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Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses

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FOR DOCUMENTS MARKED AS A TOTAL REWRITE, REVISION, OR PAGE CHANGE, REPLACE THE PRIOR REVISION AND DISCARD / DESTROY ALL COPIES OF THE PRIOR REVISION.

☐ New ☐ Total Rewrite ☒ Revision ☐ Page Change

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
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.
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. Signature on File 10/10/2007 Donald E. Bihl, Document Owner Signature on File 10/03/2007 John M. Byrne, Task 3 Manager Signature on File 10/03/2007 Edward F. Maher, Task 5 Manager Signature on File 11/13/2007 Kate Kimpan, Project Director Brant A. Ulsh Signature on File for 11/19/2007 James W. Neton, Associate Director for Science
06/13/2013	01	Total rewrite of the document to implement the following: a limiting reactor approach in place of averaging, an expanded range of reactor modeling parameters, more rigorous treatment of radionuclide ingrowth and decay and parent/progeny behavior, updated beta and gamma yield values, removal of the counting adjustment factors, and explicit consideration of changes in the beta and gamma activity of urine samples as a function of time between sample collection and counting. Additional changes were made in response to comments received from the Procedures Subcommittee and to provide clarifications with respect to methods and application of the results. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Initiated by Robert E. Burns, Jr. and Donald E. Bihl.
03/06/2014	02	Revision initiated to correct an error with the Pm-147 intake fractions in Tables 7-3b and 7-3c. The values had mistakenly been entered as zeros. No changes occurred as a result of formal internal and NIOSH review. Initiated by Robert E. Burns, Jr.

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
02/06/2015	03	Revision initiated for changes to Table H-5 on page 71. The ingestion intake values given in the last column of Table H-5 were revised to account for the ratio of 250 work days to 365 calendar days. Updated reference to OTIB-0060 in the Reference Section and text. No changes occurred as a result of formal internal and NIOSH review. Training required: as determined by the Objective Manager. Initiated by Robert E. Burns, Jr.
08/27/2015	04	Revision initiated to correct typographical errors in Tables B-1 and G-4. The elemental release fractions given in Table B-1 for arsenic, bromine, iodine, potassium, sodium, phosphorus, and sulfur were corrected from 0.05 to 0.50. In Table G-4 the column headings for the data for the N Reactor 1 (N Rx 1) and N Reactor 2 (N Rx 2) cases were corrected by reformatting to fit the Attachment G tables to one page each. They had been mislabeled as FFTF 1 and FFTF 2. No changes occurred as a result of formal internal and NIOSH review. Training required: as determined by the Objective Manager. Initiated by Robert E. Burns, Jr.

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ACRONYMS AND ABBREVIATIONS

ATR	Advanced Test Reactor
Bq	becquerel
d	day
DCF	dose conversion factor
DOE	U.S. Department of Energy
FFTF	Fast Flux Test Facility
g	gram
hr	hour
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
IREP	Interactive RadioEpidemiological Program
IRF	intake retention fraction
keV	kiloelectron-volt, 1 thousand electron-volts
m	meter
MAP	mixed activation product
MDA	minimum detectable activity
MeV	megaelectron-volt, 1 million electron-volts
MFP	mixed fission product
mL	milliliter
MT	metric ton
MTHM	metric tons of heavy metal
MTU	metric tons of uranium
MW	megawatt
MWd	megawatt-day
NIF	Normalized Intake Fraction
NIOSH	National Institute for Occupational Safety and Health
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
pCi	picocurie
PWR	pressurized-water reactor
RaLa	radioactive lanthanum
s	second
SRDB Ref ID	Site Research Database Reference Identification (number)
SRS	Savannah River Site
SS	stainless steel
t	ton (short)
TIB	technical information bulletin

TRIGA	Training, Research, Isotopes, General Atomics
U.S.C.	United States Code
yr	year
µCi	microcurie
§	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 historical 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 site(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 a basis for assigning 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 for dose reconstruction involving reactor-related source terms if site-specific guidance is not available.

3.0 SCOPE

The guidance in this document applies, if 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. The activity and intake fraction values were derived to address gross beta or gross gamma results specifically from air samples or urinalyses, but they 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., the Hanford Site's Fast Flux Test Reactor (FFTF), Argonne National Laboratory–West's experimental breeder reactors]. It does not apply to operations involving decay times shorter than 10 days (e.g., early radioactive lanthanum (RaLa) processing). It also does not apply to determination of intakes if 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 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 selection of the representative reactor cases and calculation of fission and activation product activity at decay intervals of interest following reactor shutdown.

5.1 ORIGEN2 MODELING FOR THE INITIAL SET OF REPRESENTATIVE REACTORS

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:

- 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 selected for fuel irradiation calculations using the ORIGEN2 isotope generation and depletion code, version 2.1 (Croff 1980). More than one reactor or one type of fuel with cladding was selected for some of the categories. The reactors were selected to cover a wide range of fuel (cladding) types, enrichments, and burnups. 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	Sodium-cooled fast reactors
ATR	High-flux reactors
TRIGA reactor (Al-clad fuel)	Research reactors
TRIGA reactor (SS-clad fuel)	Research reactors
PWR	Generic reactor

Eleven ORIGEN2 runs were performed for the seven reactor cases listed 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. The production and TRIGA reactor cases were run with two cross-section and fission product yield libraries to evaluate the choice of library on the inventory results. Reactor-specific ORIGEN2 libraries

were available for the FFTF, the ATR, and the pressurized-water reactor (PWR). 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 to be representative of the compositions of typical source terms encountered by workers. Decay times are discussed further in Section 5.4.

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, FFTF, and 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 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. Nominal values for burnup and specific power were selected as appropriate given the composition basis. The fuel plus 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.

5.2 EVALUATION OF THE ORIGEN2 RESULTS FOR THE INITIAL SET OF REPRESENTATIVE REACTORS

The results from the 11 ORIGEN2 runs were compared on the basis of relative activity (to ^{137}Cs) 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 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 reactors were selected as having isotopic inventories and ratios that would be representative of the variety of reactors and fuel combinations under consideration:

- The ATR
- The FFTF
- The N Reactor
- The TRIGA reactor

5.3 FISSION AND ACTIVATION PRODUCT INVENTORY CALCULATIONS FOR THE REPRESENTATIVE REACTOR CASES

To facilitate the fission and activation product inventory calculations needed for this TIB, case-specific cross-section libraries were created for the ORIGEN-S code using the TRITON analysis sequence of the Oak Ridge National Laboratory (ORNL) SCALE code system. TRITON automates the process of resonance self-shielding correction and sequential fuel depletion calculations to create binary libraries that can be used by ORIGEN-S. Custom libraries derived from modern (ENDF/B-VII) evaluated cross-section data were developed for the ATR, FFTF, N Reactor, and TRIGA reactor. The ORIGEN-S libraries generated by the TRITON models were used in radionuclide inventory calculations to

obtain the needed fission and activation product inventories as a function of decay time. ORIGEN-S, in conjunction with the ORIGEN-ARP preprocessing utility, is the current version of the ORIGEN code.

An ORIGEN-S input specifies the cross-section data to be used, the core composition of interest (fuel, cladding, structural materials, etc.), the specific power level (MW per unit mass of heavy metal), the irradiation duration, and the decay intervals of interest. To ensure broad representation of the fission and activation product inventory data for the four representative reactors, at least two (and in one instance three) sets of irradiation parameters were defined for each. This resulted in nine representative cases for which fission and activation product inventory data were computed using the custom-built cross-section libraries developed for each reactor. The parameters selected for the nine representative cases are listed in Table 5-2. Three cases were defined for the ATR and two were defined for the other three reactors.

Table 5-2. ORIGEN-S irradiation parameters for the nine representative reactor cases.

Case	Parameters	Basis
ATR 1	Specific power = 2,379.1 MW/MTU Irradiation time = 132.27 days Burnup = 314,684 MWd/MTU	Maximum burnup at nominal power.
ATR 2	Specific power = 8,651.2 MW/MTU Irradiation time = 36.4 days Burnup =314,904 MWd/MTU	Maximum burnup at maximum assembly power.
ATR 3	Specific power = 2,379.1 MW/MTU Irradiation time = 56 days Burnup = 133,230 MWd/MTU	Nominal burnup at nominal power.
FFTF 1	Specific power = 163.8 MW/MTHM Irradiation time = 929.4 days Burnup = 152,230 MWd/MTHM	Maximum burnup at nominal power.
FFTF 2	Specific power = 163.8 MW/MTHM Irradiation time = 488.3 days Burnup = 79,984 MWd/MTHM	Nominal burnup at nominal power.
N Reactor 1	Specific power = 10.4 MW/MTU Irradiation time = 114.2 days Burnup = 1,188 MWd/MTU	Production of weapons-grade plutonium (nominal 6% Pu-240 content) at nominal power.
N Reactor 2	Specific power = 10.4 MW/MTU Irradiation time = 285.6 days Burnup = 2,970 MWd/MTU	Production of fuel-grade plutonium (nominal 12% Pu-240 content) at nominal power.
TRIGA 1	Specific power = 15.57 MWd/MTU Irradiation time = 730.1 days Burnup = 11,368 MWd/MTU	Maximum burnup at nominal power.
TRIGA 2	Specific power = 15.57 MW/MTU Irradiation time = 115.2 days Burnup = 1994 MWd/MTU	Nominal burnup at nominal power.

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 of 10 days, 40 days, 180 days, and 1 year 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. 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 if sufficient information is available. 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, a decay time of 180 days or 1 year should be chosen.

The work activities described for each of the four decay intervals in Table 5-3 can be used for general guidance. As an alternative, if an appropriate decay time cannot be readily established, intakes and organ doses for all four decay times should be determined and the set that is most favorable to the claimant should be selected for dose reconstruction.

Table 5-3. Fission and activation product decay times for general steps in the fuel cycle.

Activity	Decay time
Reactor operations, spent fuel storage, fuel examination: activity from fuel leakage/failure or segmentation.	10 d
Dissolution of fuels from 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 DETERMINATION OF INTAKE FRACTIONS AND THE DOSIMETRICALLY SIGNIFICANT RADIONUCLIDES

This section describes the process of determining the dosimetrically significant constituents of the radionuclide inventory data from the ORIGEN-S calculations through the use of element-specific exposure fractions and radionuclide- and absorption type-specific dose conversion factors (DCFs).

6.1 INTAKE FRACTIONS

The fission and activation product content of irradiated reactor fuel and 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 parents were 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 only for 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 ^{129}I). In later years (approximately mid-1950s onward), it was rare to process fuel aged for less than 100 days.

Conversion of the radionuclide inventory data from the nine ORIGEN-S cases to intake fractions begins by extracting the fission and activation product activity values from the output files for the desired decay intervals and writing them to spreadsheets. The fission and activation product activity data are then summed as appropriate (ORIGEN-S reports them separately) and the radionuclides showing an activity value of zero at ≥ 10 days of decay are removed. ("Zero" in this content refers to the double-precision threshold for the hardware/software environment, typically less than 10^{-30}. Values below this threshold are set to zero by the code.) Noble gases and radionuclides that do not have specifically defined inhalation DCFs (for committed organ dose) in the RadToolbox database are also removed. Radionuclides that do not have DCFs include those that are inexorably associated with a given parent (e.g., ^{137m}Ba) and those with very short or very long half-lives. A complete list of radionuclides removed from the fission and activation product inventory data (because they were noble gases or did have inhalation DCFs) is provided in Table A-1 in Attachment A. Although the noble gases showing nonzero activity at ≥ 10 days decay are removed, their progeny are not.

Each of the source terms for the nine representative reactor cases was processed as described above. Release fractions from DOE Standard 1027 (DOE 1997) were then used to convert from activity at each of the four decay intervals to intake fraction. For each radionuclide and decay time in a given source term, the product of the activity and release fraction was summed and the values were restated on a normalized basis. These values are referred to as Normalized Intake Fractions (NIFs) because the product of the radionuclide activity and the release fraction was defined as an intake fraction. This means the activity units are arbitrary as long as they are the same for each case.

The release fractions used to compute the NIF values were assigned as follows: gases (e.g., tritium) 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 particulates and solids, the DOE standard 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. Hence, the smallest release fraction used for any radionuclide was 0.01, an order of magnitude greater than the minimum value in DOE Standard 1027 (DOE 1997).

The large majority of the radionuclides in the source terms considered in this document were nonvolatile particulates. Therefore, the choice of the release fractions has a limited effect on the results. The release fractions affect the ratio of the intakes of volatile and semivolatile elements relative to the nonvolatile particulates. The only volatile or semivolatile species of dosimetric importance are the radioiodines. The choice of the release fractions has no effect on the activity fractions presented in Sections 7.1 and 7.2.

The release fractions used to convert from activity to intake for each radionuclide of interest are listed in Table B-1 of Attachment B. The values are assigned on an elemental (rather than isotopic) basis.

6.2 DEVELOPMENT OF DOSIMETRICALLY SIGNIFICANT REACTOR SOURCE TERM

The calculations of the NIF values for the nine representative reactor cases resulted in source terms consisting of approximately 300 radionuclides. Committed organ doses were computed using the NIF values for each reactor case and decay interval to determine the dosimetrically important constituents across the nine representative cases.

Committed organ doses for each reactor case and decay interval were computed using inhalation DCF values from the RadToolbox software, version 2.0.0, distributed by the Oak Ridge National Laboratory (ORNL 2003). The DCFs from RadToolbox are consistent with those from International Commission on Radiological Protection (ICRP) Publication 68 (ICRP 1995). The inhalation DCFs for the collective set of radionuclides of interest across the nine reactor cases were partitioned into three solubility categories based on their ICRP-defined absorption types. The most soluble type for each nuclide was assigned to the soluble category and the least soluble types were assigned to the insoluble category. A moderately soluble category was also defined. Type F materials were always assigned to the soluble category, type M to moderately soluble, and type S to insoluble. If a radionuclide had only one absorption type defined, it was used in all three categories. Radionuclides with two absorption types were assigned as follows:

- Types F and M: Type M was assigned to the insoluble category.
- Types F and S: Type F was assigned to the moderately soluble category.
- Types M and S: Type M was assigned to the soluble category.

In some cases, absorption types were limited based on appropriate physical or chemical constraints for discharged reactor fuels. Examples include radiostrontium isotopes, for which only type F was assigned, and radioiodines, for which only the vapor form was considered.

Three tables of inhalation DCF values were compiled containing entries for each radionuclide of interest. Table C-1 in Attachment C presents the complete list of radionuclides considered in the committed organ dose calculations and the absorption types assigned for each of the three solubility categories. Three sets of committed organ doses (one for each solubility category) were computed for each reactor case and decay interval for the 27 ICRP Publication 68 (ICRP 1995) organs considered in RadToolbox (including effective and remainder). The collective set of organ dose results (across all nine cases) was then evaluated to identify all radionuclides that contributed ≥ 1% of the dose to any organ for any reactor case, decay interval, or solubility category. This resulted in a total of 36 dosimetrically important radionuclides being defined for the nine representative reactor cases. Next, the NIF values for each reactor case and decay interval were renormalized using just these dosimetrically significant radionuclides. The resulting NIF values are listed in Table D-1 in Attachment D. Renormalizing the NIF values in this manner is favorable to the claimant because it serves to increase the intake fraction values for the most important radionuclides over what the fraction would be if all radionuclides were included.

To reduce the workload for dose reconstructors, a further reduction in the number of dosimetrically significant radionuclides was performed. The Table D-1 NIF values were used to calculate effective dose for each reactor case, decay interval, and solubility category. Radionuclides that contributed ≥ 1% to effective dose for any case, decay interval, or solubility category were then identified and a subsequent renormalization of the corresponding NIF values was performed. This reduced list of 17 dosimetrically important radionuclides and the corresponding NIF values are presented in Table E-1 in Attachment E. As with the first reduction in the number of radionuclides of interest, this subsequent reduction is favorable to claimants with respect to assigned intakes.

The reduced list of dosimetrically important radionuclides (i.e., Table E-1) is used only to establish the relative intake fractions for the other components of an intake once the intake for the primary (indicator) radionuclide has been established. Intake assignments for the indicator radionuclides reflect the more comprehensive 36 radionuclide (Table D-1) list. This process is detailed in the following section.

7.0 DETERMINATION OF INTAKES OF MIXED FISSION AND ACTIVATION PRODUCTS FROM NONSPECIFIC RADIONUCLIDE ANALYSES

This section describes the calculation of activity and intake fraction values to be used for assigning intakes from gross beta or gross gamma assay data. The general process is to determine the activity fraction of a principal (indicator) radionuclide that should always be present in a given type of sample. The activity of the indicator radionuclide in relation to the total sample activity is then used to determine the corresponding intake given the specifics of the claim being evaluated. Once the indicator radionuclide intake has been established, the intake for the remaining radionuclides in the mix is assigned using the ratios established from Table E-1. Absorption types for each radionuclide are then selected to yield the most favorable dose assignment for the organ of interest. This can be done for a specific decay time or across all decay times, as appropriate for each case.

Two indicator radionuclides are considered in this document: ^{90}Sr for gross beta analyses and ^{137}Cs for gross gamma analyses.

7.1 ACTIVITY FRACTIONS FOR URINALYSES

This section describes the calculation of indicator radionuclide activity fractions relative to the total beta or gamma activity for a 24-hour urine sample following a 2-year (730-day) chronic intake.

The gross beta or gross gamma activity for a given sample depends on the radionuclide mixture at intake; the intake period (only chronic intake is considered), biokinetics (urinary excretion);

radionuclide decay and ingrowth that occur in the sample in the time between collection and counting; separations chemistry (for sample preparation), if any; and the beta or gamma radiation yields for the radioisotopes present at the time of counting. All of these factors were considered in deriving the indicator radionuclide activity fractions for urinalyses.

7.1.1 RADIONUCLIDE MIXTURE IN THE SAMPLE AT THE TIME OF COLLECTION

For each radionuclide present in the intake, the contribution to the urinary excretion is given by the product of the NIF (Table D-1 value) and the intake retention fraction (IRF) for a 24-hour urine sample. Progeny radionuclides that grow in from parent radionuclides in the intake also contribute to the excreted activity. Some of these progeny radionuclides are also present in the intake mixture, meaning there are both supported and unsupported components of their total activity in urine.

IRF values (see Table F-1 in Attachment F) were obtained from the Integrated Modules for Bioassay Analysis (IMBA) software for 24-hour urinalyses following a 2-year chronic intake period. IRFs were determined for each radionuclide in the Table D-1 mix. Absorption type F was used for all radiostrontium isotopes. Otherwise, only the least soluble absorption type was used for radionuclides for which multiple selections were possible. Limiting the activity fraction calculations to the least soluble absorption types maximizes the indicator radionuclide activity fractions because both of them can only be type F. However, this does not mean dose reconstructors should limit the absorption types used for determining assigned dose to the least soluble forms.

Following ICRP conventions, parent and progeny radionuclides present in the intake were treated independently (i.e., specific IRF values were used for each radionuclide). In addition, progeny radionuclides that would accumulate in the body from the decay of the corresponding parent radionuclides over the course of a 730-day chronic intake were also accounted for. The Dose and Risk Calculation (DCAL) program was used to determine the progeny activity in urine relative to the parent activity following a 730-day chronic intake. The specific progeny radionuclides considered and their activity fractions in urine were:

- ^{90}Y (from ^{90}Sr): 0.685
- ^{95}Nb (from ^{95}Zr): 0.401
- ^{95m}Nb (from ^{95}Zr): 0.0056
- ^{125m}Te (from ^{125}Sb): 0.0927
- ^{140}La (from ^{140}Ba): 0.517

The progeny fractions above were used in conjunction with the parent radionuclide activity values (given by the product of the intake fraction and the corresponding IRF) to compute the supported component of the progeny activity in urine. The unsupported component, likewise the product of the intake fraction and IRF, was then added to get the total.

Radioiodines were not included in the excreted activity calculations under the assumption they would be lost in the sample preparation process and thus not present at the time of counting. This is favorable to claimants because any radioiodine activity that was present would increase the measured beta or gamma activity and therefore the assigned indicator radionuclide intake.

7.1.2 DECAY AND INGROWTH AFTER COLLECTION AND IMPACT ON SAMPLE ACTIVITY

Radioactive decay and ingrowth that occur in a urine sample during the time between collection and counting affect the sample's total beta and gamma activity and thus the indicator radionuclide activity fractions. The magnitude of this effect depends on the delay time between collection and counting and the contribution to the total beta or gamma activity from short-lived radionuclides in the mixture at excretion.

The radionuclide mixture and corresponding beta and gamma activity were determined for each reactor case and decay time at 1-day intervals from 1 to 30 days. Decay and ingrowth were computed through a combination of standard radioactive decay calculations (for radionuclides without transient or secular equilibrium products) and using the ORIGEN-S code (for parent-progeny relationships). The half-lives used for the standalone decay corrections are given in Table F-2 in Attachment F.

There are limited urinalysis data available that include information on both when a sample was collected and when it was counted. However, such data are available for gross beta and gross gamma assays by Hanford, Idaho National Laboratory, and ORNL. Review of count data from these sites showed a 30-day maximum to be appropriate for delays between sample collection and counting. Limiting the maximum delay time is necessary because in most cases the indicator radionuclide contribution to the sample activity increases with increasing delay.

7.1.3 ACTIVITY FRACTIONS FOR DIFFERENT TYPES OF URINALYSES

Activity fractions for the indicator radionuclides in gross beta or gross gamma urinalyses were established for the following three types of analyses:

- Gross beta analysis that includes all fission and activation products except radioiodines and beta emitters naturally in urine, such as ^{40}K,
- 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 from the late 1940s through the early 1960s. It was assumed that the chemical recoveries for all the elements captured by separations chemistry were essentially equivalent. Therefore, explicit consideration of chemical recovery values was unnecessary: the activity fractions are independent of the chemical recoveries for processed samples unless there are significant differences in recovery for some of the elements. Because the gross beta urinalysis procedure at the Savannah River Site (SRS) is known to have differed from that assumed here, the guidance that is provided here for chemically processed gross beta samples should not be applied to cases from the SRS. Rather, guidance on the interpretation of gross beta urinalyses performed at SRS will be provided in the SRS Site Profile. The SRS-specific guidance should be used in lieu of this document.

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. However, if such information is not available it is favorable to the claimant to assume all of the activity is attributable to occupational exposure.

The sample activity data and beta and gamma radiation yield data (see Table F-3 in Attachment F) were used to compute the total beta and gamma activity for each of the three urinalyses types described above. Beta and gamma activity were computed as a function of delay time (i.e., the time between sample collection and counting) for each of the 36 radionuclide intake mixtures (four fuel decay times for each of the nine reactor cases). For processed samples, delay time between chemical separation and counting was assumed to be negligible.

The beta and gamma yield values used to convert from radionuclide activity to beta or gamma activity reflect energy thresholds that were applied to omit beta or gamma activity that likely would not have been detected. A threshold of 50 keV (average energy) was used for beta emitters, and a 100-keV threshold was applied for gamma lines. Beta and gamma yield values were combined for cases where parent/progeny pairs establish equilibrium within 24 hours. These include $^{89}Sr/^{89m}Y$, $^{99}Mo/^{99m}Tc$, $^{129m}Te/^{129}Te$, $^{136}Cs/^{136m}Ba$, $^{137}Cs/^{137m}Ba$, and $^{144}Ce/Pr$.

The indicator radionuclide activity fractions are simply the fraction of the total beta or gamma activity that is contributed by the indicator radionuclide for the reactor case, decay time, and analysis type of interest. Delay between sample collection and counting is accounted for by selecting the maximum indicator radionuclide activity fraction that occurs between 1 and 30 days delay. This ensures favorability to claimants. The analysis type determines if the beta or gamma activity is used and if some radioelements are removed to reflect preparation chemistry. The beta or gamma activity data used to establish the indicator radionuclide activity fractions for each reactor case, decay time, and analysis type are given in Tables G-1 through G-3 in Attachment G.

The applicable indicator radionuclide activity fractions for gross beta and gross gamma urinalyses are given in Tables 7-1a (for gross beta urinalysis without chemical separations), 7-1b (for gross gamma urinalysis without chemical separations), and 7-2 (for gross beta urinalysis following separations chemistry). The values in Table 7-1a and 7-1b should be applied to cases where minor chemistry might have been performed to remove natural emitters such as ^{40}K. The Table 7-2 values should be used for cases where significant chemistry was performed before counting.

The activity fractions in Tables 7-1a, 7-1b and 7-2 are used, as appropriate, to determine the indicator radionuclide activity associated with a given gross beta or gross gamma urinalysis result. The resulting indicator radionuclide activity values are then used to determine the corresponding intake. Once the indicator radionuclide intake has been established, intakes for the other radionuclides in the intake are assigned using Tables 7-3a through 7-3i below. These tables list the intake fractions for the Table E-1 radionuclide mix for each representative reactor case stated relative to the two indicator radionuclide intakes.

The intake ratios for ^{131}I in Tables 7-3a through 7-3i should not be used for sites where iodine was well controlled or if individual iodine monitoring data are available. The ^{131}I intake ratios would primarily apply to the earliest programs, before the use of ventilation and collection systems. For later years, it is reasonable to assume there were no significant intakes from iodine at most sites.

7.2 ACTIVITY FRACTIONS FOR AIR SAMPLES OR SURFACE CONTAMINATION MEASUREMENTS

Activity fractions for gross beta or gross gamma assays for air samples or surface contamination measurements were computed in essentially the same manner as described in the previous section for urinalyses. The key differences are that IRF values are not needed and delay time between sample collection and counting was considered negligible. The calculations are identical otherwise. As was the case for the urinalysis evaluations, radioiodines were not included in the activity fraction calculations for the air sample/surface contamination measurements. This is an overestimating assumption for cases in which the measured results did include radioiodines.

Tables G-4 and G-5 in Attachment G list the contributions from the Table D-1 radionuclide mixtures to gross beta and gross gamma activity for air concentration or surface contamination. The indicator radionuclide activity fractions for gross beta and gross gamma air concentration or surface contamination assays are listed in Tables 7-4a and 7-4b, respectively. Assay results are multiplied by the appropriate Table 7-4a or 7-4b value to compute the indicator radionuclide activity or activity concentration on the sample. Indicator radionuclide intake is established based on the corresponding

Table 7-1a. Gross beta activity fractions in urine for minimally processed samples.

Radionuclide	Activity fractions for gross beta urinalyses following minimal chemical processing[a]											
Sr-90 (gross beta)	Fraction of beta activity in urine: ATR 1				Fraction of beta activity in urine: ATR 2				Fraction of beta activity in urine: ATR 3			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
	2.14E-02	3.21E-02	1.18E-01	2.12E-01	1.24E-02	1.99E-02	9.19E-02	2.07E-01	1.43E-02	2.25E-02	1.02E-01	2.26E-01
	Fraction of beta activity in urine: FFTF 1				Fraction of beta activity in urine: FFTF 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	4.13E-02	4.97E-02	7.06E-02	8.27E-02	3.21E-02	4.19E-02	7.27E-02	8.99E-02				
	Fraction of beta activity in urine: N Rx 1				Fraction of beta activity in urine: N Rx 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	1.96E-02	3.00E-02	1.19E-01	2.27E-01	3.68E-02	5.30E-02	1.47E-01	2.17E-01				
	Fraction of beta activity in urine: TRIGA 1				Fraction of beta activity in urine: TRIGA 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	1.40E-01	1.68E-01	2.47E-01	2.77E-01	2.01E-02	3.07E-02	1.26E-01	2.50E-01				

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.

