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☐ New ☐ Total Rewrite ☐ Revision ☒ Page Change

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## PUBLICATION RECORD

| EFFECTIVE DATE | REVISION NUMBER | DESCRIPTION |
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| 05/11/2007 | 00 | Approved new technical information bulletin to provide guidance on the assignment of radionuclide-specific intakes of mixed fission and activation products when air sampling or urinalysis data associated with reactors or reactor fuels are available only as gross or total beta activity or gross or total gamma activity. Incorporates formal internal and NIOSH review comments. Corrected editing in Tables G-1 through G-4. Added examples in a new Attachment H. Added Attributions and Annotations section. There is no change to the assigned dose and no PER is required. Training required: as determined by the Task Manager. Initiated by David W. Hearnsberger and Cindy W. Bloom.<br><br>Signature on File 04/18/2007<br>Donald E. Bihl, Document Owner<br><br>Signature on File 04/18/2007<br>John M. Byrne, Task 3 Manager<br><br>Keith A. McCartney Signature on File for 04/18/2007<br>Edward F. Maher, Task 5 Manager<br><br>Signature on File 04/19/2007<br>Kate Kimpan, Project Director<br><br>Brant A. Ulsh Signature on File for 05/11/2007<br>James W. Neton, Associate Director for Science |
| 11/19/2007 | 00 PC-1 | Approved page change revision initiated to add Information regarding the distribution to be used in IREP in Section 8.0 on pages 29 and 30. This information is now covered in added Sections 8.1 and 8.2. Added reference on page 32. No sections were deleted. Incorporates formal internal and NIOSH review comments. Training required: as determined by the Task Manager. Initiated by Elizabeth M. Brackett. |

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## ACRONYMS AND ABBREVIATIONS

| | |
|---|---|
| ATR | Advanced Test Reactor |
| CANDU | Canada deuterium–uranium (reactor) |
| cm | centimeter |
| d | day |
| DOE | U.S. Department of Energy |
| dpm | disintegrations per minute |
| FFTF | Fast Flux Test Facility |
| g | gram |
| h | hour |
| ICRP | International Commission on Radiological Protection |
| IMBA | Integrated Modules for Bioassay Analysis |
| in. | inch |
| IRF | intake retention fraction |
| $m^3$ | cubic meter |
| MAP | mixed activation product |
| MeV | megaelectron-volt, 1 million electron-volts |
| MFP | mixed fission product |
| mL | milliliter |
| MTHM | metric tons of heavy metal |
| MW | megawatt |
| MWd | megawatt-day |
| NIOSH | National Institute for Occupational Safety and Health |
| pCi | picocurie |
| PWR | pressurized-water reactor |
| RaLa | radioactive lanthanum |
| s | second |
| SS | stainless steel |
| TIB | technical information bulletin |
| TRIGA | Training, Research, Isotopes, General Atomics |
| U.S.C. | United States Code |
| wt% | weight percent |
| y | year |
| µCi | microcurie |
| *Z* | atomic number |

§ section or sections

## 1.0 INTRODUCTION

Technical information bulletins (TIBs) are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historic background information and guidance to assist in the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected sites(s). TIBs may be used to assist NIOSH staff in the completion of individual dose reconstructions.

In this document the word “facility” is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an “atomic weapons employer facility” or a “Department of Energy [DOE] facility” as defined in the Energy Employees Occupational Illness Compensation Program Act of 2000 [42 U.S.C. § 7384l(5) and (12)].

## 2.0 PURPOSE

The purpose of this document is to provide guidance on the assignment of radionuclide-specific intakes of mixed fission and activation products when air sampling or urinalysis data associated with reactors or reactor fuels are available only as gross or total beta activity or gross or total gamma activity. This document standardizes the approaches to fission and activation product interpretation, and is for dose reconstruction use to address reactor-related source terms when site-specific guidance is not available.

## 3.0 SCOPE

The guidance in this document applies, when radionuclide-specific information is lacking, to intakes of fission and activation products associated with most reactor operations, destructive fuel examination, fuel dissolution, and high-level waste management. This document provides default source terms for the case when no site-specific information is available. The source terms were derived to address gross beta or gross gamma results specifically from air samples or urinalyses, but the source terms can also be used with other monitoring methods (e.g., *in vivo* or fecal bioassay) or for cases in which not all radionuclides of concern were measured.

The guidance does not address predominately alpha-emitting radionuclides in irradiated fuel (although a similar approach could be taken to address exposure to alpha emitters at sites where monitoring was not radionuclide-specific or the source term was not well defined). It does not include radionuclides generated outside the fuel and its cladding (e.g., tritium or $^{24}$Na in reactor coolant, europium from a samarium poison system).

This guidance applies to a broad scope of reactor operations including plutonium production reactors (low enrichment, low burnup, Zircaloy or aluminum cladding), research reactors [modest enrichment, modest burnup, stainless-steel (SS) or Zircaloy cladding; e.g., Training, Research, Isotope General Atomics (TRIGA) reactors], high-enrichment, high-burnup reactors [e.g., Idaho National Laboratory’s Advanced Test Reactor (ATR), fuel from naval reactors], and fast breeder reactors [e.g., Hanford’s Fast Flux Test Reactor (FFTF), Argonne National Laboratory–West’s experimental breeder reactors]. It does not apply to radioactive lanthanum (RaLa) operations, which involved very short-cooled fuel but was performed in hot cells. It also does not apply to determination of intakes where radionuclides have been purposely extracted and concentrated as for heat generation sources, medical uses, or waste handling operations that caused significant alteration to the source term to which workers were exposed.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 9.0.

## 4.0 BACKGROUND

Exposure to fission and activation products was monitored by combinations of workplace or personal surveys, air sampling, and/or bioassay. Often the records do not have sufficient information to link the results of the monitoring to specific radionuclides. For instance, air samples and urine samples were often measured using techniques that measured a single radiation type within a large range of energies. These were usually referred to as gross beta or gross gamma results. These techniques or results were called by a variety of names, such as gross beta, total beta, gross gamma, total gamma, mixed fission product (MFP), fission product, or mixed activation product (MAP). Interpretation of the results of these gross analyses of air samples requires knowledge of the activity ratios of the various fission and activation products in the air breathed by the worker and the beta or gamma yield of each product as measured by the detector. Interpretation of gross urinalysis results requires knowledge of the ratios of the products in urine at some time after the start of chronic intake and whether those ratios changed due to sample processing activities.

## 5.0 REACTOR MODELING AND DETERMINATION OF FISSION AND ACTIVATION PRODUCT ACTIVITIES

This section describes the derivation of the reactor source terms.

### 5.1 REACTOR SELECTION AND MODELING

Five basic categories of nuclear reactors were selected to represent the sources of fission and activation products encountered by workers at DOE and contractor sites. These were:

- Plutonium production reactors
- Experimental sodium-cooled reactors
- Advanced test reactors (high-enrichment uranium)
- Research reactors
- A generic reactor

At least one reactor from each of these categories was then selected for fuel irradiation calculations using the ORIGEN2 isotope generation and depletion code (Croff 1980). More than one reactor or one type of fuel-with-cladding was selected for some of the categories. A total of seven cases were considered, as listed in Table 5-1.

Table 5-1. Reactors used to represent categories.

| Reactor | Category |
|---|---|
| Hanford N Reactor | Plutonium production reactors |
| Hanford single-pass reactors | Plutonium production reactors |
| FFTF | Experimental sodium-cooled reactors |
| ATR | Advanced test reactors |
| TRIGA reactor (Al-clad fuel) | Research reactors |
| TRIGA reactor (SS-clad fuel) | Research reactors |
| Pressurized-water reactor (PWR) | Generic reactor |

Eleven ORIGEN2 runs were performed for the seven reactor cases above. The cases for the production reactors and the TRIGA reactors were run with more than one ORIGEN2 cross-section and fission product yield library because specific libraries for these cases were not available. Each of the production and TRIGA reactor cases was run with two cross-section and fission product yield libraries to evaluate the choice of library on the inventory results. All cases were decayed for intervals of 10 days, 40 days, 60 days, 90 days, 180 days, 1 year, and 3 years after irradiation, because this was judged by the authors to be representative of the compositions of typical source terms encountered by workers; this is discussed further in Section 5.4.

The ORIGEN2 cross-section and fission product yield libraries are single-group values determined for typical neutron spectra for a given core design at a number of different burnups. They provide a simulation of the burnup, depletion, irradiation, and decay of nuclear fuel and associated structural materials based on a generic fuel cycle (Ludwig and Renier 1989). The libraries are best suited for use in computing average properties for an entire core.

The ORIGEN2 cross-section and fission product yield libraries used for each case were selected to match fuel enrichment as closely as possible. For the ATR, the FFTF, and the pressurized-water reactor (PWR), this analysis used the libraries specifically developed for those reactors. For the other cases, pairs of libraries were selected from the standard models available (i.e., those distributed with ORIGEN2) on the basis of the best matches in terms of uranium enrichment. Thus, the N Reactor, single-pass reactor, and TRIGA reactor cases were run twice using different libraries to effectively bound the irradiation results for comparison of the various reactors.

The materials irradiated in the ORIGEN2 cases for the various reactor types were those from active fuel regions only. These consisted of the elemental composition of the fuel cladding, including trace constituents, and the elemental composition of the fuel (heavy metal and impurities). Materials from low-flux regions such as the ends of fuel pins were not included. Material compositions were specified on the basis of a single fuel element or per metric ton of fuel. Burnup and power density were selected accordingly (see Section 5.2). The fuel–cladding composition for a given reactor was irradiated at constant power for the time required to give the desired burnup. No decay (shutdown) intervals or other power history considerations were included. Activity ratios for short-lived radionuclides are therefore generally overstated.

The results from the 11 ORIGEN2 runs were compared on the basis of relative activity (to Cs-137) at 10 days of decay [1]. Fission and activation product inventories from the 11 runs were compared separately so any dependence on the choice of cross-section libraries could be evaluated independently for these different modes of production. Each ORIGEN2 run gave activity (in curies) for 879 fission product nuclides and 688 activation product nuclides. These include some stable species, and many of these nuclides appear in both categories (i.e., they are produced by both fission and activation).

Based on comparison of the fission and activation product relative activity data for the 11 ORIGEN2 runs, four cases were selected as having isotopic inventories that would be representative for the variety of reactors and fuel combinations:

- The ATR
- The FFTF
- The N Reactor (run with a PWR cross-section library)
- The TRIGA reactor (with SS cladding; also run with a PWR cross-section library)

The fission and activation product relative inventory data for these four runs were summed, as appropriate, to yield tables of relative activity for 1,192 nuclides for the seven decay times.

Section 5.2 details the ORIGEN2 calculations for each of the 11 ORIGEN2 runs.

## 5.2 ORIGEN2 RUNS FOR VARIOUS REACTORS

### 5.2.1 Advanced Test Reactor

The ATR is a light-water reactor designed specifically to study the effects of intense radiation on reactor fuels and other materials (DOE 2003). It uses highly enriched (nominally 93.15 wt%) uranium fuel irradiated to very high burnups. The ATR uses plate-type fuel elements. The core is serpentine, so the fuel elements consist of curved plates. The fuel meat is an aluminum-uranium compound clad with 6061T aluminum. The aluminum in the fuel meat is a higher purity Type 1100.

The composition used in the ORIGEN2 run for the ATR represented one fuel element that contained 1,151.33 g of 93.2 wt%-enriched uranium. The heavy-metal composition was:

U-234 = 12.29 g
U-235 = 1,073.00 g
U-236 = 2.40 g
U-238 = 63.64 g

An ATR fuel element contains 1,174.32 g of Type 1100 aluminum and 7,764.10 g of 6061T aluminum (DOE 2003). The composition data in Table 5-2 were obtained with the use of the material constituent and impurity concentration data for these materials in *Source Term Estimates for DOE Spent Nuclear Fuels* (DOE 2003).

Table 5-2. Elemental composition data for ATR.

| Total aluminum | | |
|---|---|---|
| **Element** | ***Z*** | **Mass (g)** |
| H | 1 | 1.66 |
| C | 6 | 1.66 |
| O | 8 | 1.66 |
| Mg | 12 | 77.64 |
| Al | 13 | 8,709.31 |
| Si | 14 | 49.52 |
| Ti | 22 | 5.82 |
| Cr | 24 | 15.14 |
| Mn | 25 | 6.12 |
| Fe | 26 | 30.11 |
| Ni | 28 | 1.66 |
| Cu | 29 | 22.82 |
| Zn | 30 | 10.29 |
| Zr | 40 | 1.66 |
| Sn | 50 | 1.66 |
| Pb | 82 | 1.66 |

These aluminum and heavy-metal composition data made up the total composition used in the ORIGEN2 run for the ATR. An upper-end burnup of 314,683 MWd/MTHM (DOE 2003) was used, which equates to 362.3 MWd for the fuel element defined above. The composition was therefore irradiated for 36.23 days at a constant power of 10 MW in increments of 10 days. The cross-section and fission product yield library was developed specifically for the ATR by the Idaho National Laboratory. However, it should be noted that this library is not reactor-specific for activation products.

Rather, it uses the activation product cross sections from the standard-burnup PWR library supplied with ORIGEN2. The cross sections for actinides and fission products were developed specifically for the ATR.

### 5.2.2 Fast Flux Test Facility

The FFTF was a liquid-sodium-cooled fast reactor using mixed oxide fuel. The driver fuel was a mixture of depleted uranium and plutonium. The uranium contained 0.2 wt% U-235 and the plutonium was 86 wt% Pu-239. The plutonium fraction was 29 wt% of the heavy metal (i.e., uranium plus plutonium) (DOE 2003). The driver assemblies were clad with 316 SS. The assemblies were generally run to high burnups, which is typical for high-value fuels. A representative burnup of 80,000 MWd/MTHM was selected for this case.

The composition used in the ORIGEN2 run for the FFTF represented one fuel element that contained 32,918.1 g of heavy metal. Table 5-3 lists the specific heavy-metal composition (DOE 2003). One FFTF driver assembly contains 21,327.8 g of 316 SS (DOE 2003). The composition data in Table 5-4 were obtained using the material constituent and impurity concentration data for 316 SS in *Source Term Estimates for DOE Spent Nuclear Fuels* (DOE 2003). The oxygen content of the mixed oxide fuel is also accounted for here.

Table 5-3. Heavy-metal composition data for FFTF.

| Nuclide | Mass (g) |
|---|---|
| Pu-239 | 8,382.9 |
| Pu-240 | 1,162.5 |
| Pu-241 | 115.4 |
| Pu-242 | 18.5 |
| Am-241 | 18.5 |
| U-235 | 49.5 |
| U-238 | 23,170.8 |

Table 5-4. Elemental composition data for FFTF.

| Element | *Z* | Weight fraction | Mass (g) |
|---|---|---|---|
| B | 5 | 0.00002 | 0.43 |
| C | 6 | 0.0006 | 12.80 |
| N | 7 | 0.0001 | 2.13 |
| O | 8 | N/A | 4,347.6 |
| Al | 13 | 0.0005 | 10.66 |
| Si | 14 | 0.0075 | 159.96 |
| P | 15 | 0.0004 | 8.53 |
| S | 16 | 0.0001 | 2.13 |
| V | 23 | 0.0004 | 8.53 |
| Cr | 24 | 0.18 | 3,839.00 |
| Mn | 25 | 0.02 | 426.56 |
| Fe | 26 | 0.61848 | 13,190.82 |
| Co | 27 | 0.0005 | 10.66 |
| Ni | 28 | 0.14 | 2,985.89 |
| Cu | 29 | 0.0004 | 8.53 |
| As | 33 | 0.0003 | 6.40 |
| Nb | 41 | 0.0005 | 10.66 |
| Mo | 42 | 0.03 | 639.83 |
| Ta | 73 | 0.0002 | 4.27 |

These composition data for the fuel and cladding made up the total composition used in the ORIGEN2 run for the FFTF. A representative burnup of 80,000 MWd/MTHM was selected, which equates to 2,633.4 MWd for the fuel assembly defined above. The composition was therefore irradiated for 487.7 days at a constant power of 5.4 MW in increments of 50 days. The cross-section and fission product yield library was developed specifically for the FFTF and was supplied with the ORIGEN2 code.

### 5.2.3 Hanford N Reactor

The Hanford N Reactor was a water-cooled, graphite-moderated plutonium production reactor using slightly enriched uranium metal fuel. The fuel considered here was the Mark IV variety, which consisted of two concentric, annular fuel elements, referred to as the inner and outer elements. In the Mark IV assembly, each element was 0.95 wt% U-235. Both elements were clad in Zircaloy 2. The fuel was irradiated to low burnup to optimize the plutonium isotopic content. A typical burnup for an inner Mark IV element was 913 MWd/MTHM. For the outer element, a typical value was 1,188 MWd/MTHM. These burnups nominally gave a discharge Pu-240 content of 6% (Schwarz 1997).

The composition used in the ORIGEN2 run for the N Reactor represented one metric ton of fuel. Table 5-5 lists the composition data for the Zircaloy 2 cladding and trace impurities in the fuel. The heavy-metal composition was (Schwarz 1997):

U-234 = 70.0 g
U-235 = 9,470.0 g
U-236 = 500.0 g
U-238 = 989,960.0 g

Table 5-5. Elemental composition data for N Reactor.

| Element | Z | Zircaloy (g/MTHM) | Trace (g/MTHM) | Total (g/MTHM) |
|---|---|---|---|---|
| H | 1 | 1.8 | 2 | 3.8 |
| Be | 4 | - | 10 | 10.0 |
| B | 5 | 0.04 | 0.25 | 0.3 |
| C | 6 | 19.3 | 550 | 569.3 |
| N | 7 | 5.6 | 75 | 80.6 |
| Na | 11 | 1.4 | - | 1.4 |
| Mg | 12 | 1.4 | 25 | 26.4 |
| Al | 13 | 5.3 | 800 | 805.3 |
| Si | 14 | 7.0 | 124 | 131.0 |
| Ti | 22 | 3.5 | - | 3.5 |
| V | 23 | 3.5 | - | 3.5 |
| Cr | 24 | 70.4 | 65 | 135.4 |
| Mn | 25 | 3.5 | 25 | 28.5 |
| Fe | 26 | 95.0 | 350 | 445.0 |
| Co | 27 | 0.7 | - | 0.7 |
| Ni | 28 | 38.7 | 100 | 138.7 |
| Cu | 29 | 3.5 | 75 | 78.5 |
| Zr | 40 | 69,041.5 | 65 | 69,106.5 |
| Mo | 42 | 3.5 | - | 3.5 |
| Cd | 48 | 0.04 | 0.25 | 0.3 |
| Sn | 50 | 1,020.1 | - | 1,020.1 |
| Hf | 72 | 14.1 | - | 14.1 |
| W | 74 | 3.5 | - | 3.5 |
| Pb | 82 | 7.0 | - | 7.0 |
| U | 92 | 0.2 | - | 0.2 |

These composition data for the fuel and cladding made up the total composition used in the ORIGEN2 run for the N Reactor. A representative burnup of 1,188 MWd/MTHM was selected. The composition was therefore irradiated for 100 days at a constant power of 11.88 MW in increments of 20 days.

A cross-section and fission product yield library developed specifically for the Hanford N Reactor exists but was not readily available. Therefore, the N Reactor composition was irradiated with the use of two available libraries so the importance of the predicted fission or activation product content on the choice of library could be evaluated. The composition was irradiated using libraries developed for a Canada deuterium–uranium (CANDU) reactor using slightly enriched uranium oxide fuel (1.2 wt% U-235) and for a U. S. PWR using 3.2 wt%-enriched uranium oxide fuel. The CANDU library provides a good match in terms of uranium enrichment, but the neutron spectrum of the CANDU is much softer than that of the N Reactor. The PWR library was developed for a somewhat higher uranium enrichment but should provide a better match for the neutron spectrum. The results from both these calculations were compared and considered along with those from all the other runs. The cross-section and fission product yield library was the only thing that changed between the two runs. All other parameters were identical.

### 5.2.4 Hanford Single-Pass Reactors

The Hanford single-pass reactors were similar in design to the N Reactor (which replaced them); they were water-cooled, graphite-moderated reactors that used natural uranium metal fuel. The fuel was in the form of solid metal cylindrical slugs contained in aluminum jackets or *cans* as they were sometimes called. The uranium was bonded to the aluminum can using an aluminum-silicon eutectic. The aluminum used in the jackets was Type 2S.

Various references give slightly different dimensions for the fuel slugs used in the Hanford single-pass reactors. The slugs were historically called W slugs because the Hanford Site was code-named Site W during the Manhattan Project. For the current work, the following physical data and assumptions were used for the W slugs [2]:

- The uranium metal was 8.0 in. long and 1.37 in. in diameter.
- The can end and bottom caps were 1.37 in. in diameter and together were 0.125 in. thick.
- The aluminum can was 8.125 in. long with a wall thickness of 0.035 in.
- The density of the uranium was 18.95 g/cm$^3$.
- The density of the aluminum was 2.71 g/cm$^3$.
- The Type 2S aluminum used in the W slugs is equivalent in composition to modern Type 1100 aluminum.

There were other variations of W slugs that came and went over time. The data above are for the slugs used during the Manhattan Project era. A typical burnup for this era was 200 MWd per short ton (2,000 pounds) or 220 MWd/MTHM.

The use of these data yields 63.9 g of aluminum and 3,662.1 g of natural uranium per W slug. This fraction was scaled up to 1 metric ton of metal for the ORIGEN2 run. The heavy-metal content of the composition was therefore:

U-234 = 55.3 g
U-235 = 7,110.0 g
U-238 = 992,834.7 g

For the aluminum can, the constituent and impurity specification for Type 1100 aluminum in DOE (2003) was used. Table 5-6 lists the composition.

Table 5-6. Elemental composition for Hanford single-pass reactors.

| Type 1100 aluminum | | | |
|---|---|---|---|
| **Element** | ***Z*** | **Weight fraction** | **Mass (g/MTHM)** |
| Al | 13 | 0.993 | 17,333.886 |
| Si | 14 | 0.0025 | 43.640196 |
| Mn | 25 | 0.00025 | 4.3640196 |
| Fe | 26 | 0.0025 | 43.640196 |
| Cu | 29 | 0.00125 | 21.820098 |
| Zn | 30 | 0.0005 | 8.7280392 |

The trace element and impurity concentration given for the N Reactor fuel in the previous section was used for the natural uranium metal.

These composition data for the fuel and cladding made up the total composition used in the ORIGEN2 run for the Hanford single-pass reactor case. A representative burnup of 220 MWd/MTHM was selected. The composition was therefore irradiated for 200 days at a constant power of 1.1 MW in increments of 25 days [3].

A cross-section and fission product yield library developed specifically for the Hanford single-pass reactors is not available. The single-pass reactor composition was therefore irradiated with the use of two available libraries so the importance of the predicted fission or activation product content on the choice of library could be evaluated. The composition was irradiated with the use of libraries developed for a CANDU reactor using natural uranium oxide fuel and for a U.S. PWR using 3.2 wt%-enriched uranium oxide fuel. The CANDU library provides a good match in terms of uranium enrichment, but the neutron spectrum of the CANDU is much softer than that of the single-pass reactors. The PWR library was developed for a higher uranium enrichment but should provide a better match for the neutron spectrum. The results from both these calculations were compared and considered along with those from all the other runs. The cross-section and fission product yield library was the only data that changed between the two runs. All other parameters were identical.

### 5.2.5 Pressurized-Water Reactor

An ORIGEN2 run for the standard burnup U.S. PWR described in the Oak Ridge National Laboratory report *Standard- and Extended-Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code* (Ludwig and Renier 1989) was included to see how the results obtained from the other models compared with those for a system with such well-known behavior. The ORIGEN2 run irradiated 1 metric ton of 3.2%-enriched uranium oxide fuel clad in Zircaloy 4.

Table 5-7 lists the constituent and impurity concentrations for the fuel and cladding, which were taken directly from Ludwig and Renier (1989). The heavy-metal content of the composition was:

U-234 = 290.0 g
U-235 = 32,000.0 g
U-238 = 967,710.0 g

Table 5-7. Elemental composition data for PWR.

| Element | *Z* | Fuel (g/MTHM) | Zircaloy 4 (g/MTHM) | Total (g/MTHM) |
|---|---|---|---|---|
| H | 1 | - | 2.90 | 2.90 |
| Li | 3 | 1 | - | 1 |
| B | 5 | 1 | 0.07 | 1.07 |
| C | 6 | 89.4 | 26.76 | 116.2 |
| N | 7 | 25 | 17.84 | 42.84 |
| O | 8 | 13,4454 | 211.85 | 13,4666 |
| F | 9 | 10.7 | - | 10.70 |
| Na | 11 | 15 | - | 15.00 |
| Mg | 12 | 2 | - | 2.00 |
| Al | 13 | 16.7 | 5.35 | 22.05 |
| Si | 14 | 12.1 | - | 12.10 |
| P | 15 | 35 | - | 35.00 |
| S | 16 | - | 7.81 | 7.81 |
| Cl | 17 | 5.3 | - | 5.30 |
| Ca | 20 | 2 | - | 2.00 |
| Ti | 22 | 1 | 4.46 | 5.46 |
| V | 23 | 3 | 4.46 | 7.46 |
| Cr | 24 | 4 | 278.75 | 282.75 |
| Mn | 25 | 1.7 | 4.46 | 6.16 |
| Fe | 26 | 18 | 501.75 | 519.75 |
| Co | 27 | 1 | 2.23 | 3.23 |
| Ni | 28 | 24 | 4.46 | 28.46 |
| Cu | 29 | 1 | 4.46 | 5.46 |
| Zn | 30 | 40.3 | - | 40.30 |
| Zr | 40 | - | 21,8341.53 | 21,8342 |
| Mo | 42 | 10 | - | 10.00 |
| Ag | 47 | 0.1 | - | 0.10 |
| Cd | 48 | 25 | 0.06 | 25.06 |
| In | 49 | 2 | - | 2.00 |
| Sn | 50 | 4 | 3,568 | 3,572 |
| Gd | 64 | 2.5 | - | 2.50 |
| Hf | 72 | - | 17.39 | 17.39 |
| W | 74 | - | 4.46 | 4.46 |
| U | 92 | - | 0.04 | 0.04 |

These composition data for the fuel and cladding made up the total composition used in the ORIGEN2 run for the PWR. A burnup of 33,000 MWd/MTHM was used. The composition was therefore irradiated for 880 days at a constant power of 37.5 MW in increments of 60 days [4]. The cross-section and fission product yield library for the standard burnup PWR supplied with ORIGEN2 was used.

### 5.2.6 TRIGA Reactors

Two varieties of TRIGA reactor fuel were considered – aluminum clad and SS clad. TRIGA reactor fuel is characterized as consisting of enriched uranium in a matrix of ZrH. The ZrH gives the system a large negative temperature coefficient of reactivity and thus allows operation of the reactor in a pulsed mode in addition to steady state.

**TRIGA with aluminum-clad fuel**

The aluminum-clad TRIGA fuel modeled was 20%-enriched U-235 at 8 wt% of the UZrH matrix. The Zr:H atom ratio was 1.0. One fuel element was considered, which consisted of 180 g of uranium and 280 g of Type 1100 aluminum (Sterbentz 1997).

Table 5-8 lists the composition of the aluminum (Sterbentz 1997). The weight fractions used to compute the heavy-metal composition reflect an average of assays performed by General Atomics on several nominal 20%-enriched TRIGA fuel elements (Schmittroth and Lessor 1996):

U-234 = 0.299 g
U-235 = 35.508 g
U-236 = 0.177 g
U-238 = 144.014 g

Table 5-8. Composition of aluminum cladding for TRIGA reactor fuel.

| Type 1100 aluminum | | | |
|---|---|---|---|
| **Element** | ***Z*** | **Weight fraction** | **Mass (g)** |
| Al | 13 | 0.993 | 278.04 |
| Si | 14 | 0.0025 | 0.70 |
| Mn | 25 | 0.00025 | 0.07 |
| Fe | 26 | 0.0025 | 0.70 |
| Cu | 29 | 0.00125 | 0.35 |
| Zn | 30 | 0.0005 | 0.14 |

Table 5-9 lists the constituent and impurity specifications from DOE (2003) used for the ZrH matrix. These composition data for the fuel and aluminum cladding made up the total composition used in the ORIGEN2 run for the TRIGA reactor with aluminum-clad fuel. A midrange burnup of 21,116 MWd/MTHM was used, which equates to 5.9 MWd for the fuel element. This burnup was selected based on information in Sterbentz (1997). The composition was irradiated for 5.384 years at a constant power of 0.003 MW in increments of 0.25 years.

**TRIGA with SS-clad fuel**

The SS-clad TRIGA fuel modeled was 20%-enriched U-235 at 8.5 wt% of the UZrH matrix. The Zr:H atom ratio was 1.7. One fuel element was considered, which consisted of 195 g of uranium and 819.4 g of Type 304 SS (Sterbentz 1997). Table 5-10 lists the composition of the 304 SS (Ludwig and Renier 1989).

Table 5-11 lists the constituent and impurity specifications from (DOE 2003) used for the ZrH matrix. For the heavy-metal composition of the fuel, the same mass fractions used above for the aluminum-clad fuel were used, giving the following:

U-234 = 0.324 g
U-235 = 38.467 g
U-236 = 0.192 g
U-238 = 156.015 g

Table 5-9. Composition of ZrH for aluminum-clad TRIGA reactor fuel.

| **ZrH** | | | |
|---|---|---|---|
| **Element** | ***Z*** | **Weight fraction** | **Mass (g)** |
| H | 1 | 0.010628 | 21.99996 |
| B | 5 | 0.0000005 | 0.001035 |
| C | 6 | 0.0002697 | 0.558238 |
| N | 7 | 0.0000799 | 0.165393 |
| O | 8 | 0.0009489 | 1.964161 |
| Al | 13 | 7.491E-05 | 0.155064 |
| Si | 14 | 0.0001199 | 0.24811 |
| P | 15 | 9.988E-05 | 0.206752 |
| S | 16 | 3.496E-05 | 0.072367 |
| Ti | 22 | 4.994E-05 | 0.103376 |
| V | 23 | 4.994E-05 | 0.103376 |
| Cr | 24 | 0.0012485 | 2.584416 |
| Mn | 25 | 4.994E-05 | 0.103376 |
| Fe | 26 | 0.0022473 | 4.651932 |
| Co | 27 | 1.998E-05 | 0.041359 |
| Ni | 28 | 6.992E-05 | 0.144734 |
| Cu | 29 | 4.994E-05 | 0.103376 |
| Zn | 30 | 9.988E-05 | 0.206752 |
| Zr | 40 | 0.989082 | 2,047.4 |
| Nb | 41 | 6.992E-05 | 0.144734 |
| Mo | 42 | 4.994E-05 | 0.103376 |
| Cd | 48 | 0.0000005 | 0.001035 |
| Sn | 50 | 0.0159809 | 33.08044 |
| Sm | 62 | 9.99E-06 | 0.020679 |
| Gd | 64 | 4.99E-06 | 0.010329 |
| Hf | 72 | 3.496E-05 | 0.072367 |
| Ta | 73 | 0.0001998 | 0.413503 |
| W | 74 | 9.988E-05 | 0.206752 |
| Pb | 82 | 9.988E-05 | 0.206752 |
| Th | 90 | 6.99E-06 | 0.014469 |
| U | 92 | 0.0000035 | 0.007245 |

The composition data for the fuel and SS cladding made up the total composition used in the ORIGEN2 run for the TRIGA reactor with SS-clad fuel. A midrange burnup of 19,492 MWd/MTHM was used, which equates to 3.8 MWd for the fuel element. This burnup was selected based on information in Sterbentz (1997). The composition was irradiated for 3.469 years at a constant power of 0.003 MW in increments of 0.25 years.

An ORIGEN2 reactor model developed for a TRIGA reactor was not available (it is not known if one exists), so, as with the Hanford reactors, the TRIGA reactor compositions (for both the aluminum- and SS-clad fuels) were irradiated using two available cross-section and fission product yield libraries. The libraries used were those for the standard burnup PWR and the ATR discussed in previous sections.

Table 5-10. Composition of cladding for SS-clad TRIGA reactor fuel.

| 304 SS | | | |
|---|---|---|---|
| **Element** | ***Z*** | **Weight fraction** | **Mass (g)** |
| C | 6 | 0.0008 | 0.66 |
| N | 7 | 0.0013 | 1.07 |
| Si | 14 | 0.01 | 8.19 |
| P | 15 | 0.00045 | 0.37 |
| S | 16 | 0.0003 | 0.25 |
| Cr | 24 | 0.19 | 155.69 |
| Mn | 25 | 0.02 | 16.39 |
| Fe | 26 | 0.68844 | 564.12 |
| Co | 27 | 0.0008 | 0.66 |
| Ni | 28 | 0.0892 | 73.09 |

Table 5-11. Composition of ZrH for SS-clad TRIGA reactor fuel.

| ZrH | | | |
|---|---|---|---|
| **Element** | ***Z*** | **Weight fraction** | **Mass (g)** |
| H | 1 | 0.018439 | 38.50063 |
| B | 5 | 0.0000005 | 0.001044 |
| C | 6 | 0.0002697 | 0.563092 |
| N | 7 | 0.0000799 | 0.166831 |
| O | 8 | 0.0009489 | 1.981241 |
| Al | 13 | 7.491E-05 | 0.156412 |
| Si | 14 | 0.0001199 | 0.250268 |
| P | 15 | 9.988E-05 | 0.208549 |
| S | 16 | 3.496E-05 | 0.072996 |
| Ti | 22 | 4.994E-05 | 0.104275 |
| V | 23 | 4.994E-05 | 0.104275 |
| Cr | 24 | 0.0012485 | 2.606889 |
| Mn | 25 | 4.994E-05 | 0.104275 |
| Fe | 26 | 0.0022473 | 4.692383 |
| Co | 27 | 1.998E-05 | 0.041718 |
| Ni | 28 | 6.992E-05 | 0.145993 |
| Cu | 29 | 4.994E-05 | 0.104275 |
| Zn | 30 | 9.988E-05 | 0.208549 |
| Zr | 40 | 0.98156 | 2,049.497 |
| Nb | 41 | 6.992E-05 | 0.145993 |
| Mo | 42 | 4.994E-05 | 0.104275 |
| Cd | 48 | 0.0000005 | 0.001044 |
| Sn | 50 | 0.0159809 | 33.3681 |
| Sm | 62 | 9.99E-06 | 0.020859 |
| Gd | 64 | 4.99E-06 | 0.010419 |
| Hf | 72 | 3.496E-05 | 0.072996 |
| Ta | 73 | 0.0001998 | 0.417099 |
| W | 74 | 9.988E-05 | 0.208549 |
| Pb | 82 | 9.988E-05 | 0.208549 |
| Th | 90 | 6.99E-06 | 0.014595 |
| U | 92 | 0.0000035 | 0.007308 |

## 5.3 REACTOR MODELING RESULTS

Once the eleven ORIGEN2 runs were completed, the fission and activation product results were imported into spreadsheets. The ORIGEN2 results were radioactivity in curies, with the results for activation products, actinides, and fission products reported separately. This is how the ORIGEN2 libraries and output tables are structured. Once the results were in spreadsheets, the activity data for the activation products and fission products were converted to activity relative to Cs-137. (Actinides are beyond the scope of this effort.) Summary tables were then created that show relative activity for fission and activation products (separately) for all nuclides that had a relative activity of at least 0.001 at 10 days of decay. These summary tables for activation and fission products are included as Attachments A and B, respectively. In the tables, the "N Reactor 1" column means the N Reactor case run using the CANDU library and the "N Reactor 2" column means the case run using the PWR library. The same is true for W Reactor cases, which are the Hanford single-pass reactors.

When reviewing the summary data, it is important to keep in mind that these results are approximate in that no effort has been made to account for reactor power history. As a result, the relative activity for short-lived radionuclides is overstated. The effect of this depends on a number of factors, but on burnup in particular. This is why there is such a long list of activation products with relative activity greater than or equal to 0.001 for the N Reactor, especially for the run using the PWR library. The overestimates for the short-lived species plus the short time for Cs-137 ingrowth result in a number of nuclides giving a result greater than or equal to 0.001. This same effect is not seen for the W Reactor cases (despite the low burnup) because the W slugs were canned in high-purity aluminum (unlike the N Reactor fuel, in which the Zircaloy 2 yields a number of activation products).

From review of the results in Tables A-1 and A-2 in Attachment A, four cases were selected as having representative radionuclide inventories that warranted further consideration:

- The ATR
- The FFTF
- The N Reactor (with the PWR cross-section library; listed as N Reactor 2 in Table A-1)
- The TRIGA reactor with SS-clad fuel (with the PWR cross-section library; listed as TRIGA SS PWR in Table A-1).

The ATR stood out because it was the only highly enriched uranium, very high burnup system considered. The FFTF similarly stood alone as the only system considered where Pu-239 was a substantial constituent of the fuel. The N Reactor was selected because it represented the low-burnup production reactors with a number of activation products in the mix. The TRIGA reactor (with SS-clad fuel) was selected because it represented moderate burnup systems, again with a number of activation products due to the SS cladding.

The fission and activation product relative inventory data for these four cases were summed, as appropriate, to yield tables of relative activity for 1,192 individual nuclides for the seven decay times. These tables are included for the ATR, FFTF, N Reactor, and TRIGA reactor as Tables A-3 to A-6, respectively.

## 5.4 DECAY TIMES

After cessation of fuel irradiation, fission and activation product activity ratios vary considerably over time. This document refers to the time since the last irradiation of the fuel as the decay time. Exposure of workers to fission and activation products would encompass a continuum of decay times from hours to years. Contamination generally included some material that had long decay times (half a year or more), although, for example, material in the reactor areas with short decay times would have been included in the mix. This document uses representative decay times for general fuel-cycle locations or processes. These decay times apply to the mix of contamination to which workers would have been exposed rather than the age of the fuel itself. The tables in Attachment A, generated as described above, list the decay times and associated reactor fuel and cladding activities relative to Cs-137. Finer resolution of decay times would not significantly change the MFP/MAP ratios of dosimetric interest, and it is unlikely that there is sufficient knowledge in most cases to apply different decay times. However, judgment in the choice of the available decay times in this document can be used for any particular dose reconstruction. For instance, if it is known that fuel examination at a given facility was restricted to fuel that had to be aged for about 6 months, then a decay time of 180 days or 1 year should be chosen. It is generally favorable to claimants to use longer decay times because that increases the relative activities of the longer lived radionuclides, which have higher 50-year dose conversion factors. If an appropriate decay time cannot be readily determined, intakes and organ doses for the two most likely decay times should be determined and the set that is most favorable to the claimant should be selected for dose reconstruction.

