# 2 SOURCE TERM CALCULATIONS IN STDOSE

The source term calculations in STDose estimate the amount of radioactive (or hazardous) material released based on a wide variety of potential radiological accident scenarios. The source term calculations performed can be generally categorized as: (1) nuclear power plant accidents, (2) spent reactor fuel accidents, (3) fuel cycle facility/$UF_6$ accidents, (4) uranium fires and explosions, (5) criticality accidents, and (6) isotopic releases (e.g. transportation, materials, etc.). Most of these calculations are performed using an extension of the method of estimating source terms from reactor accident conditions described in NUREG-1228 (McKenna and Giitter, 1988).

STDose divides the source term calculation into three parts: first, it calculates the amount of activity available for release based on the source term type; second, this activity is reduced, based on the reduction mechanisms that the user has selected for the release pathway and; third, the activity is released according to the release rate selected. This can be shown as:

$$S_{ijk} = I_i \times ARF_{ij} \times \prod_k RDF_{ik} \times LF \tag{2-1}$$

where

- $S_{ijk}$ is the total amount of radionuclide i released under accident conditions j and release pathway k,
- $I_i$ is the inventory of radionuclide i,
- $ARF_{ij}$ is the airborne release fraction of radionuclide i under accident conditions j,
- $RDF_{ik}$ is the reduction factor that applies to radionuclide i for reduction mechanism k, and
- $LF$ is the leakage fraction selected for this release pathway.

In RASCAL 3.0 nuclear power plant accident, criticality accident, and isotopic source terms may change with time. Source terms are re-computed at intervals of varying length, called time steps. A new time step starts whenever the user changes any of the time-dependent data or every 15 mins, whichever is less. At the end of each time step the inventory of activity is adjusted due the effects of radiologic decay, so that no more than the total inventory can be released. Time steps may be no less than 1 min and must be an integral number of minutes. Before passing the source term to the atmospheric transport model, STDose converts the user-selected time steps to the 15-min time steps used by the atmospheric transport models. Note that neither the type of source term calculation nor release pathway may change with time.

STDose allows eleven release pathways. Only those pathways that are appropriate to the accident type, location, and method of source calculation are available to the user in any assessment. Releases from nuclear power plant accidents occur through one of six types of release pathways. Accidents at other types of facilities only have a single release pathway available. All release pathways only allow those reduction mechanisms that are possible for that type of accident. Table 2.1 shows the relationships among the types of source terms and release pathways.

**Table 2.1. Relationship of Source Terms to Release Pathways**

| Event type | Source term | Release pathway |
|---|---|---|
| Nuclear power plant | Coolant sample | Steam generator tube rupture<br>Containment bypass |
| Nuclear power plant | Containment air sample<br>Containment radiation monitor | Containment leakage or failure |
| Nuclear power plant | Effluent gross concentrations release rate | Direct |
| Nuclear power plant<br>PWR only | Core cooling status<br>User-defined core damage estimate | Containment leakage or failure<br>Steam generator tube rupture<br>Containment bypass |
| Nuclear power plant<br>BWR only | Core cooling status<br>User-defined core damage estimate | Wet well<br>Dry well<br>Containment bypass |
| Reactor spent fuel | Pool storage - uncovered fuel<br>Pool storage - damaged assembly under water<br>Dry storage cask event | Spent fuel release |
| Fuel cycle facility<br>$UF_6$ | $UF_6$ release from cylinder | Cylinder release |
| Fuel cycle facility<br>$UF_6$ | $UF_6$ release from cascade | $UF_6$ cascade release |
| Fuel cycle facility<br>$UF_6$ | Fire involving uranium oxide<br>Explosion involving uranium | Fire and explosion release |
| Criticality | Criticality accident | Criticality release |
| Materials<br>Transportation<br>Other | Isotopic release rates<br>Isotopic concentrations<br>Sources and materials in a fire | Direct |

The following discussion of the STDose calculations is divided into three parts. First all the source term calculations are presented. These are followed by discussions of the reductions possible according to the release pathway. Finally, the calculation of leakage factor is presented.

## 2.1 CALCULATION OF THE AMOUNT OF ACTIVITY AVAILABLE FOR RELEASE

### 2.1.1 Nuclear Power Plant Accidents

Six types of source term data can be used to estimate the source term for a nuclear power plant accident: (1) core cooling status, (2) containment radiation monitor, (3) coolant sample, (4) containment air sample, (5) user-defined core damage estimate, and (6) effluent gross concentrations release rates. In the core cooling status and containment radiation monitor the user enters data that are used to estimate the core damage state or the activity in the coolant. In all other source term types the user enters a direct estimate of activity available for release. The calculations performed for each of these types of data are discussed in the following sections.

Nuclear power plant nuclide inventory data are used in computing nuclear power plant and spent fuel accidents. The inventories used for PWRs and BWRs are from NUREG-1150 (USNRC, 1987). Normal coolant concentrations are from (ANSI, 1984). These data are shown in Table 2.2. In all cases, if the reactor is shut down prior to the start of the release, the inventory or coolant activity is decayed over the shutdown period prior to any other calculations being performed.

#### 2.1.1.1 Core cooling status

When the user enters a core uncovery time, STDose uses this time to compute the duration that the core has been uncovered for each time step in the assessment. STDose will compute the core damage state from the core uncovery duration. The core damage state and the amount of core activity available in containment or coolant for a given core uncovery duration are based on the core release fractions in Tables 3-12 and 3-13 in NUREG-1465 (Soffer et al., 1992). These are shown in Tables 2.3 and 2.4 below according to Source Term Code Package (STCP) categories as revised in NUREG-1465 (Soffer, 1992) The chemical elements in each of the STCP categories are listed in Table 2.5. The isotopes of all elements not listed in this table are not included in the initial source term calculations.

STDose re-computes the core-damage state every time-step or 15 mins, whichever is less. First it computes the core uncovery duration, based on the core uncovery time entered by the user and the current time in the scenario calculations. The core damage state selected is the one into which the current core uncovery duration falls. For example, when a BWR core has been uncovered for 1 hr and 45 mins, it has a core damage state of 'core melt'; that is 1.75 hrs is greater than the total cladding failure duration (0.5) and less than the total cladding failure plus core melt durations (0.5 + 1.5). The fraction of that core-damage state, $F_{CDS}$ that has been released from the core at any time is determined as:

$$F_{CDS} = \frac{D_U - D_{CDS-1}}{D_{CDS}} \qquad (2\text{-}2)$$

where

$D_U$ is the current core uncovery duration (h),
$D_{CDS-1}$ is the core uncovery duration (h) at the beginning of core damage state CDS, and
$D_{CDS}$ is the total duration that core damage state CDS is expected to occur.

In the example given above, $D_U$ is 1.75, $D_{CDS-1}$ is 0.5 and $D_{CDS}$ is 1.5. So $F_{CDS}$ is (1.75-0.5)/1.5 or 0.833

