

APPENDIX E

EVALUATION OF HUMAN HEALTH EFFECTS FROM NORMAL OPERATIONS

E.1 INTRODUCTION

This appendix provides a brief general discussion on radiation, and its associated health effects, and describes the method and assumptions used for estimating the potential impacts and risks to individuals and the general public from exposure to the releases of radioactivity and hazardous chemicals during normal operations at the proposed facilities. Information regarding potential radiological impacts resulting from facility accidents is provided in Appendix F of this environmental impact statement (EIS).

This appendix presents numerical information using engineering and/or scientific notation. For example, the number 100,000 can also be expressed as 1×10^5 . The fraction 0.00001 can also be expressed as 1×10^{-5} . The following chart defines the equivalent numerical notations that may be used in this appendix.

FRACTIONS AND MULTIPLES OF UNITS			
<i>Multiple</i>	<i>Decimal Equivalent</i>	<i>Prefix</i>	<i>Symbol</i>
1×10^6	1,000,000	mega-	M
1×10^3	1,000	kilo-	k
1×10^2	100	hecto-	h
1×10	10	deka-	da
1×10^{-1}	0.1	deci-	d
1×10^{-2}	0.01	centi-	c
1×10^{-3}	0.001	milli-	m
1×10^{-6}	0.000001	micro-	μ
1×10^{-9}	0.000000001	nano-	n
1×10^{-12}	0.000000000001	pico-	p
1×10^{-15}	0.000000000000001	femto-	f
1×10^{-18}	0.000000000000000001	atto-	a

E.2 RADIOLOGICAL IMPACTS ON HUMAN HEALTH

Radiation exposure and its consequences are topics of interest to the general public. For this reason, this EIS places much emphasis on the consequences of exposure to radiation, provides the reader with background information on the nature of radiation, and explains the basic concepts used in the evaluation of radiation health effects.

E.2.1 Nature of Radiation and Its Effects on Humans

What Is Radiation?

Radiation is energy transferred in the form of particles or waves. Globally, human beings are exposed constantly to radiation from the solar system and from the earth's rocks and soil. This radiation contributes to the natural background radiation that always surrounds us. Manmade sources of radiation also exist, including medical and dental x-rays, household smoke detectors, and materials released from nuclear and coal-fired power plants.

All matter in the universe is composed of atoms. Radiation comes from the activity of tiny particles within an atom. An atom consists of a positively charged nucleus (central part of an atom) with a number of negatively charged electron particles in various orbits around the nucleus. There are two types of particles in the nucleus: neutrons that are electrically neutral and protons that are positively charged. Atoms of different types are known as elements. There are more than 100 natural and manmade elements. An element has equal numbers of electrons and protons. When atoms of an element differ in their number of neutrons, they are called isotopes of that element. All elements have three or more isotopes, some or all of which could be unstable (i.e., decay with time).

Unstable isotopes undergo spontaneous change, known as radioactive disintegration or radioactive decay. The process of continuously undergoing spontaneous disintegration is called radioactivity. The radioactivity of a material decreases with time. The time it takes a material to lose half of its original radioactivity is its half-life. An isotope's half-life is a measure of its decay rate. For example, an isotope with a half-life of eight days will lose one-half of its radioactivity in that amount of time. In eight more days, one-half of the remaining radioactivity will be lost, and so on. Each radioactive element has a characteristic half-life. The half-lives of various radioactive elements may vary from millionths of a second to millions of years.

As unstable isotopes change into more stable forms, they emit electrically charged particles. These particles may be either an alpha particle (a helium nucleus) or a beta particle (an electron), with various levels of kinetic energy. Sometimes these particles are emitted in conjunction with gamma rays. The alpha and beta particles are frequently referred to as ionizing radiation. Ionizing radiation refers to the fact that the charged particle energy force can ionize, or electrically charge, an atom by stripping off one of its electrons. Gamma rays, even though they do not carry an electric charge as they pass through an element, can ionize its atoms by ejecting electrons. Thus, they cause ionization indirectly. Ionizing radiation can cause a change in the chemical composition of many things, including living tissue (organs), which can affect the way they function.

When a radioactive isotope of an element emits a particle, it changes to an entirely different element, one that may or may not be radioactive. Eventually, a stable element is formed. This transformation, which may take several steps, is known as a decay chain. For example, radium, which is a member of the radioactive decay chain of uranium, has a half-life of 1,622 years. It emits an alpha particle and becomes radon, a radioactive gas with a half-life of only 3.8 days. Radon decays first to polonium, then through a series of further decay steps to bismuth, and ultimately to lead, which is a stable element. Meanwhile, the decay products will build up and will eventually die away as time progresses.

The characteristics of various forms of ionizing radiation are briefly described below and in the box at right (see Glossary for further definition):

Alpha (α)

Alpha particles are the heaviest type of ionizing radiation. They can travel only a couple of centimeters in air. Alpha particles lose their energy almost as soon as they collide with anything. They can be stopped easily by a sheet of paper or by the skin's surface.

Beta (β)

Beta particles are much (7,330 times) lighter than alpha particles. They can travel a longer distance than alpha particles in the air. A high energy beta particle can travel a few meters in the air. Beta particles can pass through a sheet of paper, but may be stopped by a thin sheet of aluminum foil or glass.

Gamma (γ)

Gamma rays (and x-rays), unlike alpha or beta particles, are waves of pure energy. Gamma rays travel at the speed of light. Gamma radiation is very penetrating and requires a thick wall of concrete, lead, or steel to stop it.

Neutrons (n)

Neutrons are particles that contribute to radiation exposure both directly and indirectly. The most prolific source of neutrons is a nuclear reactor. Indirect radiation exposure occurs when gamma rays and alpha particles are emitted following neutron capture in matter. A neutron has about one quarter the weight of an alpha particle. It will travel in the air until it is absorbed in another element.

Units of Radiation Measure

During the early days of radiological experience, there was no precise unit of radiation measure. Therefore, a variety of units were used to measure radiation. These units were used to determine the amount, type, and intensity of radiation. Just as heat can be measured in terms of its intensity or effects using units of calories or degrees, amounts of radiation or its effects can be measured in units of curies, radiation absorbed dose (rad), or dose equivalent (rem). The following summarizes those units (see also the definition in the Glossary).

Curie

The curie, named after the French scientists Marie and Pierre Curie, describes the “intensity” of a sample of radioactive material. The rate of decay of 1 gram of radium is the basis of this unit of measure. It is equal to 3.7×10^{10} disintegrations (decays) per second.

Radiation Type	Typical Travel Distance in Air	Barrier
α	Couple of centimeters	Sheet of paper or skin's surface
β	Few meters	Thin sheet of aluminum foil or glass
γ	Very Large ^a	Thick wall of concrete, lead, or steel
n	Very Large	Water, paraffin, graphite

^a Would be infinite in a vacuum

Rad

The rad is the unit of measurement for the physical absorption of radiation. The total energy absorbed per unit quantity of tissue is referred to as absorbed dose (or simply dose). As sunlight heats pavement by giving up an amount of energy to it, radiation similarly gives up rads of energy to objects in its path. One rad is equal to the amount of radiation that leads to the deposition of 0.01 joule of energy per kilogram of absorbing material.

**Radiation Units
and Conversions to
International System of Units**

1 curie = 3.7×10^{10} becquerel
1 rad = 0.01 gray
1 rem = 0.01 sievert
1 gray = 1 joule/kilogram
1 becquerel = 1 disintegration per second

Rem

A rem is a measurement of the dose equivalent from radiation based on its biological effects. The rem is used in measuring the effects of radiation on the body as degrees centigrade are used in measuring the effects of sunlight heating pavement. Thus, 1 rem of one type of radiation is presumed to have the same biological effects as 1 rem of any other kind of radiation. This allows comparison of the biological effects of radionuclides that emit different types of radiation.

The units of radiation measure in the International Systems of Units are: becquerel (a measure of source intensity [activity]), gray (a measure of absorbed dose), and sievert (a measure of dose equivalent).

An individual may be exposed to ionizing radiation externally (from a radioactive source outside the body) or internally (from ingesting or inhaling radioactive material). The external dose is different from the internal dose because an external dose is delivered only during the actual time of exposure to the external radiation source, but an internal dose continues to be delivered as long as the radioactive source is in the body. The dose from internal exposure is calculated over 50 years following the initial exposure. Both radioactive decay and elimination of the radionuclide by ordinary metabolic processes decrease the dose rate with the passage of time.

Sources of Radiation

The average American receives a total of approximately 364 millirem per year from all sources of radiation, both natural and manmade, of which approximately 300 millirem per year are from natural sources (NCRP 1987). The sources of radiation can be divided into 6 different categories: (1) cosmic radiation, (2) terrestrial radiation, (3) internal radiation, (4) consumer products, (5) medical diagnosis and therapy, and (6) other sources (NCRP 1987). These categories are discussed in the following paragraphs.