Table 7-1b. Gross gamma activity fractions in urine for minimally processed samples.

Radionuclide	Activity fractions for gross gamma urinalyses following minimal chemical processing[a]											
Cs-137 (gross gamma)	Fraction of gamma activity in urine: ATR 1				Fraction of gamma activity in urine: ATR 2				Fraction of gamma activity in urine: ATR 3			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
	1.49E-01	2.51E-01	3.90E-01	4.50E-01	7.66E-02	1.90E-01	4.39E-01	5.25E-01	1.03E-01	2.48E-01	5.82E-01	6.92E-01
	Fraction of gamma activity in urine: FFTF 1				Fraction of gamma activity in urine: FFTF 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	2.14E-01	2.47E-01	2.97E-01	3.38E-01	2.37E-01	3.05E-01	4.07E-01	4.66E-01				
	Fraction of gamma activity in urine: N Rx 1				Fraction of gamma activity in urine: N Rx 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	1.67E-01	3.43E-01	7.06E-01	8.18E-01	2.67E-01	4.12E-01	6.08E-01	6.78E-01				
	Fraction of gamma activity in urine: TRIGA 1				Fraction of gamma activity in urine: TRIGA 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	6.23E-01	7.05E-01	7.75E-01	8.06E-01	1.75E-01	3.72E-01	7.86E-01	9.08E-01				

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.

Table 7-2. Urine activity fractions for samples following major chemical processing.

Radionuclide	Activity fractions for gross beta urinalyses following major chemical processing											
Sr-90 (gross beta)	Fraction of beta activity in urine: ATR 1				Fraction of beta activity in urine: ATR 2				Fraction of beta activity in urine: ATR 3			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
	2.37E-02	3.68E-02	1.77E-01	4.71E-01	1.34E-02	2.18E-02	1.22E-01	4.12E-01	1.54E-02	2.47E-02	1.33E-01	4.27E-01
	Fraction of beta activity in urine: FFTF 1				Fraction of beta activity in urine: FFTF 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	1.03E-01	1.50E-01	3.75E-01	5.63E-01	6.00E-02	9.25E-02	3.07E-01	5.42E-01				
	Fraction of beta activity in urine: N Rx 1				Fraction of beta activity in urine: N Rx 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	2.17E-02	3.40E-02	1.68E-01	4.63E-01	4.41E-02	6.67E-02	2.58E-01	5.23E-01				
	Fraction of beta activity in urine: TRIGA 1				Fraction of beta activity in urine: TRIGA 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	1.98E-01	2.55E-01	4.76E-01	5.81E-01	2.17E-02	3.39E-02	1.67E-01	4.63E-01				

Table 7-3a. Associated radionuclide activity fractions for assigning intakes: ATR 1.

Table 7-3 values: ATR 1								
	Intake relative to Sr-90				Intake relative to Cs-137			
Nuclide	**10 d**	**40 d**	**180 d**	**1 y**	**10 d**	**40 d**	**180 d**	**1 y**
Co-60	2.05E-04	2.03E-04	1.95E-04	1.85E-04	2.03E-04	2.01E-04	1.93E-04	1.83E-04
Sr-89	6.72E+01	4.46E+01	6.61E+00	5.29E-01	6.66E+01	4.42E+01	6.54E+00	5.24E-01
Sr-90	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**	9.90E-01	9.90E-01	9.89E-01	9.89E-01
Y-90	1.00E+00	1.00E+00	1.00E+00	1.00E+00	9.93E-01	9.90E-01	9.90E-01	9.89E-01
Y-91	8.04E+01	5.64E+01	1.09E+01	1.23E+00	7.96E+01	5.59E+01	1.07E+01	1.21E+00
Zr-95	8.67E+01	6.28E+01	1.39E+01	1.90E+00	8.59E+01	6.22E+01	1.38E+01	1.88E+00
Nb-95	7.49E+01	7.41E+01	2.66E+01	4.08E+00	7.42E+01	7.34E+01	2.63E+01	4.04E+00
Ru-103	4.50E+01	2.65E+01	2.26E+00	8.73E-02	4.45E+01	2.63E+01	2.24E+00	8.64E-02
Ru-106	1.74E+00	1.65E+00	1.28E+00	9.17E-01	1.72E+00	1.63E+00	1.27E+00	9.07E-01
I-131	1.21E+03	9.13E+01	5.29E-04	6.33E-11	1.20E+03	9.05E+01	5.23E-04	6.26E-11
Cs-134	7.48E-01	7.29E-01	6.47E-01	5.52E-01	7.41E-01	7.22E-01	6.40E-01	5.46E-01
Cs-137	1.01E+00	1.01E+00	1.01E+00	1.01E+00	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**
Ce-141	8.62E+01	4.55E+01	2.32E+00	4.54E-02	8.54E+01	4.51E+01	2.30E+00	4.49E-02
Ce-144	2.89E+01	2.69E+01	1.93E+01	1.25E+01	2.86E+01	2.66E+01	1.91E+01	1.23E+01
Pr-143	7.60E+01	1.65E+01	1.30E-02	1.04E-06	7.52E+01	1.63E+01	1.29E-02	1.03E-06
Pm-147	2.81E+00	2.96E+00	2.74E+00	2.42E+00	2.78E+00	2.94E+00	2.71E+00	2.40E+00
Eu-154	2.36E-02	2.35E-02	2.30E-02	2.23E-02	2.34E-02	2.32E-02	2.27E-02	2.21E-02

Table 7-3b. Associated radionuclide activity fractions for assigning intakes: ATR 2.

Table 7-3 values: ATR 2								
	Intake relative to Sr-90				Intake relative to Cs-137			
Nuclide	**10 d**	**40 d**	**180 d**	**1 y**	**10 d**	**40 d**	**180 d**	**1 y**
Co-60	2.07E-04	2.05E-04	1.97E-04	1.86E-04	2.05E-04	2.03E-04	1.95E-04	1.85E-04
Sr-89	1.14E+02	7.61E+01	1.13E+01	9.02E-01	1.13E+02	7.53E+01	1.11E+01	8.92E-01
Sr-90	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**	9.91E-01	9.91E-01	9.90E-01	9.89E-01
Y-90	9.98E-01	1.00E+00	1.00E+00	1.00E+00	9.90E-01	9.91E-01	9.90E-01	9.89E-01
Y-91	1.30E+02	9.16E+01	1.76E+01	1.99E+00	1.29E+02	9.07E+01	1.74E+01	1.97E+00
Zr-95	1.34E+02	9.74E+01	2.16E+01	2.95E+00	1.33E+02	9.65E+01	2.14E+01	2.92E+00
Nb-95	6.28E+01	8.55E+01	3.94E+01	6.29E+00	6.22E+01	8.47E+01	3.91E+01	6.22E+00
Ru-103	8.53E+01	5.04E+01	4.29E+00	1.66E-01	8.45E+01	4.99E+01	4.25E+00	1.64E-01
Ru-106	1.88E+00	1.78E+00	1.38E+00	9.93E-01	1.86E+00	1.77E+00	1.37E+00	9.83E-01
I-131	4.24E+03	3.20E+02	1.85E-03	2.22E-10	4.21E+03	3.17E+02	1.83E-03	2.20E-10
Cs-134	5.40E-01	5.27E-01	4.68E-01	3.99E-01	5.36E-01	5.22E-01	4.63E-01	3.95E-01
Cs-137	1.01E+00	1.01E+00	1.01E+00	1.01E+00	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**
Ce-141	1.79E+02	9.46E+01	4.82E+00	9.44E-02	1.77E+02	9.37E+01	4.77E+00	9.33E-02
Ce-144	3.22E+01	3.00E+01	2.15E+01	1.39E+01	3.19E+01	2.97E+01	2.13E+01	1.38E+01
Pr-143	2.29E+02	4.97E+01	3.93E-02	3.13E-06	2.27E+02	4.92E+01	3.89E-02	3.10E-06
Pm-147	2.44E+00	3.02E+00	2.86E+00	2.53E+00	2.42E+00	2.99E+00	2.83E+00	2.51E+00
Eu-154	2.02E-02	2.01E-02	1.97E-02	1.91E-02	2.00E-02	1.99E-02	1.95E-02	1.89E-02

Table 7-3c. Associated radionuclide activity fractions for assigning intakes: ATR 3.

Table 7-3 values: ATR 3								
	Intake relative to Sr-90				Intake relative to Cs-137			
Nuclide	**10 d**	**40 d**	**180 d**	**1 y**	**10 d**	**40 d**	**180 d**	**1 y**
Co-60	1.97E-04	1.95E-04	1.87E-04	1.77E-04	1.95E-04	1.94E-04	1.86E-04	1.76E-04
Sr-89	1.02E+02	6.75E+01	9.99E+00	8.00E-01	1.01E+02	6.69E+01	9.90E+00	7.92E-01
Sr-90	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**	9.92E-01	9.92E-01	9.91E-01	9.90E-01
Y-90	9.97E-01	1.00E+00	1.00E+00	1.00E+00	9.89E-01	9.92E-01	9.91E-01	9.91E-01
Y-91	1.16E+02	8.16E+01	1.57E+01	1.77E+00	1.15E+02	8.09E+01	1.55E+01	1.76E+00
Zr-95	1.22E+02	8.84E+01	1.96E+01	2.68E+00	1.21E+02	8.77E+01	1.94E+01	2.65E+00
Nb-95	7.09E+01	8.52E+01	3.63E+01	5.72E+00	7.03E+01	8.46E+01	3.60E+01	5.66E+00
Ru-103	7.35E+01	4.34E+01	3.70E+00	1.43E-01	7.29E+01	4.30E+01	3.66E+00	1.41E-01
Ru-106	1.82E+00	1.72E+00	1.34E+00	9.61E-01	1.81E+00	1.71E+00	1.33E+00	9.52E-01
I-131	2.86E+03	2.16E+02	1.25E-03	1.50E-10	2.84E+03	2.14E+02	1.24E-03	1.48E-10
Cs-134	2.52E-01	2.46E-01	2.18E-01	1.86E-01	2.50E-01	2.44E-01	2.16E-01	1.85E-01
Cs-137	1.01E+00	1.01E+00	1.01E+00	1.01E+00	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**
Ce-141	1.51E+02	7.99E+01	4.07E+00	7.98E-02	1.50E+02	7.93E+01	4.04E+00	7.90E-02
Ce-144	3.15E+01	2.93E+01	2.10E+01	1.36E+01	3.12E+01	2.91E+01	2.09E+01	1.35E+01
Pr-143	1.71E+02	3.70E+01	2.92E-02	2.33E-06	1.69E+02	3.67E+01	2.90E-02	2.31E-06
Pm-147	3.13E+00	3.56E+00	3.33E+00	2.95E+00	3.11E+00	3.53E+00	3.30E+00	2.92E+00
Eu-154	8.34E-03	8.30E-03	8.13E-03	7.90E-03	8.28E-03	8.24E-03	8.06E-03	7.82E-03

Table 7-3d. Associated radionuclide activity fractions for assigning intakes: FFTF 1.

Table 7-3 values: FFTF 1								
	Intake relative to Sr-90				Intake relative to Cs-137			
Nuclide	**10 d**	**40 d**	**180 d**	**1 y**	**10 d**	**40 d**	**180 d**	**1 y**
Co-60	5.79E-02	5.74E-02	5.51E-02	5.22E-02	2.01E-02	1.99E-02	1.91E-02	1.81E-02
Sr-89	1.21E+01	8.04E+00	1.19E+00	9.52E-02	4.20E+00	2.78E+00	4.12E-01	3.29E-02
Sr-90	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**	3.46E-01	3.46E-01	3.46E-01	3.46E-01
Y-90	1.01E+00	1.00E+00	1.00E+00	1.00E+00	3.48E-01	3.46E-01	3.46E-01	3.46E-01
Y-91	1.81E+01	1.27E+01	2.44E+00	2.76E-01	6.25E+00	4.39E+00	8.44E-01	9.54E-02
Zr-95	3.26E+01	2.36E+01	5.24E+00	7.16E-01	1.13E+01	8.18E+00	1.81E+00	2.48E-01
Nb-95	3.60E+01	3.22E+01	1.03E+01	1.55E+00	1.25E+01	1.12E+01	3.56E+00	5.35E-01
Ru-103	4.51E+01	2.66E+01	2.27E+00	8.76E-02	1.56E+01	9.21E+00	7.85E-01	3.03E-02
Ru-106	2.64E+01	2.50E+01	1.95E+01	1.40E+01	9.14E+00	8.66E+00	6.74E+00	4.84E+00
I-131	6.25E+02	4.68E+01	2.63E-04	3.04E-11	2.16E+02	1.62E+01	9.11E-05	1.05E-11
Cs-134	3.31E+00	3.23E+00	2.87E+00	2.45E+00	1.15E+00	1.12E+00	9.92E-01	8.47E-01
Cs-137	2.89E+00	2.89E+00	2.89E+00	2.89E+00	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**
Ce-141	3.22E+01	1.70E+01	8.67E-01	1.70E-02	1.11E+01	5.89E+00	3.00E-01	5.88E-03
Ce-144	2.57E+01	2.39E+01	1.72E+01	1.11E+01	8.89E+00	8.28E+00	5.94E+00	3.83E+00
Pr-143	2.22E+01	4.81E+00	3.81E-03	3.03E-07	7.69E+00	1.67E+00	1.32E-03	1.05E-07
Pm-147	6.25E+00	6.21E+00	5.68E+00	5.03E+00	2.16E+00	2.15E+00	1.96E+00	1.74E+00
Eu-154	2.09E-01	2.08E-01	2.04E-01	1.98E-01	7.25E-02	7.21E-02	7.06E-02	6.85E-02

Table 7-3e. Associated radionuclide activity fractions for assigning intakes: FFTF 2.

Table 7-3 values: FFTF 2								
	Intake relative to Sr-90				Intake relative to Cs-137			
Nuclide	**10 d**	**40 d**	**180 d**	**1 y**	**10 d**	**40 d**	**180 d**	**1 y**
Co-60	5.17E-02	5.12E-02	4.92E-02	4.66E-02	1.80E-02	1.78E-02	1.71E-02	1.62E-02
Sr-89	2.30E+01	1.52E+01	2.25E+00	1.80E-01	7.98E+00	5.30E+00	7.84E-01	6.26E-02
Sr-90	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**	3.48E-01	3.48E-01	3.48E-01	3.47E-01
Y-90	1.00E+00	1.00E+00	1.00E+00	1.00E+00	3.49E-01	3.48E-01	3.48E-01	3.47E-01
Y-91	3.41E+01	2.39E+01	4.60E+00	5.20E-01	1.18E+01	8.32E+00	1.60E+00	1.81E-01
Zr-95	6.10E+01	4.41E+01	9.79E+00	1.34E+00	2.12E+01	1.53E+01	3.40E+00	4.64E-01
Nb-95	6.68E+01	5.99E+01	1.92E+01	2.89E+00	2.32E+01	2.08E+01	6.68E+00	1.00E+00
Ru-103	8.25E+01	4.87E+01	4.15E+00	1.60E-01	2.87E+01	1.69E+01	1.44E+00	5.56E-02
Ru-106	3.57E+01	3.38E+01	2.63E+01	1.89E+01	1.24E+01	1.18E+01	9.15E+00	6.56E+00
I-131	1.18E+03	8.87E+01	4.99E-04	5.75E-11	4.11E+02	3.08E+01	1.73E-04	2.00E-11
Cs-134	1.75E+00	1.71E+00	1.52E+00	1.30E+00	6.10E-01	5.95E-01	5.28E-01	4.50E-01
Cs-137	2.88E+00	2.88E+00	2.88E+00	2.88E+00	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**
Ce-141	6.05E+01	3.20E+01	1.63E+00	3.20E-02	2.10E+01	1.11E+01	5.67E-01	1.11E-02
Ce-144	3.72E+01	3.47E+01	2.49E+01	1.61E+01	1.29E+01	1.20E+01	8.65E+00	5.58E+00
Pr-143	4.13E+01	8.95E+00	7.08E-03	5.64E-07	1.44E+01	3.11E+00	2.46E-03	1.96E-07
Pm-147	7.82E+00	7.82E+00	7.15E+00	6.33E+00	2.72E+00	2.72E+00	2.49E+00	2.20E+00
Eu-154	1.37E-01	1.36E-01	1.33E-01	1.29E-01	4.75E-02	4.72E-02	4.62E-02	4.49E-02

Table 7-3f. Associated radionuclide activity fractions for assigning intakes: N Reactor 1.

Table 7-3 values: N Reactor 1								
	Intake relative to Sr-90				Intake relative to Cs-137			
Nuclide	**10 d**	**40 d**	**180 d**	**1 y**	**10 d**	**40 d**	**180 d**	**1 y**
Co-60	3.01E-03	2.98E-03	2.86E-03	2.71E-03	2.73E-03	2.70E-03	2.59E-03	2.46E-03
Sr-89	7.29E+01	4.84E+01	7.17E+00	5.74E-01	6.60E+01	4.39E+01	6.49E+00	5.20E-01
Sr-90	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**	9.07E-01	9.07E-01	9.06E-01	9.05E-01
Y-90	1.00E+00	1.00E+00	1.00E+00	1.00E+00	9.06E-01	9.07E-01	9.06E-01	9.05E-01
Y-91	8.78E+01	6.17E+01	1.19E+01	1.34E+00	7.96E+01	5.59E+01	1.07E+01	1.21E+00
Zr-95	9.90E+01	7.17E+01	1.59E+01	2.17E+00	8.97E+01	6.50E+01	1.44E+01	1.97E+00
Nb-95	8.10E+01	8.21E+01	3.02E+01	4.66E+00	7.34E+01	7.45E+01	2.74E+01	4.22E+00
Ru-103	6.58E+01	3.88E+01	3.31E+00	1.28E-01	5.96E+01	3.52E+01	3.00E+00	1.16E-01
Ru-106	4.07E+00	3.86E+00	3.00E+00	2.15E+00	3.69E+00	3.50E+00	2.72E+00	1.95E+00
I-131	1.62E+03	1.22E+02	7.07E-04	8.48E-11	1.47E+03	1.11E+02	6.41E-04	7.67E-11
Cs-134	1.15E-01	1.12E-01	9.95E-02	8.50E-02	1.04E-01	1.02E-01	9.02E-02	7.69E-02
Cs-137	1.10E+00	1.10E+00	1.10E+00	1.10E+00	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**
Ce-141	1.04E+02	5.51E+01	2.81E+00	5.49E-02	9.44E+01	4.99E+01	2.54E+00	4.97E-02
Ce-144	3.09E+01	2.87E+01	2.06E+01	1.33E+01	2.80E+01	2.60E+01	1.87E+01	1.21E+01
Pr-143	8.95E+01	1.94E+01	1.54E-02	1.22E-06	8.12E+01	1.76E+01	1.39E-02	1.11E-06
Pm-147	3.90E+00	4.10E+00	3.79E+00	3.36E+00	3.54E+00	3.72E+00	3.43E+00	3.04E+00
Eu-154	5.48E-03	5.45E-03	5.34E-03	5.19E-03	4.97E-03	4.94E-03	4.83E-03	4.69E-03

Table 7-3g. Associated radionuclide activity fractions for assigning intakes: N Reactor 2.

Table 7-3 values: N Reactor 2								
	Intake relative to Sr-90				Intake relative to Cs-137			
Nuclide	**10 d**	**40 d**	**180 d**	**1 y**	**10 d**	**40 d**	**180 d**	**1 y**
Co-60	3.10E-03	3.07E-03	2.95E-03	2.79E-03	2.61E-03	2.59E-03	2.48E-03	2.35E-03
Sr-89	3.46E+01	2.30E+01	3.41E+00	2.73E-01	2.92E+01	1.94E+01	2.87E+00	2.29E-01
Sr-90	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**	8.42E-01	8.42E-01	8.42E-01	8.41E-01
Y-90	1.00E+00	1.00E+00	1.00E+00	1.00E+00	8.45E-01	8.42E-01	8.42E-01	8.41E-01
Y-91	4.47E+01	3.14E+01	6.03E+00	6.82E-01	3.76E+01	2.64E+01	5.08E+00	5.74E-01
Zr-95	5.49E+01	3.98E+01	8.82E+00	1.21E+00	4.63E+01	3.35E+01	7.42E+00	1.01E+00
Nb-95	5.85E+01	5.31E+01	1.73E+01	2.60E+00	4.93E+01	4.47E+01	1.45E+01	2.18E+00
Ru-103	3.78E+01	2.23E+01	1.90E+00	7.34E-02	3.18E+01	1.88E+01	1.60E+00	6.17E-02
Ru-106	5.53E+00	5.24E+00	4.07E+00	2.92E+00	4.66E+00	4.41E+00	3.43E+00	2.46E+00
I-131	7.31E+02	5.52E+01	3.19E-04	3.82E-11	6.16E+02	4.64E+01	2.69E-04	3.22E-11
Cs-134	3.12E-01	3.04E-01	2.70E-01	2.31E-01	2.63E-01	2.56E-01	2.27E-01	1.94E-01
Cs-137	1.19E+00	1.19E+00	1.19E+00	1.19E+00	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**
Ce-141	4.80E+01	2.54E+01	1.29E+00	2.53E-02	4.04E+01	2.14E+01	1.09E+00	2.13E-02
Ce-144	2.64E+01	2.46E+01	1.77E+01	1.14E+01	2.22E+01	2.07E+01	1.49E+01	9.59E+00
Pr-143	3.68E+01	7.97E+00	6.31E-03	5.03E-07	3.10E+01	6.71E+00	5.31E-03	4.23E-07
Pm-147	3.87E+00	3.91E+00	3.59E+00	3.18E+00	3.26E+00	3.30E+00	3.02E+00	2.67E+00
Eu-154	1.51E-02	1.50E-02	1.47E-02	1.43E-02	1.27E-02	1.27E-02	1.24E-02	1.20E-02

Table 7-3h. Associated radionuclide activity fractions for assigning intakes: TRIGA 1.

Table 7-3 values: TRIGA 1								
	Intake relative to Sr-90				Intake relative to Cs-137			
Nuclide	**10 d**	**40 d**	**180 d**	**1 y**	**10 d**	**40 d**	**180 d**	**1 y**
Co-60	3.21E-01	3.18E-01	3.06E-01	2.89E-01	3.13E-01	3.11E-01	2.98E-01	2.82E-01
Sr-89	5.15E+00	3.42E+00	5.06E-01	4.05E-02	5.02E+00	3.33E+00	4.93E-01	3.95E-02
Sr-90	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**	9.76E-01	9.76E-01	9.75E-01	9.75E-01
Y-90	1.00E+00	1.00E+00	1.00E+00	1.00E+00	9.77E-01	9.76E-01	9.76E-01	9.75E-01
Y-91	6.52E+00	4.58E+00	8.80E-01	9.95E-02	6.36E+00	4.47E+00	8.59E-01	9.70E-02
Zr-95	7.89E+00	5.71E+00	1.27E+00	1.73E-01	7.70E+00	5.57E+00	1.23E+00	1.69E-01
Nb-95	8.70E+00	7.79E+00	2.49E+00	3.73E-01	8.49E+00	7.60E+00	2.43E+00	3.64E-01
Ru-103	3.38E+00	1.99E+00	1.70E-01	6.56E-03	3.30E+00	1.95E+00	1.66E-01	6.40E-03
Ru-106	6.18E-01	5.86E-01	4.56E-01	3.26E-01	6.03E-01	5.71E-01	4.44E-01	3.18E-01
I-131	8.05E+01	6.07E+00	3.51E-05	4.21E-12	7.85E+01	5.92E+00	3.43E-05	4.10E-12
Cs-134	1.25E-01	1.22E-01	1.08E-01	9.22E-02	1.22E-01	1.19E-01	1.05E-01	8.99E-02
Cs-137	1.03E+00	1.03E+00	1.03E+00	1.03E+00	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**
Ce-141	6.01E+00	3.18E+00	1.62E-01	3.17E-03	5.86E+00	3.10E+00	1.58E-01	3.09E-03
Ce-144	6.76E+00	6.29E+00	4.52E+00	2.92E+00	6.59E+00	6.14E+00	4.41E+00	2.84E+00
Pr-143	4.90E+00	1.06E+00	8.41E-04	6.70E-08	4.78E+00	1.04E+00	8.20E-04	6.53E-08
Pm-147	2.15E+00	2.13E+00	1.94E+00	1.72E+00	2.10E+00	2.08E+00	1.90E+00	1.68E+00
Eu-154	7.10E-03	7.06E-03	6.92E-03	6.72E-03	6.92E-03	6.89E-03	6.75E-03	6.55E-03

Table 7-3i. Associated radionuclide activity fractions for assigning intakes: TRIGA 2.

Table 7-3 values: TRIGA 2								
	Intake relative to Sr-90				Intake relative to Cs-137			
Nuclide	**10 d**	**40 d**	**180 d**	**1 y**	**10 d**	**40 d**	**180 d**	**1 y**
Co-60	3.88E-01	3.84E-01	3.69E-01	3.49E-01	3.84E-01	3.81E-01	3.65E-01	3.46E-01
Sr-89	7.33E+01	4.87E+01	7.21E+00	5.78E-01	7.27E+01	4.83E+01	7.14E+00	5.72E-01
Sr-90	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**	9.91E-01	9.91E-01	9.90E-01	9.90E-01
Y-90	1.00E+00	1.00E+00	1.00E+00	1.00E+00	9.94E-01	9.91E-01	9.91E-01	9.90E-01
Y-91	8.69E+01	6.11E+01	1.17E+01	1.33E+00	8.61E+01	6.05E+01	1.16E+01	1.31E+00
Zr-95	9.89E+01	7.16E+01	1.59E+01	2.17E+00	9.80E+01	7.09E+01	1.57E+01	2.15E+00
Nb-95	8.07E+01	8.19E+01	3.02E+01	4.65E+00	8.00E+01	8.12E+01	2.99E+01	4.60E+00
Ru-103	4.98E+01	2.94E+01	2.50E+00	9.68E-02	4.94E+01	2.91E+01	2.48E+00	9.57E-02
Ru-106	1.75E+00	1.66E+00	1.29E+00	9.23E-01	1.73E+00	1.64E+00	1.27E+00	9.13E-01
I-131	1.41E+03	1.06E+02	6.15E-04	7.37E-11	1.40E+03	1.05E+02	6.09E-04	7.29E-11
Cs-134	8.51E-03	8.30E-03	7.37E-03	6.29E-03	8.44E-03	8.22E-03	7.29E-03	6.22E-03
Cs-137	1.01E+00	1.01E+00	1.01E+00	1.01E+00	**1.00E+00**	**1.00E+00**	**1.00E+00**	**1.00E+00**
Ce-141	9.71E+01	5.13E+01	2.61E+00	5.12E-02	9.62E+01	5.08E+01	2.59E+00	5.07E-02
Ce-144	2.94E+01	2.74E+01	1.97E+01	1.27E+01	2.92E+01	2.72E+01	1.95E+01	1.26E+01
Pr-143	8.88E+01	1.92E+01	1.52E-02	1.21E-06	8.80E+01	1.91E+01	1.51E-02	1.20E-06
Pm-147	3.72E+00	3.91E+00	3.61E+00	3.20E+00	3.69E+00	3.87E+00	3.57E+00	3.16E+00
Eu-154	3.72E-04	3.70E-04	3.62E-04	3.52E-04	3.69E-04	3.67E-04	3.59E-04	3.49E-04

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

Radionuclide	Activity fractions for gross beta air or swipe samples[a]											
Sr-90 (gross beta)	Fraction of beta activity on sample: ATR 1				Fraction of beta activity on sample: ATR 2				Fraction of beta activity on sample: ATR 3			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
	1.37E-03	2.66E-03	9.74E-03	2.05E-02	6.29E-04	1.51E-03	7.51E-03	1.80E-02	7.82E-04	1.75E-03	7.99E-03	1.85E-02
	Fraction of beta activity on sample: FFTF 1				Fraction of beta activity on sample: FFTF 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	2.62E-03	4.12E-03	8.58E-03	1.33E-02	1.58E-03	2.61E-03	6.11E-03	9.88E-03				
	Fraction of beta activity on sample: N Rx 1				Fraction of beta activity on sample: N Rx 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	1.15E-03	2.28E-03	8.64E-03	1.82E-02	2.19E-03	3.86E-03	1.12E-02	2.05E-02				
	Fraction of beta activity on sample: TRIGA 1				Fraction of beta activity on sample: TRIGA 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	1.35E-02	2.09E-02	4.37E-02	6.73E-02	1.22E-03	2.41E-03	9.16E-03	1.97E-02				

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.