**Table 2.2. Nuclear Power Plant Core Inventory and Coolant Concentrations**

| Nuclide | Inventory* Ci/MW(t) | PWR coolant** Ci/g | BWR coolant** Ci/g | Nuclide | Inventory* Ci/MW(t) | PWR coolant** Ci/MW(t) | BWR coolant** Ci/MW(t) |
|---|---|---|---|---|---|---|---|
| Ba-140 | 5.30e+00 | 1.30e-08 | ~~4.00e-11~~ 4.00e-10 | Mo-99 | 5.30e+04 | 6.40e-09 | 2.00e-09 |
| Ce-144 | 2.80e+04 | 4.00e-09 | 3.00e-12 | Np-239 | 5.50e+05 | 2.20e-09 | 8.00e-09 |
| Co-58 | 0 | 4.60e-09 | 2.00e-10 | Ru-103 | 3.70e+04 | 7.50e-09 | 2.00e-11 |
| Co-60 | 0 | 5.30e-10 | 4.00e-10 | Ru-106 | 8.00e+03 | ~~9.00e-08~~ 9.00e-8 | 3.00e-12 |
| Cs-134 | 2.50e+03 | 7.10e-09 | 3.00e-11 | Sb-127 | 2.00e+03 | Not Given | Not Given |
| Cs-136 | 1.00e+03 | 8.70e-10 | 2.00e-11 | Sb-129 | 1.10e+04 | Not Given | Not Given |
| Cs-137 | 1.60e+03 | 9.40e-09 | 8.00e-11 | Sr-89 | 3.10e+04 | 1.40e-10 | 1.00e-10 |
| H-3 | 0 | 1.00e-06 | 1.00e-08 | Sr-90 | 1.20e+03 | 1.20e-11 | 7.00e-12 |
| I-131 | 2.80e+04 | 4.50e-08 | 2.20e-09 | Sr-91 | 3.70e+04 | 9.60e-10 | 4.00e-09 |
| I-132 | 4.00e+04 | 2.10e-07 | 2.20e-08 | Tc-99m | 0 | 4.70e-09 | 2.00e-09 |
| I-133 | 5.70e+04 | 1.40e-07 | 1.50e-08 | Te-129m | 1.80e+03 | 1.90e-10 | 4.00e-11 |
| I-134 | 6.30e+04 | 3.40e-07 | 4.30e-08 | Te-131m | 4.00e+03 | 1.50e-09 | 1.00e-10 |
| I-135 | 5.00e+04 | 2.60e-07 | 2.20e-08 | Te-132 | 4.00e+04 | 1.70e-09 | 1.00e-11 |
| Kr-85 | 1.90e+02 | 4.30e-07 | 0 | Xe-131m | 3.30e+02 | 7.30e-07 | 0 |
| Kr-85m | 8.00e+03 | 1.60e-07 | 0 | Xe-133 | 5.70e+04 | 2.60e-06 | 0 |
| Kr-87 | 1.60e+04 | 1.50e-07 | 0 | Xe-135 | 1.10e+04 | 8.50e-07 | 0 |
| Kr-88 | 2.30e+04 | 2.80e-07 | 0 | Xe-138 | 5.70e+04 | 1.20e-07 | 0 |
| La-140 | 5.30e+04 | 2.50e-08 | 4.00e-10 | Y-91 | 4.00e+04 | 5.20e-12 | 4.00e-11 |
| Mn-54 | 0 | 1.60e-09 | 7.00e-11 | | | | |

*The inventories are from NUREG-1150 (USNRC, 1987).
**Coolant concentrations are from (ANSI, 1984).

If the user has entered a time at which the core is recovered, then core damage stops, and the core damage state remains the same. As the scenario proceeds and activity is removed from the core, STDose reduces the amount of activity in the core so that no more than the total amount of activity in the core may be released.

If the reactor core has never been uncovered, the only coolant activity may be released. Note that rapid increases in the iodine and fission products concentrations in the coolant as high as 3 orders of magnitude may be seen following reactor shutdown, startup, rapid power change, and reactor coolant system depressurization. Such increases are referred to as iodine spikes. The user may select a spiking factor from 1 to 100. The activity of all radionuclides in the coolant except noble gases are multiplied by this spiking factor. If the core has not been uncovered only the steam generator tube rupture release pathway is available.

**Table 2.3. BWR Core Activity Release Fractions***

| | **Cladding failure** | **Core melt** | **Vessel melt through** |
|---|---|---|---|
| Duration of core damage state (h) | 0.5 | 1.5 | 3.0 |
| Core uncovery duration at which this core damage state begins (h) | 0.0 | 0.5 | 2.0 |
| Noble gases | 0.05 | 0.95 | 0 |
| Halogens | 0.05 | 0.25 | 0.30 |
| Alkali metals | 0.05 | 0.20 | 0.35 |
| Tellurium group | 0 | 0.05 | 0.25 |
| Barium, strontium | 0 | 0.02 | 0.1 |
| Noble metals | 0 | 0.0025 | 0.0025 |
| Cerium group | 0 | 0.0005 | 0.005 |
| Lanthanides | 0 | 0.0002 | 0.005 |

*Table 3-12 from NUREG-1465 (Soffer et al., 1992)

### 2.1.1.2 Containment radiation monitor

STDose can use a containment radiation monitor reading to estimate the core damage state or the amount of activity in the coolant. The user can enter changes in the monitor readings with time, so this calculation can result in a time-varying source term. The fractional core damage state or coolant condition is determined based on the data in Figs. A.5 - A.12 in RTM-96 (McKenna, et al, 1996), which are shown as Figs. 2.1 - 2.8 below. The bars in these figures represent the range from 1 to 100% of the labeled core damage state. The data in these figures were calculated assuming a reactor power of 3000 MW(t). Note that these figures use the older terminology 'gap' and 'in-vessel melt' to refer to cladding failure and core melt, respectively. In this section only, the older terms are used for ease of comparison with the figures.

To estimate the core damage state, STDose first determines which set of data in the figures should be used. It selects the figure to use, based on the containment type and whether or not sprays are active and for BWRs whether the monitor is in the wet well or dry well. Since the figures include data for 1 hr and

## Table 2.4. PWR Core Activity Release Fractions*

| | Cladding failure | Core melt | Vessel melt through |
|---|---|---|---|
| Duration of core damage state (h) | 0.5 | 1.3 | 2.0 |
| Core uncovery duration at which this core damage state begins (h) | 0.0 | 0.5 | 1.8 |
| Noble gases | 0.05 | 0.95 | 0 |
| Halogens | 0.05 | 0.35 | 0.25 |
| Alkali metals | 0.05 | 0.25 | 0.35 |
| Tellurium group | 0 | 0.05 | 0.25 |
| Barium, strontium | 0 | 0.02 | 0.1 |
| Noble metals | 0 | 0.0025 | 0.0025 |
| Cerium group | 0 | 0.0005 | 0.005 |
| Lanthanides | 0 | 0.0002 | 0.005 |

*Table 3-13 from NUREG-1465 (Soffer et al., 1992)

## Table 2.5. Elements in Revised STCP Categories*

| STCP Category | Category name | Elements |
|---|---|---|
| 1 | Noble gases | Xe, Kr |
| 2 | Halogens | I, Br |
| 3 | Alkali metals | Cs, Rb |
| 4 | Tellurium group | Te, Sb, Se |
| 5 | Barium, strontium | Ba, Sr |
| 6 | Noble metals | Ru, Rh, Pd, Mo, Tc, Co |
| 7 switch | Cerium group | La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am |
| 8 | Lanthanides | Ce, Pu, Np |

*Table 3-8 from NUREG-1465 (Soffer et al., 1992)

24 hrs of hold up time, STDose adjusts these data for the actual hold up time entered by the user. If the holdup time entered is less than one hr or greater than 24 hrs, the data for the respective time is used as shown. If the holdup time is between one and 24 hrs, then STDose does a linearly interpolation to determine the data to use at the time selected.

Once the correct data are selected, STDose scales the monitor reading entered by the user to account for the difference in the reactor power used to produce the figure and the reactor power entered by the user in the accident scenario. This scaled monitor reading, R, is calculated by:

$$R = \frac{3000 \times MR}{Power} \tag{2-3}$$

where

| | |
|---|---|
| MR | is the containment monitor reading entered and |
| Power | is the reactor power (MW(t)). |

STDose then determines for which of the core damage states or coolant conditions this reading falls between the 1 and 100% level. Most containment monitor readings can represent either (1) in-vessel melt or gap core damage states or (2) normal or spiked coolant conditions. If no other data are available, STDose assumes that a reading representing between 1% and 100% gap is gap. A reading greater than 100% gap is assumed to be vessel melt. A reading representing between 1% and 100% normal coolant is assumed to be normal coolant. A reading between 100% normal coolant and 100% spiked coolant is assumed to be spiked coolant. If the monitor reading is between the 100% spiked coolant level and the 1% gap level, the core damage state is assumed to be gap. The user may enter additional information to help STDose determine which is the appropriate core or coolant state. If the user enters a core uncovery time, it can be used to distinguish between the overlapping scales of the gap and in-vessel states. If the user enters data on the primary system pressure event history, these data may be used to distinguish between normal and spiked coolant.

When STDose determines that the estimated core damage state is gap or in-vessel melt, it then computes the fraction of either state that the reading represents. The percentage, P, of the damage state selected is calculated by:

$$P = 100 \times \frac{R}{P_{1D}} \tag{2-4}$$

where

$P_{1D}$ is the meter reading assumed for 1% of the core damage state for a 3000MW(t) reactor.

If the monitor reading represents spiked coolant, STDose uses a spiking factor of 100. No 'fractional' coolant states are computed for spiked or normal coolant.