Table 5-12. Fission and activation product decay times for general steps in the fuel cycle.

| Activities | Decay time |
|---|---|
| Reactor operations, spent fuel storage, fuel examination | 10 d |
| Fuel dissolution: Early production reactors (e.g., Hanford production reactors 1940s–1950s) | 40 d |
| Fuel dissolution: General or later years | 180 d |
| Waste management | 1 y |

## 6.0 DOSE COMPARISONS BY RADIONUCLIDE FOR EACH OF THE REACTOR SOURCE TERMS AND DETERMINATION OF DOSIMETRICALLY SIGNIFICANT RADIONUCLIDES

This section uses the reactor source terms from Attachment A to determine dosimetrically significant radionuclides by adjusting the source term by element-specific exposure fractions and radionuclide- and absorption type-specific dose conversion factors. The purpose of this section is to simplify the source term.

## 6.1 EXPOSURE FRACTIONS

The source term in the reactor fuel and its cladding is not of interest until it becomes available for intake by a worker. The more volatile a material is, the more likely it is to be released to the atmosphere. This is true for gases and volatile materials as well as for radionuclides whose parent was a gas or volatile material. Exposure to these volatile materials is most likely to occur during reactor-related or fuel-dissolution operations. At some sites and in the earlier days of operations, fuel was removed from the reactor and allowed to decay for only brief periods before processing. Ventilation systems in fuel-dissolution facilities were designed to remove the gases and vapors from the dissolution areas quickly and exhaust them through stacks. By the 1950s, many sites had ventilation systems and special filtration and collection systems to capture many of the volatile and semivolatile materials, especially radioiodines, much of which decayed in the filtration system (with the exception of I-129). In later years, it was rare to process fuel aged for less than 100 days.

Release fractions from DOE Standard 1027 (DOE 1997) were used to derive the exposure fraction of each radionuclide from the reactor source terms. Gases were assigned a value of 1, volatile materials such as iodine were assigned a value of 0.5, and semivolatile materials, such as cesium, were assigned a value of 0.01. For particulate and solids, the DOE value of 0.001 was increased by a factor of 10 to account for buildup of nonvolatile radionuclides over time and to better address exposures during maintenance-type activities. Table B-1 in Attachment B lists the exposure fractions assigned to each element.

All radionuclides that had reported activity from the ORIGEN2 runs were carried through the calculations, but exposures to the more volatile radionuclides, such as radioiodines, are rarely seen in operations that used adequate ventilation and collection, holdup, or filtration to control exposures. In addition, some sites had separate monitoring for radioiodines. The larger exposure fractions used in the calculations for radioiodines resulted in larger intake ratios relative to particulates, but the latter were more significant sources of internal exposure at most sites. To determine if radioiodine intakes from this document should be used, site profile and claim-specific information should be reviewed to determine if chronic iodine intakes were feasible. When available, individual monitoring data should be used to determine dose from specific radionuclides.

## 6.2 DEVELOPMENT OF DOSIMETRICALLY SIGNIFICANT REACTOR SOURCE TERM

To determine the dosimetric significance of each radionuclide, inhalation dose conversion factors were obtained for many of the radionuclides in the reactor source term. The ORIGEN2 source term tables originally included a list of 1,192 radionuclides, most of which had no reported activities. Almost 300 radionuclides had some activity. Dose conversion factors were obtained for 738 radionuclides and were available for all but 40 of the radionuclides with associated activity. The exceptions are listed in Table C-1 in Attachment C and include the noble gases, which do not contribute significantly to internal dose under most circumstances. The other unmatched radionuclides had relatively small activities and were for the most part either very short-lived (seconds to hours) or very long-lived (more than 2 E14 years); they would not contribute significantly to internal dose. These radionuclides were not considered further.

The fractional activities of the reactor source terms (activation and fission products, not actinides) were multiplied by the exposure fractions to determine the relative exposure activity for each radionuclide. The exposure activity was multiplied by organ dose conversion factors for each of the four reactors and the associated four decay times. Dose conversion factors were taken from the Radiological Toolbox, Version 1.0.0 software (ORNL 2003) and are consistent with International Commission on Radiological Protection (ICRP) Publication 68 (ICRP 1995) values. Because many radionuclides have multiple lung absorption types and because the chemical form can vary depending on the release mechanism or chemical treatment of the dissolved fuel or waste, doses were generated for three inhalation categories: soluble, moderately soluble, and insoluble. For radioelements like cesium, only a single type was specified ICRP Publication 68 (ICRP 1995), so this single type was used in all three categories. For radioelements with two solubility types, the most soluble type was designated as soluble and the least soluble type as insoluble. The moderately soluble type was designated as type M if available; if type M was not a choice, the most soluble of the types was used. Tritium (created in the fuel and cladding) and iodines were assumed to be in vapor form. Internal dose from noble gases and the radionuclides listed in Table C-1 were not estimated because they were deemed insignificant. (In these calculations it was assumed that strontium was either type F or S, but in the urinalysis calculations only type F was assumed because it was felt that strontium titanate would not have been a component of mixed fission products, but rather would have been specially manufactured in the workplace).

The relative radionuclide doses were summed for each organ. The organ dose from each radionuclide was then compared to its sum. Those radionuclides that contributed at least 1% to the total organ dose or to the effective dose were identified for each reactor scenario, which resulted in the list shown in Attachment D, Table D-1, of 39 different radionuclides for the four reactors and four decay times.

The relative exposure fractions (the initial activity times the exposure fraction divided by the sum of the exposure fractions from the 1% source terms) were determined for each radionuclide that contributed more than 1% of the dose from each of the 16 reactor scenarios, and these were assumed to be the intake fractions. The relative exposure fractions were assumed to be the fractional activities of the radionuclides at intake.

The reactor types were compared and found to have dosimetrically similar source term components and values. For many organs, the inclusion of just those radionuclides that contributed more than 15% to dose accounted for at least 50% of the dose and in most cases accounted for more than 90% of the dose. Therefore, it was decided to qualitatively simplify the list further and use only those 17 radionuclides that contributed the most to dose and to activity. Because the relative exposure fractions were assumed to be radionuclide intake fractions, this effectively increased doses from the truncated reactor source term. These radionuclides represent the more dosimetrically significant radionuclides and are assumed to account for all the activity, although it is likely that a portion of measured activity comes from less dosimetrically significant radionuclides. In addition, a default reactor source term was established by averaging (mean) the four reactor source terms for each decay period. Tables E-1 and Tables E-2 in Attachment E list the simplified source terms and default source terms.

## 6.3 VERIFICATION THAT DEFAULT SOURCE TERMS DO NOT UNDERESTIMATE DOSE

It should be pointed out that while reducing the number of radionuclides to be analyzed simplifies and speeds up calculations and report preparation, it also has the effect of increasing dose. This was verified by comparing the doses derived from intakes from the complete source terms to the doses derived from intakes from the simplified source terms and the default source term. When compared to doses from the complete ORIGEN2 reactor source terms, the results show that 50-year doses from just the radionuclides in the simplified reactor source terms with the activities from the 39-radionuclide source term are higher by factors of 14 to 309, depending on organ and solubility assumptions, with the median factor at about 93. For the default source terms, the factors increase to 28 to 371 with a median of about 110. When the simplified source term was adjusted to relative activity fraction for the 17 radionuclides, the 50-year doses from the ORIGEN2 reactor source terms are higher by factors of 20 to 335, with the median factor at about 110. The information shown here considers only the effect of reducing the source term to the most significant dose contributors. Uncertainties associated with the applications of the associated radionuclide activity factors include:

- The actual periods of fuel irradiation and decay,
- The chemical and physical processes in the work area that modify the source term ratios,
- The urinalysis measurement practices (including time between sample collection and counting, sample chemical processing and efficiency assumptions, and measurement reporting protocols) for determination of gross beta and gamma activity in the actual source term and in urine, and
- The temporal assumptions about exposure periods and sample measurement.

In addition, this document's method for assigning intakes from counts of a given radiation type is consistent with methods used at many sites with reactor source terms, for example initially assigning gross beta results as Sr-90 when other information is not available (and when the effective dose is the dose of concern). Because of the many uncertainties associated with the application of the associated radionuclide activity factors and the relatively low dose consequences in most cases, these possibly biased high associated activity factors are deemed acceptable by the authors of this document.

Based on verification that the default source terms do not underestimate dose, the default scenarios can be used to represent the source terms in all cases. Consideration was given to applying a dose reduction factor to reduce the overestimates, but because (1) the dose comparison was based on 50-year rather than annual doses, (2) the significant number of unknowns in a particular workplace exposure, and (3) the unknowns associated with the measuring scenarios (correction for decay between sample collection and counting, counting efficiency, yield efficiencies, chemical processing of samples prior to counting, etc.), no generic reduction method appeared to be reasonable.

The use of this source term is likely to provide an upper bound of dose from a nonspecific radioanalysis. Therefore, it would be reasonable to assign resultant doses as upper bounds (i.e., constant distributions) and to ignore any further uncertainty from metabolic models, etc.

In the simplest of cases, these default activity fractions can be assumed to be intake fractions when activity is reported as total activity. To use this information in reconstructing dose from other types of nonspecific radioactivity results, such as gross beta contamination measurements, further information about interpretation of monitoring results is needed.

## 7.0 INTERPRETATION OF NONSPECIFIC RADIONUCLIDE ANALYSES FOR DETERMINATION OF INTAKES OF MIXED FISSION AND ACTIVATION PRODUCTS

This section provides the internal dose parameters and their derivation for use in interpretation of gross beta and/or gamma activity measurements related to reactor source terms. These are generic parameters and models; site- and claim-specific information should be considered when applying these models. Models developed in site profiles based on site-specific information take precedence over the models in this document. The models developed in this document may be used to estimate intakes at specific sites, but the selection of parameters to use, e.g. fuel age or inclusion of radioiodine, should be explained in site profiles, or if a site profile is not available in dose reconstruction reports.

### 7.1 URINALYSIS PARAMETERS

Guidance for estimating intakes of MFPs/MAPs from urinalysis data is provided for three types of data:

- Gross beta analysis that includes all fission and activation products except radioiodines and beta emitters naturally in urine such as K-40,
- Gross gamma analysis that includes all fission and activation products except radioiodines and natural gamma emitters, and
- Gross beta analysis with chemical processing that captured strontium, yttrium, barium, lanthanum, cerium, praseodymium, neodymium, promethium, samarium, europium, gadolinium, terbium, dysprosium, holmium, erbium, thallium, ytterbium, and lutetium.

The latter analysis was used at the Hanford Site and probably at the Savannah River Site in the late 1940s through the early 1960s. For sites that did not remove the natural beta or gamma emitters before counting, site profiles or other Project resources might provide information about the protocol a site used to distinguish natural radionuclide counts from counts attributable to fission and activation product exposures.

To determine detected activity in urine for radionuclides that contribute the most to dose, 24-hour urine intake retention fractions (IRFs) at several times after the beginning of a chronic inhalation exposure were determined with Integrated Modules for Bioassay Analysis (IMBA), Version 4.0.9, software for the range of the ICRP Publication 68 (ICRP 1995) solubility types used in previous calculations (as noted previously, only type F strontium was considered in this set of calculations). It was noted that 2 years after the start of a chronic intake, most radionuclides had essentially reached equilibrium in the urine; that is, the chronic IRF became constant. IRFs for some of the 39 radionuclides and their solubility types were compared at 20 years and 2 years. The maximum ratio of the reviewed IRFs for the 20- to 2-year intake period was 2.3 (type S Pm-147).

A radiation yield and counting adjustment factor, based on several factors including the type and energy of radiation emissions, and a radiation abundance factor were used to estimate the amount of a radionuclide that might be detected in a particular analysis. Low-energy beta particles (less than the 0.0494-MeV average energy of C-14) were not included in determining beta yield because it was unlikely that they would have been detected by the early counting systems. Some judgment was used because not all the counting systems were alike and the exact detection efficiencies for the low-energy radiations are not known. It was assumed that only half of the beta particles with average energies less than 0.1 MeV were counted and that photons of less than 0.1 MeV were not counted.

Examples of the application of the above criteria include the following. The 0.062-MeV (average) beta particles from Pm-147 had a yield of 1 and a counting factor of 0.5, which resulted in an adjustment factor of 0.5. The gamma adjustment factor for La-140 was 2.12 to account for its multiple greater-than-0.1-MeV photon emissions. Short-lived progeny that quickly build to equilibrium were considered part of the total beta or photon emissions of the parent; for instance, the 1.207-MeV beta yield of Pr-144 (abundance of 1) was added to half of the 0.08108-MeV Ce-144 beta (abundance of 1), which resulted in a total adjustment factor of 1.5 for Ce-144. Table F-1 in Attachment F lists the IRFs and the yield and counting adjustment factors.

To evaluate activity in urine, two sample collection times were chosen by the document authors: 90 days and 2 years after the start of intake. For each sampling time and each reactor scenario, exposure activity fractions were multiplied by urine IRFs and by the yield/counting adjustment factor to obtain the percentage of each radionuclide's contribution to beta and/or gamma urine counts. The results for the four reactors were averaged and are presented in Attachment G, Tables G-1 to G-4 (rounding in some cases causes summing of the columnar values to differ from 100% by -0.1% to +0.2%). Note that these tables are contributions to counts, not to activity. To convert to a radionuclide's activity contribution, the percentages in Tables G-1 to G-4 must be divided by the appropriate yield factors in Table F-1 in Attachment F. The trends are similar for all solubility categories, and so the results for each solubility category were averaged to simplify the final calculations. To simplify final results and because it is more representative of an equilibrium condition, the 2-year sampling period was chosen to establish urinalysis activity fractions.

Two radionuclides were selected as indicator radionuclides for determining intakes from gross beta or gamma analyses. The fractions of activity in urine due to a chronic inhalation intake were determined for Sr-90 and Cs-137. These radionuclides were selected because they had reported activity for all the default source terms and decay periods. In addition, Sr-90-to-total-radiostrontium ratios are provided for interpretation of strontium measurements. The strontium isotopes are beta emitters and are used with gross beta results. Although Cs-137 is a beta as well as gamma emitter, it could have been removed from the sample matrix during chemical processing; it is, therefore, best used only to interpret gross gamma urinalyses. However, because some sites might have included the radionuclides in specific analyses, the beta information is presented for the rare occasion where it might be of use. Table 7-1 lists the resulting urine activity fractions.

Table 7-1. Urine activity fraction to be used for the indicator radionuclides.

| Radionuclide | Average fraction of beta activity in urine sample[a, b] | | | |
|---|---|---|---|---|
| | **10 d** | **40 d** | **180 d** | **1 y** |
| Cs-137 | 0.055 | 0.077 | 0.20 | 0.34 |
| Sr-90 | **0.024** | **0.034** | **0.090** | **0.16** |
| | **Average fraction of Sr-90 activity in Sr urine sample[c]** | | | |
| | **10 d** | **40 d** | **180 d** | **1 y** |
| Sr-90 | 0.045 | 0.061 | 0.23 | 0.75 |
| | **Average fraction of gamma activity in urine sample** | | | |
| | **10 d** | **40 d** | **180 d** | **1 y** |
| Cs-137 | 0.12 | 0.20 | 0.44 | 0.63 |

a. These fractions assume there is no I-131 in the final sample due to its volatility and short half-life. The presence of I-131 in the sample when counted will lead to an overestimate of the intake.
b. In general Cs-137 should not be used to interpret gross beta analyses.
c. The fractions of Sr-90 in radiostrontium samples are provided for use in interpreting strontium bioassay results associated with reactor work.

The same process was used to evaluate average contributions to the beta activity for the chemically processed gross beta urinalyses; Table 7-2 lists the results. The calculations considered only barium, cerium, europium, lanthanum, neodymium, promethium, praseodymium, samarium, strontium, and yttrium. A quick review of one reactor run showed that activities of gadolinium, terbium, dysprosium, holmium, erbium, thallium, ytterbium, and lutetium were lower than the considered radionuclides.

Table 7-2. Urine activity fraction for chemically processed beta samples.

| Radionuclide | Average fraction of beta activity in urine sample | | | |
|---|---|---|---|---|
| | **10 d** | **40 d** | **180 d** | **1 y** |
| Sr-90 | 0.037 | 0.057 | 0.20 | 0.43 |

To determine chronic intakes from nonspecific radionuclide analyses, each urinalysis result should be multiplied by the fraction of activity from one of the appropriate indicator radionuclides in Table 7-1 or Table 7-2, which provides the urine activity ratio of the indicator radionuclide for use in intake analyses. Table 7-1 fractions should be used for beta-counted urine samples that have not been chemically processed, radiostrontium urinalyses, and gamma-counted urinalyses reported as gross activity. The activity fraction is provided for gross radiostrontium results because early strontium analyses were not always radionuclide-specific. For decay periods of greater than a couple of years, the Sr-89 (50.5-day half-life) activity will be less than 0.01% of its original value, and the activity can be assumed to be all Sr-90 (29.12-year half-life). Table 7-2 should be used for beta-counted samples that were chemically processed to remove certain radionuclides. When the intake of the indicator radionuclide has been determined, intakes from the associated radionuclides can be determined by multiplying the intake by the appropriate ratios (selected from the applicable fuel-decay column) from Table 7-3 based on the default source term. Note that typically Cs-137 is used to interpret gamma analyses and Sr-90 is used to interpret beta analyses. The resulting intakes are then used to determine the appropriate organ doses by the methods summarized in Section 8.0.

Table 7-3. Indicator and associated radionuclide ratios for estimating intakes.

| Intake activity relative to Sr-90 | | | | | Intake activity relative to Cs-137 | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Nuclide | **10 d** | **40 d** | **180 d** | **1 y** | Nuclide | **10 d** | **40 d** | **180 d** | **1 y** |
| Ba-140 | 26 | 7.2 | | | Ba-140 | 21 | 5.5 | | |
| Ce-141 | 30 | 21 | 1.4 | 2.1 | Ce-141 | 24 | 16 | 1.1 | 1.6 |
| Ce-144 | 16 | 17 | 14 | 7.8 | Ce-144 | 13 | 13 | 11 | 6.0 |
| Cs-134 | 0.30 | 0.35 | 0.34 | 0.31 | Cs-134 | 0.24 | 0.27 | 0.26 | 0.23 |
| Cs-137 | 1.3 | 1.3 | 1.3 | 1.3 | **Cs-137** | **1** | **1** | **1** | **1** |
| Eu-155 | 0.072 | 0.087 | 0.090 | 0.081 | Eu-155 | 0.057 | 0.067 | 0.068 | 0.062 |
| Fe-55 | 1.7 | 1.5 | 1.1 | 0.81 | Fe-55 | 1.3 | 1.1 | 0.83 | 0.62 |
| I-131[a] | 532 | 57 | | | I-131[a] | 425 | 44 | | |
| La-140 | 30 | 7.1 | | | La-140 | 24 | 5.5 | | |
| Nb-95 | 28 | 33 | 16 | 3.0 | Nb-95 | 23 | 25 | 12 | 2.3 |
| Pm-147 | 3.4 | 3.6 | 3.5 | 3.1 | Pm-147 | 2.7 | 2.8 | 2.6 | 2.4 |
| Ru-103 | 23 | 18 | 2.0 | | Ru-103 | 18 | 14 | 1.5 | |
| Ru-106 | 5.5 | 6.4 | 5.4 | 3.8 | Ru-106 | 4.4 | 4.9 | 4.1 | 2.9 |
| Sr-89 | 19 | 17 | 3.6 | 0.31 | Sr-89 | 15 | 13 | 2.7 | 0.23 |
| **Sr-90** | **1** | **1** | **1** | **1** | Sr-90 | 0.80 | 0.77 | 0.75 | 0.76 |
| Y-91 | 23 | 21 | 5.8 | 0.77 | Y-91 | 19 | 16 | 4.4 | 0.59 |
| Zr-95 | 31 | 28 | 8.4 | 1.4 | Zr-95 | 24 | 21 | 6.3 | 1.1 |

a. For sites where iodine was well controlled or if individual iodine monitoring data are available, these iodine intake fractions should not be used. The iodine intake activity ratios would primarily apply to the earliest programs, before the use of ventilation and collection systems. In later years, it is reasonable to assume that there were no significant intakes from iodine at most sites.

## 7.2 PARAMETERS FOR INTERPRETATION OF AIR OR SURFACE CONTAMINATION

An approach similar to that discussed in Section 7.1 can be used to estimate intakes from gross beta air-sampling results. It is assumed that the volatile radionuclides (e.g., I-131) are not included in the results of the nonspecific air-sampling methods (this is an overestimating assumption in the case where the measured results include volatile radionuclides). The radionuclides in the simplified source term that contribute to the detected beta activity on an air sample were determined, and their relative contributions to activity were calculated with the use of the default intake fractions and yield and counting adjustment factors. The ratio of the activity from all the default radionuclides to the beta-emitting radionuclides was used to adjust the beta activity on the sample to activity for each of the default radionuclides. The resulting activity ratios are listed in Table 7-4, and one of these should be selected (usually Sr-90 for results reported as beta activity) to adjust the gross beta air concentrations to a radionuclide-specific air concentration that should then be used with the ratios for the appropriate fuel-age category in Table 7-3 to estimate inhalation intakes when multiplied by the amount of air inhaled. See Attachment H for an example of the use of these parameters. A similar approach could be used to address surface concentrations; again, contribution from the volatile radionuclides to the surface would be considered insignificant.

Table 7-4. Activity ratios for use when gross beta results are reported for air samples and workplace measurements.

| | Average fraction of beta activity by radionuclide | | | |
|---|---|---|---|---|
| **Radionuclide** | **10 d** | **40 d** | **180 d** | **1 y** |
| Cs-137 | 0.0016 | 0.0064 | 0.025 | 0.051 |
| Sr-90 | 0.0013 | 0.0049 | 0.019 | 0.039 |
| Sr-90 in gross Sr result | 0.050 | 0.057 | 0.22 | 0.77 |

# 8.0 MIXED FISSION AND ACTIVATION PRODUCT DOSE ASSIGNMENT SUMMARY

Radiation monitoring of reactor-related source terms can be interpreted with the default information in this document, when site-specific guidance is not available. Selection of source term decay times is based on worker categories in Table 5-12, as well as consideration of site and claim information.

Iodine intakes should be addressed with site- and individual-specific information when available. The source term does not directly account for radionuclides produced due to neutron interactions with material other than reactor fuel and cladding, such as tritium (although in most cases, however, the doses are sufficiently bounded such that dose from these other radionuclides could be adequately addressed with use of this source term).

## 8.1 INTAKE DETERMINATION

To determine inhalation intakes of the indicator radionuclide from gross urinalysis results:

- Per the guidance in Section 7.1, multiply the gross urine sample result by the fraction of an indicator radionuclide from Table 7-1 (for unprocessed beta, strontium, or gross gamma analyzed urine samples) or 7-2 (chemically processed, beta-counted urine samples). (All fractions were based on chronic intakes.)

- Determine the intake of the indicator radionuclide using Project guidance (ORAUT 2007). The specific method, e.g., missed or fitted dose, will depend on the value of the result and the details of the case.

For coworker data sets, the intakes have already been calculated and are documented in a site-specific document. Adjustments from Table 7-1 or 7-2 may be needed, depending on the assumptions used when the data were analyzed for the coworker study. When individual monitoring or coworker results are not available, gross activity air concentrations or surface contamination results can be used to estimate intakes. These should be adjusted to an indicator radionuclide activity with the use of one of the ratios in Table 7-4, and the intake for the indicator radionuclide should then be calculated based on generic, site-specific, or claim-specific exposure parameters.

Once the intake of the indicator radionuclide has been determined:

- Multiply the intake of the indicator radionuclide by the appropriate intake ratios of the associated radionuclides in Table 7-3.

- Calculate the doses for the indicator and associated radionuclides. For radionuclides with multiple possible absorption types, select the type that produces the largest dose to the organ of interest, with the exceptions of strontium, which would be in the soluble form (type F) for a reactor source term, and iodine, which is assumed to be a vapor intake of elemental iodine.

Attachment H shows examples of three intake calculations using gross beta bioassay data, unprocessed gross beta bioassay data and air-sampling data.

## 8.2 INTERACTIVE RADIOEPIDEMIOLOGICAL PROGRAM (IREP) DISTRIBUTION ASSIGNMENT

The indicator radionuclide doses calculated in Section 8.1 are entered into IREP with the distribution appropriate for the manner in which the intake was determined. For example, if there were positive urine sample results and a fitted dose was calculated from these results, a lognormal distribution with a GSD equal to 3 would be assigned (ORAUT 2007). For missed dose, a triangular distribution is assigned (ORAUT 2007).

The doses from the associated radionuclides, calculated in Section 8.1, are entered as constants in IREP. As discussed in Section 6.3, the simplifying assumptions used to pare down the list of associated radionuclides to a manageable number result in overestimates of the doses relative to the complete list. Because of this, these doses are entered into IREP with a constant distribution. The doses from the radionuclides in the simplified source term are estimated to be somewhere between a factor of 20 and 335 larger than those in the complete ORIGEN2 reactor source term, so the application of a constant distribution is appropriate.

Fitted and coworker doses are typically modeled with a lognormal distribution. The GSD that gives a factor of 20 at the 95% confidence level is

$$GSD^{1.645} = 20$$

$$GSD = 6.179$$

A GSD of 6.179 would therefore be required to encompass values up to 20 times larger than the geometric mean.

Missed dose is typically modeled with a triangular distribution, with a maximum value of twice the geometric mean. The GSD for fitted doses is 3, so the 95th percentile of both of these distributions falls well under the factor of 20. The GSD applied to a coworker-derived intake will vary with the data. The constant distribution can be used to apply the factors in this OTIB to coworker studies for years in which the GSD is less than 6.179.

When the GSD of a coworker intake rate is greater than 6.179, assign the associated radionuclides to a lognormal distribution in IREP and apply the GSD associated with the indicator radionuclide.

## 9.0 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this document, bracketed callouts have been inserted to indicate information, conclusions, and recommendations provided to assist in the process of worker dose reconstruction. These callouts are listed here in the Attributions and Annotations section, with information to identify the source and justification for each associated item. Conventional references, which are provided in the next section of this document, link data, quotations, and other information to documents available for review on the Project's Site Research Database.

[1] Burns, Jr. Robert E., CHP. Shonka Research Associates. Senior Health Physicist. March 2007.
Ten days of decay was selected to ensure short-lived nuclides were considered. The data are presented relative to Cs-137 as a convenience. This is a common practice since Cs-137 is the most prominent fission product and is easily quantified.

[2] Burns, Jr. Robert E., CHP. Shonka Research Associates. Senior Health Physicist. March 2007.
The nominal diameter for the uranium metal in a Hanford slug was 1.36 in. (e.g. see duPont 1945 or Clayton 1954) with the aluminum can having an ID of 1.37 in. (duPont

1945). Other references assert the slug diameter was 1.37 in. (e.g. see Ballinger and Hall 1991). The slug diameter was taken to be 1.37 in. for conservatism and simplicity.

The total thickness of 0.125 in. for the can bottom and end cap (IAEA 1959) reflects an assumption that the total thickness is the same for Hanford slugs as those used in the Oak Ridge graphite reactor. The aluminum cans for the Hanford and ORNL slugs were the same material and wall thickness. Only the total thickness is needed to compute the mass of the aluminum required for the ORIGEN calculation. Other dimensions for the aluminum cans used for the Hanford slugs are from (duPont 1945).

[3] Burns, Jr. Robert E., CHP. Shonka Research Associates. Senior Health Physicist. March 2007.
The average specific power for the Hanford single pass reactors was 1 MW/ton. The target burnup was 200 MWd/ton (Heeb 1993). The irradiation time is broken up into smaller segments to insure that plutonium ingrowth in sufficiently accounted for in the fuel composition. Calculation of fission product inventory could be biased otherwise.

[4] Burns, Jr. Robert E., CHP. Shonka Research Associates. Senior Health Physicist. March 2007.
The average specific power for a PWR is 37.5 MW/MT (Ludwig and Renier 1989). The irradiation time is broken up into smaller segments to ensure that plutonium ingrowth in sufficiently accounted for in the fuel composition. Calculation of fission product inventory could be biased otherwise.

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# ATTACHMENT A
# REACTOR SOURCE TERMS


Table A-1. Relative ratios for activation products for different reactor models at 10 days post-shutdown (ratio to Cs-137 ≥ 0.001).

| Nuclide | ATR | FFTF | N Reactor 1 | N Reactor 2 | PWR | TRIGA (Al-ATR) | TRIGA (Al-PWR) | TRIGA (SS-ATR) | TRIGA (SS-PWR) | W Reactor 1 | W Reactor 2 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| H-3 | -- | -- | -- | -- | 0.003 | -- | -- | -- | -- | -- | -- |
| P-32 | -- | 0.004 | -- | -- | -- | 0.002 | 0.002 | 0.009 | 0.008 | -- | -- |
| Cr-51 | 0.386 | 0.252 | 1.668 | 0.857 | -- | 0.057 | 0.053 | 4.697 | 4.249 | 0.046 | 0.039 |
| Mn-54 | 0.002 | 0.835 | -- | 0.005 | -- | 0.002 | 0.002 | 0.332 | 0.227 | -- | 0.001 |
| Fe-55 | 0.007 | 0.055 | 0.046 | 0.025 | -- | 0.020 | 0.018 | 2.255 | 2.027 | 0.009 | 0.008 |
| Fe-59 | 0.003 | 0.009 | 0.012 | 0.007 | -- | -- | -- | 0.096 | 0.087 | 0.001 | 0.001 |
| Co-58 | 0.002 | 3.883 | 0.008 | 0.072 | -- | -- | -- | 0.468 | 0.464 | 0.001 | 0.012 |
| Co-60 | -- | 0.028 | 0.075 | 0.044 | 0.002 | -- | 0.055 | 0.624 | 0.910 | -- | -- |
| Ni-63 | -- | 0.001 | 0.002 | 0.001 | -- | -- | -- | 0.045 | 0.040 | -- | -- |
| Zn-65 | 0.031 | -- | -- | 0.001 | 0.005 | 0.005 | 0.004 | 0.004 | 0.002 | 0.002 | |
| Sr-89 | -- | -- | 0.002 | 0.019 | -- | 0.002 | 0.001 | 0.002 | 0.002 | -- | -- |
| Y-90 | -- | -- | 0.008 | 0.076 | -- | 0.005 | 0.004 | 0.007 | 0.005 | -- | -- |
| Y-91 | -- | -- | 0.005 | 0.046 | -- | 0.005 | 0.003 | 0.006 | 0.004 | -- | -- |
| Nb-92 | -- | 0.005 | -- | -- | -- | -- | -- | -- | -- | -- | |
| Zr-95 | -- | -- | 17.537 | 21.635 | 0.002 | 1.564 | 1.567 | 2.099 | 2.049 | -- | 0.001 |
| Zr-97 | -- | -- | -- | 0.003 | -- | -- | -- | -- | -- | -- | -- |
| Nb-95 | -- | 0.004 | 13.471 | 16.937 | 0.002 | 1.719 | 1.723 | 2.307 | 2.253 | -- | 0.001 |
| Nb-95m | -- | -- | 0.128 | 0.158 | -- | 0.012 | 0.012 | 0.015 | 0.015 | -- | -- |
| Nb-97 | -- | -- | -- | 0.003 | -- | -- | -- | -- | -- | -- | -- |
| Nb-97m | -- | -- | -- | 0.003 | -- | -- | -- | -- | -- | -- | -- |
| Mo-99 | -- | 0.093 | -- | 0.012 | -- | -- | -- | -- | -- | -- | -- |
| In-113m | -- | -- | 0.156 | 0.255 | -- | 0.031 | 0.029 | 0.043 | 0.038 | -- | -- |
| In-114 | -- | -- | -- | 0.001 | -- | -- | -- | -- | -- | -- | -- |
| In-114m | -- | -- | -- | 0.001 | -- | -- | -- | -- | -- | -- | -- |
| Sn-113 | -- | -- | 0.156 | 0.255 | -- | 0.031 | 0.029 | 0.043 | 0.038 | -- | -- |
| Sn-117m | 0.013 | -- | 1.101 | 2.620 | -- | 0.156 | 0.144 | 0.211 | 0.190 | -- | -- |
| Sn-119m | 0.002 | -- | 0.473 | 1.150 | -- | 0.255 | 0.235 | 0.339 | 0.304 | -- | -- |
| Sn-121 | -- | -- | 0.003 | 0.004 | -- | -- | -- | -- | -- | -- | -- |
| Sn-123 | -- | -- | 0.104 | 0.093 | -- | 0.012 | 0.012 | 0.016 | 0.015 | -- | -- |
| Sn-125 | 0.004 | -- | 0.296 | 0.722 | -- | 0.039 | 0.040 | 0.053 | 0.053 | -- | -- |
| Sb-122 | -- | -- | -- | 0.002 | -- | -- | -- | -- | -- | -- | -- |
| Sb-125 | -- | -- | 0.075 | 0.171 | -- | 0.100 | 0.096 | 0.107 | 0.102 | -- | -- |
| Sb-126 | -- | -- | -- | 0.002 | -- | -- | -- | -- | -- | -- | -- |
| Te-125m | -- | -- | 0.008 | 0.018 | -- | 0.023 | 0.022 | 0.024 | 0.022 | -- | -- |
| Sm-153 | -- | -- | -- | -- | -- | 0.001 | 0.001 | 0.002 | 0.002 | -- | -- |
| Eu-154 | -- | -- | -- | -- | -- | 0.002 | 0.002 | 0.001 | 0.001 | -- | -- |
| Eu-156 | -- | -- | -- | -- | 0.001 | -- | -- | -- | -- | -- | |
| Hf-175 | -- | -- | 0.063 | 0.039 | -- | -- | -- | -- | -- | -- | -- |
| Hf-181 | -- | -- | 0.652 | 0.518 | -- | 0.006 | 0.005 | 0.008 | 0.007 | -- | -- |
| Ta-182 | -- | 0.136 | 0.001 | 0.002 | -- | 0.699 | 0.660 | 1.006 | 0.924 | -- | -- |
| Ta-183 | -- | 0.003 | 0.001 | -- | -- | 0.050 | 0.044 | 0.063 | 0.052 | -- | -- |
| W-185 | -- | -- | 0.019 | 0.021 | -- | 0.004 | 0.003 | 0.005 | 0.004 | -- | -- |
| W-188 | -- | -- | -- | 0.001 | -- | -- | -- | -- | -- | -- | -- |
| Re-188 | -- | -- | -- | 0.001 | -- | -- | -- | -- | -- | -- | -- |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


Table A-2. Relative ratios for fission products for different reactor models at 10 days post-shutdown (ratio to Cs-137 ≥ 0.001).

