Fuel Cycle Potential Waste Inventory for Disposition

Fuel Cycle Research & Development

Prepared for
U.S. Department of Energy
Used Nuclear Fuel
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SUMMARY

The United States (U.S.) currently utilizes a once-through fuel cycle where used nuclear fuel is stored on-site in either wet pools or in dry storage systems with ultimate disposal envisioned in a deep mined geologic repository. Within the Department of Energy's (DOE) Office of Nuclear Energy (DOE-NE), Fuel Cycle Research and Development (FCR&D) develops options to the current commercial fuel cycle management strategy to enable the safe, secure, economic, and sustainable expansion of nuclear energy while minimizing proliferation risks by conducting research and development focused on nuclear fuel recycling and waste management to meet U.S. needs.

The safe management and disposition of used nuclear fuel (UNF) and/or nuclear waste is a fundamental aspect of any nuclear fuel cycle. Yet, the routine disposal of UNF and radioactive waste remains problematic. An expansion of nuclear power in the U. S. would increase the amount of all classes of waste and requires the availability of routine disposal pathways.

To address these challenges, the mission of the Used Fuel Disposition Campaign is to identify alternatives and conduct scientific research and technology development to enable storage, transportation, and disposal of UNF and wastes generated by existing and future nuclear fuel cycles. Different commercial fuel cycle alternatives are expected to generate different waste streams where a variety of waste forms could potentially be used to isolate the hazardous materials.

The purpose of this report is to provide an estimate of potential waste inventory and waste form characteristics for the DOE UNF and HLW and a variety of commercial fuel cycle alternatives in order to support subsequent system-level evaluations of disposal system performance. This report is envisioned as a "living document" which will be revised as specific alternative fuel cycles are developed.

The initial issue (revision 0) of this study analyzes potential waste quantities and properties generated by:

- DOE activities (Section 2.0). This includes DOE UNF, Navy UNF and DOE HLW resulting from the reprocessing of DOE and navy used fuel.
- Commercial UNF potentially generated by uranium fuel light water reactors (LWR). The four once through fuel cycles (Section 3) analyzed in this study differ by varying the assumed expansion/contraction of nuclear power. The four once through fuel cycle scenarios analyzed are:
  - Scenario 1 assumes no replacement of existing nuclear generation reactors.
  - Scenario 2 assumes the amount of current nuclear generation is maintained at the current levels (100 GWe/yr (giga-watts days per year)) with new reactors replacing the existing reactors as the existing reactors are decommissioned.
  - Scenario 3 assumes the amount of nuclear generation will increase to 200 GWe/yr from 2020 to 2060.
  - Scenario 4 assumes the amount of nuclear generation will increase to 400 GWe/yr from 2020 to 2060.

  These scenarios indicate a potential range from 140,000 MT to 680,000 MT of used fuel which will be generated by the end of the century.
- The four alternative LWR recycling processes analyzed differ in the reprocessing method (aqueous vs. electro-chemical), complexity (Pu only or full transuranic (TRU) recovery) and waste forms generated. Potential waste quantities and inventories have been developed on unit base of each MT reprocessed.

Revision 1 of the report adds information regarding alternative waste forms generated by aqueous reprocessing LWR UNF.

Revision 2 of the report adds information regarding spent Mixed Oxide Fuels (MOX) and waste generated by the reprocessing of Fast Reactor (FR) fuels.

MOX fuels derive their fissile content from various sources. Section 5 discusses two potential sources, Pu from the LWR UNF reprocessing, and Weapons Grade (WG) plutonium from surplus materials.

TRU which is derived from LWR reprocessing is proposed to be converted into FR fuel. In closed fuel cycles the used FR fuel is reprocessed. Section 6 discusses the potential waste generated by reprocessing both metal and oxide fuels discharged from FR operation at conditions such that the TRU conversion ratio is approximately 0.5 or 0.75.

Revision 3 of the report:

- Updates current commercial UNF inventory summaries through December 2010,
- Provides isotopic composition and decay heat as a function of time for a 100 GWd/MT burn-up (8.32% enrichment) LWR UOX fuel as an example of an alternative fuel cycle from continuous improvement of the current U.S. reactor fleet
- Provides decay heat as a function of time for LWR UOX, MOX fuel derived from the recovered U/Pu by reprocessing 50 GWd/MT LWR UOX fuel, and the sodium FR fuels discussed in prior revisions of this report.
- Provides decay heat as a function of time for selected reprocessing wastes which are needed to continue generic repository investigations.
- Includes additional dry used fuel storage data.

Revision 3 also made minor editorial and formatting changed which did not impact the technical content of the prior materials.

Revision 4 of this report:

- Provides an initial waste inventory estimate from advanced once thru reactor concepts (Section 7.0). The reactors studied include the Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy (CANDLE) Reactor, the Substainable Sodium-Cooled Fast Reactor, the Mixed Fast Spectrum Reactor, the Ultra-Long Life Fast Reactor, the Energy Multiplier Module, and the Traveling Wave Reactor. The mass, radionuclide content, and decay heat as a function of time are provided for these discharged fuels.
- Provides an initial waste inventory estimate from recycling uranium oxide fuel from a representative prismatic block high temperature gas reactor fuel (Section 8.0).
- Provides an initial estimate for the isotopic distributions of discharged fuel from a thorium based multi-recycle fuel cycle. The wastes resulting from reprocessing this fuel have not been estimated. Inclusion of reprocessing wastes will be considered for future revisions.
- Corrected errors in the volume of Cs/Sr ceramic waste in Table 4-3.
- Corrected errors in the volume of Lanthanide glass in Table 4-3.

The data presented in this report is available in electronic format from the lead author.

Seven new elements identified by the FEP s process as potentially important to system performance were added to all material balance calculations. These additional elements are actinium, chlorine, lead, protactinium, selenium, and antimony. These elements are all minor contributors to the mass, volume, and package count of the waste forms (< 0.5 g/MT) but they are potentially important to dose projections. These isotopics are now tracked in the excel workbook associated with this Report and extracted data used by others should be regenerated to ensure these new elements are included.

Revision 5 of this report provides additonal information on the decay heat for DOE SNF and HLW, and provides update commercial UNF discharge projections thru December 2011 and updated commercial UNF in dry storage. Four new independent UNF storage facilities have started operations since the last revision in October 2011.

The excel work book associated with this report was subjected to a quality assurance (QA) audit. The audit was performed by Pacific Northwest National Laboratory personnel who provided an independent review for the Savannah River National Laboratory (SRNL). The audit reveal minor errors in the caculations to estimate wastes from reprocessing but, these did not have significant impact on the estimated mass and volume of these wastes. The QA report is referenced in the body of this report.

Future revisions to this report will include waste inventory summaries properties generated by additional alternative fuel cycles. The additional fuel cycles have not yet been determined.

CONTENTS

SUMMARY iii

CONTENTS vi

Acronyms xvii

1. INTRODUCTION 1
   - 1.1 Initial Issue (Revision 0) 1
   - 1.2 Revision 1 2
   - 1.3 Revision 2 2
   - 1.4 Revision 3 2
   - 1.5 Revision 4 3
   - 1.6 Revision 5 3
   - 1.7 Study Limitations 3

2. DOE ACTIVITY WASTE 4
   - 2.1 DOE Used Nuclear Fuel 4
     - 2.1.1 DOE UNF Inventory 5
     - 2.1.2 DOE UNF Radionuclide Inventory 6
     - 2.1.3 DOE UNF Storage/Canisters 6
   - 2.2 NAVAL UNF 7
     - 2.2.1 Naval UNF Inventory 7
     - 2.2.2 Naval UNF Radionuclide Inventory 8
     - 2.2.3 Naval UNF Storage/Canisters 8
   - 2.3 DOE HIGH LEVEL WASTE 8
     - 2.3.1 Current DOE HLW Inventory 9
     - 2.3.2 Projected DOE HLW Inventory 9
     - 2.3.3 DOE HLW Radionuclide Inventory 12
     - 2.3.4 DOE HLW Storage 13

3. COMMERCIAL LIGHT WATER REACTOR UNF- ONCE-THRU FUEL CYCLE 13
   - 3.1 Current Commercial UNF Inventory 13
   - 3.2 Projected Commercial UNF Inventory from Light Water Reactors 18
     - 3.2.1 Scenario 1 – No Replacement Nuclear Generation 18
     - 3.2.2 Scenario 2 – Maintain Current Nuclear Generation 20
     - 3.2.3 Scenario 3 – Projected Growth to 200 GWe/yr Nuclear Generation in 2060 20
     - 3.2.4 Scenario 4 – Projected Growth to 400 GWe/yr Nuclear Generation in 2060 20
   - 3.3 Commercial UNF Radionuclide Inventory 29
   - 3.4 Commercial UNF Storage 35

4. PROCESS WASTES GENERATED BY REPROCESSING COMMERCIAL LIGHT WATER REACTOR UNF 40
   - 4.1 Reprocessing Methods 40
     - 4.1.1 Co-Extraction 40
     - 4.1.2 New Extraction 40

4.1.3 UREX 40
4.1.4 Electro-Chemical 41
4.1.5 Material Balance Calculations Bases and Assumptions 41
4.2 Representative LWR Fuels 41
4.3 Baseline Waste forms 41
4.3.1 Off-Gas Waste forms 41
4.3.2 Metal Waste forms 42
4.3.3 Principle Fission Product Waste forms 47
4.4 UNF Equivalent Waste Projections 47
4.4.1 Off-Gas Waste Trends 47
4.4.2 Metal Waste Trends 48
4.4.3 Principle Fission Product Waste Trends 52
4.4.4 Characteristics of the Heat Generating Wastes from LWR UOX Processes 52
4.5 Alternative Light Water Reactor Reprocessing Waste forms 60
4.5.1 Alternative Off-Gas Waste Forms 60
4.5.2 Alternative Fission Product Waste Forms 61

5. MIXED OXIDE FUELS 65
5.1 MOX Fuel Derived from Reprocessing LWR UOX fuel 65
5.2 Weapons Grade Plutonium Derived MOX 68

6. POTENTIAL WASTE DERIVED FROM FULL RECYCLING ALTERNATIVES 68
6.1 Advanced Burner Reactor Design and Operating Parameters 69
6.1.1 Recycling ABR Oxide Fuel 72
6.1.2 Recycling ABR Metal Fuel 72
6.1.3 Potential Waste Summary from Recycling ABR fuels 72
6.1.4 Characteristics of the Heat Generating Wastes from SFR Processes 76

7. ADVANCED ONCE-THROUGH REACTOR POTENTIAL WASTE 78
7.1 Advanced Once-Through Reactor Design and Core Performance Parameters 78
7.2 Advanced Once-Through UNF Masses and Characteristics 80
7.3 Advanced Once-Through UNF Decay Heat 82

8. High Temperature Gas Reactor Potential Waste 86
8.1 HTGR Description 86
8.2 HTGR Applications 86
8.3 HTGR Core Designs 86
8.4 Reprocessing 87
8.4.1 Head-End Process Description 87
8.4.2 Aqueous Processes 89
8.4.3 Waste Estimation Methodology 90
8.4.4 Waste Summary 90
8.5 Waste Observations and Trends 94

9. Thorium Based Fuel Cycle Potential Waste 94
9.1 Background 95
9.2 Thorium as a Fuel 95

9.3 Modeling and Calculation Summary 96
9.4 Isotope Trends and Summary 97

10. Potential Waste From Molten Salt Reactors 98
10.1 Description of Molten Salt Reactors 98
10.2 MSR Experience 99
10.2.1 Aircraft Reactor Experiment (ARE) 99
10.2.2 Molten Salt Reactor Experiment (MSRE) 99
10.3 Reference Reactors 100
10.3.1 Molten Salt Breeder Reactor (MSBR) 100
10.3.2 Denatured Molten Salt Reactor (DMSR) 100
10.4 MSR Waste 101
10.4.1 Operation Wastes 101
10.4.2 Fuel Recovery 101
10.4.3 Disposition of the Balance of MSR Fuel Salt 102
10.5 Next Steps 102

11. References 103

Appendix A Commercial Nuclear Fuel Characteristics 107

Appendix B Commercial Used Nuclear Fuel Projections 119

Appendix C Commercial Used Fuel Radionuclide Characteristics 138

Appendix D DOE Used Nuclear Fuel Characteristics 200

Appendix E Navy Used Nuclear Fuel Radionuclide Inventory 215

Appendix F High Level Waste Radionuclide Inventory 218

Appendix G Alternative LWR Reprocessing Waste Forms 227
G-1 Low Temperature Glass Encapsulants for I-129 227
G-2 Iodine Capture and Immobilization with Functionalized Silica Aerogels 229
G-3 Chalcogenide Aerogel 229
G-4 Krypton Implantation in Silicon Carbide 230
G-5 Iron Phosphate Based Glasses 233
G-6 Glass ceramic from a melt process 235
G-7 HLW Cermet 236
G-8 Heat-tolerant Cermets 236
G-9 Immobilization of 99Tc as epsilon metal 237

Appendix H Constraints on Waste Packaging 238

Appendix I Commercial MOX Fuel Radionuclide Characteristics 242

Appendix K Sodium Cooled Fast Reactor Material Balance 257

Appendix L Decay Heat for Waste from Reprocessing LWR UOX Fuel 262

Appendix M Decay Heat for Waste from Reprocessing Advanced Burner Sodium Cooled Fast Reactor Fuel 287

Appendix N High Temperature Gas Reactor Fuel Characteristic 294

Appendix O Isotopic Distribution of Discharged Thorium Based Fuel 302

FIGURES

Figure3-1 Percentage of Assemblies per Burn-up Range Current Inventory 18

Figure 3-2 Annual UNF Assemblies Discharged for the No Replacement, Maintain Current, 200 GWe/yr, and 400 GWe/yr. 21

Figure 3-3 Cumulative UNF Assemblies Discharged for the No Replacement, Maintain Current, 200 GWe/yr, and 400 GWe/yr. 22

Figure 3-4 Annual UNF Mass Discharged for the No Replacement, Maintain Current, 200 GWe/yr, and 400 GWe/yr Cases. 23

Figure 3-5 Cumulative UNF Mass Discharged for the No Replacement, Maintain Current, 200 GWe/yr, and 400 GWe/yr Cases. 24

Figure 3-6 Percentage of Assemblies per Burn-up Range - No replacement Case 25

Figure 3-7 Percentage of Assemblies per Burn-up Range – Maintain Current Nuclear Generation Case 26

Figure 3-8 Percentage of Assemblies per Burn-up Range – 200 GWe/yr Nuclear Generation by 2060 Case 27

Figure 3-10 PWR 40 GWd/MT Used Fuel Decay Heat 30

Figure 3-11 PWR 60 GWd/MT Used Fuel Decay Heat 31

Figure 3-12 BWR 30 GWd/MT Used Fuel Decay Heat 32

Figure 3-13 BWR 50 GWd/MT Used Fuel Decay Heat 33

Figure 3-14 PWR 100 GWd/MT Used Fuel Decay Heat 34

Figure 4-1 Co-Extraction reprocessing block diagram. 43

Figure 4-2 New Extraction reprocessing block diagram 44

Figure 4-3 UREX reprocessing block diagram. 45

Figure 4-4 Electro-chemical reprocessing block diagram 46

Figure 4-5 Effect of Burn-up on Borosilicate Glass Decay Heat Generated by Co-Extraction, New Extraction, and UREX Processing 54

Figure 4-6 Effect of Fuel Age on Borosilicate Glass Decay Heat Generated by Co-Extraction, New Extraction, and UREX Processing 55

Figure 4-7 Effect of Reactor Type on Borosilicate Glass Decay Heat Generated by Co-Extraction, New Extraction, and UREX Processing 56

Figure 4-8 UREX Process Waste Decay Heat Summary 57

Figure 4-9 Electro-Chemical Process Waste Decay Heat Summary 58

Figure 4-10 Comparison of Ceramic Waste Decay Heat Generated by UREX and Electro-Chemical Processing 59
Figure 5-1 MO Fuel 50 GWd/MT Used Fuel Decay Heat 67
Figure 6-1 Sodium FR Used Fuel Decay Heat 71
Figure 6-2 Sodium FR Used Fuel Reprocessing Waste Decay Heat Summary 77
Figure 7-1 Comparison of Discharge UNF Masses 80
Figure 7-2 Normalized Decay Heat per Unit Electricity Generation 85
Figure 8-1 Head-End Process 88
Figure 9-1 Fuel loading by percent for each cycle. From Raitses et al., 2009. 97
Figure H-1Conceptual heat transfer model for waste container in storage 239
Figure K-1 Block Flow Diagram for the Electro-Chemical Reprocessing of Advanced Burner Reactor UNF 261
Figure L-1 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 40 GWd/MT Burn-up 5 Year Cooled 263
Figure L-2 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 40 GWd/MT Burn-up 5 Year Cooled 264
Figure L-3 Borosilicate Glass Decay Heat Generated by UREX Processing of 40 GWd/MT 5 Year Cooled PWR Fuel 265
Figure L-4 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 40 GWd/MT 5 Year Cooled PWR Fuel 266
Figure L-5 Glass Bonded Zeolite Decay Heat Generated by Electro-Chemical Processing of 40 GWd/MT 5 year Cooled PWR Fuel 267
Figure L-6 Lanthanide Glass Decay Heat Generated by Electro-Chemical Processing of 40 GWd/MT 5 Year Cooled PWR Fuel 268
Figure L-7 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 60 GWd/MT Burn-up 5 Year Cooled 269
Figure L-8 Borosilicate Glass Decay heat Generated by New Extraction Processing of 60 GWd/MT Burn-up 5 Year Cooled 270
Figure L-9 Borosilicate Glass Decay Heat Generated by UREX Processing of 60 GWd/MT 5 Year Cooled PWR Fuel 271
Figure L-10 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 60 GWd/MT 5 Year Cooled PWR Fuel 272
Figure L-11 Glass Bonded Zeolite Bonded Decay Het Generated by Electro-Chemical Processing of 60 GWd/MT 5 Year Cooled PWR Fuel 273
Figure L-12 Lanthanide Glass Decay Heat Generated by Electo-Chemical Processing of 60 GWd/MT 5 Year Cooled PWR Fuel 274
Figure L-13 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 60 GWd/MT Burn-up 30 Year Cooled 275
Figure L-14 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 60 GWd/MT Burn-up 30 Year Cooled 276

Figure L-15 Borosilicate Glass Decay Heat Generated by UREX Processing of 60 GWd/MT 30 Year PWR Fuel 277
Figure L-16 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 60 GWd/MT 30 Year Cooled PWR Fuel 278
Figure L-17 Glass Bonded Zeolite Decay Heat Generated by Electo-Chemical Processing of 60 GWd/MT 30 Year Cooled PWR Fuel 279
Figure L-18 Lanthanide Glass Decay Heat Generated by Electo-Chemical Processing of 60 GWd/MT 30 Year Cooled PWR Fuel 280
Figure L-19 Borosilicate Glass Decay Het Generated by Co-Extraction Processing of BWR 50 GWd/MT Burn-up 5 Year Cooled 281
Figure L-20 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 50 GWd/MT Burn-up 5 Year Cooled 282
Figure L-21 Borosilicate Glass Decay Heat Generated by UREX Processing of 50 GWd.MT 5 year Cooled BWR Fuel 283
Figure L-22 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 50 GWd/MT 5 Year Cooled BWR Fuel 284
Figure L-23 Glass Bonded Zeolite Decay Heat Generated by Electro-Chemical Processing of 50 GWd.MT 5 year Cooled BWR Fuel 285
Figure L-24 Lanthanide Glass Decay Het Generated by Electro-Chemical Processing of 50 GWd/MT 5 Year Cooled BWR Fuel 286
Figure M-1 New Extraction Glass Decay Heat Generated by Processing Sodium FR Oxide 0.75 CR 131 GWd/MT Burn-up 2 Year Cooled 288
Figure M-2 New Extraction Glass Decay Heat Generated by Processing Sodium FR Oxide 0.5 CR 166 GWd/MT Burn-up 2 Year Cooled 289
Figure M-3 Electro-Chemical Glass Bonded Zeolite Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.75 290
Figure M-4 Electro-Chemical Lanthanide Glass Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.75 291
Figure M-5 Electro-Chemical Glass Bonded Zeolite Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.5 292
Figure M-6 Electro-Chemical Lanthanide Glass Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.5 293

TABLES

Table 2-1 DOE Spent Nuclear Fuel Canister Decay Heat 7
Table 2-2 Current high level waste inventory 9
Table 2-3 Savannah River Canister Decay Heat Distribution 10
Table 2-4 Hanford and Idaho waste Inventory 11
Table 2-5 Projected total number of high level waste canisters 11
Table 2-6 Decay Heat for All DOE HLW 12

Table 3-2 Commercial UNF discharged based on burn-up through April 2005. 15
Table 3-3 Commercial UNF discharged based on year of discharge through April 2005. 16
Table 3-4 Commercial UNF estimated discharges through2011. 17
Table 3-5 Comparison of Calvin and UFD Scenario 1 Assumptions 19
Table 3-6 Scenario 1, No Nuclear Replacement Results Potential Range 19
Table 3-7 No replacement nuclear generation. 25
Table 3-8 Maintain current nuclear generation. 26
Table 3-9 Projected growth to 200 GWe/yr nuclear generation in 2060. 27
Table 3-10 Projected growth to 400 GWe/yr nuclear generation in 2060. 28
Table 3-11PWR 40 GWd/MT Used Fuel Decay Heat 30
Table 3-12 PWR60 GWd/MT Used Fuel Decay Heat 31
Table 3-13 BWR 30 GWd/MT Used Fuel Decay Heat 32
Table 3-14 BWR 50 GWd/MT Used Fuel Decay Heat 33
Table 3-15 PWR 100 GWd/MT Used Fuel Decay Heat 34
Table 3-16 Commercial UNF Dry Storage by State 35
Table 3-17 Commercial UNF Dry Storage by Cask System 38
Table 4-1 Summary of off-gas waste forms. 49
Table 4-2 Summary of metal waste forms. 50
Table 4-3 Summary of principle fission product waste forms. 51
Table 4-4 Summary of Alternative Waste Forms for Captured Off-gas 63
Table 4-5 Summary of Alternative Fission Product Waste Forms 64
Table 5-1 LWR Derived MOX Fuel Summary 66
Table 5-2 MOX Fuel 50 GWd/MT Used Fuel Decay Heat 67
Table 6-1 Reactor Parameter Summary 70
Table 6-2 Overall Reactor Material Balance Result 70
Table 6-3 Advanced Burner Reactor Fuel Reprocessing Off-Gas Waste Summary 74
Table 6-4 Advanced Burner Reactor Fuel Reprocessing Metal Waste Summary 74
Table 6-5 Advanced Burner Reactor Fuel Reprocessing Fission Product Waste Summary 75
Table 7-1 Summary of Design and Core Performance Parameters of Once-Through Nuclear Systems 79
Table 7-2 Normalized UNF Production Rates (at Discharge State) 81
Table 7-3 Comparison of UNF Decay Heat at Discharge 83
Table 7-4 Decay Heat of Leading Contributors (W/t-UNF) 84
Table 8-2 Head-End Wastes 91
Table 8-3Tritium Waste 91

Table 8-4 Iodine Waste 91
Table 8-5 Krypton Waste 92
Table 8-6 Borosilicate Glass 92
Table 8-7 Silicon Carbide Hulls and UDS 93
Table 8-8 Cesium/Strontium Ceramic 93
Table 8-9 Metal Alloy 93
Table 8-10 Recovered Uranium 94
Table 9-1 Initial isotope distribution of fuel inserted into the reactor. Adapted from Raitses et al., 2009. 96
Table A-1 Physical characteristics of pressurized water reactor assembly classes. 108
TableA-2 Physical characteristics of boiling water reactor assembly classes. 110
Table A-3 Assembly types and their main characteristics as of December 31, 2002. 113
Table B-1 Scenario 1 – no replacement nuclear generation. 119
Table B-2 Scenario 2 – maintain current nuclear generation. 123
Table B-3 Scenario 3 – 200 GWe/yr projected nuclear growth. 128
Table B-4 Scenario 4 400 GWe/yr projected nuclear growth. 133
Table C-1 Pressurized water reactor radionuclide inventory at 5 and 30 year decay. 138
Table C-2 Pressurized water reactor radionuclide inventory at 100 and 500 year decay. 156
Table C-3 Boiling water reactor radionuclide inventory at 5 and 30 year decay 168
Table C-4 Boiling water reactor radionuclide inventory at 100 and 500 year decay 181
Table C-5 100 GWd/MT PWR Fuel Radionuclide Inventory at 5, 30, 100 and 500 years 191
Table D-1 Ranges of nominal properties for DOE spent nuclear fuel. 206
Table D-2 Total DOE SNF radionuclide inventory. 211
Table E–1 Radionuclide inventory for a representative Naval SNF canister 5 years after reactor shutdown 215
Table F-1 Total radionuclide inventory for each HLW glass type at 2017 218
Table F-2 Estimated total radionuclide inventory for SRS HLW as of 11/16/2006 223
Table F-3 Radionuclide Inventory for Electro-Chemical Processing of Sodium Bonded UNF. 225
Table H-2 Maximum Decay Heat (W) per Container for Borosilicate Glass 240
Table H-3 Properties of High Decay Heat Waste Forms 241
Table I-1 Commercial MOX Fuel Isotopic Data at 5, 30, 100 and 500 years Decay Time 242
Table K-1 Metal Equilibrium LWR SNF CR=0.75 257
Table K-2 Oxide Equilibrium Core LWR SNF CR=0.75 258
Table K-3 Oxide Equilibrium Core LWR SNF CR=0.5 259
Table K-4 Metal Equilibrium LWR SNF CR=0.5 260