(50% or greater)

These calculations should provide the maximum reading expected under the conditions stated. The calculations assume: (1) a prompt release to containment of all the fission products in the coolant, spiked coolant, gap, or from in-vessel melt; (2) uniform mixing in the containment; and (3) an unshielded monitor that can see a large fraction of the containment volume. Because the mix is most likely different from that assumed in the calibration of the monitor, the actual reading at the upper end of the scale could differ by a factor of 10-100 if a shielded detector is used for the higher radiation measurements.

The levels of damage indicated in the figures should be considered minimum levels unless there are inconsistent monitor readings. Inconsistent readings may be caused by uneven mixing in containment [e.g. steam rising to top of dome, not enough time for uniform mixing to occur (it may take hours)].

When the radiation monitor reading implies no core damage, there is no activity released. In RASCAL 3.0 if a containment monitor reading implies a normal or spiked coolant activity, no release can occur because the only release pathway available for containment monitor source terms, "containment leakage/failure", does not allow releases of coolant.

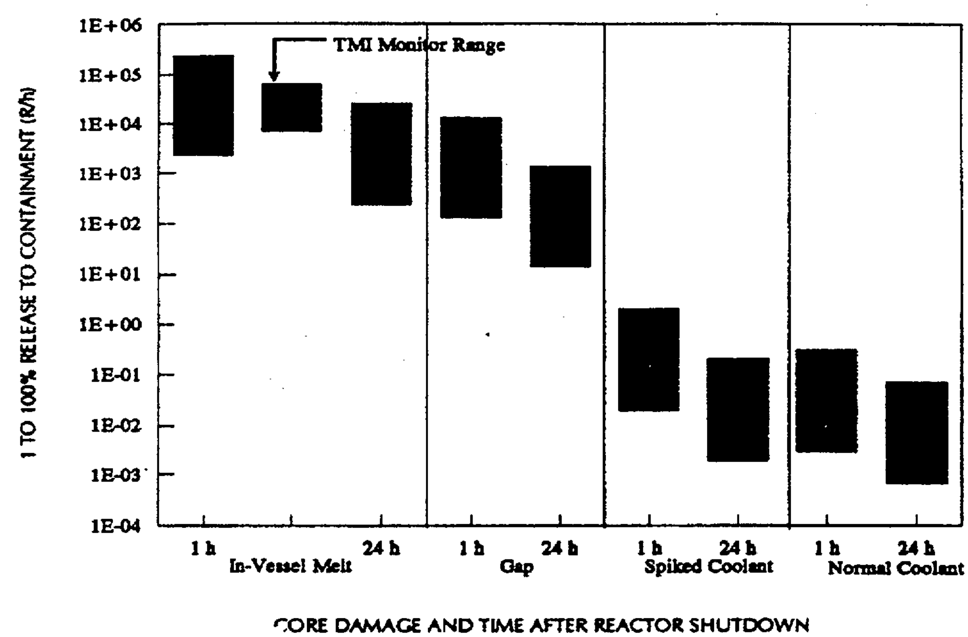


**Figure 2.1. PWR containment monitor response (sprays on).**

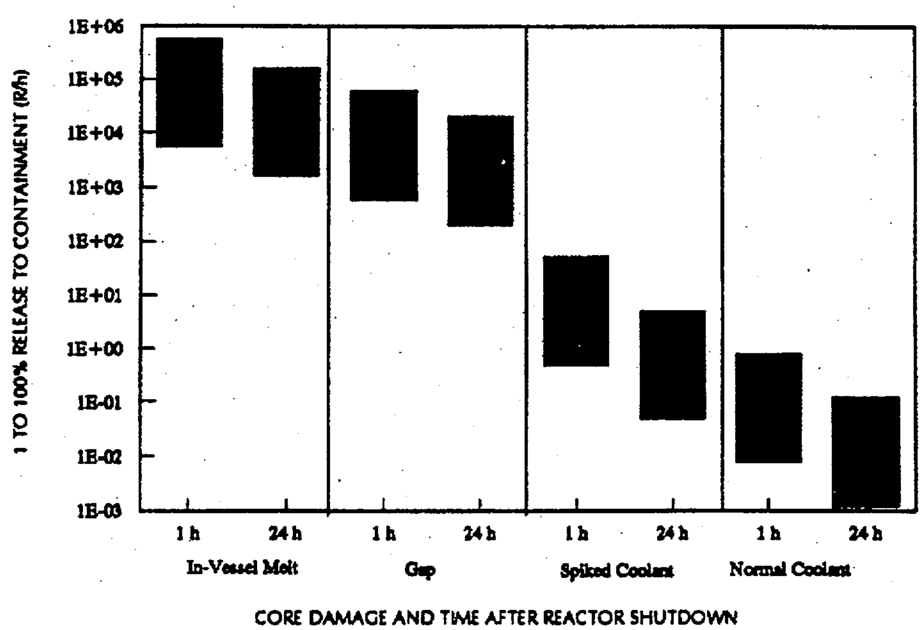


Figure 2.2. PWR containment monitor response (sprays off).

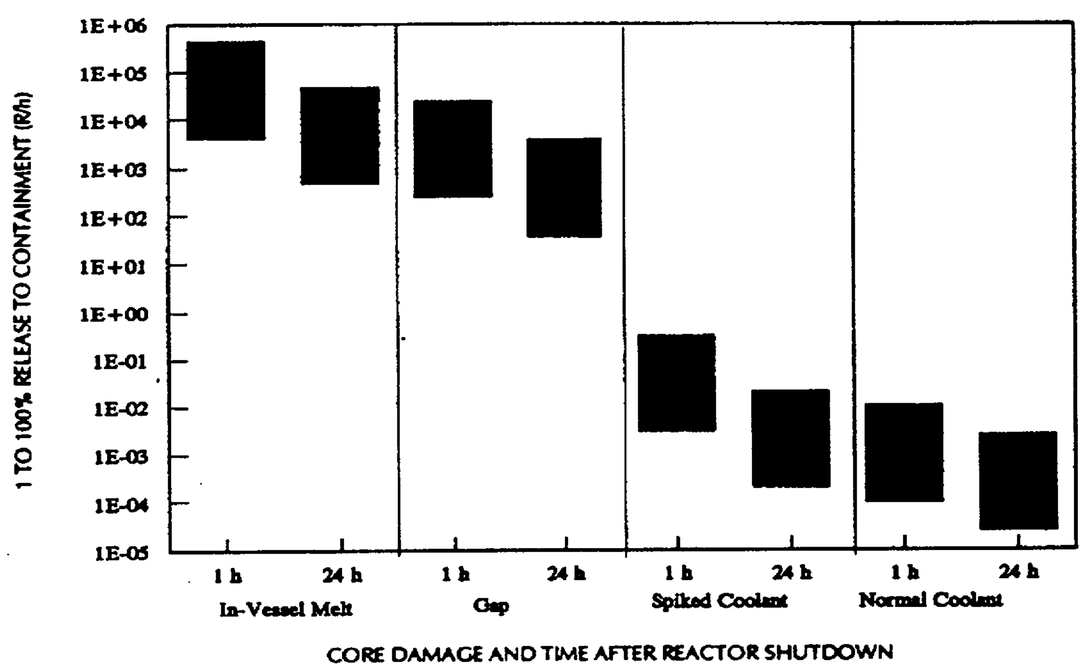


Figure 2.3. BWR Mark I & II dry well containment monitor response (sprays on).