| Nuclide | ATR | FFTF | N Reactor 1 | N Reactor 2 | PWR | TRIGA (Al-ATR) | TRIGA (Al-PWR) | TRIGA (SS-ATR) | TRIGA (SS-PWR) | W Reactor 1 | W Reactor 2 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| H-3 | 5.096 | 0.007 | 0.005 | 0.005 | 0.005 | 0.004 | 0.004 | 0.004 | 0.004 | 0.005 | 0.006 |
| As-77 | 0.011 | 0.001 | 0.004 | 0.004 | -- | -- | -- | -- | -- | 0.002 | 0.002 |
| Br-82 | 0.003 | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- |
| Rb-86 | 0.061 | 0.056 | 0.007 | 0.009 | 0.014 | 0.001 | 0.001 | -- | 0.001 | 0.001 | 0.002 |
| Sr-89 | 116.610 | 8.709 | 75.295 | 67.764 | 7.353 | 5.582 | 5.593 | 8.723 | 8.739 | 49.256 | 44.986 |
| Sr-90 | 0.963 | 0.365 | 0.905 | 0.815 | 0.705 | 0.948 | 0.950 | 0.953 | 0.954 | 0.940 | 0.857 |
| Y-90 | 0.964 | 0.368 | 0.905 | 0.814 | 0.708 | 0.949 | 0.950 | 0.953 | 0.954 | 0.941 | 0.858 |
| Y-91 | 129.709 | 12.147 | 88.107 | 80.658 | 9.895 | 6.996 | 7.013 | 10.922 | 10.939 | 59.830 | 55.432 |
| Zr-95 | 133.134 | 22.513 | 95.983 | 93.098 | 14.228 | 7.804 | 7.815 | 12.140 | 12.149 | 65.443 | 63.844 |
| Nb-95 | 61.952 | 24.634 | 73.878 | 71.562 | 15.829 | 8.618 | 8.629 | 13.401 | 13.409 | 64.465 | 62.949 |
| Nb-95m | 0.961 | 0.167 | 0.703 | 0.682 | 0.105 | 0.058 | 0.058 | 0.090 | 0.090 | 0.484 | 0.472 |
| Zr-97 | 0.021 | 0.001 | 0.008 | 0.008 | -- | -- | -- | -- | -- | 0.004 | 0.004 |
| Nb-97 | 0.021 | 0.001 | 0.008 | 0.008 | -- | -- | -- | -- | -- | 0.004 | 0.004 |
| Nb-97m | 0.020 | 0.001 | 0.008 | 0.008 | -- | -- | -- | -- | -- | 0.004 | 0.004 |
| Mo-99 | 33.844 | 2.502 | 12.586 | 12.822 | 1.459 | 0.658 | 0.661 | 1.020 | 1.023 | 6.316 | 6.410 |
| Tc-99m | 32.603 | 2.411 | 12.124 | 12.354 | 1.407 | 0.634 | 0.636 | 0.982 | 0.985 | 6.085 | 6.175 |
| Ru-103 | 88.014 | 29.656 | 64.663 | 75.923 | 13.269 | 3.774 | 3.757 | 5.764 | 5.748 | 35.748 | 42.130 |
| Rh-103m | 79.375 | 26.731 | 58.283 | 68.432 | 11.965 | 3.402 | 3.387 | 5.196 | 5.181 | 32.232 | 37.974 |
| Rh-105 | 0.472 | 0.274 | 0.422 | 0.675 | 0.109 | 0.017 | 0.017 | 0.025 | 0.025 | 0.180 | 0.282 |
| Ru-106 | 1.938 | 13.469 | 3.410 | 5.829 | 5.438 | 0.698 | 0.683 | 0.924 | 0.911 | 2.363 | 4.438 |
| Rh-106 | 1.938 | 13.469 | 3.410 | 5.829 | 5.438 | 0.698 | 0.683 | 0.924 | 0.911 | 2.363 | 4.438 |
| Ag-110m | 0.003 | 0.058 | 0.001 | 0.003 | 0.039 | -- | -- | -- | -- | -- | -- |
| Ag-111 | 0.714 | 0.822 | 0.792 | 1.716 | 0.306 | 0.022 | 0.021 | 0.031 | 0.030 | 0.288 | 0.831 |
| Cd-115 | 0.047 | 0.025 | 0.036 | 0.093 | 0.009 | 0.001 | 0.001 | 0.002 | 0.002 | 0.017 | 0.059 |
| Cd-115m | 0.037 | 0.047 | 0.051 | 0.138 | 0.016 | 0.002 | 0.002 | 0.003 | 0.003 | 0.030 | 0.105 |
| In-115m | 0.051 | 0.028 | 0.039 | 0.101 | 0.009 | 0.001 | 0.001 | 0.002 | 0.002 | 0.019 | 0.064 |
| Sn-119m | 0.002 | 0.005 | 0.003 | 0.007 | 0.002 | -- | -- | -- | -- | 0.002 | 0.007 |
| Sn-121 | 0.003 | -- | 0.002 | 0.004 | -- | -- | -- | -- | -- | -- | 0.002 |
| Sb-122 | 0.006 | 0.002 | -- | 0.001 | 0.002 | -- | -- | -- | -- | -- | -- |
| Sn-123 | 0.068 | 0.107 | 0.081 | 0.167 | 0.035 | 0.008 | 0.008 | 0.013 | 0.013 | 0.062 | 0.158 |
| Sb-124 | 0.011 | 0.015 | 0.001 | 0.005 | 0.012 | -- | -- | -- | -- | -- | 0.001 |
| Sn-125 | 0.497 | 0.176 | 0.334 | 0.654 | 0.068 | 0.012 | 0.012 | 0.019 | 0.018 | 0.150 | 0.366 |
| Sb-125 | 0.065 | 0.311 | 0.102 | 0.221 | 0.138 | 0.048 | 0.047 | 0.056 | 0.055 | 0.094 | 0.251 |
| Te-125m | 0.004 | 0.062 | 0.011 | 0.023 | 0.029 | 0.011 | 0.011 | 0.012 | 0.012 | 0.014 | 0.038 |
| Sb-126 | 0.041 | 0.035 | 0.021 | 0.035 | 0.007 | -- | -- | 0.001 | 0.001 | 0.009 | 0.016 |
| Sb-127 | 1.875 | 0.494 | 0.954 | 1.527 | 0.189 | 0.041 | 0.040 | 0.061 | 0.061 | 0.413 | 0.757 |
| Te-127 | 2.097 | 0.840 | 1.234 | 1.982 | 0.320 | 0.071 | 0.070 | 0.107 | 0.105 | 0.620 | 1.136 |
| Te-127m | 0.307 | 0.374 | 0.327 | 0.529 | 0.142 | 0.032 | 0.032 | 0.049 | 0.048 | 0.229 | 0.420 |
| Te-129 | 2.030 | 0.579 | 1.369 | 1.709 | 0.256 | 0.077 | 0.077 | 0.119 | 0.118 | 0.726 | 0.932 |
| Te-129m | 3.118 | 0.890 | 2.103 | 2.625 | 0.393 | 0.119 | 0.118 | 0.182 | 0.181 | 1.115 | 1.432 |
| Te-131 | 0.022 | 0.003 | 0.009 | 0.011 | 0.001 | -- | -- | -- | -- | 0.004 | 0.005 |
| Te-131m | 0.099 | 0.012 | 0.042 | 0.047 | 0.006 | 0.002 | 0.002 | 0.003 | 0.003 | 0.019 | 0.021 |
| I-131 | 84.452 | 8.907 | 33.578 | 36.490 | 4.285 | 1.725 | 1.723 | 2.661 | 2.658 | 16.456 | 17.934 |
| Te-132 | 35.839 | 3.211 | 13.618 | 14.492 | 1.663 | 0.701 | 0.700 | 1.085 | 1.085 | 6.723 | 7.152 |
| I-132 | 36.926 | 3.307 | 14.030 | 14.930 | 1.713 | 0.722 | 0.721 | 1.118 | 1.117 | 6.927 | 7.370 |
| I-133 | 0.159 | 0.012 | 0.061 | 0.062 | 0.007 | 0.003 | 0.003 | 0.005 | 0.005 | 0.030 | 0.031 |
| Cs-134 | 0.623 | 0.479 | 0.133 | 0.146 | 1.553 | 0.261 | 0.239 | 0.171 | 0.153 | 0.029 | 0.041 |
| Cs-136 | 0.364 | 0.607 | 0.329 | 0.592 | 0.304 | 0.044 | 0.055 | 0.042 | 0.051 | 0.108 | 0.188 |
| Ba-136m | 0.060 | 0.100 | 0.054 | 0.098 | 0.050 | 0.007 | 0.009 | 0.007 | 0.008 | 0.018 | 0.031 |
| Cs-137 | 1.000 | 1.000 | 1.000 | 1.000 | 1.000 | 1.000 | 1.000 | 1.000 | 1.000 | 1.000 | 1.000 |
| Ba-137m | 0.946 | 0.946 | 0.946 | 0.946 | 0.946 | 0.946 | 0.946 | 0.946 | 0.946 | 0.946 | 0.946 |
| Ba-140 | 220.120 | 16.199 | 91.678 | 90.021 | 9.770 | 4.905 | 4.909 | 7.621 | 7.626 | 46.704 | 45.621 |
| La-140 | 252.312 | 18.588 | 105.251 | 103.344 | 11.218 | 5.625 | 5.631 | 8.739 | 8.748 | 53.565 | 52.330 |
| Ce-141 | 181.421 | 23.391 | 106.983 | 105.725 | 12.943 | 6.510 | 6.516 | 10.102 | 10.110 | 60.638 | 59.573 |
| Ce-143 | 2.670 | 0.150 | 0.961 | 0.929 | 0.094 | 0.052 | 0.052 | 0.081 | 0.081 | 0.494 | 0.474 |
| Pr-143 | 227.055 | 14.993 | 94.566 | 91.065 | 9.406 | 5.235 | 5.241 | 8.129 | 8.136 | 49.142 | 47.035 |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


| Nuclide | ATR | FFTF | N Reactor 1 | N Reactor 2 | PWR | TRIGA (Al-ATR) | TRIGA (Al-PWR) | TRIGA (SS-ATR) | TRIGA (SS-PWR) | W Reactor 1 | W Reactor 2 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Ce-144 | 31.918 | 13.213 | 28.774 | 27.421 | 11.831 | 7.111 | 7.122 | 10.643 | 10.643 | 26.223 | 25.177 |
| Pr-144 | 31.918 | 13.213 | 28.774 | 27.421 | 11.831 | 7.111 | 7.122 | 10.643 | 10.651 | 26.223 | 25.177 |
| Pr-144m | 0.383 | 0.159 | 0.345 | 0.329 | 0.142 | 0.085 | 0.085 | 0.128 | 0.128 | 0.315 | 0.302 |
| Nd-147 | 72.149 | 6.034 | 31.137 | 31.487 | 3.418 | 1.662 | 1.662 | 2.580 | 2.580 | 15.790 | 16.174 |
| Pm-147 | 2.509 | 2.741 | 3.507 | 3.502 | 1.264 | 2.105 | 2.063 | 2.674 | 2.644 | 3.766 | 3.834 |
| Pm-148 | 2.818 | 0.444 | 1.187 | 1.150 | 0.773 | 0.097 | 0.115 | 0.107 | 0.125 | 0.161 | 0.213 |
| Pm-148m | 0.399 | 1.358 | 0.341 | 0.425 | 0.219 | 0.112 | 0.116 | 0.125 | 0.129 | 0.090 | 0.176 |
| Pm-149 | 4.813 | 0.334 | 1.372 | 1.521 | 0.273 | 0.070 | 0.070 | 0.107 | 0.107 | 0.656 | 0.726 |
| Pm-151 | 0.080 | 0.013 | 0.036 | 0.044 | 0.006 | 0.002 | 0.002 | 0.003 | 0.003 | 0.017 | 0.021 |
| Sm-151 | 0.004 | 0.039 | 0.005 | 0.018 | 0.004 | 0.008 | 0.009 | 0.012 | 0.013 | 0.014 | 0.025 |
| Sm-153 | 1.260 | 0.069 | 0.218 | 0.250 | 0.140 | 0.016 | 0.013 | 0.017 | 0.015 | 0.073 | 0.095 |
| Eu-154 | 0.031 | 0.029 | 0.010 | 0.007 | 0.088 | 0.015 | 0.011 | 0.008 | 0.006 | 0.002 | 0.002 |
| Eu-155 | 0.022 | 0.166 | 0.026 | 0.054 | 0.056 | 0.020 | 0.019 | 0.023 | 0.023 | 0.034 | 0.059 |
| Eu-156 | 1.997 | 0.662 | 1.079 | 1.265 | 1.173 | 0.042 | 0.041 | 0.050 | 0.049 | 0.279 | 0.419 |
| Tb-160 | 0.002 | 0.024 | -- | 0.001 | 0.009 | -- | -- | -- | -- | -- | -- |
| Tb-161 | 0.005 | 0.020 | 0.010 | 0.021 | 0.006 | -- | -- | -- | -- | 0.003 | 0.008 |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**
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Table A-3. ATR activity relative to Cs-137.

| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| H-3 | 4.363E-03 | 4.350E-03 | 4.345E-03 | 4.331E-03 | 4.298E-03 | 4.225E-03 | 3.956E-03 |
| Be-10 | 2.989E-11 | 2.995E-11 | 3.000E-11 | 3.005E-11 | 3.023E-11 | 3.057E-11 | 3.203E-11 |
| C-14 | 1.641E-09 | 1.644E-09 | 1.647E-09 | 1.649E-09 | 1.659E-09 | 1.678E-09 | 1.758E-09 |
| Na-24 | 5.396E-05 | 1.920E-19 | 4.479E-29 | -- | -- | -- | -- |
| Mg-28 | 6.845E-11 | 2.960E-21 | 3.648E-28 | 1.573E-38 | -- | -- | -- |
| Al-28 | 6.857E-11 | 2.965E-21 | 3.655E-28 | -- | -- | -- | -- |
| Si-31 | 9.238E-30 | -- | -- | -- | -- | -- | -- |
| Si-32 | 5.370E-12 | 5.379E-12 | 5.387E-12 | 5.397E-12 | 5.428E-12 | 5.487E-12 | 5.736E-12 |
| P-32 | 1.484E-06 | 3.472E-07 | 1.319E-07 | 3.087E-08 | 4.012E-10 | 5.538E-12 | 5.737E-12 |
| P-33 | 2.693E-17 | 1.174E-17 | 6.754E-18 | 2.945E-18 | 2.441E-19 | 1.264E-21 | 2.117E-30 |
| Ar-39 | 2.267E-22 | 2.271E-22 | 2.275E-22 | 2.279E-22 | 2.291E-22 | 2.314E-22 | 2.412E-22 |
| Ar-42 | 1.081E-19 | 1.081E-19 | 1.082E-19 | 1.082E-19 | 1.083E-19 | 1.084E-19 | 1.089E-19 |
| K-42 | 3.854E-19 | 1.081E-19 | 1.082E-19 | 1.082E-19 | 1.083E-19 | 1.084E-19 | 1.089E-19 |
| K-43 | 6.746E-13 | 1.736E-22 | 7.024E-29 | 1.806E-38 | -- | -- | -- |
| Ca-45 | 1.849E-06 | 1.630E-06 | 1.500E-06 | 1.323E-06 | 9.074E-07 | 4.173E-07 | 1.956E-08 |
| Sc-46 | 4.113E-04 | 3.215E-04 | 2.729E-04 | 2.133E-04 | 1.019E-04 | 2.228E-05 | 5.543E-08 |
| Ca-47 | 1.801E-08 | 1.842E-10 | 8.686E-12 | 8.881E-14 | 9.506E-20 | 4.884E-32 | -- |
| Sc-47 | 4.015E-04 | 8.109E-07 | 1.299E-08 | 2.650E-11 | 5.800E-19 | 1.870E-31 | -- |
| Sc-48 | 6.871E-06 | 7.761E-11 | 3.910E-14 | 4.417E-19 | 6.371E-34 | -- | -- |
| V-50 | 7.027E-21 | 7.039E-21 | 7.052E-21 | 7.064E-21 | 7.106E-21 | 7.187E-21 | 7.530E-21 |
| Cr-51 | 3.864E-01 | 1.828E-01 | 1.110E-01 | 5.250E-02 | 5.558E-03 | 5.461E-05 | 6.626E-13 |
| Mn-54 | 2.073E-03 | 1.943E-03 | 1.862E-03 | 1.745E-03 | 1.438E-03 | 9.641E-04 | 1.998E-04 |
| Fe-55 | 7.470E-03 | 7.321E-03 | 7.228E-03 | 7.083E-03 | 6.673E-03 | 5.895E-03 | 3.624E-03 |
| Mn-56 | 8.741E-28 | -- | -- | -- | -- | -- | -- |
| Co-58 | 1.582E-03 | 1.182E-03 | 9.734E-04 | 7.265E-04 | 3.028E-04 | 4.989E-05 | 4.085E-08 |
| Fe-59 | 3.136E-03 | 1.979E-03 | 1.457E-03 | 9.191E-04 | 2.312E-04 | 1.348E-05 | 1.833E-10 |
| Ni-59 | 3.296E-07 | 3.302E-07 | 3.308E-07 | 3.313E-07 | 3.333E-07 | 3.371E-07 | 3.532E-07 |
| Co-60 | 3.630E-05 | 3.597E-05 | 3.577E-05 | 3.545E-05 | 3.453E-05 | 3.266E-05 | 2.631E-05 |
| Ni-63 | 6.449E-05 | 6.457E-05 | 6.466E-05 | 6.472E-05 | 6.500E-05 | 6.548E-05 | 6.758E-05 |
| Cu-64 | 1.336E-05 | 1.148E-22 | 4.823E-34 | -- | -- | -- | -- |
| Ni-65 | 1.801E-31 | -- | -- | -- | -- | -- | -- |
| Zn-65 | 3.096E-02 | 2.847E-02 | 2.695E-02 | 2.478E-02 | 1.931E-02 | 1.153E-02 | 1.516E-03 |
| Ni- | 66 | 4.037E-08 | 4.345E-12 | 9.837E-15 | 1.059E-18 | 1.321E-30 | -- |
| Cu-66 | 4.044E-08 | 4.352E-12 | 9.854E-15 | 1.060E-18 | 1.323E-30 | -- | -- |
| Cu-67 | 1.931E-06 | 6.065E-10 | 2.804E-12 | 8.812E-16 | 2.732E-26 | -- | -- |
| Zn-69 | 1.092E-07 | 1.944E-23 | 6.156E-34 | -- | -- | -- | -- |
| Zn-69m | 1.017E-07 | 1.810E-23 | 5.733E-34 | -- | -- | -- | -- |
| Zn-71 | 7.932E-27 | -- | -- | -- | -- | -- | -- |
| Zn-71m | 1.570E-23 | -- | -- | -- | -- | -- | -- |
| Ge-71 | 4.724E-07 | 8.123E-08 | 2.513E-08 | 4.322E-09 | 2.200E-11 | 4.182E-16 | 1.014E-34 |
| Zn-72 | 1.369E-04 | 2.992E-09 | 2.341E-12 | 5.116E-17 | 5.349E-31 | -- | -- |
| Ga-72 | 1.965E-04 | 4.294E-09 | 3.359E-12 | 7.343E-17 | 7.677E-31 | -- | -- |
| Ga-73 | 2.640E-17 | -- | -- | -- | -- | -- | -- |
| Ge-73m | 2.640E-17 | -- | -- | -- | -- | -- | -- |
| As-76 | 5.158E-06 | 3.007E-14 | 9.742E-20 | 5.682E-28 | -- | -- | -- |
| Ge-77 | 9.692E-08 | 6.401E-27 | 1.002E-39 | -- | -- | -- | -- |
| As-77 | 1.091E-02 | 2.838E-08 | 5.373E-12 | 1.398E-17 | 2.468E-34 | -- | -- |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Se-77m | 2.705E-05 | 7.039E-11 | 1.332E-14 | 3.469E-20 | 7.825E-37 | -- | -- |
| Se-79 | 4.003E-06 | 4.009E-06 | 4.016E-06 | 4.023E-06 | 4.048E-06 | 4.094E-06 | 4.289E-06 |
| Kr-79 | 2.634E-13 | 1.625E-19 | 1.179E-23 | 7.275E-30 | -- | -- | -- |
| Br-80 | 9.418E-22 | -- | -- | -- | -- | -- | -- |
| Br-80m | 8.801E-22 | -- | -- | -- | -- | -- | -- |
| Kr-81 | 2.800E-13 | 2.804E-13 | 2.809E-13 | 2.814E-13 | 2.831E-13 | 2.863E-13 | 3.000E-13 |
| Br-82 | 2.789E-03 | 2.029E-09 | 1.642E-13 | 1.194E-19 | 4.600E-38 | -- | -- |
| Br-83 | 2.299E-29 | -- | -- | -- | -- | -- | -- |
| Kr-83m | 9.803E-29 | -- | -- | -- | -- | -- | -- |
| Kr-85 | 1.235E-01 | 1.231E-01 | 1.229E-01 | 1.224E-01 | 1.212E-01 | 1.187E-01 | 1.092E-01 |
| Kr-85m | 6.503E-15 | -- | -- | -- | -- | -- | -- |
| Rb-86 | 6.107E-02 | 2.007E-02 | 9.562E-03 | 3.143E-03 | 1.116E-04 | 1.164E-07 | 1.994E-19 |
| Rb-87 | 2.690E-10 | 2.695E-10 | 2.699E-10 | 2.704E-10 | 2.720E-10 | 2.751E-10 | 2.883E-10 |
| Sr-87m | 1.921E-30 | -- | -- | -- | -- | -- | -- |
| Kr-88 | 8.716E-24 | -- | -- | -- | -- | -- | -- |
| Rb-88 | 9.735E-24 | -- | -- | -- | -- | -- | -- |
| Sr-89 | 1.166E+02 | 7.738E+01 | 5.891E+01 | 3.909E+01 | 1.144E+01 | 9.098E-01 | 4.210E-05 |
| Sr-90 | 9.632E-01 | 9.631E-01 | 9.639E-01 | 9.630E-01 | 9.636E-01 | 9.632E-01 | 9.615E-01 |
| Y-90 | 9.641E-01 | 9.640E-01 | 9.639E-01 | 9.639E-01 | 9.636E-01 | 9.632E-01 | 9.624E-01 |
| Y-90m | 4.582E-27 | -- | -- | -- | -- | -- | -- |
| Sr-91 | 9.795E-06 | -- | -- | -- | -- | -- | -- |
| Y-91 | 1.297E+02 | 9.108E+01 | 7.198E+01 | 5.054E+01 | 1.751E+01 | 1.972E+00 | 3.603E-04 |
| Y-91m | 6.225E-06 | -- | -- | -- | -- | -- | -- |
| Sr-92 | 8.836E-25 | -- | -- | -- | -- | -- | -- |
| Y-92 | 6.625E-18 | -- | -- | -- | -- | -- | -- |
| Nb-92 | 1.449E-12 | 1.876E-13 | 4.800E-14 | 6.211E-15 | 1.345E-17 | 4.414E-23 | -- |
| Y-93 | 3.088E-05 | 1.074E-26 | -- | -- | -- | -- | -- |
| Zr-93 | 2.062E-05 | 2.065E-05 | 2.069E-05 | 2.072E-05 | 2.085E-05 | 2.109E-05 | 2.209E-05 |
| Nb-93m | 7.966E-08 | 1.614E-07 | 2.160E-07 | 2.977E-07 | 5.429E-07 | 1.046E-06 | 3.023E-06 |
| Nb-94 | 2.075E-10 | 2.079E-10 | 2.082E-10 | 2.086E-10 | 2.099E-10 | 2.123E-10 | 2.224E-10 |
| Zr-95 | 1.331E+02 | 9.631E+01 | 7.771E+01 | 5.624E+01 | 2.134E+01 | 2.901E+00 | 1.111E-03 |
| Nb-95 | 6.195E+01 | 8.440E+01 | 8.508E+01 | 7.639E+01 | 3.911E+01 | 6.222E+00 | 2.467E-03 |
| Nb-95m | 9.615E-01 | 7.147E-01 | 5.765E-01 | 4.172E-01 | 1.584E-01 | 2.152E-02 | 8.244E-06 |
| Nb-96 | 5.165E-04 | 2.702E-13 | 1.754E-19 | 9.174E-29 | -- | -- | -- |
| Zr-97 | 2.139E-02 | 3.206E-15 | 9.055E-24 | 1.357E-36 | -- | -- | -- |
| Nb-97 | 2.150E-02 | 3.455E-15 | 9.751E-24 | 1.463E-36 | -- | -- | -- |
| Nb-97m | 2.026E-02 | 3.037E-15 | 8.574E-24 | 1.361E-36 | -- | -- | -- |
| Tc-98 | 1.519E-11 | 1.521E-11 | 1.524E-11 | 1.527E-11 | 1.536E-11 | 1.553E-11 | 1.628E-11 |
| Mo-99 | 3.384E+01 | 1.763E-02 | 1.143E-04 | 5.951E-08 | 8.420E-18 | 4.482E-38 | -- |
| Tc-99 | 1.336E-04 | 1.351E-04 | 1.353E-04 | 1.355E-04 | 1.364E-04 | 1.379E-04 | 1.445E-04 |
| Tc-99m | 3.260E+01 | 1.698E-02 | 1.101E-04 | 5.733E-08 | 8.112E-18 | 4.346E-38 | -- |
| Rh-102 | 9.110E-07 | 8.954E-07 | 8.849E-07 | 8.692E-07 | 8.244E-07 | 7.386E-07 | 4.798E-07 |
| Ru-103 | 8.801E+01 | 5.195E+01 | 3.656E+01 | 2.157E+01 | 4.435E+00 | 1.707E-01 | 4.512E-07 |
| Rh-103m | 7.938E+01 | 4.684E+01 | 3.296E+01 | 1.945E+01 | 3.997E+00 | 1.539E-01 | 4.068E-07 |
| Ru-105 | 3.848E-15 | -- | -- | -- | -- | -- | -- |
| Rh-105 | 4.723E-01 | 3.512E-07 | 2.884E-11 | 2.145E-17 | 8.831E-36 | -- | -- |
| Rh-105m | 1.080E-15 | -- | -- | -- | -- | -- | -- |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Ru-106 | 1.938E+00 | 1.834E+00 | 1.769E+00 | 1.675E+00 | 1.423E+00 | 1.015E+00 | 2.688E-01 |
| Rh-106 | 1.938E+00 | 1.834E+00 | 1.769E+00 | 1.675E+00 | 1.423E+00 | 1.015E+00 | 2.688E-01 |
| Rh-106m | 1.313E-32 | -- | -- | -- | -- | -- | -- |
| Ag-106 | 7.852E-14 | 6.812E-15 | 1.336E-15 | 1.158E-16 | 7.572E-20 | 2.106E-26 | -- |
| Pd-107 | 1.454E-07 | 1.456E-07 | 1.459E-07 | 1.461E-07 | 1.470E-07 | 1.487E-07 | 1.558E-07 |
| Ag-108 | 5.455E-14 | 5.462E-14 | 5.470E-14 | 5.477E-14 | 5.502E-14 | 5.550E-14 | 5.751E-14 |
| Ag-108m | 6.129E-13 | 6.137E-13 | 6.146E-13 | 6.153E-13 | 6.183E-13 | 6.236E-13 | 6.462E-13 |
| Pd-109 | 1.585E-05 | 1.257E-21 | 2.323E-32 | -- | -- | -- | -- |
| Ag-109m | 1.587E-05 | 2.383E-10 | 2.317E-10 | 2.219E-10 | 1.952E-10 | 1.496E-10 | 5.266E-11 |
| Cd-109 | 2.488E-10 | 2.383E-10 | 2.317E-10 | 2.219E-10 | 1.952E-10 | 1.496E-10 | 5.266E-11 |
| Ag-110 | 4.354E-05 | 4.013E-05 | 3.802E-05 | 3.505E-05 | 2.747E-05 | 1.662E-05 | 2.295E-06 |
| Ag-110m | 3.273E-03 | 3.017E-03 | 2.859E-03 | 2.635E-03 | 2.066E-03 | 1.250E-03 | 1.726E-04 |
| Pd-111 | 8.973E-16 | -- | -- | -- | -- | -- | -- |
| Pd-111m | 1.232E-15 | -- | -- | -- | -- | -- | -- |
| Ag-111 | 7.145E-01 | 4.391E-02 | 6.842E-03 | 4.205E-04 | 9.766E-08 | 3.234E-15 | -- |
| Ag-111m | 1.289E-15 | -- | -- | -- | -- | -- | -- |
| Pd-112 | 3.335E-04 | 5.503E-15 | 3.570E-22 | 5.892E-33 | -- | -- | -- |
| Ag-112 | 3.950E-04 | 6.519E-15 | 4.229E-22 | 6.978E-33 | -- | -- | -- |
| Ag-113 | 2.640E-14 | -- | -- | -- | -- | -- | -- |
| Cd-113m | 1.286E-04 | 1.283E-04 | 1.282E-04 | 1.279E-04 | 1.272E-04 | 1.256E-04 | 1.196E-04 |
| In-113m | 5.928E-04 | 4.957E-04 | 4.402E-04 | 3.681E-04 | 2.154E-04 | 7.139E-05 | 9.193E-07 |
| Sn-113 | 5.925E-04 | 4.955E-04 | 4.399E-04 | 3.679E-04 | 2.152E-04 | 7.135E-05 | 9.183E-07 |
| In-114 | 1.462E-05 | 9.629E-06 | 7.287E-06 | 4.796E-06 | 1.369E-06 | 1.035E-07 | 3.927E-12 |
| In-114m | 1.528E-05 | 1.006E-05 | 7.614E-06 | 5.012E-06 | 1.430E-06 | 1.081E-07 | 4.104E-12 |
| Cd-115 | 4.672E-02 | 4.140E-06 | 8.230E-09 | 7.291E-13 | 5.074E-25 | -- | -- |
| Cd-115m | 3.678E-02 | 2.311E-02 | 1.697E-02 | 1.066E-02 | 2.648E-03 | 1.504E-04 | 1.847E-09 |
| In-115 | 9.666E-17 | 1.008E-16 | 1.025E-16 | 1.042E-16 | 1.068E-16 | 1.086E-16 | 1.139E-16 |
| In-115m | 5.081E-02 | 6.126E-06 | 1.202E-06 | 7.494E-07 | 1.861E-07 | 1.057E-08 | 1.298E-13 |
| Cd-117 | 1.188E-28 | -- | -- | -- | -- | -- | -- |
| Cd-117m | 2.201E-22 | -- | -- | -- | -- | -- | -- |
| In-117 | 2.923E-22 | -- | -- | -- | -- | -- | -- |
| In-117m | 2.255E-22 | -- | -- | -- | -- | -- | -- |
| Sn-117m | 1.328E-02 | 3.013E-03 | 1.121E-03 | 2.545E-04 | 2.977E-06 | 3.138E-10 | 6.528E-26 |
| Sn-119m | 4.183E-03 | 3.849E-03 | 3.643E-03 | 3.353E-03 | 2.615E-03 | 1.566E-03 | 2.077E-04 |
| Sn-121 | 2.664E-03 | 2.182E-11 | 8.877E-17 | 7.269E-25 | -- | -- | -- |
| Sn-121m | 9.591E-07 | 9.598E-07 | 9.605E-07 | 9.612E-07 | 9.632E-07 | 9.676E-07 | 9.861E-07 |
| Sb-122 | 5.691E-03 | 2.579E-06 | 1.523E-08 | 6.900E-12 | 6.427E-22 | -- | -- |
| Sn-123 | 6.823E-02 | 5.819E-02 | 5.235E-02 | 4.465E-02 | 2.771E-02 | 1.037E-02 | 2.156E-04 |
| Te-123 | 7.813E-19 | 8.380E-19 | 8.717E-19 | 9.149E-19 | 1.010E-18 | 1.109E-18 | 1.209E-18 |
| Te-123m | 1.060E-05 | 8.926E-06 | 7.961E-06 | 6.702E-06 | 4.004E-06 | 1.385E-06 | 2.110E-08 |
| Sb-124 | 1.145E-02 | 8.114E-03 | 6.456E-03 | 4.578E-03 | 1.634E-03 | 1.958E-04 | 4.560E-08 |
| Sn-125 | 5.009E-01 | 5.803E-02 | 1.380E-02 | 1.599E-03 | 2.498E-06 | 4.147E-12 | -- |
| Sb-125 | 6.486E-02 | 6.783E-02 | 6.744E-02 | 6.629E-02 | 6.272E-02 | 5.588E-02 | 3.549E-02 |
| Te-125m | 3.836E-03 | 7.340E-03 | 9.105E-03 | 1.101E-02 | 1.353E-02 | 1.343E-02 | 8.661E-03 |
| Sn-126 | 3.557E-06 | 3.563E-06 | 3.569E-06 | 3.575E-06 | 3.597E-06 | 3.637E-06 | 3.811E-06 |
| Sb-126 | 4.122E-02 | 7.715E-03 | 2.526E-03 | 4.732E-04 | 3.604E-06 | 5.094E-07 | 5.336E-07 |
| Sb-126m | 3.557E-06 | 3.563E-06 | 3.569E-06 | 3.575E-06 | 3.597E-06 | 3.637E-06 | 3.811E-06 |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Sn-127 | 2.716E-34 | -- | -- | -- | -- | -- | -- |
| Sb-127 | 1.875E+00 | 8.467E-03 | 2.314E-04 | 1.046E-06 | 9.636E-14 | 3.181E-28 | -- |
| Te-127 | 2.097E+00 | 2.646E-01 | 2.265E-01 | 1.873E-01 | 1.063E-01 | 3.312E-02 | 3.334E-04 |
| Te-127m | 3.069E-01 | 2.619E-01 | 2.311E-01 | 1.912E-01 | 1.086E-01 | 3.381E-02 | 3.404E-04 |
| Xe-127 | 9.418E-07 | 5.328E-07 | 3.648E-07 | 2.065E-07 | 3.744E-08 | 1.114E-09 | 1.066E-15 |
| Sb-128 | 1.050E-08 | -- | -- | -- | -- | -- | -- |
| Sb-129 | 8.853E-16 | -- | -- | -- | -- | -- | -- |
| Te-129 | 2.030E+00 | 1.095E+00 | 7.260E-01 | 3.917E-01 | 6.155E-02 | 1.363E-03 | 4.072E-10 |
| Te-129m | 3.118E+00 | 1.682E+00 | 1.115E+00 | 6.017E-01 | 9.455E-02 | 2.094E-03 | 6.256E-10 |
| I-129 | 2.025E-07 | 2.113E-07 | 2.150E-07 | 2.184E-07 | 2.228E-07 | 2.258E-07 | 2.366E-07 |
| Xe-129m | 2.498E-04 | 1.860E-05 | 3.294E-06 | 2.453E-07 | 1.013E-10 | 1.096E-17 | -- |
| I-130 | 2.279E-06 | 6.674E-24 | 1.367E-35 | -- | -- | -- | -- |
| Te-131 | 2.236E-02 | 1.335E-09 | 2.040E-14 | 1.219E-21 | -- | -- | -- |
| Te-131m | 9.932E-02 | 5.931E-09 | 9.064E-14 | 5.412E-21 | -- | -- | -- |
| I-131 | 8.445E+01 | 6.372E+00 | 1.138E+00 | 8.586E-02 | 3.689E-05 | 4.327E-12 | -- |
| Xe-131m | 1.539E+00 | 4.621E-01 | 1.641E-01 | 3.124E-02 | 1.758E-04 | 3.673E-09 | 1.269E-27 |
| Te-132 | 3.584E+01 | 6.080E-02 | 8.643E-04 | 1.466E-06 | 7.132E-15 | 5.523E-32 | -- |
| I-132 | 3.693E+01 | 6.263E-02 | 8.909E-04 | 1.510E-06 | 7.349E-15 | 5.690E-32 | -- |
| Cs-132 | 6.129E-04 | 2.473E-05 | 2.912E-06 | 1.175E-07 | 7.730E-12 | 1.905E-20 | -- |
| I-133 | 1.593E-01 | 6.066E-12 | 6.868E-19 | 2.614E-29 | -- | -- | |
| Xe-133 | 1.440E+02 | 2.752E+00 | 1.961E-01 | 3.730E-03 | 2.566E-08 | 6.064E-19 | -- |
| Xe-133m | 9.521E-01 | 7.190E-05 | 1.283E-07 | 9.656E-12 | 4.125E-24 | -- | -- |
| Cs-134 | 6.230E-01 | 6.071E-01 | 5.971E-01 | 5.818E-01 | 5.388E-01 | 4.595E-01 | 2.458E-01 |
| Cs-134m | 5.863E-25 | -- | -- | -- | -- | -- | -- |
| I-135 | 5.003E-09 | -- | -- | -- | -- | -- | -- |
| Xe-135 | 1.289E-05 | -- | -- | -- | -- | -- | -- |
| Xe-135m | 8.015E-10 | -- | -- | -- | -- | -- | -- |
| Cs-135 | 5.996E-07 | 6.006E-07 | 6.016E-07 | 6.027E-07 | 6.063E-07 | 6.132E-07 | 6.425E-07 |
| Ba-135m | 3.640E-07 | 1.020E-14 | 9.407E-20 | 2.635E-27 | -- | -- | -- |
| Cs-136 | 3.645E-01 | 7.466E-02 | 2.596E-02 | 5.318E-03 | 4.577E-05 | 2.565E-09 | 4.418E-26 |
| Ba-136m | 6.006E-02 | 1.231E-02 | 4.278E-03 | 8.761E-04 | 7.542E-06 | 4.228E-10 | 7.282E-27 |
| Cs-137 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 |
| Ba-137m | 9.461E-01 | 9.460E-01 | 9.459E-01 | 9.458E-01 | 9.463E-01 | 9.457E-01 | 9.459E-01 |
| La-138 | 1.855E-15 | 1.858E-15 | 1.862E-15 | 1.865E-15 | 1.876E-15 | 1.898E-15 | 1.988E-15 |
| Ba-140 | 2.201E+02 | 4.337E+01 | 1.470E+01 | 2.896E+00 | 2.218E-02 | 9.790E-07 | 6.557E-24 |
| La-140 | 2.523E+02 | 4.991E+01 | 1.692E+01 | 3.333E+00 | 2.553E-02 | 1.126E-06 | 7.545E-24 |
| La-141 | 1.800E-16 | -- | -- | -- | -- | -- | -- |
| Ce-141 | 1.814E+02 | 9.588E+01 | 6.269E+01 | 3.312E+01 | 4.892E+00 | 9.527E-02 | 1.720E-08 |
| Ce-142 | 2.783E-10 | 2.787E-10 | 2.792E-10 | 2.797E-10 | 2.814E-10 | 2.846E-10 | 2.982E-10 |
| Pr-142 | 3.491E-04 | 1.637E-15 | 4.590E-23 | 2.152E-34 | -- | -- | -- |
| Ce-143 | 2.670E+00 | 7.233E-07 | 3.030E-11 | 8.209E-18 | 1.633E-37 | -- | -- |
| Pr-143 | 2.271E+02 | 4.917E+01 | 1.772E+01 | 3.834E+00 | 3.881E-02 | 3.039E-06 | 1.958E-22 |
| Ce-144 | 3.192E+01 | 2.972E+01 | 2.835E+01 | 2.639E+01 | 2.132E+01 | 1.373E+01 | 2.422E+00 |
| Pr-144 | 3.192E+01 | 2.972E+01 | 2.835E+01 | 2.639E+01 | 2.132E+01 | 1.373E+01 | 2.422E+00 |
| Pr-144m | 3.830E-01 | 3.566E-01 | 3.402E-01 | 3.168E-01 | 2.558E-01 | 1.647E-01 | 2.906E-02 |
| Nd-144 | 1.961E-15 | 2.800E-15 | 3.331E-15 | 4.079E-15 | 6.045E-15 | 9.019E-15 | 1.389E-14 |
| Pr-145 | 2.211E-10 | -- | -- | -- | -- | -- | -- |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**
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| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Pm-146 | 1.025E-05 | 1.016E-05 | 1.011E-05 | 1.003E-05 | 9.775E-06 | 9.273E-06 | 7.553E-06 |
| Sm-146 | 1.622E-14 | 1.931E-14 | 2.138E-14 | 2.445E-14 | 3.357E-14 | 5.174E-14 | 1.172E-13 |
| Nd-147 | 7.215E+01 | 1.103E+01 | 3.154E+00 | 4.820E-01 | 1.723E-03 | 1.581E-08 | 2.172E-28 |
| Pm-147 | 2.509E+00 | 3.156E+00 | 3.207E+00 | 3.174E+00 | 2.997E+00 | 2.651E+00 | 1.638E+00 |
| Sm-147 | 8.570E-13 | 2.423E-12 | 3.559E-12 | 5.267E-12 | 1.024E-11 | 1.968E-11 | 4.856E-11 |
| Pm-148 | 2.818E+00 | 7.190E-02 | 1.418E-02 | 5.998E-03 | 1.312E-03 | 5.921E-05 | 2.936E-10 |
| Pm-148m | 3.995E-01 | 2.419E-01 | 1.732E-01 | 1.048E-01 | 2.329E-02 | 1.052E-03 | 5.212E-09 |
| Sm-148 | 5.042E-17 | 5.780E-17 | 5.898E-17 | 6.007E-17 | 6.160E-17 | 6.262E-17 | 6.562E-17 |
| Pm-149 | 4.813E+00 | 3.982E-04 | 7.580E-07 | 6.271E-11 | 3.555E-23 | -- | -- |
| Sm-149 | 7.448E-17 | 7.752E-17 | 7.765E-17 | 7.779E-17 | 7.826E-17 | 7.915E-17 | 8.293E-17 |
| Pm-150 | 1.596E-26 | -- | -- | -- | -- | -- | -- |
| Eu-150 | 1.373E-11 | 1.373E-11 | 1.374E-11 | 1.374E-11 | 1.376E-11 | 1.378E-11 | 1.389E-11 |
| Pm-151 | 7.995E-02 | 1.857E-09 | 1.513E-14 | 3.514E-22 | -- | -- | -- |
| Sm-151 | 4.338E-03 | 4.346E-03 | 4.351E-03 | 4.355E-03 | 4.374E-03 | 4.406E-03 | 4.546E-03 |
| Eu-152 | 9.546E-06 | 9.528E-06 | 9.519E-06 | 9.492E-06 | 9.429E-06 | 9.299E-06 | 8.794E-06 |
| Eu-152m | 3.723E-11 | -- | -- | -- | -- | -- | -- |
| Gd-152 | 4.150E-19 | 4.172E-19 | 4.187E-19 | 4.209E-19 | 4.276E-19 | 4.410E-19 | 4.951E-19 |
| Sm-153 | 1.260E+00 | 2.882E-05 | 2.322E-08 | 5.308E-13 | 6.345E-27 | -- | -- |
| Gd-153 | 5.384E-05 | 4.949E-05 | 4.682E-05 | 4.304E-05 | 3.346E-05 | 1.991E-05 | 2.574E-06 |
| Eu-154 | 3.132E-02 | 3.117E-02 | 3.108E-02 | 3.093E-02 | 3.050E-02 | 2.961E-02 | 2.640E-02 |
| Eu-155 | 2.190E-02 | 2.169E-02 | 2.156E-02 | 2.135E-02 | 2.075E-02 | 1.955E-02 | 1.549E-02 |
| Sm-156 | 2.094E-08 | -- | -- | -- | -- | -- | -- |
| Eu-156 | 1.997E+00 | 5.087E-01 | 2.046E-01 | 5.209E-02 | 8.615E-04 | 1.853E-07 | 6.406E-22 |
| Eu-157 | 3.503E-05 | 1.931E-19 | 6.032E-29 | -- | -- | -- | -- |
| Gd-159 | 1.703E-05 | 3.795E-17 | 6.479E-25 | 1.443E-36 | -- | -- | -- |
| Tb-160 | 1.747E-03 | 1.313E-03 | 1.086E-03 | 8.157E-04 | 3.463E-04 | 5.931E-05 | 5.649E-08 |
| Tb-161 | 4.673E-03 | 2.319E-04 | 3.134E-05 | 1.555E-06 | 1.903E-10 | 1.682E-18 | -- |
| Tb-162m | 5.684E-37 | -- | -- | -- | -- | -- | -- |
| Dy-165 | 3.369E-34 | -- | -- | -- | -- | -- | -- |
| Dy-166 | 2.880E-05 | 6.328E-08 | 1.070E-09 | 2.347E-12 | 2.482E-20 | 9.501E-37 | -- |
| Ho-166 | 4.316E-05 | 9.425E-08 | 1.594E-09 | 3.497E-12 | 3.699E-20 | 1.416E-36 | -- |
| Ho-166m | 2.356E-10 | 2.360E-10 | 2.364E-10 | 2.368E-10 | 2.383E-10 | 2.409E-10 | 2.521E-10 |
| Er-169 | 1.781E-07 | 1.952E-08 | 4.475E-09 | 4.907E-10 | 6.495E-13 | 7.673E-19 | -- |
| Tm-170 | 1.839E-09 | 1.567E-09 | 1.409E-09 | 1.201E-09 | 7.437E-10 | 2.771E-10 | 5.660E-12 |
| Er-171 | 1.333E-24 | -- | -- | -- | -- | -- | -- |
| Tm-171 | 1.256E-11 | 1.221E-11 | 1.199E-11 | 1.166E-11 | 1.074E-11 | 9.037E-12 | 4.601E-12 |
| Er-172 | 4.296E-19 | 1.623E-23 | 1.626E-23 | 6.148E-28 | -- | -- | -- |
| Tm-172 | 5.920E-13 | 2.318E-16 | 1.241E-18 | 4.857E-22 | 2.919E-32 | -- | -- |
| Pb-204 | 2.450E-19 | 2.455E-19 | 2.459E-19 | 2.463E-19 | 2.478E-19 | 2.506E-19 | 2.626E-19 |
| Pb-205 | 1.063E-12 | 1.064E-12 | 1.066E-12 | 1.068E-12 | 1.074E-12 | 1.087E-12 | 1.139E-12 |
| Tl-206 | 2.869E-18 | 2.874E-18 | 2.879E-18 | 2.884E-18 | 2.901E-18 | 2.934E-18 | 3.074E-18 |
| Bi-208 | 4.376E-18 | 4.383E-18 | 4.391E-18 | 4.398E-18 | 4.425E-18 | 4.475E-18 | 4.689E-18 |
| Pb-209 | 1.305E-27 | -- | -- | -- | -- | -- | -- |
| Bi-210 | 7.841E-11 | 1.239E-12 | 7.807E-14 | 1.234E-15 | 4.874E-21 | 3.676E-32 | -- |
| Bi-210m | 2.881E-18 | 2.886E-18 | 2.891E-18 | 2.896E-18 | 2.913E-18 | 2.947E-18 | 3.087E-18 |
| Po-210 | 2.604E-11 | 2.494E-11 | 2.264E-11 | 1.952E-11 | 1.251E-11 | 5.005E-12 | 1.352E-13 |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


