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Evaluation of EBR-II and BORAX-IV for ORAUT-OTIB-0054 Applicability

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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
05/08/2020	00	New report generated to provide the results of evaluations of the Experimental Breeder Reactor No. 2 and the Boiling Water Reactor Experiment No. 4 to determine if ORAUT-OTIB-0054 Revision 04 provides bounding internal dose guidance when only gross beta or gross gamma assay measurements are available. Incorporates formal internal and NIOSH review comments. Training required: As determined by the Objective Manager. Initiated by William M. Findley.

TABLE OF CONTENTS

LIST OF TABLES

TABLE **TITLE** **PAGE**

ACRONYMS AND ABBREVIATIONS

ABRWH	Advisory Board on Radiation and Worker Health
ANL-W	Argonne National Laboratory-West
at%	atom percent (with reference to burnup of all heavy atoms in irradiated fuel)
ATR	Advanced Test Reactor
BORAX-IV	Boiling Water Reactor Experiment No. 4
Ci	curie
cm	centimeter
CPP	Chemical Processing Plant
d	day
DCF	dose conversion factor
DOE	U.S. Department of Energy
EBR-II	Experimental Breeder Reactor No. 2
FFTF	Fast Flux Test Facility
g	gram
HEU	highly enriched uranium
INL	Idaho National Laboratory
IRF	intake retention fraction
LOFT	Loss of Fluid Test facility
MFAP	mixed fission and activation product
MTHM	metric tons of heavy metal
MW(t)	megawatt-thermal
NIF	normalized intake fraction
NIOSH	National Institute for Occupational Safety and Health
NRTS	National Reactor Testing Station
ORAU	Oak Ridge Associated Universities
pCi	picocurie
SC&A	S. Cohen & Associates
SEC	Special Exposure Cohort
SPERT	Special Power Excursion Reactor Tests
SRDB Ref ID	Site Research Database Reference Identification (number)
TLD	thermoluminescent dosimeter
TRIGA	Training, Research, Isotopes, General Atomics
wt%	weight percent
yr	year

1.0 INTRODUCTION

After the release of Revision 0 of the Special Exposure Cohort (SEC) petition evaluation report for the Idaho National Laboratory (INL) [Petition SEC-00219; NIOSH 2015], the Advisory Board on Radiation and Worker Health (ABRWH) requested that S. Cohen & Associates (SC&A) review the report in relation to (1) petitioner class definition and (2) dose reconstructability and gap analysis [SC&A/Saliant 2015a]. While SC&A's dose reconstructability and gap analysis review included the proposed SEC period, it focused on the assumption that doses can be reconstructed with sufficient accuracy for the site locations and time periods that lie outside the SEC class definition. The same requested review was made after the release of the SEC petition evaluation report for Argonne National Laboratory-West (ANL-W) [Petition SEC-00224; NIOSH 2016a]. ANL-W is on the INL site but did not combine with INL until 2005. As a result, there were two separate petitions and evaluations.

For INL, the class evaluated by the National Institute for Occupational Safety and Health (NIOSH) included all employees who worked in any area at INL from January 1, 1949, through December 31, 1970. The NIOSH-proposed class to be added to the SEC in Revision 2 of the Petition SEC-00219 evaluation report is [NIOSH 2017, p. 1]:

> *All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Idaho National Laboratory (INL) in Scoville, Idaho, and (a) who were monitored for external radiation at the Idaho Chemical Processing Plant (CPP) (e.g., at least one film badge or TLD* [thermoluminescent dosimeter] *dosimeter from CPP) between January 1, 1963 and February 28, 1970; or (b) who were monitored for external radiation at INL (e.g., at least one film badge or TLD dosimeter) between March 1, 1970 and December 31, 1974 for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort.*

The decision to extend the proposed SEC class past the end of the evaluation period was due to the identified dose infeasibility known to have existed through at least December 31, 1974, per INL documents and personnel interviews.

For ANL-W, the class evaluated by NIOSH included all employees who worked in any area at ANL-W from April 10, 1951, through December 31, 1979. The NIOSH-proposed class to be added to the SEC in Revision 0 of the Petition SEC-00224 evaluation report is [NIOSH 2016a, p. 1]:

> *All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Argonne National Laboratory-West between April 10, 1951 and December 31, 1957 for a number of work days aggregating at least 250 work days, occurring either solely under this employment, or in combination with work days within the parameters established for one or more other classes of employees in the Special Exposure Cohort.*

Operations at the site began in 1949, when the U.S. Atomic Energy Commission established the National Reactor Testing Station (NRTS). As the name implies, it was the primary nuclear reactor development facility in the United States. The missions of INL and ANL-W were to perform scientific experiments involving nuclear reactors using primarily uranium fuel for testing and demonstrating advanced nuclear reactor components and designs. Fifty-two reactors were constructed (only 50 operated) at INL and ANL-W with many unique experiments, handling and processing of irradiated nuclear fuel, and disposal of radioactive waste [Stacy 2000]. INL was known as NRTS from 1949 to 1973, Idaho National Engineering Laboratory from 1974 to 1996, and Idaho National Engineering and Environmental Laboratory from 1997 to 2004. ANL-W also began operations in 1949. However, it was

operated separately by the University of Chicago under the U.S. Department of Energy (DOE) and its predecessor the U.S. Atomic Energy Commission's Chicago Operations Office. In February 2005, the ANL-W facilities merged with INL under the DOE Idaho Operations Office. The ANL-W facilities were collectively renamed the Materials and Fuels Complex after the merger.

Because the majority of the bioassay measurements performed for INL and ANL-W workers did not determine the potential mixtures of the activation and fission products (MFAPs) that the workers were exposed to and because those potential mixtures have a significant impact on the workers' doses, a method for estimating the mixtures of the activation and fission products needed to be selected for the INL and ANL-W Sites. As indicated in Section 5.5 of the internal dosimetry technical basis document for the INL and ANL-W sites [ORAUT 2010], the method described in ORAUT-OTIB-0054, *Technical Information Bulletin: Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* [ORAUT 2015] was determined to be appropriate and applicable for INL and ANL-W because some INL reactor data were used to develop that method. That technical information bulletin is referred to as OTIB-0054 throughout the remainder of this report.

OTIB-0054 describes the basis for assigning radionuclide-specific intakes of MFAPs when air sampling or urinalysis data are available only as total beta or total gamma activity. Such estimates are based on the MFAP intensities relative to an indicator radionuclide in the gross beta or gross gamma urinalysis results. The analytical model extrapolates the estimated dose contribution from these indicator radionuclides (^{90}Sr in gross beta analyses, and ^{137}Cs in gross gamma) to the dose contribution by other dosimetrically important MFAPs in the source term.

Given the experimental nature of many of the INL and ANL-W reactors, SC&A/Saliant [2016a, 2016b] categorized the reactors as high, medium, and low based on the potential to underestimate radiation exposures using the OTIB-0054 approach due to reactor design characteristics (i.e., fuel type, enrichment, cladding, etc.). NIOSH reviewed the proposed categorization of the reactors for additional evaluation and agreed with most of what was recommended. NIOSH did propose that the INL and ANL-W high-priority category reactors be merged and that any concerns about the medium- and low-priority category reactors be addressed after the completion of the evaluations of the high-priority category reactors. NIOSH and SC&A/Saliant agreed to the NIOSH proposals but recommended that the Special Power Excursion Reactor Tests (SPERT) "bounding case" be justified in its evaluation and the Loss of Fluid Test (LOFT) evaluation be conducted as a site profile exercise because it did not begin radiological operations until after the SEC evaluation period [NIOSH 2016b; SC&A/Saliant 2016c].