Cosmic Radiation

Cosmic radiation is ionizing radiation resulting from energetic charged particles from space continuously hitting the earth's atmosphere. These particles and the secondary particles and photons they create comprise cosmic radiation. Because the atmosphere provides some shielding against cosmic radiation, the intensity of this radiation increases with the altitude above sea level. The average dose to people in the United States from this source is approximately 27 millirem per year.

External Terrestrial Radiation

External terrestrial radiation is the radiation emitted from the radioactive materials in the Earth's rocks and soils. The average dose from external terrestrial radiation is approximately 28 millirem per year.

Internal Radiation

Internal radiation results from the human body metabolizing natural radioactive material that has entered the body by inhalation or ingestion. Natural radionuclides in the body include isotopes of uranium, thorium, radium, radon, polonium, bismuth, potassium, rubidium, and carbon. The major contributor to the annual dose equivalent for internal radioactivity are the short-lived decay products of radon, which contribute approximately 200 millirem per year. The average dose from other internal radionuclides is approximately 39 millirem per year.

Consumer Products

Consumer products also contain sources of ionizing radiation. In some products, such as smoke detectors and airport x-ray machines, the radiation source is essential to the products' operation. In other products, such as televisions and tobacco, the radiation occurs as the product's function. The average dose from consumer products is approximately 10 millirem per year.

Medical Diagnosis and Therapy

Radiation is an important diagnostic medical tool and cancer treatment. Diagnostic x-rays result in an average exposure of 39 millirem per year. Nuclear medical procedures result in an average exposure of 14 millirem per year.

Other Sources

There are a few additional sources of radiation that contribute minor doses to individuals in the United States. The dose from nuclear fuel cycle facilities (e.g., uranium mines, mills, and fuel processing plants), nuclear power plants, and transportation routes has been estimated to be less than 1 millirem per year. Radioactive fallout from atmospheric atomic bomb tests, emissions of radioactive material from nuclear facilities, emissions from certain mineral extraction facilities, and transportation of radioactive materials contribute less than 1 millirem per year to the average dose to an individual. Air travel contributes approximately 1 millirem per year to the average dose.

Exposure Pathways

As stated earlier, an individual may be exposed to ionizing radiation both externally and internally. The different ways that could result in radiation exposure to an individual are called exposure pathways. Each type of exposure is discussed separately in the following paragraphs.

External Exposure

External exposure can result from several different pathways, all having in common the fact that the radiation causing the exposure is external to the body. These pathways include exposure to a cloud of radiation passing over the receptor (e.g., an individual member of the public) standing on ground that is contaminated with radioactivity and swimming or boating in contaminated water. If the receptor departs from the source of radiation exposure, the dose rate will be reduced. It is assumed that external exposure occurs uniformly during the year. The appropriate measure of dose is called the effective dose equivalent.

Internal Exposure

Internal exposure results from a radiation source entering the human body through either inhalation of contaminated air or ingestion of contaminated food and water. In contrast to external exposure, once a

radiation source enters the body, it remains there for a period of time that varies depending on decay and biological half-life. The absorbed dose to each organ of the body is calculated for a period 50 years following the intake. The dose equivalent of this absorbed dose is called the committed dose equivalent. Various organs have different susceptibilities to harm from radiation. The quantity that takes these different susceptibilities into account is called the committed effective dose equivalent, and it provides a broad indicator of the risk to the health of an individual from radiation. The committed effective dose equivalent is a weighted sum of the committed dose equivalent in each major organ or tissue. The concept of committed effective dose equivalent applies only to internal pathways.

Radiation Protection Guides

Various organizations have issued radiation protection guides. The responsibilities of the main radiation safety organizations, particularly those that affect policies in the United States, are summarized.

International Commission on Radiological Protection

This commission has the responsibility for providing guidance in matters of radiation safety. The operating policy of this organization is to prepare recommendations to deal with basic principles of radiation protection and to leave to the various national protection committees the responsibility of introducing the detailed technical regulations, recommendations, or codes of practice best suited to the needs of their countries.

National Council on Radiation Protection and Measurements

In the United States, this council is the national organization that has the responsibility to adapt and provide detailed technical guidelines for implementing the International Commission on Radiological Protection recommendations. The organization consists of technical experts who are specialists in radiation protection and scientists who are experts in disciplines that form the basis for radiation protection.

National Research Council/National Academy of Sciences

The National Research Council is an organization within the National Academy of Sciences that associates the broad community of science and technology with the Academy's purposes of furthering knowledge and advising the Federal Government.

Limits of Radiation Exposure

Limits of exposure to members of the public and radiation workers are based on International Commission on Radiological Protection recommendations. Each regulatory organization adopts the International Commission on Radiological Protection's recommendations and sets specific annual exposure limits (usually less than those specified by the commission). The U.S. Department of Energy (DOE) has established a set of limits for radiation workers in 10 CFR 835. **Table E-1** provides the various exposure limits set by the DOE and the U.S. Environmental Protection Agency (EPA) for radiation workers and members of the public.

E.2.2 Health Effects

Radiation exposure and its consequences are topics of interest to the general public. To provide the background for discussions of impacts, this section explains the basic concepts used in the evaluation of radiation effects.

Table E–1 Exposure Limits for Members of the Public and Radiation Workers

<i>Guidance Criteria (Organization)</i>	<i>Public Exposure Limits at the Site Boundary</i>	<i>Worker Exposure Limits</i>
40 CFR 190 (EPA)	25 millirem per year (all pathways)	—
10 CFR 835 (DOE)	—	5,000 millirem per year
DOE Order N441.1 (DOE)	—	2,000 millirem per year
DOE Order 5400.5 (DOE)	10 millirem per year (all air pathways) 4 millirem per year (drinking water pathway) 100 millirem per year (all pathways)	—
40 CFR 61 (EPA)	10 millirem per year (all air pathways)	—

Radiation can cause a variety of damaging health effects in people. The most significant effects are induced cancer fatalities. These effects are referred to as “latent” cancer fatalities because the cancer may take many years to develop. In the discussions that follow, all fatal cancers are considered latent; therefore, the term “latent” is not used.

The National Research Council’s Committee on the Biological Effects of Ionizing Radiation (BEIR) has prepared a series of reports to advise the U.S. Government on the health consequences of radiation exposures. *Health Effects of Exposure to Low Levels of Ionizing Radiation*, BEIR V (National Research Council 1990), provides the most current estimates for excess mortality from leukemia and cancers, other than leukemia, that are expected to result from exposure to ionizing radiation. BEIR V provides estimates that are consistently higher than those in its predecessor, BEIR III. This increase is attributed to several factors, including the use of a linear dose response model for cancers other than leukemia, revised dosimetry for the Japanese atomic bomb survivors, and additional follow-up studies of the atomic bomb survivors and others associated with them. BEIR III employs constant, relative, and absolute risk models, with separate coefficients for each of several sex and age-at-exposure groups. BEIR V develops models in which the excess relative risk is expressed as a function of age at exposure, time after exposure, and sex for each of several cancer categories. The BEIR III models were based on the assumption that absolute risks are comparable between the atomic bomb survivors and the U.S. population. BEIR V models were based on the assumption that the relative risks are comparable. For a disease such as lung cancer, where baseline risks in the United States are much larger than those in Japan, the BEIR V approach leads to larger risk estimates than the BEIR III approach.

The models and risk coefficients in BEIR V were derived through analyses of relevant epidemiologic data that included the Japanese atomic bomb survivors, ankylosis spondylitis patients, Canadian and Massachusetts fluoroscopy (breast cancer) patients, New York postpartum mastitis (breast cancer) patients, Israeli tinea capitis (thyroid cancer) patients, and Rochester thymus (thyroid cancer) patients. Models for leukemia, respiratory cancer, digestive cancer, and other cancers used only the atomic bomb survivor data, although results of analyses of the ankylosis spondylitis patients were considered. Atomic bomb survivor analyses were based on revised dosimetry, with an assumed relative biological effectiveness of 20 for neutrons, and were restricted to doses less than 400 rads. Estimates of risks of fatal cancers, other than leukemia, were obtained by totaling the estimates for breast cancer, respiratory cancer, digestive cancer, and other cancers.

The National Council on Radiation Protection and Measurements (NCRP 1993), based on the radiation risk estimates provided in BEIR V and the International Commission on Radiological Protection Publication 60

recommendations (ICRP 1991), has estimated the total detriment resulting from low dose¹ or low dose rate exposure to ionizing radiation to be 0.00073 per rem for the general population, and 0.00056 per rem for the working population. The total detriment includes fatal and nonfatal cancer and severe hereditary (genetic) effects. The major contribution to the total detriment is from fatal cancer and is estimated to be 0.0004 and 0.0005 per rem for the radiation workers and the general population, respectively. **Table E-2** provides the breakdown of the risk factors for both the workers and the general population. Nonfatal cancers and genetic effects are less probable consequences of radiation exposure. To simplify the presentation of the impacts, estimated effects of radiation are calculated only in terms of latent cancer fatalities.