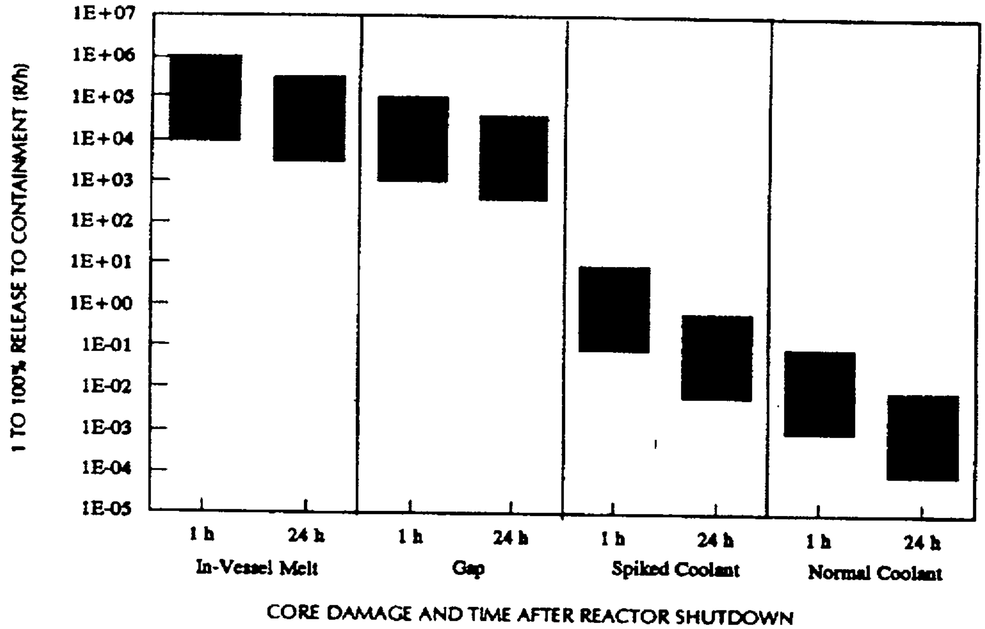


Figure 2.4. BWR Mark I & II dry well containment monitor response (sprays off).

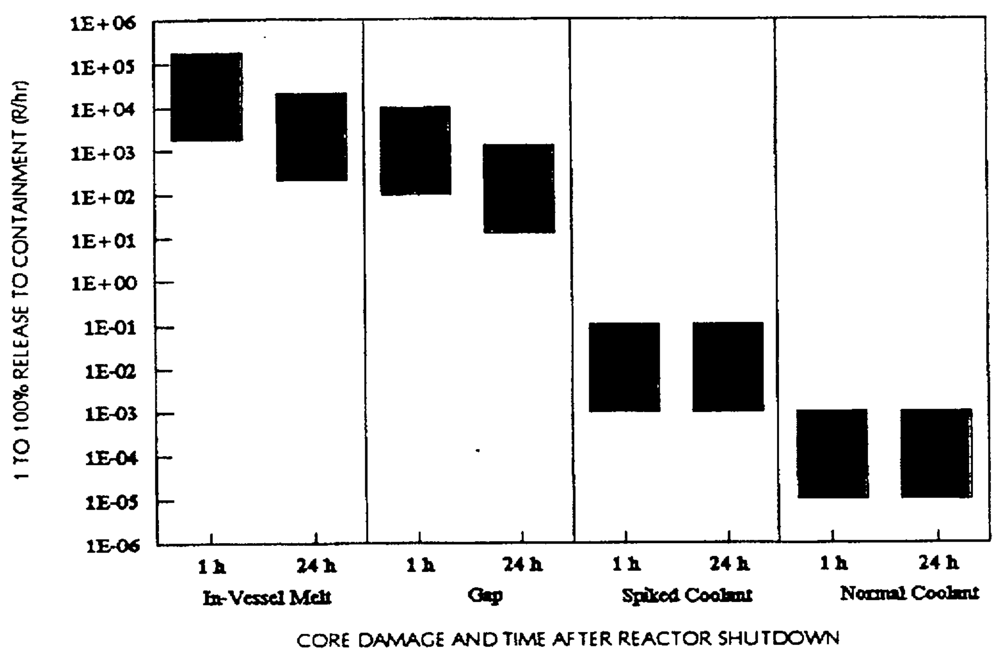


Figure 2.5. BWR Mark I & II wet well containment monitor response.