Table A-4. FFTF activity relative to Cs-137.

| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| H-3 | 6.850E-03 | 6.831E-03 | 6.820E-03 | 6.801E-03 | 6.745E-03 | 6.634E-03 | 6.209E-03 |
| Be-10 | 1.680E-10 | 1.683E-10 | 1.686E-10 | 1.689E-10 | 1.698E-10 | 1.719E-10 | 1.800E-10 |
| C-14 | 7.903E-06 | 7.918E-06 | 7.928E-06 | 7.943E-06 | 7.988E-06 | 8.081E-06 | 8.462E-06 |
| Na-22 | 2.175E-16 | 2.132E-16 | 2.103E-16 | 2.062E-16 | 1.942E-16 | 1.717E-16 | 1.055E-16 |
| Na-24 | 5.941E-09 | 2.115E-23 | 4.930E-33 | -- | -- | -- | -- |
| Mg-28 | 5.598E-17 | 2.422E-27 | 2.984E-34 | -- | -- | -- | -- |
| Al-28 | 5.609E-17 | 2.425E-27 | 2.989E-34 | -- | -- | -- | -- |
| Si-31 | 3.651E-30 | -- | -- | -- | -- | -- | -- |
| Si-32 | 2.697E-13 | 2.702E-13 | 2.706E-13 | 2.710E-13 | 2.725E-13 | 2.756E-13 | 2.880E-13 |
| P-32 | 3.685E-03 | 8.624E-04 | 3.276E-04 | 7.665E-05 | 9.826E-07 | 1.255E-10 | 2.880E-13 |
| P-33 | 3.858E-09 | 1.683E-09 | 9.675E-10 | 4.219E-10 | 3.499E-11 | 2.082E-13 | 3.487E-22 |
| S-35 | 3.000E-06 | 2.373E-06 | 2.030E-06 | 1.606E-06 | 7.948E-07 | 1.869E-07 | 6.208E-10 |
| Cl-36 | 1.126E-14 | 1.128E-14 | 1.130E-14 | 1.132E-14 | 1.138E-14 | 1.152E-14 | 1.206E-14 |
| Ar-37 | 1.052E-18 | 5.822E-19 | 3.924E-19 | 2.171E-19 | 3.676E-20 | 9.240E-22 | 5.093E-28 |
| Ar-39 | 2.774E-20 | 2.779E-20 | 2.782E-20 | 2.787E-20 | 2.802E-20 | 2.831E-20 | 2.949E-20 |
| Ar-42 | 2.690E-23 | 2.695E-23 | 2.699E-23 | 2.704E-23 | 2.704E-23 | 2.708E-23 | 2.715E-23 |
| K-42 | 1.324E-21 | 2.695E-23 | 2.699E-23 | 2.704E-23 | 2.704E-23 | 2.708E-23 | 2.718E-23 |
| Ca-45 | 1.316E-11 | 1.161E-11 | 1.067E-11 | 9.412E-12 | 6.454E-12 | 2.971E-12 | 1.392E-13 |
| Sc-46 | 4.142E-15 | 3.237E-15 | 2.748E-15 | 2.147E-15 | 1.026E-15 | 2.243E-16 | 5.582E-19 |
| Ca | 4.700E+01 | 8.325E-11 | 8.515E-13 | 4.013E-14 | 4.104E-16 | 4.393E-22 | 2.256E-34 |
| Sc-47 | 2.622E-07 | 5.323E-10 | 8.611E-12 | 1.863E-14 | 1.822E-21 | 8.636E-34 | -- |
| Sc-48 | 2.081E-08 | 2.351E-13 | 1.184E-16 | 1.338E-21 | 1.929E-36 | -- | -- |
| V-50 | 3.063E-18 | 3.069E-18 | 3.073E-18 | 3.079E-18 | 3.096E-18 | 3.133E-18 | 3.281E-18 |
| Cr-51 | 2.521E-01 | 1.192E-01 | 7.240E-02 | 3.424E-02 | 3.624E-03 | 3.563E-05 | 4.320E-13 |
| Mn-54 | 8.347E-01 | 7.825E-01 | 7.496E-01 | 7.026E-01 | 5.787E-01 | 3.882E-01 | 8.044E-02 |
| Fe-55 | 5.480E-02 | 5.371E-02 | 5.301E-02 | 5.195E-02 | 4.892E-02 | 4.324E-02 | 2.657E-02 |
| Mn-56 | 3.118E-28 | -- | -- | -- | -- | -- | -- |
| Co-58 | 3.883E+00 | 2.900E+00 | 2.387E+00 | 1.782E+00 | 7.426E-01 | 1.225E-01 | 1.002E-04 |
| Fe-59 | 9.014E-03 | 5.688E-03 | 4.186E-03 | 2.642E-03 | 6.642E-04 | 3.874E-05 | 5.266E-10 |
| Ni-59 | 9.100E-06 | 9.117E-06 | 9.129E-06 | 9.146E-06 | 9.198E-06 | 9.307E-06 | 9.747E-06 |
| Co-60 | 2.835E-02 | 2.810E-02 | 2.794E-02 | 2.769E-02 | 2.696E-02 | 2.552E-02 | 2.055E-02 |
| Ni-63 | 1.312E-03 | 1.314E-03 | 1.315E-03 | 1.317E-03 | 1.322E-03 | 1.333E-03 | 1.375E-03 |
| Cu-64 | 1.070E-08 | 9.200E-26 | 3.863E-37 | -- | -- | -- | -- |
| Ni-65 | 1.243E-31 | -- | -- | -- | -- | -- | -- |
| Zn-65 | 2.530E-07 | 2.328E-07 | 2.202E-07 | 2.026E-07 | 1.577E-07 | 9.427E-08 | 1.238E-08 |
| Ni-66 | 2.281E-10 | 2.455E-14 | 5.564E-17 | 5.988E-21 | 7.470E-33 | -- | -- |
| Cu-66 | 2.284E-10 | 2.459E-14 | 5.572E-17 | 5.998E-21 | 7.482E-33 | -- | -- |
| Cu-67 | 1.181E-11 | 3.710E-15 | 1.714E-17 | 5.387E-21 | 1.671E-31 | -- | -- |
| Zn-69 | 4.582E-20 | 7.800E-36 | -- | -- | -- | -- | -- |
| Zn-69m | 4.081E-20 | 7.265E-36 | -- | -- | -- | -- | -- |
| Ge-71 | 4.695E-21 | 8.476E-22 | 2.080E-22 | -- | -- | -- | -- |
| Zn-72 | 9.977E-05 | 2.181E-09 | 1.706E-12 | 3.729E-17 | 3.897E-31 | -- | -- |
| Ga-72 | 1.432E-04 | 3.131E-09 | 2.448E-12 | 5.351E-17 | 5.592E-31 | -- | -- |
| Ga-73 | 8.119E-18 | -- | -- | -- | -- | -- | -- |
| Ge-73m | 8.119E-18 | -- | -- | -- | -- | -- | -- |
| As-76 | 3.592E-04 | 2.094E-12 | 6.786E-18 | 3.957E-26 | -- | -- | -- |
| Ge-77 | 1.018E-08 | 6.729E-28 | 9.508E-41 | -- | -- | -- | -- |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| As-77 | 1.029E-03 | 2.679E-09 | 5.070E-13 | 1.319E-18 | 2.327E-35 | -- | -- |
| Se-77m | 2.552E-06 | 6.644E-12 | 1.257E-15 | 3.273E-21 | 1.113E-37 | -- | -- |
| Se-79 | 3.336E-06 | 3.342E-06 | 3.347E-06 | 3.353E-06 | 3.372E-06 | 3.412E-06 | 3.573E-06 |
| Kr-79 | 9.681E-13 | 5.975E-19 | 4.332E-23 | 2.674E-29 | -- | -- | -- |
| Br-80 | 4.320E-22 | -- | -- | -- | -- | -- | -- |
| Br-80m | 4.036E-22 | -- | -- | -- | -- | -- | -- |
| Kr-81 | 3.047E-12 | 3.053E-12 | 3.057E-12 | 3.063E-12 | 3.080E-12 | 3.116E-12 | 3.264E-12 |
| Br-82 | 3.434E-04 | 2.497E-10 | 2.020E-14 | 1.470E-20 | 5.657E-39 | -- | -- |
| Br-83 | 1.219E-30 | -- | -- | -- | -- | -- | -- |
| Kr-83m | 5.200E-30 | -- | -- | -- | -- | -- | -- |
| Kr-85 | 6.069E-02 | 6.048E-02 | 6.036E-02 | 6.014E-02 | 5.953E-02 | 5.829E-02 | 5.364E-02 |
| Kr-85m | 2.449E-16 | -- | -- | -- | -- | -- | -- |
| Rb-86 | 5.624E-02 | 1.849E-02 | 8.804E-03 | 2.894E-03 | 1.028E-04 | 1.075E-07 | 1.842E-19 |
| Rb-87 | 1.191E-10 | 1.193E-10 | 1.195E-10 | 1.197E-10 | 1.204E-10 | 1.218E-10 | 1.275E-10 |
| Sr-87m | 2.194E-30 | -- | -- | -- | -- | -- | -- |
| Kr-88 | 2.601E-25 | -- | -- | -- | -- | -- | -- |
| Rb-88 | 2.905E-25 | -- | -- | -- | -- | -- | -- |
| Sr-89 | 8.709E+00 | 5.780E+00 | 4.398E+00 | 2.919E+00 | 8.535E-01 | 6.791E-02 | 3.144E-06 |
| Y-89m | 5.595E-06 | 9.687E-09 | 1.395E-10 | 2.413E-13 | 1.246E-21 | -- | -- |
| Zr-89 | 5.603E-06 | 9.702E-09 | 1.398E-10 | 2.412E-13 | 1.247E-21 | 1.095E-38 | -- |
| Sr-90 | 3.653E-01 | 3.652E-01 | 3.652E-01 | 3.652E-01 | 3.652E-01 | 3.650E-01 | 3.645E-01 |
| Y-90 | 3.679E-01 | 3.655E-01 | 3.654E-01 | 3.654E-01 | 3.653E-01 | 3.651E-01 | 3.646E-01 |
| Y-90m | 3.003E-27 | -- | -- | -- | -- | -- | -- |
| Sr-91 | 3.373E-07 | -- | -- | -- | -- | -- | -- |
| Y-91 | 1.215E+01 | 8.529E+00 | 6.739E+00 | 4.731E+00 | 1.639E+00 | 1.846E-01 | 3.372E-05 |
| Y-91m | 2.143E-07 | -- | -- | -- | -- | -- | -- |
| Sr-92 | 3.517E-26 | -- | -- | -- | -- | -- | -- |
| Y-92 | 2.647E-19 | -- | -- | -- | -- | -- | -- |
| Nb-92 | 4.821E-03 | 6.238E-04 | 1.596E-04 | 2.065E-05 | 4.476E-08 | 1.470E-13 | 3.493E-35 |
| Y-93 | 1.452E-06 | 5.046E-28 | -- | -- | -- | -- | -- |
| Zr-93 | 1.241E-05 | 1.243E-05 | 1.245E-05 | 1.247E-05 | 1.254E-05 | 1.269E-05 | 1.329E-05 |
| Nb-93m | 5.148E-07 | 5.628E-07 | 5.949E-07 | 6.430E-07 | 7.873E-07 | 1.084E-06 | 2.248E-06 |
| Mo-93 | 1.430E-06 | 1.432E-06 | 1.434E-06 | 1.437E-06 | 1.445E-06 | 1.462E-06 | 1.531E-06 |
| Mo-93m | 8.596E-16 | -- | -- | -- | -- | -- | -- |
| Nb-94 | 6.682E-06 | 6.694E-06 | 6.703E-06 | 6.716E-06 | 6.754E-06 | 6.834E-06 | 7.155E-06 |
| Zr-95 | 2.251E+01 | 1.630E+01 | 1.315E+01 | 9.511E+00 | 3.608E+00 | 4.907E-01 | 1.879E-04 |
| Nb-95 | 2.464E+01 | 2.213E+01 | 1.969E+01 | 1.585E+01 | 7.112E+00 | 1.065E+00 | 4.171E-04 |
| Nb-95m | 1.667E-01 | 1.209E-01 | 9.748E-02 | 7.056E-02 | 2.677E-02 | 3.640E-03 | 1.394E-06 |
| Nb-96 | 1.288E-04 | 6.735E-14 | 4.373E-20 | 2.286E-29 | -- | -- | -- |
| Zr-97 | 1.446E-03 | 2.168E-16 | 6.119E-25 | 9.172E-38 | -- | -- | -- |
| Nb-97 | 1.453E-03 | 2.336E-16 | 6.593E-25 | 9.898E-38 | -- | -- | -- |
| Nb-97m | 1.370E-03 | 2.054E-16 | 5.796E-25 | 9.688E-38 | -- | -- | -- |
| Tc-98 | 2.010E-10 | 2.013E-10 | 2.016E-10 | 2.020E-10 | 2.031E-10 | 2.055E-10 | 2.152E-10 |
| Mo-99 | 2.595E+00 | 1.353E-03 | 8.754E-06 | 4.561E-09 | 6.451E-19 | 3.435E-39 | -- |
| Tc-99 | 1.315E-04 | 1.318E-04 | 1.320E-04 | 1.323E-04 | 1.330E-04 | 1.346E-04 | 1.409E-04 |
| Tc-99m | 2.411E+00 | 1.255E-03 | 8.133E-06 | 4.236E-09 | 5.992E-19 | 3.184E-39 | -- |
| Rh-102 | 8.893E-05 | 8.736E-05 | 8.634E-05 | 8.482E-05 | 8.042E-05 | 7.208E-05 | 4.680E-05 |

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| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Ru-103 | 2.966E+01 | 1.750E+01 | 1.231E+01 | 7.266E+00 | 1.493E+00 | 5.748E-02 | 1.520E-07 |
| Rh-103m | 2.673E+01 | 1.578E+01 | 1.110E+01 | 6.549E+00 | 1.345E+00 | 5.182E-02 | 1.369E-07 |
| Ru-105 | 1.449E-15 | -- | -- | -- | -- | -- | -- |
| Rh-105 | 2.744E-01 | 2.041E-07 | 1.675E-11 | 1.246E-17 | 5.128E-36 | -- | -- |
| Rh-105m | 4.070E-16 | -- | -- | -- | -- | -- | -- |
| Ru-106 | 1.347E+01 | 1.276E+01 | 1.230E+01 | 1.165E+01 | 9.888E+00 | 7.059E+00 | 1.868E+00 |
| Rh-106 | 1.347E+01 | 1.276E+01 | 1.230E+01 | 1.165E+01 | 9.888E+00 | 7.059E+00 | 1.868E+00 |
| Rh-106m | 1.934E-34 | -- | -- | -- | -- | -- | -- |
| Ag-106 | 2.243E-11 | 1.945E-12 | 3.814E-13 | 3.309E-14 | 2.161E-17 | 6.015E-24 | -- |
| Pd-107 | 2.162E-06 | 2.166E-06 | 2.169E-06 | 2.173E-06 | 2.186E-06 | 2.211E-06 | 2.316E-06 |
| Ag-108 | 9.827E-11 | 9.840E-11 | 9.851E-11 | 9.864E-11 | 9.906E-11 | 9.996E-11 | 1.036E-10 |
| Ag-108m | 1.104E-09 | 1.106E-09 | 1.107E-09 | 1.108E-09 | 1.113E-09 | 1.123E-09 | 1.163E-09 |
| Pd-109 | 3.682E-05 | 2.921E-21 | 5.395E-32 | -- | -- | -- | -- |
| Ag-109m | 3.803E-05 | 1.109E-06 | 1.077E-06 | 1.032E-06 | 9.074E-07 | 6.961E-07 | 2.448E-07 |
| Cd-109 | 1.157E-06 | 1.109E-06 | 1.077E-06 | 1.032E-06 | 9.074E-07 | 6.961E-07 | 2.448E-07 |
| Ag-110 | 7.650E-04 | 7.052E-04 | 6.681E-04 | 6.158E-04 | 4.826E-04 | 2.921E-04 | 4.031E-05 |
| Ag-110m | 5.752E-02 | 5.302E-02 | 5.023E-02 | 4.631E-02 | 3.628E-02 | 2.196E-02 | 3.031E-03 |
| Pd-111 | 1.742E-15 | -- | -- | -- | -- | -- | -- |
| Pd-111m | 2.392E-15 | -- | -- | -- | -- | -- | -- |
| Ag-111 | 8.220E-01 | 5.052E-02 | 7.870E-03 | 4.838E-04 | 1.123E-07 | 3.720E-15 | -- |
| Ag-111m | 2.501E-15 | -- | -- | -- | -- | -- | -- |
| Pd-112 | 2.991E-04 | 4.937E-15 | 3.201E-22 | 5.284E-33 | -- | -- | -- |
| Ag-112 | 3.543E-04 | 5.848E-15 | 3.792E-22 | 6.259E-33 | -- | -- | -- |
| Ag-113 | 1.784E-14 | -- | -- | -- | -- | -- | -- |
| Cd-113m | 9.458E-04 | 9.438E-04 | 9.427E-04 | 9.407E-04 | 9.350E-04 | 9.236E-04 | 8.795E-04 |
| In-114 | 1.200E-05 | 7.900E-06 | 5.979E-06 | 3.936E-06 | 1.123E-06 | 8.495E-08 | 3.221E-12 |
| In-114m | 1.254E-05 | 8.255E-06 | 6.247E-06 | 4.112E-06 | 1.173E-06 | 8.877E-08 | 3.366E-12 |
| Cd-115 | 2.534E-02 | 2.244E-06 | 4.461E-09 | 3.952E-13 | 2.750E-25 | -- | -- |
| Cd-115m | 4.651E-02 | 2.923E-02 | 2.145E-02 | 1.348E-02 | 3.347E-03 | 1.902E-04 | 2.334E-09 |
| In-115 | 7.823E-16 | 7.884E-16 | 7.913E-16 | 7.947E-16 | 8.018E-16 | 8.120E-16 | 8.504E-16 |
| In-115m | 2.756E-02 | 4.496E-06 | 1.513E-06 | 9.474E-07 | 2.352E-07 | 1.336E-08 | 1.641E-13 |
| Cd-117 | 4.970E-29 | -- | -- | -- | -- | -- | -- |
| Cd-117m | 9.249E-23 | -- | -- | -- | -- | -- | -- |
| In-117 | 1.229E-22 | -- | -- | -- | -- | -- | -- |
| In-117m | 9.477E-23 | -- | -- | -- | -- | -- | -- |
| Sn-117m | 2.204E-04 | 5.002E-05 | 1.861E-05 | 4.224E-06 | 4.939E-08 | 5.209E-12 | 1.083E-27 |
| Sn-119m | 5.045E-03 | 4.643E-03 | 4.393E-03 | 4.044E-03 | 3.153E-03 | 1.889E-03 | 2.505E-04 |
| Sn-121 | 9.664E-04 | 7.917E-12 | 3.218E-17 | 2.636E-25 | -- | -- | -- |
| Sn-121m | 1.167E-05 | 1.167E-05 | 1.168E-05 | 1.169E-05 | 1.171E-05 | 1.177E-05 | 1.199E-05 |
| Sb-122 | 2.494E-03 | 1.130E-06 | 6.669E-09 | 3.022E-12 | 2.814E-22 | -- | -- |
| Sn-123 | 1.066E-01 | 9.090E-02 | 8.176E-02 | 6.973E-02 | 4.327E-02 | 1.621E-02 | 3.366E-04 |
| Te-123 | 2.288E-17 | 2.320E-17 | 2.340E-17 | 2.366E-17 | 2.425E-17 | 2.498E-17 | 2.640E-17 |
| Te-123m | 5.390E-05 | 4.539E-05 | 4.047E-05 | 3.408E-05 | 2.035E-05 | 7.043E-06 | 1.072E-07 |
| Sb-124 | 1.513E-02 | 1.073E-02 | 8.533E-03 | 6.052E-03 | 2.158E-03 | 2.588E-04 | 6.025E-08 |
| Sn-125 | 1.761E-01 | 2.041E-02 | 4.851E-03 | 5.620E-04 | 8.788E-07 | 1.460E-12 | -- |
| Sb-125 | 3.114E-01 | 3.071E-01 | 3.035E-01 | 2.979E-01 | 2.816E-01 | 2.511E-01 | 1.594E-01 |
| Te-125m | 6.197E-02 | 6.483E-02 | 6.606E-02 | 6.708E-02 | 6.680E-02 | 6.103E-02 | 3.889E-02 |

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| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Sn-126 | 1.405E-05 | 1.408E-05 | 1.410E-05 | 1.412E-05 | 1.420E-05 | 1.437E-05 | 1.505E-05 |
| Sb-126 | 3.545E-02 | 6.636E-03 | 2.173E-03 | 4.083E-04 | 4.654E-06 | 2.013E-06 | 2.108E-06 |
| Sb-126m | 1.405E-05 | 1.408E-05 | 1.410E-05 | 1.412E-05 | 1.420E-05 | 1.437E-05 | 1.505E-05 |
| Sn-127 | 6.959E-35 | -- | -- | -- | -- | -- | -- |
| Sb-127 | 4.944E-01 | 2.233E-03 | 6.104E-05 | 2.758E-07 | 2.542E-14 | 8.391E-29 | -- |
| Te-127 | 8.401E-01 | 3.075E-01 | 2.693E-01 | 2.228E-01 | 1.264E-01 | 3.939E-02 | 3.965E-04 |
| Te-127m | 3.740E-01 | 3.116E-01 | 2.749E-01 | 2.275E-01 | 1.291E-01 | 4.023E-02 | 4.047E-04 |
| Xe-127 | 3.120E-06 | 1.766E-06 | 1.208E-06 | 6.839E-07 | 1.239E-07 | 3.690E-09 | 3.530E-15 |
| Sb-128 | 4.236E-09 | -- | -- | -- | -- | -- | -- |
| Sb-129 | 1.383E-16 | -- | -- | -- | -- | -- | -- |
| Te-129 | 5.794E-01 | 3.126E-01 | 2.071E-01 | 1.118E-01 | 1.756E-02 | 3.889E-04 | 1.162E-10 |
| Te-129m | 8.901E-01 | 4.802E-01 | 3.183E-01 | 1.718E-01 | 2.697E-02 | 5.975E-04 | 1.785E-10 |
| I-129 | 4.328E-07 | 4.361E-07 | 4.375E-07 | 4.392E-07 | 4.425E-07 | 4.480E-07 | 4.692E-07 |
| Xe-129m | 7.201E-06 | 5.363E-07 | 9.493E-08 | 7.068E-09 | 2.919E-12 | 3.159E-19 | -- |
| I-130 | 6.301E-07 | 1.845E-24 | 3.777E-36 | -- | -- | -- | -- |
| Te-131 | 2.797E-03 | 1.671E-10 | 2.553E-15 | 1.524E-22 | -- | -- | -- |
| Te-131m | 1.243E-02 | 7.421E-10 | 1.134E-14 | 6.771E-22 | -- | -- | -- |
| I-131 | 8.907E+00 | 6.721E-01 | 1.200E-01 | 9.054E-03 | 3.888E-06 | 4.564E-13 | -- |
| Xe-131m | 1.937E-01 | 5.422E-02 | 1.901E-02 | 3.593E-03 | 2.011E-05 | 4.202E-10 | 1.450E-28 |
| Te-132 | 3.211E+00 | 5.439E-03 | 7.729E-05 | 1.312E-07 | 6.383E-16 | 4.944E-33 | -- |
| I-132 | 3.307E+00 | 5.604E-03 | 7.964E-05 | 1.352E-07 | 6.576E-16 | 5.093E-33 | -- |
| Cs-132 | 9.219E-04 | 3.721E-05 | 4.379E-06 | 1.768E-07 | 1.162E-11 | 2.866E-20 | -- |
| I-133 | 1.198E-02 | 4.560E-13 | 5.160E-20 | 1.964E-30 | -- | -- | -- |
| Xe-133 | 1.124E+01 | 2.149E-01 | 1.531E-02 | 2.912E-04 | 2.002E-09 | 4.734E-20 | -- |
| Xe-133m | 7.930E-02 | 5.989E-06 | 1.071E-08 | 8.064E-13 | 3.442E-25 | -- | -- |
| Cs-134 | 4.787E-01 | 4.665E-01 | 4.587E-01 | 4.469E-01 | 4.137E-01 | 3.530E-01 | 1.888E-01 |
| Cs-134m | 3.592E-26 | -- | -- | -- | -- | -- | -- |
| I-135 | 3.964E-10 | -- | -- | -- | -- | -- | -- |
| Xe-135 | 1.440E-06 | -- | -- | -- | -- | -- | -- |
| Xe-135m | 6.348E-11 | -- | -- | -- | -- | -- | -- |
| Cs-135 | 1.533E-05 | 1.536E-05 | 1.538E-05 | 1.541E-05 | 1.550E-05 | 1.568E-05 | 1.642E-05 |
| Ba-135m | 2.355E-07 | 6.597E-15 | 6.086E-20 | 1.705E-27 | -- | -- | -- |
| Cs-136 | 6.074E-01 | 1.244E-01 | 4.326E-02 | 8.863E-03 | 7.624E-05 | 4.276E-09 | 7.362E-26 |
| Ba-136m | 1.001E-01 | 2.052E-02 | 7.129E-03 | 1.460E-03 | 1.257E-05 | 7.046E-10 | 1.213E-26 |
| Cs-137 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 |
| Ba-137m | 9.460E-01 | 9.460E-01 | 9.461E-01 | 9.460E-01 | 9.459E-01 | 9.460E-01 | 9.460E-01 |
| La-138 | 3.704E-16 | 3.711E-16 | 3.716E-16 | 3.723E-16 | 3.744E-16 | 3.788E-16 | 3.967E-16 |
| Ba-140 | 1.620E+01 | 3.192E+00 | 1.081E+00 | 2.132E-01 | 1.631E-03 | 7.201E-08 | 4.822E-25 |
| La-140 | 1.859E+01 | 3.673E+00 | 1.245E+00 | 2.453E-01 | 1.878E-03 | 8.287E-08 | 5.550E-25 |
| La-141 | 1.292E-17 | -- | -- | -- | -- | -- | -- |
| Ce-141 | 2.339E+01 | 1.236E+01 | 8.083E+00 | 4.272E+00 | 6.305E-01 | 1.229E-02 | 2.217E-09 |
| Ce-142 | 2.248E-10 | 2.252E-10 | 2.255E-10 | 2.259E-10 | 2.272E-10 | 2.299E-10 | 2.407E-10 |
| Pr-142 | 9.564E-05 | 4.485E-16 | 1.257E-23 | 5.895E-35 | -- | -- | -- |
| Ce-143 | 1.503E-01 | 4.073E-08 | 1.706E-12 | 4.621E-19 | 9.184E-39 | -- | -- |
| Pr-143 | 1.499E+01 | 3.247E+00 | 1.170E+00 | 2.531E-01 | 2.561E-03 | 2.007E-07 | 1.292E-23 |
| Ce-144 | 1.321E+01 | 1.231E+01 | 1.174E+01 | 1.093E+01 | 8.825E+00 | 5.683E+00 | 1.002E+00 |
| Pr-144 | 1.321E+01 | 1.231E+01 | 1.174E+01 | 1.093E+01 | 8.825E+00 | 5.684E+00 | 1.002E+00 |

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**REACTOR SOURCE TERMS**


| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Pr-144m | 1.586E-01 | 1.476E-01 | 1.409E-01 | 1.311E-01 | 1.059E-01 | 6.820E-02 | 1.203E-02 |
| Nd-144 | 3.881E-15 | 4.234E-15 | 4.457E-15 | 4.773E-15 | 5.603E-15 | 6.872E-15 | 9.034E-15 |
| Pr-145 | 1.364E-11 | -- | -- | -- | -- | -- | -- |
| Pm-146 | 1.828E-04 | 1.813E-04 | 1.804E-04 | 1.787E-04 | 1.743E-04 | 1.654E-04 | 1.347E-04 |
| Sm-146 | 4.175E-12 | 4.237E-12 | 4.280E-12 | 4.342E-12 | 4.525E-12 | 4.897E-12 | 6.251E-12 |
| Nd-147 | 6.034E+00 | 9.222E-01 | 2.636E-01 | 4.030E-02 | 1.440E-04 | 1.323E-09 | 1.816E-29 |
| Pm-147 | 2.741E+00 | 2.746E+00 | 2.717E+00 | 2.667E+00 | 2.513E+00 | 2.224E+00 | 1.373E+00 |
| Sm-147 | 1.268E-11 | 1.417E-11 | 1.515E-11 | 1.661E-11 | 2.085E-11 | 2.892E-11 | 5.372E-11 |
| Pm-148 | 4.441E-01 | 5.399E-02 | 3.374E-02 | 2.009E-02 | 4.457E-03 | 2.013E-04 | 9.975E-10 |
| Pm-148m | 1.358E+00 | 8.224E-01 | 5.887E-01 | 3.565E-01 | 7.914E-02 | 3.574E-03 | 1.772E-08 |
| Sm-148 | 1.725E-16 | 1.811E-16 | 1.848E-16 | 1.884E-16 | 1.934E-16 | 1.968E-16 | 2.061E-16 |
| Pm-149 | 3.336E-01 | 2.761E-05 | 5.245E-08 | 4.340E-12 | 2.459E-24 | -- | -- |
| Sm-149 | 6.141E-16 | 6.153E-16 | 6.162E-16 | 6.173E-16 | 6.208E-16 | 6.281E-16 | 6.578E-16 |
| Pm-150 | 2.864E-29 | -- | -- | -- | -- | -- | -- |
| Eu-150 | 1.699E-08 | 1.700E-08 | 1.701E-08 | 1.700E-08 | 1.703E-08 | 1.705E-08 | 1.719E-08 |
| Pm-151 | 1.284E-02 | 2.981E-10 | 2.427E-15 | 5.639E-23 | -- | -- | -- |
| Sm-151 | 3.922E-02 | 3.927E-02 | 3.930E-02 | 3.935E-02 | 3.950E-02 | 3.982E-02 | 4.106E-02 |
| Eu-152 | 1.715E-04 | 1.712E-04 | 1.709E-04 | 1.705E-04 | 1.693E-04 | 1.669E-04 | 1.580E-04 |
| Eu-152m | 8.091E-11 | -- | -- | -- | -- | -- | -- |
| Gd-152 | 8.419E-18 | 8.459E-18 | 8.487E-18 | 8.529E-18 | 8.652E-18 | 8.908E-18 | 9.923E-18 |
| Sm-153 | 6.873E-02 | 1.572E-06 | 1.266E-09 | 2.894E-14 | 3.458E-28 | -- | -- |
| Gd-153 | 1.911E-04 | 1.757E-04 | 1.662E-04 | 1.528E-04 | 1.187E-04 | 7.067E-05 | 9.134E-06 |
| Eu-154 | 2.884E-02 | 2.870E-02 | 2.861E-02 | 2.847E-02 | 2.808E-02 | 2.726E-02 | 2.431E-02 |
| Eu-155 | 1.663E-01 | 1.646E-01 | 1.636E-01 | 1.621E-01 | 1.574E-01 | 1.485E-01 | 1.175E-01 |
| Sm-156 | 1.481E-08 | -- | -- | -- | -- | -- | -- |
| Eu-156 | 6.616E-01 | 1.685E-01 | 6.774E-02 | 1.726E-02 | 2.852E-04 | 6.137E-08 | 2.121E-22 |
| Eu-157 | 9.403E-06 | 5.185E-20 | 1.619E-29 | -- | -- | -- | -- |
| Gd-159 | 2.772E-05 | 6.175E-17 | 1.054E-24 | 2.348E-36 | -- | -- | -- |
| Tb-160 | 2.435E-02 | 1.831E-02 | 1.513E-02 | 1.137E-02 | 4.824E-03 | 8.266E-04 | 7.869E-07 |
| Tb-161 | 1.962E-02 | 9.739E-04 | 1.316E-04 | 6.529E-06 | 7.984E-10 | 7.062E-18 | -- |
| Tb-162m | 4.269E-36 | -- | -- | -- | -- | -- | -- |
| Dy-165 | 9.215E-34 | -- | -- | -- | -- | -- | -- |
| Dy-166 | 3.638E-04 | 7.980E-07 | 1.348E-08 | 2.958E-11 | 3.128E-19 | 1.198E-35 | -- |
| Ho-166 | 5.411E-04 | 1.190E-06 | 2.009E-08 | 4.408E-11 | 4.661E-19 | 1.785E-35 | -- |
| Ho-166m | 1.889E-08 | 1.893E-08 | 1.895E-08 | 1.899E-08 | 1.910E-08 | 1.931E-08 | 2.021E-08 |
| Er-169 | 4.467E-07 | 4.899E-08 | 1.123E-08 | 1.231E-09 | 1.627E-12 | 1.925E-18 | -- |
| Tm-170 | 4.004E-08 | 3.412E-08 | 3.068E-08 | 2.614E-08 | 1.619E-08 | 6.033E-09 | 1.232E-10 |
| Er-171 | 1.466E-23 | -- | -- | -- | -- | -- | -- |
| Tm-171 | 1.535E-10 | 1.493E-10 | 1.466E-10 | 1.426E-10 | 1.312E-10 | 1.105E-10 | 5.621E-11 |
| Er-172 | 5.580E-20 | 2.109E-24 | 2.112E-24 | 7.984E-29 | -- | -- | -- |
| Tm-172 | 6.722E-13 | 2.633E-16 | 1.410E-18 | 5.521E-22 | 3.318E-32 | -- | -- |
| Ta-180 | 6.748E-18 | 6.760E-18 | 6.769E-18 | 6.782E-18 | 6.820E-18 | 6.901E-18 | 7.227E-18 |
| W-181 | 2.115E-19 | 1.784E-19 | 1.593E-19 | 1.345E-19 | 8.081E-20 | 2.833E-20 | 4.449E-22 |
| Ta-182 | 1.364E-01 | 1.140E-01 | 1.012E-01 | 8.463E-02 | 4.948E-02 | 1.639E-02 | 2.101E-04 |
| Ta-183 | 3.052E-03 | 5.183E-05 | 3.425E-06 | 5.816E-08 | 2.850E-13 | 3.353E-24 | -- |
| W-185 | 6.092E-07 | 4.627E-07 | 3.852E-07 | 2.925E-07 | 1.282E-07 | 2.347E-08 | 2.905E-11 |
| Re-186 | 8.825E-09 | 3.591E-11 | 9.155E-13 | 3.725E-15 | 2.511E-22 | 4.357E-37 | -- |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| W-187 | 6.725E-14 | 5.752E-23 | 5.183E-29 | 4.433E-38 | -- | -- | -- |
| Re-187 | 1.938E-22 | 1.942E-22 | 1.944E-22 | 1.948E-22 | 1.959E-22 | 1.982E-22 | 2.076E-22 |
| W-188 | 5.391E-14 | 4.003E-14 | 3.282E-14 | 2.437E-14 | 9.975E-15 | 1.587E-15 | 1.127E-18 |
| Re-188 | 5.451E-14 | 4.044E-14 | 3.316E-14 | 2.462E-14 | 1.008E-14 | 1.603E-15 | 1.138E-18 |
| Os-191 | 1.866E-20 | 4.885E-21 | 2.009E-21 | 5.831E-22 | 6.270E-23 | 1.522E-26 | 8.531E-41 |
| Os-191m | 1.209E-21 | 2.575E-38 | -- | -- | -- | -- | -- |
| Ir-192 | 4.732E-22 | 3.408E-22 | 2.775E-22 | 1.998E-22 | 1.030E-22 | 2.888E-23 | 9.691E-24 |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