Table L-1 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 40 GWd/MT Burn-up 5 Year Cooled 263
Table L-2 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 40 GWd/MT Burn-up 5 Year Cooled 264
Table L-3 Borosilicate Glass Decay Heat Generated by UREX Processing of 40 GWd/MT 5 Year Cooled PWR Fuel 265
Table L-5 Glass Bonded Zeolite Decay Heat Generated by Electro-Chemical Processing of 40 GWd/MT 5 year Cooled PWR Fuel 267
Table L-6 Lanthanide Glass Decay Heat Generated by Electro-Chemical Processing of 40 GWd/MT 5 Year Cooled PWR Fuel 268
Table L-7 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 60 GWd/MT Burn-up 5 Year Cooled 269
Table L-8 Borosilicate Glass Decay heat Generated by New Extraction Processing of 60 GWd/MT Burn-up 5 Year Cooled 270
Table L-9 Borosilicate Glass Decay Heat Generated by UREX Processing of 60 GWd/MT 5 year Cooled PWR Fuel 271
Table L-10 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 60 GWd/MT 5 Year Cooled PWR Fuel 272
Table L-11 Glass Bonded Zeolite Decay Heat Generated by Electro-Chemical Processing of 60 GWd.MT 5 Year Cooled PRW Fuel 273
Table L-12 Lanthanide Glass Decay Heat Generated by Electro-Chemical Processing of 60 GWd/MT 5 yer Cooled PWR Fuel 274
Table L-13 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 60 GWd/MT Burn-up 30 Year Cooled 275
Table L-14 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 60 GWd/MT Burn-up 30 Year Cooled 276
Table L-15 Borosilicate Glass Decay Heat Generated by UREX Processing of 60 GWd.MT 30 Yer Cooled PWR Fuel 277
Table L-16 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 60 GWd/MT 30 Yer Cooled PWR Fuel 278
Table L-17 Glass Bonded Zeolite Decay Heat Generated by Electro- Chemical Processing of 60 GWd/MT 30 Year Cooled PWR Fuel 279
Table L-18 Lanthanide Glass Decay Heat Generated by Electo-Chemical Processing of 60 GWd.MT 30 Year Cooled PWR Fuel 280
Table L-19 Borosilicate Glass Decay Het Generated by Co-Extraction Processing of BWR 50 GWd/MT Burn-up 5 Year Cooled 281
Table L-20 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 50 GWd/MT Burn-up 5 Year Cooled 282
Table L-21 Borosilicate Glass Decay Heat Generated by UREX Processing of 50 GWd.MT 5 Year Cooled BWR Fuel 283

Table L-22 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 50 GWd/MT 5 Year Cooled BWR Fuel 284

Table L-23 Glass Bonded Zeolite Decay Heat Generated by Electro-Chemical Processing of 50 GWd/MT 5 year Cooled BWR Fuel 285

Table L-24 Lanthanide Glass Decay Heat Generated by Electro-Chemical Processing of 50 GWd/MT 5 year Cooled BWR Fuel 286

Table M-1 New Extraction Glass Decay Heat Generated by Processing Sodium FR Oxide 0.75 CR 131 GWd/MT Burn-up 2 Year Cooled 288

Table M-2 New Extraction Glass Decay Heat Generated by Processing Sodium FR Oxide 0.5 CR 166 GWd/MT Burn-up 2 Year Cooled 289

Table M-3 Electro-Chemical Glass Bonded Zeolite Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ration of 0.75 290

Table M-4 Electro-Chemical Lanthanide Glass Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.75 291

Table M-5 Electro-Chemical Glass Bonded Zeolite Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.5 292

Table M-6 Electro-Chemical Lanthanide Glass Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.5 293

Table N-1 TRISO Particle Parameters 294

Table N-2 Fuel Compact Parameters 294

Table N-3 Fuel Element Parameters 294

Table N-4 Isotopic Distribution of Spent HTGR Fuel 295

Table N-5 Isotopic Distribution of Spent HTGR Fuel 298

Table O-1 Isotopic Distribution at the end of the first cycle 302

Table O-2 Isotopic Distribution at the end of the fourth cycle 304

Table O-3 Isotopic Distribution at the end of the eight cycle 307

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Acronyms

ABR	Advanced Burner Reactor
AEC	Atomic Energy Commission
ALWR	Advanced Light Water Reactor
ARE	Aircraft Reactor Experiment
BWR	Boiling Water Reactor
BOL	Beginning of Life
CANDLE	Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy
CDB	Characteristic Data Base
CR	TRU Conversion Ratio
DMSR	Denatured Molten Salt Reactor (DMSR)
DOE	Department of Energy
DOE-NE	Department of Energy Office of Nuclear Energy
EOL	End of Life
Em^2	Energy Multiplier Module
FCR&D	Fuel Cycle Research and Development
FMSR	Fast Mixed Spectrum Reactor
FOEAS	Follow-on Engineering Alterative Studies
FPEX	Fission Product Extraction
FR	Fast Reactor
GA	General Atomics
GSWB	Glass Waste Storage Building
GWd	Giga-Watt days
GWd/MT	Giga-Watt days per Metric Ton
GWe	Giga-Watts electric
HIP	Hot Isostatic Pressing
HLW	High Level Waste
HM	Heavy Metal
HTGR	Hight Temperature Gas Reactor
IC	Inner Core
INL	Idaho National Laboratory
LLW	Low Level Waste
LWR	Light Water Reactor

KALC	Kryton Absorption by Liquid Carbon Dioxide
MC	Middle Core
MCO	Multicanister Overpack
MOX	Mixed Oxide
MSB	Molten Salt Breeder Reactor
MSR	Molten Salt Reactor
MSRE	Molten Salt Reactor Experiment
MT	Metric Ton
MTHM	Metric Tons of Heavy Metal
MTIHM	Metric Tons of Initial Heavy Metal
MTU	Metric Tons of Uranium
MWd	MegaWatt days
NGNP	Next Generation Nuclear Plant
NNPP	Naval Nuclear Propulsion Program
OC	Outer Core
OCRWM	Office of Civilian Radioactive Waste Management
PNNL	Pacific Northwest National Laboratory
PWR	Pressurized Water Reactors
PyC	Pyrolytic Carbon
R&D	Research and Development
RW	Radioactive Waste
SRNL	Savannah River National Laboratory
SFR	Sodium-Cooled Fast Reactor
SIC	Silicon Carbide
SNF	Spent Nuclear Fuel
SRS	Savannah River Site
SSFR	Sustainable Sodium-Cooled Fast Reactor
TALSPEAK	Trivalent Actinide Lanthanide Separation by Phosphorus-based Aqueous Komplexes
TBD	To Be Determined
Tc	Technetium
THOREX	Thorium Extraction Process
TRIGA	Training, Research, and Isotope Reactors (built by GA)
TRISO	Tristructural- Isotopic
TRU	Transuranic

TRUEX	Transuranic Extraction
TWR	Traveling Wave Reactor
UDS	Undissolved Solids
UFD	Used Fuel Disposition
ULFR	Ultra-Long Life Fast Reactor
UNF	Used Nuclear Fuel
UOX	Uranium Oxide Fuel
UREX	Uranium Extraction
WG	Weapons Grade

FUEL CYCLE TECHNOLOGY

USED NUCLEAR FUEL DISPOSITION
FUEL CYCLE POTENTIAL WASTE INVENTORY FOR DISPOSITION

1. INTRODUCTION

The United States (U.S.) currently utilizes a once-through fuel cycle where used nuclear fuel is stored on-site in either wet pools or in dry storage systems with ultimate disposal envisioned in a deep mined geologic repository. Within the Department of Energy's Office of Nuclear Energy (DOE-NE), Fuel Cycle Research and Development Program (FCR&D) develops options to the current commercial fuel cycle management strategy to enable the safe, secure, economic, and sustainable expansion of nuclear energy while minimizing proliferation risks by conducting research and development focused on nuclear fuel recycling and waste management to meet U.S. needs.

The safe management and disposition of used nuclear fuel (UNF) and/or nuclear waste is a fundamental aspect of any nuclear fuel cycle. The disposal of radioactive waste of all classifications (low-, intermediate-, high-level waste (HLW), and UNF) has been investigated world-wide since the inception of nuclear power. While significant progress has been made regarding disposal, the routine disposal of UNF and radioactive waste remains problematic. Experience with the Yucca Mountain Project has illustrated the challenges of siting, characterizing, designing, and licensing a geologic repository. The capacity for disposing of low level waste (LLW) is limited, potential disposal pathways for Greater Than Class C (GTCC) LLW have yet to be identified, and the disposal of UNF and high level waste has not been demonstrated. An expansion of nuclear power in the U. S. would increase the amount of all classes of waste and requires the availability of routine disposal pathways.

To address these challenges, the mission of the Used Fuel Disposition (UFD) Campaign is to identify alternatives and conduct scientific research and technology development to enable storage, transportation, and disposal of UNF and wastes generated by existing and future nuclear fuel cycles UNF. The scope of the campaign also includes UNF and HLW generated by DOE.

Different commercial fuel cycle alternatives are expected to generate different waste streams where a variety of waste forms could potentially be used to isolate the hazardous materials. The purpose of this report is to provide an estimate of potential waste inventory and waste form characteristics for the DOE UNF and HLW and a variety of commercial fuel cycle alternatives in order to support subsequent system-level evaluations of disposal system performance. This report is envisioned as a "living document" which will be revised as specific alternative fuel cycles are developed.

The fuel cycles are expected to be categorized by three families of alternatives: 1. once-through, as currently practiced in the U.S., 2. "full fuel recycling" approaches in which no UNF is disposed (only radioactive waste from reprocessing) or 3. "modified open" cycles which are developed to achieve better fuel utilization but may still dispose of UNF as well as radioactive wastes.

1.1 Initial Issue (Revision 0)

The initial issue (revision 0) of this study analyzes potential waste quantities and properties generated by:

- DOE activities (Section 2.0). This includes DOE UNF, Naval UNF and DOE HLW resulting from the reprocessing of DOE and navy used fuel.

- Commercial UNF potentially generated by uranium fuel light water reactors (LWR). The four once through fuel cycles (Section 3) analyzed in this study differ by varying the assumed expansion/contraction of nuclear power.
- The four alternative LWR recycling processes analyzed differ in the reprocessing method (aqueous vs. electrochemical), complexity (Pu only or full transuranic (TRU) recovery), and waste forms generated. Those alternatives with full TRU recovery reflect "full recycling" category and others reflect only partial reprocessing activities which best fit the "modified open" category.

This report does not include LLW or TRU waste generated by DOE activities since these waste are disposed as they are generated. Potential LLW generated by the backend of the commercial fuel cycle are documented elsewhere.[60,61]

1.2 Revision 1

Revision 1 of the report adds information regarding alternative waste forms generated by aqueous reprocessing LWR UNF (Section 4.5 and Appendix G). Revision 1 also added Appendix H which documents container size determination for the various waste forms.

1.3 Revision 2

Revision 2 of the report adds information regarding spent Mixed Oxide (MOX) Fuel and waste generated by the reprocessing of Fast Reactor (FR) fuels.

MOX fuels derive their fissile content from various sources. Two potential sources are discussed in Section 5, Pu from the LWR UNF reprocessing, and weapons grade (WG) plutonium from surplus materials.

TRU waste derived from LWR reprocessing is proposed to be converted into FR fuel. In closed fuel cycles the used FR fuel is reprocessed. Section 6 discusses the potential waste generated through reprocessing both metal and oxide fuels discharged from FR operation at conditions such that the TRU conversion ratio (CR) is approximately 0.5 or 0.75.

Revision 2 also made minor editorial and formatting changes which did not impact the technical content of the prior materials.

1.4 Revision 3

Revision 3 of the report:

- Updates current commercial UNF inventory summaries through December 2010,
- Provides isotopic composition and decay heat as a function of time for a 100 GWd/MT (giga-watts days/metric ton) burn-up (8.32% enrichment) LWR uranium oxide (UOX) fuel as an example of an alternative fuel cycle from continuous improvement of the current U.S. reactor fleet
- Provides decay heat as a function of time for LWR UOX, MOX fuel derived from the recovered U/Pu by reprocessing 50 GWd/MT LWR UOX fuel, and the sodium FR fuels discussed in prior revisions of this report.
- Provides decay heat as a function of time for selected reprocessing wastes which are needed to continue generic repository investigations.
- Includes additional dry used fuel storage data.

Revision 3 also made minor editorial and formatting changes which did not impact the technical content of the prior materials.

1.5 Revision 4

Revision 4 of this report:

- Provides an initial waste inventory estimate from advanced once thru reactor concepts (Section 7.0). The reactors studied include the Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy (CANDLE) Reactor, the Substainable Sodium-Cooled Fast Reactor, the Mixed Fast Spectrum Reactor, the Ultra-Long Life Fast Reactor, the Energy Multiplier Module, and the Traveling Wave Reactor. The mass, radionuclide content, and decay heat as a function of time are provided for these discharged fuels.
- Provides an initial waste inventory estimate from recycling uranium oxide fuel from a representative prismatic block high temperature gas reactor fuel (Section 8.0).
- Provides an initial estimate for the isotopic distributions of discharged fuel from a thorium based multi-recycle fuel cycle. The wastes resulting from reprocessing this fuel have not been estimated. Inclusion of reprocessing wastes will be considered for future revisions.
- Corrects errors in the volume of Cs/Sr ceramic waste in Table 4-3.
- Corrects errors in the volume of Lanthanide Glass in Tables 4-3.

The data presented in this report is available in electronic format from the lead author.

Seven new elements identified by the FEP s process as potentially important to system performance were added to all material balance calculations. These additional elements are actinium, chlorine, lead, protactinium, selenium, and antimony. These elements are all minor contributors to the mass, volume, and package count of the waste forms (< 0.5 g/MT) but they are potentially important to dose projections. These isotopics are now tracked in the excel workbook associated with this Report and extracted data used by others should be regenerated to ensure these new elements areincluded.

1.6 Revision 5

Revision 5 of this report provides additional information on the decay heat for DOE SNF and HLW, updates commercial UNF discharge projections thru December 2011 and updated commercial UNF in dry storage. Four new independent UNF storage facilities have started operations since the last revision in October 2011.

The excel work book associated with this report was subjected to a quality assurance (QA) audit. The audit was performed by Pacific Northwest National Laboratory personnel who provided an independent review for the Savannah River National Laboratory (SRNL). The audit reveal minor errors in the caculations to estimate wastes from reprocessing but, these did not have significant impact on the estimated mass and volume of these wastes. The QA reports are referenced in the body of this report.

1.7 Study Limitations

This study provides a common data source for current inventory and projections that are traceable, transparent, and reproducible. The intent of this study is to provide enough information to support both a general description (piece count, metric tons of uranium [MTU], etc) and specific details (isotopes, burn-up, etc.) of the material. The high level description may be used to support general discussions or presentations. Either or both levels of descriptions may be used in specific analyses or modeling.

The type and amount of inventory information needed are dependent upon the specific use of the data. Although this study attempts to provide sufficient inventory information for a wide range of uses, the specific needs of the users may dictate varying the data set. Therefore, further information sets can be created using special queries on request. Information can be provided in spreadsheet format to facilitate data transfer without the need for retyping, which is inefficient and causes errors.

The very nature of future waste characteristics and inventory projections makes them somewhat uncertain. For example, this study includes investigation of the expansion/contraction of nuclear power generation by use of four scenarios. Inclusion of any specific scenario in this study is not an evaluation or endorsement of the scenario. Rather a range of scenarios is used to illustrate the range of waste quantity that may result. A similar approach is taken to illustrate the range in waste form characteristics by using various reprocessing methods. The burden is on the user to utilize an appropriate data set for their task.

The parameters provided for each waste form are based on the specific waste form and the available information. For example, those waste forms that are in canisters (or are planned to be) include the canister information (numbers, dimensions, etc). Radionuclide inventory is supplied for each waste form, although the level of details may vary depending on the availability of information (isotopes, decay time, maximum vs. nominal, etc).

This report uses initial, also known as beginning of life (BOL), uranium mass (i.e., MTU) values when reporting inventory for commercial UNF. This is the value typically reported by utilities and the units in which the data were collected. Initial MTU and Metric Tons Initial Heavy Metal (MTIHM) are the same for commercial UNF since uranium is the only heavy metal present. The difference between initial MTU and end of life (EOL) MTU is small (<4%) for commercial UNF because fissile uranium is less than 5% of the heavy metal.

For DOE and Naval UNF, final or end of life (EOL) MTHM (i.e., uranium, plutonium, and thorium) quantities are reported. This is considered appropriate because: 1) some DOE fuels are plutonium or thorium based so MTU is not appropriate, and 2) because of high-enrichment in some DOE and all Naval fuel, burn-up can reduce the MTHM by over 50% from BOL to EOL.

2. DOE ACTIVITY WASTE

Since the inception of nuclear reactors, the DOE and its predecessor agencies operated or sponsored a variety of research, test, training, and other experimental reactors both domestically and overseas. The Naval Nuclear Propulsion Program (NNPP) has generated UNF from operation of nuclear powered submarines and surface ships, operation of land-based prototype reactor plants, operation of moored training ship reactor plants, early development of commercial nuclear power, and irradiation test programs.

Aqueous reprocessing of DOE UNF has occurred at the Hanford Site, the Idaho National Laboratory (INL), and the Savannah River Site (SRS). The INL is pursuing the use of electro-chemical processing to treat 60 MTHM of sodium bonded UNF. DOE is also responsible for clean-up of the commercial UNF reprocessing site at West Valley, New York.

The waste requiring disposition from these DOE activities are fairly well understood and documented. This section summarizes these wastes.

2.1 DOE Used Nuclear Fuel

DOE UNF is primarily generated by DOE production reactors, demonstration commercial power reactors, and domestic and foreign research and training reactors. DOE UNF includes some commercial UNF not in the possession of NRC-licensed commercial utilities.

Over the past several decades, since the inception of nuclear reactors, the DOE and its predecessor agencies operated or sponsored a variety of research, test, training, and other experimental reactors with different characteristics from the commercial power reactors of today. DOE UNF generated in production reactors supported weapons and other isotope production programs. An example of UNF existing today from production reactors is the N Reactor UNF stored at Hanford. Some UNF from commercial power reactors, such as Shippingport, Peach Bottom, and Fort St. Vrain, is stored within the

DOE complex. This UNF was generated for commercial power demonstration purposes or as part of research projects. Also, the Three Mile Island Unit 2 UNF debris is stored at the INL.

DOE has sponsored nuclear research activities in the U. S. and overseas. There are numerous university and government research reactor sites within the United States. UNF from research reactors is stored primarily at the INL and SRS. Examples of research reactor UNF being stored within the DOE complex include the High-Flux Beam Reactor UNF stored at the SRS; the Fast Flux Test Facility UNF stored at Hanford and the INL; training, research, and isotope reactors (built by GA) (TRIGA) UNF stored at Hanford and the INL; and the Advanced Test Reactor UNF stored at the Idaho National Laboratory. Additional research reactor UNF is being returned to the U. S. from foreign research reactors as part of the DOE Foreign Research Reactor Spent Nuclear Fuel Return Program.

2.1.1 DOE UNF Inventory

The source of current inventory data for this study is information collected in support of the Office of Civilian Radioactive Waste's Management (OCRWM) efforts for licensing the Yucca Mountain Repository.[1] Complex wide DOE UNF data has been collected and is maintained in the Spent Fuel Database. The Spent Fuel Database contains the following information: inventory data by site, area, and facility; physical characteristics; chemical composition of the fuel compound, cladding, and other significant constituent components; burn-up data; source term data. Based on the available data, queries can be used to sort this data and present information on any combination of the fields identified above. Based on these queries, reports or excel files can be generated.

The majority of DOE UNF (about 2500 metric tons of heavy metal (MTHM) that has been generated is in storage. DOE continues to operate several research reactors and will be receiving UNF from universities and the foreign research reactor return program. Projected material amounts are relatively small (about 50 MTHM) and there is some uncertainty as to the total amount that will be generated or received. The inventory discussed below covers all DOE UNF, UNF which is currently in DOE storage and projected UNF generation or receipts.

DOE UNF comes from a wide range of reactor types, such as light- and heavy-water-moderated reactors, graphite-moderated reactors, and breeder reactors, with various cladding materials and enrichments, varying from depleted uranium to over 93% enriched U235. Many of these reactors, now decommissioned, had unique design features, such as core configuration, fuel element and assembly geometry, moderator and coolant materials, operational characteristics, and neutron spatial and spectral properties.

As described below, there is a large diversity of reactor and fuel designs. In addition, there is a relatively large number (over 200,000) of fuel pieces or assemblies, which range from a large number of pieces for some reactors (N Reactor) to a few individual pieces for other unique reactors (Chicago Pile-5 converter cylinders).

There are several hundred distinct types of DOE UNF, and it is not practical to attempt to determine the impact of each individual type when performing analyses covering all of this material such as considering repository performance. To support these types of analyses, the DOE UNF inventory was reduced to 34 DOE UNF groups based on fuel matrix, cladding, cladding condition, and enrichment. These parameters are the fuel characteristics that were determined to have major impacts on the release of radionuclides from the DOE UNF and contributed to nuclear criticality scenarios.

A discussion of each of the 34 groupings is presented in Appendix D. The discussions of each of the 34 groups provide a description of the fuel group and an example of fuel that makes up the group. When appropriate, a more detailed description of a fuel with the largest percentage of MTHM within each group is provided. This discussion is not intended to address each fuel in the group.

Appendix D Table D-1 describes the typical ranges of the nominal properties for DOE UNF in the 34 groups.

2.1.2 DOE UNF Radionuclide Inventory

Process knowledge and the best available information regarding fuel fabrication, operations, and storage for DOE UNF is used to develop a conservative source-term estimate. The DOE UNF characterization process relies on precalculated results that provide radionuclide inventories for typical UNF at a range of decay times. These results are used as templates that are scaled to estimate radionuclide inventories for other similar fuels.

To estimate an UNF source term, the appropriate template is selected to model the production of activation products and transuranics by matching the reactor moderator and fuel cladding, constituents, and beginning-of-life enrichment. Precalculated radionuclide inventories are extracted from the appropriate template at the desired decay period and then scaled to account for differences in fuel mass and specific burn-up. Appendix D, Table D-2 lists the projected radionuclide inventory of DOE UNF for the nominal and bounding cases as of 2010. The nominal case is the expected or average inventory. The bounding case represents the highest burn-up assembly or accounts for uncertainties if fuel burn-up is not known.

2.1.3 DOE UNF Storage/Canisters

Although DOE UNF is stored throughout the U. S. at numerous facilities, a decision was made in 1995 to consolidate DOE UNF at three existing DOE sites; Hanford Site in Washington, the INL in Idaho, and the SRS in South Carolina. The vast majority of DOE UNFs are currently stored at these three sites. The storage configurations vary for each of the sites and include both dry and wet storage. On a MTHM basis, a large portion of the UNF (about 2100 MTHM) is contained in about 400 sealed canisters. The majority of the remaining UNF will most likely be placed in canisters.

In support of the Yucca Mountain project it was anticipated that all DOE UNF, except a very small amount of DOE UNF of commercial origin, would be placed in sealed disposal canisters. At the Hanford Site, about 400 Multicanister Overpacks (MCO) were used to package and store UNF. The MCO is a sealed stainless steel canister which is about 24 inches in diameter and about 14 feet long. For the remaining DOE UNF, a standardized disposal canister design was developed which included canisters of 18 and 24 inch diameter and 10 and 15 feet length. Because of uncertainty in disposal and packaging efficiencies the total number of canisters to be generated ranged from 2,500 to 5,000 with a point estimate of 3,542. Currently no UNF has been packaged into the standardized disposal canister designDecay heat of DOE Spent Nuclear Fuel (SNF) is based on the estimated radionuclide inventory. In support of the Yucca Mountain License Application, an analytical process using process knowledge and the best available information regarding fuel fabrication, operations, and storage for DOE SNF was used to develop a conservative radionuclide inventory estimate. This methodology was applied to each fuel in the DOE SNF inventory to develop a radionuclide estimate. Also in support of the Yucca Mountain License Application, a packaging plan was developed using the DOE standardized canisters. These two data sources are used to estimate the decay heat per canister for DOE SNF.

The radionuclide and resulting decay heat was calculated in the year 2010 and 2030 to support the Yucca Mountain repository. Considering the time required before a repository for DOE SNF would be open to accept waste, these values are considered adequate for this scoping evaluation.

Table 2-1 provides the distribution of DOE SNF canisters based on the 2010 and 2030 nominal decay heat using the 2035 total canister count. The 2010 data indicates approximately 35% of the DOE SNF canisters will be less than 50 watts. Approximately 90% of the DOE SNF canisters will be less than 300 watts. Nearly all the DOE SNF canisters (>99%) will be less than 1 kW. Since the methodology used to

calculate the radionuclide inventory is very conservative, some fuels have radionuclide amounts based on bounding assumptions resulting in extreme decay heat values.

