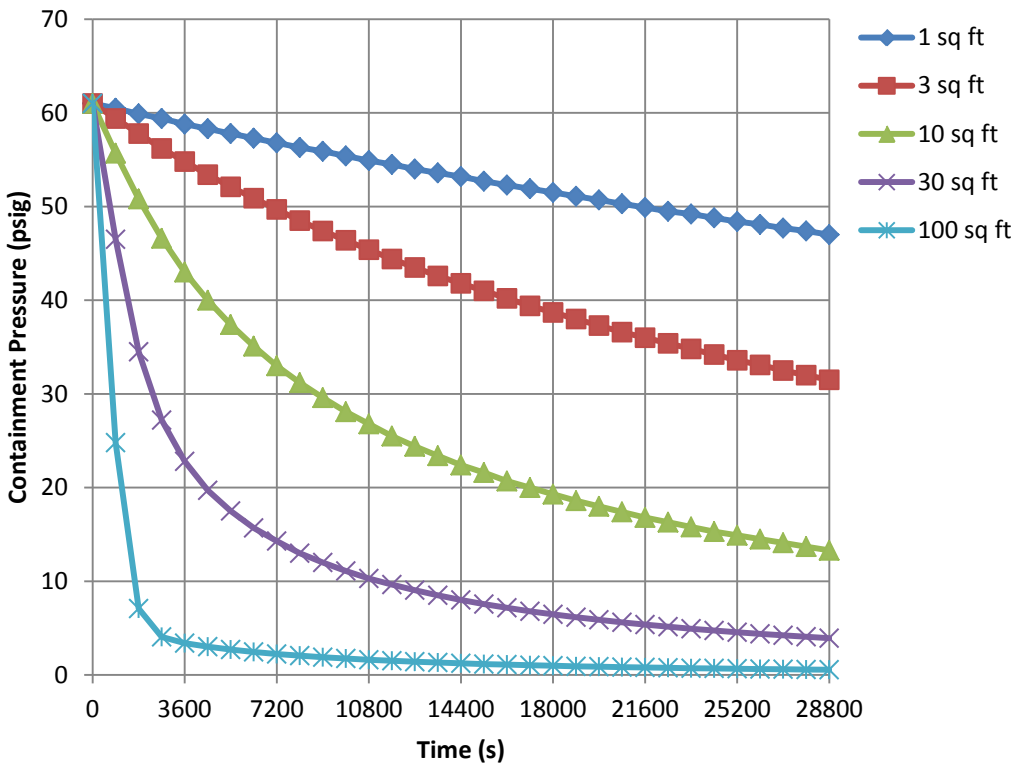


Figure 1-9 Reduction in PWR Containment Atmosphere Pressure (psia) with Time

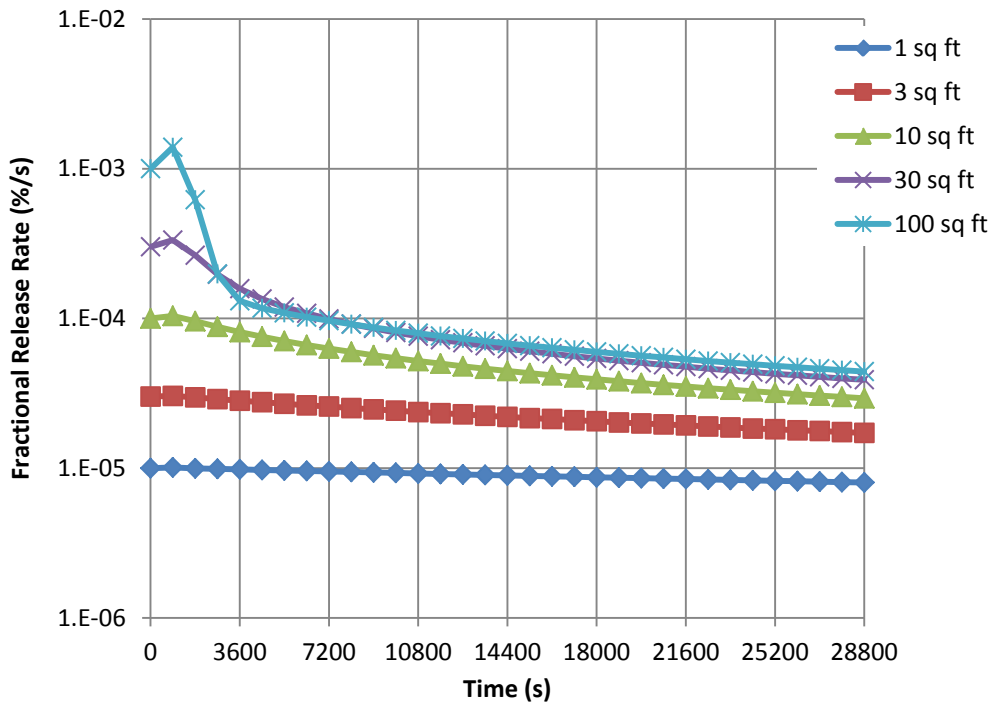


Figure 1-10 Change in Fractional Release Rate (%/s) from PWR Containments to the Atmosphere with Time

In reality, the pressure should decrease to atmospheric pressure. However, RASCAL assumes that the containment pressure will only decrease to a pressure that is 2 percent above atmospheric pressure. This 2 percent residual pressure differential is used to provide leakage to the environment through the open hole that is greater than the design containment leak rate. If containment integrity is reestablished, the leak rate may be reset to the design leak rate by setting the hole size to 1 in^2. Typical design leak rates for PWRs range from about 1.2×10^{-6} %/s to about 5.8×10^{-6} %/s. Specific containment design leak rates for reactors are provided in the RASCAL 4.3 Case Summary.

In BWRs the relationships among containment pressure, hole size, and time are similar to the relationships shown in Figure 1-9. GE BWR reactors with Mark I and Mark II containments typically have containment design pressures in the 45 to 62 psig range. BWR reactors with Mark III containments have design pressures of about 15 psig.

The relationships among containment fractional leak rate, hole size, and time for BWR are also similar to the PWR relationships. However, because of the smaller size of Mark I and Mark II containments, the initial fractional flow out of containment is larger for these containments than for most PWRs. The initial fractional flow out of containment for Mark III containments is about the same as for PWRs. The design containment leak rate for Mark I and II containments is typically about 5.8×10^{-6} %/s. For Mark III containments it ranges from about 4.1×10^{-6} %/s to about 7.5×10^{-6} %/s.

In calculating the release through a hole, RASCAL updates the containment pressure and leak rate every 15 min. This procedure causes an over estimation of the release rate early in the release period. If the release period is long enough, the over estimation of release rates will be followed by a period

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. Figure 1-11 shows the effect of the overestimation of early release rates on total release at the end of the first 15 min and the first 2 hr of the release period as a function of hole size. For a hole size approximately equal to a 3.5 ft-diameter pipe, the overestimation of release in both the first 15 min and first 2 hr is less than 5 percent. Similarly for a hole size equivalent to a 6.2 ft-diameter pipe the overestimation is less than 15%. These values are well within in the uncertainty in the underlying assumptions in the calculation and the uncertainty of the data likely to be available in the event of an accident creating the hole.

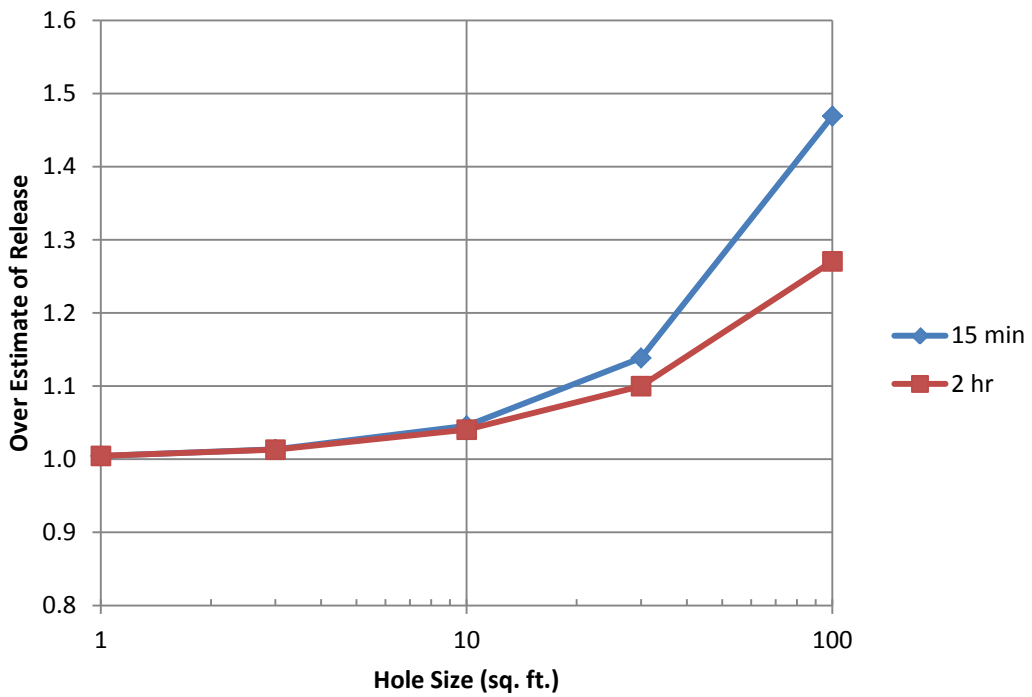


Figure 1-11 RASCAL 4.3 Overestimates of Early Releases using the Containment Pressure-Hole Size Option.

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.

1.5 Spent Fuel and Fuel Cycle Facility Accidents

RASCAL 4.3 features an increase in the modularization of the computer codes to enhance the maintainability of the codes. This increase includes the separation of the calculation of the spent fuel pool 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 have not been any significant changes the calculation of the spent fuel pool or fuel cycle facility source terms related to release paths.

There are two components to the RASCAL 4.3 spent fuel accident source term calculation. The first is the calculation of the nuclide inventory at risk, and the second is the calculation of the portion of the inventory at risk 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 nuclide inventory at risk.

1.5.1 Drained Spent Fuel Pool

The nuclide inventory at risk when the fuel in the spent fuel pool 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 inventory at risk starting with the fuel from the most recent refueling outage. If the spent fuel pool 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 nuclide inventory for each batch of fuel in the core offload. Otherwise, RASCAL only calculates the inventory for the oldest batch of fuel in the reactor at the last refueling. The nuclides inventories of fuel in the spent fuel pool 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 spent fuel pool from other reactors. This is likely to be the case when reactors share a common spent fuel pool. RASCAL 4.3 includes an option to add the inventory of nuclides in spent fuel from a second reactor to the spent fuel pool inventory at risk 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 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 nuclide inventory in a batch of fuel at the other reactor at the time of shut down will be the same as the inventory of full cycle fuel from the first reactor. RASCAL 4.3 calculates the at risk nuclide inventory of the youngest batch of other reactor fuel by decaying the shutdown inventory of the reactor for a time specified by the user. This time should be the time between the date of last exposure of fuel and the time that the pool is drained. The decayed inventory of the other

reactor is added to the at risk inventory in the pool. The inventory of the second youngest batch of spent fuel from the other reactor is then calculated from the inventory of the youngest batch by decaying youngest batch inventory for a period equal to the refueling interval. The decayed inventory is added to the at risk spent fuel pool inventory. The process is repeated until the at risk inventory in the spent fuel pool includes the contribution from each other reactor fuel batch.

1.5.2 Damaged Fuel

The at risk nuclide inventory in a damaged fuel accident is limited to the nuclide inventories in the damaged fuel assemblies. RASCAL 4.3 calculates the at risk nuclide inventory in the damaged fuel assemblies based on the burnup of the damaged fuel assemblies and the time since exposure. The burnup is estimated using the reactor power and the number of days that the fuel was exposed in the reactor. The default time since exposure is based on technical specification for each reactor type related to 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.

2 Changes to Transport, Dispersion, and Dose Calculations

This chapter describes the changes to the atmospheric transport and dispersion calculations and the dose calculations implemented in RASCAL 4.3.

2.1 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 Dai-ichi event, RASCAL 4.3 has added a fourth Cartesian computational grid which increases the RASCAL 4.3 domain from a 50 mi (80 km) radius to a 100 mi (160 km) radius. The 100-mi grid has 10-mi (16 km) spacing between computational nodes. The close-in polar grid associated with 100-mi Cartesian grid extends to 10 mi just as it does with the 50-mi Cartesian grid.

Figure 2-1 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 (090°, 180°, 270°, or 360°) with a 4 mi/hr wind speed and D 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 Value Table and in the Detailed Results for the Cartesian grid may be significantly lower than centerline doses. These points are illustrated in Figure 2-2.