Table E-2 Nominal Health Effects Coefficients (Risk Factors) from Ionizing Radiation

<i>Exposed Population</i>	<i>Fatal Cancer</i> ^{a,c}	<i>Nonfatal Cancer</i> ^b	<i>Genetic Disorders</i> ^b	<i>Total</i>
Working Population	0.0004	0.00008	0.00008	0.00056
General Population	0.0005	0.0001	0.00013	0.00073

^a For fatal cancer, the health effect coefficient is the same as the probability coefficient.

^b In determining a means of assessing health effects from radiation exposure, the International Commission on Radiological Protection has developed a weighting method for nonfatal cancers and genetic effects. Genetic effects only can be applied to a population, not individuals.

^c For high individual exposures (greater than or equal to 20 rem), the health factors are multiplied by a factor of 2.

Source: NCRP 1993.

The numerical estimates of fatal cancers presented in this EIS were obtained using a linear extrapolation from the nominal risk estimated for lifetime total cancer mortality, which is 0.1 Gray (10 rad). Other methods of extrapolation to the low-dose region could yield higher or lower numerical estimates of fatal cancers. Studies of human populations exposed to low doses are inadequate to demonstrate the actual level of risk. There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiologic observation, and the possibility of no risk cannot be excluded (CIRRPC 1992).

Health Effect Risk Factors Used in This EIS

Health impacts from radiation exposure, whether from sources external or internal to the body, generally are identified as “somatic” (i.e., affecting the exposed individual) or “genetic” (i.e., affecting descendants of the exposed individual). Radiation is more likely to produce somatic effects than genetic effects. The somatic risks of most importance are induced cancers. Except for leukemia, which can have an induction period (time between exposure to carcinogen and cancer diagnosis) of as little as 2 to 7 years, most cancers have an induction period of more than 20 years.

For a uniform irradiation of the body, the incidence of cancer varies among organs and tissues; the thyroid and skin demonstrate a greater sensitivity than other organs. Such cancers, however, also produce relatively low mortality rates because they are relatively amenable to medical treatment. Because fatal cancer is the most probable serious effect of environmental and occupational radiation exposures, estimates of cancer fatalities rather than cancer incidence are presented in this EIS. The numbers of fatal cancers can be used to compare the risks among the various alternatives.

¹The low dose is defined as the dose level where DNA repair can occur in a few hours after irradiation-induced damage. Currently, a dose level of about 0.2 Grays (20 rad), or a dose rate of 0.1 milligrays (0.01 rad) per minute is considered to allow the DNA to repair itself in a short period (EPA 1994).

Based on the preceding discussion and the values presented in Table E-2, the fatal cancers to the general public during normal operations and for accidents in which individual doses are less than 20 rem are calculated using a health risk factor of 0.0005 per person-rem. For workers, a risk factor of 0.0004 excess fatal cancer per person-rem is used. This lower value reflects the absence of children (who are more radiosensitive than adults) in the workforce. Nonfatal cancer and genetic disorders among the public are 20 and 26 percent, respectively, of the fatal cancer risk factor. For workers, the health risk estimators are both 20 percent of the fatal cancer risk factor. These factors are not used in this EIS.

The risk factors are used to calculate the statistical expectation of the effects of exposing a population to radiation. For example, in a population of 100,000 people exposed only to natural background radiation (300 millirem per year), it is expected that about 15 latent cancer fatalities per year of exposure would result from this radiation ($100,000 \text{ persons} \times 0.3 \text{ rem per year} \times 0.0005 \text{ latent cancer fatalities per person-rem} = 15 \text{ latent cancer fatalities per year}$).

Calculations of the number of excess fatal cancers associated with radiation exposure do not always yield whole numbers; calculations may yield numbers less than 1.0, especially in environmental impact applications. For example, if a population of 100,000 were exposed to a total dose of only 0.001 rem per person, the collective dose would be 100 person-rem, and the corresponding estimated number of latent cancer fatalities would be 0.05 ($100,000 \text{ persons} \times 0.001 \text{ rem} \times 0.0005 \text{ latent cancer fatalities per person-rem} = 0.05 \text{ latent cancer fatalities}$). The 0.05 latent cancer fatalities is the *expected* number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people. In most groups, no person (0 people) would incur a latent fatal cancer from the 0.001 rem dose each member would have received. In a small fraction of the groups, 1 latent cancer fatality would result; in exceptionally few groups, 2 or more latent cancer fatalities would occur. The *average* expected number of deaths over all the groups would be 0.05 latent cancer fatalities (just as the average of 0, 0, 0, and 1 is 1/4, or 0.25). The most likely outcome is 0 latent cancer fatalities.

These same concepts apply to estimating the effects of radiation exposure on a single individual. Consider the effects, for example, of exposure to background radiation over a lifetime. The “number of latent cancer fatalities” corresponding to a single individual’s exposure over a (presumed) 72-year lifetime to 0.3 rem per year is 0.011 latent cancer fatalities ($1 \text{ person} \times 0.3 \text{ rem per year} \times 72 \text{ year} \times 0.0005 \text{ latent cancer fatalities per person-rem} = 0.011 \text{ latent cancer fatalities}$).

Again, this is a statistical estimate. That is, the estimated effect of background radiation exposure on the exposed individual would produce a 1.1 percent chance that the individual might incur risk of a latent cancer fatality caused by the exposure over his full lifetime. Presented another way, this method estimates that approximately 1.1 percent of the population might die of cancers induced by background radiation.

E.3 METHODOLOGY FOR ESTIMATING RADIOLOGICAL IMPACTS

The radiological impacts from normal operation of the facilities were calculated using Version 1.485 of the GENII computer code (PNL 1988). Site-specific input data were used, including location, meteorology, population, and source terms. Section E.3.1 briefly describes GENII and outlines the approach used for normal operations.

E.3.1 GENII Computer Code

The GENII computer model, developed by Pacific Northwest National Laboratory, is an integrated system of various computer modules that analyze environmental contamination resulting from acute or chronic releases to, or initial contamination in, air, water, or soil. The model calculates radiation doses to individuals and populations. The GENII computer model is well documented for assumptions, technical approach, method,

and quality assurance issues (PNL 1988). The GENII computer model has gone through extensive quality assurance and quality control steps, including comparing results from model computations with those from hand calculations and performing internal and external peer reviews. Recommendations given in these reports were incorporated into the final GENII computer model, as appropriate.

For this EIS, only the ENVIN, ENV, and DOSE computer modules were used. The codes are connected through data transfer files. The output of one code is stored in a file that can be used by the next code in the system. The functions of the three GENII computer modules used in this EIS are discussed below.

ENVIN

The ENVIN module of the GENII code controls the reading of input files and organizes the input for optimal use in the environmental transport and exposure module, ENV. The ENVIN code interprets the basic input, reads the basic GENII data libraries and other optional input files, and organizes the input into sequential segments based on radionuclide decay chains.

A standardized file that contains scenario, control, and inventory parameters is used as input to ENVIN. Radionuclide inventories can be entered as functions of releases to air or water, concentrations in basic environmental media (air, soil, or water), or concentrations in foods. If certain atmospheric dispersion options have been selected, this module can generate tables of atmospheric dispersion parameters that will be used in later calculations. If the finite plume air submersion option is requested in addition to the atmospheric dispersion calculations, preliminary energy-dependent finite plume dose factors can be prepared as well. The ENVIN module prepares the data transfer files that are used as input by the ENV module; ENVIN generates the first portion of the calculation documentation—the run input parameters report.

ENV

The ENV module calculates the environmental transfer, uptake, and human exposure to radionuclides that result from the chosen scenario for the user-specified source term. The code reads the input files from ENVIN and then, for each radionuclide chain, sequentially performs the precalculations to establish the conditions at the start of the exposure scenario. Environmental concentrations of radionuclides are established at the beginning of the scenario by assuming decay of preexisting sources, considering biotic transport of existing subsurface contamination, and defining soil contamination from continuing atmospheric or irrigation depositions. For each year of postulated exposure, the code then estimates the air, surface soil, deep soil, groundwater, and surface water concentrations of each radionuclide in the chain. Human exposures and intakes of each radionuclide are calculated for: (1) pathways of external exposure from finite atmospheric plumes; (2) inhalation; (3) external exposure from contaminated soil, sediments, and water; (4) external exposure from special geometries; and (5) internal exposures from consumption of terrestrial foods, aquatic foods, drinking water, animal products, and inadvertent intake of soil. The intermediate information on annual media concentrations and intake rates are written to data transfer files. Although these may be accessed directly, they are usually used as input to the DOSE module of GENII.

DOSE

The DOSE module reads the intake and exposure rates defined by the ENV module and converts the data to radiation dose.

E.3.2 Data and General Assumptions

To perform the dose assessments for this EIS, different types of data were collected and generated. This section discusses the various data that were collected and/or generated, along with the assumptions made (WSRC 1999) for performing the dose assessments in this EIS.