Table 2-1 DOE Spent Nuclear Fuel Canister Decay Heat

	2010		2030	
Decay heat per canister (watts)	Number of canisters	Cumulative %	Number of canisters	Cumulative %
<50	1228	34.7%	1670	47.1%
50-100	565	50.6%	392	58.2%
100-220	789	72.9%	690	77.7%
220-300	633	90.8%	586	94.2%
300-500	241	97.6%	140	98.2%
500-1000	55	99.1%	41	99.4%
1000-1500	10	99.4%	4	99.5%
1500 - 2000	1	99.4%	5	99.6%
>2000	20	100.0%	13	100.0%
Total	**3542**		**3542**	

Does not include the Savannah River Site SRE fuel

2.2 NAVAL UNF

The Naval Nuclear Propulsion Program (NNPP) has generated UNF from operation of nuclear powered submarines and surface ships, operation of land-based prototype reactor plants, operation of moored training ship reactor plants, early development of commercial nuclear power, and irradiation test programs. The source of naval UNF information for this report is the unclassified portion of the Yucca Mountain Repository License Application.[1] Since most details regarding naval UNF are classified, only limited information is presented herein. Before using the information in this section for studies involving naval UNF, contact the NNPP Program Manager, Navy Spent Nuclear Fuel at (202) 781-6214.

2.2.1 Naval UNF Inventory

Naval UNF consists of solid metal and metallic components that are nonflammable, highly corrosion-resistant, and neither pyrophoric, explosive, combustible, chemically reactive, nor subject to gas generation by chemical reaction or off-gassing. Approximately 27 MTHM of Naval UNF currently exists with a projected inventory of less than 65 MTHM in 2035.

New naval nuclear fuel is highly enriched (approximately 93 wt % to 97 wt %) in the isotope U235. As a result of the high uranium enrichment, very small amounts of transuranics (TRU) are generated by end of life when compared to commercial UNF. The Zircalloy cladding of naval nuclear fuel provides primary containment for the radioactive fission products. Development of naval fuel systems has been supported by a long-standing program of examination of irradiated test specimens and naval UNF after removal from prototype reactor plants and operating ships. A small percentage of Navy UNF has been disassembled for examinations. In most cases, the disassembled naval UNF assemblies have intact cladding and no exposed actinides or fission products. In a few cases, destructive evaluations of disassembled components result in nonintact cladding and exposed fission products and actinides; some test specimens have nonintact cladding either because they were tested until the cladding failed or were tested with intentional defects.

2.2.2 Naval UNF Radionuclide Inventory

In support of the Yucca Mountain repository, three different methods for packaging naval UNF into naval UNF canisters were developed; however, the design of the naval UNF canister is the same irrespective of packaging method. These packaging methods are based on the type of naval UNF assemblies and whether the naval UNF cladding is intact or nonintact. Each naval UNF canister would be loaded such that thermal, shielding, criticality, and other characteristics of the received waste would be within the proposed repository waste acceptance requirement limits. As a result, a radionuclide inventory for a representative naval UNF canister, five years after reactor shutdown, was developed for use in the repository source term analyses (Appendix E, Table E-1). Different packaging designs may be needed dependant upon the future disposal options.

2.2.3 Naval UNF Storage/Canisters

UNF from the NNPP is temporarily stored at the INL. To accommodate different naval fuel assembly designs, naval UNF is loaded in either a naval short UNF canister or a naval long UNF canister. Both were sized to fit within the proposed design for the Yucca Mountain repository waste package.

The outer diameter of the naval UNF canister is 66 in. nominal (66.5 inches maximum). The maximum external dimensions ensure naval UNF canisters fit into the waste packages. The naval short UNF canister is 185.5 inches (nominal) in length (187 inches maximum), and the naval long UNF canister is 210.5 inches (nominal) in length (212 inches maximum). With the exception of length, the other characteristics of naval UNF canisters are identical.

Approximately 400 naval UNF canisters (310 long and 90 short) are currently planned to be packaged and temporarily stored pending shipment. The average thermal load is 4,250 watts/container. Maximum is 11,800 watts/container. The NNPP is responsible for preparing and loading naval UNF canisters and began canister loading operations in 2002. As of February 2010, 27 naval UNF canisters have been loaded and are being temporarily stored at INL.

2.3 DOE HIGH LEVEL WASTE

High-level radioactive waste is the highly radioactive material resulting from the reprocessing of UNF. Following aqueous reprocessing, HLW is in a liquid form and initially stored in underground metal storage tanks. Long term storage of HLW requires stabilization of the wastes into a form that will not react, nor degrade, for an extended period of time. Two treatment methods used for stabilization of the waste are vitrification or calcination. Vitrification is the transition of the HLW into a glass by mixing with a combination of silica sand and other constituents or glass forming chemicals that are melted together and poured into stainless steel canisters. HLW canisters have a nominal diameter of 2 feet and have heights of 10 or 15 feet. Calcination of HLW is accomplished by injecting the waste with calcining additives into a fluidized bed to evaporate the water and decompose the remaining constituents into a granular solid material.

Aqueous reprocessing of DOE UNF has occurred at the Hanford Site, the INL, and the SRS. Commercial UNF was reprocessed at West Valley, New York.

In addition to aqueous reprocessing, the INL is pursuing the use of electro-chemical processing to treat 60 MTHM of sodium bonded UNF. The process converts the bond sodium into sodium chloride and separates the UNF into a uranium product and HLW. The HLW is produced in two forms, ceramic and metal. The ceramic waste form primarily contains the salt electrolyte with active metal fission products and the metal waste is primarily the cladding hulls and undissolved noble metals. The process has been demonstrated and used to treat about 4 MTHM of sodium bonded UNF to date.

2.3.1 Current DOE HLW Inventory

The source of inventory data for this study is information collected in support of the OCRWM efforts for licensing the Yucca Mountain Repository.[1] In addition, site treatment plans were also used.[8, 9]

A commercial fuel reprocessing plant located at West Valley, New York operated from 1966 through 1972 and reprocessed approximately 640 metric tons of UNF to recover the unused uranium. Of the UNF reprocessed at West Valley, about 260 metric tons was commercial used fuel and about 380 metric tons was DOE N Reactor used fuel. During operations about 2,500 m^3 of liquid HLW was generated. The liquid HLW was vitrified between 1996 and 2001 producing 275 HLW canisters that are stored at West Valley.[11]

The INL reprocessed UNF from naval propulsion reactors, test reactors, and research reactors to recover uranium and generated approximately 30,000 m^3 of liquid HLW. Between 1960 and 1997, the INL converted all of their liquid HLW into about 4,400 m^3 of a solid waste form called calcine (a granular solid with the consistency of powder laundry soap). These solids are stored retrievably on-site in stainless steel bins (like grain silos but smaller) within concrete vaults.

The SRS has reprocessed defense reactor UNF and nuclear targets to recover valuable isotopes since 1954 producing more than 530,000 m^3 of liquid HLW. Through evaporation and vitrification of the waste, SRS has reduced this inventory to the current level about 136,000 m^3 of liquid HLW[17]. SRS began vitrifying liquid HLW in 1996 and through June 2012 has produced 3,496 HLW canisters[17] (2 feet × 10 ft).

The Hanford Site reprocessed defense reactor UNF since the 1940s and has generated about 220,000 m^3 of liquid HLW to recover the plutonium, uranium, and other elements for defense and other federal programs. Construction of a vitrification facility is currently underway with startup scheduled for 2019.

Table 2-2 summarizes the current HLW inventory.

Table 2-2 Current high level waste inventory.

	HLW Canisters[1]	Liquid HLW[2] (m^3)	Dry HLW[3] (m^3)
West Valley	275	N/A	N/A
Hanford	N/A	220,000	N/A
INL	N/A	N/A	4,400
SRS	3,496	136,000	N/A

1. Vitrified HLW in stainless steel canisters
2. HLW stored in tanks
3. Calcined HLW stored in bins.

The Hanford Site encapsulated Cs and Sr separated from the HLW between 1974 and 1985. Some of these capsules were leased to companies as radiation sources. After one of the capsules developed a microscopic leak, the capsules were recalled. Hanford is storing 1,335 Cs capsules and 601 Sr capsules, approximately 109 million curies (as of 8/8/06).[10]

2.3.2 Projected DOE HLW Inventory

SRS currently has the only operating reprocessing facility, H Canyon. It is estimated that an additional 17,000 m^3 of liquid HLW may be generated with continued canyon operations (approximately 2019).

The projected number of HLW canisters to be generated at each site will be dependent on actual loading and final waste form. Because of this uncertainty, the actual number of HLW canisters produced may vary significantly from what is anticipated today.

SRS began conversion of the liquid defense waste into borosilicate glass in 1996 and is the only DOE site with HLW in a packaged configuration. A total of 3496 canisters have been produced through June, 2012. Therefore, the SRS inventory can be described as those canisters in the current inventory and those projected from future operations. Decay heat of the current inventory is based on radiological inventories contained in the production records for those canisters. The decay heat of future canisters is estimated based the on radionuclide composition of the HLW inventory remaining in the liquid waste storage tanks. The radionuclide and resulting decay heat was calculated based on the year the canister is/will be produced. The total Savannah River canister count is based on information supporting Savannah River Liquid Waste Disposition Plan revision 16.

Table 2-3 provides the canister distribution of SRS canisters based on the nominal decay heat at the time of production. The data indicates: 39% of the Savannah River canisters will be less than 50 watts; 96% of the Savannah River canisters will be less than 300 watts; all the SRS canisters will be less than 500 watts.

Table 2-3 Savannah River Canister Decay Heat Distribution

	Savannah River	
Decay heat per canister (watts)	Number of canisters	Cumulative %
<50	2948	39.0%
50-100	459	45.1%
100-220	3891	96.5%
220-300	0	96.5%
300-500	264	100.0%
500-1000	0	100.0%
1000-1500	0	100.0%
1500 - 2000	0	100.0%
>2000	0	100.0%
Total	**7,562**	
Total Decay Heat (watts)	**805,500**	

The Hanford Waste Treatment Project (WTP) is currently under construction and therefore the Hanford borosilicate glass canisters are based on a projected inventory for their future production taken from the January 2011 Waste Treatment Plant document titled “2010 Tank Utilization Assessment”. The data in Table 2-4 indicates: 83% of the Hanford canisters will be less than 50 watts; and 100% of the Hanford canisters will be less than 300 watts.

At INL several options were considered for ultimate disposal of the calcine. Alternatives included direct disposal, vitrification, or hot isostatic pressing (HIP) to compress the calcine into a volume reduced monolithic waste form. A Record of Decision issued December 2009 determined that DOE will use the HIP technology to treat the calcine.

Decay heat of DOE HLW that has been calcined and is currently stored at the Idaho site is taken from the October 2005 Idaho Cleanup Project document titled “Decay Heat and Radiation from Direct Disposed Calcine”, EDF-6258 revision 0. Report EDF-6258 provides this data for direct disposal of the calcine waste. The current Record of Decision for disposal of the calcine is for it to be treated using a hot isostatic

pressing (HIP), which will result in an approximate 50% increase in the volume of material in each disposal canister and a 50% increase in the decay heat per canister.

Table 2-4 provides the distribution of DOE calcine canisters based on the nominal decay heat in the year 2016. The data indicates that 100% of calcine canisters will be less than 50 watts.

Table 2-4 Hanford and Idaho waste Inventory

	Hanford Borosilicate Glass		**Idaho Calcine**	
Decay heat per canister (watts)	Number of canisters	Cumulative %	Number of canisters	Cumulative %
<50	9291	83.9%	4391	100.0%
50-100	1237	95.0%		
100-220	523	99.7%		
220-300	28	100.0%		
300-500	0	100.0%		
500-1000	0	100.0%		
1000-1500	0	100.0%		
1500 - 2000	0	100.0%		
>2000	0	100.0%		
Total	**11,079**		**4391**	

Table 2-5 shows the estimated number of HLW canisters to be produced. The current best estimate and a potential range are provided.[12, 13, 17]

Table 2-5 Projected total number of high level waste canisters.

	HLW Canisters[1] Best Estimate	Potential HLW Canister Range
West Valley	275[2]	NA[2]
Hanford	11,079	9,667-14,111
INL (Calcine)	4,391	1,190-11,200
INL (Electro-chemical processing)	102	82-135
SRS	7,562	5,862-7,900
Total	98,759	17,100-33600[3]

1. With the exception of Hanford all HLW canisters are 2 feet × 10 feet, Hanford HLW canisters are 2 feet × 15 feet
2. All the West Valley HLW canisters currently exist
3. Rounded to nearest 100 canister

The combined inventory from all three sites, which is used in Cases 2 and 3 is presented in Table 2-6. The data indicates: 72% of the HLW canisters will be less than 50 watts; ~80% of the canisters will be less than 100 watts; almost 99% will be less than 300 watts and all the canisters will be less than 500 watts. The total decay heat to be emplaced in these cases is 1.2 million watts.

Table 2-6 Decay Heat for All DOE HLW

	All DOE HLW Canisters	
Decay heat per canister (watts)	Number of canisters	Cumulative %
<50	16630	72.2%
50-100	1696	79.6%
100-220	4414	98.7%
220-300	28	98.9%
300-500	264	100.0%
500-1000	0	100.0%
1000-1500	0	100.0%
1500 - 2000	0	100.0%
>2000	0	100.0%
Total	**23,032**	
Total Decay Heat (watts)	**1,203,103**	

Not included in Table 3-3 are a) 275 HLW canisters from West Valley which have low heat values, and b) the Idaho HLW to be processed through the Integrated Waste Treatment Unit and then per the associated Record of Decision will be disposed of as RH-TRU.

2.3.3 DOE HLW Radionuclide Inventory

Appendix F, Table F-1 lists the total HLW radionuclide inventory for each of the generating sites decayed to 2017. This inventory was developed by RW for the license application of Yucca Mountain based on estimates provided by the sites.[1] Although, there may be some variation in the number of canisters produced for the sites that have not completed waste treatment, the total amount of radionuclide will not change.

RW used the "projected maximum" inventory on a per canister basis for the HLW curie content supplied by SRS. The use of the "projected maximum" on a per canister basis resulted in a conservative total curie content for SRS that is approximately twice the actual SRS tank farm inventory. The expected curie content of SRS HLW is presented in Table F2.[14]

SRS is also the only site continuing reprocessing and the DOE-EM program periodically requires disposal of special isotopes via the reprocessing facility and the vitrification process. For example excess weapons plutonium has been disposed which results in the Pu concentration of some SRS canisters to be above the projected maximum inventory used in the licensing of Yucca Mountain. The potential for future EM special isotope disposal campaigns has not been assessed in this study.

The total radionuclide inventory for treatment of sodium bonded UNF is shown in Table F-3.[13]

2.3.4 DOE HLW Storage

The HLW vitrified glass at SRS is stored in below grade concrete vaults, called Glass Waste Storage Buildings (GWSB), containing support frames for vertical storage of 2,262 HLW canisters. SRS currently has two GWSB constructed and a third planned. The HLW canisters at West Valley are currently stored in a shielded cell in the former reprocessing plant. Alternate interim storage options are being considered at West Valley to allow decommissioning of the reprocessing facility.

3. COMMERCIAL LIGHT WATER REACTOR UNF- ONCE-THRU FUEL CYCLE

Commercial nuclear power plants have operated in the U. S. since about 1960. There are currently 104 operating nuclear power plants. UNF from these operating plants is currently stored on site in pools or dry storage casks with disposal in a geologic repository envisioned in a once-thru fuel cycle. In addition, UNF from 14 shutdown reactors is currently stored on the reactor sites. The General Electric facility at Morris, Illinois is currently the only UNF licensed storage facility in operation that is not collocated at a reactor site.

Commercial UNF includes irradiated fuel discharged from pressurized water reactors (PWRs) and boiling water reactors (BWRs). The fuel used in these reactors consists of uranium dioxide pellets encased in zirconium alloy (Zircaloy) or stainless steel rods. The fuel assemblies vary in physical configuration, depending upon reactor type and manufacturer.

Commercial UNF assemblies are categorized by physical configuration into 22 classes: 16 PWR and 6 BWR fuel assembly classes. Commercial UNF data has been collected by the Energy Information Administration for the Office of Civilian Radioactive Waste Management. Appendix A, Tables A-1 and A-2 present the assembly class, array size, fuel manufacturer, assembly version, assembly type code, length, width, and cladding material of commercial PWR UNF and commercial BWR UNF, respectively. Physical dimensions are those of unirradiated assemblies. Within an assembly class, assembly types are of a similar size. There are 134 individual fuel assembly types in these classes. Appendix A, Table A-3 presents the manufacturer, initial uranium load, enrichment, and burn-up characteristics of commercial UNF assembly types.

3.1 Current Commercial UNF Inventory

The source of current inventory data for this study is information collected in support of the Office of Civilian Radioactive Waste's (RW) efforts for licensing the Yucca Mountain Repository.[1.] Information collected from RW-859 forms is available on an assembly basis for UNF discharges from 1968 through 2002. Data is also available that was collected to support RW activities on a batch bases for fuel discharges from 1968 through April 2005.[2]

The specific UNF data available from these sources are:

1. reactor type (PWR or BWR)
2. number of assemblies
3. burn-up by assembly or batch
4. date of discharge
5. initial uranium loading
6. initial enrichment

Queries are used to sort the available data and present information on any combination of the fields identified above. Based on these queries, reports or excel files can be generated. Using these results, with the radionuclide inventory presented in Section 3.3, the total radionuclide inventory can be

calculated at a specified decay time. This information could also be used to calculate thermal output for the fuel at a specified decay time.

Table 3-1 provides a summary of the UNF discharged through April 2005. Table 3-2 and Table 3-3 provide data based on burn-up ranges and year of discharge respectively. These are examples of the type of queries that can be developed for this information.

Table 3-1 Summary of commercial UNF discharges through April 2005.

	Number of Assemblies Discharged	Total Initial Uranium (MTU)	Initial Uranium Loading		Initial Enrichment		Discharge Burn-up	
			Average (kg per assembly)	Maximum (kg per assembly)	Average (wt % ^{235}U)	Maximum (wt % ^{235}U)	Average (MWd/MTU)	Maximum (MWd/MTU)
PWR	79,242	34,230	432.0	546.6	3.56	4.95	37,440	69,452
BWR	105,417	18,788	178.2	197.6	2.88	4.24	30,108	65,149
Total	184,659	53,018						

Table 3-2 Commercial UNF discharged based on burn-up through April 2005.

Burn-up Bins (MWd/MT)	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Latest Discharge Date	
	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
0 - 5,000	57	2,297	2,354	15	421	436	2.79	1.29	Oct-97	Aug-97
5,000 - 10,000	375	2,052	2,427	146	372	518	3.28	1.64	Dec-96	Apr-98
10,000 - 15,000	1,383	4,916	6,299	579	893	1,472	2.14	2.11	Sep-97	May-98
15,000 - 20,000	3,531	7,539	11,070	1,544	1,342	2,886	2.29	2.16	Sep-97	Oct-98
20,000 - 25,000	2,860	13,073	15,933	1,252	2,396	3,648	2.71	2.40	Oct-98	Sep-98
25,000 - 30,000	7,713	19,017	26,730	3,294	3,457	6,751	2.95	2.70	Feb-99	Mar-05
30,000 - 35,000	12,709	18,342	31,051	5,489	3,275	8,764	3.24	3.03	Sep-04	Apr-05
35,000 - 40,000	17,063	21,164	38,227	7,397	3,691	11,088	3.57	3.32	Apr-05	Apr-05
40,000 - 45,000	16,673	11,016	27,689	7,224	1,908	9,132	3.92	3.58	Apr-05	Apr-05
45,000 - 50,000	11,636	4,344	15,980	5,040	748	5,788	4.16	3.81	Apr-05	Apr-05
50,000 - 55,000	4,254	1,429	5,683	1,830	245	2,074	4.37	3.99	Apr-05	Apr-05
55,000 - 60,000	890	228	1,118	380	39	419	4.56	4.24	Apr-05	Apr-05
60,000 - 65,000	98	0	98	40	N/A	40	4.73	N/A	Apr-05	N/A
Totals:	79,242	105,417	184,659	34,230	18,788	53,018				

Table 3-3 Commercial UNF discharged based on year of discharge through April 2005.

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up (MWd/MTU)	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
1968	0	5	5	0	1	1	N/A	2.87	N/A	1,727
1969	0	96	96	0	10	10	N/A	2.00	N/A	15,267
1970	99	29	128	39	6	45	3.11	2.13	18,422	329
1971	113	413	526	44	65	109	3.32	2.10	23,818	7,819
1972	282	801	1,083	100	146	246	3.36	2.12	22,064	7,172
1973	165	564	729	67	93	161	3.05	2.38	24,159	12,965
1974	575	1,290	1,865	208	242	449	3.02	2.27	18,430	12,900
1975	797	1,223	2,020	322	226	548	2.68	2.19	18,260	16,882
1976	932	1,666	2,598	401	298	700	2.61	1.96	22,411	13,322
1977	1,106	2,047	3,153	466	383	850	2.71	2.06	25,224	16,704
1978	1,665	2,239	3,904	699	384	1,082	2.77	2.21	26,385	19,273
1979	1,656	2,131	3,787	718	400	1,118	2.73	2.32	27,201	22,307
1980	1,456	3,330	4,786	618	620	1,238	2.91	2.37	29,839	22,372
1981	1,626	2,467	4,093	694	459	1,153	2.96	2.47	30,319	23,860
1982	1,491	1,951	3,442	640	357	998	2.93	2.51	29,851	24,616
1983	1,787	2,646	4,433	775	482	1,257	2.96	2.63	30,230	26,659
1984	1,933	2,735	4,668	839	498	1,337	2.93	2.70	29,493	25,740
1985	2,034	2,989	5,023	860	543	1,403	3.05	2.52	31,970	23,450
1986	2,254	2,551	4,805	979	458	1,437	2.97	2.23	30,666	21,237
1987	2,567	3,393	5,960	1,097	611	1,708	3.03	2.38	31,400	21,961
1988	2,583	2,956	5,539	1,098	536	1,633	3.18	2.41	33,724	24,204
1989	2,742	3,803	6,545	1,195	693	1,888	3.18	2.33	32,728	22,505
1990	3,474	3,487	6,961	1,500	633	2,133	3.24	2.43	34,422	25,074
1991	2,810	3,192	6,002	1,222	576	1,798	3.38	2.67	35,446	28,329
1992	3,629	3,808	7,437	1,567	690	2,257	3.49	2.68	36,689	29,247
1993	3,423	3,883	7,306	1,486	700	2,186	3.63	2.83	39,122	30,646
1994	2,800	3,776	6,576	1,199	676	1,874	3.69	3.08	40,222	33,404
1995	3,808	4,425	8,233	1,660	787	2,447	3.72	3.03	40,615	33,066
1996	3,594	4,823	8,417	1,540	856	2,396	3.76	3.11	38,884	35,085
1997	3,532	3,896	7,428	1,563	682	2,244	3.78	3.18	40,331	35,820
1998	2,322	3,880	6,202	1,002	677	1,679	3.96	3.23	43,303	36,419
1999	3,677	4,184	7,861	1,610	723	2,333	4.04	3.25	43,554	37,228
2000	3,154	4,405	7,559	1,385	764	2,149	4.13	3.36	44,968	38,138
2001	3,190	4,585	7,775	1,402	792	2,194	4.19	3.48	45,242	40,335
2002	3,498	3,966	7,464	1,521	688	2,208	4.29	3.67	46,770	40,960

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up (MWd/MTU)	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2003	3,504	4,661	8,165	1,547	806	2,353	4.31	3.69	46,350	42,627
2004	3,201	4,105	7,306	1,401	712	2,113	4.33	3.85	46,950	43,447
April 2005	1,763	3,016	4,779	765	520	1,285	4.33	3.82	46,663	42,964
Totals:	79,242	105,417	184,659	34,230	18,788	53,018				

To develop an inventory estimate through2011, fuel discharge predictions developed for the Nuclear Energy Institute in 2005 were used to estimate the number of assemblies and metric tons of uranium[3]. To estimate the average enrichment and burn-up through2011, projections made by utilities as part of the RW-859 surveys were used. These projections are documented in Office of Civilian Radioactive Waste Management's (DOE) (OCRWM's) "Calculation Method for the Projection of Future Spent Fuel Discharges", February 2002.[4] These projections identified a burn-up increase of 2.38% per year for BWR fuel and 1.11% per year for PWR fuel through 2010. The enrichment increased at the same rate as burn-up. Comparison of these projections made in 1998 to actual data collected through 2004 show very good agreement (PWR - actual 46,950 MWd/MTU vs. projected 46,922 MWd/MTU; BWR - actual 43,447 MWd/MTU vs 42,787 projected MWd/MTU). Table 3-4 provides an estimate of the commercial UNF discharged through2011.

Table 3-4 Commercial UNF estimated discharges through2011.

Total Number of Assemblies[b]			**Total Initial Uranium (MTU)[a]**			**Average Enrichment**		**Average Burnup (MWd/MTU)[c]**		**Average Age (Yr)**		**Total Radioactivity (Ci)**	
PWR	**BWR**	**Totals**	**PWR**	**BWR**	**Totals**	**PWR**	**BWR**	**PWR**	**BWR**	**PWR**	**BWR**	**PWR**	**BWR**
101,100	133,000	234,200	43,900	23,700	67,600	3.77	3.17	40,000	33,900	15.3	15.9	16 billion	7 billion
a the estimated fuel discharged has been rounded to the nearest 100 MTU, totals may no appear to sum correctly													
b the number of assemblies has been rounded to the nearest 200, totals may not appear to sum correctly													
c the burn-up has been rounded to the next 100 MWd/MT													

Figure 3-1 provides a distribution of this estimated inventory as a function of burn-up. Nearly 100% of the fuel currently being discharged exceeds the "high burn-up" threshold of 45,000 MWd/MTU (mega-watts days/metric ton uranium).