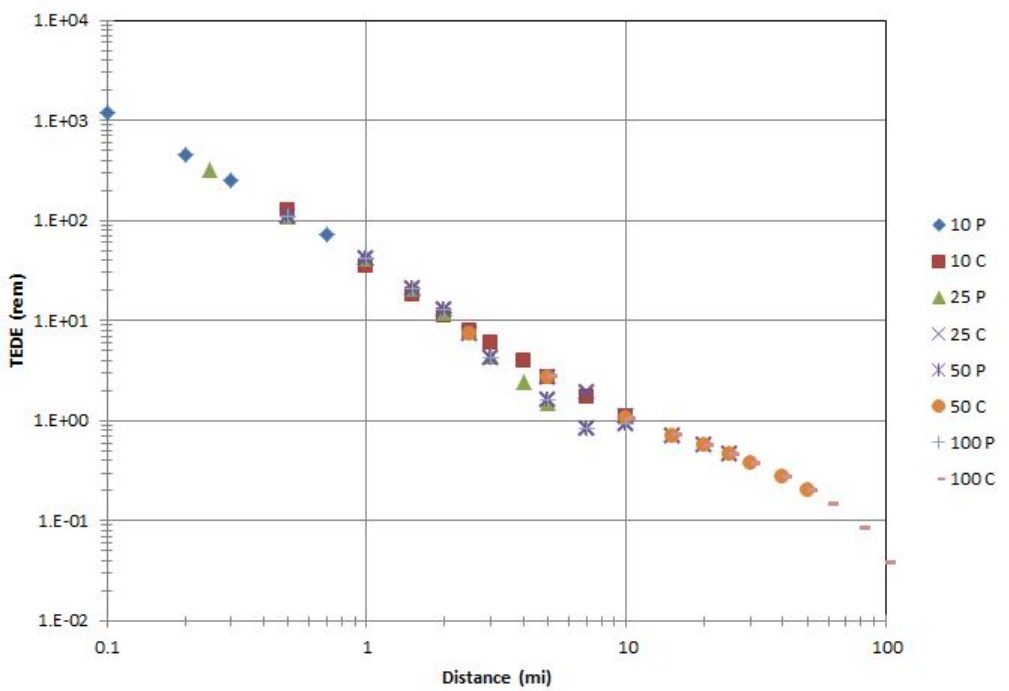


Figure 2-1 Dose Consistency Among the Eight RASCAL 4.3 Computational Grids

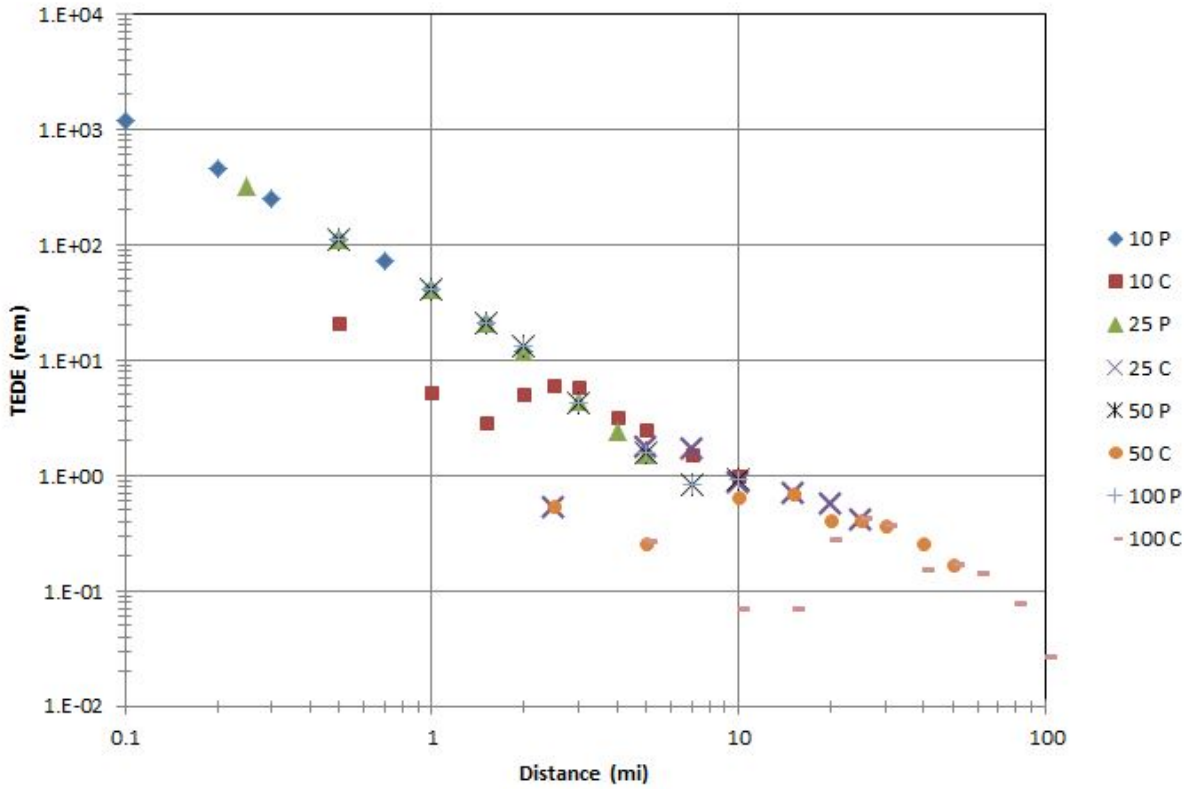


Figure 2-2 Maximum Doses as a Function of Distance for an 10° Off-Cardinal Wind Direction

The only difference in the RASCAL 4.3 input values for the calculations leading to Figures 2-1 and 2.2 is the wind direction. A cardinal wind direction was used for the calculations for Figure 2-1 and a wind direction 10° off a cardinal direction was used for the calculations for Figure 2-2. 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 Value Table. Note also that the doses are correct for the nodes but the nodes are not on the plume centerline.

Table 2-1 lists several Cartesian grid node locations for nodes at nominal distances of 1 and 10 miles on the 10 mile grid. With a wind direction of 270° (west wind), a plume centerline would pass directly over nodes at both nominal distances at 090° (east). With a wind direction of 280°, the plume centerline would pass between nodes at both distances. At the 1-mi distance, there is a 10° separation

Table 2-1 Cartesian Grid Node Positions at Nominal Distances 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

between the plume centerline (100°) and the closest computational node; at the 10-mi distance the separation is only 1.3°. 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.

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.

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 m. The topographic and surface roughness files are described more fully in Chapter 3.

2.2 Child Thyroid Dose Calculations

The Environmental Protection Agency (EPA) has released a draft PAG manual for interim use and public comment (EPA 2013). Section 2.3 of the manual deals with 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 iodines 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 manual also discusses Food and Drug Administration (FDA) recommendations for administration of KI to adults (FDA 2001; FDA 2002). The child thyroid doses are calculated when the ICRP-60 (FGR-13) dose factors are selected. The child thyroid dose is not calculated when FGR-11 (ICRP-26) dose factors are selected.

RASCAL 4.3 child thyroid dose conversion factors are based on the dose conversion factors for a 1-year old child in file FGR13INH.HDB included with FGR13PAK (Eckerman and Leggett 2011). For consistent treatment of iodine the child dose conversion factors in RASCAL 4.3 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. I_2), and 45 percent is associated with nonreactive gases (e.g. CH_3I).

The child thyroid doses calculated by RASCAL 4.3 in the STDose and FMDose 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 nuclides.

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 conversion factors. Table 2-2 lists the light-activity breathing rates and the composite thyroid dose conversion factors for the iodine isotopes that are typically associated with reactor accidents.

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

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

The values in Table 2-2 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 years). Figure 2-3, which is based on the data in Table 2-2, shows appropriate scaling factors. According to FDA, there is generally little value in use of KI for adults over the age of 40.

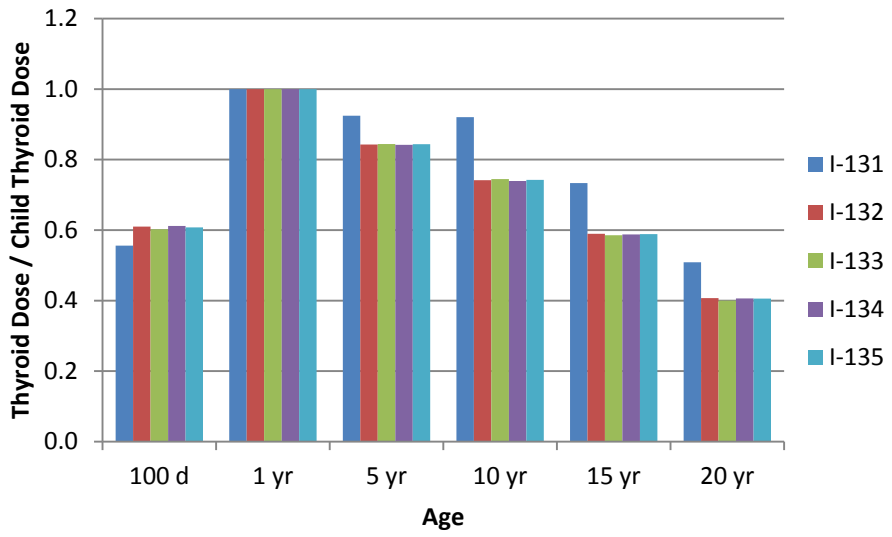


Figure 2-3 Scaling Factors (Thyroid Dose/Child Thyroid Dose) for Adjusting RASCAL 4.3 1-Yr Child Thyroid Doses to Doses for Other Ages

2.3 Ingestion Dose Factors

Ingestion dose conversion factors have been added to the radionuclide data base in RASCAL 4.3. The data base includes ingestion dose conversion factors from both Federal Guidance Report No. 11 (Eckerman, et al. 1988) and Federal Guidance Report No.13 (Eckerman and Leggett 2011). The FGR-13 ingestion dose conversion factors are for adults. These dose conversion factors 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 Chapter 6.

2.4 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 tool that 1) estimates external and total doses for field teams prior to exposure, and 2) provides 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. Select the time of exposure and display the results. To view the dose and conversion factor estimates, place the cursor over the field team location and left click to display the dose rate vs time plot for the location. The doses and conversion factors are found in the table on the right side of the screen.

2.4.1 Projected DRD Reading and Doses

The estimated 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 min exposure period. The calculation of these exposure rates is discussed in RASCAL Technical Document Section 4.9.4.

The estimated total external and internal dose at the time and location selected for field team that has not taken KI is the sum of the groundshine, cloudshine, and inhalation CEDE assuming a 15 min exposure period. Calculation of these dose rates are described in RASCAL Technical Document Sections 4.9.1, 4.9.2 and 4.9.3.

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

$$\Delta D_{KI} = 0.9 \times 0.03 \times D_{thy} \qquad (2\text{-}1)$$

where ΔD_{KI} is the dose reduction due to administration of KI (mrem)

0.9 is the effectiveness of KI in reducing the thyroid dose

0.03 is the thyroid weighting factor in calculation of the inhalation CEDE

D_{thy} is the thyroid dose (mrem).

Calculation of the thyroid dose is discussed in RASCAL Technical Document Section 4.9.1. The total dose estimate for field team that have taken KI is then calculated by subtracting the KI dose reduction from the dose estimate for field teams that not taken KI.

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

2.4.2 Post Exposure Dose Estimates

The projected DRD reading 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 estimates conversion factors that may be used to estimate total dose commitment (external dose + internal dose) from DRD readings. These conversion factors are found at the bottom of the table that presents the projected DRD reading and doses described above.

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

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

where CF is the conversion factor for field teams that have not taken KI (mrem/mR)

D_{gs} is the projected groundshine dose (mrem)

D_{cs} is the projected cloudshine dose (mrem)

D_{inh} is the projected inhalation commited effective dose equivalent (mrem)

DRD_p is the projected DRD reading (mR).

Given the actual DRD reading and the conversion factor, the total dose may be estimated by the user as

$$D = CF \times DRD_a \tag{2-3}$$

where D is the estimated external + internal dose (mrem)

DRD_a is 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 conversion factor. In this case RASCAL calculates the conversion factor as

$$CF_{KI} = \frac{D_{gs} + D_{cs} + D_{inh} - \Delta D_{KI}}{DRD_p} \tag{2-4}$$

where CF_{KI} is the conversion factor for field teams that have taken KI (mrem/mR)

ΔD_{KI} is the dose reduction associated with KI (mrem).

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

$$D_{KI} = CF_{KI} \times DRD_a \tag{2-5}$$

where D_{KI} is the 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 conversion factor is based on the radionuclide mix assumed in calculating the projected DRD reading and doses.

Table 2-3 shows an example of the variation in the conversion factors 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 conversion factors seen in the table are the result of variation in source term radionuclide mix with time, the transit time from the release point to

Table 2-3 Conversion Factor Variation with Time and Distance from the Release Point

Time after LOCA (hr)	Distance (mi)	Conversion Factor Without KI	Conversion Factor 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

the receptor, and the changes in radionuclide mix in the plume caused by decay and deposition between the release point ant 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 min. If the exposure time significantly exceeds 15 min, consideration should be given to calculating conversion factors for successive 15 min 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.

3 Meteorological Data Acquisition and Processing

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

3.1 Meteorological Data

Observation 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) (NWS 2013). Since MetFetch downloads observations and forecasts from remote servers, an internet connection is required to use the utility.

Figure 3-1 shows the MetFetch utility interface. Users can select stations by entering either a site latitude/longitude location or selecting a site location and a search radius. To the right of the site latitude/longitude boxes is a choice list for Station Types. The options in this list are used to filter the list of available meteorological stations. The “Preferred Station for RASCAL” option will limit the stations shown in the “Stations Available for Download” box to a subset of the available stations that has been selected by the RASCAL developers. The “National Weather Service (NWS) Stations Only” option limits the stations shown in the “Stations Available for Download” box to NWS Stations, and the “All Available Stations” option presents the full list. Note that for many sites, the “All Available Stations” list is far larger than is needed or useful.

Upon clicking the “Find Stations” button, a list of available stations appears in the “Stations Available for Download” box. The stations are sorted by direction from the site, starting from north and progressing clockwise around the compass. Within each direction, the stations are sorted in order of increasing distance from the site. Users should select a few meteorological stations from the list of available stations that are well distributed around the site to download (10 to 20 is adequate for most sites). Then, the user should click the “Download” button to download the observations and forecasts files; these files can then be imported for use in a RASCAL run.

Observations are generally available for a given station from the present date/time back to the previous calendar month. Forecasts are available from the present date/time out to one week. Additional information about the MetFetch utility is available in the Chapter 6 of the RASCAL 4.3 User’s Guide.

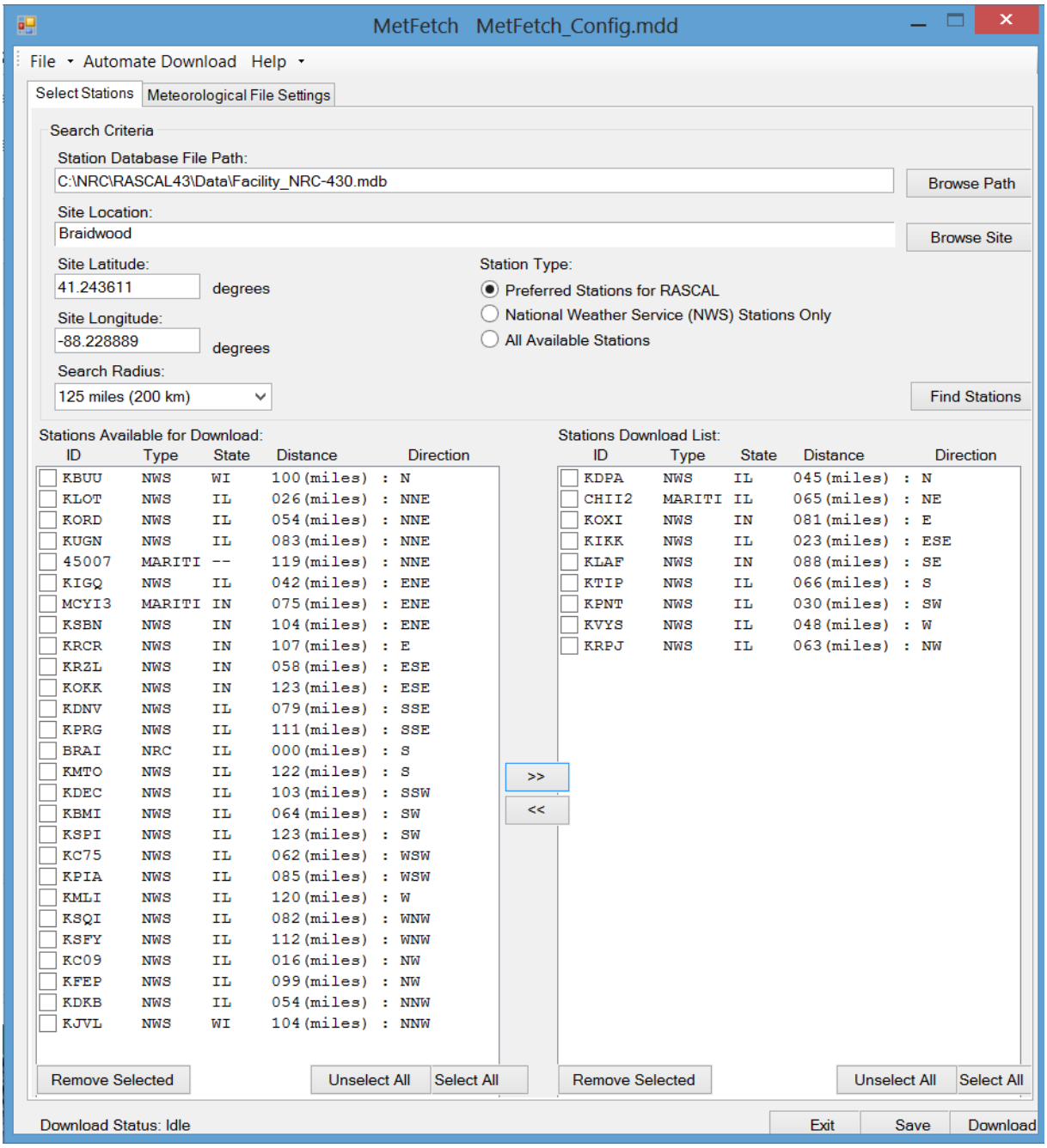


Figure 3-1 MetFetch User Interface Used for Downloading Meteorological Observations and Forecasts for Use in a RASCAL Run

3.2 Topographic Data

RASCAL gridded terrain files (*.top) have been derived for each site using elevation data from a Global Digital Elevation Model (GTOPO30) file (USGS 1996). 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 miles, 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 by developed by the U.S. Geological Survey's (USGS) Earth Resources Observation and Science (EROS) Data Center in 1996 using a variety of data sources. Figure 3-2 is a raster image of the GTOPO30 file used to derive site terrain files for RASCAL. Darker shades of grey represent lower terrain, lighter shades grey represent higher terrain. Oceans are colored white.