Meteorological Data

The meteorological data used for all normal operational scenarios discussed in this EIS were in the form of joint frequency data files. A joint frequency data file is a table listing the fractions of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The joint frequency data files were based on measurements taken over a period of several years at different heights at each of the management facilities and/or sites.

Population Data

Population distributions were based on the *1990 Census of Population and Housing* data (DOC 1992). Projections were determined for the year 2010 (representative year for operations) for areas within 80 kilometers (50 miles) of the release locations. The site population in 2010, assumed to be representative of the population over the operational period evaluated, was used in the impact assessments. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances up to 80 kilometers (50 miles). The grid was centered at the precise location from which the radionuclides were assumed to be released.

Source Term Data

The source terms used to calculate the impacts of normal operations are provided in Section E.4.

Food Production and Consumption Data

Production and consumption rates used in GENII were those established in the U.S. Nuclear Regulatory Commission Regulatory Guide 1.109 (NRC 1977), for the maximally exposed individual and the general public (average individual), see **Tables E-3 and E-4** for details.

Exposed Population

Dose assessments were performed for both members of the general public and workers for each management facility examined in this EIS. These assessments were made to determine the incremental doses that would be associated with the processing alternatives addressed in this EIS. Incremental doses for members of the public were calculated (via GENII) for two different types of receptors:

- **Maximally Exposed Offsite Individual**—The maximally exposed individual was assumed to be a hypothetical individual located at a position on the site boundary that would yield the highest impacts during normal operations.
- **Population**—The general population living within 80 kilometers (50 miles) of the facility in the year 2010.

Table E-3 GENII Usage Parameters for Production and Consumption of Terrestrial Food

Food Type	Maximally Exposed Offsite Individual				General Population			
	Growing Time (days)	Yield (kilograms per square meter)	Holdup Time (days)	Consumption Rate (kilograms per year)	Growing Time (days)	Yield (kilograms per square meter)	Holdup Time (days)	Consumption Rate (kilograms per year)
Leafy Vegetables	90.0	1.5	1.0	30.0	90.0	1.5	14.0	15.0
Root Vegetables	90.0	4.0	5.0	220.0	90.0	4.0	14.0	140.0
Fruit	90.0	2.0	5.0	330.0	90.0	2.0	14.0	64.0
Grains/Cereals	90.0	0.8	180.0	80.0	90.0	0.8	180.0	72.0

Source: PNL 1988, NRC 1977.

Table E-4 GENII Usage Parameters for Production and Consumption of Animal Products

Food Type	Human Consumption Rate (kilograms per year)	Holdup Time (days)	Animal Stored Feed				Animal Fresh Forage			
			Diet Fraction	Growing Time (days)	Yield (kilograms per square meter)	Storage Time (days)	Diet Fraction	Growing Time (days)	Yield (kilograms per square meter)	Storage Time (days)
<i>Maximally Exposed Offsite Individual</i>										
Beef	80.0	15.0	0.25	90.0	0.80	180.0	0.75	45.0	2.00	100.0
Poultry	18.0	1.0	1.00	90.0	0.80	180.0	–	–	–	–
Milk	270.0	1.0	0.25	45.0	2.00	100.0	0.75	30.0	1.50	0.00
Eggs	30.0	1.0	1.00	90.0	0.80	180.0	–	–	–	–
<i>General Population</i>										
Beef	70.0	34.0	0.25	90.0	0.80	180.0	0.75	45.0	2.00	100.0
Poultry	8.5	34.0	1.0	90.0	0.80	180.0	–	–	–	–
Milk	230.0	3.0	0.25	45.0	2.00	100.0	0.75	30.0	1.50	0.00
Eggs	20.0	18.0	1.0	90.0	0.80	180.0	–	–	–	–

Source: PNL 1988, NRC 1977.

Basic Assumptions

To estimate radiological impacts from normal operations, the following additional assumptions and factors were considered in using GENII:

Radiological airborne gaseous and particulate emissions were assumed to be released to the atmosphere through the plant stacks. See Section E.4 for the specifics at each management facility.

- Ground surfaces were assumed to have no previous deposition of radionuclides.
- The annual external exposure time to the plume and to soil contamination was 0.7 years (16.8 hours per day) for the maximally exposed offsite individual (NRC 1977).
- The annual external exposure time to the plume and to soil contamination was 0.5 years (12 hours per day) for the population (NRC 1977).

- The inhalation exposure time to the plume was 1 year for the maximally exposed individual and general population.
- The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of an adult human.
- A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure; inhalation; and ingestion of food crops and animal products contaminated by deposition of radioactivity from the air.
- Resuspension of particulates was not considered because calculations of dust loading in the atmosphere shows that this pathway is negligible compared to the other pathways.
- Reported release heights were used for atmospheric releases and were assumed to be the effective stack heights. The resultant doses were conservative, as use of the actual stack heights negates plume rise.
- The calculated doses were 50-year committed doses from 1 year of intake.

The exposure, uptake, and usage parameters used in the GENII model for normal operations are provided in Tables E-3, E-4, and E-5.

Table E-5 GENII Exposure Parameters to Plumes and Soil Contamination (Normal Operations)

<i>Maximally Exposed Offsite Individual</i>				<i>General Population</i>			
<i>External Exposure</i>		<i>Inhalation of Plume</i>		<i>External Exposure</i>		<i>Inhalation of Plume</i>	
<i>Plume (hours)</i>	<i>Soil Contamination (hours)</i>	<i>Exposure Time (hours)</i>	<i>Breathing Rate (cubic centimeters per second)</i>	<i>Plume (hours)</i>	<i>Soil Contamination (hours)</i>	<i>Exposure Time (hours)</i>	<i>Breathing Rate (cubic centimeters per second)</i>
6,136	6,136	8,766	270	4,383	4,383	8,766	270

Source: PNL 1988, NRC 1977.

Worker doses associated with the processing alternatives were determined from historical data associated with similar operations. See Section E.4 for details.

E.3.3 Uncertainties

The sequence of analyses performed to generate the radiological impact estimates from normal operation include: (1) selection of normal operational modes, (2) estimation of source terms, (3) estimation of environmental transport and uptake of radionuclides, (4) calculation of radiation doses to exposed individuals, and (5) estimation of health effects. There are uncertainties associated with each of these steps. Uncertainties exist in the way the physical systems being analyzed are represented by the computational models and in the data required to exercise the models (due to measurement, sampling, or natural variability).

In principle, one can estimate the uncertainty associated with each source and predict the remaining uncertainty in the results of each set of calculations. Thus, one can propagate the uncertainties from one set of calculations to the next and estimate the uncertainty in the final results. However, conducting such a full-scale quantitative uncertainty analysis is neither practical nor a standard practice for a study of this type. Instead, the analysis is designed to ensure—through judicious selection of release scenarios, models, and parameters—that the results represent the potential risks. This is accomplished by making conservative assumptions in the calculations at each step. The models, parameters, and release scenarios used in the calculations are selected

in such a way that most intermediate results and, consequently, the final estimates of impacts, are greater than would be expected. As a result, even though the range of uncertainty in a quantity might be large, the value calculated for the quantity would be close to one of the extremes in the range of possible values, so the chance of the actual quantity being greater than the calculated value would be low (or the chance of the quantity being less than the calculated value if the criteria are such that the quantity has to be maximized). The goal of the radiological assessment for normal operation in this study has been to produce results that are conservative.

The degree of conservatism in the calculated results is closely related to the range of possible values the quantity can have. This range is determined by what can be expected to realistically occur. Thus, the only processes considered are those that are credible for the conditions under which the physical system being modeled operates. This consideration has been employed for the normal operation analyses.

Although the radionuclide composition of source terms are reasonable estimates, there are uncertainties in the radionuclide inventory and release reactions that affect estimated impacts.

E.4 RADIOLOGICAL RELEASES TO THE ENVIRONMENT AND ASSOCIATED IMPACTS

This section summarizes the estimated radiological releases to the environment as well as resulting impacts associated with the various alternatives assessed in this EIS. Impacts to workers from these alternatives are also discussed. The methodology for estimating radiological impacts, as well as associated input data and analytical assumptions are provided in Section E.3.

E.4.1 Electrometallurgical Treatment of Both Blanket and Driver Fuels (Alternative 1)

Under this alternative, releases of radioactive material would occur during normal operational processing of the sodium-bonded fuel rods in the argon cell at the Fuel Conditioning Facility. Fuel assemblies would be disassembled in the Fuel Conditioning Facility air cell, and individual fuel elements would then be transferred to the argon cell for chopping and treatment in one of the electrorefiners. The entire inventory of gaseous fission products, mainly tritium (H-3) and krypton-85 (Kr-85) is assumed to be released during processing in the Fuel Conditioning Facility. The likelihood of release of radionuclides other than the gaseous fission products is judged to be very small. No radionuclides would be released from the packaged salt and packaged metal waste material transferred from the Fuel Conditioning Facility to the Hot Fuel Examination Facility.

