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Page 1 of 2

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13a. Description of Change 1) Incorporate ECN 645035 (which revised the tables to reflect: improved cross section data, a decision to use accountability data as the basis for total Pu, a corrected methodology for selection of the heat generation basis feed, and a revised decay date); 2) Add section 3.3.3.1 ("Limitations of the Methodology") to expand the description of the approach used to calculate the inventory values and explain why that approach yields conservative results; 3) Change the pre-irradiation braze beryllium value (in sec 3.3.1) from 137 g per assembly to 1.37 g per assembly (this was a "type-o"--the 142 kg value remains unchanged in Table 3.4). Revised chemical tables (Tables 2.1 and 3.4) 4) Incorporate miscellaneous, non-substantive, editorial changes throughout the document, including elimination of Fig. 3.2 ("Fuel Storage Areas...")			
14a. Justification (mark one) Criteria Change ☐ Design Improvement ☐ Environmental ☐ Facility Deactivation ☐ As-Found ☐ Facilitate Const ☐ Const. Error/Omission ☐ Design Error/Omission ☐			
14b. Justification Details Availability of improved neutron cross section data justified changes to tables -- See ECN 645035 for additional details. Typographical error on beryllium value in section 3.3.1 required correction. Fig. 3.2 conveyed no unique information. The quantities listed for fission products in the "chemical tables" (Tables 2.1 and 3.4) were not consistent with results from the ORIGEN2 calculations.			
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ADDITIONAL

105-K BASIN MATERIAL DESIGN BASIS FEED DESCRIPTION FOR SPENT NUCLEAR FUEL PROJECT FACILITIES

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Abstract: Revisions 0 and 0A of this document provided estimated chemical and radionuclide inventories of spent nuclear fuel and sludge currently stored within the Hanford Site's 105-K Basins. This Revision (Rev. 1) incorporates the following changes into Revision 0A: 1) updates the tables to reflect: improved cross section data, a decision to use accountability data as the basis for total Pu, a corrected methodology for selection of the heat generation basis fee, and a revised decay date; 2) adds section 3.3.3.1 to expand the description of the approach used to calculate the inventory values and explain why that approach yields conservative results; 3) changes the pre-irradiation braze beryllium value (in Sec. 3.3.1) from 137 g per assembly to 1.37 g per assembly to correct a typographical error (the 142 kg value remains unchanged in Table 3.4).

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105-K Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities

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0A	Released via ECN# 191383 (12/20/95): Replace ⁶⁰ Co and ²⁴⁴ Cm values in tables 2.2, 2.4, 2.5, 2.7, 3.5, 3.6, 3.7, 3.8, 3.9, and 3.10 to reflect better estimates of actual inventory. Change wording as shown on pages 29 and D-3 to account for change in ⁶⁰ Co inventory numbers.	WL Willis 12/12/95	JR Frederickson 12/20/95
RS 1	Released via ECN# 645037 (12/12/97): Incorporate ECN#645035 (which revised the tables to reflect: improved cross section data; a decision to use accountability data as the basis for total Pu; a corrected methodology for selection of the heat generation basis feed; and a revised decay date). Replace RADNUC output listings in Appendix B with updated runs for revised decay date. Add section 3.3.3.1 ("Limitations of the Methodology"). Change the pre-irradiation braze beryllium value (in sec 3.3.1) from 137 g per assembly to 1.37 g per assembly (this was a "type-o"--the 142 kg value remains unchanged in Table 3.4). Remove Figure 3.2, "Fuel Storage Areas..." and renumber subsequent figures in sec 3. Incorporate miscellaneous, non-substantive editorial changes throughout the document. Revise chemical tables (Tables 2.1 & 3.4).	DL Sherrell 12/12/97 <i>DL Sherrell</i>	JR Frederickson 12/12/97 <i>JR Frederickson</i> 12/11/97

**105-K BASIN MATERIAL DESIGN BASIS FEED
DESCRIPTION FOR SPENT NUCLEAR FUEL PROJECT FACILITIES**

W.L. Willis and A.N. Praga

NOVEMBER 1997

Table of Contents

1.0 INTRODUCTION	1
1.1 BACKGROUND	1
1.2 PURPOSE AND SCOPE	1
1.2.1 Purpose and Scope of Revision 1	2
2.0 SUMMARY	3
2.1 NOMINAL DESIGN FEED	4
2.1.1 Inventory Totals	4
2.1.2 Unit Mass Uranium Inventory	6
2.2 SHIELDING DESIGN FEED BASES	8
2.3 SAFETY/REGULATORY ASSESSMENT FEED	8
2.4 HEAT GENERATION BASIS FEED	9
2.5 CRITICALITY ASSESSMENT FEED	10
3.0 N REACTOR FUEL	11
3.1 K BASIN DESCRIPTION	11
3.1.1 Irradiated Fuel Storage at the KE and KW Basins	11
3.2 N REACTOR FUEL DESCRIPTION	13
3.2.1 Fuel Fabrication	16
3.2.2 Burnup and Time Since Discharge	17
3.3 NOMINAL N REACTOR DESIGN BASIS FEED	20
3.3.1 N Reactor Fuel Chemical Content Methodology	20
3.3.2 N Reactor Fuel Chemical Inventory	21
3.3.3 N Reactor Fuel Radionuclide Content Methodology	22
3.3.3.1 Limitations of the Methodology	22
3.3.4 N Reactor Fuel Radionuclide Content	23
3.4 SHIELDING DESIGN BASIS FEED	27
3.5 SAFETY/REGULATORY ASSESSMENT DESIGN BASIS FEED	27
3.6 HEAT GENERATION BASIS FEED	28
3.7 CRITICALITY DESIGN BASIS FEED	28
4.0 SINGLE PASS REACTOR FUEL	33
4.1 BACKGROUND	33
4.2 FUEL DESCRIPTION	34
4.2.1 Fuel Fabrication	36
4.3 FUEL INVENTORY	37
4.4 RADIONUCLIDE CONTENT	38
4.5 FUEL STORAGE	39
5.0 105-K BASIN SLUDGE	40
5.1 SLUDGE SOURCES	40
5.2 SLUDGE LOCATION	40
5.3 SLUDGE DEPTH	43
5.4 CURRENT INVENTORY BASIS	43
5.5 105-K EAST BASIN SLUDGE	44
5.6 105-K WEST BASIN SLUDGE	48
6.0 REFERENCES	49
APPENDIX A	A-1
APPENDIX B	B-1

APPENDIX C	C-1
APPENDIX D	D-1
APPENDIX E	E-1
APPENDIX F	F-1
APPENDIX G	G-i

List of Figures

Figure 3.1: 105-K Basins	12
Figure 3.2: N Reactor Fuel Storage Canisters	14
Figure 3.3: N Reactor Mark IV Fuel Assembly	15
Figure 3.4: K Basin N Reactor Fuel Inventory Characteristics	19
Figure 3.5: ¹³⁷ Cs Shielding Design Basis Selection	29
Figure 4.1: Single Pass Reactor Fuel Elements	35
Figure 5.1: Sludge Location in the 105-K Basins	42

List of Tables

Table 1.1: Spent Fuel Project Facility Design Feeds	2
Table 2.1: Summary Chemical Constituents	5
Table 2.2: Summary Radionuclide Constituents	6
Table 2.3: Average Chemical Inventory/Unit Mass Uranium Summary	7
Table 2.4: Average Radionuclide Inventory/Unit Mass Uranium Summary	7
Table 2.5: Shielding Design Basis Summary	8
Table 2.6: Safety/Regulatory Assessment Design Basis Summary	9
Table 2.7: Heat Generation Basis Summary	10
Table 3.1: 105-N Reactor Fuel Element Description	17
Table 3.2: N Reactor Mark IV Fuel Burnup Summary	18
Table 3.3: N Reactor Mark 1A Fuel Burnup Summary	18
Table 3.4: N Reactor Fuel Chemical Inventory	21
Table 3.5: Radionuclide Inventory of the Combined K Basins	24
Table 3.6: K-East Basin Radionuclide Inventory	24
Table 3.7: K-West Basin Radionuclide Inventory	25
Table 3.8: Shielding Design Basis	30
Table 3.9: Safety/Regulatory Assessment Design Basis	31
Table 3.10: Safety/Regulatory Assessment Basis	32
Table 4.1: Operating Periods of the Hanford Single Pass Reactors	33
Table 4.2: Physical Characteristics of Single Pass Reactor Fuel	36
Table 4.3: Total Chemical Inventory in Single Pass Reactor Fuel Metals	36
Table 4.4: Single Pass Reactor Fuel Elements Stored in the K Basins	38
Table 4.5: Total Activity of Selected Radionuclides in SPR Fuel	38
Table 5.1: Physical Properties of 105-K East Basin Sludge	45
Table 5.2: Chemical Constituents in 105-K East Basin	46
Table 5.3: Radionuclides in 105-K East Basin Sludge	46
Table 5.4: 105-K East Basin Radionuclide Totals	47
Table 5.5: Physical Properties of 105-K West Basin Sludge	48
Table 1.1: Spent Fuel Project Facility Design Feeds	2
Table 2.1: Summary Chemical Constituents	5
Table 2.2: Summary Radionuclide Constituents	6
Table 2.3: Average Chemical Inventory/Unit Mass Uranium Summary	7
Table 2.4: Average Radionuclide Inventory/Unit Mass Uranium Summary	7

Table 2.5:	Shielding Design Basis Summary	8
Table 2.6:	Safety/Regulatory Assessment Design Basis Summary	9
Table 2.7:	Heat Generation Basis Summary	10
Table 3.1:	105-N Reactor Fuel Element Description	17
Table 3.2:	N Reactor Mark IV Fuel Burnup Summary	18
Table 3.3:	N Reactor Mark 1A Fuel Burnup Summary	18
Table 3.4:	N Reactor Fuel Chemical Inventory	21
Table 3.5:	Radionuclide Inventory of the Combined K Basins	24
Table 3.6:	K-East Basin Radionuclide Inventory	25
Table 3.7:	K-West Basin Radionuclide Inventory	26
Table 3.8:	Shielding Design Basis	30
Table 3.9:	Safety/Regulatory Assessment Design Basis	31
Table 3.10:	Safety/Regulatory Assessment Basis	32
Table 4.1:	Operating Periods of the Hanford Single Pass Reactors	33
Table 4.2:	Physical Characteristics of Single Pass Reactor Fuel	36
Table 4.3:	Total Chemical Inventory in Single Pass Reactor Fuel Metals	37
Table 4.4:	Single Pass Reactor Fuel Elements Stored in the K Basins	38
Table 4.5:	Total Activity of Selected Radionuclides in SPR Fuel	38
Table 5.1:	Physical Properties of 105-K East Basin Sludge	45
Table 5.2:	Chemical Constituents in 105-K East Basin	46
Table 5.3:	Radionuclides in 105-K East Basin Sludge	47
Table 5.4:	105-K East Basin Radionuclide Totals	47
Table 5.5:	Physical Properties of 105-K West Basin Sludge	48

List of Acronyms

CEDE	cumulative effective dose equivalents
DESH	DE&S Hanford Inc.
DOE	U.S. Department of Energy
EIS	Environmental Impact Statement
GEA	gamma energy analysis
I&E	internally and externally cooled
MCO	multi-canister overpack
MWD	megawatt-day(s)
MT	metric ton(s)
MTHM	metric ton(s) heavy metal
MTU	metric ton(s) uranium
PUREX	plutonium uranium extraction
SPR	single pass reactor
SNF	spent nuclear fuel
USQ	unreviewed safety question

105-K BASIN MATERIAL DESIGN BASIS FEED DESCRIPTION FOR SPENT NUCLEAR FUEL PROJECT FACILITIES

1.0 INTRODUCTION

1.1 BACKGROUND

Metallic uranium Spent Nuclear Fuel (SNF) is currently stored within two water filled pools, 105-KE Basin (KE Basin) and 105-KW Basin (KW Basin), at the United States Department of Energy (U.S. DOE) Hanford Site, in south eastern Washington State. DE&S Hanford Inc. (DESH) Spent Nuclear Fuel Project (SNF Project) is responsible to DOE for operation of these fuel storage pools and for the 2100 metric tons of SNF materials that they contain. The SNF Project mission includes safe removal and transportation of all SNF from these storage basins to a new storage facility in the 200 East Area. To accomplish this mission, the SNF Project modifies the existing KE Basin and KW Basin facilities and constructs two new facilities: the 100 K Area Cold Vacuum Drying Facility (CVDF), which drains and dries the SNF; and the 200 East Area Canister Storage Building (CSB), which stores the SNF.

1.2 PURPOSE AND SCOPE

The purpose of this document is to describe the design basis feed compositions for materials stored or processed by SNF Project facilities and activities. This document is not intended to replace the *Hanford Spent Fuel Inventory Baseline* (WHC 1994b), but only to supplement it by providing more detail on the chemical and radiological inventories in the fuel and sludge.

A variety of feed definitions is required to support evaluation of specific facility and process considerations during the development of these new facilities. Six separate feed types have been identified for development of new storage or processing facilities. The approach for using each feed during design evaluations is to calculate the proposed facility flowsheet assuming each feed. The process flowsheet would then provide a basis for material compositions and quantities which are used in follow-on calculations.

Table 1-1 provides a list of the proposed feeds and the intended purpose of each feed definition.

Table 1.1: Spent Fuel Project Facility Design Feeds

Design Feed	Description/Purpose
Nominal Feed	Average feed to the facility over the plant life. Used to design systems that provide suitable performance over the entire plant life. Among other things, this feed is used to design shielding that will maintain radiation doses to workers as low as reasonably achievable (ALARA).
Shielding Basis Feed	A feed with the highest gamma shielding requirements as compared to the nominal feed (^{137}Cs is the dominate gamma contributor in either case). This feed is used to evaluate gamma shielding provided by the plant structure and equipment to determine if potential peak radiation dose rates are acceptable. This feed represents material which would be in the facility for short campaigns and would not be used for evaluations of performance over the entire plant life.
Safety/Regulatory Assessment Feed	A feed with bounding ^{241}Am , plutonium isotopes, and ^{90}Sr radionuclides which, if released, dominate the estimate of dose to personnel (both onsite and offsite). This feed is used to analyze design basis accidents and support definition of mitigation systems. This feed represents material which would be in the facility for short campaigns and would not be used for evaluations of performance over the entire plant life.
Heat Generation Basis Feed	A feed with bounding amounts of those isotopes that dominate heat dissipation requirements. This feed is used to define heat removal requirements provided by the facility systems. The Heat removal basis feed represents the upper bound of the operating envelope for feed material which can be handled by the facility without special operating procedures or evaluations to determine if supplemental heat removal is required. This feed represents material which would be in the facility for short campaigns and would not be used for evaluations of performance over the entire plant life.
Criticality Assessment Feed	A feed with bounding fissile material content used to define criticality controls (e.g., geometric favorable equipment designs) which may impact the system design. This feed represents an upper bound of the operating envelope for material fissile content which could be introduced into the facility without special operating procedures or additional criticality safety evaluations. The feed represents material which would be in the facility for short campaigns and would not be used for evaluations of performance over the entire plant life. (Analysis to be completed in future revision).

1.2.1 Purpose and Scope of Revision 1

The primary purpose of Revision 1 is to incorporate changes into the radionuclide tables that result from: 1) the availability of improved cross section data, 2) a decision to use the accountability data as the basis for total Pu, and 3) additional radioactive decay that would accrue between the original project startup date (12/31/97) and a May 31, 1998 startup date. Most of the radionuclide concentrations have been reduced with the exception of $^{239,240}\text{Pu}$ and Am, which have increased in some tables.

In addition to incorporation of the adjusted inventory values, Revision 1 also provides additional references for certification of the computer programs used to calculate the inventory values and adds a section (3.3.3.1) to explain the limitations of the calculation method.

Finally, Revision 1 incorporates miscellaneous editorial changes.

2.0 SUMMARY

Chemical and radionuclide inventories are estimated for fuel and sludge stored in the KE and KW Basins. Inventory estimates are presented for the nominal (or overall average) inventory, selected shielding design basis, shielding design basis, selected safety/regulatory assessment design basis, selected heat generation basis, and selected criticality assessment basis materials. This document is intended to be revised as needed to support the Spent Nuclear Fuel Project.

The nominal design basis is presented as a total material inventory in the combined basins and as an average inventory per metric ton uranium (MTU). Bounding design basis feed compositions are also presented on a unit mass of uranium basis. The bounding feeds were selected by considering the range of radionuclide compositions in N Reactor and Single Pass Reactor fuel predicted to exist within the basins as a result of varying exposures and decay periods reported for each key (or group of fuel assemblies) in the accountability database.

Accountability records were used as a basis for the quantity, exposure variation, and decay time variation of stored fuel. Chemical constituents were estimated from the mid-point of fuel fabrication material impurity specification ranges. Variation of the chemical constituents over the specification ranges was not incorporated in the current design basis feed estimates. Radionuclide estimates were effectively derived from the following two computer models:

- 1) ORIGEN2 runs for N Reactor fuel provided radionuclide compositions for discharged fuel at selected exposure levels.
- 2) RADNUC 2A combined the ORIGEN2 fuel compositions at the selected points with accountability data on exposure and decay time to generate a composite radionuclide inventory decayed to a common date. Interpolations were used to estimate radionuclides in blocks of fuel irradiated to exposure levels between the ORIGEN 2 point projections.

Tests of the radionuclide estimate method indicate the interpolation results are within 2% of the ORIGEN2 projections for exposures near the interpolation points and within 10% of expected results for exposures between interpolation points. A comparison of ORIGEN2 and RADNUC outputs is included in the RADNUC documentation (WHC 1995a).

The sludge inventory is currently limited to estimates of components in the K-East Basin. The component estimates are based on chemical and radionuclide analyses of four samples, two samples from the weasel pit, one sample from the sandfilter backwash pit, and one sample from the approximate center of the basin floor. While not quantified, significant uncertainty in the sludge component estimates should be expected.

Development of the feed basis data emphasized best estimates of the actual bounds of material compositions, with a minimum of conservatism. This approach was used in order to allow future designs based on the bounds to apply known safety factors and avoid excessive over-design. Some conservatism is currently included in the radionuclide design basis bounds as a result of

the following:

- 1) The decay date selected for presenting radionuclide inventories and bounds is May 31, 1998. This is conservative (i.e., early) in view of the July, 1999 official project startup date, and therefore, radionuclide inventories (and associated heat loads) could be reduced to reflect additional decay.
- 2) The radionuclide inventories were developed using the computer codes as described above. The plutonium inventory predicted was larger than that reported in the accountability data. While, the plutonium 239 and 240 isotope values in the overall total tables have been adjusted to reflect the accountability plutonium inventory numbers, no adjustment has been made to the other isotope numbers in the tables (see section 3.3.3.1).
- 3) The shielding design basis includes a limited degree of conservatism due to selecting 16% ^{240}Pu content fuel, aged 13.5 years to 5/31/1998, as the composition basis. This represents material containing approximately 6.5% more ^{137}Cs and ^{90}Sr per MTU than the highest actual fuel content identified in the basin inventory (15.7% ^{240}Pu content Mark IV fuel, aged 14.5 years to 5/31/1998).

2.1 NOMINAL DESIGN FEED

The nominal design feed is an average of the current inventory of spent fuel stored in the K Basins. Tables in this section provide a summary of the current nominal design basis feed.

2.1.1 Inventory Totals

Tables 2.1 and 2.2 contain a summary of the nominal design basis feed information estimated for the entire inventory of fuel and sludge in the combined K Basins. Table 2.1 includes the chemical inventories, and Table 2.2 contains a summary of the radionuclide inventory for N Reactor fuel, Single Pass Reactor (SPR) fuel, K Basin sludge, and totals for each constituent.

Table 2.1: Summary Chemical Constituents

Chemical	N Reactor Fuel (kg)	SPR Fuel (kg)	Sludge (kg)	Total (kg)
Al	1.70e+03	1.20e+02	4.32e+02	2.25e+03
As			8.06e-01	8.06e-01
B	6.05e-01	1.20e-03		6.06e-01
Ba			2.19e+00	2.19e+00
Be	1.63e+02		2.00e-01	1.63e+02
C	1.20e+03	2.54e+00		1.20e+03
Ca			8.34e+01	8.34e+01
Cd	6.05e-01	3.60e-03	1.77e+00	2.38e+00
Co	1.54e+00	1.20e-03	4.11e-01	1.95e+00
Cr	2.88e+02	2.20e-01	1.72e+01	3.05e+02
Cu	1.66e+02	1.86e-01	6.81e+00	1.73e+02
Fe	9.41e+02	1.38e+00	4.51e+03	5.45e+03
H	8.06e+00			8.06e+00
Hf	3.02e+01			3.02e+01
K			2.93e+00	2.93e+00
Li		9.60e-03	1.44e+00	1.45e+00
Mg	5.58e+01	8.45e-02	1.84e+01	7.43e+01
Mn	6.03e+01	8.45e-02	7.02e+00	6.74e+01
Mo	7.54e+00		6.91e-01	8.23e+00
N	1.70e+02	3.38e-01		1.70e+02
Na	3.02e+00		3.80e+00	6.82e+00
Nd			2.51e+00	2.51e+00
Ni	2.94e+02	1.74e+00	1.75e+00	2.97e+02
O	6.53e+00			6.53e+00
P			7.84e+00	7.84e+00
Pb	1.52e+01	4.58e-01	9.43e+00	2.51e+01
S			6.42e+00	6.42e+00
Sb			4.04e+00	4.04e+00
Si	2.77e+02		3.64e+01	3.13e+02
Sm	6.54e+00		8.97e+00	1.55e+01
Sn	2.19e+03			2.19e+03
Sr	7.33e+01		6.10e-01	7.39e+01
Ti	7.54e+00		2.58e+00	1.01e+01
Tl			2.87e+00	2.87e+00
V	7.54e+00		6.97e+00	1.45e+01
W	7.68e+00			7.68e+00
Kr	7.71e+01			7.71e+01
Xe	1.04e+03			1.04e+03
Zn			1.92e+01	1.92e+01
Zr	1.48e+05		3.22e+00	1.48e+05
U	2.10e+06	3.38e+03	1.46e+03	2.10e+06
Np	8.11e+01			8.11e+01
Pu	4.12e+03		4.88e+00	4.12e+03
Am	1.09e+02		2.28e-01	1.09e+02
Cm	1.78e-02			1.78e-02
Se	1.20e+01			1.20e+01
Tc	1.70e+02			1.70e+02
Pd	1.33e+02			1.33e+02
I	4.61e+01			4.61e+01
Cs	4.65e+02			4.65e+02
Pm	5.0e+01			5.0e-01
other			1.73e+04	1.73e+04
Total	2.26e+06	3.51e+03	2.40e+04	2.29e+06

Table 2.2: Summary Radionuclide Constituents

(Decay date basis: May 31, 1998)

Isotope	N Fuel (Ci)	SPR Fuel (Ci)	Sludge (Ci)	Total (Ci)
H-3	3.66e+04	5.94e+02		3.72e+04
C-14	6.93e+02			6.93e+02
Fe-55	1.84e+03			1.84e+03
Co-60	3.96e+03		2.30e+01	3.98e+03
Ni-63	4.49e+03	6.62e+01		4.56e+03
Kr-85	5.90e+05	6.04e+02		5.91e+05
Se-79	8.62e+01			8.62e+01
Sr-90	1.01e+07	1.27e+04	1.28e+03	1.01e+07
Y-90	1.01e+07	1.27e+04	1.28e+03	1.01e+07
Zr-93	4.00e+02			4.00e+02
Nb-93m	2.47e+02			2.47e+02
Tc-99	2.88e+03	6.76e+00		2.89e+03
Ru-106	1.82e+03			1.82e+03
Rh-106	1.82e+03			1.82e+03
Sn-126	1.56e+02			1.56e+02
Sb-125	3.35e+04			3.35e+04
Te-125m	8.18e+03			8.18e+03
I-129	6.37e+00			6.37e+00
Cs-134	1.59e+04	2.47e+00		1.59e+04
Cs-135	7.75e+01			7.75e+01
Cs-137	1.32e+07	3.25e+04	9.72e+02	1.32e+07
Ba-137m	1.25e+07	3.08e+04	9.20e+02	1.25e+07
Ce-144	9.14e+02			9.14e+02
Pr-144	9.03e+02			9.03e+02
Pm-147	4.62e+05			4.62e+05
Sm-151	1.76e+05	3.97e+02		1.76e+05
Eu-152	9.45e+02			9.45e+02
Eu-154	1.07e+05	3.69e+02	2.61e+01	1.07e+05
Eu-155	2.21e+04	7.40e+01	1.35e+01	2.22e+04
U-234	8.74e+02			8.74e+02
U-235	3.37e+01			3.37e+01
U-236	1.27e+02			1.27e+02
U-238	6.96e+02			6.96e+02
Np-237	5.72e+01			5.72e+01
Pu-238	1.11e+05	1.42e+02	6.51e+01	1.11e+05
Pu-239	2.18e+05	4.06e+02	2.60e+02	2.19e+05
Pu-240	1.19e+05	6.42e+02	1.43e+02	1.20e+05
Pu-241	6.68e+06	1.46e+04	5.22e+03	6.70e+06
Pu-242	5.49e+01		4.36e-02	5.49e+01
Am-241	3.75e+05	1.52e+03	7.82e+02	3.77e+05
Cm-242	1.62e+02			1.62e+02
Cm-244	1.44e+03			1.44e+03

Note: This table provides a summary of key radionuclides for N Reactor fuel. A complete listing of estimated radionuclides in N Reactor fuel is provided in Table 3.3.

2.1.2 Unit Mass Uranium Inventory

Tables 2.3 and 2.4 contain an estimate of the chemical and radionuclide nominal design basis inventory on a uranium mass basis. Table 2.3 is a summary of the chemical composition of the fuel on a uranium mass basis. Table 2.4 is a summary of the radionuclide inventory on a uranium mass basis.

Table 2.3: Average Chemical Inventory/Unit Mass Uranium Summary

Chemical	Mass (kg/MTU)	Chemical	Mass (kg/MTU)
aluminum	8.1e-01	beryllium	7.8e-02
carbon	5.7e-01	cesium	1.0e-01
chromium	1.4e-01	copper	7.9e-02
iron	4.5e-01	manganese	2.9e-02
molybdenum	4.6e-03	nitrogen	8.1e-02
nickel	1.4e-01	plutonium	1.9e+00
silicon	1.3e-01	strontium	3.5e-02
technetium	7.6e-02	tin	1.0e+00
uranium	1.0e+03	zirconium	7.0e+01

Table 2.4: Average Radionuclide Inventory/Unit Mass Uranium Summary
(Decay date basis: May 31, 1998)

Nuclide	Activity (Ci/MTU)	Heat Generation (W/MTU)
H-3	1.74e+01	5.86e-04
C-14	3.30e-01	9.62e-05
Fe-55	8.76e-01	2.91e-05
Co-60	1.89e+00	2.90e-02
Ni-63	2.14e+00	2.16e-04
Kr-85	2.81e+02	4.21e-01
Sr-90	4.81e+03	5.57e+00
Y-90	4.81e+03	2.67e+01
Tc-99	1.37e+00	6.86e-04
Ru-106	8.67e-01	5.14e-05
Rh-106	8.67e-01	8.29e-03
Cd-113m	1.69e+00	1.86e-03
Sb-125	1.60e+01	5.00e-02
Te-125m	3.90e+00	3.28e-03
I-129	3.03e-03	1.42e-06
Cs-134	7.57e+00	7.71e-02
Cs-137	6.29e+03	6.33e+00
Ba-137m	5.95e+03	2.33e+01
Ce-144	4.35e-01	2.88e-04
Pr-144	4.30e-01	3.15e-03
Pm-147	2.20e+02	8.10e-02
Sm-151	8.38e+01	9.81e-03
Eu-152	4.50e-01	2.04e-03
Eu-154	5.09e+01	4.59e-01
Eu-155	1.05e+01	7.62e-03
U-234	4.16e-01	1.18e-02
U-235	1.60e-02	4.34e-04
U-236	6.05e-02	1.64e-03
U-238	3.31e-01	8.43e-03
Pu-238	5.28e+01	1.73e+00
Pu-239	1.04e+02	3.20e+00
Pu-240	5.67e+01	1.77e+00
Pu-241	3.18e+03	9.86e-02
Am-241	1.79e+02	5.86e+00
Cm-244	6.86e-01	2.36e-02
Totals ¹	2.62e+04	7.67e+01

¹ Totals include additional nuclides which are not itemized in this summary table. A complete listing of the estimated radionuclide inventory can be derived from Table 3.3.

2.2 SHIELDING DESIGN FEED BASES

The shielding design basis is defined here as Mark IV fuel, irradiated to 16% ^{240}Pu content, that has been discharged from the reactor for 13.5 years. This fuel was chosen for the shielding basis because it represents a bounding limit to the ^{137}Cs concentration. The shielding design basis radionuclide concentrations are summarized in Table 2.5.

The shielding basis ^{137}Cs content is approximately 1.7 times the average ^{137}Cs content in the basins. The shielding basis allows any combination of fuel canisters which may be encountered within the K Basins to be loaded into an MCO without concern for exceeding shielding requirements to limit peak (as opposed to average) dose rates.

Table 2.5: Shielding Design Basis Summary
(Based on MK IV fuel at 16% ^{240}Pu content, aged 13.5 years)

Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)
Cobalt-60	6.27e+00	9.64e-02
Cesium-137	1.13e+04	1.14e+01
Barium-137m	1.07e+04	4.18e+01
Samarium-151	1.08e+02	1.26e-02
Europium-154	2.02e+02	1.81e+00

Note: This summary table is limited to key isotopes which are important for gamma shielding calculations. Table 3.8 provides a complete list.

2.3 SAFETY/REGULATORY ASSESSMENT FEED

The safety/regulatory assessment design basis was selected by identifying the isotopic mixture of components expected to yield the largest dose to people per unit of material released. The selection process used to compare the isotopic mixtures focused on radionuclides that are anticipated to dominate dose calculations for accident scenarios. These included: all plutonium isotopes, ^{241}Am , ^{90}Sr , and ^{137}Cs . The dose from a unit release of material was summed for these nuclides assuming dose conversion factors modelling an inhalation exposure pathway. N Reactor fuel that was irradiated to 16.72% ^{240}Pu and discharged from the reactor during the month of February 1979 was selected as the safety/regulatory assessment basis from this comparison. The safety design basis is summarized on a unit mass basis in Table 2.6.

Safety Analysis documentation will identify accident scenarios and potential exposure pathways to establish a safety basis for new facilities. The intent of this document is not to present the accident scenarios or exposure pathways, but only to supply worst case information to be used in the development of the safety documentation.

Table 2.6: Safety/Regulatory Assessment Design Basis Summary
(Based on MK IV fuel at 16.72% ^{240}Pu , aged approximately 19 years)

Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)
Plutonium-238	1.33e+02	4.34e+00
Plutonium-239	1.73e+02	5.28e+00
Plutonium-240	1.37e+02	4.19e+00
Plutonium-241	6.82e+03	2.11e-01
Plutonium-242	8.71e-02	2.55e-03
Americium-241	4.34e+02	1.42e+01
Curium-242	3.08e-01	1.11e-02
Curium-244	4.47e+00	1.54e-01
Strontium-90	6.93e+03	8.03e+00
Cesium-137	9.66e+03	9.78e+00

Note: This summary is limited to key dose contributors. Table 3.9 provides a complete list.

2.4 HEAT GENERATION BASIS FEED

Whereas the Table 2.7 values in Revisions 0 and 0A of this document were based on selection of the shielding basis fuel material to also represent the heat generation basis feed material, those values have been adjusted downward in this revision (Revision 1) to reflect selection of the safety/regulatory assessment basis fuel material to replace the shielding basis fuel to represent the heat generation basis feed material.

The heat generation basis feed material selected for Revision 1 by choosing the fuel lot capable of generating the most heat within a fully loaded MCO, as opposed to the most heat per MTU. A comparison was made of different groups of fuel in the basins including the most recently discharged fuel, the shielding design basis fuel, the safety/regulatory assessment fuel, and a 13.4% ^{240}Pu key which has been discharged for approximately 8.5 years.

Although the shielding design basis fuel has the largest heat production on a per mass basis, the safety/regulatory assessment fuel has the highest heat generation on a per element basis. The reason for this apparent discrepancy is that the safety/regulatory assessment fuel is longer than the shielding basis fuel (E length vs S length (see table 3.1)). Current plans call for loading the Multi-canister Overpacks (MCO) with a given number of fuel pieces rather than a given mass of fuel. An MCO loaded with the safety/regulatory assessment fuel will produce more heat than an MCO loaded with the shielding basis fuel. The safety/regulatory assessment basis fuel was therefore selected to represent the heat generation basis fuel.

Table 2.7 provides a listing of important heat generating isotopes in the heat generation basis feed.

Table 2.7: Heat Generation Basis Summary
 (Based on MK IV fuel at 16.72% ²⁴⁰Pu content, aged approximately 19 years)

Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)
Co-60	2.09e+00	3.23e-02
Kr-85	3.70e+02	5.53e-01
Sr-90	6.93e+03	8.03e+00
Y-90	6.93e+03	3.84e+01
Cs-134	6.47e+00	6.59e-02
Cs-137	9.66e+03	9.78e+00
Ba-137m	9.14e+03	3.59e+01
Eu-154	1.13e+02	1.01e+00
Pu-238	1.33e+02	4.34e+00
Pu-239	1.73e+02	5.28e+00
Pu-240	1.37e+02	4.19e+00
Am-241	4.34e+02	1.42e+01
Cm-244	4.47e+00	1.54e-01

Notes: This summary is limited to key heat contributors. Table 3.9 provides a complete list.

2.5 CRITICALITY ASSESSMENT FEED

The primary purpose of constructing a criticality assessment design basis feed is to identify special equipment geometries or operating limits which may be imposed by criticality safety evaluations. The criticality assessment feed is chosen as unirradiated fuel. Unirradiated Mark 1A fuel has an outer element with a ²³⁵U enrichment of 1.25%, and an inner element ²³⁵U enrichment of 0.947%. Unirradiated Mark IV fuel has ²³⁵U enrichments of 0.947% in both inner and outer elements. The criticality assessment feed is therefore unirradiated Mark 1A fuel.

3.0 N REACTOR FUEL

3.1 K BASIN DESCRIPTION

The 105-K East and 105-K West (referred to as KE and KW) Reactors are located in the 100 Area of the Hanford Site, 0.4 km (0.25 mi) from the Columbia River. The two reactors, built in the early 1950's, were graphite-moderated plutonium-production reactors using once-through (single-pass) cooling. Each reactor was provided with a large basin for underwater storage of its irradiated fuel. The KW Reactor was shut down in February 1970, and the KE Reactor was shut down in February 1971. Essentially all stored fuel was shipped to the 200 East Area for processing, leaving the 105-KE and 105-KW fuel storage basins idle but filled with water. The two basins are nearly identical. A three dimensional representation of one basin is depicted in Figure 3.1.

In the mid 1970's, need for additional storage space for irradiated N Reactor fuel was identified. Projects H-501 (Gydesen 1977) and H-508 (Gydesen 1975) modified the KE and KW Basins, respectively, so that the basins could be used for storage and handling of irradiated N Reactor fuel.

3.1.1 Irradiated Fuel Storage at the KE and KW Basins

The fuel at the KE and KW Basins is stored in canisters on the bottom of the basins in single stacked storage racks. Each canister consists of two 20 cm (8 in.) I.D. by 70 cm (28 in.) long open top cylinders, welded together in binocular fashion. Spacers at each end of the assembly, between the two cylinders, serve as lifting lugs. Each cylinder holds up to seven fuel assemblies, i.e., seven inner elements and seven outer elements. Three different types of canisters (Mark 0, Mark I, and Mark II) are currently deployed within the basins (see Figure 3.2). Mark 0 canisters have perforated bottoms and open tops, and are made of aluminum. Mark I and Mark II canisters have solid bottoms and sealable tops; however, Mark I and Mark II canisters located in the KE Basin do not have the tops installed. Mark I canisters are made of either aluminum or stainless steel. All Mark II canisters are made of stainless steel. There are approximately 3050 Mark II canisters, 2925 Mark I stainless steel canisters, and 1415 Mark 0 or Mark I aluminum canisters in the K Basins. There are 93 canisters in the KE Basin that were covered with debris and not identified as Mark 0, Mark I, or Mark II. In addition to the N Reactor fuel, there are approximately 30 containers of SPR fuel; see section 4.3 below. (Pitner 1995 and WHC 1995b)

The KE Basin stores 3,672 canisters containing 51,073 fuel assemblies, elements, or pieces. The KW Basin stores 3,841 canisters containing 53,964 fuel assemblies, elements, or pieces (WHC 1994b). The total mass of fuel elements at the KE Basin is approximately 1,233 metric tons. The total mass of fuel elements in the KW Basin is approximately 1,038 metric tons. Of this total mass, there are approximately 1,143,600 kg of uranium and 2,155 kg of plutonium in the KE Basin and approximately 951,900 kg of uranium and 1,875 kg of plutonium in the KW Basin.

Figure 3.1: 105-K Basins

The cladding on a significant fraction of the fuel was damaged at reactor discharge or during subsequent fuel handling. Video imaging done in the summer of 1994 indicates that approximately 40% of the outer elements and 20% of the inner elements have breached cladding (Pitner 1995). As a result of the cladding damage, the uranium in some elements was exposed to the water and has oxidized during storage. The uranium oxidation causes the fuel elements to swell and leads to further damage of the cladding, thus exposing fresh uranium to the basin water. The loss of cladding integrity and oxidation of the uranium allows soluble and gaseous fission products to dissolve into the canister water. The open canisters used in the KE Basin allow free exchange of water between the canister and the basin; therefore, soluble and gaseous fission products are continually released to the water of the KE Basin.

All of the canisters in the KW Basin have closed lids and bottoms; therefore, with the exception of gaseous products, contamination resulting from damaged and/or oxidized fuel has been contained within the canisters. As compared to the KE Basin, the water in the KW Basin is relatively free of contamination.

In 1983 and 1984, Mark IV fuel in the KE Basin was segregated into fuel grade and weapons grade inventories. The segregation process involved retrieving the fuel canisters from the storage racks, removing the fuel from the canisters, separating the inner elements from the outer elements, measuring the ^{240}Pu content of the elements, reassembling similar grade elements, repackaging the fuel into canisters, and returning the canisters to the storage racks. Although some of the segregated fuels were later moved to the KW Basin no large scale segregation was undertaken at that facility.

A significant amount of sludge has accumulated on the KE Basin's floor. The composition of the sludge is described in Section 5.0 of this document.

3.2 N REACTOR FUEL DESCRIPTION

The N Reactor fuel assemblies consist of two concentric tubes (termed inner and outer elements) made of uranium metal co-extruded into Zircaloy-2 cladding. There are two basic types of fuel assemblies differentiated by their uranium enrichment. Mark IV fuel assemblies have a pre-irradiation enrichment of 0.947% ^{235}U in both elements and an average uranium weight of 22.7 kg (50 lb). The Mark IV assemblies have an outside diameter of 6.1 cm (2.42 in) and lengths of 44.2, 58.9, 62.5, or 66.3 cm (17.4, 23.2, 24.6, or 26.1 in). Mark IA fuel assemblies have a pre-irradiation enrichment of 1.25% ^{235}U in the outer element and 0.947% ^{235}U in the inner element. The Mark IA assemblies have an average uranium weight of 16.3 kg (35.9 lb). Mark IA fuel assemblies have an outside diameter of 6.1 cm (2.40 in) and lengths of 37.8, 49.8, 53.1, or 66.3 cm (14.9, 19.6, 20.9, or 26.1 in). A Mark IV fuel assembly is shown schematically in Figure 3.3. Mark IA fuel assemblies are similar. There is also a small amount of fuel with 0.71% ^{235}U content that was designated as Mark IVB fuel. The Mark IVB fuel has the same dimensions and weights as the 66.3 cm (26.1 in) Mark IV fuel.

Figure 3.2: N Reactor Fuel Storage Canisters

105-N REACTOR FUEL STORAGE CANISTERS

N REACTOR MARK IV FUEL ASSEMBLY

(DS00015A)

Figure 3.3: N Reactor Mark IV Fuel Assembly

3.2.1 Fuel Fabrication (WHC 1984, WHC 1989)

In the co-extrusion fabrication process, a tubular billet of uranium was encased in an inner and outer sleeve of Zircaloy-2 with an outer sheath of copper. This billet/cladding assembly was evacuated and sealed to prevent oxidation during preheating and extrusion. As the billet/cladding assembly was extruded under high pressure and elevated temperature, a solid state diffusion bond formed between the uranium core and the Zircaloy-2 cladding.

The extruded fuel was cut to the desired lengths and a recess was machined into the uranium at each end of the fuel section. The copper sheath was then stripped off by an acid bath, after which an acid etching process was used to remove residual uranium from the cladding. The tube ends were closed by placing a Zircaloy-2/5% beryllium braze ring and a Zircaloy-2 end cap into the recess in each end of the fuel and then induction heating the assembly to brazing temperature (approximately 1050°C) in a vacuum. The junction of the end cap, braze, and cladding was fusion welded to alloy the braze material with the cladding, improving corrosion resistance and providing a hermetic seal. The fuel was then heat treated and cleaned with an abrasive grit blast and acid baths.

Six spacers were welded to the outside of the inner element to ensure proper alignment and engagement with the outer element when the two elements were assembled together. Eight Zircaloy-2 support clips were welded to the outside of the outer tube to ensure proper alignment of the fuel when inserted into the N Reactor core. Low-carbon steel "shoes" were crimped onto these Zircaloy-2 clips to minimize the tendency for galling to occur between the assemblies and the reactor tubes during fuel insertion and removal. The shoes vary in thickness from 0.25 mm to 0.46 mm (10 to 18 mils) and contain 0.42 to 0.74 grams of iron. Finishing operations on each tube included nondestructive testing, heating in an autoclave, inspecting, and inserting an inner element into an outer element to produce the finished fuel assembly. Table 3.1 provides additional details concerning the two fuel assembly types (the Mark IV and Mark IA) that are prevalent in the basins.

Table 3.1: 105-N Reactor Fuel Element Description

	Mark IV				Mark IA			
Post-irradiation enrichment of ^{235}U	0.947% Enriched				1.25-0.947% Enriched			
Type-Length code ¹	E	S	A	C	E	M	T	F
Length (cm)	66.3	62.5	58.9	44.2	66.3	53.1	49.8	37.8
Element diameter (cm)								
1. Outer of outer	6.15				6.10			
2. Inner of outer	4.32				4.50			
3. Outer of inner	3.25				3.18			
4. Inner of inner	1.22				1.11			
Cladding mass (kg)								
1. Outer element	1.09	1.04	0.99	0.79	1.07	0.88	0.83	0.66
2. Inner element	0.55	0.52	0.50	0.40	0.66	0.54	0.51	0.40
Mass of uranium in outer (kg)								
1. (0.947% ^{235}U)	16.0	15.0	14.1	10.5				
2. (1.25% ^{235}U)					13.8	11.1	10.4	7.85
Mass of uranium in inner (kg) 0.947% ^{235}U	7.48	7.03	6.62	4.94	6.84	5.49	5.12	3.90
Weighted average of uranium in element (kg)	22.7				16.3			
Ratio of Zircaloy-2 to uranium (kg/MTU)	70.0	70.8	71.6	77.1	83.8	85.5	86.3	90.4
Weighted ave. (kg/MTU)	70.3				85.7			
% of total elements	63				37			
% of length type of each fuel	78	10	7	5	0.03 ²	87	10	3
Displacement volume (L/MTU)	67				67			

1. Letter code differentiates the different lengths of the Mark IV or Mark IA fuel elements, i.e. a type "E" element is 66.3 centimeters long.
2. There are only 12 of the 66.3 cm Mark IA assemblies.

3.2.2 Burnup and Time Since Discharge

Fuel burnup (or exposure level) and time since discharge determine the radionuclide content of a fuel assembly or group of assemblies. The K Basin inventory of N Reactor fuel is composed of elements which experienced a range of burnups and were discharged from the reactor between January 1971 and April 1987. The burnups range from 0 to approximately 6000 megawatt-days (MWD) per MTU (Roblyer 1995).

Accountability records have been used as the basis for estimating N Reactor fuel radionuclide content. The accountability record run data listing, dated November 17, 1994, includes discharge date, fuel type, and other information for 497 groups of fuel elements (the groups are also known as keys). Each group (or key) includes elements of the same type, and with the same burnup, that were discharged from the reactor at the same time. The

mass of uranium associated with each key varies from 2.37×10^{-3} to 67.4 MTU. Appendix A lists the accountability database that forms the inventory basis. Tables 3.2 and 3.3 present breakdowns of the fuel burnups listed in the accountability database for Mark 1A and Mark IV fuel.

Table 3.2: N Reactor Mark IV Fuel Burnup Summary

Mark IV Fuel Burn-up Summary		
$\%^{240}\text{Pu}$ Range	Mass	% of total mass
$\leq 5\%$	166.94 MTU	11.35%
$>5\% - 7\%$	125.25 MTU	8.51%
$>7\% - 9\%$	0.059761 MTU	0.00%
$>9\% - 11\%$	62.988 MTU	4.28%
$>11\% - 13\%$	270.56 MTU	18.39%
$>13\% - 15\%$	714.01 MTU	48.54%
$>15\%$	131.29 MTU	8.92%
Total	1471.1 MTU	100%

Table 3.3: N Reactor Mark 1A Fuel Burnup Summary

Mark 1A Fuel Burn-up Summary		
$\%^{240}\text{Pu}$ Range	Mass	% of total mass
$\leq 5\%$	36.124 MTU	5.75%
$>5\% - 7\%$	3.3729 MTU	0.54%
$>7\% - 9\%$	0 MTU	0.00%
$>9\% - 11\%$	68.008 MTU	10.83%
$>11\% - 13\%$	118.59 MTU	18.88%
$>13\% - 15\%$	401.88 MTU	64.00%
$>15\%$	0 MTU	0.00%
Total	627.98 MTU	100

Figure 3.4: K Basin N Reactor Fuel Inventory Characteristics

3.3 NOMINAL N REACTOR DESIGN BASIS FEED

3.3.1 N Reactor Fuel Chemical Content Methodology

The chemical content of the fuel is taken from concentration ranges and maximums found in the *Hanford Spent Fuel Inventory Baseline* (WHC 1994b) and totaled for all of the N Reactor fuel in the K Basins. The chemical content of the fuel is on a pre-irradiation basis derived by applying the reported pre-irradiation concentration range to the uranium, zirconium, and braze filling total in N Reactor fuel of approximately 2,100,000, 145,000, and 3000 kg, respectively. The pre-irradiation uranium and zircaloy-2 cladding quantities are based on information in the fuel inventory baseline (WHC 1994b). The pre-irradiation braze quantity is based on a total of 1.37 g of beryllium per assembly (Weakley 1994) in the braze, and the braze component listing in the fuel inventory baseline (WHC 1994b). In the post irradiated fuel, a small percentage of the uranium will have been fissioned or converted to plutonium, and some of the other constituents will have been activated by neutron bombardment, this is particularly true for the boron and hafnium which have relatively high neutron capture cross sections. Major actinide and fission product amounts were derived using the specific activities of the nuclides and the nuclide amounts calculated in section 3.3.4 of this report. The specific activities and half lives of the nuclides listed in this document are found in appendix C. The overall total chemical inventory is a composite of the uranium fuel, the zircaloy 2 cladding and the zirconium/beryllium braze filler material using the midpoint where a range is listed.