The assignment of reactors to the priority categories considered reactor design factors such as the type of fuel, enrichment, cladding, moderator, coolant, operational mode, length of operation, and if the reactor performed within design limits or was deliberately or inadvertently taken outside those limits. In addition, SC&A considered "four factors that reflect the scope of the population potentially 'at risk' of uncontrolled/unmonitored exposures" [SC&A/Saliant 2016b, p. 8]. Those factors were:

- Duration reactor was in operation,
- Frequency and intensity of operation,
- Where possible, the approximate number of potentially exposed workers during its operation (it was determined later that this information was not available); and
- Incidents or other factors with potential to contribute to the risk of unintended or unprotected exposures.

Reactors categorized as high priority were to be evaluated first based on the agreement made during the May 2017 INL/ANL-W Work Group meeting. This report provides the analysis methods and conclusions reached from the evaluations of the Experimental Breeder Reactor No. 2 (EBR-II) and the Boiling Water Reactor Experiment No. 4 (BORAX-IV), two of the high-priority category reactors for the ANL-W site (SEC-00224), to verify that OTIB-0054 does not underestimate the MFAP exposures.

2.0 SCOPE

The scope of this document is to present the potential organ doses associated with the BORAX-IV and EBR-II reactors at the ANL-W site. The evaluations were based on comparative analyses between the following analytical approaches:

1. Direct use of the OTIB-0054 Tool, with its built-in MFAP source terms for the Advanced Test Reactor (ATR), Fast Flux Test Facility (FFTF), Hanford N-Reactor, and Training, Research, Isotopes, General Atomics (TRIGA) reactor; and

2. Reviewed and approved methods established for OTIB-0054, as implemented through use of spreadsheets, along with the BORAX-IV or EBR-II MFAP source term.

The exposure scenario was assumed to be the same in both approaches, the only exception being the source term. The dose estimation was based on assumed gross beta and gross gamma measurements of urine samples with consideration given to:

- Gross beta analysis of minimally processed urinalysis samples with and without radioiodines,
- Gross beta analysis of chemically processed samples with and without radioiodines, and
- Gross gamma analysis of minimally processed samples with and without radioiodines.

It is emphasized that no claimant information, including exposures, was used in any respect or for any purpose.

The evaluation method consisted of the following general steps [ORAUT 2013a]:

- Process the radionuclide inventory data at any given postirradiation decay time to remove actinides, noble gases, and radionuclides that have no activity after the said decay, which ensures that progeny inventories are subjected to the same screening at the time of interest. Use normalized intake fractions (NIFs), intake fraction (product of activity from ORIGEN and appropriate release fraction) normalized for the decay times in Table 6-1, and inhalation dose conversion factors (DCFs) to determine dosimetrically important radionuclides.

- Determine the urinary excretion the indicator radionuclide for the 2- and 10-year chronic exposure periods.

- Calculate the total urine sample activity accounting for delay between collection and counting, from 1 to 30 d.

- Calculate the indicator radionuclide activity fractions using 2-year chronic IRFs for raw and processed urine samples (^{90}Sr in gross beta analysis and ^{137}Cs in gross gamma analysis) and estimate the corresponding activities of the dosimetrically important radionuclides in Table D-1 of OTIB-0054.

- Calculate the respective organ doses using reactor-specific parameters and compare these results to those calculated using generic reactor parameters as described in OTIB-0054.

Because of the computationally complex nature of the evaluations for EBR-II and BORAX-IV, only essential information and data are provided in this report. A calculations roadmap [ORAUT 2019a] provides an overview of the analytical approach and input requirements for the dose comparison analyses used in the generation of this report. All of the detailed development work supporting this summary report can be found in the Site Research Database (SRDB) [ORAUT 2019a,b,c].

3.0 BACKGROUND

OTIB-0054 was developed to provide a basis and standardized approach for assigning radionuclide-specific intakes of MFAPs when air sampling or urinalysis data associated with reactors or reactor fuels are available only as gross beta activity or gross gamma activity. OTIB-0054 also accommodates samples that went through major chemical processing prior to the gross beta analysis. The OTIB-0054 approach is only intended to be used when site-specific guidance is not available for assessing exposures to MFAPs. This approach assigns MFAP intakes and doses based on intakes of indicator radionuclides. The OTIB-0054 approach is based on four different types of reactors and 2-3 representative reactor cases (operating scenarios) for each of those reactors, for a total of nine operating scenarios. The representative reactors used in OTIB-0054 included the ATR, FFTF, N Reactor, and a TRIGA reactor. The assumptions used for those nine operating reactors were selected to ensure favorability to the claimants. As a result, doses assigned based on the OTIB-0054 approach are likely to represent upper bounds [ORAUT 2015].

OTIB-0054 guidance applies to a broad scope of reactor operations. It does not apply to the determination of intakes where radionuclides have been purposely extracted and concentrated or to waste handling operations that caused significant alteration to the source term to which workers were exposed. For INL and ANL-W dose reconstructions, OTIB-0054 is the primary approach used for accounting exposures to unknown mixtures of the fission and activation products when intakes and doses attributable to MFAPs are assessed based on bioassay data. The OTIB-0054 source terms were derived to address gross beta or gross gamma results specifically from air samples or urinalyses, but the source terms can also be used with other monitoring methods, such as in vivo or fecal bioassay, or for cases in which not all radionuclides of concern were measured. For INL and ANL-W dose reconstructions, it is also used with ^{90}Sr in urine data and the ^{137}Cs results from whole-body counts. However, when the bioassay results are specifically in terms of ^{90}Sr and ^{137}Cs (i.e., not as gross beta or gross gamma measurements), all other MFAPs can be excluded from the urine activity fractions. Nonetheless, a reactor-specific source term will still be needed to determine the intake and resultant dose. In 2015, SC&A performed two separate evaluations to determine if OTIB-0054 was applicable to six identified INL reactors. The first evaluation assessed the three Heat Transfer Reactor Experiments, using three separate reactors named HTRE-1, HTRE-2, and HTRE-3, used in the Aircraft Nuclear Propulsion Program in Test Area North [SC&A/Saliant 2015a]. The second evaluation examined the Materials Testing Reactor, the Engineering Test Reactor, and the ATR, all in the Test Reactor Area [SC&A/Saliant 2015b]. During the November 2015 INL/ANL-W Work Group meeting, SC&A was directed to screen the remaining INL and ANL-W reactors and create a prioritized list for detailed examination at a later date with respect to OTIB-0054 applicability.

In March 2016, SC&A released SCA-TR-2016-SEC002 Revision 0, *INL SEC-00219 Reactor Prioritization for Evaluation of ORAUT-OTIB-0054 Applicability*, which detailed their review of INL and ANL-W reactors including a listing of reactors they deemed to be in the high, medium, and low categories [SC&A/Saliant 2016a]. The screening and binning of reactors into the three priority categories was based on the criteria described in Section 1.0 of this report to establish the relative likelihood of OTIB-0054 not providing a bounding approach. After a series of discussions among the ABRWH, SC&A, NIOSH, and the Oak Ridge Associated Universities (ORAU) Team, concurrence was reached on the reactor categorization with agreement that the high-priority category reactors should be evaluated first to ascertain if OTIB-0054 provides a bounding approach for those identified reactors. The seven reactors in the high-priority category include the Experimental Breeder Reactor

No. 1 and No. 2 (ANL-W), SPERT 1 to 4 (INL), Organic Moderated Reactor Experiment (INL), BORAX-IV (ANL-W), Power Burst Facility (INL), and LOFT (INL). The final prioritization of reactor studies can be found in SCA-TR-2016-SEC012, Revision 0, *INL SEC-00219 and ANL-W SEC-00224: SC&A Response to NIOSH Reactor Analysis Plan and Consolidation of All Reactor Modeling Comments* [SC&A/Saliant 2016c].