Estimated radioactive releases during normal operations at ANL-W were calculated using a conservative methodology. First, assumptions were made to estimate a maximum annual throughput of material to be processed at the Fuel Conditioning Facility. There would be two electrorefiners in the Fuel Conditioning Facility argon cell; blanket material would be treated in one of the two electrorefiners and driver material would be treated in the other. Both driver and blanket material could be processed each year. Based on an annual operational processing limit of 5,000 kilograms (11,023 pounds) of total heavy metal fuel material consisting of no more than 600 kilograms (1,320 pounds) of heavy metal driver material, it was assumed that driver fuels would be processed at the maximum rate until all driver fuel was processed. In addition, it was assumed that EBR-II fuel (driver and blanket) currently at ANL-W would be processed first. Using these assumptions, annual mass processing throughputs were developed for the purposes of estimating releases of radioactive material during normal operations, and are presented in **Table E-6**.

Table E-6 Annual Processing Assumptions for Estimation of Air Releases of Radionuclides During Normal Operations for Alternative 1 at ANL-W

Year of processing	Driver Fuel (kilograms per year)		Blanket Fuel (kilograms per year)		Total Fuel (kilograms per year)		
	EBR-II ^a	Fast Flux Test Facility ^b	EBR-II ^c	Fermi-1	Driver	Blanket	Driver + Blanket
1	6.0 E+02	0	4.4 E+03	0	6.0 E+02	4.4 E+03	5.0 E+03
2	6.0 E+02	0	4.4 E+03	0	6.0 E+02	4.4 E+03	5.0 E+03
3	6.0 E+02	0	4.4 E+03	0	6.0 E+02	4.4 E+03	5.0 E+03
4	6.0 E+02	0	4.4 E+03	0	6.0 E+02	4.4 E+03	5.0 E+03
5	6.0 E+02	0	4.4 E+03	0	6.0 E+02	4.4 E+03	5.0 E+03
6	1.0 E+02	2.9 E+02	4.0 E+02	4.2 E+03	3.9 E+02	4.6 E+03	5.0 E+03
7	0	0	0	5.0 E+03	0	5.0 E+03	5.0 E+03
8	0	0	0	5.0 E+03	0	5.0 E+03	5.0 E+03
9	0	0	0	5.0 E+03	0	5.0 E+03	5.0 E+03
10	0	0	0	5.0 E+03	0	5.0 E+03	5.0 E+03
11	0	0	0	5.0 E+03	0	5.0 E+03	5.0 E+03
12	0	0	0	5.0 E+03	0	5.0 E+03	5.0 E+03
Totals (kg)	3.10 E+03	2.90 E+02	2.24 E+04	3.42 E+04	3.39 E+03	5.66 E+04	6.00 E+04

E is exponential notation equivalent to scientific notation ($1.0\text{E}-05 = 1.0 \times 10^{-5}$).

^a EBR-II driver consists of 1,100 kilograms of EBR-II driver fuel at ANL-W and 2,000 kilograms at INTEC.

^b The Fast Flux Test Facility driver consists of 250 kilograms of sodium-bonded Fast Flux Test Facility driver fuel at Hanford, plus 40 kilograms of miscellaneous fuel at INTEC, Sandia National Laboratory, and Oak Ridge Reservation.

^c EBR-II blanket consists of EBR-II blanket fuel at ANL-W.

Radioactive releases from the Fuel Conditioning Facility argon cell during fuel treatment were estimated next. Radioactivity associated with the fuel to be processed was determined using the fuel radioactivity inventory values discussed in Appendix D. Estimated releases were based on a methodology developed in support of ANL-W's State of Idaho and National Emission Standards for Hazardous Air Pollutants air permitting activities, and agreed upon by the State of Idaho's Department of Environmental Quality (Bauer 1992). From this methodology, equilibrium concentrations in the argon cell (curies per cubic meter per curie processed) were calculated and applied to the inventory associated with the assumed annual throughputs shown in Table E-6. Annual radioactive releases to the atmosphere were calculated as the product of the radionuclide equilibrium concentrations in the argon cell, the annual argon cell atmosphere exhaust (74,400 cubic meters per year), and a conservative adjustment (0.00001) to account for the combined filtration of the two banks of high-efficiency particulate air filters that the cell exhaust must pass through before entering the environment. This filtration adjustment was not applied to tritium or krypton-85, as 100 percent of these radionuclides were assumed to be released.

The Fuel Conditioning Facility stack was modeled with an effective stack height of 60.96 meters. This is the actual stack height, and for conservatism, no plume rise was included in the atmospheric dispersion modeling.

The dose resulting from the release of tritium (H-3) depends heavily on its chemical form. The inhalation dose from oxidized tritium (HTO or T₂O) is 25,000 times higher than for tritium in elemental form (HT or T₂) (ICRP 1982). The dose conversion factors used in the GENII code assume that tritium released to the environment is in the oxidized form and are therefore very conservative for releases that involve elemental tritium. Because of the argon atmosphere in the Fuel Conditioning Facility argon cell, releases of tritium to the cell atmosphere would not become oxidized, and stack releases of tritium would most likely be in the elemental form. The oxidation of elemental tritium to HTO or T₂O has been shown to occur slowly in the environment, and the long-term dose from elemental tritium releases is conservatively estimated to be 1 percent

of that for the oxidized form for this EIS (DOE 1997). Therefore, the inventory of tritium for each year of electrometallurgical treatment processing at the Fuel Conditioning Facility was multiplied by a factor of 0.01 to convert them to an equivalent release of tritium oxide for use as input to the GENII code.

Radiological Gaseous Emissions:

The estimated annual and total atmospheric releases are tabulated in **Table E-7**. This table lists only those radionuclides that resulted from a screening procedure to indicate potential significant dose contribution. The source term listed in Table E-7 for each of the first 5 years of processing (years 1 - 5) represents the source term that results in the highest annual offsite dose, and is therefore used for the maximum annual dose calculations. The project lifetime total values in Table E-7 represent the total estimated releases over the 12 years of processing at ANL-W.

Table E-7 Annual and Total Radioactive Releases During Normal Operations for Alternative 1

Isotope ^a	Annual Releases (curies per year)			Project Lifetime Total (curies)
	Years 1-5	Year 6	Years 7-12	
H-3	7.7E+02	6.8E+02	3.8E-01	4.5E+03
C-14	5.2E-07	8.7E-08	2.3E-16	2.7E-06
Fe-55	1.4E-08	2.4E-08	5.8E-13	9.6E-08
Ni-63	6.5E-10	1.7E-10	1.0E-12	3.4E-09
Kr-85	1.2E+04	8.8E+03	3.3E+00	6.7E+04
Sr-90	7.0E-08	5.2E-08	4.7E-11	4.0E-07
Y-90	7.0E-08	5.2E-08	4.7E-11	4.0E-07
Ru-106	3.2E-08	2.9E-08	7.6E-17	1.9E-07
Rh-106	3.2E-08	2.9E-08	7.6E-17	1.9E-07
Cd-113m	6.7E-10	5.2E-10	3.1E-13	3.9E-09
Sb-125	4.1E-08	3.5E-08	3.2E-13	2.4E-07
Te-125m	4.5E-10	3.9E-10	4.0E-15	2.6E-09
I-129	1.4E-12	9.7E-13	1.8E-15	8.2E-12
Cs-134	3.2E-08	4.0E-08	9.5E-16	2.0E-07
Cs-137	4.0E-06	2.9E-06	3.5E-09	2.3E-05
Ba-137m	3.8E-06	2.8E-06	3.3E-09	2.2E-05
Ce-144	1.2E-09	1.8E-09	1.9E-20	7.7E-09
Pr-144	1.2E-09	1.8E-09	1.9E-20	7.7E-09
Pm-147	2.9E-03	2.6E-03	2.3E-08	1.7E-02
Sm-151	2.1E-09	1.4E-09	3.7E-12	1.2E-08
Eu-154	2.1E-10	2.0E-10	2.2E-15	1.3E-09
Eu-155	1.4E-09	1.1E-09	1.9E-13	8.3E-09
Th-228	1.6E-14	1.3E-14	3.2E-19	9.1E-14
U-234	1.2E-11	7.8E-12	7.8E-17	6.7E-11
U-235	3.9E-13	2.6E-13	1.8E-14	2.3E-12
U-236	3.7E-13	2.6E-13	2.7E-16	2.1E-12
U-238	7.4E-13	7.7E-13	8.1E-13	9.4E-12
Np-237	3.2E-13	2.7E-13	2.1E-15	1.9E-12
Pu-238	2.9E-10	2.2E-10	3.4E-14	1.6E-09
Pu-239	7.1E-09	1.2E-09	1.4E-10	3.7E-08

Isotope ^a	Annual Releases (curies per year)			Project Lifetime Total (curies)
	Years 1-5	Year 6	Years 7-12	
Pu-240	4.7E-10	1.2E-10	1.1E-13	2.5E-09
Pu-241	1.9E-09	1.1E-09	3.6E-15	1.1E-08
Am-241	6.2E-12	1.8E-12	1.5E-17	3.3E-11
Am-242m	6.4E-14	9.3E-15	2.6E-24	3.3E-13
Totals	1.2E+04	9.5E+03	3.7E+00	7.1E+04

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0 × 10⁻⁵).