3.3.2 N Reactor Fuel Chemical Inventory

Table 3.4 lists kilogram quantities of the chemical inventory of N Reactor fuel currently stored in the K Basins.

Table 3.4: N Reactor Fuel Chemical Inventory

Element	Uranium Alloy 601 (kg)	Zircaloy-2 Cladding (kg)	Braze Filler (kg)	Totals (kg) ²
Al	1,480 - 1,900	11.1	0.411	1,700
B	0.530	0.074	0.00142	0.605
Be	21.0	-	142	163
C	769 - 1,550	40.7	1.42	1,200
Cd	0.530	0.074	0.00142	0.605
Co	-	1.48	0.0567	1.54
Cr	137	74 - 222	1.42 - 4.26	288
Cu	158	7.40	0.170	166
Fe	632 - 843	104 - 296	1.70 - 5.96	941
H	4.22	3.70	0.142	8.06
Hf	-	29.6	0.567	30.2
Mg	52.7	2.96	0.170	55.8
Mn	52.7	7.40	0.170	60.3
Mo	-	7.40	0.142	7.54
N	158	11.8	0.567	170
Na	-	2.96	0.0567	3.02
Ni	211	44.4 - 118	0.851 - 2.27	294
O	-	-	6.53	6.53
Pb	-	14.8	0.369	15.2
Si	261	14.8	0.709	277
Sn	-	1,780 - 2,520	32.3 - 48.2	2,190
Ti	-	7.40	0.142	7.54
V	-	7.40	0.142	7.54
W	-	7.40	0.284	7.68
Zr	753	145,000	2,780	148,000
Actinides				
U	2,100,000	0.518	0.0113	2,100,000
Np	81.1	-	-	81.1
Pu	4,120	-	-	4,120
Am	109	-	-	109
Cm	0.018	-	-	0.018
Fission Products ^{2a}				
Se	12.0	-	-	12.0
Sr	152	-	-	152
Tc	170	-	-	170
Pd	133	-	-	133
Kr	77.1	-	-	77.1
I	46.1	-	-	46.1
Cs	4.65	-	-	4.65
Pm	0.50	-	-	0.50
Sm	176	-	-	176
Xe	1040	-	-	1040

² For the values with a range, the midpoint of the range is used

^{2a} Stable fission product are based on the ratio of ⁹⁹Tc in Table 3.5 and the relative concentration of stable fission products from ORIGEN2 for 12% ²⁴⁰Pu Mark IV assemblies.

3.3.3 N Reactor Fuel Radionuclide Content Methodology

Enabling assumptions used in the development of these data are listed in appendix D of this document. The basis for the radionuclide content estimates in the N Reactor fuel, with the exception of the plutonium 239 and 240, is the "RADNUC" computer program (WHC 1995a). The plutonium 239 and 240 data were taken from the *Westinghouse Hanford Company 1996 Materials Management Plan* (WHC 1996).

The RADNUC program was originally written to estimate the radionuclide content in tank waste. This code was selected for adaptation to the K Basin inventory estimates because it represented an available tool for taking multiple discharge dates and burnups of N Reactor fuel and developing a composite radionuclide inventory at a common decay date. The code has also maintained a quality assurance pedigree. RADNUC's fuel burnup estimates are based on the spent fuel's ^{240}Pu content, which increases with increasing burnup. The K Basin N Reactor fuel inventory, which includes fuel with burnups from 0 to 6000 MWD/MTU (Roblyer 1995), ranges from 0 to almost 17 percent ^{240}Pu .

Data are input to RADNUC from the output of the "ORIGEN2" program (Schwarz 97). ORIGEN2 estimates total radionuclide content of fuel with a given burnup in the reactor, and then decays the nuclides and predicts future radionuclide amounts. The results produced by ORIGEN2 are somewhat unwieldy because ORIGEN2 tracks a very large number of nuclides. Additionally, one ORIGEN2 run only produces results for one burnup level, and so a large number of runs would be required to directly estimate the radionuclide content in the N Reactor Fuel in the K Basins. For these reasons, a complete set of ORIGEN2 runs has not been thought to be practical for the N Reactor fuel. Instead, runs for burnups corresponding to ^{240}Pu contents of 6%, 9%, 12%, and 16% have been completed and used to estimate radionuclides in the fuel.

RADNUC uses ORIGEN2 estimates of eighty-nine nuclides that have been decayed to levels that would be present one year after discharge from the reactor. The one year decay period was introduced to reduce the number of nuclides to a manageable size by eliminating the short half life nuclides which would have decayed completely after one year. Tables of the eighty-nine nuclides for burnups corresponding to ^{240}Pu contents of 6%, 9%, 12%, and 16%, are included in the code for both Mark 1A and Mark IV fuel. The RADNUC user specifies a data file for each run. The data file must have at least one line, but may have any number of lines. Each line in the data file contains; the date the fuel was discharged from the reactor, fuel type (Mark IV or Mark 1A), ^{240}Pu content, and total mass of uranium in the fuel. RADNUC uses a logarithmic interpolation scheme to compute the amount of each of the eighty-nine nuclides. RADNUC then decays the nuclides to a date selected by the user. The output includes a summation of the total curies and heat production for each nuclide as well as overall totals.

The N Reactor fuel radionuclide activity and heat generation rates were derived directly from RADNUC output using the accountability records presented in Appendix A as input data and selecting a common decay date of May 31, 1998. The database was converted into RADNUC input files, which are presented in Appendix B. The RADNUC output files are presented in Appendix B as well. The output files were reformatted to generate the tables in this document.

3.3.3.1 Limitations of the Methodology. The method used to calculate radionuclide inventories (see section 3.3.3) produces conservative results because: 1) the calculations are based on irradiation at a single exposure

value, whereas the actual fuel keys are blends of higher and lower exposure fuel; and 2) the ^{240}Pu content of the spent fuel is not a linear function of exposure. Consequently, a burnup calculation that treats the average ^{240}Pu concentration of a high/low burnup blend as if it resulted from a single exposure level overestimates the average burnup of the blend.

The resulting conservatism is acceptable for establishing the bounding case, but the more accurate accountability values for $^{239,240}\text{Pu}$ have been used, in the tables for basin totals. The accountability data for the combined basin result in an estimate that is 5% less than the ORIGEN2 based estimate. The information used to calculate the accountability concentration of Pu is not readily available for use in calculating other radionuclides, which, therefore, remain overstated. The decay heat is also overstated due to the same conservatism.

3.3.4 N Reactor Fuel Radionuclide Content

Tables 3.5, 3.6, and 3.7 list the radionuclide inventory of the N Reactor Fuel in storage at the K Basins. Table 3.5 provides an inventory estimate for the combined total for both basins. Table 3.6 provides an inventory estimate specific to the KE Basin, and table 3.7 provides an inventory estimate specific to the KW Basin. The ^{85}Kr data provided in both tables assume there has been no release of this gaseous nuclide from the fuel.

Table 3.5: Radionuclide Inventory of the Combined K Basins³

Isotope	Activity (Ci)	Mass (Kg)	Heat Generation (W)	Isotope	Activity (Ci)	Mass (Kg)	Heat Generation (W)
Fission and Activation Products							
H-3	3.66e+04	3.79e-03	1.23e+00	Sn-123	1.74e-05	2.12e-12	5.42e-08
C-14	6.93e+02	1.55e-01	2.02e-01	Sn-126	1.56e+02	5.50e+00	4.78e-02
Fe-55	1.84e+03	7.36e-04	6.12e-02	Sb-124	3.03e-18	1.73e-25	4.01e-20
Co-60	3.96e+03	3.50e-03	6.09e+01	Sb-125	3.35e+04	3.24e-02	1.05e+02
Ni-59	4.10e+01	5.41e-01	1.63e-03	Sb-126	2.18e+01	2.61e-07	3.96e-01
Ni-63	4.49e+03	7.28e-02	4.54e-01	Sb-126m	1.56e+02	1.99e-09	2.00e+00
Se-79	8.62e+01	1.24e+00	2.66e-02	Te-123m	2.76e-11	3.11e-18	3.98e-14
Kr-85	5.90e+05	1.50e+00	8.85e+02	Te-123m	8.18e+03	4.54e-04	6.89e+00
Sr-89	0.00	0.00	0.00	Te-127	9.49e-07	3.60e-16	1.28e-09
Sr-90	1.01e+07	7.40e+01	1.17e+04	Te-127m	9.69e-07	1.03e-13	4.81e-10
Y-90	1.01e+07	1.86e-02	5.60e+04	Te-129	0.00	0.00	0.00
Y-91	2.23e-14	9.09e-22	8.00e-17	Te-129m	0.00	0.00	0.00
Zr-93	4.00e+02	1.59e+02	4.61e-02	I-129	6.37e+00	3.61e+01	2.99e-03
Zr-95	1.69e-12	7.85e-20	8.49e-15	Cs-134	1.59e+04	2.32e-02	1.62e+02
Nb-93m	2.47e+02	8.74e-04	4.42e-02	Cs-135	7.75e+01	6.73e+01	2.59e-02
Nb-95	3.74e-12	9.55e-20	1.79e-14	Cs-137	1.32e+07	1.52e+02	1.33e+04
Nb-95m	1.25e-14	3.29e-23	1.65e-17	Ba-137m	1.25e+07	2.32e-05	4.89e+04
Tc-99	2.88e+03	1.70e+02	1.44e+00	Ce-141	0.00	0.00	0.00
Ru-103	0.00	0.00	0.00	Ce-144	9.14e+02	2.86e-04	6.04e-01
Ru-106	1.82e+03	5.44e-04	1.08e-01	Pr-143	0.00	0.00	0.00
Rh-103m	0.00	0.00	0.00	Pr-144	9.03e+02	1.20e-08	6.62e+00
Rh-106	1.82e+03	5.11e-10	1.74e+01	Pr-144m	1.10e+01	6.06e-11	3.69e-03
Pd-107	1.63e+01	3.17e+01	8.94e-04	Pm-147	4.62e+05	4.98e-01	1.70e+02
Ag-110	2.84e-04	6.81e-17	2.13e-09	Pm-148	0.00	0.00	0.00
Ag-110m	2.14e-02	4.50e-09	3.55e-04	Pm-148m	0.00	0.00	0.00
Cd-113m	3.55e+03	1.64e-02	3.90e+00	Sm-151	1.76e+05	6.69e+00	2.06e+01
Cd-115m	0.00	0.00	0.00	Eu-152	9.45e+02	5.46e-03	4.28e+00
In-113m	2.14e-07	1.28e-17	4.95e-10	Eu-154	1.07e+05	3.96e-01	9.64e+02
Sn-113	2.14e-07	2.13e-14	3.55e-11	Eu-155	2.21e+04	4.75e-02	1.60e+01
Sn-119m	2.97e-01	6.62e-08	1.53e-04	Gd-153	1.28e-04	3.63e-11	1.15e-07
Sn-121m	7.96e+01	1.35e-03	8.36e-02	Tb-160	2.77e-15	2.45e-22	2.21e-17
Fission and Activation Product Totals					4.74e+07	7.07e+02	1.32e+05
Actinides							
U-234	8.74e+02	1.40e+02	2.47e+01	Pu-241	6.68e+06	6.48e+01	2.07e+02
U-235	3.37e+01	1.56e+04	9.14e-01	Pu-242	5.49e+01	1.44e+01	1.60e+00
U-236	1.27e+02	1.96e+03	3.40e+00	Am-241	3.75e+05	1.09e+02	1.23e+04
U-238	6.96e+02	2.07e+06	1.73e+01	Am-242	1.95e+02	2.41e-07	1.62e+00
Np-237	5.72e+01	8.11e+01	1.65e+00	Am-242m	1.96e+02	2.02e-02	2.99e-01
Pu-238	1.11e+05	6.48e+00	3.63e+03	Am-243	1.20e+02	6.02e-01	3.81e+00
Pu-239	2.18e+05	3.51e+03	6.72e+03	Cm-242	1.62e+02	4.90e-05	5.86e+00
Pu-240	1.19e+05	5.22e+02	3.71e+03	Cm-244	1.44e+03	1.78e-02	4.95e+01
Actinide Totals					7.51e+06	2.09e+06	2.66e+04

Table 3.6: K-East Basin Radionuclide Inventory⁴

³ 06/02/1997 RADNUC2A run for combined K Basins, with Results decayed to 05/31/1998. Total fuel mass in the combined basins 2.10e+03 MTU. Total fuel activity in the combined basins 5.50e+07 Ci. Total fuel heat generation in the combined basins 1.61e+05 W. Pu-239 and Pu-240 have been adjusted to ensure consistency with data previously reported by Safeguards.

⁴ 06/02/1997 RADNUC2A run for K-East Basin fuel. Results decayed to 05/31/1998. Total fuel mass in K-East 1.15e+03 MTU. Total fuel activity in K-East 2.73e+07 Ci. Total fuel heat generation in K-East 8.06e+04 W. Pu-239 and Pu-240 have been adjusted to ensure consistency with data previously reported by Safeguards.

Isotope	Activity (Ci)	Mass (Kg)	Heat Generation (W)	Isotope	Activity (Ci)	Mass (Kg)	Heat Generation (W)
Fission and Activation Products							
H-3	1.80e+04	1.86e-03	6.04e-01	Sn-123	1.46e-05	1.78e-12	4.57e-08
C-14	3.62e+02	8.12e-02	1.06e-01	Sn-126	8.07e+01	2.84e+00	2.48e-02
Fe-55	9.64e+02	3.86e-04	3.22e-02	Sb-124	2.49e-18	1.42e-25	3.28e-20
Co-60	1.86e+03	1.64e-03	2.86e+01	Sb-125	1.69e+04	1.64e-02	5.31e-01
Ni-59	2.11e+01	2.79e-01	8.38e-04	Sb-126	1.13e+01	1.35e-07	2.05e-01
Ni-63	2.30e+03	3.75e-02	2.33e-01	Sb-126m	8.07e+01	1.03e-09	1.03e+00
Se-79	4.35e+01	6.24e-01	1.34e-02	Te-123m	2.33e-11	2.63e-18	3.37e-14
Kr-85	2.84e+05	7.24e-01	4.25e+02	Te-125m	4.13e+03	2.29e-04	3.46e+00
Sr-89	0.00	0.00	0.00	Te-127	7.90e-07	2.99e-16	1.07e-09
Sr-90	4.96e+06	3.64e+01	5.77e+03	Te-127m	8.07e-07	8.55e-14	3.98e-10
Y-90	4.97e+06	9.13e-03	2.75e+04	Te-129	0.00	0.00	0.00
Y-91	1.78e-14	7.26e-22	6.39e-17	Te-129m	0.00	0.00	0.00
Zr-93	2.01e+02	7.98e+01	2.30e-02	I-129	3.26e+00	1.85e+01	1.52e-03
Zr-95	1.35e-12	6.30e-20	6.83e-15	Cs-134	6.95e+03	5.37e-03	7.09e+01
Nb-93m	1.26e+02	4.46e-04	2.25e-02	Cs-135	3.96e+01	3.44e+01	1.32e-02
Nb-95	3.01e-12	7.69e-20	1.44e-14	Cs-137	6.55e+06	7.53e+01	6.62e+03
Nb-95m	1.00e-14	2.64e-23	1.33e-17	Ba-137m	6.19e+06	1.15e-05	2.43e+04
Tc-99	1.45e+03	8.55e+01	7.27e-01	Ce-141	0.00	0.00	0.00
Ru-103	0.00	0.00	0.00	Ce-144	7.56e+02	2.37e-04	4.98e-01
Ru-106	1.39e+03	4.15e-04	8.26e-02	Pr-143	0.00	0.00	0.00
Rh-103m	0.00	0.00	0.00	Pr-144	7.47e+02	9.89e-09	5.48e+00
Rh-106	1.39e+03	3.90e-10	1.32e+01	Pr-144m	9.07e+00	5.00e-11	3.05e-03
Pd-107	8.59e+00	1.67e+01	4.72e-04	Pm-147	2.45e+05	2.64e-01	9.00e+01
Ag-110	2.28e-04	5.47e-17	1.71e-09	Pm-148	0.00	0.00	0.00
Ag-110m	1.71e-02	3.60e-09	2.84e-04	Pm-148m	0.00	0.00	0.00
Cd-113m	1.80e+03	8.30e-03	1.98e+00	Sm-151	8.92e+04	3.39e+00	1.04e+01
Cd-115m	0.00	0.00	0.00	Eu-152	4.67e+02	2.70e-03	2.11e+00
In-113m	1.77e-07	1.06e-17	4.10e-10	Eu-154	5.30e+04	1.96e-01	4.75e+02
Sn-113	1.77e-07	1.76e-14	2.93e-11	Eu-155	1.12e+04	2.41e-02	8.12e+00
Sn-119m	2.50e-01	5.57e-08	1.28e-04	Gd-153	9.59e-05	2.72e-11	8.64e-08
Sn-121m	4.01e+01	6.78e-04	4.20e-02	Tb-160	2.29e-15	2.03e-22	1.83e-17
Fission and Activation Product Totals					2.34e+07	3.55e+02	6.53e+04
Actinides							
U-234	4.66e+02	7.46e+01	1.32e+01	Pu-241	3.42e+06	3.32e+01	1.06e+02
U-235	1.77e+01	8.19e+03	4.81e-01	Pu-242	3.07e+01	8.04e+00	8.94e-01
U-236	6.61e+01	1.02e+03	1.77e+00	Am-241	2.66e+05	6.00e+01	6.74e+03
U-238	3.80e+02	1.13e+06	9.46e+00	Am-242	1.13e+02	1.40e-07	9.43e-01
Np-237	3.02e+01	4.28e+01	8.70e-01	Am-242m	1.14e+02	1.17e-02	1.74e-01
Pu-238	6.05e+04	3.53e+00	1.97e+03	Am-243	7.12e+01	3.57e-01	2.25e+00
Pu-239	1.16e+05	1.87e+03	3.59e+03	Cm-242	9.40e+01	2.84e-05	3.40e+00
Pu-240	6.37e+04	2.80e+02	1.98e+03	Cm-244	8.71e+02	1.08e-02	2.99e+01
Actinide Totals					3.87e+06	1.14e+06	1.45e+04

Table 3.7: K-West Basin Radionuclide Inventory⁵

Isotope	Activity (Ci)	Mass (Kg)	Heat Generation (W)	Isotope	Activity (Ci)	Mass (Kg)	Heat Generation (W)
Fission and Activation Products							

⁵ 06/02/1997 RADNUC2A run for K-West Basin. Results decayed to 05/31/1998. Total fuel mass in K-West 9.53e+02 MTU. Total fuel activity in K-West 2.77e+07 Ci. Total Heat Generation of fuel in K-West 8.03e+04 W. Pu-239 and Pu-240 have been adjusted to ensure consistency with data previously reported by Safeguards.

H-3	1.86e+04	1.93e-03	6.24e-01	Sn-123	2.75e-06	3.35e-13	8.58e-09
C-14	3.51e+02	7.45e-02	9.67e-02	Sn-126	7.50e+01	2.64e+00	2.30e-02
Fe-55	8.75e+02	3.50e-04	2.92e-02	Sb-124	5.65e-19	3.12e-26	7.21e-21
Co-60	2.10e+03	1.86e-03	3.22e+01	Sb-125	1.66e+04	1.61e-02	5.23e+01
Ni-59	1.99e+01	2.63e-01	7.88e-04	Sb-126	1.05e+01	1.26e-07	1.90e-01
Ni-63	2.19e+03	3.55e-02	2.21e-01	Sb-126m	7.50e+01	9.55e-10	9.61e-01
Se-79	4.28e+01	6.14e-01	1.32e-02	Te-123m	4.28e-12	4.82e-19	6.21e-15
Kr-85	3.06e+05	7.80e-01	4.60e+02	Te-125m	4.05e+03	2.25e-04	3.40e+00
Sr-89	0.00	0.00	0.00	Te-127	1.59e-07	6.02e-17	2.14e-10
Sr-90	5.17e+06	3.79e+01	6.01e+03	Te-127m	1.62e-07	1.72e-14	8.03e-11
Y-90	5.17e+06	9.50e-03	2.86e+04	Te-129	0.00	0.00	0.00
Y-91	4.47e-15	1.82e-22	1.60e-17	Te-129m	0.00	0.00	0.00
Zr-93	2.01e+02	7.98e+01	2.30e-02	I-129	3.11e+00	1.76e+01	1.46e-03
Zr-95	3.30e-13	1.54e-20	1.66e-15	Cs-134	8.96e+03	6.92e-03	9.14e+01
Nb-93m	1.22e+02	4.32e-04	2.18e-02	Cs-135	3.79e+01	3.29e+01	1.27e-02
Nb-95	7.33e-13	1.87e-20	3.52e-15	Cs-137	6.64e+06	7.63e+01	6.71e+03
Nb-95m	2.45e-15	6.44e-24	3.23e-18	Ba-137m	6.28e+06	1.17e-05	2.46e+04
Tc-99	1.43e+03	8.43e+01	7.18e-01	Ce-141	0.00	0.00	0.00
Ru-103	0.00	0.00	0.00	Ce-144	1.58e+02	4.95e-05	1.04e-01
Ru-106	4.34e+02	1.30e-04	2.58e-02	Pr-143	0.00	0.00	0.00
Rh-103m	0.00	0.00	0.00	Pr-144	1.56e+02	2.06e-09	1.15e+00
Rh-106	4.34e+02	1.22e-10	4.13e+00	Pr-144m	1.90e+00	1.05e-11	6.39e-04
Pd-107	7.68e+00	1.49e+01	4.22e-04	Pm-147	2.17e+05	2.34e-01	7.97e+01
Ag-110	5.66e-05	1.36e-17	4.25e-10	Pm-148	0.00	0.00	0.00
Ag-110m	4.26e-03	8.97e-10	7.06e-05	Pm-148m	0.00	0.00	0.00
Cd-113m	1.75e+03	8.07e-03	1.92e+00	Sm-151	8.66e+04	3.29e+00	1.01e+01
Cd-115m	0.00	0.00	0.00	Eu-152	4.77e+02	2.76e-03	2.16e+00
In-113m	3.74e-08	2.24e-18	8.64e-11	Eu-154	5.44e+04	2.02e-01	4.89e+02
Sn-113	3.74e-08	3.73e-15	6.21e-12	Eu-155	1.08e+04	2.32e-02	7.85e+00
Sn-119m	4.75e-02	1.06e-08	2.45e-05	Gd-153	3.23e-05	9.16e-12	2.91e-08
Sn-121m	3.95e+01	6.68e-04	4.15e-02	Tb-160	4.77e-16	4.23e-23	3.81e-18
Fission and Activation Product Totals					2.40e+07	3.52e+02	6.72e+04
Actinides							
U-234	4.08e+02	6.53e+01	1.15e+01	Pu-241	3.26e+06	3.16e+01	1.01e+02
U-235	1.60e+01	7.40e+03	4.34e-01	Pu-242	2.42e+01	6.34e+00	7.03e-01
U-236	6.11e+01	9.44e+02	1.63e+00	Am-241	1.69e+05	4.92e+01	5.54e+03
U-238	3.16e+02	9.40e+05	7.85e+00	Am-242	8.15e+01	1.01e-07	6.77e-01
Np-237	2.70e+01	3.83e+01	7.79e-01	Am-242m	8.19e+01	8.43e-03	1.25e-01
Pu-238	5.10e+04	2.98e+00	1.66e+03	Am-243	4.89e+01	2.45e-01	1.54e+00
Pu-239	1.01e+05	1.63e+03	3.13e+03	Cm-242	6.76e+01	2.04e-05	2.45e+00
Pu-240	5.53e+04	2.42e+02	1.72e+03	Cm-244	5.72e+02	7.07e-03	1.97e+01
Actinide Totals					3.64e+06	9.50e+05	1.22e+04

3.4 SHIELDING DESIGN BASIS FEED

The shielding design basis feed material was selected to include the fuel with the highest ^{137}Cs content. The ^{137}Cs content was used as the criterion because of that isotope's relative abundance in the fuel and its significant gamma radiation. The lines in figure 3.6 represent the burnup levels required at different times to obtain the 1/1/1995 ^{137}Cs concentrations shown in the table. The accountability key data are also included in figure 3.6. As in figure 3.5, each "x" in Table 3.6 represents one key from the accountability database. The key nearest the highest ^{137}Cs concentration line is the key with the highest ^{137}Cs concentration. The fuel with the highest actual cesium content is Mark IV fuel with a ^{240}Pu Content of 15.74% discharged from the reactor in 1984. For the sake of conservatism, the design basis was selected as Mk IV fuel with 16% ^{240}Pu content aged ten years to 1/1/1995. The bounding case chosen here has a ^{137}Cs content approximately 6.5% higher than the maximum fuel in inventory. RADNUC was used to estimate the radionuclide content and heat generation rates of the 16% ^{240}Pu content fuel. Table 3.8 is a tabulation of the results. The RADNUC input and output files in their raw form are listed in Appendix B.

3.5 SAFETY/REGULATORY ASSESSMENT DESIGN BASIS FEED

The safety/regulatory assessment design basis feed material was selected by determining which fuel would result in the highest estimated dose to people if exposed to a unit of material. A potential dose was obtained using cumulative effective dose equivalents (CEDE) (DOE 1988) and nuclide levels generated by RADNUC to estimate a dose resulting from each nuclide. The potential dose resulting from inhalation and ingestion pathways was evaluated, and the inhalation pathway was selected as most likely. Americium, plutonium, strontium, and cesium are estimated to dominate the dose impacts of other nuclides in the fuel, therefore only these isotopes were included in the safety/regulatory assessment basis development.

Two parameters were considered in selecting the fuel with the highest concentration of the isotopes used for determining the safety/regulatory assessment design basis. The first is burnup; all else being equal, the concentrations of radionuclides increase with increasing burnup. The second is age (time since discharge); with the notable exception of ^{241}Am , radionuclide concentrations decrease with increasing time since discharge from the reactor. Because it results in the largest dose, ^{241}Am is the most important nuclide for development of the safety/regulatory assessment basis.

A small amount of ^{241}Am is produced directly as an activation product in the irradiated fuel, but a far larger amount results from the decay of ^{241}Pu . Because all of the ^{241}Pu will eventually decay to ^{241}Am , and because ^{241}Am has a half life of approximately 433 years whereas the half life of ^{241}Pu is only 14 years, the concentration of ^{241}Am continues to increase for approximately 60 years following reactor discharge. With half lives of 24,100 and 6,560 years respectively, ^{239}Pu and ^{240}Pu are also among the most important radionuclides for the safety/regulatory assessment design basis. Over a 50 year time scale, the concentration of $^{239,240}\text{Pu}$ will be nearly constant. Considering the two factors above, the highest burnup, oldest fuels should contain the largest concentration of the most important radionuclides for the safety/regulatory assessment design basis.

Single Pass Reactor and N Reactor fuels were both considered in the evaluation. In the case of the single pass reactor fuel, the americium,

cesium, plutonium, and strontium were taken from the fuel inventory baseline document (WHC 1994b). The results of the evaluation indicate that N Reactor Mark IV fuel discharged from the reactor 2/20/1979 with a ^{240}Pu content of 16.72% represents the mix of isotopes with the largest potential dose impact on people. There are 6.9 MTU total of this 16.72% fuel stored at K Basins. The selected safety/regulatory assessment basis fuel is shown in Table 3.9. Through decay, the safety/regulatory assessment basis radionuclide content changes over time. Table 3.10 lists the radionuclide content of the safety/regulatory assessment basis fuel decayed to 05/31/2038 (approximately 40 years after the projected date of first fuel movement from the K Basins).

Appendix B lists the RADNUC input and output files in their raw form. Appendix F provides comparisons of the safety basis fuel (16.72% ^{240}Pu Mk IV N Reactor fuel aged 19 years) with other fuels and provides additional details regarding its selection.

3.6 HEAT GENERATION BASIS FEED

The heat generation basis feed material was selected by choosing the fuel lot capable of generating the most heat within a fully loaded MCO, as opposed to the most heat per MTU. A comparison was made of different groups of fuel in the basins including: the most recently discharged fuel, the shielding design basis fuel, the safety/regulatory assessment fuel, and a 13.4% ^{240}Pu key which has been discharged for approximately 8.5 years. The heat generation from the most recently discharged fuel was 29 W/MTU; the shielding design basis fuel was 147 W/MTU; the safety/regulatory assessment fuel was 137 W/MTU; and the 8.5 year old 13.4% ^{240}Pu fuel was 113 W/MTU. Although the shielding design basis fuel has a higher heat generation rate per MTU, the shielding design basis fuel keys include only S and C length fuel. The safety/regulatory assessment fuel keys contain the longer E length fuel. The safety/regulatory assessment fuel has the highest heat generation on a per assembly basis. The current strategy is to load up to 270 assemblies into an MCO. Therefore, an MCO fully loaded with the safety/regulatory assessment fuel would generate more heat than an MCO fully loaded with the shielding basis fuel. The heat generation basis fuel is therefore the same as the safety/regulatory assessment basis fuel already listed in Tables 3.9 and 3.10.

3.7 CRITICALITY DESIGN BASIS FEED

The criticality design basis feed was determined by selecting the fuel with the highest reactivity. In a conventional light water reactor such as the Hanford site production reactors (well moderated non-breeder reactor), more fissile atoms are used up than are created. The reactivity of the fuel therefore decreases with increasing burn-up. The highest reactivity fuels are the unirradiated Mark IA fuels. Current planning calls for the fuel to be segregated by type so that a given MCO may contain either Mark IA or Mark IV fuel, but not a mixture of the two. In the MCOs with Mark IV fuel, the criticality design basis is again un-irradiated Mark IV fuel.

Characteristics of K Basin Inventory Cs-137 Shielding Design Basis Selection

Figure 3.5: ^{137}Cs Shielding Design Basis Selection

Table 3.8: Shielding Design Basis⁶

(Based on MK IV fuel at 16% Pu-240 aged 13.5 years)

Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)	Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)
Fission and Activation Products					
H-3	4.03e+01	1.36e-03	Sn-123	4.08e-09	1.26e-11
C-14	5.27e-01	1.54e-04	Sn-126	1.22e-01	3.75e-05
Fe-55	5.23e+00	1.75e-04	Sb-124	0.00	0.00
Co-60	6.27e+00	9.64e-02	Sb-125	0.00	0.00
Ni-59	3.03e-02	1.21e-06	Sb-126	1.71e-02	3.10e-04
Ni-63	3.53e+00	3.57e-04	Sb-126m	1.22e-01	1.57e-03
Se-79	6.23e-02	1.93e-05	Te-123m	1.19e-13	1.72e-16
Kr-85	6.23e+02	9.33e-01	Te-125m	0.00	0.00
Sr-89	0.00	0.00	Te-127	1.16e-10	1.57e-13
Sr-90	8.19e+03	9.50e+00	Te-127m	1.19e-10	5.88e-14
Y-90	8.19e+03	4.53e+01	Te-129	0.00	0.00
Y-91	0.00	0.00	Te-129m	0.00	0.00
Zr-93	2.83e-01	3.25e-05	I-129	4.88e-03	2.29e-06
Zr-95	4.20e-18	2.12e-20	Cs-134	1.07e+02	1.10e+00
Nb-93m	1.38e-01	2.47e-05	Cs-135	5.77e-02	1.93e-05
Nb-95	9.33e-18	4.48e-20	Cs-137	1.13e+04	1.14e+01
Nb-95m	3.12e-20	4.12e-23	Ba-137m	1.07e+04	4.18e+01
Tc-99	2.08e+00	1.05e-03	Ce-141	0.00	0.00
Ru-103	0.00	0.00	Ce-144	1.75e+00	1.15e-03
Ru-106	9.38e+00	5.59e-04	Pr-143	0.00	0.00
Rh-103m	0.00	0.00	Pr-144	1.73e+00	1.26e-02
Rh-106	9.38e+00	8.97e-02	Pr-144m	2.10e-02	7.05e-06
Pd-107	1.44e-02	7.94e-07	Pm-147	1.06e+03	3.88e-01
Ag-110	4.15e-06	3.11e-11	Pm-148	0.00	0.00
Ag-110m	3.12e-04	5.18e-06	Pm-148m	0.00	0.00
Cd-113m	3.96e+00	4.35e-03	Sm-151	1.08e+02	1.26e-02
Cd-115m	0.00	0.00	Eu-152	1.22e+00	5.50e-03
In-113m	2.59e-11	5.98e-14	Eu-154	2.02e+02	1.81e+00
Sn-113	2.59e-11	4.30e-15	Eu-155	3.42e+01	2.48e-02
Sn-119m	0.00	0.00	Gd-153	3.92e-06	3.53e-09
Sn-121m	0.00	0.00	Tb-160	2.22e-19	1.78e-21
Fission and Activation Product Totals				4.05e+04	1.12e+02
Actinides					
U-234	3.92e-01	1.11e-02	Pu-241	9.42e+03	2.92e-01
U-235	1.31e-02	3.54e-04	Pu-242	7.46e-02	2.16e-03
U-236	7.12e-02	1.89e-03	Am-241	2.92e+02	9.57e+00
U-238	3.35e-01	8.35e-03	Am-242	3.19e-01	2.66e-03
Np-237	4.42e-02	1.27e-03	Am-242m	3.21e-01	4.91e-04
Pu-238	1.28e+02	4.16e+00	Am-243	2.22e-01	7.02e-03
Pu-239	1.68e+02	5.14e+00	Cm-242	2.65e-01	9.59e-03
Pu-240	1.28e+02	3.90e+00	Cm-244	4.62e+00	1.59e-01
Actinide Totals				1.01e+04	2.33e+01

⁶ 06/02/1997 RADNUC2A run for shielding design basis (Mk IV fuel at 16% Pu-240). Results decayed to 05/31/1998. Total mass of fuel in this run 2.60e+00 MTU. Total activity of fuel in this run 1.32e+05 Ci. Total heat generation from the fuel in this run 3.55e+02 W.

Table 3.9: Safety/Regulatory Assessment Design Basis⁷
 (Based on Mk IV fuel at 16.72% Pu-240)

Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)	Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)
Fission and Activation Products					
H-3	2.61e+01	8.77e-04	Sn-123	1.72e-16	5.35e-19
C-14	5.53e-01	1.62e-04	Sn-126	1.29e-01	3.97e-05
Fe-55	5.41e-01	1.80e-05	Sb-124	0.00	0.00
Co-60	2.09e+00	3.23e-02	Sb-125	0.00	0.00
Ni-59	3.18e-02	1.26e-06	Sb-126	1.81e-02	3.28e-04
Ni-63	3.47e+00	3.51e-04	Sb-126m	1.29e-01	1.66e-03
Se-79	6.54e-02	2.02e-05	Te-123m	1.50e-21	2.18e-24
Kr-85	3.70e+02	5.53e-01	Te-125m	0.00	0.00
Sr-89	0.00	0.00	Te-127	2.12e-19	2.85e-22
Sr-90	6.93e+03	8.03e+00	Te-127m	2.16e-19	1.07e-22
Y-90	6.93e+03	3.84e+01	Te-129	0.00	0.00
Y-91	0.00	0.00	Te-129m	0.00	0.00
Zr-93	2.95e-01	3.40e-05	I-129	5.16e-03	2.41e-06
Zr-95	0.00	0.00	Cs-134	6.47e+00	6.59e-02
Nb-93m	1.93e-01	3.45e-05	Cs-135	6.04e-02	2.02e-05
Nb-95	0.00	0.00	Cs-137	9.66e+03	9.78e+00
Nb-95m	0.00	0.00	Ba-137m	9.14e+03	3.59e+01
Tc-99	2.19e+00	1.10e-03	Ce-141	0.00	0.00
Ru-103	0.00	0.00	Ce-144	7.91e-04	5.23e-07
Ru-106	2.56e-02	1.52e-06	Pr-143	0.00	0.00
Rh-103m	0.00	0.00	Pr-144	7.82e-04	5.73e-06
Rh-106	2.56e-02	2.45e-04	Pr-144m	9.48e-06	3.18e-09
Pd-107	1.56e-02	8.56e-07	Pm-147	1.09e+02	4.02e-02
Ag-110	7.17e-10	5.38e-15	Pm-148	0.00	0.00
Ag-110m	5.39e-08	8.94e-10	Pm-148m	0.00	0.00
Cd-113m	2.78e+00	3.06e-03	Sm-151	1.02e+02	1.19e-02
Cd-115m	0.00	0.00	Eu-152	8.45e-01	3.81e-03
In-113m	1.36e-19	3.13e-22	Eu-154	1.13e+02	1.01e+00
Sn-113	1.36e-19	2.26e-23	Eu-155	1.06e+01	7.65e-03
Sn-119m	6.14e-08	3.16e-11	Gd-153	5.19e-10	4.67e-13
Sn-121m	6.27e-02	6.59e-05	Tb-160	0.00	0.00
Fission and Activation Product Totals				3.34e+04	9.38e+01
Actinides					
U-234	3.84e-01	1.09e-02	Pu-241	6.82e+03	2.11e-01
U-235	1.27e-02	3.44e-04	Pu-242	8.71e-02	2.55e-03
U-236	7.16e-02	1.91e-03	Am-241	4.34e+02	1.42e+01
U-238	3.31e-01	8.23e-03	Am-242	3.71e-01	3.08e-03
Np-237	4.66e-02	1.34e-03	Am-242m	3.72e-01	5.71e-04
Pu-238	1.33e+02	4.34e+00	Am-243	2.78e-01	8.82e-03
Pu-239	1.75e+02	5.28e+00	Cm-242	3.08e-01	1.11e-02
Pu-240	1.37e+02	4.19e+00	Cm-244	4.47e+00	1.54e-01
Actinide Totals				7.70e+03	2.85e+01

⁷ 06/02/1997 RADNUC2A run for the safety basis (Mk IV fuel at 16.72% Pu-240). Results decayed to 05/31/1998. Total mass of the fuel in this run 1.16e+01 MTU. Total activity of the fuel in this run 4.77e+05 Ci. Total heat generation of the fuel in this run 1.42e+03 W.

Table 3.10: Safety/Regulatory Assessment Basis⁸
(Decayed to 05/31/2038)

Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)	Isotope	Activity (Ci/MTU)	Heat Generation (W/MTU)
Fission and Activation Products					
H-3	3.32e+00	1.11e-04	Sn-123	0.00	0.00
C-14	5.64e-01	1.65e-04	Sn-126	1.33e-01	4.08e-05
Fe-55	2.72e-05	9.09e-10	Sb-124	0.00	0.00
Co-60	1.61e-02	2.47e-04	Sb-125	1.10e-03	3.44e-06
Ni-59	3.25e-02	1.29e-06	Sb-126	1.86e-02	3.36e-04
Ni-63	2.68e+00	2.71e-04	Sb-126m	1.33e-01	1.70e-03
Se-79	6.68e-02	2.06e-05	Te-123m	0.00	0.00
Kr-85	3.41e+01	5.10e-02	Te-125m	2.68e-04	2.25e-07
Sr-89	0.00	0.00	Te-127	0.00	0.00
Sr-90	2.91e+03	3.38e+00	Te-127m	0.00	0.00
Y-90	2.91e+03	1.61e+01	Te-129	0.00	0.00
Y-91	0.00	0.00	Te-129m	0.00	0.00
Zr-93	3.02e-01	3.47e-05	I-129	5.29e-03	2.48e-06
Zr-95	0.00	0.00	Cs-134	2.52e-05	2.58e-07
Nb-93m	2.73e-01	4.88e-05	Cs-135	6.19e-02	2.07e-05
Nb-95	0.00	0.00	Cs-137	4.19e+03	4.23e+00
Nb-95m	0.00	0.00	Ba-137m	3.96e+03	1.55e+01
Tc-99	2.23e+00	1.12e-03	Ce-141	0.00	0.00
Ru-103	0.00	0.00	Ce-144	0.00	0.00
Ru-106	2.06e-13	1.22e-17	Pr-143	0.00	0.00
Rh-103m	0.00	0.00	Pr-144	0.00	0.00
Rh-106	2.06e-13	1.96e-15	Pr-144m	0.00	0.00
Pd-107	1.61e-02	8.87e-07	Pm-147	5.97e-03	2.19e-06
Ag-110	0.00	0.00	Pm-148	0.00	0.00
Ag-110m	0.00	0.00	Pm-148m	0.00	0.00
Cd-113m	4.87e-01	5.35e-04	Sm-151	7.67e+01	8.96e-03
Cd-115m	0.00	0.00	Eu-152	1.32e-01	5.94e-04
In-113m	0.00	0.00	Eu-154	5.93e+00	5.31e-02
Sn-113	0.00	0.00	Eu-155	5.94e-02	4.29e-05
Sn-119m	0.00	0.00	Gd-153	0.00	0.00
Sn-121m	3.84e-02	4.04e-05	Tb-160	0.00	0.00
Fission and Activation Product Totals				1.41e+04	3.94e+01
Actinides					
U-234	3.81e-01	1.08e-02	Pu-241	1.19e+03	3.69e-02
U-235	1.25e-02	3.38e-04	Pu-242	9.39e-02	2.73e-03
U-236	7.19e-02	1.92e-03	Am-241	6.17e-02	2.03e+01
U-238	3.29e-01	8.20e-03	Am-242	3.42e-01	2.85e-03
Np-237	4.77e-02	1.37e-03	Am-242m	3.43e-01	5.27e-04
Pu-238	1.04e+02	3.40e+00	Am-243	3.09e-01	9.77e-03
Pu-239	1.75e+02	5.35e+00	Cm-242	2.84e-01	1.03e-02
Pu-240	1.41e+02	4.33e+00	Cm-244	1.24e+00	4.25e-02
Actinide Totals				2.23e+03	3.35e+01

⁸ 06/04/1997 RADNUC2A run for safety basis fuel (following 40 years CSB storage). Results decayed to 05/31/2038. Total mass of fuel in this run 6.90e+00 MTU. Total activity of the fuel in this run 1.13e+05 Ci. Total heat generation from the fuel in this run 5.04e+02 W.

4.0 SINGLE PASS REACTOR FUEL

The information included in this section was taken from the *Hanford Spent Fuel Inventory Baseline* (WHC 1994b).

4.1 BACKGROUND

Eight single-pass, water-cooled, graphite moderated reactors, commonly referred to as single pass reactors (SPR), operated at Hanford to produce plutonium, beginning in 1944 with the 105-B Reactor start-up and ending in 1971 with the 105-KE Reactor shut down (HAPO 1944, HAPO 1961, HAPO 1963a, HAPO 1963b, PNL 1991). The reactors and their operating periods are listed in Table 4.1.

Table 4.1: Operating Periods of the Hanford Single Pass Reactors

Reactor	Hanford Area	Operating Period	
		Startup	Shutdown
105-B	100-B/C	1944	1968
105-C	100-B/C	1952	1969
105-D	100-D/DR	1944	1967
105-DR	100-D/DR	1950	1964
105-F	100-F	1945	1965
105-H	100-H	1949	1965
105-KW	100-KW/KE	1955	1970
105-KE	100-KW/KE	1955	1971

The older reactors (105-B, 105-C, 105-D, 105-DR, 105-F, and 105-H) were similar in design. Their graphite block cores were 11 meters (36 ft) high by 11 meters (36 ft) wide by 8.5 meters (28 ft) long. A total of 2004 aluminum horizontal process tubes held the fuel and contained the cooling water. Most of the older reactors had nine horizontal, boron-containing, water-cooled control rods running perpendicular to the process tubes. The 105-C and 105-H Reactors each had 15 of these control rods. The control rods entered the reactor from the side and provided operating reactivity control, neutron flux shaping, and emergency shutdown control. A completely independent emergency shutdown control was provided by vertical channels penetrating the core which could be gravity filled with either boron-containing safety rods or boron-containing steel balls.

The basic design of the 105-KE and 105-KW Reactors was similar to the older reactors, with the following exceptions. Their graphite cores were 12.5 meters (41 ft) high by 12.5 meters (41 ft) wide by 10.7 meters (35 ft) long. There were 3220 aluminum horizontal process tubes, 2400 of which were later replaced with zirconium tubes. There were 20 horizontal control rods.

4.2 FUEL DESCRIPTION (HAPO 1955, DUN 1967)

The single pass reactor fuel consisted of a machined uranium core that was sealed within, and metallurgically bonded to, an aluminum can. There were three basic designs for SPR fuel elements; solid core cylinders, hollow core cylinders, and tubes. All three types are shown in Figure 4.1. The solid cylinder fuel was used from 1944 until the late 1950s. This design was adequate until 1951, when increasing reactor power levels and fuel exposure began to result in significant fuel failures. The hollow cored fuel was used concurrently with the solid fuel from late-1954 until 1957 in an attempt to reduce fuel failures, but this strategy was unsuccessful. Beginning in late-1955, the fuel failure rate was successfully reduced by introduction of a tubular fuel design called an internally and externally cooled (I&E) fuel.

Many variations of these basic designs were created in attempts either to improve cooling of the fuel element during irradiation or to simplify the fabrication process. In one successful variation, called the tru-line design, one end of the fuel element had a raised face while the other end had a recessed face. The raised face mated with the recessed face of the adjacent fuel element in the tube to improve cooling through better alignment of the fuel elements within the reactor tubes. Tru-line features are shown on the I&E fuel element in Figure 4.1.

The SPR fuel was made primarily in two ^{235}U concentrations, natural (0.71%) and 0.947%; although other enriched fuel (including 1.25%, 1.7%, and 2.1% ^{235}U) and depleted fuel (including 0.14% and 0.22% ^{235}U) were also made. The lengths of the fuel elements range from approximately 12 cm (4.7 in) to 23 cm (9.0 in), with specific lengths generally associated with specific ranges of enrichment. The diameters of most of the fuel elements vary from 3.45 cm (1.36 in) to 3.63 cm (1.43 in) to accommodate slight differences in the process tubes of different reactors. The fuel for the KE and KW Reactors was generally the largest, and even included some 5.1 cm (2 in) diameter elements. The fuel for the 105-C Reactor was larger than that used in the other five older reactors. These variations in length, diameter, and secondary features, such as tru-line ends, lead to a large number of similar, but distinct, fuel element types. To allow for easy differentiation, each fuel element is stamped with a six character code. The second letter of the code identifies the specific model of fuel element, including the enrichment, dimensions, and group of reactors to receive it.

The physical characteristics of some SPR fuel element types are listed in Table 4.2 and reference drawings H-3-7350 and H-3-7352 (HAPO 1963b). Additional details are available in the reference drawings.