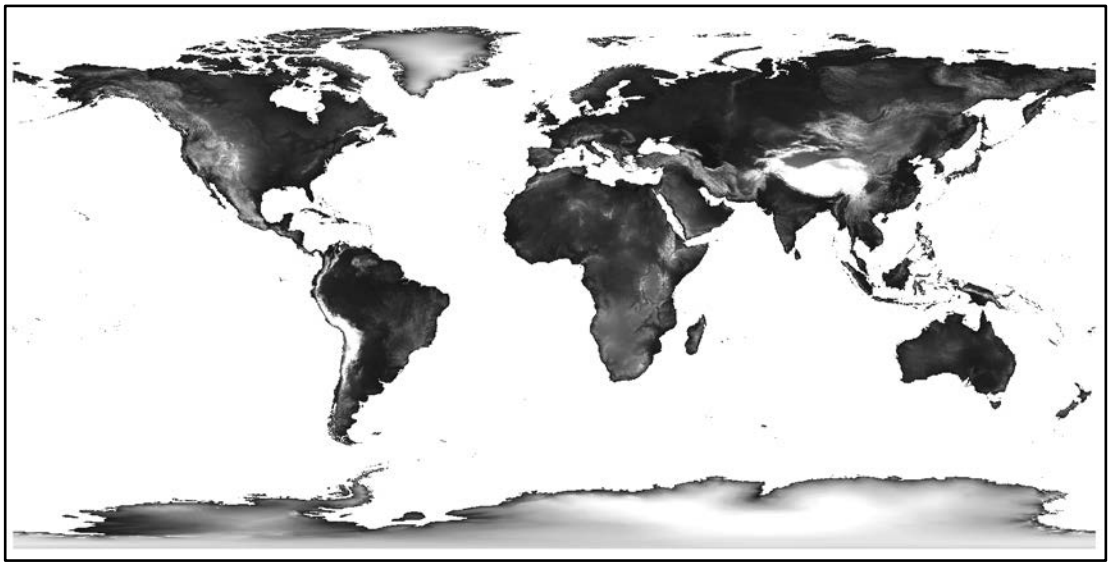


Figure 3-2 Raster Image of the GTOPO30 File Used to Derive Site Terrain Elevation Files (*.gz0) for RASCAL

3.3 Surface Roughness Data

RASCAL gridded surface roughness files (*.gz0) have been derived for each site using land use/land cover data from the National Land Cover Dataset 1992 (NLCD1992) file (Vogelmann et. al 2001). A RASCAL terrain *.gz0 file contains four surface roughness grids with 22 points in the X and Y directions, at grid spacings of 10.0, 5.0, 2.5, and 1.0 miles, respectively.

The NLCD1992 file utilizes a 21-class land cover classification scheme that has been applied to the continental U.S. at a spatial resolution of 30 meters. NLCD92 is based primarily on the classification

of Landsat Thematic Mapper (TM) 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 U.S. Environmental Protection Agency (EPA) AERSURFACE User's Guide (EPA 2013) is then used to convert the 21 land-cover codes to corresponding surface roughness values. Figure 3-3 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.

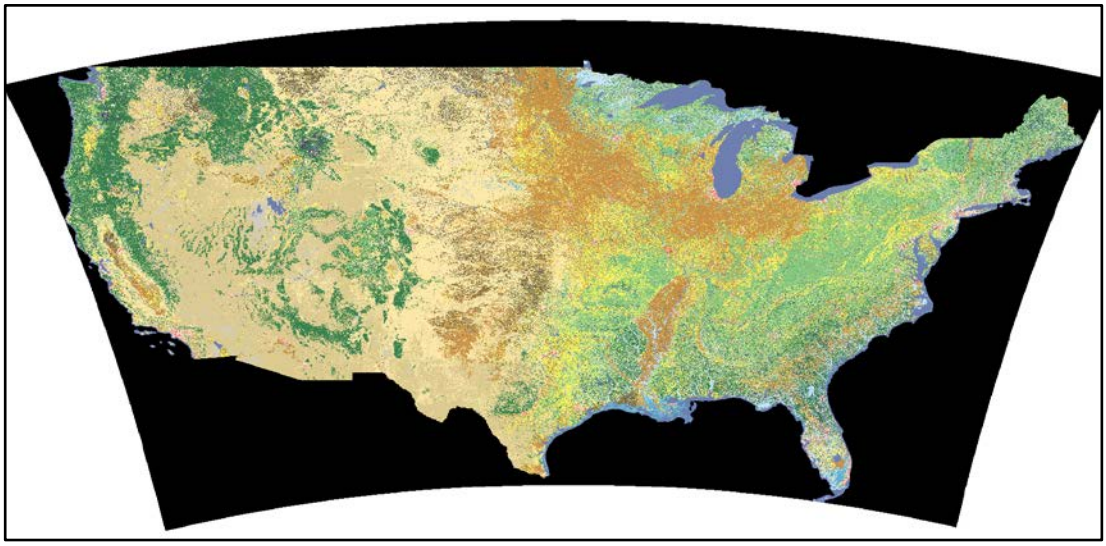


Figure 3-3 Raster Image of the NLCD1992 File Used to Derive Site Surface Roughness Files (*.gz0) for RASCAL

4 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 MWd/MTU. Similarly, earlier versions of RASCAL used tabled values reactor coolant nuclide concentrations which were scaled up and down based on reactor power.

RASCAL 4.3 uses these same assumptions for its default reactor nuclide inventories. However RASCAL 4.3 also has an option to estimate reactor core and coolant nuclide inventories at the time of an accident using reactor specific information on reactor operations prior to the accident. This option is implemented in a utility program that is bundled with RASCAL 4.3 and may be called from the RASCAL 4.3 user interface. This chapter describes the updated inventory models and utility code implementing them. Sections 4.1 through 4.4 deal with the reactor core inventory; Section 4.5 deals with the reactor coolant inventory. Section 4.6 addresses the input to and output from the code, Rx_Inventory_S, that implements the updated inventory model.

4.1 Reactor Core Activity Model

Production rate P'_i of a nuclide *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 nuclides. Production of the actinides increases with burnup, so we assume a production rate that is an average for peak rod burnup of 42 GWd/MTU and 62 GWd/MTU.

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

$$\frac{dA_i(t)}{dt} = P_i' - \lambda_i A_{i(t)} \tag{4-1}$$

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

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

$$A_i\,(t) = \frac{P_i'}{\lambda_i} \times [1.0 - \exp(-\lambda_i t)] \tag{4-2}$$

Routine reactor operation involves alternating periods when the reactor is at power and when it is shut down for refueling or maintenance. While the reactor is shutdown, the activity in core decreases due to radioactive decay. During refueling a portion of the exposed fuel, typically 1/3 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 shut down. As a result, the activity of each nuclide at restart cannot be calculated directly from Equation (4-2). The activity remaining at the time of restart is calculated as

$$A_i(t_{rs}) = A_i(t_{sd})\exp[-\lambda_i(t_{rs} - t_{sd})] \tag{4-3}$$

where t_{rs} = the time of restart
t_{sd} = the time of shut down.

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

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

Finally, the activity during the second and subsequent fuel cycles is calculated using Equation (4-2) replacing t, the time since the start of initial exposure, with

$$t_i = t_{effi} + (t - t_{rs}) \tag{4-5}$$

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

Figure 4-1 shows the activity for selected nuclides in a fuel batch as a function of time from initial exposure to final discharge for a typical reactor with a 540 day fuel cycle assuming a 30 day refueling outage during which one third of the fuel in the core is replaced. Table 4-1 gives the production and decay parameters for the selected nuclides. 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 core.

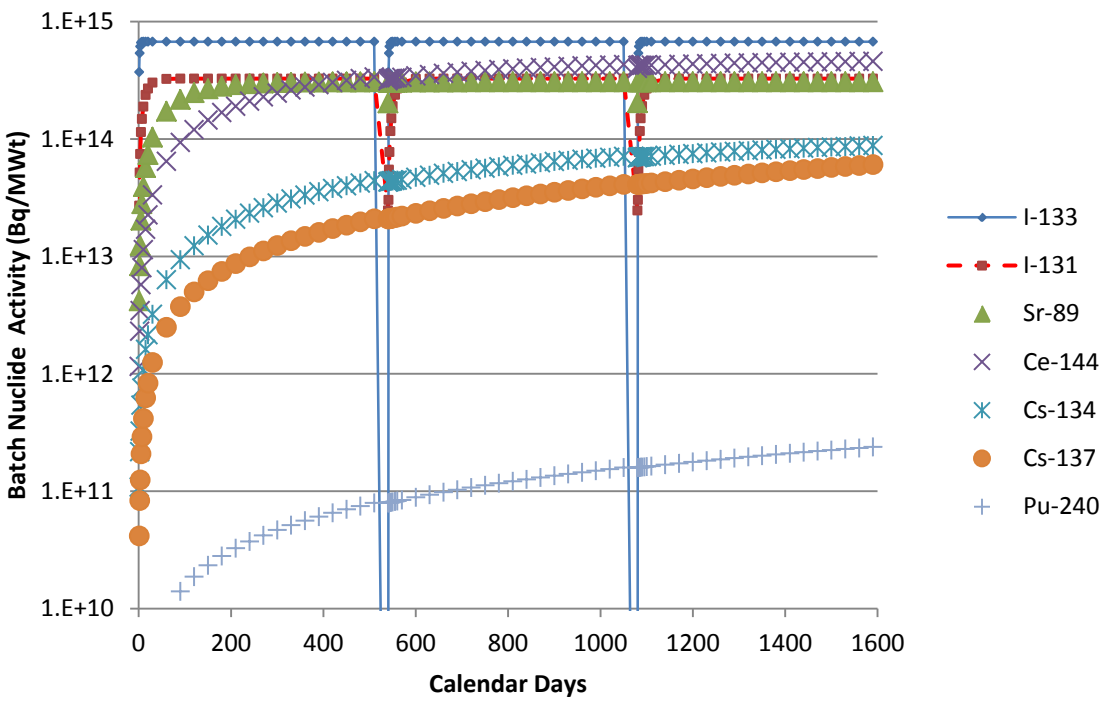


Figure 4-1 Nuclide Activity as a Function of a Fuel Batch Exposure Time

Table 4-1 Production and Decay Parameters for Selected Nuclides

Nuclide	I-133	I-131	Sr-89	Ce-144	Cs-134	Cs-137	Pu-240
Half Life (d)	8.67E-01	8.04E+00	5.05E+01	2.84E+02	7.53E+02	1.10E+04	2.39E+06
Decay Const (1/d)	7.99E-01	8.62E-02	1.37E-02	2.44E-03	9.21E-04	6.30E-05	2.90E-07
Production (Bq/MWt)/d	1.617E+15	8.458E+13	1.263E+13	3.464E+12	3.254E+11	1.249E+11	4.670E+08

The I-131 and I-133 activities in Figure 4-1 clearly show the refueling outages. These nuclides also show that the activity level for short-lived nuclides is reasonably constant through-out 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 nuclides such as Cs-137 and Pu-240 continue to increase throughout the fuel life. The activity does not decrease significantly during refueling outages and does not reach equilibrium. Assuming 3-cycle operation for a typical reactor, Figure 4-2 shows the variation in core activity during a cycle for a mature reactor core (core with fuel in 1st, 2nd, and 3rd cycles). For short-lived nuclides, the activity increases from zero to equilibrium, while for long-lived nuclides, the activity level increases by a factor of 2 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.

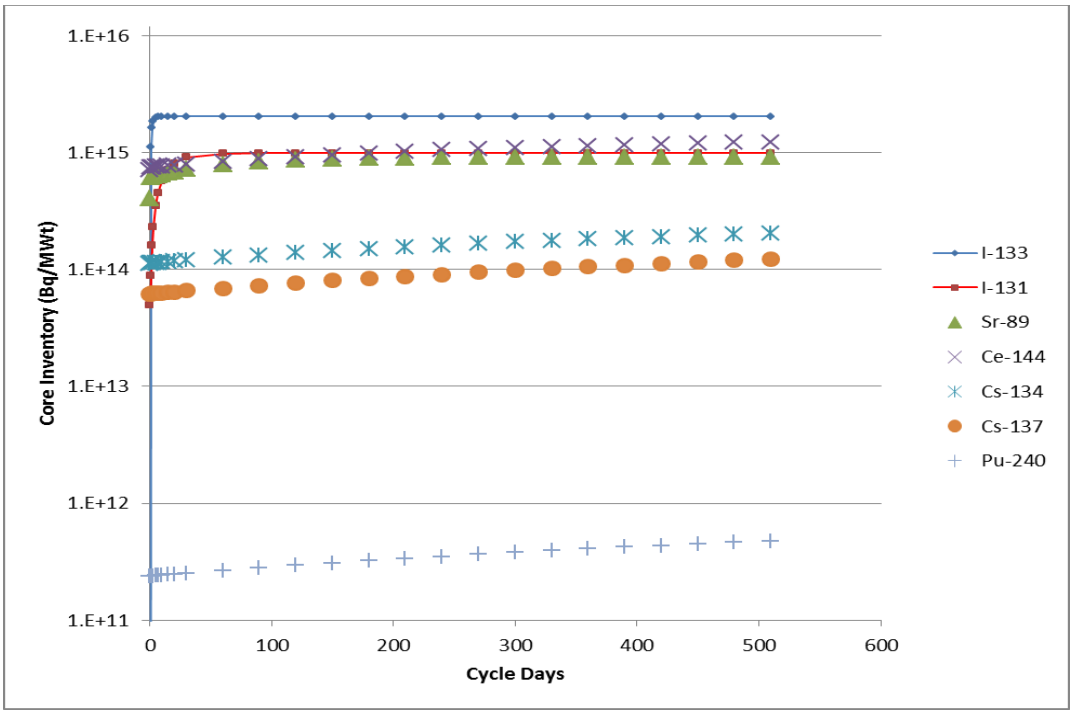


Figure 4-2 Core Activity as a Function of Time During a Cycle

4.2 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 nuclide 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 nuclide specific parameters are the half-life and production rate of each nuclide. NUREG/CR-6703 (Ramsdell et al 2003) lists the most important nuclides in terms of dose. The half-lives of these nuclides are listed in DCFPAK 2 (Eckerman and Leggett 2002). Appendix E of NUREG/CR-6703 lists core-average activity for important nuclides for PWRs and BWRs for peak rod burnups ranging from 22 GWd/MTU to 75 GWd/MTU based on a series of ORIGEN_ARP calculations. The nuclide 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 calculations. Table 4-2 lists the nuclide specific parameter values for the updated model.