Table 7-4b. Activity ratios for use if gross gamma results are reported for air samples and workplace measurements.

Radionuclide	Activity fractions for gross gamma air or swipe samples[a]											
Cs-137 (gross gamma)	Fraction of gamma activity on sample: ATR 1				Fraction of gamma activity on sample: ATR 2				Fraction of gamma activity on sample: ATR 3			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
	1.89E-03	3.91E-03	1.83E-02	8.58E-02	8.35E-04	2.22E-03	1.24E-02	6.57E-02	1.06E-03	2.58E-03	1.36E-02	7.31E-02
	Fraction of gamma activity on sample: FFTF 1				Fraction of gamma activity on sample: FFTF 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	9.97E-03	1.82E-02	6.31E-02	1.37E-01	5.82E-03	1.09E-02	4.39E-02	1.22E-01				
	Fraction of gamma activity on sample: N Rx 1				Fraction of gamma activity on sample: N Rx 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	1.71E-03	3.62E-03	1.77E-02	8.97E-02	3.57E-03	6.90E-03	3.10E-02	1.28E-01				
	Fraction of gamma activity on sample: TRIGA 1				Fraction of gamma activity on sample: TRIGA 2							
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y				
	2.24E-02	4.01E-02	1.35E-01	3.15E-01	1.67E-03	3.47E-03	1.62E-02	7.99E-02				

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.

air concentration and assumed breathing rate. Table 7-3 is used to assign intakes for the other radionuclides in the mix once the indicator radionuclide intake is established.

8.0 MIXED FISSION AND ACTIVATION PRODUCT DOSE ASSIGNMENT SUMMARY

This document provides activity and intake fraction values for use in interpreting gross beta or gross gamma activity measurements related to reactor source terms. The values were derived using representative parameters and models; site- and claim-specific information, if available, should take precedence over the guidance in this document. This includes site-specific models and guidance in Site Profiles.

Individual monitoring data should be used to determine doses from specific radionuclides if available. Site Profile and claim-specific information should be reviewed to determine if chronic iodine intakes were feasible. 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. The methods used in this document to compute intake ratios resulted in much larger values for ^{131}I at short decay times than for particulates. However, particulates were more significant sources of internal exposure at most sites.

To apply this document, the activity fraction data from Table 7-1a/b, 7-2, or 7-4a/b is applied as appropriate to determine the indicator radionuclide activity values associated with a given gross beta or gross gamma assay result. Selection of decay times should be based on the worker categories in Table 5-3 and consideration of site and claim information. If a specific decay time or times cannot be established, the dose evaluation will need to consider all four possibilities. Similarly, in most instances all nine of the representative reactor cases will need to be considered. If the dose reconstructor does not know if significant separations chemistry was performed prior to a gross beta urinalysis, it is favorable to the claimant to assume significant chemistry was performed. The activity fraction data from Table 7-2 should therefore be used for those cases. If both gross beta and gross gamma results are available for a given assay then the more sensitive measure [i.e., that with the lowest minimum detectable activity (MDA)] should be used. If the dose reconstructor does not know if a given assay result was a gross beta or gross gamma count (and both are realistic choices for the site under consideration) then both should be evaluated to determine which yields the larger assigned dose.

Corresponding radionuclide intakes are determined (using IMBA and activity fraction ratios) once the appropriate number of indicator radionuclide activity values are established. Table 7-3 is then used to establish the corresponding intakes for the other radionuclides in the mix. The goal is to find the combination of reactor, decay time, and absorption type assumptions that result in the most favorable dose being assigned. Recall that ^{89}Sr and ^{90}Sr are always treated as type F and ^{131}I is always type V when applying this document.

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

In general, the information in this document is intended for application to gross beta or gross gamma assays. In some cases, it could be applied to a radionuclide-specific assay such as a ^{90}Sr bioassay or an *in vivo* count as long as the assumptions and bases employed in developing the associated radionuclide activity fractions in Tables 7-3a through 7-3i are considered and deemed appropriate. Such considerations would be site or case specific and are outside the scope of the guidance in this document for gross beta and gross gamma assays.

If a site used a specific chemical recovery fraction to state gross beta counting results in terms of activity for a specific radionuclide, the fraction should be multiplied by the reported result to convert it

back to gross beta activity. For example, gross beta results at Hanford were divided by a chemical recovery for ^{90}Sr and reported as ^{90}Sr-equivalent activity. The ^{90}Sr-equivalent result should be multiplied by the recovery fraction to get back to gross beta activity before applying the methods in this document. However, if the dose reconstructor does not know the recovery fraction, then it is favorable to the claimant to treat the result as gross beta activity.

8.1 INTERACTIVE RADIOEPIDEMIOLOGICAL PROGRAM (IREP) DISTRIBUTION ASSIGNMENT

The indicator radionuclide doses calculated as above 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 geometric standard deviation equal to 3 would be assigned (ORAUT 2014). For missed dose, a triangular distribution is assigned (ORAUT 2014).

Given the conservatism in how they are assigned, the doses from the associated radionuclides should be entered as constants in IREP. Measures in this document to ensure favorability to claimants in relation to assigned doses include the following:

- Not accounting for reactor power history in the calculation of radionuclide inventories maximizes the short-lived radionuclide content in relation to the long-lived content.
- Increasing the release fractions for particulates by an order of magnitude higher than those in DOE Standard 1027 (DOE 1997) increases the intake fractions for those radionuclides.
- Renormalizing the Table D-1 and E-1 mixtures to reduce the number of radionuclides considered further increases the intake fractions for the dosimetrically important radionuclides.
- Computing indicator radionuclide activity fractions using only the most insoluble form of the other radionuclides in the mix maximizes those values and, thus, the assigned intakes.
- The dose assigned for a given claim is the maximum obtained by considering all possible absorption types for all nine representative reactor cases and in some instances across all four decay times (if there is not sufficient information available to select a specific decay time or times).

In addition to ensuring favorability to the claimants, the ranges of irradiation parameters (specific power, exposure time, etc.) to establish the fission and activation product inventory data for the representative reactor cases in conjunction with the conservatisms and the maximizing approach discussed above were used to encompass uncertainties associated with dose assignments per this document. Such uncertainties 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.

Doses assigned per this document are likely to represent upper bounds. Therefore, it would be reasonable to assign them as such (i.e., constant distributions) and to ignore any further uncertainty from metabolic models, etc.

When intakes are determined using urinalysis data and other bioassay data are available (e.g., whole-body counts), the reasonableness of the intakes can be checked. Predicted body burdens of gamma-emitting radionuclides based on the urine-based intakes for those that were also measured in whole body counts or would have been reported if they had been detected, and had well-defined detection levels, can be compared to levels that were measured in the whole-body counts or to MDAs, as appropriate. If predicted versus measured body burdens do not agree well, users should discuss the proper course of action with their Group Manager or the Principal Internal Dosimetrist. In general, these instances would be an issue for best-estimate cases or those in which the assigned dose from application of this document could have a significant effect on the probability of causation.

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 (SRDB).

[1] Burns Jr., Robert E., CHP. ORAU Team. Senior Health Physicist. March 2007.
Ten days of decay was selected to ensure short-lived nuclides were considered. The data were evaluated relative to ^{137}Cs as a convenience. This is a common practice because ^{137}Cs is a prominent fission product and is easily quantified.

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ATTACHMENT A
RADIONUCLIDES FROM REACTOR SOURCE TERMS NOT INCLUDED IN DOSE ANALYSES

The following radionuclides are those identified in the collective set of fission and activation product inventory data for the nine representative reactor cases for which dose conversion factors were not readily available:

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

Radionuclide	Radionuclide
Al-28	Cd-111m
Ca-48	Cd-116
Sc-45m	In-114
V-50	Sb-118
Cu-66	Te-118
Ge-71m	Te-119
Ge-73m	Te-128
Ge-76	Ba-136m
As-75m	Ba-137m
Se-72	La-133
Se-77m	Ce-139m
Se-82	Pr-140
Y-87m	Pr-144m
Y-89m	Nd-140
Nb-91	Nd-144
Nb-91m	Nd-150
Nb-92	Sm-148
Nb-92m	Gd-153m
Nb-97m	Ho-163
Mo-100	Er-167m
Rh-105m	Tm-165
Rh-106	Tm-168
Pd-111	Hf-174
Pd-111m	W-180
Pd-112	W-183m
Ag-107m	Os-186
Ag-108	Ir-191m
Ag-109m	Pt-190
Ag-110	Tl-206
Ag-111m	Bi-208
Ag-113	

ATTACHMENT B
RELEASE FRACTIONS

These fractions, adapted from DOE Standard 1027 (DOE 1997), were used to calculate intake fractions from the fission and activation product inventory data for the nine representative reactor cases.

Table B-1. Elemental exposure fractions.

Element	Release fraction		Element	Release fraction
Ag	0.01		Na	0.50
As	0.50		Nb	0.01
Au	0.01		Nd	0.01
Ba	0.01		Ni	0.01
Be	0.01		Os	0.01
Bi	0.01		P	0.50
Br	0.50		Pb	0.01
C	0.01		Pd	0.01
Ca	0.01		Pm	0.01
Cd	0.01		Po	0.01
Ce	0.01		Pr	0.01
Cl	1		Pt	0.01
Co	0.01		Rb	0.01
Cr	0.01		Re	0.01
Cs	0.01		Rh	0.01
Cu	0.01		Ru	0.01
Dy	0.01		S	0.50
Er	0.01		Sb	0.01
Eu	0.01		Sc	0.01
Fe	0.01		Se	0.01
Ga	0.01		Si	0.01
Gd	0.01		Sm	0.01
Ge	0.01		Sn	0.01
H	1		Sr	0.01
Hf	0.01		Ta	0.01
Hg	0.01		Tb	0.01
Ho	0.01		Tc	0.01
I	0.50		Te	0.01
In	0.01		Tl	0.01
Ir	0.01		Tm	0.01
K	0.50		V	0.01
La	0.01		W	0.01
Lu	0.01		Y	0.01
Mg	0.01		Yb	0.01
Mn	0.01		Zn	0.01
Mo	0.01		Zr	0.01

ATTACHMENT C
RADIONUCLIDES CONSIDERED IN COMMITTED ORGAN DOSE CALCULATIONS AND THE ABSORPTION TYPES ASSIGNED TO EACH SOLUBILITY CATEGORY

Table C-1. Radionuclides considered in committed organ dose calculations and the absorption types assigned to each solubility category.

	Absorption type assigned		
Radionuclide	**Soluble**	**Mod. soluble**	**Insoluble**
H-3	V (HTO)	V (HTO)	V (HTO)
Be-7	M	M	S
Be-10	M	M	S
C-14	V (CO_2)	V (CO_2)	V (CO_2)
Na-22	F	F	F
Na-24	F	F	F
Mg-28	F	M	M
Si-32	F	M	S
P-32	F	M	M
P-33	F	M	M
S-35	F	M	M
Cl-36	F	M	M
K-40	F	F	F
K-42	F	F	F
K-43	F	F	F
Ca-41	M	M	M
Ca-45	M	M	M
Ca-47	M	M	M
Sc-46	S	S	S
Sc-47	S	S	S
Sc-48	S	S	S
V-49	F	M	M
Cr-51	F	M	S
Mn-54	F	M	M
Fe-55	F	M	M
Fe-59	F	M	M
Co-58	M	M	S
Co-58m	M	M	S
Co-60	M	M	S
Ni-59	F	M	M
Ni-63	F	M	M
Ni-66	F	M	M
Cu-64	F	M	S
Cu-67	F	M	S
Zn-65	S	S	S
Zn-69	S	S	S
Zn-69m	S	S	S
Zn-71m	S	S	S
Zn-72	S	S	S
Ga-66	F	M	M
Ga-67	F	M	M
Ga-68	F	M	M
Ga-72	F	M	M
Ga-73	F	M	M
Ge-68	F	M	M
Ge-69	F	M	M

ATTACHMENT C
RADIONUCLIDES CONSIDERED IN COMMITTED ORGAN DOSE CALCULATIONS AND THE ABSORPTION TYPES ASSIGNED TO EACH SOLUBILITY CATEGORY (continued)

Radionuclide	Absorption type assigned		
	Soluble	Mod. soluble	Insoluble
Ge-71	F	M	M
Ge-77	F	M	M
As-71	M	M	M
As-72	M	M	M
As-73	M	M	M
As-74	M	M	M
As-76	M	M	M
As-77	M	M	M
Se-73	F	M	M
Se-75	F	M	M
Se-79	F	M	M
Br-77	F	M	M
Br-80	F	M	M
Br-80m	F	M	M
Br-82	F	M	M
Rb-81	F	F	F
Rb-83	F	F	F
Rb-84	F	F	F
Rb-86	F	F	F
Rb-87	F	F	F
Sr-83	F	F	F
Sr-85	F	F	F
Sr-89	F	F	F
Sr-90	F	F	F
Sr-91	F	F	F
Y-87	M	M	S
Y-88	M	M	S
Y-90	M	M	S
Y-91	M	M	S
Y-91m	M	M	S
Y-92	M	M	S
Y-93	M	M	S
Zr-88	F	M	S
Zr-89	F	M	S
Zr-93	F	M	S
Zr-95	F	M	S
Zr-97	F	M	S
Nb-90	M	M	S
Nb-93m	M	M	S
Nb-94	M	M	S
Nb-95	M	M	S
Nb-95m	M	M	S
Nb-96	M	M	S
Nb-97	M	M	S
Mo-93	F	F	S
Mo-93m	F	F	S
Mo-99	F	F	S
Tc-95	F	M	M
Tc-95m	F	M	M

ATTACHMENT C
RADIONUCLIDES CONSIDERED IN COMMITTED ORGAN DOSE CALCULATIONS AND THE ABSORPTION TYPES ASSIGNED TO EACH SOLUBILITY CATEGORY (continued)

Radionuclide	Absorption type assigned		
	Soluble	Mod. soluble	Insoluble
Tc-96	F	M	M
Tc-97	F	M	M
Tc-97m	F	M	M
Tc-98	F	M	M
Tc-99	F	M	M
Tc-99m	F	M	M
Ru-97	F	M	S
Ru-103	F	M	S
Ru-105	F	M	S
Ru-106	F	M	S
Rh-99	F	M	S
Rh-99m	F	M	S
Rh-101	F	M	S
Rh-101m	F	M	S
Rh-102	F	M	S
Rh-102m	F	M	S
Rh-103m	F	M	S
Rh-105	F	M	S
Pd-101	F	M	S
Pd-103	F	M	S
Pd-107	F	M	S
Pd-109	F	M	S
Ag-105	F	M	S
Ag-106m	F	M	S
Ag-108m	F	M	S
Ag-110m	F	M	S
Ag-111	F	M	S
Ag-112	F	M	S
Cd-107	F	M	S
Cd-109	F	M	S
Cd-113	F	M	S
Cd-113m	F	M	S
Cd-115	F	M	S
Cd-115m	F	M	S
Cd-117m	F	M	S
In-109	F	M	M
In-111	F	M	M
In-113m	F	M	M
In-114m	F	M	M
In-115	F	M	M
In-115m	F	M	M
In-117	F	M	M
In-117m	F	M	M
Sn-113	F	M	M
Sn-117m	F	M	M
Sn-119m	F	M	M
Sn-121	F	M	M
Sn-121m	F	M	M
Sn-123	F	M	M

ATTACHMENT C
RADIONUCLIDES CONSIDERED IN COMMITTED ORGAN DOSE CALCULATIONS AND THE ABSORPTION TYPES ASSIGNED TO EACH SOLUBILITY CATEGORY (continued)

Radionuclide	Absorption type assigned		
	Soluble	Mod. soluble	Insoluble
Sn-125	F	M	M
Sn-126	F	M	M
Sb-118m	F	M	M
Sb-119	F	M	M
Sb-120m	F	M	M
Sb-122	F	M	M
Sb-124	F	M	M
Sb-125	F	M	M
Sb-126	F	M	M
Sb-126m	F	M	M
Sb-127	F	M	M
Sb-128	F	M	M
Sb-129	F	M	M
Te-121	F	M	M
Te-121m	F	M	M
Te-123	F	M	M
Te-123m	F	M	M
Te-125m	F	M	M
Te-127	F	M	M
Te-127m	F	M	M
Te-129	F	M	M
Te-129m	F	M	M
Te-131	F	M	M
Te-131m	F	M	M
Te-132	F	M	M
I-123	V (I2)	V (I2)	V (I2)
I-124	V (I2)	V (I2)	V (I2)
I-125	V (I2)	V (I2)	V (I2)
I-126	V (I2)	V (I2)	V (I2)
I-129	V (I2)	V (I2)	V (I2)
I-130	V (I2)	V (I2)	V (I2)
I-131	V (I2)	V (I2)	V (I2)
I-132	V (I2)	V (I2)	V (I2)
I-133	V (I2)	V (I2)	V (I2)
I-135	V (I2)	V (I2)	V (I2)
Cs-127	F	F	F
Cs-129	F	F	F
Cs-131	F	F	F
Cs-132	F	F	F
Cs-134	F	F	F
Cs-135	F	F	F
Cs-136	F	F	F
Cs-137	F	F	F
Ba-131	F	F	F
Ba-133	F	F	F
Ba-133m	F	F	F
Ba-135m	F	F	F
Ba-140	F	F	F
La-135	F	M	M

ATTACHMENT C
RADIONUCLIDES CONSIDERED IN COMMITTED ORGAN DOSE CALCULATIONS AND THE ABSORPTION TYPES ASSIGNED TO EACH SOLUBILITY CATEGORY (continued)

Radionuclide	Absorption type assigned		
	Soluble	Mod. soluble	Insoluble
La-137	F	M	M
La-138	F	M	M
La-140	F	M	M
La-141	F	M	M
Ce-135	M	M	S
Ce-137	M	M	S
Ce-137m	M	M	S
Ce-139	M	M	S
Ce-141	M	M	S
Ce-143	M	M	S
Ce-144	M	M	S
Pr-139	M	M	S
Pr-142	M	M	S
Pr-143	M	M	S
Pr-144	M	M	S
Pr-145	M	M	S
Nd-147	M	M	S
Pm-143	M	M	S
Pm-144	M	M	S
Pm-145	M	M	S
Pm-146	M	M	S
Pm-147	M	M	S
Pm-148	M	M	S
Pm-148m	M	M	S
Pm-149	M	M	S
Pm-151	M	M	S
Sm-145	M	M	M
Sm-146	M	M	M
Sm-147	M	M	M
Sm-151	M	M	M
Sm-153	M	M	M
Sm-156	M	M	M
Eu-147	M	M	M
Eu-149	M	M	M
Eu-152	M	M	M
Eu-152m	M	M	M
Eu-154	M	M	M
Eu-155	M	M	M
Eu-156	M	M	M
Eu-157	M	M	M
Gd-147	F	M	M
Gd-149	F	M	M
Gd-151	F	M	M
Gd-152	F	M	M
Gd-153	F	M	M
Gd-159	F	M	M
Tb-153	M	M	M
Tb-155	M	M	M
Tb-156	M	M	M

ATTACHMENT C
RADIONUCLIDES CONSIDERED IN COMMITTED ORGAN DOSE CALCULATIONS AND THE ABSORPTION TYPES ASSIGNED TO EACH SOLUBILITY CATEGORY (continued)

Radionuclide	Absorption type assigned		
	Soluble	Mod. soluble	Insoluble
Tb-156m	M	M	M
Tb-157	M	M	M
Tb-158	M	M	M
Tb-160	M	M	M
Tb-161	M	M	M
Dy-155	M	M	M
Dy-157	M	M	M
Dy-159	M	M	M
Dy-166	M	M	M
Ho-166	M	M	M
Ho-166m	M	M	M
Er-165	M	M	M
Er-169	M	M	M
Er-171	M	M	M
Er-172	M	M	M
Tm-166	M	M	M
Tm-167	M	M	M
Tm-170	M	M	M
Tm-171	M	M	M
Tm-172	M	M	M
Tm-173	M	M	M
Yb-166	M	M	S
Yb-169	M	M	S
Yb-175	M	M	S
Lu-169	M	M	S
Lu-171	M	M	S
Lu-172	M	M	S
Lu-176	M	M	S
Lu-176m	M	M	S
Lu-177	M	M	S
Lu-177m	M	M	S
Hf-175	F	M	M
Hf-180m	F	M	M
Hf-181	F	M	M
Hf-182	F	M	M
Ta-180	M	M	M
Ta-182	M	M	M
Ta-183	M	M	M
W-181	F	F	F
W-185	F	F	F
W-187	F	F	F
W-188	F	F	F
Re-186	F	M	M
Re-187	F	M	M
Re-188	F	M	M
Re-189	F	M	M
Os-185	F	M	S
Os-189m	F	M	S
Os-191	F	M	S

ATTACHMENT C
RADIONUCLIDES CONSIDERED IN COMMITTED ORGAN DOSE CALCULATIONS AND THE ABSORPTION TYPES ASSIGNED TO EACH SOLUBILITY CATEGORY (continued)

Radionuclide	Absorption type assigned		
	Soluble	Mod. soluble	Insoluble
Os-191m	F	M	S
Os-193	F	M	S
Os-194	F	M	S
Ir-192	F	M	S
Ir-192m	F	M	S
Ir-193m	F	M	S
Ir-194	F	M	S
Ir-194m	F	M	S
Pt-193	F	F	F
Pt-193m	F	F	F
Pt-195m	F	F	F
Pt-197	F	F	F
Au-198	F	M	S
Au-199	F	M	S
Hg-197	F	M	M
Hg-197m	F	M	M
Hg-203	F	M	M
Tl-204	F	F	F
Pb-205	F	F	F
Bi-210	F	M	M
Bi-210m	F	M	M
Po-210	F	M	M

ATTACHMENT D
NORMALIZED INTAKE FRACTIONS FOR THE DOSIMETRICALLY IMPORTANT RADIONUCLIDES

Table D-1. NIFs for radionuclides that contribute ≥1% dose to committed organ dose.