SINGLE PASS REACTOR FUEL ELEMENTS

(FBI0002D)

Figure 4.1: Single Pass Reactor Fuel Elements

Table 4.2: Physical Characteristics of Single Pass Reactor Fuel¹

		Fuel Element					Uranium Core			Cap	
		Length	Diameter		Cladding Wall Thickness		Length	Diameter		Weight	Length
Code ²	% ²³⁵ U		Outer	Inner	Outer	Inner		Outer	Inner		
C2N	0.714	22.8	3.72	0.953	0.10	0.114	21.3	3.48	1.22	3.34	0.635
K4N	0.714	22.8	3.71	0.978	0.094	0.13	21.3	3.48	1.26	3.30	0.635
K5N	0.714	22.5	3.86	1.07	0.094	0.13	21.1	3.63	1.35	3.56	0.635
O3N	0.714	22.8	3.67	0.787	0.094	0.13	21.3	3.44	1.07	3.37	0.635
C6N	0.714	22.5	5.04	0.848	0.11	0.13	21.1	4.78	1.15	6.75	0.635
C3E	0.947	16.9	3.71	0.953	0.094	0.13	15.4	3.48	1.24	2.40	0.635
K5E	0.947	16.6	3.83	1.10	0.094	0.13	15.2	3.60	1.38	2.49	0.635
O3E	0.947	16.9	3.67	0.787	0.094	0.13	15.4	3.44	1.07	2.43	0.635
K4W	0.714	16.9	3.71	1.02	0.094	0.13	15.4	4.48	1.30	2.36	0.635
K5(1.25)	1.25	14.1	3.82	1.12	0.094	0.13	12.7	3.61	1.38	2.08	0.635

- All dimensions are in centimeters, except that the uranium core weight is in kilograms.
- The first letter of the code indicates in which reactors the fuel was used. "K" designates the 105-KE and 105-KW Reactors. "C" designates the 105-C Reactor. "O" designates the other five reactors. The number indicates the number of times the design has been modified. The third character indicates the general uranium content and other features. "N" indicates natural uranium content (0.714% ²³⁵U). "E" indicates 0.947% ²³⁵U enrichment. "W" indicates natural enrichment with ²³⁵U cooling water mixing attachment. "D" indicates depleted uranium. Enrichments above 0.947% ²³⁵U are indicated by the specific enrichment enclosed within parentheses.

4.2.1 Fuel Fabrication (HAPO 1955, HAPO 1958)

In the lead dip fabrication process used since 1954, a machined uranium core was sealed within, and metallurgically bonded to, an aluminum can. The core was first subjected to a degreasing and pickling operation to clean its surface. Then the core was heated to bonding temperature by immersion in a "duplex" bath of aluminum-silicon alloy (Al-Si) floating on lead. The core was preheated to brazing temperature by the lead layer and then pre-wet with the brazing alloy by brief agitation within the Al-Si layer. The core was then transferred to a heated Al-Si canning bath, where it was inserted into a preheated aluminum can and topped with an aluminum cap. After quenching, the fuel element was welded to create a continuous bond between the uranium core, the Al-Si braze, and the aluminum can. The braze layer was approximately 20 mils thick. Next, the ends of the fuel element were machined to the proper length and contour. Then, the exposed braze area was closed by a tungsten inert-gas weld bead. Finally, the completed fuel element was heated in an autoclave and inspected visually and by other non-destructive methods.

Table 4.3 lists the chemical inventory in the SPR currently stored at Hanford. This information was calculated using typical impurities of the materials in the SPR fuel elements found in fuel manufacturing documentation (HAPO 1955, HAPO 1958).

Table 4.3: Total Chemical Inventory in Single Pass Reactor Fuel Metals
(All quantities in kg)

<u>Impurity</u>	<u>Uranium</u> <u>Alloy 501</u>	<u>Aluminum</u> <u>Alloy X-8001</u>	<u>Al-Si Braze</u>
Boron	NA ¹	1.20e-03	NA
Cadmium	NA	3.60e-03	NA
Carbon	2.54e+00	NA	NA
Chromium	2.20e-01	NA	NA
Cobalt	NA	1.20e-03	NA
Copper	NA	1.77e-01	9.30e-03
Iron	5.07e-01	8.40e-01	3.00e-02
Lithium	NA	9.60e-03	NA
Magnesium	8.45e-02	NA	NA
Manganese	8.45e-02	NA	NA
Nickel	3.38e-01	1.40e+00	NA
Nitrogen	3.38e-01	NA	NA
Silicon	2.54e-01	2.05e-01	NA
Others; each	NA	6.00e-02	3.00e-03
Others, total	NA	1.80e-01	1.20e-02

¹ NA indicates that the impurity is not present at any appreciable level.

4.3 FUEL INVENTORY

There are 3.38 metric tons [as total] heavy metal (MTHM) of SPR fuel at Hanford. The majority consists of 2.87 metric tons (MT) of highly depleted uranium and 8.7 kilograms of plutonium, contained in 779 fuel elements. This highly depleted SPR fuel, stored until 1995 at the PUREX facility, has been moved to the KW Basin. The highly depleted fuel is a portion of the fuel which was irradiated in the KE and KW Reactors between December 1966 and May 1968 to produce high ²⁴⁰Pu content plutonium (WHC 1994b). The pre-irradiation ²³⁵U content of the fuel was 0.14% and the fuel was irradiated to an average exposure of 3000 MWD/MT. The post-irradiation ²³⁵U content is 0.064% and the plutonium contains 26% ²⁴⁰Pu.

The remaining 0.50 MTHM of the SPR fuel is natural or enriched uranium contained in 185 fuel elements; 0.40 MTHM in 138 elements within the KE Basin and 0.10 MTHM in 47 elements within the KW Basin (WHC 1991). The fuel is predominately the I&E tru-line design. Most of this fuel is residual material from the KE and the KW Reactors. A small amount of fuel from the clean-out of the 105-C and the 105-D Storage Basins is also included in the 0.50 MTHM total. The specific types and numbers of fuel elements are listed below.

Table 4.4: Single Pass Reactor Fuel Elements Stored in the K Basins

<u>Fuel Type</u>	<u>Basin</u>	<u>Quantity</u>
C2N	KE	2
K4N	KE	3
K5N	KE	42
O3N	KE	15
C3E	KE	42
K5E	KE	29
O3E	KE	4
K4W	KE	1
Uncertain ¹	KW	47

1) Existing records of these fuel elements do not indicate the specific fuel type, however their ²³⁵U enrichment does not exceed 1.25% (Larson 1990).

4.4 RADIONUCLIDE CONTENT

The radionuclide content of the depleted SPR fuel, estimated using the ORIGEN2 and WIMS-E computer codes, is shown in Table 4.5 (Schwinkendorf 1993). Radionuclide concentration and decay heat of the other SPR fuel stored in the KE and KW Basins are not available because of its small quantity, varied sources, and uncertain discharge and exposure history. However, nuclide concentrations and decay heat should be similar to that of N Reactor Mark IA fuel irradiated to a 6% ²⁴⁰Pu content because typical SPR fuel irradiation was less than 1000 MWD/MTU (HAPO 1944). It has been at least 20 years since the fuel was irradiated because the last SPR was shut down in 1971.

Table 4.5: Total Activity of Selected Radionuclides in SPR Fuel
(decayed to 1/1/94)

<u>Radionuclide</u>	<u>Activity (Ci)</u>
³ H	7.03e+02
⁶³ Ni	6.76e+01
⁸⁵ Kr	7.33e+02
⁹⁰ Sr	1.36e+04
⁹⁰ Y	1.36e+04
⁹⁹ Tc	6.67e+00
^{113m} Cd	1.69e+01
¹³⁴ Cs	6.76e+00
¹³⁷ Cs	3.48e+04
^{137m} Ba	3.30e+04
¹⁵¹ Sm	4.06e+02
¹⁵⁴ Eu	4.70e+02
¹⁵⁵ Eu	1.15e+02
²³⁸ Pu	1.45e+02
²³⁹ Pu	4.06e+02
²⁴⁰ Pu	6.42e+02
²⁴¹ Pu	1.69e+04
²⁴¹ Am	1.45e+03
Total	7.05e+04

4.5 FUEL STORAGE

The majority of the SPR fuel was stored until 1995 at the PUREX storage basin, although smaller amounts were also stored at the K Basins. As of September 1995, when the fuel from PUREX was transferred into the KW Basin, all SPR fuel has been located in the K Basins.

The fuel formerly stored at PUREX was stored in four buckets. Three of the four fuel buckets were briefly lifted out of the water so that the condition of the fuel could be observed and videotaped. The buckets were full of fuel elements. There were several larger elements that appeared to be either the 2 inch (5.1 centimeter) diameter fuel elements or overpack containers. Tru-line end features could be seen on a few elements, although end features on most elements could not be seen. There appeared to be sludge or corrosion product on many of the fuel elements.

5.0 105-K BASIN SLUDGE

5.1 SLUDGE SOURCES

Fuel handling operations initially caused some cladding damage in N-Reactor fuel. Subsequent fuel oxidation resulted in fuel and fission products accumulating in fuel canisters and on the floor of the KE Basin discharge chute where fuel handling occurred. Aluminum oxide, iron oxide, concrete grit, and other materials have also accumulated and mixed with the fuel corrosion products to form a sludge on the basin floors.

The sludge on the KE Basin floor is being generated from a variety of sources and is moved through the basins by water circulation and necessary in-pool operations. The sludge sources are:

- Fuel elements, which are oxidizing where cladding has been breached, thereby contributing uranium oxide and fission products to the sludge. Some oxidized fuel may have fallen to the floor of the basin through screens in the bottoms of some canisters (the Mark 0s) in the basin.
- The KE Basin is an unlined concrete pool. As the facility ages, concrete grit comes loose from the walls and falls to the basin floor.
- The facility is not airtight. High winds cause dust and other pollutants (dirt, insects, tumbleweeds, etc.) from the outside environment to enter the facility, and ultimately, the storage pool.
- There are carbon steel storage racks sitting in the storage pool to hold the fuel canisters in place. As these racks age, they present a source of rust corrosion and paint chips.
- The aluminum canisters were corroded over a two year period when chlorine was used as a biological control agent.

Some of the canisters in the KE Basin have solid bottoms (the Mark 1 and Mark 2 canisters). Sludge accumulation is likely within those canisters as a result of fuel oxidation due to breached cladding.

Because all the canisters in the KW Basin are closed from the basin pool environment, any sludge that has resulted from fuel oxidation in breached fuel assemblies has been retained within the canisters. Some sludge is present on the KW Basin floor due to rusting storage racks and dust or other dirt that has entered the facility from the outside environment.

5.2 SLUDGE LOCATION

A schematic of the K Basins is shown in Figure 5.1. The six pits connected to each basin are the Sandfilter Backwash Pit, the South Loadout Pit, the Wash Pit (also designated on drawings as the Dummy Elevator Pit), the Tech View Pit, the Weasel Pit, and the Discharge Chute. Figure 5.1 also indicates which areas of the basins contain sludge.

The Sandfilter Backwash Pit, also designated on drawings as the North Loadout Pit, is a concrete pool (connected to the main basin by a closed-off fuel canister transfer channel) that is currently used to collect the backwash residues from the fuel pool sandfilter. This pit is isolated from the basin proper to provide a storage area for sludge that has been backwashed from the sand filter. This backwash process has deposited significant quantities of sludge into the pit.

The basin water is recirculated through the pool cooling and cleanup system in order to (WHC 1993):

- (1) Remove the decay heat generated by the irradiated fuel stored in the basin.
- (2) Maintain water quality and clarity.
- (3) Control the concentration of soluble and particulate radioactive nuclides in the water to minimize personnel radiation exposure.

This recirculation of basin water results in sludge migration throughout the basin. In addition there is sufficient decay heat from the fuel canisters to drive convection currents that further promote sludge migration.

Some intentional activities in the KE Basin have redistributed the basin sludge. The Fuel Segregation Campaign, which took place in the early 1980s, was initiated to separate the fuel according to enrichment. Fuel canisters were transported one by one from the main basin to the Discharge/Pickup Chute Area. There the canisters were emptied of their contents and repackaged. As the canisters were emptied, any sludge that had accumulated in the canister was also dumped into the Discharge Chute. Sludge was pumped from the Discharge Chute to the Weasel Pit in 1992, in order to clean up the Discharge Chute Area. Another sludge pumping campaign to move sludge from the Discharge Chute to the Weasel Pit was recently completed in order to prepare the basins for installation of barrier isolation doors to enhance the seismic integrity of the basins.

The back of the Weasel Pit is screened from the main basin by a 37 micron screen in order to retain most of the pumped sludge within the pit until it has had time to settle out. However, some fine particulate sludge still escapes back into the main basin, along with all dissolved material.

As a result of the sludge pumping activities, the Weasel Pit is expected to primarily contain fuel corrosion products (including fuel pieces $\leq$ 6 mm) that originated in canisters containing fuel with breached cladding. Therefore, the Weasel Pit sludge composition is currently assumed to be characteristic of the sludge in fuel canisters with broken fuel.

Mean Sludge Depth in KE Basin

INDEX	LOCATION	MEAN DEPTH (in)	INDEX	LOCATION	MEAN DEPTH (in)
1.	Tech View Pit - North Channel	0.97	8.	South Loadout Pit	5.25
2.	Tech View Pit - South Channel	0.21	9.	South Loadout Pit Channel	5.25
3.	Tech View Pit	0.88	10.	Dummy Elevator Pit / Wash Pit	6.00
4.	Weasel Pit Channel	12.00	11.	Basin Floor	1.90
5.	Weasel Pit	39.00	12.	Fuel Canister - Empty	0.22
6.	Sandfilter Backwash Pit	33.00	12.	Fuel Canister - Full, Good	0.50
7.	Sandfilter Backwash Pit Channel	20.00	12.	Fuel Canister - Full, Breached	4.00

Figure 5.1: Sludge Location in the 105-K Basins

5.3 SLUDGE DEPTH

A series of measurements of depth of sludge in the KE Basin is presented in *Sludge Measurement Results for 105 KE Basin* (Meling 1994). Data were obtained and recorded on video cassettes as well as on data sheets. The data provide physical measurements of sludge depth in the main basin and some pits, and a sampling of canister sludge depths. In the South Loadout Pit and Wash Pit, where the presence of debris made physical measurements infeasible, the sludge depth in the nearest main basin area was assigned to the area/pit in question. Using the depth measurements, sludge volume was estimated by analyzing the data with several different techniques (Meling 1994).

The *Summary Status of K Basins Sludge Characterization* (Baker 1995) provided a qualitative summary of sludge characteristic observations from a review of videotapes. Based on this review, the KE Basin sludge was described as having a thin, light colored crust that contained lower layers of flocculent brown material.

The table in Figure 5.1 above presents a summary of the mean sludge depth reported for major KE Basin regions. Additional details on sludge depth measurements, including a description of the approximate depth measurement locations, are provided in the Baker document and in *Sludge Measurement Results for 105 KE Basin* (Meling 1994).

Measurements of sludge depth were made recently in the KW Basin Sandfilter Backwash Pit and Transfer Channel. Brief reviews of the video tapes show the sludge in the Sandfilter Backwash Pit to be composed of a thin crust with underlying flocculent material. The top crust material settles fairly rapidly, while the underlying finer material tends to settle slowly. Differences noted between the sludge material in KE Basin and KW Basin include a darker, almost black appearance of the KW Basin flocculent sludge material, and a greater tendency to agglomerate (Baker 1995).

5.4 CURRENT INVENTORY BASIS

As shown on Figure 5.1, the basins can be divided into seven major areas for estimating the sludge inventory: Weasel Pit, Tech View Pit, Wash Pit, Sandfilter Backwash Pit, South Loadout Pit, Fuel Canisters, and Basin Floor. Four sludge samples were taken from the KE Basin in May 1993 for analysis (Bechtold 1993). The samples were taken as follows:

Sample S3-032-01	West side of Weasel Pit screen
Sample S3-032-02	East side of Weasel Pit screen
Sample S3-032-03	Center of KE Basin (grating #3718)
Sample S3-032-04*	Channel of North Loadout Pit (Sandfilter Backwash Pit)

*Alternate samples analysis used, see explanation following.

Chemical analysis of the sludge was achieved by filtering sludge sub-samples from the associated liquid, oven-drying, then digesting the residual solids in nitric acid solution. Chemical and radiochemical analysis produced results for the dried residual solids (see Appendix E, Tables E-1 and E-4).

Routine operational sampling of the Sandfilter Backwash Pit in May 1993 indicated that levels of plutonium had risen above allowable limits as a result of continuing corrosion of irradiated N-Reactor fuel in the basin water. Consequently an Unreviewed Safety Question (USQ) was declared regarding continued sand filter backwash operation. A sampling and analysis

plan was prepared that called for a sampling program to obtain sludge samples (Bechtold 1994). The results of this sample analysis were used to determine the chemical and radionuclide constituents in the Sandfilter Backwash Pit, as presented in Table 5.2.

Analyses of samples from the Tech View Pit, Wash Pit, South Loadout Pit, and Fuel Canisters are not currently available. Therefore, the current sludge inventory estimates are based on assuming sludge in these areas will be similar to, or bounded by, the composition of sludges from areas that have been sampled. Of the sample analyses available, the Weasel Pit sludge was found to contain the highest concentration of radionuclides. This result was expected based on the sludge pumping activities described in Section 5.2. Therefore, in order to bound the estimate of radionuclides in the sludge inventory, the Weasel Pit sludge composition is used to model sludge in the Tech View Pit, Wash Pit, and South Loadout Pit areas.

The Fuel Canisters have been separated into 1) empty canisters, 2) full canisters containing unbreached fuel, and 3) full canisters containing breached fuel. The sludge contained in the empty canisters and full canisters with unbreached fuel is expected to be composed mainly of dust, concrete, and other environmental pollutants. Therefore, the Basin Floor sludge sample is used to model the sludge composition in these areas. The sludge contained in the full canisters with breached fuel is expected to include corrosion products along with environmental pollutants. The Weasel Pit sample composition is used as a model of sludge in these canisters.

5.5 105-K EAST BASIN SLUDGE

The following table depicts the physical properties of the "wet" sludge (i.e., the sludge as it sits in the basin) in various areas of the KE Basin. These results are based on *Summary Status of K Basins Sludge Characterization* (Baker 1995). The exception is the Sandfilter Backwash Pit; its sludge properties came from *Safety Evaluation of the Plutonium and Uranium Content of the K East Basin Sandfilter Backwash Pit* (Warner 1994), and the chemical and radionuclide constituent data came from *Statistical Evaluation of the Data Obtained from the K East Basin Sandfilter Backwash Pit Samples* (Welsh 1994).

Table 5.1: Physical Properties of 105-K East Basin Sludge

		Volume ¹ (m ³)	Mass ¹ (Mg)	Density ¹ (g/ml)	Particle Size ² (μm)	Solids ³ (weight %)
WEASEL PIT		11.86	18.20	1.535	1.53	51.29
TECH VIEW PIT		0.43	0.66	1.535	1.53	51.29
WASH PIT		1.37	2.11	1.535	1.53	51.29
SANDFILTER BACKWASH PIT		6.27	8.25	1.31	1.57	32.00
SOUTH LOADOUT PIT		1.09	1.67	1.535	1.53	51.29
FUEL CANISTERS	Empty	0.39	0.44	1.13	1.68	17.52
	Full, Good	0.81	0.92	1.13	1.68	17.52
	Full, Breech	6.64	10.19	1.535	1.53	51.29
BASIN FLOOR		21.65	24.46	1.13	1.68	17.52
TOTAL		50.51	66.90			

- 1) Volume, mass, and density is of wet sludge, after settling.
- 2) Particle size is average size of all particles in sample.
Most particles are <2μm diameter; over half of volume is in particles >20-40 μm.
- 3) Weight percent of solids is the fractions of solids in the sludge (the remainder is water).

Table 6 in the Baker document includes a range of bounding sludge volumes. These estimates show that the total KE Basin sludge volume could be as high as 67 m³.

Inductively coupled plasma analysis, uranium analysis, gamma energy analysis (GEA), and radiochemical analysis of the digestates produced chemical and radionuclide compositions of sludge samples on a dry bases. The average of the two Weasel Pit samples was used to characterize sludge in the Weasel Pit. Total component inventories were calculated for each area from compositions and total sludge mass in the area. These calculations are presented in Appendix E. In order to determine the inventory of chemical constituents in the basin, the mass of wet sludge in each basin area was multiplied by the component weight fraction as determined through chemical analysis, and then multiplied by the solids concentration of wet sludge versus dried sludge. The resultant chemical inventory for the KE Basin sludge is presented in Table 5.2.

Table 5.2: Chemical Constituents in 105-K East Basin

	Weasel Pit (Kg)	Tech View Pit (Kg)	Wash Pit (Kg)	Sandfilter Backwash Pit (Kg)	South Loadout Pit (Kg)	Fuel ¹ Canisters (Kg)	Basin Floor (Kg)	Total (Kg)
Al	1.71e+02	6.19e+00	1.98e+01	3.00e+01	1.57e+01	1.01e+02	8.91e+01	4.32e+02
As	4.05e-01	1.47e-02	4.70e-02		3.72e-02	2.31e-01	7.11e-02	8.06e-01
Be	8.56e-01	3.10e-02	9.92e-02	4.01e-01	7.85e-02	4.92e-01	2.30e-01	2.19e+00
Ba	3.28e-02	1.19e-03	3.80e-03	2.92e-02	3.01e-03	2.43e-02	1.06e-01	2.00e-01
Ca	3.19e+01	1.16e+00	3.70e+00	2.27e+01	2.93e+00	1.80e+01	2.97e+00	8.34e+01
Cd	7.19e-01	2.61e-02	8.34e-02	2.06e-01	6.60e-02	4.17e-01	2.54e-01	1.77e+00
Co	2.06e-01	7.46e-03	2.39e-02		1.89e-02	1.17e-01	3.75e-02	4.11e-01
Cr	8.64e+00	3.13e-01	1.00e+00	4.94e-01	7.93e-01	4.89e+00	1.02e+00	1.72e+01
Cu	2.67e+00	9.66e-02	3.09e-01	1.04e+00	2.45e-01	1.54e+00	9.13e-01	6.81e+00
Fe	2.26e+03	8.21e-01	2.62e+02	8.98e+01	2.08e+02	1.28e+03	3.15e+02	4.51e+03
K	1.50e+00	5.43e-02	1.74e-01		1.37e-01	8.51e-01	2.15e-01	2.93e+00
Li	7.98e-01	2.89e-02	9.25e-02		7.32e-02	4.47e-01	4.84e-03	1.44e+00
Mg	7.85e+00	2.85e-01	9.10e-01	2.86e+00	7.20e-01	4.46e+00	1.29e+00	1.84e+01
Mn	2.81e+00	1.02e-01	3.26e-01	1.24e+00	2.58e-01	1.61e+00	6.69e-01	7.02e+00
Mo	3.45e-01	1.25e-02	4.00e-02		3.17e-02	1.97e-01	6.43e-02	6.91e-01
Na	1.09e+00	3.95e-02	1.26e-01	1.63e+00	1.00e-01	6.21e-01	1.89e-01	3.80e+00
Nd	1.22e+00	4.43e-02	1.42e-01		1.12e-01	7.00e-01	2.87e-01	2.51e+00
Ni	8.70e-01	3.16e-02	1.01e-01		7.99e-02	4.97e-01	1.73e-01	1.75e+00
P	3.87e+00	1.40e-01	4.49e-01		3.55e-01	2.21e+00	8.19e-01	7.84e+00
Pb	4.55e+00	1.65e-01	5.27e-01	3.47e-01	4.17e-01	2.59e+00	8.36e-01	9.43e+00
S	3.02e+00	1.10e-01	3.50e-01		2.77e-01	1.74e+00	9.17e-01	6.42e+00
Sb	2.02e+00	7.73e-02	2.34e-01		1.85e-01	1.15e+00	3.77e-01	4.04e+00
Si	1.33e+01	4.81e-01	1.54e+00	3.44e+00	1.22e+00	7.90e+00	8.53e+00	3.64e+01
Sm	4.41e+00	1.60e-01	5.11e-01		4.05e-01	2.52e+00	9.64e-01	8.97e+00
Sr	3.09e-01	1.12e-02	3.59e-02		2.84e-02	1.76e-01	4.89e-02	6.10e-01
Ti	1.32e+00	4.77e-02	1.53e-01		1.21e-01	7.48e-01	1.92e-01	2.58e+00
Tl	1.34e+00	4.87e-02	1.56e-01		1.23e-01	7.75e-01	4.29e-01	2.87e+00
V	3.45e+00	1.25e-01	4.00e-01		3.16e-01	1.97e+00	7.07e-01	6.97e+00
Zn	9.27e+00	3.36e-01	1.08e+00	8.88e-01	8.51e-01	5.28e+00	1.51e+00	1.92e+01
Zr	1.41e+00	5.13e-02	1.64e-01		8.27e-01	3.27e-01	6.34e-01	3.22e+00
U	6.97e+02	2.53e+01	8.08e+01	3.90e+01	6.40e+01	3.99e+02	1.50e+02	1.46e+03
Pu	2.43e+00	8.80e-02	2.81e-01	9.75e-02	2.23e-01	1.38e+00	3.86e-01	4.88e+00
Other	6.09e+03	2.21e+02	7.06e+02	2.45e+03	5.59e+02	3.62e+03	3.71e+03	1.73e+04
TOTAL Dry Sludge	9.33e+03	3.39e+02	1.08e+03	2.64e+03	8.57e+02	5.46e+03	4.29e+03	2.40e+04
Inter- stitial Water	8.87e+03	3.21e+02	1.03e+03	5.61e+03	8.13e+02	6.09e+03	2.02e+04	4.29e+04
TOTAL Wet Sludge	1.82e+04	6.60e+02	2.11e+03	8.25e+03	1.67e+03	1.16e+04	2.45e+04	6.69e+04

1) This column entitled "Fuel Canisters" is inclusive of all three categories of fuel canisters: the empty canisters, full canisters of unbreached fuel, and full canisters of breached fuel.

Radionuclides present in the KE Basin sludge were estimated by the same method used to estimate chemical components. The results are presented in Table 5.3, assuming the decay date basis for radionuclides is identical to the date data were reported. The exception is plutonium, whose activity was calculated based on the mass provided in Table 5.2. The Pu isotope mass fractions were taken from the Warner document.

Table 5.3: Radionuclides in 105-K East Basin Sludge

(Decay Date: May 1993)

	WEASEL PIT	TECH VIEW PIT	WASH PIT	SANDFILTER BACKWASH PIT	SOUTH LOADOUT PIT	FUEL ¹ CANISTERS	BASIN FLOOR	TOTAL
	Ci	Ci	Ci	Ci	Ci	Ci	Ci	Ci
Co 60	1.10e+01	3.99e-01	1.28e+00	1.36e+00	1.01e+00	6.56e+00	6.99e+00	2.86e+01
Sr 90	5.51e+02	2.00e+01	6.39e+01		5.06e+01	3.26e+02	3.21e+02	1.33e+03
Y 90	5.51e+02	2.00e+01	6.39e+01		5.06e+01	3.26e+02	3.21e+02	1.33e+03
Cs 137	4.21e+02	1.53e+01	4.89e+01	6.14e+01	3.87e+01	2.46e+02	1.78e+02	1.01e+03
Ba 137m	3.99e+02	1.45e+01	4.62e+01	5.81e+01	3.66e+01	2.33e+02	1.68e+02	9.55e+02
Eu 154	1.16e+01	4.21e-01	1.35e+00	1.36e+00	1.07e+00	6.90e+00	7.11e+00	2.98e+01
Eu 155	6.57e+00	2.58e-01	7.62e-01	7.46e-01	6.03e-01	3.91e+00	4.17e+00	1.70e+01
Pu 238	3.28e+01	1.19e+00	3.81e+00	1.32e+00	3.01e+00	1.87e+01	5.22e+00	6.60e+01
Pu 239	1.29e+02	4.69e+00	1.50e+01	5.20e+00	1.19e+01	7.36e+01	2.06e+01	2.60e+02
Pu 240	7.10e+01	2.57e+00	8.23e+00	2.85e+00	6.51e+00	4.04e+01	1.13e+01	1.43e+02
Pu 241	2.81e+03	1.02e+02	3.26e+02	1.13e+02	2.58e+02	1.60e+03	4.47e+02	5.66e+03
Pu 242	2.17e-02	7.86e-04	2.51e-03	8.71e-04	1.99e-03	1.23e-02	3.44e-03	4.36e-02
Am 241	2.50e+02	9.06e+00	2.90e+01	6.65e+00	2.29e+01	1.56e+02	2.95e+02	7.69e+02
TOTAL	5.24e+03	1.90e+02	6.08e+02	2.52e+02	4.82e+02	3.04e+03	1.79e+03	1.16e+04

1) This column entitled "Fuel Canisters" is inclusive of all three categories of fuel canisters: the empty canisters, full canisters of unbreached fuel, and full canisters of breached fuel.

Table 5.4 provides an estimate of the total radionuclides for a composite of the 105-K East Basin sludge. The mass and heat generation rate of each isotope based on a decay date of 1/1/95 is derived from the activities estimated in Table 5.3, combined with specific activities and specific heat generation rates for the individual isotopes presented in Appendix C.

Table 5.4: 105-K East Basin Radionuclide Totals

RADIONUCLIDE	ACTIVITY decay date: 5/1/93 (Ci)	ACTIVITY decay date: 1/1/95 (Ci)	MASS decay date: 1/1/95 (Kg)	HEAT GENERATION decay date: 1/1/95 (BTU/hr)
Co 60	2.86e+01	2.30e+01	2.03e-05	1.21e+00
Sr 90	1.33e+03	1.28e+03	9.38e-03	5.07e+00
Y 90	1.33e+03	1.28e+03	2.35e-06	2.42e+01
Cs 137	1.01e+03	9.72e+02	1.12e-02	3.67e+00
Ba 137m	9.55e+02	9.20e+02	1.71e-09	1.23e+01
Eu 154	2.98e+01	2.61e+01	9.67e-05	7.97e-01
Eu 155	1.70e+01	1.35e+01	2.90e-05	3.35e-02
Pu 238	6.60e+01	6.51e+01	3.86e-03	7.47e+00
Pu 239	2.60e+02	2.60e+02	4.18e+00	2.74e+01
Pu 240	1.43e+02	1.43e+02	6.28e-01	1.52e+01
Pu 241	5.66e+03	5.22e+03	5.50e-02	5.99e-01
Pu 242	4.36e-02	4.36e-02	1.15e-02	4.39e-03
Am 241	7.69e+02	7.82e+02	2.28e-01	8.87e+01
TOTAL	1.16e+04	1.10e+04	5.13e+00	1.87e+02

5.6 105-K WEST BASIN SLUDGE

Extensive information is not currently available on sludge in the KW Basin. The canisters in the KW Basin are all closed and therefore isolated from the basin environment. Sludge that has accumulated in this basin and related pits is expected to be from environmental pollutants such as dust, dirt, paint chips, etc..

Table 5.5 presents information known at this time regarding sludge in the KW Basin. The volume of sludge in the Sandfilter Backwash Pit was obtained from integration of sludge depth measurements. It was also noted that significant sludge exists in the South Loadout Pit (from decapping canisters prior to shipment to PUREX) and the Tech View Pit (from changing filter cartridges) (Baker 1995).

Table 5.5: Physical Properties of 105-K West Basin Sludge

	WEASEL PIT	TECH VIEW PIT	WASH PIT	SANDFILTER BACKWASH PIT	SOUTH LOADOUT PIT	FUEL CANISTERS	BASIN FLOOR	TOTAL
Volume ¹ (m ³)				3.60				
Mass ¹ (Mg)								
Density ¹ (g/ml)								
Particle Size ² (μm)								
Solids ³ (wt. %)								

- 1) Volume, mass, and density is of wet sludge, after settling.
- 2) Particle size is average size of all particles in sample.
Most particles are <2μm diameter; over half of volume is in particles >20-40 μm.
- 3) Weight percent of solids is the fractions of solids in the sludge (the remainder is water).

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APPENDIX A

ACCOUNTABILITY DATA BASE FOR N REACTOR FUEL IN THE K BASINS

The following pages contain the database used in the development of the radionuclide tables in the document. The database includes the following 8 fields:

- The first field is a location code a 108 in this field indicates the 105 K-East Basin and a 107 indicates the 105 K-West Basin.
- The second field is the key number which corresponds to the date the fuel was discharged from the reactor. The numbers are HAPO days and increase by one each day. HAPO day number one occurred September 26, 1944, the day that B Reactor achieved initial criticality.

A few records have key numbers in excess of 20000, (which correspond to dates no earlier than 1999) these few had 10000 added to differentiate fuel types with two different burnup rates discharged on the same day. 10000 was subtracted from the key date of all records with key dates in excess of 20000 when the database was converted into a data file.

- The third field is the fuel model, and was not used.
- The fourth field is the material type and this explains why there is from one to three lines for each record. The numbers are codes for uranium, plutonium, and neptunium as follows: 10 = depleted uranium (contains less than 0.71% ^{235}U); 20 = enriched uranium (contains more than 0.71% ^{235}U); 50 = plutonium; 81 = naturally occurring uranium (contains 0.71% ^{235}U); and 82 = neptunium. Most of the records contain a line for uranium, a line for plutonium, and a line for neptunium; some contain only uranium and plutonium; and a few contain only uranium.
- The fifth field is a sub ID code, and was not used.
- The sixth field is the fuel type, V = mark IV fuel, and I = mark 1A fuel.
- The seventh field contains isotopic breakdown information. In the uranium lines the seventh field is the post-irradiation ^{235}U content given as a percent of the total uranium. In the plutonium lines the seventh field is the ^{240}Pu content given as a percent of the total plutonium. The seventh field has no meaning in the neptunium lines.
- The eighth field is the discharge mass in metric tons. In the uranium line it is the total uranium mass, in the plutonium line it is the total plutonium mass.

Of the approximately 500 records in the accountability database, 9 records may be single pass reactor fuel. These 9 records account for 0.32 MTU or about 0.015% of the total inventory in the K Basins. All 9 records contain 5.49% ^{240}Pu and are modeled as N Reactor Mark IV fuel. This fuel does not significantly impact either the chemical or radiological inventory of N Reactor fuel in the K Basins and therefore is modeled as N Reactor fuel.

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	09621	C2N	10		V	0.61	0.006668
108	09621	C2N	50		V	5.94	0.000005
108	09621	C3E	10		V	0.61	0.076639
108	09621	C3E	50		V	5.94	0.000055
108	09621	C3E	20	1	V	0.84	0.023952
108	09621	C3E	50	1	V	6.01	0.00002
108	09621	K4N	10		V	0.62	0.009893
108	09621	K4N	50		V	5.94	0.000007
108	09621	K4W	10		V	0.61	0.002372
108	09621	K4W	50		V	5.94	0.000002
108	09621	K5E	10		V	0.61	0.072602
108	09621	K5E	50		V	5.94	0.000052
108	09621	K5N	10		V	0.61	0.149739
108	09621	K5N	50		V	5.94	0.000108
108	09621	O3E	10		V	0.61	0.009721
108	09621	O3E	50		V	5.94	0.000007
108	09621	O3N	10		V	0.61	0.050439
108	09621	O3N	50		V	5.94	0.000036
108	10001	B09C	20		V	0.79	0.042041
108	10001	B09C	50		V	12.4	0.000078
108	10001	B09C	82		V	0	0.000001
108	10001	B09S	20		V	0.73	0.014875
108	10001	B09S	50		V	12.4	0.000027
108	10001	B09S	82		V	0	3.60e-07
108	10001	X09A	20		V	0.79	0.006627
108	10001	X09A	50		V	12.4	0.000012
108	10001	X09A	82		V	0	1.60e-07
108	10001	X09C	20		V	0.79	0.009892
108	10001	X09C	50		V	12.4	0.000018
108	10001	X09C	82		V	0	2.40e-07
108	10001	X09E	20		V	0.73	0.029876
108	10001	X09E	50		V	12.4	0.000055
108	10001	X09E	82		V	0	7.20e-07
108	10001	X09S	20		V	0.73	0.0035
108	10001	X09S	50		V	12.4	0.000065
108	10001	X09S	82		V	0	8.50e-07
108	10201	A09A	20		V	0.77	1.073737
108	10201	A09A	50		V	9.09	0.001702
108	10201	A09A	82		V	0	0.00002
108	10201	B09E	20		V	0.76	0.365445
108	10201	B09E	50		V	9.09	0.000579
108	10201	B09E	82		V	0	0.000007
108	10201	B09S	20		V	0.77	0.343367
108	10201	B09S	50		V	9.09	0.000544
108	10201	B09S	82		V	0	0.000006
108	10201	N09A	20		V	0.8	0.53617
108	10201	N09A	50		V	9.09	0.000849
108	10201	N09A	82		V	0	0.000001
108	10201	X09E	20		V	0.8	0.171798
108	10201	X09E	50		V	9.09	0.000272
108	10201	X09E	82		V	0	0.000003

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	10201	X09S	20		V	0.78	0.16156
108	10201	X09S	50		V	9.09	0.000256
108	10201	X09S	82		V	0	0.000003
108	10259	A09A	20		V	0.76	0.05052
108	10259	A09A	50		V	10.41	0.000091
108	10259	A09A	82		V	0	0.000001
108	10259	B09C	20		V	0.76	1.474138
108	10259	B09C	50		V	10.41	0.002666
108	10259	B09C	82		V	0	0.000034
108	10259	B09E	20		V	0.74	12.32238
108	10259	B09E	50		V	10.41	0.022279
108	10259	B09E	82		V	0	0.000279
108	10259	N09A	20		V	0.78	0.03153
108	10259	N09A	50		V	10.41	0.000057
108	10259	N09A	82		V	0	7.00e-07
108	10259	X09C	20		V	0.79	0.717121
108	10259	X09C	50		V	10.41	0.001296
108	10259	X09C	82		V	0	0.000018
108	10259	X09E	20		V	0.77	5.748173
108	10259	X09E	50		V	10.41	0.010395
108	10259	X09E	82		V	0	0.000131
108	10294	A07B	10		V	0.54	0.032558
108	10294	A07B	50		V	8.94	0.000049
108	10294	A07B	82		V	0	6.00e-07
108	10294	A09A	20		V	0.74	0.012625
108	10294	A09A	50		V	10.4	0.000023
108	10294	A09A	82		V	0	2.90e-07
108	10294	B09C	20		V	0.76	0.810745
108	10294	B09C	50		V	10.4	0.001462
108	10294	B09C	82		V	0	0.000018
108	10294	B09E	20		V	0.74	24.10675
108	10294	B09E	50		V	10.4	0.043476
108	10294	B09E	82		V	0	0.000546
108	10294	B09S	20		V	0.74	0.760911
108	10294	B09S	50		V	10.4	0.001372
108	10294	B09S	82		V	0	0.000017
108	10294	N07B	10		V	0.56	0.027111
108	10294	N07B	50		V	8.94	0.000041
108	10294	N07B	82		V	0	5.00e-07
108	10294	N09A	20		V	0.77	0.006305
108	10294	N09A	50		V	10.4	0.000011
108	10294	N09A	82		V	0	1.40e-07
108	10294	X09C	20		V	0.78	0.380707
108	10294	X09C	50		V	10.4	0.000687
108	10294	X09C	82		V	0	0.000008
108	10294	X09E	20		V	0.77	11.33217
108	10294	X09E	50		V	10.4	0.020432
108	10294	X09E	82		V	0	0.000258
108	10294	X09S	20		V	0.77	0.344011
108	10294	X09S	50		V	10.4	0.00062
108	10294	X09S	82		V	0	0.000008
108	10350	B09C	20		V	0.77	0.031594
108	10350	B09C	50		V	9.28	0.000051
108	10350	B09C	82		V	0	6.30e-07

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	10350	B09E	20		V	0.77	0.238337
108	10350	B09E	50		V	9.28	0.000386
108	10350	B09E	82		V	0	0.000005
108	10350	B09S	20		V	0.77	0.08958
108	10350	B09S	50		V	9.28	0.000145
108	10350	B09S	82		V	0	0.000002
108	10350	X09C	20		V	0.79	0.014837
108	10350	X09C	50		V	9.28	0.000024
108	10350	X09C	82		V	0	3.00e-07
108	10350	X09E	20		V	0.79	0.11202
108	10350	X09E	50		V	9.28	0.000182
108	10350	X09E	82		V	0	0.000002
108	10350	X09S	20		V	0.78	0.042138
108	10350	X09S	50		V	9.28	0.000068
108	10350	X09S	82		V	0	8.40e-07
108	10385	A07B	10		V	0.51	0.130147
108	10385	A07B	50		V	10.21	0.000234
108	10385	A07B	82		V	0	0.000003
108	10385	B09A	20		V	0.72	0.14074
108	10385	B09A	50		V	11.21	0.000274
108	10385	B09A	82		V	0	0.000004
108	10385	B09E	20		V	0.72	23.76333
108	10385	B09E	50		V	11.21	0.046287
108	10385	B09E	82		V	0	0.000614
108	10385	B09S	20		V	0.73	0.298316
108	10385	B09S	50		V	11.21	0.000581
108	10385	B09S	82		V	0	0.000008
108	10385	N07B	10		V	0.53	0.078856
108	10385	N07B	50		V	10.21	0.000136
108	10385	N07B	82		V	0	0.000002
108	10385	X09A	20		V	0.76	0.059597
108	10385	X09A	50		V	11.21	0.000116
108	10385	X09A	82		V	0	0.000002
108	10385	X09E	20		V	0.76	11.15723
108	10385	X09E	50		V	11.21	0.021722
108	10385	X09E	82		V	0	0.000284
108	10385	X09S	20		V	0.76	0.154433
108	10385	X09S	50		V	11.21	0.000301
108	10385	X09S	82		V	0	0.000004
108	10456	A9D	20		V	0.72	0.00953
108	10456	A9D	50		V	11.24	0.000018
108	10456	A9D	82		V	0	2.40e-07
108	10456	B09E	20		V	0.88	0.031856
108	10456	B09E	50		V	11.24	0.000061
108	10456	B09E	82		V	0	8.00e-07
108	10456	BXA	20		V	0.73	0.413963
108	10456	BXA	50		V	11.24	0.000793
108	10456	BXA	82		V	0	0.000011
108	10456	BXC	20		V	0.74	1.345611
108	10456	BXC	50		V	11.24	0.002577
108	10456	BXC	82		V	0	0.000035
108	10456	BXE	20		V	0.73	21.40138
108	10456	BXE	50		V	11.24	0.040972
108	10456	BXE	82		V	0	0.000541

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	10456	BXS	20		V	0.74	3.180601
108	10456	BXS	50		V	11.24	0.00609
108	10456	BXS	82		V	0	0.000081
108	10576	BXE	20		V	0.74	3.874377
108	10576	BXE	50		V	11.09	0.00738
108	10576	BXE	82		V	0	0.000098
108	10576	BXS	20		V	0.74	0.043871
108	10576	BXS	50		V	11.09	0.000084
108	10576	BXS	82		V	0	0.000001
108	10602	A7B	10		V	0.48	0.081265
108	10602	A7B	50		V	13.75	0.000188
108	10602	A7B	82		V	0	0.000003
108	10602	B09E	20		V	0.95	0.031895
108	10602	B09E	50		V	11.09	0.00003
108	10602	B09E	82		V	0	4.00e-07
108	10602	BXA	10		V	0.69	0.558238
108	10602	BXA	50		V	13.75	0.001293
108	10602	BXA	82		V	0	0.000019
108	10602	BXC	20		V	0.74	0.030929
108	10602	BXC	50		V	11.09	0.000029
108	10602	BXC	82		V	0	3.80e-07
108	10602	BXE	10		V	0.69	24.73497
108	10602	BXE	50		V	13.75	0.057294
108	10602	BXE	82		V	0	0.000828
108	10602	BXS	10		V	0.64	1.225649
108	10602	BXS	50		V	13.75	0.002839
108	10602	BXS	82		V	0	0.000041
108	10679	A7B	10		V	0.48	0.308771
108	10679	A7B	50		V	14.09	0.000728
108	10679	A7B	82		V	0	0.000011
108	10679	B09A	20		V	0.83	0.014109
108	10679	B09A	50		V	11.09	0.000027
108	10679	B09A	82		V	0	3.50e-07
108	10679	BXA	10		V	0.67	0.123994
108	10679	BXA	50		V	14.09	0.000292
108	10679	BXA	82		V	0	0.000004
108	10679	BXC	20		V	0.74	1.252617
108	10679	BXC	50		V	11.09	0.002385
108	10679	BXC	82		V	0	0.000031
108	10679	BXE	10		V	0.68	22.91124
108	10679	BXE	50		V	14.09	0.054006
108	10679	BXE	82		V	0	0.000796
108	10679	BXS	10		V	0.71	1.293334
108	10679	BXS	50		V	14.09	0.003049
108	10679	BXS	82		V	0	0.000045
108	10742	B09C	10		V	0.7	1.167243
108	10742	B09C	50		V	14.25	0.002818
108	10742	B09C	82		V	0	0.000042
108	10742	B09E	10		V	0.66	12.33092
108	10742	B09E	50		V	14.25	0.029774
108	10742	B09E	82		V	0	0.000443
108	10742	B09S	10		V	0.67	0.461833
108	10742	B09S	50		V	14.25	0.001115
108	10742	B09S	82		V	0	0.000017

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	10742	X09C	20		V	0.74	0.548368
108	10742	X09C	50		V	10.8	0.001091
108	10742	X09C	82		V	0	0.000017
108	10742	X09E	10		V	0.71	5.791929
108	10742	X09E	50		V	14.25	0.013986
108	10742	X09E	82		V	0	0.00021
108	10742	X09S	20		V	0.72	0.224427
108	10742	X09S	50		V	10.8	0.000447
108	10742	X09S	82		V	0	0.000007
108	10764	B09S	20		V	0.95	0.052151
108	10764	B7E	81		V	0	0.04681
108	10764	BXA	20		V	0.95	0.446186
108	10764	BXE	20		V	0.95	0.365569
108	10764	BXR	20		V	0.95	0.373599
108	10764	BXS	20		V	0.95	0.681973
108	10764	X09A	20		V	0.93	0.00572
108	10764	X09E	20		V	0.94	0.013134
108	10764	X09R	20		V	0.95	0.00537
108	10770	BXC	20		V	0.73	0.680305
108	10770	BXC	50		V	11.66	0.001345
108	10770	BXC	82		V	0	0.000018
108	10770	BXE	10		V	0.69	14.07941
108	10770	BXE	50		V	13.91	0.032713
108	10770	BXE	82		V	0	0.000489
108	10770	BXS	10		V	0.69	4.426014
108	10770	BXS	50		V	13.91	0.010284
108	10770	BXS	82		V	0	0.000152
108	10832	A7B	10		V	0.49	0.260079
108	10832	A7B	50		V	13.65	0.0006
108	10832	A7B	82		V	0	0.000009
108	10832	BXA	10		V	0.66	0.041328
108	10832	BXA	50		V	13.65	0.000095
108	10832	BXA	82		V	0	0.000001
108	10832	BXC	10		V	0.71	2.039846
108	10832	BXC	50		V	13.65	0.004703
108	10832	BXC	82		V	0	0.000007
108	10832	BXE	10		V	0.69	47.20901
108	10832	BXE	50		V	13.65	0.108844
108	10832	BXE	82		V	0	0.0016
108	10832	BXS	10		V	0.66	3.131132
108	10832	BXS	50		V	13.65	0.00722
108	10832	BXS	82		V	0	0.000106
108	10982	A07B	10		V	0.48	0.173428
108	10982	A07B	50		V	14.97	0.000433
108	10982	A07B	82		V	0	0.000007
108	10982	B09A	10		V	0.69	0.126571
108	10982	B09A	50		V	14.97	0.000316
108	10982	B09A	82		V	0	0.000005
108	10982	B09C	20		V	0.71	0.704745
108	10982	B09C	50		V	11.7	0.001436
108	10982	B09C	82		V	0	0.000002

