AIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910
301/587-8202

Centimeter

Inches

MANUFACTURED TO AIM STANDARDS
BY APPLIED IMAGE, INC.

Radionuclide Releases to the Columbia River from Hanford Operations, 1944-1971

Hanford Environmental Dose Reconstruction Project

C. M. Heeb

D. J. Bates

May 1994

**Prepared for the Technical Steering Panel
and the Centers for Disease Control and Prevention
under Contract 200-92-0503(CDC)/18620(BNW)**

**Battelle
Pacific Northwest Laboratories
Richland, Washington 99352**

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

Radionuclide Releases to the Columbia River from Hanford Operations, 1944-1971

May 1994

This document has been reviewed and
approved by the Technical Steering Panel.

J. E. Till, Chair
Technical Steering Panel
Hanford Environmental
Dose Reconstruction Project

April 18, 1994
Date

Preface

In 1987, the U.S. Department of Energy (DOE) directed the Pacific Northwest Laboratory, which is operated by Battelle Memorial Institute, to conduct the Hanford Environmental Dose Reconstruction (HEDR) Project. The DOE directive to begin project work followed a 1986 recommendation by the Hanford Health Effects Review Panel (HHERP). The HHERP was formed to consider the potential health implications of past releases of radioactive materials from the Hanford Site near Richland, Washington.

Members of a Technical Steering Panel (TSP) were selected to direct the HEDR Project work. The TSP consists of experts in the various technical fields relevant to HEDR Project work and representatives from the states of Washington, Oregon, and Idaho; Native American tribes; and the public. The technical members on the panel were selected by the vice presidents for research at major universities in Washington and Oregon. The state representatives were selected by the respective state governments. The Native American tribes and public representatives were selected by the other panel members.

A December 1990 Memorandum of Understanding between the Secretaries of the DOE and the U.S. Department of Health and Human Services (DHHS) transferred responsibility for managing the DOE's dose reconstruction and exposure assessment studies to the DHHS. This transfer resulted in the current contract between Battelle, Pacific Northwest Laboratories and the Centers for Disease Control and Prevention, an agency of the DHHS.

The purpose of the HEDR Project is to estimate the radiation dose that individuals could have received as a result of radionuclide emissions since 1944 from the Hanford Site.

5,6

The HEDR Project work is conducted under several technical and administrative tasks, among which is the Source Term Task. The broad scope of this task is to provide estimates of the amount and timing of radionuclide releases to air and the Columbia River. Radionuclide releases to air occurred at the top of a vent stack. Radionuclide discharges to the Columbia River occurred at the facility outfall pipe. The environmental transport of the releases and *the calculation of radiation doses* form the scope of other HEDR tasks.

The TSP originally directed that the study of radionuclide releases cover the years 1944-1972. However, the releases to the Columbia River were from the eight single-pass production reactors which discharged cooling water directly to the river. These single-pass production reactors operated at the Hanford Site from 1944-1971. This report addresses then the historic release of radionuclides to the Columbia River from those eight reactors, 1944-1971. The report describes how the actual monthly release estimates were made. These estimates have been provided to the HEDR Surface-Water Transport Subtask to calculate radionuclide concentrations in the Columbia River from the Hanford Site to the mouth of the Columbia River. Radionuclide concentration data for the Columbia River are being used to estimate radiation doses to individuals for several exposure pathways at specified river locations.

The report completes HEDR Project Milestone 0304B. It is the final report, replacing the previous version dated January 1994. Appendix E is a record of the TSP comments and BNW responses that have been addressed in this final report. Changes made in response to the comments are denoted by numbers in the left margin and italicized text.

Summary

The purpose of the Hanford Environmental Dose Reconstruction (HEDR) Project is to estimate the radiation dose that individuals could have received as a result of radionuclide emissions since 1944 from the Hanford Site. One source of radionuclide releases to the Columbia River was from production reactor operations. This report provides a quantitative estimate of the amount of radioactivity released each month (1944-1971) to the Columbia River from eleven radionuclides as well as from gross beta activity.

Background

Beginning in September 1944 with the initial startup of B Reactor, eight single-pass production reactors were operated at the Hanford Site. Single-pass reactors used Columbia River water to cool the fuel elements in the reactor core. The cooling water flowed past the fuel elements in the process tubes, was stored temporarily in retention basins, then was released to the river. (A ninth reactor, N, did not discharge directly to the Columbia River and so was not considered in this study.) Operation of the single-pass reactors continued until the last one was shut down in 1971.

Radionuclides were created when neutrons in the reactor core activated native elements present in the inlet cooling water from the Columbia River, as well as elements added to the water by water treatment processes. Reactor neutrons also produced radionuclides by activating elements in the alloys used for process tubes and fuel cladding and activating materials held in the films deposited on the tube and jacket surfaces. The resulting radionuclides were released in the cooling water discharged to the Columbia River. Uranium fuel-element failures caused additional radionuclide releases.

Scope

Although a large number of different radionuclides were discharged, most of them had very small inventories and/or very short half-lives. As a result of a study done by Napier (1993a), the following eleven radionuclides, as well as gross nonvolatile beta, were designated by the Technical Steering Panel of the HEDR Project to be studied:^(a) sodium-24, phosphorus-32, scandium-46, chromium-51, manganese-56, zinc-65, gallium-72, arsenic-76, yttrium-90, iodine-131, and neptunium-239. From these eleven radionuclides, the TSP designated the five most significant radionuclides (Napier 1993a) to be those provided as input to the Columbia River transport model for downriver water and biota concentration estimates and dose calculation. The five used as input were sodium-24, phosphorus-32, zinc-65, arsenic-76, and neptunium-239. Two of the radionuclides designated by the TSP for study, chromium-51 and scandium-46, were provided for validation purposes

(a) Letter (HEDR Project Document No. 07930232), "Key Radionuclides for River Pathway," from J. E. Till (TSP) to D. B. Shipler (BNW), April 12, 1993.

only. Scandium-46, however, was not used for validation because not enough data were available. All available monitoring data for the eleven radionuclides, plus gross beta, were used to estimate monthly releases from 1944-1971.

Technical Approach

To estimate the radionuclides discharged to the Columbia River, the actual curie quantities were reconstructed for each month using reactor operating history and measurements of radionuclide concentrations, where the latter were available. Missing data were reconstructed using statistical analysis of existing data coupled with Monte Carlo modeling techniques.

Results

Table S.1 provides the lifetime totals for the eleven radionuclides and gross nonvolatile beta. These totals are provided in curies/year as median values of 100 realizations. Monte Carlo stochastic modeling was used to account for uncertainties in the source term release estimates. The radionuclides were corrected for decay from the time of release from the reactors to the time of discharge to the Columbia River. Gross beta was decayed for a standard 4 hours from the time of release from the reactor to the time of release to the Columbia River. The line at the bottom of the table represents an estimate of the total release of the radionuclide to the river over the operating lifetime of the eight single-pass reactors.

The uncertainty in these estimates is complex and time-dependent. A general overview of uncertainty can be gained by examining Figures 4.5 through 4.16, which show the release history for each radionuclide. These figures show the minimum, median, and maximum monthly release values of 100 Monte Carlo simulations of the monthly releases.

Table S.1. Total Releases to the Columbia River from Eight Single-Pass Hanford Production Reactors (Median Values of 100 Realizations in Curies per Year)

Yr	Na24	P32	Sc46	Cr51	Mn56	Zn65	Ga72	As76	Y90	I131	Np239	Beta
44	905	216	57	4211	18530	708	4820	1178	300	2	17060	34300
45	34900	2890	2533	44920	95930	10518	75150	20340	3279	54	192100	114230
46	28360	2190	2058	29090	142281	8688	64780	14212	2452	41	153400	68140
47	25450	1860	1839	17785	57049	7457	57360	12319	1436	35	127840	39990
48	33970	2219	2161	30180	98137	8362	69960	15772	2351	47	151100	51630
49	46870	3175	2961	49520	204630	11713	96070	24660	2617	75	214600	105130
50	72590	4027	3762	66090	256090	14546	131730	30510	3636	116	279900	282200
51	99020	3333	2495	86430	232300	11153	113690	23660	6044	694	261700	549300
52	132690	5050	1799	127380	317000	9037	98400	34740	9909	1091	259000	989600
53	202500	8688	2253	173700	962300	8691	120410	98940	13102	821	316200	1481100
54	243400	7261	6545	153860	815000	21910	206900	91380	9938	1562	391600	1702400
55	318000	7184	5269	276600	1471000	26670	267400	139530	15008	3070	419400	2158000
56	407800	7722	6522	252400	1611800	31940	317500	134280	16569	2645	450300	2264000
57	644700	12325	6334	381300	4602000	27560	268600	212130	24230	3500	500100	5065000
58	751400	18484	5931	405800	4081000	27180	258420	293300	32180	5029	422300	4613000
59	1019000	17993	4753	418400	5068000	32030	271978	218400	31010	1986	275100	5321000
60	1382900	19490	9069	592800	7160000	42720	299650	236900	53270	4944	354800	6411000
61	1096300	21526	3313	634400	6370000	47110	236170	166890	49670	4825	243910	6551000
62	1094200	13845	3988	1085600	7965000	56010	92580	86660	37760	4302	257100	5407000
63	887900	11738	6432	592400	5586000	14920	101660	100630	29810	3115	211800	4733000
64	960000	12311	7469	536600	5873000	15710	144080	114480	24980	2827	247500	4425000
65	764500	12126	7968	390200	6364000	13379	94400	124600	21200	2076	168400	3554000
66	613000	7365	6485	256220	4000700	9656	80190	74600	13421	778	78950	2712700
67	671900	10118	8616	298200	5324000	15360	77260	94010	17889	1407	114970	2853000
68	499500	8632	3589	148940	4207000	7734	59090	71670	9798	1150	99790	2397000
69	359200	5478	4571	75730	5618500	6451	41602	61250	5095	473	59820	1523300
70	178041	1759	1135	55297	1026430	3394	32720	20253	7198	1194	36879	822860
71	13200	235	31	7040	85400	386	2660	2440	566	20	3540	72100
Sum	12582196	229239	119937	7191093	79613077	490993	3685230	2519734	444718	47878	6309159	66300980

iii

Acknowledgments

Jim Notch (Boeing) wrote the Source Term River Release Model computer code which made this estimation of releases to the Columbia River possible. S. P. Gydesen (BNW) and R. B. Hall (BNW) searched for and recovered the Hanford documents which created the information base for this reconstruction of radionuclide releases. Radiochemist R. W. Perkins (BNW) provided guidance and added substance to the research. J. K. Soldat (BNW) and R. B. Hall (BNW) gave us the benefit of their vast experience and insight.

9 Reconstructing the operation of the eight single-pass Hanford reactors on a monthly basis forms the backbone of the release estimates. Nuclear engineers E. C. Hawkes, E. F. Love, and U. P. Jenquin, all BNW staff members, contributed to this key task. S. P. Gydesen (BNW) compared the information entered in the database for each fuel-element failure against the original document, providing 100-percent quality assurance of the fuel-element failure data. Thanks are given to TSP Source Term Subcommittee Chair, M. A. Robkin, for his pilot study of possible correlations involving effluent radionuclide activities.

Glossary

actinide - any of a series of chemically similar, radioactive elements with atomic numbers ranging from 89 (actinium) through 103 (lawrencium)

10,11,23 **activation product** - radionuclide produced by neutron absorption by a *radioactive or nonradioactive* parent nuclide

alpha particle - nucleus of the helium atom consisting of two protons and two neutrons

burnup - thermal energy generation; units are megawatt days (MWd)

cfs - cubic feet per second; volumetric flow rate used when referring to the rate of Columbia River flow

Ci - curie; a unit of radioactivity corresponding to 3.7×10^{10} disintegrations per second

coefficient of variation - ratio of the standard deviation to the mean

deterministic - a mode of running a Monte Carlo code with the statistical sampling of distribution functions disabled. Also, a calculation performed using a single set of parameters to generate a single-value answer.

effective holdup time - length of time a parent radionuclide must be exposed to the reactor neutron flux in order to produce a measured activity concentration. It is usually much longer than the physical transit time of the cooling water through the reactor core because dissolved parent radionuclides were adsorbed on the process tubes or fuel-element oxide films, then released much later. The difference between the effective holdup time and the physical transit time is a measure of the degree of interaction between the cooling water and the fuel and process tube surfaces.

effective retention time - length of time effluent water was held in a retention basin before discharge to the Columbia River, normally expressed in hours

FPD - full power days; the number of days a reactor is operated at full power. It is calculated in Equation (3.1) as the ratio of monthly production to average monthly power.

fuel element - aluminum-clad, cylindrical (or tubular in the case of I&E fuel) uranium rod used in Hanford reactors; also called a slug

12 **gamma ray** - *high-energy, short wave-length electromagnetic radiation (a packet of energy) emitted from the nucleus*

gpm - gallons per minute

15 **gross beta** - in this context, the term refers to the total activity of beta-emitting radionuclides, which could not be distinguished separately by the instrumentation in use when the measurements were taken *and did not include volatile beta-emitting radionuclides*

half-life - time required for an initial number of radioactive atoms to be reduced to half that number by transformations

helium - element with atomic number 2

holdup time - measure of the length of time cooling water remained in a process tube (see "effective holdup time")

I&E - later model fuel element with a hole down the center for internal as well as external cooling

KER - KE Reactor experimental facility consisting of a number of process tubes equipped to recirculate cooling water independently from the normal cooling system

Kgpm - reactor flow in thousands of gallons per minute

lognormal - distribution function in which the logarithm of the random variable follows a normal distribution

Mgal - million gallons; in the context of this document, the reactor retention basin capacity

Monte Carlo simulation - technique of representing the effect of uncertainty in one or more contributing parameters on the overall uncertainty by randomly sampling distribution functions which express parameter uncertainty. The results of each complete sampling (realization) are recorded for analysis of computed uncertainty of the outcome.

MW - megawatts; units of reactor thermal power

MWd - megawatt days; units of thermal energy generation (burnup) by the reactor

neutron - electrically neutral particle which, in conjunction with the proton, forms all atomic nuclei

13,14 **normal distribution** - distribution function; *proportional to* e^{-u} , where $u = (x-m)^2/(2s^2)$, x is the statistical variable, m is its mean, and s is the standard deviation

power - reactor thermal power; units are megawatts (MW)

ppb - parts per billion by weight

ppm - parts per million by weight

process tube - aluminum tube which held the uranium fuel elements (slugs) and cooling water

proton - positively charged particle which, in conjunction with the neutron, forms all atomic nuclei

realization - a particular pass through a Monte Carlo simulation where all stochastic parameters have been assigned a value. The simulation represents a "possible reality."

radionuclide - isotope of an element that exhibits radioactivity

regression - algorithm involving one or more independent variables in which the parameters have been adjusted to minimize some function of the difference between the set of actual values and the calculated values of the dependent variable. Typically, the function minimized is the sum of the squares of the difference between the dependent variable values and the values calculated by the algorithm (least squares).

rupture - uranium fuel element in which the aluminum cladding had been breached, exposing the uranium metal to cooling water, also referred to as a "failure"

single-pass reactor - refers to B, C, D, DR, F, H, KE, and KW Reactors which did not recirculate Columbia River water as N Reactor did

slug - aluminum-clad, cylindrical (or tubular in the case of an I&E) uranium fuel element used in Hanford reactors

source term - amount of radioactivity (curies) of a radionuclide released to the environment from an industrial facility at the point of loss of engineering control

spectrometer - in this context, an instrument used to measure intensity and energy of gamma or beta radiation

standard deviation - statistic used as a measure of the dispersion or variation in a distribution, equal to the square root of the mean of the squares of the deviations from the mean of the distribution

STRRM - computer code which provides estimates of the monthly releases to the Columbia River (Source Term River Release Model)

Contents

Preface	iii
Summary	v
Acknowledgments	ix
Glossary	xi
1.0 Introduction	1.1
1.1 Background	1.1
1.2 Purpose	1.1
1.3 Scope	1.1
1.4 Preview of Report	1.2
2.0 Data Quality Objectives	2.1
2.1 Accuracy	2.1
2.2 Completeness	2.1
2.3 Representativeness	2.1
2.4 Comparability	2.2
3.0 Technical Approach	3.1
3.1 Calculation of Radionuclide Releases to the Columbia River	3.1
3.2 Routine Reactor Operations	3.1
3.2.1 Reactor Core and Coolant System	3.1
3.2.2 Purges	3.2
3.2.3 Fuel-Element Failures (Ruptures)	3.3
3.2.4 Reactor Operating Data	3.3
3.2.5 Activity Concentration Measurements	3.4

3.3 Statistical Analysis of Activity Data	3.5
3.3.1 Uncertainty in Activity Concentration Measurements	3.6
3.3.2 Development of Estimated Regressions	3.6
3.3.3 Development of Default Distributions	3.10
3.4 STRRM - Source Term River Release Model	3.11
3.4.1 Outline of Basic Equations	3.11
3.4.2 Uncertainty Distributions	3.13
3.4.3 Input and Output Information	3.13
4.0 Application and Results	4.1
4.1 Uncertainty Distribution Functions	4.1
4.2 Monthly Reactor Operating History	4.8
4.3 STRRM Results Summary	4.8
4.3.1 Sodium-24 (15-hour half-life)	4.11
4.3.2 Phosphorus-32 (14.3-day half-life)	4.13
4.3.3 Scandium-46 (83.7-day half-life)	4.15
4.3.4 Chromium-51 (27.7-day half-life)	4.17
4.3.5 Manganese-56 (2.5-hour half-life)	4.19
4.3.6 Zinc-65 (245-day half-life)	4.21
4.3.7 Gallium-72 (14-hour half-life)	4.23
4.3.8 Arsenic-76 (26.3-hour half-life)	4.25
4.3.9 Yttrium-90 (64-hour half-life)	4.26
4.3.10 Iodine-131 (8-day half-life)	4.27
4.3.11 Neptunium-239 (2.4-day half-life)	4.29
4.3.12 Gross Nonvolatile Beta	4.31

5.0 Conclusions	5.1
6.0 References	6.1
Appendix A - Operating History of Hanford Single-Pass Production Reactors	A.1
Appendix B - Monthly Releases from Eight Single-Pass Hanford Production Reactors	B.1
Appendix C - Bibliography of Reactor Reference Sources	C.1
Appendix D - Description of STRRM Input and Output File Information	D.1
Appendix E - Summary of Technical Steering Panel Comments and Battelle, Pacific Northwest Laboratories Responses	E.1

Figures

4.1	Ratio of Measured to Estimated Retention Time	4.2
4.2	Distribution of Total Uranium Lost per Failure	4.6
4.3	Histogram of Average Tube Burnup at Failure	4.6
4.4	Normalized Rupture Potential at Various Inlet Temperatures	4.8
4.5	Reconstructed Sodium-24 Release History	4.10
4.6	Reconstructed Phosphorus-32 Release History	4.12
4.7	Reconstructed Scandium-46 Release History	4.14
4.8	Reconstructed Chromium-51 Release History	4.16
4.9	Reconstructed Manganese-56 Release History	4.18
4.10	Reconstructed Zinc-65 Release History	4.20
4.11	Reconstructed Gallium-72 Release History	4.22
4.12	Reconstructed Arsenic-76 Release History	4.24
4.13	Reconstructed Yttrium-90 Release History	4.26
4.14	Reconstructed Iodine-131 Release History	4.27
4.15	Reconstructed Neptunium-239 Release History	4.28
4.16	Reconstructed Gross Beta Activity Release History	4.30

Tables

S.1	Total Releases to the Columbia River from Eight Single-Pass Hanford Production Reactors	vii
4.1	Classification of Fuel-Element Failures	4.5
A.1	Hanford Single-Pass Production Reactor Operating History of B, C, D, and DR Reactors (1944-1971)	A.1

A.2	Hanford Single-Pass Production Reactor Operating History of F, H, KE, and KW Reactors (1944-1971)	A.12
B.1	Total Monthly Releases of Na-24, P-32, Sc-46, Cr-51, Mn-56, and Zn-65 from Eight Single-Pass Hanford Production Reactors (Ci/month)	B.1
B.2	Total Monthly Releases of Ga-72, As-76, Y-90, I-131, Np-239, and Gross Beta from Eight Single-Pass Hanford Production Reactors (Ci/month)	B.12

1.0 Introduction

17

As a *part* of the Hanford Environmental Dose Reconstruction (HEDR) Project, this report presents a reconstruction of the historic releases of eleven radionuclides to the Columbia River during the operation of eight of the nine Hanford Site production reactors: B, C, D, DR, F, H, KE, and KW. The eight original reactors were termed "single-pass" reactors because they discharged effluent cooling water directly to the river following holdup in retention basins. N Reactor, the last operating production reactor, recirculated water within its core and did not discharge directly to the river. Therefore, it is not included in the scope of this report. All available effluent water activity measurements were used. Monte Carlo modeling techniques were employed to assess uncertainty and to assign estimates to time periods when measurements were not made.

1.1 Background

The first Hanford reactor (B Reactor) began operation in September 1944. This was followed by construction and operation of seven more reactors of similar design. The last single-pass reactor was shut down early in 1971. N Reactor continued operation until 1989. During the operation of the single-pass production reactors, large quantities of radionuclides were released to the Columbia River. Initial HEDR scoping studies (Napier 1993a) indicated that eleven of these radionuclides were the largest contributors to dose. The eleven radionuclides were approved for studying by the HEDR Technical Steering Panel (TSP). Five of these eleven radionuclides were designated as input to the Columbia River transport model.^(a) Two of the eleven radionuclides were added to the five river transport modeling input set, to be used for river transport validation.

1.2 Purpose

4,16,18

The possible consequences to individuals from the radiological releases can be addressed by starting with estimates of the amount and timing of those releases. These estimates are derived from the large amount of information concerning radioactive releases to the Columbia River from Hanford reactor operations that exist in government- and contractor-generated documents, plus articles in technical journals. The purpose of the Source Term Task within the HEDR Project is to provide monthly radionuclide release estimates for eleven radionuclides over the entire period of single-pass reactor operation, 1944-1971. *Five of the eleven will be used by the river transport model as discussed in the next section. The source from each reactor to the river at the reactor outfall is provided to the river transport model.*

(a) Letter (HEDR Project Document No. 07930232), "Key Radionuclides for River Pathway," from J. E. Till (TSP) to D. B. Shipler (BNW), April 12, 1993.

1.3 Scope

The following eleven radionuclides, as well as gross nonvolatile beta, were designated by the TSP to be studied: sodium-24, phosphorus-32, scandium-46, chromium-51, manganese-56, zinc-65, gallium-72, arsenic-76, yttrium-90, iodine-131, and neptunium-239. The five most significant radionuclides (Napier 1993a) provided as input to the Columbia River transport model for estimating downriver distribution estimates and dose calculations are sodium-24, phosphorus-32, zinc-65, arsenic-76, and neptunium-239. Two of the radionuclides designated by the TSP for study, chromium-51 and scandium-46, were provided for validation purposes only. Scandium-46, however, was not used for validation because not enough data were available. Although radionuclide release monitoring data are plentiful, the number of radionuclides covered and the time periods addressed are limited. The data in the Hanford literature are generally reported on a monthly basis. Although some daily reactor operation information does exist, it does not cover the entire 1944-1971 time period. Therefore, release information is provided on a monthly basis. The radionuclide half-lives are also provided, to be used as reference for later discussion.

<u>Radionuclide</u>	<u>Half-life</u>
sodium-24	15.0 hours
phosphorus-32	14.3 days
scandium-46	83.7 days
chromium-51	27.7 days
manganese-56	2.5 hours
zinc-65	245 days
gallium-72	14 hours
arsenic-76	26.3 hours
yttrium-90	64 hours
iodine -131	8 days
neptunium-239	2.4 days

1.4 Preview of Report

The method used to reconstruct monthly radionuclide data is discussed in the technical approach section. This is followed by a section summarizing monthly release estimates. A table of monthly reactor operating data is included in Appendix A. The estimated monthly releases and their attendant uncertainty ranges are tabulated in Appendix B. Appendix C provides a bibliography of reactor reference sources, and Appendix D provides a description of Source Term River Release Model (STRRM) input and output file information.

2.0 Data Quality Objectives

The data quality objectives for estimating radionuclide releases to the Columbia River are defined in Shipler (1993). Each objective is presented and discussed in this section.

2.1 Accuracy

20 The task plan for the Source Term Task requires that accurate estimates of source terms based on all known measurements be provided. Holding out a subset of data for accuracy testing would have *increased the uncertainty* of the release estimates. Therefore, all available historical concentration data (Gydesen 1992a, 1992c, 1993) were used to develop the estimates.

2.2 Completeness

The completeness objective requires that release information cover the period 1944 through 1971 and identify those radionuclides which contribute up to 99 percent of the potential radiation dose from the Columbia River pathway. Completeness is also dependent on generating a complete set of values.

21 Because the eight single-pass reactors were shut down by the end of 1971 and N reactor did not discharge directly to the Columbia River, no study for 1972 was necessary. Napier (1993a, p. 47) determined that five radionuclides accounted for more than 95 percent of the potential total dose from the Columbia River. The five radionuclides are the ones being used as input to the Columbia River transport model. *Originally, two radionuclides were to be used for model validation. Subsequently, it was found that data for one of these radionuclides (scandium-46) was insufficient, and the radionuclide was excluded from validation studies.* Four additional radionuclides plus gross nonvolatile beta were considered of enough interest by the TSP to be included in this study. A compendium of historical concentration data (Gydesen 1992a, 1992c, 1993) was used in developing the estimates.

2.3 Representativeness

The representativeness objective requires data to represent all known activities, operating variations, and accidental releases. To ensure representativeness, all recorded incidents must be included in the development of the release data, which will be verified through peer review.

All of the requirements of the representativeness objective were met. In the case of operating variations, a complete record of thermal production, flow, and power level was reconstructed from operating data report summaries. This record is provided in Appendix A. There were no accidental releases to the river because the effluent water system design did not provide a means of accidental release. The "accidental releases" refer to purges (discussed in Section 3.2.2) and fuel-element failures (discussed in Section 3.2.3). The purges were planned but irregular in frequency. The

subject of purges was investigated and found to be inconsequential in the scale of monthly releases as explained in Section 3.2.2 of this document. All known natural and 0.947 weight-percent uranium-235 fuel-element failures (1,963) were recorded. Their contributions to the release estimates were computed in the Source Term River Release Model (STRRM) model and included in the release calculations. HEDR staff performed an independent quality assurance check on 100 percent of the fuel-element failure data.

2.4 Comparability

The comparability objective requires cross-comparisons to be made with multiple historical reference sources. Any discrepancies larger than 10 percent must be resolved.

Because all available historical release data were used in the source term reconstruction, no outside sources were available for comparison. However, the HEDR validation plan (Napier 1993b) imposes more stringent comparability tests. It examines the source term, the river transport, and river activity concentration measurements to establish comparability between both source term and transport modeling.

Various annual environmental reports, which describe the environmental monitoring program, were examined from their beginning in 1958 through 1971. No estimates, either monthly or annually, of curie releases to the Columbia River were made in any of the reports. The reports do contain substantial radionuclide concentration information at points downstream of the reactors, but this information cannot be compared with at-reactor releases without recourse to river transport modeling assumptions.

3.0 Technical Approach

This section describes the technical approach used to reconstruct monthly radionuclide release values.

3.1 Calculation of Radionuclide Releases to the Columbia River

All primary radioactive materials generated at the Hanford Site were produced either by neutron-induced fission in uranium, bred actinides, or fission products or by neutron activation of previously nonradioactive materials (activation products). Neutron activation occurs in three ways:

- 22 • radiative capture: a neutron is absorbed, followed by gamma-ray emission (n, gamma)
- neutron, proton: a neutron is absorbed, followed by proton emission (n, p)
- neutron, alpha: a neutron is absorbed, followed by alpha-particle emission (n, alpha)

4 A quantitative reconstruction of reactor operations was necessary to determine the amount of radioactive materials produced. *Release estimates in this study were based on the concentration of radionuclides in effluent water entering the river from retention basins.* Hanford records provided this information. Monthly radionuclide release information is the product of the following factors:

- activity concentration in the reactor effluent (decay-corrected for the delay between discharge from the reactor to discharge to the river)
- reactor flow rate
- number of days of operation during the month.

Estimates of the missing monitoring data were supplied for months where no activity measurements were found. These estimates were generated using statistical analysis of available monitoring data coupled with Monte Carlo uncertainty modeling (see Section 3.3).

3.2 Routine Reactor Operations

A brief description of those aspects of routine, single-pass reactor operations which relate to radionuclide releases to the Columbia River is provided in this section.

3.2.1 Reactor Core and Coolant System

The Hanford production reactors described in Ballinger and Hall (1991) were large graphite cubes with between 1000 and 4000 horizontal aluminum process tubes, the exact number depending on the reactor design. Six- to eight-inch long uranium fuel elements were inserted into the process tubes. Until 1958, the fuel elements were solid cylinders of uranium metal clad in aluminum alloy.

After 1958, the solid "slugs," as the fuel elements were called, were replaced with hollow cylinders completely clad in aluminum. The newer model slug was called the "I&E" because it was both internally and externally cooled.

To cool the fuel elements, inlet water from the Columbia River was pumped into the water treatment plant. There chemicals were added to adjust the pH, decrease turbidity, and inhibit corrosion of the supply piping and reactor process tubes. The processed river water was then filtered, held in clear wells, and pumped into large holding tanks. From the tanks it was pumped to the reactor inlet. Cooling water passed from the inlet piping into the annular gap between the slug surface and the process tube. If the process tube was loaded with I&E slugs, some of the water went down the center hole of the slug.

During its brief passage through the reactor core region (1 to 2 seconds), the water was heated from the inlet temperature (0 to 20°C) to over 100°C in the highest-powered tubes. The cooling water was also subjected to a neutron flux of between 10^{13} and 10^{14} neutrons per square centimeter per second. The hot effluent water (bulk temperature as high as 95°C) was discharged from the reactor into external retention basins located near the Columbia River. After cooling thermally and allowing time for the shortest-lived radionuclides to decay, the basin water was discharged to the Columbia River. The retention basin capacities were designed to produce an effective retention time of 2.4 to 4 hours (Hall and Jerman 1960). With gradual design modifications to increase reactor power, however, the reactor bulk flows in B, D, DR, F, and H Reactors were increased to almost three times the originally designed flows; this resulted in effective retention times nearer to 1 hour.

The retention basins received reactor effluent water at the inlet end and discharged water to the river at the outlet end. Basin water could flow directly from the inlet end to the outlet end with little mixing. Baffles were included in some of the earlier designs to minimize direct passage through the basin.

3.2.2 Purges

As reactor operation continued, films of oxides and entrained materials built up on both the process tubes and the fuel elements. This reduced the gap between the tube and the clad fuel element surface, which resulted in a reduction in bulk flow. Diatomaceous earth was used periodically to remove film buildup and regain flow. A slurry of the abrasive diatomaceous earth was injected into the inlet cooling water during full power operation. This material mechanically removed some of the film from the slugs and process tubes. These purges began in 1945 and continued until final shutdown in 1971. Before 1951, the purges were conducted during reactor outages; after that, purging was done at full power. Because the film being removed contained radionuclides, purges resulted in increased radioactive discharges to the Columbia River.

Radionuclide releases to the river during diatomaceous earth purges were determined to be minor compared to routine operational releases, fuel-element failures, and activation of corrosion products in the process tubes. For a period of almost four years, the incremental increase in the amount of phosphorus-32, chromium-51, zinc-65, arsenic-76, and neptunium-239 released with each purge was reported in a series of Reactor Effluent Analytical Data Reports (Hall 1962; Corley 1962; Corley 1963; Koop 1963a; Koop 1963b; Koop 1963c; Koop 1963d; Koop 1964a; Koop 1964b;

Koop 1964c; Koop 1964d; Koop 1965a; Koop 1965b) covering July 1961 through March 1965. This information was reported along with the monthly total of curies released. These reports show that the maximum percentage of radionuclides released during the purges, compared to the monthly total release of all subject radionuclides, occurred for chromium-51 in October 1963. The five purges for the month amounted to only 2.9 percent of the monthly radionuclide release total. This places the diatomaceous purge releases in the minor release category. Hence, they were considered insignificant relative to routine operational releases and fuel-element failures.

Another type of purge was the use of proprietary chemical compounds to dissolve oxide film. These operations were performed during shutdowns at greatly reduced flow. The purge water was retained long enough to reduce the amount of short half-life radionuclides to acceptable levels before release to the Columbia River.

3.2.3 Fuel-Element Failures (Ruptures)

23

Nearly 2000 fuel-element failures occurred in the eight original Hanford reactors. A failure occurred when the aluminum cladding on the fuel element was breached, allowing coolant water direct access to the irradiated uranium. The uranium-water reaction proceeds rapidly at process temperatures. The result was a release of fission products and actinides to the effluent water.

Every attempt was made to remove the ruptured slug as soon as possible. The reactor was shut down as soon as a rupture was indicated. With the development of scintillation counting in 1951, the primary detection system in place was a gamma spectrometer set to detect the presence of specific fission products in the effluent water. The instrument sequentially sampled water from each tube row during operation. Thus, when a fuel-element failure was detected, the tube row containing the failed element was usually known at shutdown. A radiation survey along the suspected tube row was conducted soon after shutdown. The radiation reading from the ruptured tube was usually high enough above the general background level within the reactor to permit identification. Attempts were then made to discharge the tube. However, the initial attempt sometimes failed when uranium oxide buildup had caused the slug to swell, lodging it securely in the process tube. Most of these "stuck ruptures" were discharged within the following 24 hours by applying higher forces than were routinely available from the normal charging machines.

Information on the reactor, the date, and the rupture classification for each rupture was extracted from Hanford reports (Gydeson 1993). This information was used to estimate the release contributions of iodine-131 and neptunium-239 from fuel-element failures. These two radionuclides, the first a fission product and the other a neutron capture product of uranium-238, were widely used indicators of the presence of fuel-element failures.

3.2.4 Reactor Operating Data

The routinely recorded operating variables were found in Gydesen (1992b; 1992d):

- thermal power level (MW) - determined by simultaneous measurement of bulk inlet and outlet temperatures and bulk flow to the reactor. Reactor power is proportional to the product of flow and the difference between inlet and outlet temperatures.

- production (MWd) - determined by calculating the integral of power over time for a given production period
- reactor flow (Kgpm) - rate of cooling water flow through the process tubes.

An evaluated reactor data set was developed using several Hanford document sources. These are detailed in Appendix C, a bibliography. These sources contain a lot of overlapping information. The same measurement was reported in more than one of the data sources. In those few instances where there were differences in measurements, a choice was made and the reasons documented in the HEDR Project records.

Monthly activity releases are the product of activity concentration (pCi/ml), monthly production (MWd) divided by monthly averaged power (MW), and multiplied by the reactor flow. Hanford operating documents provide information on production, power, and flow.

3.2.5 Activity Concentration Measurements

24 Before 1951, only the gross nonvolatile total beta activity in reactor effluent was routinely measured. The General Electric Health Instruments Operations Group collected effluent water samples at the retention basin outlet to the river. The beta activity was measured with an electroscope exposed to a shallow pan of effluent water. Effluent gamma activity was measured by an ionization chamber suspended over the discharge of the retention basin. Individual radionuclide activities could only be determined by first chemically separating the elements, then determining their activity by half-life identification from the decay curve. No data on *radionuclide*-specific activities were found from 1947 through 1949 (Hall 1991).