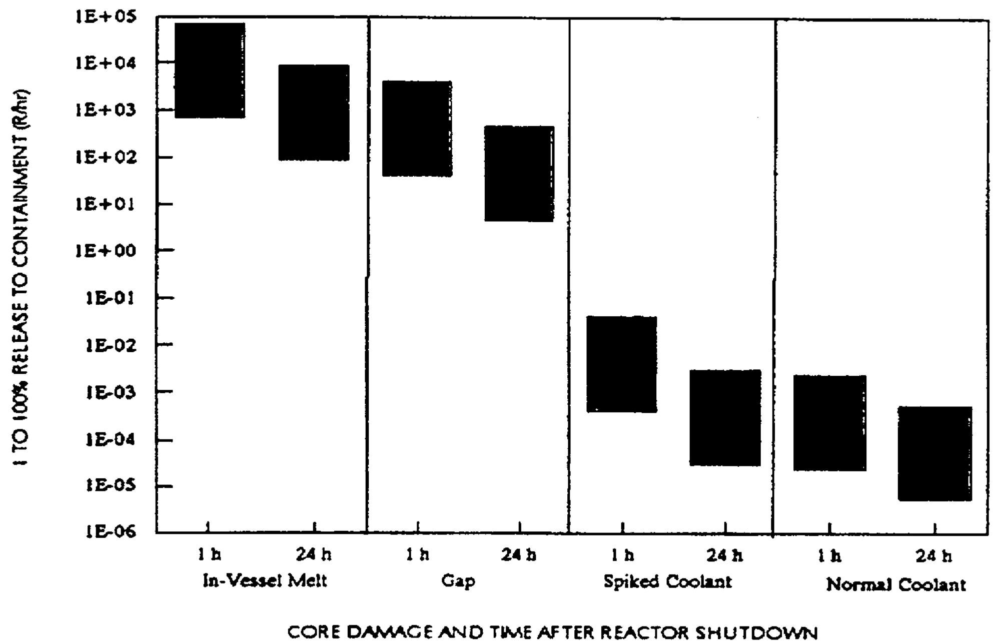


**Figure 2.6. BWR Mark III dry well containment monitor response (sprays on).**

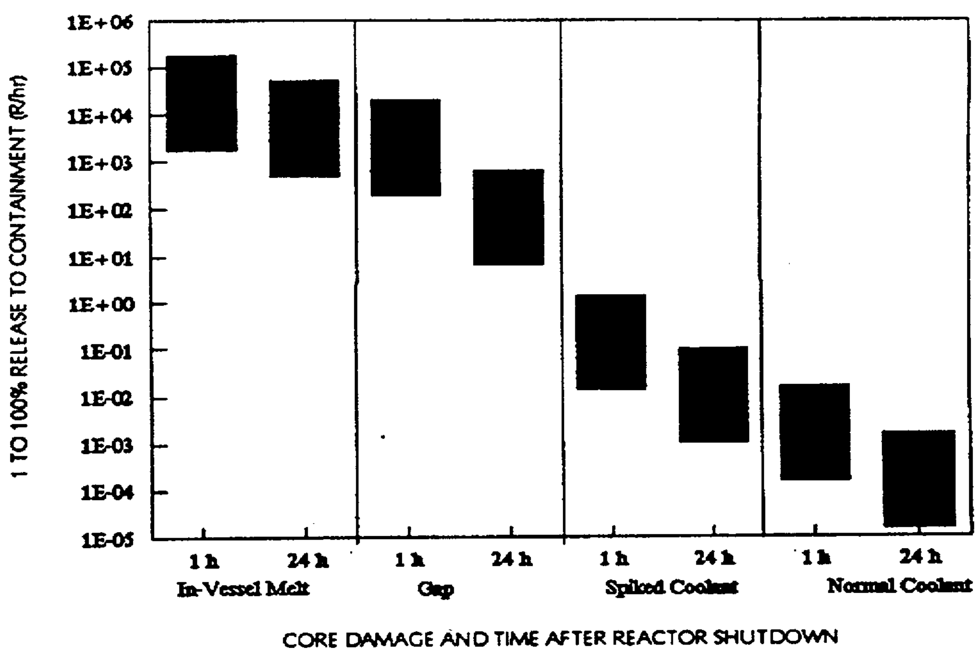


**Figure 2.7. BWR Mark III dry well containment monitor response (sprays off).**

*

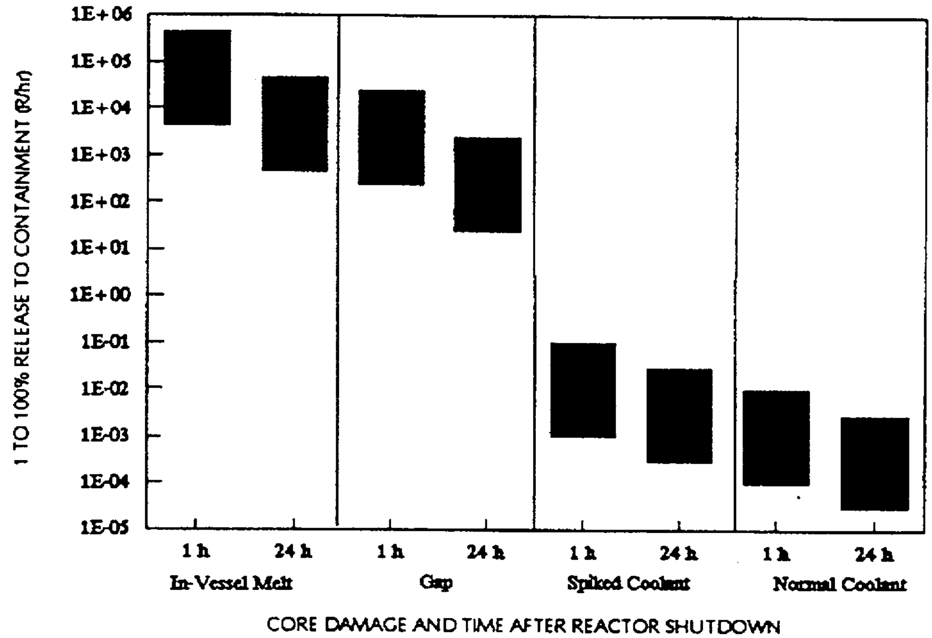


**Figure 2.8. BWR Mark III wet well containment monitor response.**

### 2.1.1.3 Coolant sample

When the user enters a coolant sample it completely defines the activity available for release. No further calculations are performed. It cannot be used to estimate core damage state. The activity can only be released through a steam generator tube rupture.

### 2.1.1.4 Containment air sample

When the user enters an air sample it completely defines the activity available for release. No further calculations are performed. It cannot be used to estimate core damage state. A containment air sample cannot be used to define the activity released in a steam generator tube rupture.

### 2.1.1.5 User-defined core damage estimate

The user can select a core damage state directly. The state may be from 1 to 100% of cladding failure or core melt or 100% vessel melt through. The core damage activity release fractions (ARFs) used are shown in Tables 2.3 and 2.4 (Soffer, 1992). The source term, $S_{ijk}$, is then computed as:

X

*

$$S_{ijk} = I_i \times ARF_{ij} \qquad (2\text{-}5)$$

ARF Nomenclature is the same as airborne release fraction on page 2.1. Nomenclature should have consistent definitions.

X

### 2.1.1.6 Effluent gross concentrations release rate

(Table 2.5)

The effluent gross concentrations release rate source term calculation is intended for assessments when the user can enter an activity release rate (i.e., a stack monitor reading) and a release duration and can estimate the radionuclide mix. The user can enter the estimated mix either of two ways: (1) as a core damage state or (2) as the percentages of the release that are represented by each of the eight STCP categories. When the user has selected the mix as a core damage state, then the mix can be modified by the effects of filters, sprays, retention in a pool, or 2 hrs of holdup. These factors will change the mix because they affect only the non-nobles. X

To compute the source term STDose first determines the source term that would result from the mix selected by the user for a reactor with a power of 1 MW(t) using the core release fractions from Tables 2.3.or 2.4 for (1) above or the fractions entered by the user for (2). The equation used is the same as Eq. (2-1), without the leakage fraction. STDose then scales all activities in the computed mix up (or down) so that they sum to the total amount of activity represented by the product of the release rate and duration that were entered by the user.

### 2.1.2 Spent Fuel Storage Accidents

Three source term scenarios are available for spent fuel: (1) pool storage - uncovered fuel, (2) pool storage - damaged assembly under water, and (3) dry storage cask event. The first scenario applies to undamaged fuel in a pool that is partially or totally drained. The fuel may be damaged by heating that results from the draining of the pool. The second scenario applies to mechanically damaged fuel that remains under water. The third scenario applies to fuel stored dry in casks. The two spent fuel pool scenarios are based on NUREG/CR-6451 (Travis, 1997). The dry cask scenario was developed by NRC staff for use in RASCAL 3.0. X

Need to document

To compute the amount of activity available for release, first the initial amount of activity in the fuel involved is determined. The activity is computed as the number of batches or assemblies times the fraction of a reactor core that they represent times the activity in a full reactor core. The core activity is computed as the reactor power entered times the inventory per MW(t) listed in Table 2.2 (USNRC, 1987). Then radiological decay is applied to that activity over the fuel storage duration. Finally, the amount of activity present is multiplied by the appropriate fuel release fractions, which are selected as described below.

STDose assumes that the fuel in a pool must be uncovered for more than two hrs in order for the fuel to reach 1200°F, which is the temperature at which cladding failures are expected. This assumption is based on the heat-up rate of 30-day-old fuel. It does not give credit for head removal by steam cooling and so would be conservative. No releases are projected if the fuel is uncovered for less than 2 hrs. After 2 hrs (at 1200°F) the fuel is projected to rupture due to internal pressure releasing the "hot gap" fraction of the fission products in the fuel. Only fuel that has been irradiated within the past year for a PWR and 180 days for a BWR is assumed to be able to reach 1200°F once uncovered. This is based on the observation that one-year-old fuel would require 10 hrs or more to reach 1200°F and with steam cooling is not expected to reach this temperature.

*

The user enters the number of fuel batches or assemblies that are one, two, and three years old. (No other fuel ages are allowed.) If the reactor is a PWR, the user enters whether the density of the fuel pool racking is high or low. The accident conditions that the user enters are: the fuel uncovery time, if and when the fuel was recovered, and if and when the pool was drained.

Fuel release fractions are shown in Table 2.6 (Travis, 1997) by STCP category. STDose determines which fuel release fractions to use based conditions in the pool, the fuel racking, and the fuel age. (1) If the pool is never totally drained, only fuel that has been in storage for less than one year is considered to be damaged and the release fractions used are those for a hot gap release. (2) In a PWR with high density pool racking or in a BWR, if the pool is totally drained for at least two hrs all fuel is damaged and the release fractions used are for a fire release. (3) In a PWR with low density pool racking if the pool is totally drained for at least two hrs a fire release occurs in all fuel that has been in storage for one year or less. Fuel that has been stored for more than one year is involved in a fire release only if there is at least one batch of fuel that has been in storage for one year or less. If all fuel is older than one year then hot gap release fractions are used for all the fuel.

X See Attachment 1 **Table 2.6. Fuel Release Fractions used in Spent Fuel Accidents***

| STCP category | Fire pool | Hot gap pool | Cold gap pool | Hot gap dry cask | Cold gap dry cask |
|---|---|---|---|---|---|
| Noble gases | 1 | 0.4 | 0.4 | 0.4 | 0.4 |
| Halogens | 0.7 | $3\times10^{2}$ | $3\times10^{-3}$ | $3\times10^{-2}$ | $3\times10^{-2}$ |
| Alkali metals | 0.5 | $3\times10^{-2}$ | $3\times10^{-3}$ | $3\times10^{-2}$ | $3\times10^{-3}$ |
| Tellurium group | $7\times10^{-3}$ | $1\times10^{-3}$ | $1\times10^{-4}$ | $1\times10^{-3}$ | $1\times10^{-4}$ |
| Barium, strontium | $7\times10^{-4}$ | $6\times10^{-6}$ | $6\times10^{-7}$ | $6\times10^{-6}$ | $6\times10^{-7}$ |
| Noble metals | $7\times10^{-6}$ | $6\times10^{-6}$ | $6\times10^{-7}$ | $6\times10^{-6}$ | $6\times10^{-7}$ |
| Cerium group | $2\times10^{-6}$ | $6\times10^{-6}$ | $6\times10^{-7}$ | $6\times10^{-6}$ | $6\times10^{-7}$ |
| Lanthanides | $2\times10^{-6}$ | $6\times10^{-6}$ | $6\times10^{-7}$ | $6\times10^{-6}$ | $6\times10^{-7}$ |

*From (Travis, 1997)

Fuel that is mechanically damaged under water results in a cold gap release. Fuel damaged under water may have been stored for any length of time, but is all considered to have the same storage duration.

For a dry storage cask event, major structural damage will cause a cold gap release. Loss of cooling for more than the thermal limit (24 hrs) causes a hot gap release. No other conditions can cause a release.