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	10982	B09E	10		V	0.65	15.16333
108	10982	B09E	50		V	14.97	0.037897
108	10982	B09E	82		V	0	0.000584
108	10982	B09S	10		V	0.67	0.938379
108	10982	B09S	50		V	14.97	0.002345
108	10982	B09S	82		V	0	0.000036
108	10982	N07B	10		V	0.52	0.08666
108	10982	N07B	50		V	14.97	0.000217
108	10982	N07B	82		V	0	0.000003
108	10982	X09A	20		V	0.76	0.059598
108	10982	X09A	50		V	11.7	0.000122
108	10982	X09A	82		V	0	0.000002
108	10982	X09C	20		V	0.75	0.331073
108	10982	X09C	50		V	11.7	0.000675
108	10982	X09C	82		V	0	0.000009
108	10982	X09E	10		V	0.7	7.072527
108	10982	X09E	50		V	14.97	0.01768
108	10982	X09E	82		V	0	0.000275
108	10982	X09S	20		V	0.72	0.497909
108	10982	X09S	50		V	11.7	0.001015
108	10982	X09S	82		V	0	0.000014
108	11036	A09D	10		V	0.68	0.006356
108	11036	A09D	50		V	14.37	0.000015
108	11036	A09D	82		V	0	2.30e-07
108	11036	B09C	20		V	0.73	0.326187
108	11036	B09C	50		V	13.31	0.000729
108	11036	B09C	82		V	0	0.000011
108	11036	B09E	10		V	0.66	7.798047
108	11036	B09E	50		V	14.37	0.018824
108	11036	B09E	82		V	0	0.000285
108	11036	B09S	10		V	0.65	0.684893
108	11036	B09S	50		V	14.37	0.001654
108	11036	B09S	82		V	0	0.000025
108	11036	N09D	20		V	0.73	0.003169
108	11036	N09D	50		V	13.31	0.000007
108	11036	N09D	82		V	0	1.00e-07
108	11036	X09A	10		V	0.69	0.013225
108	11036	X09A	50		V	14.37	0.000032
108	11036	X09A	82		V	0	4.80e-07
108	11036	X09C	20		V	0.76	0.123564
108	11036	X09C	50		V	13.31	0.000276
108	11036	X09C	82		V	0	0.000004
108	11036	X09E	10		V	0.71	3.699834
108	11036	X09E	50		V	14.37	0.00891
108	11036	X09E	82		V	0	0.000134
108	11036	X09S	10		V	0.7	0.329476
108	11036	X09S	50		V	14.37	0.000795
108	11036	X09S	82		V	0	0.000012
108	11109	B09A	10		V	0.69	0.16879
108	11109	B09A	50		V	14.62	0.000413
108	11109	B09A	82		V	0	0.000006
108	11109	B09C	10		V	0.71	0.673131
108	11109	B09C	50		V	14.62	0.001648
108	11109	B09C	82		V	0	0.000026

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	11109	B09E	10		V	0.65	15.07112
108	11109	B09E	50		V	14.62	0.036899
108	11109	B09E	82		V	0	0.000571
108	11109	B09S	10		V	0.65	1.235751
108	11109	B09S	50		V	14.62	0.003025
108	11109	B09S	82		V	0	0.000046
108	11109	X09A	20		V	0.73	0.079413
108	11109	X09A	50		V	10.32	0.00015
108	11109	X09A	82		V	0	0.000002
108	11109	X09C	20		V	0.75	0.331074
108	11109	X09C	50		V	10.32	0.000628
108	11109	X09C	82		V	0	0.000009
108	11109	X09E	10		V	0.7	7.073428
108	11109	X09E	50		V	14.62	0.017319
108	11109	X09E	82		V	0	0.000266
108	11109	X09S	10		V	0.7	0.574785
108	11109	X09S	50		V	14.62	0.001407
108	11109	X09S	82		V	0	0.000021
108	11183	A07B	10		V	0.44	0.184081
108	11183	A07B	50		V	14.49	0.00045
108	11183	A07B	82		V	0	0.000007
108	11183	B09A	10		V	0.62	0.140425
108	11183	B09A	50		V	14.49	0.000343
108	11183	B09A	82		V	0	0.000005
108	11183	B09C	10		V	0.69	0.189234
108	11183	B09C	50		V	14.49	0.000462
108	11183	B09C	82		V	0	0.000007
108	11183	B09E	10		V	0.66	32.82243
108	11183	B09E	50		V	14.49	0.080189
108	11183	B09E	82		V	0	0.001222
108	11183	B09S	10		V	0.63	1.012043
108	11183	B09S	50		V	14.49	0.002472
108	11183	B09S	82		V	0	0.000038
108	11183	BXC	10		V	0.71	0.030941
108	11183	BXC	50		V	14.49	0.000076
108	11183	BXC	82		V	0	0.000001
108	11183	BXE	10		V	0.71	0.209737
108	11183	BXE	50		V	14.49	0.000512
108	11183	BXE	82		V	0	0.000008
108	11183	BXS	10		V	0.71	0.043783
108	11183	BXS	50		V	14.49	0.000107
108	11183	BXS	82		V	0	0.000002
108	11183	N07B	10		V	0.49	0.097432
108	11183	N07B	50		V	14.49	0.000235
108	11183	N07B	82		V	0	0.000004
108	11183	X09A	10		V	0.68	0.06605
108	11183	X09A	50		V	14.49	0.000161
108	11183	X09A	82		V	0	0.000003
108	11183	X09C	20		V	0.73	0.088899
108	11183	X09C	50		V	11.02	0.000183
108	11183	X09C	82		V	0	0.000003
108	11183	X09E	10		V	0.71	15.39268
108	11183	X09E	50		V	14.49	0.037604
108	11183	X09E	82		V	0	0.000578

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	11183	X09S	10		V	0.69	0.49053
108	11183	X09S	50		V	14.49	0.001198
108	11183	X09S	82		V	0	0.000018
108	11372	A07B	10		V	0.47	0.130046
108	11372	A07B	50		V	15.17	0.000323
108	11372	A07B	82		V	0	0.000005
108	11372	B09A	10		V	0.63	0.042131
108	11372	B09A	50		V	15.17	0.000105
108	11372	B09A	82		V	0	0.000002
108	11372	B09C	10		V	0.69	0.420552
108	11372	B09C	50		V	15.17	0.001044
108	11372	B09C	82		V	0	0.000016
108	11372	B09E	10		V	0.66	19.86085
108	11372	B09E	50		V	15.17	0.04928
108	11372	B09E	82		V	0	0.000752
108	11372	B09S	10		V	0.65	0.684839
108	11372	B09S	50		V	15.17	0.001699
108	11372	B09S	82		V	0	0.000026
108	11372	N07B	10		V	0.51	0.059569
108	11372	N07B	50		V	15.17	0.000148
108	11372	N07B	82		V	0	0.000002
108	11372	X09A	20		V	0.72	0.019848
108	11372	X09A	50		V	12.05	0.000043
108	11372	X09A	82		V	0	6.60e-07
108	11372	X09C	20		V	0.74	0.177825
108	11372	X09C	50		V	12.05	0.000388
108	11372	X09C	82		V	0	0.000006
108	11372	X09E	20		V	0.71	9.393451
108	11372	X09E	50		V	12.05	0.020513
108	11372	X09E	82		V	0	0.000315
108	11372	X09S	10		V	0.7	0.30841
108	11372	X09S	50		V	15.17	0.000765
108	11372	X09S	82		V	0	0.000011
108	11449	B09A	10		V	0.66	0.196778
108	11449	B09A	50		V	14.37	0.000471
108	11449	B09A	82		V	0	0.000007
108	11449	B09C	10		V	0.69	0.094623
108	11449	B09C	50		V	14.37	0.000226
108	11449	B09C	82		V	0	0.000004
108	11449	B09E	10		V	0.66	16.23012
108	11449	B09E	50		V	14.37	0.03882
108	11449	B09E	82		V	0	0.000594
108	11449	B09S	10		V	0.69	3.353249
108	11449	B09S	50		V	14.37	0.008021
108	11449	B09S	82		V	0	0.000124
108	11449	X09A	10		V	0.71	0.085984
108	11449	X09A	50		V	14.37	0.000208
108	11449	X09A	82		V	0	0.000003
108	11449	X09C	20		V	0.74	0.039517
108	11449	X09C	50		V	11.03	0.00008
108	11449	X09C	82		V	0	0.000001
108	11449	X09E	10		V	0.71	7.611167
108	11449	X09E	50		V	14.37	0.018204
108	11449	X09E	82		V	0	0.000276

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	11449	X09S	20		V	0.73	1.613429
108	11449	X09S	50		V	11.03	0.003255
108	11449	X09S	82		V	0	0.000048
108	11540	A07B	10		V	0.5	0.151821
108	11540	A07B	50		V	11.66	0.0003
108	11540	A07B	82		V	0	0.000004
108	11540	B09A	20		V	0.72	0.140757
108	11540	B09A	50		V	11.3	0.000274
108	11540	B09A	82		V	0	0.000004
108	11540	B09C	20		V	0.72	0.042084
108	11540	B09C	50		V	11.3	0.000082
108	11540	B09C	82		V	0	0.000001
108	11540	B09E	20		V	0.73	7.588326
108	11540	B09E	50		V	11.3	0.014761
108	11540	B09E	82		V	0	0.000206
108	11540	B09S	20		V	0.72	0.596613
108	11540	B09S	50		V	11.3	0.001161
108	11540	B09S	82		V	0	0.000016
108	11540	BXE	20		V	0.72	0.980828
108	11540	BXE	50		V	11.3	0.001908
108	11540	BXE	82		V	0	0.000026
108	11540	N07B	10		V	0.53	0.075864
108	11540	N07B	50		V	11.66	0.00015
108	11540	N07B	82		V	0	0.000002
108	11540	X09A	20		V	0.75	0.072834
108	11540	X09A	50		V	11.3	0.000142
108	11540	X09A	82		V	0	0.000002
108	11540	X09C	20		V	0.76	0.02471
108	11540	X09C	50		V	11.3	0.000048
108	11540	X09C	82		V	0	6.50e-07
108	11540	X09E	20		V	0.76	3.560047
108	11540	X09E	50		V	11.3	0.006926
108	11540	X09E	82		V	0	0.000095
108	11540	X09S	20		V	0.75	0.273709
108	11540	X09S	50		V	11.3	0.000532
108	11540	X09S	82		V	0	0.000007
108	11733	B09E	10		V	0.7	0.349069
108	11733	B09E	50		V	12.82	0.000762
108	11733	B09E	82		V	0	0.000011
108	11733	B09S	10		V	0.7	0.089444
108	11733	B09S	50		V	12.82	0.000195
108	11733	B09S	82		V	0	0.000003
108	11733	X09E	20		V	0.74	0.156671
108	11733	X09E	50		V	10.36	0.000304
108	11733	X09E	82		V	0	0.000004
108	11733	X09S	20		V	0.74	0.049116
108	11733	X09S	50		V	10.36	0.000095
108	11733	X09S	82		V	0	0.000001
108	11806	A07B	10		V	0.47	0.162563
108	11806	A07B	50		V	15.03	0.000399
108	11806	A07B	82		V	0	0.000006
108	11806	B09A	20		V	0.73	0.084459
108	11806	B09A	50		V	11.94	0.000181
108	11806	B09A	82		V	0	0.000003

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	11806	B09C	20		V	0.74	0.021051
108	11806	B09C	50		V	11.94	0.000045
108	11806	B09C	82		V	0	7.00e-07
108	11806	B09E	10		V	0.67	18.21424
108	11806	B09E	50		V	15.03	0.044708
108	11806	B09E	82		V	0	0.000678
108	11806	B09S	10		V	0.63	0.267865
108	11806	B09S	50		V	15.03	0.000657
108	11806	B09S	82		V	0	0.00001
108	11806	BXE	20		V	0.72	0.326966
108	11806	BXE	50		V	11.94	0.0007
108	11806	BXE	82		V	0	0.000011
108	11806	N07B	10		V	0.51	0.070407
108	11806	N07B	50		V	15.03	0.000173
108	11806	N07B	82		V	0	0.000003
108	11806	X09A	20		V	0.76	0.033111
108	11806	X09A	50		V	11.94	0.000071
108	11806	X09A	82		V	0	0.000001
108	11806	X09C	20		V	0.78	0.009888
108	11806	X09C	50		V	11.94	0.000021
108	11806	X09C	82		V	0	3.20e-07
108	11806	X09E	20		V	0.72	8.589093
108	11806	X09E	50		V	11.94	0.018397
108	11806	X09E	82		V	0	0.000288
108	11806	X09S	10		V	0.69	0.126152
108	11806	X09S	50		V	15.03	0.00031
108	11806	X09S	82		V	0	0.000005
108	11897	B09C	20		V	0.72	3.040089
108	11897	B09C	50		V	11.69	0.00602
108	11897	B09C	82		V	0	0.000084
108	11897	B09E	10		V	0.63	20.32167
108	11897	B09E	50		V	15.47	0.053014
108	11897	B09E	82		V	0	0.000847
108	11897	B09S	10		V	0.64	1.607559
108	11897	B09S	50		V	15.47	0.004194
108	11897	B09S	82		V	0	0.000067
108	11897	X09A	10		V	0.69	0.006612
108	11897	X09A	50		V	15.47	0.000017
108	11897	X09A	82		V	0	2.80e-07
108	11897	X09C	20		V	0.76	1.428212
108	11897	X09C	50		V	11.69	0.002829
108	11897	X09C	82		V	0	0.00004
108	11897	X09E	10		V	0.69	9.55962
108	11897	X09E	50		V	15.47	0.024942
108	11897	X09E	82		V	0	0.000398
108	11897	X09S	10		V	0.69	0.749897
108	11897	X09S	50		V	15.47	0.001956
108	11897	X09S	82		V	0	0.000031
108	11979	A7B	10		V	0.49	0.081282
108	11979	A7B	50		V	15.34	0.000208
108	11979	A7B	82		V	0	0.000003
108	11979	B09C	10		V	0.68	1.849931
108	11979	B09C	50		V	15.34	0.004734
108	11979	B09C	82		V	0	0.000076

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	11979	B09E	10		V	0.63	14.22384
108	11979	B09E	50		V	15.34	0.036396
108	11979	B09E	82		V	0	0.000584
108	11979	B08S	10		V	0.66	1.921296
108	11979	B08S	50		V	15.34	0.004916
108	11979	B08S	82		V	0	0.000079
108	11979	BXC	10		V	0.69	0.370705
108	11979	BXC	50		V	15.34	0.000949
108	11979	BXC	82		V	0	0.000015
108	11979	BXE	10		V	0.66	9.597243
108	11979	BXE	50		V	15.34	0.024559
108	11979	BXE	82		V	0	0.000391
108	11979	BXS	10		V	0.67	0.459883
108	11979	BXS	50		V	15.34	0.001177
108	11979	BXS	82		V	0	0.000019
108	11979	X09C	20		V	0.73	0.893948
108	11979	X09C	50		V	11.7	0.001882
108	11979	X09C	82		V	0	0.000029
108	11979	X09E	10		V	0.69	6.676392
108	11979	X09E	50		V	15.34	0.017087
108	11979	X09E	82		V	0	0.000269
108	11979	X09S	20		V	0.71	0.890476
108	11979	X09S	50		V	11.7	0.001875
108	11979	X09S	82		V	0	0.000029
108	12127	A7B	10		V	0.45	0.146157
108	12127	A7B	50		V	14.73	0.000365
108	12127	A7B	82		V	0	0.000006
108	12127	BXA	10		V	0.67	0.496056
108	12127	BXA	50		V	14.73	0.001237
108	12127	BXA	82		V	0	0.000019
108	12127	BXC	10		V	0.69	0.339818
108	12127	BXC	50		V	14.73	0.000848
108	12127	BXC	82		V	0	0.000013
108	12127	BXE	10		V	0.67	26.16562
108	12127	BXE	50		V	14.73	0.065269
108	12127	BXE	82		V	0	0.001022
108	12127	BXS	10		V	0.66	3.327976
108	12127	BXS	50		V	14.73	0.008302
108	12127	BXS	82		V	0	0.000129
108	12201	A7B	10		V	0.49	0.130051
108	12201	A7B	50		V	13.98	0.000305
108	12201	A7B	82		V	0	0.000005
108	12201	B09C	20		V	0.95	0.01057
108	12201	B09C	50		V	11.38	0.000019
108	12201	B09C	82		V	0	2.70e-07
108	12201	BXC	20		V	0.74	0.448464
108	12201	BXC	50		V	11.38	0.000823
108	12201	BXC	82		V	0	0.000012
108	12201	BXE	10		V	0.69	15.898
108	12201	BXE	50		V	13.98	0.037237
108	12201	BXE	82		V	0	0.000566
108	12201	BXS	10		V	0.68	5.651549
108	12201	BXS	50		V	13.98	0.013238
108	12201	BXS	82		V	0	0.000201

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	12201	X09E	20		V	0.95	0.007493
108	12201	X09E	50		V	11.38	0.000014
108	12201	X09E	82		V	0	1.90e-07
108	12240	B09E	10		V	0.68	0.063436
108	12240	B09E	50		V	14.51	0.000146
108	12240	B09E	82		V	0	0.000002
108	12240	X09E	20		V	0.73	0.029835
108	12240	X09E	50		V	11.22	0.00006
108	12240	X09E	82		V	0	8.80e-07
108	12285	B09A	10		V	0.64	0.014048
108	12285	B09A	50		V	14.45	0.000034
108	12285	B09A	82		V	0	5.30e-07
108	12285	B09E	10		V	0.67	0.031693
108	12285	B09E	50		V	14.45	0.000077
108	12285	B09E	82		V	0	0.000001
108	12285	BXC	20		V	0.72	0.757435
108	12285	BXC	50		V	12.1	0.001555
108	12285	BXC	82		V	0	0.000022
108	12285	BXE	10		V	0.67	57.56088
108	12285	BXE	50		V	14.45	0.140543
108	12285	BXE	82		V	0	0.002174
108	12285	BXS	10		V	0.66	6.086724
108	12285	BXS	50		V	14.45	0.014859
108	12285	BXS	82		V	0	0.000228
108	12285	X09C	20		V	0.95	0.004962
108	12285	X09C	50		V	12.1	0.00001
108	12285	X09C	82		V	0	1.40e-07
108	12480	BXA	10		V	0.66	0.578595
108	12480	BXA	50		V	14.29	0.001403
108	12480	BXA	82		V	0	0.000022
108	12480	BXC	20		V	0.72	0.247324
108	12480	BXC	50		V	12.38	0.000509
108	12480	BXC	82		V	0	0.000007
108	12480	BXE	10		V	0.68	49.49961
108	12480	BXE	50		V	14.29	0.119985
108	12480	BXE	82		V	0	0.001848
108	12480	BXS	10		V	0.67	3.723296
108	12480	BXS	50		V	14.29	0.009025
108	12480	BXS	82		V	0	0.000138
108	12565	BXA	10		V	0.69	0.289453
108	12565	BXA	50		V	14.15	0.00069
108	12565	BXA	82		V	0	0.000011
108	12565	BXC	10		V	0.68	0.13899
108	12565	BXC	50		V	14.15	0.000331
108	12565	BXC	82		V	0	0.000005
108	12565	BXE	10		V	0.68	34.70278
108	12565	BXE	50		V	14.15	0.082699
108	12565	BXE	82		V	0	0.001266
108	12565	BXS	10		V	0.7	3.637973
108	12565	BXS	50		V	14.15	0.00867
108	12565	BXS	82		V	0	0.000133
108	12565	X09E	20		V	0.95	0.014986
108	12565	X09E	50		V	0	1.20e-07

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	12639	B09E	10		V	0.69	0.031703
108	12639	B09E	50		V	13.84	0.000074
108	12639	B09E	82		V	0	0.000001
108	12639	B7E	10		V	0.55	0.023035
108	12639	B7E	50		V	13.84	0.000054
108	12639	B7E	82		V	0	8.20e-07
108	12639	BXE	10		V	0.69	40.23534
108	12639	BXE	50		V	13.84	0.094447
108	12639	BXE	82		V	0	0.001415
108	12639	BXS	10		V	0.67	5.387353
108	12639	BXS	50		V	13.84	0.012647
108	12639	BXS	82		V	0	0.000192
108	12852	BXE	20		V	0.74	2.520529
108	12852	BXE	50		V	11.43	0.004749
108	12852	BXE	82		V	0	0.000065
108	13015	BXC	20		V	0.72	0.21617
108	13015	BXC	50		V	13.19	0.000428
108	13015	BXC	82		V	0	0.000006
108	13015	BXE	20		V	0.72	0.652963
108	13015	BXE	50		V	13.19	0.001293
108	13015	BXE	82		V	0	0.000018
108	13016	BXS	20		V	0.86	0.021922
108	13016	BXS	50		V	5.63	0.000018
108	13016	BXS	82		V	0	2.10e-07
108	13371	BXE	20		V	0.84	0.021108
108	13371	BXE	50		V	6.58	0.000002
108	13371	BXE	82		V	0	2.40e-07
108	13649	B7E	10		V	0.57	0.374046
108	13649	B7E	50		V	11.4	0.000631
108	13649	B7E	82		V	0	0.000008
108	13649	BXC	20		V	0.72	0.587287
108	13649	BXC	50		V	13.25	0.001159
108	13649	BXC	82		V	0	0.000017
108	13649	BXE	20		V	0.72	3.965021
108	13649	BXE	50		V	13.25	0.007825
108	13649	BXE	82		V	0	0.000116
108	13685	BXE	20		V	0.72	2.612148
108	13685	BXE	50		V	13.34	0.005193
108	13685	BXE	82		V	0	0.000077
108	14065	X09E	20		V	0.86	0.023268
108	14065	X09E	50		V	5.62	0.000019
108	14065	X09E	82		V	0	2.40e-07
108	14355	B09C	10		V	0.66	0.094561
108	14355	B09C	50		V	15.74	0.000253
108	14355	B09C	82		V	0	0.000004
108	14355	B09E	20		V	0.72	6.189737
108	14355	B09E	50		V	12	0.012254
108	14355	B09E	82		V	0	0.000172
108	14355	B09S	10		V	0.61	0.312363
108	14355	B09S	50		V	15.74	0.000836
108	14355	B09S	82		V	0	0.000013
108	14355	X09C	10		V	0.71	0.044429
108	14355	X09C	50		V	15.74	0.000119
108	14355	X09C	82		V	0	0.000002

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	14355	X09E	20		V	0.76	2.910359
108	14355	X09E	50		V	12	0.00576
108	14355	X09E	82		V	0	0.000082
108	14355	X09S	10		V	0.67	0.147105
108	14355	X09S	50		V	15.74	0.000394
108	14355	X09S	82		V	0	0.000006
108	14356	A09A	20		V	0.81	0.180411
108	14356	A09A	50		V	11.28	0.000283
108	14356	A09A	82		V	0	0.000003
108	14356	A9A	20		V	0.81	3.202667
108	14356	A9A	50		V	11.28	0.005031
108	14356	A9A	82		V	0	0.000059
108	14356	B09S	20		V	0.78	0.022
108	14356	B09S	50		V	11.28	0.000035
108	14356	B09S	82		V	0	4.00e-07
108	14356	BXA	20		V	0.75	8.148949
108	14356	BXA	50		V	11.28	0.012802
108	14356	BXA	82		V	0	0.00015
108	14356	BXE	20		V	0.76	1.349992
108	14356	BXE	50		V	11.28	0.002121
108	14356	BXE	82		V	0	0.000025
108	14356	BXS	20		V	0.74	0.371997
108	14356	BXS	50		V	11.28	0.000584
108	14356	BXS	82		V	0	0.000007
108	14356	N09A	20		V	0.81	0.084899
108	14356	N09A	50		V	11.28	0.000133
108	14356	N09A	82		V	0	-0.000002
108	14356	X09E	20		V	0.81	0.1035
108	14356	X09E	50		V	11.28	0.000163
108	14356	X09E	82		V	0	0.000002
108	14722	B7E	10		V	0.65	0.068792
108	14722	B7E	50		V	5.05	0.00005
108	14722	B7E	82		V	0	5.40e-07
108	15204	B09E	20		V	0.85	0.015905
108	15204	B09E	50		V	6.01	0.000014
108	15204	B09E	82		V	0	1.60e-07
108	15204	B7E	10		V	0.65	0.35096
108	15204	B7E	50		V	5.53	0.000281
108	15204	B7E	82		V	0	0.000003
108	15204	BXA	20		V	0.85	14.29324
108	15204	BXA	50		V	6.01	0.01245
108	15204	BXA	82		V	0	0.000145
108	15204	BXC	20		V	0.85	3.16191
108	15204	BXC	50		V	6.01	0.002754
108	15204	BXC	82		V	0	0.000033
108	15204	BXE	20		V	0.85	59.433
108	15204	BXE	50		V	6.01	0.051786
108	15204	BXE	82		V	0	0.00061
108	15204	BXS	20		V	0.85	20.09126
108	15204	BXS	50		V	6.01	0.017503
108	15204	BXS	82		V	0	0.000201
108	15204	X09E	20		V	0.85	0.007485
108	15204	X09E	50		V	6.01	0.000007
108	15204	X09E	82		V	0	8.00e-08

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	15240	B09E	20		V	0.86	0.015906
108	15240	B09E	50		V	5.6	0.000013
108	15240	B09E	82		V	0	1.40e-07
108	15240	B7E	10		V	0.65	0.374369
108	15240	B7E	50		V	5.71	0.000043
108	15240	B7E	82		V	0	0.000005
108	15240	BXS	20		V	0.86	0.087938
108	15240	BXS	50		V	5.6	0.000069
108	15240	BXS	82		V	0	7.60e-07
108	15240	X09A	20		V	0.87	0.006639
108	15240	X09A	50		V	5.6	0.000005
108	15240	X09A	82		V	0	6.00e-08
108	15240	X09S	20		V	0.86	0.007035
108	15240	X09S	50		V	5.6	0.000006
108	15240	X09S	82		V	0	6.00e-08
108	15316	BXA	20		V	0.85	10.03809
108	15316	BXA	50		V	6.1	0.008775
108	15316	BXA	82		V	0	0.000106
108	15316	BXC	20		V	0.86	0.185878
108	15316	BXC	50		V	6.1	0.000163
108	15316	BXC	82		V	0	0.000002
108	15316	BXE	20		V	0.85	4.958521
108	15316	BXE	50		V	6.1	0.004338
108	15316	BXE	82		V	0	0.000053
108	15316	BXS	20		V	0.86	4.330962
108	15316	BXS	50		V	6.1	0.003786
108	15316	BXS	82		V	0	0.000047
108	15316	X09E	20		V	0.85	0.007485
108	15316	X09E	50		V	6.1	0.000007
108	15316	X09E	82		V	0	8.00e-08
108	15399	BXE	20		V	0.95	0.374485
108	15444	BXA	20		V	0.88	0.103645
108	15444	BXA	50		V	4.7	0.00006
108	15444	BXA	82		V	0	6.50e-07
108	15444	BXE	20		V	0.89	0.116888
108	15444	BXE	50		V	4.7	0.000068
108	15444	BXE	82		V	0	7.50e-07
108	15444	BXS	20		V	0.89	0.08789
108	15444	BXS	50		V	4.7	0.000051
108	15444	BXS	82		V	0	5.20e-07
108	15451	B7E	10		V	0.66	1.496212
108	15451	B7E	50		V	4.97	0.000914
108	15451	B7E	82		V	0	0.00001
108	15451	BXA	20		V	0.83	6.792276
108	15451	BXA	50		V	6.79	0.007239
108	15451	BXA	82		V	0	0.000082
108	15451	BXC	20		V	0.83	0.092584
108	15451	BXC	50		V	6.79	0.000099
108	15451	BXC	82		V	0	0.000001
108	15451	BXS	20		V	0.9	0.04396
108	15451	BXS	50		V	6.79	0.000047
108	15451	BXS	82		V	0	5.40e-07
108	15455	BXA	20		V	0.86	0.372934
108	15455	BXA	50		V	4.94	0.000225
108	15455	BXA	82		V	0	0.000002

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
108	15455	BXE	20		V	0.88	0.747968
108	15455	BXE	50		V	4.94	0.000452
108	15455	BXE	82		V	0	0.000005
108	15455	BXS	20		V	0.92	0.373759
108	15455	BXS	50		V	4.94	0.000226
108	15455	BXS	82		V	0	0.000003
108	15457	B7E	10		V	0.69	0.046788
108	15457	B7E	50		V	1.69	0.000012
108	15457	B7E	82		V	0	1.20e-07
108	15457	BXA	20		V	0.87	0.290174
108	15457	BXA	50		V	4.35	0.000198
108	15457	BXA	82		V	0	0.000002
108	15458	B7E	10		V	0.67	0.374152
108	15458	B7E	50		V	3.44	0.000181
108	15458	B7E	82		V	0	0.000002
108	15458	BXA	20		V	0.88	21.26727
108	15458	BXA	50		V	4.26	0.011707
108	15458	BXA	82		V	0	0.000123
108	15458	BXC	20		V	0.9	2.596366
108	15458	BXC	50		V	4.26	0.00143
108	15458	BXC	82		V	0	0.000015
108	15458	BXE	20		V	0.89	67.3323
108	15458	BXE	50		V	4.26	0.037094
108	15458	BXE	82		V	0	0.000374
108	15458	BXR	20		V	0.91	0.373308
108	15458	BXR	50		V	4.26	0.000206
108	15458	BXR	82		V	0	0.000002
108	15458	BXS	20		V	0.89	40.34478
108	15458	BXS	50		V	4.26	0.022216
108	15458	BXS	82		V	0	0.000239
108	15460	B7E	10		V	0.69	2.947289
108	15460	B7E	50		V	2.83	0.000971
108	15460	B7E	82		V	0	0.000009
108	15460	BXA	20		V	0.88	14.92423
108	15460	BXA	50		V	4.29	0.00972
108	15460	BXA	82		V	0	0.000108
108	15461	BXE	20		V	0.93	12.72751
108	15461	BXE	50		V	1.55	0.00247
108	15461	BXE	82		V	0	0.000022
107	09621	K5A	20		I	1.11	0.105496
107	09621	K5A	50		I	6.04	0.000103
107	09964	A2M	20		I	1.01	0.017123
107	09964	A2M	50		I	6.18	0.000017
107	09964	A2M	82		I	0	1.90e-07
107	09964	A2T	20		I	1.01	0.278637
107	09964	A2T	50		I	6.18	0.000278
107	09964	A2T	82		I	0	0.000003
107	09964	BXA	20		V	0.85	0.041658
107	09964	BXA	50		V	6.26	0.000038
107	09964	BXA	82		V	0	3.80e-07
107	09964	N09E	20		I	0.81	0.013771
107	09964	N09E	50		I	6.18	0.000014
107	09964	N09E	82		I	0	1.60e-07

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
107	09964	N09F	20		I	0.81	0.003879
107	09964	N09F	50		I	6.18	0.000004
107	09964	N09F	82		I	0	4.00e-08
107	09964	N09M	20		I	0.81	0.00548
107	09964	N09M	50		I	6.18	0.000005
107	09964	N09M	82		I	0	6.00e-08
107	09964	X09A	20		V	0.85	0.03334
107	09964	X09A	50		V	6.26	0.00003
107	09964	X09A	82		V	0	3.00e-07
107	09993	A2F	20		I	0.91	0.246184
107	09993	A2F	50		I	10.29	0.000405
107	09993	A2F	82		I	0	0.000005
107	09993	A2M	20		I	0.91	4.636134
107	09993	A2M	50		I	10.29	0.007627
107	09993	A2M	82		I	0	0.000104
107	09993	A2T	20		I	0.91	3.566373
107	09993	A2T	50		I	10.29	0.005865
107	09993	A2T	82		I	0	0.000078
107	10001	B09C	10		V	0.71	0.03156
107	10001	B09C	50		V	13.59	0.000064
107	10001	B09C	82		V	0	9.00e-07
107	10001	B09E	10		V	0.71	0.03169
107	10001	B09E	50		V	13.59	0.000064
107	10001	B09E	82		V	0	9.00e-07
107	10001	B09R	20		V	0.88	0.013353
107	10001	B09R	50		V	5.61	0.000011
107	10001	B09R	82		V	0	1.30e-07
107	10001	B09S	10		V	0.71	0.029772
107	10001	B09S	50		V	13.59	0.00006
107	10001	B09S	82		V	0	8.60e-07
107	10001	BXE	10		V	0.71	0.72243
107	10001	BXE	50		V	13.59	0.001455
107	10001	BXE	82		V	0	0.00002
107	10001	BXR	20		V	0.88	0.019638
107	10001	BXR	50		V	5.61	0.000016
107	10001	BXR	82		V	0	1.90e-07
107	10001	BXS	20		V	0.86	0.021922
107	10001	BXS	50		V	5.61	0.000018
107	10001	BXS	82		V	0	2.10e-07
107	10001	X09E	10		V	0.7	0.007459
107	10001	X09E	50		V	13.59	0.000015
107	10001	X09E	82		V	0	2.10e-07
107	10001	X09R	20		V	0.88	0.006284
107	10001	X09R	50		V	5.61	0.000005
107	10001	X09R	82		V	0	6.00e-08
107	10001	X09S	10		V	0.71	0.01401
107	10001	X09S	50		V	13.59	0.000028
107	10001	X09S	82		V	0	4.00e-07
107	10201	A2E	20		I	0.94	0.24942
107	10201	A2E	50		I	9.22	0.000368
107	10201	A2E	82		I	0	0.000005
107	10201	A2F	20		I	0.94	0.489995
107	10201	A2F	50		I	9.22	0.000723
107	10201	A2F	82		I	0	0.000009

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
107	10201	A2M	20		I	0.94	12.12241
107	10201	A2M	50		I	9.22	0.017897
107	10201	A2M	82		I	0	0.000228
107	10201	A2T	20		I	0.94	1.234542
107	10201	A2T	50		I	9.22	0.001822
107	10201	A2T	82		I	0	0.000023
107	10259	A2F	20		I	0.91	0.734838
107	10259	A2F	50		I	10.35	0.001217
107	10259	A2F	82		I	0	0.000016
107	10259	A2M	20		I	0.91	2.225036
107	10259	A2M	50		I	10.35	0.003683
107	10259	A2M	82		I	0	0.00005
107	10294	A2M	20		I	0.93	19.05676
107	10294	A2M	50		I	9.8	0.02994
107	10294	A2M	82		I	0	0.000393
107	10294	A2T	20		I	0.93	0.06176
107	10294	A2T	50		I	9.8	0.000097
107	10294	A2T	82		I	0	0.000001
107	10350	A12M	20		I	0.9	0.01099
107	10350	A12M	50		I	10.65	0.000019
107	10350	A12M	82		I	0	2.50e-07
107	10350	A2F	20		I	0.9	1.306433
107	10350	A2F	50		I	10.65	0.002225
107	10350	A2F	82		I	0	0.000031
107	10350	A2M	20		I	0.9	13.56264
107	10350	A2M	50		I	10.65	0.023102
107	10350	A2M	82		I	0	0.000313
107	10350	A2T	20		I	0.9	0.895177
107	10350	A2T	50		I	10.65	0.001524
107	10350	A2T	82		I	0	0.000021
107	10350	N09M	20		I	0.9	0.005494
107	10350	N09M	50		I	10.65	0.000009
107	10350	N09M	82		I	0	1.30e-07
107	10385	A12M	20		I	0.91	0.011208
107	10385	A12M	50		I	10.42	0.000019
107	10385	A12M	82		I	0	2.50e-07
107	10385	A2M	20		I	0.91	7.481947
107	10385	A2M	50		I	10.42	0.012467
107	10385	A2M	82		I	0	0.000168
107	10385	A2T	20		I	0.91	0.740783
107	10385	A2T	50		I	10.42	0.001235
107	10385	A2T	82		I	0	0.000017
107	10456	A2M	20		I	0.89	15.58376
107	10456	A2M	50		I	11.37	0.028304
107	10456	A2M	82		I	0	0.000407
107	10456	A2T	20		I	0.89	0.956295
107	10456	A2T	50		I	11.37	0.001737
107	10456	A2T	82		I	0	0.000025
107	10602	A12M	20		I	0.89	0.011218
107	10602	A12M	50		I	11.11	0.00002
107	10602	A12M	82		I	0	2.80e-07
107	10602	A2M	20		I	0.89	13.84097
107	10602	A2M	50		I	11.11	0.024583
107	10602	A2M	82		I	0	0.000344

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
107	10602	A2T	20		I	0.89	0.64864
107	10602	A2T	50		I	11.11	0.001152
107	10602	A2T	82		I	0	0.000016
107	10679	A2F	20		I	0.87	0.163593
107	10679	A2F	50		I	11.76	0.000308
107	10679	A2F	82		I	0	0.000004
107	10679	A2M	20		I	0.87	7.750192
107	10679	A2M	50		I	11.76	0.01457
107	10679	A2M	82		I	0	0.000212
107	10679	A2T	20		I	0.87	3.551796
107	10679	A2T	50		I	11.76	0.006677
107	10679	A2T	82		I	0	0.000097
107	10742	A12F	20		I	0.85	0.007946
107	10742	A12F	50		I	12.77	0.000016
107	10742	A12F	82		I	0	2.50e-07
107	10742	A2F	20		I	0.85	0.21033
107	10742	A2F	50		I	12.77	0.000428
107	10742	A2F	82		I	0	0.000007
107	10742	A2M	20		I	0.85	3.642564
107	10742	A2M	50		I	12.77	0.007417
107	10742	A2M	82		I	0	0.000115
107	10742	A2T	20		I	0.85	2.777783
107	10742	A2T	50		I	12.77	0.005657
107	10742	A2T	82		I	0	0.000086
107	10742	N09M	20		I	0.85	0.005274
107	10742	N09M	50		I	12.77	0.000011
107	10742	N09M	82		I	0	1.70e-07
107	10764	A2M	20		I	1.15	0.016656
107	10764	BXE	20		V	0.95	0.304268
107	10770	A12T	20		I	0.84	0.010337
107	10770	A12T	50		I	13.16	0.000022
107	10770	A12T	82		I	0	3.40e-07
107	10770	A2M	20		I	0.84	6.146783
107	10770	A2M	50		I	13.16	0.012895
107	10770	A2M	82		I	0	0.000201
107	10770	A2T	20		I	0.84	2.005478
107	10770	A2T	50		I	13.16	0.004206
107	10770	A2T	82		I	0	0.000066
107	10770	N09T	20		I	0.86	0.005095
107	10770	N09T	50		I	13.16	0.000011
107	10770	N09T	82		I	0	1.70e-07
107	10832	A2M	20		I	0.84	13.05217
107	10832	A2M	50		I	12.92	0.02688
107	10832	A2M	82		I	0	0.00042
107	10832	A2T	20		I	0.84	3.6888
107	10832	A2T	50		I	12.92	0.007598
107	10832	A2T	82		I	0	0.00012
107	10982	A12T	20		I	0.82	0.020936
107	10982	A12T	50		I	13.73	0.000046
107	10982	A12T	82		I	0	7.40e-07
107	10982	A2F	20		I	0.82	0.023339
107	10982	A2F	50		I	13.73	0.000051
107	10982	A2F	82		I	0	8.20e-07

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
107	10982	A2M	20		I	0.82	16.45954
107	10982	A2M	50		I	13.73	0.036006
107	10982	A2M	82		I	0	0.000581
107	10982	A2T	20		I	0.82	0.692735
107	10982	A2T	50		I	13.73	0.001515
107	10982	A2T	82		I	0	0.000024
107	10982	N09M	20		I	0.82	0.010524
107	10982	N09M	50		I	13.73	0.000023
107	10982	N09M	82		I	0	3.80e-07
107	11036	A2M	20		I	0.82	7.596712
107	11036	A2M	50		I	13.6	0.01647
107	11036	A2M	82		I	0	0.000263
107	11109	A2F	20		I	0.82	0.467032
107	11109	A2F	50		I	13.7	0.00102
107	11109	A2F	82		I	0	0.000016
107	11109	A2M	20		I	0.82	14.60046
107	11109	A2M	50		I	13.7	0.031879
107	11109	A2M	82		I	0	0.000524
107	11109	N09M	20		I	0.82	0.005263
107	11109	N09M	50		I	13.7	0.000011
107	11109	N09M	82		I	0	1.90e-07
107	11183	A12M	20		I	0.83	0.022046
107	11183	A12M	50		I	13.31	0.000047
107	11183	A12M	82		I	0	7.60e-07
107	11183	A2F	20		I	0.83	0.677131
107	11183	A2F	50		I	13.31	0.001439
107	11183	A2F	82		I	0	0.000023
107	11183	A2M	20		I	0.83	9.341152
107	11183	A2M	50		I	13.31	0.019857
107	11183	A2M	82		I	0	0.000318
107	11183	A2T	20		I	0.83	0.246348
107	11183	A2T	50		I	13.31	0.000524
107	11183	A2T	82		I	0	0.000008
107	11183	N09M	20		I	0.83	0.010858
107	11183	N09M	50		I	13.31	0.000023
107	11183	N09M	82		I	0	3.60e-07
107	11372	A2F	20		I	0.82	0.688631
107	11372	A2F	50		I	13.77	0.001513
107	11372	A2F	82		I	0	0.000025
107	11372	A2M	20		I	0.82	18.52989
107	11372	A2M	50		I	13.77	0.040707
107	11372	A2M	82		I	0	0.000665
107	11372	A2T	20		I	0.82	0.708117
107	11372	A2T	50		I	13.77	0.001556
107	11372	A2T	82		I	0	0.000026
107	11449	A2F	20		I	0.83	0.105028
107	11449	A2F	50		I	13.45	0.000226
107	11449	A2F	82		I	0	0.000004
107	11449	A2M	20		I	0.83	18.54239
107	11449	A2M	50		I	13.45	0.039825
107	11449	A2M	82		I	0	0.000641
107	11449	A2T	20		I	0.83	1.295873
107	11449	A2T	50		I	13.45	0.002783
107	11449	A2T	82		I	0	0.000045

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
107	11540	A2M	20		I	0.83	12.46058
107	11540	A2M	50		I	13.26	0.026407
107	11540	A2M	82		I	0	0.000423
107	11806	A2F	20		I	0.82	0.209843
107	11806	A2F	50		I	13.86	0.000465
107	11806	A2F	82		I	0	0.000008
107	11806	A2M	20		I	0.82	19.19476
107	11806	A2M	50		I	13.86	0.042523
107	11806	A2M	82		I	0	0.000699
107	11806	A2T	20		I	0.82	0.539924
107	11806	A2T	50		I	13.86	0.001196
107	11806	A2T	82		I	0	0.00002
107	11897	A12F	20		I	0.8	0.023429
107	11897	A12F	50		I	14.4	0.000054
107	11897	A12F	82		I	0	9.30e-07
107	11897	A2F	20		I	0.8	1.981819
107	11897	A2F	50		I	14.4	0.004559
107	11897	A2F	82		I	0	0.000078
107	11897	A2M	20		I	0.8	17.6565
107	11897	A2M	50		I	14.4	0.040607
107	11897	A2M	82		I	0	0.000697
107	11897	N09F	20		I	0.8	0.003847
107	11897	N09F	50		I	14.4	0.000009
107	11897	N09F	82		I	0	1.50e-07
107	11979	A12F	20		I	0.8	0.007963
107	11979	A12F	50		I	14.27	0.000018
107	11979	A12F	82		I	0	3.10e-07
107	11979	A12M	20		I	0.8	0.011208
107	11979	A12M	50		I	14.27	0.000026
107	11979	A12M	82		I	0	4.30e-07
107	11979	A12T	20		I	0.8	0.010483
107	11979	A12T	50		I	14.27	0.000024
107	11979	A12T	82		I	0	4.10e-07
107	11979	A2F	20		I	0.8	1.947591
107	11979	A2F	50		I	14.27	0.004442
107	11979	A2F	82		I	0	0.000075
107	11979	A2M	20		I	0.8	14.46003
107	11979	A2M	50		I	14.27	0.032986
107	11979	A2M	82		I	0	0.000553
107	11979	A2T	20		I	0.8	8.864286
107	11979	A2T	50		I	14.27	0.020223
107	11979	A2T	82		I	0	0.000345
107	11979	N09M	20		I	0.8	0.010706
107	11979	N09M	50		I	14.27	0.000024
107	11979	N09M	82		I	0	4.20e-07
107	12127	A2F	20		I	0.81	0.980225
107	12127	A2F	50		I	14.22	0.00223
107	12127	A2F	82		I	0	0.000037
107	12127	A2M	20		I	0.81	13.29014
107	12127	A2M	50		I	14.22	0.030238
107	12127	A2M	82		I	0	0.000508
107	12127	A2T	20		I	0.81	1.47972
107	12127	A2T	50		I	14.22	0.003366
107	12127	A2T	82		I	0	0.000057

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
107	12201	A2F	20		I	0.83	0.746713
107	12201	A2F	50		I	13.24	0.001585
107	12201	A2F	82		I	0	0.000026
107	12201	A2M	20		I	0.83	24.3225
107	12201	A2M	50		I	13.24	0.051616
107	12201	A2M	82		I	0	0.000827
107	12285	A2F	20		I	0.84	0.245655
107	12285	A2F	50		I	13.17	0.000519
107	12285	A2F	82		I	0	0.000008
107	12285	A2M	20		I	0.84	19.5256
107	12285	A2M	50		I	13.17	0.04124
107	12285	A2M	82		I	0	0.000662
107	12480	A12M	20		I	0.82	0.022123
107	12480	A12M	50		I	13.78	0.000049
107	12480	A12M	82		I	0	8.00e-07
107	12480	A2M	20		I	0.82	19.38647
107	12480	A2M	50		I	13.78	0.042851
107	12480	A2M	82		I	0	0.000704
107	12565	A12M	20		I	0.82	0.011067
107	12565	A12M	50		I	13.7	0.000024
107	12565	A12M	82		I	0	4.00e-07
107	12565	A2M	20		I	0.82	25.18379
107	12565	A2M	50		I	13.7	0.055373
107	12565	A2M	82		I	0	0.000915
107	12639	A12M	20		I	0.84	0.011071
107	12639	A12M	50		I	13.06	0.000023
107	12639	A12M	82		I	0	3.70e-07
107	12639	A2M	20		I	0.84	23.55778
107	12639	A2M	50		I	13.06	0.049425
107	12639	A2M	82		I	0	0.000799
107	12639	N09M	20		I	0.84	0.005453
107	12639	N09M	50		I	13.06	0.000011
107	12639	N09M	82		I	0	1.80e-07
107	12823	A2M	20		I	0.83	7.198119
107	12823	A2M	50		I	13.17	0.015247
107	12823	A2M	82		I	0	0.000244
107	12823	A2T	20		I	0.83	0.635473
107	12823	A2T	50		I	13.17	0.001346
107	12823	A2T	82		I	0	0.000022
107	12852	A2M	20		I	0.84	10.83763
107	12852	A2M	50		I	13.01	0.022683
107	12852	A2M	82		I	0	0.00036
107	12852	A2T	20		I	0.84	2.869037
107	12852	A2T	50		I	13.01	0.006
107	12852	A2T	82		I	0	0.000096
107	12852	B09E	20		V	0.72	0.047572
107	12852	B09E	50		V	13.42	0.000095
107	12852	B09E	82		V	0	0.000001
107	12852	BXE	20		V	0.72	66.47092
107	12852	BXE	50		V	13.42	0.132468
107	12852	BXE	82		V	0	0.001881
107	12852	BXS	20		V	0.72	0.372719
107	12852	BXS	50		V	13.42	0.000743
107	12852	BXS	82		V	0	0.000011