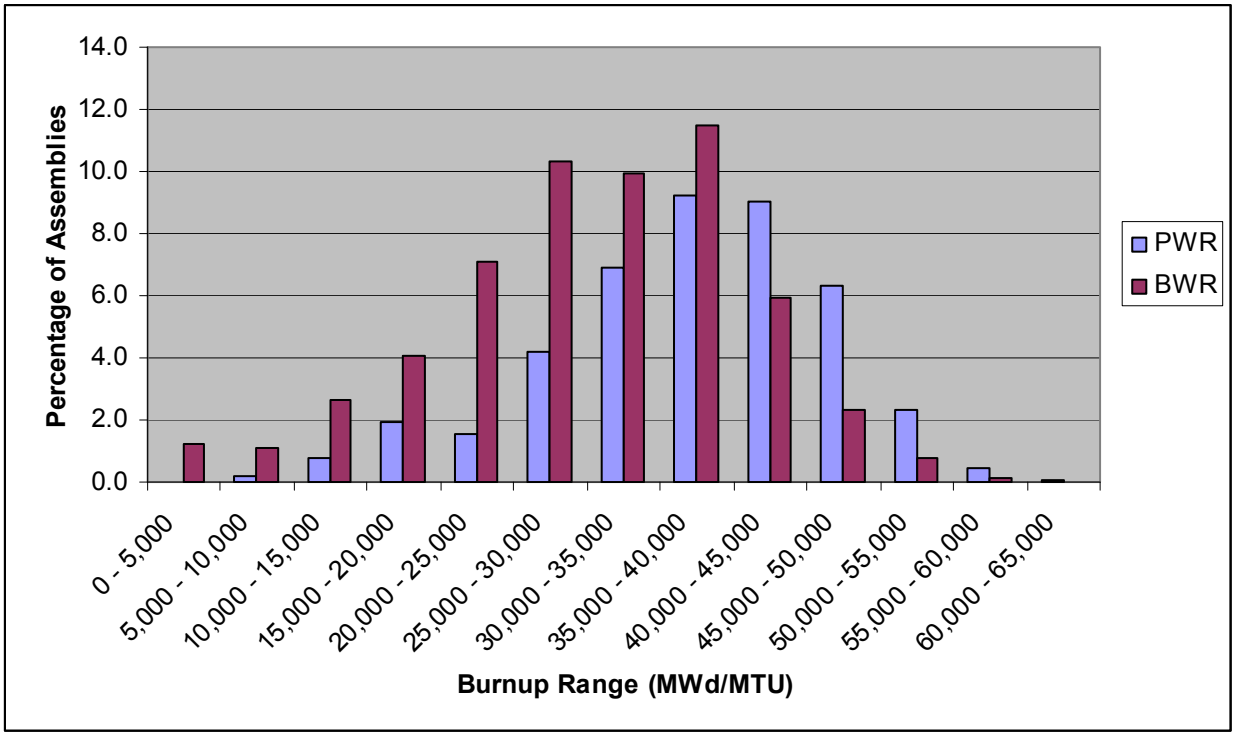


Figure3-1 Percentage of Assemblies per Burn-up Range Current Inventory

3.2 Projected Commercial UNF Inventory from Light Water Reactors

The UFD campaign requested an estimate of the potential UNF inventory. Four scenarios were used to evaluate the projected increases in the commercial LWR UNF inventory. The four scenarios were selected from those previously evaluated by DOE. Use of these scenarios does not constitute an endorsement; the scenarios were selected to provide a wide range of LWR fuel inventory for use in future analysis.

- Scenario 1: assumes no replacement of existing nuclear generation reactors.
- Scenario 2: assumes the amount of current nuclear generation is maintained at the current levels (100 GWe/yr) with new reactors replacing the existing reactors as the existing reactors are decommissioned.
- Scenario 3: assumes the amount of nuclear generation will increase to 200 GWe/yr from 2020 to 2060.
- Scenario 4: assumes the amount of nuclear generation will increase to 400 GWe/yr from 2020 to 2060.

Figure 3-2 and Figure 3-3 project the number of assemblies annually and cumulatively through the end of the century. Figure 3-4 and Figure 3-5 project the mass (MT of uranium) annually and cumulatively.

3.2.1 Scenario 1 – No Replacement Nuclear Generation

The first scenario assumes no replacement nuclear generating plants are built. The existing plants are assumed to have one 20 year life extension and will be decommissioned after 60 years of operation. Applying these assumptions the last nuclear generator finishes operations in 2055.

UNF discharge predictions developed for Nuclear Energy Institute in 2005 were used to estimate the number of assemblies and metric tons of uranium.[3] To estimate the average enrichment and burn-up through 2009, projections made by utilities as part of the RW-859 surveys were used. These projections

are documented in OCRWMs "Calculation Method for the Projection of Future Spent Fuel Discharges", February 2002.[4] These projections identified a burn-up increase of 2.38% per year for BWR UNF and 1.11% per year for PWR UNF. The enrichment increased at the same rate as burn-up until reaching the current enrichment limit of 5%. Once the 5% enrichment limit is reached, enrichment and burn-up are assumed to remain constant. Table 3-7 shows a summary of the results for this scenario. Figure 3-6 provides the percentage of the assemblies discharged as a function of the burn-up range. Complete results are in Appendix B, Table B-1.

3.2.1.1 Scenario 1 Potential Range

In FY-12 UFD has sponsored additional UNF projections for this scenario as a part of the logistical modeling update effort. As part of this work [Calvin Database Update in Support of UFD System Architecture Study, Kalinina] slightly different assumptions were used for the major variables that effect the projection.[59] The Calvin Database Study Assumptions and the UFD inventory assumptions are summaried in Table 3-5.

Table 3-5 Comparison of Calvin and UFD Scenario 1 Assumptions

Parameter	UFD Inventory	Calvin
Maximum U-235 Enrichment %	5.0%	5.0%
Maxiumum PWR Burn-up (GWd/MT)	54.2%	62.0%
Maximum BWR Burn-up (GWd/MT)	56.3%	57.0%
Annual Increase in Burn-up (%)	1.11%	0 & 1.3%
Decrease in Average Capacity due to age %	0.0%	1 & .1%

These alternate parameters effectively provide a range of the expected scenario 1. The results of all four cases are summarized in Table 3-6. Inspection of the table indicates a narrow range and similar results can be expected for the 2011 discharge projection and scenarios 2,3, and 4 discussed below.

Table 3-6 Scenario 1, No Nuclear Replacement Results Potential Range

			Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment (%)		Average Burn-up (GWd/MT)	
Study	Annual Burn-up Increase %	Decrease in Average Capacity %	PWR	BWR	Total	PWR	BWR	Total	PWR	BWR	PWR	BWR
UFD Inventory	1.11	0	209,000	273,000	482,000	91,000	49,000	140,000	4.40	4.09	47.30	45.30
Calvin (No Burnup)	0	0	206,000	276,400	482,400	89,800	48,900	138,700	3.80	3.39	41.60	36.00
Calvin	1.3	0	191,800	258,700	450,500	83,600	45,700	129,300	3.96	3.53	44.29	38.56
Calvin (NEI)	1.11	0	208,700	273,400	482,100	91,100	48,900	140,000	N/A	N/A	N/A	N/A

3.2.2 Scenario 2 – Maintain Current Nuclear Generation

The second scenario assumes the current nuclear generation rate (about 100GWe/yr) is maintained. UNF discharge predictions used in Scenario 1 were used up to 2020. The average amount of UNF discharge from 2011 to 2020 was used for each year past 2020, maintaining the same enrichment and burn-up (5% max enrichment). The current PWR/BWR ratio is maintained. Table 3-8 shows a summary of the results for this scenario. Figure 3-7 provides the percentage of the assemblies discharged as a function of the burn-up range. Complete results are in Appendix B, Table B-2.

3.2.3 Scenario 3 – Projected Growth to 200 GWe/yr Nuclear Generation in 2060

The third scenario assumes the nuclear generation rate will increase to 200 GWe/yr in 2060. New nuclear generating capacity is assumed to come on line in 2020 and increase linearly till 2060. Nuclear generation is assumed to remain constant past 2060. UNF discharge predictions used in Scenario 1 were used up to 2020. The average amount of UNF discharge up to 2020 was used to increase the amount of UNF beginning in 2021 linearly through 2060, maintaining the same enrichment and burn-up (5% max enrichment). The current PWR/BWR ratio is maintained. Table 3-9 shows a summary of the results for this scenario. Figure 3-8 provides the percentage of the assemblies discharged as a function of the burn-up range. Complete results are in Appendix B, Table B-3.

3.2.4 Scenario 4 – Projected Growth to 400 GWe/yr Nuclear Generation in 2060

The third scenario assumes the nuclear generation rate will increase to 400 GWe/yr in 2060. New nuclear generating capacity is assumed to come on line in 2020 and increase linearly till 2060. Nuclear generation is assumed to remain constant past 2060. UNF discharge predictions used in Scenario 1 were used up to 2020. The average amount of UNF discharge up to 2020 was used as the basis to linearly increase the amount of UNF beginning in 2021 through 2060, maintaining the same enrichment and burn-up (5% max enrichment). The current PWR/BWR ratio is maintained. Table 3-10 shows a summary of the results for this scenario. Figure 3-9 provides the percentage of the assemblies discharged as a function of the burn-up range. Complete results are in Appendix B, Table B-4.

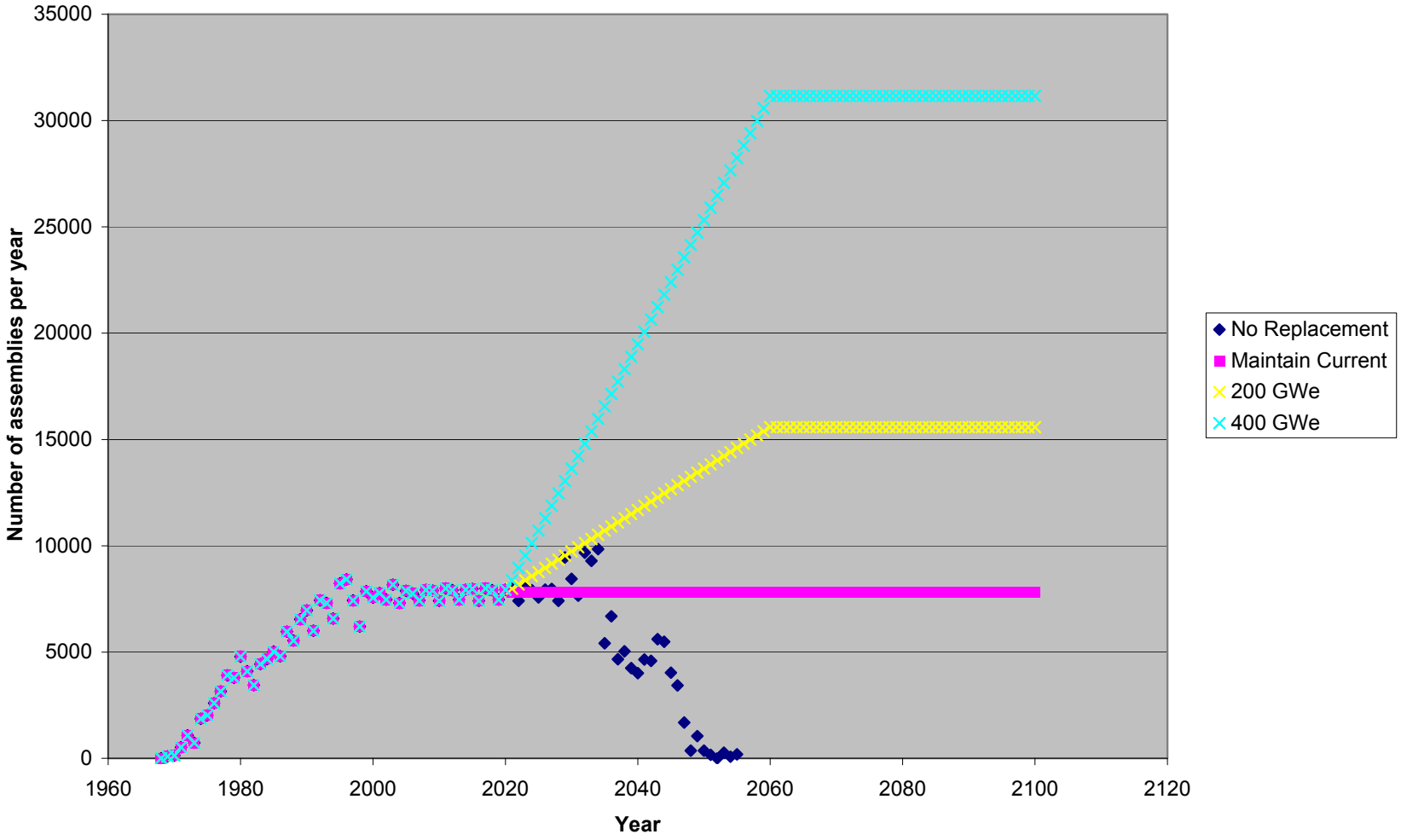


Figure 3-2 Annual UNF Assemblies Discharged for the No Replacement, Maintain Current, 200 GWe/yr, and 400 GWe/yr.

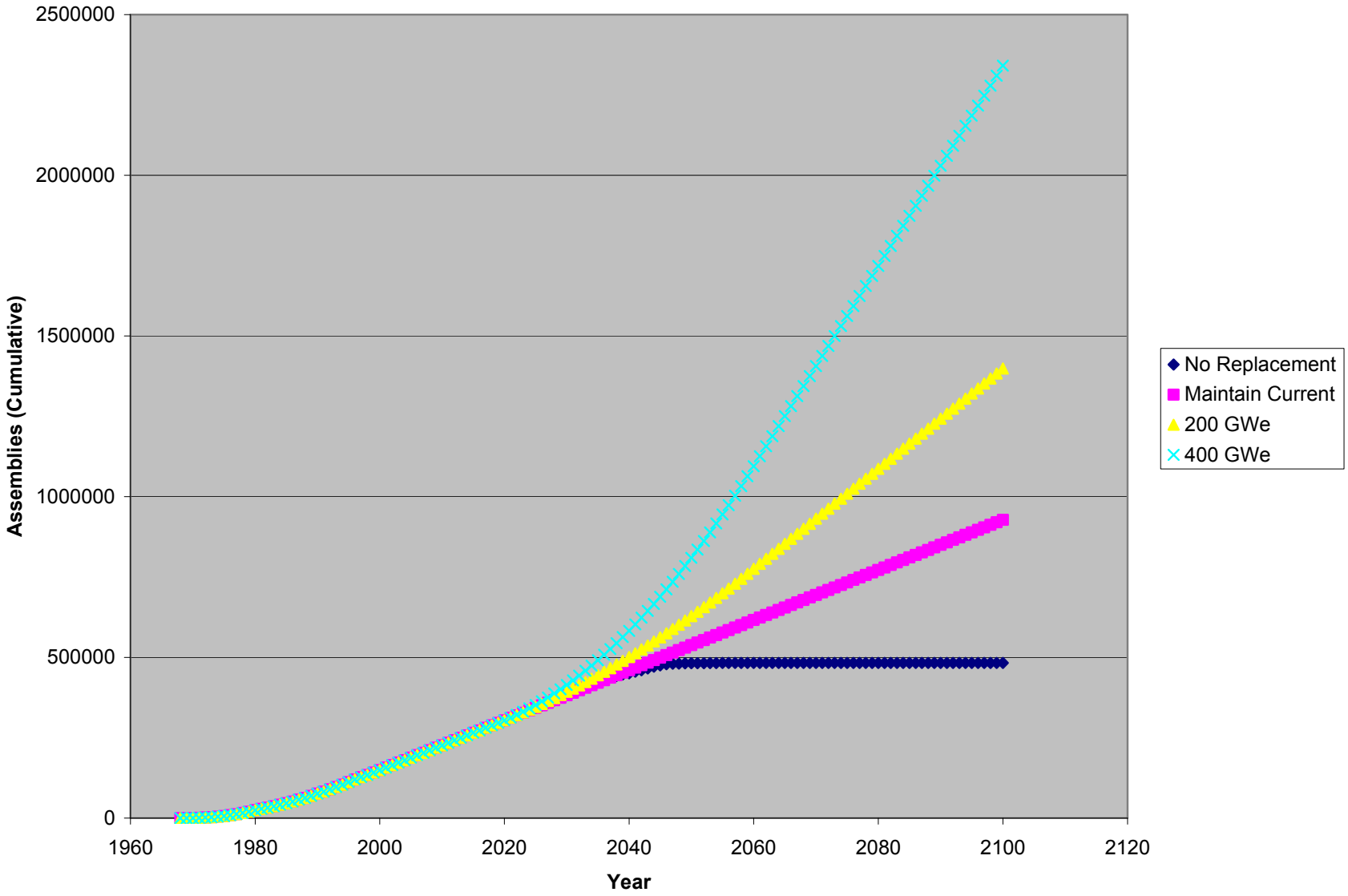


Figure 3-3 Cumulative UNF Assemblies Discharged for the No Replacement, Maintain Current, 200 GWe/yr, and 400 GWe/yr.

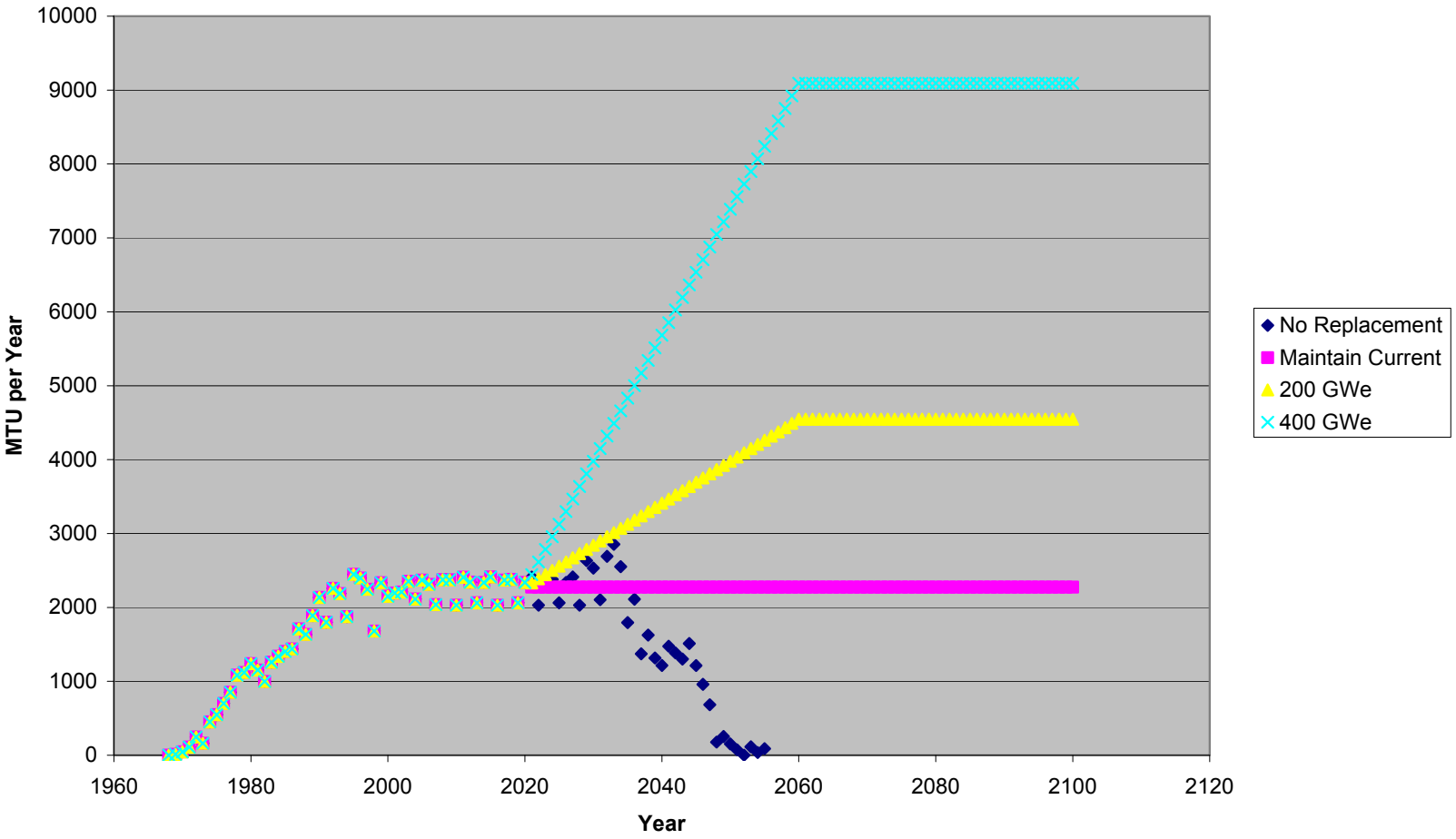


Figure 3-4 Annual UNF Mass Discharged for the No Replacement, Maintain Current, 200 GWe/yr, and 400 GWe/yr Cases.

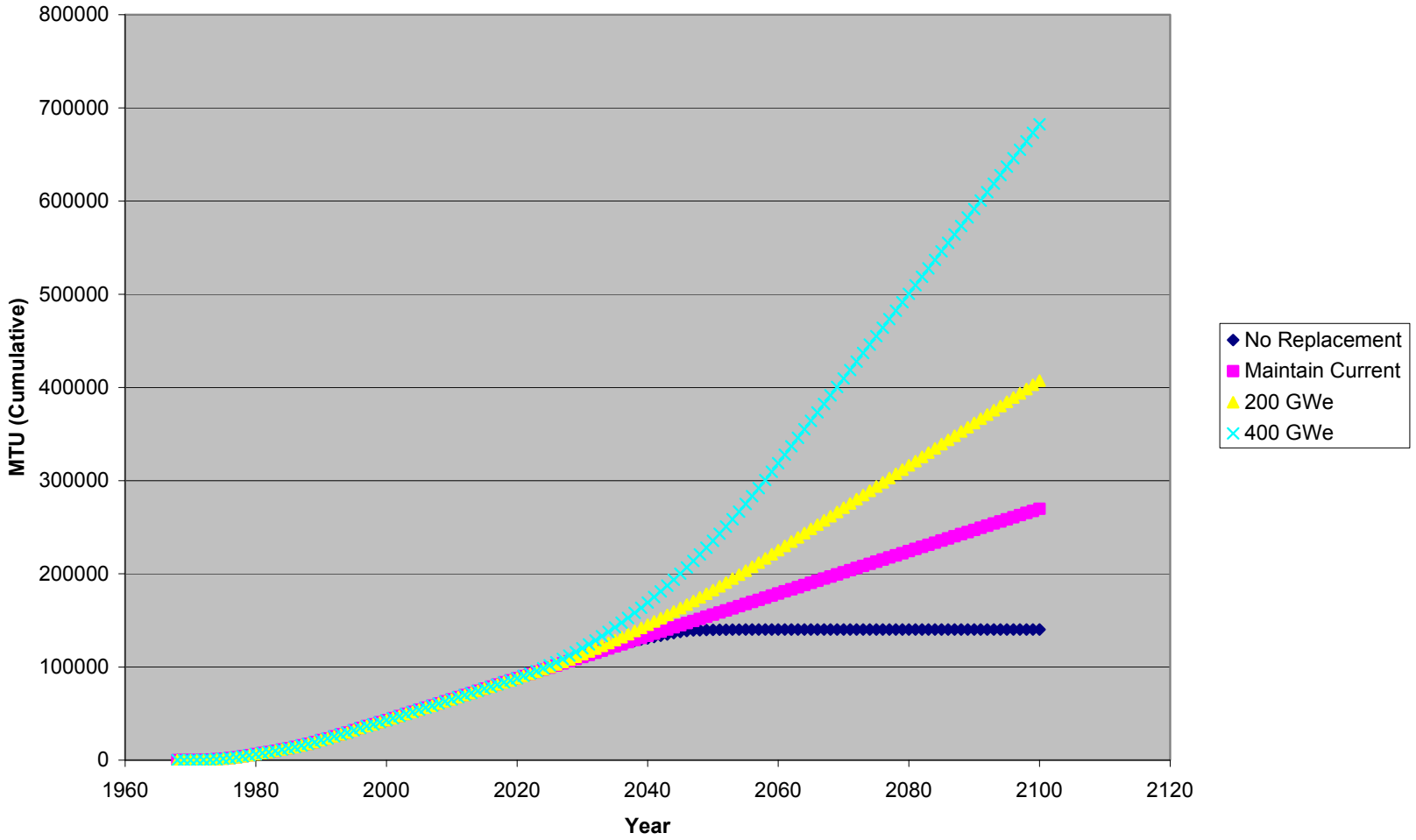


Figure 3-5 Cumulative UNF Mass Discharged for the No Replacement, Maintain Current, 200 GWe/yr, and 400 GWe/yr Cases.

Table 3-7 No replacement nuclear generation.

	Number of Assemblies[b]			Total Initial Uranium (MTU)[a]			Average Enrichment		Average Burn-up (MWd/MTU)[c]	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2010	97,400	128,600	226,000	42,300	23,000	65,200	3.74	3.12	39,600	33,300
2030	165,000	219,200	384,200	72,000	39,200	111,100	4.24	3.87	45,400	42,600
2055	209,000	273,000	483,000	91,000	49,000	140,000	4.40	4.09	47,300	45,300

a the estimated fuel discharged has been rounded to the nearest 100 MTU prior to 2050 and the nearest 1,000 thereafter, totals may not appear to sum correctly

b the estimated number of assemblies has been rounded to the nearest 200 prior to 2050 and nearest 1000 thereafter, totals may not appear to sum correctly

c the burn-up has been rounded to the next 100 MWd/MT

The complete data in Appendix B has not been rounded to allow for independent reproduction of the calculations.