X

### 2.1.3 $UF_6$ Releases from Cylinders and Cascade Systems

Two types of source term scenarios are available that release $UF_6$: $UF_6$ cascade releases, and $UF_6$ cylinder releases. Cascade release only occur at gaseous diffusion plants. Cylinder releases can occur at any facility that stores $UF_6$.

In the cascade scenario the user enters the amount of $UF_6$ in the cascade in the pounds of $UF_6$ per cell and the number of units or cells in the cascade that are involved in the release or as a total mass. Ten cells per unit is the default. When the number of units or cells is entered, the amount of material available is the product of the number of units or cells times their inventory.

In the cylinder scenario the user enters the amount of $UF_6$ in a cylinder release as the mass of the available inventory or as the number of each 14-ton, 10-ton, and 2.5 ton cylinders involved. The 2.5 ton cylinder is assumed to contain 2277 kg $UF_6$, the 10-ton cylinder is assumed to contain 9539 kg, and the 14-ton cylinder 12338 kg. The amount of material is the sum of the number of each type of cylinder times its inventory.

In both cases the user enters the uranium enrichment level. STDose converts the mass of $UF_6$ to activity using the enrichment and the specific activity. Enrichment level affects the specific activity, SA, of the uranium according to the equations :

$$SA = E_U \times 0.01 \times SA_{U^{234}} + (100.-E_U) \times 0.01 \times SA_{U^{238}} \qquad (2\text{-}6)$$

where the enrichment of $U^{234}$ by weight, Eu is

$$E_U = (E_{U^{235}})^{1.4} \times 0.003 \qquad (2\text{-}7)$$

and

$E_U^{235}$ is the enrichment of $U^{235}$ by weight in percent.

RTM 96 has the same table and is included in the References

This relationship was developed based on unpublished data and is in good agreement with RTM~~98~~ 96 Table E-4 and 10 CFR Part 71, App. A. Table A-3. X

The user enters the maximum release rate and maximum release fraction from the cascade. STDose uses these to limit the amount of activity released is limited . If the release rate and duration chosen result in a release that is greater than the maximum release fraction, the total amount released is reduced to this value. No radiologic decay is computed for $UF_6$ accident scenarios.

### 2.1.4 Fires and Explosions Involving Uranium Oxide

In STDose accidents involving uranium oxide can be modeled as either fires or as explosions. For both the user enters the amount of material ($UO_2$) at risk and its enrichment level. For both the user chooses

X

Reference document refers to IF's AS RF's

among a variety of accident conditions. These accident conditions are listed in Tables 2.7 and 2.8. Each set of accident conditions is associated with a default set of airborne release fractions (ARFs) and inhalation fractions (IFs). The ARFs and IFs are considered to be conservative. The IF is the fraction of the material released that is expected to be inhaled and is defined as all vapors and any particulate material that has a diameter of less that 10 $\mu$m. (Note that the IFs are not used in the source term calculation. They are used in the calculation of inhalation dose to reduce the amount of material inhaled.) The default values for the ARFs and IFs are shown in Tables 2.7 and 2.8 (NRC, 1998). The selection of these parameters is discussed in that document.

The scenarios available in STDose were selected to span a wide range of possible accidents. Uranium oxide fires may occur in several different types of facilites. In the milling of uranium ore, a fire can occur in a drum of milled ore or in the process of extracting solvent. Once the ore is milled, the production of reactor fuel begins with creating a powder from the $UO_2$. Both wet and dry processes are used to produce this powder. Uranium-oxide-contaminated waste can be stored in several forms and any of these can be involved in a fire. Uranium oxide explosions are characterized as those caused by the detonation of high explosives in contact with the material, those that are caused by a fire (deflagration), and those that are caused by a sudden change in pressure in the container of the material (venting). The $UO_2$ in the explosion may be in liquid, solid, or powder form or may simply be surface contamination.

Y

**Table 2.7. ARFs and ~~R~~IFs used in Uranium Oxide Fires ***

| Type of fire | Condition | ARF | IF |
|---|---|---|---|
| Production process | Dry process | $1x10^{-3}$ | 1 |
| | Wet process | $3x10^{-5}$ | 1 |
| HEPA filter | At high temperature | $1x10^{-4}$ | 1 |
| | Failure | 1 | 1 |
| Incinerator exhaust | | $4x10^{-1}$ | 1 |
| Waste fire | Solid packaged in drums | $5x10^{-4}$ | 1 |
| | Solid loosely packed | $5x10^{-2}$ | 1 |
| | Combustible liquid | $3x10^{-2}$ | 1 |
| | Non-combustible liquid | $2x10^{-3}$ | 1 |
| Uranium mill | Drum in a fire | $1x10^{-3}$ | 1 |
| | Solvent extraction | $3x10^{-2}$ | 1 |

*From (NRC, 1998)

STDose calculates the source term by first converting the mass of $UO_2$ to U by multiplying by 0.88. Then the amount of material from mass to activity as described in the Section 2.1.3. The source term is the

X

product of this activity times the ARF value.

### 2.1.5 Criticality Accidents

A criticality accident results from the uncontrolled release of energy from an assemblage of fissile material. In RASCAL 3.0 a criticality accident may be modeled using the assumptions in NUREG-1320 (USNRC, 1998). In addition, the criticality data may be entered directly by the user. The physical systems modeled are listed in Table 2.9 (USNRC, 1998). This table also presents the assumed number of fissions in the first burst and the total yield. The assumed amounts of each radionuclide released per $10^{19}$ fissions (total released) are listed in Table 2.10 (USNRC, 1998). These values are based on ORIGEN (ORNL, 1989) calculations.

**Table 2.8. ARFs and IFs used in Uranium Explosions ***

| Explosion characteristics | Material form | ARF | IF |
|---|---|---|---|
| Detonation | Liquid | 1 | 1 |
| | Solid | 1 | 2x10$^{-1}$ |
| | Powder | 1 | 2x10$^{-1}$ |
| | Surface contamination | 1x10$^{-3}$ | 1 |
| Deflagration | Liquid | ~~1x10$^{-6}$~~ | 1 |
| | Solid | 0 | 0 |
| | Powder | 5x10$^{-3}$ | 1x10$^{-3}$ |
| | Surface contamination | 3x10$^{-1}$ | switch |
| Venting | Liquid | 2x10$^{-3}$ | 1 |
| | Solid | 0 | 0 |
| | Powder | 1x10$^{-1}$ | 7x10$^{-1}$ |
| | Surface contamination | 1x10$^{-3}$ | 1 |

4x10$^{-3}$

X

X

1.0

*From (NRC, 1998)

In the preset scenarios, the bursts are assumed to be 10 mins apart. The fission may not continue for more than 48 bursts. The number of fissions in all but the first burst is:

$$FB = \frac{Y_T - Y_I}{(48-1)} \tag{2-8}$$

where

$Y_T$ is the total yield of the criticality (column 3 in Table 2.9), and
$Y_I$ is the yield of the initial burst (column 2 in Table 2.9).

To calculate the source term, STDose first determines the initial activity present as the product of the yield of the first burst (in $10^{19}$ fissions) and the activity per $10^{19}$ fissions listed in Table 2.10. For each following time-step STDose (1) determines if the criticality is still occurring and if enough time has passed for one or more bursts to have occurred, and if so, adds the appropriate activity, (2) reduced the amount of activity for the amount released, and (3) applies radiological decay to the result. A criticality will end when either the total number of allowed bursts have been accounted for or when the 'end of criticality' time entered by the user has been reached. Note that if the user selects a release duration that is not long enough to include all 48 bursts, the total activity released will be less than the amount listed in Table 2.9.

For criticality accidents, the criticality shine dose is computed with the source term. The shielding thicknesses entered by the user are only used in this calculation. The dose in rem, $D_{crit}$, at 10 ft is computed as (Broadhead, et al, 1997):

$$D_{CRIT} = 1 \times 10^{-15} \times F \times e^{-(0.386 \times S + 0.147 \times W + 0.092 \times C)} + 1 \times 10^{-14} \times F \times e^{-(0.236 \times S + 0.240 \times W + 0.227 \times C)} \tag{2-9}$$

where

F is the total number of fissions,
S is the thickness of steel shielding in inches,
W is the thickness of water shielding in inches, and
C is the thickness of concrete shielding in inches.