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
107	12852	X09E	20		V	0.72	0.022379
107	12852	X09E	50		V	13.42	0.000045
107	12852	X09E	82		V	0	6.30e-07
107	12942	A2F	20		I	0.84	0.739118
107	12942	A2F	50		I	13.03	0.00155
107	12942	A2F	82		I	0	0.000025
107	12942	A2M	20		I	0.84	17.40925
107	12942	A2M	50		I	13.03	0.036494
107	12942	A2M	82		I	0	0.000589
107	12942	A2T	20		I	0.84	2.246109
107	12942	A2T	50		I	13.03	0.00471
107	12942	A2T	82		I	0	0.000075
107	12942	B7E	10		V	0.56	0.373946
107	12942	B7E	50		V	12.1	0.000671
107	12942	B7E	82		V	0	0.000009
107	12942	BXA	20		V	0.73	0.369594
107	12942	BXA	50		V	12.82	0.000702
107	12942	BXA	82		V	0	0.00001
107	12942	BXC	20		V	0.73	0.556128
107	12942	BXC	50		V	12.82	0.001057
107	12942	BXC	82		V	0	0.000015
107	12942	BXE	20		V	0.73	50.04592
107	12942	BXE	50		V	12.82	0.095109
107	12942	BXE	82		V	0	0.00133
107	12942	BXS	20		V	0.73	1.468644
107	12942	BXS	50		V	12.82	0.002791
107	12942	BXS	82		V	0	0.00004
107	13015	B09C	20		V	0.72	0.010503
107	13015	B09C	50		V	13.19	0.000021
107	13015	B09C	82		V	0	2.90e-07
107	13015	B7E	10		V	0.56	0.373937
107	13015	B7E	50		V	12.13	0.000673
107	13015	B7E	82		V	0	0.000009
107	13015	BXC	20		V	0.72	4.585912
107	13015	BXC	50		V	13.19	0.009088
107	13015	BXC	82		V	0	0.000128
107	13015	BXE	20		V	0.72	36.5893
107	13015	BXE	50		V	13.19	0.072519
107	13015	BXE	82		V	0	0.00102
107	13015	BXS	20		V	0.72	2.609064
107	13015	BXS	50		V	13.19	0.005172
107	13015	BXS	82		V	0	0.000074
107	13015	X09C	20		V	0.72	0.004942
107	13015	X09C	50		V	13.19	0.00001
107	13015	X09C	82		V	0	1.40e-07
107	13016	A2F	20		I	0.83	0.762811
107	13016	A2F	50		I	13.18	0.001639
107	13016	A2F	82		I	0	0.000026
107	13016	A2M	20		I	0.83	19.57218
107	13016	A2M	50		I	13.18	0.042032
107	13016	A2M	82		I	0	0.000674
107	13016	A2T	20		I	0.83	0.247751
107	13016	A2T	50		I	13.18	0.000532
107	13016	A2T	82		I	0	0.000008

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
107	13016	N09M	20		I	0.83	0.005235
107	13016	N09M	50		I	13.18	0.000011
107	13016	N09M	82		I	0	1.80e-07
107	13016	N09T	20		I	0.83	0.015329
107	13016	N09T	50		I	13.18	0.000033
107	13016	N09T	82		I	0	5.40e-07
107	13017	BXC	20		V	0.76	1.129932
107	13017	BXC	50		V	11.39	0.001898
107	13017	BXC	82		V	0	0.000025
107	13017	BXE	20		V	0.76	25.38336
107	13017	BXE	50		V	11.39	0.042626
107	13017	BXE	82		V	0	0.000565
107	13017	BXS	20		V	0.76	0.373695
107	13017	BXS	50		V	11.39	0.000627
107	13017	BXS	82		V	0	0.000008
107	13371	A12F	20		I	0.84	0.015875
107	13371	A12F	50		I	12.95	0.000033
107	13371	A12F	82		I	0	5.40e-07
107	13371	A2F	20		I	0.84	0.245132
107	13371	A2F	50		I	12.95	0.000511
107	13371	A2F	82		I	0	0.000008
107	13371	A2M	20		I	0.84	4.713549
107	13371	A2M	50		I	12.95	0.009835
107	13371	A2M	82		I	0	0.000162
107	13371	N09T	20		I	0.84	0.010219
107	13371	N09T	50		I	12.95	0.000021
107	13371	N09T	82		I	0	3.60e-07
107	13372	B09S	20		V	0.72	0.014931
107	13372	B09S	50		V	13.02	0.000029
107	13372	B09S	82		V	0	4.20e-07
107	13372	BXC	20		V	0.72	3.170657
107	13372	BXC	50		V	13.02	0.006132
107	13372	BXC	82		V	0	0.00009
107	13372	BXE	20		V	0.72	11.76885
107	13372	BXE	50		V	13.02	0.022756
107	13372	BXE	82		V	0	0.000333
107	13372	BXS	20		V	0.72	4.324834
107	13372	BXS	50		V	13.02	0.008363
107	13372	BXS	82		V	0	0.000122
107	13372	X09C	20		V	0.72	0.004951
107	13372	X09C	50		V	13.02	0.00001
107	13372	X09C	82		V	0	1.40e-07
107	13524	BXA	20		V	0.72	4.098228
107	13524	BXA	50		V	13.13	0.008005
107	13524	BXA	82		V	0	0.000117
107	13524	BXC	20		V	0.72	2.923896
107	13524	BXC	50		V	13.13	0.005712
107	13524	BXC	82		V	0	0.000083
107	13524	BXE	20		V	0.72	25.92515
107	13524	BXE	50		V	13.13	0.050638
107	13524	BXE	82		V	0	0.000744
107	13524	BXS	20		V	0.72	4.630184
107	13524	BXS	50		V	13.13	0.009046
107	13524	BXS	82		V	0	0.000133

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
107	13525	A2F	20		I	0.82	1.217087
107	13525	A2F	50		I	13.52	0.002653
107	13525	A2F	82		I	0	0.000045
107	13525	A2M	20		I	0.82	16.33464
107	13525	A2M	50		I	13.52	0.035608
107	13525	A2M	82		I	0	0.000613
107	13648	A2M	20		I	0.88	0.032513
107	13648	A2M	50		I	11.67	0.000061
107	13648	A2M	82		I	0	9.40e-07
107	13650	A2F	20		I	0.83	1.435141
107	13650	A2F	50		I	13.41	0.003108
107	13650	A2F	82		I	0	0.000053
107	13650	A2M	20		I	0.83	12.39377
107	13650	A2M	50		I	13.41	0.026841
107	13650	A2M	82		I	0	0.000451
107	13650	A2T	20		I	0.83	2.7182
107	13650	A2T	50		I	13.41	0.005885
107	13650	A2T	82		I	0	0.000099
107	13686	A2F	20		I	0.86	0.023339
107	13686	A2F	50		I	12.41	0.000047
107	13686	A2F	82		I	0	7.60e-07
107	13686	A2M	20		I	0.86	17.5855
107	13686	A2M	50		I	12.41	0.035252
107	13686	A2M	82		I	0	0.000564
107	13686	A2T	20		I	0.86	0.247249
107	13686	A2T	50		I	12.41	0.000496
107	13686	A2T	82		I	0	0.000008
107	13724	A2T	20		I	1.02	0.015508
107	13724	A2T	50		I	5.99	0.000015
107	13724	A2T	82		I	0	1.80e-07
107	13858	B7E	10		V	0.56	0.373955
107	13858	B7E	50		V	12.01	0.000668
107	13858	B7E	82		V	0	0.000009
107	13858	BXC	20		V	0.74	3.122185
107	13858	BXC	50		V	12.04	0.005575
107	13858	BXC	82		V	0	0.000077
107	13858	BXE	20		V	0.74	48.26557
107	13858	BXE	50		V	12.04	0.086215
107	13858	BXE	82		V	0	0.00122
107	13858	BXS	20		V	0.74	1.009438
107	13858	BXS	50		V	12.04	0.001803
107	13858	BXS	82		V	0	0.000026
107	13858	X09C	20		V	0.74	0.004946
107	13858	X09C	50		V	12.04	0.000009
107	13858	X09C	82		V	0	1.20e-07
107	13859	A12F	20		I	0.86	0.007911
107	13859	A12F	50		I	12.28	0.000016
107	13859	A12F	82		I	0	2.50e-07
107	13859	A2F	20		I	0.86	0.011634
107	13859	A2F	50		I	12.28	0.000023
107	13859	A2F	82		I	0	3.70e-07
107	13859	A2M	20		I	0.86	25.82511
107	13859	A2M	50		I	12.28	0.051268
107	13859	A2M	82		I	0	0.000829

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
107	13859	A2T	20		I	0.86	3.743876
107	13859	A2T	50		I	12.28	0.007432
107	13859	A2T	82		I	0	0.000119
107	13859	N09M	20		I	0.86	0.005235
107	13859	N09M	50		I	12.28	0.00001
107	13859	N09M	82		I	0	1.70e-07
107	14055	BXE	20		V	0.91	0.327834
107	14055	BXE	50		V	2.39	0.000111
107	14055	BXE	82		V	0	5.60e-07
107	14065	B09E	20		V	0.86	0.015889
107	14065	B09E	50		V	5.67	0.000013
107	14065	B09E	82		V	0	1.50e-07
107	14065	BXE	20		V	0.86	0.071892
107	14065	BXE	50		V	5.67	0.000059
107	14065	BXE	82		V	0	6.60e-07
107	14065	X09C	20		V	0.86	0.004456
107	14065	X09C	50		V	5.67	0.000004
107	14065	X09C	82		V	0	4.00e-08
107	14065	X09E	20		V	0.86	0.024224
107	14065	X09E	50		V	5.67	0.00002
107	14065	X09E	82		V	0	2.40e-07
107	14065	X09S	20		V	0.86	0.007015
107	14065	X09S	50		V	5.67	0.000006
107	14065	X09S	82		V	0	7.00e-08
107	14448	B09S	20		V	0.82	0.01226
107	14448	B09S	50		V	5.99	0.000012
107	14448	B09S	82		V	0	1.50e-07
107	14883	BXE	20		V	0.86	0.022244
107	14883	BXE	50		V	5.66	0.000018
107	14883	BXE	82		V	0	2.10e-07
107	15204	B09E	20		V	0.85	0.015905
107	15204	B09E	50		V	5.92	0.000014
107	15204	B09E	82		V	0	1.50e-07
107	15244	A12F	20		I	0.95	0.007942
107	15244	A12F	50		I	5.44	0.000007
107	15244	A12F	82		I	0	9.00e-08
107	15244	A12M	20		I	1.03	0.011251
107	15244	A12M	50		I	5.44	0.000009
107	15244	A12M	82		I	0	1.20e-07
107	15244	A12T	20		I	1.03	0.042134
107	15244	A12T	50		I	5.44	0.000035
107	15244	A12T	82		I	0	4.80e-07
107	15244	A2F	20		I	0.95	0.02336
107	15244	A2F	50		I	5.44	0.000019
107	15244	A2F	82		I	0	2.60e-07
107	15244	A2M	20		I	1.03	0.132363
107	15244	A2M	50		I	5.44	0.000011
107	15244	A2M	82		I	0	0.000001
107	15244	A2T	20		I	1.03	0.123923
107	15244	A2T	50		I	5.44	0.000103
107	15244	A2T	82		I	0	0.000001
107	15244	N09F	20		I	0.95	0.007475
107	15244	N09F	50		I	5.44	0.000006
107	15244	N09F	82		I	0	8.00e-08

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
107	15244	N09T	20		I	1.03	0.004957
107	15244	N09T	50		I	5.44	0.000004
107	15244	N09T	82		I	0	5.00e-08
107	15244	X09S	10		V	0.71	0.007005
107	15244	X09S	50		V	13.58	0.000014
107	15244	X09S	82		V	0	2.00e-07
107	15347	A12F	20		I	0.95	0.007941
107	15347	A12F	50		I	6.16	0.000008
107	15347	A12F	82		I	0	1.00e-07
107	15347	A2F	20		I	0.95	0.011679
107	15347	A2F	50		I	6.16	0.000011
107	15347	A2F	82		I	0	1.50e-07
107	15347	A2M	20		I	1.02	0.083087
107	15347	A2M	50		I	6.16	0.00008
107	15347	A2M	82		I	0	0.000001
107	15347	A2T	20		I	1.02	0.031107
107	15347	A2T	50		I	6.16	0.00003
107	15347	A2T	82		I	0	4.20e-07
107	15347	N09M	20		I	1.02	0.010635
107	15347	N09M	50		I	6.16	0.00001
107	15347	N09M	82		I	0	1.40e-07
107	15445	A2M	20		I	1.05	1.296922
107	15445	A2M	50		I	5.23	0.000984
107	15445	A2M	82		I	0	0.000012
107	15445	A2T	20		I	1.03	0.668979
107	15445	A2T	50		I	5.23	0.000508
107	15445	A2T	82		I	0	0.000006
107	15448	A12F	20		I	1.02	0.007997
107	15448	A12F	50		I	4.54	0.000005
107	15448	A12F	82		I	0	7.00e-08
107	15448	A2T	20		I	1.03	0.015801
107	15448	A2T	50		I	4.54	0.000011
107	15448	A2T	82		I	0	1.30e-07
107	15448	N09F	20		I	1.02	0.003763
107	15448	N09F	50		I	4.54	0.000003
107	15448	N09F	82		I	0	3.00e-08
107	15448	N09M	20		I	1.07	0.005323
107	15448	N09M	50		I	4.54	0.000004
107	15448	N09M	82		I	0	4.00e-08
107	15451	A2M	20		I	1.04	0.465448
107	15451	A2M	50		I	5.31	0.000387
107	15451	A2M	82		I	0	0.000005
107	15457	A2M	20		I	1.09	0.033277
107	15457	A2M	50		I	2.85	0.000015
107	15457	A2M	82		I	0	1.80e-07
107	15459	A2M	20		I	1.07	25.26509
107	15459	A2M	50		I	4.06	0.01519
107	15459	A2M	82		I	0	0.000182
107	15459	A2T	20		I	1.07	8.655769
107	15459	A2T	50		I	4.06	0.00521
107	15459	A2T	82		I	0	0.000067
107	15460	A2M	20		I	1.08	2.098243
107	15460	A2M	50		I	3.32	0.001042
107	15460	A2M	82		I	0	0.000011

Location Code	Key Number	Fuel Model	Material Type	sub ID Code	Fuel Type	Isotopic Breakdown	Discharge Mass
107	15555	A12F	20		I	1.02	0.007997
107	15555	A12F	50		I	4	0.000005
107	15555	A12F	82		I	0	5.00e-08
107	15555	A12M	20		I	1.08	0.011313
107	15555	A12M	50		I	4	0.000006
107	15555	A12M	82		I	0	8.00e-08
107	21540	B09C	10		V	0.68	0.031533
107	21540	B09C	50		V	16.49	0.000089
107	21540	B09C	82		V	0	0.000001
107	21540	B09E	10		V	0.6	6.853472
107	21540	B09E	50		V	16.49	0.019269
107	21540	B09E	82		V	0	0.00032
107	21540	B09S	10		V	0.59	0.981302
107	21540	B09S	50		V	16.49	0.002759
107	21540	B09S	82		V	0	0.000046
107	21540	X09C	20		V	0.73	0.009877
107	21540	X09C	50		V	13.73	0.000025
107	21540	X09C	82		V	0	4.00e-07
107	21540	X09E	10		V	0.67	3.231976
107	21540	X09E	50		V	16.49	0.009088
107	21540	X09E	82		V	0	0.000152
107	21540	X09S	10		V	0.66	0.455183
107	21540	X09S	50		V	16.49	0.00128
107	21540	X09S	82		V	0	0.000021
107	22565	BXE	10		V	0.61	6.655418
107	22565	BXE	50		V	16.72	0.019233
107	22565	BXE	82		V	0	0.000323
107	22565	BXS	10		V	0.64	0.175075
107	22565	BXS	50		V	16.72	0.000506
107	22565	BXS	82		V	0	0.000008
107	22852	BXE	10		V	0.62	4.188897
107	22852	BXE	50		V	16.62	0.0121
107	22852	BXE	82		V	0	0.000202
108	14356	B09A	20		V	0.75	0.014029
108	14356	B09A	50		V	11.28	0.000022
108	14356	B09A	82		V	0	2.60e-07
108	14356	B09C	20		V	0.75	0.010454
108	14356	B09C	50		V	11.28	0.000016
108	14356	B09C	82		V	0	1.90e-07

APPENDIX B

RADNUC INPUT AND OUTPUT FILES

Selected information was taken from the accountability database (Appendix A) to develop the following data files which were used as input files in the RADNUC program. Five separate data files were used; one for the combined basins, one for K-East alone, one for K-West alone, one for the shielding basis, and one for the safety/regulatory assessment basis. Eleven keys in the K East Basin (2.38 MTU) and two keys in the K West Basin (0.32 MTU) in the accountability database were un-irradiated fuel, these keys are not included in the RADNUC input files. The uranium from the keys not included in the RADNUC input is added to the uranium mass, activity, and heat generation tables within the document.

The first four lines of the data files are used to identify the information, and are ignored by RADNUC. RADNUC uses the information following the first four lines to estimate the radionuclide content. Each line represents one record from the accountability database, and contains the following information:

- **First Field** The first field is the Key Number. This is the date (HAPO day) that the fuel was discharged from the reactor.
- **Second Field** The second field is the grade and is ignored by RADNUC. The grade corresponds to the percent ^{240}Pu of the total plutonium.
- **Third Field** The third field is the type fuel, and indicates the initial ^{235}U enrichment. This field contains either a 1 corresponding to mark 1A fuel or a 4 corresponding to mark 4 fuel.
- **Fourth Field** The fourth field is the number of elements in the record. This field is ignored by RADNUC.
- **Fifth Field** The fifth field is the percent ^{240}Pu of the total plutonium.
- **Sixth Field** The sixth field is the combined masses of uranium and plutonium from the accountability database. The intent is to estimate the pre-irradiation uranium content of the fuel this will be a slight underestimation of the pre-irradiation uranium because it does not account for fission-product mass.
- **Seventh Field** The seventh field is the month the fuel was processed in PUREX. It is important to note that the RADNUC program was written to estimate radionuclides in tank waste. The numbers used in the seventh and eighth fields is described in the eighth fields definition below.
- **Eighth Field** The eighth field is the year the fuel was processed in PUREX. The seventh and eighth fields were set to 12 and 99 corresponding to december 1999, a date later than 1/1/1995 (decay date).

"105-K Basins charge data"

"Key" "Grade" "Fuel" "Pieces" "%Pu-240" "Uranium" "Mo" "Yr"

09621	1 4	000	5.94	0.006673	12	99
09621	1 4	000	5.94	0.076694	12	99
09621	1 4	000	6.01	0.023972	12	99
09621	1 4	000	5.94	0.0099	12	99
09621	1 4	000	5.94	0.002374	12	99
09621	1 4	000	5.94	0.072655	12	99
09621	1 4	000	5.94	0.149847	12	99
09621	1 4	000	5.94	0.009728	12	99
09621	1 4	000	5.94	0.050475	12	99
10001	3 4	000	12.4	0.04212	12	99
10001	3 4	000	12.4	0.014903	12	99
10001	3 4	000	12.4	0.006639	12	99
10001	3 4	000	12.4	0.00991	12	99
10001	3 4	000	12.4	0.029932	12	99
10001	3 4	000	12.4	0.035065	12	99
10201	2 4	000	9.09	1.075459	12	99
10201	2 4	000	9.09	0.366031	12	99
10201	2 4	000	9.09	0.343918	12	99
10201	2 4	000	9.09	0.537029	12	99
10201	2 4	000	9.09	0.172074	12	99
10201	2 4	000	9.09	0.161819	12	99
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10259	3 4	000	10.41	5.758699	12	99
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10294	3 4	000	10.4	24.15077	12	99
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10350	2 4	000	9.28	0.089727	12	99
10350	2 4	000	9.28	0.014861	12	99
10350	2 4	000	9.28	0.112204	12	99
10350	2 4	000	9.28	0.042207	12	99
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10385	3 4	000	11.21	0.141018	12	99

10385	3	4	000	11.21	23.81023	12	99
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10385	3	4	000	10.21	0.075994	12	99
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10385	3	4	000	11.21	11.17924	12	99
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10982	4	4	000	14.97	15.20181	12	99
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10982	4	4	000	14.97	0.08688	12	99
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11109	4	4	000	14.62	15.10859	12	99
11109	4	4	000	14.62	1.238823	12	99
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11897	4 4	000	15.47	0.751884	12	99
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11979	4 4	000	15.34	14.26082	12	99
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11979	4 4	000	15.34	0.371669	12	99
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11979	3 4	000	11.7	0.89586	12	99
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11979	3 4	000	11.7	0.892379	12	99
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12127	4 4	000	14.73	0.497313	12	99
12127	4 4	000	14.73	0.340679	12	99
12127	4 4	000	14.73	26.23191	12	99
12127	4 4	000	14.73	3.336408	12	99
12201	4 4	000	13.98	0.13036	12	99
12201	3 4	000	11.38	0.01059	12	99
12201	3 4	000	11.38	0.449298	12	99
12201	4 4	000	13.98	15.9358	12	99

12201	4	4	000	13.98	5.664989	12	99
12201	3	4	000	11.38	0.007507	12	99
12240	4	4	000	14.51	0.063584	12	99
12240	3	4	000	11.22	0.029896	12	99
12285	4	4	000	14.45	0.014083	12	99
12285	4	4	000	14.45	0.031772	12	99
12285	3	4	000	12.1	0.759011	12	99
12285	4	4	000	14.45	57.7036	12	99
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12285	3	4	000	12.1	0.004972	12	99
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12565	4	4	000	14.15	0.290153	12	99
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12565	4	4	000	14.15	34.78675	12	99
12565	4	4	000	14.15	3.646776	12	99
12639	4	4	000	13.84	0.02309	12	99
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12639	4	4	000	13.84	5.400192	12	99
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13015	4	4	000	13.19	0.654274	12	99
13016	1	4	000	5.63	0.02194	12	99
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13649	4	4	000	13.25	0.588463	12	99
13649	4	4	000	13.25	3.972961	12	99
13685	4	4	000	13.34	2.617419	12	99
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14722	1	4	000	5.05	0.068843	12	99
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15204	1	4	000	6.01	59.48539	12	99
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15316	1	4	000	6.1	0.186043	12	99
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15316	1	4	000	6.1	4.334796	12	99
15316	1	4	000	6.1	0.007492	12	99
15444	1	4	000	4.7	0.103706	12	99
15444	1	4	000	4.7	0.116956	12	99
15444	1	4	000	4.7	0.087941	12	99
15451	1	4	000	4.97	1.497135	12	99
15451	1	4	000	6.79	6.799597	12	99
15451	1	4	000	6.79	0.092684	12	99
15451	1	4	000	6.79	0.044007	12	99
15455	1	4	000	4.94	0.373161	12	99
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15455	1	4	000	4.94	0.373987	12	99
15457	1	4	000	1.69	0.046801	12	99
15457	1	4	000	4.35	0.290374	12	99
15458	1	4	000	3.44	0.374335	12	99
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15458	1	4	000	4.26	67.36977	12	99
15458	1	4	000	4.26	0.373516	12	99
15458	1	4	000	4.26	40.36723	12	99
15460	1	4	000	2.83	2.948269	12	99
15460	1	4	000	4.29	14.93406	12	99
15461	1	4	000	1.55	12.73	12	99
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14356	3	4	000	11.28	0.010471	12	99
09621	1	1	000	6.04	0.105599	12	99
09964	1	1	000	6.18	0.01714	12	99
09964	1	1	000	6.18	0.279918	12	99
09964	1	4	000	6.26	0.041696	12	99
09964	1	1	000	6.18	0.013785	12	99
09964	1	1	000	6.18	0.003883	12	99
09964	1	1	000	6.18	0.005486	12	99
09964	1	4	000	6.26	0.033371	12	99
09993	3	1	000	10.29	0.246594	12	99
09993	3	1	000	10.29	4.643864	12	99
09993	3	1	000	10.29	3.572316	12	99
10001	4	4	000	13.59	0.031624	12	99
10001	4	4	000	13.59	0.031755	12	99
10001	1	4	000	5.61	0.013364	12	99

10001	4	4	000	13.59	0.029833	12	99
10001	4	4	000	13.59	0.723905	12	99
10001	1	4	000	5.61	0.019654	12	99
10001	1	4	000	5.61	0.02194	12	99
10001	4	4	000	13.59	0.007474	12	99
10001	1	4	000	5.61	0.006289	12	99
10001	4	4	000	13.59	0.014039	12	99
10201	2	1	000	9.22	0.249793	12	99
10201	2	1	000	9.22	0.490728	12	99
10201	2	1	000	9.22	12.14053	12	99
10201	2	1	000	9.22	1.236388	12	99
10259	3	1	000	10.35	0.736071	12	99
10259	3	1	000	10.35	2.228769	12	99
10294	2	1	000	9.8	19.0871	12	99
10294	2	1	000	9.8	0.061858	12	99
10350	3	1	000	10.65	0.011009	12	99
10350	3	1	000	10.65	1.30869	12	99
10350	3	1	000	10.65	13.58606	12	99
10350	3	1	000	10.65	0.896723	12	99
10350	3	1	000	10.65	0.005503	12	99
10385	3	1	000	10.42	0.011227	12	99
10385	3	1	000	10.42	7.494582	12	99
10456	3	1	000	11.37	15.61247	12	99
10456	3	1	000	11.37	0.958057	12	99
10602	3	1	000	11.11	0.011238	12	99
10602	3	1	000	11.11	13.86589	12	99
10602	3	1	000	11.11	0.649808	12	99
10679	3	1	000	11.76	0.163905	12	99
10679	3	1	000	11.76	7.764973	12	99
10679	3	1	000	11.76	3.558569	12	99
10742	3	1	000	12.77	0.007962	12	99
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10832	3	1	000	12.92	13.07947	12	99
10832	3	1	000	12.92	3.696517	12	99
10982	4	1	000	13.73	0.020983	12	99
10982	4	1	000	13.73	0.023391	12	99
10982	4	1	000	13.73	16.49613	12	99
10982	4	1	000	13.73	0.694274	12	99
10982	4	1	000	13.73	0.010547	12	99
11036	4	1	000	13.6	7.613446	12	99
11109	4	1	000	13.7	0.468068	12	99
11109	4	1	000	13.7	14.63286	12	99
11109	4	1	000	13.7	0.005275	12	99

11183	4	1	000	13.31	0.022094	12	99
11183	4	1	000	13.31	0.678594	12	99
11183	4	1	000	13.31	9.361327	12	99
11183	4	1	000	13.31	0.24688	12	99
11183	4	1	000	13.31	0.010881	12	99
11372	4	1	000	13.77	0.690168	12	99
11372	4	1	000	13.77	18.57126	12	99
11372	4	1	000	13.77	0.709699	12	99
11449	4	1	000	13.45	0.105257	12	99
11449	4	1	000	13.45	18.58285	12	99
11449	4	1	000	13.45	1.298701	12	99
11540	4	1	000	13.26	12.48741	12	99
11806	4	1	000	13.86	0.210315	12	99
11806	4	1	000	13.86	19.23798	12	99
11806	4	1	000	13.86	0.54114	12	99
11897	4	1	000	14.4	0.023484	12	99
11897	4	1	000	14.4	1.986457	12	99
11897	4	1	000	14.4	17.69781	12	99
11897	4	1	000	14.4	0.003856	12	99
11979	4	1	000	14.27	0.007981	12	99
11979	4	1	000	14.27	0.011234	12	99
11979	4	1	000	14.27	0.010507	12	99
11979	4	1	000	14.27	1.952108	12	99
11979	4	1	000	14.27	14.49357	12	99
11979	4	1	000	14.27	8.884854	12	99
11979	4	1	000	14.27	0.010731	12	99
12127	4	1	000	14.22	0.982493	12	99
12127	4	1	000	14.22	13.32088	12	99
12127	4	1	000	14.22	1.483142	12	99
12201	4	1	000	13.24	0.748323	12	99
12201	4	1	000	13.24	24.37494	12	99
12285	4	1	000	13.17	0.246182	12	99
12285	4	1	000	13.17	19.5675	12	99
12480	4	1	000	13.78	0.022173	12	99
12480	4	1	000	13.78	19.43003	12	99
12565	4	1	000	13.7	0.011092	12	99
12565	4	1	000	13.7	25.24008	12	99
12639	4	1	000	13.06	0.011095	12	99
12639	4	1	000	13.06	23.608	12	99
12639	4	1	000	13.06	0.005465	12	99
12823	4	1	000	13.17	7.213609	12	99
12823	4	1	000	13.17	0.63684	12	99
12852	4	1	000	13.01	10.86065	12	99
12852	4	1	000	13.01	2.874132	12	99
12852	4	4	000	13.42	0.047668	12	99
12852	4	4	000	13.42	66.60527	12	99
12852	4	4	000	13.42	0.373472	12	99
12852	4	4	000	13.42	0.022424	12	99
12942	4	1	000	13.03	0.740693	12	99
12942	4	1	000	13.03	17.44634	12	99

12942	4	1	000	13.03	2.250894	12	99
12942	3	4	000	12.1	0.374625	12	99
12942	3	4	000	12.82	0.370306	12	99
12942	3	4	000	12.82	0.557199	12	99
12942	3	4	000	12.82	50.14236	12	99
12942	3	4	000	12.82	1.471475	12	99
13015	4	4	000	13.19	0.010524	12	99
13015	3	4	000	12.13	0.374619	12	99
13015	4	4	000	13.19	4.595128	12	99
13015	4	4	000	13.19	36.66284	12	99
13015	4	4	000	13.19	2.61431	12	99
13015	4	4	000	13.19	0.004952	12	99
13016	4	1	000	13.18	0.764476	12	99
13016	4	1	000	13.18	19.61489	12	99
13016	4	1	000	13.18	0.248292	12	99
13016	4	1	000	13.18	0.005246	12	99
13016	4	1	000	13.18	0.015362	12	99
13017	3	4	000	11.39	1.131855	12	99
13017	3	4	000	11.39	25.42655	12	99
13017	3	4	000	11.39	0.374331	12	99
13371	3	1	000	12.95	0.015909	12	99
13371	3	1	000	12.95	0.245652	12	99
13371	3	1	000	12.95	4.723547	12	99
13371	3	1	000	12.95	0.010241	12	99
13372	4	4	000	13.02	0.01496	12	99
13372	4	4	000	13.02	3.176879	12	99
13372	4	4	000	13.02	11.79194	12	99
13372	4	4	000	13.02	4.333318	12	99
13372	4	4	000	13.02	0.004961	12	99
13524	4	4	000	13.13	4.10635	12	99
13524	4	4	000	13.13	2.929691	12	99
13524	4	4	000	13.13	25.97653	12	99
13524	4	4	000	13.13	4.639363	12	99
13525	4	1	000	13.52	1.219785	12	99
13525	4	1	000	13.52	16.37086	12	99
13648	3	1	000	11.67	0.032575	12	99
13650	4	1	000	13.41	1.438302	12	99
13650	4	1	000	13.41	12.42106	12	99
13650	4	1	000	13.41	2.724184	12	99
13686	3	1	000	12.41	0.023387	12	99
13686	3	1	000	12.41	17.62132	12	99
13686	3	1	000	12.41	0.247753	12	99
13724	1	1	000	5.99	0.015523	12	99
13858	3	4	000	12.01	0.374632	12	99
13858	3	4	000	12.04	3.127837	12	99
13858	3	4	000	12.04	48.353	12	99
13858	3	4	000	12.04	1.011267	12	99
13858	3	4	000	12.04	0.004955	12	99
13859	3	1	000	12.28	0.007927	12	99
13859	3	1	000	12.28	0.011657	12	99

13859	3	1	000	12.28	25.87721	12	99
13859	3	1	000	12.28	3.751426	12	99
13859	3	1	000	12.28	0.005246	12	99
14055	1	4	000	2.39	0.327945	12	99
14065	1	4	000	5.67	0.015902	12	99
14065	1	4	000	5.67	0.071951	12	99
14065	1	4	000	5.67	0.00446	12	99
14065	1	4	000	5.67	0.024244	12	99
14065	1	4	000	5.67	0.007021	12	99
14448	1	4	000	5.99	0.012272	12	99
14883	1	4	000	5.66	0.022262	12	99
15204	1	4	000	5.92	0.015919	12	99
15244	1	1	000	5.44	0.007949	12	99
15244	1	1	000	5.44	0.01126	12	99
15244	1	1	000	5.44	0.042169	12	99
15244	1	1	000	5.44	0.02338	12	99
15244	1	1	000	5.44	0.132474	12	99
15244	1	1	000	5.44	0.124027	12	99
15244	1	1	000	5.44	0.007481	12	99
15244	1	1	000	5.44	0.004961	12	99
15244	4	4	000	13.58	0.007019	12	99
15347	1	1	000	6.16	0.007949	12	99
15347	1	1	000	6.16	0.01169	12	99
15347	1	1	000	6.16	0.083168	12	99
15347	1	1	000	6.16	0.031137	12	99
15347	1	1	000	6.16	0.010645	12	99
15445	1	1	000	5.23	1.297919	12	99
15445	1	1	000	5.23	0.669493	12	99
15448	1	1	000	4.54	0.008002	12	99
15448	1	1	000	4.54	0.015812	12	99
15448	1	1	000	4.54	0.003766	12	99
15448	1	1	000	4.54	0.005327	12	99
15451	1	1	000	5.31	0.46584	12	99
15457	1	1	000	2.85	0.033293	12	99
15459	1	1	000	4.06	25.28046	12	99
15459	1	1	000	4.06	8.661046	12	99
15460	1	1	000	3.32	2.097296	12	99
15555	1	1	000	4	0.008002	12	99
15555	1	1	000	4	0.01132	12	99
11540	4	4	000	16.49	0.031623	12	99
11540	4	4	000	16.49	6.873061	12	99
11540	4	4	000	16.49	0.984106	12	99
11540	4	4	000	13.73	0.009902	12	99
11540	4	4	000	16.49	3.241216	12	99
11540	4	4	000	16.49	0.456485	12	99
12565	4	4	000	16.72	6.674973	12	99
12565	4	4	000	16.72	0.175589	12	99
12852	4	4	000	16.62	4.201198	12	99

"06/02/1997 RADNUC2A run for combined K Basins "
 " Results decayed to 05/31/1998 "
 " Total MTU " 2099.1
 " Total Curies " 5.50E+07
 " Total BTU " 5.49E+05

"Isotope"	"Curies"	"BTUs"
"H-3	"3.59E+04	4.12E+00
"C-14	"2.35E-02	2.34E-05
"Se-79	"8.62E+01	9.07E-02
"Kr-85	"5.90E+05	3.02E+03
"Sr-89	"0.00E+00	0.00E+00
"Sr-90	"1.01E+07	4.01E+04
"Y-90	"1.01E+07	1.91E+05
"Y-91	"2.23E-14	2.73E-16
"Zr-93	"3.87E+02	1.52E-01
"Nb-93m	"2.39E+02	1.46E-01
"Zr-95	"1.67E-12	2.87E-14
"Nb-95	"3.70E-12	6.06E-14
"Nb-95m	"1.24E-14	5.58E-17
"Tc-99	"2.88E+03	4.93E+00
"Ru-103	"0.00E+00	0.00E+00
"Rh-103m	"0.00E+00	0.00E+00
"Ru-106	"1.82E+03	3.70E-01
"Rh-106	"1.82E+03	5.94E+01
"Pd-107	"1.63E+01	3.05E-03
"Ag-110	"2.84E-04	7.28E-09
"Ag-110m	"2.14E-02	1.21E-03
"Cd-113m	"3.55E+03	1.33E+01
"Cd-115m	"0.00E+00	0.00E+00
"Sn-119m	"2.76E-02	4.85E-05
"Sn-121m	"2.69E+01	9.64E-02
"Sn-123	"1.74E-05	1.85E-07
"Te-123m	"2.76E-11	1.36E-13
"Sb-124	"3.03E-18	1.37E-19
"Sb-125	"3.23E+04	3.47E+02
"Te-125m	"7.89E+03	2.27E+01
"Sn-126	"1.56E+02	1.63E-01
"Sb-126	"2.18E+01	1.35E+00
"Sb-126m	"1.56E+02	6.81E+00
"Te-127	"9.49E-07	4.37E-09
"Te-127m	"9.69E-07	1.64E-09
"Te-129	"0.00E+00	0.00E+00
"Te-129m	"0.00E+00	0.00E+00
"I-129	"6.37E+00	1.02E-02
"Cs-134	"1.59E+04	5.54E+02
"Cs-135	"7.75E+01	8.84E-02
"Cs-137	"1.32E+07	4.55E+04
"Ba-137m	"1.25E+07	1.67E+05
"Ce-141	"0.00E+00	0.00E+00
"Pr-143	"0.00E+00	0.00E+00
"Ce-144	"9.14E+02	2.06E+00
"Pr-144	"9.03E+02	2.26E+01
"Pr-144m	"1.10E+01	1.26E-02
"Pm-147	"4.62E+05	5.79E+02
"Pm-148	"0.00E+00	0.00E+00
"Pm-148m	"0.00E+00	0.00E+00

"Sm-151	"1.76E+05	7.02E+01
"Eu-152	"9.45E+02	1.46E+01
"Gd-153	"1.28E-04	3.94E-07
"Eu-154	"1.07E+05	3.29E+03
"Eu-155	"2.21E+04	5.45E+01
"Tb-160	"2.77E-15	7.55E-17
"H-3	"7.24E+02	8.31E-02
"C-14	"6.93E+02	6.91E-01
"Fe-55	"1.84E+03	2.09E-01
"Ni-59	"4.10E+01	5.55E-03
"Co-60	"3.96E+03	2.08E+02
"Ni-63	"4.49E+03	1.55E+00
"Zr-93	"1.32E+01	5.20E-03
"Nb-93m	"8.18E+00	5.00E-03
"Zr-95	"1.64E-14	2.81E-16
"Nb-95	"3.63E-14	5.94E-16
"Nb-95m	"1.21E-16	5.47E-19
"In-113m	"2.14E-07	1.69E-09
"Sn-113	"2.14E-07	1.21E-10
"Sn-119m	"2.69E-01	4.73E-04
"Sn-121m	"5.27E+01	1.89E-01
"Sb-125	"1.19E+03	1.27E+01
"Te-125m	"2.89E+02	8.32E-01
"U-234	"8.74E+02	8.44E+01
"U-235	"3.37E+01	3.12E+00
"U-236	"1.27E+02	1.16E+01
"U-238	"6.96E+02	5.92E+01
"Np-237	"5.72E+01	5.63E+00
"Pu-238	"1.11E+05	1.24E+04
"Pu-239	"2.51E+05	2.62E+04
"Pu-240	"1.44E+05	1.51E+04
"Pu-241	"6.68E+06	7.07E+02
"Pu-242	"5.49E+01	5.45E+00
"Am-241	"3.75E+05	4.19E+04
"Am-242m	"1.96E+02	1.02E+00
"Am-242	"1.95E+02	5.53E+00
"Am-243	"1.20E+02	1.30E+01
"Cm-242	"1.62E+02	2.00E+01
"Cm-244	"1.44E+03	1.69E+02

"105 K-East charge data"

"Key"	"Grade"	"Fuel"	"Pieces"	"%Pu-240"	"Uranium"	"Mo"	"Yr"
09621	1	4	000	5.94	0.006673	12	99
09621	1	4	000	5.94	0.076694	12	99
09621	1	4	000	6.01	0.023972	12	99
09621	1	4	000	5.94	0.0099	12	99
09621	1	4	000	5.94	0.002374	12	99
09621	1	4	000	5.94	0.072655	12	99
09621	1	4	000	5.94	0.149847	12	99
09621	1	4	000	5.94	0.009728	12	99
09621	1	4	000	5.94	0.050475	12	99
10001	3	4	000	12.4	0.04212	12	99
10001	3	4	000	12.4	0.014903	12	99
10001	3	4	000	12.4	0.006639	12	99
10001	3	4	000	12.4	0.00991	12	99
10001	3	4	000	12.4	0.029932	12	99
10001	3	4	000	12.4	0.035065	12	99
10201	2	4	000	9.09	1.075459	12	99
10201	2	4	000	9.09	0.366031	12	99
10201	2	4	000	9.09	0.343918	12	99
10201	2	4	000	9.09	0.537029	12	99
10201	2	4	000	9.09	0.172074	12	99
10201	2	4	000	9.09	0.161819	12	99
10259	3	4	000	10.41	0.050612	12	99
10259	3	4	000	10.41	1.476838	12	99
10259	3	4	000	10.41	12.34494	12	99
10259	3	4	000	10.41	0.031588	12	99
10259	3	4	000	10.41	0.718433	12	99
10259	3	4	000	10.41	5.758699	12	99
10294	2	4	000	8.94	0.032608	12	99
10294	3	4	000	10.4	0.012648	12	99
10294	3	4	000	10.4	0.812226	12	99
10294	3	4	000	10.4	24.15077	12	99
10294	3	4	000	10.4	0.762301	12	99
10294	2	4	000	8.94	0.027153	12	99
10294	3	4	000	10.4	0.006317	12	99
10294	3	4	000	10.4	0.381403	12	99
10294	3	4	000	10.4	11.35286	12	99
10294	3	4	000	10.4	0.344639	12	99
10350	2	4	000	9.28	0.031646	12	99
10350	2	4	000	9.28	0.238728	12	99
10350	2	4	000	9.28	0.089727	12	99
10350	2	4	000	9.28	0.014861	12	99
10350	2	4	000	9.28	0.112204	12	99
10350	2	4	000	9.28	0.042207	12	99
10385	3	4	000	10.21	0.130384	12	99
10385	3	4	000	11.21	0.141018	12	99
10385	3	4	000	11.21	23.81023	12	99
10385	3	4	000	11.21	0.298905	12	99
10385	3	4	000	10.21	0.075994	12	99
10385	3	4	000	11.21	0.059715	12	99
10385	3	4	000	11.21	11.17924	12	99
10385	3	4	000	11.21	0.154738	12	99
10456	3	4	000	11.24	0.009548	12	99
10456	3	4	000	11.24	0.031918	12	99
10456	3	4	000	11.24	0.414766	12	99
10456	3	4	000	11.24	1.348222	12	99
10456	3	4	000	11.24	21.44289	12	99

10456	3	4	000	11.24	3.186772	12	99
10576	3	4	000	11.09	3.881855	12	99
10576	3	4	000	11.09	0.043956	12	99
10602	4	4	000	13.75	0.081456	12	99
10602	3	4	000	11.09	0.031925	12	99
10602	4	4	000	13.75	0.55955	12	99
10602	3	4	000	11.09	0.030958	12	99
10602	4	4	000	13.75	24.79309	12	99
10602	4	4	000	13.75	1.22853	12	99
10679	4	4	000	14.09	0.309509	12	99
10679	3	4	000	11.09	0.014136	12	99
10679	4	4	000	14.09	0.124291	12	99
10679	3	4	000	11.09	1.255033	12	99
10679	4	4	000	14.09	22.96605	12	99
10679	4	4	000	14.09	1.296428	12	99
10742	4	4	000	14.25	1.170103	12	99
10742	4	4	000	14.25	12.36114	12	99
10742	4	4	000	14.25	0.462965	12	99
10742	3	4	000	10.8	0.549476	12	99
10742	4	4	000	14.25	5.806125	12	99
10742	3	4	000	10.8	0.22488	12	99
10770	3	4	000	11.66	0.681667	12	99
10770	4	4	000	13.91	14.11261	12	99
10770	4	4	000	13.91	4.436449	12	99
10832	4	4	000	13.65	0.260688	12	99
10832	4	4	000	13.65	0.041425	12	99
10832	4	4	000	13.65	2.044619	12	99
10832	4	4	000	13.65	47.31945	12	99
10832	4	4	000	13.65	3.138458	12	99
10982	4	4	000	14.97	0.173868	12	99
10982	4	4	000	14.97	0.126892	12	99
10982	3	4	000	11.7	0.706201	12	99
10982	4	4	000	14.97	15.20181	12	99
10982	4	4	000	14.97	0.94076	12	99
10982	4	4	000	14.97	0.08688	12	99
10982	3	4	000	11.7	0.059721	12	99
10982	3	4	000	11.7	0.331757	12	99
10982	4	4	000	14.97	7.090482	12	99
10982	3	4	000	11.7	0.498938	12	99
11036	4	4	000	14.37	0.006372	12	99
11036	4	4	000	13.31	0.326927	12	99
11036	4	4	000	14.37	7.817157	12	99
11036	4	4	000	14.37	0.686572	12	99
11036	4	4	000	13.31	0.003176	12	99
11036	4	4	000	14.37	0.013257	12	99
11036	4	4	000	13.31	0.123844	12	99
11036	4	4	000	14.37	3.698878	12	99
11036	4	4	000	14.37	0.330283	12	99
11109	4	4	000	14.62	0.16921	12	99
11109	4	4	000	14.62	0.674804	12	99
11109	4	4	000	14.62	15.10859	12	99
11109	4	4	000	14.62	1.238823	12	99
11109	3	4	000	10.32	0.079566	12	99
11109	3	4	000	10.32	0.331711	12	99
11109	4	4	000	14.62	7.091013	12	99
11109	4	4	000	14.62	0.576213	12	99
11183	4	4	000	14.49	0.184538	12	99
11183	4	4	000	14.49	0.140773	12	99

11183	4	4	000	14.49	0.189703	12	99
11183	4	4	000	14.49	32.90384	12	99
11183	4	4	000	14.49	1.014554	12	99
11183	4	4	000	14.49	0.031018	12	99
11183	4	4	000	14.49	0.210257	12	99
11183	4	4	000	14.49	0.043892	12	99
11183	4	4	000	14.49	0.097674	12	99
11183	4	4	000	14.49	0.066214	12	99
11183	3	4	000	11.02	0.089084	12	99
11183	4	4	000	14.49	15.43086	12	99
11183	4	4	000	14.49	0.491747	12	99
11372	4	4	000	15.17	0.130374	12	99
11372	4	4	000	15.17	0.042237	12	99
11372	4	4	000	15.17	0.421612	12	99
11372	4	4	000	15.17	19.91088	12	99
11372	4	4	000	15.17	0.686564	12	99
11372	4	4	000	15.17	0.059719	12	99
11372	3	4	000	12.05	0.019892	12	99
11372	3	4	000	12.05	0.178219	12	99
11372	3	4	000	12.05	9.414279	12	99
11372	4	4	000	15.17	0.309187	12	99
11449	4	4	000	14.37	0.197256	12	99
11449	4	4	000	14.37	0.094853	12	99
11449	4	4	000	14.37	16.26954	12	99
11449	4	4	000	14.37	3.361394	12	99
11449	4	4	000	14.37	0.086193	12	99
11449	3	4	000	11.03	0.039598	12	99
11449	4	4	000	14.37	7.629647	12	99
11449	3	4	000	11.03	1.616732	12	99
11540	3	4	000	11.66	0.152125	12	99
11540	3	4	000	11.3	0.141035	12	99
11540	3	4	000	11.3	0.042167	12	99
11540	3	4	000	11.3	7.603293	12	99
11540	3	4	000	11.3	0.59779	12	99
11540	3	4	000	11.3	0.982763	12	99
11540	3	4	000	11.66	0.076006	12	99
11540	3	4	000	11.3	0.072978	12	99
11540	3	4	000	11.3	0.024759	12	99
11540	3	4	000	11.3	3.567068	12	99
11540	3	4	000	11.3	0.274249	12	99
11733	3	4	000	12.82	0.349841	12	99
11733	3	4	000	12.82	0.089642	12	99
11733	3	4	000	10.36	0.156979	12	99
11733	3	4	000	10.36	0.049213	12	99
11806	4	4	000	15.03	0.162968	12	99
11806	3	4	000	11.94	0.084643	12	99
11806	3	4	000	11.94	0.021097	12	99
11806	4	4	000	15.03	18.25962	12	99
11806	4	4	000	15.03	0.268532	12	99
11806	3	4	000	11.94	0.327677	12	99
11806	4	4	000	15.03	0.070582	12	99
11806	3	4	000	11.94	0.033183	12	99
11806	3	4	000	11.94	0.00991	12	99
11806	3	4	000	11.94	8.607779	12	99
11806	4	4	000	15.03	0.126466	12	99
11897	3	4	000	11.69	3.046193	12	99
11897	4	4	000	15.47	20.37553	12	99
11897	4	4	000	15.47	1.61182	12	99