25,26 Beginning in 1951 with the development and application of scintillation counting methods, the measurement of individual radionuclide activities became feasible on a routine basis. Initial attempts were made to identify all radionuclides contributing to the gross beta activity. It was, however, found that the parts did not add up to the whole because of imperfect analytical procedures and because most of the radionuclides of interest contributed only a small fraction of the total gross beta activity (Hall 1991). *The major short-lived components of the nonvolatile gross beta measurements were never definitively identified and their effect on dose is unknown. It was generally known that manganese-56 was the largest single contributor (see Section 4.3.12).*

Historical documents from the post-1951 period provided over 53,000 radionuclide concentration data points. The points were mostly daily measurements made during operation on one or more days during the month. About 3 percent of the data points were average curies per day values for the month.

The major gaps in the information base were the lack of specific radionuclide concentration measurements before 1951 and the absence of data on the eleven radionuclides of interest during some months.

3.3 Statistical Analysis of Activity Data

In order to provide a continuous release history of the eleven radionuclides and gross beta for those months without measured data, monthly estimates were made of radionuclide concentrations in reactor effluent from the eight operating single-pass reactors (B, C, D, DR, F, H, KE, and KW) during their period of operation, 1944 through 1971. Monthly effluent activity concentration data were available in Hanford reports for many months. Often several measurements were made during the month. A statistical analysis was made of the recorded effluent concentration data to estimate effluent radionuclide concentrations for those months where no historical data are available. Once radionuclide activity was estimated, the uncertainty in the estimates provided a basis for Monte Carlo modeling.

Several alternative distributions to model activity measurements were tried. The lognormal distribution was chosen to represent the activity measurements because it provided the best fit to the widest range of data. Its use eliminates the possibility of negative results as a Monte Carlo sampling outcome. It also works better than the normal distribution in returning the occasional high values that were noticeable in the data.

The Monte Carlo model had to provide a value of the activity concentration for each of the radionuclides measured in the reactor effluent for each month. Concentration data were grouped in a hierarchy according to the following criteria:

- 27 1. If one or more effluent monitoring measurements existed for the month, the value used in a given Monte Carlo realization was obtained by *multiplying the monthly average by a value obtained from sampling a lognormal distribution with a unit mean and a standard deviation equal to the value for that radionuclide (see table in Section 3.3.1). The standard deviation was divided by the square root of the number of measurements in the monthly average. This procedure results in a lognormally distributed monthly mean value with a standard deviation equal to the estimated measurement uncertainty in a sample size equal to the number of measurements made during the month.*
2. If the information required for Group 1 was not available, an average Ci/day value was used for the month. Lognormal distribution was sampled as in Group 1 *for a single measurement during the month.*
3. If neither Group 1 nor Group 2 information was available, an estimated regression value was used, *lognormally distributed with the standard deviation of the estimated value.*
4. If no information was available from Groups 1, 2, or 3 *above*, the mean and standard deviation of all the measured values at all reactors over all the months were used in lognormal sampling.

4 Group 3 data analysis is called "estimated regression" analysis. Group 4 analysis results are referred to as "default" distributions. The monthly releases are calculated from these data according to Equations (3.1) and (3.2) in Section 3.4.1. *Tables in Appendix B show the range between minimum and maximum realizations. The extent of the range is an indication of what group the concentration data falls into. Group 1 realizations have the narrowest range, and Group 4 realizations the widest. The complete data source table is not appended because of its length.*

3.3.1 Uncertainty in Activity Concentration Measurements

Measurement uncertainty is normally determined by analysis of repeated measurements of the same quantity. No information was found on replicate measurements of radionuclide concentration. Hence, an assessment of the measurement uncertainty was made by choosing periods when the activity concentrations were not changing very much and examining the distribution of measurements. This was as close to replicate measurements as the data allow. It was found that the variation could be represented by a lognormal distribution with a unit mean and the following standard deviations:

<u>Radionuclide</u>	<u>Standard Deviation</u>
sodium-24	0.23
phosphorus-32	0.44
scandium-46	0.67
chromium-51	0.40
manganese-56	0.30
zinc-65	0.48
gallium-72	0.41
arsenic-76	0.39
yttrium-90	0.38
iodine-131	0.42
neptunium-239	0.40
gross beta	0.22

When there was more than one measurement in a given month (Group 1), these standard deviations were divided by the square root of the number of measurements to reduce the variance before the lognormal distribution was sampled in the Monte Carlo simulation. When an average Ci/day value was given (Group 2), lognormal sampling was done using the standard deviation values listed above because the computation must have been based on at least one measurement. In both cases, the sampled value from the unit mean lognormal distribution was multiplied by the mean pCi/ml or Ci/day for the month to produce a value for that particular realization.

3.3.2 Development of Estimated Regressions

For those reactor operating months where no historical data exist, regression methods were used to estimate monthly reactor effluent radionuclide concentrations from available data. Four general classes of potential independent variables were explored in the regression work: natural precursor conditions, inlet water treatment variables, reactor operating conditions, and concentrations of other radionuclides in the effluent water. These are discussed in detail in the following sections.

3.3.2.1 Natural Precursor Conditions

Concentrations of parent nuclides in Columbia River water should be related to radionuclide activities in the effluent. However, the only continuously available seasonal variables that might be related to parent nuclide concentrations were

- monthly Columbia River flow (cfs)
- monthly inlet water temperature (°C).

3.3.2.2 Inlet Water Treatment Variables

28

Water treatment chemicals added to Columbia River water should be directly related to the resulting radionuclide activities in the effluent for some radionuclides. However, there were many time intervals without data, so that even if water treatment proved to be a useful predictor, the necessary data were not always available. *In many cases, water treatment chemicals contained parent radionuclides. The release estimates are based on measurements that include whatever additives were being used.* Types of water treatments reviewed are listed below along with a very brief description of when they were used:

- coagulants (used to improve filtration)
ferric sulfate (Fe_2SO_4) (phased out from November 1951 to April 1954)
aluminum sulfate (Alum) ($\text{Al}_2(\text{SO}_4)_3$)
activated silica (seasonal)
Separan 2600 (seasonal)
- corrosion inhibitors
sodium dichromate ($\text{Na}_2\text{Cr}_2\text{O}_4$) (continuous, constant)
- film inhibitors
sodium silicate (Na_2SiO_4) (discontinued ~ August 1949)
- pH control
Lime (CaO) (discontinued ~ May 1955)
aluminum sulfate (used temporarily for pH control; continued as a coagulant)
sulfuric acid (H_2SO_4)
- algae control
chlorine (Cl_2) (continuous).

3.3.2.3 Reactor Operating Conditions

Information about reactor operating conditions should be significantly related to the resulting radionuclide activities in the effluent water.

The following operating parameters were available on a monthly basis:

- reactor power (MW)
- reactor production (MWd)
- reactor flow (Kgpm)
- effluent outflow temperature (°C).

3.3.2.4 Concentrations of Other Radionuclides

Some radionuclide activities are significantly related to the activities of other radionuclides. Therefore, when data for particular radionuclide activities are known, regressions can be used to estimate the activities of those radionuclides where data are missing. Unfortunately, there are some time periods when almost no radionuclide activity concentration data were available (including those radionuclide activity concentrations that could be used as possible predictor variables).

The regression estimation approach was limited by the availability of correlation variables. A correlation variable is the independent variable used to calculate the radionuclide activity concentration. In order to estimate radionuclide releases using the regression approach, values for all of the independent variables must be available over a selected time period. Major difficulties were encountered in finding long stretches of time where independent variables were always available. In the end, this led to a choice between developing literally hundreds of regression equations that would work for short periods of time for selected radionuclides for only some of the reactors or developing only a few regression equations that would contain fewer independent variables but would frequently give prediction results for most reactors for most months. The latter method was chosen because there were concerns about the reliability of predictions from regression equations based on short time periods; namely, a few extreme results in the available data could lead to even more extreme predictions.

29

The historical data of effluent measurements were almost complete for the time period from August 1958 through June 1962, so regressions were not used to replace the relatively few missing radionuclide results during this time. *Different independent variable data were available for the periods of overlapping time. Regressions that had a lower uncertainty were estimated using specific data for these periods.* Those time periods where data were incomplete and, therefore, regression equations were necessary are:

- July 1962 through December 1969
- September 1944 through July 1958
- Selected periods between January 1951 and June 1957.

In general, results of the regressions could be put into two groups: the two K reactors exhibited one pattern and the other six reactors exhibited another pattern. The reactors were, therefore, divided into these two groups for the purposes of the regression work.

July 1962 through December 1969: Data recorded during this time period provided estimates for six of the radionuclides: sodium-24, scandium-46, manganese-56, gallium-72, yttrium-90, and iodine-131; and gross beta activity. The records for this period were complete for the two natural precursor conditions (river flow and inlet water temperature), for one of the water treatment variables (pH), and for the reactor operating conditions (reactor power, reactor production, reactor flow, and effluent outflow temperature). Data were also fairly complete for phosphorus-32, zinc-65, chromium-51, arsenic-76, and neptunium-239 levels, so these radionuclides were used as correlation variables.

The results of the regression from this period are shown with the R^2 values obtained for each regression. R^2 is a measure of the regression accuracy in fitting the data, where $R^2 = 1.0$ indicates a perfect fit and $R^2 = 0.0$ indicates that there is no correlation between the predictor and response variables. The independent variables in this regression are

- river flow
- inlet temperature
- pH
- concentration of other radionuclides.

<u>Nuclide</u>	<u>R^2 Values</u>	
	<u>B.C.D.DR.F.H</u>	<u>KE. KW</u>
sodium-24	0.36	0.47
scandium-46	0.45	0.62
manganese-56	0.30	0.23
gallium-72	0.30	0.45
yttrium-90	0.27	0.35
iodine-131	0.07	0.23
gross beta	0.54	0.61

September 1944 (reactor startup) through July 1958: Estimates over this period were provided for all eleven radionuclides and for gross beta. However, the only independent variables regularly available were one precursor variable, river flow (cfs), and the four reactor operating conditions (reactor power, reactor production, reactor flow, and effluent outflow temperature). The regressed radionuclides are shown below with tabulated R^2 values.

<u>Nuclide</u>	<u>R^2 Values</u>	
	<u>B.C.D.DR.F.H</u>	<u>KE. KW</u>
sodium-24	0.32	0.28
phosphorus-32	0.11	0.16
scandium-46	0.26	0.10
chromium-51	0.10	0.14
manganese-56	0.26	0.11
zinc-65	0.10	0.17
gallium-72	0.18	0.29
arsenic-76	0.17	0.14
yttrium-90	0.10	0.14
iodine-131	0.03	0.09
neptunium-239	0.57	0.38
gross beta	0.38	0.36

Selected time periods (January 1951 - December 1953, July 1955 - September 1955, October 1956 - June 1957): These three time periods provided measured concentration values of sodium-24, phosphorus-32, manganese-56, zinc-65, arsenic-76 and gross beta. These were used, along with reactor and water treatment variables, to provide a better fit for seven radionuclides: scandium-46, chromium-51, zinc-65, gallium-72, yttrium-90, iodine-131, and neptunium-239. The improvement in the fit can be judged by comparing the following improved fit R^2 values with R^2 values in the previous table. For example, scandium-46 R^2 value was improved from 0.26 to 0.36 by including the additional correlation variables. The improved fits were used to provide the release estimates where possible. The effect of low R^2 values is to increase the uncertainty modeled in STRRM.

<u>Nuclide</u>	<u>R^2 Values</u>	
	<u>B.C.D.DR.F.H</u>	<u>KE. KW</u>
scandium-46	0.36	0.31
chromium-51	0.20	0.42
zinc-65	0.23	0.33
gallium-72	0.28	0.36
yttrium-90	0.28	0.28
iodine-131	0.13	0.22
neptunium-239	0.72	0.56

3.3.3 Development of Default Distributions

The means and standard deviations of release estimates over all reactors and all months were calculated for each of the eleven radionuclides and gross beta. These mean values and standard deviations are contained in the default distribution file (see Appendix D). The mean radionuclide concentration (pCi/ml), standard deviation, and coefficient of variation are shown below:

<u>Radionuclide</u>	<u>Mean</u>	<u>Standard Deviation</u>	<u>Coefficient of Variation</u>
sodium-24	699.6	318.8	0.456
phosphorus-32	11.92	8.183	0.686
scandium-46	4.417	9.97	2.257
chromium-51	434.7	362.1	0.833
manganese-56	6180	5600	0.906
zinc-65	21.9	26.32	1.202
gallium-72	159.2	145.2	0.912
arsenic-76	137.2	113.1	0.824
yttrium-90	33.75	30.48	0.903
iodine-131	1.605	3.036	1.892
neptunium-239	186.2	158.4	0.851
gross beta	3420	2168	0.634

The largest uncertainties (large coefficient of variation) are for scandium-46, zinc-65, and iodine-131. Twenty-seven percent of the scandium-46 estimates, 12 percent of the zinc-65 estimates, and 13 percent of the iodine-131 estimates were made using the default distribution (Group 4 in Section 3.3).

3.4 STRRM - Source Term River Release Model

The STRRM computer code is used to simulate releases from one reactor at a time. It applies the Monte Carlo technique to estimate the monthly releases of the eleven radionuclides and gross beta. The Monte Carlo technique was used to provide estimated monthly release values, as well as the associated uncertainty, for every radionuclide and gross beta where measured effluent concentrations were not available. In the Monte Carlo technique, the process of providing a release estimate is performed repeatedly according to a specific algorithm or set of algorithms. Missing input data and missing specific values within an uncertainty range in known input data are supplied by randomly sampling from a distribution function which expresses parameter uncertainty. This results in a number of "realizations" of possible release values. The variation among realizations reflects the uncertainty in the release when all modeled input uncertainties are taken into account.

3.4.1 Outline of Basic Equations

Estimates of monthly radionuclide releases resulting from activation (discussed in Sections 4.3.1 through 4.3.12) were calculated using Equation (3.1):

$$R = 0.00545 * C * F * PD / PW \quad (3.1)$$

where R = release for a given reactor-year-month-nuclide combination (Ci/month)
 C = average concentration of the nuclide (pCi/ml)
 F = reactor flow (Kgpm)
 PD/PW = number of full power operating days computed as a ratio of
 monthly production (PD) to the average monthly power (PW).

The constant in Equation (3.1) is required for the conversion of units to a consistent set.

For months when an average curie per day release for the month was provided but no average activity concentration was given, Equation (3.2) provided the monthly release estimate:

$$R = CD * DM \quad (3.2)$$

where R = release for a given month and radionuclide combination (Ci/month)
 CD = monthly average curies per day
 DM = number of calendar days in the month.

Radionuclide releases were calculated at the discharge point from the reactor. The amount of radionuclide release to the river was reduced by the radioactive decay that occurred during holdup in the retention basin (effective retention time), as shown in Equation (3.3):

$$RR = R * e^{-\lambda * HTP} \quad (3.3)$$

where RR = average curie release to the river for the month (Ci/month)
 R = release from the reactor with no decay (Ci/month)
 λ = decay constant (hours⁻¹)
 HTP = effective retention time (hours).

Releases of iodine-131 and neptunium-239 resulting from any fuel-element failures that occurred during the month were added to the monthly releases as computed by Equations (3.1) and (3.2). Fuel-element failure releases are calculated by Equation (3.4):

$$RF = W * FP * e^{-\lambda * HTP} \quad (3.4)$$

where RF = number of curies of iodine-131 or neptunium-239 released by the reactor (decayed by the number of holdup hours before release to the river)
 W = number of grams of uranium released from the failure
 FP = number of curies of iodine-131 or neptunium-239 per gram of uranium (a function of the burnup and specific power)
 λ = decay constant (hours⁻¹)
 HTP = effective retention time (hours).

The iodine-131 and neptunium-239 release estimates were calculated using the four best fits of ORIGEN2 (Croff 1980) output data for these radionuclides. There is one fit for natural uranium for iodine-131 and neptunium-239 and another for enriched fuel for each radionuclide, a total of four fits for both. The ORIGEN2 cases were run over a range of power densities from 1.0 MW/ton to 10 MW/ton and a range of burnups from 100 MWd/ton to 2,500 MWd/ton. The results were fitted to bivariate polynomial expressions in PD (power density) and BU (burnup). The ORIGEN2 fits for iodine-131 are

$$CI131N = PD*(26450. + 0.9132*BU - 5.448E-05*BU^2 - 63501.*PD/BU - 5655000.*PD/BU^2)$$

$$CI131E = PD*(26100. + 0.4593*BU + 5.78E-05*BU^2 - 57270.*PD/BU - 5934000.*PD/BU^2)$$

The ORIGEN2 fits for neptunium-239 are

$$CNP239N = PD*(1069300. - 280.67*BU + 0.11281*BU^2 - 1.786E-05*BU^3)$$

$$CNP239E = PD*(820840. - 109.48*BU + 0.03306*BU^2 - 3.917E-06*BU^3)$$

The dependent variables are

$CI131N$ = iodine-131 content in natural uranium (Ci/ton)
 $CI131E$ = iodine-131 content in enriched (0.947 weight-percent) uranium (Ci/ton)
 $CI239N$ = neptunium-239 content in natural uranium (Ci/ton)
 $CI239E$ = neptunium-239 content in enriched (0.947 weight-percent) uranium (Ci/ton)

The independent variables are

PD = power density (MW/ton)

BU = burnup (MWd)

The regression (R^2) values for the iodine-131 fits in ORIGEN2 are 0.987 and 0.984 for natural and enriched, respectively. The R^2 values for neptunium in ORIGEN2 are 0.993 and 0.979 for natural and enriched, respectively.

3.4.2 Uncertainty Distributions

31

The quantities in the previous sections were not always known at the time the computation was made. Values were supplied by sampling the following distributions, which are more fully defined in Section 4.1:

- DHUPT - retention basin holdup time (supplies a value for HTP, effective retention time)
- DPOWER - reactor power measurement uncertainty (supplies a value for specific power, *an independent variable* in the iodine-131 and neptunium-239 regressions)
- DFIT - uncertainty in the iodine-131 and neptunium-239 regression from ORIGEN2 (Croff 1980) which supplies the curie content of uranium as a function of specific power and burnup (supplies a value for FP, the number of curies per gram uranium of iodine-131 and neptunium-239)
- DGRAMS - number of grams released per fuel-element failure (supplies a value for W, the number of grams of uranium released from a fuel-element failure)
- DRUPBURN - burnup distribution of fuel-element failures (supplies a value for the burnup of the failed fuel element which is used in the ORIGEN2 fits)
- DRPOT - position of the failed slug in the process tube.

STRRM provides a value of the activity concentration for each of the radionuclides measured in the reactor effluent for each month. Development of the concentration values and the data availability hierarchy is discussed in Section 3.3. The logic of the data hierarchy is coded into STRRM; that is, STRRM uses Group 1 data preferentially. If no Group 1 data are available, STRRM uses Group 2 data, etc.

3.4.3 Input and Output Information

A detailed description of the information contained in the STRRM input and output files is shown in Appendix D. The size of the files makes it impractical to include printouts in this

document. Readers interested in the information contained in either the input or the output files may obtain electronic copies in ASCII format from:

Technical Steering Panel, c/o K. CharLee
Office of Nuclear Waste Management
Department of Ecology
Technical Support and Publication Information Section
P.O. Box 47651
Olympia, Washington 98504-7651

STRRM is used to simulate releases from one reactor at a time. A deterministic case may be run which produces the single best estimate value of each uncertainty distribution. Otherwise, a specified set of Monte Carlo realizations may be run wherein all values used are sampled from the appropriate distribution function.

4.0 Application and Results

This section describes the actual selection of specific distribution functions and gives the results of 100 Monte Carlo STRRM realizations of the Columbia River releases.

4.1 Uncertainty Distribution Functions

The distribution functions chosen to represent uncertainty are discussed below. Those files relating to activity concentration measurement uncertainty were described in Section 3.3.

DHUART models all retention time uncertainty. The equation for retention time, T , is

$$T = (\text{basin volume})/(\text{reactor flow}) * \text{DHUART}.$$

The ratio of basin volume to reactor flow is the calculated (or design) retention time. DHUART is the uncertainty ratio between actual (for a given realization) retention time and calculated retention time. Retention time uncertainty comes from two considerations:

- Not knowing the effect of mixing that occurred within the basin system.
- Not knowing exactly how many of the available basins were in service.

32

Both are considered in choosing a distribution function form for DHUART. An experiment was conducted in the 1950s to measure effective mixing within the basins (Soldat and Quimby 1953). The effective retention time was determined by comparing the counting rates of water samples collected at the inlet and outlet ends of each basin. The time required for the beta emitter counting rate of the water sample collected at the inlet to equal the counting rate (*as determined by the effluent water beta decay curve*) of the sample collected at the outlet was considered to be the effective retention time.

Injection of uranyl nitrate at the inlet to the basin was also used to determine the effective retention time. The effective retention time in this method was defined as the time lapse between maximum concentration of uranyl nitrate at the inlet and maximum concentration at the outlet.

Soldat and Quimby (1953) list 62 effective residence time measurements at reactors equipped with dual rectangular retention basins (B, D, DR, F, and H). The results are expressed as a ratio of observed retention time to the calculated retention time. The results are plotted in Figure 4.1.

A triangular distribution with a minimum at 0.45, a mode at 0.80, and a maximum of 1.15 appeared to represent the data best, and was selected to model the retention time uncertainty at the dual rectangular basin reactors (B, D, DR, F, and H). The higher values beyond 1.15 were assumed to result from questionable data. The upper cutoff at 1.15 eliminated the most extreme measured values.

B, D, and F Reactors were all equipped with a rectangular basin divided into two halves of 6.1 Mgal each. Reactor operations used both halves in parallel until 1946 (Hall and Jerman 1960). After that, effluent from normal operations was discharged to one half, and the other half was held on

Figure 4.1. Ratio of Measured to Estimated Retention Time
(Mean: 0.85, Median: 0.795, for 62 Observations)

standby for high-radiation effluent from purges and fuel-element failures. By 1954, however, parallel operation of the two tanks was the rule. This continued until final shutdown.

DR and H Reactors were each equipped with a single rectangular basin divided into two halves of 9.1 Mgal each. Single-basin operation with one half on standby was used until 1954 when parallel operation was instituted as at B, D, and F. The BDF triangular distribution applies to DR and H because all of the basins were rectangular.

Three measurements of effective-to-calculated retention time made on the C East basin (Soldat and Quimby 1953) gave a minimum of 0.476 with a standard deviation of 0.045 and a maximum of 0.619 with a 0.017 standard deviation. Because there were only three measurements, a uniform distribution was chosen to represent mixing uncertainty. First, the lower bound was calculated by subtracting three standard deviations from the measured minimum: $0.476 - (3 \times 0.045) = 0.341$. Then the upper bound was calculated by adding three standard deviations to the measured maximum: $0.619 + (3 \times 0.017) = 0.670$. Thus, mixing uncertainty was represented by a uniform distribution from 0.341 to 0.670.

C Reactor had two cylindrical steel basins of 10.2 Mgal capacity (Hall and Jerman 1960). Initial operation had one tank receiving effluent and the other on standby. For an unspecific period after January 1954, both single and parallel tank operation was used. According to Hall and Jerman (1960, p. 2), "The use of the tanks singly or in parallel has not been consistent." With the added uncertainty in the mode of operation, the distribution function must represent not only the mixing uncertainty within a given tank, but also whether one or two tanks were in use. Given no other information, one would conclude that the capacity of the C basin system would be 1.5 tanks or $1.5 \times 10.2 = 15.3$ Mgal. This could range from single-tank operation ($10.2/15.3 = 0.667$) of average capacity to two-tank operation ($20.4/15.3 = 1.33$) of average capacity. The final uniform distribution would then range from a low of $0.341 \times 0.667 = 0.228$ to a high of $0.670 \times 1.33 = 0.891$. With a nominal flow of 65,000 gpm, minimum retention time would be $0.23 \times 15,300/65 = 54$ minutes. The maximum would be $0.89 \times 15300/65 = 209$ minutes (3.49 hours).

K East (KE) and K West (KW) Reactors were each equipped with three cylindrical basins of 9.0 Mgal (Hall and Jerman 1960). Initial operation at KW briefly used the three basins in series. Beginning in April 1955, normal operation for both KE and KW was two basins in parallel and one on standby. The three measurements of effective retention time done at C East basin were assumed to apply also to the identical cylindrical basins at the two K reactors. Because two basins were used, the only uncertainty is due to mixing. The following uniform distribution calculation can be used. Minimum: $0.476 - (3 \times 0.045) = 0.341$. Maximum: $0.619 + (3 \times 0.017) = 0.670$. With a capacity of 18 Mgal and a flow rate of 185,000 gpm, the retention time with these limits would range from 33 to 65 minutes.

DPOWER describes the uncertainty in measured reactor power. The uncertainties in flow and temperature measurement are ± 5 percent, uniformly distributed.

DFIT describes the uncertainty in the iodine-131 and neptunium-239 activity in uranium fuel at various specific powers and burnups. This was addressed in Heeb (1991, p. 3.7). The uncertainty is ± 3.6 percent for iodine-131 and ± 19.2 percent for neptunium-239. The basis for the uncertainty is a bounding calculation comparison of two spectrum-averaged cross section values. A uniform distribution with the appropriate range was, therefore, selected.

DGRAMS provides the number of grams of uranium released per fuel-element failure (W in Equation 3.4). There were essentially no direct measurements of the number of grams released per failure found in the Hanford documents. Such a measurement would be very difficult to make. It would involve 1) determining the initial uranium weight of the slug itself (individual slugs were only identified with a lot number; hence, individual slug weights were not normally available), 2) stripping the cladding and uranium oxide from the ruptured fuel element, and 3) weighing the remaining uranium metal. The operation would have to be performed in an inert atmosphere (clean uranium metal surfaces are susceptible to spontaneous combustion) in a radiometallurgical hot cell facility. Many radiometallurgical examinations of ruptures were conducted, but none were found which had these necessary data.

Fortunately the routine fuel-failure data (Gydesen 1993) included a failure classification. The definitions of the rupture classes are discussed in Bloomstrand (1957). The following weight-loss ranges were assigned to each recorded failure type based on the author's experience, which includes examining and classifying approximately 60 fuel-element failures while working as a B Reactor

process engineer from 1958-1961. Weight losses for the lower categories overlap because the same number of grams could be lost by failures in two different classes.

Class 1 worst failures: "The largest single release of fission products to the river at Hanford" was "about 450 g" (Jerman et al. 1965, p. 6). This particular failure occurred on May 12, 1963 at KE and is listed as unclassified (Class 6) in the fuel-element failure data file, presumably because it did not fit neatly into any of the other classes. Two of the 1570 fuel-element failures used for classification are listed as "disintegrated." These must have released a large amount of uranium, but they must have been estimated at less than 450 grams. Therefore, a range of 200 to 450 grams is assigned to Class 1. In addition to the two disintegration failures, an additional 20 failures from the unknown category were added to Class 1 for the purposes of this study to account for possible failures listed as Class 1 failures.

Class 2 severe failures: These involved splits in the uranium core and aluminum cladding which exposed a large surface of uranium metal. A range of 100 to 200 grams was assigned to this class.

Class 3 medium failures: These were predominantly side failures. Included in this classification (Bloomstrand 1957) were side hot spots and side reduced ribs, a failure on the side of the element between the process tube ribs, thought to be due to reduced rib height. The medium fuel-element failures all shared a slug side failure, which usually started on a small area on the side of the aluminum cladding and grew larger, the final size depending on how long it took the operators to establish that a fuel-element failure had occurred and to shut the reactor down. Because the amount of uranium surface exposed to the coolant was usually less with this type of failure than with the two more severe failure classes, a range of 50 to 150 grams was assigned to this class.

Class 4 low failures: End failures of all kinds were included in this category. According to Bloomstrand (1957, p. 3), these failures might have been caused by corrosion penetration or uranium core dimensional changes that resulted in separation of the cap. The amount of exposed uranium ranged from very small penetrations of the cap weld to complete separation of the cap, in which case only the area of the cylinder end was exposed. A range of 10 to 100 grams was assigned to this class.

Class 5 I&E hole failure: After the solid slugs were replaced by I&E fuel elements, some failures were experienced in the center coolant hole. Because of the small diameter of the hole (0.310 inches at the oldest reactors), the amount of exposed uranium was small when compared to failures in the annulus. Even a partial plugging of the hole resulted in a detectable reduction in tube flow. This caused the reactor to be shut down sooner than when side failures occurred. Accordingly, a range of 1 to 10 grams was assigned to this class.

Class 6 failure: This class included all fuel-element failures not assigned to Classes 1 through 5.

Listed in Table 4.1 are the fuel-element failure types included in each of the original five severity classifications, along with the number of failures in each class.

Table 4.1. Classification of Fuel-Element Failures

<u>Class</u>	<u>Failures in Original Classifications</u>	<u>Number in Class</u>
Class 1 Worst (200 to 450 grams)	Disintegration, plus 20 from unknowns for purposes of this study	22
Class 2 Severe (100 to 200 grams)	Cleavage, uranium split, split, split transverse, core, jacket split, side & cap, split longitudinal, can split	331
Class 3 Medium (50 to 150 grams)	Cap, cpd. cap end, end-female, end-male, end other	290
Class 4 Low (10 to 100 grams)	Side, side tear, side other, side groove, side ledge, side wall, canister wall, side hot spot	813
Class 5 I&E Hole (1 to 10 grams)	Hole, inside other	114
Class 6 All Others	Blank field, mechanical damage, not available, not classified or unknown (less 20 put in Class 1 for purposes of this study)	393
Total all classes		1963
Total of classes 1 to 5		1570

The total number of natural and enriched fuel-element failures was 1963. Of these, 413 were assigned no classification or were not classifiable for a variety of reasons. Twenty of the 413 unclassified failures were included in the current Class 1, leaving 393 not included in the ranking.

Figure 4.2 shows the STRRM distribution function based on these six classes. This histogram is broken up into six non-overlapping classes: 200-450, 150-200, 100-150, 50-100, 10-50, and 1-10 grams. The first of these is the same as Class 1 (i.e., it contains 22 failures). The second one (150-200) contains half of Class 2, so the number of failures assigned to it is $(331/2) = 166$. The third non-overlapping class (100-150) is composed of the remaining half of Class 2 and half of Class 3: $166 + (290/2) = 311$. The fourth class (50-100) is composed of the remaining half of Class 3 plus $5/9$ of Class 4: $(290/2) + (5/9)(813) = 597$. The fifth class (10-50) is composed of the remaining $4/9$ of Class 4: $(4/9)(813) = 361$. The sixth class (1-10) is the same as Class 5 (i.e., it contains 114 failures). This piecewise uniform distribution places the most likely mass loss in the 50- to 100-gram range, but does provide an upper range of 200 to 450 grams for 22 out of 1570 failures (1.4 percent).

DRUPBURN describes the burnup distribution of fuel-element failures. Records of many of the fuel-element failures did not include data for the average burnup of the uranium metal in the tube at failure. To supply the missing information, an analysis of failures with burnup data supplied was performed. A histogram of the results is shown in Figure 4.3.

Figure 4.2. Distribution of Total Uranium Lost per Failure (1570 Failures Included)

Figure 4.3. Histogram of Average Tube Burnup at Failure

A triangular distribution with a minimum at 0 MWd/ton, peak at 550 MWd/ton, and a maximum of 1000 MWd/ton appeared to represent the data best and was chosen to represent the distribution.

DRPOT describes the position of the failed slug within the process tube. An analysis of fuel-element failures by Bloomstrand and Neef (1958) showed that the failure rate as function of burnup, power, and water temperature could be expressed in the following form:

$$RP = KRP * P^{3.3} * T^{8.7} * B^{5.0} \quad (4.1)$$

where RP = rupture potential
 KRP = normalization constant
 P = power (MW)
 T = coolant temperature (°C)
 B = burnup (MWd/ton).

RP can be expressed as a function of x, the distance down the slug column from inlet to outlet because 1) power distribution down the length of a process tube determines the local slug power as well as the local cooling water temperature, 2) coolant temperature is proportional to inlet temperature plus a constant times the definite integral of P(x) from the inlet end to the point x in the process tube, and 3) slug burnup distribution will have the same form as the power distribution, P(x), if the shape of the power distribution is constant.

Rupture potential can, therefore, be transformed into a function of x, assuming a constant power shape over time:

$$B(x) = K * P(x)$$

where x = distance down the slug column from inlet to outlet
 B(x) = burnup at point x
 K = normalization constant
 P(x) = power at point x

so that $P^{3.3} * B^{5.0}$ in Equation (4.1) can be replaced by $P^{8.3}$ and

$$RP(x) = NKRP * P(x)^{8.3} * T(x)^{8.7} \quad (4.2)$$

where RP(x) = rupture potential at point x
 x = distance down the slug column from inlet to outlet
 NKRP = normalization constant
 P(x) = power at point x
 T(x) = coolant temperature at point x.

Using a representative power distribution (truncated or chopped cosine), RP(x) was calculated with an assumed power peak of 1.42 (ratio of peak to average power). Three RP(x) curves were calculated for inlet water temperatures of 0, 10, and 20°C. Figure 4.4 shows that the rupture potential is not sensitive to variations of inlet water temperature. A similar calculation showed that the shape of the rupture potential distribution was not unduly sensitive to power peaking assumptions.

Figure 4.4. Normalized Rupture Potential at Various Inlet Temperatures (0, 10, and 20°C Inlet, Power Peak 1.42)

A triangular distribution appeared to represent the data best, and was chosen to represent the rupture potential. Figure 4.4 shows a minimum 47 percent of the way down the fuel column (about $x = 9$). The mode is at 68 percent ($x = 13$) and the maximum is at 95 percent ($x = 18$).

4.2 Monthly Reactor Operating History

The monthly operating history of all the single-pass reactors is given in Appendix A. Power, production, and flow information for the period of reactor operations (1944 through 1971) is provided.

4.3 STRRM Results Summary

A complete listing of the computed sum of monthly releases of the eleven radionuclides and gross beta activity from the eight single-pass reactors is provided in Appendix B. The minimum, median, and maximum value of 100 Monte Carlo realizations of the sum are given for each month. The releases are computed at the point of release to the Columbia River. Gross beta, manganese-56 (2.5-hour half-life), gallium-72 (14-hour half-life), and sodium-24 (15-hour half-life) would have decayed significantly in the transport from B to F Reactor. Therefore, the summary numbers for gross beta and these three radionuclides do not constitute the river source below F Reactor (the

reactor furthest downstream). The STRRM output release files list the minimum, maximum, median, standard deviation, and coefficient of variation for each reactor-month-radionuclide entry.

STRRM is used to simulate releases from one reactor at a time. This is appropriate for the source term input files to the Columbia River transport model because the specific location of each source is required to route released radionuclides downriver from B Reactor to F Reactor. River transit time is less significant for the longer-lived radionuclides. The sum of releases from the eight reactors represents the maximum value of Hanford source term releases to the Columbia River.