^a The listed isotopes are present within the argon cell at the Fuel Conditioning Facility. Due to lack (scarcity) of oxygen in the argon cell, the tritium released to the cell would be in molecular (elemental) form (i.e., T₂, or HT).

Population Impacts:

The estimated annual radiological impacts due to the source term for the maximally exposed offsite individual and the 80-kilometer (50-mile) population surrounding ANL-W are tabulated in **Table E-8**. Calculated impacts are shown for each year of processing as well as for each of the fuel types to be processed. Impacts resulting from releases during processing EBR-II driver and EBR-II blanket during each of the first 5 years are listed (years 1 - 5), for processing some of all 4 fuel types during the 6th year (year 6), and for processing Fermi-1 blanket fuel during each of the final 6 years (years 7 - 12). The impacts to the maximally exposed offsite individual and the surrounding population result primarily from estimated releases of tritium and krypton-85. Together, these two radionuclides account for greater than 99.9 percent of the estimated impacts.

Table E-8 Annual Radiological Impacts to the Public From Operational Activities Associated With Alternative 1 at ANL-W

Year(s) of Processing	Fuel Type	Offsite Population		Maximally Exposed Offsite Individual	
		Collective Dose (person-rem per year)	Latent Cancer Fatalities (number of cancers)	Annual Dose (millirem per year)	Latent Cancer Fatality Risk
1 - 5	EBR-II Driver	2.8 E-03	1.4 E-06	3.3 E-04	1.6 E-10
	FFTF Driver	0	0	0	0
	EBR-II Blanket	8.4 E-05	4.2 E-08	1.0 E-05	5.0 E-12
	Fermi Blanket	0	0	0	0
	All Fuels, Years 1 - 5	2.9E-03	1.4 E-06	3.4E-04	1.7 E-10
6	EBR-II Driver	4.6 E-04	2.3 E-07	5.4 E-05	2.7 E-11
	FFTF Driver	1.8 E-03	9.2 E-07	2.2 E-04	1.1 E-10
	EBR-II Blanket	7.6 E-06	3.8 E-09	9.2 E-07	4.6 E-13
	Fermi Blanket	9.1 E-07	4.6 E-10	1.1 E-07	5.5 E-14
	All Fuels, Year 6	2.3 E-03	1.2 E-06	2.8 E-04	1.4 E-10
7 - 12	EBR-II Driver	0	0	0	0
	FFTF Driver	0	0	0	0
	EBR-II Blanket	0	0	0	0
	Fermi Blanket	1.1E-06	5.4 E-10	1.3 E-07	6.5 E-14
	All Fuels, Years 7-12	1.1E-06	5.4 E-10	1.3 E-07	6.5 E-14

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0 × 10⁻⁵).

Total cumulative radiological impacts over the projected 13 years of operations under this alternative are tabulated in **Table E-9**. This table shows the sum of the calculated impacts to the maximally exposed offsite individual and the surrounding population for each of the 12 years of processing.

Table E-9 Cumulative Maximum Radiological Impacts From Operational Activities Associated With Alternative 1 at ANL-W

	<i>80-km Offsite Population</i>		<i>Maximally Exposed Offsite Individual</i>	
	<i>Collective Dose (person-rem)</i>	<i>Latent Cancer Fatalities (number of cancers)</i>	<i>Dose (millirem)</i>	<i>Latent Cancer Fatality Risk</i>
Total Project Impacts ^a	1.68-02	8.4E-06	2.0 E-03	9.8 E-10

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0 × 10⁻⁵).

^a Total impacts are estimated for the 12-year duration of fuel treatment.

Worker Impacts:

Workers involved with electrometallurgical treatment activities at ANL-W could receive radiation doses during handling activities, such as receiving and unloading fuel casks, and transferring in-process waste material from the Fuel Conditioning Facility to the Hot Fuel Examination Facility. Doses received during in-cell activities would likely be very small. A maximally exposed worker dose estimate for this EIS is based on the regulatory limit of 5,000 millirem per year for radiation workers at DOE sites. If an individual worker received this dose each year of the 13 years of the electrometallurgical treatment project, the total maximally exposed worker dose would be 65,000 millirem, with an associated risk of 0.026 latent cancer fatalities.

However, actual worker doses are likely to be much lower than this maximum estimate. The ANL-W radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. In addition, ANL-W has established an administrative goal of 1,500 millirem per year to any individual. The general design goals at the Fuel Conditioning Facility, for example, were to maintain radiation fields below 0.5 millirem per hour at all workstations. This means that for an individual working at the Fuel Conditioning Facility for a full-time occupational work year of 2,000 hours, the annual dose would be 1,000 millirem.

Worker population doses were estimated by examining the type and duration of various operations performed by workers involved with the electrometallurgical treatment project. Doses can be estimated based on previous doses from similar activities at ANL-W. Based on information from ANL-W, the total worker population dose estimate is 22 person-rem per year, averaging out to an individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates are extended out over the 13 years of operational activities (12 years of fuel treatment and a final year of high level radioactive waste conversion activities), the cumulative worker population dose is 286 person-rem and the associated risk is 0.11 latent cancer fatalities. The estimated population impacts to the workers associated with this alternative are summarized in **Table E-10**.

Table E-10 Annual and Total Impacts to Electrometallurgical Treatment Workers at ANL-W

	<i>Worker Population</i>	
	<i>Collective Dose (person-rem)</i>	<i>Latent Cancer Fatalities</i>
Annual Impacts	2.2 E+01	8.8 E-03
Total Project Impacts ^a	2.9 E+02	1.1 E-01

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0 × 10⁻⁵).

^a Total impacts are estimated for the 13-year duration of processing.

E.4.2 Blanket Fuel Preparation and Electrometallurgical Treatment of Driver Fuel at ANL-W (Alternatives 2 Through 5)

In Alternatives 2 through 5, the blanket fuel assemblies will need to be prepared at the ANL-W facilities prior to packaging in high-integrity cans or processing in either the PUREX process at SRS or the melt and dilute process at SRS or ANL-W. When the blanket fuel is to be processed at SRS, Alternative 3 (PUREX processing) and 5 (melt and dilute processing), the blanket fuel will be declad and cleaned at ANL-W in the argon cell of the Hot Fuel Examination Facility. Processing of the blanket fuel assemblies at ANL-W (Alternative 2, placing the blanket fuel in high-integrity cans, and Alternatives 4 and 6, melt and dilute) do not require decladding of the blanket fuel. This activity would also be performed in the argon cell of the Hot Fuel Examination Facility. The preparation of the blanket fuel under these alternatives require only that the fuel be cut into segments and cleaned (see Appendix C for details). The following discussion addresses the radiological impact of normal operations at ANL-W for the preparation of the blanket fuel elements and the electrometallurgical treatment of the driver fuel elements. This analysis is applicable to Alternatives 2 through 5.

Gaseous Emissions:

Blanket fuel preparation would occur at the Hot Fuel Examination Facility. These activities would cause gaseous fission products to be released into the argon cell. As stated earlier in Section E.4.1, krypton-85 and elemental tritium are the most prevalent gaseous radionuclides that would be released to the environment. The released tritium into the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1 provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies), and are released to the atmosphere through the facility stack, along with the krypton-85 and elemental tritium. The maximum released curies of radioactive gaseous emissions occurs when preparation of the blanket fuel and chopping of the driver fuel (for electrometallurgical treatment processing) are performed simultaneously. This simultaneous operation was estimated to occur over a 6-year period starting in 2003. Based on a blanket fuel preparation throughput of 10 metric tons of heavy metal and an electrometallurgical treatment process rate of about 0.6 metric tons of heavy metal of driver fuel elements annually, about 809 curies of elemental tritium and 11,860 curies of krypton-85 would be released to the atmosphere annually, see **Table E-11**. This release rate would last about 2 years, or until all of the EBR-II blanket fuels are processed, then afterward the release rate would drop during the processing of the Fermi-1 blanket fuels (The release rate for the processing of 10 metric tons of heavy metal of Fermi-1 blanket fuel is less than 1 curie of elemental tritium and 6.6 curies of krypton-85.)