NIOSH and the ORAU Team decided to evaluate the EBR-II and BORAX-IV reactors first. EBR-II primarily operated in a steady state much like the reactors used in the development of OTIB-0054, while BORAX-IV operated for a relatively short time. The inclusion of a reactor evaluation with a short operating history was deemed to be important as it would provide some insights as to whether OTIB-0054 would still provide bounding guidance. Corresponding analyses for the other high-priority reactors on the INL/ANL-W site are currently in progress and will be reported separately.

4.0 PURPOSE

This report provides a summary of the evaluations of EBR-II and BORAX-IV. The organ doses calculated from gross beta and gross gamma assay using reactor-specific source terms generated for both EBR-II and BORAX-IV are compared to those calculated by the OTIB-0054 method to determine if OTIB-0054 provides a bounding internal dose approach.

5.0 OVERVIEW OF REACTORS AND RATIONALE FOR EVALUATION SELECTION

5.1 EBR-II

EBR-II was categorized as a high-priority reactor for evaluation because the EBR-II fuel types were significantly different from those used in FFTF, the only sodium-cooled fast-neutron reactor evaluated for OTIB-0054. EBR-II used highly enriched uranium (HEU) metal alloy fuel; whereas, the FFTF used mixed plutonium and uranium oxide fuel.

The EBR-II reactor was a second-generation liquid-metal fast breeder reactor that operated from September 30, 1961, to September 30, 1994. EBR-II was an unmoderated sodium-cooled fast-neutron power reactor that was designed to produce 62.5 MW(t). The EBR-II reactor was essentially a scaled-up version of the EBR-I reactor given that it was approximately 20 times larger than EBR-I. Another difference was the liquid-metal coolants. EBR-I used a eutectic alloy of sodium and potassium while EBR-II used only sodium [Stacy 2000]. Both EBR-I and EBR-II were ANL-W reactors. Additional information on EBR-II can be found in the SEC-00224 evaluation report [NIOSH 2016a].

As a breeder reactor, EBR-II was designed to operate with both HEU metal alloy fuel and plutonium-uranium metal alloy fuel. However, it was only operated using HEU metal alloy fuel. Experimental subassemblies were typically placed in the core for irradiation of fuel samples and cladding material, as well as for testing of other reactor materials of interest in the liquid-metal fast breeder reactor development program. Consequently, the EBR-II concept was based on the need to use all means available to achieve a high power density in the reactor while minimizing the total fuel inventory [ANL-W no date]. EBR-II continued fast neutron breeder reactor development, including onsite reprocessing of spent fuel into new fuel pins, to demonstrate the feasibility of a closed fuel cycle.

5.2 BORAX-IV

BORAX-IV was categorized as a high-priority reactor for evaluation because of the use of thoria-urania fuel (ThO_2-UO_2), a mixture of thorium and uranium oxides, during the operational period.

The BORAX series consisted of five different reactor configurations built at the NRTS to determine the self-limiting characteristics of water-cooled reactors and the operating characteristics of boiling water

reactors. The precipitating event for the BORAX series was a steam explosion within a critical assembly mockup in 1952 at Argonne National Laboratory-East in Lemont, Illinois. It was noted afterward that the accidental power excursion, which resulted in boiling of the water coolant, was surprisingly mild and that the steam formation in direct boiling reactors might help stabilize reactor operation [Haroldsen 2008]. This event contradicted the assumption at that time that boiling within a water-cooled reactor would cause the reactor to become unstable. As a result of this accident, the BORAX series of reactor experiments were planned and conducted to understand the safety parameters for operating a boiling-water reactor. These experiments needed to be performed in a less populated area, so the experiments were performed at the NRTS.

In evaluating their safety characteristics, the BORAX reactors were run in both steady-state and transient modes, with the latter intentionally leading to core damage in some cases. For this reason, the reactors were remotely controlled and an exclusion area was maintained around them during operation. Total burnup for each reactor was small because each reactor operated for only a short period and not continuously [SC&A/Saliant 2016b].

BORAX-IV experiments began after the replacement of the BORAX-III core with a maximum of 72 ceramic ThO_2-UO_2 fuel elements to test and demonstrate the feasibility of stable operation with a fuel that (1) could operate at higher temperatures than a uranium core, and (2) was considered less reactive with water coolant in case of cladding rupture [Stacy 2000].

6.0 REACTOR MODELING AND DETERMINATION OF FISSION AND ACTIVATION PRODUCT ACTIVITIES

This section describes the calculation of fission and activation product activity at decay intervals of interest for EBR-II and BORAX-IV.

6.1 SOURCE TERM GENERATION

The analytical tools employed in the source term generation include the TRITON and ORIGEN modules in the SCALE Version 6.2.3 system [Rearden and Jessee 2018]. TRITON was used to model nuclear reactor fuel lattices for reactor core simulation and the generation of radionuclide concentrations (Ci/MTHM) at the end-of-fuel burnup. ORIGEN is a general-purpose point depletion and decay code to calculate isotopic concentrations, decay heat, radiation source terms, and curie levels. ORIGEN was used for the generation of decayed radioactivity at the same time intervals (10 days, 40 days, 180 days, and 1 year) used in the OTIB-0054 method.

For EBR-II, use of TRITON was limited to the generation of radionuclide concentrations (Ci/MTHM) in the EBR-II subassemblies as a function of burnup. The radionuclide concentrations at the end-of-fuel burnup were then converted to curies per subassembly (Mark-IA, Mark-II, or experimental subassemblies) and decayed through the use of ORIGEN. Because this was a one-time ORIGEN application for each subassembly model, the use of ORIGEN-ARP, the cross-section preprocessor for ORIGEN, was not warranted and was circumvented.

For the EBR-II analytical approach, the TRITON model was designed to consist of the equivalent of three Mark-IA or Mark-II subassemblies, or a combination of portions of Mark-IA subassemblies and the experimental subassembly. Details can be found in ORAUT [2019b]. This was done for the following reasons:

1. To ensure proper representation of the reflective surfaces (in a fully occupied lattice), and

2. To provide a model suitable for all the geometries of interest for the present application, specifically a geometry that consists only of subassemblies of the same type (such as Mark-IA

or Mark-II) or a geometry that consists of different subassembly types (such as a combination of Mark-IA and experimental subassemblies).

It should be noted that in earlier versions of SCALE, specifically SCALE 6.0, which was used during the initial development of OTIB-0054, it was necessary to change the default cross-section library used by TRITON to one that would be suitable for fast reactor analyses, as is the case with EBR-II. This is no longer necessary because the issue was resolved in SCALE 6.2, where all the data needed by TRITON and ORIGEN are now collapsed on the fly from fine group cross sections, the fission yields are adjusted according to the energy of fission, and all cross sections use total flux normalization.

For BORAX-IV, use of TRITON was limited to the generation of radionuclide concentrations (Ci/MTHM) in the BORAX-IV quarter core as a function of burnup. The radionuclide concentrations at the end-of-fuel burnup of the ThO_2-UO_2 fuel were then converted to core curies and decayed through the use of ORIGEN. As with EBR-II, this was a one-time ORIGEN application for this core model, so the use of ORIGEN-ARP was not warranted and was circumvented.

The OPUS module, which provides the ability to extract specific data from ORIGEN output libraries, perform unit conversions, and generate plot data in the TRITON sequence, was called three times for printouts of the following:

1. The atoms per barn-cm of the uranium isotopes in the fuel versus irradiation time for the purpose of quantifying the burnup in terms of at% of total heavy atoms (1 barn = 1×10^{-28} m^2),

2. The uranium and plutonium mass concentrations (g/MTHM) versus burnup for quantifying the generation of plutonium versus burnup, and

3. The concentration (Ci/MTHM) of all radionuclides in the TRITON data library (for elements 1 through 100 in the periodic table, hydrogen to fermium).