Figures 4.5 through 4.16 provide an overview of the estimated total release of each radionuclide from the eight single-pass reactors. These figures show the minimum, median, and maximum monthly curie release computed over 100 STRRM realizations for each of the eleven radionuclides and gross beta activity. The total monthly production is plotted separately as a reference. This provides a guide from which to view the computed activity releases. A given release is the product of the monthly production (thermal output of all eight reactors) and an activity concentration. A high release value occurs when there is a high production during the month combined with a high activity concentration. A comparison of the release versus the production allows an assessment of the reasons for the release. In Figures 4.5 through 4.16, the left vertical axis shows the scale for radionuclide release. The right vertical axis shows the scale for monthly thermal energy generated by the eight reactors (MWd). The dark bar chart at the bottom is the monthly production profile that refers to the right vertical axis.

Production variations from month to month are less than the release variations. The reasons for seasonal release variations are apparent for those years when production was stable during the year, but the release shows a similar monthly pattern for several years. These occur because the activity concentrations were found to correlate with the two seasonal variables: inlet water temperature and total flow. Release variations due to temperature variations are caused by seasonal fluctuations in inlet temperature. At constant power level, the temperature rise from reactor inlet to outlet is approximately constant because the reactor flow generally held fairly constant during the year. Hence the outlet temperature often mirrored the inlet temperature. An exception to this is when the reactor power level was limited by the bulk outlet temperature limit during the summer months. Reactor power was then reduced during the summer because of higher ambient water temperatures in the Columbia River. The adsorption and desorption rate of parent radionuclides on the aluminum oxide process-tube and fuel-element films, as well as aluminum corrosion rates, is highly temperature sensitive.

River flow variations are significant because concentrations of parent species may be covariant with river flow. Using the river flow as an independent variable is a convenient surrogate for missing knowledge concerning parent nuclide inlet concentrations. The water treatment process was seasonal because of ambient water quality in the Columbia River. This seasonal fluctuation meant that the concentration of additives required to produce water to constant properties such as pH, turbidity, resistivity, and hardness, varied seasonally with inlet water conditions.

The following sections discuss the possible sources of the eleven radionuclides. The presence of their precursors as contaminants in cooling water, in water treatment chemicals, in the aluminum used as cladding, and in the process tubes is discussed. The sources of uncertainty are also reviewed,

Three upper curves = Minimum, median, and maximum of 100 realizations
 ■ = Total monthly production (MWd)

Figure 4.5. Reconstructed Sodium-24 Release History

including the percentage of values obtained from direct measurements, from Ci/day data, and from estimated regressions and default distributions.

4.3.1 Sodium-24 (15-hour half-life)

This radionuclide is produced by two different neutron reactions. The first is an n, alpha reaction on aluminum-27; the second is an n, gamma reaction on sodium-23. The aluminum-27 reaction occurred on the aluminum process tubes and fuel cladding. Ewing (1957) estimates that 25 percent of sodium-24 came from activation of sodium-23 and 75 percent from aluminum-27.

Perkins (1959) concluded that process tube aluminum was not an important source, based on comparison of normal in-service process tube effluent concentrations with effluent from cleaned process tubes. In a later study, Perkins (1961) concluded that 25 percent of the released sodium-24 came from sodium-23 in water treatment chemicals. He noted a concentration of 810 ppb of sodium-23 in the river water and 1300 ppb of sodium-23 in the process water in November and December 1960. He also noted that the water treatment changes starting in these two months increased sodium-24 releases by 60 percent. The median STRRM releases (Appendix B) show a sodium-24 release value of 85,300 curies in October 1960, rising to 165,000 curies in November, then back down to 102,000 curies in December.

35

Perkins (1961) measured the effective holdup time in the reactor at 3.8 seconds. This compares with a flow time of only about 1 second. The difference between the effective holdup time and the flow time is a measure of the degree of interaction between *the elements* in the cooling water and the fuel and process tube surfaces.

Silker and Thomas (1964a, 1964b) used experimental loop facilities at the KE Reactor (known as the KER facility) to estimate the relative sodium-24 contribution from process tubes versus fuel cladding. They measured the sodium-24 release first using new fuel in new aluminum tubes, and then using new fuel in zirconium tubes. They estimated a relative contribution of 30 percent from process tubes versus 70 percent from fuel cladding. This conflicts with Perkins' (1959) finding that the contribution from process tubes was not significant.

The uncertainty in the sodium-24 release as indicated in Figure 4.5 is smallest in the early 1960s and is an order of magnitude greater in the earlier and later years. This largely reflects the availability of concentration measurements. The peak monthly releases of 100,000 Ci in the early 1960s are the least uncertain. Fifty percent of the data supporting sodium-24 releases comes from one or more measured pCi/ml values per month. Forty-seven percent comes from estimated regressions, and the remaining 3 percent comes from default distributions.

Three upper curves = Minimum, median, and maximum of 100 realizations
 ■ = Total monthly production (MWd)

Figure 4.6. Reconstructed Phosphorus-32 Release History

4.3.2 Phosphorus-32 (14.3-day half-life)

This radionuclide is produced by two different neutron reactions. The first is an n, gamma reaction on phosphorus-31. The second is an n, p reaction on sulfur-32.

Ewing (1957) tried to measure phosphorus-32 production by KER activation experiments on phosphorus-31 and sulfur-32 additions to the coolant but did not get a good measurement. He concluded that sulfur-32 activation is a minor source of phosphorus-32. In another effort, he calculated that phosphorus-31 impurities in the aluminum alloy used for fuel cladding and process tubes could not be a major source of phosphorus-32. He based this conclusion on an analysis of aluminum 1100 alloy by Foster and Gaitanis (1955) and on corrosion rate correlation by Groot and Wilson (1957). He determined an effective holdup time of 2.5 days, which indicates relatively strong adsorption-desorption reactions with tube and slug films.

In 1959, Perkins (1959) concluded that process tube films were an important source of phosphorus-32, based on measurements of effluent activity from cleaned process tubes.

In 1961, Silker (1961) conducted KER experiments on two aluminum process tubes and one zirconium control tube to determine the relative importance of the two neutron reactions responsible for the presence of phosphorus-32 in the effluent water. He concluded that two-thirds of the phosphorus-32 was produced by the thermal neutron reaction with phosphorus-31, and the remaining one-third was from the fast neutron reaction with sulfur-32. Silker also examined the time-dependent behavior of adsorption-desorption and provided some experimental insight into the complexity of these reactions. After an equilibrium activity concentration of phosphorus-32 was reached with a steady addition of phosphorus-31 to the inlet of a KER tube, the addition of phosphorus-31 was stopped. The post-addition decay curves for phosphorus-32 activity were analyzed in terms of three component system models with half-lives of 7 hours, 30 hours, and 27 days. This indicated that at least three different adsorption-desorption mechanisms were at work, each with a characteristic half-life.

Perkins (1961) concluded that 40 percent of phosphorus-32 came from activation of sulfur-32. This is relatively close to Silker's (1961) estimate of 33 percent. Perkins calculated a holdup time of 2300 seconds for a phosphorus-31 parent and 9.6 seconds for a sulfur-32 parent.

KER loop experiments (Silker and Thomas 1964a, 1964b) on new fuel in new aluminum tubes versus new fuel in zirconium tubes permitted an estimate of the relative contribution from the tube versus the slug. The results indicated that 55 percent of phosphorus-32 came from process tube films and 45 percent from slugs.

Estimated phosphorus-32 releases peaked in the late 1950s and early 1960s. The early estimates are based on estimated regressions, and the widest uncertainty (in 1970) is based on the default distribution. Sixty-seven percent of the phosphorus-32 data is based on one or more measured concentrations during the month; 9 percent is based on average Ci/day value for the month; 23 percent is based on estimated regressions; and only 1 percent is based on the default distribution.

Three upper curves = Minimum, median, and maximum of 100 realizations
 ■ = Total monthly production (MWd)

Figure 4.7. Reconstructed Scandium-46 Release History

4.3.3 Scandium-46 (83.7-day half-life)

This radionuclide is produced by two different neutron reactions. The dominant source is an n, gamma reaction on scandium-45; a lesser source (Perkins 1961) is an n, p reaction on titanium-46.

Perkins (1961) calculated an effective holdup time of 9.6 seconds for scandium-45, indicating major adsorption-desorption on process films. The scandium-45 impurity in aluminum tube alloys averaged 75 ppb. In slug cladding alloys, it was 68 ppb in Alcoa C64F and 120 ppb in X8001F. Thus, tube and slug corrosion may have been a contributing source of scandium-46.

The scandium-46 source as shown in Figure 4.7 has a rather broad uncertainty because only 24 percent of the estimate comes directly from one or more measurements during the month. Scandium releases were extensively measured from 1958 through 1962, with a few final measurements in 1969. Forty-nine percent of the estimates are based on estimated regressions and 27 percent are based on default distributions.

The measured data show a real decrease in the amount of scandium-46 released from June through December 1961. No reason for this was found in the Hanford literature.

Three upper curves = Minimum, median, and maximum of 100 realizations
 ■ = Total monthly production (MWd)

Figure 4.8. Reconstructed Chromium-51 Release History

4.3.4 Chromium-51 (27.7-day half-life)

Chromium-51 comes directly from an n, gamma reaction on chromium-50. Perkins (1959) compared normal in-service process tube effluent concentrations with effluent from cleaned process tubes and concluded that process-tube films were an important source for this radionuclide. In a subsequent study, Perkins (1961) further concluded that most of the chromium-51 came directly from the sodium dichromate used as an aluminum corrosion inhibitor in the water treatment process. His measurement of the effective holdup time was 640 seconds, indicating strong interaction with tube and slug films.

Chromium-50 impurity in aluminum tube alloys averaged 89 ppm. In slug cladding alloys, it was 8.6 ppm in Alcoa C64F and 25.0 ppm in X8001F.

KER loop experiments (Silker and Thomas 1964a, 1964b) on new fuel in new aluminum tubes versus new fuel in zirconium tubes permitted an estimate of the relative contribution from tube versus slug. The results showed that tubes and slugs contributed an equal amount of chromium-51.

Fifty-one percent of the chromium-51 estimates are based on one or more activity measurements during the month. These started in 1958 and continued through 1969. Figure 4.8 shows a large reduction in uncertainty due to the abundant measurements. Twenty-five percent of the chromium-51 release estimates are based on estimated regressions, 21 percent on default distributions, and only 3 percent on average CI/day data for the month.

Three upper curves = Minimum, median, and maximum of 100 realizations
 ■ = Total monthly production (MWd)

Figure 4.9. Reconstructed Manganese-56 Release History

4.3.5 Manganese-56 (2.5-hour half-life)

The dominant means of manganese-56 production is by an n, gamma reaction on manganese-55. The measured effective holdup time was 440 seconds (Perkins 1961), indicating interaction with tube and slug films. Aluminum corrosion might have also been a contributor, with manganese-55 impurities averaging 292 ppm in tube alloys, 46 ppm in Alcoa C64F, and 110 ppm in X8001F alloy. Perkins (1961) concluded that a significant amount of manganese-56 might have come from impurities in water treatment additives.

KER loop experiments (Silker and Thomas 1964a, 1964b) on new fuel in new aluminum tubes versus new fuel in zirconium tubes showed a relative manganese-56 contribution of 15 percent from tubes and 85 percent from slugs.

One or more daily measurements per month were available in 1951 through 1953 and again from 1959 through 1963. Therefore, these periods show up in Figure 4.9 with smaller uncertainty. Overall, the manganese-56 release estimates are based on 51 percent measured values, 3 percent Ci/day, 25 percent estimated regressions, and 21 percent default distributions.

Three upper curves = Minimum, median, and maximum of 100 realizations
 ■ = Total monthly production (MWd)

Figure 4.10. Reconstructed Zinc-65 Release History

4.3.6 Zinc-65 (245-day half-life)

The dominant means of zinc-65 production is by an n, gamma reaction on zinc-64. Perkins (1961) measured an effective holdup time of 18,000 seconds, indicating strong interaction with tube and slug films and with aluminum corrosion. He noted that zinc-65 releases increased by 22 percent by 1960 because of changes in water treatment. Nielsen (1961) also noted a sensitivity to water treatment. A change in alum feed from 3 to 5 ppm to 18.5 ppm, or substitution of 20 ppm $\text{Al}(\text{NO}_3)_3$ for normal alum feed of 3 to 5 ppm, reduced zinc-65 effluent activity concentration by 25 to 30 percent. This decrease presumably resulted from a more efficient removal of zinc-64 by the alum substitute.

Zinc-64 impurities in aluminum tube alloys ranged from 150 ppm to 880 ppm. In slug cladding alloys, Alcoa C64F had 39 ppm zinc-64 impurities and X8001F had 71 ppm zinc-64 impurities. KER loop experiments (Silker and Thomas 1964a, 1964b) on raw fuel in new aluminum tubes versus new fuel in zirconium tubes indicated a relative zinc-65 contribution of 90 percent from process tubes and only 10 percent from slugs.

Figure 4.10 shows a reduction in uncertainty as daily measurements became abundant in 1958. Measurements were discontinued in 1970 and 1971, forcing a reliance on the default distribution, which increased the uncertainty almost two orders of magnitude for those years. Overall, 50 percent of the estimates are based on daily measurements, 35 percent on estimated regressions, 3 percent on Ci/day, and 12 percent on default distributions.

Three upper curves = Minimum, median, and maximum of 100 realizations
 ■ = Total monthly production (MWd)

Figure 4.11. Reconstructed Gallium-72 Release History

4.3.7 Gallium-72 (14-hour half-life)

The predominant method of gallium-72 production is an n, gamma reaction on gallium-71. Daily measurements of this radionuclide began in 1958 and continued through early 1964. The uncertainty shown in Figure 4.11 reflects this. Thirty percent of the release estimates are based on daily measurements and 65 percent are based on estimated regressions, leaving 5 percent for the default distributions (mainly in 1970 and 1971).

Three upper curves = Minimum, median, and maximum of 100 realizations
 ■ = Total monthly production (MWd)

Figure 4.12. Reconstructed Arsenic-76 Release History

4.3.8 Arsenic-76 (26.3-hour half-life)

The predominant method of arsenic-76 production is by an n, gamma reaction on arsenic-75. Perkins (1961) determined an effective holdup time of 2300 seconds, indicating strong interaction with tube and slug films. Arsenic-76 release was reduced 31 percent by water treatment processes initiated in 1960.

A comparison (Perkins 1959) of normal effluent concentrations with effluent from cleaned process tubes indicated that process tube films were an important source of arsenic-76. Arsenic-75 impurities in aluminum tube alloys ranged from less than 500 ppb to 1700 ppb. In slug clad alloys, arsenic-75 impurities were less than 150 ppb in Alcoa C64F and less than 840 ppb in X8001F.

KER loop experiments (Silker 1960) were performed to determine the kinetics of retention on and discharge from the aluminum cladding and process tubes. It was assumed that discharge occurs 1) by immediate recoil and 2) by slower desorption. Equations were derived for assumptions 1 and 2 and tested against experiments using two KER in-pile loops in which varying concentrations of sodium biarsenate (NaHAsO_4) were added to the loop. The arsenic-75 concentration was increased twenty-fold for 8 days, then the arsenic-76 release rate was measured. The best fit to the measured data was obtained for assumption 2. According to this analysis, no fewer than four detachment half-lives of 20 minutes, 7 hours, 38 hours, and 300 hours were required to fit the observed results.

KER loop experiments (Silker and Thomas 1964a, 1964b) on new fuel in new aluminum tubes versus new fuel in zirconium tubes showed a 40-percent contribution from process tube film and 60-percent contribution from slugs.

Sixty-seven percent of the releases shown in Figure 4.12 are based on daily measurements, 27 percent are from estimated regressions, 5 percent are based on the average number of curies per day and 1 percent is based on the default distribution.

Three upper curves = Minimum, median, and maximum of 100 realizations
 ■ = Total monthly production (MWD)

Figure 4.13. Reconstructed Yttrium-90 Release History

4.3.9 Yttrium-90 (64-hour half-life)

Yttrium-90 is a fission product which came primarily from fissioning of native uranium present in Columbia River water. Twenty percent of the estimated releases shown in Figure 4.13 are based on daily measurements performed from 1958 to 1962. Fifty-three percent are based on estimated regressions and 27 percent on the default distribution.

Three upper curves = Minimum, median, and maximum of 100 realizations
 ■ = Total monthly production (MWD)

Figure 4.14. Reconstructed Iodine-131 Release History

4.3.10 Iodine-131 (8-day half-life)

Iodine-131 came from fissioning of native uranium present in Columbia River water and from fuel-element failures. Summing the median values of the 100 realizations from each reactor showed that over the span of operation from 1944 through 1971, 44.9 percent of iodine-131 releases came from fuel-element failures.

Forty-five percent of the estimates shown in Figure 4.14 are based on daily measurements, 38 percent are from estimated regressions, 13 percent are from the default distribution, and 4 percent are based on average curies per day. The measurement period was from 1955 through 1969, with a few measurements in 1964 and 1965.

Three upper curves = Minimum, median, and maximum of 100 realizations
 ■ = Total monthly production (MWd)

Figure 4.15. Reconstructed Neptunium-239 Release History

4.3.11 Neptunium-239 (2.4-day half-life)

Neptunium-239 results from an n, gamma reaction on uranium-238 which produces uranium-239. This decays by beta emission with a 23.5-minute half-life to neptunium-239. Releases of neptunium-239 resulted from this reaction on native uranium present in Columbia River water and from fuel-element failures. Summing the median values of the 100 realizations from each reactor showed that over the span of operation from 1944 through 1971, 11.9 percent of neptunium-239 releases came from fuel-element failures.

Fifty-six percent of the estimates shown in Figure 4.15 are based on daily measurements, 38 percent are from estimated regressions, 2 percent are from the default distribution, and 4 percent are based on average curies per day. The measurement period was from 1956 through 1969.

The combination of many measurements and a low uncertainty resulted in a relatively low uncertainty for the neptunium-239 release values compared to those of other radionuclides. The largest uncertainty is in 1970, when no measurements were made and the default distribution was used. Two percent of the estimated neptunium-239 release values are based on the default distribution.

Three upper curves = Minimum, median, and maximum of 100 realizations
 ■ = Total monthly production (MWd)

Figure 4.16. Reconstructed Gross Beta Activity Release History

4.3.12 Gross Nonvolatile Beta

The methods of measurement did not include volatile beta emitters. Hence, these are not included in the activity estimates. The gross beta activity included many short half-lived radionuclides. The Hanford convention was to correct all gross beta measurements to a standard 4 hours-out-of reactor, using a composite decay curve which represented the mixture of contributing beta emitters. One result of this is that the estimates of manganese-56 (2.5-hour half-life) often exceed the gross beta estimates because the STRRM-modeled retention time is much less than the standard 4-hour time delay in the gross beta measurement. Early attempts to determine the identity of radionuclide contributors to the gross beta activity put the manganese-56 fraction in the 15- to 30-percent range.

The uncertainty in gross beta is relatively small because daily measurements were available starting in 1947 and, except for 1954, continued until early in 1964. For 1944 through 1947 data, an estimated regression was used which actually had a larger uncertainty than the default distribution which applied only in 1970 and the two months of 1971.

Fifty-seven percent of the estimates are based on daily measurements, 31 percent are from estimated regressions, 9 percent are Ci/day averages for the month, and only 3 percent are based on the default distribution.

5.0 Conclusions

This report provides a quantitative estimate of the amount of radioactivity released each month (1944-1971) to the Columbia River from eleven radionuclides as well as from gross beta activity. Estimates are provided for the entire operating history of the eight Hanford single-pass reactors. Uncertainties in these estimates were determined. All of the release estimates are based on either activity concentration measurements made during the time period of the release or on inferred values resulting from a statistical analysis of data from other time periods. The uncertainties reflect the available number of measurements and are, therefore, larger where no measurements were made.

36

The release estimates are based on all activity concentration measurements available in the historic Hanford documents *reviewed*. Further reductions in the uncertainty are, in theory, possible through the use of more detailed analysis. Perhaps experiments could shed some light on activation mechanisms involved. However, such experiments would be very expensive and might lead to activation algorithms that require a knowledge of independent variables for which no measurements exist (for example, parent nuclide concentrations in inlet water). The present results are sufficient for dose calculations. Improvements in the estimates would be expensive and it is doubtful whether improved estimates would significantly reduce the uncertainty.

The river releases presented, and their attendant uncertainty estimates, form the most accurate reconstruction of actual production reactor radioactivity releases to the Columbia River that can be made at this time. They are an adequate source term for river dose calculations.

6.0 References

- Ballinger, M. Y., and R. B. Hall. 1991. *A History of Major Hanford Facilities and Processes Involving Radioactive Material*. PNL-6964 HEDR, Pacific Northwest Laboratory, Richland, Washington.
- Bloomstrand, R. R. 1957. *Definitions of Terms Used in the Periodic Ruptured Slug Data Reports*. HW-54316, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Bloomstrand, R. R., and W. I. Neef. 1958. *Analyses and Correlations of Hapo Rupture Experience with Natural Uranium Material*. HW-55377, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Corley, J. P. 1962. *Reactor Effluent Analytical Data January Through June 1962*. HW-75023, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Corley, J. P. 1963. *Reactor Effluent Analytical Data July, August, and September 1962*. HW-76103, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Croff, A. G. 1980. *ORIGEN2-A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*. ORNL-5621, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
- Ewing, R. E. 1957. *The Origin of Certain Radioisotopes in Hanford Reactor Cooling Water*. HW-49485, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Foster, L. M., and C. D. Gaitanis. 1955. "Determination of Phosphorus in Aluminum and Alumina by Radioactivation Analysis." *Anal. Chem.* 27(8):1342-1344.
- Groot, C., and R. E. Wilson. 1957. *In-Reactor Corrosion of Aluminum*. HW-44996, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Gydesen, S. P. 1992a. *Documents Containing Operating Data for Hanford Separations Processes, 1944-1972*. PNWD-2028 HEDR, Battelle, Pacific Northwest Laboratories, Richland, Washington.
- Gydesen, S. P. 1992b. *Selected Hanford Reactor and Separations Operating Data for 1960-1964*. PNWD-2017 HEDR, Battelle, Pacific Northwest Laboratories, Richland, Washington.
- Gydesen, S. P. 1992c. *Status of Document Search and Data Quality Objective Efforts*. PNWD-2047 HEDR, Battelle, Pacific Northwest Laboratories, Richland, Washington.
- Gydesen, S. P. 1992d. *Title Listing of Daily Operating Data on Hanford Single-Pass Reactors, 1944-1971*. PNL-7998 HEDR, Pacific Northwest Laboratory, Richland, Washington.
- Gydesen, S. P. 1993. *Fuel-Element Failures in Hanford Single-Pass Reactors 1944-1971*. PNWD-2161 HEDR, Battelle, Pacific Northwest Laboratories, Richland, Washington.

- Hall, R. B., and P. C. Jerman. 1960. *Reactor Effluent Water Disposal*. HW-63653, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Hall, R. B. 1962. *Reactor Effluent Water Analyses Data July 1961 Through December 1961*. HW-73073, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Hall, R. B. 1991. *References for Radioactive Releases to the Columbia River from Hanford Operations, 1944-1957*. PNL-7869 HEDR, Pacific Northwest Laboratory, Richland, Washington.
- Heeb, C. M. 1991. *Uncertainties in Source Term Calculations Generated by the ORIGEN2 Computer Code for Hanford Production Reactors*. PNL-7223 HEDR, Pacific Northwest Laboratory, Richland, Washington.
- Jerman, P. C., W. N. Koop, and F. E. Owen. 1965. *Release of Radioactivity to the Columbia River from Irradiated Fuel Element Ruptures*. RL-REA-2160, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Koop, W. N. 1963a. *Reactor Effluent Analytical Data October, November, and December 1962*. HW-76894, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Koop, W. N. 1963b. *Reactor Effluent Analytical Data January, February, and March 1963*. HW-77766, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Koop, W. N. 1963c. *Reactor Effluent Analytical Data April, May, and June 1963*. HW-78820, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Koop, W. N. 1963d. *Reactor Effluent Analytical Data July, August, and September 1963*. HW-79348, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Koop, W. N. 1964a. *Reactor Effluent Analytical Data October, November, December 1963*. HW-80706, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Koop, W. N. 1964b. *Reactor Effluent Analytical Data January, February, March 1964*. HW-81983, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Koop, W. N. 1964c. *Reactor Effluent Analytical Data April, May, June 1964*. HW-83423, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Koop, W. N. 1964d. *Reactor Effluent Analytical Data July, August, September 1964*. RL-REA-43, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Koop, W. N. 1965a. *Reactor Effluent Analytical Data October, November, December 1964*. RL-REA-498, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- Koop, W. N. 1965b. *Reactor Effluent Data January, February, March 1965*. RL-REA-2028, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

Napier, B. A. 1993a. *Determination of Key Radionuclides and Parameters Related to Dose from the Columbia River Pathway*. BN-SA-3768 HEDR, Battelle, Pacific Northwest Laboratories, Richland, Washington.

Napier, B. A. 1993b. *HEDR Model Validation Plan*. PNWD-2156 HEDR, Battelle, Pacific Northwest Laboratories, Richland, Washington.

Nielsen, J. M. 1961. *Radiological Chemistry Operation Annual Report, January-December 1960*. HW-68533, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

Perkins, R. W. 1959. *Radiological Analysis of Reactor Effluent Waste Materials at Hanford*. HW-SA-1809, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

Perkins, R. W. 1961. *Source of Reactor Effluent Water Radioisotopes*. HW-69969, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

Shipler, D. B. 1993. *Integrated Task Plans, Hanford Environmental Dose Reconstruction Project, June 1992 Through May 1994*. PNWD-2187 HEDR, Battelle, Pacific Northwest Laboratories, Richland, Washington.

Silker, W. B. 1960. *Kinetic of Arsenic Retention in Hanford Reactor Cooling Systems*. HW-63633, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

Silker, W. B. 1961. *On the Origin of Radiophosphorus in Hanford Reactor Effluent Water*. HW-69083, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

Silker, W. B., and C. W. Thomas. 1964a. *The Effect of Deionized Water on Reactor Effluent Activities, Part I: New Tubes and New Fuel Charges*. HW-81412, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

Silker, W. B., and C. W. Thomas. 1964b. *The Effect of Deionized Water on Reactor Effluent Activities, Part II: Zirconium and Aluminum Clad Fuel*. HW-81413, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

Soldat, J. K., and G. R. Quimby. 1953. *Effective Retention Time of the Hanford 107 Reactor Effluent Retention Basins*. HW-28830, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

Appendix A

Operating History of Hanford Single-Pass Production Reactors

Table A.1. Hanford Single-Pass Production Reactor Operating History of B, C, D, and DR Reactors (1944-1971)

Yr	Mo	B Reactor			C Reactor			D Reactor			DR Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
44	9	0	0	0	0	0	0	0	0	0	0	0	0
44	10	62	1190	32	0	0	0	0	0	0	0	0	0
44	11	74	1710	32	0	0	0	0	0	0	0	0	0
44	12	126	2630	32	0	0	0	147	1990	32	0	0	0
45	1	205	5830	32	0	0	0	200	6080	32	0	0	0
45	2	248	6690	32	0	0	0	247	6780	32	0	0	0
45	3	246	7150	32	0	0	0	247	7240	32	0	0	0
45	4	248	7030	32	0	0	0	250	7250	32	0	0	0
45	5	248	7030	32	0	0	0	261	7730	32	0	0	0
45	6	247	7290	32	0	0	0	261	7730	32	0	0	0
45	7	247	7290	32	0	0	0	275	7780	32	0	0	0
45	8	248	7110	32	0	0	0	279	8170	32	0	0	0
45	9	249	7380	32	0	0	0	276	7860	32	0	0	0
45	10	244	7260	32	0	0	0	276	7860	32	0	0	0
45	11	224	6310	32	0	0	0	251	7210	32	0	0	0
45	12	224	6310	32	0	0	0	247	7220	32	0	0	0
46	1	225	6610	32	0	0	0	247	7220	32	0	0	0
46	2	226	6370	32	0	0	0	250	7020	32	0	0	0
46	3	228	2390	32	0	0	0	259	7500	32	0	0	0
46	4	226	6620	32	0	0	0	226	6030	32	0	0	0
46	5	223	5870	32	0	0	0	224	5990	32	0	0	0
46	6	166	2650	32	0	0	0	237	7050	32	0	0	0
46	7	0	0	0	0	0	0	249	6840	32	0	0	0
46	8	0	0	0	0	0	0	248	6960	32	0	0	0
46	9	0	0	0	0	0	0	248	5960	32	0	0	0
46	10	0	0	0	0	0	0	248	5960	32	0	0	0
46	11	0	0	0	0	0	0	247	6620	32	0	0	0
46	12	0	0	0	0	0	0	248	6820	32	0	0	0
47	1	0	0	0	0	0	0	248	6710	32	0	0	0
47	2	0	0	0	0	0	0	239	5840	32	0	0	0
47	3	0	0	0	0	0	0	249	7000	32	0	0	0
47	4	0	0	0	0	0	0	248	7260	32	0	0	0
47	5	0	0	0	0	0	0	252	6990	31.8	0	0	0
47	6	0	0	0	0	0	0	247	6230	31.35	0	0	0

A.1

Table A.1. (contd)

Yr	Mo	B Reactor			C Reactor			D Reactor			DR Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
47	3	0	0	0	0	0	0	248	7060	31.31	0	0	0
47	4	0	0	0	0	0	0	248	6750	31.42	0	0	0
47	5	0	0	0	0	0	0	248	7070	31.81	0	0	0
47	6	0	0	0	0	0	0	247	6570	32.05	0	0	0
47	7	0	0	0	0	0	0	247	6770	31.81	0	0	0
47	8	0	0	0	0	0	0	247	6700	32.13	0	0	0
47	9	0	0	0	0	0	0	247	6240	31.83	0	0	0
47	10	0	0	0	0	0	0	254	6890	32.02	0	0	0
47	11	0	0	0	0	0	0	271	5790	31.95	0	0	0
47	12	0	0	0	0	0	0	272	5810	31.65	0	0	0
48	1	0	0	0	0	0	0	273	6880	31.54	0	0	0
48	2	0	0	0	0	0	0	272	5550	31.43	0	0	0
48	3	0	0	0	0	0	0	273	6630	31.41	0	0	0
48	4	0	0	0	0	0	0	269	5990	31.43	0	0	0
48	5	0	0	0	0	0	0	271	5980	31.42	0	0	0
48	6	0	0	0	0	0	0	270	7390	31.79	0	0	0
48	7	223	6440	31.11	0	0	0	270	7400	32.08	0	0	0
48	8	272	7790	31.7	0	0	0	270	7450	32.34	0	0	0
48	9	235	5920	31.42	0	0	0	269	7100	32.86	0	0	0
48	10	256	7220	31.54	0	0	0	186	6920	32.15	0	0	0
48	11	272	7270	32.14	0	0	0	268	7270	30.84	0	0	0
48	12	274	7830	31.44	0	0	0	264	7060	30.97	0	0	0
49	1	273	7800	31.79	0	0	0	272	7930	30.97	0	0	0
49	2	273	6960	31.8	0	0	0	273	6760	31.2	0	0	0
49	3	273	8050	31.88	0	0	0	271	7410	31.28	0	0	0
49	4	271	7730	31.48	0	0	0	270	6910	31.2	0	0	0
49	5	272	7680	31.75	0	0	0	287	7490	31.58	0	0	0
49	6	270	7500	31.63	0	0	0	298	7930	32.56	0	0	0
49	7	269	7540	31.8	0	0	0	295	8040	32.25	0	0	0
49	8	268	7550	31.75	0	0	0	298	8030	31.92	0	0	0
49	9	269	7400	32.21	0	0	0	297	7670	33.21	0	0	0

Table A.1. (contd)

Yr	Mo	B Reactor			C Reactor			D Reactor			DR Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
49	10	270	7960	32.06	0	0	0	299	8650	33.18	0	0	0
49	11	270	7240	31.94	0	0	0	300	8300	32.74	0	0	0
49	12	271	7770	31.92	0	0	0	287	9030	32.68	0	0	0
50	1	271	7440	31.77	0	0	0	300	8920	32.56	0	0	0
50	2	272	7100	32	0	0	0	300	7350	32.13	0	0	0
50	3	273	7860	31.72	0	0	0	301	8590	32.32	0	0	0
50	4	273	7610	31.14	0	0	0	301	8340	31.93	0	0	0
50	5	288	8320	32.34	0	0	0	297	8480	32.56	0	0	0
50	6	300	8540	32.73	0	0	0	296	6570	32.68	0	0	0
50	7	315	8730	32.6	0	0	0	297	8610	32.29	0	0	0
50	8	330	9360	32.25	0	0	0	315	5550	32.25	0	0	0
50	9	325	9180	32.6	0	0	0	315	6050	33.7	0	0	0
50	10	337	4800	33.2	0	0	0	318	9280	33.5	387	7830	36.2
50	11	356	9990	33.75	0	0	0	321	8750	34	410	9780	36.7
50	12	341	9900	33.43	0	0	0	310	8990	33.21	430	13300	36.7
51	1	395	10592	33.4	0	0	0	370	9830	32.8	461	12019	36.6
51	2	388	10038	33.1	0	0	0	361	8834	33.2	448	12130	36.15
51	3	397	11200	31.11	0	0	0	376	10720	31.41	494	15310	36.37
51	4	412	11960	33.28	0	0	0	402	11600	33.5	469	12440	36.2
51	5	398	11570	33.16	0	0	0	401	10880	34.7	422	11160	36.7
51	6	432	12930	33.64	0	0	0	431	12410	34.61	441	11390	36.78
51	7	392	10730	34.6	0	0	0	403	10610	33.9	499	13580	38.14
51	8	398	9640	36.94	0	0	0	439	11230	36.53	528	15630	40.3
51	9	450	12990	36.5	0	0	0	434	9660	37.4	528	15240	41.3
51	10	430	10900	36.9	0	0	0	440	11920	36.6	521	14710	40.8
51	11	445	11630	36.7	0	0	0	456	11200	36.7	466	11990	40.1
51	12	431	11600	36.4	0	0	0	477	13870	38.1	489	12990	40.7
52	1	440	11590	36.4	0	0	0	462	12930	37.5	361	9570	40.2
52	2	490	12650	36.1	0	0	0	478	12330	38.1	380	8310	40.7
52	3	524	15290	35	0	0	0	448	12670	38.1	536	15670	41.1
52	4	452	11570	36.3	0	0	0	513	14560	38.7	532	14380	40.8

Table A.1. (contd)