Table E-11 Maximum Annual Radiological Gaseous Emission From Activities Associated With Alternative 2 Through 5 at ANL-W

	Facility	Maximum Processing Rate (metric tons of heavy metal per year)	Duration (years)	Annual Release (curies)	
				Tritium	Krypton-85
Driver Fuel	Fuel Conditioning Facility	0.6	6	738	11,340
Blanket Fuel	Hot Fuel Examination Facility	10	6	71.2	520

Population Impacts:

The doses to the maximally exposed offsite individual and the general public residing within the 80 kilometers (50 miles) surrounding ANL-W are presented in **Table E-12**. As stated in Section E.4.1 the dose resulting from the release of tritium is highly dependent upon its chemical form. The doses presented in Table E-12 result from releases that are assumed to be one percent oxidized tritium, the same assumption used in the

analysis of Alternative 1. These impacts are calculated for the preparation of the blanket fuel assemblies, for the processing of the driver fuel assemblies using the EMT process, and the total impacts. These total impacts are applicable to the processing of blanket and driver materials under Alternatives 2, 3, 4, and 5.

Table E-12 Maximum Annual Radiological Impacts to the Public From Operational Activities Associated With Alternatives 2 Through 5 at ANL-W

	<i>Offsite Population</i>		<i>Maximally Exposed Offsite Individual</i>	
	<i>Dose (person-rem)</i>	<i>Latent Cancer Fatalities</i>	<i>Dose (millirem)</i>	<i>Latent Cancer Fatality Risk</i>
Driver Fuel	0.0028	1.4×10^{-6}	0.00033	1.6×10^{-10}
Blanket Fuel	0.00028	1.4×10^{-7}	0.000048	2.4×10^{-11}
Total	0.0031	1.6×10^{-6}	0.00038	1.9×10^{-10}

Worker Impacts:

The annual worker dose and worker population dose would be similar to those provided in Section E.4.1.

E.4.3 PUREX Processing at SRS (Alternative 3)

PUREX processing at F-Canyon would release radioactive gaseous fission products during treatment of about 57 metric tons of heavy metal of EBR-II and Fermi-1 blanket fuels. Since declad and cleaned blanket fuels are packaged and sent to SRS, no additional gaseous fission products are expected to be present in that fuel. However, for conservatism, it was assumed that the gaseous fission products in the blanket fuels would remain within the fuel matrix and would be released to the environment during PUREX processing at SRS. As a result, there would be incurred doses to the public associated with PUREX operations. The duration of PUREX operations was estimated to be about 6 months, based on the F-Canyon’s throughput and consistent with assumptions made for the treatment duration of a similar type fuel at SRS in the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998).

Gaseous Emissions:

According to SRS Spent Nuclear Fuel EIS data (DOE 1997), tritium and krypton-85 are the only isotopes that would be expected to be released during PUREX processing operations. Based on the assumption that the entire fission gas inventory would remain within the fuel matrix after the decladding and cleaning process, it was assumed that the inventory of krypton-85 and tritium would be released. Using the gaseous fission product inventory provided in Appendix D for the EBR-II and Fermi-1 blanket fuels, the potential airborne radiological release quantities were estimated and presented in **Table E-13**. This inventory was used to calculate the population doses from air emissions.

Liquid Effluent:

PUREX processing is the only process among the alternatives considered that would release measurable radioactive nuclides to the surface water. This release occurs through the cooling water system. The expected radiological effluents from processing of declad and cleaned blanket fuels at F-Canyon were estimated based on the measured data from various effluent streams at F-Area as presented in the SRS Environmental data for 1997 (Arnett and Mamatey 1998). Since the mechanism associated with releases of liquid effluent from PUREX processing at F-Canyon are essentially the same for almost every fuel type processed, the F-Area 1997 effluent data were used to conservatively represent the potential releases from a 6-month operation of F-Canyon. Table E-13 lists the radionuclides and their corresponding curies that are estimated to be released during PUREX processing of blanket fuels.

Table E–13 Estimated Incremental Releases of Radiological Air Emissions and Liquid Effluent During Normal Operations of PUREX Processing at F-Canyon

<i>Isotope</i>	<i>Releases to Air (curies)</i>	<i>Releases to Liquid (curies) ^a</i>
H-3	162.1	1.54
Kr-85	1,187.5	-
Sr-89/90	-	3.13×10^{-5}
Cs-137	-	2.17×10^{-3}
U-234	-	8.48×10^{-5}
U-235	-	1.10×10^{-5}
U-238	-	1.93×10^{-4}
Pu-238	-	1.56×10^{-5}
Pu-239	-	7.76×10^{-6}

^a Estimated curies using the information provided in Arnett and Mamatey 1998.

Population Impacts:

Estimated annual radiological impacts associated with the F-Canyon PUREX operations for the maximally exposed individual and the general population residing within the 80-kilometer (50-mile) population surrounding F-Canyon are presented in **Table E–14**. This table provides the radiological doses to the public from air emissions and liquid effluents separately. According to SRS analytical assumptions (WSRC 1998), a maximally exposed individual associated with liquid releases is an individual who lives downriver of SRS 365 days per year, drinks 2 liters of untreated water per day from the Savannah River, consumes a large amount of Savannah River fish, and spends the majority of time on or near the river. The general population liquid effluent dose is calculated for the discrete population groups at Beaufort-Jasper and Port Wentworth, as well as for other diffuse population groups that make use of the Savannah River; the majority of this dose is due to the drinking water pathway.

Table E–14 Incremental Radiological Impacts to Public from Normal Operational Releases at F-Canyon During PUREX Processing

<i>Offsite Population</i>				<i>Maximally Exposed Offsite Individual</i>			
<i>Air Dose (person-rem)</i>	<i>Liquid Dose ^a (person-rem)</i>	<i>Total Dose (person-rem)</i>	<i>Latent Cancer Fatalities</i>	<i>Air Dose (millirem)</i>	<i>Liquid Dose ^a (millirem)</i>	<i>Total Dose (millirem)</i>	<i>Latent Cancer Fatality Risk</i>
0.019	0.00068	0.020	0.000010	0.00039	0.00012	0.00051	2.6×10^{-10}

^a The dose values were estimated based on the results presented in DOE 1995 for processing a similar fuel.

For conservatism, as well as being able to show compliance with DOE Order 5400.5 (100 millirem annual dose limit to an individual from all pathways), the incremental airborne and liquid doses associated with the F-Canyon processing were summed together even though it is two distinct individuals who receive a maximum airborne and maximum liquid dose. In addition, for analysis purposes, it was assumed that tritium would be released to the atmosphere in oxide form. The public impacts from radiological liquid effluent were estimated based on the results provided in the SRS's *Interim Management of Nuclear Materials Environmental Impact Statement* (DOE 1995). This is consistent with the approach used in the recent SRS *Spent Nuclear Fuel Management Draft Environmental Impact Statement* (DOE 1998). The SRS Spent Nuclear Fuel management EIS (DOE 1998) used "per unit" values (per metric tons of fuels processed) to estimate liquid doses associated with the PUREX processing of 20 metric tons of heavy metal of declad blanket fuel. This EIS uses the same

approach to estimate the radiological doses to the public from potential radiological liquid effluent from PUREX processing.

Worker Impacts:

Worker impacts were estimated by examining historical doses associated with PUREX processing at SRS; these values were based on those presented in the SRS Spent Nuclear Fuel Management Environmental Impact Statement (DOE 1998). The SRS radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. Doses and associated impacts are based on a 6-month processing period. **Table E-15** presents estimated values to both the average worker and entire workforce population.

Table E-15 Incremental Worker Radiological Impacts Due to Normal Operations at F-Canyon

<i>Worker Population</i>		<i>Individual Worker</i>	
<i>Collective Dose (person-rem/yr)</i>	<i>Latent Cancer Fatalities from 6 Months of Processing</i>	<i>Individual Dose (millirem per year)</i>	<i>Latent Cancer Fatality Risk from 6 Months of Processing</i>
75 ^a	0.015	500 ^a	1.0 × 10 ⁻⁴

^a Processing of blanket fuel will require six months of F-Canyon operation, yielding half of the annual doses presented.

E.4.4 Melt and Dilute at SRS Building 105-L Radiological Releases and Impacts (Alternative 5)

Melt and dilute processing at Building 105-L would release radioactive gaseous fission products during treatment of about 57 metric tons of heavy metal of EBR-II and Fermi-1 blanket fuels. Since declad and clean blanket fuels are packaged and sent to SRS, no additional gaseous fission products are expected to be present in that fuel. However, for conservatism, it was assumed that the gaseous fission products in the blanket fuels would remain within the fuel matrix and would be released to the environment during melt and dilute at SRS. As a result, there would be incurred doses to the public associated with these operations. The duration of the melt and dilute process was estimated to be about 3 years, based on the current design throughput of the melter, and an assumption that the final metallic high-level radioactive waste product from this process would contain about 30 percent depleted uranium in aluminum alloy (WSRC 1999).

Gaseous Emissions:

Based on the assumption that the entire fission gas inventory would remain within the fuel matrix after the decladding and cleaning process, it was assumed the inventory of krypton-85 and tritium would be released during the melt and dilute process. Using the gaseous fission product inventory provided in Appendix D for the EBR-II and Fermi-1 blanket fuels, the potential airborne radiological release quantities were estimated and presented in **Table E-16**. These inventories were then used to calculate the population doses from air emissions.