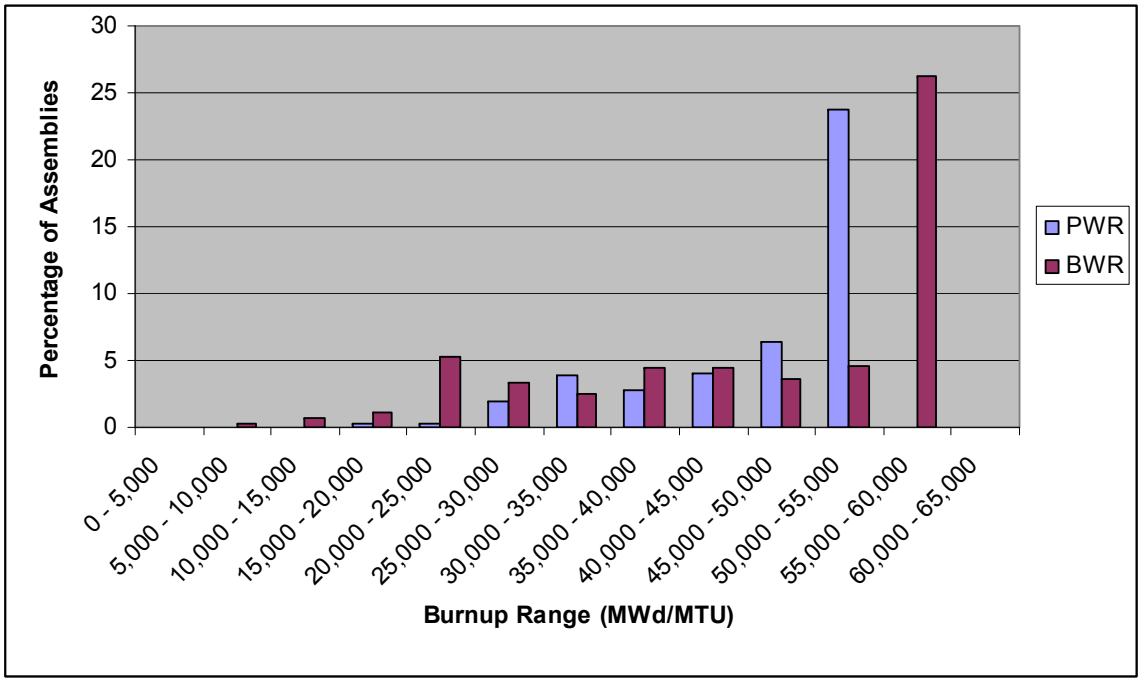


Figure 3-6 Percentage of Assemblies per Burn-up Range - No replacement Case

Table 3-8 Maintain current nuclear generation.

	Number of Assemblies[b]			Total Initial Uranium (MTU)[a]			Average Enrichment		Average Burn-up (MWd/MTU)[c]	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2010	97,400	128,600	226,000	42,300	23,000	65,200	3.74	3.12	39,600	33,300
2020	131,000	173,000	304,200	57,000	30,900	88,000	4.04	3.57	43,100	39,000
2040	198,600	261,600	460,200	86,600	46,800	133,400	4.36	4.05	46,900	44,900
2060	266,000	350,000	616,000	116,000	63,000	179,000	4.52	4.29	48,800	47,800
2080	333,000	439,000	772,000	146,000	79,000	224,000	4.62	4.43	49,900	49,500
2100	401,000	527,000	928,000	175,000	95,000	270,000	4.68	4.53	50,600	50,600

a the estimated fuel discharged has been rounded to the nearest 100 MTU prior to 2050 and the nearest 1,000 thereafter, totals may not appear to sum correctly

b the estimated number of assemblies has been rounded to the nearest 200 prior to 2050 and nearest 1000 thereafter, totals may not appear to sum correctly

c the burn-up has been rounded to the next 100 MWd/MT

The complete data in Appendix B has not been rounded to allow for independent reproduction of the calculations.

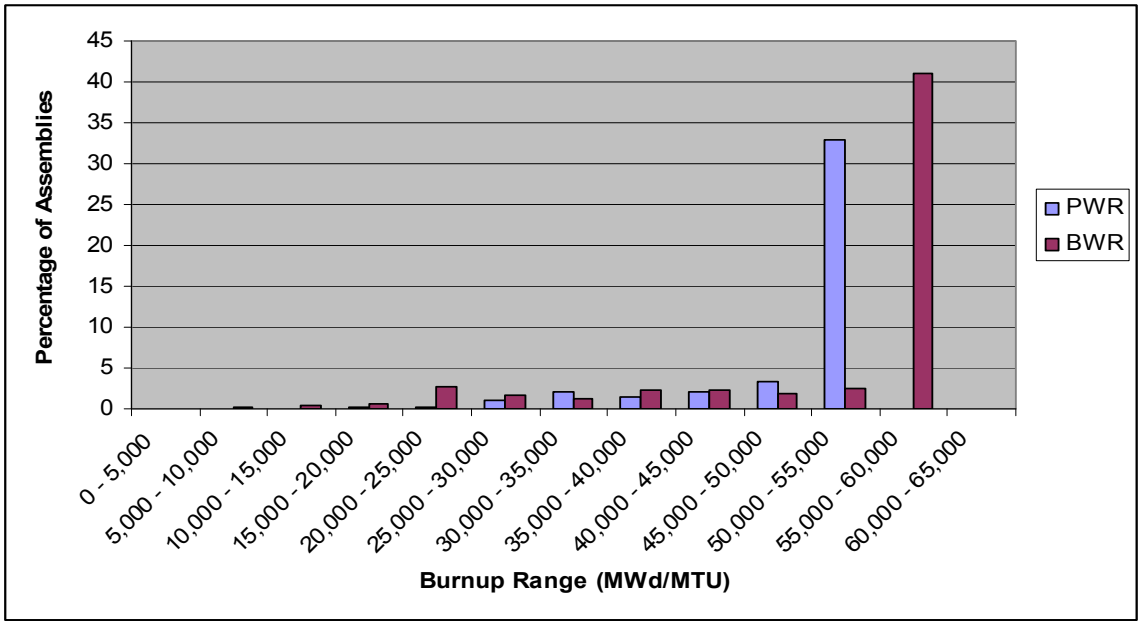


Figure 3-7 Percentage of Assemblies per Burn-up Range – Maintain Current Nuclear Generation Case

Table 3-9 Projected growth to 200 GWe/yr nuclear generation in 2060.

	Number of Assemblies[b]			Total Initial Uranium (MTU)[a]			Average Enrichment		Average Burn-up (MWd/MTU)[c]	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2010	97,400	128,600	226,000	42,300	23,000	65,200	3.74	3.12	39,600	33,300
2020	131,000	173,000	304,200	57,000	30,900	88,000	4.04	3.57	43,100	39,000
2040	216,200	284,600	501,000	94,400	51,000	145,400	4.42	4.13	47,500	45,800
2060	335,000	440,000	775,000	147,000	79,000	226,000	4.62	4.43	49,900	49,500
2080	470,000	617,000	1,087,000	206,000	111,000	316,000	4.73	4.59	51,200	51,500
2100	605,000	794,000	1,397,000	265,000	143,000	407,000	4.79	4.68	51,800	52,600

a the estimated fuel discharged has been rounded to the nearest 100 MTU prior to 2050 and the nearest 1,000 thereafter, totals may not appear to sum correctly
b the estimated number of assemblies has been rounded to the nearest 200 prior to 2050 and nearest 1000 thereafter, totals may not appear to sum correctly
c the burn-up has been rounded to the next 100 MWd/MT
The complete data in Appendix B has not been rounded to allow for independent reproduction of the calculations.

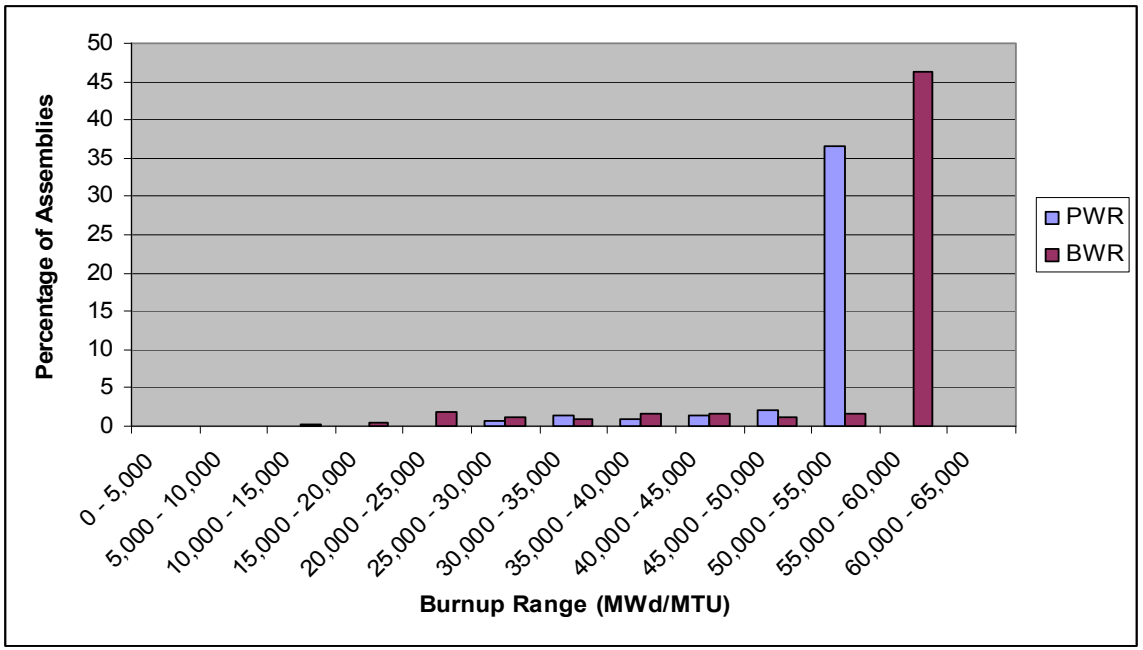


Figure 3-8 Percentage of Assemblies per Burn-up Range – 200 GWe/yr Nuclear Generation by 2060 Case

Table 3-10 Projected growth to 400 GWe/yr nuclear generation in 2060.

	Number of Assembliesb			Total Initial Uranium (MTU)a			Average Enrichment		Average Burn-up (MWd/MTU)c	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2010	97,400	128,600	226,000	42,300	23,000	65,2000	3.74	3.12	39,600	33,3000
2020	131,000	173,000	304,200	57,000	30,900	88,000	4.04	3.57	43,100	38,900
2040	251,600	331,000	582,600	109,900	59,300	169,200	4.50	4.25	48,500	47,300
2060	473,000	622,000	1,095,000	207,000	112,000	319,000	4.73	4.60	51,200	51,500
2080	743,000	975,000	1,718,000	325,000	175,000	501,000	4.83	4.74	52,300	53,300
2100	1,013,000	1,329,000	2,341,000	444,000	239,000	682,000	4.87	4.80	52,800	54,100

a the estimated fuel discharged has been rounded to the nearest 100 MTU prior to 2050 and the nearest 1,000 thereafter, totals may not appear to sum correctly

b the estimated number of assemblies has been rounded to the nearest 200 prior to 2050 and nearest 1000 thereafter, totals may not appear to sum correctly

c the burn-up has been rounded to the next 100 MWd/MTThe complete data in Appendix B has not been rounded to allow for independent reproduction of the calculations.

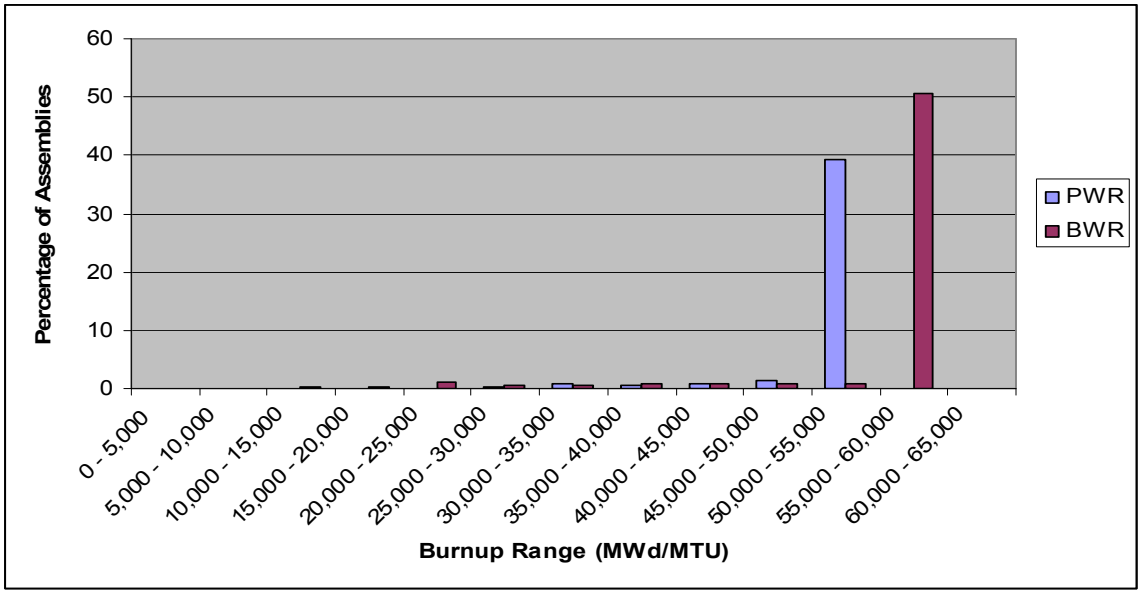


Figure 3-9 Percentage of Assemblies per Burn-up Range – 400 GWe/yr nuclear generation in 2060 Case

3.3 Commercial UNF Radionuclide Inventory

To provide a tool for evaluation 24 LWR UNF compositions were selected to represent the range of compositions in the current and projected future inventory. These include three burn-ups each for PWR and BWR reactor types, with the range of fuel burn-up taken from Tables 3-5 to 3-8, and fuel ages. Specifically, 20, 40 and 60 GWd/MT burn-up PWR and 15, 30 and 50 GWd/MT BWR for 5, 30, 100 and 500 years of cooling were selected as representative fuels.

The Characteristic Data Base (CDB) developed for RW was used to generate a radionuclide inventory for a given burn-up and decay time[5]. The maximum burn-up available in CDB for BWRs is 50,000 MWd/MTIHM and PWRs is 60,000 MWd/MTIHM. Decay times from 1 year to 1 million years can be provided. The number of radionuclides included in the output can also be varied based on their contribution to the total. The settings include 1.00%, 0.10%, 0.01% of the total, or the database cutoff which includes approximately 450 radionuclides. The inventory is reported on a gram/MTIHM basis so it can be applied to any inventory (current or projected). Future results can also be presented in Curies/MTIHM if desired. Appendix C, Tables C-1 through C-4 present the radionuclide inventory for the representative fuels.

The radionuclide decay algorithms and isotopic parameters from ORIGEN 2.2 were used in estimating the decay heat in an excel spreadsheet developed by the Systems Analysis Working Group. This method was adapted for use by the Used Fuel Disposition Working Group. The CDB output for 1 year cooled fuel was used to develop decay heat projections vs. time for the representative fuels.

Tables 3-11 to 3-14 and Figure 3-10 to 3-13 provide the decay heat projections for 40 GWd/MT PWR, 60 GWd/MT PWR, 30 GWd/MT BWR and 50 GWd/MT BWR fuel respectively. The figures and tables provide the total decay heat and isotopic groups with similar isotopic parameters.

The representative fuels discussed above are all limited by the current licensing basis for U.S enrichment plants to 5% U-235. To provide an example of an alternative fuel cycle from continuous improvement of the current U.S. reactor fleet, Table 3-15 and Figure 3-14 provide decay heat information for 100 GWd/MT LWR UOX used fuel. This burn-up requires an enrichment of approximately 8.3% U-235. The out of reactor isotopic composition was obtained from the transmutation database maintained by the Systems Analysis Working Group. Appendix C Table C-5 provides the isotopic composition for this fuel at 5, 30, 100 and 500 years cooling.

Table 3-11PWR 40 GWd/MT Used Fuel Decay Heat

Time (years) Decay Heat (Watts/MT)	1	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	2,765	1,054	566	354	222	110	1	0
Noble Metals Ag, Pd, Ru, Rh	2,752	11	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	3,593	64	10	2	0	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	819	348	332	309	287	258	159	116
Others	515	15	2	1	0	0	0	0
Total	**10,444**	**1,492**	**910**	**666**	**509**	**368**	**160**	**116**

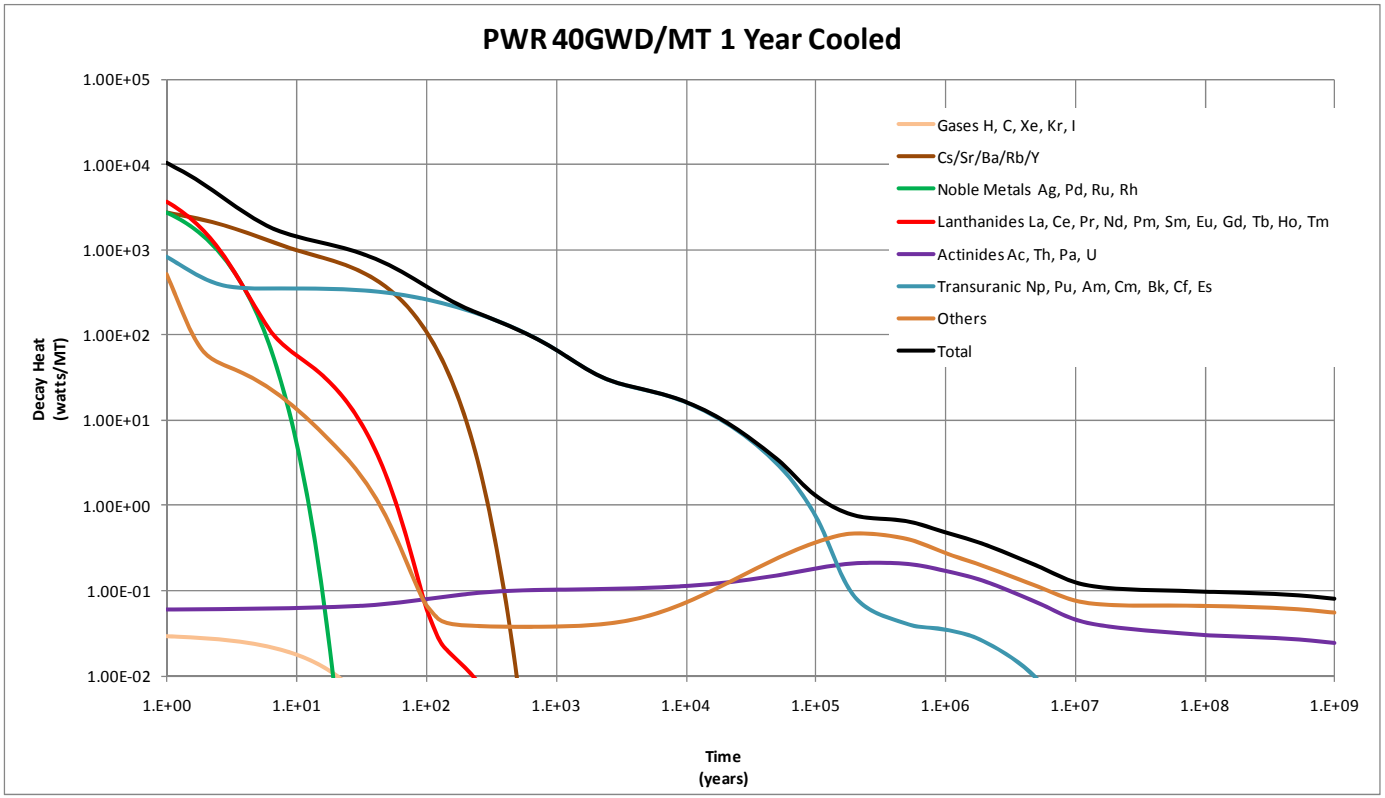


Figure 3-10 PWR 40 GWd/MT Used Fuel Decay Heat

Table 3-12 PWR60 GWd/MT Used Fuel Decay Heat

Time (years) / Decay Heat (Watts/MT)	1	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	4,608	1,576	824	516	323	160	1	0
Noble Metals Ag, Pd, Ru, Rh	3,447	14	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	3,843	109	17	3	1	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	1,515	785	613	516	449	381	199	139
Others	522	21	3	1	0	0	0	0
Total	**13,936**	**2,505**	**1,458**	**1,036**	**773**	**541**	**201**	**139**

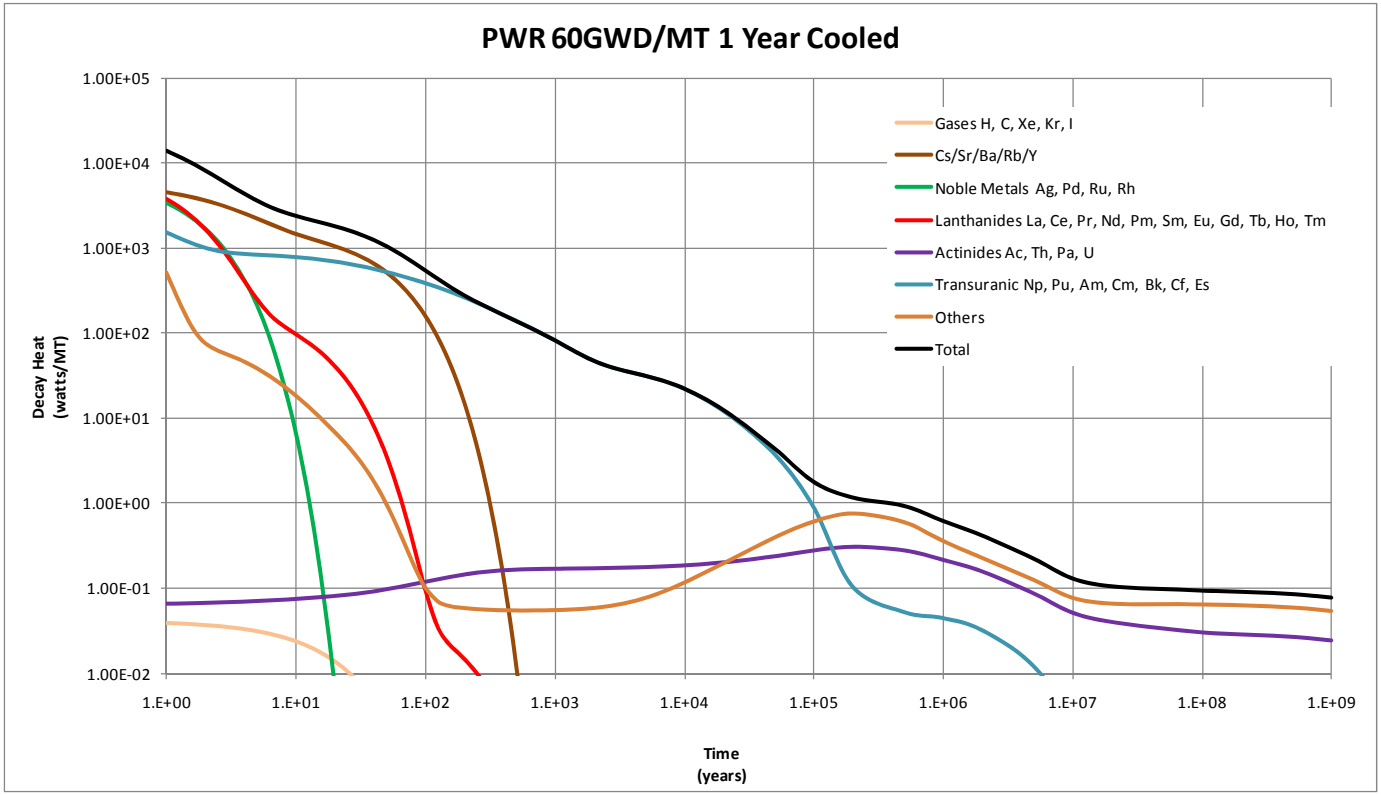


Figure 3-11 PWR 60 GWd/MT Used Fuel Decay Heat

Table 3-13 BWR 30 GWd/MT Used Fuel Decay Heat

Time (years) / Decay Heat (Watts/MT)	1	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	1,895	778	425	266	166	82	1	0
Noble Metals Ag, Pd, Ru, Rh	2,042	8	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	2,675	43	6	1	0	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	588	225	234	225	213	196	127	94
Others	403	12	2	0	0	0	0	0
Total	**7,603**	**1,067**	**667**	**493**	**380**	**278**	**128**	**94**