Yr	Mo	B Reactor			C Reactor			D Reactor			DR Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kggm)	Power (MW)	Prod. (MWd)	Flow (Kggm)	Power (MW)	Prod. (MWd)	Flow (Kggm)	Power (MW)	Prod. (MWd)	Flow (Kggm)
52	5	467	12940	36.5	0	0	0	532	15370	39	548	16480	41.6
52	6	524	14130	36.4	0	0	0	551	15620	39.7	546	15770	42
52	7	494	14210	37.8	0	0	0	540	14880	39.4	560	17360	41.9
52	8	482	11710	38.3	0	0	0	509	11660	39.7	561	16280	42.1
52	9	521	14220	38.8	0	0	0	565	15410	40	539	15040	40
52	10	532	14640	39	0	0	0	550	15170	39	566	16530	43
52	11	496	12900	38.1	400	4670	65.1	558	16670	38.9	559	15260	42.7
52	12	553	15720	37.8	686	19690	64.8	561	15980	38.6	572	16470	42.7
53	1	466	2660	37.8	697	21540	64.1	540	12470	38.5	563	16160	42.6
53	2	494	11400	38.3	733	19270	64	384	1490	44.5	558	14300	42.5
53	3	526	14840	38.1	764	21620	71.2	543	14190	44.9	485	260	0
53	4	533	12600	38.3	852	22540	72	548	15240	44.7	500	11580	42.6
53	5	539	15620	38.5	876	22020	73.9	594	16590	46.7	415	10330	42.9
53	6	559	14910	38.6	871	22920	75.1	633	17470	47.2	369	10470	43.3
53	7	519	14790	38.5	866	26500	74.5	605	15470	47.5	530	9380	43.6
53	8	557	16040	39	939	25970	74.1	598	16220	47.3	334	10340	43.4
53	9	521	12320	38.9	978	23590	73.7	640	17780	47.4	351	10520	43.2
53	10	541	13630	39.3	945	20920	72.7	663	19200	47.1	531	15110	42.9
53	11	625	17120	39.1	1000	27600	72.2	649	16140	46.5	512	13500	42.5
53	12	664	18260	39.5	976	20760	82.5	739	18760	46.7	544	16830	42.8
54	1	707	19790	40	1074	27570	81.7	814	25180	47.9	593	17630	42.5
54	2	673	12780	40	1096	21300	81.5	796	19570	47.8	388	10220	42.7
54	3	628	9070	45.5	1191	30390	81.7	787	21880	47.7	566	13280	43.1
54	4	708	15870	48.4	1172	24930	82	741	17290	46.8	642	15990	43
54	5	731	20350	48.5	1191	27840	88.7	770	20580	47.7	652	19180	43.1
54	6	821	21630	48.7	1227	27090	89.2	760	21930	47.5	620	14470	42.8
54	7	817	19960	47.4	1182	27740	88.8	739	21120	48.3	610	16940	42.8
54	8	829	19300	48.7	1201	30160	89.9	755	21510	48.3	662	20480	43
54	9	809	21530	48.1	1276	30090	90.6	735	22030	48.8	639	19170	42.7
54	10	851	21870	47.7	1247	31390	87.2	766	20750	48.5	599	16140	42.7
54	11	897	23840	47.7	1268	25680	85.4	766	17280	50.8	654	17600	41.7

Table A.1. (contd)

Yr	Mo	B Reactor			C Reactor			D Reactor			DR Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
54	12	961	26010	47.8	1307	26340	84.3	910	20280	51.7	689	19350	42.2
55	1	918	21490	48.1	1234	27430	82.3	996	26710	52.7	734	21890	42
55	2	841	15920	48.2	1358	24980	82.8	888	18850	58.8	680	15800	41.5
55	3	867	22740	48.8	1350	29090	81.8	951	21790	53.2	735	21190	41.1
55	4	794	20790	48.7	907	18040	82.7	708	16890	52.9	350	5260	41
55	5	819	22720	48.8	838	16550	83	641	15560	52.4	593	14300	49.8
55	6	773	19900	48.5	933	19450	83.4	481	8730	52.4	799	22290	49.8
55	7	922	7320	48.4	1365	31400	84.1	866	11170	52.8	885	22690	50.1
55	8	938	25090	48.4	1351	33340	83	863	19640	53.2	872	24000	50.8
55	9	861	23530	48.2	1278	21550	83.1	900	16310	53.1	641	15460	51
55	10	912	27250	47.8	1358	30690	83	853	11290	53.2	948	21860	51
55	11	875	16750	47.5	1313	22690	82.1	878	12720	52.1	1033	29290	50.5
55	12	895	25220	46.6	1413	29920	83.8	843	5120	52	862	20010	49.6
56	1	784	14050	47.1	1432	25040	84	885	16080	51.1	833	18950	50
56	2	826	13880	46.1	1461	30150	83.8	633	7410	49.9	828	16760	49.8
56	3	841	15750	46.1	1504	34640	83.2	886	17220	49.9	898	20520	50.3
56	4	897	23410	45.8	1532	36580	83.9	922	18890	49.8	924	23590	50.2
56	5	842	21740	46	1400	32380	83	831	14250	49.9	901	22210	51.2
56	6	834	20140	45.8	1492	37780	84.4	853	18810	50.1	933	26620	51
56	7	815	24080	46.4	1447	34540	85.8	764	12290	51.2	901	24800	51.7
56	8	772	18620	46.5	1418	32480	85	847	20530	51	886	25460	51.8
56	9	784	16180	46.2	1299	29730	86	841	21980	50.3	902	24600	51.5
56	10	0	0	0	1276	36750	84.2	879	24850	49.5	904	24100	51.5
56	11	0	0	0	1308	29430	84	932	19240	51	938	24310	52
56	12	938	17800	69	1411	34510	84	963	25350	51.3	1077	33390	51.8
57	1	979	27070	71	1326	34160	81.2	1031	27040	51.1	1106	2400	8.7
57	2	1030	19900	70.2	1265	22700	81.6	1029	10890	25.9	1042	9340	71.5
57	3	1147	21260	71	1324	28650	82.9	0	0	0	1100	26630	68.9
57	4	1227	31800	71	1348	29970	83	1084	26150	67.2	1205	30980	69.1
57	5	1152	24070	70.5	1427	41850	83.6	1116	22070	71	1151	27830	69.2
57	6	1123	24800	71.7	1335	34680	85	1216	28130	71	1254	31830	70.5

Table A.1. (contd)

Yr	Mo	B Reactor			C Reactor			D Reactor			DR Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
57	7	1123	25450	71	1335	21060	85.5	1216	31440	70.9	1254	30400	71.8
57	8	1235	27600	71	1368	31510	85.8	1204	31280	70.5	1244	33940	70.7
57	9	1244	26760	71	1435	30870	87	1218	26020	71	1268	31190	70
57	10	1298	34120	71.6	1329	35060	82	1242	35420	71.2	1253	27770	70.4
57	11	1298	31930	71.5	1329	30340	82.5	1242	29360	71	1253	28870	69.8
57	12	1237	32100	71	1372	33470	71.2	1272	32730	70.4	1222	22310	69.1
58	1	1333	30620	70.8	1472	41840	84.5	1265	34310	71	1338	37410	70.7
58	2	1286	29090	70.8	1356	27220	85.2	1291	32060	70.9	1331	28470	70.5
58	3	1218	27070	70.6	1454	36330	85.5	1305	32160	71.2	1323	32110	69.6
58	4	1306	33150	70.8	1460	37760	90.2	1335	34000	70.2	1261	24250	70.6
58	5	1330	32080	71	1425	35210	92.3	1275	32490	70.7	1256	26940	73.1
58	6	1361	33110	71	1582	41810	94	1224	30290	71	1348	37000	73.4
58	7	1266	32180	73.3	1460	34900	94.5	1270	30980	75.2	1347	27940	73.6
58	8	1273	31330	72.3	1482	32660	94	1242	33500	77.5	1278	26620	73.2
58	9	1267	25620	73.9	1481	33010	95	1333	33990	78.7	1278	27410	74.2
58	10	1297	31000	75.3	1482	36060	93.8	1399	40980	78.7	1215	21730	76.5
58	11	1354	34890	76.2	1284	24000	93.5	1399	38700	78.8	1245	23570	76.9
58	12	1355	36380	76.8	1581	40730	94.2	1403	39230	80.2	1336	30070	78.9
59	1	1345	22220	78.2	1509	32770	94.8	1404	37650	79.4	1360	33260	79.4
59	2	1458	35800	78	1695	35830	94.5	1435	36970	79.9	1502	42010	79.9
59	3	1468	39270	79	1760	40540	95.3	1437	35060	78.4	1466	37670	80.2
59	4	1468	40560	79.9	1625	34270	94.2	1458	36650	78.4	1426	34520	80.2
59	5	1444	33040	80	1737	44260	95	1457	41820	78.6	1479	40940	79.3
59	6	1471	36530	83.1	1693	41600	95.2	1412	31010	78.9	1471	35920	79.5
59	7	1477	40800	83	1715	45080	95.3	1429	40130	81	1399	31140	82
59	8	1457	35000	82.8	1731	44810	95.8	1484	38180	81	1547	43110	83.4
59	9	1499	37460	82.8	1615	31690	96.4	1520	43000	81.1	1498	42420	82
59	10	1454	35880	80.3	1760	51500	95.2	1548	43140	79	1497	37500	80.2
59	11	1619	42640	80	1812	42350	95	1449	31340	78.8	1603	45110	80
59	12	1405	32840	82.9	1756	41920	96	1597	44560	79	1543	36020	79.9
60	1	1540	39000	80.5	1743	43170	94.8	1595	36790	77.7	1556	42880	79.6

Table A.1. (contd)

Yr	Mo	B Reactor			C Reactor			D Reactor			DR Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
60	2	1594	33380	80	1781	36400	94.5	1639	39790	78.2	1533	29700	80
60	3	1544	38340	81	1783	45100	95	1632	40070	78.9	1733	45450	80
60	4	1619	34920	80.5	1842	38960	95.5	1710	46940	82	1754	47090	82.8
60	5	1635	44910	81	1851	49690	94.5	1601	40300	82.2	1580	35800	82.5
60	6	1584	39110	84	1805	42670	95.8	1659	44200	83.7	1594	44280	84.1
60	7	1567	39590	85.3	1792	43610	95.8	1556	35120	84.8	1618	45040	83.9
60	8	1604	39680	85.3	1825	53290	96	1571	41490	85	1454	38580	83.5
60	9	1508	35560	85.2	1809	43140	95.8	1577	32030	84.8	1569	36570	84.8
60	10	1587	37240	85.7	1835	45620	96	1632	45830	84.8	1665	33700	85
60	11	1677	43420	84.8	1929	50750	95	1718	44380	84.9	1574	38910	85
60	12	1719	43800	85	1922	46350	95.3	1773	50900	85.3	1715	47330	84.2
61	1	1758	42560	84.5	2011	54550	95.5	1750	40850	85.6	1746	39840	83
61	2	1783	42590	84.8	2072	46990	96.3	1864	43790	86	1871	49400	86.1
61	3	1733	39110	85.8	1971	46440	96	1871	48140	86.3	1764	42600	86.4
61	4	1787	45190	86.2	2006	49230	95.5	1799	42470	86.8	1628	32840	87.2
61	5	1726	39220	87.5	1971	48880	98.5	1834	47530	87.2	1803	44600	87.8
61	6	1760	41340	88	1908	40410	99.8	1732	35640	87.5	1769	43940	90.1
61	7	1709	42280	88	1990	58540	100	1656	36910	87.9	1129	14770	88
61	8	1596	37500	88.4	1863	35110	100.2	1607	35270	88.2	1369	34550	88.5
61	9	1669	34250	88.6	1933	49520	101.5	1541	28520	88	1614	31470	88.7
61	10	1612	35030	88.2	1970	48920	100	1618	38570	88	1591	29250	88
61	11	1717	24310	88	2081	52880	100	1687	32140	88	1756	43780	87.8
61	12	1648	33260	88	1965	29000	97.7	1924	37810	87.8	1746	43300	87.7
62	1	1631	17800	88	2082	47830	97.7	1610	32790	87.1	1734	27200	84.8
62	2	1769	33240	88	2173	42040	98	1631	28450	87	1825	46550	87.7
62	3	1730	44460	88	2216	43900	101.5	1930	38830	87	1728	19120	87.2
62	4	1684	16820	88.2	2140	49560	100.5	1762	37480	87	1648	34210	88
62	5	1761	46350	89	2074	33560	101.5	1813	47040	87	1734	48000	88
62	6	1761	42320	89.5	1909	25540	100.4	1745	43080	87	1619	18600	87
62	7	1679	39560	89.2	1898	41010	100.5	1714	44630	87.5	1633	37110	86.3
62	8	1673	41650	88.5	1766	15330	100	1646	38520	87	1654	46970	86.1

Table A.1. (contd)

Yr	Mo	B Reactor			C Reactor			D Reactor			DR Reactor		
		Power (MW)	Prod. (MWh/d)	Flow (Kgpm)	Power (MW)	Prod. (MWh/d)	Flow (Kgpm)	Power (MW)	Prod. (MWh/d)	Flow (Kgpm)	Power (MW)	Prod. (MWh/d)	Flow (Kgpm)
62	9	1642	39950	88.2	1928	41470	100.3	1569	11200	87.7	1587	35800	86.2
62	10	1686	35230	88.9	1878	39700	101	1626	34180	87.3	1472	18750	86.1
62	11	1609	20320	89	2003	44410	99.8	1721	37230	87.3	1726	43060	85.9
62	12	1727	45670	89	1780	5850	96.5	1791	42250	87.8	1676	48220	86
63	1	1841	40060	89.3	2021	51370	97.8	1790	44060	87.8	1646	41330	86.1
63	2	1679	24680	89	2069	41250	97.8	1893	40650	87.8	1721	21200	85.6
63	3	1864	52060	89	2146	41780	97.5	1879	26270	87.3	1658	37310	86.3
63	4	1805	42240	89.3	2062	47320	97.5	1867	47330	87.7	1810	47620	86.9
63	5	1785	48310	89.8	2181	64300	100	1823	53520	87.5	1749	48960	87.4
63	6	1740	28400	90.7	2039	46240	100	1737	24390	87.3	1737	44810	87
63	7	1718	42550	90.5	1978	55000	100	1672	42550	87.8	1691	43090	97
63	8	1698	44950	90.4	1915	48560	100.2	1674	51840	87.5	1657	44540	87.5
63	9	1725	45900	90	1868	34520	100	1627	40170	87.7	1651	45070	87.1
63	10	1748	46760	90.4	2020	58240	97.5	1675	39360	87.8	1707	43970	87.6
63	11	1895	52640	89.5	2018	46980	96	1821	51900	87.8	1781	44610	86.3
63	12	1894	49730	89.9	2134	49680	99.5	1914	53820	88	1813	43730	87.1
64	1	1853	47740	86.5	2201	52540	96.2	1949	48270	88.8	1973	54370	87.6
64	2	1895	43240	89.3	2262	50350	97.8	1999	48860	88.8	1928	46910	87.6
64	3	2030	32530	89.2	2237	59430	96.2	1996	56620	88	1943	48300	87.5
64	4	1853	42860	89.5	2178	54620	96	1823	21060	88.8	1880	38580	87.2
64	5	1808	42370	89.9	2101	52630	99.6	1892	52200	88.8	1835	48470	87.2
64	6	1855	51140	90	2035	49390	99.9	1860	46650	89	1807	47760	87.2
64	7	1786	36150	89.2	2041	24740	99.5	1800	52900	89.2	1762	48400	87.8
64	8	1651	42530	89	1976	52680	100	1748	49310	89	1706	41300	87.8
64	9	1636	38920	88.2	1985	47640	101.3	1743	45540	89	1696	43150	88
64	10	1687	36240	88.9	1980	42170	99.2	1660	23470	88.4	1627	36060	88
64	11	1676	38210	89.1	2092	44310	99.5	1713	36080	89.3	1807	51500	87.5
64	12	1754	45180	87.5	2204	50290	100.1	1901	50890	89	1837	52220	87.1
65	1	1888	47580	87.5	2157	56700	99.8	1812	45440	89	0	0	0
65	2	1947	43560	88.5	2167	49090	99.5	1757	32370	89	0	0	0
65	3	1855	47270	89.1	2175	50840	99.8	1811	51310	86	0	0	0

Table A.1. (contd)

Yr	Mo	B Reactor				C Reactor				D Reactor				DR Reactor			
		Power (MW)	Prod. (MWd)	Flow (Kgpm)		Power (MW)	Prod. (MWd)	Flow (Kgpm)		Power (MW)	Prod. (MWd)	Flow (Kgpm)		Power (MW)	Prod. (MWd)	Flow (Kgpm)	
65	4	1950	51010	89.2		2176	57380	100.6		1774	43690	86.1		0	0	0	0
65	5	1799	43280	90		2052	43070	100.8		1711	29330	86.3		0	0	0	0
65	6	1818	50010	89.5		2060	50060	101.2		1889	54230	90		0	0	0	0
65	7	1710	43840	89.5		1832	37820	101.2		1761	44160	89.9		0	0	0	0
65	8	1712	44690	89.5		1925	51620	102.4		1673	39210	90.2		0	0	0	0
65	9	1740	43740	89		1922	42730	102		1711	46400	90		0	0	0	0
65	10	1745	46200	89.1		1987	27720	102.2		1722	43030	90.4		0	0	0	0
65	11	1836	43510	88.7		2166	56920	108.8		1875	22280	92.2		0	0	0	0
65	12	1974	52870	88.9		2151	42610	107.7		2004	58330	92.2		0	0	0	0
66	1	1961	49970	89.4		2348	66730	105.9		2051	47430	93		0	0	0	0
66	2	1965	42700	89.2		2358	48200	106.7		2103	53760	92.7		0	0	0	0
66	3	2010	53460	90.3		2387	62160	107.3		2102	47960	92.4		0	0	0	0
66	4	1975	53090	87.6		2366	58200	107.5		2003	56970	92.6		0	0	0	0
66	5	1938	60080	89.8		2286	61230	107.8		1977	43760	92.5		0	0	0	0
66	6	1745	32300	90.5		2208	43630	108.4		1953	50740	94.2		0	0	0	0
66	7	1765	12910	90.6		2199	10040	108.1		1730	8690	93.5		0	0	0	0
66	8	1663	10360	90.6		2065	13440	108.5		1804	14990	93.9		0	0	0	0
66	9	1758	40560	90.5		2133	59770	108.9		1759	32770	94		0	0	0	0
66	10	1805	50640	89.9		2204	53220	108.3		1877	58190	94		0	0	0	0
66	11	1855	15970	93.6		2217	53140	104.9		1896	40560	94		0	0	0	0
66	12	1982	55050	94.4		2335	61240	107.4		2050	63550	94.3		0	0	0	0
67	1	2056	48380	95		2417	60620	107.2		2027	47690	93.7		0	0	0	0
67	2	2187	61240	94.3		2410	56680	107.6		2043	49310	93.8		0	0	0	0
67	3	1982	39570	95		2455	56770	107.3		2081	47740	94.4		0	0	0	0
67	4	2125	63750	94.9		2317	58810	104.6		2001	48980	89.8		0	0	0	0
67	5	1985	47630	95.2		2282	70120	104.4		1937	47920	93		0	0	0	0
67	6	1994	59820	94.4		2078	36720	103.7		1924	56620	92.8		0	0	0	0
67	7	1870	49040	95.5		2074	60440	103.5		0	0	0		0	0	0	0
67	8	1626	42090	94.8		2017	62530	103.3		0	0	0		0	0	0	0
67	9	1774	36300	95.8		1958	35300	103.4		0	0	0		0	0	0	0
67	10	1727	42030	95.4		2081	55350	101.7		0	0	0		0	0	0	0

Table A.1. (contd)

Yr	Mo	B Reactor			C Reactor			D Reactor			DR Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
67	11	1855	41460	95.8	2110	41150	102.2	0	0	0	0	0	0
67	12	2071	60800	94.7	2279	52420	105.1	0	0	0	0	0	0
68	1	1613	34450	93.6	2331	43000	105.9	0	0	0	0	0	0
68	2	1674	43920	93.2	2236	34000	106.3	0	0	0	0	0	0
68	3	0	0	0	2418	70160	106.2	0	0	0	0	0	0
68	4	0	0	0	2286	51090	107.7	0	0	0	0	0	0
68	5	0	0	0	2528	68490	107.7	0	0	0	0	0	0
68	6	0	0	0	2176	53400	107.3	0	0	0	0	0	0
68	7	0	0	0	2076	51230	106.2	0	0	0	0	0	0
68	8	0	0	0	1940	47870	106.2	0	0	0	0	0	0
68	9	0	0	0	1935	18920	106.6	0	0	0	0	0	0
68	10	0	0	0	2117	57820	105.7	0	0	0	0	0	0
68	11	0	0	0	2173	45370	105.4	0	0	0	0	0	0
68	12	0	0	0	2225	58080	105	0	0	0	0	0	0
69	1	0	0	0	2196	35260	100.8	0	0	0	0	0	0
69	2	0	0	0	2311	42510	98.9	0	0	0	0	0	0
69	3	0	0	0	2183	51160	99.7	0	0	0	0	0	0
69	4	0	0	0	2069	50150	101.4	0	0	0	0	0	0
69	5	0	0	0	0	0	0	0	0	0	0	0	0
69	6	0	0	0	0	0	0	0	0	0	0	0	0
69	7	0	0	0	0	0	0	0	0	0	0	0	0
69	8	0	0	0	0	0	0	0	0	0	0	0	0
69	9	0	0	0	0	0	0	0	0	0	0	0	0
69	10	0	0	0	0	0	0	0	0	0	0	0	0
69	11	0	0	0	0	0	0	0	0	0	0	0	0
69	12	0	0	0	0	0	0	0	0	0	0	0	0
70	1	0	0	0	0	0	0	0	0	0	0	0	0
70	2	0	0	0	0	0	0	0	0	0	0	0	0
70	3	0	0	0	0	0	0	0	0	0	0	0	0
70	4	0	0	0	0	0	0	0	0	0	0	0	0
70	5	0	0	0	0	0	0	0	0	0	0	0	0

Table A.1. (contd)

Yr	Mo	B Reactor			C Reactor			D Reactor			DR Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
70	6	0	0	0	0	0	0	0	0	0	0	0	0
70	7	0	0	0	0	0	0	0	0	0	0	0	0
70	8	0	0	0	0	0	0	0	0	0	0	0	0
70	9	0	0	0	0	0	0	0	0	0	0	0	0
70	10	0	0	0	0	0	0	0	0	0	0	0	0
70	11	0	0	0	0	0	0	0	0	0	0	0	0
70	12	0	0	0	0	0	0	0	0	0	0	0	0
71	1	0	0	0	0	0	0	0	0	0	0	0	0
71	2	0	0	0	0	0	0	0	0	0	0	0	0
71	3	0	0	0	0	0	0	0	0	0	0	0	0
71	4	0	0	0	0	0	0	0	0	0	0	0	0
71	5	0	0	0	0	0	0	0	0	0	0	0	0
71	6	0	0	0	0	0	0	0	0	0	0	0	0
71	7	0	0	0	0	0	0	0	0	0	0	0	0
71	8	0	0	0	0	0	0	0	0	0	0	0	0
71	9	0	0	0	0	0	0	0	0	0	0	0	0
71	10	0	0	0	0	0	0	0	0	0	0	0	0
71	11	0	0	0	0	0	0	0	0	0	0	0	0
71	12	0	0	0	0	0	0	0	0	0	0	0	0

Table A.2. Hanford Single-Pass Production Reactor Operating History of F, H, KE, and KW Reactors (1944-1971)

Yr	Mo	F Reactor			H Reactor			KE Reactor			KW Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
44	9	0	0	0	0	0	0	0	0	0	0	0	0
44	10	0	0	0	0	0	0	0	0	0	0	0	0
44	11	0	0	0	0	0	0	0	0	0	0	0	0
44	12	0	0	0	0	0	0	0	0	0	0	0	0
45	1	0	0	0	0	0	0	0	0	0	0	0	0
45	2	130	390	32	0	0	0	0	0	0	0	0	0
45	3	233	6960	32	0	0	0	0	0	0	0	0	0
45	4	250	7500	32	0	0	0	0	0	0	0	0	0
45	5	249	7240	32	0	0	0	0	0	0	0	0	0
45	6	258	7530	32	0	0	0	0	0	0	0	0	0
45	7	262	7710	32	0	0	0	0	0	0	0	0	0
45	8	259	7250	32	0	0	0	0	0	0	0	0	0
45	9	223	6180	32	0	0	0	0	0	0	0	0	0
45	10	225	6660	32	0	0	0	0	0	0	0	0	0
45	11	225	6150	32	0	0	0	0	0	0	0	0	0
45	12	235	6980	32	0	0	0	0	0	0	0	0	0
46	1	225	5980	32	0	0	0	0	0	0	0	0	0
46	2	225	5950	32	0	0	0	0	0	0	0	0	0
46	3	211	6200	32	0	0	0	0	0	0	0	0	0
46	4	199	5590	32	0	0	0	0	0	0	0	0	0
46	5	199	5430	32	0	0	0	0	0	0	0	0	0
46	6	199	5490	32	0	0	0	0	0	0	0	0	0
46	7	199	5630	32	0	0	0	0	0	0	0	0	0
46	8	199	5580	32	0	0	0	0	0	0	0	0	0
46	9	199	5480	32	0	0	0	0	0	0	0	0	0
46	10	199	5650	32	0	0	0	0	0	0	0	0	0
46	11	197	4730	32	0	0	0	0	0	0	0	0	0
46	12	199	5730	32	0	0	0	0	0	0	0	0	0
47	1	198	5480	31.07	0	0	0	0	0	0	0	0	0
47	2	199	5250	31.04	0	0	0	0	0	0	0	0	0

Table A.2. (contd)

Yr	Mo	F Reactor				H Reactor				KE Reactor				KW Reactor			
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)
47	3	199	5650	31.37	0	0	0	0	0	0	0	0	0	0	0	0	0
47	4	199	5420	31.22	0	0	0	0	0	0	0	0	0	0	0	0	0
47	5	199	5660	30.4	0	0	0	0	0	0	0	0	0	0	0	0	0
47	6	198	5310	31.47	0	0	0	0	0	0	0	0	0	0	0	0	0
47	7	199	5640	31.78	0	0	0	0	0	0	0	0	0	0	0	0	0
47	8	185	4230	31.19	0	0	0	0	0	0	0	0	0	0	0	0	0
47	9	199	5210	30.42	0	0	0	0	0	0	0	0	0	0	0	0	0
47	10	205	5620	31.15	0	0	0	0	0	0	0	0	0	0	0	0	0
47	11	262	5680	31.34	0	0	0	0	0	0	0	0	0	0	0	0	0
47	12	271	6220	30.64	0	0	0	0	0	0	0	0	0	0	0	0	0
48	1	272	6310	31.05	0	0	0	0	0	0	0	0	0	0	0	0	0
48	2	268	5160	30.08	0	0	0	0	0	0	0	0	0	0	0	0	0
48	3	272	6380	30.94	0	0	0	0	0	0	0	0	0	0	0	0	0
48	4	272	6200	30.65	0	0	0	0	0	0	0	0	0	0	0	0	0
48	5	262	6260	31.23	0	0	0	0	0	0	0	0	0	0	0	0	0
48	6	270	3480	31.23	0	0	0	0	0	0	0	0	0	0	0	0	0
48	7	272	7540	31.65	0	0	0	0	0	0	0	0	0	0	0	0	0
48	8	267	6700	32.35	0	0	0	0	0	0	0	0	0	0	0	0	0
48	9	272	5640	32.48	0	0	0	0	0	0	0	0	0	0	0	0	0
48	10	213	4790	32.34	0	0	0	0	0	0	0	0	0	0	0	0	0
48	11	221	5630	31.98	0	0	0	0	0	0	0	0	0	0	0	0	0
48	12	268	7440	31.42	0	0	0	0	0	0	0	0	0	0	0	0	0
49	1	269	7280	31.32	0	0	0	0	0	0	0	0	0	0	0	0	0
49	2	270	6270	30.97	0	0	0	0	0	0	0	0	0	0	0	0	0
49	3	271	7630	31.09	0	0	0	0	0	0	0	0	0	0	0	0	0
49	4	268	7260	31.29	0	0	0	0	0	0	0	0	0	0	0	0	0
49	5	271	7320	31.3	0	0	0	0	0	0	0	0	0	0	0	0	0
49	6	272	7520	31.48	0	0	0	0	0	0	0	0	0	0	0	0	0
49	7	266	7420	31.38	0	0	0	0	0	0	0	0	0	0	0	0	0
49	8	271	7710	31.36	0	0	0	0	0	0	0	0	0	0	0	0	0
49	9	270	7430	31.42	0	0	0	0	0	0	0	0	0	0	0	0	0

Table A.2. (contd)

Yr	Mo	F Reactor			H Reactor			KE Reactor			KW Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
49	10	271	7580	31.84	53	530	40.1	0	0	0	0	0	0
49	11	272	7460	31.65	267	7650	40.6	0	0	0	0	0	0
49	12	270	7680	31.44	275	8530	40.5	0	0	0	0	0	0
50	1	301	7560	30.89	328	9600	40.1	0	0	0	0	0	0
50	2	301	7640	31.12	329	9170	40.5	0	0	0	0	0	0
50	3	301	8650	30.95	344	9880	40.95	0	0	0	0	0	0
50	4	301	8440	30.76	369	10760	40.7	0	0	0	0	0	0
50	5	299	8670	31.23	398	11230	40.2	0	0	0	0	0	0
50	6	297	8480	30.6	389	9870	39.8	0	0	0	0	0	0
50	7	301	8420	31.05	377	10490	40.8	0	0	0	0	0	0
50	8	297	7710	31.4	389	11290	41.2	0	0	0	0	0	0
50	9	298	8300	31.3	384	9020	40.8	0	0	0	0	0	0
50	10	292	8070	30.81	433	11430	41.4	0	0	0	0	0	0
50	11	301	8740	30.5	445	11990	41.11	0	0	0	0	0	0
50	12	286	7900	30.3	454	12190	40.4	0	0	0	0	0	0
51	1	325	2650	31.5	485	11878	41.6	0	0	0	0	0	0
51	2	359	9258	31.7	469	7932	41	0	0	0	0	0	0
51	3	391	11650	31.01	493	10940	39.48	0	0	0	0	0	0
51	4	394	9930	31.85	490	9450	42.15	0	0	0	0	0	0
51	5	372	10130	32.5	491	9270	42.6	0	0	0	0	0	0
51	6	422	11670	33	489	10150	42.5	0	0	0	0	0	0
51	7	386	7080	33.2	514	12010	42.6	0	0	0	0	0	0
51	8	477	13770	37.63	512	11320	42.76	0	0	0	0	0	0
51	9	436	9900	37.7	538	12350	41.4	0	0	0	0	0	0
51	10	405	9350	37.9	513	15190	41.6	0	0	0	0	0	0
51	11	418	12500	37.7	529	15240	42.5	0	0	0	0	0	0
51	12	394	6550	37.4	545	15590	42.5	0	0	0	0	0	0
52	1	451	11650	36.7	559	16770	42.6	0	0	0	0	0	0
52	2	512	11310	37.5	558	14330	41.1	0	0	0	0	0	0
52	3	496	7900	36.8	559	15350	42.4	0	0	0	0	0	0
52	4	496	12590	36.5	575	15820	42.9	0	0	0	0	0	0

Table A.2. (contd)

Yr	Mo	F Reactor			H Reactor			KE Reactor			KW Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
52	5	509	15760	37	506	12020	43.4	0	0	0	0	0	0
52	6	461	10880	36.8	532	14380	43.4	0	0	0	0	0	0
52	7	486	11300	37.2	554	15580	49.3	0	0	0	0	0	0
52	8	440	9810	37.5	583	16900	45.7	0	0	0	0	0	0
52	9	488	12440	38.1	591	16540	45.8	0	0	0	0	0	0
52	10	494	13480	37.7	604	17770	45.8	0	0	0	0	0	0
52	11	519	7880	37.5	597	15900	46	0	0	0	0	0	0
52	12	342	920	0	624	17890	47.3	0	0	0	0	0	0
53	1	448	8250	38.3	617	17710	47.7	0	0	0	0	0	0
53	2	543	13930	38.4	655	17350	47.8	0	0	0	0	0	0
53	3	548	14780	38.7	641	19140	47.3	0	0	0	0	0	0
53	4	562	15360	39	660	3070	0	0	0	0	0	0	0
53	5	595	17120	39.3	586	11950	49.7	0	0	0	0	0	0
53	6	576	9490	39.2	634	16660	51	0	0	0	0	0	0
53	7	584	16290	38.9	762	22420	52.6	0	0	0	0	0	0
53	8	562	16080	38.5	732	20600	52.5	0	0	0	0	0	0
53	9	581	15230	39.2	638	12440	52.5	0	0	0	0	0	0
53	10	631	18700	39.2	707	18190	52.7	0	0	0	0	0	0
53	11	633	16500	40	795	21540	52.2	0	0	0	0	0	0
53	12	658	17910	39.3	869	25160	52.7	0	0	0	0	0	0
54	1	704	16870	38.8	754	16760	52.8	0	0	0	0	0	0
54	2	628	12310	45.2	889	22300	52.5	0	0	0	0	0	0
54	3	713	17400	45.5	877	22050	52.5	0	0	0	0	0	0
54	4	737	18200	45.6	891	22480	52.4	0	0	0	0	0	0
54	5	701	14910	45	797	16480	52.3	0	0	0	0	0	0
54	6	718	19170	45.9	899	25780	52.6	0	0	0	0	0	0
54	7	781	23270	45.5	908	25310	52.5	0	0	0	0	0	0
54	8	755	20830	46.3	857	23300	52.5	0	0	0	0	0	0
54	9	760	20020	46.3	851	24180	52.1	0	0	0	0	0	0
54	10	816	24260	46.4	703	12550	52	0	0	0	0	0	0
54	11	757	15940	46.2	863	24470	51.9	0	0	0	0	0	0

Table A.2. (contd)

Yr	Mo	F Reactor			H Reactor			KE Reactor			KW Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
54	12	897	22750	46.6	925	26750	51.5	0	0	0	0	0	0
55	1	933	25130	46.5	952	22130	52.5	0	0	0	0	0	0
55	2	952	3650	0	944	21300	52.2	0	0	0	0	0	0
55	3	864	25610	45.9	959	23340	52.6	0	0	0	861	24930	124
55	4	566	10950	52.9	920	25360	0	168	3220	126.3	734	17020	124.3
55	5	627	13160	47.6	724	17460	52.9	924	25690	128.2	903	24720	127.1
55	6	446	6770	46.8	715	16560	53	999	28140	129.3	970	25640	127.3
55	7	836	17390	47.3	931	26610	53.5	1086	33530	128.6	1055	27150	128.7
55	8	801	15370	47.2	937	16060	57.1	1006	25920	128.5	1036	23730	130
55	9	747	8700	46.9	965	20320	57.1	1068	28770	126.4	1099	18730	128.8
55	10	820	11640	47.3	1020	20080	56.9	1240	24520	136.7	1386	34460	127.4
55	11	784	11150	45.9	1071	27340	56.7	1537	39610	125.3	1592	37110	127.9
55	12	867	18710	47	915	18410	56.1	1846	51850	125.5	1763	49240	125
56	1	829	9300	45.9	1042	20670	55.7	1882	43760	124	1814	44710	122.5
56	2	888	11260	47	957	18620	55.4	1766	42770	139	1849	47370	136
56	3	781	11740	46.9	1052	25310	55.6	1930	53850	140	1963	52760	134
56	4	834	9880	47	1119	23530	56.4	1979	52300	140	1823	46100	138
56	5	834	14970	46.7	1015	19820	56	1887	53820	138	1789	46700	140
56	6	858	11270	43.8	942	17300	56	2038	53930	141	1875	49840	136
56	7	772	12540	47.6	955	18150	57.2	2146	55420	143	1894	53140	146
56	8	790	18880	47.4	1010	22360	56.7	1986	49620	145.5	2069	55610	147
56	9	793	17200	47	982	22270	56.4	1811	38790	149	2202	54430	147
56	10	805	18770	48	1034	29170	56.5	2261	59790	149	2337	63390	149
56	11	875	23260	48	1079	28100	56.3	2464	63720	149.5	2448	61540	148.5
56	12	914	20740	47.7	1045	23680	56	2294	40960	147	2573	70350	148
57	1	972	20490	47.8	1112	28300	56.1	2586	64530	146	2513	62710	147
57	2	1008	23480	48.2	1161	27570	56.1	2686	63250	148	2529	60120	150
57	3	1010	22480	47.4	1153	25810	56.9	2628	58660	149.5	2637	73490	149
57	4	0	0	0	1140	26950	56.1	2678	68850	145.5	2824	71080	150.5
57	5	1058	13250	72.9	981	7690	0	2478	61300	150	2663	72320	150.5
57	6	1222	27500	73	0	0	0	2603	62630	161	2848	72620	155.5