	ATR 1				ATR 2				ATR 3			
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**
Mn-54	1.91E-07	8.69E-07	3.04E-06	5.59E-06	6.77E-08	4.70E-07	2.41E-06	5.15E-06	9.30E-08	5.56E-07	2.49E-06	5.12E-06
Fe-55	3.33E-06	1.58E-05	6.85E-05	1.67E-04	1.11E-06	8.05E-06	5.12E-05	1.45E-04	1.49E-06	9.34E-06	5.18E-05	1.41E-04
Co-58	2.12E-07	7.66E-07	9.29E-07	4.23E-07	8.03E-08	4.45E-07	7.93E-07	4.19E-07	2.10E-07	1.00E-06	1.56E-06	7.93E-07
Co-60	8.21E-08	3.94E-07	1.79E-06	4.64E-06	2.68E-08	1.97E-07	1.31E-06	3.95E-06	3.65E-08	2.31E-07	1.34E-06	3.89E-06
Sr-89	2.69E-02	8.67E-02	6.06E-02	1.33E-02	1.48E-02	7.30E-02	7.50E-02	1.91E-02	1.88E-02	7.99E-02	7.16E-02	1.76E-02
Sr-90	4.01E-04	1.94E-03	9.17E-03	2.51E-02	1.30E-04	9.60E-04	6.66E-03	2.12E-02	1.85E-04	1.18E-03	7.16E-03	2.20E-02
Y-90	4.02E-04	1.94E-03	9.18E-03	2.51E-02	1.29E-04	9.61E-04	6.66E-03	2.12E-02	1.85E-04	1.18E-03	7.17E-03	2.20E-02
Y-91	3.22E-02	1.10E-01	9.95E-02	3.08E-02	1.69E-02	8.79E-02	1.17E-01	4.22E-02	2.15E-02	9.66E-02	1.12E-01	3.89E-02
Zr-95	3.48E-02	1.22E-01	1.28E-01	4.78E-02	1.74E-02	9.35E-02	1.44E-01	6.25E-02	2.26E-02	1.05E-01	1.40E-01	5.88E-02
Nb-95	3.00E-02	1.44E-01	2.44E-01	1.03E-01	8.13E-03	8.21E-02	2.63E-01	1.33E-01	1.31E-02	1.01E-01	2.60E-01	1.26E-01
Mo-99	3.78E-03	9.46E-06	2.05E-20	3.03E-40	4.47E-03	1.71E-05	5.46E-20	9.36E-40	4.18E-03	1.38E-05	3.83E-20	6.33E-40
Ru-103	1.80E-02	5.15E-02	2.07E-02	2.19E-03	1.10E-02	4.83E-02	2.86E-02	3.51E-03	1.36E-02	5.14E-02	2.65E-02	3.13E-03
Ru-106	6.96E-04	3.19E-03	1.17E-02	2.30E-02	2.44E-04	1.71E-03	9.23E-03	2.10E-02	3.38E-04	2.04E-03	9.61E-03	2.11E-02
Cd-113m	7.23E-10	3.49E-09	1.63E-08	4.42E-08	2.88E-10	2.13E-09	1.46E-08	4.59E-08	1.04E-09	6.62E-09	3.97E-08	1.20E-07
Cd-115m	2.79E-06	8.49E-06	4.59E-06	7.19E-07	1.60E-06	7.47E-06	5.94E-06	1.08E-06	2.02E-06	8.09E-06	5.61E-06	9.81E-07
Sb-125	2.32E-05	1.12E-04	4.84E-04	1.18E-03	7.43E-06	5.66E-05	3.62E-04	1.02E-03	1.07E-05	6.92E-05	3.85E-04	1.05E-03
Te-129m	5.89E-04	1.54E-03	4.08E-04	2.49E-05	3.89E-04	1.56E-03	6.07E-04	4.30E-05	4.69E-04	1.61E-03	5.49E-04	3.75E-05
Te-132	3.95E-03	3.24E-05	1.79E-17	4.01E-34	4.67E-03	5.87E-05	4.76E-17	1.24E-33	4.36E-03	4.72E-05	3.34E-17	8.35E-34
I-131	4.85E-01	1.77E-01	4.85E-06	1.59E-12	5.50E-01	3.08E-01	1.23E-05	4.70E-12	5.31E-01	2.56E-01	8.95E-06	3.29E-12
I-132	2.04E-01	1.67E-03	9.23E-16	2.06E-32	2.41E-01	3.02E-03	2.45E-15	6.36E-32	2.25E-01	2.43E-03	1.72E-15	4.30E-32
Cs-134	3.00E-04	1.42E-03	5.93E-03	1.39E-02	7.00E-05	5.06E-04	3.12E-03	8.46E-03	4.68E-05	2.91E-04	1.56E-03	4.09E-03
Cs-136	8.78E-05	8.77E-05	2.62E-07	4.27E-11	4.01E-05	6.13E-05	2.69E-07	5.08E-11	4.70E-05	6.20E-05	2.38E-07	4.32E-11
Cs-137	4.05E-04	1.96E-03	9.27E-03	2.54E-02	1.31E-04	9.69E-04	6.73E-03	2.14E-02	1.87E-04	1.19E-03	7.23E-03	2.22E-02
Ba-140	2.81E-02	2.67E-02	6.30E-05	7.49E-09	2.85E-02	4.14E-02	1.44E-04	1.98E-08	2.93E-02	3.67E-02	1.11E-04	1.48E-08
La-140	3.23E-02	3.08E-02	7.25E-05	8.63E-09	3.27E-02	4.77E-02	1.65E-04	2.28E-08	3.36E-02	4.23E-02	1.28E-04	1.70E-08
Ce-141	3.45E-02	8.84E-02	2.13E-02	1.14E-03	2.32E-02	9.08E-02	3.21E-02	2.00E-03	2.81E-02	9.47E-02	2.92E-02	1.75E-03
Ce-144	1.16E-02	5.22E-02	1.77E-01	3.13E-01	4.17E-03	2.88E-02	1.44E-01	2.94E-01	5.84E-03	3.47E-02	1.51E-01	2.98E-01
Pr-143	3.05E-02	3.20E-02	1.19E-04	2.61E-08	2.97E-02	4.77E-02	2.62E-04	6.63E-08	3.16E-02	4.38E-02	2.10E-04	5.12E-08
Pr-144	1.16E-02	5.22E-02	1.77E-01	3.13E-01	4.17E-03	2.88E-02	1.44E-01	2.94E-01	5.84E-03	3.47E-02	1.51E-01	2.98E-01
Nd-147	8.74E-03	6.39E-03	4.42E-06	1.04E-10	8.40E-03	9.40E-03	9.54E-06	2.60E-10	9.48E-03	9.13E-03	8.09E-06	2.13E-10
Pm-147	1.13E-03	5.75E-03	2.51E-02	6.09E-02	3.17E-04	2.90E-03	1.91E-02	5.36E-02	5.81E-04	4.22E-03	2.39E-02	6.48E-02
Pm-148m	1.78E-04	5.24E-04	2.38E-04	2.96E-05	4.13E-05	1.85E-04	1.24E-04	1.78E-05	8.72E-05	3.37E-04	1.96E-04	2.73E-05
Sm-151	1.23E-06	5.95E-06	2.83E-05	7.82E-05	4.84E-07	3.60E-06	2.51E-05	8.05E-05	1.41E-06	8.99E-06	5.47E-05	1.69E-04
Eu-154	9.45E-06	4.56E-05	2.11E-04	5.61E-04	2.61E-06	1.93E-05	1.31E-04	4.05E-04	1.55E-06	9.83E-06	5.82E-05	1.73E-04
Eu-155	5.47E-06	2.62E-05	1.18E-04	3.04E-04	1.58E-06	1.16E-05	7.67E-05	2.29E-04	2.49E-06	1.57E-05	9.07E-05	2.61E-04
Ta-182	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

ATTACHMENT D
NORMALIZED INTAKE FRACTIONS FOR THE DOSIMETRICALLY IMPORTANT RADIONUCLIDES (continued)

Radionuclide	FFTF 1				FFTF 2			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
Mn-54	1.17E-03	4.77E-03	1.05E-02	1.22E-02	8.15E-04	3.61E-03	9.23E-03	1.18E-02
Fe-55	1.57E-03	6.69E-03	1.82E-02	2.82E-02	8.60E-04	3.99E-03	1.26E-02	2.14E-02
Co-58	7.42E-03	2.40E-02	1.83E-02	5.28E-03	6.40E-03	2.26E-02	2.01E-02	6.33E-03
Co-60	4.69E-05	2.02E-04	5.76E-04	9.47E-04	2.28E-05	1.07E-04	3.54E-04	6.39E-04
Sr-89	9.82E-03	2.83E-02	1.24E-02	1.73E-03	1.01E-02	3.18E-02	1.63E-02	2.48E-03
Sr-90	8.10E-04	3.51E-03	1.05E-02	1.82E-02	4.41E-04	2.09E-03	7.21E-03	1.37E-02
Y-90	8.15E-04	3.51E-03	1.05E-02	1.82E-02	4.43E-04	2.09E-03	7.21E-03	1.37E-02
Y-91	1.46E-02	4.46E-02	2.55E-02	5.01E-03	1.50E-02	4.99E-02	3.31E-02	7.14E-03
Zr-95	2.64E-02	8.30E-02	5.48E-02	1.30E-02	2.69E-02	9.21E-02	7.06E-02	1.84E-02
Nb-95	2.92E-02	1.13E-01	1.08E-01	2.81E-02	2.95E-02	1.25E-01	1.39E-01	3.96E-02
Mo-99	3.31E-03	7.42E-06	1.01E-20	9.49E-41	3.26E-03	7.96E-06	1.26E-20	1.30E-40
Ru-103	3.65E-02	9.35E-02	2.37E-02	1.59E-03	3.64E-02	1.02E-01	2.99E-02	2.20E-03
Ru-106	2.14E-02	8.79E-02	2.04E-01	2.54E-01	1.57E-02	7.05E-02	1.90E-01	2.59E-01
Cd-113m	2.11E-06	9.15E-06	2.70E-05	4.62E-05	8.57E-07	4.05E-06	1.38E-05	2.60E-05
Cd-115m	2.63E-05	7.18E-05	2.44E-05	2.41E-06	2.53E-05	7.52E-05	2.97E-05	3.22E-06
Sb-125	4.66E-04	2.00E-03	5.45E-03	8.43E-03	2.91E-04	1.37E-03	4.33E-03	7.35E-03
Te-129m	5.82E-06	1.36E-05	2.28E-06	8.81E-08	3.57E-06	9.11E-06	1.77E-06	7.50E-08
Te-132	3.62E-03	2.39E-05	5.03E-18	3.67E-35	3.73E-03	2.68E-05	6.57E-18	5.26E-35
I-131	5.06E-01	1.65E-01	2.75E-06	5.51E-13	5.22E-01	1.85E-01	3.59E-06	7.90E-13
I-132	1.86E-01	1.23E-03	2.59E-16	1.89E-33	1.92E-01	1.38E-03	3.38E-16	2.71E-33
Cs-134	2.69E-03	1.13E-02	3.00E-02	4.44E-02	7.74E-04	3.57E-03	1.09E-02	1.78E-02
Cs-136	3.18E-03	2.84E-03	5.36E-06	5.52E-10	1.48E-03	1.44E-03	3.16E-06	3.57E-10
Cs-137	2.34E-03	1.01E-02	3.02E-02	5.25E-02	1.27E-03	6.00E-03	2.07E-02	3.95E-02
Ba-140	1.93E-02	1.65E-02	2.45E-05	1.84E-09	1.99E-02	1.84E-02	3.18E-05	2.63E-09
La-140	2.22E-02	1.89E-02	2.82E-05	2.12E-09	2.28E-02	2.12E-02	3.66E-05	3.03E-09
Ce-141	2.61E-02	5.98E-02	9.07E-03	3.09E-04	2.67E-02	6.67E-02	1.18E-02	4.39E-04
Ce-144	2.08E-02	8.40E-02	1.80E-01	2.01E-01	1.64E-02	7.23E-02	1.79E-01	2.20E-01
Pr-143	1.80E-02	1.69E-02	3.98E-05	5.50E-09	1.82E-02	1.87E-02	5.10E-05	7.74E-09
Pr-144	2.08E-02	8.41E-02	1.80E-01	2.01E-01	1.64E-02	7.23E-02	1.79E-01	2.20E-01
Nd-147	7.13E-03	4.67E-03	2.03E-06	3.03E-11	7.05E-03	5.03E-03	2.55E-06	4.16E-11
Pm-147	5.06E-03	2.18E-02	5.94E-02	9.12E-02	3.45E-03	1.63E-02	5.16E-02	8.70E-02
Pm-148m	1.71E-03	4.48E-03	1.28E-03	1.01E-04	8.72E-04	2.50E-03	8.30E-04	7.17E-05
Sm-151	5.54E-05	2.40E-04	7.19E-04	1.26E-03	3.91E-05	1.85E-04	6.43E-04	1.24E-03
Eu-154	1.70E-04	7.32E-04	2.13E-03	3.60E-03	6.02E-05	2.83E-04	9.58E-04	1.77E-03
Eu-155	4.57E-04	1.96E-03	5.58E-03	9.11E-03	2.32E-04	1.08E-03	3.58E-03	6.41E-03
Ta-182	1.55E-04	5.63E-04	7.24E-04	4.15E-04	1.52E-04	5.99E-04	8.95E-04	5.62E-04

ATTACHMENT D
NORMALIZED INTAKE FRACTIONS FOR THE DOSIMETRICALLY IMPORTANT RADIONUCLIDES (continued)

	N Rx 1				N Rx 2			
Radionuclide	**10 d**	**40 d**	**180 d**	**1 y**	**10 d**	**40 d**	**180 d**	**1 y**
Mn-54	1.72E-07	8.46E-07	3.12E-06	5.79E-06	3.40E-07	1.42E-06	4.18E-06	6.42E-06
Fe-55	4.79E-06	2.47E-05	1.13E-04	2.77E-04	1.03E-05	4.50E-05	1.63E-04	3.33E-04
Co-58	1.29E-06	5.08E-06	6.50E-06	2.99E-06	1.32E-06	4.40E-06	4.48E-06	1.70E-06
Co-60	9.37E-07	4.88E-06	2.33E-05	6.11E-05	2.05E-06	9.08E-06	3.45E-05	7.48E-05
Sr-89	2.27E-02	7.91E-02	5.84E-02	1.29E-02	2.29E-02	6.79E-02	3.99E-02	7.31E-03
Sr-90	3.11E-04	1.63E-03	8.14E-03	2.25E-02	6.61E-04	2.95E-03	1.17E-02	2.68E-02
Y-90	3.11E-04	1.64E-03	8.15E-03	2.25E-02	6.63E-04	2.95E-03	1.17E-02	2.68E-02
Y-91	2.73E-02	1.01E-01	9.66E-02	3.02E-02	2.95E-02	9.26E-02	7.06E-02	1.83E-02
Zr-95	3.08E-02	1.17E-01	1.29E-01	4.89E-02	3.63E-02	1.18E-01	1.03E-01	3.23E-02
Nb-95	2.52E-02	1.34E-01	2.46E-01	1.05E-01	3.87E-02	1.57E-01	2.02E-01	6.96E-02
Mo-99	3.77E-03	1.02E-05	2.34E-20	3.49E-40	3.44E-03	7.94E-06	1.45E-20	1.78E-40
Ru-103	2.05E-02	6.34E-02	2.69E-02	2.88E-03	2.50E-02	6.58E-02	2.23E-02	1.97E-03
Ru-106	1.27E-03	6.31E-03	2.44E-02	4.84E-02	3.65E-03	1.55E-02	4.77E-02	7.82E-02
Cd-113m	3.35E-09	1.76E-08	8.66E-08	2.37E-07	4.60E-09	2.05E-08	8.06E-08	1.82E-07
Cd-115m	3.72E-06	1.23E-05	7.01E-06	1.11E-06	5.20E-06	1.46E-05	6.63E-06	8.66E-07
Sb-125	2.50E-05	1.31E-04	6.00E-04	1.48E-03	6.34E-05	2.81E-04	1.02E-03	2.08E-03
Te-129m	6.79E-04	1.92E-03	5.39E-04	3.32E-05	8.13E-04	1.96E-03	4.37E-04	2.23E-05
Te-132	4.08E-03	3.63E-05	2.12E-17	4.78E-34	3.82E-03	2.89E-05	1.34E-17	2.50E-34
I-131	5.05E-01	2.00E-01	5.76E-06	1.91E-12	4.83E-01	1.63E-01	3.74E-06	1.02E-12
I-132	2.10E-01	1.87E-03	1.09E-15	2.46E-32	1.97E-01	1.49E-03	6.91E-16	1.29E-32
Cs-134	3.58E-05	1.83E-04	8.11E-04	1.91E-03	2.06E-04	8.99E-04	3.16E-03	6.18E-03
Cs-136	9.71E-05	1.05E-04	3.32E-07	5.45E-11	1.84E-04	1.69E-04	4.25E-07	5.77E-11
Cs-137	3.43E-04	1.80E-03	8.99E-03	2.49E-02	7.85E-04	3.51E-03	1.39E-02	3.19E-02
Ba-140	2.70E-02	2.78E-02	6.92E-05	8.30E-09	2.40E-02	2.11E-02	4.18E-05	4.15E-09
La-140	3.10E-02	3.20E-02	7.97E-05	9.56E-09	2.76E-02	2.43E-02	4.81E-05	4.77E-09
Ce-141	3.24E-02	9.00E-02	2.29E-02	1.24E-03	3.17E-02	7.49E-02	1.51E-02	6.78E-04
Ce-144	9.61E-03	4.70E-02	1.68E-01	3.00E-01	1.75E-02	7.27E-02	2.07E-01	3.06E-01
Pr-143	2.79E-02	3.17E-02	1.25E-04	2.76E-08	2.43E-02	2.35E-02	7.39E-05	1.35E-08
Pr-144	9.61E-03	4.70E-02	1.68E-01	3.00E-01	1.75E-02	7.27E-02	2.07E-01	3.06E-01
Nd-147	9.00E-03	7.12E-03	5.20E-06	1.23E-10	8.07E-03	5.44E-03	3.16E-06	6.20E-11
Pm-147	1.22E-03	6.71E-03	3.09E-02	7.56E-02	2.56E-03	1.16E-02	4.21E-02	8.52E-02
Pm-148m	6.28E-05	2.00E-04	9.58E-05	1.20E-05	1.46E-04	3.95E-04	1.51E-04	1.56E-05
Sm-151	3.50E-06	1.84E-05	9.23E-05	2.57E-04	4.24E-06	1.90E-05	7.56E-05	1.74E-04
Eu-154	1.71E-06	8.91E-06	4.35E-05	1.17E-04	9.99E-06	4.44E-05	1.72E-04	3.83E-04
Eu-155	1.03E-05	5.33E-05	2.53E-04	6.58E-04	1.95E-05	8.63E-05	3.27E-04	7.02E-04
Ta-182	1.72E-08	7.56E-08	1.64E-07	1.50E-07	6.42E-08	2.40E-07	4.13E-07	3.14E-07

ATTACHMENT D
NORMALIZED INTAKE FRACTIONS FOR THE DOSIMETRICALLY IMPORTANT RADIONUCLIDES (continued)

	TRIGA 1				TRIGA 2			
Radionuclide	**10 d**	**40 d**	**180 d**	**1 y**	**10 d**	**40 d**	**180 d**	**1 y**
Mn-54	8.69E-05	2.79E-04	5.49E-04	6.20E-04	2.17E-05	1.01E-04	3.62E-04	6.81E-04
Fe-55	2.06E-03	6.93E-03	1.69E-02	2.53E-02	2.12E-04	1.04E-03	4.59E-03	1.14E-02
Co-58	1.62E-04	4.13E-04	2.82E-04	7.89E-05	1.22E-04	4.56E-04	5.65E-04	2.62E-04
Co-60	1.69E-03	5.73E-03	1.46E-02	2.33E-02	1.35E-04	6.67E-04	3.09E-03	8.18E-03
Sr-89	2.71E-02	6.15E-02	2.42E-02	3.27E-03	2.55E-02	8.45E-02	6.03E-02	1.35E-02
Sr-90	5.26E-03	1.80E-02	4.79E-02	8.07E-02	3.48E-04	1.74E-03	8.37E-03	2.34E-02
Y-90	5.27E-03	1.80E-02	4.80E-02	8.07E-02	3.49E-04	1.74E-03	8.37E-03	2.34E-02
Y-91	3.43E-02	8.25E-02	4.22E-02	8.03E-03	3.03E-02	1.06E-01	9.82E-02	3.11E-02
Zr-95	4.15E-02	1.03E-01	6.07E-02	1.39E-02	3.44E-02	1.24E-01	1.33E-01	5.08E-02
Nb-95	4.58E-02	1.40E-01	1.19E-01	3.01E-02	2.81E-02	1.42E-01	2.53E-01	1.09E-01
Mo-99	3.29E-03	5.81E-06	7.10E-21	6.45E-41	3.84E-03	9.89E-06	2.19E-20	3.31E-40
Ru-103	1.78E-02	3.59E-02	8.15E-03	5.29E-04	1.73E-02	5.10E-02	2.10E-02	2.27E-03
Ru-106	3.25E-03	1.05E-02	2.18E-02	2.63E-02	6.09E-04	2.87E-03	1.08E-02	2.16E-02
Cd-113m	2.75E-08	9.39E-08	2.47E-07	4.11E-07	1.93E-08	9.58E-08	4.58E-07	1.27E-06
Cd-115m	2.98E-06	6.41E-06	1.95E-06	1.88E-07	2.67E-06	8.36E-06	4.62E-06	7.39E-07
Sb-125	2.85E-04	9.59E-04	2.34E-03	3.50E-03	3.09E-05	1.53E-04	6.76E-04	1.68E-03
Te-129m	5.62E-04	1.04E-03	1.55E-04	5.82E-06	5.69E-04	1.53E-03	4.15E-04	2.59E-05
Te-132	3.45E-03	2.00E-05	6.22E-18	8.54E-35	4.00E-03	3.38E-05	1.91E-17	4.36E-34
I-131	4.23E-01	1.09E-01	1.68E-06	3.40E-13	4.91E-01	1.84E-01	5.15E-06	1.73E-12
I-132	1.77E-01	1.03E-03	3.20E-16	4.40E-33	2.06E-01	1.74E-03	9.83E-16	2.25E-32
Cs-134	6.57E-04	2.19E-03	5.18E-03	7.44E-03	2.96E-06	1.44E-05	6.17E-05	1.47E-04
Cs-136	1.44E-04	1.02E-04	1.71E-07	1.71E-11	3.10E-05	3.19E-05	9.74E-08	1.62E-11
Cs-137	5.39E-03	1.85E-02	4.91E-02	8.27E-02	3.51E-04	1.75E-03	8.45E-03	2.37E-02
Ba-140	2.41E-02	1.62E-02	2.15E-05	1.57E-09	2.81E-02	2.75E-02	6.63E-05	8.05E-09
La-140	2.76E-02	1.86E-02	2.47E-05	1.81E-09	3.23E-02	3.17E-02	7.64E-05	9.28E-09
Ce-141	3.16E-02	5.72E-02	7.76E-03	2.56E-04	3.38E-02	8.90E-02	2.19E-02	1.20E-03
Ce-144	3.56E-02	1.13E-01	2.17E-01	2.35E-01	1.02E-02	4.76E-02	1.65E-01	2.98E-01
Pr-143	2.58E-02	1.91E-02	4.03E-05	5.40E-09	3.09E-02	3.34E-02	1.28E-04	2.84E-08
Pr-144	3.56E-02	1.13E-01	2.17E-01	2.35E-01	1.02E-02	4.76E-02	1.65E-01	2.98E-01
Nd-147	7.98E-03	4.12E-03	1.60E-06	2.32E-11	9.32E-03	7.01E-03	4.95E-06	1.19E-10
Pm-147	1.13E-02	3.83E-02	9.31E-02	1.39E-01	1.30E-03	6.79E-03	3.02E-02	7.49E-02
Pm-148m	3.04E-04	6.30E-04	1.61E-04	1.23E-05	1.83E-05	5.53E-05	2.57E-05	3.26E-06
Sm-151	3.00E-05	1.03E-04	2.75E-04	4.67E-04	7.24E-06	3.61E-05	1.75E-04	4.95E-04
Eu-154	3.74E-05	1.27E-04	3.31E-04	5.42E-04	1.29E-07	6.42E-07	3.03E-06	8.25E-06
Eu-155	7.61E-05	2.58E-04	6.54E-04	1.03E-03	1.11E-05	5.49E-05	2.53E-04	6.64E-04
Ta-182	9.34E-04	2.67E-03	3.09E-03	1.72E-03	7.56E-04	3.15E-03	6.59E-03	6.13E-03

ATTACHMENT E
NORMALIZED INTAKE FRACTIONS FOR THE REDUCED LIST OF DOSIMETRICALLY IMPORTANT RADIONUCLIDES

Table E-1. Normalized intake fractions for the reduced list of dosimetrically important radionuclides.

	ATR 1				ATR 2				ATR 3			
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**
Co-60	1.16E-07	4.48E-07	2.18E-06	6.77E-06	3.96E-08	2.27E-07	1.53E-06	5.61E-06	5.31E-08	2.65E-07	1.58E-06	5.56E-06
Sr-89	3.81E-02	9.85E-02	7.38E-02	1.94E-02	2.19E-02	8.41E-02	8.77E-02	2.71E-02	2.74E-02	9.15E-02	8.44E-02	2.51E-02
Sr-90	5.67E-04	2.21E-03	1.12E-02	3.67E-02	1.92E-04	1.11E-03	7.79E-03	3.01E-02	2.70E-04	1.36E-03	8.45E-03	3.14E-02
Y-90	5.68E-04	2.21E-03	1.12E-02	3.67E-02	1.91E-04	1.11E-03	7.79E-03	3.01E-02	2.69E-04	1.36E-03	8.45E-03	3.14E-02
Y-91	4.56E-02	1.25E-01	1.21E-01	4.50E-02	2.50E-02	1.01E-01	1.37E-01	5.99E-02	3.13E-02	1.11E-01	1.33E-01	5.56E-02
Zr-95	4.92E-02	1.39E-01	1.55E-01	6.97E-02	2.58E-02	1.08E-01	1.68E-01	8.87E-02	3.29E-02	1.20E-01	1.66E-01	8.39E-02
Nb-95	4.25E-02	1.64E-01	2.97E-01	1.50E-01	1.20E-02	9.46E-02	3.07E-01	1.89E-01	1.91E-02	1.16E-01	3.07E-01	1.79E-01
Ru-103	2.55E-02	5.85E-02	2.52E-02	3.20E-03	1.63E-02	5.57E-02	3.34E-02	4.98E-03	1.98E-02	5.89E-02	3.12E-02	4.48E-03
Ru-106	9.85E-04	3.63E-03	1.43E-02	3.36E-02	3.60E-04	1.97E-03	1.08E-02	2.99E-02	4.91E-04	2.34E-03	1.13E-02	3.01E-02
I-131	6.86E-01	2.02E-01	5.90E-06	2.32E-12	8.13E-01	3.54E-01	1.44E-05	6.68E-12	7.72E-01	2.93E-01	1.06E-05	4.70E-12
Cs-134	4.24E-04	1.61E-03	7.22E-03	2.03E-02	1.04E-04	5.83E-04	3.64E-03	1.20E-02	6.80E-05	3.34E-04	1.85E-03	5.84E-03
Cs-137	5.72E-04	2.23E-03	1.13E-02	3.71E-02	1.93E-04	1.12E-03	7.87E-03	3.04E-02	2.72E-04	1.37E-03	8.53E-03	3.17E-02
Ce-141	4.89E-02	1.00E-01	2.59E-02	1.67E-03	3.43E-02	1.05E-01	3.76E-02	2.84E-03	4.08E-02	1.08E-01	3.44E-02	2.50E-03
Ce-144	1.64E-02	5.93E-02	2.15E-01	4.57E-01	6.17E-03	3.32E-02	1.68E-01	4.18E-01	8.48E-03	3.98E-02	1.78E-01	4.26E-01
Pr-143	4.31E-02	3.63E-02	1.45E-04	3.81E-08	4.39E-02	5.50E-02	3.06E-04	9.42E-08	4.60E-02	5.02E-02	2.47E-04	7.31E-08
Pm-147	1.59E-03	6.54E-03	3.06E-02	8.89E-02	4.68E-04	3.34E-03	2.23E-02	7.62E-02	8.45E-04	4.83E-03	2.82E-02	9.25E-02
Eu-154	1.34E-05	5.18E-05	2.56E-04	8.19E-04	3.87E-06	2.22E-05	1.53E-04	5.75E-04	2.25E-06	1.13E-05	6.87E-05	2.48E-04

	FFTF 1				FFTF 2				N Rx 1			
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**
Co-60	6.51E-05	2.44E-04	7.59E-04	1.29E-03	3.15E-05	1.27E-04	4.61E-04	8.83E-04	1.33E-06	5.53E-06	2.81E-05	8.76E-05
Sr-89	1.36E-02	3.42E-02	1.64E-02	2.36E-03	1.40E-02	3.77E-02	2.12E-02	3.42E-03	3.22E-02	8.97E-02	7.03E-02	1.85E-02
Sr-90	1.12E-03	4.25E-03	1.38E-02	2.47E-02	6.10E-04	2.47E-03	9.38E-03	1.90E-02	4.42E-04	1.85E-03	9.81E-03	3.23E-02
Y-90	1.13E-03	4.25E-03	1.38E-02	2.48E-02	6.13E-04	2.47E-03	9.38E-03	1.90E-02	4.42E-04	1.85E-03	9.81E-03	3.23E-02
Y-91	2.03E-02	5.39E-02	3.36E-02	6.82E-03	2.08E-02	5.91E-02	4.31E-02	9.86E-03	3.88E-02	1.14E-01	1.16E-01	4.33E-02
Zr-95	3.67E-02	1.00E-01	7.21E-02	1.77E-02	3.72E-02	1.09E-01	9.18E-02	2.53E-02	4.37E-02	1.33E-01	1.56E-01	7.01E-02
Nb-95	4.04E-02	1.37E-01	1.42E-01	3.82E-02	4.07E-02	1.48E-01	1.80E-01	5.47E-02	3.58E-02	1.52E-01	2.97E-01	1.50E-01
Ru-103	5.07E-02	1.13E-01	3.12E-02	2.17E-03	5.03E-02	1.20E-01	3.89E-02	3.04E-03	2.91E-02	7.19E-02	3.25E-02	4.13E-03
Ru-106	2.97E-02	1.06E-01	2.68E-01	3.46E-01	2.18E-02	8.36E-02	2.47E-01	3.58E-01	1.80E-03	7.15E-03	2.94E-02	6.95E-02
I-131	7.02E-01	1.99E-01	3.63E-06	7.51E-13	7.22E-01	2.19E-01	4.68E-06	1.09E-12	7.16E-01	2.27E-01	6.94E-06	2.74E-12
Cs-134	3.72E-03	1.37E-02	3.95E-02	6.06E-02	1.07E-03	4.23E-03	1.42E-02	2.46E-02	5.08E-05	2.08E-04	9.77E-04	2.75E-03
Cs-137	3.25E-03	1.23E-02	3.98E-02	7.15E-02	1.75E-03	7.11E-03	2.70E-02	5.46E-02	4.87E-04	2.05E-03	1.08E-02	3.57E-02
Ce-141	3.62E-02	7.23E-02	1.19E-02	4.21E-04	3.69E-02	7.90E-02	1.53E-02	6.06E-04	4.60E-02	1.02E-01	2.75E-02	1.78E-03
Ce-144	2.89E-02	1.02E-01	2.36E-01	2.74E-01	2.27E-02	8.57E-02	2.33E-01	3.04E-01	1.36E-02	5.33E-02	2.02E-01	4.30E-01
Pr-143	2.49E-02	2.04E-02	5.24E-05	7.50E-09	2.52E-02	2.21E-02	6.64E-05	1.07E-08	3.96E-02	3.60E-02	1.51E-04	3.95E-08
Pm-147	7.02E-03	2.64E-02	7.81E-02	1.24E-01	4.77E-03	1.93E-02	6.71E-02	1.20E-01	1.72E-03	7.61E-03	3.72E-02	1.08E-01
Eu-154	2.35E-04	8.85E-04	2.81E-03	4.90E-03	8.32E-05	3.36E-04	1.25E-03	2.45E-03	2.42E-06	1.01E-05	5.24E-05	1.68E-04