Table A-5. N Reactor activity relative to Cs-137.

| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| H-3 | 5.497E-03 | 5.483E-03 | 5.472E-03 | 5.457E-03 | 5.412E-03 | 5.322E-03 | 4.982E-03 |
| Be-10 | 1.694E-10 | 1.697E-10 | 1.699E-10 | 1.703E-10 | 1.712E-10 | 1.732E-10 | 1.814E-10 |
| C-14 | 3.264E-05 | 3.270E-05 | 3.274E-05 | 3.281E-05 | 3.299E-05 | 3.338E-05 | 3.496E-05 |
| Na-24 | 1.383E-07 | 4.921E-22 | 1.147E-31 | -- | -- | -- | -- |
| Mg-28 | 4.106E-15 | 1.776E-25 | 2.188E-32 | -- | -- | -- | -- |
| Al-28 | 4.114E-15 | 1.779E-25 | 2.192E-32 | -- | -- | -- | -- |
| Si-31 | 3.943E-31 | -- | -- | -- | -- | -- | -- |
| Si-32 | 5.829E-14 | 5.840E-14 | 5.848E-14 | 5.859E-14 | 5.890E-14 | 5.956E-14 | 6.226E-14 |
| P-32 | 2.219E-08 | 5.194E-09 | 1.973E-09 | 4.617E-10 | 5.979E-12 | 6.033E-14 | 6.226E-14 |
| P-33 | 4.291E-20 | 1.869E-20 | 1.069E-20 | 4.692E-21 | 4.281E-22 | 9.018E-23 | 1.510E-31 |
| Ar-39 | 1.521E-25 | 1.523E-25 | 1.525E-25 | 1.528E-25 | 1.537E-25 | 1.555E-25 | 1.629E-25 |
| Ar-42 | 2.642E-23 | 2.647E-23 | 2.651E-23 | 2.656E-23 | 2.671E-23 | 2.702E-23 | 2.711E-23 |
| K-42 | 2.279E-21 | 2.647E-23 | 2.651E-23 | 2.656E-23 | 2.671E-23 | 2.702E-23 | 2.714E-23 |
| K-43 | 7.638E-16 | 1.965E-25 | 7.952E-32 | -- | -- | -- | -- |
| Ca-45 | 4.323E-08 | 3.814E-08 | 3.508E-08 | 3.092E-08 | 2.121E-08 | 9.759E-09 | 4.570E-10 |
| Sc-46 | 8.686E-06 | 6.792E-06 | 5.762E-06 | 4.504E-06 | 2.152E-06 | 4.703E-07 | 1.170E-09 |
| Ca-47 | 8.414E-11 | 8.606E-13 | 4.055E-14 | 4.149E-16 | 4.440E-22 | 2.280E-34 | -- |
| Sc-47 | 3.090E-06 | 6.256E-09 | 1.001E-10 | 2.034E-13 | 3.367E-21 | 8.728E-34 | -- |
| Sc-48 | 5.281E-08 | 5.966E-13 | 3.003E-16 | 3.393E-21 | 4.895E-36 | -- | -- |
| V-50 | 4.077E-19 | 4.085E-19 | 4.090E-19 | 4.098E-19 | 4.121E-19 | 4.170E-19 | 4.367E-19 |
| Cr-51 | 1.090E-01 | 5.157E-02 | 3.129E-02 | 1.481E-02 | 1.567E-03 | 1.541E-05 | 1.868E-13 |
| Mn-54 | 1.298E-03 | 1.217E-03 | 1.166E-03 | 1.092E-03 | 8.999E-04 | 6.038E-04 | 1.251E-04 |
| Fe-55 | 6.592E-03 | 6.462E-03 | 6.377E-03 | 6.249E-03 | 5.884E-03 | 5.201E-03 | 3.195E-03 |
| Mn-56 | 6.859E-29 | -- | -- | -- | -- | -- | -- |
| Co-58 | 1.527E-02 | 1.140E-02 | 9.388E-03 | 7.013E-03 | 2.920E-03 | 4.815E-04 | 3.943E-07 |
| Fe-59 | 1.766E-03 | 1.114E-03 | 8.202E-04 | 5.176E-04 | 1.301E-04 | 7.590E-06 | 1.032E-10 |
| Ni-59 | 1.394E-06 | 1.396E-06 | 1.398E-06 | 1.401E-06 | 1.409E-06 | 1.425E-06 | 1.493E-06 |
| Co-60 | 3.109E-03 | 3.082E-03 | 3.062E-03 | 3.036E-03 | 2.956E-03 | 2.799E-03 | 2.252E-03 |
| Ni-63 | 2.195E-04 | 2.198E-04 | 2.200E-04 | 2.202E-04 | 2.211E-04 | 2.228E-04 | 2.299E-04 |
| Cu-64 | 7.442E-07 | 6.400E-24 | 2.687E-35 | -- | -- | -- | -- |
| Ni-65 | 6.132E-32 | -- | -- | -- | -- | -- | -- |
| Zn-65 | 3.291E-06 | 3.026E-06 | 2.864E-06 | 2.634E-06 | 2.051E-06 | 1.226E-06 | 1.609E-07 |
| Ni-66 | 7.999E-10 | 8.612E-14 | 1.952E-16 | 2.101E-20 | 2.621E-32 | -- | -- |
| Cu-66 | 8.012E-10 | 8.625E-14 | 1.955E-16 | 2.104E-20 | 2.625E-32 | -- | -- |
| Cu-67 | 5.872E-09 | 1.845E-12 | 8.529E-15 | 2.680E-18 | 8.307E-29 | -- | -- |
| Zn-69 | 8.245E-14 | 1.468E-29 | -- | -- | -- | -- | -- |
| Zn-69m | 7.678E-14 | 1.367E-29 | 4.283E-40 | -- | -- | -- | -- |
| Zn-71 | 5.696E-31 | -- | -- | -- | -- | -- | -- |
| Zn-71m | 1.127E-27 | -- | -- | -- | -- | -- | -- |
| Ge-71 | 1.234E-20 | 2.228E-21 | 5.467E-22 | -- | -- | -- | -- |
| Zn-72 | 2.257E-04 | 4.934E-09 | 3.859E-12 | 8.435E-17 | 8.816E-31 | -- | -- |
| Ga-72 | 3.240E-04 | 7.081E-09 | 5.537E-12 | 1.211E-16 | 1.265E-30 | -- | -- |
| Ga-73 | 2.556E-17 | -- | -- | -- | -- | -- | -- |
| Ge-73m | 2.556E-17 | -- | -- | -- | -- | -- | -- |
| As-76 | 5.072E-07 | 2.959E-15 | 9.587E-21 | 5.590E-29 | -- | -- | -- |
| Ge-77 | 4.037E-08 | 2.669E-27 | 4.173E-40 | -- | -- | -- | -- |
| As-77 | 4.481E-03 | 1.166E-08 | 2.207E-12 | 5.746E-18 | 1.014E-34 | -- | -- |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Se-77m | 1.111E-05 | 2.892E-11 | 5.475E-15 | 1.425E-20 | 2.444E-37 | -- | -- |
| Se-79 | 4.304E-06 | 4.313E-06 | 4.318E-06 | 4.326E-06 | 4.351E-06 | 4.402E-06 | 4.610E-06 |
| Kr-79 | 2.179E-14 | 1.345E-20 | 9.750E-25 | 6.018E-31 | -- | -- | -- |
| Br-80 | 1.579E-21 | -- | -- | -- | -- | -- | -- |
| Br-80m | 1.476E-21 | -- | -- | -- | -- | -- | -- |
| Kr-81 | 1.264E-12 | 1.267E-12 | 1.268E-12 | 1.271E-12 | 1.278E-12 | 1.293E-12 | 1.354E-12 |
| Br-82 | 5.161E-04 | 3.755E-10 | 3.038E-14 | 2.209E-20 | 8.513E-39 | -- | -- |
| Br-83 | 8.066E-30 | -- | -- | -- | -- | -- | -- |
| Kr-83m | 3.440E-29 | -- | -- | -- | -- | -- | -- |
| Kr-85 | 1.082E-01 | 1.078E-01 | 1.076E-01 | 1.072E-01 | 1.061E-01 | 1.039E-01 | 9.562E-02 |
| Kr-85m | 2.086E-15 | -- | -- | -- | -- | -- | -- |
| Rb-86 | 9.414E-03 | 3.093E-03 | 1.474E-03 | 4.843E-04 | 1.720E-05 | 1.792E-08 | 3.069E-20 |
| Rb-87 | 2.316E-10 | 2.321E-10 | 2.324E-10 | 2.328E-10 | 2.341E-10 | 2.369E-10 | 2.481E-10 |
| Sr-87m | 2.180E-30 | -- | -- | -- | -- | -- | -- |
| Kr-88 | 2.702E-24 | -- | -- | -- | -- | -- | -- |
| Rb-88 | 3.018E-24 | -- | -- | -- | -- | -- | -- |
| Sr-89 | 6.776E+01 | 4.498E+01 | 3.422E+01 | 2.271E+01 | 6.642E+00 | 5.283E-01 | 2.445E-05 |
| Y-89m | 4.109E-09 | 7.105E-12 | 1.021E-13 | 1.765E-16 | 9.108E-25 | -- | -- |
| Zr-89 | 4.117E-09 | 7.097E-12 | 1.023E-13 | 1.765E-16 | 9.124E-25 | 1.532E-41 | -- |
| Sr-90 | 8.151E-01 | 8.151E-01 | 8.151E-01 | 8.150E-01 | 8.148E-01 | 8.145E-01 | 8.132E-01 |
| Y-90 | 8.197E-01 | 8.156E-01 | 8.151E-01 | 8.155E-01 | 8.150E-01 | 8.148E-01 | 8.135E-01 |
| Y-90m | 4.577E-27 | -- | -- | -- | -- | -- | -- |
| Sr-91 | 3.114E-06 | -- | -- | -- | -- | -- | -- |
| Y-91 | 8.066E+01 | 5.664E+01 | 4.474E+01 | 3.143E+01 | 1.088E+01 | 1.226E+00 | 2.239E-04 |
| Y-91m | 1.979E-06 | -- | -- | -- | -- | -- | -- |
| Sr-92 | 2.892E-25 | -- | -- | -- | -- | -- | -- |
| Y-92 | 2.174E-18 | -- | -- | -- | -- | -- | -- |
| Nb-92 | 3.491E-06 | 4.519E-07 | 1.156E-07 | 1.496E-08 | 3.245E-11 | 1.065E-16 | 2.532E-38 |
| Y-93 | 1.060E-05 | 3.688E-27 | -- | -- | -- | -- | -- |
| Zr-93 | 2.069E-05 | 2.073E-05 | 2.076E-05 | 2.080E-05 | 2.092E-05 | 2.116E-05 | 2.216E-05 |
| Nb-93m | 1.798E-07 | 2.617E-07 | 3.163E-07 | 3.982E-07 | 6.433E-07 | 1.147E-06 | 3.126E-06 |
| Mo-93 | 4.663E-09 | 4.672E-09 | 4.678E-09 | 4.687E-09 | 4.713E-09 | 4.769E-09 | 4.991E-09 |
| Mo-93m | 7.421E-17 | -- | -- | -- | -- | -- | -- |
| Nb-94 | 7.703E-10 | 7.718E-10 | 7.728E-10 | 7.743E-10 | 7.787E-10 | 7.878E-10 | 8.248E-10 |
| Zr-95 | 9.462E+01 | 6.848E+01 | 5.521E+01 | 3.998E+01 | 1.516E+01 | 2.062E+00 | 7.895E-04 |
| Nb-95 | 7.275E+01 | 7.593E+01 | 7.120E+01 | 6.026E+01 | 2.880E+01 | 4.449E+00 | 1.753E-03 |
| Nb-95m | 6.931E-01 | 5.080E-01 | 4.097E-01 | 2.965E-01 | 1.125E-01 | 1.530E-02 | 5.856E-06 |
| Nb-96 | 7.570E-05 | 3.960E-14 | 2.570E-20 | 1.344E-29 | -- | -- | -- |
| Zr-97 | 8.236E-03 | 1.235E-15 | 3.484E-24 | 5.225E-37 | -- | -- | -- |
| Nb-97 | 8.278E-03 | 1.330E-15 | 3.756E-24 | 5.634E-37 | -- | -- | -- |
| Nb-97m | 7.804E-03 | 1.170E-15 | 3.300E-24 | 4.961E-37 | -- | -- | -- |
| Tc-98 | 2.742E-12 | 2.747E-12 | 2.751E-12 | 2.756E-12 | 2.772E-12 | 2.804E-12 | 2.937E-12 |
| Mo-99 | 1.282E+01 | 6.682E-03 | 4.327E-05 | 2.254E-08 | 3.188E-18 | 1.699E-38 | -- |
| Tc-99 | 1.396E-04 | 1.404E-04 | 1.406E-04 | 1.408E-04 | 1.416E-04 | 1.433E-04 | 1.501E-04 |
| Tc-99m | 1.235E+01 | 6.435E-03 | 4.168E-05 | 2.172E-08 | 3.072E-18 | 1.637E-38 | -- |
| Rh-102 | 3.735E-07 | 3.669E-07 | 3.626E-07 | 3.560E-07 | 3.378E-07 | 3.026E-07 | 1.965E-07 |
| Ru-103 | 7.592E+01 | 4.479E+01 | 3.151E+01 | 1.860E+01 | 3.821E+00 | 1.471E-01 | 3.888E-07 |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Rh-103m | 6.843E+01 | 4.039E+01 | 2.842E+01 | 1.677E+01 | 3.445E+00 | 1.326E-01 | 3.504E-07 |
| Ru-105 | 3.727E-15 | -- | -- | -- | -- | -- | -- |
| Rh-105 | 6.752E-01 | 5.020E-07 | 4.122E-11 | 3.065E-17 | 1.261E-35 | -- | -- |
| Rh-105m | 1.047E-15 | -- | -- | -- | -- | -- | -- |
| Ru-106 | 5.829E+00 | 5.519E+00 | 5.322E+00 | 5.039E+00 | 4.278E+00 | 3.053E+00 | 8.083E-01 |
| Rh-106 | 5.829E+00 | 5.519E+00 | 5.322E+00 | 5.039E+00 | 4.278E+00 | 3.053E+00 | 8.083E-01 |
| Rh-106m | 1.712E-33 | -- | -- | -- | -- | -- | -- |
| Ag-106 | 7.694E-13 | 6.676E-14 | 1.308E-14 | 1.135E-15 | 7.418E-19 | 2.063E-25 | -- |
| Pd-107 | 6.346E-07 | 6.358E-07 | 6.366E-07 | 6.378E-07 | 6.414E-07 | 6.490E-07 | 6.797E-07 |
| Ag-108 | 1.100E-11 | 1.102E-11 | 1.103E-11 | 1.104E-11 | 1.109E-11 | 1.119E-11 | 1.159E-11 |
| Ag-108m | 1.236E-10 | 1.238E-10 | 1.239E-10 | 1.241E-10 | 1.246E-10 | 1.257E-10 | 1.302E-10 |
| Pd-109 | 5.952E-05 | 4.723E-21 | 8.720E-32 | -- | -- | -- | -- |
| Ag-109m | 6.037E-05 | 7.816E-07 | 7.596E-07 | 7.277E-07 | 6.398E-07 | 4.909E-07 | 1.726E-07 |
| Cd-109 | 8.160E-07 | 7.816E-07 | 7.596E-07 | 7.277E-07 | 6.398E-07 | 4.909E-07 | 1.726E-07 |
| Ag-110 | 4.647E-05 | 4.283E-05 | 4.058E-05 | 3.740E-05 | 2.931E-05 | 1.774E-05 | 2.449E-06 |
| Ag-110m | 3.494E-03 | 3.222E-03 | 3.052E-03 | 2.813E-03 | 2.204E-03 | 1.334E-03 | 1.842E-04 |
| Pd-111 | 2.548E-15 | -- | -- | -- | -- | -- | -- |
| Pd-111m | 3.497E-15 | -- | -- | -- | -- | -- | -- |
| Ag-111 | 1.716E+00 | 1.055E-01 | 1.643E-02 | 1.010E-03 | 2.345E-07 | 7.765E-15 | -- |
| Ag-111m | 3.657E-15 | -- | -- | -- | -- | -- | -- |
| Pd-112 | 7.729E-04 | 1.276E-14 | 8.272E-22 | 1.365E-32 | -- | -- | -- |
| Ag-112 | 9.155E-04 | 1.511E-14 | 9.796E-22 | 1.617E-32 | -- | -- | -- |
| Ag-113 | 5.631E-14 | -- | -- | -- | -- | -- | -- |
| Cd-113m | 5.642E-04 | 5.631E-04 | 5.623E-04 | 5.612E-04 | 5.579E-04 | 5.510E-04 | 5.246E-04 |
| In-113m | 1.792E-02 | 1.498E-02 | 1.330E-02 | 1.112E-02 | 6.506E-03 | 2.157E-03 | 2.777E-05 |
| Sn-113 | 1.791E-02 | 1.497E-02 | 1.329E-02 | 1.112E-02 | 6.504E-03 | 2.156E-03 | 2.775E-05 |
| In-114 | 9.659E-05 | 6.359E-05 | 4.813E-05 | 3.169E-05 | 9.039E-06 | 6.838E-07 | 2.593E-11 |
| In-114m | 1.009E-04 | 6.645E-05 | 5.029E-05 | 3.310E-05 | 9.445E-06 | 7.144E-07 | 2.709E-11 |
| Cd-115 | 9.329E-02 | 8.266E-06 | 1.640E-08 | 1.453E-12 | 1.011E-24 | | |
| Cd-115m | 1.379E-01 | 8.665E-02 | 6.360E-02 | 3.997E-02 | 9.922E-03 | 5.638E-04 | 6.922E-09 |
| In-115 | 7.167E-16 | 7.320E-16 | 7.386E-16 | 7.456E-16 | 7.575E-16 | 7.686E-16 | 8.052E-16 |
| In-115m | 1.014E-01 | 1.507E-05 | 4.485E-06 | 2.807E-06 | 6.971E-07 | 3.962E-08 | 4.862E-13 |
| Cd-117 | 2.036E-28 | -- | -- | -- | -- | -- | -- |
| Cd-117m | 3.791E-22 | -- | -- | -- | -- | -- | -- |
| In-117 | 5.035E-22 | -- | -- | -- | -- | -- | -- |
| In-117m | 3.884E-22 | -- | -- | -- | -- | -- | -- |
| Sn-117m | 1.843E-01 | 4.183E-02 | 1.557E-02 | 3.534E-03 | 4.131E-05 | 4.357E-09 | 9.060E-25 |
| Sn-119m | 8.744E-02 | 8.048E-02 | 7.615E-02 | 7.008E-02 | 5.464E-02 | 3.273E-02 | 4.341E-03 |
| Sn-121 | 4.213E-03 | 3.454E-11 | 1.403E-16 | 1.150E-24 | -- | -- | -- |
| Sn-121m | 4.770E-06 | 4.772E-06 | 4.775E-06 | 4.780E-06 | 4.791E-06 | 4.813E-06 | 4.903E-06 |
| Sb-122 | 1.613E-03 | 7.310E-07 | 4.314E-09 | 1.956E-12 | 1.821E-22 | -- | -- |
| Sn-123 | 1.741E-01 | 1.485E-01 | 1.335E-01 | 1.139E-01 | 7.064E-02 | 2.645E-02 | 5.496E-04 |
| Te-123 | 3.027E-19 | 3.164E-19 | 3.244E-19 | 3.347E-19 | 3.578E-19 | 3.825E-19 | 4.118E-19 |
| Te-123m | 2.500E-06 | 2.105E-06 | 1.877E-06 | 1.581E-06 | 9.440E-07 | 3.265E-07 | 4.972E-09 |
| Sb-124 | 4.921E-03 | 3.489E-03 | 2.776E-03 | 1.969E-03 | 7.025E-04 | 8.420E-05 | 1.960E-08 |
| Sn-125 | 7.043E-01 | 8.160E-02 | 1.940E-02 | 2.248E-03 | 3.506E-06 | 5.820E-12 | -- |
| Sb-125 | 2.327E-01 | 2.342E-01 | 2.320E-01 | 2.278E-01 | 2.154E-01 | 1.920E-01 | 1.219E-01 |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Te-125m | 2.462E-02 | 3.350E-02 | 3.782E-02 | 4.241E-02 | 4.805E-02 | 4.634E-02 | 2.975E-02 |
| Sn-126 | 8.590E-06 | 8.606E-06 | 8.618E-06 | 8.634E-06 | 8.683E-06 | 8.785E-06 | 9.201E-06 |
| Sb-126 | 3.513E-02 | 6.576E-03 | 2.153E-03 | 4.040E-04 | 3.856E-06 | 1.230E-06 | 1.288E-06 |
| Sb-126m | 8.590E-06 | 8.606E-06 | 8.618E-06 | 8.634E-06 | 8.683E-06 | 8.785E-06 | 9.201E-06 |
| Sn-127 | 2.410E-34 | -- | -- | -- | -- | -- | -- |
| Sb-127 | 1.527E+00 | 6.896E-03 | 1.884E-04 | 8.510E-07 | 7.845E-14 | 2.590E-28 | -- |
| Te-127 | 1.982E+00 | 4.420E-01 | 3.841E-01 | 3.178E-01 | 1.804E-01 | 5.620E-02 | 5.656E-04 |
| Te-127m | 5.292E-01 | 4.447E-01 | 3.921E-01 | 3.245E-01 | 1.842E-01 | 5.737E-02 | 5.774E-04 |
| Xe-127 | 6.466E-09 | 3.661E-09 | 2.505E-09 | 1.418E-09 | 2.570E-10 | 7.647E-12 | 7.315E-18 |
| Sb-128 | 1.031E-08 | -- | -- | -- | -- | -- | -- |
| Sb-129 | 5.144E-16 | -- | -- | -- | -- | -- | -- |
| Te-129 | 1.709E+00 | 9.220E-01 | 6.111E-01 | 3.297E-01 | 5.178E-02 | 1.147E-03 | 3.427E-10 |
| Te-129m | 2.625E+00 | 1.416E+00 | 9.388E-01 | 5.066E-01 | 7.956E-02 | 1.762E-03 | 5.264E-10 |
| I-129 | 2.892E-07 | 2.967E-07 | 3.001E-07 | 3.030E-07 | 3.075E-07 | 3.114E-07 | 3.261E-07 |
| Xe-129m | 9.403E-07 | 7.001E-08 | 1.239E-08 | 9.231E-10 | 3.810E-13 | 4.126E-20 | -- |
| I-130 | 3.569E-07 | 1.045E-24 | 2.139E-36 | -- | -- | -- | -- |
| Te-131 | 1.050E-02 | 6.272E-10 | 9.581E-15 | 5.722E-22 | -- | -- | -- |
| Te-131m | 4.666E-02 | 2.785E-09 | 4.257E-14 | 2.542E-21 | -- | -- | -- |
| I-131 | 3.649E+01 | 2.755E+00 | 4.919E-01 | 3.711E-02 | 1.594E-05 | 1.870E-12 | -- |
| Xe-131m | 7.846E-01 | 2.206E-01 | 7.740E-02 | 1.464E-02 | 8.199E-05 | 1.713E-09 | 5.914E-28 |
| Te-132 | 1.449E+01 | 2.447E-02 | 3.478E-04 | 5.910E-07 | 2.875E-15 | 2.227E-32 | -- |
| I-132 | 1.493E+01 | 2.530E-02 | 3.583E-04 | 6.090E-07 | 2.961E-15 | 2.294E-32 | -- |
| Cs-132 | 4.005E-04 | 1.617E-05 | 1.903E-06 | 7.679E-08 | 5.051E-12 | 1.245E-20 | -- |
| I-133 | 6.185E-02 | 2.355E-12 | 2.665E-19 | 1.014E-29 | -- | -- | -- |
| Xe-133 | 5.583E+01 | 1.067E+00 | 7.601E-02 | 1.446E-03 | 9.941E-09 | 2.350E-19 | -- |
| Xe-133m | 3.625E-01 | 2.737E-05 | 4.874E-08 | 3.670E-12 | 1.567E-24 | -- | -- |
| Cs-134 | 1.460E-01 | 1.422E-01 | 1.399E-01 | 1.363E-01 | 1.262E-01 | 1.076E-01 | 5.756E-02 |
| Cs-134m | 3.689E-26 | -- | -- | -- | -- | -- | -- |
| I-135 | 1.967E-09 | -- | -- | -- | -- | -- | -- |
| Xe-135 | 5.639E-06 | -- | -- | -- | -- | -- | -- |
| Xe-135m | 3.151E-10 | -- | -- | -- | -- | -- | -- |
| Cs-135 | 4.200E-06 | 4.208E-06 | 4.214E-06 | 4.222E-06 | 4.246E-06 | 4.295E-06 | 4.499E-06 |
| Ba-135m | 9.751E-08 | 2.731E-15 | 2.520E-20 | 7.061E-28 | -- | -- | -- |
| Cs-136 | 5.920E-01 | 1.213E-01 | 4.216E-02 | 8.642E-03 | 7.434E-05 | 4.167E-09 | 7.178E-26 |
| Ba-136m | 9.759E-02 | 1.999E-02 | 6.951E-03 | 1.424E-03 | 1.225E-05 | 6.870E-10 | 1.183E-26 |
| Cs-137 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 |
| Ba-137m | 9.460E-01 | 9.459E-01 | 9.461E-01 | 9.460E-01 | 9.459E-01 | 9.461E-01 | 9.458E-01 |
| La-138 | 1.553E-15 | 1.556E-15 | 1.558E-15 | 1.561E-15 | 1.570E-15 | 1.588E-15 | 1.663E-15 |
| Ba-140 | 9.002E+01 | 1.774E+01 | 6.009E+00 | 1.184E+00 | 9.070E-03 | 4.003E-07 | 2.681E-24 |
| La-140 | 1.033E+02 | 2.042E+01 | 6.922E+00 | 1.363E+00 | 1.044E-02 | 4.607E-07 | 3.086E-24 |
| La-141 | 6.763E-17 | -- | -- | -- | -- | -- | -- |
| Ce-141 | 1.057E+02 | 5.588E+01 | 3.653E+01 | 1.930E+01 | 2.850E+00 | 5.554E-02 | 1.002E-08 |
| Ce-142 | 2.632E-10 | 2.637E-10 | 2.640E-10 | 2.645E-10 | 2.660E-10 | 2.692E-10 | 2.819E-10 |
| Pr-142 | 4.971E-05 | 2.332E-16 | 6.535E-24 | 3.065E-35 | -- | -- | -- |
| Ce-143 | 9.286E-01 | 2.516E-07 | 1.053E-11 | 2.856E-18 | 5.676E-38 | -- | -- |
| Pr-143 | 9.106E+01 | 1.972E+01 | 7.107E+00 | 1.537E+00 | 1.555E-02 | 1.219E-06 | 7.845E-23 |
| Ce-144 | 2.742E+01 | 2.554E+01 | 2.436E+01 | 2.268E+01 | 1.832E+01 | 1.179E+01 | 2.081E+00 |

# ATTACHMENT A
# REACTOR SOURCE TERMS



| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Pr-144 | 2.742E+01 | 2.554E+01 | 2.436E+01 | 2.268E+01 | 1.832E+01 | 1.179E+01 | 2.081E+00 |
| Pr-144m | 3.291E-01 | 3.064E-01 | 2.923E-01 | 2.721E-01 | 2.198E-01 | 1.416E-01 | 2.497E-02 |
| Nd-144 | 1.852E-15 | 2.575E-15 | 3.030E-15 | 3.673E-15 | 5.362E-15 | 7.923E-15 | 1.211E-14 |
| Pr-145 | 8.167E-11 | -- | -- | -- | -- | -- | -- |
| Pm-146 | 2.844E-06 | 2.820E-06 | 2.802E-06 | 2.780E-06 | 2.710E-06 | 2.572E-06 | 2.093E-06 |
| Sm-146 | 1.269E-14 | 1.356E-14 | 1.415E-14 | 1.501E-14 | 1.759E-14 | 2.273E-14 | 4.126E-14 |
| Nd-147 | 3.149E+01 | 4.814E+00 | 1.377E+00 | 2.104E-01 | 7.518E-04 | 6.903E-09 | 9.479E-29 |
| Pm-147 | 3.502E+00 | 3.739E+00 | 3.731E+00 | 3.670E+00 | 3.459E+00 | 3.062E+00 | 1.890E+00 |
| Sm-147 | 3.109E-12 | 5.066E-12 | 6.401E-12 | 8.384E-12 | 1.413E-11 | 2.506E-11 | 5.851E-11 |
| Pm-148 | 1.150E+00 | 3.798E-02 | 1.216E-02 | 6.324E-03 | 1.396E-03 | 6.304E-05 | 3.123E-10 |
| Pm-148m | 4.254E-01 | 2.575E-01 | 1.844E-01 | 1.116E-01 | 2.478E-02 | 1.119E-03 | 5.544E-09 |
| Sm-148 | 5.375E-17 | 5.827E-17 | 5.942E-17 | 6.058E-17 | 6.217E-17 | 6.326E-17 | 6.625E-17 |
| Pm-149 | 1.521E+00 | 1.258E-04 | 2.397E-07 | 1.983E-11 | 1.124E-23 | -- | -- |
| Sm-149 | 7.076E-17 | 7.180E-17 | 7.190E-17 | 7.204E-17 | 7.244E-17 | 7.330E-17 | 7.676E-17 |
| Pm-150 | 3.028E-28 | -- | -- | -- | -- | -- | -- |
| Eu-150 | 2.067E-11 | 2.067E-11 | 2.068E-11 | 2.069E-11 | 2.071E-11 | 2.074E-11 | 2.091E-11 |
| Pm-151 | 4.398E-02 | 1.022E-09 | 8.323E-15 | 1.933E-22 | -- | -- | -- |
| Sm-151 | 1.780E-02 | 1.782E-02 | 1.784E-02 | 1.786E-02 | 1.793E-02 | 1.807E-02 | 1.863E-02 |
| Eu-152 | 4.361E-05 | 4.353E-05 | 4.345E-05 | 4.334E-05 | 4.305E-05 | 4.244E-05 | 4.014E-05 |
| Eu-152m | 5.375E-11 | -- | -- | -- | -- | -- | -- |
| Gd-152 | 1.686E-18 | 1.696E-18 | 1.703E-18 | 1.712E-18 | 1.741E-18 | 1.800E-18 | 2.037E-18 |
| Sm-153 | 2.503E-01 | 5.722E-06 | 4.611E-09 | 1.054E-13 | 1.260E-27 | -- | -- |
| Gd-153 | 1.706E-06 | 1.568E-06 | 1.483E-06 | 1.363E-06 | 1.059E-06 | 6.306E-07 | 8.152E-08 |
| Eu-154 | 6.661E-03 | 6.628E-03 | 6.608E-03 | 6.577E-03 | 6.485E-03 | 6.298E-03 | 5.613E-03 |
| Eu-155 | 5.420E-02 | 5.366E-02 | 5.333E-02 | 5.281E-02 | 5.133E-02 | 4.837E-02 | 3.831E-02 |
| Sm-156 | 2.937E-08 | -- | -- | -- | -- | -- | -- |
| Eu-156 | 1.265E+00 | 3.222E-01 | 1.295E-01 | 3.299E-02 | 5.454E-04 | 1.173E-07 | 4.054E-22 |
| Eu-157 | 1.717E-05 | 9.467E-20 | 2.955E-29 | -- | -- | -- | -- |
| Gd-159 | 3.357E-05 | 7.483E-17 | 1.277E-24 | 2.845E-36 | -- | -- | -- |
| Tb-160 | 1.454E-03 | 1.092E-03 | 9.031E-04 | 6.787E-04 | 2.880E-04 | 4.933E-05 | 4.696E-08 |
| Tb-161 | 2.119E-02 | 1.052E-03 | 1.421E-04 | 7.053E-06 | 8.626E-10 | 7.628E-18 | -- |
| Tb-162m | 4.120E-36 | -- | -- | -- | -- | -- | -- |
| Dy-165 | 1.112E-33 | -- | -- | -- | -- | -- | -- |
| Dy-166 | 2.844E-04 | 6.250E-07 | 1.056E-08 | 2.318E-11 | 2.451E-19 | 9.387E-36 | -- |
| Ho-166 | 4.219E-04 | 9.308E-07 | 1.574E-08 | 3.453E-11 | 3.651E-19 | 1.399E-35 | -- |
| Ho-166m | 8.400E-10 | 8.416E-10 | 8.427E-10 | 8.443E-10 | 8.488E-10 | 8.585E-10 | 8.983E-10 |
| Er-169 | 6.862E-08 | 7.526E-09 | 1.724E-09 | 1.891E-10 | 2.494E-13 | 2.947E-19 | -- |
| Tm-170 | 4.296E-10 | 3.664E-10 | 3.293E-10 | 2.807E-10 | 1.737E-10 | 6.476E-11 | 1.322E-12 |
| Er-171 | 5.268E-26 | -- | -- | -- | -- | -- | -- |
| Tm-171 | 5.214E-13 | 5.071E-13 | 4.979E-13 | 4.843E-13 | 4.456E-13 | 3.754E-13 | 1.909E-13 |
| Er-172 | 2.125E-21 | 8.033E-26 | 8.043E-26 | 3.041E-30 | -- | -- | -- |
| Tm-172 | 1.225E-12 | 4.799E-16 | 2.569E-18 | 1.006E-21 | 6.048E-32 | -- | -- |
| Tm-173 | 4.256E-21 | -- | -- | -- | -- | -- | -- |
| Yb-175 | 4.778E-18 | 3.399E-20 | 1.591E-21 | -- | -- | -- | -- |
| Hf-175 | 2.747E-03 | 2.044E-03 | 1.679E-03 | 1.250E-03 | 5.157E-04 | 8.331E-05 | 6.301E-08 |
| Lu-176 | 4.928E-17 | 4.937E-17 | 4.944E-17 | 4.953E-17 | 4.981E-17 | 5.040E-17 | 5.278E-17 |
| Lu-176m | 1.390E-24 | -- | -- | -- | -- | -- | -- |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Lu-177 | 7.547E-07 | 3.428E-08 | 4.503E-09 | 3.536E-10 | 1.066E-10 | 4.709E-11 | 1.881E-12 |
| Lu-177m | 9.807E-10 | 8.593E-10 | 7.866E-10 | 6.892E-10 | 4.635E-10 | 2.048E-10 | 8.178E-12 |
| Hf-180m | 1.141E-16 | -- | -- | -- | -- | -- | -- |
| Hf-181 | 3.649E-02 | 2.239E-02 | 1.617E-02 | 9.919E-03 | 2.290E-03 | 1.121E-04 | 7.633E-10 |
| W-181 | 4.898E-05 | 4.136E-05 | 3.693E-05 | 3.116E-05 | 1.873E-05 | 6.566E-06 | 1.054E-07 |
| Hf-182 | 2.456E-12 | 2.460E-12 | 2.464E-12 | 2.468E-12 | 2.482E-12 | 2.512E-12 | 2.630E-12 |
| Ta-182 | 1.323E-04 | 1.107E-04 | 9.823E-05 | 8.212E-05 | 4.803E-05 | 1.590E-05 | 2.039E-07 |
| Ta-183 | 6.140E-05 | 1.043E-06 | 6.889E-08 | 1.170E-09 | 5.733E-15 | 6.747E-26 | -- |
| W-185 | 1.464E-03 | 1.112E-03 | 9.259E-04 | 7.034E-04 | 3.083E-04 | 5.642E-05 | 6.968E-08 |
| Re-186 | 1.204E-05 | 4.897E-08 | 1.248E-09 | 5.077E-12 | 3.423E-19 | 5.937E-34 | -- |
| W-187 | 6.720E-05 | 5.749E-14 | 5.180E-20 | 4.431E-29 | -- | -- | -- |
| Re-187 | 2.892E-13 | 2.897E-13 | 2.901E-13 | 2.907E-13 | 2.923E-13 | 2.958E-13 | 3.097E-13 |
| W-188 | 8.274E-05 | 6.143E-05 | 5.038E-05 | 3.740E-05 | 1.531E-05 | 2.435E-06 | 1.729E-09 |
| Re-188 | 8.371E-05 | 6.207E-05 | 5.091E-05 | 3.778E-05 | 1.547E-05 | 2.460E-06 | 1.747E-09 |
| Re-189 | 6.394E-12 | 7.711E-21 | 8.739E-27 | 1.054E-35 | -- | -- | -- |
| Os-191 | 2.225E-11 | 5.781E-12 | 2.354E-12 | 6.115E-13 | 1.072E-14 | 2.601E-18 | 1.448E-32 |
| Os-191m | 1.267E-16 | 2.699E-33 | -- | -- | -- | -- | -- |
| Ir-192 | 2.132E-13 | 1.613E-13 | 1.339E-13 | 1.013E-13 | 4.386E-14 | 7.830E-15 | 8.825E-18 |
| Ir-192m | 9.334E-20 | 9.349E-20 | 9.361E-20 | 9.376E-20 | 9.424E-20 | 9.521E-20 | 9.914E-20 |
| Os-193 | 4.668E-21 | 4.768E-28 | 1.042E-32 | 1.067E-39 | -- | -- | -- |
| Pt-193 | 7.932E-21 | 8.033E-21 | 8.043E-21 | 8.059E-21 | 8.102E-21 | 8.192E-21 | 8.556E-21 |
| Pt-193m | 3.692E-18 | 2.962E-20 | 1.408E-21 | -- | -- | -- | -- |
| Os-194 | 1.051E-23 | 1.053E-23 | 1.055E-23 | 1.057E-23 | 1.063E-23 | 1.075E-23 | 8.659E-24 |
| Ir-194 | 6.969E-20 | 1.054E-23 | 1.055E-23 | 1.057E-23 | 1.063E-23 | 1.076E-23 | 8.940E-24 |
| Pt-195m | 1.715E-22 | -- | -- | -- | -- | -- | -- |
| Pb-204 | 3.232E-19 | 3.238E-19 | 3.242E-19 | 3.248E-19 | 3.267E-19 | 3.305E-19 | 3.461E-19 |
| Pb-205 | 2.681E-13 | 2.686E-13 | 2.689E-13 | 2.694E-13 | 2.710E-13 | 2.741E-13 | 2.871E-13 |
| Tl-206 | 1.378E-19 | 1.381E-19 | 1.382E-19 | 1.385E-19 | 1.393E-19 | 1.409E-19 | 1.476E-19 |
| Bi-208 | 1.600E-19 | 1.603E-19 | 1.605E-19 | 1.608E-19 | 1.617E-19 | 1.636E-19 | 1.713E-19 |
| Pb-209 | 8.909E-29 | -- | -- | -- | -- | -- | -- |
| Bi-210 | 1.296E-12 | 2.049E-14 | 1.290E-15 | 2.039E-17 | 8.053E-23 | 6.077E-34 | -- |
| Bi-210m | 1.383E-19 | 1.386E-19 | 1.388E-19 | 1.390E-19 | 1.398E-19 | 1.415E-19 | 1.482E-19 |
| Po-210 | 1.193E-12 | 1.070E-12 | 9.697E-13 | 8.360E-13 | 5.357E-13 | 2.144E-13 | 5.791E-15 |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