Table A.2. (contd)

Yr	Mo	F Reactor				H Reactor				KE Reactor				KW Reactor			
		Power (kW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
57	7	1222	34280	73.2	1124	22200	69.4	2603	64880			162	2848	66830			165
57	8	1207	28590	72.5	1186	29490	69	1576	22330			165.5	2706	68530			165.5
57	9	1276	30620	71.5	1198	27820	70.3	2300	58650			164	2511	62370			162
57	10	1246	21940	70.2	1237	30260	71.4	2794	73540			153.5	2874	76440			164
57	11	1246	30690	71.2	1237	31280	69	2794	68980			152	2874	69580			152.5
57	12	1260	29920	71.5	1245	32300	70.3	2632	62090			153	2918	74180			155.5
58	1	1297	28710	71.4	1299	34710	70.3	2765	64200			152.5	2847	69630			154
58	2	1329	33230	72.4	1335	31550	70.8	2877	63400			153.5	2018	42720			154.5
58	3	1342	31910	72.2	1331	33300	71.1	2790	58550			158	2454	53560			155.5
58	4	1351	34130	72.5	1374	18840	71.5	2550	56460			173	2796	76410			157
58	5	1348	34060	73	1386	37250	71.5	2462	55490			157.5	2589	66530			172.5
58	6	1392	36160	73	1363	34350	74.1	2853	78910			157	2814	74800			172.5
58	7	1357	36140	73.7	1326	34820	74.5	2707	78040			147	2536	63360			173.5
58	8	1321	33620	75.5	1280	34920	74.9	2711	67320			174	2573	69390			173
58	9	1227	28790	75.1	1273	30020	76.8	2838	72710			175.5	2804	73180			172.5
58	10	1317	39770	75.2	1265	29800	79.3	2508	59790			172	2900	82440			172
58	11	1252	24680	80.5	1365	38370	82.9	2757	75760			172.5	3090	7230			21.9
58	12	1439	44610	80	1440	44640	80.6	2859	74540			172.5	2385	59810			171
59	1	1362	32090	81.2	1368	29940	81.4	2922	82070			171.5	2824	75030			171.5
59	2	1498	41940	81.2	1494	34930	81.2	3189	82510			172.5	3038	78600			171.5
59	3	1415	28690	81.4	1599	49570	81.8	2775	67530			171.5	3119	84600			171
59	4	1509	39290	82.3	1530	38280	82.5	2747	37330			157.5	3148	79800			173.5
59	5	1399	23160	82.3	1480	37390	83	2493	41110			174.5	2752	69100			176.5
59	6	1356	33850	82.2	1428	34440	85.2	3040	80990			175	2882	76950			178
59	7	1483	37150	82.2	1593	49380	85.4	2739	64530			175	2939	77720			178.5
59	8	1462	39750	85.4	1448	35730	85.3	3044	78790			174.5	3193	90770			178
59	9	1401	29630	83.6	1543	38610	85.8	3142	83140			174.5	3300	87120			178
59	10	1460	37790	83.4	1581	41320	84.5	3153	80990			175.5	3302	81790			178.5
59	11	1274	35850	83.6	1631	41690	83.4	3317	92350			174	3415	93130			179
59	12	1610	40280	83.4	1636	45140	83.2	3325	82670			176	3406	95130			179.5
60	1	1624	39820	83.3	1684	43020	83.5	3549	93850			178	3507	93170			179.5

Table A.2. (contd)

Yr	Mo	F Reactor			H Reactor			KB Reactor			KW Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
60	2	1699	32730	83.5	1708	37830	84	3372	73930	180	3598	83110	179.5
60	3	1590	36870	82.4	1761	43240	81.2	3289	86870	180.5	3540	90320	180
60	4	1553	39280	83.4	1759	40690	87.5	3378	81980	181	3560	93240	189.5
60	5	1604	44300	86.1	1765	43220	86.8	3574	103370	182	3327	93240	180
60	6	1531	38170	86.5	1675	31860	86.6	3589	98950	183.5	3553	100730	180
60	7	1550	33590	86.9	1622	38870	87.2	3368	82060	180.5	3413	91200	189
60	8	1575	42670	87.3	1631	43840	87.2	3169	72110	180.5	3343	87050	182
60	9	1613	43840	86.1	1596	41660	88	3515	99650	180	3493	92950	181.5
60	10	1605	33530	86.8	1564	35150	87.2	3620	96620	180.5	3447	92860	181.5
60	11	1723	43730	86.8	1673	40850	87.1	3742	93850	189.5	3701	100370	182
60	12	1760	45500	86.7	1834	52590	86.7	3497	81200	180.5	3872	103590	182
61	1	1775	44190	86.7	1709	34700	86.2	3802	100420	181	3900	109170	181.5
61	2	1755	35430	84.4	1659	29450	86.3	3505	83030	181	3930	103220	182.5
61	3	1800	38220	88.3	1836	46840	86.4	3818	85930	182	3810	87400	182.5
61	4	1860	48880	89.1	1752	43890	86.4	3793	91830	183	3867	99300	183.5
61	5	1759	38390	90.3	1616	27850	86.2	3851	102790	182.5	3821	103880	184
61	6	1742	40500	89.9	1655	38680	87.8	3804	98830	183.5	3807	95370	184.5
61	7	1657	26510	89.9	1645	47780	87.1	3647	98700	183.5	3641	101700	185
61	8	1401	38830	90	1554	35500	86.3	3558	93750	184	3506	86190	185.5
61	9	1663	33780	90.1	1442	22280	86.3	3508	87140	184.5	3629	95700	184.5
61	10	1738	36910	90.3	1605	40750	86	3745	94270	185.5	3752	103750	185.5
61	11	1799	38640	89.9	1570	30760	86	3816	102350	184.5	3973	107990	185
61	12	1822	42530	90.3	1710	40130	84.9	3870	93580	184	4023	114610	184.5
62	1	1756	47300	90.8	1613	27350	85.3	4248	113520	188	3977	90490	185
62	2	1611	16280	90.7	1723	26390	86.1	4326	110830	188.8	3917	90920	185.5
62	3	1249	34110	88.7	1746	44110	85.7	4298	104460	187.6	4222	118450	186.5
62	4	1790	43660	89.5	1698	24910	86.3	4155	102090	188.6	4085	101100	188
62	5	1762	27370	88.9	1693	21570	86.1	3988	103110	193.7	4051	108750	188
62	6	1724	35220	89.3	1689	30200	86.1	4087	106430	200.8	3983	105870	188.5
62	7	1647	40280	89.4	1652	43170	86.3	3800	74330	200.9	3620	64190	198.5
62	8	1656	33730	89.1	1573	37010	85.7	4007	105710	201.7	3934	107930	201.5

Table A.2. (contd)

Yr	Mo	F Reactor			H Reactor			KE Reactor			KW Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
62	9	1715	47690	89.9	1562	22260	85.6	3917	95300	206	3909	98150	204.5
62	10	1713	39720	89.9	1612	25940	85.6	4269	125060	206.5	3997	75340	204
62	11	1695	38750	90.7	1692	37920	85.5	3867	48140	206	4297	115630	205
62	12	1763	47330	90.3	1635	40240	85.3	4188	102300	205	4297	108160	205.5
63	1	1726	27070	90.3	1489	12920	0	4162	108380	206	4349	110960	205.5
63	2	1843	46390	90.1	1805	48620	84.8	3989	69470	205	4253	70850	204
63	3	1835	47560	89.5	1861	53590	84.3	4215	107670	205	4353	128060	204
63	4	1792	50640	89.3	1827	47580	84.2	4336	120970	205.5	4206	98930	206
63	5	1791	48410	89.9	1687	36970	84.6	4399	48680	205	3804	35850	207
63	6	1751	38820	90.2	1719	51570	84.5	4201	95530	205.8	4337	130110	206.5
63	7	1773	51200	97	1652	44910	83.7	3991	94150	206	3957	89910	206.5
63	8	1727	48130	87.5	1573	43250	92.5	4023	80320	206.2	3555	21270	210
63	9	1666	36390	87.1	1554	27180	84.6	3682	52140	206	4144	124320	208
63	10	1755	47710	87.6	1663	48200	84.7	4111	30080	205	3957	98990	208
63	11	1778	40700	86.3	1782	53460	84.9	4304	118790	205.5	3870	31230	208
63	12	1805	43090	89.3	1771	46890	84.5	4148	97730	206	4194	107390	208
64	1	1906	30670	89.1	1840	19390	93.3	4351	91850	205.5	4393	136180	207.5
64	2	1900	46120	88.8	1967	51660	93.3	4102	49500	207.5	4396	51080	207.5
64	3	1954	40160	89.1	2042	58930	93.3	4402	136460	206	4314	94550	208
64	4	1749	37520	89.6	1709	22250	93.3	4189	85960	207	4362	116600	209.5
64	5	1843	46160	89	1949	58300	93.3	4330	127250	207	4369	114850	209.5
64	6	1841	47830	89	1864	53740	93.2	4305	98020	208.5	4329	106230	209.5
64	7	1741	47930	89.3	1851	43380	93.3	4277	119060	209	4250	108300	209.5
64	8	1732	49130	89.9	1832	56790	92.8	4235	78380	209	4127	104780	208.5
64	9	1681	43570	89.9	1756	40090	92.6	4102	76910	210	4270	122080	210
64	10	1768	47680	89.9	1802	49270	93.9	4265	112780	209	3963	40420	210
64	11	1812	44580	89.8	1794	46340	93.2	4347	110850	210	4320	109380	209
64	12	1895	48290	87.1	1934	51200	93.3	4331	115060	209	4169	108560	206
65	1	1987	61600	88.7	2033	57100	92.9	4332	99380	208.2	3936	81020	204
65	2	1872	38470	89.5	1927	41650	92.4	4351	107330	208	4375	98490	203
65	3	1973	51560	90.2	1893	52640	93.1	4365	116100	207	4192	108120	205

Table A.2. (contd)

Yr	Mo	F Reactor			H Reactor			KE Reactor			KW Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)	Power (MW)	Prod. (MWd)	Flow (Kgpm)
65	4	1877	49160	90.1	1719	45020	93.2	4219	86070	207	4221	96620	206.5
65	5	1817	50020	89.8	0	0	0	4325	98950	207.5	4336	116670	206.5
65	6	1737	44920	89.8	0	0	0	4370	106450	208	4174	96540	208
65	7	0	0	0	0	0	0	4314	126910	208.5	4148	80110	209.8
65	8	0	0	0	0	0	0	4098	96290	209.5	4028	95270	209
65	9	0	0	0	0	0	0	4223	95780	211.5	4165	92340	208.5
65	10	0	0	0	0	0	0	4274	108510	210	4296	122790	211
65	11	0	0	0	0	0	0	4251	104060	210	4096	74830	209.5
65	12	0	0	0	0	0	0	4371	111650	208	4358	100650	209.5
66	1	0	0	0	0	0	0	4284	101990	208	4225	93910	209.5
66	2	0	0	0	0	0	0	4263	80330	213	4349	84390	214
66	3	0	0	0	0	0	0	4380	129940	213.5	4383	121330	214
66	4	0	0	0	0	0	0	4352	108890	213.5	4287	54270	214
66	5	0	0	0	0	0	0	4362	106010	214	4240	107650	214.5
66	6	0	0	0	0	0	0	4373	117810	213	4194	80400	215.3
66	7	0	0	0	0	0	0	4395	27790	213	4394	33780	215
66	8	0	0	0	0	0	0	3757	17120	213.5	4213	30690	215
66	9	0	0	0	0	0	0	4188	116850	213	4307	92770	216
66	10	0	0	0	0	0	0	4322	59620	212	4211	105480	198
66	11	0	0	0	0	0	0	4373	111640	212	4365	111700	215.5
66	12	0	0	0	0	0	0	4360	115020	212.5	4302	93750	215.5
67	1	0	0	0	0	0	0	4354	105820	212	4387	135590	215
67	2	0	0	0	0	0	0	4380	95780	213.5	4305	96310	216.5
67	3	0	0	0	0	0	0	4303	107110	197.5	4295	106380	216.5
67	4	0	0	0	0	0	0	4394	131820	197.5	4363	108770	216.5
67	5	0	0	0	0	0	0	4278	87000	214.5	4188	81920	217
67	6	0	0	0	0	0	0	4369	114160	213	4373	114270	216.5
67	7	0	0	0	0	0	0	4262	105040	212.5	4310	114900	216.5
67	8	0	0	0	0	0	0	4239	131410	214	4211	102080	216.5
67	9	0	0	0	0	0	0	4145	96000	214	4207	105640	216.5
67	10	0	0	0	0	0	0	4223	94390	214.5	4301	96260	215.5

Table A.2. (contd)

Yr	Mo	F Reactor			H Reactor			KE Reactor			KW Reactor		
		Power (MW)	Prod. (MWd)	Flow (Kkgpm)	Power (MW)	Prod. (MWd)	Flow (Kkgpm)	Power (MW)	Prod. (MWd)	Flow (Kkgpm)	Power (MW)	Prod. (MWd)	Flow (Kkgpm)
67	11	0	0	0	0	0	0	4311	111610	196	4400	132000	215
67	12	0	0	0	0	0	0	4355	113670	194.5	4349	100840	215.5
68	1	0	0	0	0	0	0	4299	91020	196	4152	97180	194.5
68	2	0	0	0	0	0	0	4340	105110	193	4256	100090	192.5
68	3	0	0	0	0	0	0	4316	98070	196	4385	111470	193.5
68	4	0	0	0	0	0	0	4355	122680	194.5	4172	94250	195
68	5	0	0	0	0	0	0	4250	102630	195	4135	100110	212
68	6	0	0	0	0	0	0	4346	114340	207.5	4259	30280	212
68	7	0	0	0	0	0	0	4183	104390	207.5	3486	10050	0
68	8	0	0	0	0	0	0	3933	71690	207.5	4070	112420	210.5
68	9	0	0	0	0	0	0	3717	90770	194.5	3942	118260	196.5
68	10	0	0	0	0	0	0	4039	125210	193	3857	72340	194.5
68	11	0	0	0	0	0	0	3963	100940	192	3845	102890	193.5
68	12	0	0	0	0	0	0	3935	97830	190.5	3934	101710	193.5
69	1	0	0	0	0	0	0	3908	90620	190	3818	100250	192
69	2	0	0	0	0	0	0	3879	89820	188	3738	75790	191.5
69	3	0	0	0	0	0	0	3792	99450	188	3652	84340	191
69	4	0	0	0	0	0	0	3369	26280	190	3953	118590	191
69	5	0	0	0	0	0	0	3851	99210	190	3757	59980	192
69	6	0	0	0	0	0	0	3902	106640	190	3777	86910	193.5
69	7	0	0	0	0	0	0	3753	50910	190	3653	90250	192.5
69	8	0	0	0	0	0	0	3512	74900	191.5	3729	100220	192.5
69	9	0	0	0	0	0	0	3651	90800	190.5	3475	68390	195
69	10	0	0	0	0	0	0	3615	63990	190.5	3757	74890	194.5
69	11	0	0	0	0	0	0	3442	51840	190	3646	85210	203.5
69	12	0	0	0	0	0	0	3741	83730	192.5	3911	121240	203
70	1	0	0	0	0	0	0	3944	109430	199.5	3946	94680	202.5
70	2	0	0	0	0	0	0	3920	1210	199.5	0	0	0
70	3	0	0	0	0	0	0	3390	23440	199.5	0	0	0
70	4	0	0	0	0	0	0	3903	97070	199.5	0	0	0
70	5	0	0	0	0	0	0	3778	83970	200	0	0	0

Table A.2. (contd)

Yr	Mo	F Reactor			H Reactor			KE Reactor			KW Reactor		
		Power (MW)	Prod. (MW/d)	Flow (Kgpm)	Power (MW)	Prod. (MW/d)	Flow (Kgpm)	Power (MW)	Prod. (MW/d)	Flow (Kgpm)	Power (MW)	Prod. (MW/d)	Flow (Kgpm)
70	6	0	0	0	0	0	0	3777	81470	200	0	0	0
70	7	0	0	0	0	0	0	3454	73350	200.5	0	0	0
70	8	0	0	0	0	0	0	3653	48920	202.5	0	0	0
70	9	0	0	0	0	0	0	3519	82980	202	0	0	0
70	10	0	0	0	0	0	0	3700	90730	202	0	0	0
70	11	0	0	0	0	0	0	3819	86160	201.5	0	0	0
70	12	0	0	0	0	0	0	3866	91560	202	0	0	0
71	1	0	0	0	0	0	0	3929	78680	202	0	0	0
71	2	0	0	0	0	0	0	0	0	0	0	0	0
71	3	0	0	0	0	0	0	0	0	0	0	0	0
71	4	0	0	0	0	0	0	0	0	0	0	0	0
71	5	0	0	0	0	0	0	0	0	0	0	0	9
71	6	0	0	0	0	0	0	0	0	0	0	0	0
71	7	0	0	0	0	0	0	0	0	0	0	0	0
71	8	0	0	0	0	0	0	0	0	0	0	0	0
71	9	0	0	0	0	0	0	0	0	0	0	0	0
71	10	0	0	0	0	0	0	0	0	0	0	0	0
71	11	0	0	0	0	0	0	0	0	0	0	0	0
71	12	0	0	0	0	0	0	0	0	0	0	0	0

Appendix B

**Monthly Releases from Eight
Single-Pass Hanford Production Reactors**

Table B.1. Total Monthly Releases of Na-24, P-32, Sc-46, Cr-51, Mn-56, and Zn-65 from Eight Single-Pass Hanford Production Reactors (Ci/month)^(a)

Yr	Mo	Na24 Min	Na24 Med	Na24 Max	P32 Min	P32 Med	P32 Max	Sc46 Min	Sc46 Med	Sc46 Max	Cr51 Min	Cr51 Med	Cr51 Max	Mn56 Min	Mn56 Med	Mn56 Max	Zn65 Min	Zn65 Med	Zn65 Max
44	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	3	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	7	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	8	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	9	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	10	6.43	122	1670	21.1	51.7	117	1.3	15	157	243	971	5210	537	4320	17900	53	179	533
44	11	8.96	122	2460	28.1	59.9	166	0.945	13.1	135	261	1160	5760	770	6120	26500	71.8	184	700
44	12	170	661	3400	51.2	104	195	5.09	29.3	206	687	2080	5840	1010	8090	28300	153	345	904
45	1	470	1600	7580	84.1	181	288	9.31	70.1	413	1440	3690	9260	3540	12700	53900	305	580	1230
45	2	597	1800	5080	97.1	163	337	20.6	99.6	410	1150	3600	15100	1070	5900	28900	226	553	1040
45	3	706	2840	6390	174	287	458	62.5	230	531	1390	3960	11000	2410	14400	39200	512	996	2220
45	4	1040	2910	8720	171	283	471	74.9	240	553	1060	4520	12500	707	5810	27000	613	972	2000
45	5	1170	3640	8370	139	257	447	85.3	245	611	1160	4070	16000	1840	8200	32000	622	996	2840
45	6	1770	4300	10200	112	221	502	85.5	201	536	1200	4410	10800	613	4840	36900	575	1030	1600
45	7	1340	3920	8530	135	236	525	126	234	754	1220	4900	12600	1520	9719	40800	538	1020	1780
45	8	1180	3430	8120	130	258	439	114	266	504	894	3190	13500	1740	8360	39300	506	860	1420
45	9	682	2730	8770	139	250	416	82.1	268	590	663	2460	9340	1560	6540	40700	483	910	1590
45	10	1100	2870	8310	139	259	501	124	270	587	591	2990	8910	1270	5940	30900	460	922	1760
45	11	990	2760	7440	137	276	396	88.3	212	506	1190	3890	16900	1320	5740	26300	579	896	1640
45	12	862	2100	8470	131	219	420	99.6	197	482	970	3240	9140	2150	7790	38300	381	783	1420
46	1	641	2510	7910	130	253	431	51	196	571	1140	4300	11100	12500	39700	103000	455	920	1670
46	2	653	2580	7620	148	270	426	69.9	189	385	1220	3930	10600	12000	38800	92000	508	932	1490
46	3	785	2550	6730	140	249	461	86	199	548	807	3040	11800	4500	22300	70400	430	884	1550
46	4	694	2260	6120	87.9	168	320	66.5	161	427	324	2040	7690	3.12	256	20200	280	690	1410
46	5	643	2630	10400	79.9	147	337	58.2	145	385	394	1620	7210	457	5890	47600	309	686	1210
46	6	963	2860	5230	53.1	124	307	45.7	157	481	292	1330	7570	1560	7610	41900	307	682	1130

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide.

2

of

2

Table B.1. (contd)

Yr	Mo	Na24 Min	Na24 Med	Na24 Max	P32 Min	P32 Med	P32 Max	So46 Min	So46 Med	So46 Max	Cr51 Min	Cr51 Med	Cr51 Max	Mn56 Min	Mn56 Med	Mn56 Max	Zn65 Min	Zn65 Med	Zn65 Max
46	7	882	2760	7780	74.6	144	345	49.5	158	409	325	2080	7130	307	3530	36900	347	666	1480
46	8	653	2330	6180	77.7	156	336	54.5	169	445	308	2170	9060	135	1730	17700	296	644	1080
46	9	680	2110	6190	78	163	385	77.5	191	418	350	2260	11400	9.94	972	14600	346	654	1150
46	10	505	1950	6580	74.2	172	355	63.8	177	440	543	2190	7970	1.91	93	8280	296	588	1170
46	11	478	1820	5620	86.1	164	294	63.9	149	427	360	1900	8470	2040	10400	64300	304	632	1290
46	12	742	2000	7950	96.7	180	436	60.2	167	403	354	2230	8430	2850	11000	50300	347	710	1430
47	1	417	2040	9150	99.2	185	347	64.2	162	386	569	1610	5560	2100	10100	60000	298	678	1300
47	2	390	2170	5760	84.9	172	285	49	141	468	385	1910	6120	1440	9070	56100	309	611	1180
47	3	680	1970	8900	102	175	385	67.2	159	468	427	1760	6460	1770	12400	68100	331	648	1580
47	4	663	2050	5850	84.3	168	336	59	146	383	500	2040	7830	6.72	353	10000	294	666	1130
47	5	766	2540	6580	76.5	147	321	58	157	456	368	1800	8550	442	7900	51600	410	718	1240
47	6	883	2580	6890	69.3	121	312	65.7	155	393	320	1930	6630	970	7590	33600	357	634	1220
47	7	796	2740	8710	87	164	296	65.6	166	485	308	2220	8280	466	4490	31800	419	666	1310
47	8	581	1880	5750	68	147	284	59.4	153	463	22.2	413	6020	81.4	1890	31600	291	561	1200
47	9	570	1930	5600	69.5	153	291	64.6	165	431	58.5	502	6740	32.9	801	19700	272	587	996
47	10	678	2060	6380	67.5	160	311	64.9	166	436	22.5	400	5570	58	1080	16400	284	624	1230
47	11	484	1620	4330	54.6	134	272	50.7	134	343	362	1700	6890	66.3	934	17300	231	528	1080
47	12	577	1870	6130	63.9	134	266	54.7	135	289	470	1500	6420	7.77	441	12600	264	536	1140
48	1	588	2040	7260	73	146	307	59.4	143	290	323	1960	6790	6.98	206	9540	285	590	1270
48	2	438	1680	5890	51.2	122	246	34.2	113	360	326	1530	7450	2.57	147	4920	169	465	869
48	3	621	2120	7000	81.5	154	295	50.2	142	364	456	1920	10000	5.72	382	7870	290	545	1000
48	4	716	1840	6800	66.2	131	252	67.3	146	439	437	1740	7570	22.4	652	12700	253	508	921
48	5	690	2480	6280	60.2	127	256	47.5	141	430	324	1350	7560	811	5200	59500	259	554	1130
48	6	1020	2720	6350	32.3	75.9	251	31.1	94.1	294	294	1490	6420	2660	12800	62100	248	500	995
48	7	1800	3880	9980	125	224	392	72	243	627	1420	3870	10300	1210	7450	36900	543	942	1590
48	8	1750	4120	11000	124	235	471	89.1	232	597	826	2400	8370	5260	20500	57500	490	868	1400
48	9	1080	2900	8600	103	224	356	90.1	201	416	648	2500	11600	2470	14700	53500	346	745	1530
48	10	1190	3720	8620	155	275	455	81.2	222	656	1790	5240	13400	2080	10000	58400	520	941	1570
48	11	1080	3060	7130	104	241	435	96.4	238	604	457	1790	7240	2240	12300	59800	491	782	1360
48	12	1410	3330	7530	156	264	478	93.5	246	575	1100	4390	9840	2340	13800	62800	522	922	1520
49	1	1430	3540	7840	135	266	444	93.3	237	505	1770	4560	10700	2050	11900	56700	440	948	1650
49	2	1300	2740	9280	119	226	400	66.4	198	438	1240	3630	18000	2070	9270	48700	457	766	1370
49	3	1190	3490	7740	135	272	488	81.9	224	497	1150	3970	12000	2560	12300	72700	530	969	1700

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide.

Table B.1. (contd)

Yr	Mo	Na24 Min	Na24 Med	Na24 Max	P32 Min	P32 Med	P32 Max	So46 Min	So46 Med	So46 Max	Cr51 Min	Cr51 Med	Cr51 Max	Mn56 Min	Mn56 Med	Mn56 Max	Zn65 Min	Zn65 Med	Zn65 Max
49	4	1220	3560	8110	133	241	441	107	214	465	1360	3810	10600	3100	13200	71900	584	883	1600
49	5	1910	4350	13500	112	217	414	77	214	620	1510	3900	10300	1070	9160	44900	566	966	1580
49	6	1880	4660	13000	97.7	206	391	78.7	217	591	1190	4180	9880	1260	9700	49500	505	940	1790
49	7	1500	4020	8270	126	233	472	122	270	562	477	2390	8830	4280	18700	58700	526	890	1520
49	8	1600	3620	8370	143	233	423	125	274	562	1040	2990	11200	4420	18100	79100	452	940	1620
49	9	1590	3270	9090	117	236	442	111	256	719	710	2810	12900	2310	16200	61700	438	821	1750
49	10	1140	4280	10900	165	304	541	133	279	592	1520	4140	14200	5840	20100	81500	552	1040	1630
49	11	1460	4420	9790	206	351	569	117	268	585	1970	5450	14800	9810	32400	91400	664	1160	1860
49	12	1580	4920	12800	216	390	668	148	310	780	3230	7690	18500	9120	33600	89900	684	1390	2230
50	1	2200	4950	13100	232	375	613	90.9	284	678	3070	7460	21900	11600	30100	83300	600	1180	2400
50	2	1890	4580	12700	191	349	574	79.8	248	606	2320	6490	13400	10700	30600	87700	675	1100	2350
50	3	2040	5210	11000	212	365	677	116	286	596	2830	7020	15400	9210	27600	99600	795	1240	2080
50	4	2300	5660	9900	181	338	541	112	298	627	1720	7100	15200	4040	14900	61800	690	1210	2020
50	5	3340	6440	11100	171	315	552	143	290	671	2760	6930	14800	6040	23300	72200	629	1280	2160
50	6	2320	6500	14900	130	248	502	110	276	697	2280	6000	13200	8320	22700	95300	671	1200	2110
50	7	3410	7030	12100	123	282	493	158	326	694	2650	7530	16700	8400	26400	81300	788	1280	1950
50	8	2200	5440	13200	146	291	575	159	311	618	815	3170	10100	1060	8590	59900	588	1040	1710
50	9	2320	5010	12400	175	291	542	117	308	538	164	1280	8760	4480	18200	79700	577	996	1740
50	10	3460	7230	13400	173	300	472	173	362	910	784	2850	10500	5840	16900	71500	707	1180	2470
50	11	3520	7990	14600	277	414	744	161	388	840	674	3440	17500	4720	24700	76200	748	1370	2810
50	12	4980	6350	10800	258	459	732	193	385	846	1950	6820	17800	8960	12100	41800	1070	1470	2550
51	1	4790	6550	11400	213	333	485	59.5	202	692	2830	6110	16800	10200	21700	70100	454	821	1490
51	2	8300	10100	15100	139	268	422	33.5	138	468	3710	8130	20600	10600	27100	96000	370	769	1550
51	3	8170	10980	15900	209	319	495	45.4	155	463	2410	6650	13300	10200	18400	60500	399	819	1620
51	4	5530	8200	13500	123	240	391	41.7	172	491	3640	7850	16100	12700	17800	46000	322	890	1950
51	5	5310	6670	11500	70.6	130	253	62.6	172	578	2760	8470	15500	13900	19700	59300	448	914	1930
51	6	6120	7890	9220	153	241	400	99.2	265	633	2700	7530	18100	14600	18900	24900	487	1050	1880
51	7	5630	7380	11500	96.4	226	423	86.9	248	533	1180	4100	14500	13100	17400	23300	669	1190	2180
51	8	6820	8810	13500	181	308	537	96.3	261	615	1360	5990	17600	14600	19600	61700	438	910	1870
51	9	5930	7930	13100	194	335	608	164	345	806	1730	6970	14300	9380	16300	80700	721	1210	2090
51	10	5470	8460	15000	205	322	518	38.2	164	518	3650	8300	18300	9180	20100	62200	429	881	1880
51	11	6490	8130	10100	177	292	478	65.5	237	531	2230	6770	17200	13200	19100	25800	545	998	1790
51	12	6450	8000	10200	201	319	447	32.3	136	512	3750	9560	19500	10400	16200	23100	343	701	1530

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide.

Table B.1. (contd)

Yr	Mo	Na24 Min	Na24 Med	Na24 Max	P32 Min	P32 Med	P32 Max	So46 Min	So46 Med	So46 Max	Cr51 Min	Cr51 Med	Cr51 Max	Mn56 Min	Mn56 Med	Mn56 Max	Zn65 Min	Zn65 Med	Zn65 Max
52	1	5680	7680	12200	212	345	634	145	302	622	1850	5890	16900	7870	13600	54900	504	1210	2280
52	2	6890	8810	10400	268	435	675	10.5	52.1	331	5030	10200	22100	10400	13000	17700	114	418	1140
52	3	8610	10200	12300	211	327	492	49.9	178	511	2520	6190	14700	20700	27700	34600	393	883	1900
52	4	9630	11800	14000	187	269	408	17.4	83	366	4640	10300	24500	29800	39400	56300	139	528	1530
52	5	8890	11200	14100	170	256	360	23.4	126	419	5010	10200	21600	21500	29000	44100	385	812	2110
52	6	7520	8800	11200	154	238	362	15.1	86.1	400	5240	11500	26600	18400	24900	37700	187	555	1390
52	7	7810	11100	15900	200	306	487	105	237	676	5000	12300	21400	13100	21600	50300	397	844	1940
52	8	7880	9800	11600	176	261	344	17.3	95.7	289	5160	11700	23300	15700	20200	26600	193	558	1260
52	9	12000	14000	17100	281	396	574	24.3	124	598	4560	11900	23400	14800	19500	26800	220	510	1980
52	10	10000	13700	19600	526	744	1050	97	259	643	6430	12900	30400	26300	37900	88000	364	875	1810
52	11	9740	11500	15400	437	671	1050	27.8	104	522	5380	13200	25900	18000	31700	104000	274	724	1580
52	12	10200	14100	18900	549	802	1170	45.5	152	716	5040	11100	25700	28500	38500	51200	453	1120	2550
53	1	10700	12800	15300	752	1120	1570	41.1	129	755	4350	10400	27900	27900	38000	52300	286	858	1890
53	2	11200	14400	17900	527	808	1190	18.5	141	518	4180	10500	25300	15900	22700	37100	331	834	2240
53	3	15900	18600	23100	645	861	1180	19.8	115	363	4320	11000	28900	29200	38100	47600	301	771	1870
53	4	14400	16000	18500	599	788	1010	27	146	409	5460	13800	33300	65000	92300	127000	32.4	164	736
53	5	16600	20100	26000	554	778	1200	51.5	203	693	7080	14000	27800	104000	131000	183000	276	660	1800
53	6	12900	15000	17700	407	555	752	32	125	520	7780	17200	32000	77600	102000	133000	90.6	379	1080
53	7	11300	13300	15300	274	390	558	47.2	200	673	6990	16300	30200	60000	78800	106000	300	801	2460
53	8	13400	16900	19400	326	528	700	32.1	211	812	9030	17900	42500	69000	90700	119000	284	931	1920
53	9	11600	14100	17900	262	415	551	124	298	762	6410	13700	32400	77400	101000	128000	359	886	2180
53	10	15500	19800	25300	370	521	823	81.1	233	679	8110	15400	29100	56900	80800	125000	388	1000	2420
53	11	16700	19900	23300	645	894	1210	86.2	260	894	7920	17000	34700	69800	96300	130000	313	746	1980
53	12	17300	21600	25000	752	1030	1440	61	192	612	8250	16500	35100	69000	90600	125000	183	661	1660
54	1	9510	18400	30100	469	690	1090	248	507	964	3490	11500	28800	14800	45400	171000	952	1810	2950
54	2	8330	13400	23200	249	561	900	230	422	857	3040	10500	24500	8780	36400	131000	848	1630	2660
54	3	12000	17400	28600	404	626	1110	181	481	833	5220	11200	29400	19800	51300	160000	939	1700	3060
54	4	10100	17700	28000	289	629	925	260	509	927	5950	12200	37500	17000	47500	161000	750	1700	3190
54	5	11300	20600	31800	281	520	1020	187	471	931	3190	9510	22700	41100	77900	218000	1050	1880	4180
54	6	15900	23200	34200	231	495	854	202	512	1170	2410	9850	35100	54800	106000	199000	1170	2170	3670
54	7	14600	24200	34400	269	531	844	323	629	1090	3680	10600	25500	60200	104000	266000	1130	2050	3610
54	8	14300	23200	35100	363	615	1130	308	672	1170	5650	16600	30300	37100	85500	165000	1080	1950	3550
54	9	12700	21600	30300	380	661	943	337	628	1300	6370	16600	39300	30700	72200	181000	867	1820	3560

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide.