ATTACHMENT E
NORMALIZED INTAKE FRACTIONS FOR THE REDUCED LIST OF DOSIMETRICALLY IMPORTANT RADIONUCLIDES (continued)

Radionuclide	N Rx 2				TRIGA 1				TRIGA 2			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
Co-60	2.85E-06	1.04E-05	4.37E-05	1.08E-04	2.36E-03	6.88E-03	1.93E-02	3.19E-02	1.92E-04	7.60E-04	3.76E-03	1.20E-02
Sr-89	3.19E-02	7.79E-02	5.04E-02	1.06E-02	3.78E-02	7.38E-02	3.19E-02	4.46E-03	3.62E-02	9.63E-02	7.35E-02	1.99E-02
Sr-90	9.21E-04	3.39E-03	1.48E-02	3.88E-02	7.35E-03	2.16E-02	6.31E-02	1.10E-01	4.94E-04	1.98E-03	1.02E-02	3.44E-02
Y-90	9.24E-04	3.39E-03	1.48E-02	3.88E-02	7.36E-03	2.16E-02	6.32E-02	1.10E-01	4.96E-04	1.98E-03	1.02E-02	3.44E-02
Y-91	4.11E-02	1.06E-01	8.93E-02	2.64E-02	4.79E-02	9.90E-02	5.56E-02	1.10E-02	4.30E-02	1.21E-01	1.20E-01	4.57E-02
Zr-95	5.06E-02	1.35E-01	1.31E-01	4.67E-02	5.80E-02	1.23E-01	7.99E-02	1.91E-02	4.89E-02	1.42E-01	1.62E-01	7.46E-02
Nb-95	5.39E-02	1.80E-01	2.56E-01	1.01E-01	6.40E-02	1.68E-01	1.57E-01	4.11E-02	3.99E-02	1.62E-01	3.08E-01	1.60E-01
Ru-103	3.48E-02	7.55E-02	2.81E-02	2.84E-03	2.49E-02	4.31E-02	1.07E-02	7.23E-04	2.46E-02	5.81E-02	2.55E-02	3.33E-03
Ru-106	5.09E-03	1.77E-02	6.03E-02	1.13E-01	4.55E-03	1.27E-02	2.88E-02	3.60E-02	8.64E-04	3.27E-03	1.31E-02	3.17E-02
I-131	6.73E-01	1.87E-01	4.72E-06	1.48E-12	5.92E-01	1.31E-01	2.22E-06	4.64E-13	6.97E-01	2.10E-01	6.27E-06	2.54E-12
Cs-134	2.88E-04	1.03E-03	4.00E-03	8.94E-03	9.18E-04	2.63E-03	6.82E-03	1.02E-02	4.21E-06	1.64E-05	7.51E-05	2.16E-04
Cs-137	1.09E-03	4.02E-03	1.76E-02	4.61E-02	7.54E-03	2.22E-02	6.47E-02	1.13E-01	4.99E-04	1.99E-03	1.03E-02	3.48E-02
Ce-141	4.42E-02	8.59E-02	1.91E-02	9.81E-04	4.42E-02	6.86E-02	1.02E-02	3.49E-04	4.80E-02	1.01E-01	2.66E-02	1.76E-03
Ce-144	2.43E-02	8.34E-02	2.62E-01	4.42E-01	4.97E-02	1.36E-01	2.85E-01	3.21E-01	1.45E-02	5.42E-02	2.01E-01	4.37E-01
Pr-143	3.39E-02	2.70E-02	9.34E-05	1.95E-08	3.61E-02	2.30E-02	5.31E-05	7.38E-09	4.39E-02	3.80E-02	1.55E-04	4.18E-08
Pm-147	3.57E-03	1.33E-02	5.32E-02	1.23E-01	1.58E-02	4.60E-02	1.23E-01	1.90E-01	1.84E-03	7.73E-03	3.68E-02	1.10E-01
Eu-154	1.39E-05	5.09E-05	2.18E-04	5.54E-04	5.22E-05	1.53E-04	4.37E-04	7.40E-04	1.84E-07	7.32E-07	3.69E-06	1.21E-05

ATTACHMENT F
SUMMARY OF INTAKE RETENTION FRACTIONS

Table F-1. Summary of IRFs used to compute urinary excretion.

Nuclide and absorption type	IRF for chronic intake (24-hr urine, 2 years)
Cd-113m S	6.15E-04
Cd-115m S	5.33E-05
Ce-141 S	6.47E-07
Ce-144 S	9.10E-06
Co-58 S	1.60E-02
Co-60 S	2.01E-02
Cs-134 F	3.39E-01
Cs-136 F	6.65E-02
Cs-137 F	3.78E-01
Eu-154 M	8.73E-03
Eu-155 M	8.49E-03
Fe-55 M	9.65E-04
I-131 V	5.83E-01
I-132 V	1.20E-02
La-140 M	4.78E-06
Mn-54 M	4.33E-02
Mo-99 S	1.54E-03
Nb-95 S	1.98E-03
Nd-147 S	2.81E-05
Pm-147 S	1.88E-04
Pm-148m S	4.22E-05
Pr-143 S	3.77E-05
Pr-144 S	1.43E-09
Ru-103 S	1.06E-02
Ru-106 S	1.56E-02
Sb-125 M	4.73E-02
Sm-151 M	6.35E-03
Sr-89 F	1.91E-01
Sr-90 F	2.31E-01
Ta-182 S	4.59E-04
Te-127m M	9.57E-02
Te-129m M	8.06E-02
Te-132 M	4.39E-02
Y-90 S	3.39E-05
Y-91 S	8.58E-05
Zr-95 S	5.81E-04

ATTACHMENT F
SUMMARY OF INTAKE RETENTION FRACTIONS (continued)

Table F-2. Half-lives used for decay corrections.

Radionuclide	Half-life
Mn-54	312.7 d
Fe-55	2.7 y
Co-58	70.8 d
Co-60	5.271 y
Sr-89	50.55 d
Sr-90	28.6 y
Y-90	64.1 h
Y-91	58.51 d
Zr-95	64.02 d
Nb-95	35.06 d
Mo-99	66.02 h
Ru-103	39.35 d
Ru-106	368.2 d
Cd-113m	13.7 y
Cd-115m	44.6 d
Sb-125	2.77 y
Te-129m	33.6 d
Te-132	78.2 h
Cs-134	2.062 y
Cs-136	13.16 y
Cs-137	30.17 y
Ba-140	12.789 d
La-140	40.22 h
Ce-141	32.5 d
Ce-144	284.3 d
Pr-143	13.56 d
Pr-144	17.28 m
Nd-147	10.98 d
Pm-147	2.6234 y
Pm-148m	41.3 d
Sm-151	90 y
Eu-154	8.8 y
Eu-155	4.96 y
Ta-182	114.74 d

ATTACHMENT F
SUMMARY OF INTAKE RETENTION FRACTIONS (continued)

Table F-3. Beta and gamma radiation yield values.

Radionuclide	Beta yield	Gamma yield
Mn-54	0	1.0
Fe-55	0	0.0
Co-58	0	1.306
Co-60	1	2.0
Sr-89/Y-89m	1	9.3E-5
Sr-90	1	0
Y-90	1	0
Y-91	1	0.003
Zr-95	1	0.996
Nb-95	0	0.998
Nb-95m	0	0.2587
Mo-99/Tc-99m	1	1.0743
Ru-103	1	0.9306
Ru-106/Rh-106	2	0.342
Cd-113m	1	0
Cd-115m	1	0.0222
Sb-125	1	0.8562
Te-125m	0	0.00274
Te-129m/Te-129	1.02	0.11567
Te-132	1	0.9199
Cs-134	1	2.22876
Cs-136/Ba-136m	1	2.8793
Cs-137/Ba-137m	1	0.8998
Ba-140	1	0.40381
La-140	1	2.1542
Ce-141	1	0.48
Ce-144/Pr-144	1.98	0.133586
Pr-143	1	0
Pr-144	1	0.0257
Nd-147	1	0.1997
Pm-147	1	0
Pm-148m	0.955	3.2178
Sm-151	0	0
Eu-154	1	1.633
Eu-155	0	0.20653
Ta-182	0.99	1.4468

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY

Table G-1. Contributions to gross beta activity for an unprocessed urinalysis.

Nuclide	ATR 1 (decay time/delay time)				ATR 2 (decay time/delay time)				ATR 3 (decay time/delay time)			
	10 d/30 d	40 d/30 d	180 d/30 d	1 y/ 1d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d
Mn-54	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	1.63E-09	7.82E-09	3.55E-08	9.30E-08	5.32E-10	3.91E-09	2.60E-08	7.92E-08	7.24E-10	4.58E-09	2.66E-08	7.81E-08
Sr-89/Y-89m	3.42E-03	1.10E-02	7.69E-03	2.51E-03	1.88E-03	9.26E-03	9.52E-03	3.61E-03	2.39E-03	1.01E-02	9.08E-03	3.32E-03
Sr-90	9.24E-05	4.48E-04	2.11E-03	5.81E-03	2.99E-05	2.21E-04	1.54E-03	4.89E-03	4.28E-05	2.73E-04	1.65E-03	5.07E-03
Y-90	9.24E-05	4.48E-04	2.11E-03	3.98E-03	2.99E-05	2.21E-04	1.54E-03	3.35E-03	4.28E-05	2.73E-04	1.65E-03	3.47E-03
Y-91	1.94E-06	6.59E-06	5.98E-06	2.61E-06	1.02E-06	5.29E-06	7.05E-06	3.57E-06	1.30E-06	5.81E-06	6.76E-06	3.30E-06
Zr-95	1.46E-05	5.12E-05	5.35E-05	2.74E-05	7.31E-06	3.92E-05	6.03E-05	3.59E-05	9.50E-06	4.39E-05	5.89E-05	3.37E-05
Nb-95m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99/Tc-99m	3.03E-09	7.58E-12	1.64E-26	3.63E-43	3.59E-09	1.37E-11	4.38E-26	1.12E-42	3.35E-09	1.11E-11	3.07E-26	7.57E-43
Ru-103	1.12E-04	3.21E-04	1.29E-04	2.28E-05	6.88E-05	3.01E-04	1.78E-04	3.64E-05	8.49E-05	3.20E-04	1.65E-04	3.25E-05
Ru-106/Rh-106	2.05E-05	9.41E-05	3.46E-04	7.17E-04	7.18E-06	5.04E-05	2.72E-04	6.54E-04	9.95E-06	6.02E-05	2.83E-04	6.57E-04
Cd-113m	4.43E-13	2.14E-12	1.00E-11	2.72E-11	1.76E-13	1.30E-12	8.96E-12	2.82E-11	6.36E-13	4.06E-12	2.43E-11	7.38E-11
Cd-115m	9.32E-11	2.84E-10	1.54E-10	3.77E-11	5.36E-11	2.50E-10	1.98E-10	5.65E-11	6.73E-11	2.70E-10	1.87E-10	5.15E-11
Sb-125	1.07E-06	5.17E-06	2.24E-05	5.58E-05	3.44E-07	2.62E-06	1.68E-05	4.84E-05	4.96E-07	3.21E-06	1.78E-05	4.96E-05
Te-125m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-129m/Te-129	2.61E-05	6.82E-05	1.81E-05	2.01E-06	1.72E-05	6.89E-05	2.69E-05	3.46E-06	2.08E-05	7.15E-05	2.43E-05	3.02E-06
Te-132	2.93E-07	2.41E-09	1.33E-21	1.42E-35	3.47E-07	4.36E-09	3.54E-21	4.38E-35	3.24E-07	3.50E-09	2.48E-21	2.96E-35
Cs-134	9.88E-05	4.66E-04	1.95E-03	4.70E-03	2.31E-05	1.67E-04	1.03E-03	2.86E-03	1.54E-05	9.59E-05	5.15E-04	1.38E-03
Cs-136/Ba-136m	1.20E-06	1.20E-06	3.59E-09	2.69E-12	5.49E-07	8.40E-07	3.69E-09	3.21E-12	6.44E-07	8.49E-07	3.25E-09	2.73E-12
Cs-137/Ba-137m	1.53E-04	7.40E-04	3.50E-03	9.61E-03	4.93E-05	3.66E-04	2.54E-03	8.09E-03	7.06E-05	4.50E-04	2.73E-03	8.38E-03
Ba-140	1.34E-04	1.27E-04	3.00E-07	1.72E-10	1.36E-04	1.97E-04	6.84E-07	4.54E-10	1.40E-04	1.75E-04	5.29E-07	3.39E-10
La-140	1.54E-04	1.47E-04	3.46E-07	8.89E-11	1.56E-04	2.27E-04	7.88E-07	2.35E-10	1.61E-04	2.01E-04	6.09E-07	1.75E-10
Ce-141	1.18E-08	3.02E-08	7.26E-09	7.24E-10	7.91E-09	3.10E-08	1.10E-08	1.27E-09	9.58E-09	3.23E-08	9.96E-09	1.11E-09
Ce-144/Pr-144	1.94E-07	8.74E-07	2.96E-06	5.63E-06	6.99E-08	4.83E-07	2.40E-06	5.29E-06	9.78E-08	5.81E-07	2.53E-06	5.36E-06
Pr-143	2.48E-07	2.60E-07	9.72E-10	9.34E-13	2.41E-07	3.88E-07	2.13E-09	2.38E-12	2.57E-07	3.56E-07	1.70E-09	1.83E-12
Pr-144	0.00E+00	0.00E+00	0.00E+00	3.69E-35	0.00E+00	0.00E+00	0.00E+00	3.47E-35	0.00E+00	0.00E+00	0.00E+00	3.51E-35
Nd-147	3.70E-08	2.70E-08	1.87E-11	2.74E-15	3.55E-08	3.97E-08	4.03E-11	6.86E-15	4.01E-08	3.86E-08	3.42E-11	5.61E-15
Pm-147	2.07E-07	1.06E-06	4.63E-06	1.15E-05	5.84E-08	5.35E-07	3.51E-06	1.01E-05	1.07E-07	7.77E-07	4.40E-06	1.22E-05
Pm-148m	4.34E-09	1.27E-08	5.79E-09	1.17E-09	1.00E-09	4.51E-09	3.01E-09	7.06E-10	2.12E-09	8.21E-09	4.78E-09	1.08E-09
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	8.20E-08	3.95E-07	1.83E-06	4.90E-06	2.27E-08	1.67E-07	1.14E-06	3.53E-06	1.34E-08	8.53E-08	5.05E-07	1.51E-06
Eu-155	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ta-182	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	FFTF 1 (decay time/delay time)				FFTF 2 (decay time/delay time)			
	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d
Mn-54	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	9.31E-07	4.00E-06	1.14E-05	1.90E-05	4.52E-07	2.12E-06	7.03E-06	1.28E-05
Sr-89/Y-89m	1.25E-03	3.59E-03	1.58E-03	3.26E-04	1.29E-03	4.04E-03	2.06E-03	4.68E-04
Sr-90	1.87E-04	8.10E-04	2.41E-03	4.19E-03	1.02E-04	4.81E-04	1.66E-03	3.17E-03
Y-90	1.87E-04	8.10E-04	2.41E-03	2.87E-03	1.02E-04	4.81E-04	1.66E-03	2.17E-03
Y-91	8.80E-07	2.68E-06	1.53E-06	4.24E-07	9.04E-07	3.00E-06	1.99E-06	6.05E-07
Zr-95	1.11E-05	3.48E-05	2.30E-05	7.46E-06	1.13E-05	3.86E-05	2.96E-05	1.05E-05
Nb-95m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99/Tc-99m	2.65E-09	5.95E-12	8.12E-27	1.13E-43	2.61E-09	6.38E-12	1.01E-26	1.55E-43
Ru-103	2.27E-04	5.82E-04	1.48E-04	1.65E-05	2.27E-04	6.32E-04	1.86E-04	2.28E-05
Ru-106/Rh-106	6.30E-04	2.59E-03	6.00E-03	7.90E-03	4.64E-04	2.08E-03	5.59E-03	8.07E-03
Cd-113m	1.29E-09	5.60E-09	1.65E-08	2.84E-08	5.25E-10	2.48E-09	8.48E-09	1.60E-08
Cd-115m	8.80E-10	2.40E-09	8.16E-10	1.27E-10	8.46E-10	2.51E-09	9.93E-10	1.69E-10
Sb-125	2.16E-05	9.24E-05	2.52E-04	3.98E-04	1.35E-05	6.33E-05	2.01E-04	3.47E-04
Te-125m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-129m/Te-129	2.58E-07	6.03E-07	1.01E-07	7.10E-09	1.58E-07	4.03E-07	7.83E-08	6.04E-09
Te-132	2.69E-07	1.77E-09	3.74E-22	1.30E-36	2.77E-07	1.99E-09	4.88E-22	1.86E-36
Cs-134	8.85E-04	3.74E-03	9.88E-03	1.50E-02	2.55E-04	1.18E-03	3.61E-03	6.02E-03
Cs-136/Ba-136m	4.35E-05	3.89E-05	7.34E-08	3.48E-11	2.03E-05	1.98E-05	4.33E-08	2.25E-11
Cs-137/Ba-137m	8.83E-04	3.83E-03	1.14E-02	1.98E-02	4.79E-04	2.26E-03	7.83E-03	1.49E-02
Ba-140	9.22E-05	7.84E-05	1.17E-07	4.23E-11	9.47E-05	8.78E-05	1.52E-07	6.03E-11
La-140	1.06E-04	9.03E-05	1.34E-07	2.19E-11	1.09E-04	1.01E-04	1.74E-07	3.12E-11
Ce-141	8.90E-09	2.04E-08	3.10E-09	1.96E-10	9.11E-09	2.28E-08	4.01E-09	2.78E-10
Ce-144/Pr-144	3.49E-07	1.41E-06	3.01E-06	3.62E-06	2.75E-07	1.21E-06	3.00E-06	3.96E-06
Pr-143	1.46E-07	1.37E-07	3.24E-10	1.97E-13	1.48E-07	1.52E-07	4.15E-10	2.77E-13
Pr-144	0.00E+00	0.00E+00	0.00E+00	2.37E-35	0.00E+00	0.00E+00	0.00E+00	2.60E-35
Nd-147	3.02E-08	1.97E-08	8.60E-12	7.98E-16	2.98E-08	2.13E-08	1.08E-11	1.10E-15
Pm-147	9.33E-07	4.02E-06	1.09E-05	1.72E-05	6.36E-07	3.01E-06	9.51E-06	1.64E-05
Pm-148m	4.15E-08	1.09E-07	3.12E-08	4.00E-09	2.12E-08	6.08E-08	2.02E-08	2.84E-09
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	1.47E-06	6.35E-06	1.85E-05	3.14E-05	5.22E-07	2.46E-06	8.31E-06	1.55E-05
Eu-155	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ta-182	5.90E-08	2.14E-07	2.75E-07	1.88E-07	5.76E-08	2.27E-07	3.40E-07	2.54E-07

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	N Rx 1 (decay time/delay time)				N Rx 2 (decay time/delay time)			
	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d
Mn-54	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	1.86E-08	9.67E-08	4.63E-07	1.22E-06	4.06E-08	1.80E-07	6.85E-07	1.50E-06
Sr-89/Y-89m	2.88E-03	1.00E-02	7.40E-03	2.44E-03	2.90E-03	8.62E-03	5.06E-03	1.38E-03
Sr-90	7.18E-05	3.77E-04	1.88E-03	5.20E-03	1.52E-04	6.81E-04	2.70E-03	6.19E-03
Y-90	7.18E-05	3.77E-04	1.88E-03	3.56E-03	1.52E-04	6.81E-04	2.70E-03	4.24E-03
Y-91	1.64E-06	6.06E-06	5.81E-06	2.56E-06	1.78E-06	5.57E-06	4.25E-06	1.55E-06
Zr-95	1.29E-05	4.92E-05	5.43E-05	2.81E-05	1.52E-05	4.93E-05	4.33E-05	1.85E-05
Nb-95m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99/Tc-99m	3.02E-09	8.20E-12	1.88E-26	4.17E-43	2.76E-09	6.37E-12	1.16E-26	2.13E-43
Ru-103	1.27E-04	3.95E-04	1.68E-04	2.99E-05	1.55E-04	4.10E-04	1.39E-04	2.04E-05
Ru-106/Rh-106	3.74E-05	1.86E-04	7.20E-04	1.51E-03	1.08E-04	4.56E-04	1.41E-03	2.43E-03
Cd-113m	2.05E-12	1.08E-11	5.31E-11	1.46E-10	2.82E-12	1.26E-11	4.94E-11	1.12E-10
Cd-115m	1.24E-10	4.10E-10	2.34E-10	5.80E-11	1.74E-10	4.88E-10	2.22E-10	4.54E-11
Sb-125	1.16E-06	6.08E-06	2.78E-05	6.98E-05	2.94E-06	1.30E-05	4.73E-05	9.83E-05
Te-125m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-129m/Te-129	3.01E-05	8.52E-05	2.39E-05	2.67E-06	3.60E-05	8.68E-05	1.93E-05	1.79E-06
Te-132	3.03E-07	2.70E-09	1.57E-21	1.70E-35	2.84E-07	2.15E-09	9.96E-22	8.89E-36
Cs-134	1.18E-05	6.04E-05	2.67E-04	6.48E-04	6.80E-05	2.96E-04	1.04E-03	2.09E-03
Cs-136/Ba-136m	1.33E-06	1.44E-06	4.55E-09	3.44E-12	2.51E-06	2.32E-06	5.82E-09	3.64E-12
Cs-137/Ba-137m	1.30E-04	6.81E-04	3.39E-03	9.41E-03	2.96E-04	1.32E-03	5.25E-03	1.20E-02
Ba-140	1.29E-04	1.32E-04	3.30E-07	1.91E-10	1.15E-04	1.00E-04	1.99E-07	9.51E-11
La-140	1.48E-04	1.53E-04	3.80E-07	9.85E-11	1.32E-04	1.16E-04	2.29E-07	4.92E-11
Ce-141	1.11E-08	3.07E-08	7.80E-09	7.84E-10	1.08E-08	2.56E-08	5.16E-09	4.29E-10
Ce-144/Pr-144	1.61E-07	7.87E-07	2.82E-06	5.40E-06	2.92E-07	1.22E-06	3.47E-06	5.49E-06
Pr-143	2.27E-07	2.58E-07	1.02E-09	9.87E-13	1.98E-07	1.92E-07	6.01E-10	4.82E-13
Pr-144	0.00E+00	0.00E+00	0.00E+00	3.53E-35	0.00E+00	0.00E+00	0.00E+00	3.60E-35
Nd-147	3.80E-08	3.01E-08	2.20E-11	3.25E-15	3.41E-08	2.30E-08	1.34E-11	1.63E-15
Pm-147	2.24E-07	1.24E-06	5.69E-06	1.42E-05	4.72E-07	2.13E-06	7.75E-06	1.60E-05
Pm-148m	1.53E-09	4.86E-09	2.33E-09	4.76E-10	3.55E-09	9.61E-09	3.67E-09	6.20E-10
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	1.48E-08	7.73E-08	3.77E-07	1.02E-06	8.66E-08	3.85E-07	1.50E-06	3.34E-06
Eu-155	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ta-182	6.53E-12	2.87E-11	6.20E-11	6.79E-11	2.44E-11	9.11E-11	1.57E-10	1.42E-10

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	TRIGA 1 (decay time/delay time)				TRIGA 2 (decay time/delay time)			
	10 d/30 d	40 d/30 d	180 d/1 d	1 y/1 d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d
Mn-54	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	3.35E-05	1.14E-04	2.94E-04	4.68E-04	2.68E-06	1.32E-05	6.13E-05	1.64E-04
Sr-89/Y-89m	3.44E-03	7.81E-03	4.58E-03	6.17E-04	3.24E-03	1.07E-02	7.66E-03	2.55E-03
Sr-90	1.21E-03	4.15E-03	1.11E-02	1.86E-02	8.02E-05	4.00E-04	1.93E-03	5.41E-03
Y-90	1.21E-03	4.15E-03	7.58E-03	1.28E-02	8.03E-05	4.00E-04	1.93E-03	3.71E-03
Y-91	2.06E-06	4.96E-06	3.58E-06	6.81E-07	1.82E-06	6.37E-06	5.91E-06	2.64E-06
Zr-95	1.74E-05	4.31E-05	3.48E-05	8.01E-06	1.44E-05	5.21E-05	5.57E-05	2.92E-05
Nb-95m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99/Tc-99m	2.63E-09	4.66E-12	8.49E-24	7.71E-44	3.08E-09	7.93E-12	1.76E-26	3.96E-43
Ru-103	1.11E-04	2.24E-04	8.45E-05	5.49E-06	1.08E-04	3.17E-04	1.31E-04	2.35E-05
Ru-106/Rh-106	9.59E-05	3.11E-04	6.79E-04	8.19E-04	1.79E-05	8.47E-05	3.18E-04	6.73E-04
Cd-113m	1.68E-11	5.75E-11	1.52E-10	2.53E-10	1.18E-11	5.87E-11	2.80E-10	7.78E-10
Cd-115m	9.96E-11	2.14E-10	1.02E-10	9.85E-12	8.92E-11	2.80E-10	1.55E-10	3.88E-11
Sb-125	1.32E-05	4.44E-05	1.10E-04	1.65E-04	1.43E-06	7.07E-06	3.13E-05	7.95E-05
Te-125m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-129m/Te-129	2.49E-05	4.60E-05	1.25E-05	4.69E-07	2.52E-05	6.78E-05	1.84E-05	2.08E-06
Te-132	2.56E-07	1.48E-09	2.21E-19	3.03E-36	2.97E-07	2.51E-09	1.42E-21	1.55E-35
Cs-134	2.17E-04	7.22E-04	1.75E-03	2.52E-03	9.76E-07	4.74E-06	2.03E-05	4.99E-05
Cs-136/Ba-136m	1.97E-06	1.39E-06	1.08E-08	1.08E-12	4.25E-07	4.37E-07	1.33E-09	1.02E-12
Cs-137/Ba-137m	2.04E-03	6.97E-03	1.86E-02	3.13E-02	1.33E-04	6.61E-04	3.19E-03	8.95E-03
Ba-140	1.15E-04	7.70E-05	4.93E-07	3.60E-11	1.34E-04	1.31E-04	3.16E-07	1.85E-10
La-140	1.32E-04	8.86E-05	2.55E-07	1.86E-11	1.54E-04	1.51E-04	3.64E-07	9.56E-11
Ce-141	1.08E-08	1.95E-08	4.92E-09	1.62E-10	1.15E-08	3.04E-08	7.47E-09	7.60E-10
Ce-144/Pr-144	5.96E-07	1.90E-06	3.89E-06	4.23E-06	1.72E-07	7.97E-07	2.76E-06	5.35E-06
Pr-143	2.10E-07	1.56E-07	1.44E-09	1.94E-13	2.52E-07	2.72E-07	1.04E-09	1.02E-12
Pr-144	0.00E+00	0.00E+00	2.55E-35	2.77E-35	0.00E+00	0.00E+00	0.00E+00	3.51E-35
Nd-147	3.37E-08	1.74E-08	4.23E-11	6.11E-16	3.94E-08	2.96E-08	2.09E-11	3.14E-15
Pm-147	2.09E-06	7.06E-06	1.75E-05	2.61E-05	2.39E-07	1.25E-06	5.57E-06	1.41E-05
Pm-148m	7.40E-09	1.53E-08	6.39E-09	4.88E-10	4.46E-10	1.35E-09	6.25E-10	1.29E-10
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	3.24E-07	1.10E-06	2.89E-06	4.73E-06	1.12E-09	5.57E-09	2.63E-08	7.20E-08
Eu-155	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ta-182	3.54E-07	1.01E-06	1.40E-06	7.80E-07	2.87E-07	1.19E-06	2.50E-06	2.77E-06

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Table G-2. Contributions to gross beta activity for urinalysis following major chemical processing.