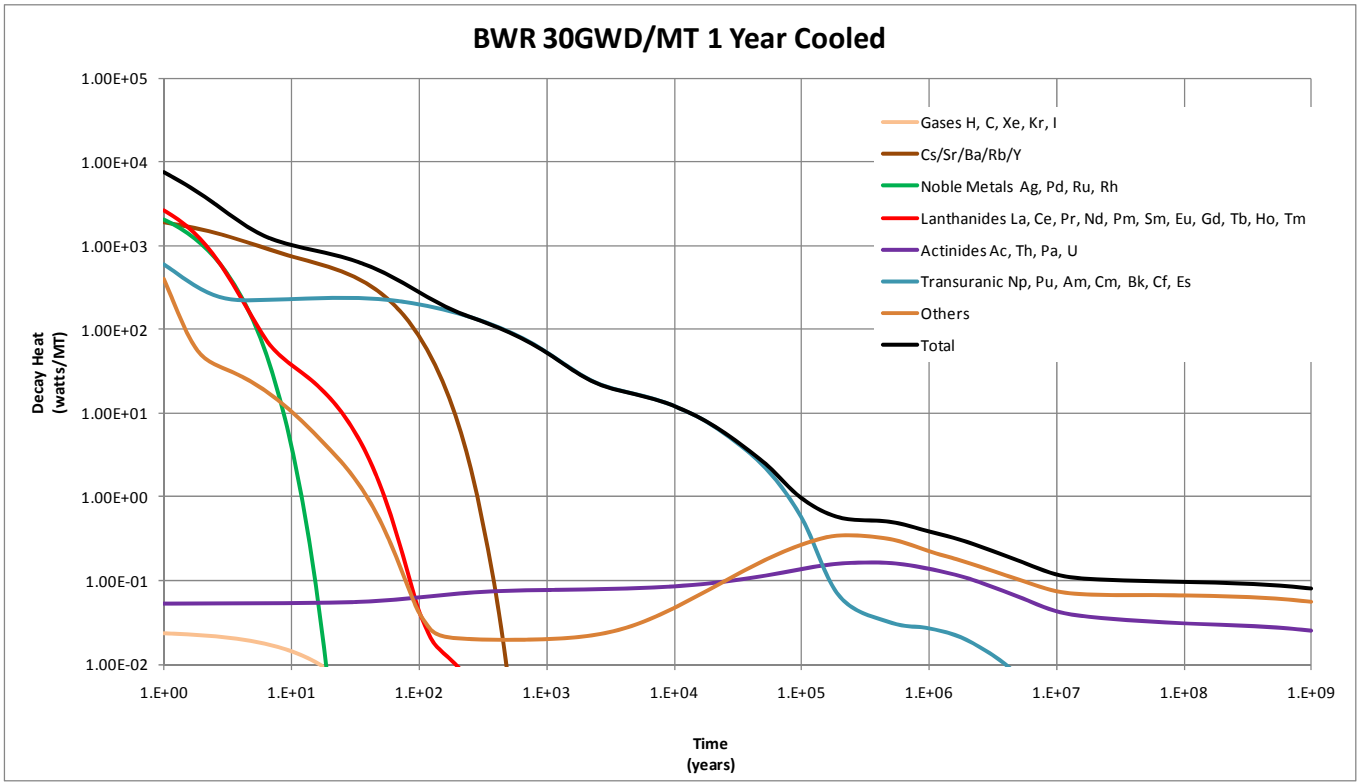


Figure 3-12 BWR 30 GWd/MT Used Fuel Decay Heat

Table 3-14 BWR 50 GWd/MT Used Fuel Decay Heat

Time (years) Decay Heat (Watts/MT)	1	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	3,558	1,257	662	414	259	128	1	0
Noble Metals Ag, Pd, Ru, Rh	2,669	11	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	2,734	92	14	3	1	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	1,627	760	591	496	433	369	199	139
Others	420	17	2	1	0	0	0	0
Total	**11,008**	**2,137**	**1,271**	**914**	**693**	**498**	**200**	**139**

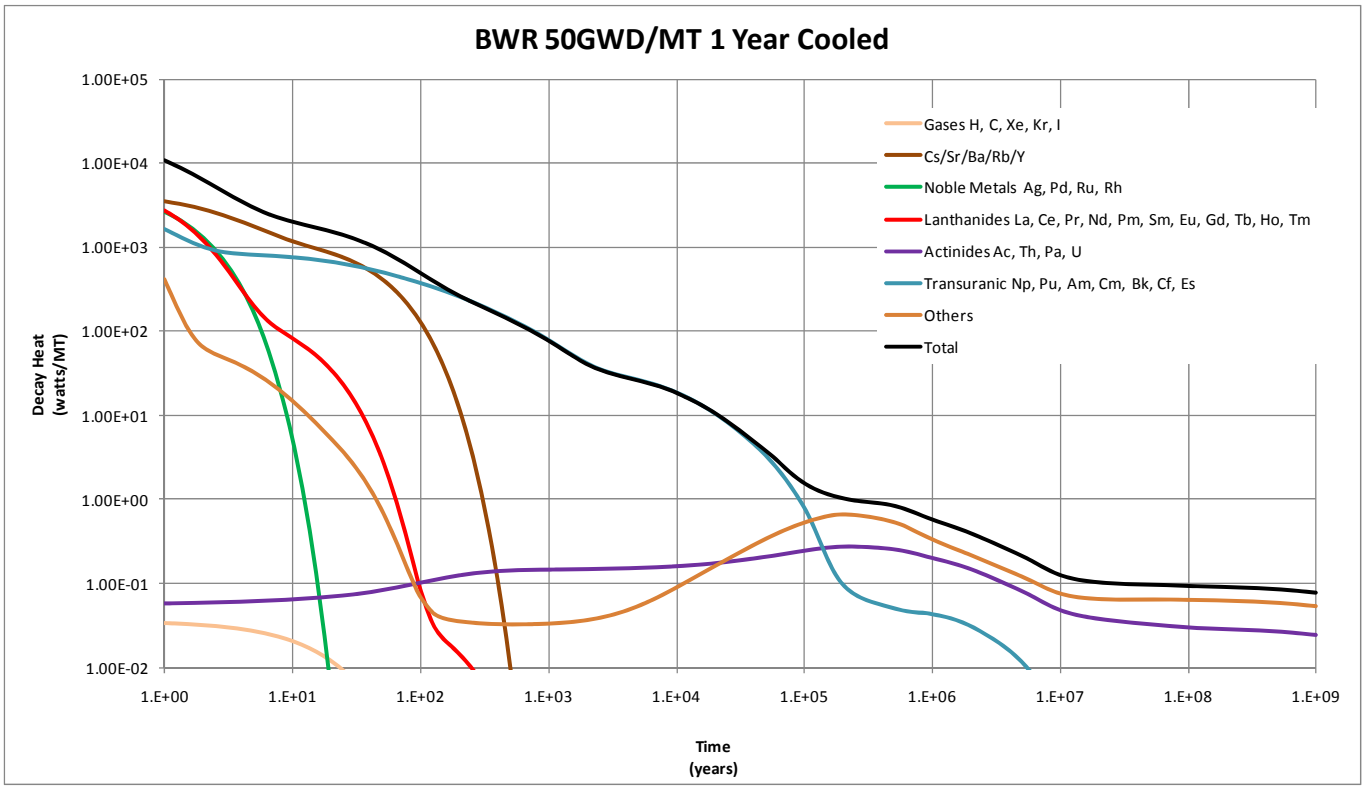


Figure 3-13 BWR 50 GWd/MT Used Fuel Decay Heat

Table 3-15 PWR 100 GWd/MT Used Fuel Decay Heat

Time (years) / Decay Heat (Watts/MT)	Discharge	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	5,544	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	22,369	2,449	1,387	868	543	269	2	0
Noble Metals Ag, Pd, Ru, Rh	18,262	8	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	52,720	163	31	6	1	0	0	0
Actinides Ac, Th, Pa, U	50,379	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	59,445	1,433	1,088	879	745	615	291	191
Others	20,343	26	5	2	0	0	0	0
Total	**229,062**	**4,079**	**2,511**	**1,755**	**1,290**	**884**	**294**	**192**

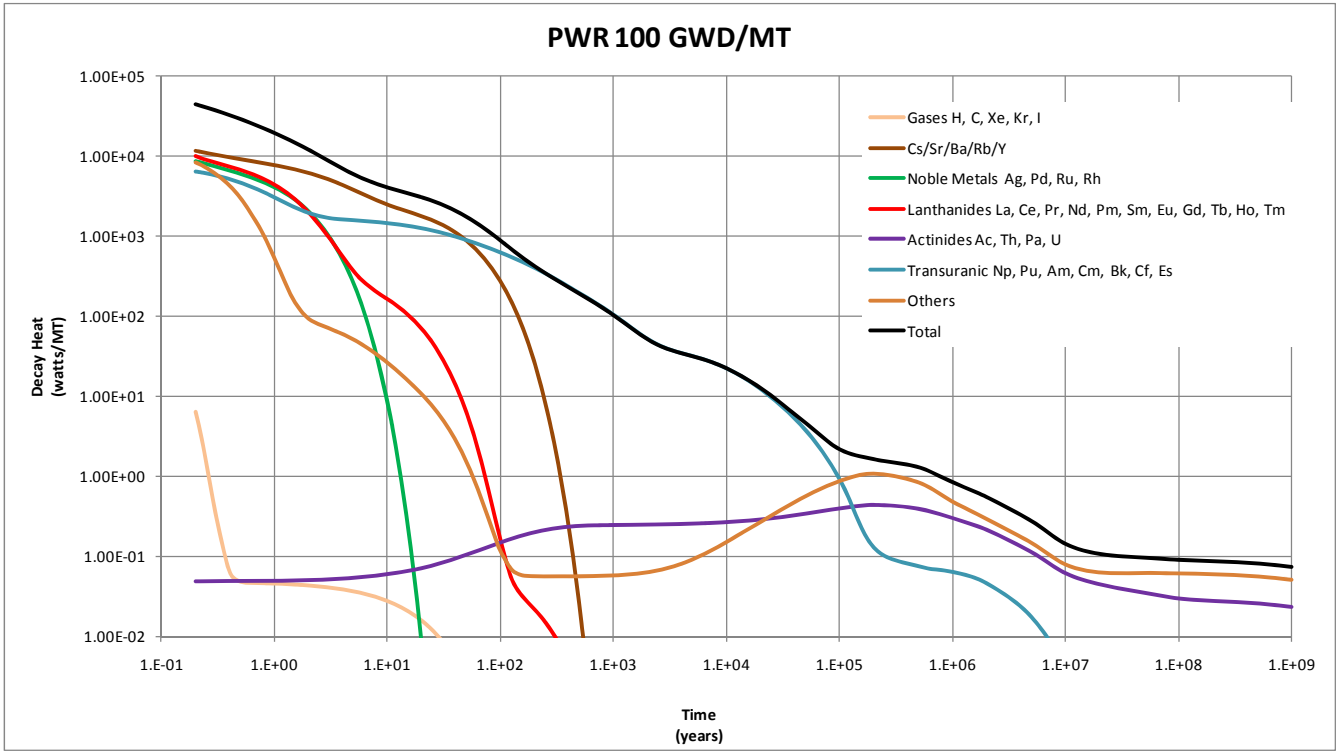


Figure 3-14 PWR 100 GWd/MT Used Fuel Decay Heat

3.4 Commercial UNF Storage

UNF is initially stored at the nuclear plants in water filled pools. These pools were not designed for long term storage and some facilities have run out of capacity to store all of the UNF in their pools. At these facilities, above ground dry storage systems are utilized to store the UNF. As more facilities run out of pool storage the amount of dry storage is increasing. Through June of 2012, 1596 dry storage casks have been loaded containing 62,597 assemblies of UNF.[24]. NEI estimates by 2020 over 30,000 MTU in about 2,600 casks will be in dry storage.[24]

Tables 3-16 and 3-17 provide additional details for the materials in dry cask storage. Table 3-16 is arranged by state and Table 3-17 is arranged by cask vendor.

Table 3-16 Commercial UNF Dry Storage by State

State	Vendor	Cask System	Canister Type	Reactor	Reactor Type	Utility	Casks	Assemblies
Alabama	Holtec	HI-STORM	MPC-68	Browns Ferry	BWR	TVA	30	2040
	Holtec	HI-STORM	MPC-32	Farley	PWR	Southern Nuclear	15	480
Arizona	NAC	NAC-UMS	UMS-24	Palo Verde	PWR	APS	94	2256
Arkansas	BFS/ES	Fuel Solutions	VSC-24	ANO	PWR	Entergy	24	576
	Holtec	HI-STORM	MPC-24	ANO	PWR	Entergy	22	528
	Holtec	HI-STORM	MPC-32	ANO	PWR	Entergy	16	512
California	Holtec	HI-STORM	MPC-32	Diabio Canyon	PWR	PG&E	23	736
	Holtec	HI-STAR	MPC-80	Humboldt Bay[1,3]	BWR	PG&E	5	390
	TN	NUHOMS	24PT	Rancho Seco[1]	PWR	SMUD	22	493
	TN	NUHOMS	24PT1	SONGS 1[1,3]	PWR	Southern Cal Edison	18	395
	TN	NUHOMS	24PT4	SONGS 2, 3	PWR	Southern Cal Edison	28	672
Colorado	DOE	Foster Wheeler	MVDS	Ft. St. Vrain	HTGR	PS Colorado		1464
Connecticut	NAC	NAC-MPC	MPC-26	Conn Yankee[2,3]	PWR	Ct. Yankee	43	1019
	TN	NUHOMS	32PT	Millstone	PWR	Dominion	14	448
Florida	TN	NUHOMS	32PTH	St. Lucie	PWR	FPL	14	448
	TN	NUHOMS	32PTH	Turkey Point	PWR	FPL	18	576
Georgia	Holtec	HI-STORM	MPC-68	Hatch	BWR	Southern Nuclear	45	3060
	Holtec	HI-STAR	MPC-68	Hatch	BWR	Southern Nuclear	3	204
Idaho	TN	NUHOMS	12T	INEEL	PWR	DOE	29	177
Illinois	Holtec	HI-STORM	MPC-32	Braidwoood	PWR	Exelon	3	96
	Holtec	HI-STORM	MPC-32	Byron	PWR	Exelon	11	352
	Holtec	HI-STORM	MPC-68	Dresden	BWR	Exelon	47	3196
	Holtec	Hi-STAR	MPC-68	Dresden	BWR	Exelon	4	272
	Holtec	HI-STORM	MPC-68	LaSalle	BWR	Exelon	6	408
	Holtec	HI-STORM	MPC-68	Quad Cities	BWR	Exelon	35	2380
Iowa	TN	NUHOMS	61BT	Duane Arnold	BWR	FPL	20	1220
Louisiana	Holtec	HI-STORM	MPC-68	River Bend	BWR	Entergy	15	1020
	Holtec	HI-STORM	MPC-32	Waterford	PWR	Entergy	9	288
Maine	NAC	NAC-UMS	UMS-24	Maine Yankee[2,3]	PWR	Maine Yankee	64	1434
Maryland	TN	NUHOMS	24P	Calvert Cliffs	PWR	Constellation	48	1152
	TN	NUHOMS	32P	Calvert Cliffs	PWR	Constellation	21	672
Massachusetts	NAC	NAC-MPC	MPC-36	Yankee Rowe[2,3]	PWR	YAEC	16	533
Michigan	BFS/ES	Fuel Solutions	W150	Big Rock Point1,[3]	BWR	Consumers	8	441
	BFS/ES	Fuel Solutions	VSC-24	Palisades	PWR	Entergy	18	432
	TN	NUHOMS	24PHT	Palisades	PWR	Entergy	13	312
	TN	NUHOMS	32PT	Palisades	PWR	Entergy	11	352

Minnesota	TN	NUHOMS	61BT	Monticello	BWR	Xcel Energy	10	610
	TN	TN Metal Casks	TN-40	Prairie Island	PWR	Ecel Energy	29	1160
Mississippi	Holtec	HI-STORM	MPC-68	Grand Gulf	BWR	Entergy	17	1156
Nebraska	TN	NUHOMS	61BT	Cooper	BWR	NPPD	8	488
	TN	NUHOMS	32PT	Fort Calhoun	PWR	OPPD	10	320

Table 3-16 (Con't)

Reactor State	Vendor	Cask System	Canister Type	Reactor	Reactor Type	Utility	Casks	Assemblies
New Hampshire	TN	NUHOMS	32PHT	Seabrook	PWR	FPL	6	192
New Jersey	Holtec	HI-STORM	MPC-68	Hope Creek	BWR	PSE&G	16	1088
	Holtec	HI-STORM	MPC-32	Salem	PWR	PSE&G	9	288
	TN	NUHOMS	61BT	Oyster Creek	BWR	Exelon	23	1403
New York	Holtec	HI-STORM	MPC-68	Fitzpatrick	BWR	Entergy	15	1020
	Holtec	HI-STORM	MPC-32	Indian Point 1[3]	PWR	Entergy	5	160
	Holtec	HI-STORM	MPC-32	Indian Point 2	PWR	Entergy	14	448
	TN	NUHOMS	32PT	Ginna	PWR	Constellation	6	192
North Carolina	NAC	NAC-UMS	UMS-24	McGuire	PWR	Duke	28	672
	TN	NUHOMS	61BTH	Brunswick	BWR	Progress	8	488
	TN	TN Metal Casks	TN-32	McGuire	PWR	Duke	10	320
Ohio	TN	NUHOMS	24P	Davis-Besse	PWR	FirstEnergy	3	72
Oregon	Holtec	TranStor	MPC-24E/EF	Trojan	PWR	Portland GE	34	780
Pennsylvania	TN	NUHOMS	61BT	Limerick	BWR	Exelon	16	976
	TN	NUHOMS	52B	Susquehanna	BWR	PPL	27	1404
	TN	NUHOMS	61BT	Susquehanna	BWR	PPL	40	2440
	TN	TN Metal Casks	TN-68	Peach Bottom	BWR	Exelon	57	3876
South Carolina	NAC	NAC-UMS	UMS-24	Catawba	PWR	Duke	24	576
	TN	NUHOMS	24PHB	Oconee	PWR	Duke	38	912
	TN	NUHOMS	24P	Oconee	PWR	Duke	84	2016
	TN	NUHOMS	24PTH	Robinson	PWR	Progress	14	336
	TN	NUHOMS	7P	Robinson	PWR	Progress	8	56
Tennessee	Holtec	HI-STORM	MPC-32	Sequoyah	PWR	TVA	32	1024
Texas	Holtec	HI-STORM	MPC-32	Comanche Pea	PWR	Luminant	8	256
Vermont	Holtec	HI-STORM	MPC-68	Vermont Yanke	BWR	Entergy	10	680
Virginia	GNB	Castor	V/21 and X33	Surry	PWR	Dominion	26	558
	NAC	NAC-I28	NAC-I28	Slurry	PWR	Dominion	2	56
	TN	NUHOMS	32PTH	North Anna	PWR	Dominion	13	416
	TN	NUHOMS	32PTH	Surry	PWR	Dominion	15	480
	TN	TN Metal Casks	TN-32	North Anna	PWR	Dominion	27	864
	TN	TN Metal Casks	TN-32	Surry	PWR	Dominion	26	832
	Westinghouse	MC-10	MC-10	Surry	PWR	Dominion	1	24
Washinton	Holtec	HI-STORM	MPC-68	Columbia	BWR	Energy Northwest	27	1836
Wisconsin	BFS/ES	Fuel Solutions	VSC-24	Point Beach	PWR	FPL	16	384
	TN	NUHOMS	32PT	Kewaunee	PWR	Dominion	8	256
	TN	NUHOMS	32PT	Point Beach	PWR	FPL	14	448

[1]One cask is storing GTCC waste is in use

[2]CY has 3 casks storing GTCC waste; Yankee Rowe has one and Maine Yankee has four casks

[3]All spent fuel from the shutdown plant

Table 3-17 Commercial UNF Dry Storage by Cask System

Vendor	Cask System	Canister Type	Reactor	Reactor Type	Reactor State	Utility	Casks	Assemblies
GNB	Castor	V/21 and X33	Surry	PWR	Virginia	Dominion	26	558
DOE	Foster Wheeler	MVDS	Ft. St. Vrain	HTGR	Colorado	PS Colorado		1464
BFS/ES	Fuel Solutions	VSC-24	ANO	PWR	Arkansas	Entergy	24	576
	Fuel Solutions	W150	Big Rock Point1,[3]	BWR	Michigan	Consumers	8	441
	Fuel Solutions	VSC-24	Palisades	PWR	Michigan	Entergy	18	432
	Fuel Solutions	VSC-24	Point Beach	PWR	Wisconsin	FPL	16	384
Holtec	HI-STAR	MPC-80	Humboldt Bay[1,3]	BWR	California	PG&E	5	390
	HI-STAR	MPC-68	Hatch	BWR	Georgia	Southern Nuclear	3	204
	Hi-STAR	MPC-68	Dresden	BWR	Illinois	Exelon	4	272
	TranStor	MPC-24E/EF	Trojan	PWR	Oregon	Portland GE	34	780
	HI-STORM	MPC-32	Braidwood	PWR	Illinois	Exelon	3	96
	HI-STORM	MPC-68	Browns Ferry	BWR	Alabama	TVA	30	2040
	HI-STORM	MPC-32	Comanche Peak	PWR	Texas	Luminant	8	256
	HI-STORM	MPC-32	Farley	PWR	Alabama	Southern Nuclear	15	480
	HI-STORM	MPC-24	ANO	PWR	Arkansas	Entergy	22	528
	HI-STORM	MPC-32	ANO	PWR	Arkansas	Entergy	16	512
	HI-STORM	MPC-32	Diabio Canyon	PWR	California	PG&E	23	736
	HI-STORM	MPC-68	Hatch	BWR	Georgia	Southern Nuclear	45	3060
	HI-STORM	MPC-32	Byron	PWR	Illinois	Exelon	11	352
	HI-STORM	MPC-68	Dresden	BWR	Illinois	Exelon	47	3196
	HI-STORM	MPC-68	LaSalle	BWR	Illinois	Exelon	6	408
	HI-STORM	MPC-68	Quad Cities	BWR	Illinois	Exelon	35	2380
	HI-STORM	MPC-68	River Bend	BWR	Louisiana	Entergy	15	1020
	HI-STORM	MPC-68	Grand Gulf	BWR	Mississippi	Entergy	17	1156
	HI-STORM	MPC-68	Hope Creek	BWR	New Jersey	PSE&G	16	1088
	HI-STORM	MPC-32	Salem	PWR	New Jersey	PSE&G	9	288
	HI-STORM	MPC-68	Fitzpatrick	BWR	New York	Entergy	15	1020
	HI-STORM	MPC-32	Indian Point 1[3]	PWR	New York	Entergy	5	160
	HI-STORM	MPC-32	Indian Point 2	PWR	New York	Entergy	14	448
	HI-STORM	MPC-32	Sequoyah	PWR	Tennessee	TVA	32	1024
	HI-STORM	MPC-68	Vermont Yanke	BWR	Vermont	Entergy	10	680
	HI-STORM	MPC-32	Waterford	PWR	Louisiana	Entergy	9	288
	HI-STORM	MPC-68	Columbia	BWR	Washington	Energy Northwest	27	1836
Westinghouse	MC-10	MC-10	Surry	PWR	Virginia	Dominion	1	24
NAC	NAC-I28	NAC-I28	Slurry	PWR	Virginia	Dominion	2	56
	NAC-MPC	MPC-26	Conn Yankee[2,3]	PWR	Connecticut	Ct. Yankee	43	1019
	NAC-MPC	MPC-36	Yankee Rowe[2,3]	PWR	Massachusetts	YAEC	16	533
	NAC-UMS	UMS-24	Palo Verde	PWR	Arizona	APS	94	2256
	NAC-UMS	UMS-24	Maine Yankee[2,3]	PWR	Maine	Maine Yankee	64	1434
	NAC-UMS	UMS-24	McGuire	PWR	North Carolina	Duke	28	672
	NAC-UMS	UMS-24	Catawba	PWR	South Carolina	Duke	24	576

Table 3-17 (Con't)

Vendor	Cask System	Canister Type	Reactor	Reactor Type	Reactor State	Utility	Casks	Assemblies
Trans Nuclear	NUHOMS	24PT	Rancho Seco[1]	PWR	California	SMUD	22	493
	NUHOMS	24PT1	SONGS 1[1,3]	PWR	California	Southern Cal Edison	18	395
	NUHOMS	24PT4	SONGS 2, 3	PWR	California	Southern Cal Edison	28	672
	NUHOMS	32PT	Millstone	PWR	Connecticut	Dominion	14	448
	NUHOMS	32PTH	St. Lucie	PWR	Florida	FPL	14	448
	NUHOMS	12T	INEEL	PWR	Idaho	DOE	29	177
	NUHOMS	61BT	Duane Arnold	BWR	Iowa	FPL	20	1220
	NUHOMS	24P	Calvert Cliffs	PWR	Maryland	Constellation	48	1152
	NUHOMS	32P	Calvert Cliffs	PWR	Maryland	Constellation	21	672
	NUHOMS	24PHT	Palisades	PWR	Michigan	Entergy	13	312
	NUHOMS	32PT	Palisades	PWR	Michigan	Entergy	11	352
	NUHOMS	61BT	Monticello	BWR	Minnesota	Xcel Energy	10	610
	NUHOMS	61BT	Cooper	BWR	Nebraska	NPPD	8	488
	NUHOMS	32PT	Fort Calhoun	PWR	Nebraska	OPPD	10	320
	NUHOMS	32PHT	Seabrook	PWR	New Hampshire	FPL	6	192
	NUHOMS	61BT	Oyster Creek	BWR	New Jersey	Exelon	23	1403
	NUHOMS	32PT	Ginna	PWR	New York	Constellation	6	192
	NUHOMS	61BTH	Brunswick	BWR	North Carolina	Progress	8	488
	NUHOMS	24P	Davis-Besse	PWR	Ohio	FirstEnergy	3	72
	NUHOMS	61BT	Limerick	BWR	Pennsylvania	Exelon	16	976
	NUHOMS	52B	Susquehanna	BWR	Pennsylvania	PPL	27	1404
	NUHOMS	61BT	Susquehanna	BWR	Pennsylvania	PPL	40	2440
	NUHOMS	24PHB	Oconee	PWR	South Carolina	Duke	38	912
	NUHOMS	24P	Oconee	PWR	South Carolina	Duke	84	2016
	NUHOMS	24PTH	Robinson	PWR	South Carolina	Progress	14	336
	NUHOMS	7P	Robinson	PWR	South Carolina	Progress	8	56
	NUHOMS	32PTH	North Anna	PWR	Virginia	Dominion	13	416
	NUHOMS	32PTH	Surry	PWR	Virginia	Dominion	15	480
	NUHOMS	32PTH	Turkey Point	PWR	Florida	FPL	18	576
	NUHOMS	32PT	Kewaunee	PWR	Wisconsin	Dominion	8	256
	NUHOMS	32PT	Point Beach	PWR	Wisconsin	FPL	14	448
	TN Metal Casks	TN-40	Prairie Island	PWR	Minnesota	Ecel Energy	29	1160
	TN Metal Casks	TN-32	McGuire	PWR	North Carolina	Duke	10	320
	TN Metal Casks	TN-68	Peach Bottom	BWR	Pennsylvania	Exelon	57	3876
	TN Metal Casks	TN-32	North Anna	PWR	Virginia	Dominion	27	864
	TN Metal Casks	TN-32	Surry	PWR	Virginia	Dominion	26	832

4. PROCESS WASTES GENERATED BY REPROCESSING COMMERCIAL LIGHT WATER REACTOR UNF

To provide a tool for evaluation of the impact of reprocessing, unit quantities of 24 representative LWR UNF compositions have been converted into their equivalent waste forms for a variety of reprocessing methods. For each of the representative fuels the mass, volume, container count and decay heat in each container has been projected for each of the baseline waste forms anticipated. Each of the reprocessing methods, representative fuels and baseline waste form are described below.