Table A-6. TRIGA activity relative to Cs-137.

| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| H-3 | 4.188E-03 | 4.175E-03 | 4.170E-03 | 4.157E-03 | 4.125E-03 | 4.054E-03 | 3.798E-03 |
| Be-10 | 5.750E-11 | 5.759E-11 | 5.769E-11 | 5.779E-11 | 5.814E-11 | 5.879E-11 | 6.161E-11 |
| C-14 | 1.789E-04 | 1.792E-04 | 1.795E-04 | 1.798E-04 | 1.808E-04 | 1.829E-04 | 1.917E-04 |
| Na-24 | 3.740E-10 | 1.331E-24 | 3.104E-34 | -- | -- | -- | -- |
| Mg-28 | 2.329E-18 | 1.007E-28 | 1.241E-35 | -- | -- | -- | -- |
| Al-28 | 2.651E-18 | 1.008E-28 | 9.992E-36 | -- | -- | -- | -- |
| Si-31 | 1.860E-30 | -- | -- | -- | -- | -- | -- |
| Si-32 | 2.732E-13 | 2.736E-13 | 2.740E-13 | 2.745E-13 | 2.760E-13 | 2.791E-13 | 2.918E-13 |
| P-32 | 7.639E-03 | 1.787E-03 | 6.791E-04 | 1.589E-04 | 2.038E-06 | 2.599E-10 | 2.918E-13 |
| P-33 | 2.987E-07 | 1.303E-07 | 7.493E-08 | 3.267E-08 | 2.711E-09 | 1.612E-11 | 2.701E-20 |
| S-35 | 2.407E-04 | 1.903E-04 | 1.629E-04 | 1.288E-04 | 6.378E-05 | 1.499E-05 | 4.985E-08 |
| Cl-36 | 6.687E-12 | 6.698E-12 | 6.710E-12 | 6.721E-12 | 6.761E-12 | 6.837E-12 | 7.166E-12 |
| Ar-37 | 1.464E-15 | 8.102E-16 | 5.463E-16 | 3.022E-16 | 5.118E-17 | 1.333E-18 | 7.352E-25 |
| Ar-39 | 1.274E-17 | 1.276E-17 | 1.278E-17 | 1.280E-17 | 1.287E-17 | 1.299E-17 | 1.354E-17 |
| Ar-42 | 3.352E-26 | 3.358E-26 | 3.363E-26 | 3.369E-26 | 3.389E-26 | 3.427E-26 | 3.592E-26 |
| K-42 | 3.358E-26 | 3.363E-26 | 3.369E-26 | 3.389E-26 | 3.427E-26 | 3.592E-26 | -- |
| K-43 | 8.215E-16 | 2.114E-25 | 8.557E-32 | -- | -- | -- | -- |
| Ca-45 | 1.091E-07 | 9.619E-08 | 8.846E-08 | 7.802E-08 | 5.352E-08 | 2.462E-08 | 1.154E-09 |
| Sc-46 | 1.372E-05 | 1.073E-05 | 9.109E-06 | 7.118E-06 | 3.401E-06 | 7.431E-07 | 1.851E-09 |
| Ca-47 | 8.335E-11 | 8.525E-13 | 4.019E-14 | 4.111E-16 | 4.400E-22 | 2.260E-34 | -- |
| Sc-47 | 2.940E-06 | 5.926E-09 | 9.491E-11 | 1.931E-13 | 3.266E-21 | 8.647E-34 | -- |
| Sc-48 | 5.043E-08 | 5.697E-13 | 2.869E-16 | 3.241E-21 | 4.676E-36 | -- | -- |
| V-50 | 5.042E-18 | 5.051E-18 | 5.059E-18 | 5.068E-18 | 5.098E-18 | 5.156E-18 | 5.403E-18 |
| Cr-51 | 4.249E+00 | 2.009E+00 | 1.221E+00 | 5.772E-01 | 6.112E-02 | 6.003E-04 | 7.286E-12 |
| Mn-54 | 2.269E-01 | 2.126E-01 | 2.037E-01 | 1.910E-01 | 1.573E-01 | 1.055E-01 | 2.188E-02 |
| Fe-55 | 2.027E+00 | 1.986E+00 | 1.961E+00 | 1.922E+00 | 1.811E+00 | 1.599E+00 | 9.837E-01 |
| Mn-56 | 1.294E-27 | -- | -- | -- | -- | -- | -- |
| Co-58 | 4.637E-01 | 3.463E-01 | 2.851E-01 | 2.129E-01 | 8.871E-02 | 1.462E-02 | 1.198E-05 |
| Fe-59 | 8.697E-02 | 5.490E-02 | 4.041E-02 | 2.550E-02 | 6.413E-03 | 3.739E-04 | 5.086E-09 |
| Ni-59 | 2.624E-04 | 2.629E-04 | 2.633E-04 | 2.638E-04 | 2.654E-04 | 2.683E-04 | 2.812E-04 |
| Co-60 | 9.103E-01 | 9.025E-01 | 8.973E-01 | 8.895E-01 | 8.657E-01 | 8.191E-01 | 6.599E-01 |
| Ni-63 | 4.025E-02 | 4.030E-02 | 4.035E-02 | 4.039E-02 | 4.056E-02 | 4.086E-02 | 4.218E-02 |
| Cu-64 | 3.157E-08 | 2.715E-25 | 1.145E-36 | -- | -- | -- | -- |
| Ni-65 | 1.006E-30 | -- | -- | -- | -- | -- | -- |
| Zn-65 | 3.897E-03 | 3.584E-03 | 3.392E-03 | 3.120E-03 | 2.430E-03 | 1.452E-03 | 1.908E-04 |
| Ni-66 | 1.331E-09 | 1.432E-13 | 3.254E-16 | 3.502E-20 | 4.370E-32 | -- | -- |
| Cu-66 | 1.333E-09 | 1.435E-13 | 3.258E-16 | 3.507E-20 | 4.376E-32 | -- | -- |
| Cu-67 | 2.536E-11 | 7.967E-15 | 3.710E-17 | 1.165E-20 | 3.614E-31 | -- | -- |
| Zn-69 | 1.140E-09 | 2.030E-25 | 6.281E-36 | -- | -- | -- | -- |
| Zn-69m | 1.062E-09 | 1.890E-25 | 5.989E-36 | -- | -- | -- | -- |
| Zn-71 | 8.316E-29 | -- | -- | -- | -- | -- | -- |
| Zn-71m | 1.647E-25 | -- | -- | -- | -- | -- | -- |
| Ge-71 | 5.889E-10 | 1.013E-10 | 3.133E-11 | 5.387E-12 | 2.743E-14 | 5.213E-19 | 1.266E-37 |
| Zn-72 | 4.994E-06 | 1.092E-10 | 8.540E-14 | 1.866E-18 | 1.951E-32 | -- | -- |
| Ga-72 | 7.168E-06 | 1.567E-10 | 1.226E-13 | 2.679E-18 | 2.801E-32 | -- | -- |
| Ga-73 | 9.036E-19 | -- | -- | -- | -- | -- | -- |
| Ge-73m | 9.036E-19 | -- | -- | -- | -- | -- | -- |

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**REACTOR SOURCE TERMS**
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| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| As-76 | 3.253E-08 | 1.897E-16 | 6.147E-22 | 3.584E-30 | -- | -- | -- |
| Ge-77 | 3.041E-09 | 2.010E-28 | -- | -- | -- | -- | -- |
| As-77 | 3.409E-04 | 8.873E-10 | 1.680E-13 | 4.372E-19 | 7.719E-36 | -- | -- |
| Se-77m | 8.458E-07 | 2.201E-12 | 4.167E-16 | 1.084E-21 | -- | -- | -- |
| Se-79 | 4.181E-06 | 4.188E-06 | 4.195E-06 | 4.202E-06 | 4.228E-06 | 4.275E-06 | 4.481E-06 |
| Kr-79 | 1.228E-15 | 7.575E-22 | 5.493E-26 | 3.390E-32 | -- | -- | -- |
| Br-80 | 3.547E-23 | -- | -- | -- | -- | -- | -- |
| Br-80m | 3.314E-23 | -- | -- | -- | -- | -- | -- |
| Kr-81 | 1.796E-13 | 1.799E-13 | 1.802E-13 | 1.805E-13 | 1.816E-13 | 1.837E-13 | 1.925E-13 |
| Br-82 | 3.284E-05 | 2.389E-11 | 1.933E-15 | 1.406E-21 | -- | -- | -- |
| Br-83 | 7.174E-31 | -- | -- | -- | -- | -- | -- |
| Kr-83m | 3.061E-30 | -- | -- | -- | -- | -- | -- |
| Kr-85 | 1.146E-01 | 1.141E-01 | 1.139E-01 | 1.134E-01 | 1.123E-01 | 1.099E-01 | 1.013E-01 |
| Kr-85m | 2.019E-16 | -- | -- | -- | -- | -- | -- |
| Rb-86 | 1.102E-03 | 3.623E-04 | 1.726E-04 | 5.673E-05 | 2.015E-06 | 2.104E-09 | 3.607E-21 |
| Rb-87 | 2.778E-10 | 2.783E-10 | 2.788E-10 | 2.793E-10 | 2.809E-10 | 2.841E-10 | 2.977E-10 |
| Sr-87m | 5.077E-32 | -- | -- | -- | -- | -- | -- |
| Kr-88 | 2.701E-25 | -- | -- | -- | -- | -- | -- |
| Rb-88 | 3.016E-25 | -- | -- | -- | -- | -- | -- |
| Sr-89 | 8.741E+00 | 5.799E+00 | 4.414E+00 | 2.929E+00 | 8.564E-01 | 6.814E-02 | 3.156E-06 |
| Y-89m | 3.910E-09 | 6.767E-12 | 9.754E-14 | 1.686E-16 | 8.717E-25 | -- | -- |
| Zr-89 | 3.917E-09 | 6.777E-12 | 9.771E-14 | 1.688E-16 | 8.734E-25 | -- | -- |
| Sr-90 | 9.543E-01 | 9.542E-01 | 9.542E-01 | 9.541E-01 | 9.538E-01 | 9.533E-01 | 9.529E-01 |
| Y-90 | 9.594E-01 | 9.542E-01 | 9.542E-01 | 9.549E-01 | 9.547E-01 | 9.533E-01 | 9.529E-01 |
| Y-90m | 1.549E-28 | -- | -- | -- | -- | -- | -- |
| Sr-91 | 3.039E-07 | -- | -- | -- | -- | -- | -- |
| Y-91 | 1.094E+01 | 7.683E+00 | 6.073E+00 | 4.264E+00 | 1.476E+00 | 1.663E-01 | 3.039E-05 |
| Y-91m | 1.931E-07 | -- | -- | -- | -- | -- | -- |
| Sr-92 | 2.748E-26 | -- | -- | -- | -- | -- | -- |
| Y-92 | 2.060E-19 | -- | -- | -- | -- | -- | -- |
| Nb-92 | 1.104E-05 | 1.428E-06 | 3.655E-07 | 4.730E-08 | 1.027E-10 | 3.368E-16 | 8.004E-38 |
| Y-93 | 9.662E-07 | 3.359E-28 | -- | -- | -- | -- | -- |
| Zr-93 | 4.051E-05 | 4.058E-05 | 4.065E-05 | 4.071E-05 | 4.096E-05 | 4.142E-05 | 4.341E-05 |
| Nb-93m | 3.238E-06 | 3.392E-06 | 3.495E-06 | 3.648E-06 | 4.109E-06 | 5.054E-06 | 8.780E-06 |
| Mo-93 | 4.963E-08 | 4.970E-08 | 4.979E-08 | 4.987E-08 | 5.017E-08 | 5.073E-08 | 5.315E-08 |
| Mo-93m | 7.064E-17 | -- | -- | -- | -- | -- | -- |
| Nb-94 | 2.180E-06 | 2.183E-06 | 2.187E-06 | 2.191E-06 | 2.204E-06 | 2.229E-06 | 2.335E-06 |
| Zr-95 | 1.420E+01 | 1.027E+01 | 8.285E+00 | 5.997E+00 | 2.275E+00 | 3.093E-01 | 1.185E-04 |
| Nb-95 | 1.566E+01 | 1.402E+01 | 1.246E+01 | 1.003E+01 | 4.490E+00 | 6.717E-01 | 2.631E-04 |
| Nb-95m | 1.048E-01 | 7.620E-02 | 6.146E-02 | 4.448E-02 | 1.688E-02 | 2.294E-03 | 8.791E-07 |
| Nb-96 | 2.155E-06 | 1.127E-15 | 7.318E-22 | 3.827E-31 | -- | -- | -- |
| Zr-97 | 8.388E-04 | 1.257E-16 | 3.550E-25 | 6.632E-38 | -- | -- | -- |
| Nb-97 | 8.431E-04 | 1.355E-16 | 3.825E-25 | -- | -- | -- | -- |
| Nb-97m | 7.946E-04 | 1.191E-16 | 3.363E-25 | -- | -- | -- | -- |
| Tc-98 | 3.613E-12 | 3.619E-12 | 3.625E-12 | 3.631E-12 | 3.653E-12 | 3.694E-12 | 3.872E-12 |
| Mo-99 | 1.024E+00 | 5.331E-04 | 3.454E-06 | 1.799E-09 | 2.545E-19 | -- | -- |
| Tc-99 | 1.423E-04 | 1.426E-04 | 1.429E-04 | 1.431E-04 | 1.440E-04 | 1.456E-04 | 1.526E-04 |

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| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Tc-99m | 9.848E-01 | 5.131E-04 | 3.324E-06 | 1.731E-09 | 2.450E-19 | -- | -- |
| Rh-102 | 7.095E-07 | 6.969E-07 | 6.890E-07 | 6.767E-07 | 6.418E-07 | 5.749E-07 | 3.736E-07 |
| Ru-103 | 5.748E+00 | 3.391E+00 | 2.387E+00 | 1.408E+00 | 2.894E-01 | 1.113E-02 | 2.946E-08 |
| Rh-103m | 5.181E+00 | 3.057E+00 | 2.152E+00 | 1.270E+00 | 2.609E-01 | 1.004E-02 | 2.655E-08 |
| Ru-105 | 1.310E-16 | -- | -- | -- | -- | -- | -- |
| Rh-105 | 2.486E-02 | 1.848E-08 | 1.518E-12 | 1.128E-18 | 4.677E-37 | -- | -- |
| Rh-105m | 3.679E-17 | -- | -- | -- | -- | -- | -- |
| Ru-106 | 9.112E-01 | 8.627E-01 | 8.322E-01 | 7.878E-01 | 6.689E-01 | 4.773E-01 | 1.265E-01 |
| Rh-106 | 9.112E-01 | 8.627E-01 | 8.322E-01 | 7.878E-01 | 6.689E-01 | 4.773E-01 | 1.265E-01 |
| Rh-106m | 6.420E-36 | -- | -- | -- | -- | -- | -- |
| Ag-106 | 1.289E-13 | 1.118E-14 | 2.192E-15 | 1.902E-16 | 1.243E-19 | 3.456E-26 | -- |
| Pd-107 | 1.809E-07 | 1.812E-07 | 1.815E-07 | 1.818E-07 | 1.829E-07 | 1.849E-07 | 1.938E-07 |
| Cd-107 | 2.287E-18 | -- | -- | -- | -- | -- | -- |
| Ag-108 | 9.599E-13 | 9.611E-13 | 9.624E-13 | 9.636E-13 | 9.681E-13 | 9.763E-13 | 1.012E-12 |
| Ag-108m | 1.079E-11 | 1.080E-11 | 1.081E-11 | 1.083E-11 | 1.088E-11 | 1.097E-11 | 1.137E-11 |
| Pd-109 | 6.975E-07 | 5.533E-23 | 1.022E-33 | -- | -- | -- | -- |
| Ag-109m | 1.202E-06 | 4.829E-07 | 4.695E-07 | 4.497E-07 | 3.954E-07 | 3.032E-07 | 1.067E-07 |
| Cd-109 | 5.042E-07 | 4.829E-07 | 4.695E-07 | 4.497E-07 | 3.954E-07 | 3.032E-07 | 1.067E-07 |
| Ag-110 | 5.306E-06 | 4.891E-06 | 4.635E-06 | 4.272E-06 | 3.348E-06 | 2.025E-06 | 2.798E-07 |
| Ag-110m | 3.990E-04 | 3.677E-04 | 3.485E-04 | 3.212E-04 | 2.518E-04 | 1.522E-04 | 2.103E-05 |
| Pd-111 | 3.747E-17 | -- | -- | -- | -- | -- | -- |
| Pd-111m | 5.142E-17 | -- | -- | -- | -- | -- | -- |
| Ag-111 | 2.964E-02 | 1.821E-03 | 2.839E-04 | 1.744E-05 | 4.053E-09 | 1.341E-16 | -- |
| Ag-111m | 5.380E-17 | -- | -- | -- | -- | -- | -- |
| Pd-112 | 1.338E-05 | 2.209E-16 | 1.433E-23 | 2.365E-34 | -- | -- | -- |
| Ag-112 | 1.585E-05 | 2.616E-16 | 1.697E-23 | 2.801E-34 | -- | -- | -- |
| Ag-113 | 1.053E-15 | -- | -- | -- | -- | -- | -- |
| Cd-113m | 1.426E-04 | 1.424E-04 | 1.422E-04 | 1.419E-04 | 1.411E-04 | 1.393E-04 | 1.327E-04 |
| In-113m | 3.833E-02 | 3.205E-02 | 2.846E-02 | 2.380E-02 | 1.393E-02 | 4.615E-03 | 5.944E-05 |
| Sn-113 | 3.831E-02 | 3.203E-02 | 2.845E-02 | 2.378E-02 | 1.392E-02 | 4.612E-03 | 5.940E-05 |
| In-114 | 4.471E-04 | 2.942E-04 | 2.228E-04 | 1.466E-04 | 4.184E-05 | 3.163E-06 | 1.200E-10 |
| In-114m | 4.671E-04 | 3.075E-04 | 2.327E-04 | 1.532E-04 | 4.372E-05 | 3.306E-06 | 1.255E-10 |
| Cd-115 | 1.773E-03 | 1.571E-07 | 3.134E-10 | 2.777E-14 | 1.932E-26 | -- | -- |
| Cd-115m | 3.327E-03 | 2.090E-03 | 1.535E-03 | 9.640E-04 | 2.395E-04 | 1.360E-05 | 1.671E-10 |
| In-115 | 1.642E-16 | 1.648E-16 | 1.652E-16 | 1.656E-16 | 1.668E-16 | 1.687E-16 | 1.768E-16 |
| In-115m | 1.925E-03 | 3.172E-07 | 1.081E-07 | 6.767E-08 | 1.681E-08 | 9.550E-10 | 1.172E-14 |
| Cd-117 | 4.503E-30 | -- | -- | -- | -- | -- | -- |
| Cd-117m | 8.346E-24 | -- | -- | -- | -- | -- | -- |
| In-117 | 1.108E-23 | -- | -- | -- | -- | -- | -- |
| In-117m | 8.553E-24 | -- | -- | -- | -- | -- | -- |
| Sn-117m | 1.899E-01 | 4.309E-02 | 1.604E-02 | 3.640E-03 | 4.257E-05 | 4.487E-09 | 9.338E-25 |
| Sn-119m | 3.045E-01 | 2.802E-01 | 2.652E-01 | 2.441E-01 | 1.904E-01 | 1.140E-01 | 1.513E-02 |
| Sn-121 | 4.215E-04 | 3.452E-12 | 1.403E-17 | 1.149E-25 | -- | -- | -- |
| Sn-121m | 3.969E-05 | 3.971E-05 | 3.975E-05 | 3.978E-05 | 3.988E-05 | 4.004E-05 | 4.082E-05 |
| Sb-122 | 2.044E-04 | 9.265E-08 | 5.470E-10 | 2.479E-13 | 2.308E-23 | -- | -- |
| Sn-123 | 2.794E-02 | 2.382E-02 | 2.143E-02 | 1.828E-02 | 1.134E-02 | 4.245E-03 | 8.826E-05 |
| Te-123 | 6.233E-19 | 6.330E-19 | 6.391E-19 | 6.468E-19 | 6.648E-19 | 6.861E-19 | 7.264E-19 |

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| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Te-123m | 1.670E-06 | 1.406E-06 | 1.254E-06 | 1.056E-06 | 6.307E-07 | 2.181E-07 | 3.324E-09 |
| Sb-124 | 4.745E-04 | 3.365E-04 | 2.677E-04 | 1.898E-04 | 6.775E-05 | 8.117E-06 | 1.891E-09 |
| Sn-125 | 7.082E-02 | 8.205E-03 | 1.951E-03 | 2.261E-04 | 3.516E-07 | 5.836E-13 | -- |
| Sb-125 | 1.566E-01 | 1.543E-01 | 1.525E-01 | 1.496E-01 | 1.415E-01 | 1.261E-01 | 8.010E-02 |
| Te-125m | 3.457E-02 | 3.497E-02 | 3.509E-02 | 3.503E-02 | 3.402E-02 | 3.071E-02 | 1.955E-02 |
| Sn-126 | 3.978E-06 | 3.985E-06 | 3.992E-06 | 3.998E-06 | 4.022E-06 | 4.067E-06 | 4.263E-06 |
| Sb-126 | 1.508E-03 | 2.827E-04 | 9.291E-05 | 1.784E-05 | 6.763E-07 | 5.695E-07 | 5.967E-07 |
| Sb-126m | 3.978E-06 | 3.985E-06 | 3.992E-06 | 3.998E-06 | 4.022E-06 | 4.067E-06 | 4.263E-06 |
| Sn-127 | 9.205E-36 | -- | -- | -- | -- | -- | -- |
| Sb-127 | 6.084E-02 | 2.747E-04 | 7.509E-06 | 3.391E-08 | 3.126E-15 | 1.031E-29 | -- |
| Te-127 | 1.055E-01 | 3.959E-02 | 3.470E-02 | 2.872E-02 | 1.630E-02 | 5.075E-03 | 5.111E-05 |
| Te-127m | 4.821E-02 | 4.015E-02 | 3.542E-02 | 2.932E-02 | 1.664E-02 | 5.182E-03 | 5.218E-05 |
| Xe-127 | 1.161E-09 | 6.568E-10 | 4.496E-10 | 2.544E-10 | 4.614E-11 | 1.372E-12 | 1.314E-18 |
| Sb-128 | 3.563E-10 | -- | -- | -- | -- | -- | -- |
| Sb-129 | 2.859E-17 | -- | -- | -- | -- | -- | -- |
| Te-129 | 1.181E-01 | 6.369E-02 | 4.223E-02 | 2.278E-02 | 3.580E-03 | 7.925E-05 | 2.369E-11 |
| Te-129m | 1.814E-01 | 9.788E-02 | 6.488E-02 | 3.500E-02 | 5.500E-03 | 1.217E-04 | 3.639E-11 |
| I-129 | 2.350E-07 | 2.359E-07 | 2.365E-07 | 2.371E-07 | 2.387E-07 | 2.414E-07 | 2.530E-07 |
| Xe-129m | 9.027E-08 | 6.723E-09 | 1.190E-09 | 8.861E-11 | 3.661E-14 | 3.960E-21 | |
| I-130 | 2.580E-08 | 7.554E-26 | 1.508E-37 | -- | -- | -- | -- |
| Te-131 | 6.920E-04 | 4.132E-11 | 6.316E-16 | 3.770E-23 | -- | -- | -- |
| Te-131m | 3.074E-03 | 1.836E-10 | 2.806E-15 | 1.675E-22 | -- | -- | -- |
| I-131 | 2.658E+00 | 2.005E-01 | 3.582E-02 | 2.702E-03 | 1.161E-06 | 1.362E-13 | -- |
| Xe-131m | 5.775E-02 | 1.617E-02 | 5.671E-03 | 1.071E-03 | 6.001E-06 | 1.253E-10 | 4.329E-29 |
| Te-132 | 1.085E+00 | 1.833E-03 | 2.607E-05 | 4.409E-08 | 2.145E-16 | 1.661E-33 | -- |
| I-132 | 1.117E+00 | 1.892E-03 | 2.686E-05 | 4.549E-08 | 2.210E-16 | 1.711E-33 | -- |
| Cs-132 | 4.986E-05 | 2.012E-06 | 2.368E-07 | 9.558E-09 | 6.287E-13 | 1.549E-21 | -- |
| I-133 | 4.919E-03 | 1.872E-13 | 2.120E-20 | 8.066E-31 | -- | -- | -- |
| Xe-133 | 4.651E+00 | 8.881E-02 | 6.332E-03 | 1.204E-04 | 8.281E-10 | 1.957E-20 | -- |
| Xe-133m | 2.919E-02 | 2.204E-06 | 3.932E-09 | 2.960E-13 | 1.264E-25 | -- | -- |
| Cs-134 | 1.531E-01 | 1.492E-01 | 1.468E-01 | 1.430E-01 | 1.324E-01 | 1.129E-01 | 6.043E-02 |
| Cs-134m | 4.124E-27 | -- | -- | -- | -- | -- | -- |
| I-135 | 1.567E-10 | -- | -- | -- | -- | -- | -- |
| Xe-135 | 5.268E-07 | -- | -- | -- | -- | -- | -- |
| Xe-135m | 2.509E-11 | -- | -- | -- | -- | -- | -- |
| Cs-135 | 1.192E-05 | 1.194E-05 | 1.196E-05 | 1.198E-05 | 1.205E-05 | 1.219E-05 | 1.277E-05 |
| Ba-135m | 2.349E-08 | 6.578E-16 | 6.070E-21 | 1.701E-28 | | -- | -- |
| Cs-136 | 5.062E-02 | 1.037E-02 | 3.605E-03 | 7.386E-04 | 6.355E-06 | 3.561E-10 | 6.137E-27 |
| Ba-136m | 8.342E-03 | 1.708E-03 | 5.941E-04 | 1.217E-04 | 1.047E-06 | 5.869E-11 | 1.011E-27 |
| Cs-137 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 | 1.000E+00 |
| Ba-137m | 9.459E-01 | 9.458E-01 | 9.457E-01 | 9.456E-01 | 9.461E-01 | 9.455E-01 | 9.465E-01 |
| La-138 | 2.100E-15 | 2.103E-15 | 2.107E-15 | 2.111E-15 | 2.123E-15 | 2.147E-15 | 2.250E-15 |
| Ba-140 | 7.626E+00 | 1.503E+00 | 5.092E-01 | 1.003E-01 | 7.687E-04 | 3.390E-08 | 2.272E-25 |
| La-140 | 8.748E+00 | 1.730E+00 | 5.864E-01 | 1.155E-01 | 8.845E-04 | 3.901E-08 | 2.615E-25 |
| Ce-141 | 1.011E+01 | 5.342E+00 | 3.493E+00 | 1.846E+00 | 2.726E-01 | 5.310E-03 | 9.583E-10 |
| Ce-142 | 2.854E-10 | 2.858E-10 | 2.863E-10 | 2.868E-10 | 2.885E-10 | 2.918E-10 | 3.058E-10 |
| Pr-142 | 8.909E-06 | 4.176E-17 | 1.171E-24 | 5.487E-36 | -- | -- | -- |

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| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Ce-143 | 8.098E-02 | 2.194E-08 | 9.194E-13 | 2.490E-19 | -- | -- | -- |
| Pr-143 | 8.136E+00 | 1.762E+00 | 6.351E-01 | 1.373E-01 | 1.390E-03 | 1.088E-07 | 7.015E-24 |
| Ce-144 | 1.064E+01 | 9.915E+00 | 9.457E+00 | 8.801E+00 | 7.112E+00 | 4.578E+00 | 8.081E-01 |
| Pr-144 | 1.065E+01 | 9.915E+00 | 9.457E+00 | 8.801E+00 | 7.113E+00 | 4.578E+00 | 8.082E-01 |
| Pr-144m | 1.277E-01 | 1.190E-01 | 1.135E-01 | 1.057E-01 | 8.535E-02 | 5.494E-02 | 9.701E-03 |
| Nd-144 | 9.602E-15 | 9.898E-15 | 1.008E-14 | 1.035E-14 | 1.106E-14 | 1.215E-14 | 1.422E-14 |
| Pr-145 | 6.908E-12 | -- | -- | -- | -- | -- | -- |
| Pm-145 | 6.057E-07 | 6.258E-07 | 6.390E-07 | 6.571E-07 | 7.057E-07 | 7.798E-07 | 8.921E-07 |
| Sm-145 | 6.768E-06 | 6.377E-06 | 6.132E-06 | 5.779E-06 | 4.839E-06 | 3.355E-06 | 7.931E-07 |
| Pm-146 | 3.154E-06 | 3.127E-06 | 3.110E-06 | 3.084E-06 | 3.008E-06 | 2.853E-06 | 2.324E-06 |
| Sm-146 | 3.471E-13 | 3.486E-13 | 3.499E-13 | 3.514E-13 | 3.563E-13 | 3.657E-13 | 4.027E-13 |
| Nd-147 | 2.580E+00 | 3.943E-01 | 1.128E-01 | 1.724E-02 | 6.163E-05 | 5.657E-10 | 7.774E-30 |
| Pm-147 | 2.644E+00 | 2.617E+00 | 2.587E+00 | 2.537E+00 | 2.391E+00 | 2.115E+00 | 1.306E+00 |
| Sm-147 | 4.005E-11 | 4.152E-11 | 4.252E-11 | 4.395E-11 | 4.816E-11 | 5.613E-11 | 8.114E-11 |
| Pm-148 | 1.248E-01 | 6.840E-03 | 3.329E-03 | 1.907E-03 | 4.227E-04 | 1.907E-05 | 9.456E-11 |
| Pm-148m | 1.288E-01 | 7.794E-02 | 5.581E-02 | 3.379E-02 | 7.504E-03 | 3.387E-04 | 1.679E-09 |
| Sm-148 | 1.701E-16 | 1.713E-16 | 1.719E-16 | 1.726E-16 | 1.739E-16 | 1.760E-16 | 1.844E-16 |
| Pm-149 | 1.070E-01 | 8.856E-06 | 1.683E-08 | 1.392E-12 | 7.890E-25 | -- | -- |
| Sm-149 | 3.685E-17 | 3.697E-17 | 3.704E-17 | 3.710E-17 | 3.732E-17 | 3.774E-17 | 3.956E-17 |
| Pm-150 | 3.202E-30 | -- | -- | -- | -- | -- | -- |
| Eu-150 | 2.212E-10 | 2.213E-10 | 2.214E-10 | 2.214E-10 | 2.216E-10 | 2.220E-10 | 2.238E-10 |
| Pm-151 | 2.624E-03 | 6.095E-11 | 4.964E-16 | 1.153E-23 | -- | -- | -- |
| Sm-151 | 1.309E-02 | 1.311E-02 | 1.313E-02 | 1.314E-02 | 1.319E-02 | 1.329E-02 | 1.372E-02 |
| Eu-152 | 4.507E-04 | 4.496E-04 | 4.491E-04 | 4.480E-04 | 4.451E-04 | 4.386E-04 | 4.151E-04 |
| Eu-152m | 4.637E-11 | -- | -- | -- | -- | -- | -- |
| Gd-152 | 5.525E-17 | 5.541E-17 | 5.555E-17 | 5.571E-17 | 5.624E-17 | 5.728E-17 | 6.159E-17 |
| Sm-153 | 1.728E-02 | 3.948E-07 | 3.182E-10 | 7.273E-15 | 8.695E-29 | -- | -- |
| Gd-153 | 5.635E-05 | 5.180E-05 | 4.899E-05 | 4.503E-05 | 3.501E-05 | 2.083E-05 | 2.694E-06 |
| Eu-154 | 6.954E-03 | 6.920E-03 | 6.901E-03 | 6.866E-03 | 6.772E-03 | 6.574E-03 | 5.864E-03 |
| Eu-155 | 2.355E-02 | 2.332E-02 | 2.317E-02 | 2.295E-02 | 2.230E-02 | 2.102E-02 | 1.665E-02 |
| Sm-156 | 7.586E-10 | -- | -- | -- | -- | -- | -- |
| Eu-156 | 4.914E-02 | 1.251E-02 | 5.031E-03 | 1.281E-03 | 2.119E-05 | 4.556E-09 | 1.576E-23 |
| Eu-157 | 3.582E-07 | 1.975E-21 | 6.166E-31 | -- | -- | -- | -- |
| Gd-159 | 7.244E-07 | 1.614E-18 | 2.755E-26 | 6.026E-38 | -- | -- | -- |
| Tb-160 | 1.399E-04 | 1.051E-04 | 8.693E-05 | 6.531E-05 | 2.772E-05 | 4.747E-06 | 4.523E-09 |
| Tb-161 | 2.786E-04 | 1.383E-05 | 1.869E-06 | 9.274E-08 | 1.135E-11 | 1.003E-19 | -- |
| Dy-165 | 8.858E-36 | -- | -- | -- | -- | -- | -- |
| Dy-166 | 1.998E-06 | 4.385E-09 | 7.409E-11 | 1.627E-13 | 1.721E-21 | 6.529E-38 | -- |
| Ho-166 | 2.966E-06 | 6.535E-09 | 1.104E-10 | 2.423E-13 | 2.564E-21 | 9.922E-38 | -- |
| Ho-166m | 9.543E-11 | 9.559E-11 | 9.576E-11 | 9.592E-11 | 9.641E-11 | 9.749E-11 | 1.021E-10 |
| Er-169 | 9.061E-10 | 9.932E-11 | 2.278E-11 | 2.497E-12 | 3.311E-15 | 3.910E-21 | -- |
| Tm-170 | 3.997E-11 | 3.406E-11 | 3.063E-11 | 2.610E-11 | 1.617E-11 | 6.022E-12 | 1.230E-13 |
| Er-171 | 9.416E-27 | -- | -- | -- | -- | -- | -- |
| Tm-171 | 5.486E-14 | 5.335E-14 | 5.239E-14 | 5.095E-14 | 4.689E-14 | 3.949E-14 | 2.010E-14 |
| Er-172 | 5.723E-22 | 2.164E-26 | 2.167E-26 | 8.190E-31 | -- | -- | -- |
| Tm-172 | 5.853E-13 | 2.292E-16 | 9.957E-19 | 3.898E-22 | 2.343E-32 | -- | -- |
| Tm-173 | 2.800E-21 | -- | -- | -- | -- | -- | -- |

**ATTACHMENT A**
**REACTOR SOURCE TERMS**


| Nuclide | 10 d | 40 d | 60 d | 90 d | 180 d | 1 y | 3 y |
|---|---|---|---|---|---|---|---|
| Yb-175 | 7.305E-18 | 1.674E-19 | 1.072E-19 | 8.580E-20 | 2.950E-26 | -- | -- |
| Hf-175 | 6.529E-04 | 4.859E-04 | 3.992E-04 | 2.972E-04 | 1.226E-04 | 1.980E-05 | 1.499E-08 |
| Lu-176 | 3.213E-16 | 3.219E-16 | 3.224E-16 | 3.230E-16 | 3.249E-16 | 3.285E-16 | 3.443E-16 |
| Lu-176m | 6.242E-25 | -- | -- | -- | -- | -- | -- |
| Lu-177 | 6.320E-07 | 2.928E-08 | 4.329E-09 | 7.877E-10 | 4.200E-10 | 1.855E-10 | 7.413E-12 |
| Lu-177m | 3.863E-09 | 3.384E-09 | 3.099E-09 | 2.715E-09 | 1.826E-09 | 8.066E-10 | 3.223E-11 |
| Hf-180m | 1.926E-17 | -- | -- | -- | -- | -- | -- |
| Ta-180 | 4.497E-16 | 4.505E-16 | 4.513E-16 | 4.520E-16 | 4.547E-16 | 4.599E-16 | 4.820E-16 |
| Hf-181 | 7.160E-03 | 4.392E-03 | 3.172E-03 | 1.946E-03 | 4.494E-04 | 2.199E-05 | 1.499E-10 |
| W-181 | 1.973E-04 | 1.664E-04 | 1.487E-04 | 1.254E-04 | 7.543E-05 | 2.644E-05 | 4.246E-07 |
| Hf-182 | 7.605E-13 | 7.618E-13 | 7.631E-13 | 7.644E-13 | 7.689E-13 | 7.776E-13 | 8.150E-13 |
| Ta-182 | 9.239E-01 | 7.726E-01 | 6.861E-01 | 5.736E-01 | 3.354E-01 | 1.111E-01 | 1.424E-03 |
| Ta-183 | 5.229E-02 | 8.881E-04 | 5.870E-05 | 9.966E-07 | 4.885E-12 | 5.745E-23 | -- |
| W-185 | 4.343E-03 | 3.298E-03 | 2.747E-03 | 2.086E-03 | 9.145E-04 | 1.673E-04 | 2.083E-07 |
| Re-186 | 7.163E-05 | 2.914E-07 | 7.433E-09 | 3.025E-11 | 2.039E-18 | 3.536E-33 | -- |
| W-187 | 1.276E-04 | 1.092E-13 | 9.839E-20 | 8.412E-29 | -- | -- | -- |
| Re-187 | 6.141E-12 | 6.152E-12 | 6.162E-12 | 6.173E-12 | 6.210E-12 | 6.279E-12 | 6.581E-12 |
| W-188 | 2.199E-05 | 1.632E-05 | 1.339E-05 | 9.940E-06 | 4.070E-06 | 6.470E-07 | 4.598E-10 |
| Re-188 | 2.243E-05 | 1.649E-05 | 1.353E-05 | 1.004E-05 | 4.112E-06 | 6.536E-07 | 4.646E-10 |
| Re-189 | 1.341E-12 | 1.617E-21 | 1.834E-27 | 2.213E-36 | -- | -- | -- |
| Os-191 | 1.198E-10 | 3.112E-11 | 1.267E-11 | 3.292E-12 | 5.772E-14 | 1.401E-17 | 7.798E-32 |
| Os-191m | 3.834E-16 | 8.165E-33 | -- | -- | -- | -- | -- |
| Ir-192 | 7.356E-12 | 5.564E-12 | 4.621E-12 | 3.496E-12 | 1.514E-12 | 2.701E-13 | 3.101E-16 |
| Ir-192m | 7.783E-18 | 7.794E-18 | 7.806E-18 | 7.817E-18 | 7.859E-18 | 7.935E-18 | 8.269E-18 |
| Os-193 | 5.082E-19 | 5.190E-26 | 1.134E-30 | 1.205E-37 | -- | -- | -- |
| Pt-193 | 4.964E-18 | 4.977E-18 | 4.986E-18 | 4.993E-18 | 5.021E-18 | 5.074E-18 | 5.304E-18 |
| Pt-193m | 2.523E-16 | 1.942E-18 | 2.289E-19 | 1.395E-19 | 7.019E-26 | 7.365E-39 | -- |
| Os-194 | 1.432E-23 | 1.435E-23 | 1.437E-23 | 1.440E-23 | 1.448E-23 | 1.465E-23 | 1.535E-23 |
| Ir-194 | 9.475E-19 | 1.435E-23 | 1.437E-23 | 1.440E-23 | 1.448E-23 | 1.465E-23 | 1.535E-23 |
| Pt-195m | 8.663E-21 | 8.678E-21 | 8.693E-21 | 8.707E-21 | 2.062E-29 | -- | -- |
| Pb-204 | 3.043E-18 | 3.048E-18 | 3.053E-18 | 3.059E-18 | 3.077E-18 | 3.112E-18 | 3.261E-18 |
| Pb-205 | 2.853E-12 | 2.858E-12 | 2.862E-12 | 2.867E-12 | 2.885E-12 | 2.917E-12 | 3.057E-12 |
| Tl-206 | 1.646E-18 | 1.648E-18 | 1.651E-18 | 1.654E-18 | 1.664E-18 | 1.683E-18 | 1.763E-18 |
| Bi-208 | 1.911E-18 | 1.914E-18 | 1.918E-18 | 1.921E-18 | 1.932E-18 | 1.954E-18 | 2.048E-18 |
| Pb-209 | 8.477E-29 | -- | -- | -- | -- | -- | -- |
| Bi-210 | 1.481E-12 | 2.341E-14 | 1.475E-15 | 2.342E-17 | 9.247E-23 | 6.975E-34 | -- |
| Bi-210m | 1.652E-18 | 1.655E-18 | 1.658E-18 | 1.661E-18 | 1.671E-18 | 1.689E-18 | 1.771E-18 |
| Po-210 | 4.714E-12 | 4.111E-12 | 3.726E-12 | 3.212E-12 | 2.059E-12 | 8.234E-13 | 2.226E-14 |

## ATTACHMENT B
## ELEMENTAL EXPOSURE FRACTIONS

These fractions were used to calculate ratios of fission and activation products available for inhalation intakes from reactor source terms.