11897	4	4	000	15.47	0.00663	12	99
11897	3	4	000	11.69	1.431082	12	99
11897	4	4	000	15.47	9.584959	12	99
11897	4	4	000	15.47	0.751884	12	99
11979	4	4	000	15.34	0.081493	12	99
11979	4	4	000	15.34	1.854741	12	99
11979	4	4	000	15.34	14.26082	12	99
11979	4	4	000	15.34	1.926291	12	99
11979	4	4	000	15.34	0.371669	12	99
11979	4	4	000	15.34	9.622194	12	99
11979	4	4	000	15.34	0.461079	12	99
11979	3	4	000	11.7	0.89586	12	99
11979	4	4	000	15.34	6.693747	12	99
11979	3	4	000	11.7	0.892379	12	99
12127	4	4	000	14.73	0.146527	12	99
12127	4	4	000	14.73	0.497313	12	99
12127	4	4	000	14.73	0.340679	12	99
12127	4	4	000	14.73	26.23191	12	99
12127	4	4	000	14.73	3.336408	12	99
12201	4	4	000	13.98	0.13036	12	99
12201	3	4	000	11.38	0.01059	12	99
12201	3	4	000	11.38	0.449298	12	99
12201	4	4	000	13.98	15.9358	12	99
12201	4	4	000	13.98	5.664989	12	99
12201	3	4	000	11.38	0.007507	12	99
12240	4	4	000	14.51	0.063584	12	99
12240	3	4	000	11.22	0.029896	12	99
12285	4	4	000	14.45	0.014083	12	99
12285	4	4	000	14.45	0.031772	12	99
12285	3	4	000	12.1	0.759011	12	99
12285	4	4	000	14.45	57.7036	12	99
12285	4	4	000	14.45	6.101811	12	99
12285	3	4	000	12.1	0.004972	12	99
12480	4	4	000	14.29	0.580019	12	99
12480	3	4	000	12.38	0.24784	12	99
12480	4	4	000	14.29	49.62144	12	99
12480	4	4	000	14.29	3.732459	12	99
12565	4	4	000	14.15	0.290153	12	99
12565	4	4	000	14.15	0.139326	12	99
12565	4	4	000	14.15	34.78675	12	99
12565	4	4	000	14.15	3.646776	12	99
12639	4	4	000	13.84	0.02309	12	99
12639	4	4	000	13.84	40.3312	12	99
12639	4	4	000	13.84	5.400192	12	99
12852	3	4	000	11.43	2.525342	12	99
13015	4	4	000	13.19	0.216604	12	99
13015	4	4	000	13.19	0.654274	12	99
13016	1	4	000	5.63	0.02194	12	99
13371	1	4	000	6.58	0.021128	12	99
13649	3	4	000	11.4	0.374685	12	99
13649	4	4	000	13.25	0.588463	12	99
13649	4	4	000	13.25	3.972961	12	99
13685	4	4	000	13.34	2.617419	12	99
14065	1	4	000	5.62	0.023287	12	99
14355	4	4	000	15.74	0.094818	12	99
14355	3	4	000	12	6.202162	12	99
14355	4	4	000	15.74	0.313212	12	99
14355	4	4	000	15.74	0.04455	12	99

14355	3	4	000	12	2.916202	12	99
14355	4	4	000	15.74	0.147505	12	99
14356	3	4	000	11.28	0.180698	12	99
14356	3	4	000	11.28	3.207757	12	99
14356	3	4	000	11.28	0.022035	12	99
14356	3	4	000	11.28	8.161901	12	99
14356	3	4	000	11.28	1.352138	12	99
14356	3	4	000	11.28	0.372588	12	99
14356	3	4	000	11.28	0.085034	12	99
14356	3	4	000	11.28	0.103665	12	99
14722	1	4	000	5.05	0.068843	12	99
15204	1	4	000	6.01	0.015919	12	99
15204	1	4	000	5.53	0.351245	12	99
15204	1	4	000	6.01	14.30584	12	99
15204	1	4	000	6.01	3.164697	12	99
15204	1	4	000	6.01	59.48539	12	99
15204	1	4	000	6.01	20.10897	12	99
15204	1	4	000	6.01	0.007492	12	99
15240	1	4	000	5.6	0.015919	12	99
15240	1	4	000	5.71	0.374804	12	99
15240	1	4	000	5.6	0.088008	12	99
15240	1	4	000	5.6	0.006644	12	99
15240	1	4	000	5.6	0.007041	12	99
15316	1	4	000	6.1	10.04697	12	99
15316	1	4	000	6.1	0.186043	12	99
15316	1	4	000	6.1	4.962912	12	99
15316	1	4	000	6.1	4.334796	12	99
15316	1	4	000	6.1	0.007492	12	99
15444	1	4	000	4.7	0.103706	12	99
15444	1	4	000	4.7	0.116956	12	99
15444	1	4	000	4.7	0.087941	12	99
15451	1	4	000	4.97	1.497135	12	99
15451	1	4	000	6.79	6.799597	12	99
15451	1	4	000	6.79	0.092684	12	99
15451	1	4	000	6.79	0.044007	12	99
15455	1	4	000	4.94	0.373161	12	99
15455	1	4	000	4.94	0.748424	12	99
15455	1	4	000	4.94	0.373987	12	99
15457	1	4	000	1.69	0.046801	12	99
15457	1	4	000	4.35	0.290374	12	99
15458	1	4	000	3.44	0.374335	12	99
15458	1	4	000	4.26	21.2791	12	99
15458	1	4	000	4.26	2.597811	12	99
15458	1	4	000	4.26	67.36977	12	99
15458	1	4	000	4.26	0.373516	12	99
15458	1	4	000	4.26	40.36723	12	99
15460	1	4	000	2.83	2.948269	12	99
15460	1	4	000	4.29	14.93406	12	99
15461	1	4	000	1.55	12.73	12	99
14356	3	4	000	11.28	0.014051	12	99
14356	3	4	000	11.28	0.010471	12	99

"06/02/1997 RADNUC2A run for K-EAST
 " Results decayed to 05/31/1998 "
 " Total MTU " 1146.0
 " Total Curies " 2.73E+07
 " Total BTU " 2.75E+05

"Isotope"	"Curies"	"BTUs"
"H-3	"1.76E+04	2.02E+00
"C-14	"1.18E-02	1.18E-05
"Se-79	"4.35E+01	4.57E-02
"Kr-85	"2.84E+05	1.45E+03
"Sr-89	"0.00E+00	0.00E+00
"Sr-90	"4.96E+06	1.97E+04
"Y-90	"4.97E+06	9.37E+04
"Y-91	"1.78E-14	2.18E-16
"Zr-93	"1.94E+02	7.60E-02
"Nb-93m	"1.22E+02	7.43E-02
"Zr-95	"1.34E-12	2.31E-14
"Nb-95	"2.98E-12	4.87E-14
"Nb-95m	"9.95E-15	4.48E-17
"Tc-99	"1.45E+03	2.48E+00
"Ru-103	"0.00E+00	0.00E+00
"Rh-103m	"0.00E+00	0.00E+00
"Ru-106	"1.39E+03	2.82E-01
"Rh-106	"1.39E+03	4.52E+01
"Pd-107	"8.59E+00	1.61E-03
"Ag-110	"2.28E-04	5.83E-09
"Ag-110m	"1.71E-02	9.70E-04
"Cd-113m	"1.80E+03	6.77E+00
"Cd-115m	"0.00E+00	0.00E+00
"Sn-119m	"2.35E-02	4.13E-05
"Sn-121m	"1.38E+01	4.94E-02
"Sn-123	"1.46E-05	1.56E-07
"Te-123m	"2.33E-11	1.15E-13
"Sb-124	"2.49E-18	1.12E-19
"Sb-125	"1.63E+04	1.75E+02
"Te-125m	"3.98E+03	1.14E+01
"Sn-126	"8.07E+01	8.46E-02
"Sb-126	"1.13E+01	6.98E-01
"Sb-126m	"8.07E+01	3.53E+00
"Te-127	"7.90E-07	3.64E-09
"Te-127m	"8.07E-07	1.36E-09
"Te-129	"0.00E+00	0.00E+00
"Te-129m	"0.00E+00	0.00E+00
"I-129	"3.26E+00	5.20E-03
"Cs-134	"6.95E+03	2.42E+02
"Cs-135	"3.96E+01	4.52E-02
"Cs-137	"6.55E+06	2.26E+04
"Ba-137m	"6.19E+06	8.29E+04
"Ce-141	"0.00E+00	0.00E+00
"Pr-143	"0.00E+00	0.00E+00
"Ce-144	"7.56E+02	1.70E+00
"Pr-144	"7.47E+02	1.87E+01
"Pr-144m	"9.07E+00	1.04E-02
"Pm-147	"2.45E+05	3.07E+02
"Pm-148	"0.00E+00	0.00E+00
"Pm-148m	"0.00E+00	0.00E+00

"Sm-151	"8.92E+04	3.56E+01
"Eu-152	"4.67E+02	7.21E+00
"Gd-153	"9.59E-05	2.95E-07
"Eu-154	"5.30E+04	1.62E+03
"Eu-155	"1.12E+04	2.77E+01
"Tb-160	"2.29E-15	6.25E-17
"H-3	"3.75E+02	4.30E-02
"C-14	"3.62E+02	3.61E-01
"Fe-55	"9.64E+02	1.10E-01
"Ni-59	"2.11E+01	2.86E-03
"Co-60	"1.86E+03	9.76E+01
"Ni-63	"2.30E+03	7.95E-01
"Zr-93	"6.58E+00	2.59E-03
"Nb-93m	"4.13E+00	2.53E-03
"Zr-95	"1.31E-14	2.26E-16
"Nb-95	"2.91E-14	4.77E-16
"Nb-95m	"9.73E-17	4.39E-19
"In-113m	"1.77E-07	1.40E-09
"Sn-113	"1.77E-07	1.00E-10
"Sn-119m	"2.26E-01	3.97E-04
"Sn-121m	"2.63E+01	9.41E-02
"Sb-125	"5.96E+02	6.39E+00
"Te-125m	"1.45E+02	4.18E-01
"U-234	"4.66E+02	4.51E+01
"U-235	"1.77E+01	1.64E+00
"U-236	"6.61E+01	6.03E+00
"U-238	"3.80E+02	3.23E+01
"Np-237	"3.02E+01	2.97E+00
"Pu-238	"6.05E+04	6.73E+03
"Pu-239	"1.32E+05	1.38E+04
"Pu-240	"7.64E+04	7.98E+03
"Pu-241	"3.42E+06	3.62E+02
"Pu-242	"3.07E+01	3.05E+00
"Am-241	"2.06E+05	2.30E+04
"Am-242m	"1.14E+02	5.95E-01
"Am-242	"1.13E+02	3.22E+00
"Am-243	"7.12E+01	7.68E+00
"Cm-242	"9.40E+01	1.16E+01
"Cm-244	"8.71E+02	1.02E+02

"150-K WEST charge data"

"Key"	"Grade"	"Fuel"	"Pieces"	"%Pu-240"	"Uranium"	"Mo"	"Yr"
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09621	1	1	000	6.04	0.105599	12	99
09964	1	1	000	6.18	0.01714	12	99
09964	1	1	000	6.18	0.279918	12	99
09964	1	4	000	6.26	0.041696	12	99
09964	1	1	000	6.18	0.013785	12	99
09964	1	1	000	6.18	0.003883	12	99
09964	1	1	000	6.18	0.005486	12	99
09964	1	4	000	6.26	0.033371	12	99
09993	3	1	000	10.29	0.246594	12	99
09993	3	1	000	10.29	4.643864	12	99
09993	3	1	000	10.29	3.572316	12	99
10001	4	4	000	13.59	0.031624	12	99
10001	4	4	000	13.59	0.031755	12	99
10001	1	4	000	5.61	0.013364	12	99
10001	4	4	000	13.59	0.029833	12	99
10001	4	4	000	13.59	0.723905	12	99
10001	1	4	000	5.61	0.019654	12	99
10001	1	4	000	5.61	0.02194	12	99
10001	4	4	000	13.59	0.007474	12	99
10001	1	4	000	5.61	0.006289	12	99
10001	4	4	000	13.59	0.014039	12	99
10201	2	1	000	9.22	0.249793	12	99
10201	2	1	000	9.22	0.490728	12	99
10201	2	1	000	9.22	12.14053	12	99
10201	2	1	000	9.22	1.236388	12	99
10259	3	1	000	10.35	0.736071	12	99
10259	3	1	000	10.35	2.228769	12	99
10294	2	1	000	9.8	19.0871	12	99
10294	2	1	000	9.8	0.061858	12	99
10350	3	1	000	10.65	0.011009	12	99
10350	3	1	000	10.65	1.30869	12	99
10350	3	1	000	10.65	13.58606	12	99
10350	3	1	000	10.65	0.896723	12	99
10350	3	1	000	10.65	0.005503	12	99
10385	3	1	000	10.42	0.011227	12	99
10385	3	1	000	10.42	7.494582	12	99
10456	3	1	000	11.37	15.61247	12	99
10456	3	1	000	11.37	0.958057	12	99
10602	3	1	000	11.11	0.011238	12	99
10602	3	1	000	11.11	13.86589	12	99
10602	3	1	000	11.11	0.649808	12	99
10679	3	1	000	11.76	0.163905	12	99
10679	3	1	000	11.76	7.764973	12	99
10679	3	1	000	11.76	3.558569	12	99
10742	3	1	000	12.77	0.007962	12	99
10742	3	1	000	12.77	0.210765	12	99
10742	3	1	000	12.77	3.650096	12	99
10742	3	1	000	12.77	2.783527	12	99
10742	3	1	000	12.77	0.005285	12	99
10770	4	1	000	13.16	0.010359	12	99
10770	4	1	000	13.16	6.159879	12	99
10770	4	1	000	13.16	2.009749	12	99
10770	4	1	000	13.16	0.005106	12	99
10832	3	1	000	12.92	13.07947	12	99

10832	3	1	000	12.92	3.696517	12	99
10982	4	1	000	13.73	0.020983	12	99
10982	4	1	000	13.73	0.023391	12	99
10982	4	1	000	13.73	16.49613	12	99
10982	4	1	000	13.73	0.694274	12	99
10982	4	1	000	13.73	0.010547	12	99
11036	4	1	000	13.6	7.613446	12	99
11109	4	1	000	13.7	0.468068	12	99
11109	4	1	000	13.7	14.63286	12	99
11109	4	1	000	13.7	0.005275	12	99
11183	4	1	000	13.31	0.022094	12	99
11183	4	1	000	13.31	0.678594	12	99
11183	4	1	000	13.31	9.361327	12	99
11183	4	1	000	13.31	0.24688	12	99
11183	4	1	000	13.31	0.010881	12	99
11372	4	1	000	13.77	0.690168	12	99
11372	4	1	000	13.77	18.57126	12	99
11372	4	1	000	13.77	0.709699	12	99
11449	4	1	000	13.45	0.105257	12	99
11449	4	1	000	13.45	18.58285	12	99
11449	4	1	000	13.45	1.298701	12	99
11540	4	1	000	13.26	12.48741	12	99
11806	4	1	000	13.86	0.210315	12	99
11806	4	1	000	13.86	19.23798	12	99
11806	4	1	000	13.86	0.54114	12	99
11897	4	1	000	14.4	0.023484	12	99
11897	4	1	000	14.4	1.986457	12	99
11897	4	1	000	14.4	17.69781	12	99
11897	4	1	000	14.4	0.003856	12	99
11979	4	1	000	14.27	0.007981	12	99
11979	4	1	000	14.27	0.011234	12	99
11979	4	1	000	14.27	0.010507	12	99
11979	4	1	000	14.27	1.952108	12	99
11979	4	1	000	14.27	14.49357	12	99
11979	4	1	000	14.27	8.884854	12	99
11979	4	1	000	14.27	0.010731	12	99
12127	4	1	000	14.22	0.982493	12	99
12127	4	1	000	14.22	13.32088	12	99
12127	4	1	000	14.22	1.483142	12	99
12201	4	1	000	13.24	0.748323	12	99
12201	4	1	000	13.24	24.37494	12	99
12285	4	1	000	13.17	0.246182	12	99
12285	4	1	000	13.17	19.5675	12	99
12480	4	1	000	13.78	0.022173	12	99
12480	4	1	000	13.78	19.43003	12	99
12565	4	1	000	13.7	0.011092	12	99
12565	4	1	000	13.7	25.24008	12	99
12639	4	1	000	13.06	0.011095	12	99
12639	4	1	000	13.06	23.608	12	99
12639	4	1	000	13.06	0.005465	12	99
12823	4	1	000	13.17	7.213609	12	99
12823	4	1	000	13.17	0.63684	12	99
12852	4	1	000	13.01	10.86065	12	99
12852	4	1	000	13.01	2.874132	12	99
12852	4	4	000	13.42	0.047668	12	99
12852	4	4	000	13.42	66.60527	12	99
12852	4	4	000	13.42	0.373472	12	99
12852	4	4	000	13.42	0.022424	12	99

12942	4	1	000	13.03	0.740693	12	99
12942	4	1	000	13.03	17.44634	12	99
12942	4	1	000	13.03	2.250894	12	99
12942	3	4	000	12.1	0.374625	12	99
12942	3	4	000	12.82	0.370306	12	99
12942	3	4	000	12.82	0.557199	12	99
12942	3	4	000	12.82	50.14236	12	99
12942	3	4	000	12.82	1.471475	12	99
13015	4	4	000	13.19	0.010524	12	99
13015	3	4	000	12.13	0.374619	12	99
13015	4	4	000	13.19	4.595128	12	99
13015	4	4	000	13.19	36.66284	12	99
13015	4	4	000	13.19	2.61431	12	99
13015	4	4	000	13.19	0.004952	12	99
13016	4	1	000	13.18	0.764476	12	99
13016	4	1	000	13.18	19.61489	12	99
13016	4	1	000	13.18	0.248292	12	99
13016	4	1	000	13.18	0.005246	12	99
13016	4	1	000	13.18	0.015362	12	99
13017	3	4	000	11.39	1.131855	12	99
13017	3	4	000	11.39	25.42655	12	99
13017	3	4	000	11.39	0.374331	12	99
13371	3	1	000	12.95	0.015909	12	99
13371	3	1	000	12.95	0.245652	12	99
13371	3	1	000	12.95	4.723547	12	99
13371	3	1	000	12.95	0.010241	12	99
13372	4	4	000	13.02	0.01496	12	99
13372	4	4	000	13.02	3.176879	12	99
13372	4	4	000	13.02	11.79194	12	99
13372	4	4	000	13.02	4.333318	12	99
13372	4	4	000	13.02	0.004961	12	99
13524	4	4	000	13.13	4.10635	12	99
13524	4	4	000	13.13	2.929691	12	99
13524	4	4	000	13.13	25.97653	12	99
13524	4	4	000	13.13	4.639363	12	99
13525	4	1	000	13.52	1.219785	12	99
13525	4	1	000	13.52	16.37086	12	99
13648	3	1	000	11.67	0.032575	12	99
13650	4	1	000	13.41	1.438302	12	99
13650	4	1	000	13.41	12.42106	12	99
13650	4	1	000	13.41	2.724184	12	99
13686	3	1	000	12.41	0.023387	12	99
13686	3	1	000	12.41	17.62132	12	99
13686	3	1	000	12.41	0.247753	12	99
13724	1	1	000	5.99	0.015523	12	99
13858	3	4	000	12.01	0.374632	12	99
13858	3	4	000	12.04	3.127837	12	99
13858	3	4	000	12.04	48.353	12	99
13858	3	4	000	12.04	1.011267	12	99
13858	3	4	000	12.04	0.004955	12	99
13859	3	1	000	12.28	0.007927	12	99
13859	3	1	000	12.28	0.011657	12	99
13859	3	1	000	12.28	25.87721	12	99
13859	3	1	000	12.28	3.751426	12	99
13859	3	1	000	12.28	0.005246	12	99
14055	1	4	000	2.39	0.327945	12	99
14065	1	4	000	5.67	0.015902	12	99
14065	1	4	000	5.67	0.071951	12	99

14065	1	4	000	5.67	0.00446	12	99
14065	1	4	000	5.67	0.024244	12	99
14065	1	4	000	5.67	0.007021	12	99
14448	1	4	000	5.99	0.012272	12	99
14883	1	4	000	5.66	0.022262	12	99
15204	1	4	000	5.92	0.015919	12	99
15244	1	1	000	5.44	0.007949	12	99
15244	1	1	000	5.44	0.01126	12	99
15244	1	1	000	5.44	0.042169	12	99
15244	1	1	000	5.44	0.02338	12	99
15244	1	1	000	5.44	0.132474	12	99
15244	1	1	000	5.44	0.124027	12	99
15244	1	1	000	5.44	0.007481	12	99
15244	1	1	000	5.44	0.004961	12	99
15244	4	4	000	13.58	0.007019	12	99
15347	1	1	000	6.16	0.007949	12	99
15347	1	1	000	6.16	0.01169	12	99
15347	1	1	000	6.16	0.083168	12	99
15347	1	1	000	6.16	0.031137	12	99
15347	1	1	000	6.16	0.010645	12	99
15445	1	1	000	5.23	1.297919	12	99
15445	1	1	000	5.23	0.669493	12	99
15448	1	1	000	4.54	0.008002	12	99
15448	1	1	000	4.54	0.015812	12	99
15448	1	1	000	4.54	0.003766	12	99
15448	1	1	000	4.54	0.005327	12	99
15451	1	1	000	5.31	0.46584	12	99
15457	1	1	000	2.85	0.033293	12	99
15459	1	1	000	4.06	25.28046	12	99
15459	1	1	000	4.06	8.661046	12	99
15460	1	1	000	3.32	2.097296	12	99
15555	1	1	000	4	0.008002	12	99
15555	1	1	000	4	0.01132	12	99
11540	4	4	000	16.49	0.031623	12	99
11540	4	4	000	16.49	6.873061	12	99
11540	4	4	000	16.49	0.984106	12	99
11540	4	4	000	13.73	0.009902	12	99
11540	4	4	000	16.49	3.241216	12	99
11540	4	4	000	16.49	0.456485	12	99
12565	4	4	000	16.72	6.674973	12	99
12565	4	4	000	16.72	0.175589	12	99
12852	4	4	000	16.62	4.201198	12	99


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"06/02/1997 RADNUC2A run for K-WEST
" Results decayed to 05/31/1998 "
" Total MTU " 953.1
" Total Curies " 2.77E+07
" Total BTU " 2.74E+05

"Isotope" "Curies" "BTUs"
"H-3 "1.83E+04 2.09E+00
"C-14 "1.17E-02 1.16E-05
"Se-79 "4.28E+01 4.50E-02
"Kr-85 "3.06E+05 1.57E+03
"Sr-89 "0.00E+00 0.00E+00
"Sr-90 "5.17E+06 2.05E+04
"Y-90 "5.17E+06 9.76E+04
"Y-91 "4.47E-15 5.46E-17
"Zr-93 "1.94E+02 7.60E-02
"Nb-93m "1.18E+02 7.20E-02
"Zr-95 "3.27E-13 5.62E-15
"Nb-95 "7.26E-13 1.19E-14
"Nb-95m "2.43E-15 1.09E-17
"Tc-99 "1.43E+03 2.45E+00
"Ru-103 "0.00E+00 0.00E+00
"Rh-103m "0.00E+00 0.00E+00
"Ru-106 "4.34E+02 8.81E-02
"Rh-106 "4.34E+02 1.41E+01
"Pd-107 "7.68E+00 1.44E-03
"Ag-110 "5.66E-05 1.45E-09
"Ag-110m "4.26E-03 2.41E-04
"Cd-113m "1.75E+03 6.56E+00
"Cd-115m "0.00E+00 0.00E+00
"Sn-119m "4.15E-03 7.29E-06
"Sn-121m "1.31E+01 4.70E-02
"Sn-123 "2.75E-06 2.93E-08
"Te-123m "4.28E-12 2.12E-14
"Sb-124 "5.45E-19 2.46E-20
"Sb-125 "1.60E+04 1.72E+02
"Te-125m "3.91E+03 1.12E+01
"Sn-126 "7.50E+01 7.86E-02
"Sb-126 "1.05E+01 6.49E-01
"Sb-126m "7.50E+01 3.28E+00
"Te-127 "1.59E-07 7.31E-10
"Te-127m "1.62E-07 2.74E-10
"Te-129 "0.00E+00 0.00E+00
"Te-129m "0.00E+00 0.00E+00
"I-129 "3.11E+00 4.98E-03
"Cs-134 "8.96E+03 3.12E+02
"Cs-135 "3.79E+01 4.32E-02
"Cs-137 "6.64E+06 2.29E+04
"Ba-137m "6.28E+06 8.41E+04
"Ce-141 "0.00E+00 0.00E+00
"Pr-143 "0.00E+00 0.00E+00
"Ce-144 "1.58E+02 3.56E-01
"Pr-144 "1.56E+02 3.91E+00
"Pr-144m "1.90E+00 2.18E-03
"Pm-147 "2.17E+05 2.72E+02
"Pm-148 "0.00E+00 0.00E+00
"Pm-148m "0.00E+00 0.00E+00

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"Sm-151	"8.66E+04	3.46E+01
"Eu-152	"4.77E+02	7.36E+00
"Gd-153	"3.23E-05	9.92E-08
"Eu-154	"5.44E+04	1.67E+03
"Eu-155	"1.08E+04	2.68E+01
"Tb-160	"4.77E-16	1.30E-17
"H-3	"3.49E+02	4.00E-02
"C-14	"3.31E+02	3.30E-01
"Fe-55	"8.75E+02	9.95E-02
"Ni-59	"1.99E+01	2.69E-03
"Co-60	"2.10E+03	1.10E+02
"Ni-63	"2.19E+03	7.54E-01
"Zr-93	"6.65E+00	2.61E-03
"Nb-93m	"4.04E+00	2.47E-03
"Zr-95	"3.23E-15	5.57E-17
"Nb-95	"7.18E-15	1.17E-16
"Nb-95m	"2.40E-17	1.08E-19
"In-113m	"3.74E-08	2.95E-10
"Sn-113	"3.74E-08	2.12E-11
"Sn-119m	"4.33E-02	7.62E-05
"Sn-121m	"2.64E+01	9.48E-02
"Sb-125	"5.91E+02	6.33E+00
"Te-125m	"1.44E+02	4.14E-01
"U-234	"4.08E+02	3.94E+01
"U-235	"1.60E+01	1.48E+00
"U-236	"6.11E+01	5.57E+00
"U-238	"3.16E+02	2.68E+01
"Np-237	"2.70E+01	2.66E+00
"Pu-238	"5.10E+04	5.68E+03
"Pu-239	"1.19E+05	1.24E+04
"Pu-240	"6.77E+04	7.07E+03
"Pu-241	"3.26E+06	3.45E+02
"Pu-242	"2.42E+01	2.40E+00
"Am-241	"1.69E+05	1.89E+04
"Am-242m	"8.19E+01	4.28E-01
"Am-242	"8.15E+01	2.31E+00
"Am-243	"4.89E+01	5.27E+00
"Cm-242	"6.76E+01	8.35E+00
"Cm-244	"5.72E+02	6.71E+01

"105-K Basins shielding basis charge data"

"Key"	"Grade"	"Fuel"	"Pieces"	"%Pu-240"	"Uranium"	"Mo"	"Yr"
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14707	4	4	000	16.00	1.000000	12	99
14707	4	4	000	16.00	1.000000	12	99
14707	4	4	000	16.00	0.630000	12	99

"06/02/1997 RADNUC2A run for shielding basis "

" Results decayed to 05/31/1998 "

" Total MTU " 2.6

" Total Curies " 1.32E+05

" Total BTU " 1.21E+03

"Isotope"	"Curies"	"BTUs"
"H-3	"1.02E+02	1.17E-02
"C-14	"4.45E-05	4.44E-08
"Se-79	"1.62E-01	1.71E-04
"Kr-85	"1.62E+03	8.28E+00
"Sr-89	"0.00E+00	0.00E+00
"Sr-90	"2.13E+04	8.43E+01
"Y-90	"2.13E+04	4.02E+02
"Y-91	"0.00E+00	0.00E+00
"Zr-93	"7.10E-01	2.79E-04
"Nb-93m	"3.47E-01	2.12E-04
"Zr-95	"1.08E-17	1.86E-19
"Nb-95	"2.40E-17	3.93E-19
"Nb-95m	"8.03E-20	3.62E-22
"Tc-99	"5.42E+00	9.28E-03
"Ru-103	"0.00E+00	0.00E+00
"Rh-103m	"0.00E+00	0.00E+00
"Ru-106	"2.44E+01	4.96E-03
"Rh-106	"2.44E+01	7.96E-01
"Pd-107	"3.75E-02	7.05E-06
"Ag-110	"1.08E-05	2.76E-10
"Ag-110m	"8.11E-04	4.60E-05
"Cd-113m	"1.03E+01	3.86E-02
"Cd-115m	"0.00E+00	0.00E+00
"Sn-119m	"1.28E-04	2.25E-07
"Sn-121m	"6.39E-02	2.29E-04
"Sn-123	"1.06E-08	1.12E-10
"Te-123m	"3.10E-13	1.53E-15
"Sb-124	"0.00E+00	0.00E+00
"Sb-125	"2.48E+02	2.65E+00
"Te-125m	"6.04E+01	1.74E-01
"Sn-126	"3.18E-01	3.33E-04
"Sb-126	"4.45E-02	2.75E-03
"Sb-126m	"3.18E-01	1.39E-02
"Te-127	"3.02E-10	1.39E-12
"Te-127m	"3.09E-10	5.22E-13
"Te-129	"0.00E+00	0.00E+00
"Te-129m	"0.00E+00	0.00E+00
"I-129	"1.27E-02	2.03E-05
"Cs-134	"2.79E+02	9.72E+00
"Cs-135	"1.50E-01	1.71E-04
"Cs-137	"2.93E+04	1.01E+02
"Ba-137m	"2.77E+04	3.71E+02
"Ce-141	"0.00E+00	0.00E+00
"Pr-143	"0.00E+00	0.00E+00
"Ce-144	"4.55E+00	1.02E-02
"Pr-144	"4.49E+00	1.12E-01
"Pr-144m	"5.45E-02	6.26E-05
"Pm-147	"2.75E+03	3.44E+00
"Pm-148	"0.00E+00	0.00E+00
"Pm-148m	"0.00E+00	0.00E+00

"Sm-151	"2.80E+02	1.12E-01
"Eu-152	"3.16E+00	4.88E-02
"Gd-153	"1.02E-05	3.13E-08
"Eu-154	"5.24E+02	1.61E+01
"Eu-155	"8.90E+01	2.20E-01
"Tb-160	"5.78E-19	1.58E-20
"H-3	"2.89E+00	3.32E-04
"C-14	"1.37E+00	1.37E-03
"Fe-55	"1.36E+01	1.55E-03
"Ni-59	"7.88E-02	1.07E-05
"Co-60	"1.63E+01	8.55E-01
"Ni-63	"9.19E+00	3.17E-03
"Zr-93	"2.49E-02	9.76E-06
"Nb-93m	"1.21E-02	7.40E-06
"Zr-95	"1.17E-19	2.02E-21
"Nb-95	"2.61E-19	4.26E-21
"Nb-95m	"8.71E-22	3.92E-24
"In-113m	"6.74E-11	5.31E-13
"Sn-113	"6.73E-11	3.82E-14
"Sn-119m	"1.08E-03	1.91E-06
"Sn-121m	"1.10E-01	3.95E-04
"Sb-125	"8.13E+00	8.72E-02
"Te-125m	"1.98E+00	5.71E-03
"U-234	"1.02E+00	9.83E-02
"U-235	"3.40E-02	3.14E-03
"U-236	"1.85E-01	1.68E-02
"U-238	"8.71E-01	7.41E-02
"Np-237	"1.15E-01	1.13E-02
"Pu-238	"3.32E+02	3.69E+01
"Pu-239	"4.38E+02	4.56E+01
"Pu-240	"3.32E+02	3.46E+01
"Pu-241	"2.45E+04	2.59E+00
"Pu-242	"1.94E-01	1.92E-02
"Am-241	"7.58E+02	8.49E+01
"Am-242m	"8.34E-01	4.36E-03
"Am-242	"8.30E-01	2.36E-02
"Am-243	"5.77E-01	6.23E-02
"Cm-242	"6.89E-01	8.51E-02
"Cm-244	"1.20E+01	1.41E+00

"105-K Basins Maximum Pu-240 charge data safety/regulatory assessment basis"

"Key" "Grade" "Fuel" "Pieces" "%Pu-240" "Uranium" "Mo" "Yr"

12565	4	4	000	16.72	6.674973	12	99
12565	4	4	000	16.72	0.175589	12	99

"06/02/1997 RADNUC2A run for Safety Basis "

" Results decayed to 05/31/1998 "

" Total MTU " 11.6

" Total Curies " 4.77E+05

" Total BTU " 4.84E+03

"Isotope"	"Curies"	"BTUs"
"H-3	"2.94E+02	3.37E-02
"C-14	"2.08E-04	2.08E-07
"Se-79	"7.59E-01	7.98E-04
"Kr-85	"4.29E+03	2.19E+01
"Sr-89	"0.00E+00	0.00E+00
"Sr-90	"8.04E+04	3.18E+02
"Y-90	"8.04E+04	1.52E+03
"Y-91	"0.00E+00	0.00E+00
"Zr-93	"3.31E+00	1.30E-03
"Nb-93m	"2.16E+00	1.32E-03
"Zr-95	"0.00E+00	0.00E+00
"Nb-95	"0.00E+00	0.00E+00
"Nb-95m	"0.00E+00	0.00E+00
"Tc-99	"2.54E+01	4.34E-02
"Ru-103	"0.00E+00	0.00E+00
"Rh-103m	"0.00E+00	0.00E+00
"Ru-106	"2.97E-01	6.03E-05
"Rh-106	"2.97E-01	9.68E-03
"Pd-107	"1.81E-01	3.39E-05
"Ag-110	"8.32E-09	2.13E-13
"Ag-110m	"6.25E-07	3.54E-08
"Cd-113m	"3.22E+01	1.21E-01
"Cd-115m	"0.00E+00	0.00E+00
"Sn-119m	"7.58E-08	1.33E-10
"Sn-121m	"2.70E-01	9.69E-04
"Sn-123	"1.99E-15	2.12E-17
"Te-123m	"1.74E-20	8.62E-23
"Sb-124	"0.00E+00	0.00E+00
"Sb-125	"1.33E+02	1.42E+00
"Te-125m	"3.24E+01	9.30E-02
"Sn-126	"1.50E+00	1.57E-03
"Sb-126	"2.10E-01	1.30E-02
"Sb-126m	"1.50E+00	6.56E-02
"Te-127	"2.46E-18	1.13E-20
"Te-127m	"2.51E-18	4.25E-21
"Te-129	"0.00E+00	0.00E+00
"Te-129m	"0.00E+00	0.00E+00
"I-129	"5.98E-02	9.55E-05
"Cs-134	"7.50E+01	2.61E+00
"Cs-135	"7.01E-01	7.99E-04
"Cs-137	"1.12E+05	3.87E+02
"Ba-137m	"1.06E+05	1.42E+03
"Ce-141	"0.00E+00	0.00E+00
"Pr-143	"0.00E+00	0.00E+00
"Ce-144	"9.18E-03	2.07E-05
"Pr-144	"9.07E-03	2.27E-04
"Pr-144m	"1.10E-04	1.26E-07
"Pm-147	"1.27E+03	1.59E+00
"Pm-148	"0.00E+00	0.00E+00
"Pm-148m	"0.00E+00	0.00E+00

"Sm-151	"1.18E+03	4.71E-01
"Eu-152	"9.80E+00	1.51E-01
"Gd-153	"6.02E-09	1.85E-11
"Eu-154	"1.31E+03	4.00E+01
"Eu-155	"1.23E+02	3.03E-01
"Tb-160	"0.00E+00	0.00E+00
"H-3	"8.78E+00	1.01E-03
"C-14	"6.42E+00	6.40E-03
"Fe-55	"6.28E+00	7.14E-04
"Ni-59	"3.69E-01	5.00E-05
"Co-60	"2.43E+01	1.28E+00
"Ni-63	"4.03E+01	1.39E-02
"Zr-93	"1.17E-01	4.58E-05
"Nb-93m	"7.61E-02	4.65E-05
"Zr-95	"0.00E+00	0.00E+00
"Nb-95	"0.00E+00	0.00E+00
"Nb-95m	"0.00E+00	0.00E+00
"In-113m	"1.58E-18	1.24E-20
"Sn-113	"1.58E-18	8.93E-22
"Sn-119m	"6.37E-07	1.12E-09
"Sn-121m	"4.57E-01	1.64E-03
"Sb-125	"4.32E+00	4.64E-02
"Te-125m	"1.05E+00	3.03E-03
"U-234	"4.46E+00	4.31E-01
"U-235	"1.47E-01	1.36E-02
"U-236	"8.31E-01	7.58E-02
"U-238	"3.84E+00	3.26E-01
"Np-237	"5.40E-01	5.31E-02
"Pu-238	"1.54E+03	1.72E+02
"Pu-239	"2.01E+03	2.09E+02
"Pu-240	"1.59E+03	1.66E+02
"Pu-241	"7.91E+04	8.37E+00
"Pu-242	"1.01E+00	1.01E-01
"Am-241	"5.03E+03	5.63E+02
"Am-242m	"4.32E+00	2.26E-02
"Am-242	"4.30E+00	1.22E-01
"Am-243	"3.23E+00	3.49E-01
"Cm-242	"3.57E+00	4.41E-01
"Cm-244	"5.19E+01	6.09E+00

"06/04/1997 RADNUC2A run for safety basis fuel (40 "
 " Results decayed to 05/31/2038 "
 " Total MTU " 6.9
 " Total Curies " 1.13E+05
 " Total BTU " 1.72E+03

"Isotope"	"Curies"	"BTUs"
"H-3	"2.22E+01	2.54E-03
"C-14	"1.26E-04	1.26E-07
"Se-79	"4.61E-01	4.85E-04
"Kr-85	"2.35E+02	1.20E+00
"Sr-89	"0.00E+00	0.00E+00
"Sr-90	"2.01E+04	7.96E+01
"Y-90	"2.01E+04	3.80E+02
"Y-91	"0.00E+00	0.00E+00
"Zr-93	"2.01E+00	7.89E-04
"Nb-93m	"1.82E+00	1.11E-03
"Zr-95	"0.00E+00	0.00E+00
"Nb-95	"0.00E+00	0.00E+00
"Nb-95m	"0.00E+00	0.00E+00
"Tc-99	"1.54E+01	2.64E-02
"Ru-103	"0.00E+00	0.00E+00
"Rh-103m	"0.00E+00	0.00E+00
"Ru-106	"1.42E-12	2.88E-16
"Rh-106	"1.42E-12	4.62E-14
"Pd-107	"1.11E-01	2.09E-05
"Ag-110	"0.00E+00	0.00E+00
"Ag-110m	"0.00E+00	0.00E+00
"Cd-113m	"3.36E+00	1.26E-02
"Cd-115m	"0.00E+00	0.00E+00
"Sn-119m	"0.00E+00	0.00E+00
"Sn-121m	"9.90E-02	3.55E-04
"Sn-123	"0.00E+00	0.00E+00
"Te-123m	"0.00E+00	0.00E+00
"Sb-124	"0.00E+00	0.00E+00
"Sb-125	"7.33E-03	7.85E-05
"Te-125m	"1.79E-03	5.14E-06
"Sn-126	"9.16E-01	9.60E-04
"Sb-126	"1.28E-01	7.92E-03
"Sb-126m	"9.16E-01	4.00E-02
"Te-127	"0.00E+00	0.00E+00
"Te-127m	"0.00E+00	0.00E+00
"Te-129	"0.00E+00	0.00E+00
"Te-129m	"0.00E+00	0.00E+00
"I-129	"3.65E-02	5.83E-05
"Cs-134	"1.74E-04	6.07E-06
"Cs-135	"4.27E-01	4.87E-04
"Cs-137	"2.89E+04	9.96E+01
"Ba-137m	"2.73E+04	3.66E+02
"Ce-141	"0.00E+00	0.00E+00
"Pr-143	"0.00E+00	0.00E+00
"Ce-144	"0.00E+00	0.00E+00
"Pr-144	"0.00E+00	0.00E+00
"Pr-144m	"0.00E+00	0.00E+00
"Pm-147	"4.12E-02	5.16E-05
"Pm-148	"0.00E+00	0.00E+00
"Pm-148m	"0.00E+00	0.00E+00

"Sm-151	"5.29E+02	2.11E-01
"Eu-152	"9.09E-01	1.40E-02
"Gd-153	"0.00E+00	0.00E+00
"Eu-154	"4.09E+01	1.25E+00
"Eu-155	"4.10E-01	1.01E-03
"Tb-160	"0.00E+00	0.00E+00
"H-3	"6.79E-01	7.78E-05
"C-14	"3.89E+00	3.88E-03
"Fe-55	"1.88E-04	2.14E-08
"Ni-59	"2.24E-01	3.04E-05
"Co-60	"1.11E-01	5.82E-03
"Ni-63	"1.85E+01	6.39E-03
"Zr-93	"7.10E-02	2.79E-05
"Nb-93m	"6.43E-02	3.93E-05
"Zr-95	"0.00E+00	0.00E+00
"Nb-95	"0.00E+00	0.00E+00
"Nb-95m	"0.00E+00	0.00E+00
"In-113m	"0.00E+00	0.00E+00
"Sn-113	"0.00E+00	0.00E+00
"Sn-119m	"0.00E+00	0.00E+00
"Sn-121m	"1.66E-01	5.96E-04
"Sb-125	"2.38E-04	2.55E-06
"Te-125m	"5.81E-05	1.67E-07
"U-234	"2.63E+00	2.54E-01
"U-235	"8.61E-02	7.96E-03
"U-236	"4.96E-01	4.53E-02
"U-238	"2.27E+00	1.93E-01
"Np-237	"3.29E-01	3.23E-02
"Pu-238	"7.20E+02	8.01E+01
"Pu-239	"1.21E+03	1.26E+02
"Pu-240	"9.75E+02	1.02E+02
"Pu-241	"8.22E+03	8.69E-01
"Pu-242	"6.48E-01	6.44E-02
"Am-241	"4.26E+03	4.77E+02
"Am-242m	"2.37E+00	1.24E-02
"Am-242	"2.36E+00	6.70E-02
"Am-243	"2.13E+00	2.30E-01
"Cm-242	"1.96E+00	2.42E-01
"Cm-244	"8.55E+00	1.00E+00

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APPENDIX C

HALF-LIFE, SPECIFIC ACTIVITY, AND SPECIFIC HEAT FOR SELECTED RADIONUCLIDES

Specific Activity and Decay Heat Generation Rates for Selected Radionuclides.						
Radio-nuclide	Atomic Weight	Half-life		Specific Activity (Ci/g)	Decay Heat Generation Rate	
					(W/Ci)	(W/g)
H-3	3	3.897e+08	s	9.651e+03	3.367e-05	3.250e-01
C-14	14	1.808e+11	s	4.458e+00	2.933e-04	1.307e-03
Fe-55	55	2.600e+00	y	2.500e+03	3.379e-05	8.448e-02
Co-60	60	1.663e+08	s	1.131e+03	1.542e-02	1.744e+01
Ni-59	59	8.000e+01	ky	7.575e-02	3.972e-05	3.009e-06
Ni-63	63	9.200e+01	y	6.169e+01	3.972e-04	2.450e-02
Se-79	79	2.050e+12	s	6.967e-02	2.490e-04	1.735e-05
Kr-85	85	3.383e+08	s	3.924e+02	1.498e-03	5.878e-01
Sr-89	89	4.363e+06	s	2.906e+04	3.457e-03	1.005e+02
Sr-90	90	9.190e+08	s	1.364e+02	1.161e-03	1.583e-01
Y-90	90	2.304e+05	s	5.441e+05	5.543e-03	3.016e+03
Y-91	91	5.055e+06	s	2.453e+04	3.592e-03	8.810e+01
Zr-93	93	4.828e+13	s	2.513e-03	1.162e-04	2.920e-07
Zr-95	95	5.528e+08	s	2.148e+04	5.066e-03	1.088e+02
Nb-93m	93	4.292e+08	s	2.827e+02	1.772e-04	5.009e-02
Nb-95	95	3.037e+06	s	3.911e+04	4.797e-03	1.876e+02
Nb-95m	95	3.118e+05	s	3.809e+05	1.390e-03	5.293e+02
Tc-99	99	6.722e+12	s	1.695e-02	5.015e-04	8.503e-06
Ru-103	103	3.394e+06	s	3.228e+04	3.346e-03	1.080e+02
Ru-106	106	3.181e+07	s	3.346e+03	5.946e-05	1.990e-01
Rh-103m	103	3.367e+03	s	3.253e+07	2.302e-04	7.489e+03
Rh-106	106	2.990e+01	s	3.560e+09	9.592e-03	3.415e+07
Pd-107	107	2.050e+14	s	5.144e-04	5.928e-05	3.049e-08
Ag-110	110	2.460e+01	s	4.170e+09	7.185e-03	2.996e+07
Ag-110m	110	2.159e+07	s	4.751e+03	1.670e-02	7.934e+01
Cd-113m	113	4.604e+08	s	2.169e+02	1.684e-03	3.651e-01
Cd-115m	115	3.853e+06	s	2.546e+04	3.730e-03	9.498e+01
In-113m	113	5.969e+03	s	1.673e+07	2.330e-03	3.897e+04
Sn-113	113	9.945e+06	s	1.004e+04	1.666e-04	1.672e+00
Sn-119m	119	2.117e+07	s	4.479e+03	5.169e-04	2.315e+00
Sn-121m	121	1.577e+09	s	5.913e+01	2.004e-03	1.185e-01
Sn-123	123	1.116e+07	s	8.220e+03	3.124e-03	2.567e+01
Sn-126	126	3.156e+12	s	2.837e-02	1.247e-03	3.539e-05
Sb-124	124	5.201e+06	s	1.750e+04	1.328e-02	2.323e+02
Sb-125	125	8.741e+07	s	1.033e+03	3.126e-03	3.229e+00
Sb-126	126	1.071e+06	s	8.361e+04	1.848e-02	1.545e+03
Sb-126m	126	1.140e+03	s	7.855e+07	1.273e-02	1.000e+06
Te-125m	125	5.011e+06	s	1.801e+04	8.406e-04	1.514e+01
Te-127	127	3.366e+04	s	2.639e+06	1.350e-03	3.564e+03
Te-127m	127	9.418e+06	s	9.433e+03	5.379e-04	5.074e+00

Specific Activity and Decay Heat Generation Rates for Selected Radionuclides.						
Radio-nuclide	Atomic Weight	Half-life		Specific Activity (Ci/g)	Decay Heat Generation Rate	
					(W/Ci)	(W/g)
Te-129	129	4.176e+03	s	2.094e+07	3.573e-03	7.483e+04
Te-129m	129	2.903e+06	s	3.013e+04	1.754e-03	5.283e+01
I-129	129	4.954e+14	s	1.766e-04	4.626e-04	8.168e-08
Cs-134	134	6.507e+07	s	1.294e+03	1.018e-02	1.317e+01
Cs-135	135	7.258e+13	s	1.152e-03	3.338e-04	3.843e-07
Cs-137	137	9.467e+08	s	8.699e+01	1.106e-03	9.623e-02
Ba-137m	137	1.531e+02	s	5.379e+08	3.927e-03	2.112e+06
Ce-141	141	2.809e+06	s	2.849e+04	1.464e-03	4.171e+01
Ce-144	144	2.456e+07	s	3.190e+03	6.634e-04	2.116e+00
Pr-143	143	1.172e+06	s	6.732e+04	1.863e-03	1.254e+02
Pr-144	144	1.037e+03	s	7.556e+07	7.351e-03	5.554e+05
Pr-144m	144	4.320e+02	s	1.814e+08	3.422e-04	6.206e+04
Pm-147	147	8.279e+07	s	9.271e+02	3.587e-04	3.326e-01
Pm-148	148	4.640e+05	s	1.643e+05	7.701e-03	1.265e+03
Pm-148m	148	3.568e+06	s	2.137e+04	1.268e-02	2.709e+02
Sm-151	151	2.840e+09	s	2.631e+01	1.173e-04	3.085e-03
Eu-152	152	4.292e+08	s	1.730e+02	7.564e-03	1.308e+00
Eu-154	154	2.714e+08	s	2.700e+02	8.946e-03	2.415e+00
Eu-155	155	1.565e+08	s	4.651e+02	7.274e-04	3.383e-01
Gd-153	153	2.091e+07	s	3.527e+03	9.034e-04	3.186e+00
Tb-160	160	6.247e+06	s	1.129e+04	8.145e-03	9.195e+01
U-234	234	7.716e+12	s	6.249e-03	2.880e-02	1.800e-04
U-235	235	2.221e+16	s	2.162e-06	2.619e-02	5.662e-08
U-236	236	7.389e+14	s	6.470e-05	2.709e-02	1.753e-06
U-238	238	1.410e+17	s	3.362e-07	2.537e-02	8.529e-09
Np-237	237	6.753e+13	s	7.050e-04	3.057e-02	2.155e-05
Pu-238	238	2.769e+09	s	1.712e+01	3.314e-02	5.675e-01
Pu-239	239	7.594e+11	s	6.217e-02	3.082e-02	1.916e-03
Pu-240	240	2.063e+11	s	2.279e-01	3.114e-02	7.096e-03
Pu-241	241	4.544e+08	s	1.030e+02	3.100e-05	3.194e-03
Pu-242	242	1.221e+13	s	3.819e-03	2.953e-02	1.128e-04
Am-241	241	1.364e+10	s	3.432e+00	3.322e-02	1.140e-01
Am-242	242	5.767e+04	s	8.085e+05	1.135e-03	9.178e+02
Am-242m	242	4.797e+09	s	9.719e+00	3.950e-04	3.840e-03
Am-243	243	2.329e+11	s	1.994e-01	3.215e-02	6.409e-03
Cm-242	242	1.410e+07	s	3.307e+03	3.685e-02	1.218e+02
Cm-244	244	5.715e+08	s	8.091e+01	3.498e-02	2.830e+00
Time abbreviations are as follows: "Gy" gigayears, "My" megayears, "Ky" kiloyears, "y" years, "d" days, "h" hours, "m" minutes, and "s" seconds.						