Table B.1. (contd)

Yr	Mo	Na24 Min	Na24 Med	Na24 Max	P32 Min	P32 Med	P32 Max	So46 Min	So46 Med	So46 Max	Cr51 Min	Cr51 Med	Cr51 Max	Mn56 Min	Mn56 Med	Mn56 Max	Zn65 Min	Zn65 Med	Zn65 Max
54	10	11700	20300	34700	388	631	1070	330	561	1000	6130	15300	30300	19200	62700	207000	1070	1810	3080
54	11	12100	19800	28000	327	642	1010	323	537	1140	4340	13300	25000	16200	59900	139000	888	1600	3210
54	12	13400	21600	32100	398	660	1020	285	616	1130	8040	16500	31800	33400	66200	152000	1070	1790	2940
55	1	14400	22000	33500	385	631	1160	374	636	1040	6320	16700	34700	39500	80100	228000	878	1680	2790
55	2	10300	17000	26200	266	441	839	119	258	634	5640	10000	21700	18900	46200	121000	557	1150	2520
55	3	20600	35600	56600	477	697	1170	329	643	1170	11400	24100	52000	61000	115000	354000	1620	2470	4880
55	4	16500	30700	52600	314	568	959	193	444	840	12900	23600	46600	55000	162000	392000	1030	1970	3600
55	5	10500	15400	30800	381	626	986	255	577	1300	17300	31700	64900	90400	201000	438000	1840	2760	4730
55	6	12700	22000	43800	206	467	898	243	439	855	9550	25700	64100	46600	111000	533000	1530	3010	4700
55	7	26600	31700	37800	278	411	582	80.6	215	707	15200	29800	59900	62800	120000	518000	1120	1970	3900
55	8	20600	24200	39200	302	389	599	69.2	171	577	17100	31900	57600	80400	173000	528000	1040	2080	3550
55	9	22000	25000	29900	284	409	665	89	207	452	11400	23700	48200	61200	151000	421000	822	1520	3100
55	10	16700	28300	51500	436	745	1250	288	558	1010	8100	17500	36700	67700	143000	352000	1450	2340	3940
55	11	22000	32300	53500	516	810	1470	297	558	1120	10100	19600	32900	31300	77400	303000	1490	2730	4270
55	12	24800	33800	56100	595	990	1560	353	563	1270	10100	22300	42900	32200	91300	321000	1740	2990	5270
56	1	16300	23200	34600	389	539	893	200	449	981	9150	17500	34800	36000	78200	245000	1860	2750	4390
56	2	14300	20100	32800	411	601	900	234	440	931	5840	13900	28700	22900	67600	331000	1250	2410	3960
56	3	22500	31700	44800	744	1230	2410	310	590	1440	8000	19300	34200	52900	93800	347000	1890	2920	4650
56	4	22300	32000	49500	538	789	1140	272	615	1350	7640	17400	33300	77600	132000	548000	1850	2960	4750
56	5	22200	31700	46100	424	613	826	307	570	1130	5020	14600	35400	77300	189000	582000	2070	3250	5080
56	6	38700	50100	67000	379	550	833	271	596	1300	5920	15700	40900	160000	276000	559000	2400	3340	5600
56	7	32000	46800	70800	256	385	547	294	606	1050	10800	20700	45500	124000	220000	649000	2080	2580	4950
56	8	28100	44700	64100	272	386	589	396	727	1790	14400	25900	42800	48000	136000	542000	1780	2850	5390
56	9	25600	37000	61300	330	485	746	308	662	1280	10600	25300	49500	51900	123000	530000	1540	2790	4620
56	10	21100	26900	37600	330	459	739	184	507	1360	8840	26700	48500	74200	103000	165000	898	1950	4210
56	11	23700	30200	40200	612	765	1290	181	418	1140	10500	28600	56300	53800	94900	400000	921	2080	5870
56	12	26100	33400	41600	618	920	1220	154	342	775	14000	26800	60500	66500	98300	378000	833	1660	3530
57	1	35800	41300	49300	973	1220	1700	308	666	1600	11900	26000	60100	78900	108000	132000	940	1750	3280
57	2	34700	39200	47500	851	1110	1430	197	414	1050	12800	24900	57100	111000	137000	180000	645	1190	2480
57	3	44400	51700	58800	778	976	1170	139	361	945	7460	21000	41700	287000	351000	444000	812	1520	2920
57	4	46800	55400	66500	816	1120	1520	227	545	1650	19600	35500	60100	690000	920000	1160000	682	1900	3130
57	5	47000	55800	64800	634	894	1200	167	445	1360	15400	27600	54500	733000	1040000	1490000	1130	2170	3820
57	6	37700	48600	67200	468	638	843	70.3	294	961	8110	19700	46700	291000	437000	826000	844	1680	4060

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide.

Table B.1. (contd)

Yr	Mo	Na24 Min	Na24 Med	Na24 Max	P32 Min	P32 Med	P32 Max	So46 Min	So46 Med	So46 Max	Cr51 Min	Cr51 Med	Cr51 Max	Mn56 Min	Mn56 Med	Mn56 Max	Zn65 Min	Zn65 Med	Zn65 Max
57	7	43200	58200	86700	334	556	913	289	557	1180	18100	36400	60100	150000	333000	881000	1520	2890	4550
57	8	35400	54300	75600	609	1050	1840	298	603	1270	18500	38700	78600	150000	286000	647000	1360	2360	4760
57	9	36000	53700	77100	591	878	1330	197	559	1640	20000	34900	61400	101000	210000	720000	1440	2610	4850
57	10	47000	64300	97500	694	978	1480	238	645	1450	22300	41400	68500	130000	264000	750000	1500	3090	5220
57	11	42400	63600	88700	669	975	1580	213	592	1600	25000	40700	58800	152000	282000	894000	1820	3260	5970
57	12	36200	58600	88400	1290	1930	3140	258	653	1260	18900	34500	66300	121000	234000	726000	1930	3140	5210
58	1	45500	65000	92200	1120	1760	2580	264	645	1340	23800	37800	50000	130000	255000	687000	1350	2130	3340
58	2	35600	51000	75300	958	1440	1970	207	502	1160	22700	31800	43500	88000	215000	610000	1650	2440	3670
58	3	45800	58000	79100	753	1190	1790	206	476	804	20900	29500	40300	118000	271000	679000	1200	1950	3290
58	4	39000	56900	81200	1470	2590	5210	269	602	1340	22300	33000	45600	121000	241000	747000	2030	3390	6660
58	5	44000	62900	89300	910	1560	2280	207	491	990	20800	31300	42000	199000	367000	1060000	1790	2830	4530
58	6	52200	81000	105000	969	1380	2250	192	498	1360	24600	36200	48500	341000	653000	1410000	921	1580	3180
58	7	49400	67300	101000	1090	2070	4130	180	543	1190	19500	26600	38300	187000	393000	1070000	1100	1880	3110
58	8	51600	58400	65300	802	1140	1680	282	506	956	34900	49600	65400	202000	249000	294000	1560	2140	2940
58	9	57400	62900	70900	761	954	1140	188	313	535	24500	33600	51300	457000	586000	719000	823	1010	1250
58	10	61300	68800	78600	1090	1310	1670	339	522	816	24800	34200	43800	322000	384000	460000	2580	3240	4710
58	11	45000	54100	60700	1010	1310	1770	338	620	1170	19600	28300	39400	172000	213000	264000	1590	2210	3080
58	12	59600	65100	73900	1480	1780	2200	154	213	352	25500	33900	47600	226000	254000	288000	1860	2380	3290
59	1	50200	57800	66000	1390	1900	2290	152	228	338	26200	33700	51800	169000	203000	239000	1060	1400	1960
59	2	44800	50300	57900	1530	1840	2230	96.9	149	194	22400	27200	35500	242000	292000	327000	1750	2180	2580
59	3	75800	87300	96100	1600	1970	2400	164	233	358	17600	26000	35000	325000	404000	482000	2200	2670	3210
59	4	87300	95200	109000	1320	1570	1990	240	331	470	27000	32800	40400	693000	791000	934000	2070	2500	3350
59	5	75300	89100	105000	915	1230	1500	102	149	186	35200	45000	64500	491000	632000	812000	1660	2430	3460
59	6	83500	91700	103000	801	963	1140	172	226	320	28500	38400	52800	341000	419000	484000	1790	2230	2780
59	7	69800	79400	91700	710	1020	1590	396	645	1140	32300	41100	51600	356000	462000	557000	2080	2580	3470
59	8	94800	109000	122000	797	976	1180	413	617	952	16800	32200	64800	356000	430000	499000	1500	1890	2520
59	9	75400	81500	93100	805	984	1250	311	444	735	23200	30500	41300	316000	392000	455000	3230	4020	5310
59	10	58500	66600	72600	1410	1760	2140	124	158	215	30700	40100	48000	338000	394000	452000	1060	1470	1830
59	11	90400	99100	113000	1660	2240	2860	298	463	724	22500	30700	40200	286000	328000	379000	3010	3780	5060
59	12	95400	112000	123000	1240	1540	2030	654	1110	2050	30200	40700	48600	264000	321000	375000	3440	4880	6580
60	1	102000	117000	133000	1780	2230	2720	1280	2130	3790	30700	40300	52800	250000	291000	357000	6110	8460	12000
60	2	99500	110000	132000	1220	1480	1820	96.6	134	173	29900	38600	49600	308000	395000	481000	1590	2070	2620
60	3	120000	137000	149000	1710	2010	2620	285	417	656	38300	48400	61900	708000	836000	1050000	2050	2730	3590

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide.

Table B.1. (contd)

Yr	Mo	Na24 Min	Na24 Med	Na24 Max	P32 Min	P32 Med	P32 Max	So46 Min	So46 Med	So46 Max	Cs137 Min	Cs137 Med	Cs137 Max	Mn56 Min	Mn56 Med	Mn56 Max	Zn65 Min	Zn65 Med	Zn65 Max
60	4	135000	150000	167000	1510	1820	2420	368	844	2110	40000	53500	80000	1070000	1240000	1480000	3700	4570	5990
60	5	117000	131000	149000	1250	1620	2000	388	538	835	38100	52800	75300	522000	619000	758000	3330	4250	5240
60	6	80800	92900	105000	992	1320	1710	370	734	1580	43600	59200	74600	419000	492000	585000	2930	3940	5910
60	7	73000	81700	92900	1070	1290	1760	548	914	1540	34400	45200	60100	417000	495000	606000	2710	3740	5190
60	8	94000	104000	119000	951	1160	1480	100	170	892	16300	21300	32100	501000	585000	674000	1020	1530	3560
60	9	96100	107000	123000	796	1000	1310	544	858	1300	29400	40000	56300	607000	705000	868000	2170	2820	3510
60	10	71800	85300	111000	849	1140	1960	574	998	1730	45600	64600	94900	172000	259000	705000	1630	2470	4570
60	11	144000	165000	193000	1420	1940	2610	552	1160	2940	52400	82300	136000	446000	573000	777000	1630	2420	5080
60	12	90000	102000	113000	1930	2480	3040	102	172	287	32700	46600	68500	568000	672000	806000	3020	3720	5060
61	1	124000	139000	155000	2320	3020	3660	438	576	753	58500	73400	96800	654000	766000	923000	4270	5250	6280
61	2	77900	86100	95000	2550	3190	4090	154	211	290	29500	37600	48600	428000	527000	693000	2540	3380	4320
61	3	124000	141000	158000	2410	3090	3680	1080	1710	2860	31100	41100	58200	842000	997000	1280000	5450	6570	8150
61	4	97900	110000	125000	2330	2940	3450	300	500	733	32400	46100	63000	869000	1010000	1180000	6220	8140	10800
61	5	102000	117000	138000	1960	2500	3040	77.3	130	208	35000	42700	49700	583000	680000	798000	3540	4290	5350
61	6	101000	112000	124000	1460	1710	2080	10.6	13.3	19.3	34500	41800	48900	414000	506000	573000	4590	5480	7250
61	7	73600	91100	109000	795	896	1000	9.2	12.9	18.6	28000	33700	43400	369000	469000	590000	1450	1740	2120
61	8	56000	64600	74000	464	521	590	27.7	45.3	63.8	40500	52300	61000	255000	312000	391000	2980	3660	5060
61	9	29300	34300	39400	700	807	929	22.8	37.2	55.9	52100	63500	77900	264000	336000	411000	2190	2750	3380
61	10	52400	60700	67700	796	937	1040	14.7	21	35.2	51300	64000	81000	228000	266000	304000	1350	1720	2050
61	11	60700	69600	77800	916	1050	1170	13.7	28.4	219	59500	70700	88400	219000	280000	368000	1910	2620	3960
61	12	61600	70900	81800	766	865	1010	9.85	27.7	310	54900	67500	91900	183000	221000	269000	1150	1510	2570
62	1	74200	81900	94300	926	1080	1210	58	89.1	141	101000	124000	151000	211000	247000	295000	3330	4290	5350
62	2	76700	86100	103000	1010	1160	1310	49.1	72.2	100	58300	75000	93700	260000	312000	373000	2220	2920	4100
62	3	89800	102000	119000	1320	1530	1730	36.7	56	79.8	64800	81100	94000	574000	682000	798000	2470	3270	4790
62	4	122000	141000	160000	1410	1570	1770	68.8	95.9	145	90700	110000	151000	1500000	1780000	2110000	5100	7650	9800
62	5	98400	116000	140000	1510	1670	1850	155	310	1180	88000	125000	185000	562000	786000	1170000	3320	4310	7220
62	6	101000	118000	142000	1010	1130	1290	40.8	70.5	127	47700	62200	82800	926000	1100000	1470000	4100	5440	6750
62	7	74200	82900	93800	757	845	991	94.9	145	212	78200	99000	120000	522000	616000	733000	4740	6480	8640
62	8	76500	90000	106000	1060	1190	1350	452	802	1270	51300	69500	87300	438000	580000	721000	3980	5640	7380
62	9	73400	84800	98500	982	1120	1260	461	946	1580	83200	104000	136000	358000	483000	583000	4460	6460	9340
62	10	61500	71900	84000	761	838	908	99.2	373	1410	70100	92000	120000	316000	371000	433000	814	2240	5540
62	11	48000	56200	64100	688	822	964	116	398	1070	53900	68800	83900	526000	632000	840000	2080	2910	4270
62	12	55300	63400	72400	803	890	1030	309	630	1290	64100	77000	96200	304000	376000	430000	3370	4400	6070

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide.

Table B.1. (contd)

Yr	M ₀	Na24 Min	Na24 Med	Na24 Max	P32 Min	P32 Med	P32 Max	Sc46 Min	Sc46 Med	Sc46 Max	Cs51 Min	Cs51 Med	Cs51 Max	Mn56 Min	Mn56 Med	Mn56 Max	Zn65 Min	Zn65 Med	Zn65 Max
63	1	52500	63100	76800	773	867	994	372	631	1580	25000	30400	40600	156000	181000	241000	1760	2310	3440
63	2	50400	60700	70100	603	702	800	126	443	799	32300	41200	58200	232000	278000	345000	1350	1980	2560
63	3	84000	106000	132000	1110	1270	1420	290	677	1290	44200	47200	53100	471000	614000	833000	679	1140	1940
63	4	93500	119000	150000	1330	1480	1700	238	623	1120	48200	57400	59400	474000	612000	840000	1700	2250	3030
63	5	47300	58900	71200	920	1020	1130	135	335	1040	35300	38800	41900	441000	589000	794000	1140	1410	1850
63	6	62200	75500	102000	912	1010	1110	219	558	1360	38200	42300	46700	434000	534000	773000	684	969	1340
63	7	61000	71900	86400	796	915	1030	219	520	1200	45500	49500	53300	598000	762000	1000000	635	845	1160
63	8	49900	63900	79300	658	733	808	123	425	1020	59100	64400	69800	410000	559000	803000	612	787	1210
63	9	53200	67600	85400	667	741	816	199	560	1150	47400	53200	57600	320000	430000	568000	434	635	1130
63	10	48900	56200	69800	821	893	1020	192	540	1210	55300	59900	64900	269000	348000	463000	676	868	1240
63	11	57200	63400	75900	826	907	1020	193	488	1320	49600	53500	57400	228000	300000	420000	619	956	1590
63	12	63100	81700	103000	1080	1200	1330	317	612	1190	54100	59600	63600	266000	379000	562000	615	770	926
64	1	65700	80700	100000	1100	1190	1390	253	602	1370	49100	53000	57900	228000	326000	433000	718	1030	1760
64	2	50500	59500	71800	1030	1150	1250	164	414	1090	43000	47700	54700	214000	263000	336000	908	1250	1590
64	3	52700	72500	88500	1440	1560	1670	221	630	1370	43300	48400	53400	271000	367000	444000	1080	1400	1790
64	4	57700	75400	94500	1070	1230	1360	334	639	1280	34500	37600	42500	195000	265000	365000	1320	1830	2470
64	5	74600	102000	125000	1380	1520	1670	259	744	1220	50600	55500	61000	436000	712000	1570000	1550	2170	2790
64	6	90500	112000	133000	889	1000	1100	287	657	1300	45400	49800	54700	682000	1080000	1730000	1160	1640	2190
64	7	63800	86300	109000	611	664	737	423	791	1450	27800	29800	32400	488000	854000	1480000	563	749	1030
64	8	54700	72200	102000	468	516	564	276	629	1360	34000	37600	40600	218000	501000	1100000	965	1390	1860
64	9	50500	68300	91000	680	751	822	361	736	1230	39200	43400	49400	181000	352000	1120000	888	1360	1890
64	10	41400	55500	75500	581	630	680	281	631	1220	30300	33500	37100	161000	351000	1140000	518	721	979
64	11	63800	84500	114000	949	1050	1170	163	518	1240	46000	49400	55100	72800	220000	578000	885	1050	1410
64	12	65700	91100	130000	982	1050	1140	222	478	936	45500	50900	56000	246000	480000	1140000	854	1120	1660
65	1	47200	71500	89800	1160	1250	1360	254	556	1270	47600	53000	60300	113000	273000	568000	1020	1330	1840
65	2	46700	65700	96900	848	951	1040	197	465	999	30900	34600	39000	133000	347000	909000	917	1460	1980
65	3	51200	75100	111000	1340	1470	1610	284	676	1590	31400	36000	39200	157000	335000	932000	1350	1900	2460
65	4	58000	75300	94100	1450	1630	1780	390	656	1200	31300	34900	40500	238000	539000	1220000	1130	1580	2130
65	5	58400	73900	91700	1160	1310	1460	326	674	1280	26100	29100	33300	403000	812000	1480000	1220	1760	2320
65	6	58200	82400	113000	768	826	932	269	635	1440	28900	32400	35400	429000	793000	1490000	1330	1630	2210
65	7	33500	54400	88500	480	543	619	281	611	1090	22500	26900	31600	293000	595000	1250000	528	814	1980
65	8	35500	51000	83300	556	631	736	405	669	1190	28700	31500	37600	228000	520000	1370000	503	662	888
65	9	27900	50600	70300	638	731	833	425	676	1590	27000	32200	37100	256000	525000	983000	501	633	813

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide.

Table B.1. (contd)

Yr	Mo	Ni24 Min	Ni24 Med	Ni24 Max	P32 Min	P32 Med	P32 Max	So46 Min	So46 Med	So46 Max	Cs137 Min	Cs137 Med	Cs137 Max	Mn56 Min	Mn56 Med	Mn56 Max	Zn65 Min	Zn65 Med	Zn65 Max
65	10	38600	56300	91200	888	1020	1170	619	953	1570	23000	25300	27700	355000	749000	1300000	278	351	433
65	11	34200	49900	74200	696	831	963	405	656	1260	23100	26100	30600	230000	458000	1200000	378	492	692
65	12	41000	58400	77100	805	933	1060	402	741	1360	24100	28200	32400	227000	418000	1150000	611	767	951
66	1	40000	58500	81100	732	860	969	266	602	1140	26200	29000	33800	154000	302000	917000	555	689	874
66	2	36900	51600	75500	544	610	689	314	569	1120	18400	21400	24100	135000	264000	762000	435	551	732
66	3	40800	60300	91700	891	1010	1090	521	827	1500	22800	25600	28500	159000	408000	893000	626	895	1120
66	4	39400	53200	76500	679	793	882	294	525	969	22900	25400	28600	167000	370000	748000	736	923	1130
66	5	41800	64600	86000	692	769	877	409	778	1440	23600	27100	29900	232000	545000	1420000	840	1140	1510
66	6	43700	59100	88300	512	578	646	361	700	1090	19000	21400	23900	306000	614000	1460000	467	607	843
66	7	13600	18500	25900	85.2	113	139	52.7	147	410	2750	3240	3960	94600	188000	408000	92	160	225
66	8	16700	23700	32500	77.9	100	120	22.7	87.8	292	3000	3480	4060	36000	79700	304000	107	161	276
66	9	43600	62200	93400	640	721	816	224	533	1040	22000	24400	27100	132000	308000	973000	790	1030	1330
66	10	36100	49700	69700	526	600	650	222	457	927	21000	23600	26500	171000	375000	933000	1060	1400	1790
66	11	26800	46900	74600	488	547	647	325	618	1240	21700	24500	28200	63400	181000	666000	671	940	1280
66	12	42200	64700	91800	577	664	705	338	641	1240	24200	27100	30900	158000	366000	832000	912	1160	1460
67	1	50900	61000	81000	613	683	743	554	946	1600	24400	27400	30600	222000	268000	638000	1200	1570	1930
67	2	46400	53800	71900	636	734	816	357	676	1180	21700	24100	27600	193000	241000	394000	1140	1420	1910
67	3	47200	57600	70100	611	683	785	365	678	1060	17700	20400	23700	210000	301000	885000	1430	1910	2710
67	4	58800	70000	92200	1080	1190	1340	491	779	1360	24500	27400	30000	489000	592000	1050000	1380	2060	2920
67	5	51300	62500	82600	1150	1290	1450	377	693	1330	23800	27200	30400	486000	622000	1210000	949	1270	1670
67	6	49500	62000	83300	965	1070	1210	539	924	1500	32400	35700	39000	590000	816000	1500000	1080	1480	1890
67	7	45700	55800	74400	719	824	925	357	736	1380	22800	25900	29500	534000	777000	1240000	888	1360	1810
67	8	49400	53200	57600	815	889	964	447	765	1460	23800	26400	29700	475000	528000	509000	648	854	1210
67	9	42700	47400	51000	666	766	888	340	660	1110	17500	20000	22300	329000	378000	430000	718	1010	1400
67	10	44400	48200	51100	652	735	823	333	638	1190	19200	21000	23500	280000	326000	358000	542	703	926
67	11	42800	46500	50000	523	568	628	301	587	1220	18200	20600	23500	224000	249000	287000	567	783	1010
67	12	49000	51900	55900	608	684	775	211	534	1300	19800	22100	25000	198000	226000	251000	660	940	1230
68	1	43000	45700	49600	561	621	686	145	398	911	14000	15700	18100	140000	163000	186000	418	585	848
68	2	50800	53800	57300	885	1020	1220	43.8	194	659	13700	15400	17300	180000	218000	240000	522	714	1000
68	3	38400	53000	76400	827	1010	1180	131	354	1200	14300	16100	17900	266000	518000	1120000	785	984	1360
68	4	32400	47900	67400	1050	1230	1420	82.4	251	569	10200	12000	14900	252000	546000	1170000	677	950	1400
68	5	32400	48400	77400	1020	1180	1320	109	299	954	13400	15400	17600	185000	483000	1130000	557	793	1070
68	6	21700	36000	60900	390	453	523	146	288	742	7840	9360	11200	263000	512000	1030000	351	574	919

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide.

Table B.1. (contd)

Yr	Mo	Na24 Min	Na24 Med	Na24 Max	P32 Min	P32 Med	P32 Max	So46 Min	So46 Med	So46 Max	Cr51 Min	Cr51 Med	Cr51 Max	Mn56 Min	Mn56 Med	Mn56 Max	Zn65 Min	Zn65 Med	Zn65 Max
68	7	14100	27700	47800	321	372	437	133	280	713	8870	10200	11400	126000	322000	936000	341	507	846
68	8	24700	39700	62300	575	655	729	233	462	1080	12100	13300	15000	131000	366000	899000	491	664	857
68	9	20000	33500	62000	506	580	663	81.7	248	761	7990	9180	10400	98800	335000	1110000	331	472	1070
68	10	23100	36300	65300	463	531	620	71.7	267	657	10100	11300	12600	84000	296000	821000	298	408	579
68	11	23400	38000	63100	389	452	512	112	295	897	8980	10400	12400	46100	244000	994000	419	584	848
68	12	22200	39500	63000	450	528	641	80.6	253	981	9390	10600	12400	53700	204000	714000	341	499	641
69	1	21000	35500	55200	431	490	561	88.3	250	581	6780	7660	8950	15500	78400	547000	440	615	849
69	2	20200	32500	48400	526	614	714	56.2	147	578	6080	6970	8280	81700	238000	752000	451	684	926
69	3	23900	38700	67300	713	824	940	408	511	613	7270	8280	9380	79600	245000	782000	592	820	1200
69	4	28100	41500	57000	688	788	945	403	521	659	8020	9090	10400	1380000	1870000	2550000	564	827	1110
69	5	21800	33200	52500	292	366	455	343	470	625	5600	6590	7470	1150000	1330000	1520000	314	548	873
69	6	22400	35500	53900	296	368	433	420	552	691	6570	7680	8560	576000	662000	758000	257	448	684
69	7	16000	26700	41000	330	398	482	288	361	450	4240	5180	6060	167000	376000	864000	324	442	616
69	8	15000	28100	53000	393	502	576	277	346	439	4970	5920	6840	71000	266000	851000	362	545	808
69	9	12800	24000	38900	314	371	425	220	282	354	4120	4840	5610	56500	212000	1090000	180	285	395
69	10	12700	21600	38800	285	352	419	129	171	222	3480	4140	4810	40000	197000	845000	256	364	482
69	11	9080	17200	31600	151	189	234	270	356	469	3480	4050	4830	21500	99500	497000	232	350	525
69	12	13900	24700	40200	181	216	266	460	604	744	4710	5330	6570	2410	44600	591000	413	523	721
70	1	20000	37200	71400	61.9	144	270	176	369	777	2660	4740	7900	70400	247000	603000	119	251	717
70	2	82.2	181	471	0.912	3.01	13.1	0.0496	0.6	13	25.8	103	614	211	1230	6670	0.662	3.96	32.1
70	3	1720	4340	11500	14.2	28.3	66.9	12.6	49.8	195	189	414	966	5090	25500	116000	18.2	91.7	907
70	4	6470	15400	40600	65.3	150	375	35	169	534	763	1540	3670	22600	83000	434000	57.9	344	2770
70	5	6480	15200	37300	23.5	57.2	148	43	161	477	846	1750	3760	17000	80500	580000	43.6	358	2860
70	6	6860	14800	37300	11.9	31.4	81	31.6	128	433	853	1950	4780	14900	79400	422000	43.6	310	2530
70	7	5350	14100	33900	59.6	212	776	2.48	35.3	719	1810	7080	32900	15300	86000	308000	39	335	2430
70	8	3590	9620	22600	39.6	151	565	1.44	24.2	600	1090	4780	23100	15000	54400	316000	31.3	243	1730
70	9	6380	18500	39500	70.9	246	1080	2.41	59.2	694	1710	8080	48000	24000	98300	464000	56.6	346	2290
70	10	6750	15600	42900	74.9	216	761	2.64	48.8	625	2140	8960	39100	21600	110000	518000	43	450	2840
70	11	6280	16200	39600	58.6	244	976	2.02	48.6	697	1570	8720	41200	16300	77700	493000	54.9	328	3110
70	12	6050	16900	42600	63.8	276	972	3.61	41	966	1770	7180	43100	18000	83400	455000	44.2	333	3070
71	1	5320	13200	32900	52.6	235	852	2.04	31.1	833	2060	7040	39300	16100	85400	444000	42.5	386	2460
71	2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	3	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide.

Table B.1. (contd)

Yr	Mo	Na24 Min	Na24 Med	Na24 Max	P32 Min	P32 Med	P32 Max	So46 Min	So46 Med	So46 Max	Cr51 Min	Cr51 Med	Cr51 Max	Mn56 Min	Mn56 Med	Mn56 Max	Zn65 Min	Zn65 Med	Zn65 Max
71	4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	7	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	8	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	9	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	10	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	12	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

B.11

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide.

Table B.2. Total Monthly Releases of Ga-72, As-76, Y-90, I-131, Np-239, and Gross Beta from Eight Single-Pass Hanford Production Reactors (Ci/month)^(a)

Yr	Mo	Ga72 Min	Ga72 Med	Ga72 Max	As76 Min	As76 Med	As76 Max	Y90 Min	Y90 Med	Y90 Max	I131 Min	I131 Med	I131 Max	Np239 Min	Np239 Med	Np239 Max	GrossBeta Min	GrossBeta Med	GrossBeta Max
44	1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	3	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	7	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	8	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	9	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
44	10	517	1060	2640	62.9	268	1380	14.7	78.1	422	0.00615	0.333	17.1	3380	4260	5340	3390	10400	36100
44	11	626	1440	3070	49.5	353	1710	16.6	83.8	530	0.00914	0.453	13.3	4250	5200	6550	3420	13400	38000
44	12	1230	2320	4100	178	554	1990	53.5	138	536	0.0929	1.05	14.1	6570	7600	9200	4140	10500	39300
45	1	2060	3950	6750	261	1140	2720	80.1	258	898	0.119	2.16	45.9	10600	12400	15000	17.8	670	17900
45	2	2560	4110	7400	366	1140	3070	57.6	265	791	0.167	2.14	36.8	10200	11900	14000	948	5130	28200
45	3	3820	6330	8620	745	1960	4770	111	331	819	0.568	5.8	41.8	16200	18300	20100	536	5050	51100
45	4	4640	6610	10100	574	1960	4300	100	312	1110	0.473	5.19	63	15300	17800	19900	1520	8180	40900
45	5	5360	7340	10500	568	1400	5660	89.8	340	979	0.157	2.25	61.4	15400	17800	20800	1550	9260	66900
45	6	5400	7950	10900	238	1160	3580	51.5	269	1020	2.21	11.1	78.2	14600	16600	18600	2830	15100	53600
45	7	5610	7620	10400	519	1670	4530	79.6	301	993	0.105	1.96	29.4	14100	16800	19200	3400	16200	61000
45	8	5080	6940	9930	663	2260	4200	60.1	206	748	0.201	3.55	30	14300	16000	18700	3810	18000	47200
45	9	4230	6680	9200	1050	2030	3780	33.8	199	855	0.376	4.11	58.7	13700	16300	19100	2360	12800	41200
45	10	4150	6580	10400	906	2380	4380	43.8	196	936	0.323	4.68	61.4	15200	17000	20100	2830	10400	48600
45	11	4080	6080	8630	761	1850	4430	130	338	1020	0.445	5.14	47.1	14000	17000	18700	1480	7310	36800
45	12	3200	4960	7600	564	1390	3720	41.1	264	819	0.241	6.12	49.4	12500	14200	16700	507	6130	29600
46	1	4140	6340	9800	714	1680	4060	100	305	935	0.29	5.12	47.1	15800	18000	20300	203	3140	26400
46	2	3660	6400	8960	687	1720	4040	66.6	321	1160	0.268	4.64	48.9	15000	17200	19500	210	2190	20600
46	3	3650	6260	9020	432	1600	3350	55.9	239	702	0.419	5.5	66.9	14700	16200	18800	3000	12700	52200
46	4	2800	4760	7430	282	1020	3120	22.2	148	628	0.0819	3.19	49	9940	11900	14300	272	4300	37200
46	5	3760	5700	9190	89.6	615	2970	43.1	297	774	0.267	5.26	58.5	9670	11600	13900	628	4940	37600
46	6	3550	5580	8860	88.9	433	1910	65.2	224	841	0.287	3.31	38.7	8700	10600	12600	1080	5040	38500

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide and gross beta.

Table B.2. (contd)

Yr	Mo	Cs72 Min	Cs72 Med	Cs72 Max	As76 Min	As76 Med	As76 Max	Y90 Min	Y90 Med	Y90 Max	I131 Min	I131 Med	I131 Max	Np239 Min	Np239 Med	Np239 Max	GrossBeta Min	GrossBeta Med	GrossBeta Max
46	7	3890	5470	7700	186	784	3330	24.6	124	641	0.0134	0.651	23.8	9350	11400	13300	1220	6700	50900
46	8	3300	5360	7950	422	1290	3370	21.5	134	568	0.0738	1.62	39.3	9250	11200	13600	1660	7650	51500
46	9	3220	4860	7780	480	1360	3610	23.4	194	881	0.175	2.68	54.9	9640	11000	13200	1790	7810	34200
46	10	2640	4420	7260	578	1240	3510	27.3	133	534	0.193	3.14	50	9110	10900	13300	545	4630	32000
46	11	2700	4530	7750	533	1220	3040	29.7	150	970	0.176	2.91	39.5	8740	11000	12500	313	4620	33300
46	12	3170	5100	8430	243	1250	3590	32.4	163	839	0.178	2.98	34	10700	12400	14400	277	4420	39100
47	1	2660	4560	7270	240	1090	3670	29.4	147	762	0.122	3.39	49.2	9440	11900	14100	190	2350	25000
47	2	2530	4280	6440	305	1030	2910	22.7	135	656	0.134	3.73	37.8	9080	10800	12600	295	2710	22700
47	3	2800	4660	7420	364	1090	2140	23.2	140	764	0.184	3.3	57.6	9960	12100	14100	225	3360	34700
47	4	2590	4680	7760	297	1070	2450	28.6	141	687	0.132	2.69	54.3	9540	11400	13200	321	3780	34400
47	5	3900	5680	8200	127	569	2680	22.2	144	597	0.394	6.61	26	10000	11600	13400	1120	3180	18100
47	6	3960	5720	7820	87	634	2660	87.5	262	777	0.373	8.08	50.8	8890	10900	13100	985	3670	31500
47	7	3620	5620	8240	269	1070	3140	24.6	124	773	0.0458	0.731	28.4	9510	11000	13400	1240	3710	32000
47	8	3210	4820	7590	377	1380	2880	25.9	126	536	0.0628	1.75	33.2	8160	9770	12300	1490	9010	39400
47	9	2630	4540	7050	432	1210	2560	1.44	30	642	0.137	1.67	44.7	8680	10100	11600	1080	1640	2490
47	10	3360	4920	8240	492	1310	3430	1.56	21.6	309	0.11	2.29	61.8	9460	10900	13600	1290	1770	2530
47	11	2430	3820	5860	371	960	2890	1.03	22.1	364	0.121	2.15	21.2	7180	8670	9940	1110	1550	2250
47	12	2560	4060	5960	449	906	2590	33.4	143	592	0.0919	2.41	31.4	7530	8800	10300	1270	3260	26800
48	1	2530	4140	5940	454	1020	2600	26.1	176	470	0.129	3.03	46.1	8510	9880	11200	1060	1570	2170
48	2	1850	3320	5060	278	714	2080	29.1	135	924	0.114	2.09	40	7160	8100	9640	1710	2520	3400
48	3	2670	4180	6920	367	978	2760	22.1	177	651	0.205	3.07	46.4	8560	9390	10700	2030	3180	4300
48	4	2110	3990	6390	294	843	2220	29.2	134	644	0.0746	2.05	28.5	7710	9150	10300	1830	2800	3820
48	5	3260	4610	6450	159	568	1840	27.6	111	538	0.899	6.02	33.4	7740	9260	10600	1660	2350	3160
48	6	3390	5200	7690	0.756	38.8	714	53.7	219	686	0.973	7.34	83.4	6620	7840	9350	2790	3700	5350
48	7	5730	8480	11500	429	1440	3680	68.5	285	1020	0.016	0.73	16.5	14600	16560	18400	5380	7390	9780
48	8	5610	7790	11500	811	2030	3850	45.9	303	1040	0.122	1.94	48.4	13600	15700	17700	3940	5680	8350
48	9	4250	6710	9040	989	1990	4210	33.3	150	681	0.063	4.21	41.5	12300	14200	15900	3690	4760	6130
48	10	5350	7610	12200	904	2290	5820	111	333	902	0.404	5.55	35.6	16180	17800	20200	4390	6340	8300
48	11	4350	6840	10400	881	1810	3890	31.4	166	995	1.51	7.31	48.1	14200	16400	19100	4130	5360	7590
48	12	4840	7090	10400	696	1980	6500	41.1	162	719	0.362	4.05	46.6	14700	16700	18900	4360	5980	8020
49	1	4970	7040	10300	778	1990	4840	29.2	174	1030	0.876	5.21	50.8	15000	17600	20200	6650	9240	12100
49	2	4450	6040	9920	575	1790	3840	22.5	135	703	0.313	4.37	44.3	13300	15200	17100	6160	8000	10600
49	3	4200	7040	9430	953	1840	3820	33.5	184	1120	0.213	5.45	49.8	15000	17900	20100	8350	11100	14600

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide and gross beta.