6.2 DECAY TIMES AND COMPARATIVE ANALYSES

The source terms for the dose comparison analyses were based on the calculated inventory for the end-of-fuel irradiation because that maximizes the inventory and therefore the potential internal exposure. For these evaluations, the end-of-fuel irradiation source term, corrected for in-transit decay to the exposure location or pending process of interest, was conservatively assumed to be applicable during the entire chronic exposure period of interest. Representative decay times of 10, 40, 180, and 365 days for general fuel-cycle locations or processes were used to be consistent with the decay times for OTIB-0054. These decay times apply to the mix of contamination to which workers would have been exposed rather than the age of the fuel itself. Table 6-1 is Table 5-3 from OTIB-0054 and provides descriptions of the work activities for each of the four established decay intervals.

Table 6-1. 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 yr

The dose comparison analyses were limited to comparative analyses between the following analytical approaches:

1. Direct use of the OTIB-0054 Tool, limited to the use of its built-in source terms for the ATR, FFTF, Hanford N Reactor, and TRIGA reactors, and not designed to accept other source terms as input; and

2. The reviewed and approved intake and dose calculation methods established by OTIB-0054, as implemented through the use of data spreadsheets [ORAUT 2013a], along with the EBR-II and BORAX-IV source terms. These spreadsheets were recently updated to improve their applicability to the INL and ANL-W sites by incorporating the dose comparison analyses. It was important to verify that the updates were correctly implemented. The verification is in a separate document [ORAUT 2019a]; it consists of a one-to-one comparison of doses generated by the spreadsheets for three of the reactor cases analyzed in OTIB-0054, namely the FFTF, ATR, and TRIGA reactors, versus the corresponding results from the OTIB-0054 Tool. Only minor round-off-error differences were identified in the comparison.

In OTIB-0054, the source term was based on continuous reactor operation at the designed power level for an irradiation period that would yield the burnup of interest. The OTIB-0054 approach minimized the irradiation time, which in turn yielded a source term that was favorable to the claimant. Specifically, the approach maximizes the inventory of an associated radionuclide in relation to that of an indicator radionuclide and therefore increases its contribution to the internal exposure; it also increases the number of radionuclides that contribute to the dose. This is a consequence of the fact that, in general, the source inventory of a short-lived radionuclide is proportional to the power level, whereas that of a long-lived radionuclide is proportional to the burnup (i.e., the product of power and irradiation interval).

Unlike the OTIB-0054 approach, the EBR-II and BORAX-IV source terms were based on as realistic as possible power histories based on available information in the literature. However, all other dose calculation parameters were selected to be the same in both approaches. The comparative analyses between the OTIB-0054 Tool and the spreadsheets was based on a single assumed scenario that did not include any worker information whatsoever. The exposure scenario was assumed to be the same in both approaches, the only difference being the source term. Specifically, the dose estimation was based on the same assumed gross beta and gross gamma measurements of urine samples for 2-year and 10-year chronic exposure periods.

7.0 SOURCE TERM DEVELOPMENT FOR EVALUATION

TRITON was used for generation of the radionuclide concentrations (Ci/MTHM) as a function of burnup. The radionuclide concentrations at the end-of-fuel burnup were then decayed through the use of ORIGEN. Tabulations of the entire lists of radionuclide inventories for EBR-II and BORAX-IV can be found in the associated TRITON and ORIGEN outputs [ORAUT 2019b,c]. The radionuclides with initial inventories less than 1×10^{-6} Ci/MTHM were excluded because they contribute 0% to the total potential dose (the definition of “zero” is provided in Section 8.0). The remaining 900+ radionuclides with initial inventories greater than 1×10^{-6} Ci/MTHM were then converted to total assembly or core curies and provided as input to ORIGEN for postirradiation decay. Subsequently, the decay-corrected inventories (with decay times equal to 10, 40, 180 and 365 days) were subjected to computerized screening based on their dose contributions to any of 27 organs in the database. Those radionuclides contributing more than 1% of the dose to any organ, at any postirradiation decay, were then selected for further processing. This process led to the elimination of all but no more than 30 of the initial 900+ radionuclides for each evaluation (see Attachment A) . The radionuclides listed in the tables in Attachment A are analogous to those in Table D-1 in OTIB-0054.

7.1 SOURCE TERM SUMMARY FOR EBR-II

The source term inventories for all of the analyzed subassemblies are too extensive for inclusion in this report. As an alternative, Table 7-1 was generated to summarize the cases for which subassembly inventories were identified for evaluation against OTIB-0054. The respective source term tables are identified. ORAUT [2019b] provides the details on the power histories for all of the various subassemblies. Blanket subassemblies were excluded from analyses after concluding that blanket subassemblies were not internal dose hazards to INL and ANL-W workers. There are no indications that the cladding on the blanket subassemblies ever failed during reactor operation. The irradiated depleted uranium blankets were initially intended to be reprocessed at the Fuel Cycle Facility at ANL-W to recover reactor-bred plutonium, but the equipment for reprocessing was never installed [Stevenson 1987]. Furthermore, there are no known processing EBR-II blanket subassemblies at the Chemical Processing Plant [ORAUT 2019b]. Given these facts, the blanket subassemblies were considered sealed sources that did not represent an internal exposure hazard.

Table 7-1. Summary of EBR-II source terms evaluated against OTIB-0054.

Description	Modeled uptime and lifetime (d)	Uptime average power level (MW(t)/MTHM)	Burnup (at%)	Source-term table in ORAUT 2019b
Mark-IA average fuel subassembly (U - 5 wt% fissium; U-235 enrichment 52.18 wt%)	116 and 149	129.0	1.77	Table 2-12
Mark-II peak fuel subassembly (U - 5 wt% fissium; U-235 enrichment 66.57 wt%)	347 and 480	168.4	6.64	Table 3-11
Experimental subassembly with peak burnup (in Row 6 of the core) (UO_2 - 20 wt% PuO_2; U-235 enrichment 92.7 wt%)	564 and 1,211	75.04	7.90	Table 4-10

7.2 SOURCE TERM SUMMARY FOR BORAX-IV

The source term inventory for the BORAX-IV ThO_2-UO_2 core is too extensive for inclusion in this report. Table 7-2 summarizes the case for which core inventory was identified for evaluation against OTIB-0054. ORAUT [2019c] provides the details on the power histories for the ThO_2-UO_2 core.

Table 7-2. Summary of BORAX-IV source term evaluated against OTIB-0054.

Description	Modeled uptime and lifetime (d)	Core-average power level (MW(t)/MTHM)	Burnup (at%)	Source-term table in ORAUT 2019c
BORAX-IV core (93.65 wt% ThO_2 and 6.35 wt% UO_2; U-235 enrichment 93.2 wt%)	21 and 356	50.45	0.11	Table 2-11

8.0 DETERMINATION OF INTAKE FRACTIONS AND DOSIMETRICALLY IMPORTANT RADIONUCLIDES

The determination of intake fractions used the method in OTIB-0054. The source-term inventories of 900+ radionuclides generated by ORIGEN for EBR-II and BORAX-IV for each of the four established decay intervals were independently processed to select the fission and activation product activity values. Radionuclides with zero activity at 10 days of decay were removed because 10 days was the first established decay interval. In OTIB-0054, zero is defined as the threshold for the hardware/software environment, typically less than 10^{-50}. Values below this threshold are set to zero by the code (ORIGEN). Radionuclides without defined inhalation DCFs in Rad Toolbox version 3.0.0

and noble gases were also removed. The removal of noble gases was because they do not contribute significantly to internal dose under most circumstances. However, progeny of noble gases were not removed as they can contribute to internal dose.