Nuclide	ATR 1 (decay time/delay time)				ATR 2 (decay time/delay time)				ATR 3 (decay time/delay time)			
	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d
Sr-89/Y-89m	3.42E-03	1.10E-02	7.69E-03	2.51E-03	1.88E-03	9.26E-03	9.52E-03	3.61E-03	2.39E-03	1.01E-02	9.08E-03	3.32E-03
Sr-90	9.24E-05	4.48E-04	2.11E-03	5.81E-03	2.99E-05	2.21E-04	1.54E-03	4.89E-03	4.28E-05	2.73E-04	1.65E-03	5.07E-03
Y-90	9.24E-05	4.48E-04	2.11E-03	3.98E-03	2.99E-05	2.21E-04	1.54E-03	3.35E-03	4.28E-05	2.73E-04	1.65E-03	3.47E-03
Y-91	1.94E-06	6.59E-06	5.98E-06	2.61E-06	1.02E-06	5.29E-06	7.05E-06	3.57E-06	1.30E-06	5.81E-06	6.76E-06	3.30E-06
Ba-140	1.34E-04	1.27E-04	3.00E-07	1.72E-10	1.36E-04	1.97E-04	6.84E-07	4.54E-10	1.40E-04	1.75E-04	5.29E-07	3.39E-10
La-140	1.54E-04	1.47E-04	3.46E-07	8.89E-11	1.56E-04	2.27E-04	7.88E-07	2.35E-10	1.61E-04	2.01E-04	6.09E-07	1.75E-10
Ce-141	1.18E-08	3.02E-08	7.26E-09	7.24E-10	7.91E-09	3.10E-08	1.10E-08	1.27E-09	9.58E-09	3.23E-08	9.96E-09	1.11E-09
Ce-144/Pr-144	1.94E-07	8.74E-07	2.96E-06	5.63E-06	6.99E-08	4.83E-07	2.40E-06	5.29E-06	9.78E-08	5.81E-07	2.53E-06	5.36E-06
Pr-143	2.48E-07	2.60E-07	9.72E-10	9.34E-13	2.41E-07	3.88E-07	2.13E-09	2.38E-12	2.57E-07	3.56E-07	1.70E-09	1.83E-12
Pr-144	0.00E+00	0.00E+00	0.00E+00	3.69E-35	0.00E+00	0.00E+00	0.00E+00	3.47E-35	0.00E+00	0.00E+00	0.00E+00	3.51E-35
Nd-147	3.70E-08	2.70E-08	1.87E-11	2.74E-15	3.55E-08	3.97E-08	4.03E-11	6.86E-15	4.01E-08	3.86E-08	3.42E-11	5.61E-15
Pm-147	2.07E-07	1.06E-06	4.63E-06	1.15E-05	5.84E-08	5.35E-07	3.51E-06	1.01E-05	1.07E-07	7.77E-07	4.40E-06	1.22E-05
Pm-148m	4.34E-09	1.27E-08	5.79E-09	1.17E-09	1.00E-09	4.51E-09	3.01E-09	7.06E-10	2.12E-09	8.21E-09	4.78E-09	1.08E-09
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	8.20E-08	3.95E-07	1.83E-06	4.90E-06	2.27E-08	1.67E-07	1.14E-06	3.53E-06	1.34E-08	8.53E-08	5.05E-07	1.51E-06
Eu-155	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Nuclide	FFTF 1 (decay time/delay time)				FFTF 2 (decay time/delay time)			
	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d
Sr-89/Y-89m	1.25E-03	3.59E-03	1.58E-03	3.26E-04	1.29E-03	4.04E-03	2.06E-03	4.68E-04
Sr-90	1.87E-04	8.10E-04	2.41E-03	4.19E-03	1.02E-04	4.81E-04	1.66E-03	3.17E-03
Y-90	1.87E-04	8.10E-04	2.41E-03	2.87E-03	1.02E-04	4.81E-04	1.66E-03	2.17E-03
Y-91	8.80E-07	2.68E-06	1.53E-06	4.24E-07	9.04E-07	3.00E-06	1.99E-06	6.05E-07
Ba-140	9.22E-05	7.84E-05	1.17E-07	4.23E-11	9.47E-05	8.78E-05	1.52E-07	6.03E-11
La-140	1.06E-04	9.03E-05	1.34E-07	2.19E-11	1.09E-04	1.01E-04	1.74E-07	3.12E-11
Ce-141	8.90E-09	2.04E-08	3.10E-09	1.96E-10	9.11E-09	2.28E-08	4.01E-09	2.78E-10
Ce-144/Pr-144	3.49E-07	1.41E-06	3.01E-06	3.62E-06	2.75E-07	1.21E-06	3.00E-06	3.96E-06
Pr-143	1.46E-07	1.37E-07	3.24E-10	1.97E-13	1.48E-07	1.52E-07	4.15E-10	2.77E-13
Pr-144	0.00E+00	0.00E+00	0.00E+00	2.37E-35	0.00E+00	0.00E+00	0.00E+00	2.60E-35
Nd-147	3.02E-08	1.97E-08	8.60E-12	7.98E-16	2.98E-08	2.13E-08	1.08E-11	1.10E-15
Pm-147	9.33E-07	4.02E-06	1.09E-05	1.72E-05	6.36E-07	3.01E-06	9.51E-06	1.64E-05
Pm-148m	4.15E-08	1.09E-07	3.12E-08	4.00E-09	2.12E-08	6.08E-08	2.02E-08	2.84E-09
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	1.47E-06	6.35E-06	1.85E-05	3.14E-05	5.22E-07	2.46E-06	8.31E-06	1.55E-05
Eu-155	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	N Rx 1 (decay time/delay time)				N Rx 2 (decay time/delay time)			
	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d
Sr-89/Y-89m	2.88E-03	1.00E-02	7.40E-03	2.44E-03	2.90E-03	8.62E-03	5.06E-03	1.38E-03
Sr-90	7.18E-05	3.77E-04	1.88E-03	5.20E-03	1.52E-04	6.81E-04	2.70E-03	6.19E-03
Y-90	7.18E-05	3.77E-04	1.88E-03	3.56E-03	1.52E-04	6.81E-04	2.70E-03	4.24E-03
Y-91	1.64E-06	6.06E-06	5.81E-06	2.56E-06	1.78E-06	5.57E-06	4.25E-06	1.55E-06
Ba-140	1.29E-04	1.32E-04	3.30E-07	1.91E-10	1.15E-04	1.00E-04	1.99E-07	9.51E-11
La-140	1.48E-04	1.53E-04	3.80E-07	9.85E-11	1.32E-04	1.16E-04	2.29E-07	4.92E-11
Ce-141	1.11E-08	3.07E-08	7.80E-09	7.84E-10	1.08E-08	2.56E-08	5.16E-09	4.29E-10
Ce-144/Pr-144	1.61E-07	7.87E-07	2.82E-06	5.40E-06	2.92E-07	1.22E-06	3.47E-06	5.49E-06
Pr-143	2.27E-07	2.58E-07	1.02E-09	9.87E-13	1.98E-07	1.92E-07	6.01E-10	4.82E-13
Pr-144	0.00E+00	0.00E+00	0.00E+00	3.53E-35	0.00E+00	0.00E+00	0.00E+00	3.60E-35
Nd-147	3.80E-08	3.01E-08	2.20E-11	3.25E-15	3.41E-08	2.30E-08	1.34E-11	1.63E-15
Pm-147	2.24E-07	1.24E-06	5.69E-06	1.42E-05	4.72E-07	2.13E-06	7.75E-06	1.60E-05
Pm-148m	1.53E-09	4.86E-09	2.33E-09	4.76E-10	3.55E-09	9.61E-09	3.67E-09	6.20E-10
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	1.48E-08	7.73E-08	3.77E-07	1.02E-06	8.66E-08	3.85E-07	1.50E-06	3.34E-06
Eu-155	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Nuclide	TRIGA 1 (decay time/delay time)				TRIGA 2 (decay time/delay time)			
	10 d/30 d	40 d/30 d	180 d/1 d	1 y/1 d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/1 d
Sr-89/Y-89m	3.44E-03	7.81E-03	4.58E-03	6.17E-04	3.24E-03	1.07E-02	7.66E-03	2.55E-03
Sr-90	1.21E-03	4.15E-03	1.11E-02	1.86E-02	8.02E-05	4.00E-04	1.93E-03	5.41E-03
Y-90	1.21E-03	4.15E-03	7.58E-03	1.28E-02	8.03E-05	4.00E-04	1.93E-03	3.71E-03
Y-91	2.06E-06	4.96E-06	3.58E-06	6.81E-07	1.82E-06	6.37E-06	5.91E-06	2.64E-06
Ba-140	1.15E-04	7.70E-05	4.93E-07	3.60E-11	1.34E-04	1.31E-04	3.16E-07	1.85E-10
La-140	1.32E-04	8.86E-05	2.55E-07	1.86E-11	1.54E-04	1.51E-04	3.64E-07	9.56E-11
Ce-141	1.08E-08	1.95E-08	4.92E-09	1.62E-10	1.15E-08	3.04E-08	7.47E-09	7.60E-10
Ce-144/Pr-144	5.96E-07	1.90E-06	3.89E-06	4.23E-06	1.72E-07	7.97E-07	2.76E-06	5.35E-06
Pr-143	2.10E-07	1.56E-07	1.44E-09	1.94E-13	2.52E-07	2.72E-07	1.04E-09	1.02E-12
Pr-144	0.00E+00	0.00E+00	2.55E-35	2.77E-35	0.00E+00	0.00E+00	0.00E+00	3.51E-35
Nd-147	3.37E-08	1.74E-08	4.23E-11	6.11E-16	3.94E-08	2.96E-08	2.09E-11	3.14E-15
Pm-147	2.09E-06	7.06E-06	1.75E-05	2.61E-05	2.39E-07	1.25E-06	5.57E-06	1.41E-05
Pm-148m	7.40E-09	1.53E-08	6.39E-09	4.88E-10	4.46E-10	1.35E-09	6.25E-10	1.29E-10
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	3.24E-07	1.10E-06	2.89E-06	4.73E-06	1.12E-09	5.57E-09	2.63E-08	7.20E-08
Eu-155	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Table G-3. Contributions to gross gamma activity for an unprocessed urinalysis.

	ATR 1 (decay time/delay time)				ATR 2 (decay time/delay time)				ATR 3 (decay time/delay time)			
Nuclide	**10 d/30 d**	**40 d/30 d**	**180 d/30 d**	**1 y/30 d**	**10 d/30 d**	**40 d/30 d**	**180 d/30 d**	**1 y/30 d**	**10 d/30 d**	**40 d/30 d**	**180 d/30 d**	**1 y/30 d**
Mn-54	7.75E-09	3.52E-08	1.23E-07	2.26E-07	2.74E-09	1.91E-08	9.78E-08	2.09E-07	3.77E-09	2.25E-08	1.01E-07	2.07E-07
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	3.30E-09	1.20E-08	1.45E-08	6.59E-09	1.25E-09	6.94E-09	1.24E-08	6.53E-09	3.28E-09	1.57E-08	2.43E-08	1.24E-08
Co-60	3.26E-09	1.56E-08	7.09E-08	1.84E-07	1.06E-09	7.81E-09	5.20E-08	1.57E-07	1.45E-09	9.17E-09	5.33E-08	1.55E-07
Sr-89/Y-89m	3.18E-07	1.02E-06	7.15E-07	1.57E-07	1.75E-07	8.61E-07	8.85E-07	2.25E-07	2.22E-07	9.43E-07	8.44E-07	2.07E-07
Sr-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	5.81E-09	1.98E-08	1.80E-08	5.56E-09	3.05E-09	1.59E-08	2.12E-08	7.61E-09	3.89E-09	1.74E-08	2.03E-08	7.03E-09
Zr-95	1.45E-05	5.10E-05	5.33E-05	2.00E-05	7.28E-06	3.91E-05	6.01E-05	2.61E-05	9.46E-06	4.37E-05	5.87E-05	2.46E-05
Nb-95m	4.43E-08	1.55E-07	1.63E-07	6.08E-08	2.22E-08	1.19E-07	1.83E-07	7.95E-08	2.88E-08	1.33E-07	1.79E-07	7.48E-08
Nb-95	4.47E-05	1.99E-04	3.11E-04	1.29E-04	1.48E-05	1.22E-04	3.36E-04	1.67E-04	2.21E-05	1.46E-04	3.32E-04	1.57E-04
Mo-99/Tc-99m	3.25E-09	8.15E-12	1.77E-26	2.61E-46	3.85E-09	1.48E-11	4.70E-26	8.07E-46	3.60E-09	1.19E-11	3.30E-26	5.46E-46
Ru-103	1.04E-04	2.98E-04	1.20E-04	1.27E-05	6.40E-05	2.80E-04	1.66E-04	2.03E-05	7.90E-05	2.97E-04	1.53E-04	1.82E-05
Ru-106/Rh-106	3.51E-06	1.61E-05	5.91E-05	1.16E-04	1.23E-06	8.62E-06	4.65E-05	1.06E-04	1.70E-06	1.03E-05	4.84E-05	1.06E-04
Cd-113m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cd-115m	2.07E-12	6.30E-12	3.41E-12	5.33E-13	1.19E-12	5.54E-12	4.41E-12	8.00E-13	1.50E-12	6.00E-12	4.16E-12	7.28E-13
Sb-125	9.19E-07	4.43E-06	1.92E-05	4.68E-05	2.95E-07	2.25E-06	1.43E-05	4.06E-05	4.25E-07	2.74E-06	1.53E-05	4.17E-05
Te-125m	3.97E-10	1.91E-09	8.29E-09	2.02E-08	1.27E-10	9.69E-10	6.19E-09	1.75E-08	1.83E-10	1.18E-09	6.59E-09	1.80E-08
Te-129m/Te-129	2.96E-06	7.73E-06	2.05E-06	1.25E-07	1.95E-06	7.81E-06	3.05E-06	2.16E-07	2.35E-06	8.11E-06	2.76E-06	1.88E-07
Te-132	2.70E-07	2.21E-09	1.22E-21	2.74E-38	3.19E-07	4.01E-09	3.25E-21	8.44E-38	2.98E-07	3.22E-09	2.28E-21	5.70E-38
Cs-134	2.20E-04	1.04E-03	4.35E-03	1.02E-02	5.14E-05	3.72E-04	2.29E-03	6.21E-03	3.44E-05	2.14E-04	1.15E-03	3.00E-03
Cs-136/Ba-136m	3.46E-06	3.46E-06	1.04E-08	1.68E-12	1.58E-06	2.42E-06	1.06E-08	2.01E-12	1.85E-06	2.44E-06	9.37E-09	1.70E-12
Cs-137/Ba-137m	1.37E-04	6.66E-04	3.15E-03	8.63E-03	4.44E-05	3.29E-04	2.28E-03	7.27E-03	6.35E-05	4.05E-04	2.45E-03	7.53E-03
Ba-140	5.41E-05	5.14E-05	1.21E-07	1.44E-11	5.48E-05	7.97E-05	2.76E-07	3.81E-11	5.64E-05	7.07E-05	2.14E-07	2.84E-11
La-140	3.32E-04	3.16E-04	7.45E-07	8.86E-11	3.37E-04	4.89E-04	1.70E-06	2.34E-10	3.47E-04	4.34E-04	1.31E-06	1.74E-10
Ce-141	5.66E-09	1.45E-08	3.49E-09	1.87E-10	3.80E-09	1.49E-08	5.26E-09	3.27E-10	4.60E-09	1.55E-08	4.78E-09	2.87E-10
Ce-144/Pr-144	1.31E-08	5.90E-08	2.00E-07	3.54E-07	4.71E-09	3.26E-08	1.62E-07	3.33E-07	6.60E-09	3.92E-08	1.70E-07	3.37E-07
Pr-143	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nd-147	7.38E-09	5.39E-09	3.73E-12	8.77E-17	7.09E-09	7.93E-09	8.06E-12	2.20E-16	8.01E-09	7.71E-09	6.83E-12	1.79E-16
Pm-147	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-148m	1.46E-08	4.29E-08	1.95E-08	2.42E-09	3.39E-09	1.52E-08	1.01E-08	1.46E-09	7.15E-09	2.77E-08	1.61E-08	2.24E-09
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	1.34E-07	6.46E-07	2.98E-06	7.95E-06	3.70E-08	2.73E-07	1.86E-06	5.73E-06	2.19E-08	1.39E-07	8.25E-07	2.46E-06
Eu-155	9.47E-09	4.54E-08	2.05E-07	5.26E-07	2.74E-09	2.01E-08	1.33E-07	3.97E-07	4.31E-09	2.72E-08	1.57E-07	4.52E-07
Ta-182	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	FFTF 1 (decay time/delay time)				FFTF 2 (decay time/delay time)			
	10 d/30 d	40 d/30 d	180 d/30 d	1 y/30 d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/30 d
Mn-54	4.75E-05	1.93E-04	4.25E-04	4.95E-04	3.30E-05	1.46E-04	3.74E-04	4.78E-04
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	1.16E-04	3.75E-04	2.86E-04	8.24E-05	9.99E-05	3.53E-04	3.13E-04	9.88E-05
Co-60	1.86E-06	8.01E-06	2.29E-05	3.76E-05	9.05E-07	4.24E-06	1.41E-05	2.54E-05
Sr-89/Y-89m	1.16E-07	3.33E-07	1.47E-07	2.04E-08	1.20E-07	3.75E-07	1.92E-07	2.92E-08
Sr-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	2.64E-09	8.04E-09	4.60E-09	9.03E-10	2.71E-09	9.00E-09	5.98E-09	1.29E-09
Zr-95	1.10E-05	3.47E-05	2.29E-05	5.43E-06	1.12E-05	3.85E-05	2.95E-05	7.67E-06
Nb-95m	3.37E-08	1.06E-07	6.98E-08	1.65E-08	3.42E-08	1.17E-07	8.99E-08	2.34E-08
Nb-95	4.09E-05	1.52E-04	1.36E-04	3.51E-05	4.14E-05	1.68E-04	1.76E-04	4.96E-05
Mo-99/Tc-99m	2.85E-09	6.39E-12	8.73E-27	8.17E-47	2.81E-09	6.86E-12	1.09E-26	1.12E-46
Ru-103	2.12E-04	5.42E-04	1.37E-04	9.21E-06	2.11E-04	5.88E-04	1.73E-04	1.27E-05
Ru-106/Rh-106	1.08E-04	4.43E-04	1.03E-03	1.28E-03	7.93E-05	3.55E-04	9.56E-04	1.31E-03
Cd-113m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cd-115m	1.95E-11	5.32E-11	1.81E-11	1.79E-12	1.88E-11	5.58E-11	2.21E-11	2.39E-12
Sb-125	1.85E-05	7.92E-05	2.16E-04	3.34E-04	1.16E-05	5.42E-05	1.72E-04	2.92E-04
Te-125m	7.98E-09	3.42E-08	9.33E-08	1.44E-07	4.99E-09	2.34E-08	7.42E-08	1.26E-07
Te-129m/Te-129	2.92E-08	6.84E-08	1.14E-08	4.43E-10	1.79E-08	4.57E-08	8.88E-09	3.77E-10
Te-132	2.47E-07	1.63E-09	3.44E-22	2.51E-39	2.55E-07	1.83E-09	4.48E-22	3.59E-39
Cs-134	1.97E-03	8.33E-03	2.20E-02	3.26E-02	5.68E-04	2.62E-03	8.04E-03	1.31E-02
Cs-136/Ba-136m	1.25E-04	1.12E-04	2.11E-07	2.18E-11	5.83E-05	5.69E-05	1.25E-07	1.41E-11
Cs-137/Ba-137m	7.95E-04	3.45E-03	1.03E-02	1.78E-02	4.31E-04	2.04E-03	7.04E-03	1.34E-02
Ba-140	3.72E-05	3.17E-05	4.71E-08	3.55E-12	3.82E-05	3.55E-05	6.12E-08	5.06E-12
La-140	2.29E-04	1.94E-04	2.89E-07	2.18E-11	2.35E-04	2.18E-04	3.76E-07	3.11E-11
Ce-141	4.27E-09	9.79E-09	1.49E-09	5.06E-11	4.37E-09	1.09E-08	1.93E-09	7.19E-11
Ce-144/Pr-144	2.35E-08	9.50E-08	2.03E-07	2.27E-07	1.86E-08	8.17E-08	2.03E-07	2.49E-07
Pr-143	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nd-147	6.02E-09	3.94E-09	1.72E-12	2.56E-17	5.95E-09	4.24E-09	2.15E-12	3.51E-17
Pm-147	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-148m	1.40E-07	3.67E-07	1.05E-07	8.28E-09	7.15E-08	2.05E-07	6.81E-08	5.88E-09
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	2.40E-06	1.04E-05	3.02E-05	5.09E-05	8.53E-07	4.01E-06	1.36E-05	2.51E-05
Eu-155	7.93E-07	3.40E-06	9.67E-06	1.58E-05	4.02E-07	1.88E-06	6.20E-06	1.11E-05
Ta-182	8.62E-08	3.12E-07	4.01E-07	2.30E-07	8.41E-08	3.32E-07	4.96E-07	3.12E-07

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	N Rx 1 (decay time/delay time)				N Rx 2 (decay time/delay time)			
	10 d/30 d	40 d/30 d	180 d/30 d	1 y/30 d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/30 d
Mn-54	6.96E-09	3.43E-08	1.26E-07	2.35E-07	1.38E-08	5.77E-08	1.69E-07	2.60E-07
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	2.02E-08	7.93E-08	1.01E-07	4.66E-08	2.06E-08	6.86E-08	7.00E-08	2.66E-08
Co-60	3.72E-08	1.93E-07	9.25E-07	2.42E-06	8.13E-08	3.60E-07	1.37E-06	2.97E-06
Sr-89/Y-89m	2.68E-07	9.34E-07	6.89E-07	1.53E-07	2.70E-07	8.02E-07	4.71E-07	8.63E-08
Sr-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	4.93E-09	1.82E-08	1.74E-08	5.45E-09	5.33E-09	1.67E-08	1.27E-08	3.30E-09
Zr-95	1.29E-05	4.90E-05	5.41E-05	2.04E-05	1.52E-05	4.91E-05	4.32E-05	1.35E-05
Nb-95m	3.92E-08	1.49E-07	1.65E-07	6.23E-08	4.62E-08	1.50E-07	1.32E-07	4.11E-08
Nb-95	3.81E-05	1.87E-04	3.13E-04	1.31E-04	5.47E-05	2.11E-04	2.56E-04	8.71E-05
Mo-99/Tc-99m	3.25E-09	8.81E-12	2.01E-26	3.01E-46	2.96E-09	6.84E-12	1.25E-26	1.54E-46
Ru-103	1.19E-04	3.67E-04	1.56E-04	1.67E-05	1.45E-04	3.81E-04	1.29E-04	1.14E-05
Ru-106/Rh-106	6.39E-06	3.18E-05	1.23E-04	2.44E-04	1.84E-05	7.80E-05	2.40E-04	3.94E-04
Cd-113m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cd-115m	2.76E-12	9.10E-12	5.20E-12	8.21E-13	3.85E-12	1.08E-11	4.92E-12	6.43E-13
Sb-125	9.90E-07	5.20E-06	2.38E-05	5.86E-05	2.51E-06	1.11E-05	4.05E-05	8.25E-05
Te-125m	4.27E-10	2.25E-09	1.03E-08	2.53E-08	1.09E-09	4.81E-09	1.75E-08	3.56E-08
Te-129m/Te-129	3.41E-06	9.66E-06	2.71E-06	1.67E-07	4.08E-06	9.84E-06	2.19E-06	1.12E-07
Te-132	2.79E-07	2.48E-09	1.45E-21	3.26E-38	2.61E-07	1.97E-09	9.16E-22	1.71E-38
Cs-134	2.63E-05	1.35E-04	5.95E-04	1.41E-03	1.52E-04	6.60E-04	2.32E-03	4.54E-03
Cs-136/Ba-136m	3.83E-06	4.15E-06	1.31E-08	2.15E-12	7.24E-06	6.68E-06	1.68E-08	2.28E-12
Cs-137/Ba-137m	1.17E-04	6.12E-04	3.05E-03	8.45E-03	2.66E-04	1.19E-03	4.72E-03	1.08E-02
Ba-140	5.19E-05	5.35E-05	1.33E-07	1.60E-11	4.63E-05	4.06E-05	8.04E-08	7.98E-12
La-140	3.19E-04	3.29E-04	8.18E-07	9.81E-11	2.84E-04	2.49E-04	4.94E-07	4.90E-11
Ce-141	5.31E-09	1.47E-08	3.74E-09	2.03E-10	5.20E-09	1.23E-08	2.48E-09	1.11E-10
Ce-144/Pr-144	1.09E-08	5.31E-08	1.90E-07	3.39E-07	1.97E-08	8.22E-08	2.34E-07	3.45E-07
Pr-143	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nd-147	7.59E-09	6.01E-09	4.39E-12	1.04E-16	6.82E-09	4.59E-09	2.67E-12	5.23E-17
Pm-147	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-148m	5.15E-09	1.64E-08	7.86E-09	9.85E-10	1.20E-08	3.24E-08	1.24E-08	1.28E-09
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	2.42E-08	1.26E-07	6.16E-07	1.65E-06	1.41E-07	6.29E-07	2.44E-06	5.43E-06
Eu-155	1.78E-08	9.24E-08	4.39E-07	1.14E-06	3.38E-08	1.50E-07	5.66E-07	1.22E-06
Ta-182	9.54E-12	4.19E-11	9.07E-11	8.33E-11	3.56E-11	1.33E-10	2.29E-10	1.74E-10