4.1 Reprocessing Methods

Commercial LWR UNF reprocessing methods vary in process complexity and technical maturity. Generally the objective of additional complexity is to lessen the potential environmental impact of the resulting waste disposition activities. To support future evaluations of potential environmental impact three aqueous reprocessing methods and one electro-chemical reprocessing method were selected.

4.1.1 Co-Extraction

The Co-Extraction method represents the simplest and most technically mature aqueous reprocessing method evaluated. The process envisioned is similar to the current generation of deployed reprocessing technology (e.g., the Rokkasaho Reprocessing Facility). U and Pu are recovered together (no pure plutonium separation). The principle fission product wastes, including the minor actinides are combined with the undissolved solids (UDS) and recovered Tc into a single borosilicate glass waste form.

The gaseous radionuclides I-129 and H-3 released during reprocessing are captured and converted to waste forms suitable for disposal while C-14 and Kr-85 are assumed to be released to the atmosphere. See Figure 4-1 Co-Extraction reprocessing block diagram.

While this process is similar in function to the industrial COEX™ process deployed by AREVA, the two processes assume different processing methods and steps and so the product and waste streams cannot be directly compared.

4.1.2 New Extraction

New Extraction is an advanced aqueous process which recovers all of the TRU elements for re-use. The process envisioned includes Transuranic Extraction (TRUEX) and the Trivalent Actinide Lanthanide Separation by Phosphorus-based Aqueous Komplexes [*sic.*] (TALSPEAK) process for complete TRU recovery. The principle fission product wastes are combined with the UDS and separated Tc into a single borosilicate glass waste form.

The principle gaseous radionuclides I-129, Kr-85, C-14 and H-3 released during reprocessing are captured and converted to waste forms suitable for disposal. See Figure 4-2 New Extraction reprocessing block diagram

While this process is similar in function to the industrial process proposed by Energy Solutions, the two processes assume different processing methods and steps and so the product and waste streams cannot be directly compared.

4.1.3 UREX

Uranium Extraction (UREX) is an advanced aqueous process which also recovers all of the TRU and in addition separates the fission product waste components into three segments. The Fission Product Extraction (FPEX) process is added to separate the Cs/Sr/Ba/Rb, which is converted to a solid ceramic waste form. The Tc and Undissolved Solids (UDS) are combined with a portion of the zirconium hulls/stainless steel hardware to form a metal alloy, and the remaining fission products are converted to a borosilicate glass. UREX is the most complex of the three aqueous processes evaluated. See Figure

4-3 UREX reprocessing block diagram.

4.1.4 Electro-Chemical

The electro-chemical process is a dry process using conductive molten salt baths to recover all the TRU elements. In this process the fission products are split between three waste streams. Elements which are more noble (as measured by electro-chemical potential) than uranium, such as fuel cladding and noble metal fission products, remain as metals and are incorporated into a metal alloy waste form. Elements less noble than uranium are converted to chloride salts. The lanthanide elements are recovered from the salt by electrolysis and converted to a lanthanide glass. Excess salt is purged; the chloride is adsorbed by zeolite and bonded with glass to make the final waste form.

The principle gaseous radionuclides I-129, Kr-85, C-14 and H-3 released during reprocessing are captured and converted to a waste form suitable for disposal, although most of the I-129 in this process is not released to the gaseous phase but is converted to a molten salt and purged with the excess salt. See Figure 4-4.

4.1.5 Material Balance Calculations Bases and Assumptions

The four reprocessing methods were investigated during the Follow-on Engineering Alterative Studies (FOEAS) conducted as a part of the Global Nuclear Energy Partnership program. The material balance basis and assumptions established in reference 15 were also used for this study including the extraction separation efficiencies. However, the material balance completed for this study determined only the elemental mass flows for the numbered streams shown in Figure 4-1 to Figure 4-4. This limited dataset encompassed all of the major waste stream data components allowing rapid determination of the waste forms for the 24 representative UNF compositions.

4.2 Representative LWR Fuels

The historical and projected UNF inventory (Sections 3.1 and 3.2) were reviewed to select fuels representative of the anticipated fuel type, burn-up and age at the time of reprocessing. Three fuel burn-ups were selected to represent the historical and future PWR and BWR reactors discharge. The selected PWR burn-ups are 20, 40 and 60 GWd/MT and the selected BWR burn-ups are 15, 30 and 50 GWd/MT.

The elapsed time until fuel reprocessing is unknown and a broad range potential ages (5, 30, 100 and 500 years) were selected. This broad range of potential ages allows evaluation of various strategic decisions (i.e., to process short or long cooled fuel).

The combination of 2 fuel types, each with 3 burn-ups and 4 potential ages results in a total of 24 representative fuels.

4.3 Baseline Waste forms

The Global Nuclear Energy Partnership Integrated Waste Management Strategy Baseline Study.[16] summarized the state-of-the-art in stabilization concepts for byproduct and waste streams, and recommended a baseline of waste forms for the safe disposition of proposed waste streams from future fuel recycling processes. This baseline has been adopted for this study as applicable to the specific reprocessing method.

4.3.1 Off-Gas Waste forms

Tritium (H-3) is not captured nor treated with current generation reprocessing methods (aqueous methods practiced commercially and electro-chemical methods practiced by INL). Tritium is currently released to the environment via atmospheric or waste water discharges. This release is assumed to be an unacceptable practice in future domestic reprocessing applications. To prevent the aqueous phases from becoming contaminated with tritium, voloxidation is used to ensure tritium is released to off-gas system

where it is captured as tritiated water. The tritiated water is converted to a grout and allowed to cure in a 10 liter container, which is subsequently contained in a double steel box.

I-129 is captured on silver mordenite. The mordenite is then grouted and allowed to cure in a 55 gal drum.

C-14 is converted to carbonate and grouted. The grout is cured in a 55 gal drum.

Kr-85 is separated from the other off-gas components (including xenon) by cryogenic methods and the Kr-85 is stored in high pressure type A gas cylinders.

There is considerable uncertainty in the need to capture and treat both the C-14 and Kr-85 released during reprocessing with many factors (e.g. reprocessing facility location and environmental regulations) influencing the final decision. To provide a comprehensive range of the waste quantities these waste forms were not assumed as part of the Co-extraction processing method (to better reflect current practice) while they have been included in the other aqueous and electrochemical alternatives (to provide a bounding inventory).

4.3.2 Metal Waste forms

Compacted hulls and hardware - After being separated from the fuel, the assembly hardware (principally stainless steels) and zirconium and stainless steel based cladding are decontaminated, compacted and placed inside a HLW canister. Each canister is 2 feet in diameter by 10 feet tall and contains 3,600 kg of waste material.

Metal alloy - To reduce the waste disposal environmental impact, UREX reprocessing proposed an alloy which contains the undissolved solids (UDS) and technetium (Tc). A portion of the hulls and hardware are diverted from the compaction line to provide the zirconium and iron required to produce a durable waste form. The metal alloy is cast into a high level waste canister. Each canister is 2 feet in diameter by 10 feet tall and contains 3,600 kg of waste material.

In the electrochemical process elements more noble (as measured by electro-chemical potential) than uranium such as the hulls, hardware and noble metal fission products remain as metals. The metal waste is decontaminated by volatilizing any adhered salts and then cast into a high level waste canister. Each canister is 2 feet in diameter by 10 feet tall and contains 3,600 kg of waste material.

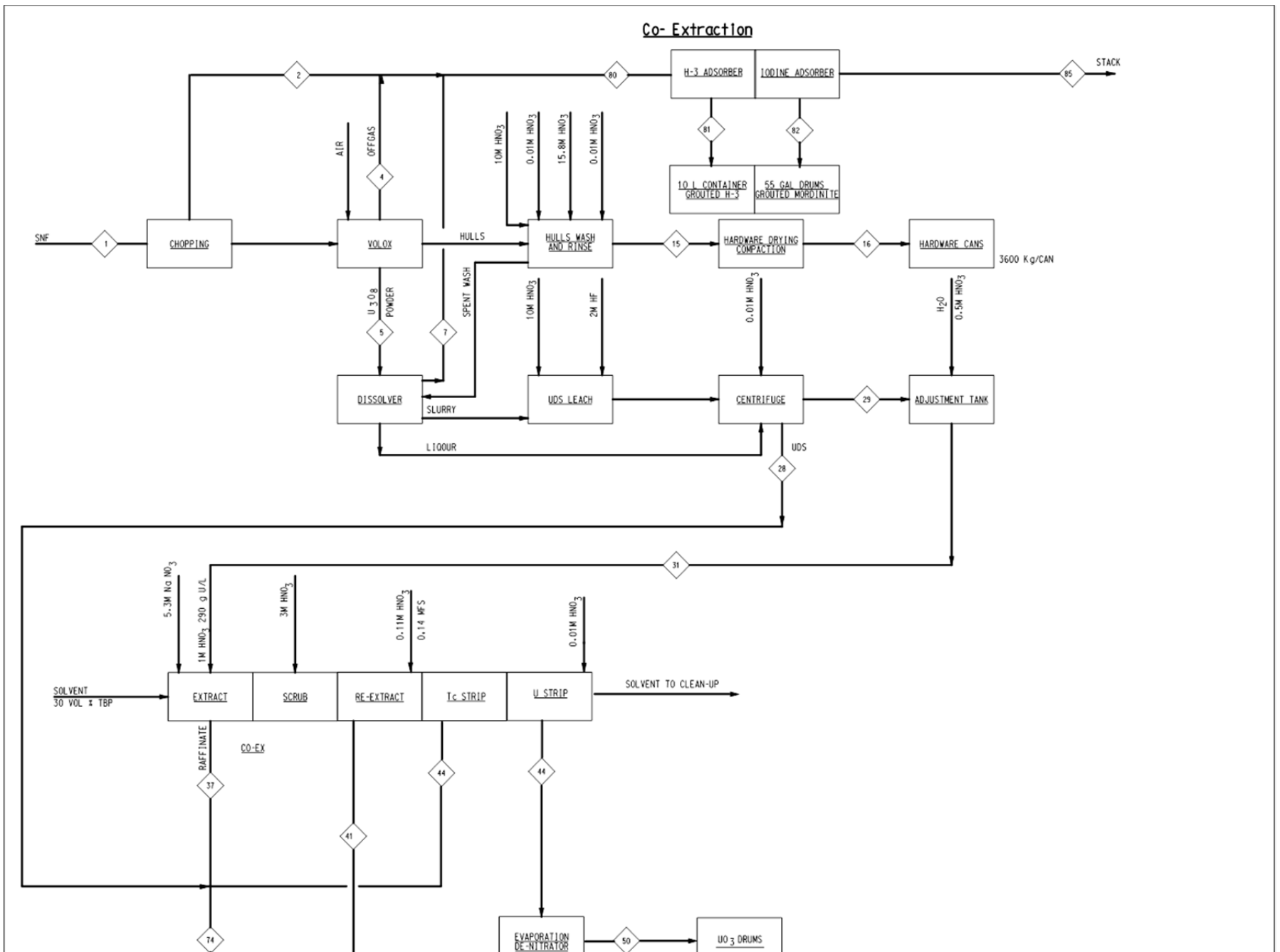


Figure 4-1 Co-Extraction reprocessing block diagram.

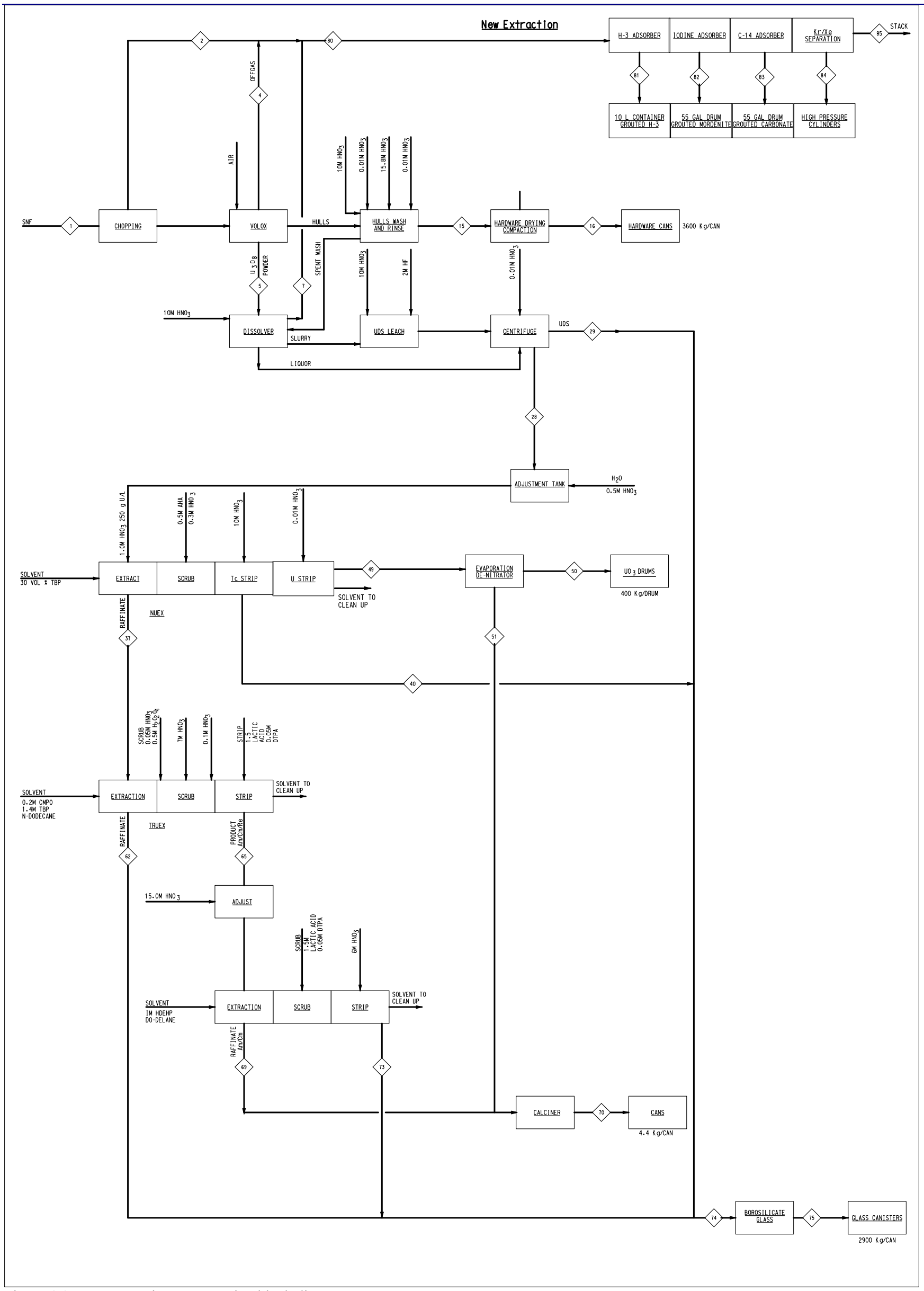


Figure 4-2 New Extraction reprocessing block diagram

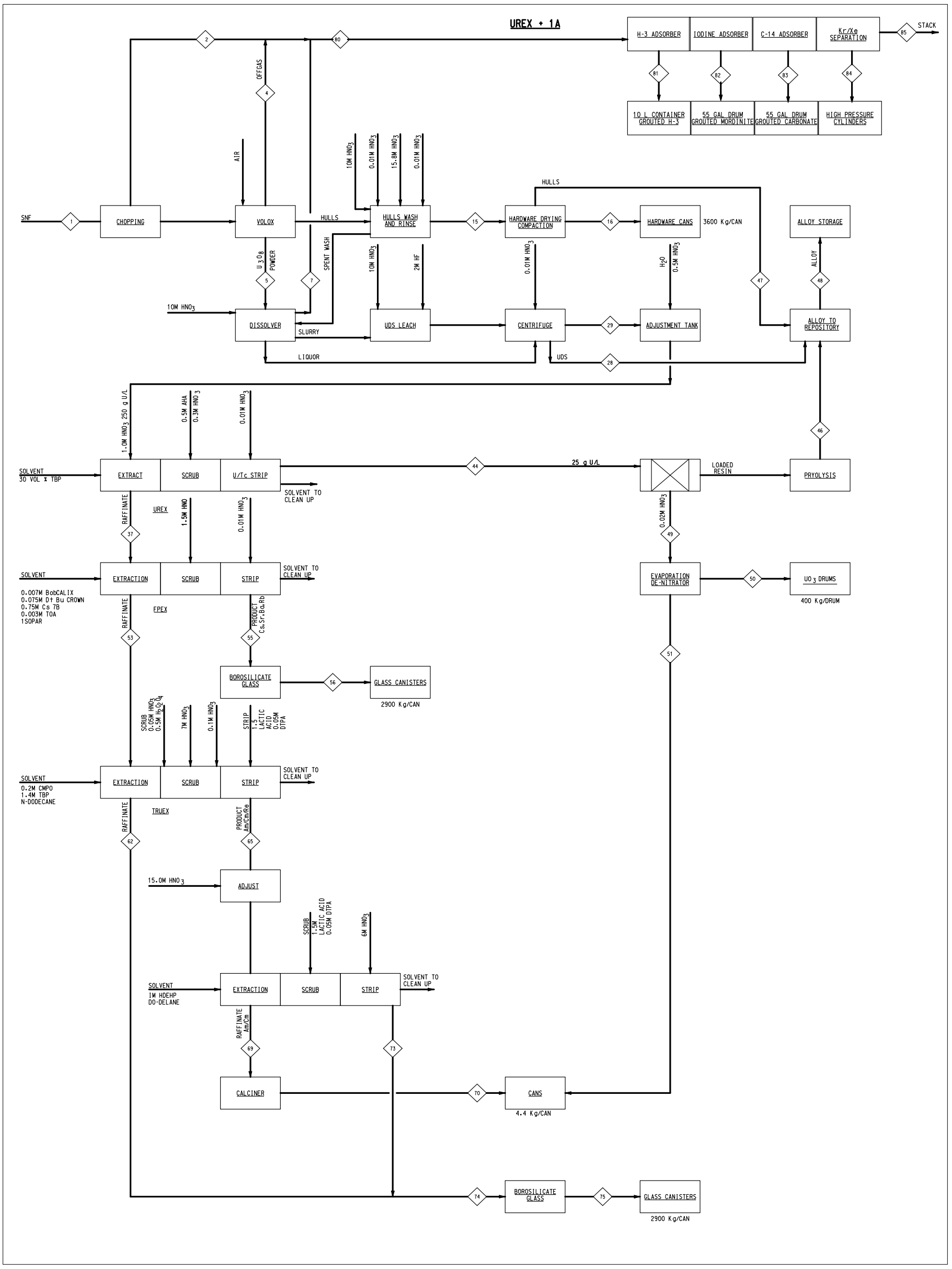


Figure 4-3 UREX reprocessing block diagram.

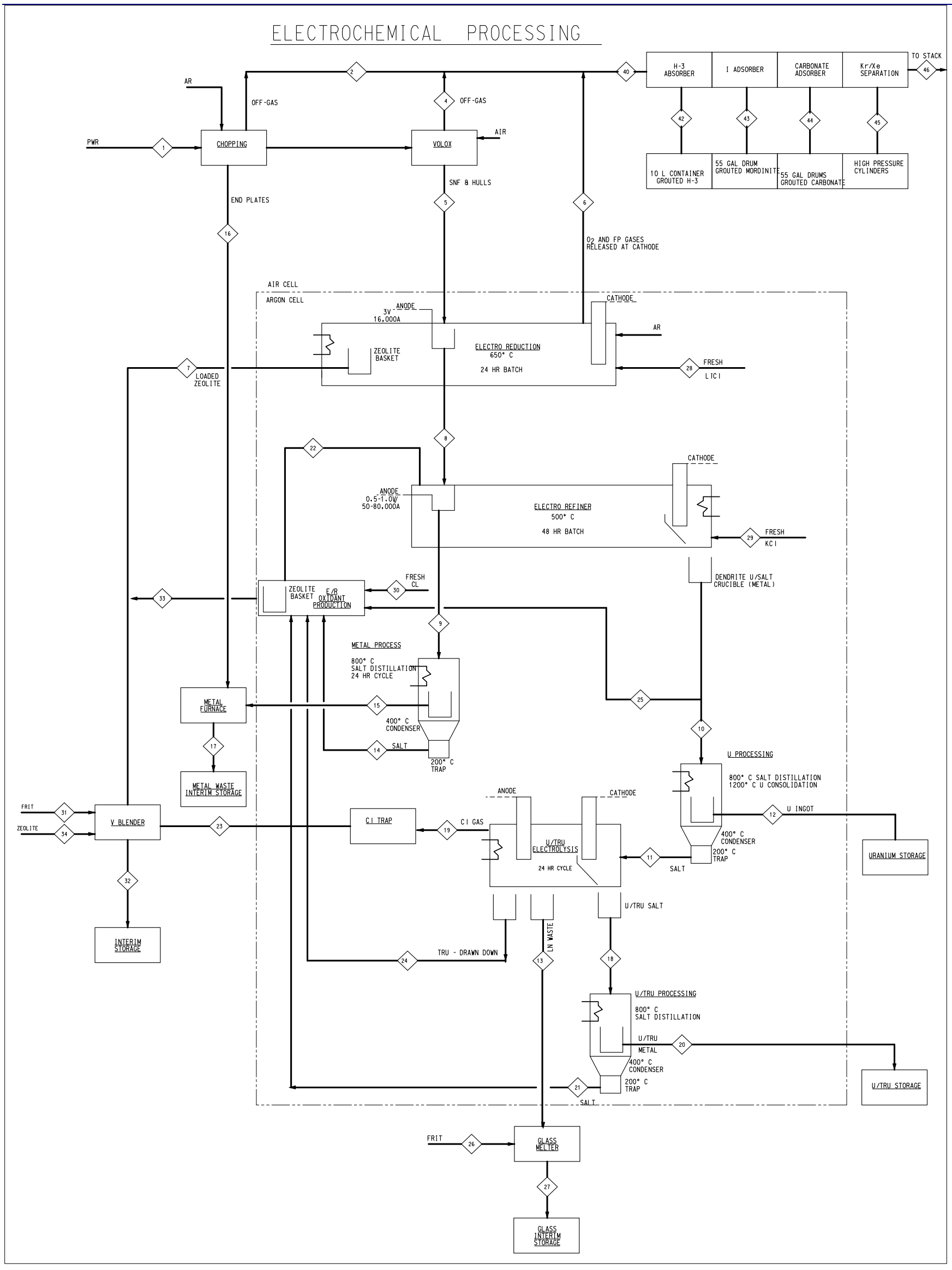


Figure 4-4 Electro-chemical reprocessing block diagram

4.3.3 Principle Fission Product Waste forms

Borosilicate Glass - In the aqueous processes most of the fission products are incorporated into a borosilicate glass. While this waste form is the accepted standard for reprocessing waste disposal, the waste form is limited by a number of attributes which must be considered in this study. The limits to avoid the formation of multiphase glasses include:

- a maximum decay heat of 14,000 watts per 2 feet diameter canister to prevent the canister centerline temperature from reaching the transition temperature
- the molybdenum trioxide solubility is limited to 2.5% by weight, and
- the noble (Ag, Pd, Rh, Ru) metals are limited to 3% by weight. The limit selected for any representative fuel allows the maximum waste loading and minimum projected waste volume, and mass. The glass is cast into a 2 feet diameter by 15 feet tall canister containing 2900kg of glass.

Cesium/strontium waste form - UREX processing proposed separating the Cs/Sr/Ba/Rb to segregate the high heat producing waste. The waste is treated by mixing the waste with a bentonite clay, high pressure pressing and high temperature sintering to produce a ceramic puck. The pucks are loaded into a 8.6 inch diameter by 86 cm tall canister. Each canister contains 120kg of ceramic waste. The waste loading is 25% Cs/Sr/Ba/Rb.