Table 4-2 Nuclide Specific Core Inventory Parameters for the Updated Inventory Model

			RASCAL 4.3 Model Production	
	Half-life	**Decay Const**	**BWR**	**PWR**
Nuclide	**(d)**	**(1/d)**	**(Bq/MWt)/d**	**(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

			RASCAL 4.3 Model Production	
	Half-life	**Decay Const**	**BWR**	**PWR**
Nuclide	**(d)**	**(1/d)**	**(Bq/MWt)/d**	**(Bq/MWt)/d**
Te-129m	3.36E+01	2.06E-02	1.253E+12	1.276E+12
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

			RASCAL 4.3 Model Production	
	Half-life	**Decay Const**	**BWR**	**PWR**
Nuclide	**(d)**	**(1/d)**	**(Bq/MWt)/d**	**(Bq/MWt)/d**
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

4.3 Comparison with RASCAL 4.2

RASCAL 4.2 core inventory estimates are based on the result of an ORIGEN calculation made in December, 2003. Full details of the ORIGEN calculations are not available. However, notes accompanying the inventory data indicate that the calculations were made for reactor power of 3479 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 results, the fuel characteristics listed above are consistent with a late generation Westinghouse PWR (possibly Sequoyah 1 or 2).

Table 1.1 of the RASCAL 4 Technical Document (Ramsdell et al. 2012) lists the core inventory assumed in RASCAL 4.2 normalized to a reactor power of 38,585 MWt. The nuclide 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 nuclides having half-lives longer than 1 year. The activities of the long-lived nuclides are scaled up by the ratio of the core-average fuel burnup (MWd/MTU) to 38,585 MWd/MTU.

Figure 4-3 shows the ratio of the normalized RASCAL 4.3 core activity to normalized RASCAL 4.2 core activity as a function of nuclide half-life for the nuclides common to both models. Ratios larger than 1.0 indicate that the RASCAL 4.3 model predicts larger values the inventory compared to RASCAL 4.2. For most nuclides, the normalized activity calculated with the RASCAL 4.3 model is nearly the same as the activity in RASCAL 4.2. Nuclides with significantly increased activity are identified in the figure. The RASCAL 4.3 model includes eight nuclides not included in RASCAL 4.2. Those nuclides are Co-58, Co-60, Pu-238, Pu-239, Pu-240, Am-241, Am-242, and Cm-244. There are also four nuclides that were included in the RASCAL 4.2 inventory that are not included in the RASCAL 4.3 model inventory. Those nuclides are Kr-83m, Sb-129, Xe-133m, and Xe-138. A fifth nuclide, 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.

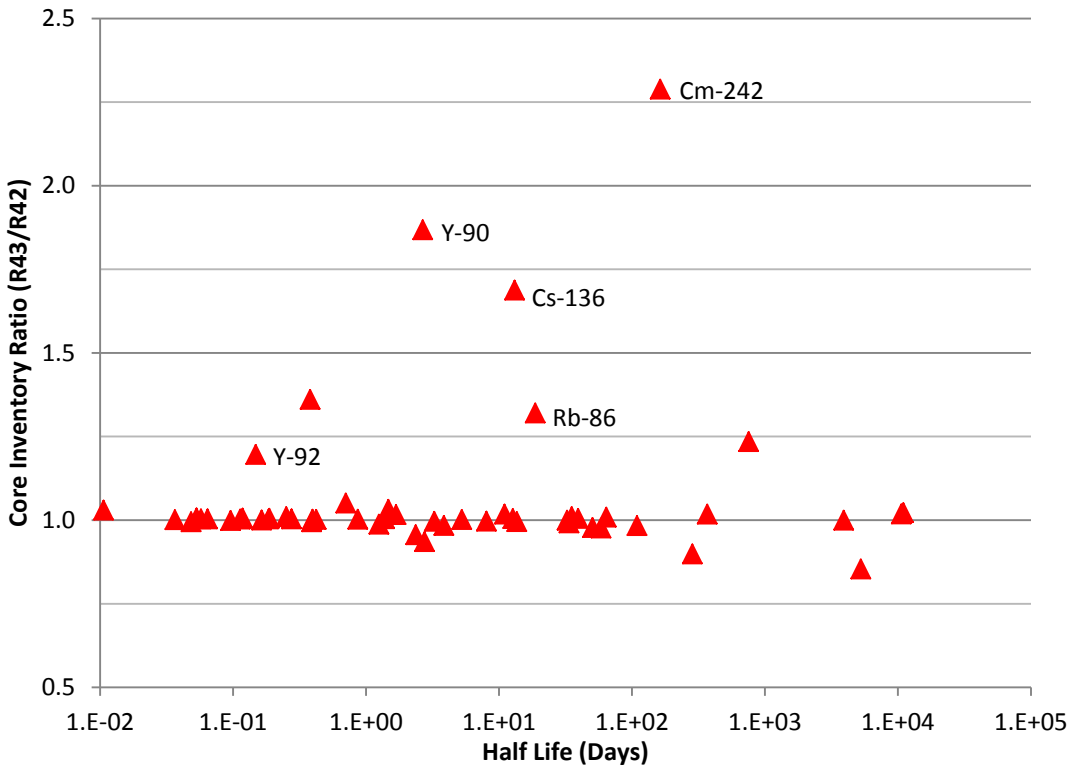


Figure 4-3 Comparison of Updated Model Nuclide Inventory with the RASCAL 4.2 Inventory as a Function of Nuclide Half-life

The RASCAL 4.3 custom fuel inventory utility provides a convenient method of obtaining the core inventory for specific time in the fuel cycle. The inventory is calculated from the dates of last startup and the time in question and the duration of the last outage. Figure 4-4 shows the variation in activity from early cycle (1 month after startup) to the end of cycle (18 months after startup) as a function nuclide half-life. Time in cycle is important for nuclides with half-lives of about 50 and longer. For some important nuclides (e.g. Cs-134 and 137), the activity in core can increase by almost a factor of 2 during the cycle. The two nuclides that show significantly larger increases than expected given their half-lives are Pr-144 (half-life = 0.012 days) and Te-125m (half-life = 58 days). These nuclides are short-lived daughters of longer lived parents (Ce-144 and Sb-125).

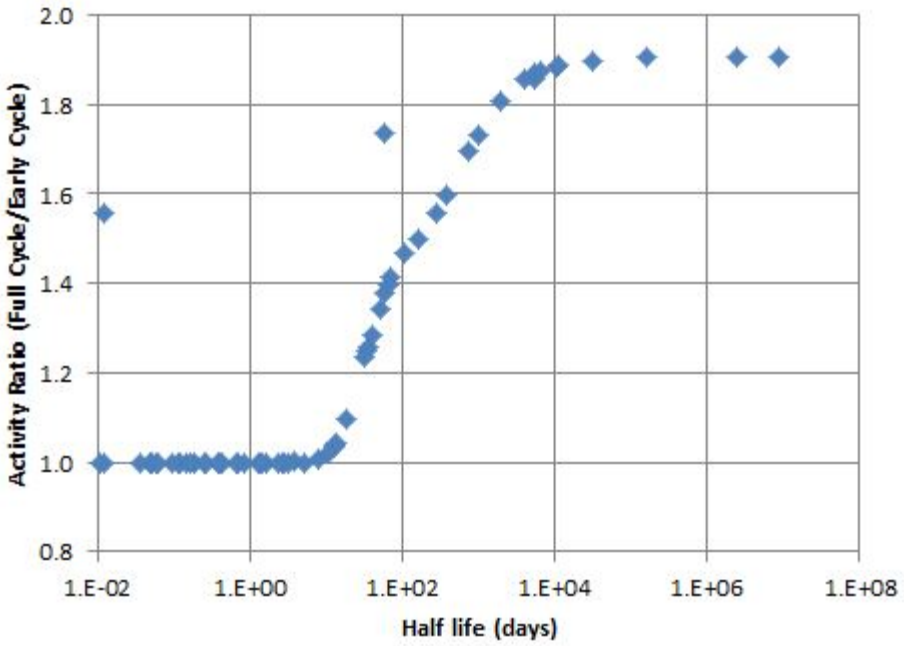


Figure 4-4 Increase in Core Inventory During the Fuel Cycle as a Function of Nuclide Half-life

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 nuclides for burn-up, they can approximate early cycle inventories using the average burn-up value for the fuel in the reactor in user interface screen used to select the plant location. The average burn-up at the beginning of a cycle is approximately one-half of the burn-up at the end of cycle (for 3 cycle fuel management). For four-cycle fuel management, the burn-up at the beginning of cycle is about 60 percent of the end of cycle burn-up. Figure 4-5 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 burn-up. Comparing Figures 4.4 and 4.5, it is clear that the approximate method based on adjustment of the default end-of-cycle inventory gives reasonable results for most nuclide. However, it does not properly account for the increase in Pr-144 and Te-125m. The choice between using a custom inventory or a burn-up adjusted end-of-cycle inventory may reasonable be made based on the availability of information at the time of an event, refueling dates, etc. or average core burn-up.

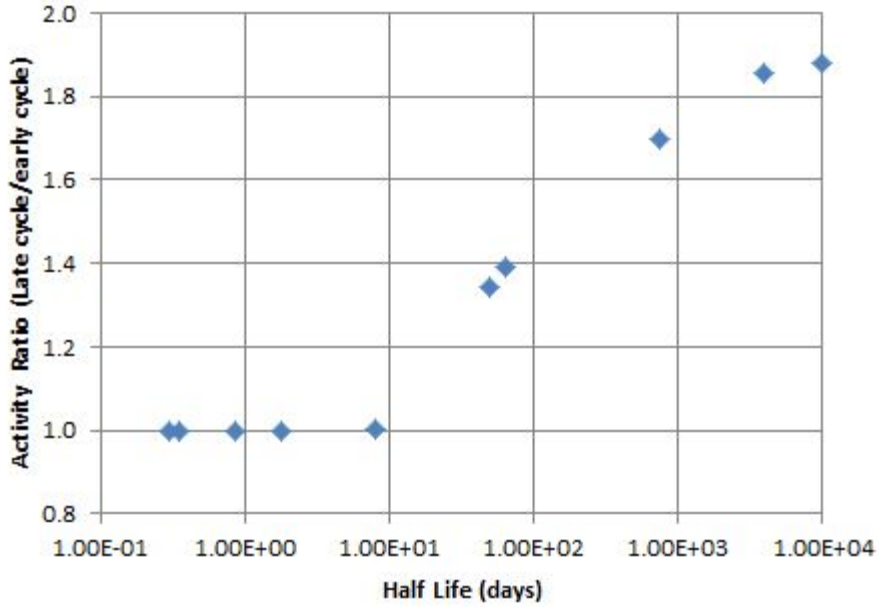


Figure 4-5 Nuclide Activity Increase During the Fuel Cycle as a Function of Half-life Estimated from End-of-cycle Inventory and Burnup

4.4 Model Validation

The updated core inventory model has been used to estimate core inventories for five current generation reactors types and four new reactors. 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 4-7 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, when available. When the data were not available, data were obtained from the reactor characteristics in NUREG/CR-6703. Except for the EPR, the number of cycles and cycle exposure 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 nuclide from the set of burnups.

Figures 4-6 and 4-7 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 4-8. 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 reactors shown in Figure 4-7 indicate that the model also provides reasonable core inventory estimates for new reactors.

Table 4-3 Reactor specific characteristics used to test the updated core inventory model

Reactor Type	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% of the ratios would fall in the range between 1/gsd and gsd, and about 95.5% of the ratios would fall in the range between 1/(2×gsd) and 2×gsd.

Table 4-4 and Table 4-5 present model statistics for each reactor type and average statistics for the 9 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% of the ratios would be expected to fall within the range 0.8 to 1.3 and about 95% of the values would be in the range of 0.4 to 2.6. Table 4-9 presents the ratios for 6 nuclides 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 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.

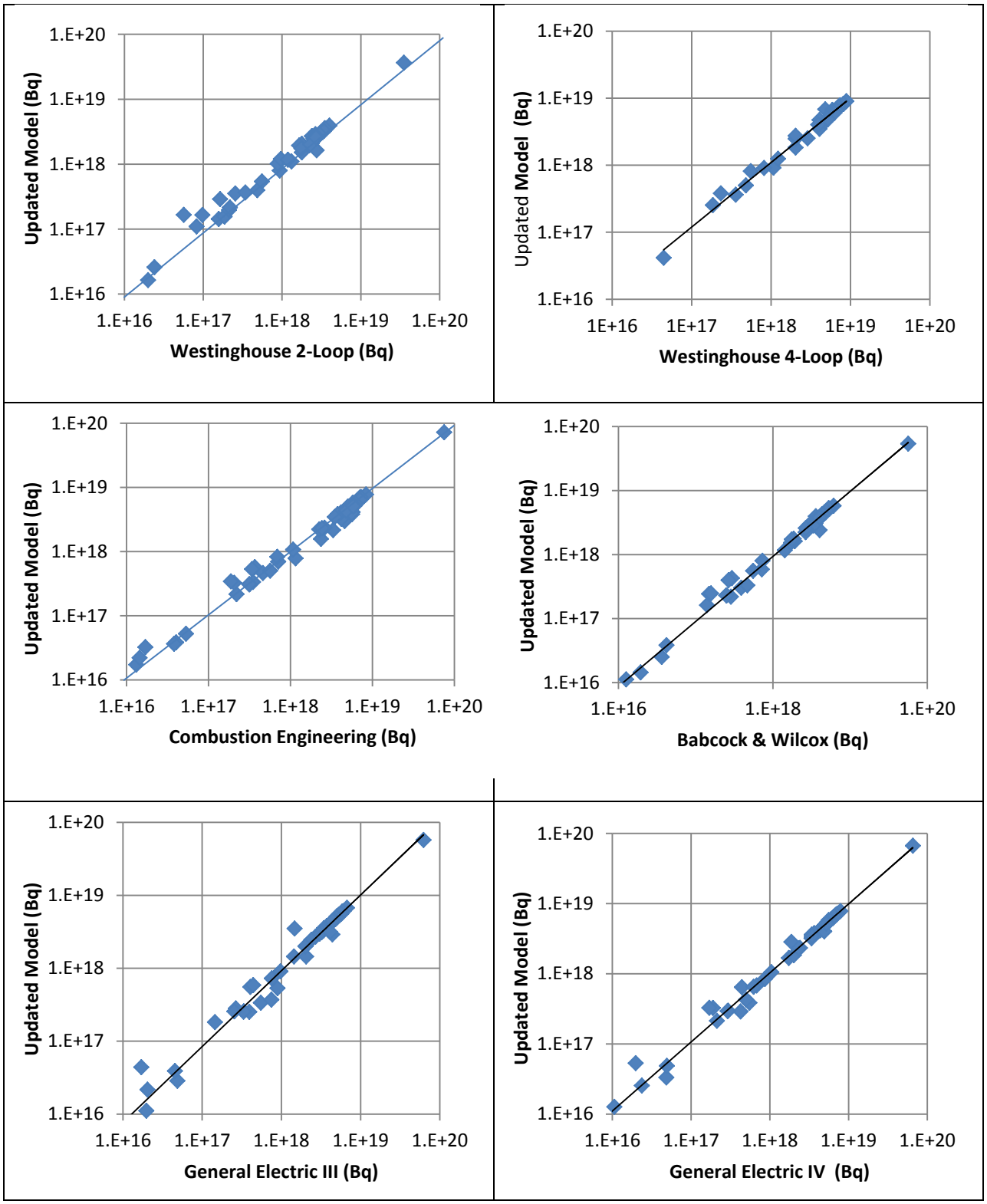


Figure 4-6 Comparison of the Updated Core Inventory Estimates with Core Inventory Estimates for Current Generation Reactors