Release fractions from DOE Standard 1027 [DOE 1997] were used to convert from activity at each of the four decay intervals to intake fraction. The release fractions are not reproduced in this report but can be found in Attachment B of OTIB-0054. For each radionuclide and decay time in each evaluated source term, the product of the activity and release fraction was summed and the values were restated on a normalized basis. The values are denoted as NIFs because the product of the radionuclide activity and the release fraction was defined as an intake fraction.

Committed organ doses for each source term case for EBR-II and BORAX-IV at the four established decay intervals were calculated using inhalation DCFs from Rad Toolbox version 3.0.0. Using the most restrictive DCFs for all solubility types per radionuclide, the collective set of organ dose results for each source term case was evaluated to identify all radionuclides that contributed ≥1% of the dose to any organ for any reactor case, decay interval, or solubility category. Next, the NIF values for each reactor case and decay interval were renormalized using just the dosimetrically significant (i.e., those contributing ≥1% of the dose to any organ) radionuclides. The resulting NIF values for each source term case for EBR-II and BORAX-IV are listed in Tables A-1 through A-4 in Attachment A. 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.

Next, the Table A-1 through Table A-4 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 as a subset and a subsequent renormalization of the corresponding NIF values was performed. This produced a reduced list of dosimetrically important radionuclides. The renormalized NIF values are listed in Tables B-1 through B-4 in Attachment B. As with the first reduction in the number of radionuclides of interest, this subsequent reduction is favorable to claimants in that it results in higher assigned intakes. Renormalizing the NIF values in this manner is favorable to claimants because it increases the intake fraction values for the most important radionuclides versus what the fraction would be if all radionuclides were included.

The reduced lists of dosimetrically important radionuclides were used only to establish the relative intake fractions for the other components of an intake once the intake for the indicator radionuclide has been established. Intake assignments for the indicator radionuclides are based on the NIFs in Tables A-1 through A-4.

Dosimetrically significant radionuclides, both in terms of quantity and total activity, differ based on reactor design and operation. Reactor design parameters such as the type of fuel, enrichment, cladding, moderator, and coolant are key factors in the source term generated under power. Operational parameters such as length of operation, steady-state or periodic operation, and operation of reactor within or outside of design criteria are also central factors. The determination of the dosimetrically significant radionuclides and the NIFs for EBR-II and BORAX-IV in the Attachment A tables demonstrate the effect of reactor design and operation.

9.0 REVIEW OF DETERMINATION OF INTAKES FROM NONSPECIFIC RADIONUCLIDE ANALYSIS

The calculation of activity and intake fraction values for assigning intakes from gross beta or gross gamma assay data made use of the method in OTIB-0054. First, the calculated intake mixture was used to determine the quantity of each radionuclide in the assay of a 24-hour urine sample after a

2-year chronic intake. The gross beta or gross gamma activity for a 24-hour urine sample depends on factors such as inhalation mode, biokinetics of the radionuclides, and decay and ingrowth in the urine sample between collection and counting. Such considerations would not be needed for an air sample as radionuclide mixture, concentration of radionuclides in the air, and exposure time are required.

The activity fraction of an indicator radionuclide, either ^{90}Sr for gross beta or ^{137}Cs for gross gamma, that should be present in a given sample type was then determined for each reactor case, decay time, and analysis type. Once the indicator radionuclide intake was established, the intake for the remaining radionuclides in the mix was assigned using activity ratios based on the NIFs in Tables A-1 through A-4 to determine the total intake. Absorption types for each radionuclide were then selected to yield the most favorable dose assignment for the organ of interest.

The gross beta or gross gamma activity for a given sample depends on:

- Radionuclide mixture at intake,
- Intake mode,
- Biokinetics,
- Radionuclide decay and ingrowth that occur in the sample during the time between collection and counting,
- Separations chemistry, if used during sample preparation, and
- 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. Chronic daily intake rates for the 2- and 10-year chronic exposures were calculated using these exposure parameters and are presented below [ORAUT 2019a]:

- ^{90}Sr: 4.329 pCi/d (2-year exposure) 4.117 pCi/d (10-year exposure)
- ^{137}Cs: 2.645 pCi/d (2-year exposure) 2.622 pCi/d (10-year exposure)

Note: The difference in the intake rates results from the slightly lower intake rate over 10 years to give the same urinary excretion (1 pCi/d) versus 2 years. Rather than multiplying the assumed unit excretion rate by the urine fraction and then calculating the indicator nuclide intake rate from that, the intake rate for a unit excretion of the indicator radionuclide is calculated and then multiplied by the applicable urine fraction.

These intake rates can then be used to determine the mixture and magnitude of the total intake. Activity fractions for the indicator radionuclides in gross beta or gross gamma urinalyses were established for the following three types of analyses with and without radioiodines:

- Gross beta analysis of minimally processed samples,
- Gross beta analysis of chemically processed samples, and
- Gross gamma analysis of minimally processed samples.

The 2-year IRFs used in the activity fraction calculation, half-lives used for decay corrections, and yields for beta and gamma from Attachment F of OTIB-0054 were used, for both the 2-year and 10-year chronic exposures.

As previously described, the potential dose comparison analyses for each source term case for EBR-II and BORAX-IV was limited to comparative analyses between the following analytical approaches:

- Direct use of the OTIB-0054 Tool, with its built-in source terms for the ATR, FFTF, Hanford N Reactor, and TRIGA reactors; and

- Reviewed and approved methods established for OTIB-0054, as implemented through use of spreadsheets, along with the EBR-II and BORAX-IV source terms.

For informational purposes, results were also calculated for workplace samples, specifically smears for removable contamination or air samples for particulates assayed by either gross beta or gross gamma counting. Calculation of the indicator radionuclide activity fractions for workplace samples was essentially the same as that for urinalyses and only used to establish the relative intake fractions for the other components of an intake once the intake for the indicator radionuclide was established [ORAUT 2013a].

In summary, the following were calculated and presented in tables in the respective development files for EBR-II and BORAX-IV for each of the evaluated source terms [ORAUT 2019b,c]:

- Beta activity in air and workplace samples based on the Attachment A normalized NIFs;

- Gamma activity in air and workplace samples based on the Attachment A normalized NIFs;

- ^{90}Sr beta and ^{137}Cs/^{137m}Ba gamma activity fractions in workplace and urine samples;

- Associated radionuclide intake fractions in urine samples relative to the indicator radionuclides;

- ^{90}Sr and ^{137}Cs intake rates for a 2-year exposure that would yield the urine sample activity fractions;

- Summary of committed organ doses from gross beta analyses with minimal chemical processing for a 2-year exposure;

- Summary of committed organ doses from gross beta analyses with chemical processing for a 2-year exposure;

- Summary of committed organ doses from gross gamma analyses with minimal chemical processing for a 2-year exposure;

- Organ dose comparison with the OTIB-0054 Tool for a 2-year exposure;

- ^{90}Sr and ^{137}Cs intake rates for a 10-year exposure that would yield the urine sample activity fractions;

- Summary of committed organ doses from gross beta analyses with minimal chemical processing for a 10-year exposure;

- Summary of committed organ doses from gross beta analyses with chemical processing for a 10-year exposure;

- Summary of committed organ doses from gross gamma analyses with minimal chemical processing for a 10-year exposure; and

- Organ dose comparison with the OTIB-0054 Tool for a 10-year exposure.