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	TRIGA 1 (decay time/delay time)				TRIGA 2 (decay time/delay time)			
	10 d/30 d	40 d/30 d	180 d/30 d	1 y/30 d	10 d/30 d	40 d/30 d	180 d/30 d	1 y/30 d
Mn-54	3.52E-06	1.13E-05	2.22E-05	2.51E-05	8.80E-07	4.11E-06	1.47E-05	2.76E-05
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	2.52E-06	6.44E-06	4.41E-06	1.23E-06	1.91E-06	7.11E-06	8.82E-06	4.10E-06
Co-60	6.71E-05	2.28E-04	5.81E-04	9.26E-04	5.36E-06	2.65E-05	1.23E-04	3.25E-04
Sr-89/Y-89m	3.20E-07	7.26E-07	2.86E-07	3.86E-08	3.01E-07	9.97E-07	7.12E-07	1.60E-07
Sr-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	6.19E-09	1.49E-08	7.61E-09	1.45E-09	5.46E-09	1.91E-08	1.77E-08	5.61E-09
Zr-95	1.73E-05	4.30E-05	2.53E-05	5.83E-06	1.44E-05	5.19E-05	5.55E-05	2.12E-05
Nb-95m	5.29E-08	1.31E-07	7.72E-08	1.78E-08	4.38E-08	1.58E-07	1.69E-07	6.47E-08
Nb-95	6.42E-05	1.88E-04	1.51E-04	3.77E-05	4.25E-05	1.98E-04	3.22E-04	1.37E-04
Mo-99/Tc-99m	2.83E-09	5.00E-12	6.12E-27	5.56E-47	3.31E-09	8.52E-12	1.89E-26	2.85E-46
Ru-103	1.03E-04	2.08E-04	4.72E-05	3.07E-06	1.00E-04	2.95E-04	1.21E-04	1.31E-05
Ru-106/Rh-106	1.64E-05	5.32E-05	1.10E-04	1.33E-04	3.07E-06	1.45E-05	5.43E-05	1.09E-04
Cd-113m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cd-115m	2.21E-12	4.75E-12	1.45E-12	1.39E-13	1.98E-12	6.21E-12	3.43E-12	5.48E-13
Sb-125	1.13E-05	3.80E-05	9.27E-05	1.39E-04	1.23E-06	6.06E-06	2.68E-05	6.68E-05
Te-125m	4.88E-09	1.64E-08	4.00E-08	5.99E-08	5.29E-10	2.61E-09	1.16E-08	2.88E-08
Te-129m/Te-129	2.82E-06	5.22E-06	7.80E-07	2.93E-08	2.86E-06	7.69E-06	2.08E-06	1.30E-07
Te-132	2.35E-07	1.36E-09	4.25E-22	5.84E-39	2.73E-07	2.31E-09	1.30E-21	2.98E-38
Cs-134	4.83E-04	1.61E-03	3.80E-03	5.46E-03	2.18E-06	1.06E-05	4.53E-05	1.08E-04
Cs-136/Ba-136m	5.68E-06	4.01E-06	6.75E-09	6.74E-13	1.22E-06	1.26E-06	3.84E-09	6.38E-13
Cs-137/Ba-137m	1.83E-03	6.27E-03	1.67E-02	2.81E-02	1.19E-04	5.95E-04	2.87E-03	8.04E-03
Ba-140	4.63E-05	3.11E-05	4.13E-08	3.02E-12	5.42E-05	5.30E-05	1.28E-07	1.55E-11
La-140	2.85E-04	1.91E-04	2.54E-07	1.85E-11	3.33E-04	3.25E-04	7.84E-07	9.52E-11
Ce-141	5.18E-09	9.37E-09	1.27E-09	4.19E-11	5.54E-09	1.46E-08	3.59E-09	1.96E-10
Ce-144/Pr-144	4.02E-08	1.28E-07	2.45E-07	2.66E-07	1.16E-08	5.38E-08	1.86E-07	3.36E-07
Pr-143	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nd-147	6.74E-09	3.47E-09	1.35E-12	1.96E-17	7.87E-09	5.92E-09	4.18E-12	1.00E-16
Pm-147	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-148m	2.49E-08	5.17E-08	1.32E-08	1.01E-09	1.50E-09	4.54E-09	2.11E-09	2.67E-10
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	5.29E-07	1.80E-06	4.69E-06	7.68E-06	1.83E-09	9.10E-09	4.30E-08	1.17E-07
Eu-155	1.32E-07	4.46E-07	1.13E-06	1.79E-06	1.93E-08	9.52E-08	4.38E-07	1.15E-06
Ta-182	5.18E-07	1.48E-06	1.71E-06	9.57E-07	4.19E-07	1.75E-06	3.66E-06	3.40E-06

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Table G-4. Contributions to gross beta activity for an air or swipe sample.

Nuclide	ATR 1				ATR 2				ATR 3			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
Mn-54	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	8.21E-08	3.94E-07	1.79E-06	4.64E-06	2.68E-08	1.97E-07	1.31E-06	3.95E-06	3.65E-08	2.31E-07	1.34E-06	3.89E-06
Sr-89/Y-89m	2.69E-02	8.67E-02	6.06E-02	1.33E-02	1.48E-02	7.30E-02	7.50E-02	1.91E-02	1.88E-02	7.99E-02	7.16E-02	1.76E-02
Sr-90	4.01E-04	1.94E-03	9.17E-03	2.51E-02	1.30E-04	9.60E-04	6.66E-03	2.12E-02	1.85E-04	1.18E-03	7.16E-03	2.20E-02
Y-90	4.02E-04	1.94E-03	9.18E-03	2.51E-02	1.29E-04	9.61E-04	6.66E-03	2.12E-02	1.85E-04	1.18E-03	7.17E-03	2.20E-02
Y-91	3.22E-02	1.10E-01	9.95E-02	3.08E-02	1.69E-02	8.79E-02	1.17E-01	4.22E-02	2.15E-02	9.66E-02	1.12E-01	3.89E-02
Zr-95	3.48E-02	1.22E-01	1.28E-01	4.78E-02	1.74E-02	9.35E-02	1.44E-01	6.25E-02	2.26E-02	1.05E-01	1.40E-01	5.88E-02
Nb-95	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99/Tc-99m	3.78E-03	9.46E-06	2.05E-20	3.03E-40	4.47E-03	1.71E-05	5.46E-20	9.36E-40	4.18E-03	1.38E-05	3.83E-20	6.33E-40
Ru-103	1.80E-02	5.15E-02	2.07E-02	2.19E-03	1.10E-02	4.83E-02	2.86E-02	3.51E-03	1.36E-02	5.14E-02	2.65E-02	3.13E-03
Ru-106/Rh-106	1.39E-03	6.39E-03	2.35E-02	4.61E-02	4.87E-04	3.42E-03	1.85E-02	4.21E-02	6.75E-04	4.09E-03	1.92E-02	4.22E-02
Cd-113m	7.23E-10	3.49E-09	1.63E-08	4.42E-08	2.88E-10	2.13E-09	1.46E-08	4.59E-08	1.04E-09	6.62E-09	3.97E-08	1.20E-07
Cd-115m	2.79E-06	8.49E-06	4.59E-06	7.19E-07	1.60E-06	7.47E-06	5.94E-06	1.08E-06	2.02E-06	8.09E-06	5.61E-06	9.81E-07
Sb-125	2.32E-05	1.12E-04	4.84E-04	1.18E-03	7.43E-06	5.66E-05	3.62E-04	1.02E-03	1.07E-05	6.92E-05	3.85E-04	1.05E-03
Te-129m/Te-129	6.00E-04	1.57E-03	4.17E-04	2.54E-05	3.97E-04	1.59E-03	6.19E-04	4.38E-05	4.78E-04	1.65E-03	5.60E-04	3.82E-05
Te-132	3.95E-03	3.24E-05	1.79E-17	4.01E-34	4.67E-03	5.87E-05	4.76E-17	1.24E-33	4.36E-03	4.72E-05	3.34E-17	8.35E-34
Cs-134	3.00E-04	1.42E-03	5.93E-03	1.39E-02	7.00E-05	5.06E-04	3.12E-03	8.46E-03	4.68E-05	2.91E-04	1.56E-03	4.09E-03
Cs-136/Ba-136m	8.78E-05	8.77E-05	2.62E-07	4.27E-11	4.01E-05	6.13E-05	2.69E-07	5.08E-11	4.70E-05	6.20E-05	2.38E-07	4.32E-11
Cs-137/Ba-137m	4.05E-04	1.96E-03	9.27E-03	2.54E-02	1.31E-04	9.69E-04	6.73E-03	2.14E-02	1.87E-04	1.19E-03	7.23E-03	2.22E-02
Ba-140	2.81E-02	2.67E-02	6.30E-05	7.49E-09	2.85E-02	4.14E-02	1.44E-04	1.98E-08	2.93E-02	3.67E-02	1.11E-04	1.48E-08
La-140	3.23E-02	3.08E-02	7.25E-05	8.63E-09	3.27E-02	4.77E-02	1.65E-04	2.28E-08	3.36E-02	4.23E-02	1.28E-04	1.70E-08
Ce-141	3.45E-02	8.84E-02	2.13E-02	1.14E-03	2.32E-02	9.08E-02	3.21E-02	2.00E-03	2.81E-02	9.47E-02	2.92E-02	1.75E-03
Ce-144/Pr-144	2.29E-02	1.03E-01	3.50E-01	6.20E-01	8.26E-03	5.70E-02	2.84E-01	5.83E-01	1.16E-02	6.87E-02	2.99E-01	5.90E-01
Pr-143	3.05E-02	3.20E-02	1.19E-04	2.61E-08	2.97E-02	4.77E-02	2.62E-04	6.63E-08	3.16E-02	4.38E-02	2.10E-04	5.12E-08
Pr-144	1.16E-02	5.22E-02	1.77E-01	3.13E-01	4.17E-03	2.88E-02	1.44E-01	2.94E-01	5.84E-03	3.47E-02	1.51E-01	2.98E-01
Nd-147	8.74E-03	6.39E-03	4.42E-06	1.04E-10	8.40E-03	9.40E-03	9.54E-06	2.60E-10	9.48E-03	9.13E-03	8.09E-06	2.13E-10
Pm-147	1.13E-03	5.75E-03	2.51E-02	6.09E-02	3.17E-04	2.90E-03	1.91E-02	5.36E-02	5.81E-04	4.22E-03	2.39E-02	6.48E-02
Pm-148m	1.70E-04	5.00E-04	2.27E-04	2.82E-05	3.94E-05	1.77E-04	1.18E-04	1.70E-05	8.33E-05	3.22E-04	1.87E-04	2.61E-05
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	9.45E-06	4.56E-05	2.11E-04	5.61E-04	2.61E-06	1.93E-05	1.31E-04	4.05E-04	1.55E-06	9.83E-06	5.82E-05	1.73E-04
Eu-155	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ta-182	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	FFTF 1				FFTF 2			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
Mn-54	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	4.69E-05	2.02E-04	5.76E-04	9.47E-04	2.28E-05	1.07E-04	3.54E-04	6.39E-04
Sr-89/Y-89m	9.82E-03	2.83E-02	1.24E-02	1.73E-03	1.01E-02	3.18E-02	1.63E-02	2.48E-03
Sr-90	8.10E-04	3.51E-03	1.05E-02	1.82E-02	4.41E-04	2.09E-03	7.21E-03	1.37E-02
Y-90	8.15E-04	3.51E-03	1.05E-02	1.82E-02	4.43E-04	2.09E-03	7.21E-03	1.37E-02
Y-91	1.46E-02	4.46E-02	2.55E-02	5.01E-03	1.50E-02	4.99E-02	3.31E-02	7.14E-03
Zr-95	2.64E-02	8.30E-02	5.48E-02	1.30E-02	2.69E-02	9.21E-02	7.06E-02	1.84E-02
Nb-95	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99/Tc-99m	3.31E-03	7.42E-06	1.01E-20	9.49E-41	3.26E-03	7.96E-06	1.26E-20	1.30E-40
Ru-103	3.65E-02	9.35E-02	2.37E-02	1.59E-03	3.64E-02	1.02E-01	2.99E-02	2.20E-03
Ru-106/Rh-106	4.28E-02	1.76E-01	4.07E-01	5.08E-01	3.15E-02	1.41E-01	3.79E-01	5.19E-01
Cd-113m	2.11E-06	9.15E-06	2.70E-05	4.62E-05	8.57E-07	4.05E-06	1.38E-05	2.60E-05
Cd-115m	2.63E-05	7.18E-05	2.44E-05	2.41E-06	2.53E-05	7.52E-05	2.97E-05	3.22E-06
Sb-125	4.66E-04	2.00E-03	5.45E-03	8.43E-03	2.91E-04	1.37E-03	4.33E-03	7.35E-03
Te-129m/Te-129	5.94E-06	1.39E-05	2.32E-06	8.99E-08	3.64E-06	9.29E-06	1.80E-06	7.65E-08
Te-132	3.62E-03	2.39E-05	5.03E-18	3.67E-35	3.73E-03	2.68E-05	6.57E-18	5.26E-35
Cs-134	2.69E-03	1.13E-02	3.00E-02	4.44E-02	7.74E-04	3.57E-03	1.09E-02	1.78E-02
Cs-136/Ba-136m	3.18E-03	2.84E-03	5.36E-06	5.52E-10	1.48E-03	1.44E-03	3.16E-06	3.57E-10
Cs-137/Ba-137m	2.34E-03	1.01E-02	3.02E-02	5.25E-02	1.27E-03	6.00E-03	2.07E-02	3.95E-02
Ba-140	1.93E-02	1.65E-02	2.45E-05	1.84E-09	1.99E-02	1.84E-02	3.18E-05	2.63E-09
La-140	2.22E-02	1.89E-02	2.82E-05	2.12E-09	2.28E-02	2.12E-02	3.66E-05	3.03E-09
Ce-141	2.61E-02	5.98E-02	9.07E-03	3.09E-04	2.67E-02	6.67E-02	1.18E-02	4.39E-04
Ce-144/Pr-144	4.12E-02	1.66E-01	3.55E-01	3.98E-01	3.25E-02	1.43E-01	3.55E-01	4.37E-01
Pr-143	1.80E-02	1.69E-02	3.98E-05	5.50E-09	1.82E-02	1.87E-02	5.10E-05	7.74E-09
Pr-144	2.08E-02	8.41E-02	1.80E-01	2.01E-01	1.64E-02	7.23E-02	1.79E-01	2.20E-01
Nd-147	7.13E-03	4.67E-03	2.03E-06	3.03E-11	7.05E-03	5.03E-03	2.55E-06	4.16E-11
Pm-147	5.06E-03	2.18E-02	5.94E-02	9.12E-02	3.45E-03	1.63E-02	5.16E-02	8.70E-02
Pm-148m	1.63E-03	4.27E-03	1.22E-03	9.64E-05	8.33E-04	2.39E-03	7.93E-04	6.85E-05
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	1.70E-04	7.32E-04	2.13E-03	3.60E-03	6.02E-05	2.83E-04	9.58E-04	1.77E-03
Eu-155	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ta-182	1.54E-04	5.57E-04	7.16E-04	4.11E-04	1.50E-04	5.93E-04	8.86E-04	5.57E-04

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	N Rx 1				N Rx 2			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
Mn-54	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	9.37E-07	4.88E-06	2.33E-05	6.11E-05	2.05E-06	9.08E-06	3.45E-05	7.48E-05
Sr-89/Y-89m	2.27E-02	7.91E-02	5.84E-02	1.29E-02	2.29E-02	6.79E-02	3.99E-02	7.31E-03
Sr-90	3.11E-04	1.63E-03	8.14E-03	2.25E-02	6.61E-04	2.95E-03	1.17E-02	2.68E-02
Y-90	3.11E-04	1.64E-03	8.15E-03	2.25E-02	6.63E-04	2.95E-03	1.17E-02	2.68E-02
Y-91	2.73E-02	1.01E-01	9.66E-02	3.02E-02	2.95E-02	9.26E-02	7.06E-02	1.83E-02
Zr-95	3.08E-02	1.17E-01	1.29E-01	4.89E-02	3.63E-02	1.18E-01	1.03E-01	3.23E-02
Nb-95	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99/Tc-99m	3.77E-03	1.02E-05	2.34E-20	3.49E-40	3.44E-03	7.94E-06	1.45E-20	1.78E-40
Ru-103	2.05E-02	6.34E-02	2.69E-02	2.88E-03	2.50E-02	6.58E-02	2.23E-02	1.97E-03
Ru-106/Rh-106	2.54E-03	1.26E-02	4.89E-02	9.68E-02	7.31E-03	3.09E-02	9.54E-02	1.56E-01
Cd-113m	3.35E-09	1.76E-08	8.66E-08	2.37E-07	4.60E-09	2.05E-08	8.06E-08	1.82E-07
Cd-115m	3.72E-06	1.23E-05	7.01E-06	1.11E-06	5.20E-06	1.46E-05	6.63E-06	8.66E-07
Sb-125	2.50E-05	1.31E-04	6.00E-04	1.48E-03	6.34E-05	2.81E-04	1.02E-03	2.08E-03
Te-129m/Te-129	6.93E-04	1.96E-03	5.50E-04	3.39E-05	8.29E-04	2.00E-03	4.46E-04	2.27E-05
Te-132	4.08E-03	3.63E-05	2.12E-17	4.78E-34	3.82E-03	2.89E-05	1.34E-17	2.50E-34
Cs-134	3.58E-05	1.83E-04	8.11E-04	1.91E-03	2.06E-04	8.99E-04	3.16E-03	6.18E-03
Cs-136/Ba-136m	9.71E-05	1.05E-04	3.32E-07	5.45E-11	1.84E-04	1.69E-04	4.25E-07	5.77E-11
Cs-137/Ba-137m	3.43E-04	1.80E-03	8.99E-03	2.49E-02	7.85E-04	3.51E-03	1.39E-02	3.19E-02
Ba-140	2.70E-02	2.78E-02	6.92E-05	8.30E-09	2.40E-02	2.11E-02	4.18E-05	4.15E-09
La-140	3.10E-02	3.20E-02	7.97E-05	9.56E-09	2.76E-02	2.43E-02	4.81E-05	4.77E-09
Ce-141	3.24E-02	9.00E-02	2.29E-02	1.24E-03	3.17E-02	7.49E-02	1.51E-02	6.78E-04
Ce-144/Pr-144	1.90E-02	9.30E-02	3.33E-01	5.94E-01	3.46E-02	1.44E-01	4.10E-01	6.05E-01
Pr-143	2.79E-02	3.17E-02	1.25E-04	2.76E-08	2.43E-02	2.35E-02	7.39E-05	1.35E-08
Pr-144	9.61E-03	4.70E-02	1.68E-01	3.00E-01	1.75E-02	7.27E-02	2.07E-01	3.06E-01
Nd-147	9.00E-03	7.12E-03	5.20E-06	1.23E-10	8.07E-03	5.44E-03	3.16E-06	6.20E-11
Pm-147	1.22E-03	6.71E-03	3.09E-02	7.56E-02	2.56E-03	1.16E-02	4.21E-02	8.52E-02
Pm-148m	6.00E-05	1.91E-04	9.15E-05	1.15E-05	1.39E-04	3.77E-04	1.44E-04	1.49E-05
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	1.71E-06	8.91E-06	4.35E-05	1.17E-04	9.99E-06	4.44E-05	1.72E-04	3.83E-04
Eu-155	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ta-182	1.70E-08	7.49E-08	1.62E-07	1.49E-07	6.36E-08	2.38E-07	4.09E-07	3.11E-07

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	TRIGA 1				TRIGA 2			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
Mn-54	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	1.69E-03	5.73E-03	1.46E-02	2.33E-02	1.35E-04	6.67E-04	3.09E-03	8.18E-03
Sr-89/Y-89m	2.71E-02	6.15E-02	2.42E-02	3.27E-03	2.55E-02	8.45E-02	6.03E-02	1.35E-02
Sr-90	5.26E-03	1.80E-02	4.79E-02	8.07E-02	3.48E-04	1.74E-03	8.37E-03	2.34E-02
Y-90	5.27E-03	1.80E-02	4.80E-02	8.07E-02	3.49E-04	1.74E-03	8.37E-03	2.34E-02
Y-91	3.43E-02	8.25E-02	4.22E-02	8.03E-03	3.03E-02	1.06E-01	9.82E-02	3.11E-02
Zr-95	4.15E-02	1.03E-01	6.07E-02	1.39E-02	3.44E-02	1.24E-01	1.33E-01	5.08E-02
Nb-95	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mo-99/Tc-99m	3.29E-03	5.81E-06	7.10E-21	6.45E-41	3.84E-03	9.89E-06	2.19E-20	3.31E-40
Ru-103	1.78E-02	3.59E-02	8.15E-03	5.29E-04	1.73E-02	5.10E-02	2.10E-02	2.27E-03
Ru-106/Rh-106	6.51E-03	2.11E-02	4.36E-02	5.27E-02	1.22E-03	5.75E-03	2.16E-02	4.32E-02
Cd-113m	2.75E-08	9.39E-08	2.47E-07	4.11E-07	1.93E-08	9.58E-08	4.58E-07	1.27E-06
Cd-115m	2.98E-06	6.41E-06	1.95E-06	1.88E-07	2.67E-06	8.36E-06	4.62E-06	7.39E-07
Sb-125	2.85E-04	9.59E-04	2.34E-03	3.50E-03	3.09E-05	1.53E-04	6.76E-04	1.68E-03
Te-129m/Te-129	5.74E-04	1.06E-03	1.58E-04	5.94E-06	5.80E-04	1.56E-03	4.23E-04	2.64E-05
Te-132	3.45E-03	2.00E-05	6.22E-18	8.54E-35	4.00E-03	3.38E-05	1.91E-17	4.36E-34
Cs-134	6.57E-04	2.19E-03	5.18E-03	7.44E-03	2.96E-06	1.44E-05	6.17E-05	1.47E-04
Cs-136/Ba-136m	1.44E-04	1.02E-04	1.71E-07	1.71E-11	3.10E-05	3.19E-05	9.74E-08	1.62E-11
Cs-137/Ba-137m	5.39E-03	1.85E-02	4.91E-02	8.27E-02	3.51E-04	1.75E-03	8.45E-03	2.37E-02
Ba-140	2.41E-02	1.62E-02	2.15E-05	1.57E-09	2.81E-02	2.75E-02	6.63E-05	8.05E-09
La-140	2.76E-02	1.86E-02	2.47E-05	1.81E-09	3.23E-02	3.17E-02	7.64E-05	9.28E-09
Ce-141	3.16E-02	5.72E-02	7.76E-03	2.56E-04	3.38E-02	8.90E-02	2.19E-02	1.20E-03
Ce-144/Pr-144	7.04E-02	2.24E-01	4.29E-01	4.66E-01	2.03E-02	9.42E-02	3.26E-01	5.89E-01
Pr-143	2.58E-02	1.91E-02	4.03E-05	5.40E-09	3.09E-02	3.34E-02	1.28E-04	2.84E-08
Pr-144	3.56E-02	1.13E-01	2.17E-01	2.35E-01	1.02E-02	4.76E-02	1.65E-01	2.98E-01
Nd-147	7.98E-03	4.12E-03	1.60E-06	2.32E-11	9.32E-03	7.01E-03	4.95E-06	1.19E-10
Pm-147	1.13E-02	3.83E-02	9.31E-02	1.39E-01	1.30E-03	6.79E-03	3.02E-02	7.49E-02
Pm-148m	2.90E-04	6.02E-04	1.54E-04	1.18E-05	1.75E-05	5.28E-05	2.45E-05	3.11E-06
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	3.74E-05	1.27E-04	3.31E-04	5.42E-04	1.29E-07	6.42E-07	3.03E-06	8.25E-06
Eu-155	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ta-182	9.25E-04	2.65E-03	3.06E-03	1.71E-03	7.48E-04	3.12E-03	6.53E-03	6.07E-03

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Table G-5. Contributions to gross gamma activity for an air or swipe sample.