Table B.2. (contd)

Yr	Mo	Ga72 Min	Ga72 Med	Ga72 Max	As76 Min	As76 Med	As76 Max	Y90 Min	Y90 Med	Y90 Max	I131 Min	I131 Med	I131 Max	Np239 Min	Np239 Med	Np239 Max	GrossBeta Min	GrossBeta Med	GrossBeta Max
49	4	4900	6900	9520	601	1500	3510	95.4	300	1420	0.285	4.01	49.3	13800	16500	18700	6530	7640	9280
49	5	5440	8480	12500	353	1040	3200	84.5	299	962	1.63	10.9	56	13800	16000	18600	5840	7200	9080
49	6	6450	9050	12800	400	1340	3930	72.4	253	886	0.815	10.6	77.6	14000	16100	18600	4850	5990	7400
49	7	5510	8270	10400	937	2220	4200	34.2	173	827	0.0937	1.74	43.4	13700	15900	17800	4560	6580	9130
49	8	5740	7870	10500	986	2200	4860	6.49	71	821	0.294	3.18	30.1	14000	15600	17900	4590	6240	8580
49	9	5390	7540	11500	1130	2410	4620	7.3	80.4	884	0.441	4.12	54.6	13900	15400	18100	5470	8240	11500
49	10	5900	8470	13500	1080	2610	5150	23.9	148	816	0.233	5.88	44.6	17400	19500	21800	8910	14200	29400
49	11	6380	9480	13300	1540	2890	5320	136	373	1100	0.674	8.7	61.4	19900	23300	26000	8140	9460	11600
49	12	6880	9890	14160	1520	2830	6930	146	387	1230	0.694	10.5	76.5	23200	25600	28000	9000	11300	12800
50	1	5410	9380	14700	1020	2580	5960	63.5	371	1140	0.862	7.16	44.1	20600	24000	27500	10200	14100	17400
50	2	6080	8880	11900	933	2080	4950	85.4	337	1470	0.503	7.76	50.4	19200	22700	25600	15200	19900	25600
50	3	5990	9770	13500	739	2740	5300	95.3	443	1150	0.379	7	58.7	21600	24700	27300	18400	24500	30900
50	4	7070	9860	14100	987	2150	5240	86.4	345	1160	0.344	6.6	43.1	20700	23500	26400	18400	24400	32300
50	5	8170	11400	16600	1080	2080	6210	195	527	1540	0.0703	1.65	40.6	20700	23300	26000	20600	26700	32800
50	6	10000	12500	15600	314	1030	4150	169	437	1220	5	18.3	79.3	18700	21100	23700	16200	20600	26300
50	7	10500	13300	15900	562	1570	4900	150	493	1200	2.23	17.2	71.3	19500	21900	24700	11400	15000	19400
50	8	7690	10400	14200	1350	2720	5510	34.3	232	1050	0.207	2.38	19.8	16300	19100	21600	11400	14300	18100
50	9	6620	9240	13000	1340	3040	5410	7.61	74.7	656	0.436	6.01	51.5	16400	19200	21600	11200	14700	19000
50	10	7830	11200	15900	1900	3540	5900	11.8	136	1120	1.75	9.91	60.4	21000	23400	25600	20800	24000	29800
50	11	9170	12500	17000	1510	3780	6420	18.7	132	872	3.74	13.7	96.7	24200	27300	31000	31700	42100	54900
50	12	9420	13300	17200	1730	3200	6300	8.55	108	952	3.63	17.9	96.4	26200	29500	34100	34300	41900	52200
51	1	4960	7560	12600	1310	2230	4160	83.7	355	1030	8.03	29.6	79.7	16600	19800	24100	26900	31000	35900
51	2	4380	7320	11600	1430	2390	4010	89.2	448	1230	3.43	23.6	91.3	17100	19200	22000	32900	39300	45100
51	3	5600	8630	12200	1690	2650	4440	178	475	1350	6.05	22.3	108	19500	21700	24800	36600	45700	54200
51	4	5250	8240	12300	1360	2290	5730	145	461	1320	21.7	43.1	108	18200	20800	24200	33200	39000	46200
51	5	6390	9920	15200	795	1260	2820	312	743	1720	19.1	47.7	172	16900	20300	25000	39300	46100	58300
51	6	8080	11300	14500	1110	2030	4610	151	541	1210	20.3	47	91.9	17300	20300	23800	38800	46400	54800
51	7	10300	13300	17600	712	1740	4270	155	432	968	28.4	53.5	115	20800	23700	27300	35000	42800	56500
51	8	7760	10700	17000	875	1400	3260	131	424	1220	28.5	50.9	121	19300	22500	26300	39200	51100	67000
51	9	7960	11900	15300	983	2210	4290	93.9	432	1340	45.7	73.9	141	21100	24400	28000	44500	53200	69900
51	10	5680	8880	13400	1050	2030	4570	230	590	1850	51.6	85.4	155	18800	22000	26500	41000	49300	57700
51	11	6050	9510	15100	652	1230	2430	113	472	1370	51.8	99.2	201	21800	24800	29700	46200	52600	61700
51	12	3960	6430	10700	1210	2200	3990	293	671	1700	68.2	118	214	19300	22200	25500	42700	50800	59000

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide and gross beta.

Table B.2. (contd)

[illegible]

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide and gross beta.

Table B.2. (contd)

Yr	Mo	Ga72 Min	Ga72 Med	Ga72 Max	As76 Min	As76 Med	As76 Max	Y90 Min	Y90 Med	Y90 Max	I131 Min	I131 Med	I131 Max	Np239 Min	Np239 Med	Np239 Max	GrossBeta Min	GrossBeta Med	GrossBeta Max
34	10	12100	16400	23000	5810	9320	13200	403	1020	2560	72	174	296	28000	33400	39400	97400	142000	198000
54	11	11700	15600	21800	4250	7980	12200	279	990	2110	14.1	37.9	154	22600	27900	32200	89200	149000	214000
54	12	12900	16900	22300	5300	8160	13600	451	1240	2910	16.9	55.3	218	24200	29000	33100	108000	152000	244000
55	1	11500	16900	21600	5630	8450	12600	568	1510	2320	85.9	141	259	25200	31800	37700	106000	158000	250000
55	2	7290	11300	16500	3540	5720	9060	225	643	1840	52.2	113	184	18800	23300	27600	67000	110000	173000
55	3	16000	21600	30800	6960	11200	16600	643	1620	4450	63.3	133	299	31500	36600	42200	137000	226000	397000
55	4	11300	19500	29600	9770	12900	18100	441	1090	3290	130	235	451	27900	32500	38100	110000	190000	340000
55	5	17800	27100	35800	11400	16600	22500	430	1410	3790	218	342	638	37900	44300	51000	165000	262000	427000
55	6	24000	32700	44400	8020	12800	16600	281	1090	2830	189	298	548	35000	40000	46000	95700	171000	353000
55	7	19000	27500	37200	7240	10600	15900	315	885	4280	129	249	485	26000	31900	38400	116000	141000	172000
55	8	14900	21300	30200	9230	12900	18100	677	1590	3130	161	260	578	27000	31800	37100	95800	114000	192000
55	9	11600	16600	24200	6920	9560	13100	629	1590	3160	148	240	464	24900	29200	35100	85100	110000	131000
55	10	16600	23600	32700	9150	12700	16600	578	1290	2930	143	272	455	30300	35000	41200	142000	191000	309000
55	11	17400	24900	34700	8830	12300	17900	544	1170	2730	191	329	495	32800	38500	43900	145000	224000	340000
55	12	15400	24400	33500	9940	13800	18000	493	1320	3130	357	458	773	39200	44500	52700	177000	261000	361000
56	1	13000	21400	30700	5360	8390	12500	314	1010	3270	280	420	619	36800	42100	49600	99000	128000	155000
56	2	14400	20500	35500	6350	9500	12600	558	1430	4270	270	384	520	31900	39100	45800	97400	115000	137000
56	3	16700	25100	36500	6910	10500	15400	663	1490	4940	181	263	542	34200	39900	47100	142000	182000	208000
56	4	20000	28700	38900	6540	9520	14100	549	1430	4010	108	183	347	32100	36600	42900	173000	200000	244000
56	5	27200	33300	42600	4970	7820	14200	281	686	1590	108	176	459	30800	35800	42400	147000	194000	234000
56	6	28500	37800	50100	3940	6870	11900	595	1580	3500	94.5	186	432	31300	36500	43600	162000	193000	254000
56	7	25400	33300	45800	5730	9680	16100	650	1580	3980	103	162	392	31000	35600	41000	174000	217000	264000
56	8	21500	31200	42000	7960	12000	16900	461	1280	2290	85.2	164	339	32000	37600	44900	156000	183000	213000
56	9	19700	26400	35200	8250	12500	17800	592	1600	3900	136	199	389	30700	37600	43300	160000	197000	249000
56	10	13900	22100	34800	8790	11300	18000	733	1430	4290	51.9	85.2	126	18200	27300	37300	166000	208000	247000
56	11	12100	19100	31600	9940	14500	19300	269	943	3880	130	180	259	25500	32400	40200	165000	195000	267000
56	12	11200	18600	27600	15200	21700	29000	960	2110	4420	173	243	326	39000	49600	61800	217000	250000	287000
57	1	9630	17700	31600	26000	31100	40200	1130	2870	6180	94.4	130	196	38400	46400	55200	239000	269000	311000
57	2	9070	15700	29400	16800	21200	26100	926	1790	3620	180	229	319	39900	53700	70300	238000	280000	315000
57	3	10900	18200	29400	9100	11800	15000	649	1520	4790	270	370	517	40800	51000	67000	366000	427000	473000
57	4	12200	18700	30100	12600	17400	25800	696	1910	4320	175	270	353	40200	50500	62800	811000	897000	1040000
57	5	19300	25900	43900	10800	14800	23300	426	1320	4410	200	267	372	37100	46300	62500	756000	880000	1020000
57	6	17700	26600	38400	7140	9730	12700	561	1540	4230	107	163	331	25400	32400	43800	354000	416000	494000

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide and gross beta.

Table B.2. (contd)

Yr	Mo	Ga72 Min	Ga72 Med	Ga72 Max	As76 Min	As76 Med	As76 Max	Y90 Min	Y90 Med	Y90 Max	I131 Min	I131 Med	I131 Max	Np239 Min	Np239 Med	Np239 Max	GrossBeta Min	GrossBeta Med	GrossBeta Max
57	7	20000	27100	37400	7200	10100	14200	712	1580	3750	121	231	424	24300	33100	40300	280000	338000	426000
57	8	16100	23000	36700	8560	11600	16100	1080	2390	4710	309	467	735	34500	44600	57500	215000	263000	369000
57	9	15200	24300	35900	10400	13800	18400	1130	2420	4850	227	348	611	33300	41200	55200	247000	292000	345000
57	10	17100	24700	35800	12100	17000	24200	1060	2450	5560	215	334	569	23200	28300	36800	276000	317000	370000
57	11	15600	23800	38100	15200	21200	28000	764	2150	6120	175	285	579	24800	30200	39700	288000	360000	448000
57	12	16800	22900	33800	23700	32400	45500	1190	2290	4780	270	406	677	34100	42400	56200	271000	326000	396000
58	1	16100	23100	35300	26100	35000	49800	1180	2520	6960	261	415	732	32700	43700	54900	329000	384000	504000
58	2	12400	20900	34400	24000	31400	42000	1190	2310	5120	227	376	579	24800	36600	47800	294000	350000	419000
58	3	15100	22900	33000	17200	24700	34300	1120	2370	5000	257	422	673	28900	38600	52600	296000	353000	425000
58	4	13500	21600	31300	14800	22600	31900	1410	2630	5350	408	611	886	27800	36000	45200	424000	558000	675000
58	5	22600	28900	40500	16400	22600	27800	1189	2250	4540	257	393	697	22600	31700	43900	382000	463000	524000
58	6	26200	37000	52900	9810	13500	18900	601	1560	3770	263	493	833	18800	24400	30800	253000	325000	404000
58	7	22100	28700	45000	12600	16800	22700	910	2020	4610	480	886	1630	22600	27500	37500	233000	285000	341000
58	8	1970	5220	17900	15000	19200	24700	2610	4360	5820	256	354	504	30200	35900	43000	299000	340000	379000
58	9	22900	33000	51700	17500	20500	25100	1320	2050	2800	137	224	300	23700	28400	33700	352000	391000	431000
58	10	10300	14400	22400	19700	24600	30000	1670	2520	3520	242	334	471	31300	38100	45100	349000	387000	447000
58	11	8310	12200	17900	20500	28200	39100	1630	2270	2990	169	242	429	31000	37900	48900	292000	336000	376000
58	12	6960	10500	16600	30200	34200	40600	3220	5300	8400	206	279	394	37600	43500	49900	394000	441000	466000
59	1	11800	17600	26800	21500	27200	33300	1520	2570	4560	128	209	406	19900	24500	29600	340000	392000	440000
59	2	562	1030	2640	25600	30900	38100	1390	2670	5410	128	173	245	26700	32800	39100	434000	464000	518000
59	3	168	848	6030	22000	26300	33600	2280	3310	5190	128	178	265	22900	27500	31800	438000	484000	526000
59	4	23000	30300	44600	17100	20400	24100	2650	3360	4970	90.6	155	229	19200	24400	30700	388000	446000	708000
59	5	26000	34400	57600	14300	17300	22000	2000	2590	4170	107	169	241	18700	22700	28300	467000	522000	599000
59	6	20100	31100	47400	9240	11400	13300	1100	1450	1880	105	173	293	14100	17700	21400	353000	387000	423000
59	7	26900	35800	50200	9210	10700	12800	1030	1500	2310	93.6	176	510	14500	17600	20900	352000	397000	451000
59	8	23700	32000	45000	10100	11900	14300	1690	2490	3520	51.1	94.9	171	12400	16100	18800	364000	395000	439000
59	9	13400	24100	38400	10000	12500	14800	1170	1620	2220	69.8	114	211	15100	18200	22200	326000	365000	397000
59	10	14100	20300	27500	15400	19300	23400	1090	1630	2200	96.6	155	283	18400	22000	27300	360000	414000	471000
59	11	13600	18300	26900	13000	15200	17100	2140	3070	4330	55.2	96.7	177	19200	22900	27100	356000	404000	456000
59	12	18800	26200	44100	11600	15300	18200	3620	4730	6500	227	292	427	23800	28700	35700	404000	451000	503000
60	1	2370	3620	6380	24800	30200	48300	5120	7730	10800	504	750	1228	28800	35900	47100	437000	498000	548000
60	2	16600	22500	33400	12300	15500	20000	3580	5010	6580	307	482	620	27300	33600	39900	386000	420000	471000
60	3	7610	10500	16500	16300	20400	26200	3990	5420	7850	307	433	575	29900	37200	44700	368000	446000	716000

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide and gross beta.

Table B.2. (contd)

Yr	Mo	Ga72 Min	Ga72 Med	Ga72 Max	As76 Min	As76 Med	As76 Max	Y90 Min	Y90 Med	Y90 Max	I131 Min	I131 Med	I131 Max	Np239 Min	Np239 Med	Np239 Max	GrossBeta Min	GrossBeta Med	GrossBeta Max
60	4	20600	30900	45800	18500	22500	30300	2420	3710	5320	178	294	423	25800	31300	42200	744000	841000	913000
60	5	34400	51200	72600	15700	18700	21600	2800	4670	7870	278	366	542	17100	21600	26300	500000	560000	615000
60	6	5230	7830	10800	12200	14200	16900	2180	3090	5110	163	232	351	19000	24100	30400	396000	436000	483000
60	7	19300	26100	39200	9480	11600	14400	1970	3040	4080	155	238	408	20600	26100	31700	367000	420000	459000
60	8	15200	24100	33600	17000	21300	26200	2380	3300	6820	309	481	642	19400	23900	30900	388000	432000	483000
60	9	24800	36300	50600	11400	14200	17800	3610	5110	8220	153	241	354	20100	24900	31400	406000	439000	510000
60	10	18200	26900	43100	15700	20100	30700	2560	3740	7570	240	359	632	18100	23500	39500	414000	511000	670000
60	11	22500	32200	54600	19200	23700	36700	1260	2480	6180	270	442	693	29200	35900	47800	517000	583000	660000
60	12	20700	27500	40300	20500	24500	30500	4160	5970	8160	442	626	857	27800	36600	45700	563000	625000	715000
61	1	22700	34000	44700	30400	34900	43100	4520	6920	10100	419	578	811	38500	50300	57700	656000	726000	844000
61	2	19400	28300	45000	13200	16700	21000	4610	6570	9890	265	403	591	30100	36000	44600	558000	630000	684000
61	3	12600	20400	27800	18800	23900	30600	3360	4570	7410	245	340	483	18800	25300	31300	777000	884000	956000
61	4	20000	27100	38800	16600	19700	23900	3690	5030	6900	235	358	459	14300	18000	22800	852000	956000	1040000
61	5	34200	46900	75800	15500	18500	22500	5070	6570	9570	146	204	360	11600	14900	18400	553000	634000	706000
61	6	21700	29500	46100	6190	7430	8670	1680	2410	3240	236	322	451	11300	15600	20000	490000	551000	607000
61	7	7980	12100	28200	6600	7190	7900	1370	2150	3420	73.9	145	326	5870	7610	9910	402000	448000	507000
61	8	4810	9560	21100	6910	7730	8420	995	1950	4230	238	347	495	8500	12400	15900	363000	404000	474000
61	9	2820	5300	10700	6140	6780	7710	3440	5280	7460	309	408	580	12700	16400	23000	264000	304000	371000
61	10	4630	7510	22100	7020	7920	8640	2630	4170	5750	531	764	1300	15500	19000	22300	326000	366000	421000
61	11	6060	10500	26200	7940	8480	9200	1350	2290	3960	510	689	984	9740	12900	17500	308000	358000	421000
61	12	3680	5000	7730	6990	7640	8450	1360	1760	3250	144	267	466	11900	15300	19600	251000	290000	333000
62	1	2890	4480	6360	5620	6230	6720	886	1380	1800	411	551	746	16100	20100	25400	364000	404000	458000
62	2	3330	4490	6300	7350	7980	8640	712	1210	2310	256	375	514	14400	18000	22900	312000	354000	406000
62	3	2180	3070	4550	11700	13000	14700	2500	3500	5020	345	505	662	20900	26300	32700	610000	696000	795000
62	4	10300	15700	24800	11600	12600	13700	3090	5010	7280	108	174	440	23800	27500	36000	648000	720000	815000
62	5	1240	2670	6100	8250	8930	9900	2960	5450	8960	569	763	1010	27700	34800	48400	478000	580000	739000
62	6	2190	3250	5930	5710	6540	7380	1420	2090	3350	326	471	686	14300	18500	23000	432000	502000	588000
62	7	5150	7720	12400	4780	5120	5560	1290	2980	6180	91	157	311	12000	14700	17900	337000	375000	410000
62	8	9540	13100	21300	5770	6260	7010	1680	3270	6910	87.1	147	271	18700	22000	26500	378000	428000	491000
62	9	9790	14000	19800	4820	5300	5930	2330	3980	7240	77.9	157	260	18200	22700	29900	324000	360000	423000
62	10	5470	7920	11400	4440	4880	5420	1560	3210	6820	370	519	696	15700	20800	26000	320000	358000	406000
62	11	6310	10600	15400	3870	4230	4590	1480	2640	4650	143	248	353	12200	15500	19200	234000	280000	331000
62	12	3790	5580	8520	5030	5590	6170	1330	3040	6460	158	235	341	13100	16200	21000	313000	350000	411000

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide and gross beta.

Table B.2. (contd)

Yr	Mo	Ga72 Min	Ga72 Med	Ga72 Max	As76 Min	As76 Med	As76 Max	Y90 Min	Y90 Med	Y90 Max	I131 Min	I131 Med	I131 Max	Np239 Min	Np239 Med	Np239 Max	GrossBeta Min	GrossBeta Med	GrossBeta Max
63	1	3780	5670	8910	3310	3660	3940	530	1630	5800	328	544	776	8610	11600	17000	286000	334000	412000
63	2	3750	4940	6660	3740	4070	4540	983	2360	6150	215	312	442	13200	17400	22100	261000	295000	341000
63	3	3480	5380	8410	12600	14000	15800	1430	3330	6100	426	613	890	18700	21300	24200	399000	507000	638000
63	4	4900	7530	12000	9590	10500	11600	1890	3610	7730	39.6	87.4	154	18200	20300	23300	519000	614000	735000
63	5	5560	7620	10300	8040	8570	9510	1300	2690	4360	67.4	106	211	13700	15900	19100	462000	529000	701000
63	6	5710	7780	10600	7480	8380	9870	1070	2300	7050	98.6	186	258	14900	16300	19100	338000	408000	508000
63	7	5690	9000	14700	7380	8030	8690	1120	3000	7530	61.1	118	218	14000	15700	19200	377000	441000	539000
63	8	7280	10700	14300	8080	8760	9570	958	2290	4630	110	177	297	14500	16500	18900	211000	262000	336000
63	9	7380	10900	15500	7470	8030	8920	525	1750	5140	126	214	341	14900	17200	19700	264000	310000	358000
63	10	8300	12200	17100	7880	8570	9430	797	2270	5830	132	266	400	20300	23900	28000	290000	346000	404000
63	11	5910	8340	13500	6920	8020	8910	786	1850	5290	166	266	474	14100	16000	18300	240000	300000	349000
63	12	7190	11400	16500	9120	10000	10800	759	2730	5770	128	226	327	16900	19300	21900	335000	387000	467000
64	1	5740	7680	11100	9650	10600	11500	558	1770	5010	174	277	404	19300	21900	26300	262000	314000	388000
64	2	5050	7100	11580	8450	9280	10100	628	1710	5030	64.2	130	234	18100	20500	23100	208000	254000	295000
64	3	6180	8290	12200	9600	10300	11000	601	1730	5590	150	229	353	19500	22100	24900	276000	341000	414000
64	4	2240	5850	15100	8570	9290	10100	999	2110	6110	337	486	774	24600	28200	33800	257000	345000	461000
64	5	7260	15400	31600	11000	12200	13200	897	1950	5280	199	338	574	30500	35200	39200	410000	504000	663000
64	6	14200	23300	37200	10100	10800	11700	790	2280	5520	19.9	92.1	255	19200	20900	22500	426000	531000	637000
64	7	9380	16600	28300	7920	8420	9130	977	2210	5530	49	122	314	12100	13800	16100	339000	431000	561000
64	8	4350	9470	26100	6260	6730	7300	964	2360	5720	55.5	138	399	12800	14400	17400	254000	336000	473000
64	9	3030	7370	21300	8520	9240	10200	447	1590	4100	105	192	414	12400	15300	20300	236000	320000	522000
64	10	1510	5420	21000	7630	8500	9510	229	930	5140	150	290	588	13200	16000	22200	180000	265000	385000
64	11	9210	17200	30900	8940	9540	10400	1260	3150	7670	134	278	576	17700	20700	24300	280000	386000	555000
64	12	11900	20400	33800	8790	9580	10500	1380	3190	6580	122	255	436	14600	18500	22500	280000	390000	545000
65	1	2810	6470	21100	4940	5420	5960	701	1820	5280	168	301	462	19000	22000	27300	179000	291000	473000
65	2	1870	4500	10800	4330	4720	5290	463	1400	4580	114	212	473	12500	15600	20300	185000	291000	413000
65	3	3910	11700	26200	7610	8290	9140	908	2600	7030	51.8	185	385	14500	16200	19800	255000	349000	499000
65	4	3800	8270	18300	9540	10400	11100	844	2420	5340	58.7	146	315	13600	16000	19000	263000	363000	504000
65	5	3430	9660	28500	11800	12900	14400	501	1670	5040	87.1	183	412	14200	17100	21200	269000	339000	535000
65	6	6180	13500	27800	8400	9140	9920	763	1960	5230	37.9	101	350	10800	12600	14700	273000	363000	549000
65	7	3240	8900	21800	7180	8130	9250	455	1610	6770	58.1	164	340	7630	11400	14600	179000	269000	422000
65	8	1970	6470	19600	9880	10800	12200	493	1750	4980	49.4	115	343	8860	10900	13300	158000	223000	363000
65	9	1060	4870	17700	9040	10400	12200	754	1830	5320	66.1	141	280	9010	11400	14000	149000	243000	360000

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide and gross beta.

Table B.2. (contd)

Yr	Mo	Ga72 Min	Ga72 Med	Ga72 Max	As76 Min	As76 Med	As76 Max	Y90 Min	Y90 Med	Y90 Max	I131 Min	I131 Med	I131 Max	Np239 Min	Np239 Med	Np239 Max	GrossBeta Min	GrossBeta Med	GrossBeta Max
65	10	2830	8090	22000	16200	18800	23000	373	1500	5090	79	162	377	10100	12300	16100	180000	294000	431000
65	11	1550	5890	16100	11200	13000	14500	381	1370	4360	107	207	395	9720	12200	15400	174000	230000	331000
65	12	2320	6080	15500	11100	12600	15000	387	1270	4800	64.5	159	423	8770	10500	16400	197000	279000	404000
66	1	1220	5850	19000	6440	7240	8690	165	1190	5840	26.2	86.5	270	6760	7940	9210	161000	255000	388000
66	2	2680	7900	18300	6660	7460	8430	197	908	4320	30.1	89.9	282	5710	7150	9350	143000	235000	320000
66	3	741	2800	8110	9730	10500	11700	139	1200	5550	10.7	68.1	276	7540	8300	9190	202000	300000	476000
66	4	1400	4310	15600	7240	8020	8880	315	1140	5250	26.3	47.8	79.5	7680	8960	12600	198000	274000	387000
66	5	2840	7360	22300	7040	7810	9130	251	1280	4680	56.9	96.5	177	9150	11400	14400	198000	278000	416000
66	6	3280	8100	21700	6280	7170	7890	186	749	3110	29.6	49.1	108	4530	5200	7640	165000	247000	363000
66	7	847	2500	8220	1160	1460	1900	79.3	381	1420	5.42	8.39	13	868	1080	1400	49500	79200	110000
66	8	6150	9840	14200	1110	1350	1760	353	1140	3040	9.34	14.5	25.3	1350	1710	2200	51500	84500	113000
66	9	11100	19500	33300	5420	6040	6690	643	2200	7200	65.1	108	181	5920	7740	10200	170000	248000	368000
66	10	838	3630	12500	5990	6770	7610	258	1050	4710	33.7	61.9	124	5150	6210	9340	175000	248000	409000
66	11	663	2530	11200	4070	4420	4900	124	773	4320	29.3	49.4	88.1	4480	5680	8890	115000	196000	337800
66	12	1880	5870	23600	5710	6360	6920	188	1410	4490	60.7	96.1	149	6240	7660	9650	198000	268000	354000
67	1	4410	11400	32100	7880	8680	9620	293	1170	4310	38.3	58.1	120	6490	7510	8800	137000	226000	357000
67	2	1270	5960	20900	4600	5190	5610	325	1150	5480	28.2	37.6	46.5	5740	6260	6950	162000	227000	339000
67	3	2350	7400	18700	6410	7020	7700	461	1580	4210	65.5	120	200	6450	8680	12100	199000	234000	418000
67	4	2850	7910	17200	10500	11500	12900	335	1610	5850	41.7	51.2	61.5	8210	9180	10000	223000	314000	490000
67	5	2600	6680	24700	8700	9510	10500	361	1730	4520	62.9	104	186	9640	11300	14400	203000	274000	436000
67	6	4880	11000	30100	10000	11100	12100	711	1890	6080	44	56.5	71.2	12300	14400	16400	219000	342000	479000
67	7	1930	9130	23400	5540	6420	7230	655	1840	5760	82	133	180	9140	10700	13300	163000	255000	380000
67	8	1590	6890	21000	6590	7500	8490	618	1730	6690	90.3	119	162	7920	9250	12000	116000	199000	405000
67	9	1570	4810	15400	4730	5260	5820	612	1660	5180	217	312	438	8890	11800	16600	106000	177000	355000
67	10	506	3220	15700	7030	7730	8690	437	1500	3260	123	187	291	6620	9010	12200	104000	184000	295000
67	11	235	1390	12000	5220	5780	6640	287	1110	4410	67.1	126	241	6780	8480	11500	86300	184000	305000
67	12	222	1470	8620	7560	8320	9100	136	919	4340	64.8	103	168	7090	8400	10300	157000	237000	369000
68	1	2940	8890	22300	8110	9130	10200	328	847	2600	55.8	80.8	125	6830	7920	9340	116000	199000	287000
68	2	2760	8560	28000	6440	7220	8440	361	1210	3710	86.5	124	166	8780	9980	13400	178000	259000	354000
68	3	2350	8260	20400	8890	9960	11400	211	992	4060	87.1	121	198	8720	10100	11800	200000	283000	390000
68	4	295	1540	7920	7920	8720	10000	189	844	3430	89.4	124	184	7280	8700	11100	188000	280000	374000
68	5	548	2690	9580	4190	4670	5240	136	954	4000	88.4	143	253	9150	10600	13900	174000	260000	364000
68	6	626	2730	14800	3460	3980	4520	186	736	2980	30.2	66.5	127	5340	7130	8720	97200	160000	277000

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide and gross beta.

Table B.2. (contd)

Yr	Mo	Ga72 Min	Ga72 Med	Ga72 Max	Aa76 Min	Aa76 Med	Aa76 Max	Y90 Min	Y90 Med	Y90 Max	I131 Min	I131 Med	I131 Max	Np239 Min	Np239 Med	Np239 Max	GrossBeta Min	GrossBeta Med	GrossBeta Max
68	7	1020	2940	11600	3300	3790	4270	180	743	3780	38.3	61.3	142	4930	5920	7440	77000	123000	206000
68	8	929	3650	14300	4130	4530	5090	161	894	3900	48.4	74.8	128	7280	8300	10000	83000	138000	269000
68	9	473	2560	17400	3120	3590	4170	213	716	4180	41.3	66.6	101	6740	8040	9520	108000	181000	291000
68	10	898	5480	19100	5080	5680	6240	191	790	3110	138	177	269	7090	8420	11500	105000	178000	333000
68	11	702	4520	19700	4440	4940	5770	34.7	413	2690	32.3	42.1	57.8	6020	6900	8130	106000	156000	315000
68	12	1440	7270	21400	4790	5440	6470	211	639	3030	46.3	68.4	110	6560	7580	9620	100000	160000	252000
69	1	1510	6940	22400	3660	4360	4920	44.2	357	2960	33.7	59.3	130	4480	5620	6930	85200	133000	299000
69	2	1020	3780	16000	3970	4710	5430	171	980	3060	26.6	49.2	133	4760	5720	8250	94500	152000	218000
69	3	177	1300	6480	6620	7460	8700	94	526	2960	39.7	81.1	157	6920	8600	12700	125000	184000	309000
69	4	810	3070	10500	6860	8000	9050	103	543	2260	35.6	57.7	147	8590	9960	12000	133000	197000	278000
69	5	227	1700	13500	5690	6600	8160	110	534	3130	13.5	19.7	27.7	4640	5340	6310	79000	137000	208000
69	6	707	4480	19100	4830	5460	6240	36.2	494	2230	15.7	25.5	37.1	4830	5500	6360	91700	148000	261000
69	7	1060	3450	13100	3820	4340	5100	17.6	312	2730	21.9	42.2	112	3010	3820	5520	76300	130000	222000
69	8	957	4370	15400	3370	3980	4780	51.3	380	2350	30.4	69.8	144	3570	5580	9390	83300	133000	231000
69	9	487	2270	13600	3920	4500	5360	28.9	303	2450	9.8	14.1	19.9	2930	3280	3760	48100	98700	181000
69	10	279	1770	8840	3530	4180	5020	17.8	210	1510	10	13.2	17.5	1520	1890	2250	52500	96600	159000
69	11	40	432	5450	2810	3360	3970	32.2	238	2620	8.99	13.5	18.1	1500	1770	2170	16200	47600	112000
69	12	2550	8040	25600	3760	4300	5010	10.6	218	2710	20.8	27.8	38.3	2260	2740	3270	18200	66400	178000
70	1	1770	7160	21700	1270	2480	4960	430	1520	4020	153	362	863	2600	5960	27900	55000	178000	399000
70	2	9.6	39.2	189	7.33	31.5	186	1.44	7.88	40.9	0.0171	0.339	3.96	10.2	59	242	266	1060	3440
70	3	145	911	4880	41.8	91.9	190	33.5	151	813	24.6	63.5	125	232	1030	5230	6400	19400	85100
70	4	561	2960	15500	659	1440	3130	118	594	3250	124	268	612	806	3980	15600	21900	83000	292000
70	5	464	2630	10400	432	936	2150	110	586	3280	89.1	190	444	611	3079	17900	18300	69200	218000
70	6	560	2520	14100	271	614	1240	116	580	2740	90	191	440	701	3100	17200	21200	63800	198000
70	7	574	2800	12800	689	2400	12400	136	588	2980	1.5	18.8	259	705	3000	17300	17000	57100	240000
70	8	351	1560	9840	340	1520	4380	66.7	420	2000	0.738	9.27	164	365	2160	10200	12800	44000	144000
70	9	485	3100	11800	531	2730	14200	122	719	3000	2.21	24.9	286	762	3300	19600	24000	77200	277000
70	10	597	3380	16900	582	2720	11000	117	782	3890	2.2	23.9	310	671	3400	17300	27000	80900	237000
70	11	554	2780	15500	520	2790	12700	101	686	2620	1.39	23.1	297	785	3120	18600	26500	65400	220000
70	12	565	2880	14800	526	2480	7670	116	564	3650	2.69	18.9	199	753	4700	20900	19500	81000	215000
71	1	516	2660	15800	461	2440	9620	90.4	566	3120	1.23	19.5	227	568	3540	15200	17900	72100	230000
71	2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	3	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide and gross beta.

Table B.2. (contd)

Yr	Mo	Ga72 Min	Ga72 Med	Ga72 Max	As76 Min	As76 Med	As76 Max	Y90 Min	Y90 Med	Y90 Max	I131 Min	I131 Med	I131 Max	Np239 Min	Np239 Med	Np239 Max	GrossBeta Min	GrossBeta Med	GrossBeta Max
71	4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	5	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	6	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	7	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	8	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	9	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	10	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	11	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
71	12	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

(a) Columns are the minimum, median, and maximum of 100 realizations for each radionuclide and gross beta.