Table E-16 Annual Radiological Releases During Normal Operations of Melt and Dilute at Building 105-L

<i>Isotope</i>	<i>Releases^a to Air (curies)</i>
H-3	54
Kr-85	399

^a There are no liquid releases associated with melt and dilute processing at SRS.

Liquid Effluent:

The melt and dilute process would not produce liquid effluent.

Population Impacts:

Estimated annual radiological impacts associated with melt and dilute operations at SRS for the maximally exposed offsite individual and the general population residing within 80-kilometers (50-miles) surrounding Building 105-L, are presented in **Table E–17**. For analysis purposes, the released tritium was assumed to be in oxide form.

Table E–17 Annual Radiological Impacts Due to Normal Operations of Melt and Dilute Process at Building 105-L

<i>Offsite Population</i>		<i>Maximally Exposed Offsite Individual</i>	
<i>Dose (person-rem)</i>	<i>Latent Cancer Fatalities</i>	<i>Dose (millirem)</i>	<i>Latent Cancer Fatality Risk</i>
0.0076	3.8×10^{-6}	0.00010	5.0×10^{-11}

Worker Impacts:

Worker impacts were estimated by examining historical doses associated with melt and dilute processing at SRS; these values were based on those presented in the SRS Spent Nuclear Fuel Management Environmental Impact Statement (DOE 1998). The SRS radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. Doses and associated impacts are based on a 3-month processing period. **Table E–18** presents estimated values to both the average worker and entire workforce population.

Table E–18 Annual and Cumulative Worker Radiological Impacts Due to Normal Operations of Melt and Dilute at Building 105-L

<i>Worker Population</i>		<i>Individual Worker</i>	
<i>Collective Dose (person-rem per year)</i>	<i>Latent Cancer Fatalities from 3 years Melt and Dilute Processing</i>	<i>Individual Dose (millirem per year)</i>	<i>Latent Cancer Fatality Risk from 3 years Melt and Dilute Processing</i>
50	0.060	500	0.00060

E.4.5 Melt and Dilute at ANL-W (Alternative 6)

In Alternative 6, the blanket and driver fuel elements will need to be prepared at the ANL-W facilities prior to their processing at ANL-W. Preparation of the fuel assemblies at ANL-W for the melt and dilute process requires only that the fuel be cleaned to remove sodium prior to melt and dilute processing; decladding of the blanket and driver fuel is not necessary. This activity would be performed in the argon cell of the Hot Fuel Examination Facility. The following discussion addresses the radiological impacts of normal operations at ANL-W for the preparation and melt and dilute treatment of the blanket and driver fuel assemblies.

Gaseous Emissions:

Fuel preparation would occur at the Hot Fuel Examination Facility. These activities would cause gaseous fission products to be released into the argon cell. As stated earlier in Section E.4.1, krypton-85 and elemental tritium are the most prevalent gaseous radionuclides that would be released to the environment. The released

tritium into the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1 provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies), and are released to the atmosphere through the facility stack, along with the krypton-85 and elemental tritium. The maximum released curies of radioactive gaseous emissions occurs when preparation of the blanket fuel and the driver fuel are performed simultaneously. This simultaneous operation was estimated to occur over a 6-year period starting in 2003. Based on a blanket fuel preparation throughput of 10 metric tons of heavy metal and a driver fuel process rate of about 1.7 metric tons of heavy metal annually, about 2,162 curies of elemental tritium and 32,650 curies of krypton-85 would be released to the atmosphere annually (see **Table E-19**). This release rate would last about 2 years, or until all of the EBR-II blanket fuels and the driver fuel assemblies are processed, then afterward the release rate would drop during the processing of the Fermi-1 blanket fuels (The release rate for the processing of 10 metric tons of heavy metal of Fermi blanket fuel is less than 1 curie of elemental tritium and 6.6 curies of krypton-85).

Table E-19 Normal Operation Radiological Emissions for Alternative 6

	Facility	Maximum Processing Rate (metric tons of heavy metal per year)	Duration (years)	Annual Release (curies)	
				Tritium	Krypton-85
Driver Fuel	Hot Fuel Examination Facility	1.7	2	2091	32,130
Blanket Fuel	Hot Fuel Examination Facility	10	6	71.2	520
Maximum Annual Release				2162	32,650

Population Impacts:

The doses to the maximally exposed offsite individual and the general public residing within the 80 kilometers (50 miles) surrounding ANL-W are presented in **Table E-20**. As stated in Section E.4.1, the dose resulting from the release of tritium is highly dependent upon its chemical form. The doses presented in Table E-20 result from releases that are assumed to be one percent oxidized tritium, the same assumption used in the analysis of Alternative 1. These impacts are calculated for the preparation and processing of the sodium-bonded blanket and driver assemblies and the total impacts.

Table E-20 Maximum Annual Radiological Impacts Due to Normal Operations Under Alternative 6

	Offsite Population		Maximally Exposed Offsite Individual	
	Dose (person-rem)	Latent Cancer Fatalities	Dose (millirem)	Latent Cancer Fatality Risk
Driver Fuel	0.012	6.0×10^{-6}	0.0019	9.5×10^{-10}
Blanket Fuel	0.00028	1.4×10^{-7}	0.000048	2.4×10^{-11}
Total	0.012	6.0×10^{-6}	0.0020	1.0×10^{-9}

E.5 IMPACTS OF EXPOSURES TO HAZARDOUS CHEMICALS ON HUMAN HEALTH

The potential impacts of exposure to hazardous chemicals released to the atmosphere were evaluated for routine operations associated with the alternatives analyzed in this EIS.

The receptors considered in these evaluations are members of the public and noninvolved worker. Impacts of exposures to hazardous chemicals for workers directly involved in the treatment process were not quantitatively evaluated because the use of personal protective equipment and engineering process controls would limit their exposure to levels within applicable Occupational Safety and Health Administration Permissible Exposure Limits or American Conference of Governmental Industrial Hygienists Threshold Limit Values.

As a result of releases from routine operations, receptors are expected to be potentially exposed to concentrations of hazardous chemicals that are below those that could cause acutely toxic health effects. Acutely toxic health effects generally result from short-term exposure to relatively high concentrations of contaminants, such as those that may be encountered during facility accidents. Long-term exposure to relatively lower concentrations of hazardous chemicals can produce adverse chronic health effects that include both carcinogenic and noncarcinogenic effects. The health effect endpoints evaluated in this analysis include excess incidences of latent cancers for carcinogenic chemicals, and a spectrum of chemical-specific noncancer health effects (e.g., headaches, membrane irritation, neurotoxicity, immunotoxicity, liver toxicity, kidney toxicity, developmental toxicity, reproductive toxicity, and genetic toxicity) for noncarcinogens.

Methodology

Annual Airborne concentrations of hazardous chemicals were estimated from the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998).

This EIS estimates the noncancer health risks by comparing annual air concentrations of contaminants to the EPA Reference Concentrations published in the Integrated Risk Information System (IRIS). For each noncarcinogenic chemical, potential health risks are estimated by dividing the estimated airborne concentration by the chemical-specific Reference Concentration value to obtain a noncancer hazard quotient:

$$\text{Noncancer Hazard Quotient} = \text{air concentration/Reference Concentrations}$$

Reference Concentrations are estimates (with an uncertainty spanning perhaps an order of magnitude) of a daily exposure to the human population (including sensitive subgroups) that is likely to be without appreciable risk of harmful effects during a lifetime. Hazard Quotients are calculated for each hazardous chemical to which receptors may be exposed. Hazard Quotients for each chemical are summed to generate a Hazard Index. The Hazard Index is an estimate of the total noncancer toxicity potential from exposure to hazardous chemicals. According to EPA risk assessment guidelines, if the Hazard Index value is less than or equal to 1.0, the exposure is unlikely to produce adverse toxic effects. If the Hazard Index exceeds 1.0, adverse noncancer health effects may result from the exposure.

For carcinogenic chemicals, risk is estimated by the following equation:

$$\text{Risk} = \text{CA} \times \text{URF}$$

where:

Risk = a unitless probability of cancer incidence.

CA = contaminant concentration in air (in micrograms/cubic meters).

URF = cancer inhalation unit risk factor (in units of cancers per micrograms/cubic meters).

Cancer unit risk factors are used in risk assessments to estimate an upper-bound lifetime probability of an individual developing cancer as a result of exposure to a particular level of a potential carcinogen.

Assumptions

The airborne pathway is assumed to be the principal exposure route by which the offsite population maximally exposed individual is exposed to hazardous chemicals released from processing facilities. No synergistic or antagonistic effects are assumed to occur from exposure to the hazardous chemicals. Synergistic effects among released contaminants may result in adverse health effects that are greater than those estimated, whereas antagonistic effects among released chemicals may result in less severe health effects than those estimated.

Analysis

The potential impacts of exposure to hazardous chemicals released to the atmosphere during routine operations of the processing facilities are presented in Chapter 4 for each alternative.

E.6 REFERENCES

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