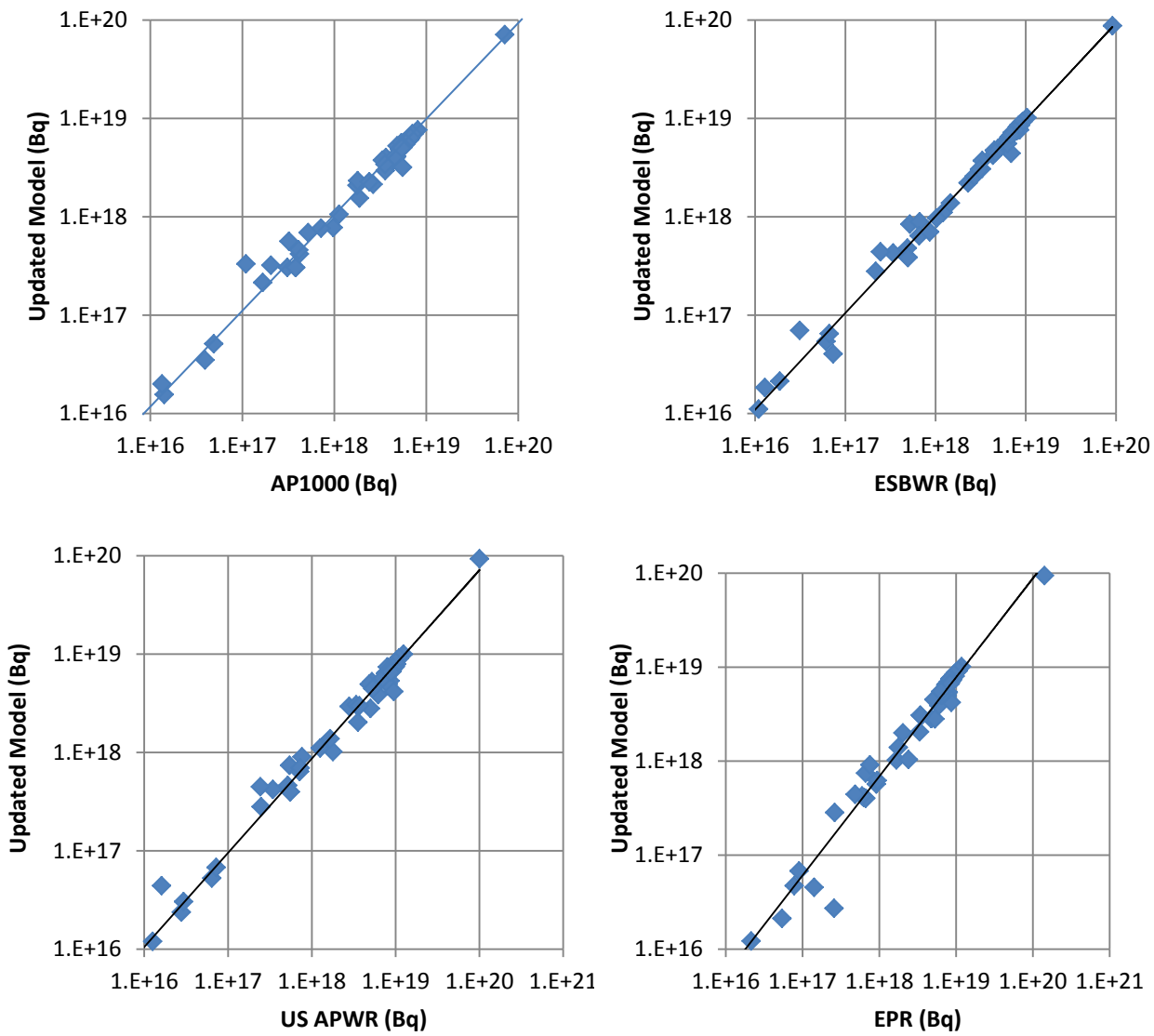


Figure 4-7 Comparison of Updated Core Inventory Estimates with Core Inventory Estimates for New Reactors

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

Reactor Type	n	median	Geo. Mean	GSD
W 2-loop	57	0.990	1.072	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
St. Dev.		0.073	0.108	0.088

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

4.5 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

$$\frac{dC_i(t)}{dt} = \frac{S_i'(t)}{M} - (\lambda_{i+} + R_i) \times C_i(t) \tag{4-6}$$

where $C_i(t)$ = the coolant concentration of nuclide i (Bq/kg) at time t
$S'_i(t)$ = the appearance rate of nuclide i in (Bq/hr) in the coolant
M = the total mass of coolant (kg)

λ_i = the decay constant (1/hr) for nuclide i

R_i = the combined removal rate (1/hr) for nuclide i for all of the processes other than decay that remove nuclides 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 nuclides at reactor startup, the core inventory of a nuclide tends to be nearly constant for periods of a few days. The removal rate for nuclides 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 nuclide 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

$$C_i(t) = \frac{S_i'(t)}{[M \times (\lambda_i + R_i)]} \tag{4-7}$$

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 (NRC 1979), NUREG-0017 (NRC 1985), and ANSI/ANS-18.1-1999 (ANS 1999). 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 nuclide 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 coolant inventory model assumes

$$S_i'(t) = A_i(t) \times Es_i \tag{4-8}$$

where $A_i(t)$ is the core inventory of nuclide i (Bq), and Es_i is an 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 nuclides, 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 4-9

$$S_i'(t) = A_i(t) \times Es_i + A_{cj}(t) \times \lambda_j \times BF_j \tag{4-9}$$

where A_{cj} is the activity of the parent in the coolant, λ_j is the decay constant for the parent, and BF_j is the 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 Cr-58, Cr-60, Zr-95 and Zr-97 and a number of other activation products. The only activation products in the RCS other than the isotopes of Cr and Zr that might contribute significantly to radiological doses are Na-24, Mn-54, Mn-56, Cu-64 and Ag-110m.

Appearance rates for these nuclides 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 the steam path through the turbine. The reactor cleanup system removal is characterized by the water flow through the cleanup system (FA) and the fraction of nuclides removed by cleanup system demineralizer (NA). (The variable symbols are the same as those used in ANSI/ANS-18.1.) Steam path removal is characterized by the nuclides carryover from water to steam(NS), the steam flow (FS), the fraction of steam flow passing through the condenser and condensate demineralizer (NC), and the fraction of nuclides removed in the demineralizer (NB). The values of NA, NS and NB are all dependent on nuclide. Thus, the removal rate for most nuclides for a BWR reactor is given by

$$R_i = \frac{(FA \times NA_i + FS \times NS_i \times NC_i \times NB_i)}{WP} \tag{4-10}$$

where WP is the total mass of water and steam in the coolant system.

For the 3400 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 1.6E+01 kg/s, the steam flow is 1.9E+03 kg/s, 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 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 4-9 defines the appearance rate on the primary side. A similar equation is used to define the appearance rate on the secondary side. It is

$$s_{i(t)} = C_i(t) \times L \times C_j(t) \times \lambda_j \times BF_j \tag{4-11}$$

where $s_i(t)$ = secondary side appearance rate (Bq/hr)
$C_i(t)$ = concentration in the primary coolant (Bq/g)
L = primary to secondary leak rate (g/hr)
$C_j(t)$ = concentration of the parent in the secondary coolant (Bq/g)
λ_j = the decay constant of the parent (1/hr)
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+03 g/hr (75 lbs/day). This value is used to estimate realistic secondary side inventories in the AP1000, EPR and 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 g/hr 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

$$R = \frac{[FB + (FD - FB) \times Y]}{WP} \tag{4-12}$$

where R = removal rate (1/hr)
FB = flow to the boron recovery system (kg/hr)
FD = flow through the purification demineralizer (kg/hr)
Y = the fraction of noble gases removed by degassing
WP = the mass of the primary coolant(kg)

The removal rate is for other nuclides, except H_3, is

$$R = \frac{[(FD \times NB_i) + (1 - NB_i) \times FA \times NA_i + (1 - NB_i) \times FB]}{WP} \tag{4-13}$$

where NB_i = fraction of activity of nuclide i removed in the purification demineralizer
FA = flow through the cation demineralizer, kg/hr
NA_i = fraction of activity of nuclide 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

$$C_i(t) = \frac{s_i(\mathrm{t})}{FS} \tag{4-14}$$

where FS is the steam flow (g/hr).

Equation (7) is used to calculate the secondary side concentrations of H_3 and other nuclides. For these nuclides, the secondary side removal rate is calculated as

$$R_i = \frac{[(FBD \times NBD) + (NS_i \times FS \times NC \times NX_i)]}{WS} \tag{4-15}$$

where FBD = blowdown flow (kg/hr)
NBD = fraction of blowdown removed from secondary side
NS_i = nuclide carryover from condensate to steam
FS = steam flow (kg/hr)
NC = fraction of steam flow passing through condensate demineralizer
NX_i = fraction of nuclide 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 nuclides in the core, have units of fraction of the core activity have units of fraction of a core activity per hour. They are included in data statements in the Rx_Inventory_S code.

4.6 Inventory Utility: Rx_Inventory_S

The inventory model Rx_Inventory_S uses two input files. One of the input files contains the nuclide decay and production data; it is named Coreinventory.dat. The other input file contains user input 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 4-10 lists the format content of each of the lines in the Reactor.tmp file.

4.6.1 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.

Table 4-6 Reactor.tmp file format and contents

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

The second line identifies the reactor type. The core inventory model recognizes seven reactor types and selects the PWR or BWR nuclide production parameters based on reactor type. The reactor types are: BWR, ABWR, ESBWR, PWR, 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 website 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 4-11gives 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 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 in is

the length in days of a typical refueling outage. Figure 4-8 shows the distribution of the duration of refueling outages in days for the 60 U.S. reactors that refueled in 2011.

Table 4-7 Fuel management characteristics

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

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

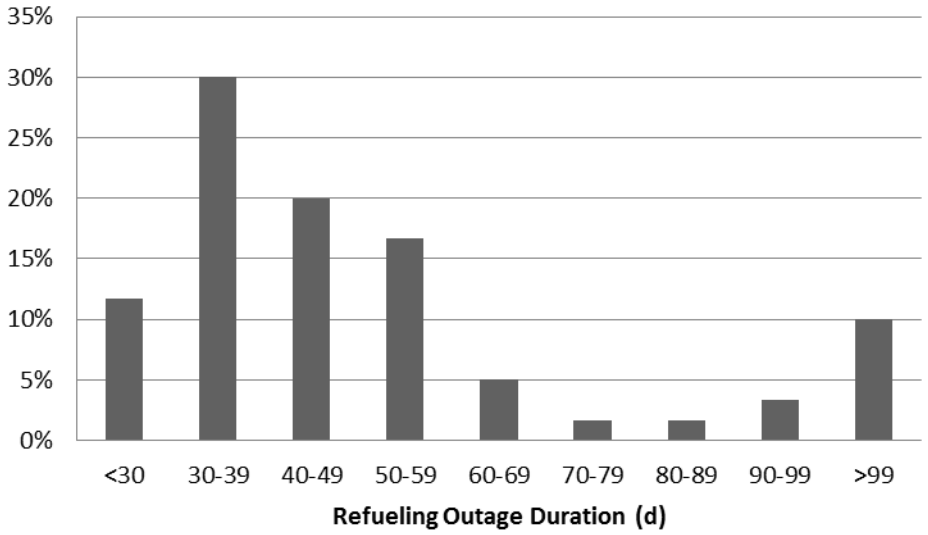


Figure 4-8 Distribution of Refueling Outage Durations in 2011

The last three lines in this section are related to the most recent fuel cycle. The fifth line in the 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 shut down.

4.6.2 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 data base. 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 4-9 shows the

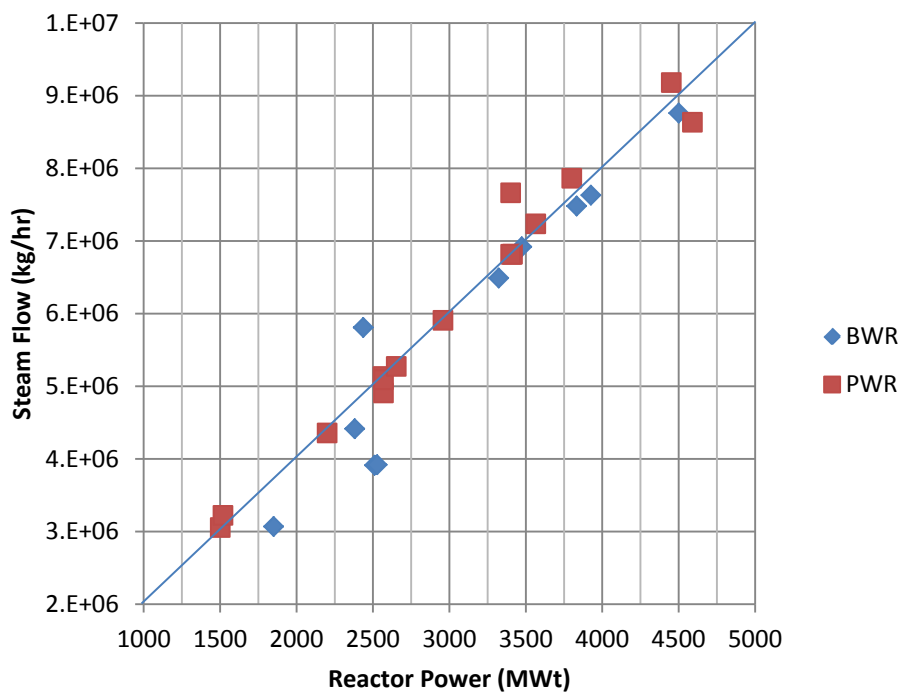


Figure 4-9 Steam Flow as Function of Reactor Power

relationship between design basis steam flow and design basis thermal power for current generation and new nuclear power plant designs. To a very good approximation, the steam flow may be estimated from the thermal power using

$$FS = \frac{2000\ (kg/h)}{MWt \times P} \tag{4-16}$$

where: FS = the steam flow (kg/h)
P = the 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 Standard gives 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, 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. (The 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 data base 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/day (3.9E-04 kg/s) is assumed in the ANSI Standard. The AP1000 DCD assumes a primary to secondary leak rate of 500 gal/day (~4000 lb/day) for design basis calculations and a primary to secondary leak rate of 75 lb/day 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 4-10 shows the relationship between secondary water mass and reactor power for current generation and new power plant designs.