Nuclide	ATR 1				ATR 2				ATR 3			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
Mn-54	1.91E-07	8.69E-07	3.04E-06	5.59E-06	6.77E-08	4.70E-07	2.41E-06	5.15E-06	9.30E-08	5.56E-07	2.49E-06	5.12E-06
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	2.76E-07	1.00E-06	1.21E-06	5.52E-07	1.05E-07	5.81E-07	1.04E-06	5.47E-07	2.75E-07	1.31E-06	2.04E-06	1.04E-06
Co-60	1.64E-07	7.89E-07	3.57E-06	9.28E-06	5.36E-08	3.94E-07	2.62E-06	7.90E-06	7.30E-08	4.62E-07	2.68E-06	7.79E-06
Sr-89/Y-89m	2.51E-06	8.06E-06	5.64E-06	1.24E-06	1.38E-06	6.79E-06	6.98E-06	1.78E-06	1.75E-06	7.43E-06	6.66E-06	1.63E-06
Sr-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	9.66E-05	3.29E-04	2.99E-04	9.25E-05	5.07E-05	2.64E-04	3.52E-04	1.26E-04	6.46E-05	2.90E-04	3.37E-04	1.17E-04
Zr-95	3.46E-02	1.21E-01	1.27E-01	4.76E-02	1.74E-02	9.31E-02	1.43E-01	6.22E-02	2.26E-02	1.04E-01	1.40E-01	5.85E-02
Nb-95	3.00E-02	1.44E-01	2.44E-01	1.02E-01	8.12E-03	8.19E-02	2.62E-01	1.33E-01	1.31E-02	1.01E-01	2.59E-01	1.25E-01
Mo-99/Tc-99m	4.06E-03	1.02E-05	2.20E-20	3.26E-40	4.81E-03	1.84E-05	5.87E-20	1.01E-39	4.49E-03	1.48E-05	4.12E-20	6.80E-40
Ru-103	1.68E-02	4.79E-02	1.93E-02	2.04E-03	1.03E-02	4.50E-02	2.66E-02	3.26E-03	1.27E-02	4.78E-02	2.46E-02	2.92E-03
Ru-106/Rh-106	2.38E-04	1.09E-03	4.01E-03	7.88E-03	8.33E-05	5.85E-04	3.16E-03	7.19E-03	1.15E-04	6.99E-04	3.29E-03	7.22E-03
Cd-113m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cd-115m	6.19E-08	1.88E-07	1.02E-07	1.60E-08	3.56E-08	1.66E-07	1.32E-07	2.39E-08	4.47E-08	1.80E-07	1.24E-07	2.18E-08
Sb-125	1.98E-05	9.56E-05	4.14E-04	1.01E-03	6.37E-06	4.85E-05	3.10E-04	8.77E-04	9.17E-06	5.93E-05	3.30E-04	9.00E-04
Te-129m/Te-129	6.81E-05	1.78E-04	4.72E-05	2.88E-06	4.50E-05	1.80E-04	7.02E-05	4.97E-06	5.42E-05	1.87E-04	6.35E-05	4.34E-06
Te-132	3.63E-03	2.98E-05	1.65E-17	3.68E-34	4.30E-03	5.40E-05	4.38E-17	1.14E-33	4.01E-03	4.34E-05	3.07E-17	7.68E-34
Cs-134	6.68E-04	3.15E-03	1.32E-02	3.09E-02	1.56E-04	1.13E-03	6.94E-03	1.88E-02	1.04E-04	6.49E-04	3.49E-03	9.12E-03
Cs-136/Ba-136m	2.53E-04	2.53E-04	7.55E-07	1.23E-10	1.15E-04	1.77E-04	7.76E-07	1.46E-10	1.35E-04	1.78E-04	6.84E-07	1.24E-10
Cs-137/Ba-137m	3.64E-04	1.76E-03	8.34E-03	2.29E-02	1.18E-04	8.72E-04	6.05E-03	1.93E-02	1.68E-04	1.07E-03	6.50E-03	1.99E-02
Ba-140	1.13E-02	1.08E-02	2.54E-05	3.03E-09	1.15E-02	1.67E-02	5.80E-05	7.99E-09	1.18E-02	1.48E-02	4.48E-05	5.96E-09
La-140	6.95E-02	6.62E-02	1.56E-04	1.86E-08	7.03E-02	1.03E-01	3.56E-04	4.91E-08	7.25E-02	9.11E-02	2.75E-04	3.66E-08
Ce-141	1.66E-02	4.24E-02	1.02E-02	5.48E-04	1.11E-02	4.36E-02	1.54E-02	9.59E-04	1.35E-02	4.54E-02	1.40E-02	8.41E-04
Ce-144/Pr-144	1.55E-03	6.97E-03	2.36E-02	4.18E-02	5.57E-04	3.85E-03	1.92E-02	3.93E-02	7.80E-04	4.64E-03	2.01E-02	3.98E-02
Pr-143	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	2.97E-04	1.34E-03	4.55E-03	8.05E-03	1.07E-04	7.40E-04	3.69E-03	7.57E-03	1.50E-04	8.92E-04	3.88E-03	7.66E-03
Nd-147	1.75E-03	1.28E-03	8.82E-07	2.07E-11	1.68E-03	1.88E-03	1.91E-06	5.20E-11	1.89E-03	1.82E-03	1.62E-06	4.25E-11
Pm-147	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-148m	5.74E-04	1.68E-03	7.66E-04	9.51E-05	1.33E-04	5.96E-04	3.98E-04	5.74E-05	2.81E-04	1.08E-03	6.32E-04	8.78E-05
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	1.54E-05	7.44E-05	3.44E-04	9.16E-04	4.27E-06	3.15E-05	2.14E-04	6.61E-04	2.53E-06	1.61E-05	9.51E-05	2.83E-04
Eu-155	1.13E-06	5.41E-06	2.44E-05	6.27E-05	3.26E-07	2.40E-06	1.59E-05	4.73E-05	5.13E-07	3.24E-06	1.87E-05	5.39E-05
Ta-182	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	FFTF 1				FFTF 2			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
Mn-54	1.17E-03	4.77E-03	1.05E-02	1.22E-02	8.15E-04	3.61E-03	9.23E-03	1.18E-02
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	9.69E-03	3.14E-02	2.40E-02	6.89E-03	8.36E-03	2.96E-02	2.62E-02	8.27E-03
Co-60	9.39E-05	4.04E-04	1.15E-03	1.89E-03	4.56E-05	2.14E-04	7.09E-04	1.28E-03
Sr-89/Y-89m	9.13E-07	2.63E-06	1.16E-06	1.61E-07	9.42E-07	2.96E-06	1.51E-06	2.30E-07
Sr-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	4.39E-05	1.34E-04	7.65E-05	1.50E-05	4.51E-05	1.50E-04	9.94E-05	2.14E-05
Zr-95	2.63E-02	8.27E-02	5.46E-02	1.29E-02	2.68E-02	9.17E-02	7.03E-02	1.83E-02
Nb-95	2.91E-02	1.13E-01	1.07E-01	2.80E-02	2.94E-02	1.25E-01	1.38E-01	3.95E-02
Mo-99/Tc-99m	3.55E-03	7.97E-06	1.09E-20	1.02E-40	3.50E-03	8.55E-06	1.36E-20	1.39E-40
Ru-103	3.40E-02	8.70E-02	2.21E-02	1.48E-03	3.39E-02	9.45E-02	2.78E-02	2.05E-03
Ru-106/Rh-106	7.31E-03	3.01E-02	6.96E-02	8.68E-02	5.38E-03	2.41E-02	6.49E-02	8.87E-02
Cd-113m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cd-115m	5.85E-07	1.59E-06	5.42E-07	5.36E-08	5.62E-07	1.67E-06	6.60E-07	7.16E-08
Sb-125	3.99E-04	1.71E-03	4.67E-03	7.22E-03	2.49E-04	1.17E-03	3.71E-03	6.30E-03
Te-129m/Te-129	6.74E-07	1.58E-06	2.64E-07	1.02E-08	4.13E-07	1.05E-06	2.04E-07	8.67E-09
Te-132	3.33E-03	2.20E-05	4.63E-18	3.38E-35	3.43E-03	2.47E-05	6.04E-18	4.84E-35
Cs-134	5.99E-03	2.53E-02	6.68E-02	9.91E-02	1.73E-03	7.95E-03	2.44E-02	3.97E-02
Cs-136/Ba-136m	9.14E-03	8.18E-03	1.54E-05	1.59E-09	4.26E-03	4.15E-03	9.09E-06	1.03E-09
Cs-137/Ba-137m	2.11E-03	9.13E-03	2.72E-02	4.72E-02	1.14E-03	5.40E-03	1.87E-02	3.56E-02
Ba-140	7.81E-03	6.64E-03	9.88E-06	7.44E-10	8.02E-03	7.44E-03	1.28E-05	1.06E-09
La-140	4.78E-02	4.08E-02	6.07E-05	4.57E-09	4.91E-02	4.57E-02	7.89E-05	6.52E-09
Ce-141	1.25E-02	2.87E-02	4.35E-03	1.48E-04	1.28E-02	3.20E-02	5.64E-03	2.11E-04
Ce-144/Pr-144	2.78E-03	1.12E-02	2.40E-02	2.69E-02	2.19E-03	9.66E-03	2.40E-02	2.95E-02
Pr-143	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	5.35E-04	2.16E-03	4.62E-03	5.17E-03	4.22E-04	1.86E-03	4.61E-03	5.67E-03
Nd-147	1.42E-03	9.32E-04	4.06E-07	6.04E-12	1.41E-03	1.00E-03	5.08E-07	8.30E-12
Pm-147	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-148m	5.49E-03	1.44E-02	4.12E-03	3.25E-04	2.81E-03	8.04E-03	2.67E-03	2.31E-04
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	2.77E-04	1.20E-03	3.48E-03	5.87E-03	9.84E-05	4.63E-04	1.56E-03	2.90E-03
Eu-155	9.45E-05	4.06E-04	1.15E-03	1.88E-03	4.79E-05	2.24E-04	7.39E-04	1.32E-03
Ta-182	2.25E-04	8.15E-04	1.05E-03	6.00E-04	2.19E-04	8.67E-04	1.29E-03	8.13E-04

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	N Rx 1				N Rx 2			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
Mn-54	1.72E-07	8.46E-07	3.12E-06	5.79E-06	3.40E-07	1.42E-06	4.18E-06	6.42E-06
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	1.69E-06	6.64E-06	8.49E-06	3.90E-06	1.72E-06	5.75E-06	5.86E-06	2.22E-06
Co-60	1.87E-06	9.75E-06	4.66E-05	1.22E-04	4.10E-06	1.82E-05	6.91E-05	1.50E-04
Sr-89/Y-89m	2.11E-06	7.36E-06	5.43E-06	1.20E-06	2.13E-06	6.32E-06	3.71E-06	6.80E-07
Sr-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	8.20E-05	3.03E-04	2.90E-04	9.06E-05	8.86E-05	2.78E-04	2.12E-04	5.48E-05
Zr-95	3.07E-02	1.17E-01	1.29E-01	4.87E-02	3.62E-02	1.17E-01	1.03E-01	3.21E-02
Nb-95	2.52E-02	1.34E-01	2.46E-01	1.05E-01	3.86E-02	1.56E-01	2.02E-01	6.94E-02
Mo-99/Tc-99m	4.05E-03	1.10E-05	2.51E-20	3.75E-40	3.69E-03	8.53E-06	1.55E-20	1.92E-40
Ru-103	1.91E-02	5.90E-02	2.51E-02	2.68E-03	2.32E-02	6.12E-02	2.07E-02	1.83E-03
Ru-106/Rh-106	4.34E-04	2.16E-03	8.36E-03	1.66E-02	1.25E-03	5.29E-03	1.63E-02	2.67E-02
Cd-113m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cd-115m	8.25E-08	2.72E-07	1.56E-07	2.46E-08	1.15E-07	3.24E-07	1.47E-07	1.92E-08
Sb-125	2.14E-05	1.12E-04	5.14E-04	1.27E-03	5.43E-05	2.41E-04	8.75E-04	1.78E-03
Te-129m/Te-129	7.85E-05	2.23E-04	6.23E-05	3.84E-06	9.40E-05	2.27E-04	5.05E-05	2.58E-06
Te-132	3.75E-03	3.34E-05	1.95E-17	4.40E-34	3.51E-03	2.66E-05	1.23E-17	2.30E-34
Cs-134	7.98E-05	4.09E-04	1.81E-03	4.27E-03	4.60E-04	2.00E-03	7.05E-03	1.38E-02
Cs-136/Ba-136m	2.80E-04	3.03E-04	9.56E-07	1.57E-10	5.29E-04	4.88E-04	1.22E-06	1.66E-10
Cs-137/Ba-137m	3.09E-04	1.62E-03	8.09E-03	2.24E-02	7.06E-04	3.16E-03	1.25E-02	2.87E-02
Ba-140	1.09E-02	1.12E-02	2.79E-05	3.35E-09	9.71E-03	8.51E-03	1.69E-05	1.67E-09
La-140	6.67E-02	6.89E-02	1.72E-04	2.06E-08	5.95E-02	5.23E-02	1.04E-04	1.03E-08
Ce-141	1.56E-02	4.32E-02	1.10E-02	5.94E-04	1.52E-02	3.59E-02	7.26E-03	3.25E-04
Ce-144/Pr-144	1.28E-03	6.27E-03	2.25E-02	4.01E-02	2.33E-03	9.71E-03	2.76E-02	4.08E-02
Pr-143	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	2.47E-04	1.21E-03	4.32E-03	7.71E-03	4.49E-04	1.87E-03	5.32E-03	7.85E-03
Nd-147	1.80E-03	1.42E-03	1.04E-06	2.46E-11	1.61E-03	1.09E-03	6.31E-07	1.24E-11
Pm-147	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-148m	2.02E-04	6.43E-04	3.08E-04	3.87E-05	4.70E-04	1.27E-03	4.85E-04	5.03E-05
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	2.78E-06	1.46E-05	7.10E-05	1.91E-04	1.63E-05	7.25E-05	2.81E-04	6.26E-04
Eu-155	2.12E-06	1.10E-05	5.23E-05	1.36E-04	4.03E-06	1.78E-05	6.74E-05	1.45E-04
Ta-182	2.49E-08	1.09E-07	2.37E-07	2.17E-07	9.29E-08	3.47E-07	5.98E-07	4.54E-07

ATTACHMENT G
RADIONUCLIDE CONTRIBUTIONS TO SAMPLE ACTIVITY (continued)

Nuclide	TRIGA 1				TRIGA 2			
	10 d	40 d	180 d	1 y	10 d	40 d	180 d	1 y
Mn-54	8.69E-05	2.79E-04	5.49E-04	6.20E-04	2.17E-05	1.01E-04	3.62E-04	6.81E-04
Fe-55	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	2.11E-04	5.39E-04	3.69E-04	1.03E-04	1.60E-04	5.95E-04	7.38E-04	3.43E-04
Co-60	3.38E-03	1.15E-02	2.93E-02	4.67E-02	2.70E-04	1.33E-03	6.17E-03	1.64E-02
Sr-89/Y-89m	2.52E-06	5.72E-06	2.25E-06	3.04E-07	2.37E-06	7.86E-06	5.61E-06	1.26E-06
Sr-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-90	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-91	1.03E-04	2.47E-04	1.27E-04	2.41E-05	9.08E-05	3.18E-04	2.95E-04	9.33E-05
Zr-95	4.13E-02	1.02E-01	6.04E-02	1.39E-02	3.43E-02	1.24E-01	1.32E-01	5.06E-02
Nb-95	4.57E-02	1.40E-01	1.19E-01	3.00E-02	2.80E-02	1.42E-01	2.52E-01	1.09E-01
Mo-99/Tc-99m	3.53E-03	6.24E-06	7.63E-21	6.93E-41	4.12E-03	1.06E-05	2.35E-20	3.55E-40
Ru-103	1.66E-02	3.34E-02	7.58E-03	4.93E-04	1.61E-02	4.74E-02	1.95E-02	2.11E-03
Ru-106/Rh-106	1.11E-03	3.61E-03	7.46E-03	9.00E-03	2.08E-04	9.83E-04	3.69E-03	7.39E-03
Cd-113m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cd-115m	6.61E-08	1.42E-07	4.34E-08	4.17E-09	5.92E-08	1.86E-07	1.03E-07	1.64E-08
Sb-125	2.44E-04	8.21E-04	2.00E-03	3.00E-03	2.64E-05	1.31E-04	5.79E-04	1.44E-03
Te-129m/Te-129	6.50E-05	1.20E-04	1.80E-05	6.74E-07	6.58E-05	1.77E-04	4.80E-05	2.99E-06
Te-132	3.17E-03	1.84E-05	5.72E-18	7.86E-35	3.68E-03	3.11E-05	1.76E-17	4.01E-34
Cs-134	1.46E-03	4.89E-03	1.15E-02	1.66E-02	6.60E-06	3.21E-05	1.37E-04	3.28E-04
Cs-136/Ba-136m	4.14E-04	2.92E-04	4.93E-07	4.92E-11	8.93E-05	9.19E-05	2.81E-07	4.66E-11
Cs-137/Ba-137m	4.85E-03	1.66E-02	4.42E-02	7.45E-02	3.16E-04	1.58E-03	7.61E-03	2.13E-02
Ba-140	9.72E-03	6.52E-03	8.67E-06	6.33E-10	1.14E-02	1.11E-02	2.68E-05	3.25E-09
La-140	5.95E-02	4.01E-02	5.33E-05	3.89E-09	6.96E-02	6.83E-02	1.65E-04	2.00E-08
Ce-141	1.52E-02	2.75E-02	3.72E-03	1.23E-04	1.62E-02	4.27E-02	1.05E-02	5.76E-04
Ce-144/Pr-144	4.75E-03	1.51E-02	2.89E-02	3.14E-02	1.37E-03	6.35E-03	2.20E-02	3.98E-02
Pr-143	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pr-144	9.14E-04	2.91E-03	5.57E-03	6.05E-03	2.63E-04	1.22E-03	4.24E-03	7.65E-03
Nd-147	1.59E-03	8.22E-04	3.20E-07	4.63E-12	1.86E-03	1.40E-03	9.89E-07	2.37E-11
Pm-147	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pm-148m	9.78E-04	2.03E-03	5.19E-04	3.96E-05	5.89E-05	1.78E-04	8.26E-05	1.05E-05
Sm-151	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Eu-154	6.10E-05	2.08E-04	5.41E-04	8.85E-04	2.11E-07	1.05E-06	4.95E-06	1.35E-05
Eu-155	1.57E-05	5.32E-05	1.35E-04	2.13E-04	2.30E-06	1.13E-05	5.22E-05	1.37E-04
Ta-182	1.35E-03	3.87E-03	4.47E-03	2.50E-03	1.09E-03	4.56E-03	9.54E-03	8.86E-03

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. Considerations about parameter selection are included.

In the examples below, one reactor case is chosen to work through the steps and mathematics. However, in dose reconstructions intakes and doses to the organ of concern have to be calculated for all the reactor cases to determine the one that maximizes the organ. This is necessary to justify use of the constant (upper bound) distribution. However, the user can choose to determine the intakes for one reactor case only and use the OTIB-0054 tool calculate the intakes (and organ doses) for the other reactor cases without having to use IMBA more than once.

Example 1. Gross beta bioassay data (with major chemical processing) from waste management worker.

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

Table H-1. Gross beta urinalysis results and ATR-1 ^{90}Sr results (pCi/24 hr).

Date	Gross beta[a]	Sr-90[b]
07/23/1964	56	26.4
10/20/1964	42	19.8
08/13/1965	33	15.5
10/21/1965	41	19.3
02/16/1966	57	26.8
05/03/1966	31	14.6
08/11/1966	17	8.0
11/02/1966	31	14.6

a. Detection threshold of 20 pCi/24 hr.
b. Calculated by multiplying 1-year Sr-90 fraction, 0.471, from Table 7-2 times the bioassay results.

Because the worker was involved in waste management activities, Table 5-3 indicates the use of a 1-year fuel decay assumption for selection of intake parameters is appropriate. For gross beta urinalyses, ^{90}Sr is the indicator radionuclide. 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, 1-year decay period, are used. There are nine options for the representative reactor cases and one must calculate nine separate intakes of ^{90}Sr using the Table 7-2 fractions; however, calculating the intake once is sufficient if a tool automating the process is used to calculate the other eight cases.

Using the ATR-1 reactor case in Table 7-2, the activity of ^{90}Sr in each gross beta urinalysis result is 0.471 times the gross beta result. These values are listed in Table H-1.

The ^{90}Sr urinalysis estimates from Table H-1 were used to fit a chronic intake of ^{90}Sr type F for the period from January 28, 1960, through January 25, 1971, and resulted in a calculated ^{90}Sr intake rate of 76.0 pCi/d. Because all biological model parameters are the same for the other eight reactor cases, the ^{90}Sr intake rates for the other eight reactor cases are the product of the ATR-1 intake rate and the ratio of the fraction of beta activity in urine for each reactor (from Table 7-2) to the ATR-1 fraction. For example the ATR-2 ^{90}Sr intake rate is

ATTACHMENT H
EXAMPLE INTAKES ESTIMATED FROM BIOASSAY AND FROM AIR SAMPLE DATA (continued)

ATR-2 ^{90}Sr intake rate = (76.0 pCi/d)[(0.412)/(0.471)] = 66.5 pCi/d.

The intake rates from the other radionuclides in the ATR-1 source term were determined by multiplying the ATR-1 1-year ^{90}Sr ratios in Table 7-3a by the ^{90}Sr intake rate; the results are listed in Table H-2. Similarly, the intake rates of the other radionuclides in the other reactor source terms are determined using their corresponding ratios in Table 7-3.

Table H-2. ATR-1 radionuclide intake rates derived from ^{90}Sr intake rate of 76.0 pCi/d.

Radionuclide	1-yr ratios[a]	pCi/d
Co-60	1.85E-4	1.41E-02
Sr-89	5.29E-1	4.02E+01
Sr-90	1.00	7.60E+01
Y-90	1.00	7.60E+01
Y-91	1.23	9.35E+01
Zr-95	1.90	1.44E+02
Nb-95	4.08	3.10E+02
Ru-103	8.73E-2	6.63E+00
Ru-106	9.17E-1	6.97E+01
I-131	6.33E-11	4.81E-09
Cs-134	5.52E-1	4.20E+01
Cs-137	1.01	7.68E+01
Ce-141	4.54E-2	3.45E+00
Ce-144	1.26E+1	9.58E+02
Pr-143	1.04E-6	7.90E-05
Pm-147	2.42	1.84E+02
Eu-154	2.23E-2	1.69E+00
Total daily intake	—	2.08E+03

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

To determine the maximum dose to the organ of concern, the radionuclide intake rates for each of the nine representative reactor cases are used. Annual doses to the organ of concern are calculated assuming the absorption type for each radionuclide that maximizes the dose to the organ, with the exception of ^{89}Sr and ^{90}Sr, which are assumed to be type F, and ^{131}I, which is assumed to be I_2 vapor. Assigning ^{131}I intake might not be appropriate for all cases (see Section 7.1.3)

Example 2. Gross beta bioassay data (with minor chemical processing) from worker involved in spent fuel storage operations.

Example 2 uses the bioassay data in Table H-1 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 5-3. For this example, it is assumed that the urinalysis did not include a chemical separation step, so the fractions of ^{90}Sr in the urine sample come from Table 7-1a; the resulting ^{90}Sr urine activity using the ATR-1 reactor case is listed in Table H-3.

ATTACHMENT H
EXAMPLE INTAKES ESTIMATED FROM BIOASSAY AND FROM AIR SAMPLE DATA (continued)

Table H-3. Gross beta urinalysis results and ATR-1 ^{90}Sr results (pCi/24 hr).

Date	Urinalysis[a]	Sr-90[b]
07/23/1964	56	1.2
10/20/1964	42	0.9
08/13/1965	33	0.7
10/21/1965	41	0.9
02/16/1966	57	1.2
05/03/1966	31	0.7
08/11/1966	17	0.4
11/02/1966	31	0.7

a. Detection threshold of 20 pCi/24 hr.
b. Calculated by multiplying 10-day ATR-1 Sr-90 fraction, 2.14E-02, from Table 7-1a times the bioassay results.

The ^{90}Sr data are used to fit a chronic intake of ^{90}Sr type F for the period from January 28, 1960, through January 25, 1971, which resulted in a calculated ^{90}Sr intake rate of 3.51 pCi/d. As discussed for Example 1, the ^{90}Sr intake rates for the other eight representative reactor cases are given by the product of the ATR-1 intake rate and the ratio of the fraction of beta activity in urine for each reactor (from Table 7-1a) to the ATR-1 fraction. For example, the ATR-2 ^{90}Sr intake rate is

ATR-2 ^{90}Sr intake rate = (3.51 pCi/d)[(0.0124)/(0.0214)] = 2.03 pCi/d.

The intake rates from the other radionuclides in the ATR-1 source term are determined by multiplying the ATR-1 10-day ^{90}Sr ratios in Table 7-3a by the ^{90}Sr intake rate; the results are listed in Table H-4. Similarly, the intake rates of the other radionuclides in the other reactor source terms are determined using their corresponding ratios in Table 7-3.

Table H-4. ATR-1 radionuclide intake rates derived from ^{90}Sr intake rate of 3.51 pCi/d.

Radionuclide	10-d ratios[a]	pCi/d
Co-60	2.05E-04	7.19E-04
Sr-89	6.72E+01	2.36E+02
Sr-90	1.00E+00	3.51E+00
Y-90	1.00E+00	3.52E+00
Y-91	8.04E+01	2.82E+02
Zr-95	8.67E+01	3.04E+02
Nb-95	7.49E+01	2.63E+02
Ru-103	4.50E+01	1.58E+02
Ru-106	1.74E+00	6.10E+00
I-131	1.21E+03	4.25E+03
Cs-134	7.48E-01	2.62E+00
Cs-137	1.01E+00	3.54E+00
Ce-141	8.62E+01	3.02E+02
Ce-144	2.89E+01	1.01E+02
Pr-143	7.60E+01	2.67E+02
Pm-147	2.81E+00	9.86E+00
Eu-154	2.36E-02	8.28E-02
Total daily intake	—	6.19E+03

a. From Table 7-3a, ATR-1, Sr-90, 10-d decay.

ATTACHMENT H
EXAMPLE INTAKES ESTIMATED FROM BIOASSAY AND FROM AIR SAMPLE DATA (continued)

Example 3. Gross gamma 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 if 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 5-3. This example assumes monthly average gamma air concentrations in the dissolution area ranged from 1×10^{-14} to 1×10^{-10} µCi/mL during the period from 1960 through 1971. It further assumes that the air concentrations were not tabulated, so no overall average value is available. The following calculation is used to bound radionuclide exposures using the upper range of the monthly average air concentrations, 1×10^{-10} µCi/mL.

Each 180-day ^{137}Cs fraction from Table 7-4b for the nine representative reactors is used to estimate the ^{137}Cs bounding air concentrations by multiplying the fraction by 1×10^{-10} µCi/mL. The daily intake rates of ^{137}Cs are given by

^{137}Cs intake, pCi/d = (Table 7-4b gamma fraction)(air conc. in µCi/mL)(10^6 pCi/µCi)(10^6 mL/m^3)(2400 m^3)/365 d.

Or for a gamma-based air concentration of 1 x10^{-10} µCi/mL

$$^{137}\text{Cs intake, pCi/d} = (6.58 \times 10^2)(\text{Table 7-4b gamma fraction}).$$

These ^{137}Cs intake rates are used with the 180-day intake fractions in Table 7-3a to estimate air concentrations and inhalation intakes of associated radionuclides. The results for ATR-1 are listed in Table H-5. An ingestion intake must also be estimated (NIOSH 2004). Table H-5 summarizes the results of the air concentrations and intakes for ATR-1.

Table H-5. ATR-1 radionuclide air concentrations and intake rates derived from a gamma air concentration of 1×10^{-10} µCi/mL.

Radionuclide	180 d[a]	Air concentration (µCi/mL)	Inhalation intake (pCi/d)	Ingestion intake (pCi/d)[b]
Co-60	1.93E-04	3.53E-16	2.32E-03	4.84E-05
Sr-89	6.54	1.20E-11	7.87E+01	1.64
Sr-90	9.89E-01	1.81E-12	1.19E+01	2.48E-01
Y-90	9.90E-01	1.81E-12	1.19E+01	2.48E-01
Y-91	1.07E+01	1.96E-11	1.29E+02	2.68
Zr-95	1.38E+01	2.53E-11	1.66E+02	3.46
Nb-95	2.63E+01	4.81E-11	3.16E+02	6.59
Ru-103	2.24	4.10E-12	2.70E+01	5.62E-01
Ru-106	1.27	2.32E-12	1.53E+01	3.18E-01
I-131	5.23E-04	9.57E-16	6.29E-03	1.31E-04
Cs-134	6.40E-01	1.17E-12	7.70E+00	1.60E-01
Cs-137	1.00	1.83E-12	1.20E+01	2.51E-01
Ce-141	2.30	4.21E-12	2.77E+01	5.77E-01
Ce-144	1.91E+01	3.50E-11	2.30E+02	4.79
Pr-143	1.29E-02	2.36E-14	1.55E-01	3.23E-03
Pm-147	2.71	4.96E-12	3.26E+01	6.79E-01
Eu-154	2.27E-02	4.15E-14	2.73E-01	5.69E-03

a. From Table 7-3a ATR-1 Cs-137 ratios.
b. From guidance in NIOSH 2004, Rev. 0.