10.0 CONCLUSIONS

In the current ORAUT-TKBS-0007-5, *Idaho National Laboratory and Argonne National Laboratory-West – Occupational Internal Dose* [ORAUT 2010], OTIB-0054 is the primary approach used for accounting exposures to unknown mixtures of fission and activation products when intakes and doses are assessed based on bioassay data. The OTIB-0054 approach is primarily intended to be used with gross beta activity or gross gamma activity in urine data. However, it can also be used with ^{90}Sr in urine data or the ^{137}Cs results from whole-body counts. However, when the bioassay results are specifically in terms of ^{90}Sr and ^{137}Cs (i.e., not as gross beta or gross gamma measurements), all other MFAPs can be excluded from the urine activity fractions.

OTIB-0054 is intended to provide a generic approach to account for exposures to unknown mixtures of the fission and activation products that are usable for all Atomic Weapons Employer and DOE sites. OTIB-0054 uses 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 conservatisms and a maximizing approach to encompass uncertainties associated with dose assignments such that the approach represents an upper bound. Because ANL-W has functioned as a reactor testing facility throughout its history, there was concern raised by the INL/ANL-W Work Group that OTIB-0054 might not provide a bounding approach and could potentially underestimate worker exposures to MFAPs for certain reactors that had unique operating conditions. This report provides a summary of the evaluations of two high-priority category ANL-W reactors and compared the organ doses calculated from the reactor-specific source terms to those generated by OTIB-0054 to verify OTIB-0054 does not underestimate the MFAP exposures for workers at EBR-II and BORAX-IV.

An extensive evaluation of the EBR-II and BORAX-IV operating histories was performed with the identification of three source term cases for EBR-II and one for BORAX-IV deemed to represent the cases most likely not to be bound by OTIB-0054 guidance. These source term cases were provided in Tables 7-1 and 7-2. The details of the full evaluation process for each reactor's history and the technical justification for which subassembly inventories were identified for evaluation against OTIB-0054 are provided in Section 1.2 of the technical reports for each EBR-II and BORAX-IV [ORAUT 2019b, 2019c]. The potential dose comparison analyses of the reactor cases was based on comparative analyses with predicted values based on the OTIB-0054 Tool. Tables 10-1 and 10-2 provide summaries for the 2- and 10-year chronic exposure intervals evaluated. Notes of interest include:

1. The tables include results for postulated gross beta and gross gamma urinalysis, specifically for minimally or chemically processed gross beta samples and minimally processed gross gamma samples, with and without radioiodines in all cases.

2. The tables include only the critical organs in the comparative analyses, specifically the organs with the smallest (OTIB-0054 Tool/EBR-II or BORAX-IV) dose ratios, and therefore those with the least bounding margin. The OTIB-0054 Tool provides bounding results for all cases analyzed.

3. There is practically no difference between the 2- and 10-year chronic exposures to any organ.

In conclusion, the ORAU Team's evaluation indicates that OTIB-0054 provides a bounding approach for the EBR-II and BORAX-IV reactors. The two reactors were part of a group of six reactors that operated during the SEC evaluation periods for INL and ANL-W that were classified as high-priority

reactors deemed to require further evaluation. Corresponding analyses for the other high-priority reactors on the INL/ANL-W site are currently in progress and will be reported separately.

Table 10-1. Summary of EBR-II dose ratios (OTIB-0054 Tool/EBR-II calculation). This summary shows the results for the organ with the smallest dose ratio.

Description	2-yr chronic exposure for Mark-IA subassembly	10-yr chronic exposure for Mark-IA subassembly	2-yr chronic exposure for Mark-II subassembly	10-yr chronic exposure for Mark-II subassembly	2-yr chronic exposure for Experimental subassembly	10-yr chronic exposure for Experimental subassembly
Minimally processed gross beta sample, with radioiodines	1.019 (liver)	1.019 (liver)	1.123 (bone surface)	1.125 (bone surface)	1.368 (bone surface)	1.370 (bone surface)
Minimally processed gross beta sample, w/o radioiodines	1.019 (liver)	1.019 (liver)	1.123 (bone surface)	1.125 (bone surface)	1.369 (bone surface)	1.370 (bone surface)
Chemically processed gross beta sample, with radioiodines	1.558 (bone surface)	1.560 (bone surface)	1.715 (bone surface)	1.720 (bone surface)	1.929 (bone surface)	1.936 (bone surface)
Chemically processed gross beta sample, w/o radioiodines	1.558 (bone surface)	1.560 (bone surface)	1.715 (bone surface)	1.720 (bone surface)	1.929 (bone surface)	1.936 (bone surface)
Gross gamma sample, with radioiodines	1.001 (thymus and esophagus)	1.001 (thymus and esophagus)	1.278 (thymus and esophagus)	1.278 (thymus and esophagus)	1.485 (testes)	1.485 (testes)
Gross gamma sample, w/o radioiodines	1.064 (liver)	1.064 (liver)	1.369 (testes)	1.369 (testes)	1.508 (testes)	1.508 (testes)
Table in ORAUT [2019b] with comparative dose summary	Table 2-23	Table 2-28	Table 3-22	Table 3-27	Table 4-21	Table 4-26

Table 10-2. Summary of BORAX-IV dose ratios (OTIB-0054 Tool/BORAX-IV calculation). This summary shows the results for the organ with the smallest dose ratio.

Description	Dose ratio for 2-yr exposure	Dose ratio for 10-yr exposure
Minimally processed gross beta sample, with radioiodines	1.031 (bone surface)	1.033 (bone surface)
Minimally processed gross beta sample, w/o radioiodines	1.031 (bone surface)	1.033 (bone surface)
Chemically processed gross beta sample, with radioiodines	1.634 (bone surface)	1.637 (bone surface)
Chemically processed gross beta sample, w/o radioiodines	1.634 (bone surface)	1.637 (bone surface)
Gross gamma sample, with radioiodines	1.046 (thymus and esophagus)	1.046 (thymus and esophagus)
Gross gamma sample, w/o radioiodines	1.067 (red bone marrow)	1.067 (red bone marrow)
Table in ORAUT [2019c] with comparative dose summary	Table 2-22	Table 2-27

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ATTACHMENT A
NORMALIZED INTAKE FRACTIONS FOR THE DOSIMETRICALLY IMPORTANT RADIONUCLIDES

LIST OF TABLES

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

The NIFs for the dosimetrically important radionuclides for EBR-II and BORAX-IV are presented in Tables A-1 through A-4. These are the radionuclides, in any elemental solubility type (solubility types are identified as F, M and S, for fast, moderate and slow rates of absorption into the blood from the respiratory tract.), contributing more than 1 % to any organ dose, used in quantification of the indicator radionuclide fractional contributions to urine sample gross beta and gross gamma measurements.

Table A-1. NIFs for EBR-II Mark-IA subassembly versus postirradiation decay time [ORAUT 2019b, Table 2-13].