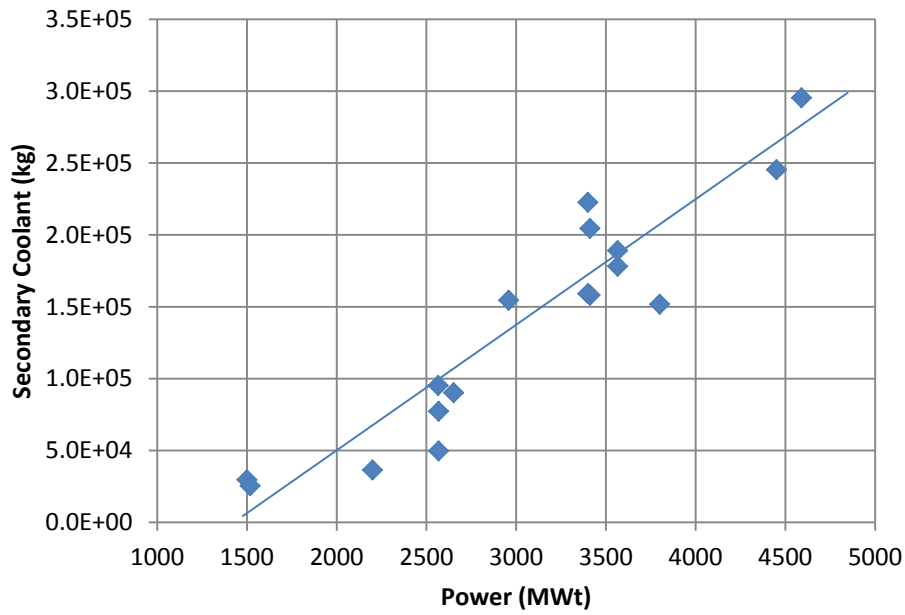


Figure 4-10 Secondary Water Mass as a Function of Reactor Power for Current Generation and New PWR Reactor Designs

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

$$WS = 85.7 \times P - 1.24 \times 10^5 \tag{4-17}$$

Where: WS = the secondary water mass (kg)
P = the reactor design power (MWt)

RASCAL 4.3 estimates the steam flow on the secondary side using Equation 4-16.

4.6.3 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. The remaining file is a debug file that contains intermediate computational results. This section describes the inventory files.

4.6.3.1 Core Inventory

The core inventory file (core_inventory.tmp) is intended for use by the reactor accident source term module. Figure 4-11 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 nuclides in the inventory. There are other nuclides in the inventory, but the nuclides in this file contribute most to potential offsite consequences. The remainder of the file is a listing of the inventory in Bq by nuclide.

```
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 4-11 Sample Core Inventory.TMP file

4.6.3.2 Coolant Inventory

Cooling system inventory estimates are stored in the file rcs_inventory.tmp. This file list 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 describe the contents of the file and the following three lines for a PWR (two lines for a BWR) list the reactor parameter input provided by the user. The next four lines list the RCS parameters that vary by nuclide. These parameters, which are included in Rx_Inventory_S data statements, are listed by the MELCOR nuclide classes plus a class for tritium. The MELCOR classes are:

1	2	3	4	5	6	7	8	9	10	11
NG	Cs, Rb	Sr, Ba	I, Br	Te	Ru	Mo	Ce, Np	La, Pr, Y	U	Sb

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 nuclide classes, NG, I, Cs and Rb, and other nuclides. 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. The following excerpt shows a portion of a PWR rcs_inventory.tmp file. Each nuclide is followed by its estimated concentrations (Bq/g) in the primary coolant, the secondary water, and in the steam.

```
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
```

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 systems; and nuclide concentrations are only provided for water and steam. An excerpt of a BWR rcs_inventory.tmp file is shown below.

```
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
…
```

4.6.3.3 Spent-Fuel Pool Inventory

The third output file is the spent-fuel pool inventory file sfpool_inventory.tmp. It contains the inventory for each batch of fuel in the reactor at the time of shut-down. 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 shut down. 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
…
```

5 Activity Balance

RASCAL source term calculations track nuclide activity in the core, 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 trouble shoot problems. However they have not been available to users. RASCAL 4.3 makes this information available to users by creating an activity balance file and displaying the contents through the user interface.

The activity balance file tracks activity for selected nuclides and nuclide groups from the reactor core and coolant systems through various pathways to the environment. For each time step the file lists the activity in 5 locations and the activity movement in 9 paths. The locations and paths are listed in Table 5-1.

Table 5-1 Locations and transfer paths for Balance activity summaries.

Col	Name	Location or Path
1	Rx Core	Activity in the reactor core
2	Rx-RCS	Activity transferred from the reactor core to primary coolant
3	Rx-Cont	Activity transferred from the reactor core to containment
4	Rx-Env	Activity transferred from the reactor core 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, 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 reactor core, primary and secondary coolant, and containment does reflect the reduction in activity by depletion and decay. The activity released to the environment reflects the reduction in activity prior to release to the environment, but it does not reflect decay or ingrowth in the environment.

An activity balance file is created for source terms based on reactor conditions. These source terms types include Long Term Station Blackout, Time Core is Uncovered, Specified Core Damage Endpoint, 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 5-2 lists the

activity locations and transfers tracked in the activity balance file for each source type and release path.

Table 5-2 Activity Balance locations and transfer paths for reactor accident categories

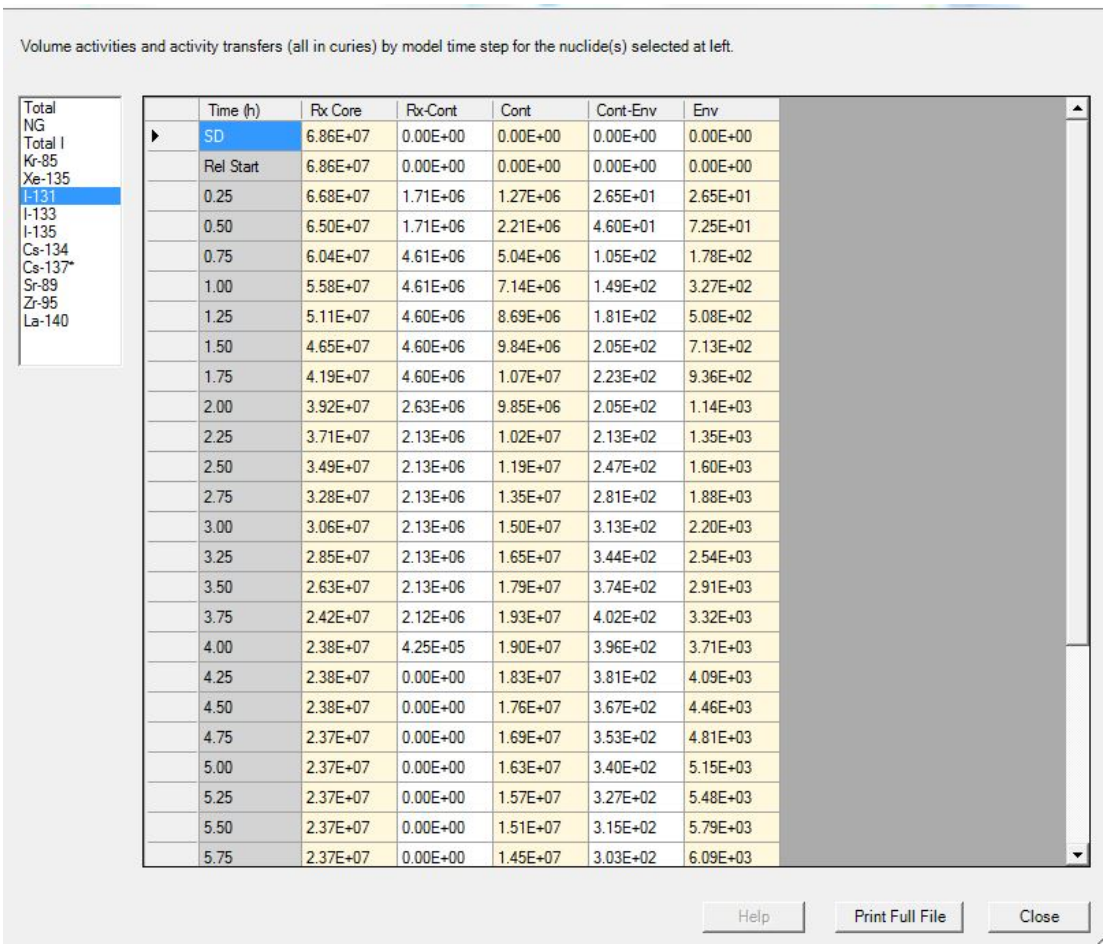
Source Term Type	Release Path	Activity Locations and Transfers
PWR Events		
Station Blackout	Cont Leak	Rx Core, Rx-Cont, Cont, Cont-Env, Env
	SGTR	Rx Core, Rx-RCS, RCS, RCS-SG, SG, SG-Env, Env
	Bypass	Rx Core, Rx-RCS, RCS,RCS-Env, Env
LB Loss-of-Coolant	Cont Leak	Rx Core, Rx-Cont, Cont, Cont-Env, Env
	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		
Station Blackout	Wet Well	Rx Core, Rx-Cont, Cont, Cont-Env, Env
	Dry Well	Rx Core, Rx-Cont, Cont, Cont-Env, Env
	Bypass	Rx Core, Rx-RCS, RCS, RCS-Env, Env
LB Loss-of-Coolant	Wet Well	Rx Core, Rx-Cont, Cont, Cont-Env, Env
	Dry Well	Rx Core, Rx-Cont, Cont, Cont-Env, Env
	Bypass	Rx Core, Rx-RCS, RCS, RCS-Env, Env
Coolant Release	Wet Well	NA
	Dry Well	NA
	Bypass	RCS, RCS-Env, Env

The activity balance file includes separate tables for 12 activity groups and nuclides. 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 of the release starts (t = 0), and the remaining times are at 900 s intervals following the beginning of release. These features are shown in Table 5-3 presents a portion of the balance file for a PWR LOCA as it is displayed in the user interface. The activity shown is “total activity” in curies. The first column is the time of the balance. The SD and Rel Start rows give the activity balance at the time of reactor shutdown and the beginning of release from the reactor core. At these times the activity is in the core. If the event involved release of activity from the RCS, the activity in the RCS would also be shown. The activity shown in the Rx Core (reactor core), Cont (containment), and Env (environment) columns is the activity at the end of the time period. The activity in the core and containment is adjusted for decay and depletion. The activity in the Env column is the total activity released to the environment, not corrected for decay. The activity shown in the Rx-Cont and Cont-Env columns is the activity transfer during the period.

The activity balance information may be accessed from the user interface using the View Balance button on the Source Term Tab. Figure 5-1 shows how the balance information is presented through the user interface.

Table 5-3 Sample of the Balance Summary Available in the User Interface

Time (h)	Rx Core	Rx-Cont	Cont	Cont-Env	Env
SD	7.07E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
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	6.86E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rel Start	6.86E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.25	6.68E+07	1.71E+06	1.27E+06	2.65E+01	2.65E+01
0.50	6.50E+07	1.71E+06	2.21E+06	4.60E+01	7.25E+01
0.75	6.04E+07	4.61E+06	5.04E+06	1.05E+02	1.78E+02
1.00	5.58E+07	4.61E+06	7.14E+06	1.49E+02	3.27E+02
1.25	5.11E+07	4.60E+06	8.69E+06	1.81E+02	5.08E+02
1.50	4.65E+07	4.60E+06	9.84E+06	2.05E+02	7.13E+02
1.75	4.19E+07	4.60E+06	1.07E+07	2.23E+02	9.36E+02
2.00	3.92E+07	2.63E+06	9.85E+06	2.05E+02	1.14E+03
2.25	3.71E+07	2.13E+06	1.02E+07	2.13E+02	1.35E+03
2.50	3.49E+07	2.13E+06	1.19E+07	2.47E+02	1.60E+03
2.75	3.28E+07	2.13E+06	1.35E+07	2.81E+02	1.88E+03
3.00	3.06E+07	2.13E+06	1.50E+07	3.13E+02	2.20E+03
3.25	2.85E+07	2.13E+06	1.65E+07	3.44E+02	2.54E+03
3.50	2.63E+07	2.13E+06	1.79E+07	3.74E+02	2.91E+03
3.75	2.42E+07	2.12E+06	1.93E+07	4.02E+02	3.32E+03
4.00	2.38E+07	4.25E+05	1.90E+07	3.96E+02	3.71E+03
4.25	2.38E+07	0.00E+00	1.83E+07	3.81E+02	4.09E+03
4.50	2.38E+07	0.00E+00	1.76E+07	3.67E+02	4.46E+03
4.75	2.37E+07	0.00E+00	1.69E+07	3.53E+02	4.81E+03
5.00	2.37E+07	0.00E+00	1.63E+07	3.40E+02	5.15E+03
5.25	2.37E+07	0.00E+00	1.57E+07	3.27E+02	5.48E+03
5.50	2.37E+07	0.00E+00	1.51E+07	3.15E+02	5.79E+03
5.75	2.37E+07	0.00E+00	1.45E+07	3.03E+02	6.09E+03

Figure 5-1 User Interface Presentation of the RASCAL 4.3 Activity Balance Information

6 Importance

During the emergency response to the Fukushima Dai-ichi accident there were frequent requests to identify a limited number of nuclides that were most important. RASCAL 4.3 includes a utility program to process the total nuclide activity released to the environment in the course of an event by evaluating the relative importance of the nuclides to four dose measures and ranking the nuclides in order of importance. The order of importance is a function of time as well as the dose measure. The importance utility calculates importance at the end of the release and at seven additional times ranging from 1 day to 10 years. The results of the importance calculation are available through the user interface.

6.1 The Importance Model

The importance model assumes that the importance of nuclides can be determined from pseudo dose calculations that are based on the dose calculation methods described in the RASCAL 4 technical documentation (NUREG-1940). The pseudo doses calculated in RASCAL 4.3 are for the submersion (cloud shine), inhalation, ground shine, and ingestion pathways, and for TEDE.

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

$$pD_i(t) = Q_i(t) \times DCF_i \tag{6-1}$$

where: $pD_i(t)$ = the pseudo dose from nuclide i at time t = 0
Q_i = the total activity release of nuclide i decayed to the release time
DCF_i = a submersion dose conversion factor for nuclide i

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

$$pD_i(t) = Q_i(t) \times DCF_i \times dv_i \times rs_i(t) \tag{6-2}$$

where: dv_i = the deposition velocity for nuclide I, (0.0 for noble gases, 0.003 m/s for iodines, and 0.001 for other nuclides)
$rs_i(t)$ = a resuspension factor (See NUREG-1940 Chapter 7) for time t after the end of release.