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APPENDIX D
ENABLING ASSUMPTIONS

ENABLING ASSUMPTIONS

The following is a list of the assumptions that were made to enable the development of this document. This list indicates the limitations of the data that have been developed. A projected method of validation is presented for each assumption. It is anticipated that a decision will be made on the need to validate each assumption, comparing the cost of validation with the uncertainty associated with retaining the assumption invalidated.

- Nuclear Accountability Database

The assumption is that the database is an accurate reflection of the fuel that is stored in the K Basins. There are at least two such databases: the one used here, and another known as Reconstructed SCATS which was developed in parallel with the Nuclear Accountability database. The Reconstructed SCATS database was used in the development of the safety analysis which was done totally independently of the feed basis. The results of feed development in this document are within about 10% of those presented in the safety documentation for ^{137}Cs and ^{241}Am , two important isotopes for shielding and safety basis calculations.

The Nuclear Accountability database was used because it represents the official accepted numbers for tracking special nuclear materials. Because the Nuclear Accountability data contains the official plutonium inventory for the Hanford site, the plutonium isotopes reported in tables 2.2, 2.4, 3.5, 3.6, and 3.7 come directly from the database (WHC 1996).

For the purpose of this development, the most important aspects of the database are: the type of fuel (Mark 1A or Mark IV); the date irradiation of the fuel ended, listed here as a key date (date of discharge from the reactor); the percentage ^{240}Pu in the fuel; and the mass of uranium in the fuel. Validation of the database, for these factors, could be obtained by direct measurement of the beta/gamma energy levels emitted by the fuel and the use those energies to estimate concentrations of particular nuclides. With knowledge of the nuclide concentrations in the fuel, time since discharge and ^{240}Pu content could be estimated accurately. The mass of uranium, at least for the undamaged elements, could be obtained by weighing the fuel, and then adjusting the weight to account for the cladding, fission, activation, and actinide products. This activity would be difficult and costly.

- ORIGEN2

The assumption is that ORIGEN2 accurately predicts fission, activation, and actinide products in irradiated N-Reactor fuel, when correct input information is used. There is reason to suspect that some impurities in the fuel were inadequately defined, and that activation products from those impurities may become important for shielding and/or safety basis definition.

The ^{60}Co amounts were specifically changed in RADNOC from those predicted in ORIGEN2 because of higher than expected ^{60}Co concentrations in tank waste samples. It is important to note that radionuclides in the tank waste originated in irradiated N Reactor fuel processed at the PUREX Plant. Subsequent samples showed somewhat lower cobalt concentrations, and RADNOC was changed to reflect these lower numbers. In the development work done here, the ^{60}Co concentrations as predicted by RADNOC produced about half the dose rates resulting from the ^{137}Cs and therefore shielding basis is derived by cesium, and not cobalt. For the present the shielding design basis should be as listed here.

Another potentially important isotope is ^{14}C which results from the activation of nitrogen in the fuel or cladding, and as a fission product. Activation of nitrogen produces orders of magnitude more ^{14}C than fission. As in the case of cobalt, the ^{14}C amounts in RADNUC were changed to reflect higher than expected concentrations in tank waste samples.

Validation of ORIGEN2 for N Reactor would need to be done by analysis of a statistically meaningful sampling of irradiated N Reactor fuels. The analysis results would need to be compared with ORIGEN2 predictions.

Levels of activation products in the fuel and cladding will depend on impurity levels in the unirradiated fuel. Since the fuel was fabricated using many different shipments of uranium and zircaloy 2, it is possible that impurity levels varied significantly over time.

- RADNUC

The assumption is that RADNUC accurately predicts nuclide concentrations in N Reactor fuel. RADNUC was originally written to predict nuclides in tank waste. The nuclides in tank waste however, originated in the irradiated fuel. Therefore, RADNUC was written to determine fission, activation, and actinide products in the fuel. As originally written, RADNUC gave the user the option of removing some of the actinides and fission products to simulate processing activities at PUREX and other facilities. RADNUC was recently changed to allow the user to specify fuel or tank waste in the case of fuel, no nuclides are removed.

RADNUC uses ORIGEN2 runs for Mark 1A and Mark IV N Reactor fuel as the basis for estimating radionuclide amounts. Output from ORIGEN2 runs for 6%, 9%, 12%, and 16% ^{240}Pu are included in the RADNUC code. It is important to note that although the runs are referred to as 6%, 9%, 12%, and 16% ^{240}Pu runs, ORIGEN2 input files only include burnup information and were iterated to arrive at the desired ^{240}Pu content. In the case of the "16% ^{240}Pu " run the actual ^{240}Pu content is only slightly over 15% at discharge and actually drops below 15% at 120 days after discharge. This apparent weakness in the available ORIGEN2 runs was factored into the RADNUC code however, and should not present a problem in the predictions.

RADNUC reads the % ^{240}Pu content from the input data file and interpolates the radionuclide per MTU content from the 2 nearest ORIGEN2 runs. For ^{240}Pu contents between 0% and 6% a direct linear interpolation scheme is used, for ^{240}Pu contents between 6% and 15% the log of ^{240}Pu content is interpolated, and for ^{240}Pu contents above 15%, the log of the ^{240}Pu content is used to extrapolate the nuclides. Each of the 88 isotopes are interpolated or extrapolated for each entry in the input data file. Each of the nuclides are summed to arrive at total for each of the 88 for the entire data file. For fission and activation products, the concentration of nuclides in the fuel is nearly linear with ^{240}Pu content, and therefore should be predicted well by RADNUC. Actinides are somewhat less linear, and therefore accurate prediction by interpolation is limited to the range well covered by the available data. Fortunately, the maximum ^{240}Pu content is only slightly above 16%.

RADNUC has been validated by comparison with ORIGEN2 runs for identical ^{240}Pu content fuels. This validation was performed and documented as part of the changes to the RADNUC code.

- Fuel vs Sludge

The assumption is that all the fuel in the K Basins is intact, and contains all of the nuclides within its cladding. In fact this assumption is not correct, much of the fuel in the basins has damaged, and in some cases severely damaged, cladding which has allowed the uranium to corrode and release fission, activation, and actinide products into the surrounding basin as either gases, soluble materials, or solids (sludge). Some of this material (^{85}Kr and ^3H) has been released to the environment, and some has been removed from the basin water in the treatment systems (filtration and ion exchange). It is difficult to account for the exact origin of the radionuclides in the basin sludge, filtration systems, ion exchange columns, and other basin systems because of the long history of fuel storage in the basins.

Extensive analyses would be required to determine the origin of radionuclides in: sludge or other basin systems; ion exchange columns; or environmental releases. These analyses are not recommended because of the high cost and small likelihood that any such results would cause significant changes to fuel storage or treatment systems. A conservative approach is to assume that the radionuclides in basin sludge, basin water systems, and any which may have been released to the environment, originated in fuel not currently stored in the basins. At the same time the most conservative assumption is that all nuclides which originated in the fuel currently in storage remain with the fuel.

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APPENDIX E

CALCULATION OF SLUDGE COMPONENT INVENTORY

APPENDIX E

CALCULATION OF SLUDGE COMPONENT INVENTORY

Table E-1 contains the raw sample data from 105-K East Basin samples (Bechtold 1993). The unit conversion columns show the change in units for further use.

Table E-2 shows conversion of the raw data into a sludge component inventory by multiplying the component weight fraction (obtained from Table E-1) by the total sludge mass and solids weight fraction in a Basin area from which the sample was obtained.

Since limited sample data were available, extrapolation was required to obtain sludge component inventory for the remaining pits and fuel canisters. Since the Weasel Pit sludge contains the highest amount of radionuclides due to sludge pumping activities, the sludge in this area is used to estimate the component inventory in the Tech View Pit, Wash Pit, South Loadout Pit, and breached fuel canisters. For the empty fuel canisters and fuel canisters with unbreached fuel, the Basin Floor sludge composition was used to model the sludge in these areas. These calculated values are provided in Table E-3.

Similar information on the radionuclide inventory in the sludge is provided in Tables E-4 through E-6. The final sludge radionuclide inventory calculations are the same as described above for the chemical components.

TABLE E-1: 105-KE BASIN SLUDGE SAMPLES (CONSTITUENTS)

	West Side Weasel Pit Screen Sample # S3-032-01	East Side Weasel Pit Screen Sample # S3-032-02	WEASEL PIT (Average of sample #'s S3-032-01/02 (micro g/g)	WEASEL PIT Unit Conversion (g/g)	KE BASIN CENTER grate# 3718 Sample # S3-032-03 (micro g/g)	KE BASIN CENTER Unit Conversion (g/g)
Al	2.43E+04	1.23E+04	1.83E+04	1.83E-02	2.08E+04	2.08E-02
As	4.14E+01	4.57E+01	4.36E+01	4.36E-05	1.66E+01	1.66E-05
Ba	1.16E+02	6.73E+01	9.17E+01	9.17E-05	5.37E+01	5.37E-05
Be	3.93E+00	3.09E+00	3.51E+00	3.51E-06	2.48E+01	2.48E-05
Ca	4.38E+03	2.46E+03	3.42E+03	3.42E-03	6.93E+02	6.93E-04
Cd	9.59E+01	5.82E+01	7.71E+01	7.71E-05	5.92E+01	5.92E-05
Co	3.03E+01	1.38E+01	2.21E+01	2.21E-05	8.74E+00	8.74E-06
Cr	1.18E+03	6.71E+02	9.26E+02	9.25E-04	2.38E+02	2.38E-04
Cu	3.69E+02	2.02E+02	2.86E+02	2.86E-04	2.13E+02	2.13E-04
Fe	2.85E+05	2.00E+05	2.43E+05	2.42E-01	7.35E+04	7.35E-02
K	2.00E+02	1.21E+02	1.60E+02	1.60E-04	5.02E+01	5.02E-05
Li	3.98E+00	1.67E+00	2.83E+00	2.83E-06	1.13E+00	1.13E-06
Mg	1.07E+03	6.11E+02	8.41E+02	8.40E-04	3.00E+02	3.00E-04
Mn	3.63E+02	2.40E+02	3.02E+02	3.02E-04	1.56E+02	1.56E-04
Mo	4.57E+01	2.83E+01	3.70E+01	3.70E-05	1.50E+01	1.50E-05
Na	1.50E+02	8.36E+01	1.17E+02	1.17E-04	4.40E+01	4.40E-05
Nd	1.87E+02	7.47E+01	1.31E+02	1.31E-04	6.70E+01	6.70E-05
Ni	1.14E+02	7.25E+01	9.33E+01	9.32E-05	4.04E+01	4.04E-05
P	5.02E+02	3.27E+02	4.15E+02	4.14E-04	1.91E+02	1.91E-04
Pb	6.18E+02	3.56E+02	4.87E+02	4.87E-04	1.95E+02	1.95E-04
S	3.44E+02	3.03E+02	3.24E+02	3.24E-04	2.14E+02	2.14E-04
Sb	3.18E+02	1.15E+02	2.17E+02	2.16E-04	8.79E+01	8.79E-05
Si	1.43E+03	1.41E+03	1.42E+03	1.42E-03	1.99E+03	1.99E-03
Sm	6.65E+02	2.80E+02	4.72E+02	4.72E-04	2.25E+02	2.25E-04
Sr	4.46E+01	2.17E+01	3.31E+01	3.31E-05	1.14E+01	1.14E-05
Ti	1.74E+02	1.08E+02	1.41E+02	1.41E-04	4.49E+01	4.49E-05
Tl	2.17E+02	7.05E+01	1.44E+02	1.44E-04	1.00E+02	1.00E-04
V	5.27E+02	2.12E+02	3.70E+02	3.69E-04	1.65E+02	1.65E-04
Zn	1.25E+03	7.37E+02	9.94E+02	9.94E-04	3.53E+02	3.53E-04
Zr	2.22E+02	8.11E+01	1.52E+02	1.52E-04	1.48E+02	1.48E-04
U	1.09E+05	4.04E+04	7.47E+04	7.47E-02	3.50E+04	3.50E-02
Other				6.53E-01		8.65E-01

Sample #'s S3-032-01 & S3-032-02 averaged
in order to obtain final Weasel Pit values

TABLE E-2: 105-KE BASIN CONSTITUENT TOTALS FROM SAMPLE FRACTIONS.

	WEASEL PIT		KE BASIN CENTER	
	Sample Fraction	Mass=18.20 Mg solids=51.29%	Sample Fraction	Mass=24.46 Mg solids=17.52%
	(g/g)	Kg Total	(g/g)	Kg Total
Al	1.83E-02	1.71E+02	2.08E-02	8.91E+01
As	4.34E-05	4.05E-01	1.66E-05	7.11E-02
Ba	9.17E-05	8.56E-01	5.37E-05	2.30E-01
Be	3.51E-06	3.28E-02	2.48E-05	1.06E-01
Ca	3.42E-03	3.19E+01	6.93E-04	2.97E+00
Cd	7.71E-05	7.19E-01	5.92E-05	2.54E-01
Co	2.21E-05	2.06E-01	8.74E-06	3.75E-02
Cr	9.25E-04	8.64E+00	2.38E-04	1.02E+00
Cu	2.86E-04	2.67E+00	2.13E-04	9.13E-01
Fe	2.42E-01	2.26E+03	7.35E-02	3.15E+02
K	1.60E-04	1.50E+00	5.02E-05	2.15E-01
Li	8.55E-05	7.98E-01	1.13E-06	4.84E-03
Mg	8.40E-04	7.85E+00	3.00E-04	1.29E+00
Mn	3.02E-04	2.81E+00	1.56E-04	6.69E-01
Mo	3.70E-05	3.45E-01	1.50E-05	6.43E-02
Na	1.17E-04	1.09E+00	4.40E-05	1.89E-01
Nd	1.31E-04	1.22E+00	6.70E-05	2.87E-01
Ni	9.32E-05	8.70E-01	4.04E-05	1.73E-01
P	4.14E-04	3.87E+00	1.91E-04	8.19E-01
Pb	4.87E-04	4.55E+00	1.95E-04	8.36E-01
S	3.24E-04	3.02E+00	2.14E-04	9.17E-01
Sb	2.16E-04	2.02E+00	8.79E-05	3.77E-01
Si	1.42E-03	1.33E+01	1.99E-03	8.53E+00
Sm	4.72E-04	4.41E+00	2.25E-04	9.64E-01
Sr	3.31E-05	3.09E-01	1.14E-05	4.89E-02
Ti	1.41E-04	1.32E+00	4.49E-05	1.92E-01
Tl	1.44E-04	1.34E+00	1.00E-04	4.29E-01
V	3.69E-04	3.45E+00	1.65E-04	7.07E-01
Zn	9.94E-04	9.27E+00	3.53E-04	1.51E+00
Zr	1.52E-04	1.41E+00	1.48E-04	6.34E-01
U	7.47E-02	6.97E+02	3.50E-02	1.50E+02
Other	6.53E-01	6.09E+03	8.65E-01	3.71E+03
TOTAL/ DRY		9.33E+03		4.29E+03
Interstitial Water		8.87E+03		2.02E+04
TOTAL/ WET		1.82E+04		2.45E+04

TABLE E-3: 105-KE BASIN CONSTITUENT TOTALS FROM EXTRAPOLATION

	TECH VIEW PIT	WASH PIT	SOUTH LOADOUT PIT	FUEL CANISTERS			FUEL CANISTERS
	Mass=0.66 Mg solids=51.29%	Mass=2.11 Mg solids=51.29%	Mass=1.67 Mg solids=51.29%	EMPTY Mass= 0.44 Mg solids=17.52%	FULL, GOOD Mass= 0.92 Mg solids=17.52%	FULL, BREECH Mass= 10.19 Mg solids=51.29%	TOTAL
	Kg Total	Kg Total	Kg Total	Kg Total	Kg Total	Kg Total	Kg Total
Al	6.19E+00	1.98E+01	1.57E+01	1.60E+00	3.35E+00	9.56E+01	1.01E+02
As	1.47E-02	4.70E-02	3.72E-02	1.28E-03	2.68E-03	2.27E-01	2.31E-01
Ba	3.10E-02	9.92E-02	7.85E-02	4.14E-03	8.66E-03	4.79E-01	4.92E-01
Be	1.19E-03	3.80E-03	3.01E-03	1.91E-03	4.00E-03	1.83E-02	2.43E-02
Ca	1.16E+00	3.70E+00	2.93E+00	5.34E-02	1.12E-01	1.79E+01	1.80E+01
Cd	2.61E-02	8.34E-02	6.60E-02	4.56E-03	9.54E-03	4.03E-01	4.17E-01
Co	7.46E-03	2.39E-02	1.89E-02	6.74E-04	1.41E-03	1.15E-01	1.17E-01
Cr	3.13E-01	1.00E+00	7.93E-01	1.83E-02	3.84E-02	4.84E+00	4.89E+00
Cu	9.66E-02	3.09E-01	2.45E-01	1.64E-02	3.43E-02	1.49E+00	1.54E+00
Fe	8.21E+01	2.62E+02	2.08E+02	5.67E+00	1.18E+01	1.27E+03	1.28E+03
K	5.43E-02	1.74E-01	1.37E-01	3.87E-03	8.09E-03	8.39E-01	8.51E-01
Li	2.89E-02	9.25E-02	7.32E-02	8.71E-05	1.82E-04	4.47E-01	4.47E-01
Mg	2.85E-01	9.10E-01	7.20E-01	2.31E-02	4.84E-02	4.39E+00	4.46E+00
Mn	1.02E-01	3.26E-01	2.58E-01	1.20E-02	2.51E-02	1.58E+00	1.61E+00
Mo	1.25E-02	4.00E-02	3.17E-02	1.16E-03	2.42E-03	1.93E-01	1.97E-01
Na	3.95E-02	1.26E-01	1.00E-01	3.39E-03	7.09E-03	6.10E-01	6.21E-01
Nd	4.43E-02	1.42E-01	1.12E-01	5.16E-03	1.08E-02	6.84E-01	7.00E-01
Ni	3.16E-02	1.01E-01	7.99E-02	3.11E-03	6.51E-03	4.87E-01	4.97E-01
P	1.40E-01	4.49E-01	3.55E-01	1.47E-02	3.08E-02	2.17E+00	2.21E+00
Pb	1.65E-01	5.27E-01	4.17E-01	1.50E-02	3.14E-02	2.55E+00	2.59E+00
S	1.10E-01	3.50E-01	2.77E-01	1.65E-02	3.45E-02	1.69E+00	1.74E+00
Sb	7.33E-02	2.34E-01	1.85E-01	6.78E-03	1.42E-02	1.13E+00	1.15E+00
Si	4.81E-01	1.54E+00	1.22E+00	1.53E-01	3.21E-01	7.42E+00	7.90E+00
Sm	1.60E-01	5.11E-01	4.05E-01	1.73E-02	3.63E-02	2.47E+00	2.52E+00
Sr	1.12E-02	3.59E-02	2.84E-02	8.79E-04	1.84E-03	1.73E-01	1.76E-01
Ti	4.77E-02	1.53E-01	1.21E-01	3.46E-03	7.24E-03	7.37E-01	7.48E-01
Tl	4.87E-02	1.56E-01	1.23E-01	7.71E-03	1.61E-02	7.51E-01	7.75E-01
V	1.25E-01	4.00E-01	3.16E-01	1.27E-02	2.66E-02	1.93E+00	1.97E+00
Zn	3.36E-01	1.08E+00	8.51E-01	2.72E-02	5.69E-02	5.19E+00	5.28E+00
Zr	5.13E-02	1.64E-01	1.30E-01	1.14E-02	2.39E-02	7.92E-01	8.27E-01
U	2.53E+01	8.08E+01	6.40E+01	2.70E+00	5.64E+00	3.90E+02	3.99E+02
Other	2.21E+02	7.06E+02	5.59E+02	6.67E+01	1.39E+02	3.41E+03	3.62E+03
TOTAL/ DRY	3.39E+02	1.08E+03	8.57E+02	7.71E+01	1.61E+02	5.23E+03	5.46E+03
Interstitial Water	3.21E+02	1.03E+03	8.13E+02	3.63E+02	7.59E+02	4.96E+03	6.09E+03
TOTAL/ WET	6.60E+02	2.11E+03	1.67E+03	4.40E+02	9.20E+02	1.02E+04	1.16E+04

Tech View Pit, Wash Pit, South Loadout Pit, and Full/Breach Fuel Canisters use Weasel Pit sample fraction numbers. The Empty Fuel Canisters and Full/Good Fuel Canisters use KE Basin Center sample fraction numbers. Weasel Pit and KE Basin Center sample fraction numbers found in Appendix Table E-2.

TABLE E-4: 105-KE BASIN SLUDGE SAMPLES (RADIONUCLIDES)

	West side Weasel Pit screen Sample # S3-032-01	East side Weasel Pit screen Sample # S3-032-02	Weasel Pit (Average of sample #'s S3-032-01/02	WEASEL PIT Unit Conversion	KE Basin Center grate# 3718 Sample # S3-032-03	KE BASIN CENTER Unit Conversion
	micro Ci/g	micro Ci/g	micro Ci/g	Ci/g	micro Ci/g	Ci/g
Co 60	1.41E+00	9.50E-01	1.18E+00	1.18E-06	1.63E+00	1.63E-06
Sr 90	8.81E+01	3.00E+01	5.90E+01	5.90E-05	7.48E+01	7.48E-05
Y 90	8.81E+01	3.00E+01	5.90E+01	5.90E-05	7.48E+01	7.48E-05
Cs 137	4.28E+01	4.75E+01	4.51E+01	4.51E-05	4.15E+01	4.15E-05
Ba 137m	4.05E+01	4.49E+01	4.27E+01	4.27E-05	3.93E+01	3.93E-05
Eu 154	1.56E+00	9.30E-01	1.25E+00	1.25E-06	1.66E+00	1.66E-06
Eu 155	9.28E-01	4.80E-01	7.04E-01	7.04E-07	9.72E-01	9.72E-07
Pu 239,240	6.83E+00	4.24E+00	5.54E+00	5.53E-06	5.87E+00	5.87E-06
Am 241	7.44E+00	4.61E+01	2.68E+01	2.68E-05	6.88E+01	6.88E-05
TOTAL						

Sample #'s S3-032-01 & S3-032-02 averaged
in order to obtain final Weasel Pit values

TABLE E-5: 105-KE BASIN RADIONUCLIDE TOTALS FROM SAMPLE FRACTIONS

	WEASEL PIT			KE BASIN CENTER	
	Sample Fraction	Mass=18.20 Mg solids=51.29%		Sample Fraction	Mass=24.46 Mg solids=17.52%
	(Ci/g)	Ci total		(Ci/g)	Ci total
Co 60	1.18E-06	1.10E+01		1.63E-06	6.99E+00
Sr 90	5.90E-05	5.51E+02		7.48E-05	3.21E+02
Y 90	5.90E-05	5.51E+02		7.48E-05	3.21E+02
Cs 137	4.51E-05	4.21E+02		4.15E-05	1.78E+02
Ba 137m	4.27E-05	3.99E+02		3.93E-05	1.68E+02
Eu 154	1.25E-06	1.16E+01		1.66E-06	7.11E+00
Eu 155	7.04E-07	6.57E+00		9.72E-07	4.17E+00
Pu 239,240	5.53E-06	5.17E+01		5.87E-06	2.52E+01
Am 241	2.68E-05	2.50E+02		6.88E-05	2.95E+02
TOTAL		2.25E+03			1.33E+03

TABLE E-6: 105-KE BASIN RADIONUCLIDE TOTALS FROM EXTRAPOLATION

	TECH VIEW PIT	WASH PIT	SOUTH LOADOUT PIT	-----FUEL CANISTERS-----			FUEL CANISTERS
	Mass=0.66 Mg solids=51.29%	Mass=2.11 Mg solids=51.29%	Mass=1.67 Mg solids=51.29%	EMPTY Mass=0.44 Mg solids=17.52%	FULL, GOOD Mass=0.92 Mg solids=17.52%	FULL, BREECH Mass=10.19 Mg solids=51.29%	TOTAL
	Ci total	Ci total	Ci total	Ci total	Ci total	Ci total	Ci total
H 3							
Co 60	3.99E-01	1.28E+00	1.01E+00	1.26E-01	2.63E-01	6.17E+00	6.56E+00
Sr 90	2.00E+01	6.39E+01	5.06E+01	5.77E+00	1.21E+01	3.09E+02	3.26E+02
Y 90	2.00E+01	6.39E+01	5.06E+01	5.77E+00	1.21E+01	3.09E+02	3.26E+02
Cs 137	1.53E+01	4.89E+01	3.87E+01	3.20E+00	6.69E+00	2.36E+02	2.46E+02
Ba 137m	1.45E+01	4.62E+01	3.66E+01	3.03E+00	6.33E+00	2.23E+02	2.33E+02
Eu 154	4.21E-01	1.35E+00	1.07E+00	1.28E-01	2.68E-01	6.51E+00	6.90E+00
Eu 155	2.38E-01	7.62E-01	6.03E-01	7.49E-02	1.57E-01	3.68E+00	3.91E+00
Pu 239,240	1.87E+00	5.99E+00	4.74E+00	4.53E-01	9.46E-01	2.89E+01	3.03E+01
Am 241	9.06E+00	2.90E+01	2.29E+01	5.30E+00	1.11E+01	1.40E+02	1.56E+02
TOTAL	8.17E+01	2.61E+02	2.07E+02				1.34E+03

Tech View Pit, Wash Pit, South Loadout Pit, and Full/Breech Fuel Canisters use Weasel Pit sample fraction numbers.

The Empty Fuel Canisters and Full/Good Fuel Canisters use KE Basin Center sample fraction numbers.

Weasel Pit and KE Basin Center sample fraction numbers found in Appendix Table E-5.

APPENDIX F

SAFETY/REGULATORY ASSESSMENT BASIS METHODOLOGY

The Safety/Regulatory assessment basis was selected by estimating a relative dose commitment to people resulting from an accidental release. The dose rates were estimated using "*Internal Dose Conversion Factors for Calculation of Dose to the Public*" (DOE 1988), which includes tables for radionuclides. The tables give the dose rate per unit intake of the radionuclide for ingestion and inhalation pathways to various organs and a total dose known as a cumulative effective dose equivalent or CEDE.

For a release of irradiated fuel, the total dose to people per unit mass released, is dominated by only a few nuclides. The dose results from a product of the activity/unit mass of fuel, and the CEDE (REM/unit activity). ^{241}Am , with a CEDE of 520 rem/ μCi for the inhalation pathway and a total of 315,000 Ci in the K Basins, results in the largest potential dose from a single nuclide in a release. Plutonium isotopes, with CEDE's of 330 rem/ μCi for ^{239}Pu and ^{240}Pu ; 300 rem/ μCi for ^{238}Pu ; and 5.7 rem/ μCi for ^{241}Pu , and totals of 225,000 Ci of ^{239}Pu , 130,000 Ci ^{240}Pu , 125,000 Ci ^{238}Pu , and 7,380,000 Ci ^{241}Pu in the K Basins, also result in significant potential dose in comparison to ^{241}Am . ^{90}Sr , with a CEDE of 1.3 rem/ μCi , and a total of 10,500,000 Ci in the K Basins, also results in a significant potential dose in comparison with ^{241}Am . ^{137}Cs , the isotope with the largest activity (13,500,000 Ci in the K Basins), and with a CEDE of 0.032 rem/ μCi results in a dose about 2.5 orders of magnitude less than the ^{241}Am . The dose resulting from other nuclides in the fuel is negligible in comparison to those above because of their lower abundance and smaller CEDE.

The fuel in the K basins containing the mix of these isotopes per unit mass of fuel with the largest potential dose consequence is used to establish a safety envelope. ^{241}Am with a half life of approximately 433 years results in the fuel primarily from the decay of ^{241}Pu . Because of the difference in half lives, ^{241}Pu has a half life of approximately 14 years, the concentration of ^{241}Am in the fuel increases for approximately 60 years following discharge from the reactor. ^{241}Am concentrations are increasing in both single pass and N Reactor fuels. The most important plutonium isotopes from a safety/regulatory assessment standpoint are ^{239}Pu and ^{240}Pu with half lives of 24,100 and 6,560 years respectively. Over a fifty year life, the concentrations of ^{239}Pu and ^{240}Pu will not change appreciably in the fuel. The concentrations of ^{90}Sr , ^{137}Cs , ^{238}Pu , and ^{241}Pu all change significantly over a fifty year facility life.

Two factors are important in determining which fuel to use in determining the safety/regulatory assessment design basis: burn-up, and time since discharge from the reactor. The safety/regulatory assessment basis should be the highest burn-up fuel in the basins. Time since discharge is also an important factor. As a first guess, the highest burn-up fuel N Reactor fuel (16.72% ^{240}Pu Mk IV fuel discharged from N Reactor 2/20/1995) is used as a starting point in the determination. Tables F.1 and F.2 show the concentrations of the safety basis nuclides in this fuel on 1/1/1995, and 1/1/2015 respectively.

Figures F.3 and F.4 are 16.49% ^{240}Pu fuel discharged from N Reactor on May 1, 1976. Figure F.5 is single Pass reactor fuel with 26% ^{240}Pu content. Note that the dose equivalent for the lower burn-up N reactor fuel and for SPR fuel is lower than for the 16.72% ^{240}Pu fuel in all cases.

**Table F.1: Maximum ^{240}Pu Content N Reactor Fuel in the K Basins
As of 1/1/1995**

Isotope	Activity (Ci/MTU)	CEDE (rem/uCi)	Dose Equivalent (rem/MTU)
Sr-90	7.84e+03	1.30e+00	1.02e+10
Cs-137	1.08e+04	3.20e-02	3.46e+08
Pu-238	1.71e+02	3.00e+02	5.13e+10
Pu-239	1.58e+02	3.30e+02	5.21e+10
Pu-240	1.28e+02	3.30e+02	4.22e+10
Pu-241	9.25e+03	5.70e+00	5.27e+10
Pu-242	1.05e-01	3.10e+02	3.25e+07
Am-241	3.58e+02	5.20e+02	1.86e+11
Total Dose Equivalent (rem/MTU)			3.95e+11
RADNUC 2.0 run 3/7/1995 for Maximum Pu-240 content K Basin fuel Results decayed to 01/01/1995. Total mass in run 6.90e+00 MTU. Total Activity in run 3.35e+05 Ci.			

**Table F.2: Maximum ^{240}Pu Content N Reactor Fuel in the K Basins
As of 1/1/2015**

Isotope	Activity (Ci/MTU)	C.E.D.E. (rem/uCi)	Dose Equivalent (rem/MTU)
Sr-90	4.87e+03	1.30e+00	6.33e+09
Cs-137	6.81e+03	3.20e-02	2.18e+08
Pu-238	1.46e+02	3.00e+02	4.39e+10
Pu-239	1.58e+02	3.30e+02	5.21e+10
Pu-240	1.28e+02	3.30e+02	4.21e+10
Pu-241	3.54e+03	5.70e+00	2.02e+10
Pu-242	1.05e-01	3.10e+02	3.25e+07
Am-241	5.33e+02	5.20e+02	2.77e+11
Total Dose Equivalent (rem/MTU)			4.42e+11
RADNUC 2.0 run 3/7/1995 for Maximum pu-240 K Basin Fuel. Results decayed to 01/01/2015. Total mass for run 6.90e+00 MTU. Total Activity for run 1.92e+05 Ci			

Table F.3: 16.49% ^{240}Pu Fuel Discharged from N Reactor 5/1/1976
as of 1/1/1995

Isotope	Activity (Ci/MTU)	CEDE (rem/uCi)	Dose Equivalent (rem/MTU)
Sr-90	7.20e+03	1.30e+00	9.36e+09
Cs-137	9.91e+03	3.20e-02	3.17e+08
Pu-238	1.59e+02	3.00e+02	4.76e+10
Pu-239	1.56e+02	3.30e+02	5.15e+10
Pu-240	1.24e+02	3.30e+02	4.10e+10
Pu-241	7.72e+03	5.70e+00	4.40e+10
Pu-242	9.74e-02	3.10e+02	3.02e+07
Am-241	3.78e+02	5.20e+02	1.96e+11
Total Equivalent Dose (rem/MTU) 3.90e+11			
RADNUC 2.0 run 1/26/1995 for Safety Basis Fuel Results decayed to 01/01/1995. Total Mass 1.16e+01 MTU. Total Activity 5.04e+05 Ci.			

Table F.4: 16.49% ^{240}Pu Fuel Discharged from N Reactor 5/1/1976
as of 1/1/2015

Isotope	Activity (Ci/MTU)	CEDE (rem/MTU)	Dose Equivalent (rem/MTU)
Sr-90	4.47e+03	1.30e+00	5.82e+09
Cs-137	6.25e+03	3.20e-02	2.00e+08
Pu-238	1.35e+02	3.00e+02	4.06e+10
Pu-239	1.56e+02	3.30e+02	5.15e+10
Pu-240	1.23e+02	3.30e+02	4.07e+10
Pu-241	2.95e+03	5.70e+00	1.68e+10
Pu-242	9.74e-02	3.10e+02	3.02e+07
Am-241	5.22e+02	5.20e+02	2.72e+11
Total Dose Equivalent (rem/MTU) 4.27e+11			
RADNUC 2.0 run 3/7/1995 for 1/1/1995 Safety Basis. Results decayed to 01/01/2015. Total Mass this run 1.16e+01 MTU. Total Activity this run 2.93e+05 Ci.			

Table F.5: 26.% ^{240}Pu SPR Fuel Currently in storage at PUREX

Isotope	Activity (Ci/MTU)	CEDE (rem/uCi)	Dose Equivalent (rem/MTU)
Sr-90	4.03e+03	1.30e+00	5.24e+09
Cs-137	1.03e+04	3.20e-02	3.30e+08
Pu-238	4.30e+01	3.00e+02	1.29e+10
Pu-239	1.20e+02	3.30e+02	3.96e+10
Pu-240	1.90e+02	3.30e+02	6.27e+10
Pu-241	5.00e+03	5.70e+00	2.85e+10
Pu-242	not available	3.10e+02	N/A
Am-241	4.30e+02	5.20e+02	2.24e+11
Total Dose Equivalent (rem/MTU) 3.73e+11			
Single Pass Reactor Fuel. Results decayed to 01/01/1994.			

The 16.72% ^{240}Pu fuel is the safety/regulatory assessment basis fuel.

APPENDIX G

INDEPENDENT REVIEW

Appendix G

INDEPENDENT REVIEW

PURPOSE

The purpose of this appendix is to assess whether the design basis feed types defined in this document are appropriate for developing path forward processes and facilities development and whether the design basis feeds are sufficiently accurate.

SCOPE

This appendix uses the main body of this report and its other appendices to assess whether:

1. The design basis feeds encompass the needs of process and facilities development.
2. The design basis feeds are sufficiently and conservatively described.

This report relies on the Accountability Data Base and the Origen 2 computer program to determine the inventories and their isotopic contents that will be included in the path forward. This data base and this computer program have been validated prior to preparation of this report; their assessment is outside the scope of this appendix.

SUMMARY

This appendix reviews three design basis feeds described in the main part of this report:

- Shielding Design Basis
- Safety/Regulatory Design Basis
- Heat Generation Design Basis

The reviews conclude that these design bases are appropriate and conservative.

This appendix also reviews Section 5, Sludge Sources. The review concludes that the data in Section 5 is appropriate and conservative except that Section 5 is based on nominal sludge volumes reported in Table 4, Baker 1995. However, Table 6, Baker 1995, observes that the sludge volumes could be approximately 50% higher if uncertainty is considered. A caveat has been added to Section 5 to note this difference between Tables 4 and 6 in Baker 1995.

The next revision of this document should address this issue.

Review of the remaining design bases, Criticality and Operational Variability, will be completed later when the sections of this report addressing these design bases are completed.

DISCUSSION

Review of Design Basis Feeds

Table 1.1 defines five design feeds and their basis. The purpose of the five feeds is to provide material quantities and isotopic content need to support the activities required for development of processes and facilities needed for the Spent Nuclear Fuel Project recommended path forward. Although details of the path forward process and facilities are not finalized, information is available in the body of this report and the Independent Technical Assessment (ITA) team report (WHC 1994a) to determine the major activities required to develop processes and facilities. These are summarized below:

1. Facility Development: These activities include design, evaluations and construction of the facilities required for the path forward. Major facilities to be developed include:
 - Multiple canister overpacks (MCOs) used during SNF transportation, conditioning and interim storage.
 - Modifications in the K Basin to prepare SNF, load it into MCOs and load MCOs into transfer casks.
 - A conditioning station where the MCOs will be dewatered, the fuel dehydrided and passivated, and the MCO will be inerted and sealed for interim storage.
 - An interim storage facility where the MCO containing dry and inerted SNF will be stored for an extended period of time.
2. Process Development
 - Processes to dewater the MCOs
 - Processes to dry, dehydride and passivate the SNF
 - Processes to inert and seal the MCOs

The above facility and process development and operation tasks will require several types of activities:

1. Design of facilities and process equipment including construction drawings and specifications.
2. Design analysis including shielding, criticality, stress and thermal effects.
3. Safety evaluations including SARs, SERs and EISs.

The five design basis feeds described in Table 1.1. address the activities listed above. These design basis feeds are sufficient for the scope of design, analysis and safety evaluation described above provided that conditions described in Table 1.1 and the following are met:

1. Using the definitions in Table 1.1, the shielding design basis should include the significant γ emitting isotopes. On the other hand, isotopes that emit only α and β need not be part of this design basis since they have no penetrating power and do not control shielding thickness.
2. The thermal design basis should include all significant radioactive isotopes including γ as well as α and β emitters, on the basis that the latter particles will be stopped and release their energy within the source and the conservative assumption that the γ 's will also be stopped and release their energy within the source.
3. The dose to personnel assessment used to develop the Safety/Regulatory design basis is based on DOE methodology. The dose methodology is contained in the reference (DOE, 1988). This document assigns values of committed effective dose equivalents (CEDEs) to radioactive isotopes. CEDEs are in units of rem/curie and, when multiplied by the curie level of an isotope, result in the radioactive intensity of the isotope in rem.

Assessment Spreadsheets

The main part of this report uses the program Origen 2 to calculate N Reactor spent fuel isotopes for various amounts of fuel burnup in megawatt-days/ton (MWD/T) for both MKIV and MKIA fuel elements. The amount of burnup used in Origen 2 was that estimated to produce Pu240 contents of 6, 9, 12 and 16 percent of total Pu. The output of Origen 2 is a complete lists of weights (grams/MTU), radioactivity (curies/MTU), and heat generation (watts/MTU) for both radioactive isotopes and elements. Only data for the significant isotopes were used for this report. Origen 2 output in the form of electronic media was made available to Mollerus Engineering Corporation (MEC) for this independent review. The Origen 2 output was for the following conditions:

For MKIV Fuel

Nominal Pu240, %	Burnup, MDW/T	Total Pu, grams/MTU	Pu240, grams/MTU	Pu240
6	1188	965.7	62.5	6.469
9	1979	1503	139.8	9.301
12	2970	2079	253.9	12.184
16	later	2758	417.8	15.159

For MKIA Fuel

Nominal Pu240, %	Burnup, MDW/t	Total Pu, grams/MTU	Pu240, grams/MTU	Pu240
6	1010	970	62.3	6.42
9	1670	1510	143.0	9.48
12	2510	2090	262.0	12.58
16	3620	2760	439.0	15.88

The above data indicates that, for the same burnup, there is little difference in Pu240 and total Pu produced in the two fuel designs stored in the K Basins. However, the Pu content (and other isotopes) is very much a linear function of burnup.

Origen 2 is a well documented and validated national laboratory computer program for calculating isotopic content of irradiated nuclear fuel. Independent review of Origen 2 and it's application to Hanford's spent fuel inventory is not part of the scope of the independent review described in this appendix. Rather, the scope of this

review is the application of the output of Origen 2 as described below.

As noted above, the output of Origen 2 is based on a pre-selected MWD/T burnup with an estimate of Pu240 content and pre-programmed decay periods since fuel discharge. However, the spent fuel stored at Hanford has a wide variety of Pu240 content and decay times since discharge from N Reactor. The approach used in other parts of this report is to use the program RADNUC to derive data for conditions different than those developed using Origen 2. The independent review described in this appendix does not use RADNUC. Instead, the Origen 2 output showing significant isotopic content after 7 1/2 years of radioactive decay has taken place has been placed into the following EXCEL based spreadsheets.

<u>IVFUELWA.xls</u>	Spreadsheet for MKIV fuel heat generation
<u>IVFUELCU.xls</u>	Spreadsheet for MKIV fuel radioactivity
<u>IAFUELWA.xls</u>	Spreadsheet for MKIA fuel heat generation
<u>IAFUELCU.xls</u>	Spreadsheet for MKIA fuel radioactivity

Tables G-1 and G-2 show the significant isotopic weights, radioactivity and heat sources calculated to exist in the MKIV and MKIA fuel after 7 1/2 years of radioactive decay following discharge from N Reactor (a period of time slightly less than the decay time associated with the last fuel to be discharged from N Reactor). Typically, the isotopic content is a nonlinear function of Pu240. The nonlinear data has been curve fitted using the equation form:

$$Y = \frac{C1 + C3 * X}{1 + C2 * X + C4 * X^2}$$

where X is the Pu240 content in fraction of total Pu. Coefficients for both radioactivity (curies/MTU) and heat generation (watts/MTU) have been calculated and added to the spreadsheets. Equations were then added to the spreadsheets to calculate radioactivity and heat generation as a function of Pu240 fraction.

Next, the spreadsheets were modified to include table of half lives, CEDE's and gamma energy data for each of the isotopes. Table G-3 shows this data. Equations were then added to the spreadsheets to calculate radioactivity and heat generation as a function of time using

$$X = X_0 e^{-\lambda t}$$

where

X_0 = the 7 1/2 year radioactivity or heat generation value

$$\lambda = \frac{0.693}{t_{1/2}}$$

t = time after 7 1/2 years in seconds

This feature gives the spreadsheet the capability to calculate radioactivity and heat generation values for decay periods greater than 7 1/2 years.

Next, the IVFUELCU.xls and IAFUELCU.xls spreadsheets were modified to multiply the radioactivity (curies/MTU) by both gamma energy and the CEDE for each significant isotope. This is used later to evaluate each isotope in terms of safety and shielding impact.

Finally, regions of the spreadsheets were set up to allow transfer of data using the COPY/SPECIAL PASTE/DATA feature of EXCEL. Data can be sorted in descending order to evaluate the impact of each of the isotopes. Areas of the spreadsheet have been set up to print portions of the sorted data that are considered to be significant.

In summary, the spreadsheets provide the capability to evaluate MKIV and MKIA fuels in terms of:

- Pu240 content
- Radioactive decay period
- Heat generation
- CEDE and shielding impacts

Table G-4 provides a roadmap of spreadsheet tables, calculation and sorting regions.

Analysis of Design Basis Feeds

1. Shielding Design Basis

Heavy shielding is needed to reduce gamma radiation. Alpha and Beta radiation can be stopped by protective clothing and relatively thin-walled structures and enclosures. A measure of the shielding load of spent fuel is its curies content and the energy in the gamma radiation. Therefore, the shielding design basis feed should include those isotopes with the combination of highest activity and highest gamma energy, i.e. (curies x Kev). According to the Accountability Data Base contained in Appendix B the following appear to be key numbers with the highest Pu240 and shortest decay periods:

Key Number	Discharge Date	Decay period, years	Pu240, %	(Curies*Kev)
14355	Jan 1984	11	15.74	3.16E+07
15244	Jun 1986	8.5	13.58	2.73E+07
12565	Jan 1979	16	16.72	1.76E+07

Key number 14355 has the highest (Curie*Kev) loading. Table G-5 shows the significant heat generating isotopes for Key Number 14355 using the IVFUELCU.xls spread sheet. The values shown in Table 2.5 are higher, indicating that the basis used for Table 2.5, 16% Pu240 and 10 year decay, is conservative.