Table B-1. Elemental exposure fractions.

| Element | Exposure fraction | Element | Exposure fraction | Element | Exposure fraction |
|---|---|---|---|---|---|
| Ac | 0.01 | Gd | 0.01 | Ra | 0.01 |
| Ag | 0.01 | Ge | 0.01 | Rb | 0.01 |
| Al | 0.01 | H | 1 | Re | 0.01 |
| Am | 0.01 | Hf | 0.01 | Rh | 0.01 |
| As | 0.05 | Hg | 0.01 | Ru | 0.01 |
| At | 0.05 | Ho | 0.01 | S | 0.05 |
| Au | 0.01 | I | 0.05 | Sb | 0.01 |
| Ba | 0.01 | In | 0.01 | Sc | 0.01 |
| Be | 0.01 | Ir | 0.01 | Se | 0.01 |
| Bi | 0.01 | K | 0.05 | Si | 0.01 |
| Br | 0.05 | La | 0.01 | Sm | 0.01 |
| C | 0.01 | Lu | 0.01 | Sn | 0.01 |
| Ca | 0.01 | Mg | 0.01 | Sr | 0.01 |
| Cd | 0.01 | Mn | 0.01 | Ta | 0.01 |
| Ce | 0.01 | Mo | 0.01 | Tb | 0.01 |
| Cf | 0.01 | Na | 0.05 | Tc | 0.01 |
| Cl | 1 | Nb | 0.01 | Te | 0.01 |
| Cm | 0.01 | Nd | 0.01 | Th | 0.01 |
| Co | 0.01 | Ni | 0.01 | Ti | 0.01 |
| Cr | 0.01 | Np | 0.01 | Tl | 0.01 |
| Cs | 0.01 | Os | 0.01 | Tm | 0.01 |
| Cu | 0.01 | P | 0.05 | U | 0.01 |
| Dy | 0.01 | Pa | 0.01 | V | 0.01 |
| Er | 0.01 | Pb | 0.01 | W | 0.01 |
| Es | 0.01 | Pd | 0.01 | Y | 0.01 |
| Eu | 0.01 | Pm | 0.01 | Yb | 0.01 |
| F | 1 | Po | 0.01 | Zn | 0.01 |
| Fe | 0.01 | Pr | 0.01 | Zr | 0.01 |
| Fr | 0.01 | Pt | 0.01 | | |
| Ga | 0.01 | Pu | 0.01 | | |

## ATTACHMENT C
## RADIONUCLIDES FROM REACTOR SOURCE TERMS NOT INCLUDED IN DOSE ANALYSES

The following radionuclides were identified in some of the reactor source terms; no internal dose conversion factors were readily available for these radionuclides, but they were judged to be insignificant contributors to internal dose and were not included in the internal dose evaluations. The noble gases contribute primarily to external dose. For radionuclides that are not noble gases, most of the half-lives are either very short or very long, which indicates that their relative doses would not significantly influence the results. In addition, the relative activities of the radionuclides that are sufficiently long-lived (greater than 1 hour) to have some dosimetric significance in the source terms were at least an order of magnitude less in activity than the radionuclides chosen to represent the source term.

Table C-1. Radionuclides from reactor source terms not included in analyses.

| Radionuclide | Half-life if particulate[a] | Radionuclide | Half-life if particulate[a] |
|---|---|---|---|
| Ag-108 | 2.42 min | Nb-92 | 13 h, >350 y, <1 h |
| Ag-109m | 39.2 s | Nb-97m | 1min |
| Ag-110 | 24.4 s | Nd-144 | 2.4e15y |
| Ag-111m | 74 s | Pd-111 | 22min |
| Ag-113 | 5.3 h/1.2 min | Pd-111m | 5.5h |
| Al-28 | 2.31 min | Pd-112 | 21h |
| Ar-37 | Noble gas | Pr-144m | 7.2min |
| Ar-39 | Noble gas | Sb-128 | 10.8min |
| Ar-42 | Noble gas | Se-77m | 17.5s |
| Ba-136m | 0.32 s | Sm-148 | >2e14y |
| Ba-137m | 2.6 min | Sm-149 | >1e15y |
| Ce-142 | >5e16 y | V-50 | 6e15y |
| Ge-73m | 0.53 s | Xe-127 | Noble gas |
| In-114 | 72 s | Xe-129m | Noble gas |
| Kr-79 | Noble gas | Xe-131m | Noble gas |
| Kr-81 | Noble gas | Xe-133 | Noble gas |
| Kr-83m | Noble gas | Xe-133m | Noble gas |
| Kr-85 | Noble gas | Xe-135 | Noble gas |
| Kr-85m | Noble gas | Xe-135m | Noble gas |
| Kr-88 | Noble gas | Y-89m | 16.1s |

a. The abbreviations are seconds (s), minutes (min), hours (h) and years (y).

# ATTACHMENT D
# RELATIVE EXPOSURE ACTIVITY FRACTIONS

Table D-1. Relative exposure activity fractions (radionuclides that contribute more than 1% dose to at least one organ or effective dose).

| | ATR | | | | FFTF | | | | N Reactor 2 | | | | TRIGA SS PWR | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Radionuclide**[a] | **10 d** | **40 d** | **180 d** | **1 y** | **10 d** | **40 d** | **180 d** | **1 y** | **10 d** | **40 d** | **180 d** | **1 y** | **10 d** | **40 d** | **180 d** | **1 y** |
| **Ba-140** | 2.9E-02 | 4.2E-02 | -- | -- | 1.9E-02 | 1.8E-02 | -- | -- | 2.6E-02 | 2.9E-02 | -- | -- | 2.4E-02 | 1.6E-02 | -- | -- |
| Cd-113m | -- | -- | -- | -- | 1.1E-06 | 5.4E-06 | 1.9E-05 | 3.6E-05 | -- | -- | 4.8E-06 | 1.3E-05 | -- | -- | 4.3E-06 | 7.4E-06 |
| Cd-115m | -- | -- | -- | -- | 5.6E-05 | 1.7E-04 | -- | -- | 4.0E-05 | 1.4E-04 | 8.6E-05 | -- | -- | -- | -- | -- |
| **Ce-141** | 2.4E-02 | 9.2E-02 | 3.3E-02 | -- | 2.8E-02 | 7.1E-02 | 1.3E-02 | 2.2E-01 | 3.1E-02 | 9.2E-02 | 2.5E-02 | -- | 3.2E-02 | 5.8E-02 | -- | -- |
| **Ce-144** | 4.2E-03 | 2.9E-02 | 1.4E-01 | 2.9E-01 | 1.6E-02 | 7.1E-02 | 1.8E-01 | -- | 7.9E-03 | 4.2E-02 | 1.6E-01 | 2.9E-01 | 3.4E-02 | 1.1E-01 | 2.1E-01 | 2.4E-01 |
| Co-58 | -- | -- | -- | -- | 4.7E-03 | 1.7E-02 | 1.5E-02 | 4.8E-03 | -- | -- | -- | -- | -- | 3.8E-03 | 2.7E-03 | -- |
| Co-60 | -- | -- | -- | -- | -- | -- | 5.5E-04 | 1.0E-03 | -- | -- | -- | -- | 2.9E-03 | 9.8E-03 | 2.6E-02 | 4.4E-02 |
| **Cs-134** | 8.1E-05 | 5.8E-04 | 3.6E-03 | 9.9E-03 | 5.8E-04 | 2.7E-03 | 8.4E-03 | 1.4E-02 | -- | 2.3E-04 | 1.1E-03 | 2.6E-03 | 4.9E-04 | 1.6E-03 | 4.0E-03 | 6.0E-03 |
| Cs-136 | -- | -- | -- | -- | 7.3E-04 | 7.1E-04 | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- |
| **Cs-137** | 1.3E-04 | 9.6E-04 | 6.7E-03 | 2.1E-02 | 1.2E-03 | 5.7E-03 | 2.0E-02 | 3.9E-02 | 2.9E-04 | 1.6E-03 | 8.6E-03 | 2.4E-02 | 3.2E-03 | 1.1E-02 | 3.0E-02 | 5.3E-02 |
| Eu-154 | 4.1E-06 | 3.0E-05 | 2.0E-04 | 6.4E-04 | 3.5E-05 | 1.6E-04 | 5.7E-04 | 1.1E-03 | -- | -- | 5.6E-05 | 1.5E-04 | 2.2E-05 | 7.5E-05 | 2.0E-04 | 3.5E-04 |
| Eu-155 | -- | -- | 1.4E-04 | 4.2E-04 | 2.0E-04 | 9.4E-04 | 3.2E-03 | 5.8E-03 | -- | 8.8E-05 | 4.4E-04 | 1.2E-03 | 7.5E-05 | 2.5E-04 | 6.7E-04 | 1.1E-03 |
| **Fe-55** | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | 6.4E-03 | 2.2E-02 | 5.5E-02 | 8.5E-02 |
| **I-131** | 5.5E-01 | 3.1E-01 | 1.2E-05 | -- | 5.4E-01 | 1.9E-01 | -- | -- | 5.3E-01 | 2.3E-01 | 6.9E-06 | -- | 4.2E-01 | 1.1E-01 | -- | -- |
| I-132 | 2.4E-01 | | -- | -- | 2.0E-01 | 2.1E-02 | -- | -- | 2.2E-01 | -- | -- | -- | 1.8E-01 | -- | -- | -- |
| **La-140** | 3.3E-02 | 4.8E-02 | -- | -- | 2.2E-02 | 4.5E-03 | -- | -- | 3.0E-02 | 3.4E-02 | -- | -- | 2.8E-02 | 1.9E-02 | -- | -- |
| Mn-54 | -- | -- | -- | -- | -- | -- | 1.2E-02 | 1.5E-02 | -- | -- | -- | -- | -- | 2.3E-03 | 4.7E-03 | 5.6E-03 |
| Mo-99 | 4.4E-03 | -- | -- | -- | 3.1E-03 | -- | -- | -- | 3.7E-03 | -- | -- | -- | 3.2E-03 | -- | -- | -- |
| **Nb-95** | 8.1E-03 | 8.1E-02 | 2.6E-01 | 1.3E-01 | 3.0E-02 | 1.3E-01 | 1.4E-01 | 4.2E-02 | 2.1E-02 | 1.3E-01 | 2.5E-01 | 1.1E-01 | 5.0E-02 | 1.5E-01 | 1.4E-01 | 3.6E-02 |
| Nd-147 | 9.4E-03 | 1.1E-02 | -- | -- | -- | -- | -- | -- | 9.1E-03 | -- | -- | -- | 8.2E-03 | -- | -- | -- |
| P-32 | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | 1.2E-03 | -- | -- | -- |
| **Pm-147** | 3.3E-04 | 3.0E-03 | 2.0E-02 | 5.7E-02 | 3.3E-03 | 1.6E-02 | 5.1E-02 | 8.8E-02 | 1.0E-03 | 6.2E-03 | 3.0E-02 | 7.5E-02 | 8.4E-03 | 2.8E-02 | 7.2E-02 | 1.1E-01 |
| Pm-148m | -- | -- | -- | -- | 1.6E-03 | 4.7E-03 | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- |
| Pr-143 | 3.0E-02 | 4.7E-02 | -- | -- | 1.8E-02 | 1.9E-02 | -- | -- | 2.6E-02 | 3.3E-02 | -- | -- | 2.6E-02 | 1.9E-02 | -- | -- |
| Pr-144 | | 2.9E-02 | 1.4E-01 | 2.9E-01 | -- | 7.1E-02 | 1.8E-01 | 2.2E-01 | -- | 4.2E-02 | 1.6E-01 | 2.9E-01 | 3.4E-02 | 1.1E-01 | 2.1E-01 | 2.4E-01 |
| **Ru-103** | 1.1E-02 | 5.0E-02 | 3.0E-02 | -- | 3.6E-02 | 1.0E-01 | 3.0E-02 | -- | 2.2E-02 | 7.4E-02 | 3.3E-02 | | 1.8E-02 | 3.7E-02 | 8.7E-03 | -- |
| **Ru-106** | 2.5E-04 | 1.8E-03 | 9.5E-03 | 2.2E-02 | 1.6E-02 | 7.3E-02 | 2.0E-01 | 2.8E-01 | 1.7E-03 | 9.1E-03 | 3.7E-02 | 7.4E-02 | 2.9E-03 | 9.3E-03 | 2.0E-02 | 2.6E-02 |
| Sb-125 | -- | -- | -- | -- | -- | 1.8E-03 | 5.7E-03 | 9.9E-03 | -- | -- | 1.9E-03 | 4.7E-03 | -- | 1.7E-03 | 4.3E-03 | 6.7E-03 |
| Sm-151 | -- | -- | -- | -- | 4.7E-05 | 2.3E-04 | 8.0E-04 | 1.6E-03 | -- | -- | 1.5E-04 | 4.4E-04 | -- | 1.4E-04 | 4.0E-04 | 7.1E-04 |
| **Sr-89** | 1.5E-02 | 7.5E-02 | 7.7E-02 | 2.0E-02 | 1.0E-02 | 3.3E-02 | 1.7E-02 | | 2.0E-02 | 7.4E-02 | 5.7E-02 | 1.3E-02 | 2.8E-02 | 6.3E-02 | 2.6E-02 | -- |
| **Sr-90** | 1.3E-04 | 9.3E-04 | 6.4E-03 | 2.1E-02 | 4.4E-04 | 2.1E-03 | 7.4E-03 | 1.4E-02 | 2.4E-04 | 1.3E-03 | 7.0E-03 | 2.0E-02 | 3.0E-03 | 1.0E-02 | 2.9E-02 | 5.1E-02 |
| Ta-182 | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | 2.9E-03 | 8.4E-03 | 1.0E-02 | 5.9E-03 |
| Te-125m | -- | -- | -- | -- | | | | 2.4E-03 | -- | -- | -- | -- | -- | -- | -- | -- |
| Te-127m | -- | 2.5E-04 | 7.3E-04 | -- | 4.5E-04 | 1.8E-03 | 2.6E-03 | 1.6E-03 | 1.5E-04 | 7.3E-04 | 1.6E-03 | 1.4E-03 | -- | -- | -- | -- |
| Te-129m | 4.1E-04 | 1.6E-03 | -- | -- | 1.1E-03 | 2.8E-03 | -- | -- | 7.6E-04 | 2.3E-03 | -- | -- | 5.7E-04 | -- | -- | -- |
| Te-132 | 4.7E-03 | -- | -- | -- | 3.9E-03 | -- | -- | -- | 4.2E-03 | -- | -- | -- | 3.4E-03 | -- | -- | -- |
| Y-90 | -- | -- | 6.4E-03 | 2.1E-02 | -- | -- | -- | 1.4E-02 | -- | -- | 7.0E-03 | 2.0E-02 | 3.0E-03 | 1.0E-02 | 2.9E-02 | 5.1E-02 |
| **Y-91** | 1.7E-02 | 8.8E-02 | 1.2E-01 | 4.2E-02 | 1.5E-02 | 4.9E-02 | 3.3E-02 | -- | 2.3E-02 | 9.3E-02 | 9.4E-02 | 3.0E-02 | 3.5E-02 | 8.3E-02 | 4.5E-02 | 8.9E-03 |
| **Zr-95** | 1.7E-02 | 9.3E-02 | 1.4E-01 | 6.2E-02 | 2.7E-02 | 9.4E-02 | 7.3E-02 | 1.9E-02 | 2.7E-02 | 1.1E-01 | 1.3E-01 | 5.0E-02 | 4.5E-02 | 1.1E-01 | 6.9E-02 | 1.7E-02 |

a. Radionuclides listed in **bold type** were used in the final default source terms.

## ATTACHMENT E
## SIMPLIFIED REACTOR SOURCE TERMS AND DEFAULT SOURCE TERM

Tables E-1 and E-2 list the reactor source terms used in the final fission and activation product intake analyses.

Table E-1. Simplified reactor source terms for intake calculations.

| | ATR | | | | FFTF | | | | N Reactor 2 | | | | TRIGA SS PWR | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Radionuclide** | **10 d** | **40 d** | **180 d** | **1 y** | **10 d** | **40 d** | **180 d** | **1 y** | **10 d** | **40 d** | **180 d** | **1 y** | **10 d** | **40 d** | **180 d** | **1 y** |
| Ba-140 | 4.0E-02 | 4.6E-02 | -- | -- | 2.6E-02 | 2.1E-02 | -- | -- | 3.5E-02 | 3.2E-02 | -- | -- | 3.3E-02 | 1.9E-02 | -- | -- |
| Ce-141 | 3.3E-02 | 1.0E-01 | 3.9E-02 | 0.0E+00 | 3.7E-02 | 8.2E-02 | 1.6E-02 | 3.1E-01 | 4.1E-02 | 1.0E-01 | 3.0E-02 | -- | 4.3E-02 | 6.9E-02 | -- | -- |
| Ce-144 | 5.9E-03 | 3.1E-02 | 1.7E-01 | 4.3E-01 | 2.1E-02 | 8.2E-02 | 2.3E-01 | -- | 1.1E-02 | 4.6E-02 | 1.9E-01 | 4.2E-01 | 4.6E-02 | 1.3E-01 | 3.0E-01 | 3.8E-01 |
| Cs-134 | 1.1E-04 | 6.4E-04 | 4.2E-03 | 1.4E-02 | 7.6E-04 | 3.1E-03 | 1.1E-02 | 1.9E-02 | -- | 2.5E-04 | 1.3E-03 | 3.8E-03 | 6.6E-04 | 1.9E-03 | 5.6E-03 | 9.4E-03 |
| Cs-137 | 1.8E-04 | 1.1E-03 | 7.9E-03 | 3.1E-02 | 1.6E-03 | 6.7E-03 | 2.6E-02 | 5.4E-02 | 3.9E-04 | 1.8E-03 | 1.0E-02 | 3.6E-02 | 4.3E-03 | 1.3E-02 | 4.3E-02 | 8.3E-02 |
| Eu-155 | -- | -- | 1.6E-04 | 6.1E-04 | 2.6E-04 | 1.1E-03 | 4.1E-03 | 8.1E-03 | | 9.6E-05 | 5.3E-04 | 1.7E-03 | 1.0E-04 | 3.0E-04 | 9.5E-04 | 1.8E-03 |
| Fe-55 | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | -- | 8.7E-03 | 2.6E-02 | 7.7E-02 | 1.3E-01 |
| I-131 | 7.8E-01 | 3.4E-01 | 1.5E-05 | -- | 7.0E-01 | 2.2E-01 | -- | -- | 7.1E-01 | 2.5E-01 | 8.3E-06 | -- | 5.7E-01 | 1.3E-01 | -- | -- |
| La-140 | 4.6E-02 | 5.3E-02 | -- | -- | 2.9E-02 | 5.2E-03 | -- | -- | 4.0E-02 | 3.6E-02 | -- | -- | 3.8E-02 | 2.2E-02 | -- | -- |
| Nb-95 | 1.1E-02 | 8.9E-02 | 3.1E-01 | 2.0E-01 | 3.9E-02 | 1.5E-01 | 1.8E-01 | 5.8E-02 | 2.8E-02 | 1.4E-01 | 3.0E-01 | 1.6E-01 | 6.7E-02 | 1.8E-01 | 1.9E-01 | 5.6E-02 |
| Pm-147 | 4.6E-04 | 3.3E-03 | 2.4E-02 | 8.3E-02 | 4.3E-03 | 1.8E-02 | 6.5E-02 | 1.2E-01 | 1.4E-03 | 6.7E-03 | 3.6E-02 | 1.1E-01 | 1.1E-02 | 3.4E-02 | 1.0E-01 | 1.8E-01 |
| Ru-103 | 1.6E-02 | 5.5E-02 | 3.5E-02 | -- | 4.7E-02 | 1.2E-01 | 3.9E-02 | -- | 3.0E-02 | 8.0E-02 | 4.0E-02 | -- | 2.5E-02 | 4.4E-02 | 1.2E-02 | -- |
| Ru-106 | 3.6E-04 | 1.9E-03 | 1.1E-02 | 3.2E-02 | 2.1E-02 | 8.5E-02 | 2.6E-01 | 3.8E-01 | 2.3E-03 | 9.9E-03 | 4.4E-02 | 1.1E-01 | 3.9E-03 | 1.1E-02 | 2.8E-02 | 4.0E-02 |
| Sr-89 | 2.1E-02 | 8.2E-02 | 9.0E-02 | 2.9E-02 | 1.4E-02 | 3.9E-02 | 2.2E-02 | -- | 2.7E-02 | 8.0E-02 | 6.9E-02 | 1.9E-02 | 3.8E-02 | 7.5E-02 | 3.6E-02 | -- |
| Sr-90 | 1.8E-04 | 1.0E-03 | 7.6E-03 | 3.0E-02 | 5.8E-04 | 2.4E-03 | 9.5E-03 | 2.0E-02 | 3.2E-04 | 1.5E-03 | 8.5E-03 | 2.9E-02 | 4.1E-03 | 1.2E-02 | 4.1E-02 | 7.9E-02 |
| Y-91 | 2.4E-02 | 9.6E-02 | 1.4E-01 | 6.2E-02 | 1.9E-02 | 5.7E-02 | 4.3E-02 | 0.0E+00 | 3.2E-02 | 1.0E-01 | 1.1E-01 | 4.4E-02 | 4.7E-02 | 9.9E-02 | 6.3E-02 | 1.4E-02 |
| Zr-95 | 2.4E-02 | 1.0E-01 | 1.7E-01 | 9.1E-02 | 3.6E-02 | 1.1E-01 | 9.4E-02 | 2.7E-02 | 3.7E-02 | 1.2E-01 | 1.6E-01 | 7.3E-02 | 6.1E-02 | 1.3E-01 | 9.7E-02 | 2.6E-02 |

Table E-2. Default source term for intake calculations.

| | Default | | | |
|---|---|---|---|---|
| **Radionuclide** | **10 d** | **40 d** | **180 d** | **1 y** |
| Ba-140 | 0.0334 | 0.0299 | -- | -- |
| Ce-141 | 0.0388 | 0.0887 | 0.0221 | 0.0818 |
| Ce-144 | 0.0209 | 0.0704 | 0.2191 | 0.3023 |
| Cs-134 | 0.0004 | 0.0014 | 0.0054 | 0.0119 |
| Cs-137 | 0.0016 | 0.0054 | 0.0208 | 0.0507 |
| Eu-155 | 0.0001 | 0.0004 | 0.0014 | 0.0031 |
| Fe-55 | 0.0022 | 0.0061 | 0.0172 | 0.0312 |
| I-131 | 0.6904 | 0.2366 | -- | -- |
| La-140 | 0.0383 | 0.0297 | -- | -- |
| Nb-95 | 0.0368 | 0.1374 | 0.2492 | 0.1170 |
| Pm-147 | 0.0044 | 0.0151 | 0.0546 | 0.1214 |
| Ru-103 | 0.0296 | 0.0739 | 0.0321 | -- |
| Ru-106 | 0.0071 | 0.0265 | 0.0844 | 0.1461 |
| Sr-89 | 0.0248 | 0.0693 | 0.0558 | 0.0119 |
| Sr-90 | 0.0013 | 0.0042 | 0.0157 | 0.0387 |
| Y-91 | 0.0304 | 0.0887 | 0.0911 | 0.0297 |
| Zr-95 | 0.0396 | 0.1162 | 0.1311 | 0.0543 |

## ATTACHMENT F
## SUMMARY OF IRFs, AND YIELD AND COUNTING ADJUSTMENT FACTORS
Page 1 of 2

Table F-1. Summary of IRFs, and yield and counting adjustment factors.

| Radionuclide Absorption type | Beta yield and counting adjustment factor | Gamma yield | Chronic 24-h urine IRF | |
|---|---|---|---|---|
| | | | 90 d | 2 y (730 d) |
| Ba-140 F | 1 | 1.76 | 2.42E-02 | 2.42E-02 |
| Cd-113m F | 1 | (a) | (a) | (a) |
| Cd-113m M | 1 | (a) | (a) | (a) |
| Cd-113m S | 1 | (a) | (a) | (a) |
| Cd-115m F | 1 | (a) | (a) | (a) |
| Cd-115m M | 1 | (a) | (a) | (a) |
| Cd-115m S | 1 | (a) | (a) | (a) |
| Ce-141 M | 1 | 0.48 | 2.67E-05 | 3.30E-05 |
| Ce-141 S | 1 | 0.48 | 5.04E-07 | 6.47E-07 |
| Ce-144 M | 1.5006 | 0.14 | 5.77E-05 | 3.23E-04 |
| Ce-144 S | 1.5006 | 0.14 | 1.12E-06 | 9.10E-06 |
| Co-58 M | 0.15 | 1 | 5.60E-02 | 5.90E-02 |
| Co-58 S | 0.15 | 1 | 1.56E-02 | 1.60E-02 |
| Co-60 M | 0.5 | 2 | 6.27E-02 | 8.11E-02 |
| Co-60 S | 0.5 | 2 | 1.69E-02 | 2.01E-02 |
| Cs-134 F | 1 | 1 | 1.81E-01 | 3.39E-01 |
| Cs-136 F | 1 | 1 | (a) | (a) |
| Cs-137 F | 1 | 0.95 | 1.87E-01 | 3.78E-01 |
| Eu-154 M | 0.998 | 1.64 | 4.58E-03 | 8.73E-03 |
| Eu-155 M | 0 | 1.64 | 4.57E-03 | 8.49E-03 |
| Fe-55 F | 0 | (a) | 1.18E-03 | 3.00E-03 |
| Fe-55 M | 0 | (a) | 3.23E-04 | 9.65E-04 |
| I-131 V | 0 | 0 | 5.83E-01 | 5.83E-01 |
| I-132 V | 0 | (a) | (a) | (a) |
| La-140 F | 1 | 2.12 | 4.63E-05 | 4.63E-05 |
| La-140 M | 1 | 2.12 | 4.80E-06 | 4.80E-06 |
| Mn-54 F | 0 | 1 | 1.21E-01 | 1.38E-01 |
| Mn-54 M | 0 | 1 | 3.26E-02 | 4.33E-02 |
| Mo-99 F | 1 | (a) | 3.34E-02 | 3.34E-02 |
| Mo-99 S | 1 | (a) | 1.54E-03 | 1.54E-03 |
| Nb-95 M | 0 | 1 | 3.01E-04 | 3.01E-04 |
| Nb-95 S | 0 | 1 | 1.93E-03 | 1.98E-03 |
| Nd-147 M | 1.5 | (a) | (a) | (a) |
| Nd-147 S | 1.5 | (a) | (a) | (a) |
| P-32 F | 1 | 0 | 1.70E-01 | 1.70E-01 |
| P-32 M | 1 | 0 | 1.36E-01 | 1.36E-01 |
| Pm-147 M | 0.5 | 0 | 2.46E-03 | 5.44E-03 |
| Pm-147 S | 0.5 | 0 | 5.10E-05 | 1.88E-04 |
| Pm-148m M | 0.954 | (a) | (a) | (a) |
| Pm-148m S | 0.954 | (a) | (a) | (a) |
| Pr-143 M | 1 | 0.38 | 2.06E-03 | 2.06E-03 |
| Pr-143 S | 1 | 0.38 | 3.24E-05 | 3.24E-05 |
| Pr-144 M | (a) | (a) | (a) | (a) |
| Pr-144 S | (a) | (a) | (a) | (a) |
| Ru-103 F | 0.5 | 0.97 | 1.37E-01 | 1.39E-01 |

## ATTACHMENT F
## SUMMARY OF IRFs, AND YIELD AND COUNTING ADJUSTMENT FACTORS
Page 2 of 2

| Radionuclide Absorption type | Beta yield and counting adjustment factor | Gamma yield | Chronic 24-h urine IRF | |
|---|---|---|---|---|
| | | | 90 d | 2 y (730 d) |
| Ru-103 M | 0.5 | 0.97 | 2.68E-02 | 2.78E-02 |
| Ru-103 S | 0.5 | 0.97 | 1.04E-02 | 1.06E-02 |
| Ru-106 F | 1 | 0.34 | 1.71E-01 | 1.89E-01 |
| Ru-106 M | 1 | 0.34 | 3.62E-02 | 4.84E-02 |
| Ru-106 S | 1 | 0.34 | 1.33E-02 | 1.56E-02 |
| Sb-125 F | 0.5 | 0.86 | 2.20E-01 | 2.29E-01 |
| Sb-125 M | 0.5 | 0.86 | 3.41E-02 | 4.73E-02 |
| Sm-151 M | 0 | | 2.49E-03 | 6.35E-03 |
| Sr-89 F | 1 | 0 | 1.90E-01 | 1.91E-01 |
| Sr-89 S | 1 | 0 | 2.70E-03 | 2.78E-03 |
| Sr-90 F | 1 | 0 | 2.13E-01 | 2.31E-01 |
| Sr-90 S | 1 | 0 | 3.17E-03 | 4.65E-03 |
| Ta-182 M | 1 | (a) | 1.13E-02 | 1.58E-02 |
| Ta-182 S | 1 | (a) | 2.86E-04 | 4.59E-04 |
| Te-125m F | 0 | (a) | (a) | (a) |
| Te-125m M | 0 | (a) | (a) | (a) |
| Te-127m F | (a) | (a) | (a) | (a) |
| Te-127m M | (a) | (a) | (a) | (a) |
| Te-129m F | 1 | (a) | (a) | (a) |
| Te-129m M | 1 | (a) | (a) | (a) |
| Te-132 F | 0.5 | (a) | (a) | (a) |
| Te-132 M | 0.5 | (a) | (a) | (a) |
| Y-90 M | 1 | 0 | 2.87E-03 | 2.87E-03 |
| Y-90 S | 1 | 0 | 3.39E-05 | 3.39E-05 |
| Y-91 M | 1 | 0.003 | 4.78E-03 | 5.03E-03 |
| Y-91 S | 1 | 0.003 | 7.37E-05 | 8.58E-05 |
| Zr-95 F | 1 | 1.1 | 1.05E-01 | 1.05E-01 |
| Zr-95 M | 1 | 1.1 | 1.47E-02 | 1.58E-02 |
| Zr-95 S | 1 | 1.1 | 5.28E-04 | 5.81E-04 |

a. The information was not obtained for these radionuclides. A review of the characteristics of these radionuclides indicates that addition of these parameters would not significantly change the results of this review.

## ATTACHMENT G - RADIONUCLIDE CONTRIBUTIONS TO GROSS ANALYSIS COUNTING RESULTS



The numbers in the following tables show radionuclide contributions to urinalysis counts. The zeros mean there was no activity, no adjustment factor, no yield factor, no IRF, or the result was less than 0.05%. To determine a radionuclide's activity contribution, the percentages must be divided by the appropriate adjustment factor in Table F-1.