Radionuclide	10 d	40 d	180 d	1 yr
Mn-54	1.595E-04	8.445E-04	3.122E-03	5.648E-03
Fe-55	4.066E-05	2.254E-04	1.032E-03	2.480E-03
Co-58	7.350E-04	3.102E-03	3.980E-03	1.774E-03
Sr-89	2.068E-02	7.757E-02	5.736E-02	1.233E-02
Sr-90	3.183E-04	1.798E-03	8.989E-03	2.424E-02
Y-90	3.178E-04	1.799E-03	8.991E-03	2.425E-02
Y-91	2.536E-02	1.006E-01	9.668E-02	2.940E-02
Zr-95	2.737E-02	1.119E-01	1.241E-01	4.561E-02
Nb-95	2.338E-02	1.314E-01	2.371E-01	9.798E-02
Mo-99	4.331E-03	1.271E-05	2.994E-20	4.231E-40
Ru-103	1.615E-02	5.380E-02	2.290E-02	2.372E-03
Ru-106	7.494E-04	4.011E-03	1.559E-02	3.013E-02
Te-127m	1.164E-04	5.528E-04	1.145E-03	9.626E-04
Te-129m	5.185E-04	1.580E-03	4.440E-04	2.654E-05
Te-132	4.426E-03	3.804E-05	1.347E-17	1.447E-34
I-131	5.082E-01	2.156E-01	6.098E-06	1.873E-12
I-132	2.281E-01	1.960E-03	6.943E-16	7.456E-33
Cs-134	8.178E-06	4.503E-05	1.998E-04	4.601E-04
Cs-137	3.384E-04	1.912E-03	9.562E-03	2.580E-02
Ba-140	2.537E-02	2.812E-02	7.036E-05	8.141E-09
La-140	2.909E-02	3.239E-02	8.102E-05	9.375E-09
Ce-141	2.810E-02	8.388E-02	2.139E-02	1.124E-03
Ce-144	9.347E-03	4.918E-02	1.765E-01	3.071E-01
Pr-143	2.748E-02	3.364E-02	1.331E-04	2.823E-08
Pr-144	9.347E-03	4.918E-02	1.765E-01	3.071E-01
Nd-147	8.715E-03	7.423E-03	5.435E-06	1.237E-10
Pm-147	1.231E-03	7.293E-03	3.365E-02	8.033E-02
Eu-155	1.327E-05	7.424E-05	3.542E-04	8.981E-04
Totals	**1.000E+00**	**1.000E+00**	**1.000E+00**	**1.000E+00**

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

Table A-2. NIFs for EBR-II Mark-II subassembly versus postirradiation decay time [ORAUT 2019b, Table 3-12].

Radionuclide	10 d	40 d	180 d	1 yr
Mn-54	3.461E-04	1.561E-03	4.481E-03	6.441E-03
Fe-55	1.201E-04	5.670E-04	2.016E-03	3.847E-03
Co-58	1.391E-03	5.001E-03	4.981E-03	1.764E-03
Sr-89	2.228E-02	7.118E-02	4.086E-02	6.982E-03
Sr-90	9.228E-04	4.440E-03	1.724E-02	3.693E-02
Y-90	9.237E-04	4.443E-03	1.724E-02	3.694E-02
Y-91	2.737E-02	9.251E-02	6.901E-02	1.668E-02
Zr-95	3.016E-02	1.051E-01	9.046E-02	2.641E-02
Nb-95	2.642E-02	1.251E-01	1.733E-01	5.677E-02
Mo-99	3.786E-03	9.466E-06	1.731E-20	1.944E-40
Ru-103	1.711E-02	4.857E-02	1.605E-02	1.321E-03
Ru-106	1.491E-03	6.800E-03	2.052E-02	3.151E-02
Cd-113m	9.003E-08	4.323E-07	1.662E-06	3.517E-06
Te-129m	5.582E-04	1.449E-03	3.162E-04	1.501E-05
Te-132	3.875E-03	2.836E-05	7.800E-18	6.655E-35
I-131	4.966E-01	1.795E-01	3.941E-06	9.616E-13
I-132	1.997E-01	1.462E-03	4.020E-16	3.430E-33
Cs-134	7.434E-05	3.487E-04	1.201E-03	2.198E-03
Cs-137	9.763E-04	4.698E-03	1.824E-02	3.911E-02
Ba-140	2.678E-02	2.529E-02	4.912E-05	4.516E-09
La-140	3.073E-02	2.912E-02	5.656E-05	5.200E-09
Ce-141	3.055E-02	7.770E-02	1.538E-02	6.424E-04
Ce-144	1.807E-02	8.098E-02	2.257E-01	3.119E-01
Pr-143	2.935E-02	3.060E-02	9.399E-05	1.584E-08
Pr-144	1.807E-02	8.098E-02	2.257E-01	3.119E-01
Nd-147	8.984E-03	6.519E-03	3.706E-06	6.703E-11
Pm-147	3.253E-03	1.576E-02	5.608E-02	1.064E-01
Sm-151	2.231E-05	1.075E-04	4.199E-04	9.073E-04
Eu-154	2.466E-06	1.181E-05	4.486E-05	9.341E-05
Eu-155	3.444E-05	1.641E-04	6.078E-04	1.224E-03
Totals	**1.000E+00**	**1.000E+00**	**1.000E+00**	**1.000E+00**

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

Table A-3. NIFs for EBR-II experimental subassembly versus postirradiation decay time [ORAUT 2019b, Table 4-11].

Radionuclide	10 d	40 d	180 d	1 yr
Mn-54	1.903E-03	5.159E-03	9.772E-03	1.167E-02
Fe-55	1.078E-03	3.059E-03	7.178E-03	1.138E-02
Co-58	3.500E-03	7.562E-03	4.970E-03	1.463E-03
Sr-89	3.215E-02	6.172E-02	2.338E-02	3.320E-03
Sr-90	3.955E-03	1.144E-02	2.930E-02	5.217E-02
Y-90	3.957E-03	1.145E-02	2.930E-02	5.218E-02
Y-91	4.337E-02	8.808E-02	4.336E-02	8.707E-03
Zr-95	5.348E-02	1.120E-01	6.361E-02	1.543E-02
Nb-95	6.997E-02	1.702E-01	1.278E-01	3.344E-02
Mo-99	4.928E-03	7.405E-06	8.936E-21	8.337E-41
Ru-103	2.810E-02	4.793E-02	1.045E-02	7.149E-04
Ru-106	1.164E-02	3.188E-02	6.349E-02	8.102E-02
Cd-113m	7.286E-07	2.103E-06	5.334E-06	9.378E-06
Te-129m	8.242E-04	1.286E-03	1.851E-04	7.305E-06
Te-132	4.677E-03	2.057E-05	3.733E-18	2.647E-35
I-131	3.032E-01	6.597E-02	9.556E-07	1.938E-13
I-132	2.410E-01	1.060E-03	1.924E-16	1.364E-33
Cs-134	3.123E-04	8.803E-04	2.001E-03	3.042E-03
Cs-137	4.896E-03	1.416E-02	3.628E-02	6.465E-02
Ba-140	1.446E-02	8.203E-03	1.051E-05	8.032E-10
La-140	1.644E-02	9.446E-03	1.211E-05	9.250E-10
Ce-141	3.546E-02	5.420E-02	7.080E-03	2.458E-04
Ce-144	4.576E-02	1.233E-01	2.267E-01	2.604E-01
Pr-143	1.583E-02	9.949E-03	2.016E-05	2.824E-09
Pr-144	4.576E-02	1.233E-01	2.267E-01	2.604E-01
Pm-147	1.284E-02	3.654E-02	8.543E-02	1.347E-01
Sm-151	1.298E-04	3.759E-04	9.689E-04	1.740E-03
Eu-154	1.550E-05	4.463E-05	1.119E-04	1.935E-04
Eu-155	2.721E-04	7.791E-04	1.905E-03	3.189E-03
Totals	**1.0000E+00**	**1.0000E+00**	**1.0000E+00**	**1.0000E+00**

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

Table A-4. NIFs for BORAX-IV versus postirradiation decay time [ORAUT 2019c, Table 2-12].