2. Safety/Regulatory Assessment Design Basis

Most safety assessments deal with the impact of the release of activation products, fission products and daughter isotopes on human health. A measure of these impact is the DOE committed effective dose equivalent (CEDE) methodology where the CEDE units are rem per curie. Table G-3 lists the CEDEs used in this appendix and the spreadsheets. The total CEDE content of N Reactor fuel increases in proportion to burnup and exponentially with Pu240 content while it decreases with decay time. According to the Accountability Data Base contained in Appendix B the following appear to be key numbers with the highest Pu240 and shortest decay periods:

Key Number	Discharge Date	Decay period, years	Pu240, %	CEDE*Curies rem/MTU
12565	Jun 1979	16	16.72	1.95E+05
14355	Jan 1984	11	15.74	1.89E+05
15244	Jun 1986	8.5	13.58	1.40E+05

Key number 12565 has the highest rem/MTU loading. Table G-6 shows the significant heat generating isotopes for Key Number 12565 using the IVFUELCU.xls spread sheet. The values shown in Table 2.8 are higher, indicating that the basis used for Table 2.8, 16.72% Pu240 and 16 year decay, is conservative.

3. Heat Generation Design Basis

The heat generation rate in N Reactor fuel increases in proportion to burnup and exponentially with Pu240 content while it decreases with decay time. According to the Accountability Data Base contained in Appendix B, the following appear to be key numbers with the highest Pu240 and shortest decay periods:

Key Number	Discharge Date	Decay period, years	Pu240, %	Watts/MTU
14355	Jan 1984	11	15.74	127.24
12565	Jun 1979	16	16.72	123.79
15244	Jun 1986	8.5	13.58	109.04

Key number 14355 has the highest watts/MTU loading. Table G-7 shows the significant heat generating isotopes for Key Number 14355 using the IVFUELWA.xls spread sheet. The values shown in Table 2.10 are higher, indicating that the basis used for Table 2.10, 16% Pu240 and 10 year decay, is conservative.

4. Criticality Design
Later
5. Operational Variability Design Basis
Later
6. Sludge Sources

Section 5 notes that metallic oxides, concrete grit, and other loose material has accumulated and mixed with spent fuel corrosion product to form sludge. Most of the sludge is on the basin floors; a small amount is in the open 105 KE canisters and the closed 105 KW canisters. Present plans call for the sludge to be dried, passivated, and loaded into MCO's using the processes and facilities provided for spent fuel. It is anticipated that, due to dilution with metallic oxides, concrete grit and other non-fuel material, the chemical and radioactive loading of sludge will be less than the spent fuel loading described in Sections 3 and 4 and reviewed above. This section of the appendix reviews the data contained in Section 5 and Appendix E.

The sludge design basis described in Section 5 relies on test data and visual observations to bracket the volumes and both chemical and isotopic content of the sludge. This methodology is appropriate since a purely analytical approach is not appropriate to a problem that is highly empirical and dependent on basin water chemistry and temperature conditions that varied widely over a long period of time. Basically, Section 5 multiplied observed sludge volumes by reported results from chemical and isotopic analysis of sludge samples. Sludge samples are limited and do not cover all parts of the K East Basin. The higher measured values have been applied to the entire basin. Sources of input used to develop the data in Tables 5.1 through 5.5 are

- Sludge volumes: Baker 1995 except Warner 1994 for the Sandfilter Backwash Pit
- Chemical and radionuclide constituent data: Welsh 1994 and Baker 1995

These sources appear to be appropriate and are the most current.

This appendix is not a review of these documents, which were reviewed by other procedures prior to their publication. Rather, this appendix reviews application of the data in these references to generate the information in Section 5.

The approach taken and the data used to develop Section 5 appear to be appropriate.

However the results may not conservatively bracket total sludge volumes, weights, and isotopic content since Section 5 is based on sludge volumes reported in Table 4, Baker 1995. However, Table 6, Baker 1995, reports maximum volumes that are approximately 50% higher. Baker reports these to be estimated based on conservative uncertainty factors.

APPENDIX G

TABLES

HNF-SD-SNF-TI-009, REV. 1

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SB126	0	0	0	0
SB126M	0	0	0	0
TE123M	5.818E-11	2.607E-10	8.061E-10	2.199E-09
TE125M	0.8592	1.414	2.055	2.876
TE127	3.753E-14	2.009E-13	8.048E-13	2.06E-12
TE127M	3.831E-14	2.051E-13	8.217E-13	2.103E-12
TOTALS	0.064688827	0.093013972	0.121837422	0.151486584
	0.09561	0.2621	0.565	1.132
ACTIVATION PRODUCTS				Thermal power, watts
Nominal Pu 240	MKIV 6%	MKIV 9%	MKIV 12% Pu	MKIV 16% Pu
H3	0.000003219	0.000008823	0.00001902	0.00003813
C14	0.00003501	0.00005912	0.00008887	
FE55	0.0002209	0.0003631	0.0005277	0.0007369
CO60	0.0432	0.07175	0.1057	0.1502
NI59	2.826E-07	4.734E-07	7.048E-07	0.000001016
NI63	0.0003367	0.0005668	0.000849	0.001233
SR89	9.34E-19	1.149E-18	1.272E-18	1.324E-18
SR90	2.594E-09	6.559E-09	1.334E-08	2.583E-08
Y90	1.239E-08	3.133E-08	6.374E-08	1.234E-07
Y91	3.993E-16	5.03E-16	5.646E-16	5.934E-16
ZR93	0.000000252	4.252E-07	6.389E-07	0.000000932
ZR95	2.19E-12	2.798E-12	3.171E-12	3.358E-12
NB93M	0.000000118	2.013E-07	3.066E-07	4.553E-07
NB95	4.603E-12	5.884E-12	6.666E-12	7.059E-12
NB95M	4.456E-15	5.695E-15	6.452E-15	6.833E-15
TC99	3.322E-10	5.589E-10	8.366E-10	1.214E-09
AG110	2.179E-14	9.251E-14	2.744E-13	6.89E-13
AG110M	3.809E-12	1.617E-11	4.796E-11	1.204E-10
CD115M	1.41E-22	1.72E-22	1.884E-22	1.945E-22
IN113M	1.337E-08	1.876E-08	2.292E-08	0.000000026
SN113	9.554E-10	1.34E-09	1.638E-09	1.858E-09
SN119M	0.0000342	0.00005242	0.00007019	0.00008833
SN121M	0.00002093	0.0000353	0.00005299	0.00007721
SN123	2.758E-08	3.935E-08	4.887E-08	5.644E-08
SB124	6.176E-18	2.225E-17	5.529E-17	1.152E-16
SB125	0.01101	0.01812	0.02633	0.03686
SB126	0	0	0	0
SB126M	0	0	0	0
TE123M	8.473E-14	3.797E-13	1.174E-12	3.202E-12
TE125M	0.0007222	0.001189	0.001728	0.002418
TE127	5.067E-17	2.712E-16	1.087E-15	2.782E-15

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Appendix G
Independent Review

Radioactivity, curies				
ACTINIDES+DAUGHTERS				
Nominal Pu 240	MKIV 6%	MKIV 9%	MKIV 12% Pu	MKIV 16% Pu
U234	0.425	0.4168	0.4072	0.3949
U235	0.01807	0.01672	0.01522	0.01355
U236	0.04464	0.05143	0.05893	0.0671
U238	0.3326	0.3323	0.332	0.3315
NP237	0.01103	0.01929	0.02985	0.0448
PU238	7.263	22.81	53.09	116.2
PU239	55.77	83.35	110	138.5
PU240	14.24	31.86	57.75	95.23
PU241	589	2076	4997	9449
PU242	0.0009554	0.005533	0.02031	0.05663
AM241	8.579	30.47	74.06	142
AM242	0.002233	0.02123	0.1086	0.396
AM242M	0.002244	0.02134	0.1091	0.398
AM243	0.0003997	0.00352	0.01687	0.05758
CM242	0.001901	0.0181	0.09271	0.3388
CM244	0.03972	0.6409	4.994	26.79
Thermal power, watts				
ACTINIDES+DAUGHTERS				
Nominal Pu 240	MKIV 6%	MKIV 9%	MKIV 12% Pu	MKIV 16% Pu
U234	0.01224	0.01201	0.01173	0.01137
U235	0.0004731	0.000438	0.0003985	0.0003547
U236	0.001209	0.001393	0.001596	0.001818
U238	0.008435	0.008428	0.00842	0.008409
NP237	0.0003371	0.0005897	0.0009124	0.001369
PU238	0.2407	0.7561	1.759	3.851
PU239	1.719	2.569	3.389	4.268
PU240	0.4434	0.992	1.798	2.965
PU241	0.01826	0.06435	0.1549	0.2929
PU242	0.00002822	0.0001634	0.0005997	0.001672
AM241	0.285	1.012	2.46	4.716
AM242	0.000002535	0.0000241	0.0001232	0.0004495
AM242M	8.866E-07	0.000008428	0.0000431	0.0001572
CM242	0.00007006	0.0006669	0.003416	0.01248
CM244	0.001389	0.02242	0.1747	0.937
Radioactivity, curies				
FISSION PRODUCTS				

Mollerus Engineering Corporation

Appendix G
Independent Review

Nominal Pu 240	MKIV 6%	MKIV 9%	MKIV 12% Pu	MKIV 16% Pu
H3	11.58	19.74	30.31	44.25
C14	0.000003926	0.00000654	0.000009852	0.00001411
SE79	0.01459	0.02418	0.03616	0.05143
KR85	244.2	391.4	565.5	771.7
SR89	1.247E-11	1.401E-11	1.427E-11	1.333E-11
SR90	2540	4060	5857	7986
Y90	2540	4061	5859	7988
Y91	2.5E-09	2.908E-09	3.04E-09	2.901E-09
ZR93	0.06846	0.1114	0.1638	0.2284
NB93M	0.02106	0.03468	0.05176	0.07357
ZR95	4.467E-08	5.452E-08	5.975E-08	5.996E-08
NB95	9.917E-08	0.000000121	1.326E-07	1.331E-07
NB95M	3.314E-10	4.044E-10	4.432E-10	4.448E-10
TC99	0.4865	0.8076	1.211	1.723
RU103	2.744E-16	3.419E-16	3.943E-16	4.226E-16
RH103M	2.474E-16	3.082E-16	3.554E-16	3.81E-16
RU106	91.73	174.2	287.1	435.1
RH106	91.73	174.2	287.1	435.1
PD107	0.001681	0.003516	0.006385	0.01082
AG110	0.00005647	0.0001924	0.0004961	0.001121
AG110M	0.004246	0.01447	0.0373	0.08427
CD113M	0.9142	1.633	2.609	4.038
CD115M	1.086E-16	1.371E-16	1.571E-16	1.695E-16
SN119M	0.006528	0.01033	0.01435	0.01871
SN121M	0.003969	0.007625	0.01298	0.0207
SN123	0.0001755	0.0002573	0.0003303	0.0003912
TE123M	4.944E-10	2.254E-09	7.125E-09	1.979E-08
SB124	2.042E-13	4.82E-13	9.125E-13	1.557E-12
SB125	85.55	148	228.6	333.5
TE125M	20.87	36.12	55.78	81.36
SN126	0.02274	0.04016	0.06347	0.09602
SB126	0.003183	0.005622	0.008885	0.01344
SB126M	0.02274	0.04016	0.06347	0.09602
TE127	0.00004109	0.00006122	0.00008019	0.00009596
TE127M	0.00004195	0.0000625	0.00008187	0.00009797
TE129	1.757E-21	2.133E-21	2.415E-21	2.55E-21
TE129M	2.699E-21	3.277E-21	3.71E-21	3.917E-21
II29	0.0009641	0.001678	0.002627	0.00391
CS134	41.55	126.7	282.2	566.5
CS135	0.01319	0.02163	0.03255	0.04651
CS137	2957	4931	7431	10640
BA137M	2797	4664	7030	10060
CE141	1.838E-20	2.042E-20	2.143E-20	2.108E-20
PR143	8.456E-56	8.693E-56	8.674E-56	8.355E-56
CE144	124.6	186.6	247.8	302.7

PR144		124.6	186.6	247.8	302.7
PR144M		1.495	2.239	2.973	3.632
PM147		1833	2752	3646	4439
PM148		1.323E-18	2.439E-18	3.431E-18	4.295E-18
PM148M		2.349E-17	4.33E-17	6.091E-17	7.625E-17
SM151		55.96	74.22	89.09	100.8
EU152		0.112	0.34	0.7188	1.208
GD153		0.00002641	0.0001441	0.000484	0.001213
EU154		14.91		103.8	224.5
EU155		28.31	37.17	48.46	65.43
TB160		1.227E-11	3.594E-11	8.12E-11	1.607E-10
Fission Products				Thermal power, watts	
Nominal Pu 240	MKIV 6%	MKIV 9%	MKIV 12% Pu	MKIV 16% Pu	
H3	0.0003899	0.0006645	0.00102	0.00149	
C14	1.151E-09	1.918E-09	2.889E-09	4.138E-09	
SE79	0.000003633	0.000006019	0.000009003	0.0000128	
KR85	0.3658	0.5862	0.8471	1.156	
SR89	4.31E-14	4.842E-14	4.934E-14	4.607E-14	
SR90	2.948	4.712	6.798	9.269	
Y90	14.08	22.51	32.47	44.27	
Y91	8.978E-12	1.044E-11	1.092E-11	1.042E-11	
ZR93	0.000007954	0.00001294	0.00001903	0.00002653	
NB93M	0.000003731	0.000006144	0.000009171	0.00001304	
ZR95	2.263E-10	2.762E-10	3.026E-10	3.037E-10	
NB95	4.757E-10	5.806E-10	6.363E-10	6.385E-10	
NB95M	4.604E-13	5.62E-13	6.159E-13	6.18E-13	
TC99	0.000244	0.000405	0.0006073	0.000864	
RUI03	9.181E-19	1.144E-18	1.319E-18	1.414E-18	
RH103M	5.694E-20	7.094E-20	8.181E-20	8.77E-20	
RUI06	0.005454	0.01036	0.01707	0.02587	
RH106	0.8798	1.671	2.754	4.173	
PDI07	9.964E-08	2.084E-07	3.785E-07	6.413E-07	
AGI10	4.057E-07	0.000001383	0.000003564	0.000008052	
AGI10M	0.0000709	0.0002416	0.0006228	0.001407	
CDI13M	0.001539	0.002749	0.004392	0.006798	
SNI19M	0.000003374	0.00000534	0.000007418	0.00000967	
SNI21M	0.000007952	0.00001528	0.000026	0.00004148	
SNI23	0.000000548	8.036E-07	0.000001032	0.000001222	
TEI23M	7.2E-13	3.283E-12	1.038E-11	2.883E-11	
SB124	2.711E-15	6.4E-15	1.212E-14	2.067E-14	
SB125	0.2674	0.4628	0.7147	1.042	
TEI25M	0.01754	0.03036	0.04689	0.06839	

SN126	0.00002836	0.00005009	0.00007915	0.0001198
SB126	0.00005882	0.0001039	0.0001642	0.0002484
SB126M	0.0002895	0.0005113	0.0008081	0.001223
TE127	5.549E-08	8.267E-08	1.083E-07	1.296E-07
TE127M	2.257E-08	3.362E-08	4.403E-08	5.269E-08
TE129	6.275E-24	7.62E-24	8.628E-24	9.11E-24
TE129M	4.732E-24	5.745E-24	6.506E-24	6.869E-24
II29	0.000000446	7.762E-07	0.000001215	0.000001809
CSI34	0.4229	1.29	2.872	5.766
CSI35	0.000004403	0.00000722	0.00001086	0.00001552
CSI37	3.271	5.454	8.22	11.77
BA137M	10.98	18.31	27.6	39.51
CE141	2.691E-23	2.99E-23	3.137E-23	3.086E-23
PRI43	1.575E-58	1.619E-58	1.616E-58	1.557E-58
CE144	0.08265	0.1237	0.1643	0.2008
PRI44	0.9159	1.371	1.821	2.225
PRI44M	0.0005116	0.000766	0.001017	0.001243
PM147	0.6575	0.987	1.308	1.592
PM148	1.019E-20	1.878E-20	2.642E-20	3.307E-20
PM148M	2.978E-19	5.49E-19	7.723E-19	9.667E-19
SM151	0.006561	0.008703	0.01045	0.01182
EU152	0.000847	0.002572	0.005437	0.009136
GD153	2.386E-08	1.301E-07	4.372E-07	0.000001096
EU154	0.1334	0.3945	0.9285	2.008
EU155	0.02059	0.02703	0.03524	0.04759
TB160	9.994E-14	2.927E-13	6.614E-13	1.308E-12

Table G-2

ORIGEN II BASED DATA FOR MKIA FUEL (7.5 YEARS DECAY)
(Data shown in Units per Metric Ton of Uranium)

PLUTONIUM		Concentrations, grams		
Pu240 file	MK1a6% Pu	MK1a 9% Pu	MK1a12% Pu	MK1a16% Pu
Pu, gm	9.70E+02	1.51E+03	2.09E+03	2.76E+03
Pu240, gm	6.23E+01	1.43E+02	2.62E+02	4.39E+02
Pu240, %	6.42%	9.48%	12.58%	15.88%
ACTIVATION PRODUCTS		Radioactivity, curies		
Pu240 file	MK1a 6% Pu	MK1a 9% Pu	MK1a12% Pu	MK1a16% Pu
H3	9.86E-02	2.68E-01	5.81E-01	1.17E+00
C14	1.27E-01	2.13E-01	3.21E-01	4.71E-01
FE55	7.06E+00	1.16E+01	1.69E+01	2.37E+01
FE59	2.88E-18	3.46E-18	1.69E+01	3.99E-18
CO60	3.50E+00	5.78E+00	8.55E+00	1.22E+01
NI59	7.89E-03	1.32E-02	1.97E-02	2.84E-02
NI63	9.38E-01	1.57E+00	2.37E+00	3.45E+00
SR89	3.27E-16	4.00E-16	4.45E-16	4.65E-16
SR90	2.69E-06	6.76E-06	1.39E-05	2.72E-05
Y90	2.69E-06	6.76E-06	1.39E-05	2.72E-05
Y91	1.35E-13	1.68E-13	1.90E-13	2.01E-13
ZR93	2.68E-03	4.51E-03	6.80E-03	9.96E-03
ZR95	5.29E-10	6.72E-10	7.66E-10	8.15E-10
NB93M	8.23E-04	1.40E-03	2.14E-03	3.19E-03
NB95	1.17E-09	1.49E-09	1.70E-09	1.81E-09
NB95M	3.92E-12	4.98E-12	5.68E-12	6.05E-12
TC99	7.96E-07	1.33E-06	2.01E-06	2.92E-06
AG110	3.11E-12	1.31E-11	3.92E-11	9.92E-11
AG110M	2.34E-10	9.85E-10	2.95E-09	7.46E-09
CD115M	3.87E-20	4.68E-20	5.16E-20	5.34E-20
IN113M	6.90E-06	9.63E-06	1.18E-05	1.35E-05
SN113	6.89E-06	9.62E-06	1.18E-05	1.35E-05
SN119M	7.87E-02	1.20E-01	1.62E-01	2.04E-01
SN121M	1.25E-02	2.10E-02	3.17E-02	4.63E-02
SN123	1.08E-05	1.54E-05	1.92E-05	2.23E-05
SB124	5.63E-16	2.01E-15	5.04E-15	1.06E-14

SB125	4.21E+00	6.89E+00	1.01E+01	1.41E+01
SB126	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SB126M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE123M	6.82E-11	3.02E-10	9.49E-10	2.63E-09
TE125M	1.03E+00	1.68E+00	2.46E+00	3.45E+00
TE127	4.38E-14	2.32E-13	9.43E-13	2.45E-12
TE127M	4.47E-14	2.36E-13	9.63E-13	2.50E-12
TOTAL	6.42%	9.48%	12.58%	15.88%
	9.86E-02	2.68E-01	5.81E-01	1.17E+00
ACTIVATION PRODUCTS		Thermal power, watts		
Pu240 file	MK1a 6% Pu	MK1a 9% Pu	MK1a12% Pu	MK1a16% Pu
H3	3.32E-06	9.01E-06	1.96E-05	3.94E-05
C14	3.71E-05	6.24E-05	9.42E-05	1.38E-04
FE55	2.39E-04	3.90E-04	5.70E-04	7.99E-04
CO60	5.39E-02	8.91E-02	1.32E-01	1.88E-01
NI59	3.13E-07	5.23E-07	7.81E-07	1.13E-06
NI63	3.73E-04	6.24E-04	9.40E-04	1.37E-03
SR89	1.13E-18	1.38E-18	1.54E-18	1.61E-18
SR90	3.12E-09	7.85E-09	1.62E-08	3.16E-08
Y90	1.49E-08	3.75E-08	7.72E-08	1.51E-07
Y91	4.83E-16	6.05E-16	6.83E-16	7.21E-16
ZR93	3.11E-07	5.23E-07	7.90E-07	1.16E-06
ZR95	2.68E-12	3.40E-12	3.88E-12	4.13E-12
NB93M	1.46E-07	2.48E-07	3.79E-07	5.65E-07
NB95	5.63E-12	7.15E-12	8.16E-12	8.68E-12
NB95M	5.45E-15	6.92E-15	7.89E-15	8.40E-15
TC99	3.99E-10	6.69E-10	1.01E-09	1.46E-09
AG110	2.24E-14	9.41E-14	2.82E-13	7.13E-13
AG110M	3.91E-12	1.65E-11	4.93E-11	1.25E-10
CD115M	1.44E-22	1.75E-22	1.92E-22	1.99E-22
IN113M	1.61E-08	2.24E-08	2.76E-08	3.14E-08
SN113	1.15E-09	1.60E-09	1.97E-09	2.25E-09
SN119M	4.07E-05	6.21E-05	8.35E-05	1.06E-04
SN121M	2.51E-05	4.21E-05	6.35E-05	9.28E-05
SN123	3.38E-08	4.79E-08	5.99E-08	6.95E-08
SB124	7.47E-18	2.67E-17	6.70E-17	1.41E-16
SB125	1.32E-02	2.16E-02	3.15E-02	4.42E-02
SB126	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SB126M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE123M	9.93E-14	4.40E-13	1.38E-12	3.82E-12
TE125M	8.63E-04	1.41E-03	2.06E-03	2.90E-03
TE127	5.91E-17	3.13E-16	1.27E-15	3.31E-15

Moller Engineering Corporation

Appendix G
Independent Review

ACTINIDES+DAUGHTERS		Radioactivity, curies		
Pu240 file	MKIIa 6% Pu	MKIIa 9% Pu	MKIIa12% Pu	MKIIa16% Pu
U234	4.66E-01	4.57E-01	4.46E-01	4.33E-01
U235	2.18E-02	2.01E-02	1.82E-02	1.60E-02
U236	4.76E-02	5.63E-02	6.59E-02	7.66E-02
U238	3.32E-01	3.32E-01	3.31E-01	3.31E-01
NP237	1.12E-02	1.98E-02	3.11E-02	4.74E-02
PU238	7.35E+00	2.33E+01	5.55E+01	1.24E+02
PU239	5.60E+01	8.33E+01	1.10E+02	1.37E+02
PU240	1.42E+01	3.25E+01	5.98E+01	1.00E+02
PU241	5.80E+02	2.07E+03	5.04E+03	9.83E+03
PU242	9.40E-04	5.65E-03	2.13E-02	6.24E-02
AM241	8.44E+00	3.04E+01	7.47E+01	1.48E+02
AM242	2.19E-03	2.09E-02	1.08E-01	4.05E-01
AM242M	2.20E-03	2.10E-02	1.09E-01	4.07E-01
AM243	3.95E-04	3.55E-03	1.76E-02	6.29E-02
CM242	1.86E-03	1.78E-02	9.26E-02	3.47E-01
CM244	3.89E-02	6.33E-01	5.12E+00	2.88E+01
ACTINIDES+DAUGHTERS		Thermal power, watts		
Pu240 file	MKIIa 6% Pu	MKIIa 9% Pu	MKIIa12% Pu	MKIV16% Pu
U234	1.34E-02	1.32E-02	1.29E-02	1.25E-02
U235	5.72E-04	5.27E-04	4.76E-04	4.18E-04
U236	1.29E-03	1.53E-03	1.79E-03	2.08E-03
U238	8.42E-03	8.41E-03	8.40E-03	8.39E-03
NP237	3.42E-04	6.04E-04	9.50E-04	1.45E-03
PU238	2.43E-01	7.73E-01	1.84E+00	4.12E+00
PU239	1.73E+00	2.57E+00	3.38E+00	4.22E+00
PU240	4.42E-01	1.01E+00	1.86E+00	3.11E+00
PU241	1.80E-02	6.42E-02	1.56E-01	3.05E-01
PU242	2.78E-05	1.67E-04	6.29E-04	1.84E-03
AM241	2.80E-01	1.01E+00	2.48E+00	4.90E+00
AM242	2.48E-06	2.37E-05	1.23E-04	4.60E-04
AM242M	8.68E-07	8.30E-06	4.31E-05	1.61E-04
CM242	6.86E-05	6.57E-04	3.41E-03	1.28E-02
CM244	1.36E-03	2.22E-02	1.79E-01	1.01E+00
FISSION PRODUCTS		Radioactivity, curies		

Pu240 file	MK1a 6% Pu	MK1a 9% Pu	MK1a12% Pu	MK1a16% Pu
H3	1.36E+01	2.31E+01	3.54E+01	5.17E+01
C14	4.72E-06	7.85E-06	1.18E-05	1.69E-05
SE79	1.75E-02	2.89E-02	4.32E-02	6.15E-02
KR85	2.98E+02	4.79E+02	6.93E+02	9.49E+02
SR89	1.53E-11	1.73E-11	1.77E-11	1.66E-11
SR90	3.11E+03	4.99E+03	7.22E+03	9.89E+03
Y90	3.11E+03	4.99E+03	7.22E+03	9.89E+03
Y91	3.06E-09	3.57E-09	3.75E-09	3.60E-09
ZR93	8.31E-02	1.36E-01	1.99E-01	2.79E-01
NB93M	2.56E-02	4.22E-02	6.30E-02	8.98E-02
ZR95	5.40E-08	6.60E-08	7.23E-08	7.27E-08
NB95	1.20E-07	1.47E-07	1.61E-07	1.62E-07
NB95M	4.01E-10	4.90E-10	5.36E-10	5.40E-10
TC99	5.84E-01	9.69E-01	1.45E+00	2.06E+00
RU103	3.20E-16	3.95E-16	4.53E-16	4.88E-16
RH103M	2.88E-16	3.56E-16	4.09E-16	4.40E-16
RU106	9.96E+01	1.88E+02	3.10E+02	4.75E+02
RH106	9.96E+01	1.88E+02	3.10E+02	4.75E+02
PD107	1.79E-03	3.74E-03	6.81E-03	1.17E-02
AG110	6.03E-05	2.04E-04	5.30E-04	1.22E-03
AG110M	4.54E-03	1.53E-02	3.98E-02	9.15E-02
CD113M	9.88E-01	1.75E+00	2.80E+00	4.35E+00
CD115M	1.17E-16	1.46E-16	1.68E-16	1.82E-16
SN119M	7.10E-03	1.12E-02	1.55E-02	2.03E-02
SN121M	4.52E-03	8.60E-03	1.46E-02	2.33E-02
SN123	1.95E-04	2.84E-04	3.64E-04	4.32E-04
TE123M	5.34E-10	2.40E-09	7.68E-09	2.16E-08
SB124	2.24E-13	5.20E-13	9.91E-13	1.70E-12
SB125	9.39E+01	1.62E+02	2.49E+02	3.64E+02
TE125M	2.29E+01	3.94E+01	6.08E+01	8.89E+01
SN126	2.55E-02	4.47E-02	7.05E-02	1.07E-01
SB126	3.56E-03	6.25E-03	9.87E-03	1.50E-02
SB126M	2.55E-02	4.47E-02	7.05E-02	1.07E-01
TE127	4.64E-05	6.86E-05	8.96E-05	1.08E-04
TE127M	4.73E-05	7.00E-05	9.15E-05	1.10E-04
TE129	2.04E-21	2.46E-21	2.77E-21	2.94E-21
TE129M	3.14E-21	3.78E-21	4.26E-21	4.51E-21
II29	1.12E-03	1.95E-03	3.04E-03	4.52E-03
CS134	5.13E+01	1.55E+02	3.45E+02	6.97E+02
CS135	1.53E-02	2.51E-02	3.76E-02	5.36E-02
CS137	3.55E+03	5.92E+03	8.89E+03	1.28E+04
BA137M	3.36E+03	5.60E+03	8.41E+03	1.21E+04
CE141	2.22E-20	2.46E-20	2.57E-20	2.54E-20
PR143	1.02E-55	1.05E-55	1.04E-55	1.01E-55
CE144	1.51E+02	2.26E+02	3.01E+02	3.68E+02

PR144	1.51E+02	2.26E+02	3.01E+02	3.68E+02
PR144M	1.81E+00	2.72E+00	3.61E+00	4.42E+00
PM147	2.20E+03	3.30E+03	4.33E+03	5.26E+03
PM148	1.60E-18	2.93E-18	4.08E-18	5.10E-18
PM148M	2.85E-17	5.21E-17	7.25E-17	9.06E-17
SM151	6.40E+01	8.41E+01	9.96E+01	1.12E+02
EU152	1.32E-01	3.95E-01	8.24E-01	1.37E+00
GD153	3.16E-05	1.67E-04	5.60E-04	1.40E-03
EU154	1.78E+01	5.20E+01	1.23E+02	2.66E+02
EU155	3.07E+01	4.04E+01	5.29E+01	7.29E+01
TB160	1.31E-11	3.78E-11	8.60E-11	1.73E-10
FISSION PRODUCTS			Thermal power, watts	
Pu240 file	MK1a 6% Pu	MK1a 9% Pu	MK1a12% Pu	MC1a16% Pu
H3	4.59E-04	7.79E-04	1.19E-03	1.74E-03
CI4	1.38E-09	2.30E-09	3.46E-09	4.96E-09
SE79	4.35E-06	7.20E-06	1.08E-05	1.53E-05
KR85	4.46E-01	7.17E-01	1.04E+00	1.42E+00
SR89	5.28E-14	5.97E-14	6.11E-14	5.74E-14
SR90	3.61E+00	5.79E+00	8.38E+00	1.15E+01
Y90	1.72E+01	2.77E+01	4.00E+01	5.48E+01
Y91	1.10E-11	1.28E-11	1.35E-11	1.29E-11
ZR93	9.66E-06	1.58E-05	2.32E-05	3.24E-05
NB93M	4.53E-06	7.48E-06	1.12E-05	1.59E-05
ZR95	2.74E-10	3.34E-10	3.66E-10	3.68E-10
NB95	5.75E-10	7.03E-10	7.70E-10	7.75E-10
NB95M	5.57E-13	6.80E-13	7.45E-13	7.50E-13
TC99	2.93E-04	4.86E-04	7.27E-04	1.04E-03
RU103	1.07E-18	1.32E-18	1.52E-18	1.63E-18
RH103M	6.63E-20	8.20E-20	9.41E-20	1.01E-19
RU106	5.92E-03	1.12E-02	1.85E-02	2.83E-02
RH106	9.55E-01	1.81E+00	2.98E+00	4.56E+00
PD107	1.06E-07	2.22E-07	4.04E-07	6.92E-07
AG110	4.33E-07	1.46E-06	3.81E-06	8.75E-06
AG110M	7.58E-05	2.56E-04	6.65E-04	1.53E-03
CD113M	1.66E-03	2.95E-03	4.72E-03	7.33E-03
SN119M	3.67E-06	5.77E-06	8.02E-06	1.05E-05
SN121M	9.05E-06	1.72E-05	2.92E-05	4.67E-05
SN123	6.09E-07	8.86E-07	1.14E-06	1.35E-06
TE123M	7.78E-13	3.49E-12	1.12E-11	3.14E-11
SB124	2.97E-15	6.91E-15	1.32E-14	2.26E-14
SB125	2.94E-01	5.05E-01	7.79E-01	1.14E+00
TE125M	1.93E-02	3.31E-02	5.11E-02	7.47E-02
SN126	3.17E-05	5.57E-05	8.79E-05	1.33E-04

SB126	6.58E-05	1.16E-04	1.82E-04	2.76E-04
SB126M	3.24E-04	5.69E-04	8.97E-04	1.36E-03
TE127	6.26E-08	9.26E-08	1.21E-07	1.45E-07
TE127M	2.55E-08	3.77E-08	4.92E-08	5.91E-08
TE129	7.29E-24	8.79E-24	9.90E-24	1.05E-23
TE129M	5.50E-24	6.62E-24	7.47E-24	7.91E-24
II29	5.20E-07	9.01E-07	1.40E-06	2.09E-06
CS134	5.22E-01	1.57E+00	3.52E+00	7.09E+00
CS135	5.10E-06	8.38E-06	1.25E-05	1.79E-05
CS137	3.93E+00	6.55E+00	9.84E+00	1.41E+01
BA137M	1.32E+01	2.20E+01	3.30E+01	4.73E+01
CE141	3.24E-23	3.60E-23	3.77E-23	3.71E-23
PR143	1.90E-58	1.96E-58	1.95E-58	1.88E-58
CE144	1.00E-01	1.50E-01	1.99E-01	2.44E-01
PR144	1.11E+00	1.66E+00	2.21E+00	2.71E+00
PR144M	6.20E-04	9.30E-04	1.23E-03	1.51E-03
PM147	7.88E-01	1.18E+00	1.55E+00	1.89E+00
PM148	1.23E-20	2.26E-20	3.14E-20	3.93E-20
PM148M	3.61E-19	6.60E-19	9.19E-19	1.15E-18
SM151	7.50E-03	9.86E-03	1.17E-02	1.32E-02
EU152	1.00E-03	2.99E-03	6.23E-03	1.04E-02
GD153	2.85E-08	1.51E-07	5.06E-07	1.26E-06
EU154	1.59E-01	4.65E-01	1.10E+00	2.38E+00
EU155	2.23E-02	2.94E-02	3.84E-02	5.30E-02
TB160	1.07E-13	3.08E-13	7.00E-13	1.41E-12

Table G-3
ISOTOPE DATA

ISOTOPE	CEDE	HALF-LIFE	See footnotes	HALF-LIFE SECONDS	GAMMA Kev
H3	0.000063	12.3	a	3.88E+08	0
C14	0.0021	5730	a	1.81E+11	0
FE55	0.0026	2.73	a	86093280	0
FE59	0.015	44.51	d	3845664	1291.6
CO60	0.03	5.271	a	1.66E+08	1332.5
NI59	0.0027	76000	a	2.4E+12	0
NI63	0.0063	100	a	3.15E+09	0
SE79	0.0039	65000	a	2.05E+12	0
KR85	0	10.73	a	3.38E+08	514
SR89	0.037	50.52	d	4364928	909.2
SR90	1.3	29.1	a	9.18E+08	0
Y90	0.0082	2.67	d - SR90	9.18E+08	479.5
Y91	0.044	58.5	d	5054400	1205
ZR93	0.32	1500000	a	4.73E+13	30.4
ZR95	0.019	64.02	d	5531328	756.7
NB93M	0.028	16.1	a - ZR93**	5.08E+08	30.4
NB95	0.0045	34.97	d - ZR95**	3021408	765.8
NB95M	0.0022	3.61	d - ZR95**	311904	235.7
TC99	0.0075	213000	a	6.72E+12	89.7
RU103	0.0078	39.27	d	3392928	497.1
RH103M	4.8E-06	56.12	m - RU103m	3392928	39.8
RU106	0.44	1.02	a	32166720	
RH106	0.00039	2.18	h - RU106	32166720	1045.8
PD107	0.013	650000	a	2.05E+13	
AG110	0	24.6	s	24.6	657.8
AG110M	0.053	249.8	d	21582720	884.7
CD113M	0.42	14.1	a	4.45E+08	263.7
CD115M	0.065	44.6	d	3853440	1290.6
IN113M	0.0001	1.658	h - CD113M	4.45E+08	391.7
SN113	0.0039	115.1	d	9944640	391.7
SN119M	0.0053	293	d	25315200	65.7
SN121M	0.0063	55	a	1.73E+09	6.3
SN123	0.03	129.2	d	11162880	1088.6
TE123M	0.0096	119.7	d	10342080	159
SB124	0.021	60.2	d	5201280	1691
SB125	0.0093	2.76	a	87039360	635.9
TE125M	0.0067	58	d - SB125	87039360	109.3
SN126	0.086	100000	a	3.15E+12	87.6
SB126	0.01	12.4	d - S126	3.15E+12	694.9

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Appendix G
Independent Review

SB126M	0.000073	11	s - S126	3.15E+12	22.7
TE127	0.00069	9.4	h	33840	417.9
TE127M	0.019	109	d	9417600	88.3
TE129	0.00019	1.16	h - I129**	4176	459.6
TE129M	0.02	33.6	d - I129**	2903040	695.9
I129	0.28	15700000	a	4.95E+14	39.6
CS134	0.074	2.065	a	65121840	795.9
CS135	0.0071	2.356	a	74298816	
CS137	0.05	30.17	a - BA137M	9.51E+08	661.7
BA137M	0	2.552	m	9.51E+08	661.7
CE141	0.0085	32.5	d	2808000	145.1
PR143	0.0073	13.57	d	1172448	742
CE144	0.36	284.5	d	24580800	133.5
PR144	0.00011	7.2	m - CE144	24580800	696.5
PR144M	0	17.28	m - CE144	24580800	2187.5
PM147	0.034	2.634	a	83065824	121.3
PM148	0.01	41.3	d	3568320	630
PM148M	0.017	5.37	d	463968	1465.1
SM151	0.029	90	a	2.84E+09	21.5
EU152	0.22	13.48	a	4.25E+08	1408
GD153	0.06747	241.4	d	20856960	103.2
EU154	0.28	8.59	a	2.71E+08	1274.5
EU155	0.039	4.71	a	1.49E+08	105.3
TB160	3.12	72.3	d	6246720	966.2
U234	130	246000	a	7.76E+12	
U235	120	7.04E+08	a	2.22E+16	
U236	120	23420000	a	7.39E+14	
U238	120	447000000	a	1.41E+15	
NP237	4.9	2140000	a	6.75E+13	
PU238	300	87.7	a	2.77E+09	
PU239	330	24100	a	7.6E+11	
PU240	330	6560	a	2.07E+11	
PU241	5.7	14.4	a	4.54E+08	
PU242	310	375000	a	1.18E+13	
AM241	0.1027	432.7	a	1.36E+10	
AM242	0.061	16.02	h	57672	
AM242M	510	141	a	4.45E+09	
AM243	520	7370	a	2.32E+11	
CM242	17	162.8	d	14065920	
CM244	270	18.1	a	5.71E+08	

Time Symbols

a = years 1 year = 31536000 seconds
d = days 1 day = 86400 seconds
h = hours 1 hour = 3600 seconds

An isotope symbol shown next to a time symbol indicates this isotope is a daughter of the isotope listed

in the left column.

Table G-4

PROGRAM IVFUEL.XLS ROADMPA

A1	Roadmap
A100	Isotope data (CEDE, half-life, gamma, beta, alpha)
A200	Curve fitting data (from IVFUELWA.XLS
AA200	Curve fitted curies, curie*Kev and Curie*CEDE by isotope Curve fitted watts by isotope
BA200	Isotope/watts sorting area
BA300	Print summary tables

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Appendix G
Independent Review

Table G-5

SIGNIFICANT ISOTOPES SORTED BY CURIES x Kev

Radioactivity in MKIV Fuel

Pu240 = 15.74 percent
Time after Discharge = 11 years

Values shown are per MTU

Individual Isotope				Cumulative, % of Total		
Isotope	Curies	Cu*Kev	Cu*CEDE	Curies	Cu*Kev	Cu*CEDE
CS137	10490.88	6941817.698	524.5441815	0.215703878	0.377940217	0.002812907
BA137M	9916.094	6561479.664	0	0.419589471	0.735173315	0.002812907
Y90	7771.321	3726348.605	63.72483538	0.579376217	0.938050582	0.003154637
KR85	651.0065	334617.3564	0	0.592761613	0.956268484	0.003154637
EU154	193.9723	247217.6783	54.31224005	0.596749893	0.969727999	0.00344589
PM147	1821.639	220964.7564	61.93571077	0.634204745	0.981758201	0.003778025
CS134	198.4937	157981.1259	14.68853288	0.638285989	0.990359323	0.003856793
SB125	153.6475	97704.46075	1.428921977	0.641445148	0.995678743	0.003864456
RH106	43.36999	45356.3331	0.016914295	0.642336882	0.998148123	0.003864547
PR144	13.89067	9674.854598	0.001527974	0.642622489	0.998674861	0.003864555
CO60	6.582655	8771.387454	0.197479642	0.642757836	0.99915241	0.003865614
EU155	41.79158	4400.653394	1.629871627	0.643617116	0.999391999	0.003874354
TE125M	37.48302	4096.894102	0.251136235	0.644387807	0.99961505	0.003875701
SM151	100.0078	2150.167644	2.900226125	0.646444075	0.999732114	0.003891254
CE144	13.89067	1854.405009	5.000642721	0.646729682	0.999833075	0.00391807
EU152	1.082745	1524.504645	0.238203851	0.646751945	0.999916075	0.003919347
CD113M	3.704964	976.9991244	1.556085067	0.646828123	0.999969267	0.003927692
PR144M	0.16668	364.6123852	0	0.64683155	0.999989118	0.003927692
TC99	1.83837	164.9018068	0.013787776	0.646869349	0.999998096	0.003927766
SB126	0.014543	10.10625547	0.000145435	0.646869648	0.999998646	0.003927767
TOTAL	48635.58	18367502.03	186477.59			

Table G-6

SIGNIFICANT ISOTOPES SORTED BY CURIES x CEDE

Radioactivity in MKIV Fuel

Pu 240 = 16.72 percent
Time after discharge = 16 years

Values shown are per MTU

Individual Isotope				Cumulative, % of Total		
Isotope	Curies	Cu*Kev	Cu*CEDE	Curies	Cu*Kev	Cu*CEDE
PU239	154.8357	0.00E+00	5.11E+04	0.343%	0.000%	26.229%
PU238	161.5706	0.00E+00	4.85E+04	0.701%	0.000%	51.111%
PU241	7557.049	0.00E+00	4.31E+04	17.448%	0.000%	73.223%
PU240	119.8451	0.00E+00	3.95E+04	17.714%	0.000%	93.525%
SR90	7539.69	0.00E+00	9.80E+03	34.422%	0.000%	98.556%
CM244	6.782464	0.00E+00	1.83E+03	34.437%	0.000%	99.496%
CS137	10405.87	6.89E+06	5.20E+02	57.497%	39.085%	99.764%
AM242M	0.289204	0.00E+00	1.47E+02	57.498%	39.085%	99.839%
Y90	7540.417	3.62E+06	6.18E+01	74.208%	59.609%	99.871%
U234	0.386429	0.00E+00	5.02E+01	74.208%	59.609%	99.897%
EU154	158.9324	2.03E+05	4.45E+01	74.561%	60.759%	99.920%
U238	0.33128	0.00E+00	3.98E+01	74.561%	60.759%	99.940%
AM241	340.3458	0.00E+00	3.50E+01	75.316%	60.759%	99.958%
AM243	0.047683	0.00E+00	2.48E+01	75.316%	60.759%	99.971%
PU242	0.061705	0.00E+00	1.91E+01	75.316%	60.759%	99.980%
PM147	511.0072	6.20E+04	1.74E+01	76.448%	61.110%	99.989%
U236	0.071531	0.00E+00	8.58E+00	76.448%	61.110%	99.994%
CS134	45.13431	3.59E+04	3.34E+00	76.548%	61.314%	99.996%
SM151	98.99248	2.13E+03	2.87E+00	76.768%	61.326%	99.997%
U235	0.012616	0.00E+00	1.51E+00	76.768%	61.326%	99.998%
TOTAL	4.51E+04	1.76E+07	1.95E+05			

Table G-7

HEAT GENERATION IN MKIV FUEL

Pu 240 = 5.74 percent

Time after discharge = 11 years

Individual Isotope			Cumulative	
Isotope	Watts/MTU	% of TOTAL	Watts/MTU	% of TOTAL
H3	1.35E-03	0.00%	0.001	0.00%
C14	-8.19E-06	0.00%	0.001	0.00%
FE55	3.23E-04	0.00%	0.002	0.00%
CO60	1.02E-01	0.08%	0.103	0.08%
NI59	1.10E-06	0.00%	0.103	0.08%
NI63	1.30E-03	0.00%	0.104	0.08%
SE79	1.37E-05	0.00%	0.104	0.08%
KR85	9.75E-01	0.77%	1.080	0.85%
SR89	1.10E-21	0.00%	1.080	0.85%
SR90	9.02E+00	7.09%	10.097	7.94%
Y90	4.31E+01	33.85%	53.169	41.79%
Y91	2.72E-18	0.00%	53.169	41.79%
ZR93	2.92E-05	0.00%	53.169	41.79%
ZR95	3.00E-16	0.00%	53.169	41.79%
NB93M	1.24E-05	0.00%	53.169	41.79%
NB95	0.00E+00	0.00%	53.169	41.79%
NB95M	0.00E+00	0.00%	53.169	41.79%
TC99	9.22E-04	0.00%	53.170	41.79%
RU103	0.00E+00	0.00%	53.170	41.79%
RH103M	0.00E+00	0.00%	53.170	41.79%
TOTAL	1.27E+02			

DISTRIBUTION SHEET

To	From	Page 1 of 1
Distribution	Process Engineering	Date 1/6/98
Project Title/Work Order		EDT No.
105-K Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities		ECN No. 645037

Name	MSIN	Text With All Attach.	Text Only	Attach./Appendix Only	EDT/ECN Only
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DE&S Hanford

C. J. Alderman	X3-85	X
R. B. Baker	H0-40	X
D. E. Ball	R3-86	X
D. W. Bergman	X3-79	X
K. H. Bergsman	X3-85	X
C. Defigh-Price	R3-86	X
J. I. Diehl	X3-80	X
D. R. Duncan	R3-86	X
J. R. Frederickson	R3-86	X
R. G. Gant	X3-79	X
L. H. Goldman	R3-86	X
J. W. Osborne	X3-80	X
A. T. Kee	R3-86	X
P. G. LeRoy	R3-15	X
M. K. Mahaffey	R3-86	X
R. H. Meichle	X3-79	X
R. P. Omberg	R3-15	X
K. L. Pearce	R3-11	X
A. N. Praga	R3-86	X
W. H. Rasin	R3-11	X
M. A. Reilly	R3-86	X
R. P. Ruth	R3-11	X
E. J. Shen	X3-75	X
D. L. Sherrell	R3-86	X (10)
D. W. Smith	R3-15	X
K. E. Smith	R3-86	X
J. A. Swenson	R3-11	X
C. A. Thompson	R3-86	X
J. E. Turnbaugh	R3-15	X
Central Files (Orig. + 1)	A3-88	X
SNF Project Files	R3-11	X

FDH

W. D. Gallo	R3-11	X
E. W. Gerber	R3-11	X
R. L. McCormack	R3-86	X

FDNW

R. D. Crowe	R3-11	X
L. J. Garvin	R3-11	X
D. A. Smith	R3-11	X

NHC

G. D. Bazinet	S8-06	X
J. J. Irwin	R3-86	X
C. R. Miska	R3-86	X
J. P. Sloughter	H5-49	X
W. L. Willis	H5-61	X

RFSH

F. M. Coony	H6-06	X
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SGN Eurisys

A. L. Pajunen	R3-86	X
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XWEST

B.D. Lorenz	R3-11	X
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