Doses, $D_{criti}$, (rem) at other distances are computed as:

$$D_{criti} = \left(\frac{10}{D_i}\right)^2 \times D_{crit} \tag{2-10}$$

where

$D_i$ is the distance to the dose point in feet.

### 2.1.6 Isotopic Source Terms

Three types of isotopic source terms are available: (1) isotopic release rates, (2) isotopic concentrations, and (3) sources and materials in a fire. Note that the source terms labeled 'effluent isotopic release rates' and 'effluent isotope concentrations' perform the same calculations as (1) and (2), respectively. Uranium oxide fires are discussed in Section 2.1.4.

**Table 2.9. Data used in Criticality Calculations***

| System modeled in the scenario | Initial burst yield (fissions) | Total yield (fissions) |
|---|---|---|
| Solution < 100 gal | $1 \times 10^{17}$ | $3 \times 10^{18}$ |
| Solution > 100 gal | $1 \times 10^{18}$ | $3 \times 10^{19}$ |
| Liquid / powder | $3 \times 10^{20}$ | $3 \times 10^{20}$ |
| Liquid / metal pieces | $3 \times 10^{18}$ | $1 \times 10^{19}$ |
| Solid uranium | $3 \times 10^{19}$ | $3 \times 10^{19}$ |
| Solid plutonium | $1 \times 10^{18}$ | $1 \times 10^{18}$ |
| Large storage arrays below prompt critical | None | $1 \times 10^{19}$ |
| Large storage arrays above prompt critical | $3 \times 10^{22}$ | $3 \times 10^{22}$ |

*From (NRC, 1998)

Isotopic release rates are simply used as entered. The user is not given the opportunity to enter reduction factors since this option assumes that the user knows what is being released. Up to three sets of rates may be entered along with their start and end times.

When isotopic concentrations are entered, the user enters an overall release rate as well. The amount of activity released is computed as the concentration times the rate times the release duration. Up to three sets of concentrations and rates may be entered. The isotopic concentrations may be decayed over a selected time period.

release fractions (ARFs in Eq. 2-1)

In a fire release, the user enters the amount of radionuclides present. These amounts are reduced by the fire ~~reduction factors~~ listed in RTM-96 (McKenna et al., 1996). No release occurs when the fire is not burning and no other types of reduction are allowed. A fire may only start and stop burning once. The user can select fire release fractions by element, by the form of the compound, or may enter them directly. The default fire release fractions used are shown in Tables 2.11 and 2.12. Note that the total amount of activity released also depends on the release duration entered in the isotopic release pathway form. For example, if the release duration is shorter than the duration of the fire, the amount of activity released is reduced.

**Table 2.10. Activity (Ci) Released in Criticality of $10^{19}$ Fissions**

| Radionuclide | Activity (Ci) | Radionuclide | Activity (Ci) |
|---|---|---|---|
| Kr-83m | 1.5E2 | I-131 | 7.3E0 |
| Kr-85m | 8.9E1 | I-132 | 1.0E3 |
| Kr-85 | 1.3E-5 | I-133 | 1.7E2 |
| Kr-87 | 1.1E3 | I-134 | 4.2E3 |
| Kr-88 | 6.6E2 | I-135 | 5.0E2 |
| Kr-89 | 4.6E4 | Sr-91 | 3.2E2 |
| Xe-133m | 1.9E-2 | Sr-92 | 1.2E3 |
| Xe-133 | 2.7E-3 | Ru-106 | 2.0E-2 |
| Xe-135m | 3.3E2 | Cs-137 | 1.0E-2 |
| Xe-135 | 5.2E0 | Ba-139 | 2.4E3 |
| Xe-137 | 2.4E4 | Ba-140 | 1.1E1 |
| Xe-138 | 1.0E4 | Ce-143 | 1.0E2 |

*From (NRC, 1998)

Note that when the user selects release units in mass, rather than activity, the source term is converted to Curies using the specific activity of each radionuclide. The user may specify the enrichment level for enriched uranium. The enrichment level for natural uranium is assumed to be 0.7% (RTM-96, Table E-5). Specific activity is computed as described in Eq. (2-6). For natural and enriched uranium radiologic decay and dose are calculated assuming the properties of $U^{238}$ and $U^{234}$, respectively.

**Table 2.11. Fire Release Fractions by Compound Form***

| Form of compound in Fire | Release fraction |
|---|---|
| Noble gas | 1.0 |
| Very mobile form | 1.0 |
| Volatile or combustible compound | 0.5 |
| Carbon | 0.01 |
| Semi-volatile compound | 0.01 |
| Non-volatile compound | 0.001 |
| U and Pu metal | 0.001 |
| Non-volatile in a flammable liquid | 0.005 |
| Non-volatile in a non-flammable liquid | 0.001 |
| Non-volatile solid | 0.0001 |

*Table F-2 from (McKenna et al., 1996)

## 2.2 CALCULATION OF RELEASE REDUCTION FOR EACH RELEASE PATHWAY

Most accident release pathways allow consideration of the reduction due to the physical systems that are present. The release pathways limit the available reduction mechanisms and the way in which the leak rate can be defined. In nuclear power plant releases, the reduction mechanisms available are dependent on the source term calculation type and the release pathway chosen. Each of the release pathways and reduction mechanisms, and the data used for them, are discussed in more detail below. Seven types of release pathways have been defined. Table 2.13 shows which reduction mechanisms are available for each pathway.

release fractions

Reduction factors (RDF in Eq. 2-1) are applied as scaling factors, as described in NUREG-1228 (McKenna and Giitter, 1988). All reduction factors, except fire reduction, are defined for three categories: noble gases (STCP category 1), halogens (STCP category 2), and other particles. Fire release fractions are element-specific. The fire ~~reduction factors~~ are shown in Tables 2.11 and 2.12 and are from NUREG-1140. The default values for all other reduction factors and their references are included in Table 2.14. In some cases the user can turn reduction mechanisms on and off as the accident progresses. Also, the reduction factors themselves may be modified for filters and for 'other' reduction.

Table 2.12. **Fire Release Fractions by Element***

| Element | Release Fraction | Element | Release Fraction | Element | Release Fraction | Element | Release Fraction |
|---|---|---|---|---|---|---|---|
| H(gas) | 0.5 | Se | 0.01 | I | 0.5 | W | 0.01 |
| C | 0.01 | Kr | 1.0 | Xe | 1.0 | Ir | 0.001 |
| Na | 0.01 | Rb | 0.01 | Cs | 0.01 | Au | 0.01 |
| P | 0.5 | Sr | 0.01 | Ba | 0.01 | Hg | 0.01 |
| S | 0.5 | Y | 0.01 | La | 0.01 | Tl | 0.01 |
| Cl | 0.5 | Zr | 0.01 | Ce | 0.01 | Pb | 0.01 |
| K | 0.01 | Nb | 0.01 | Pr | 0.01 | Bi | 0.01 |
| Ca | 0.01 | Mo | 0.01 | Pm | 0.01 | Po | 0.01 |
| Sc | 0.01 | Tc | 0.01 | Sm | 0..01 | Ra | 0.001 |
| Ti | 0.01 | Ru | 0.1 | Eu | 0.01 | Ac | 0.001 |
| V | 0.01 | Rh | 0.01 | Gd | 0.01 | Th | 0.001 |
| Cr | 0.01 | Ag | 0.01 | Tb | 0.01 | Pa | 0.001 |
| Mn | 0.01 | Cd | 0.01 | Ho | 0.01 | U | 0.001 |
| Fe | 0.01 | In | 0.01 | Tm | 0.01 | Np | 0.001 |
| Co | 0.001 | Sn | 0.01 | Yb | 0.01 | Pu | 0.001 |
| Zn | 0.01 | Sb | 0.01 | Hf | 0.01 | Am | 0.001 |
| Ge | 0.01 | Te | 0.01 | Ta | 0.001 | | |

*Table F-3 from (McKenna et al., 1996). The release fraction for ruthenium was changed from the value of 0.01 in NUREG-1140 to a value of 0.1. NUREG-1140 assumed that ruthenium was non-volatile. However, more recent research in NUREG/CR-6218, "A Review of the Technical Issue of Air Ingression During Severe Reactor Accidents," 1994, indicates (in Table 5) that at high temperatures ruthenium starts to become volatile. The ruthenium release fraction of 0.1 is less than the value of 0.5 used in NUREG-1140 for volatile compounds because ruthenium is less volatile than those other volatile compounds becoming highly volatile only at temperatures not normally reached in building fires.