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

$$pD_i(t) = Q_i(t) \times vb \times DCF_i \tag{6-3}$$

where: vb = the breathing rate
DCF_i = an inhalation dose conversion factor.

After plume passage the inhalation pseudo dose is calculated as

$$pD_i(t) = Q_i(t) \times dv_i \times rs_i(t) \times vb \times DCF_i \tag{6-4}$$

The pseudo doses for the groundshine pathway are calculated by

$$pD_i(t) = Q_i(t) \times DCF_i \times dv_i \times wx(t) \tag{6-5}$$

where: DCF_i = a groundshine dose conversion factor.
wx(t) = a 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 groundshine pathways.

Finally, the ingestion pathway pseudo dose is calculated as

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

where: DCF_i = an ingestion dose conversion factor.
wxlv(t) = a weathering factor for leafy vegetables

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

$$I_i(t) = \frac{pD_i(t)}{\sum_i pD_i(t)} \tag{6-7}$$

where $I_i(t)$ is the importance of nuclide I and the summation over all nuclides releases.

The sum of pseudo doses of all nuclides is referred to as the dose potential for the pathway.
Note that RASCAL 4.3 uses the dose conversion factors for the submersion, inhalation, and groundshine pathways that were selected in the user interface, either Federal Guidance Report (FGR) 11 and 12 dose conversion factors (Eckerman et al 1988; Eckerman and Ryman 1993) or FGR-13 dose conversion factors (Eckerman and Leggett 2011). The FGR-13 dose conversion factors are for an adult.

The importance of each nuclide in the source term includes the contributions to pseudo dose from all daughters. When a parent nuclide and a daughter nuclide 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 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 nuclides 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 6-8. The weathering function is defined as

$$wxlv(t) = exp^{-\ln(2)\times\frac{t}{14}} \tag{6-8}$$

where 14 is the effective half-life of the weathering process. The weathering process is assumed to end after 183 days (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 isn't considered in RASCAL importance calculations. The dose conversion factor and the consumption are both functions of age. Neither the variation dose conversion factor 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.

6.2 Importance Results

The importance calculation results are presented in two parts. The first part establishes a context for interpretation of the nuclide 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 groundshine pathways to TEDE. Figure 6-1 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    1.0E-07    5.8E-08    4.7E-08    1.4E-08    7.2E-09

Relative Importance of Pathways to TEDE
Days           0.         1.         7.        30.       183.       365.      1825.      3650.
cs        3.3E-02    4.2E-04    2.8E-04    5.7E-05    3.0E-07    2.4E-07    9.1E-08    8.8E-08
inh       9.7E-01    4.0E-02    4.0E-02    1.4E-02    9.5E-05    7.9E-05    5.5E-05    8.1E-05
gs        8.3E-07    9.6E-01    9.6E-01    9.9E-01    1.0E+00    1.0E+00    1.0E+00    1.0E+00
```

Figure 6-1 An example of the contextual portion of the Importance results

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

The second portion of the Importance results file shows the importance of specific nuclides. Figure 6-2 shows a portion of the Balance file that presents the importance of specific nuclides for a PWR LOCA event.

```
Cumulative Importance to dose at t =       1. days
        Submersion        Inhalation        Ground shine      Ingestion
  1  I-132       0.34  I-131       0.25  I-132       0.21  I-131       0.31    1
  2  La-140      0.45  Sr-90       0.44  La-140      0.41  Cs-134      0.53    2
  3  I-133       0.56  Pu-241      0.55  Cs-134      0.52  Te-132      0.65    3
  4  I-131       0.65  Sr-89       0.62  I-133       0.60  Cs-137*     0.75    4
  5  Xe-133      0.71  Cm-242      0.70  Te-132      0.67  Sr-90       0.80    5

Cumulative Importance to dose at t =       7. days
        Submersion        Inhalation        Ground shine      Ingestion
  1  La-140      0.36  Sr-90       0.24  La-140      0.43  Cs-134      0.31    1
  2  Cs-134      0.52  I-131       0.43  Cs-134      0.63  I-131       0.57    2
  3  I-131       0.65  Pu-241      0.56  Cs-136      0.70  Cs-137*     0.71    3
  4  Xe-133      0.73  Sr-89       0.65  I-131       0.76  Sr-90       0.78    4
  5  I-132       0.79  Cm-242      0.73  Cs-137*     0.81  Sr-89       0.83    5

Cumulative Importance to dose at t =      30. days
        Submersion        Inhalation        Ground shine      Ingestion
  1  Cs-134      0.53  Sr-90       0.32  Cs-134      0.53  Cs-134      0.47    1
  2  Cs-137*     0.66  Pu-241      0.49  Cs-137*     0.67  Cs-137*     0.70    2
  3  La-140      0.77  Cm-242      0.60  La-140      0.77  Sr-90       0.81    3
  4  I-131       0.83  Cs-134      0.69  Cs-136      0.83  Sr-89       0.87    4
  5  Cs-136      0.89  Sr-89       0.78  Ba-140      0.87  I-131       0.93    5
Cumulative Importance to dose at t =     183. days
        Submersion        Inhalation        Ground shine      Ingestion
  1  Cs-134      0.76  Sr-90       0.41  Cs-134      0.74  Cs-134      0.53    1
  2  Cs-137*     0.98  Pu-241      0.63  Cs-137*     0.97  Cs-137*     0.82    2
  3  Zr-95       0.99  Cs-134      0.74  Y-90        0.97  Sr-90       0.95    3
  4  Nb-95       0.99  Cm-242      0.81  Zr-95       0.98  Ce-144*     0.97    4
  5  Ru-106*     0.99  Ce-144*     0.88  Sr-89       0.99  Y-90        0.98    5
```

Figure 6-2 Sample of specific nuclide importance file for a PWR LOCA

The example only shows the five most important nuclides for four time periods. The file lists the importance of all nuclides (typically 70 to 80) for eight time periods. The time periods not shown in the figure are 0 days (end of model run), 365 days (1 yr), 1825 days (5 yr), and 3650 days (10 yr). For each time period and pathway, the nuclides are listed in order of decreasing importance. The number following the nuclide is the fraction of dose potential contributed by that nuclide and the nuclides that are above it in the list.

The importance lists in Figure 6-2 illustrate features that are common to most importance lists. First, the most important nuclides are different for different dose pathways and change with increasing time after the release. Second the number of nuclides that contribute significantly to dose potential decreases with increasing time. These points are clearly illustrated in the groundshine importance lists by comparing the lists for 1 and 183 days. The two most important nuclides for ground shine at day 1 contribute about 41 percent of the dose potential. Neither nuclide is in the top five list for 183 days. Further, the top two nuclides at day 183 contribute about 97 percent of the dose potential.

7 Source Term Import, Export and Merge

RASCAL 4.3 includes an option to import a source term instead of using the RASCAL 4.3 source term module. RASCAL 4.3 also includes a new tool to export and merge source terms. Combined, these new features give RASCAL 4.3 the capability to deal with simultaneous or nearly simultaneous releases from two or more reactors at a single site or with simultaneous or nearly simultaneous releases from a reactor and an adjacent spent fuel pool. The general procedure to be followed in either case is to run RASCAL 4.3 for each release individually and save the cases as usual. When all individual cases have been run, uses the Source Term Merge / Export tool to create and export a combined source term file. Re-run RASCAL 4.3 importing the merged source term. The RASCAL 4.3 results will show the consequences of the combined source term.

Note that this procedure 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 following sections describe the Import source term option and the Source Term Merge / Export tool.

7.1 Source Term Import Option

The Source Term Import Option is available for all RASCAL 4.3 Event types. It is selected using the Import check box below Source Term button on the main Source Term to Dose data entry form. When a source term is imported, RASCAL 4.3 obtains the source term and release height and location from the imported file. The RASCAL 4.3 modules that usually calculate the source term and release information are bypassed.

RASCAL 4.3 accepts imported source terms in either XML or CSV format. The file format should be indicated by the file extension -- name.xml or name.csv. The RASCAL 4.3 Source Term Merge/Export tool (see the following sections) can be used to create either type file from an existing RASCAL case file, or an import file can be created by another program as long as the file format is followed. The following figures and tables show the required import file formats for releases to the atmosphere.

```
<?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="Test run"
  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 7-1 Example of a RASCAL 4.3 XML Format Atmospheric Release Imported Source Term File