Table G-1. Average radionuclide contributions to beta counting results for sampling at 90 days.

| | Soluble | | | | | Moderately Soluble | | | | | Insoluble | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Nuclide Type** | **10 d** | **40 d** | **180 d** | **1 y** | **Nuclide Type** | **10 d** | **40 d** | **180 d** | **1 y** | **Nuclide Type** | **10 d** | **40 d** | **180 d** | **1 y** |
| **Ba-140 F** | 6.4% | 1.9% | 0.0% | 0.0% | **Ba-140 F** | 11.2% | 3.5% | 0.0% | 0.0% | **Ba-140 F** | 13.3% | 4.2% | 0.0% | 0.0% |
| **Cd-113m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-113m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-113m S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Cd-115m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-115m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-115m S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ce-141 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ce-141 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ce-141 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ce-144 M** | 0.0% | 0.0% | 0.0% | 0.1% | **Ce-144 M** | 0.0% | 0.0% | 0.1% | 0.1% | **Ce-144 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Co-58 M** | 0.1% | 0.1% | 0.1% | 0.0% | **Co-58 M** | 0.2% | 0.3% | 0.2% | 0.0% | **Co-58 S** | 0.1% | 0.1% | 0.1% | 0.0% |
| **Co-60 M** | 0.2% | 0.2% | 0.6% | 0.8% | **Co-60 M** | 0.2% | 0.3% | 0.8% | 1.0% | **Co-60 S** | 0.1% | 0.1% | 0.2% | 0.3% |
| **Cs-134 F** | 0.4% | 0.6% | 1.7% | 3.5% | **Cs-134 F** | 0.9% | 1.3% | 3.5% | 5.9% | **Cs-134 F** | 1.1% | 1.7% | 4.5% | 7.1% |
| **Cs-136 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cs-136 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cs-136 F** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Cs-137 F** | 1.7% | 2.5% | 7.4% | 15.5% | **Cs-137 F** | 3.2% | 4.7% | 13.2% | 24.5% | **Cs-137 F** | 4.0% | 5.8% | 16.3% | 28.7% |
| **Eu-154 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Eu-154 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Eu-154 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Eu-155 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Eu-155 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Eu-155 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Fe-55 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Fe-55 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Fe-55 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **I-131 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-131 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-131 V** | 0.0% | 0.0% | 0.0% | 0.0% |
| **I-132 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-132 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-132 V** | 0.0% | 0.0% | 0.0% | 0.0% |
| **La-140 F** | 0.0% | 0.0% | 0.0% | 0.0% | **La-140 M** | 0.0% | 0.0% | 0.0% | 0.0% | **La-140 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Mn-54 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Mn-54 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Mn-54 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Mo-99 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Mo-99 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Mo-99 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Nb-95 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Nb-95 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Nb-95 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Nd-147 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Nd-147 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Nd-147 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **P-32 F** | 0.0% | 0.0% | 0.0% | 0.0% | **P-32 M** | 0.0% | 0.0% | 0.0% | 0.0% | **P-32 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pm-147 M** | 0.0% | 0.0% | 0.1% | 0.3% | **Pm-147 M** | 0.1% | 0.1% | 0.2% | 0.4% | **Pm-147 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pm-148m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pm-148m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pm-148m S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pr-143 M** | 0.6% | 0.2% | 0.0% | 0.0% | **Pr-143 M** | 1.0% | 0.3% | 0.0% | 0.0% | **Pr-143 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pr-144 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pr-144 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pr-144 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ru-103 F** | 14.7% | 12.7% | 4.1% | 0.0% | **Ru-103 M** | 5.5% | 5.0% | 1.6% | 0.0% | **Ru-103 S** | 2.7% | 2.4% | 0.8% | 0.0% |
| **Ru-106 F** | 8.0% | 10.5% | 22.6% | 32.8% | **Ru-106 M** | 3.8% | 5.3% | 11.4% | 14.4% | **Ru-106 S** | 1.8% | 2.6% | 5.7% | 6.7% |
| **Sb-125 F** | 0.0% | 0.3% | 0.7% | 1.2% | **Sb-125 M** | 0.0% | 0.1% | 0.2% | 0.3% | **Sb-125 M** | 0.0% | 0.1% | 0.3% | 0.4% |
| **Sm-151 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Sm-151 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Sm-151 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Sr-89 F** | 34.5% | 34.3% | 21.3% | 5.0% | **Sr-89 F** | 59.3% | 61.2% | 38.2% | 7.6% | **Sr-89 F** | 70.3% | 73.3% | 45.8% | 8.6% |
| **Sr-90 F** | 3.1% | 4.4% | 13.5% | 29.1% | **Sr-90 F** | 5.3% | 7.7% | 22.0% | 42.4% | **Sr-90 F** | 6.3% | 9.2% | 26.1% | 48.1% |
| **Ta-182 M** | 0.1% | 0.1% | 0.1% | 0.0% | **Ta-182 M** | 0.1% | 0.1% | 0.1% | 0.0% | **Ta-182 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-125m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-125m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-125m M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-127m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-127m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-127m M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-129m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-129m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-129m M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-132 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-132 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-132 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Y-90 M** | 0.0% | 0.0% | 0.1% | 0.2% | **Y-90 M** | 0.0% | 0.0% | 0.1% | 0.3% | **Y-90 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Y-91 M** | 1.0% | 1.1% | 0.9% | 0.3% | **Y-91 M** | 1.8% | 2.0% | 1.6% | 0.5% | **Y-91 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Zr-95 F** | 29.2% | 31.1% | 26.9% | 11.2% | **Zr-95 M** | 7.3% | 8.1% | 6.9% | 2.5% | **Zr-95 S** | 0.3% | 0.3% | 0.3% | 0.1% |

## ATTACHMENT G - RADIONUCLIDE CONTRIBUTIONS TO GROSS ANALYSIS COUNTING RESULTS



Table G-2. Average radionuclide contributions to beta counting results for sampling at 2 years.

| | Soluble | | | | | Moderately Soluble | | | | | Insoluble | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Nuclide Type** | **10 d** | **40 d** | **180 d** | **1 y** | **Nuclide Type** | **10 d** | **40 d** | **180 d** | **1 y** | **Nuclide Type** | **10 d** | **40 d** | **180 d** | **1 y** |
| **Ba-140 F** | 6.2% | 1.8% | 0.0% | 0.0% | **Ba-140 F** | 10.5% | 3.2% | 0.0% | 0.0% | **Ba-140 F** | 12.6% | 3.9% | 0.0% | 0.0% |
| **Cd-113m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-113m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-113m S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Cd-115m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-115m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-115m S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ce-141 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ce-141 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ce-141 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ce-144 M** | 0.1% | 0.1% | 0.2% | 0.2% | **Ce-144 M** | 0.1% | 0.1% | 0.3% | 0.3% | **Ce-144 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Co-58 M** | 0.1% | 0.1% | 0.1% | 0.0% | **Co-58 M** | 0.2% | 0.2% | 0.1% | 0.0% | **Co-58 S** | 0.1% | 0.1% | 0.0% | 0.0% |
| **Co-60 M** | 0.2% | 0.3% | 0.6% | 0.8% | **Co-60 M** | 0.3% | 0.4% | 0.8% | 0.9% | **Co-60 S** | 0.1% | 0.1% | 0.2% | 0.2% |
| **Cs-134 F** | 0.8% | 1.1% | 2.8% | 5.2% | **Cs-134 F** | 1.5% | 2.2% | 5.0% | 7.8% | **Cs-134 F** | 1.9% | 2.8% | 6.4% | 9.2% |
| **Cs-136 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cs-136 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cs-136 F** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Cs-137 F** | 3.3% | 4.6% | 12.7% | 24.7% | **Cs-137 F** | 5.9% | 8.3% | 20.9% | 35.5% | **Cs-137 F** | 7.3% | 10.3% | 25.5% | 40.9% |
| **Eu-154 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Eu-154 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Eu-154 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Eu-155 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Eu-155 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Eu-155 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Fe-55 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Fe-55 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Fe-55 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **I-131 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-131 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-131 V** | 0.0% | 0.0% | 0.0% | 0.0% |
| **I-132 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-132 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-132 V** | 0.0% | 0.0% | 0.0% | 0.0% |
| **La-140 F** | 0.0% | 0.0% | 0.0% | 0.0% | **La-140 M** | 0.0% | 0.0% | 0.0% | 0.0% | **La-140 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Mn-54 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Mn-54 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Mn-54 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Mo-99 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Mo-99 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Mo-99 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Nb-95 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Nb-95 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Nb-95 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Nd-147 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Nd-147 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Nd-147 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **P-32 F** | 0.0% | 0.0% | 0.0% | 0.0% | **P-32 M** | 0.0% | 0.0% | 0.0% | 0.0% | **P-32 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pm-147 M** | 0.1% | 0.1% | 0.2% | 0.4% | **Pm-147 M** | 0.1% | 0.2% | 0.4% | 0.6% | **Pm-147 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pm-148m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pm-148m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pm-148m S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pr-143 M** | 0.5% | 0.2% | 0.0% | 0.0% | **Pr-143 M** | 0.9% | 0.3% | 0.0% | 0.0% | **Pr-143 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pr-144 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pr-144 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pr-144 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ru-103 F** | 14.3% | 12.2% | 3.8% | 0.0% | **Ru-103 M** | 5.3% | 4.7% | 1.4% | 0.0% | **Ru-103 S** | 2.5% | 2.2% | 0.6% | 0.0% |
| **Ru-106 F** | 8.3% | 10.8% | 21.7% | 29.3% | **Ru-106 M** | 4.5% | 6.1% | 11.5% | 13.3% | **Ru-106 S** | 2.0% | 2.7% | 4.9% | 5.3% |
| **Sb-125 F** | 0.0% | 0.2% | 0.6% | 1.0% | **Sb-125 M** | 0.0% | 0.1% | 0.2% | 0.3% | **Sb-125 M** | 0.0% | 0.1% | 0.3% | 0.4% |
| **Sm-151 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Sm-151 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Sm-151 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Sr-89 F** | 33.6% | 33.1% | 19.5% | 4.1% | **Sr-89 F** | 56.2% | 57.2% | 33.4% | 5.8% | **Sr-89 F** | 66.9% | 68.5% | 39.3% | 6.4% |
| **Sr-90 F** | 3.2% | 4.4% | 12.4% | 24.7% | **Sr-90 F** | 5.2% | 7.3% | 19.0% | 33.3% | **Sr-90 F** | 6.2% | 8.8% | 22.3% | 37.4% |
| **Ta-182 M** | 0.1% | 0.1% | 0.1% | 0.0% | **Ta-182 M** | 0.1% | 0.1% | 0.1% | 0.0% | **Ta-182 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-125m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-125m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-125m M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-127m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-127m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-127m M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-129m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-129m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-129m M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-132 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-132 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-132 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Y-90 M** | 0.0% | 0.0% | 0.1% | 0.2% | **Y-90 M** | 0.0% | 0.0% | 0.1% | 0.2% | **Y-90 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Y-91 M** | 1.1% | 1.1% | 0.8% | 0.3% | **Y-91 M** | 1.8% | 1.9% | 1.4% | 0.4% | **Y-91 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Zr-95 F** | 28.1% | 29.7% | 24.4% | 9.1% | **Zr-95 M** | 7.3% | 7.6% | 5.3% | 1.5% | **Zr-95 S** | 0.3% | 0.4% | 0.3% | 0.1% |

## ATTACHMENT G- RADIONUCLIDE CONTRIBUTIONS TO GROSS ANALYSIS COUNTING RESULTS



Table G-3. Average radionuclide contributions to gamma counting results for sampling at 90 days.

| | Soluble | | | | | Moderately Soluble | | | | | Insoluble | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Nuclide Type** | **10 d** | **40 d** | **180 d** | **1 y** | **Nuclide Type** | **10 d** | **40 d** | **180 d** | **1 y** | **Nuclide Type** | **10 d** | **40 d** | **180 d** | **1 y** |
| **Ba-140 F** | 14.7% | 4.9% | 0.0% | 0.0% | **Ba-140 F** | 43.5% | 19.4% | 0.0% | 0.0% | **Ba-140 F** | 65.3% | 36.4% | 0.0% | 0.0% |
| **Cd-113m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-113m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-113m S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Cd-115m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-115m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-115m S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ce-141 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ce-141 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ce-141 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ce-144 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ce-144 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ce-144 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Co-58 M** | 0.6% | 1.0% | 0.8% | 0.2% | **Co-58 M** | 2.2% | 3.5% | 2.2% | 0.5% | **Co-58 S** | 1.1% | 1.9% | 0.9% | 0.2% |
| **Co-60 M** | 0.9% | 1.3% | 3.9% | 6.5% | **Co-60 M** | 2.7% | 4.1% | 7.3% | 8.3% | **Co-60 S** | 1.2% | 2.0% | 2.9% | 3.0% |
| **Cs-134 F** | 0.6% | 0.9% | 2.9% | 7.2% | **Cs-134 F** | 1.7% | 3.2% | 9.1% | 14.2% | **Cs-134 F** | 2.8% | 6.0% | 15.1% | 17.1% |
| **Cs-136 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cs-136 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cs-136 F** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Cs-137 F** | 2.2% | 3.4% | 12.0% | 30.0% | **Cs-137 F** | 6.9% | 11.7% | 32.9% | 55.8% | **Cs-137 F** | 11.1% | 21.7% | 53.4% | 69.0% |
| **Eu-154 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Eu-154 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Eu-154 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Eu-155 M** | 0.0% | 0.0% | 0.0% | 0.1% | **Eu-155 M** | 0.0% | 0.0% | 0.1% | 0.1% | **Eu-155 M** | 0.0% | 0.1% | 0.1% | 0.2% |
| **Fe-55 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Fe-55 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Fe-55 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **I-131 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-131 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-131 V** | 0.0% | 0.0% | 0.0% | 0.0% |
| **I-132 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-132 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-132 V** | 0.0% | 0.0% | 0.0% | 0.0% |
| **La-140 F** | 0.0% | 0.0% | 0.0% | 0.0% | **La-140 M** | 0.0% | 0.0% | 0.0% | 0.0% | **La-140 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Mn-54 F** | 0.0% | 0.3% | 1.8% | 2.2% | **Mn-54 M** | 0.0% | 0.3% | 1.2% | 1.1% | **Mn-54 M** | 0.0% | 0.4% | 1.8% | 1.4% |
| **Mo-99 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Mo-99 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Mo-99 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Nb-95 M** | 0.1% | 0.1% | 0.2% | 0.1% | **Nb-95 M** | 0.3% | 0.6% | 0.9% | 0.3% | **Nb-95 S** | 2.9% | 6.8% | 10.5% | 2.3% |
| **Nd-147 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Nd-147 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Nd-147 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **P-32 F** | 0.0% | 0.0% | 0.0% | 0.0% | **P-32 M** | 0.0% | 0.0% | 0.0% | 0.0% | **P-32 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pm-147 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pm-147 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pm-147 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pm-148m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pm-148m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pm-148m S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pr-143 M** | 0.3% | 0.1% | 0.0% | 0.0% | **Pr-143 M** | 0.8% | 0.4% | 0.0% | 0.0% | **Pr-143 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pr-144 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pr-144 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pr-144 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ru-103 F** | 35.5% | 34.0% | 13.3% | 0.0% | **Ru-103 M** | 21.7% | 26.3% | 10.1% | 0.0% | **Ru-103 S** | 13.3% | 19.6% | 6.9% | 0.0% |
| **Ru-106 F** | 3.1% | 4.6% | 13.2% | 23.4% | **Ru-106 M** | 2.2% | 3.7% | 8.8% | 10.7% | **Ru-106 S** | 1.4% | 2.7% | 5.2% | 4.8% |
| **Sb-125 F** | 0.0% | 0.6% | 2.1% | 4.3% | **Sb-125 M** | 0.0% | 0.3% | 0.9% | 1.3% | **Sb-125 M** | 0.0% | 0.6% | 1.4% | 1.6% |
| **Sm-151 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Sm-151 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Sm-151 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Sr-89 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Sr-89 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Sr-89 F** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Sr-90 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Sr-90 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Sr-90 F** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ta-182 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ta-182 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ta-182 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-125m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-125m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-125m M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-127m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-127m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-127m M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-129m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-129m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-129m M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-132 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-132 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-132 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Y-90 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Y-90 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Y-90 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Y-91 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Y-91 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Y-91 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Zr-95 F** | 41.9% | 48.9% | 49.7% | 25.9% | **Zr-95 M** | 17.8% | 26.4% | 26.3% | 7.6% | **Zr-95 S** | 1.0% | 1.8% | 1.7% | 0.3% |

Table G-4. Average radionuclide contributions to gamma counting results for sampling at 2 years.

| Soluble | | | | | Moderately Soluble | | | | | Insoluble | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Nuclide Type** | **10 d** | **40 d** | **180 d** | **1 y** | **Nuclide Type** | **10 Days** | **40 d** | **180 d** | **1 y** | **Nuclide Type** | **10 d** | **40 d** | **180 d** | **1 y** |
| **Ba-140 F** | 14.3% | 4.7% | 0.0% | 0.0% | **Ba-140 F** | 39.8% | 17.0% | 0.0% | 0.0% | **Ba-140 F** | 58.9% | 30.7% | 0.0% | 0.0% |
| **Cd-113m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-113m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-113m S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Cd-115m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-115m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Cd-115m S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ce-141 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ce-141 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ce-141 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ce-144 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ce-144 M** | 0.0% | 0.0% | 0.1% | 0.1% | **Ce-144 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Co-58 M** | 0.6% | 0.9% | 0.7% | 0.2% | **Co-58 M** | 2.0% | 2.9% | 1.5% | 0.3% | **Co-58 S** | 0.9% | 1.4% | 0.5% | 0.1% |
| **Co-60 M** | 1.1% | 1.5% | 3.8% | 5.3% | **Co-60 M** | 2.8% | 3.9% | 5.8% | 6.2% | **Co-60 S** | 1.0% | 1.5% | 1.9% | 1.9% |
| **Cs-134 F** | 1.0% | 1.5% | 4.6% | 9.5% | **Cs-134 F** | 2.8% | 4.8% | 11.4% | 14.9% | **Cs-134 F** | 4.3% | 8.2% | 16.5% | 17.0% |
| **Cs-136 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cs-136 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Cs-136 F** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Cs-137 F** | 4.3% | 6.2% | 19.5% | 42.3% | **Cs-137 F** | 11.7% | 18.5% | 43.8% | 63.7% | **Cs-137 F** | 18.0% | 31.4% | 62.9% | 74.1% |
| **Eu-154 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Eu-154 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Eu-154 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Eu-155 M** | 0.0% | 0.0% | 0.0% | 0.1% | **Eu-155 M** | 0.0% | 0.0% | 0.1% | 0.1% | **Eu-155 M** | 0.0% | 0.1% | 0.1% | 0.2% |
| **Fe-55 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Fe-55 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Fe-55 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **I-131 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-131 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-131 V** | 0.0% | 0.0% | 0.0% | 0.0% |
| **I-132 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-132 V** | 0.0% | 0.0% | 0.0% | 0.0% | **I-132 V** | 0.0% | 0.0% | 0.0% | 0.0% |
| **La-140 F** | 0.0% | 0.0% | 0.0% | 0.0% | **La-140 M** | 0.0% | 0.0% | 0.0% | 0.0% | **La-140 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Mn-54 F** | 0.0% | 0.3% | 1.6% | 1.8% | **Mn-54 M** | 0.0% | 0.2% | 1.0% | 0.9% | **Mn-54 M** | 0.0% | 0.4% | 1.3% | 1.0% |
| **Mo-99 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Mo-99 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Mo-99 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Nb-95 M** | 0.1% | 0.1% | 0.2% | 0.1% | **Nb-95 M** | 0.3% | 0.5% | 0.6% | 0.2% | **Nb-95 S** | 2.6% | 5.6% | 6.5% | 1.3% |
| **Nd-147 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Nd-147 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Nd-147 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **P-32 F** | 0.0% | 0.0% | 0.0% | 0.0% | **P-32 M** | 0.0% | 0.0% | 0.0% | 0.0% | **P-32 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pm-147 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pm-147 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pm-147 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pm-148m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pm-148m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pm-148m S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pr-143 M** | 0.3% | 0.1% | 0.0% | 0.0% | **Pr-143 M** | 0.7% | 0.3% | 0.0% | 0.0% | **Pr-143 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Pr-144 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pr-144 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Pr-144 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ru-103 F** | 34.7% | 32.7% | 11.9% | 0.0% | **Ru-103 M** | 20.2% | 23.4% | 7.3% | 0.0% | **Ru-103 S** | 11.9% | 16.2% | 4.2% | 0.0% |
| **Ru-106 F** | 3.3% | 4.7% | 12.2% | 18.8% | **Ru-106 M** | 2.6% | 4.0% | 7.8% | 8.1% | **Ru-106 S** | 1.3% | 2.3% | 3.6% | 3.0% |
| **Sb-125 F** | 0.0% | 0.6% | 1.8% | 3.2% | **Sb-125 M** | 0.0% | 0.3% | 0.8% | 1.0% | **Sb-125 M** | 0.0% | 0.6% | 1.2% | 1.2% |
| **Sm-151 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Sm-151 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Sm-151 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Sr-89 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Sr-89 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Sr-89 F** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Sr-90 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Sr-90 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Sr-90 F** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Ta-182 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ta-182 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Ta-182 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-125m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-125m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-125m M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-127m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-127m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-127m M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-129m F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-129m M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-129m M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Te-132 F** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-132 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Te-132 M** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Y-90 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Y-90 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Y-90 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Y-91 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Y-91 M** | 0.0% | 0.0% | 0.0% | 0.0% | **Y-91 S** | 0.0% | 0.0% | 0.0% | 0.0% |
| **Zr-95 F** | 40.4% | 46.5% | 43.6% | 18.7% | **Zr-95 M** | 17.1% | 24.1% | 19.8% | 4.5% | **Zr-95 S** | 0.9% | 1.6% | 1.1% | 0.2% |

# ATTACHMENT H
# EXAMPLE INTAKES ESTIMATED FROM BIOASSAY AND FROM AIR SAMPLE DATA


The information in Sections 7 and 8 of this document is used with example gross beta bioassay data and gross beta air sample data to provide examples of reconstructing intakes using information in claims, site profiles and this document. Some considerations about parameter selection are included, and these examples highlight two important points about the use of this document.

The first point is that in some cases the sum of the fractions of a parameter of interest will not appear to add up to one, e.g., the air concentrations from the listed radionuclides in Example 3 add up to 20% more than the initially assumed beta air concentration. In this particular case, this is because the initial calculations were based on what would be detected from an air sample, not from all the radioactivity in the air. In addition, in the process of simplification of the source term and in estimation of ratios from a variety of complex kinetic equations, the precision of values is reduced. However, on the whole, the parameter selection and mathematical uncertainty introduced into these values tend to produce dose estimates more favorable to the claimant than a more precise calculation might produce. (Although this last statement seems to imply that a more accurate dose reconstruction method might be produced from more precise assignment of some parameters, this is unlikely to be the case because of the inherent uncertainty related to selection of dose parameters for a specific source term and an individual's exposure in the first place.)

The second point is the importance of understanding the exposure scenario, source term and exposure conditions to best assess an individual exposure scenario. Examples 1 and 2 demonstrate this issue, because the estimated intakes based on urinalysis and the predicted whole body counts indicate that the source terms and/or intake scenarios might need adjustment based on additional information that might be available for the claim and/or the site. While it is fairly straightforward to adjust parameters until the numbers "make sense," it is equally important to be sure the exposure scenarios make sense. For best estimates of dose, the goal is to not underestimate dose (unless compensability can be readily determined), but neither should dose be grossly overestimated. Example 1 indicates that the Cs-137 intakes are high by a factor of 4, and the predicted whole body counts exceed the 95th-percentile whole body counts (the reported whole body count plus the two-sigma value) by a factor of 2. This might be close enough for some claims, while other claims might require more rigorous consideration of the different parameters chosen to estimate intakes. Supervisors, principal dosimetrists, and site profile team leaders and subject experts are available to provide additional guidance and/or check for additional information when compensability determinations are initially ambiguous.

**Example 1. Gross beta bioassay data from waste management worker.**

In this example the bioassay data in Tables H-1 and H-2 were assumed for a (reactor) waste management worker employed from January 28, 1960 through January 25, 1971.

Because the worker was involved in waste management activities, Table 5-12 indicates the use of a 1-year fuel decay assumption for selection of intake parameters is appropriate. For gross beta urinalyses, the strontium fraction would usually be selected because of the likelihood that it indeed was part of the matrix. In this example, it is assumed that the chemical processing of the urine resulted in removal of the cesium, ruthenium and similar chemicals, so the fractions from Table 7-2 are used and 0.43 for the 1-year, Sr-90 indicator radionuclide is used to estimate the fraction of Sr-90 in the gross beta urinalysis results. The product of the gross beta urinalysis results and the Sr-90 factor of 0.43 produces the Sr-90 results column in Table H-1.

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## EXAMPLE INTAKES ESTIMATED FROM BIOASSAY AND FROM AIR SAMPLE DATA


Table H-1. Gross beta urinalysis results (pCi/24 h).

| Date | Gross beta[a] | Sr-90[b] |
|---|---|---|
| 07/23/1964 | 56 | 24.08 |
| 10/20/1964 | 42 | 18.06 |
| 08/13/1965 | 33 | 14.19 |
| 10/21/1965 | 41 | 17.63 |
| 02/16/1966 | 57 | 24.51 |
| 05/03/1966 | 31 | 13.33 |
| 08/11/1966 | 17 | 7.31 |
| 11/02/1966 | 31 | 13.33 |

a. Detection threshold of 20 pCi/24 h.
b. Calculated by multiplying 1-y Sr-90 fraction, 0.43, from Table 7-2, times the bioassay results.

Table H-2. Cesium-137 whole body count results and mean whole body activity due to fallout (pCi).

| Date | Cs-137 | 2-sigma | Detection threshold | Fallout[a] |
|---|---|---|---|---|
| 01/23/1967 | 3,000 | - | - | 5,670 |
| 01/16/1968 | 4,400 | 2,200 | - | 3,510 |
| 01/20/1969 | 2,100 | 2,400 | 2,500 | 2,700 |
| 01/25/1971 | 0 | 2,300 | - | 2,700 |

a. Mean whole body Cs-137 activity in the United States due to fallout. Adjusted from becquerels to picocuries by multiplying NCRP Report 94(NCRP 1987, Table B.5) results by 27 picocuries per becquerel.

The Sr-90 urinalysis estimates from Table H-1 were used to fit a chronic intake of Sr-90 type F (no strontium titanates were used at the facility, so type S is not assumed) for the period, January 28, 1960 through January 25, 1971, and resulted in a calculated Sr-90 intake rate of 67.14 pCi/d. The intake rates from the other radionuclides in the source term were determined by multiplying the 1-year Sr-90 ratios in Table 7-3 by the Sr-90 intake rate; the results are listed in Table H-3.

To check whether the intake results in Table H-3 are reasonable, the Cs-137 intake rate was used to predict the whole body count results for comparison with the measured results in Table H-2. The IMBA-predicted whole body counts for a Cs-137 intake rate of 87.3 pCi/d from January 28, 1960 through January 25, 1971, are 8,000 pCi on January 23, 1967 to 10,000 pCi on January 25, 1971. The whole body counts are over-predicted, especially for later years. If these results are deemed too large for the particular claim, it would be reasonable to review the claim record to determine if the exposure period to fission products might have been shorter. Another alternative to correlate the Cs-137 whole body predicted results more precisely to the estimated intake would be to eliminate the Cs-137 intake from the urinalysis and instead calculate the Cs-137 intake from the whole body count results. Because there are many ways that the ratios of the radionuclides can be modified during production and measurement processes, it is not generally recommended that all the radionuclide intakes be reduced by the same factor as Cs-137, unless there is additional information that clearly would support such a reduction.

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**EXAMPLE INTAKES ESTIMATED FROM BIOASSAY AND FROM AIR SAMPLE DATA**


Table H-3. Radionuclide intake rates derived from Sr-90 intake rate of 67.14 pCi/d.

| Radionuclide | 1-y factor[a] | pCi/d |
|---|---|---|
| Ba-140 | - | 0.00E+00 |
| Ce-141 | 2.1 | 1.41E+02 |
| Ce-144 | 7.8 | 5.24E+02 |
| Cs-134 | 0.31 | 2.08E+01 |
| Cs-137 | 1.3 | 8.73E+01 |
| Eu-155 | 0.081 | 5.44E+00 |
| Fe-55 | 0.81 | 5.44E+01 |
| I-131 | - | 0.00E+00 |
| La-140 | - | 0.00E+00 |
| Nb-95 | 3 | 2.01E+02 |
| Pm-147 | 3.1 | 2.08E+02 |
| Ru-103 | - | 0.00E+00 |
| Ru-106 | 3.8 | 2.55E+02 |
| Sr-89 | 0.31 | 2.08E+01 |
| Sr-90 | **1** | 6.71E+01 |
| Y-91 | 0.77 | 5.17E+01 |
| Zr-95 | 1.4 | 9.40E+01 |
| Total daily intake | - | 1.73E+03 |

a. From Table 7-3 Sr-90 factors.

Another check on the reasonableness of the Example 1 intake calculation is an estimate of the air concentration that would produce this chronic intake. The air concentration can be derived by multiplying the Total Daily Intake rate in Table H-3 by 365 calendar days divided by 2,400 cubic meters, the default annual light work air intake volume, which results in a calculated air concentration of 2.6 E-10 µCi/mL (or 580 dpm/m$^3$) when units are appropriately adjusted. This calculated concentration is in the range historical limiting air concentrations for Sr-90, 2E-10 to 2E-9 µCi/mL and for beta/gamma emitters other than Sr-90, 4E-9 and 2E-6 µCi/mL (ORAUT 2005, Table 2-1), which indicates that such a chronic exposure is conceivable (although perhaps on the high side for some sites). This scoping calculation is just to show whether the intake estimate is reasonable; some radionuclides such as Fe-55 and Pm-147 might not be readily detected by some measurement systems and some radionuclides might decay significantly before counting. The source term is for dose estimation and so is not necessarily representative of the actual activity in the air (even though, for the purpose of dose reconstruction, it appears to provide a sufficiently accurate bound for the intake estimate).

To provide further perspective on this intake, the 50-year organ doses derived from the product of the intake rates in Table H-3, 4,015 days of exposure and Radiological Toolbox dose inhalation dose conversion factors are listed in Table H-4. Except for Sr-90, which was assumed to be type F, the absorption type that resulted in the maximum dose was selected when multiple absorption types were possible.

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## EXAMPLE INTAKES ESTIMATED FROM BIOASSAY AND FROM AIR SAMPLE DATA


Table H-4. 50-year doses based on chronic inhalation intake rates specified in Table H-3 from January 28, 1960 through January 25, 1971 (4,015 days).

| Organ | Dose (rem) |
|---|---|
| Adrenals | 6.06E-02 |
| Urinary bladder | 6.42E-02 |
| Bone surface | 1.03E+00 |
| Brain | 5.40E-02 |
| Breast | 5.35E-02 |
| Esophagus | 5.66E-02 |
| Stomach | 6.19E-02 |
| Small intestine | 7.53E-02 |
| Upper large intestine | 1.67E-01 |
| Lower large intestine | 4.11E-01 |
| Colon | 2.68E-01 |
| Kidneys | 5.78E-02 |
| Liver | 8.77E-01 |
| Muscle | 5.50E-02 |
| Ovaries | 5.95E-02 |
| Pancreas | 5.94E-02 |
| Red marrow | 4.24E-01 |
| Extrathoracic airways | 4.77E-01 |
| Lungs | 2.94E+00 |
| Skin | 5.17E-02 |
| Spleen | 6.26E-02 |
| Testes | 5.35E-02 |
| Thymus | 5.66E-02 |
| Thyroid | 5.54E-02 |
| Uterus | 5.73E-02 |

**Example 2. Gross beta bioassay data from worker involved in spent fuel storage operations.**

Example 2 also uses the bioassay data provided in Tables H-1 and H-2 to estimate intakes, but the worker is assumed to have worked in a spent fuel storage operation and to be exposed to 10-day decayed reactor fuel based on the information in Table 5-12. For this example it is assumed that the urinalysis did not include a chemical separation step, so the fraction of Sr-90 in the urine sample would be 0.024 from Table 7-1; the resulting Sr-90 urine activity is listed in Table H-5.

The Sr-90 data are used to fit a chronic intake of Sr-90 type F (no strontium titanates were used at the facility, so type S is not assumed) for the period, January 28, 1960 through January 25, 1971, which resulted in a calculated Sr-90 intake rate of 3.75 pCi/d. The intake rates from the other radionuclides in the source term are determined by multiplying the 10-day Sr-90 ratios in Table 7-3 by the Sr-90 intake rate and the results are listed in Table H-6.

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## EXAMPLE INTAKES ESTIMATED FROM BIOASSAY AND FROM AIR SAMPLE DATA



Table H-5. Gross beta urinalysis results (pCi/24 h).

| Date | Urinalysis[a] | Sr-90[b] |
|---|---|---|
| 07/23/1964 | 56 | 1.344 |
| 10/20/1964 | 42 | 1.008 |
| 08/13/1965 | 33 | 0.792 |
| 10/21/1965 | 41 | 0.984 |
| 02/16/1966 | 57 | 1.368 |
| 05/03/1966 | 31 | 0.744 |
| 08/11/1966 | 17 | 0.408 |
| 11/02/1966 | 31 | 0.744 |

a. Detection threshold of 20 pCi/24 h.
b. Calculated by multiplying 10-d Sr-90 fraction, 0.024, from Table 7-1 times the bioassay results.

Table H-6. Radionuclide intake rates derived from Sr-90 intake rate of 3.748 pCi/d.

| Radionuclide | 10-d factors[a] | pCi/d |
|---|---|---|
| Ba-140 | 26 | 9.74E+01 |
| Ce-141 | 30 | 1.12E+02 |
| Ce-144 | 16 | 6.00E+01 |
| Cs-134 | 0.3 | 1.12E+00 |
| Cs-137 | 1.3 | 4.87E+00 |
| Eu-155 | 0.072 | 2.70E-01 |
| Fe-55 | 1.7 | 6.37E+00 |
| I-131 | 532 | 1.99E+03 |
| La-140 | 30 | 1.12E+02 |
| Nb-95 | 28 | 1.05E+02 |
| Pm-147 | 3.4 | 1.27E+01 |
| Ru-103 | 23 | 8.62E+01 |
| Ru-106 | 5.5 | 2.06E+01 |
| Sr-89 | 19 | 7.12E+01 |
| Sr-90 | 1 | 3.75E+00 |
| Y-91 | 23 | 8.62E+01 |
| Zr-95 | 31 | 1.16E+02 |
| Total daily intake | - | 2.89E+03 |

a. From Table 7-3 Sr-90 factors.

To check if the intake results in Table H-6 are reasonable, the Cs-137 intake rate is used to predict the whole body count results in Table H-2. The IMBA-predicted whole body counts for a Cs-137 intake rate of 4.87 pCi/d from January 28, 1960 through January 25, 1971, range from about 400 pCi on January 23, 1967 to 600 pCi on January 25, 1971. The whole body counts are under-predicted, but during this period the mean Cs-137 whole body activity due to fallout was reported as 2,700 pCi or larger and could account for the elevated Cs-137 counts, so the intake estimate is deemed reasonable. Depending on the particular site and particular years, an additional intake of Cs-137 might be assumed to account for under-predicted whole body count results that cannot be explained by fallout.

Another check on the reasonableness of the Example 2 intakes is to estimate the air concentration that would produce the chronic intakes in Table H-6. The air concentration can be derived by

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multiplying the Total Daily Intake rate in Table H-6 by 365 calendar days divided by 2,400 cubic meters, the default annual light work air intake volume, which results in a calculated air concentration of 4.4E-10 µCi/mL (980 dpm/$m^3$). This calculated air concentration is a little bit high compared to the range of historical limiting air concentrations for Sr-90, 2E-10 to 2E-9 µCi/mL, but not when compared to limiting values for beta/gamma emitters other than Sr-90, 4E-9 and 2E-6 µCi/mL (ORAUT 2005, Table 2-1). This concentration does not appear inconceivable (although perhaps on the high side for some sites). Note that this is just a scoping calculation to show whether the intake estimate is reasonable; some radionuclides such as Fe-55, I-131 and Pm-147 might not be readily collected and/or detected by some measurement systems. If it were assumed that the iodine in air was not reported, the beta air concentration would be 1.4E-10 µCi/mL (300 dpm/$m^3$). The iodine intake would be included or excluded in dose calculations based on what is known about site control and monitoring of radioiodine exposures. It is favorable to the claimant to include the iodine intake.

To provide further perspective on the intakes listed in Table H-6, the 50-year organ doses derived from the product of the intake rates in Table H-6, 4,015 days of exposure and Radiological Toolbox dose inhalation dose conversion factors are listed in Table H-7. Except for Sr-90, which was assumed to be type F, and I-131, which was assumed to be $I_2$ vapor, the absorption type that resulted in the maximum dose was selected when multiple absorption types were possible.

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## EXAMPLE INTAKES ESTIMATED FROM BIOASSAY AND FROM AIR SAMPLE DATA



Table H-7. 50-year doses based on chronic inhalation intake rates specified in Table H-6 from January 28, 1960 through January 25, 1971(4,015 days).

| Organ | Dose (rem) |
|---|---|
| Adrenals | 1.24E-02 |
| Urinary bladder | 3.20E-02 |
| Bone surface | 2.00E-01 |
| Brain | 1.20E-02 |
| Breast | 8.88E-03 |
| Esophagus | 1.22E-02 |
| Stomach | 1.26E-02 |
| Small intestine | 1.66E-02 |
| Upper large intestine | 4.49E-02 |
| Lower large intestine | 1.12E-01 |
| Colon | 7.41E-02 |
| Kidneys | 1.06E-02 |
| Liver | 1.09E-01 |
| Muscle | 1.17E-02 |
| Ovaries | 1.27E-02 |
| Pancreas | 1.07E-02 |
| Red marrow | 6.46E-02 |
| Extrathoracic airways | 1.97E-01 |
| Lungs | 4.83E-01 |
| Skin | 8.70E-03 |
| Spleen | 1.00E-02 |
| Testes | 7.71E-03 |
| Thymus | 1.22E-02 |
| Thyroid | 1.16E+01 |
| Uterus | 1.04E-02 |

**Example 3. Gross beta air sample data from fuel dissolution area.**

Example 3 is an estimate of intake rates from site air concentration measurements. This method does not directly apply to exposure from effluents where radionuclide ratios might be significantly altered due to the use of radionuclide filtration and holdup systems from those initially in the air stream.

Exposure due to work in a spent fuel dissolution area in the 1960s is assumed. The 180-day reactor fuel decay time is selected based on Table 5-12. This example assumes monthly average beta air concentrations in the dissolution area ranged from 1E-14 to 1E-10 µCi/mL during the period, 1960 through 1971. It is further assumed that the air concentrations were not tabulated, so no average value is available. The following calculation is used to bound radionuclide exposures, using the upper range of the monthly average air concentrations, 1E-10 µCi/mL.

The 180-day Sr-90 fraction from Table 7-4, 0.019, is used to estimate the Sr-90 bounding air concentration, 0.019 times 1E-10 µCi/mL, which equals 1.9 E-12 µCi/mL. That in turn would produce an annual intake rate of 4,560 pCi for an annual air intake of 2,400 cubic meters per work year and a daily (calendar day) intake rate of 12.5 pCi for use in dose calculations. These values are used with

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the 180-day intake fractions in Table 7-3 to estimate air concentrations and inhalation intakes of associated radionuclides. Note that the total calculated air concentration in Table H-8 is slightly higher than the starting air concentration. This is primarily due to the assumption that some of the beta emitters are not measured in the air sample. An ingestion intake must also be estimated; it is achieved by calculating the intake per workday by multiplying the air concentration in units of activity per cubic meter by 0.2 (NIOSH 2004). The ingestion intake in picocuries (pCi) in a year would be 0.2 times the air concentration in µCi/mL times 1E+6 mL/$m^3$ times 1E+6 pCi/µCi times 250 workdays per year. To obtain the daily intake rate the annual intake rate is divided by 365 calendar days. Table H-8 summarizes the results of the air concentrations and intakes for Example 3.

Table H-8. Radionuclide air concentrations and intake rates derived from a beta air concentration of 1E-10 µCi/mL.

| Radionuclide | 180 d[a] | Air concentration (µCi/mL) | Inhalation intake (pCi/d) | Ingestion Intake (pCi/d) |
|---|---|---|---|---|
| Ba-140 | - | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| Ce-141 | 1.4 | 2.66E-12 | 1.75E+01 | 3.64E-01 |
| Ce-144 | 14 | 2.66E-11 | 1.75E+02 | 3.64E+00 |
| Cs-134 | 0.34 | 6.46E-13 | 4.25E+00 | 8.85E-02 |
| Cs-137 | 1.3 | 2.47E-12 | 1.62E+01 | 3.38E-01 |
| Eu-155 | 0.09 | 1.71E-13 | 1.12E+00 | 2.34E-02 |
| Fe-55 | 1.1 | 2.09E-12 | 1.37E+01 | 2.86E-01 |
| I-131 | - | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| La-140 | - | 0.00E+00 | 0.00E+00 | 0.00E+00 |
| Nb-95 | 16 | 3.04E-11 | 2.00E+02 | 4.16E+00 |
| Pm-147 | 3.5 | 6.65E-12 | 4.37E+01 | 9.11E-01 |
| Ru-103 | 2 | 3.80E-12 | 2.50E+01 | 5.21E-01 |
| Ru-106 | 5.4 | 1.03E-11 | 6.75E+01 | 1.41E+00 |
| Sr-89 | 3.6 | 6.84E-12 | 4.50E+01 | 9.37E-01 |
| Sr-90 | **1** | 1.90E-12 | 1.25E+01 | 2.60E-01 |
| Y-91 | 5.8 | 1.10E-11 | 7.25E+01 | 1.51E+00 |
| Zr-95 | 8.4 | 1.60E-11 | 1.05E+02 | 2.19E+00 |
| Total | - | 1.21E-10 | 7.99E+02 | 1.66E+01 |

a. From Table 7-3 Sr-90 factors.

To provide perspective on the intakes estimated for Example 3, the 50-year organ doses, derived from the product of the intake rates in Table H-8, 4,015 days of exposure and Radiological Toolbox dose inhalation dose conversion factors, are listed in Table H-9. Except for Sr-90, which was assumed to be type F, the absorption type that resulted in the maximum dose was selected when multiple absorption types were possible. The organ doses associated with Example 3 are listed in Table H-9.

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Table H-9. 50-year doses based on chronic inhalation and ingestion intake rates specified in Table H-8 from January 28, 1960 through January 25, 1971 (4,015 days).

| Organ | Dose (rem) |
|---|---|
| Adrenals | 1.99E-02 |
| Urinary bladder | 2.07E-02 |
| Bone surface | 3.29E-01 |
| Brain | 1.66E-02 |
| Breast | 1.62E-02 |
| Esophagus | 1.72E-02 |
| Stomach | 1.91E-02 |
| Small intestine | 2.49E-02 |
| Upper large intestine | 5.92E-02 |
| Lower large intestine | 1.48E-01 |
| Colon | 9.63E-02 |
| Kidneys | 1.82E-02 |
| Liver | 2.87E-01 |
| Muscle | 1.67E-02 |
| Ovaries | 1.92E-02 |
| Pancreas | 1.84E-02 |
| Red marrow | 1.24E-01 |
| Extrathoracic airways | 1.80E-01 |
| Lungs | 9.78E-01 |
| Skin | 1.53E-02 |
| Spleen | 1.86E-02 |
| Testes | 1.54E-02 |
| Thymus | 1.72E-02 |
| Thyroid | 1.67E-02 |
| Uterus | 1.74E-02 |