STDose reactor source term calculations include a maximum effectiveness for sprays and a maximum effectiveness for all reduction, excluding filters. For each, the appropriate reduction factor or product of reduction factors computed at each time-step is compared to the maximum and is not allow to surpass it.

Steam generator condition and fire release fractions do not affect the total amount of activity present; they only reduce the amount of activity released. Activity removed by all other mechanisms is made permanently unavailable for release. Note that sprays and holdup will not be applied simultaneously and that hold-up only applies in large containment or auxiliary buildings. Note also that natural processes only apply to bypass accidents and will not be combined with any other reduction mechanism.

All reduction mechanisms are ~~be~~ characterized as constant factor except for the reduction due to the action of sprays and holdup in the containment. The RDFs for sprays and holdup are computed at each time step as an exponential function of time. Two factors each are used, an 'initial $\lambda_I$' which is only applied for the initial time interval listed in Table 2.14 and a "continuing" $\lambda_C$ which is applied thereafter. The reduction factors used for sprays and holdup are dependent on the total amount of time that each mechanism has been acting. So initially the RDF for sprays or holdup is:

There is no initial time interval specified in Table 2.14

$$RDF = e^{-\lambda_I T} \tag{2-11}$$

where

T is the total amount of time the sprays or holdup have been acting.

Once the amount of time that the reduction mechanism is active has passed the initial time interval, $T_I$, the RDF is:

$$RDF = e^{-\lambda_I T_I} + e^{-\lambda_C (T - T_I)} \tag{2-12}$$

Since the user can enter reduction data that changes with time, it is possible to turn the sprays on and off several times. The initial spray $\lambda_I$ applies to (1) all the activity in containment the first time the sprays are turned on and to (2) all the activity that enters the containment the first time that sprays are active. If the sprays are turned off and then turned back on, only the continuing $\lambda_C$ applies. The initial $\lambda_I$ for holdup only applies if the sprays were never turned on. Otherwise the continuing $\lambda_C$ applies.

## 2.3 CALCULATION OF LEAKAGE FACTORS

Four types of leak rates are used in RASCAL 3.0. They are: (1) percent of total amount per unit time, (2) based on containment pressure and hole size, (3) flow rate (volume or mass per unit time), and (4) a "direct" release, with all activity released during the selected release duration. The types of flow rates available in each type of release pathway are shown in Table 2.15. Because the atmospheric transport models expect the source term to be in terms of a rate, rather than a total amount, the leakage data that the user enters is always converted to a leakage rate fraction (LRF) by STDose.

**Table 2.13 Reduction Factors Available in each Release Pathway**

| Reduction mechanism \ Pathway | Isotopic | Containment leakage | Containment bypass | Wet well leakage | Dry well leakage | Steam generator tube rupture | Spent fuel | $UF_6$ cylinder | Cascade | Criticality | Fire or explosion |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Fire | X[1] | | | | | | | | | | |
| Filters | | | X[2] | X | X | | X[3] | X | | | X |
| Sprays | | X | X[2] | | X | | | | | | X |
| Containment holdup | | X | | X | X | | | | | | |
| Natural processes | | | X[4] | | | | | | | | |
| Sub-cooled or saturated pool | | | | X | | | | | | | |
| Ice condenser | | X[5] | | | | | | | | | |
| Steam generator - partitioned or not partitioned | | | | | | X[6] | | | | | |
| Steam jet air ejector release | | | | | | X[6] | | | | | |
| Building retention | | | | | | | | X | X | | X |
| Other | | | | | | | | | | X | |

[1] Isotopes and materials in a fire only
[2] Only when through a building
[3] Not for dry cask event
[4] Only if no other reduction is active
[5] PWR with ice condenser only
[6] PWR only

X

**Table 2.14 Reduction Factors**

| Reduction | Noble gas | Halogen | Other | Ref. NUREG |
|---|---|---|---|---|
| Filters | 1 | 0.01 | 0.01 | 1228 |
| Sub-cooled pool | 1 | 0.01 | 0.01 | 1228 |
| Saturated pool | 1 | 0.05 | 0.05 | 1228 |
| Sprays (exponential time dependence) | $e^{0t} = 1$ | hr factor 0.25 $e^{12t}$ > $e^{0.2t}$ | hr factor 0.25 $e^{12t}$ > $e^{0.2t}$ | /CR-4722 (Fig. 5) |
| Containment holdup (exponential time dependence) | $e^{0t} = 1$ | hr factor 2 $e^{1.2t}$ > $e^{0.15t}$ | hr factor 2 $e^{1.2t}$ > $e^{0.15t}$ | 1150 App. B |
| Natural processes | 1 | 0.4 | 0.4 | 1228 |
| Ice condenser - no fans or re-circulation | 1 | 0.5 | 0.5 | 1228 |
| Ice condenser - 1 h or more re-circulation | 1 | 0.25 | 0.25 | 1228 |
| Steam generator tube rupture - to secondary * | 1 | 0.02 | 0.02 | 1228 |
| Steam generator tube rupture - not partitioned or solid (secondary side "U" tube (liquid release)) | 1 | 0.5 | 0.5 | 1228 |
| Steam generator tube rupture - steam jet air ejector release ** | 1 | 0.5 | 0.5 | 1228 |
| Other reduction | 1 | 1 | 1 | 1228 |
| *Minimum reduction, except for filters* | *0* | *0.001* | *0.001* | *1228* |
| *Minimum reduction, sprays only* | *0* | *0.03* | *0.03* | *1228* |

X X X

* Normal partially filled "U" tube (liquid release)

** Normal once-through

Could not determine where these numbers are in Reference

**Table 2.15 Leak Rate Types Available in each Release Pathway**

| Pathway | %/hr | Pressure and hole size | cfm, kg/s, gal/m | Direct |
|---|---|---|---|---|
| Bypass | X | | X | |
| Containment leakage | X | X | | |
| Wet eell | X | X | X | |
| Dry eell | X | X | | |
| Spent fuel | X | | | |
| Steam generator tube rupture | | | X | X |
| $UF_6$ cylinder | X | | X | |
| $UF_6$ cascade | | | X | X |
| Criticality | X | | | |
| $UO_2$ fire or explosion | | | | X |
| Isotopic | | | X | X |

### 2.3.1 Percent per hour

The %/hr release rate releases a fixed fraction of the source term per unit time. The actual leakage rate fraction used is computed based on the length of the time step entered by the user. This release rate results in the source term being released at a constant rate over time.

### 2.3.2 Containment pressure and hole size

The containment pressure and hole size option is only available for a facility with a defined design pressure and containment volume. In this case, the equation (Blevins, 1984) for the mass flow rate, MFR, is:

$$MFR = C\left(\frac{\pi D^2}{4}\right)\sqrt{2\rho\,(P_1 - P_2)g} \tag{2-11}$$

where

C is 0.63,
D is hole diameter,

ρ is density of air,
$P_1$ is pressure in containment,
$P_2$ is atmospheric pressure, and
g is acceleration of gravity.

The leakage rate fraction LRF of the containment volume atmosphere released, then is

$$LRF = \frac{MFR \times D_R}{\rho V_C} \quad (2\text{-}12)$$

where

$D_R$ is the release duration and
$V_C$ is the containment volume.

When the user enters a containment pressure is less than the design pressure, STDose sets LF to zero. STDose does not compute the change in containment pressure. The user may enter changing containment pressures as the assessment proceeds.

### 2.3.3 Flow rate

When a flow rate is entered by the user, the leakage rate fraction is computed as the flow rate divided by the total volume that contains the source term. This volume may be the containment volume, a building volume, or the volume of coolant, depending on the source term type.

### 2.3.4 Direct

In a 'direct' release, the user only enters the release duration. The leakage rate fraction is computed differently, depending on the source term type. For a steam generator tube rupture, a release rate of 100%/hr is assumed. For the $UF_6$ cascade and the $UO_2$ fire or explosion, the leakage rate fraction is set at 100% divided by the release duration. For isotopic releases the release rates, those rates are used as entered by the user. No leakage rate fraction is needed, so none is computed.

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