Table 7-1 Description of the Lines in an XML Format Imported 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 nuclides 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 nuclide in the release step. Nuclide_Name Released_Amount

```
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, Testing the export process
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 7-2 Example of a RASCAL 4.3 CSV Format Atmospheric Release Imported Source Term File

Table 7-2 Description of the Lines in an CSV Format Imported Atmospheric Release Source Term File

Keyword	Description
Creator	Model name and version; result type
File_Created	Date and time the export file was created
Site_Name	Name of the location
Release_Latitude	Latitude in decimal degrees of the model release point.
Release_Longitude	Longitude in decimal degrees of the model release point.
UTC_Offset	Hours from UTC
Release_Height	Release height and units
Case_Title	Title given to the modeling run
Case_Runtime	When the modeling run occurred
Activity_Units	Units for the reported activities
Other_Info	Status of the importance filter option
Interval	The dates of each time step
Start	The times of each time step
The above is followed by a line for each nuclide in the release consisting of the nuclide name, a comma, and the activity for that nuclide.	

Note that the time interval for RASCAL 4.3 source terms is 15 minutes. Note, also, that it is the user's responsibility to ensure that the list of nuclides in the imported source term file includes all daughter nuclides that will grow in during the RASCAL run. **Warning:** If the file to be imported was created using the RASCAL 4.3 Source Term Merge/Export tool with the Importance filter option, this may not be the case. If a daughter is not included in the imported file, the atmospheric transport and dispersion module will end in an error mode. The error message will indicate the missing daughter. As a matter of good practice files to be imported into RASCAL 4.3 should not be created using the import filter in the RASCAL 4.3 Source Term Merge/Export tool.

7.2 RASCAL 4.3 Source Term Merge/Export Tool

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

7.2.1 Base Folder

The base folder is the location where all the processing of the source term files occurs. The software will create files and sub-folders as needed to contain the RASCAL case information. This folder can be anywhere on your system. The recommended location is just under the folder containing the

program itself. The basic installation creates the folder \TempCases for this purpose if you choose to use it. During program execution, one of more sub-folder will be created under this base folder. All the sub-folders and their files are deleted when the program is closed.

7.2.2 Select Cases

Use the Add button to select one or more RASCAL 4.3 STDose cases to work with. If you select only one case, you can export it. If you select more than one, you will be able to merge them and then export the new source term. 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.

7.2.3 Load Cases

After at least one case has been added, the Load Cases button becomes active. Clicking this button will do the following for each case in the list:

1. Create a sub-folder to hold the case information
2. Open the case file and extract the needed files
3. Optionally, display the source term related case data as it is loaded
4. Optionally, run the importance calculation on the source term and display the results.

7.2.4 Merge

If more than one case has been loaded, then the Merge Source terms button becomes active. Clicking the button starts the merge process. The following must be specified:

- release height
- latitude / longitude
- case title
- analyst name
- site name

Before the merge begins, you need to review these parameters and select which value to use. The release height and latitude/longitude cannot be merged. You must select from the options available. These options are the values from the cases that are to be merged. For example, only a single release height is allowed. If your cases used differing release heights, you 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, site name can be edited as needed.

The case descriptions for the cases to be merged are concatenated and presented in an editable form in the text box on the right side of the Merge form. Modify this text to adequately describe the merged source terms.

Like the load case operation, you have the option to display the case information and the importance data for the merged source term.

7.2.5 Export

After the loading of a single case or the merging of multiple cases, you can export the source term file. The export function support 2 formats: the new XML format being proposed for the exchange of source term information, and the CSV (comma-separated variables) format. The CSV format is the same format used from within RASCAL itself but with additional information about the release point. Both formats include all the information about the release with no need to open the case in RASCAL.

There are 2 options for the export file (use the Options button):
1. Strip * characters from nuclide names. This is useful when the export file is to be used with other software (e.g. NARAC) that handles daughters differently.
2. Use the RASCAL 4.2 format. This will not include all the release information but is retained for backward compatibility. This option applies to the CSV format only.

The export file may be filtered by radionuclide importance. Select the option button to enable and then select the filter options: pathway, filter method, and time period. **Warning:** 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.

7.2.6 Case Information

This form displays the case information related to the modeling run and the generated source term. It is particularly important to include detailed information about the case or cased if the output file is to be exported or recalled at a later date.

7.2.7 Importance

The importance calculation examines the contribution of each nuclide in a source term to 5 dose pathways: TEDE, air immersion, groundshine, inhalation, and ingestion. Eight time periods are examined from 0 days out to 10 years. The user may specify the filter to export only the top X contributors to dose or to export all the nuclide needed to get X% of the dose.

The importance display shows the calculated contributions for each pathway and time period. The lists are truncated when the cumulative percentage dose becomes greater than 99.5%. Those nuclides that are not shown in the truncated list contribute a combined total of less than 0.5% of the dose. In some instances the list may be empty because all nuclides in the source term will have decayed to an extremely small value that is essentially 0.0.

See Chapter 6 for a detailed discussion of the RASCAL 4.3 importance calculation.

8 Exposure Guidelines

Various guidelines have been established for limiting exposure to chemicals based on health effects. RASCAL does not calculate health effects, but it does reference various exposure guidelines set to protect public health. RASCAL results exceeding guidelines are highlighted in the Maximum Value Table and in the Detailed Results displays.

8.1 EPA Protective Action Guides

Protective Action Guides (PAGs) suggest precautions that state and local authorities can take during an emergency to keep people from receiving an amount of radiation that might be dangerous to their health. EPA developed the PAG Manual to provide guidance on actions to protect the public, such as having people evacuate an area or stay indoors.

The EPA manual provides recommended numerical protective action guides for the principal protective actions available to public officials during a radiological incident. A PAG is defined for purposes of this document as the projected dose to an individual from a release of radioactive material at which a specific protective action to reduce or avoid that dose is recommended. PAGs are guides to help officials select protective actions under emergency conditions during which exposures would occur for relatively short time periods. They are not meant to be applied as strict numeric criteria, but rather as guidelines to be considered in the context of incident-specific factors. PAGs do not establish an acceptable level of risk for normal, nonemergency conditions, nor do they represent the boundary between safe and unsafe conditions. The PAGs are not legally binding regulations or standards and do not supersede any environmental laws. For information on roles, responsibilities and authorities during emergency response and recovery, please refer to the National Response Framework: http://www.fema.gov/national-response-framework and specifically for radiological incidents, the Nuclear Radiological Incident Annex: http://www.fema.gov/pdf/about/divisions/thd/IncidentNucRad.pdf (FEMA 2008a,b).

Some protective actions are not associated with a numerical PAG. For example, the control of access to areas is a protective action implemented in concert with other protective actions; it does not have its own PAG. Any reasonable action to reduce radiation dose is encouraged even if it is not associated with a PAG, such as recommending that individuals use ad hoc respiratory protection with a handkerchief or piece of folded cloth. Or in areas where PAGs are not exceeded, but airborne radioactivity is present, people might be asked to stay indoors to the extent practicable to reduce their exposures. To further develop radiological emergency plans, brief planning guides have been provided for reentry to relocation areas, the cleanup planning process and considerations for radioactive waste disposal. (Source: http://www.epa.gov/radiation/rert/pags.html accessed 9/9/2013)

8.2 Immediately Dangerous to Life or Health (IDLH)

The immediately dangerous to life or health (IDLH) values used by the National Institute for Occupational Safety and Health (NIOSH) as respirator selection criteria were first developed in the mid-1970's. The Documentation for Immediately Dangerous to Life or Health (IDLH) Concentrations is a compilation of the rationale and sources of information used by NIOSH during the original

determination of 387 IDLH values. In addition, NIOSH continues to review, document, and revise the science and methodology behind the existing IDLH values when appropriate, and derive new IDLH. (Source: http://www.cdc.gov/niosh/idlh/idlhintr.html accessed 9/9/2013)

8.3 Emergency Response Planning Guidelines (ERPGs)

ERPGs estimate the concentrations at which most people will begin to experience health effects if they are exposed to a hazardous airborne chemical for 1 hour. (Sensitive members of the public—such as old, sick, or very young people—aren't covered by these guidelines and they may experience adverse effects at concentrations below the ERPG values.) A chemical may have up to three ERPG values, each of which corresponds to a specific tier of health effects.

The three ERPG tiers are defined as follows:

- **ERPG-3** is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing or developing life-threatening health effects.
- **ERPG-2** is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing or developing irreversible or other serious health effects or symptoms which could impair an individual's ability to take protective action.
- **ERPG-1** is the maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing other than mild transient health effects or perceiving a clearly defined, objectionable odor.

(Source: http://response.restoration.noaa.gov/erpgs accessed 9/9/2013))

8.4 Acute Exposure Guideline Levels (AEGLs)

AEGLs represent threshold exposure limits for the general public and are applicable to emergency exposure periods ranging from 10 minutes to 8 hours. AEGL-2 and AEGL-3, and AEGL-1 values as appropriate, will be developed for each of five exposure periods (10 and 30 minutes, 1 hour, 4 hours, and 8 hours) and will be distinguished by varying degrees of severity of toxic effects. It is believed that the recommended exposure levels are applicable to the general population including infants and children, and other individuals who may be susceptible. The three AEGLs have been defined as follows:

- **AEGL-1** is the airborne concentration, expressed as parts per million or milligrams per cubic meter (ppm or mg/m3) of a substance above which it is predicted that the general population, including susceptible individuals, could experience notable discomfort, irritation, or certain asymptomatic nonsensory effects. However, the effects are not disabling and are transient and reversible upon cessation of exposure.
- **AEGL-2** is the airborne concentration (expressed as ppm or mg/m3) of a substance above which it is predicted that the general population, including susceptible individuals, could experience irreversible or other serious, long-lasting adverse health effects or an impaired ability to escape.

- **AEGL-3** is the airborne concentration (expressed as ppm or mg/m3) of a substance above which it is predicted that the general population, including susceptible individuals, could experience life-threatening health effects or death.

Airborne concentrations below the AEGL-1 represent exposure levels that can produce mild and progressively increasing but transient and nondisabling odor, taste, and sensory irritation or certain asymptomatic, nonsensory effects. With increasing airborne concentrations above each AEGL, there is a progressive increase in the likelihood of occurrence and the severity of effects described for each corresponding AEGL. Although the AEGL values represent threshold levels for the general public, including susceptible subpopulations, such as infants, children, the elderly, persons with asthma, and those with other illnesses, it is recognized that individuals, subject to unique or idiosyncratic responses, could experience the effects described at concentrations below the corresponding AEGL. (Source: http://www.epa.gov/oppt/aegl/pubs/define.htm accessed 9/9/2013)

8.5 Use of Potassium Iodide (KI)

NRC has published a "Spotlight" article on Consideration of Potassium Iodide in Emergency Planning (http://wwwnrc.gov/about-nrc/emerg-preparedness/potassium-iodide.html) on its web site. NRC emergency preparedness regulations require that States with a population within the 10-mile emergency planning guide consider including potassium iodide (KI) as a protective measure for the general public. The Food and Drug Administration (FDA) is the definitive medical authority in the United States on the use of KI. FDA guidance related to the use of KI may be found at http://www.fda.gov/Drugs/EmergencyPreparedness/BioterrorismandDrugPreparedness/ucm319791.htm. The Environmental Protection Agency Draft PAG Manual for Interim Use and Public Comment (http://www.epa.gov/radiation/rert/pags/html) includes specific guidance for the administration of KI in response to releases of radioactive iodine. The EPA guidance is based on child (1-yr old) thyroid dose.

RASCAL 4.3 calculates a child thyroid dose for the purpose of evaluating recommendations related to administration of KI. Child thyroid doses are only calculated if the ICRP-60/72 dose conversion factors are selected on the Calculate Doses data entry form.

9 Health Effects

RASCAL does not calculate health effects. The health effects that are presented in this supplement, The RASCAL User's Guide, and RASCAL Help summarize the work of others. They are background material only and should only be used to provide context for evaluating the consequences of RASCAL 4.3 predictions of doses and concentrations.

9.1 Radiation Health Effects

The health effects of radiation depend on the duration and the amount of radiation exposure. Exposure pathways also play an important role in how radiation affects the body. Internal exposure occurs when a radionuclide is inhaled, ingested, or enters the body through cuts, wounds, or injections. External exposure occurs when an individual is exposed to an external radioactive source. Radiation health effects can be broken down into two categories as described by the EPA (EPA 2013): stochastic health effects and non-stochastic health effects. Each are described below.

9.1.1 Stochastic Health Effects

Stochastic effects are associated with long-term, low-level (chronic) exposure to radiation ("Stochastic" refers to the likelihood that something will happen). Increased levels of exposure make these health effects more likely to occur, but do not influence the type or severity of the effect.

Cancer is considered by most people the primary health effect from radiation exposure. Simply put, cancer is the uncontrolled growth of cells. Ordinarily, natural processes control the rate at which cells grow and replace themselves. They also control the body's processes for repairing or replacing damaged tissue. Damage occurring at the cellular or molecular level, can disrupt the control processes, permitting the uncontrolled growth of cells. This is why ionizing radiation's ability to break chemical bonds in atoms and molecules makes it such a potent carcinogen.

Other stochastic effects also occur. Radiation can cause changes in DNA, the "blueprints" that ensure cell repair and replacement produces a perfect copy of the original cell. Changes in DNA are called mutations.

Sometimes the body fails to repair these mutations or even creates mutations during repair. The mutations can be teratogenic or genetic. Teratogenic mutations are caused by exposure of the fetus in the uterus and affect only the individual who was exposed. Genetic mutations are passed on to offspring.

9.1.2 Non-Stochastic Health Effects

Non-stochastic effects appear in cases of exposure to high levels of radiation, and become more severe as the exposure increases. Short-term, high-level exposure is referred to as 'acute' exposure.

Many non-cancerous health effects of radiation are non-stochastic. Unlike cancer, health effects from 'acute' exposure to radiation usually appear quickly. Acute health effects include burns and radiation sickness. Radiation sickness is also called 'radiation poisoning.' It can cause premature aging or even death. If the dose is fatal, death usually occurs within two months. The symptoms of radiation sickness include: nausea, weakness, hair loss, skin burns or diminished organ function.

Medical patients receiving radiation treatments often experience acute effects, because they are receiving relatively high "bursts" of radiation during treatment."

9.1.3 Summary of Radiation Health Effects

The following table presents a summary of radiation-induced health effects from whole-body irradiation. It is adapted from ICRP Publication 96.

Table 9-1 Radiation Health Effects with Dose

Radiation-Induced Health Effects – Whole-body Irradiation		
Expected Dose	**Effects**	**Outcome**
Very-low dose: ~ 10 mSv ~1 rem	No acute effects, Extremely small additional cancer risk, <0.1%	No observable increase in incidence of cancer, even in large exposure population, 10^6
Low dose: toward 100 mSv 10 rem	No acute effects, subsequent additional cancer risk, <1%	Possible observable increase in the incidence of cancer in a large population, ~ 10^5
Moderate dose: towards 1 Sv 100 rem	Nausea, vomiting possible, mild bone marrow depression; subsequent additional cancer risk of ~ 10%	Probable observable increase in the incidence of cancer if the exposed population is more than a few hundred.
High dose: >1 Gy >100 rad	Certain nausea, likely bone-marrow syndrome, high risk of death above 4 Gy (400 rad) of acute dose without medical treatment. Significant additional cancer risk.	Observable increase in the incidence of cancer.
[a]ICRP Publication 96, *Ann. ICRP 35*(1), 2005. (Source: K. Eckerman, ORNL, Radiation Toolbox 3.0)		

9.1.4 Acute Radiation Syndrome

Acute Radiation Syndrome (ARS) is an illness that results when a person is exposed to high levels of radiation over a short period of time. Depending on the amount of dose from a whole body (or a significant portion of it) the signs and symptoms are readily predictable. In general, the higher the radiation dose the greater the health effects.

According to the Centers for Disease Control (CDC) (CDC 2013),

> **The required conditions for Acute Radiation Syndrome (ARS) are:**
>
> - **The radiation dose must be large** (i.e., greater than 0.7 Gray (Gy) or 70 rads). Mild symptoms may be observed with doses as low as 0.3 Gy or 30 rads.
> - **The dose usually must be external** (i.e., the source of radiation is outside of the patient's body). Radioactive materials deposited inside the body have produced some ARS effects only in extremely rare cases.
> - **The radiation must be penetrating** (i.e., able to reach the internal organs). High energy X-rays, gamma rays, and neutrons are penetrating radiations.
> - **The entire body** (or a significant portion of it) must have received the dose. Most radiation injuries are local, frequently involving the hands, and these local injuries seldom cause classical signs of ARS.
> - **The dose must have been delivered in a short time** (usually a matter of minutes). Fractionated doses are often used in radiation therapy. These large total doses are delivered in small daily amounts over a period of time. Fractionated doses are less effective at inducing ARS than a single dose of the same magnitude.

The three classic ARS syndromes are: bone marrow syndrome, gastrointestinal (GI) syndrome, and cardiovascular (CV) / central nervous system (CNS) syndrome. The CDC describes the syndromes as:

Bone Marrow Syndrome

Bone marrow syndrome (sometimes referred to as hematopoietic syndrome): the full syndrome will usually occur with a dose greater than approximately 0.7 Gy (70 rads) although mild symptoms may occur as low as 0.3 Gy or 30 rads. The survival rate of patients with this syndrome decreases with increasing dose. The primary cause of death is the destruction of the bone marrow, resulting in infection and hemorrhage.

Gastrointestinal (GI) Syndrome

Gastrointestinal (GI) syndrome will usually occur with a dose greater than approximately 10 Gy (1000 rads) although some symptoms may occur as low as 6 Gy or 600 rads. Survival is extremely unlikely with this syndrome. Destructive and irreparable changes in the GI tract and bone marrow usually cause infection, dehydration, and electrolyte imbalance. Death usually occurs within 2 weeks.

Cardiovascular (CV)/ Central Nervous System (CNS) Syndrome

Cardiovascular (CV)/ Central Nervous System (CNS) syndrome will usually occur with a dose greater than approximately 50 Gy (5000 rads) although some symptoms may occur as low as 20 Gy or 2000 rads. Death occurs within 3 days. Death likely is due to collapse of the circulatory system as well as increased pressure in the confining cranial vault as the result of increased fluid content caused by edema, vasculitis, and meningitis.

9.1.5 ARS phases

ARS Syndromes follow four phases, a prodromal period, a latent period, a period of illness, and period of recovery or death. The radiation Emergency Assistance Center/Training Site (REAC/TS) guidance for radiation accident management states (REAC/TS 2013):

> "During the prodromal period patients might experience loss of appetite, nausea, vomiting, fatigue, and diarrhea; after extremely high doses, additional symptoms such as fever, prostration, respiratory distress, and hyperexcitability can occur. However, all of these symptoms usually disappear in a day or two, and a symptom-free, latent period follows, varying in length depending upon the size of the radiation dose. A period of overt illness follows, and can be characterized by infection, electrolyte imbalance, diarrhea, bleeding, cardiovascular collapse, and sometimes short periods of unconsciousness. Death or a period of recovery follows the period of overt illness."

9.1.6 Cutaneous Radiation Syndrome

The CDC also mentions another type of syndrome often associated with acute radiation exposure, the Cutaneous Radiation Syndrome (CRS) (CDC 2013). This is described as:

> "The concept of cutaneous radiation syndrome (CRS) was introduced in recent years to describe the complex pathological syndrome that results from acute radiation exposure to the skin. ARS usually will be accompanied by some skin damage. It is also possible to receive a damaging dose to the skin without symptoms of ARS, especially with acute exposures to beta radiation or X-rays. Sometimes this occurs when radioactive materials contaminate a patient's skin or clothes.
>
> When the basal cell layer of the skin is damaged by radiation, inflammation, erythema, and dry or moist desquamation can occur. Also, hair follicles may be damaged, causing epilation. Within a few hours after irradiation, a transient and inconsistent erythema (associated with itching) can occur. Then, a latent phase may occur and last from a few days up to several weeks, when intense reddening, blistering, and ulceration of the irradiated site are visible. In most cases, healing occurs by regenerative means; however, very large skin doses can cause permanent hair loss, damaged sebaceous and sweat glands, atrophy, fibrosis, decreased or increased skin pigmentation, and ulceration or necrosis of the exposed tissue."

9.2 Chemical Health Effects

RASCAL does not calculate health effects. The health effects that are presented in the RASCAL 4.3 Technical Supplement and RASCAL Help summarize the work of others. They are presented as background material only.

9.2.1 HF Lung -- HF

Acute exposure guideline levels (AEGLs) have been established for exposure to HF. They are applicable to the general population, including infants and children and other individuals who may be susceptible. The AEGLs are as follows:

	Hydrogen fluoride	7664-39-3	(Final)		
	ppm				
	10 min	30 min	60 min	4 hr	8 hr
AEGL 1	1.0	1.0	1.0	1.0	1.0
AEGL 2	95	34	24	12	12
AEGL 3	170	62	44	22	22

(Source: http://www.epa.gov/oppt/aegl/pubs/results53.htm accessed 9/9/2013)

9.2.2 HF Lung -- 1 Hour Equivalent

A 1-hour equivalent HF concentration in the lungs of about 1070 mg/m3 for one hour has about a 50% chance of fatality without emergency medical treatment. A concentration of 1670 mg/m3 for 30 minutes has about a 50% chance of fatality without emergency medical treatment. The NIOSH Immediately Dangerous to Life or Health (IDLH) concentration is 25 mg/m3 for 30 minutes. IDLH is defined as "a maximum concentration from which one could escape within 30 minutes without any escape-impairing symptoms or any irreversible health effects." (NUREG-1391, Chemical Toxicity of Uranium Hexafluoride Compared to Acute Effects of Radiation, 1991).

9.2.3 Inhaled Uranium

Uranium intake provides the best measure of potential acute kidney damage due to heavy metal poisoning. There are no known long-term chemical injuries from uranium intake that are sub-lethal.

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