Radionuclide	10 d	40 d	180 d	1 yr
Sr-89	2.187E-02	7.602E-02	4.477E-02	8.083E-03
Sr-90	6.883E-04	3.603E-03	1.435E-02	3.249E-02
Y-90	6.873E-04	3.605E-03	1.435E-02	3.250E-02
Y-91	2.710E-02	9.963E-02	7.625E-02	1.947E-02
Zr-95	2.996E-02	1.136E-01	1.003E-01	3.094E-02
Nb-95	3.112E-02	1.492E-01	1.956E-01	6.669E-02
Mo-99	5.863E-03	1.595E-05	2.992E-20	3.549E-40
Ru-103	1.361E-02	4.204E-02	1.426E-02	1.240E-03
Ru-106	9.809E-04	4.865E-03	1.506E-02	2.444E-02
Te-129m	4.059E-04	1.146E-03	2.565E-04	1.287E-05
Te-132	5.355E-03	4.264E-05	1.203E-17	1.085E-34
I-131	4.444E-01	1.749E-01	3.939E-06	1.016E-12
I-132	2.760E-01	2.198E-03	6.201E-16	5.591E-33
Cs-134	1.271E-05	6.485E-05	2.292E-04	4.430E-04
Cs-137	7.046E-04	3.688E-03	1.469E-02	3.329E-02
Ba-140	2.260E-02	2.321E-02	4.625E-05	4.494E-09
La-140	2.576E-02	2.673E-02	5.327E-05	5.174E-09
Ce-141	2.602E-02	7.200E-02	1.462E-02	6.453E-04
Ce-144	1.600E-02	7.801E-02	2.230E-01	3.257E-01
Pr-143	2.454E-02	2.790E-02	8.791E-05	1.566E-08
Pr-144	1.600E-02	7.801E-02	2.230E-01	3.257E-01
Nd-147	7.731E-03	6.102E-03	3.559E-06	6.802E-11
Pm-147	2.520E-03	1.332E-02	4.864E-02	9.751E-02
Eu-155	1.925E-05	9.974E-05	3.791E-04	8.069E-04
Totals	**1.000E+00**	**1.000E+00**	**1.000E+00**	**1.000E+00**

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

LIST OF TABLES

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

The NIFs for the reduced list of dosimetrically important radionuclides for EBR-II and BORAX-IV are presented in Tables B-1 through B-4. These are a subset of the radionuclides in Attachment A contributing more than 1 % of the committed effective dose. The NIFs in Tables B-1 through B-4 were used to calculate effective dose for each reactor case, decay interval, and solubility category. Radionuclides that contributed less than 1% to effective dose in any case were identified with a subsequent renormalization of the corresponding NIF values performed. This reduced list of dosimetrically important radionuclides is only used to establish the relative intake fractions for the other components of an intake once the intake for the indicator radionuclide has been established.

Table B-1. NIFs for reduced list of dosimetrically important radionuclides for EBR-II Mark-IA subassembly versus postirradiation decay time [ORAUT 2019b, Table 2-14].

Radionuclide	10 d	40 d	180 d	1 yr
Sr-89	3.128E-02	9.245E-02	7.135E-02	1.879E-02
Sr-90	4.814E-04	2.143E-03	1.118E-02	3.693E-02
Y-91	3.836E-02	1.199E-01	1.203E-01	4.480E-02
Zr-95	4.139E-02	1.334E-01	1.544E-01	6.948E-02
Nb-95	3.536E-02	1.566E-01	2.950E-01	1.493E-01
Ru-103	2.442E-02	6.412E-02	2.849E-02	3.614E-03
Ru-106	1.133E-03	4.781E-03	1.939E-02	4.590E-02
I-131	7.686E-01	2.570E-01	7.585E-06	2.853E-12
Cs-137	5.118E-04	2.279E-03	1.189E-02	3.931E-02
Ce-141	4.249E-02	9.997E-02	2.661E-02	1.713E-03
Ce-144	1.414E-02	5.862E-02	2.196E-01	4.678E-01
Pm-147	1.862E-03	8.692E-03	4.185E-02	1.224E-01
Totals	**1.000E+00**	**1.000E+00**	**1.000E+00**	**1.000E+00**

Table B-2. NIFs for reduced list of dosimetrically important radionuclides for EBR-II Mark-II subassembly versus postirradiation decay time [ORAUT 2019b, Table 3-13].

Radionuclide	10 d	40 d	180 d	1 yr
Sr-89	3.299E-02	8.763E-02	5.501E-02	1.100E-02
Sr-90	1.367E-03	5.466E-03	2.320E-02	5.819E-02
Y-91	4.054E-02	1.139E-01	9.290E-02	2.628E-02
Zr-95	4.467E-02	1.294E-01	1.218E-01	4.162E-02
Nb-95	3.913E-02	1.540E-01	2.333E-01	8.945E-02
Ru-103	2.534E-02	5.979E-02	2.161E-02	2.082E-03
Ru-106	2.209E-03	8.371E-03	2.762E-02	4.965E-02
I-131	7.355E-01	2.210E-01	5.305E-06	1.515E-12
Cs-137	1.446E-03	5.784E-03	2.456E-02	6.163E-02
Ce-141	4.524E-02	9.564E-02	2.071E-02	1.012E-03
Ce-144	2.675E-02	9.969E-02	3.038E-01	4.915E-01
Pm-147	4.818E-03	1.941E-02	7.549E-02	1.676E-01
Totals	**1.0000E+00**	**1.0000E+00**	**1.0000E+00**	**1.0000E+00**

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

Table B-3. NIFs for reduced list of dosimetrically important radionuclides for EBR-II experimental subassembly versus postirradiation decay time [ORAUT 2019b, Table 4-12].

Radionuclide	10 d	40 d	180 d	1 yr
Sr-89	4.985E-02	7.551E-02	3.262E-02	5.071E-03
Sr-90	6.134E-03	1.399E-02	4.087E-02	7.968E-02
Y-91	6.725E-02	1.077E-01	6.048E-02	1.330E-02
Zr-95	8.294E-02	1.370E-01	8.873E-02	2.357E-02
Nb-95	1.085E-01	2.083E-01	1.783E-01	5.108E-02
Ru-103	4.357E-02	5.863E-02	1.458E-02	1.092E-03
Ru-106	1.805E-02	3.900E-02	8.857E-02	1.237E-01
I-131	4.702E-01	8.070E-02	1.333E-06	2.960E-13
Cs-137	7.593E-03	1.732E-02	5.061E-02	9.873E-02
Ce-141	5.499E-02	6.631E-02	9.877E-03	3.753E-04
Ce-144	7.097E-02	1.508E-01	3.162E-01	3.977E-01
Pm-147	1.992E-02	4.471E-02	1.192E-01	2.057E-01
Totals	**1.0000E+00**	**1.0000E+00**	**1.0000E+00**	**1.0000E+00**

Table B-4. NIFs for reduced list of dosimetrically important radionuclides for BORAX-IV versus postirradiation decay time [ORAUT 2019c, Table 2-13].

Radionuclide	10 d	40 d	180 d	1 yr
Sr-89	3.556E-02	9.149E-02	5.879E-02	1.262E-02
Sr-90	1.119E-03	4.336E-03	1.884E-02	5.072E-02
Y-91	4.407E-02	1.199E-01	1.001E-01	3.040E-02
Zr-95	4.871E-02	1.367E-01	1.317E-01	4.830E-02
Nb-95	5.060E-02	1.796E-01	2.569E-01	1.041E-01
Ru-103	2.214E-02	5.060E-02	1.872E-02	1.935E-03
Ru-106	1.595E-03	5.855E-03	1.978E-02	3.816E-02
I-131	7.226E-01	2.105E-01	5.173E-06	1.586E-12
Cs-137	1.146E-03	4.439E-03	1.929E-02	5.197E-02
Ce-141	4.231E-02	8.665E-02	1.920E-02	1.007E-03
Ce-144	2.601E-02	9.389E-02	2.929E-01	5.085E-01
Pm-147	4.097E-03	1.603E-02	6.387E-02	1.522E-01
Totals	**1.000E+00**	**1.000E+00**	**1.000E+00**	**1.000E+00**