Appendix C

Bibliography of Reactor Reference Sources

Appendix C

Bibliography of Reactor Reference Sources

Reactor Operation References

1. Kannberg, L.D. 1992. *Hanford Production Reactor Heat Releases 1951-1971*. PNL-2689, Pacific Northwest Laboratory, Richland, Washington.
2. Gydesen, S.P. 1992. *Selected Hanford Reactor and Separations Operating Data for 1960-1964*. PNWD-2017 HEDR, Battelle, Pacific Northwest Laboratories, Richland, Washington.
3. Gydesen, S.P. 1992. Letter Report: *Title Listing of Daily Operating Data on Hanford Single-Pass Reactors, 1944-1971*. PNL-7998 HEDR, Pacific Northwest Laboratory, Richland, Washington.
4. Jungfleisch, F.M. 1992. *TRAC Data Files (U)*. WHC92-00001 Disk (2A), Westinghouse Hanford Company, Richland, Washington.
5. General Electric Company. *P Department Daily Reports 10/5/44 Through 12/3/1948*. HAN-45954, (Parts 1 through 4 covering 1944 - 1946), General Electric Company, Hanford Works, Richland, Washington.
6. General Electric Company. *P Department Daily Reports 10/5/44 through 12/3/1948*. HAN-54954 (Parts 5 through 8 covering 1947 and 1948), General Electric Company, Hanford Works, Richland, Washington.
7. General Electric Company. 1949. *Daily Operating Conditions 100 Areas October 1948-December 1949*. HW-89079, General Electric Company, Hanford Works, Richland, Washington.
8. General Electric Company. 1950. *Daily Operating Conditions 100 Areas 1950*. HW-89080, General Electric Company, Hanford Works, Richland, Washington.
9. General Electric Company. 1951. *Daily Operating Conditions 100 Areas 1951*. HW-89081, General Electric Company, Hanford Works, Richland, Washington.

Effluent Activity Measurement References

1. Carey, Z.E. 1952. *Summary of Radiochemical Analysis of Pile Effluent Water*. HW-25044, General Electric Company, Hanford Works, Richland, Washington.
2. Paas, H.J. 1953. *Radioactive Contamination in the Hanford Environs for the Period January, February, March 1953*. HW-28009, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

3. Pilcher, G.E., and H.T. Norton. 1953. *The Calculation of Beta Particle Emitter Concentration in Hanford Reactor Effluent Water*. HW-27584, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
4. Paas, H.J. 1953. *Radioactive Contamination in the Hanford Environs for the Period April, May, June 1953*. HW-29514, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
5. Paas, H.J. 1953. *Radioactive Contamination in the Hanford Environs for the Period July, August, September 1953*. HW-30174, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
6. Paas, H.J. 1954. *Radioactive Contamination in the Hanford Environs for the Period October, November, December 1953*. HW-30744, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
7. Watters, R.L., and E.E. Larkin. 1953. *Interim Report on the Radiochemical Analysis of Hanford Reactor Effluent Water*. HW-24956, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
8. Paas, H.J. 1954. *Radioactive Contamination in the Hanford Environs for the Period January, February, March 1954*. HW-31818, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
9. Paas, H.J., and G.E. Pilcher. 1954. *Radioactive Contamination in the Hanford Environs for the Period April, May, June 1954*. HW-33896, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
10. Pilcher, G.E., J.K. Soldat, and Z.E. Carey. 1955. *Radioactive Contamination in the Hanford Environs for the Period July, August, September 1954*. HW-36504, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
11. Regional Radiation Measurements Unit. 1955. *Radioactive Contamination in the Hanford Environs for the Period October, November, December 1954*. HW-36505, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
12. Regional Radiation Measurements Unit. 1955. *Radioactive Contamination in the Hanford Environs for the Period January, February, March 1955*. HW-36506, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
13. Regional Radiation Measurements Unit. 1955. *Radioactive Contamination in the Hanford Environs for the Period April, May, June 1955*. HW-38566, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
14. Matsumoto, W.Y., D.L. Reid, and Z.E. Carey. 1954. *Report on the Radiochemical Analysis of Hanford Reactor Effluent Water for the Period January 1951 to January 1954*. HW-32141, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

15. Andersen, B.V., and J.K. Soldat. 1955. *Radioactive Contamination in the Hanford Environs for the Period July, August, September 1955*. HW-39429, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
16. Andersen, B.V., and J.K. Soldat. 1956. *Radioactive Contamination in the Hanford Environs for the Period October, November, December 1955*. HW-40871, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
17. Matsumoto, W.Y., and D.L. Reid. 1956. *Quarterly Report, Radiochemical Analysis of Reactor Effluent Water July, August, September 1955*. HW-41632, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
18. Andersen, B.V., and J.K. Soldat. 1956. *Radioactive Contamination in the Hanford Environs for the Period January, February, March 1956*. HW-43012, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
19. Andersen, B.V., and J.K. Soldat. 1956. *Radioactive Contamination in the Hanford Environs for the Period April, May, June 1956*. HW-44215, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
20. Andersen, B.V., M.W. McConiga, and J.K. Soldat. 1956. *Radioactive Contamination in the Hanford Environs for the Period July, August, September 1956*. HW-46726, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
21. Andersen, B.V., M.W. McConiga, and J.K. Soldat. 1957. *Radioactive Contamination in the Hanford Environs for the Period October, November, December 1956*. HW-48374, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
22. McConiga, M.W., and J.K. Soldat. 1957. *Radioactive Contamination in the Hanford Environs for the Period January, February, March 1957*. HW-51009, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
23. McConiga, M.W., and J.K. Soldat. 1957. *Radioactive Contamination in the Hanford Environs for the Period April, May, June 1957*. HW-52803, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
24. McConiga, M.W., J.K. Soldat, and J.M. Selby. 1958. *Radioactive Contamination in the Hanford Environs for the Period July, August, September 1957*. HW-54841, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
25. McConiga, M.W., J.M. Selby, and J.K. Soldat. 1958. *Radioactive Contamination in the Hanford Environs for the Period October, November, December 1957*. HW-54842, General Electric Company, Hanford Atomic Products Operation, Richland, Washington. .
26. Jackson, P.O. 1956. *Tabulation of Reactor Effluent Water Contaminants in the Columbia River January - May 1956*. HW-45092, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

27. Polinsky, D.M. 1957. *Tabulation of Reactor Effluent Water Contaminants in the Columbia River October 1956*. HW-47811, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
28. Polinsky, D.M. 1956. *Tabulation of Reactor Effluent Water Contaminants in the Columbia River June, July, August 1956*. HW-46709, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
29. Polinsky, D.M. 1956. *Tabulation of Reactor Effluent Water Contaminants in the Columbia River September 1956*. HW-47261, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
- 29a. Polinsky, D.M. 1956. *Tabulation of Reactor Effluent Water Contaminants in the Columbia River October 1956*. HW-47811, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
30. Polinsky, D.M. 1957. *Tabulation of Reactor Effluent Water Contaminants in the Columbia River November - December 1956*. HW-48530, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
31. Reid, D.L. 1957. *Reactor Effluent Water Analytical Data*. HW-48350, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
32. Reid, D.L. 1957. *Reactor Effluent Water Analytical Data January 1957*. HW-49146, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
33. Reid, D.L. 1957. *Reactor Effluent Water Analytical Data February 1957*. HW-49588, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
34. Norton, H.T. 1957. *Tabulation of Reactor Effluent Water Contaminants in the Columbia River January-March 1957*. HW-49843, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
35. Reid, D.L. 1957. *Reactor Effluent Water Analytical Data March 1957*. HW-50292, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
36. Norton, H.T. 1957. *Tabulation of Reactor Effluent Water Contaminants in the Columbia River April 1957*. HW-50769, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
37. Reid, D.L. 1957. *Reactor Effluent Water Analytical Data April - May 1957*. HW-51383, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
38. Reid, D.L. 1957. *Tabulation of Reactor Effluent Water Contaminants in the Columbia River May 1957*. HW-51397, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

39. Reid, D.L. 1957. *Reactor Effluent Water Analytical Data 1957*. HW-51743, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
40. Hall, R.B., and P.C. Jerman. 1960. *Reactor Effluent Water Disposal*. HW-63653, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
41. Hall, R.B. 1960. *Reactor Effluent Water Analyses August 1958 - December 1959*. HW-64552, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
42. Hall, R.B. 1960. *Reactor Effluent Water Analyses January and February 1960*. HW-64824, General Electric Company, Hanford Atomic Products Operation, Richland, Washington..
43. Hall, R.B. 1960. *Reactor Effluent Water Analyses March and April 1960*. HW-66453, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
44. Koop, W.N. 1964. *Reactor Effluent Analytical Data January, February, March, 1964*. HW-81983, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
45. Koop, W.N. 1964. *Reactor Effluent Analytical Data April, May, June, 1964*. HW-83423, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
46. Koop, W.N. 1964. *Reactor Effluent Analytical Data July, August, September, 1964*. RL-REA-43, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
47. Koop, W.N. 1965. *Reactor Effluent Analytical Data October, November, December, 1964*. RL-REA-498, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
48. Koop, W.N. 1965. *Reactor Effluent Analytical Data January, February, March, 1965*. RL-REA-2028, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
49. Owen, F.E. 1965. *Reactor Effluent Analytical Data July, August, September, 1965*. DUN-110, Douglas United Nuclear, Inc., Richland, Washington.
50. Owen, F.E. 1965. *Reactor Effluent Analytical Data April, May, June, 1965*. RL-REA-2428, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
51. Owen, F.E. 1966. *Reactor Effluent Water Analytical Data January, February, March, 1966*. DUN-1035, Douglas United Nuclear, Inc., Richland, Washington.
52. Owen, F.E. 1966. *Reactor Effluent Analytical Data October, November, December, 1965*. DUN-658, Douglas United Nuclear, Inc., Richland, Washington.

53. Owen, F.E. 1966. *Reactor Effluent Analytical Data April, May, June, 1966*. DUN-1035 SUP1, Douglas United Nuclear, Inc., Richland, Washington.
54. Owen, F.E. 1966. *Reactor Effluent Analytical Data July, August, September 1966*. DUN-1035 SUP2, Douglas United Nuclear, Inc., Richland, Washington.
- 54a. Owen, F.E. 1967. *Reactor Effluent Analytical Data October, November, December 1966*. DUN-1035 SUP3, Douglas United Nuclear, Inc., Richland, Washington.
55. Dabrowski, T.E. 1971. *Douglas United Nuclear, Inc. Reactor and Fuel Facilities 1970 Environmental Release Report*. DUN-7613, Douglas United Nuclear, Inc., Richland, Washington.
56. Geier, R.G. 1969. *Reactor Effluent Water Analytical Data 1967 and 1968*. DUN-5405, Douglas United Nuclear, Inc., Richland, Washington.
57. Larrick, A.P. 1970. *Reactor Effluent Water Analytical Data July - December 1969*. DUN-6631, Douglas United Nuclear, Inc., Richland, Washington.
58. Geier, R.G. 1969. *Reactor Effluent Water Analytical Data January - June, 1969*. DUN-6035, Douglas United Nuclear, Inc., Richland, Washington.
59. Geier, R.G. 1970. *Reactor Effluent Water Analytical Data 1962 and 1963*. DUN-6738, Douglas United Nuclear, Inc., Richland, Washington.
60. Owen, F.E. 1967. *Reactor Effluent Water Analytical Data 1964, 1965, 1966*. DUN-2457, Douglas United Nuclear, Inc., Richland, Washington.
61. Hall, R.B., and J. R. Jones. 1961. *Reactor Effluent Water Analyses May 1960 Through June 1961*. HW-70733, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
62. Corley, J.P. 1962. *Reactor Effluent Analytical Data January 1962 Through June 1962*. HW-75023, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.
63. Hall, R.B. 1962. *Reactor Effluent Water Analyses Data July 1961 Through December 1961*. HW-73073, General Electric Company, Hanford Atomic Products Operation, Richland, Washington.

Appendix D

Description of STRRM Input and Output File Information

Appendix D

Description of STRRM Input and Output File Information

Input data files

- **Reactor file**

Single-record field definitions:

1. Reactor identification
2. Year
3. Month
4. PW, monthly power level (MW)
5. PD, monthly production (MWd)
6. F, monthly flow (1000 gpm)

The following entries are in sets of three, one set for each of the eleven radionuclides plus gross beta (twelve total sets)

7. A, monthly average activity concentration (pCi/ml), nuclide 1
8. nm, number of values in the average, nuclide 1
9. C, average daily release rate for the month (Ci/day), nuclide 1

.
.
.

11 additional A,nm,C triples in fields 10 through 42

- **Fuel-element failure file**

Single-record field definitions:

1. Reactor identification
2. Year
3. Month
4. W, uranium mass released per failure (grams)
5. B, burnup (MWd/ton)
6. Enr (Enr=0, natural uranium; Enr=1, enriched uranium)

- **Estimated regression distribution file**

Single-record field definitions:

1. Reactor identification
2. Year
3. Month

(12 sets of 2 entries each follow, one set for each of the 11 radionuclides and one set for gross beta)

4. Estimated activity concentration of nuclide 1 from applicable regression (pCi/ml)
5. Standard deviation of nuclide 1
- .
24. Estimated activity concentration of nuclide 11 from applicable regression (pCi/ml)
25. Standard deviation of nuclide 11
26. Estimated activity concentration of gross beta from applicable regression (pCi/ml)
27. Standard deviation of nuclide gross beta

- **Default distribution file**

Single-record field definitions:

1. Reactor identification
2. Radionuclide name
3. Year
4. Month
5. Concentration (pCi/ml)
6. Distribution type, only 'LN' for lognormal is used
7. Repeat concentration (pCi/ml)
8. Standard deviation (pCi/ml)

(12 sets of 2 entries follow, one set for each of the 11 radionuclides and one set for gross beta)

9. AM(1), mean value of distribution, nuclide 1
10. SD(1), standard deviation, nuclide 1
- .
- .
31. AM(12), mean value of distribution, gross beta
32. SD(12), standard deviation, gross beta

- **River flow file**

Single-record field definitions:

1. Year
2. Month
3. Columbia River average monthly flow (cfs)

- **Specific power file**

One record per year 1944 through 1972

Single-record field definitions:

1. Year
2. B average reactor specific power (MW/ton)
3. C average reactor specific power (MW/ton)
4. D average reactor specific power (MW/ton)
5. DR average reactor specific power (MW/ton)
6. F average reactor specific power (MW/ton)
7. H average reactor specific power (MW/ton)
8. KE average reactor specific power (MW/ton)
9. KW average reactor specific power (MW/ton)

- **Basin volume file**

The monthly retention volume for each reactor

One record per month

Single-record field definitions:

1. Year
2. Month
3. B reactor retention basin capacity (Kgal)
4. C reactor retention basin capacity (Kgal)
5. D reactor retention basin capacity (Kgal)
6. DR reactor retention basin capacity (Kgal)
7. F reactor retention basin capacity (Kgal)
8. H reactor retention basin capacity (Kgal)
9. KE reactor retention basin capacity (Kgal)
10. KW reactor retention basin capacity (Kgal)

Output files

- **Report file**

Tables of minimum, maximum, median, mean, standard deviation, and coefficient of variation for each of the eleven radionuclides and gross beta. These are aggregated over month, year, and all years of operation in different tables.

- **Fuel-element failure file**

Tables of minimum, maximum, median, mean, standard deviation, and coefficient of variation for iodine-131 and neptunium-239 from fuel-element failures during the month.

- **River file**

This file contains information on sodium-24, phosphorus-32, scandium-46, chromium-51, zinc-65, arsenic-76, and neptunium-239 and is primarily a river transport model input file for this subset of radionuclides. One file is produced for each Monte Carlo realization.

- **Statistics file**

This file contains a record of the result of sampling each distribution function. It is normally used for statistical verification purposes.

Appendix E

Summary of Technical Steering Panel Comments and Battelle, Pacific Northwest Laboratories Responses

Appendix E

Summary of Technical Steering Panel Comments and Battelle, Pacific Northwest Laboratories Responses

Document Number: PNWD-2223 HEDR

Document Title: "Radionuclide Releases to the Columbia River
from Hanford Operations, 1944-1971"

Summary Comments by: M. A. Robkin

Generally the report was very well-received. The report represents an excellent effort by C.M. Heeb and his BNW team members on the Source Term task. Explicit individual comments are included as an attachment to this summary. Generally comments included:

E.1

- There were editorial comments regarding organization, presentation or the need for additional inclusion of material, typos, and some suggested wording changes.
- There were only few comments on methodology or results mainly on the need for a more extensive justification for some of the procedures. There is also a need for a more complete discussion of some of the explicit items addressed by individual reviewers.
- Some particular points raised were the need to explain the discrepancies between the sum of individual nuclide activities and gross beta activity, to address the question of volatility of nuclides and to address the question of the division of data between use for determining distributions and use for validation.

Comment Number	Commenter	Page, Paragraph	Comment Summary	Resolution
TSP COMMENTS				
1.	B. Shleien	Full Report	This is very fine work, consistent with the iodine-131 source term work.	NA - Thank you.
2.	P. Klingeman	Full Report	No specific comments.	NA
3.	J. Till	General Comment	No comments on this report. This is a fine piece of work.	NA - Thank you.
4.	B. Schleien	General Comment	During certain periods the concentration of specific radionuclides or their daily releases based on concentration measurements were employed to estimate the source term (Groups 1, 2). To some extent the period during which these measurements were employed is indicated (page 3.8). However, only at the end of the document are the percentages given for each radionuclide determined by these methodologies. It would be highly preferable if a table was included indicating how the source term for each radionuclide was determined (Group 1, 2, 3, or 4), the period during which a method was employed and the uncertainty incurred. Also explain the uncertainty modality employed with each mode of estimation of the source term (see my comments in other reports for review, suggesting discussion of methods of expressing uncertainty in an appendix).	Agree. Text added in Section 3.3: "Tables in Appendix B show the range between minimum and maximum realizations. The extent of the range is an indication of what group the concentration data falls into. Group 1 realizations have the narrowest range, and Group 4 realizations the widest. The complete data source table is not appended due to its length." It would be informative to have a data source table, but the table would be too big for the report. It would have 28 years x 12 months/year x 12 (11 nuclides + gross beta) x 8 reactors = 32,256 entries. Each entry would have a "1,2,3, or 4" to indicate the source of the data. The results table in appendix B has 3 entries and all of the reactors are added together. It is 22 pages long. At a crude estimate the printed file would be 58.7 pages.

NA - No action.

Comment Number	Commenter	Page, Paragraph	Comment Summary	Resolution
4. (contd)			<p>The point above is particularly important because the source term information was employed to calculate radionuclide concentrations using the WSU-CHARIMA model (PNWD-2225 HEDR DRAFT). Some of the concentration levels indicated in PNWD-2225 HEDR may have the same measurement basis as those in PNWD-2223 HEDR. It is important for the reader to be aware of this.</p> <p>Up-front in the report there should be an indication that the source term, taken from each reactor release, has been employed to calculate concentration values of various radionuclides at various locations. These concentrations were in turn used to calculate radiation doses.</p>	<p>Agree. Text added in Section 3.1: "Release estimates in this study were based on the concentration of radionuclides in effluent water entering the river from retention basins." The measurement basis of PNWD-2223 HEDR and PNWD-2225 HEDR is completely different.</p> <p>Agree. Text added in Section 1.2: "Five of the eleven will be used by the river transport model as discussed in the next section. The source from each reactor to the river at the reactor outfall is provided to the river transport model."</p>
5.	M. Robkin	Page iii, Para 5, Line 5	HEDR is not addressing biological effects.	Agree. Changed wording to "the calculation of radiation doses."
6.	G. Roessler	Pg iii, Para 5, Line 5	We are not doing biological effects! I suggest you say "the calculation of radiation doses."	Agree. Changed wording to "the calculation of radiation doses."
7.	G. Roessler	Pg vi, Para 3 Line 6	Why was gross beta decayed for a standard 4 hours?	NA. Originally 4 hours decay was selected because that was the designed holdup time in the retention basins. See Section 4.3.12.
8.	M. Robkin	Pg vii, Table S.1	Please put in the number punctuation; i.e. x,xxx,xxx.	NA. Software (Lotus) in which table was generated does not permit number punctuation.
9.	G. Roessler	Pg ix, Last Para, Line 3	Take out space in "com pared".	Agree. Space removed.
10.	M. Robkin	Pg xi, Glossary	"activation product". An activation product may be formed by neutron capture in either a stable or radioactive parent.	Agree. Changed definition to "a radioactive or nonradioactive parent nuclide."
11.	G. Roessler	Pg xi, Activation Product	It could also be formed from a radioactive parent.	Agree. Changed definition to "a radioactive or nonradioactive parent nuclide."

NA - No action.

Comment Number	Commenter	Page, Paragraph	Comment Summary	Resolution
12.	G. Roessler	Pg xi, Gamma Ray	This is not always true. Suggest (from RHH): High-energy, short wavelength electromagnetic radiation (a packet of energy) emitted from the nucleus.	Agree. Changed definition as suggested
13.	M. Robkin	Pg xii, Normal Dist.	Is $c^{\#}$ meant to be the standardized unit variable normal distribution consistent with the definition given for μ . If so, there is a factor of $1/(2\pi)^{1/2}$ missing. Otherwise there is a factor of $1/(2\pi s^2)^{1/2}$ missing.	Agree. Replaced "defined as" with "proportional to," which makes it unnecessary to define the normalization factor.
14.	G. Roessler	Pg xii, Normal Dist.	$c^{\#}$ should be $s\sqrt{2\pi} c^{\#}$	Agree. Replaced "defined as" with "proportional to" which makes it unnecessary to define the normalization factor.
15.	S. Davis	Pg xii	"Gross beta" should be qualified as <u>nonvolatile</u> .	Agree. Added "and did not include volatile beta-emitting radionuclides."
16.	P. McGavran	Pg 1.1, Para 1.2	plus scattered particles should be articles	Agree. Changed "particles" to "articles."
17.	G. Roessler	Pg 1.1, Para 1, Line 1	Instead of "As a result...", "As a part..."	Agree. Changed "As result..." to "As a part..."
18.	G. Roessler	Pg 1.1, Para 3, Line 5	I think you mean articles instead of particles. If you mean particles then please specify whether they are neutrons, protons, or alphas.	Agree. Changed "particles" to "articles."
19.	M. Robkin	Pg 1.2	It seems mutually contradictory to say that Cr-51 was provided for validation purposes and at the same time there was not enough Cr-51 data available. More explanation is needed.	NA. Text says "Scandium-46, however, was not used for validation because not enough data were available." There was enough Cr-51 data for validation.

NA - No action.

Comment Number	Commenter	Page, Paragraph	Comment Summary	Resolution
20.	M. Robkin	Pg 2.1, Accuracy	The argument given about using all of the data on the basis of accuracy at the expense of validation seems very weak. Surely there is a best distribution of data to optimize the mutual requirements of accuracy and validation that does not fall at using 100% of the data for accuracy. More justification of not holding back any data for validation is needed. In particular, some words about how much accuracy would be lost for a given holdback of data are needed.	Agree. Changed "lowered the accuracy" to "increased the uncertainty." Although the increase in uncertainty would be significant, it is not possible to quantify it accurately.
21.	P. McGavran	Pg 2.1, Para 2	"Two others are being used for validation." This is confusing since Scandium is not being used.	Agree. Changed text to read "Originally, two radionuclides were to be used for model validation. Subsequently, it was found that data for one of these radionuclides (scandium-46) was insufficient, and the radionuclide was excluded from validation studies."
22.	G. Roessler	Pg 3.1, Para 3	Gamma-ray and alpha-particle.	Agree. Changed to include hypkens.
23.	G. Caldwell	Pg 3.3, Para 2, Line 4	The meaning of this sentence is unclear. What are "bred actinides"? Are they the same as activation products?	"Bred actinides" changed to "actinides."
24.	M. Robkin	Pg 3.4, Section 3.2.5, Para 1, Last Sentence	Does the phrase "specific activities" refer to activity density as in Ci/ml or to specific nuclides? "Specific activity" has the formal meaning in the literature of "activity density" that may not be meant here.	Agree. Added "radionuclide-specific activities" in place of "specific activities."
25.	B. Shleien	Pg 3.4, Sec 3.2.5, Para 2, Last Line	If most of the radionuclides of interest contributed only a small fraction of the total gross beta activity, what was the majority of this activity due to? If due to short-lived radionuclides it would be important to note, and also indicate the range of their half-lives. If tritium (volatile) and/or S-35 (also volatile to a lesser extent) were present, it's also important to indicate because of their biological importance.	Agree. Text added in Section 3.2.5: "The major short-lived components of the non-volatile gross beta measurements were never definitively identified and their effect on dose is unknown. It was generally known that mangano-56 was the largest single contributor (see Section 4.3.12)."

Comment Number	Commenter	Page, Paragraph	Comment Summary	Resolution
26.	S. Davis	Pg 3.4, Para 3 from bottom of page	This paragraph states that the sum of beta emissions from individual radionuclides is less than the total gross beta. However, Table S-1 shows a sum of curies from Mn-56 alone which is greater than gross beta. This discrepancy was explained verbally by Mr. Heeb in the January 1994, TSP meeting. Nevertheless, it should be included in this document.	Agree. Text added in Section 3.2.5: "The major short-lived components of the non-volatile gross beta measurements were never definitively identified and their effect on dose is unknown. It was generally known that manganese-56 was the largest single contributor (see Section 4.3.12)."
27.	M. Robkin	Pg 3.5, Section 3.3, Para 3, Item 4	How are the mean and standard deviation computed? Category 4 seems to cover the case of no data at all. More words are needed to explain what data are used to provide category 4 information.	Agree. Items 1 through 4 were rewritten for a more accurate description of the actual lognormal sampling means and standard deviations.
		Section 3.3.1, Line 5	Replace "allows" by "allow".	Agree. Text changed.
28.	G. Caldwell	Pg 3.7, Lines 4-15	Were these substances activated by passage through the reactor and if so how much radioactivity was created and released to the river. What quantities of these substances were used, how much was activated and released? Were any of these the source of the key radionuclides?	Agree. Text added in Section 3.3.2.2: "In many cases, water treatment chemicals contained parent radionuclides. The release estimates are based on measurements that include whatever additives were being used."
29.	M. Robkin	Pg 3.8, Para 2	Second bullet gives time period as 1944-1958. Third bullet gives periods in 1951 and 1957. These two periods overlap and it is confusing to say that data were incomplete in the second bullet and then say it again in the third bullet. Why are not the third bullet dates included in the second bullet? A few more words would help.	Agree. Text added in Section 3.3.2.4: "Different independent variable data were available for the periods of overlapping time. Regressions that had a lower uncertainty were estimated using specific data for these periods."
30.	M. Robkin	Pg 3.12, Equation 3.4	Where is it explained how "W" is determined. There is no explanation or reference given.	NA. Explained in Section 4.0 under DGRAMS.

NA - No action.

Comment Number	Commenter	Page, Paragraph	Comment Summary	Resolution
31.	G. Caldwell	Pg 3.13, Para 3, Line 2	Is "argument" the correct word? If so, does it mean the same as "factor"?	Agree. Replaced "as an argument" with "an independent variable." It is an argument if the iodine-131 and neptunium-239 regressions are regarded as functions of two arguments: specific power and burnup. It is more accurate to identify it as an independent variable in a regression.
32.	M. Robkin	Pg 4.1, Section 4.1, Para 2,	What is meant by the "initial counting rate of the sample collected at the outlet". Does this mean the count rate taken at the instant the sample was collected? Does it mean count rate corrected for decay from collection time to counting time, and if so, how was the half-life known? More explanation is needed.	Agree. Removed "initial" and added "as determined by the effluent water beta decay curve."
33.	M. Robkin	Pg 4.5, Para 2 below Table, Line 5	According to the table, Class 3 ranges from 50-100 g. The third histogram class is 100-150 g. Why are there any of "Class 3 Medium" releases in this histogram class since the amounts do not overlap?	Agree. Changed class 3 from "50 to 100 grams" to "50 to 150 grams." This makes the allocation correct; i.e., half of it is in the 100 to 150 class and half of it is in the 50 to 100 class. The DGRAMS distribution function used is unchanged. The original calculation used 50 to 150 grams, but in writing the text an error was made in the range for class 3.
34.	B. Shleien	Pg 4.8, Last Para	Indicates that for gross beta, Mn-56, Ga-72, and Na-24 the summary values do not constitute a source term for these materials below the F reactor. Calculation of doses from the river pathway using a simple dilution/decay model gives significant doses to the lower large intestine from Ga-72 and Na-24 for a one-day average transit time equivalent to a Pasco, WA location (see my memo to J. Till, January 20, 1994). A different transit time would certainly change this. Please clarify, if possible, if there is point along the river which is applicable, on the average, to the total source term release.	NA. Report size dictated that a summary of releases be given rather than specific data from the eight individual source terms. If a more accurate source term is required, individual reactor sources are available on electronic medium.
35.	M. Robkin	Pg 4.11, Section 4.3.1, Para 3, Line 3	Insert "elements in the" between "the" and "cooling water"	Agree. Changed as suggested.

NA - No action.

Comment Number	Commenter	Page, Paragraph	Comment Summary	Resolution
36.	M.L. Blazek	Pg 5.1	States, in part, that "The release estimates are based on all activity concentration measurements available in the historic Hanford documents." This statement should be changed to read "available in those historic documents reviewed."	Changed as suggested. The documents reviewed form a complete record of the only routine activity measurements that were made. That is, they cover the time periods with reports of whatever radionuclide activity concentrations were being measured in that period. There could have been measurements made by cleared and authorized researchers from other agencies or institutions, but the lack of known records suggests that is unlikely.

E.8

NA - No action.

Distribution

No. of
Copies

No. of
Copies

Offsite

Technical Steering Panel

D. S. Barth
University of Nevada
4505 Maryland Parkway
Las Vegas, NV 89154

W. A. Bishop
P.O. Box 47651
Olympia, WA 98504-7651

M. L. Blazek
Oregon Department of Energy
625 Marion Street N.E.
Salem, OR 97310

G. G. Caldwell
Tulsa City-County Health Dept.
4616 East 15th Street
Tulsa, OK 74112

S. N. Davis
Dept. of Hydrology and Water Resources
Building 11
University of Arizona
Tucson, AZ 85721

N. J. Germond
224 Iron Mountain Blvd.
Lake Oswego, OR 97034

P. C. Klingeman
Civil Engineering Dept.
Apperson Hall 202
Oregon State University
Corvallis, OR 97331-2302

K. J. Kopecky
Fred Hutchinson Cancer Research Center
1124 Columbia Street
Seattle, WA 98104

P. D. McGavran
Dept. of Health and Welfare
450 W. State Street, 4th Floor
Boise, ID 83720-5450

R. L. Morrill
Dept. of Geography, DP-10
University of Washington
Seattle, WA 98195

A. H. Murphy
c/o Max Planck Institute for Meteorology
Bundesstrasse 55
20146 Hamburg
Germany

D. W. Price
Agricultural Economics
Hulbert Hall Room 211
Washington State University
Pullman, WA 99164-6210

M. A. Robkin
Radiological Sciences, SB-75
University of Washington
Seattle, WA 98195

G. S. Roessler
Route 1, Box 139H
Elysian, MN 56028

B. Shleien
2421 Homestead Drive
Silver Springs, MD 20902

**No. of
Copies**

**No. of
Copies**

A. P. Slickpoo, Sr.
809 Nez Perce Lane
P.O. Box 311
Kamiah, ID 83536

J. E. Till
Route 2 Box 122
Neeses, SC 29107

D. E. Walker, Jr.
P.O. Box 4147
Boulder, CO 80306

Other

- 2 DOE Office of Scientific and
Technical Information
Technical Information Center
P.O. Box 62
Oak Ridge, TN 37830

B. Aripa
Environmental Trust
Colville Confederated Tribes
P.O. Box 150
Nespelem, WA 99155

D. Bonga
Kalispel Tribe
P.O. Box 39
Usk, WA 99180

B. G. Brooks, EH-421
Room J-112
U.S. Department of Energy
Germantown, MD 20545

- 18 K. CharLee
Technical Support and
Public Information
P.O. Box 47651
Olympia, WA 98504-7651

J. P. Corley
2213 Torbett
Richland, WA 99352

L. E. Denham
Centers for Disease Control
and Prevention
4770 Buford Highway N.E., F-35
Atlanta, GA 30341-3724

DNR Hanford Projects
Attn: J. R. Wilkinson
Confederated Tribes of the Umatilla
Indian Reservation
P.O. Box 638
Pendleton, OR 97801

M. R. Donnelly
Public Health Service
Centers for Disease Control
and Prevention
2201 Sixth Avenue, RX-22
Seattle, WA 98121

D. Ernst
Planning Department
Spokane Tribe
P.O. Box 100
Wellpinit, WA 99040

H. A. Haerer
Golder Associates, Inc.
4104 148th N.E.
Redmond, WA 98052

R. Jim
Confederated Tribes and Bands
of the Yakama Indian Nation
P.O. Box 151
Toppenish, WA 98948

No. of
Copies

D. Powaukes
Nez Perce Tribe
P.O. Box 365
Lapwai, ID 83540

M. Queahpama
Vital Statistics
Confederated Tribes of the Warm
Spring Reservation
P.O. Box C
Warm Springs, OR 97761

M. C. Richmond
Department of Civil and
Environmental Engineering
Washington State University
Pullman, WA 99164-2910

L. Seyler
Coeur d'Alene Tribe
Plummer, ID 83851

J. C. Simpson
Golder Associates, Inc.
4104 148th N.E.
Redmond, WA 98052

M. Squeochs
Confederated Tribes and Bands
of the Yakama Indian Nation
P.O. Box 151
Toppenish, WA 98948

L. Stembridge
HEAL
1408 W. Broadway Avenue
Spokane, WA 99201-1902

W. L. Templeton
2331 Enterprise
Richland, WA 99352

No. of
Copies

J. P. Thomas
HHIN Resource Center
1719 Smith Tower
506 Second Avenue
Seattle, WA 98104

Onsite

6 DOE Richland Operations Office

R. F. Brich, TSD A5-55
Public Reading Room (5) A1-65

36 Battelle, Pacific Northwest
Laboratories

D. M. Anderson	K8-15
D. J. Bates	K7-34
G. L. Black (5)	B1-34
S. D. Cannon	B1-34
D. E. Deonigi	K8-11
P. W. Eslinger	B1-34
W. T. Farris	K3-54
M. D. Freshley	K6-77
R. O. Gilbert	K7-34
W. A. Glass (2)	K3-53
R. H. Gray	B1-34
S. P. Gydesen	P8-55
G. L. Harvey	K2-02
C. M. Heeb (5)	K8-34
T. A. Ikenberry	K3-54
B. A. Napier	K3-54
J. V. Ramsdell	K6-55
D. B. Shipler	B1-34
J. K. Soldat	K3-54
S. A. Stage	K6-55
M. E. Thiede	K6-52
W. H. Walters	K6-60
Publishing Coordination	K1-06
Records Center (2)	K3-70
Technical Library (2)	P8-55

DATE

FILMED

7/19/94

END