Glass Bonded Zeolite - The electrochemical process purges excess salt and fission products which have been adsorbed onto zeolite. Additional zeolite is added to sequester the excess salt chloride and then bonded with borosilicate glass. The glass bonded zeolite is cast into a 2 feet diameter by 15 feet tall canister containing 2900kg of glass. The waste form is 25% glass binder.

Lanthanide Glass - The electrochemical process also separates the lanthanide which are converted to a lanthanide based glass. The glass is cast into a 6”diameter by 60 in tall canister containing 500kg of glass. The waste loading is 50% lanthanides.

4.4 UNF Equivalent Waste Projections

The waste mass, volume, container count and decay heat per container has been estimated for each of the 24 representative fuels for each of the 4 reprocessing methods. The mass, volume, and container count results (Table 4-1 to Table 4-3) are normalized for a metric ton of the representative UNF. The decay heat per container is determined at the time of production (reprocessing). Some observations and trends for each of the waste categories are discussed below.

4.4.1 Off-Gas Waste Trends

Table 4-1 summarizes the projected off-gas waste forms. This table assumes that all reprocessing methods will capture and treat the H-3 and I-129, while Kr-85 and C-14 capture may be optional. Trends include:

- The quantity of the grouted tritium waste form is dominated by the water captured concurrently with the tritium and therefore does not vary with burn-up or age in the same manner as the tritium. Although the air used for the voloxidation step is assumed to be nearly water free (-50°C dew point) the material balance accounts for a limited amount of humid air in leakage. This water dominates the tritium present in the UNF.
- Due to the short half-life of H-3 (~12 years) and Kr-85 (~11 years) it may not be necessary to capture and treat these radionuclides if reprocessing occurs after 10 half-lives (~100yrs). The waste form projections include this data for completeness. Disposal methods for these wastes differ if the waste form is disposed immediately after production or following decay storage.
- The quantity of the grouted silver mordenite (captured I-129) projected for the electro-chemical process is only about 1% of the quantity projected for the aqueous reprocessing for any given representative fuel. This is because most of the iodine is converted and purged with the excess

salt. This long lived radioisotope may impact the disposal alternatives for the glass bonded zeolite resulting from the electrochemical process.
- The projected quantity of grouted carbonate increases only slightly with increasing burn-up corresponding to the total carbon increase in the UNF which is not linear with burn-up.
- The quantity of Kr captured is nearly constant regardless of fuel age even as the decay heat per cylinder declines due to the decay of Kr-85 (to stable Rb-85).
- The quantity of grouted carbonate projected for electrochemical reprocessing is about 2X the quantity for aqueous reprocessing for any given representative fuel. The aqueous processes are assumed to release about 50% of the carbon during voloxidation which is captured and treated for disposal. (The remaining carbon is likely released to the environment via atmospheric releases throughout the reprocessing facility and will likely be extremely difficult to capture. These losses are not included in the waste estimates for the aqueous reprocessing methods.)

The electro-chemical process includes voloxidation and an electro-reduction step in which the remaining carbon is assumed to be released.

The estimates assume a small contribution of carbon from the CO_2 in the off-gas system air in leakage (the voloxidation air is assumed be CO_2 free).

4.4.2 Metal Waste Trends

Table 4-2 summarizes the metal waste forms.

Trends include:

- The compacted hulls and hardware projected from the BWR fuels are higher than from PWR fuels on a unit base due to the lower ratio of fuel to structural materials.
- The compacted hulls and hardware projected from the UREX process are slightly lower than the other aqueous processes for a given representative fuel since a portion of the hulls and hardware have been included in the metal alloy.
- The UREX metal alloy has a fairly high initial decay heat (due to the inclusion of the UDS) if "young" UNF is processed. The decay heat is significantly reduced if the UNF is cooled for 30 or more years prior to reprocessing.
- The quantity of the metal alloy waste projected from electro-chemical processing is nearly 2X the quantity projected from aqueous reprocessing. This is due to inclusion of discarded electrode baskets and process crucibles in the metal waste stream. Although the baskets and crucibles are used multiple (~10X) times the mass of this additional waste is significant.
- The electro-chemical metal alloy has a fairly high initial decay heat (from the noble metal fission products) if "young" UNF is processed. The decay heat is significantly reduced if the UNF is cooled for 30 or more years prior to reprocessing

Table 4-1 Summary of off-gas waste forms

Off-Gas Waste Summary																											
		Off-Gas waste Common to All Aqueous Processes							Proposed Additonal Off-Gas Waste for Aqueous Processes							Electrochemical											
		Captured Tritium Grouted				Captured I on Silver Mordenite Grouted			Captured C-14 as Carbonate Grouted			Captured Kr in High Pressure Cyclinders				Captured Tritium Grouted				Captured I on Silver Mordenite Grouted			Captured C-14 as Carbonate Grouted			Captured Kr in High Pressure Cyclinders			
		Containers: 10 liter poly bottle contained within a double steel box. Each bottle contains 23 kg of cured grout				Containers: 55 gallon drum. Each drum contains 460 kg of cured grout			Containers: 55 gallon drum. Each drum contains 460 kg of cured grout			Containers: Standard Type 1 A high pressure cylinders containing 43.8 liters at 50 atm pressure.				Containers: 10 liter poly bottle contained within a double steel box. Each bottle contains 23 kg of cured grout				Containers: 55 gallon drum. Each drum contains 460 kg of cured grout			Containers: 55 gallon drum. Each drum contains 460 kg of cured grout			Containers: Standard Type 1 A high pressure cylinders containing 43.8 liters at 50 atm pressure.			
Burn-up (GWd/MT)	Age (years)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Mass (kg/MT)	Volume (ft^2/MT)	Containers per MT	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)
PWR																													
20 GWD/MT																													
	5	2.10	0.09	0.09	0.12	4.72	0.08	0.010	9.04	0.14	0.020	0.28	1.52	0.035	182	2.10	0.09	0.091	0.12	0.05	0.0008	0.00010	18.07	0.29	0.039	0.28	1.52	0.035	182
	30	2.10	0.09	0.09	0.03	4.72	0.08	0.010	9.04	0.14	0.020	0.27	1.47	0.033	37	2.10	0.09	0.091	0.03	0.05	0.0008	0.00010	18.07	0.29	0.039	0.27	1.47	0.033	37
	100	2.10	0.09	0.09	0.00	4.72	0.08	0.010	9.04	0.14	0.020	0.27	1.45	0.033	0	2.10	0.09	0.091	0.00	0.05	0.0008	0.00010	18.07	0.29	0.039	0.27	1.45	0.033	0
	500	2.10	0.09	0.09	0.00	4.72	0.08	0.010	9.04	0.14	0.020	0.27	1.45	0.033	0	2.10	0.09	0.091	0.00	0.05	0.0008	0.00010	18.07	0.29	0.039	0.27	1.45	0.033	0
40 GWD/MT																													
	5	2.10	0.09	0.09	0.20	9.22	0.15	0.020	9.31	0.15	0.020	0.55	2.95	0.067	171	2.10	0.09	0.091	0.20	0.09	0.0015	0.00020	18.62	0.30	0.040	0.55	2.95	0.067	171
	30	2.10	0.09	0.09	0.05	9.22	0.15	0.020	9.31	0.15	0.020	0.54	2.86	0.065	35	2.10	0.09	0.091	0.05	0.09	0.0015	0.00020	18.61	0.30	0.040	0.54	2.86	0.065	35
	100	2.10	0.09	0.09	0.00	9.22	0.15	0.020	9.31	0.15	0.020	0.53	2.84	0.065	0	2.10	0.09	0.091	0.00	0.09	0.0015	0.00020	18.61	0.30	0.040	0.53	2.84	0.065	0
	500	2.10	0.09	0.09	0.00	9.22	0.15	0.020	9.31	0.15	0.020	0.53	2.84	0.065	0	2.10	0.09	0.091	0.00	0.09	0.0015	0.00020	18.61	0.30	0.040	0.53	2.84	0.065	0
60 GWD/MT																													
	5	2.10	0.09	0.09	0.27	13.37	0.21	0.029	9.56	0.15	0.021	0.82	4.37	0.100	162	2.10	0.09	0.091	0.27	0.13	0.0021	0.00029	19.12	0.31	0.042	0.82	4.37	0.100	162
	30	2.10	0.09	0.09	0.07	13.37	0.21	0.029	9.56	0.15	0.021	0.79	4.24	0.097	33	2.10	0.09	0.091	0.07	0.13	0.0021	0.00029	19.12	0.31	0.042	0.79	4.24	0.097	33
	100	2.10	0.09	0.09	0.00	13.37	0.21	0.029	9.56	0.15	0.021	0.79	4.21	0.096	0	2.10	0.09	0.091	0.00	0.13	0.0021	0.00029	19.12	0.31	0.042	0.79	4.21	0.096	0
	500	2.10	0.09	0.09	0.00	13.37	0.21	0.029	9.56	0.15	0.021	0.79	4.21	0.096	0	2.10	0.09	0.091	0.00	0.13	0.0021	0.00029	19.11	0.31	0.042	0.79	4.21	0.096	0
BWR																													
15 GWD/MT																													
	5	2.10	0.09	0.09	0.10	3.74	0.06	0.008	10.90	0.17	0.024	0.20	1.09	0.025	181	2.10	0.09	0.091	0.10	0.04	0.0006	0.00008	21.80	0.35	0.047	0.20	1.09	0.025	181
	30	2.10	0.09	0.09	0.03	3.74	0.06	0.008	10.90	0.17	0.024	0.20	1.06	0.024	37	2.10	0.09	0.091	0.03	0.04	0.0006	0.00008	21.80	0.35	0.047	0.20	1.06	0.024	37
	100	2.10	0.09	0.09	0.00	3.74	0.06	0.008	10.90	0.17	0.024	0.20	1.05	0.024	0	2.10	0.09	0.091	0.00	0.04	0.0006	0.00008	21.80	0.35	0.047	0.20	1.05	0.024	0
	500	2.10	0.09	0.09	0.00	3.74	0.06	0.008	10.90	0.17	0.024	0.20	1.05	0.024	0	2.10	0.09	0.091	0.00	0.04	0.0006	0.00008	21.80	0.35	0.047	0.20	1.05	0.024	0
30 GWD/MT																													
	5	2.10	0.09	0.09	0.16	6.96	0.11	0.015	11.04	0.18	0.024	0.41	2.21	0.050	171	2.10	0.09	0.091	0.16	0.07	0.0011	0.00015	22.09	0.35	0.048	0.41	2.21	0.050	171
	30	2.10	0.09	0.09	0.04	6.96	0.11	0.015	11.04	0.18	0.024	0.40	2.14	0.049	35	2.10	0.09	0.091	0.04	0.07	0.0011	0.00015	22.09	0.35	0.048	0.40	2.14	0.049	35
	100	2.10	0.09	0.09	0.00	6.96	0.11	0.015	11.04	0.18	0.024	0.40	2.12	0.048	0	2.10	0.09	0.091	0.00	0.07	0.0011	0.00015	22.09	0.35	0.048	0.40	2.12	0.048	0
	500	2.10	0.09	0.09	0.00	6.96	0.11	0.015	11.04	0.18	0.024	0.40	2.12	0.048	0	2.10	0.09	0.091	0.00	0.07	0.0011	0.00015	22.09	0.35	0.048	0.40	2.12	0.048	0
													0.00																
50 GWD/MT																													
	5	2.10	0.09	0.09	0.23	11.61	0.19	0.025	11.37	0.18	0.025	0.66	3.51	0.080	155	2.10	0.09	0.091	0.23	0.12	0.0019	0.00025	22.75	0.36	0.049	0.66	3.51	0.080	155
	30	2.10	0.09	0.09	0.06	11.61	0.19	0.025	11.37	0.18	0.025	0.64	3.41	0.078	32	2.10	0.09	0.091	0.06	0.12	0.0019	0.00025	22.75	0.36	0.049	0.64	3.41	0.078	32
	100	2.10	0.09	0.09	0.00	11.61	0.19	0.025	11.37	0.18	0.025	0.64	3.39	0.077	0	2.10	0.09	0.091	0.00	0.12	0.0019	0.00025	22.75	0.36	0.049	0.64	3.39	0.077	0
	500	2.10	0.09	0.09	0.00	11.61	0.19	0.025	11.37	0.18	0.025	0.64	3.39	0.077	0	2.10	0.09	0.091	0.00	0.12	0.0019	0.00025	22.74	0.36	0.049	0.64	3.39	0.077	0

Table 4-2 Summary of metal waste forms.

		Metal Waste Summary															
		COEX			NUEX			UREX							Electrochemical			
		Compacted Hulls and Hardware			Compacted Hulls and Hardware			Compacted Hulls and Hardware			Metal Alloy				Metal Alloy			
		Containers: 2 ft diameter x 10 ft tall canisters. Each Canister Contains 3,600 kg.			Containers: 2 ft diameter x 10 ft tall canisters. Each Canister Contains 3,600 kg.			Containers: 2 ft diameter x 10 ft tall canisters. Each Canister Contains 3,600 kg.			Containers: 2 ft diameter x 10 ft tall canisters. Each Canister Contains 3,600 kg.				Containers: 2 ft diameter x 10 ft tall canisters. Each Canister Contains 3,600 kg.			
Burn-up (GWd/MT)	Age (years)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)
PWR																		
20 GWD/MT																		
	5	300.46	2.62	0.08	300.46	2.62	0.08	293.33	2.56	0.08	8.44	0.07	0.00	10827	562.21	4.90	0.16	812
	30	300.66	2.62	0.08	300.66	2.62	0.08	293.54	2.56	0.08	8.43	0.07	0.00	5	562.26	4.90	0.16	0
	100	300.76	2.62	0.08	300.76	2.62	0.08	293.64	2.56	0.08	8.43	0.07	0.00	3	562.29	4.90	0.16	0
	500	300.76	2.62	0.08	300.76	2.62	0.08	293.63	2.56	0.08	8.43	0.07	0.00	0	562.25	4.90	0.16	0
40 GWD/MT																		
	5	302.54	2.64	0.08	302.54	2.64	0.08	290.25	2.53	0.08	14.62	0.13	0.00	9258	573.36	5.00	0.16	1150
	30	302.84	2.64	0.08	302.84	2.64	0.08	290.57	2.53	0.08	14.60	0.13	0.00	15	573.40	5.00	0.16	0
	100	303.04	2.64	0.08	303.04	2.64	0.08	290.76	2.54	0.08	14.60	0.13	0.00	8	573.43	5.00	0.16	0
	500	303.13	2.64	0.08	303.13	2.64	0.08	290.86	2.54	0.08	14.60	0.13	0.00	0	573.40	5.00	0.16	0
60 GWD/MT																		
	5	304.68	2.66	0.08	304.68	2.66	0.08	287.05	2.50	0.08	20.93	0.18	0.01	8077	583.90	5.09	0.16	1385
	30	304.98	2.66	0.08	304.98	2.66	0.08	287.37	2.51	0.08	20.92	0.18	0.01	24	583.88	5.09	0.16	0
	100	305.38	2.66	0.08	305.38	2.66	0.08	287.77	2.51	0.08	20.92	0.18	0.01	14	584.00	5.09	0.16	0
	500	305.47	2.66	0.08	305.47	2.66	0.08	287.86	2.51	0.08	20.92	0.18	0.01	1	583.74	5.09	0.16	0
BWR																		
15 GWD/MT																		
	5	579.25	5.05	0.16	579.25	5.05	0.16	573.63	5.00	0.16	6.63	0.06	0.00	11522	871.57	7.60	0.24	436
	30	579.35	5.05	0.16	579.35	5.05	0.16	573.74	5.00	0.16	6.62	0.06	0.00	5	871.52	7.60	0.24	0
	100	579.45	5.05	0.16	579.45	5.05	0.16	573.84	5.01	0.16	6.62	0.06	0.00	3	871.55	7.60	0.24	0
	500	579.45	5.05	0.16	579.45	5.05	0.16	573.84	5.01	0.16	6.62	0.06	0.00	0	871.54	7.60	0.24	0
30 GWD/MT																		
	5	580.95	5.07	0.16	580.95	5.07	0.16	571.58	4.99	0.16	11.16	0.10	0.00	9008	878.93	7.67	0.24	564
	30	581.15	5.07	0.16	581.15	5.07	0.16	571.79	4.99	0.16	11.15	0.10	0.00	11	878.92	7.67	0.24	0
	100	581.35	5.07	0.16	581.35	5.07	0.16	571.99	4.99	0.16	11.15	0.10	0.00	6	879.00	7.67	0.24	0
	500	581.35	5.07	0.16	581.35	5.07	0.16	571.99	4.99	0.16	11.15	0.10	0.00	0	878.94	7.67	0.24	0
50 GWD/MT																		
	5	582.78	5.08	0.16	582.78	5.08	0.16	567.81	4.95	0.16	17.78	0.16	0.00	7359	890.55	7.77	0.25	707
	30	583.08	5.09	0.16	583.08	5.09	0.16	568.12	4.96	0.16	17.77	0.15	0.00	24	890.53	7.77	0.25	0
	100	583.38	5.09	0.16	583.38	5.09	0.16	568.42	4.96	0.16	17.77	0.15	0.00	14	890.57	7.77	0.25	0
	500	583.48	5.09	0.16	583.48	5.09	0.16	568.51	4.96	0.16	17.77	0.15	0.00	1	890.42	7.77	0.25	0

Table 4-3 Summary of principle fission product waste forms.

		Fission Product Waste Summary																			
		Co-Extraction				New-Extraction				UREX						Electrochemical					
		Borosilicate Glass				Borosilicate Glass				Borosilicate Glass				Cs/Sr Ceramic				Glass Bonded Zeolite				Lanthanide Glass			
Burn-up (GWd/MT)	Age (years)	Containers: 2 ft diameter x 15 ft tall canisters. Each Canister Contains 2,900 kg.				Containers: 2 ft diameter x 15 ft tall canisters. Each Canister Contains 2,900 kg.				Containers: 2 ft diameter x 15 ft tall canisters. Each Canister Contains 2,900 kg.				Containers: 22cm diameter x 220cm tall canisters. Each Canister Contains 120 kg.				Containers: 2 ft diameter x 15 ft tall canisters. Each Canister Contains 2,900 kg.				Containers: 6in diameter x 60in tall canisters. Each Canister Contains 500 kg.			
PWR		Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)
20 GWD/MT																									
	5	198.53	3.23	0.07	14000	139.30	2.26	0.05	12124	116.84	1.90	0.04	4819	14.56	0.36	0.12	5697	274.36	4.46	0.09	7160	13.66	0.03	0.03	4257
	30	147.61	2.40	0.05	7766	139.30	2.26	0.05	2911	116.84	1.90	0.04	104	14.11	0.35	0.12	2572	270.53	4.40	0.09	3239	13.67	0.03	0.03	220
	100	147.61	2.40	0.05	3059	139.30	2.26	0.05	562	116.84	1.90	0.04	7	13.64	0.34	0.11	515	266.42	4.33	0.09	637	13.68	0.03	0.03	22
	500	147.61	2.40	0.05	1004	139.30	2.26	0.05	0	116.85	1.90	0.04	0	13.52	0.33	0.11	0	264.31	4.29	0.09	0	13.69	0.03	0.03	0
40 GWD/MT																									
	5	410.33	6.67	0.14	14000	253.53	4.12	0.09	13111	212.66	3.46	0.07	4035	29.16	0.72	0.24	6034	418.04	6.79	0.14	10016	27.15	0.05	0.05	3489
	30	268.66	4.37	0.09	8367	253.53	4.12	0.09	3157	212.66	3.46	0.07	162	28.32	0.70	0.24	2463	410.35	6.67	0.14	4094	27.15	0.05	0.05	307
	100	268.66	4.37	0.09	2928	253.53	4.12	0.09	601	212.66	3.46	0.07	8	27.42	0.67	0.23	492	401.93	6.53	0.14	810	27.17	0.05	0.05	25
	500	268.66	4.37	0.09	884	253.53	4.12	0.09	0	212.66	3.46	0.07	0	27.22	0.67	0.23	0	397.60	6.46	0.14	0	27.19	0.05	0.05	0
60 GWD/MT																									
	5	658.47	10.70	0.23	14000	366.12	5.95	0.13	13851	307.10	4.99	0.11	3671	43.52	1.07	0.36	6314	553.12	8.99	0.19	11828	40.47	0.08	0.08	3321
	30	387.97	6.30	0.13	8667	366.12	5.95	0.13	3233	307.10	4.99	0.11	192	42.33	1.04	0.35	2398	541.65	8.80	0.19	4503	40.49	0.08	0.08	366
	100	387.97	6.30	0.13	2546	366.12	5.95	0.13	611	307.10	4.99	0.11	8	41.05	1.01	0.34	478	528.77	8.59	0.18	894	40.51	0.08	0.08	29
	500	387.98	6.30	0.13	654	366.13	5.95	0.13	0	307.10	4.99	0.11	0	40.75	1.00	0.34	0	521.77	8.48	0.18	0	40.54	0.08	0.08	0
BWR																									
15 GWD/MT																									
	5	148.83	2.42	0.05	14000	83.93	1.36	0.03	15191	75.53	1.23	0.03	5810	11.17	0.27	0.09	5395	248.80	4.04	0.09	5715	13.82	0.03	0.03	3133
	30	95.43	1.55	0.03	9457	90.05	1.46	0.03	3362	75.53	1.23	0.03	118	10.86	0.27	0.09	2441	246.13	4.00	0.08	2593	13.83	0.03	0.03	182
	100	95.43	1.55	0.03	4234	90.05	1.46	0.03	650	75.53	1.23	0.03	8	10.53	0.26	0.09	488	243.26	3.95	0.08	508	13.84	0.03	0.03	21
	500	95.43	1.55	0.03	1546	90.05	1.46	0.03	0	75.53	1.23	0.03	0	10.45	0.26	0.09	0	241.59	3.93	0.08	0	13.85	0.03	0.03	0
30 GWD/MT																									
	5	292.86	4.76	0.10	14000	177.56	2.89	0.06	13318	148.94	2.42	0.05	4191	22.27	0.55	0.19	5672	348.64	5.67	0.12	8618	24.00	0.05	0.05	2799
	30	188.15	3.06	0.06	8999	177.56	2.89	0.06	3369	148.94	2.42	0.05	156	21.65	0.53	0.18	2416	342.98	5.57	0.12	3672	24.00	0.05	0.05	245
	100	188.15	3.06	0.06	3337	177.56	2.89	0.06	644	148.94	2.42	0.05	8	20.98	0.52	0.17	482	336.85	5.47	0.12	724	24.01	0.05	0.05	22
	500	188.16	3.06	0.06	1058	177.56	2.89	0.06	0	148.94	2.42	0.05	0	20.82	0.51	0.17	0	333.72	5.42	0.12	0	24.03	0.05	0.05	0
50 GWD/MT																									
	5	542.03	8.81	0.19	14000	292.63	4.76	0.10	13634	245.46	3.99	0.08	3585	36.70	0.90	0.31	5894	492.86	8.01	0.17	10455	37.03	0.07	0.07	2826
	30	310.09	5.04	0.11	9449	292.63	4.76	0.10	3320	245.46	3.99	0.08	201	31.32	0.77	0.26	2605	464.14	7.54	0.16	4219	37.04	0.07	0.07	342
	100	310.09	5.04	0.11	3027	292.63	4.76	0.10	627	245.46	3.99	0.08	8	34.80	0.86	0.29	454	473.54	7.69	0.16	803	37.07	0.07	0.07	25
	500	310.09	5.04	0.11	892	292.63	4.76	0.10	0	245.46	3.99	0.08	0	34.57	0.85	0.29	0	467.68	7.60	0.16	0	37.09	0.07	0.07	0
EAS Blended for QA Check																									
60 GWD/MT																									
	5	698.29	11.35	0.24	14000	372.72	6.06	0.13	14000	305.00	4.96	0.11	3887	43.42	1.07	0.36	6303	552.04	8.97	0.19	11794	40.00	0.08	0.08	3432

4.4.3 Principle Fission Product Waste Trends

Table 4-3 summarizes the fission product waste forms. Trends include:

- Projections of borosilicate glass quantity from the Co-extraction process is limited by decay heat (14,000 watts per canister) when "young" UNF is processed.
- Projections of borosilicate glass quantity from the Co-extraction process when UNF older than 30 years are limited by molybdenum trioxide solubility at (2.5 wt %). In these cases the mass, volume and containers per metric ton is a constant regardless of the age of the UNF processed although, the decay heat continues to decline with fuel age.
- Recovery of the Am/Cm in other aqueous reprocessing methods and separation of the Cs/Sr in the UREX reprocessing reduces the decay heat such that the waste loading is limited by molybdenum trioxide solubility at (2.5 wt %). In these cases the mass, volume and containers per metric ton is a constant regardless of the age of the UNF processed although, the decay heat continues to decline with fuel age.
- Many of the fission product waste forms included in this study significantly exceed the 1500 watts/canister limit of the Yucca Mountain license application. Disposal in a TBD alternative repository could require additional decay storage time prior to disposal, or a more dilute waste form.

4.4.4 Characteristics of the Heat Generating Wastes from LWR UOX Processes

A consideration of any repository design is the decay heat being generated by the emplaced waste. To assess the potential repository thermal loads, decay heat as a function of time was calculated for the heat generating wastes resulting from the four reprocessing methods discussed.

Figure 4-5 provides the effect of fuel burn-up on borosilicate glass generated by the Co-Extraction, New Extraction and uranium extraction (UREX) processes. Since the glass waste form is limited by three principle waste loading considerations (decay heat, molybdenum solubility and noble metals solubility) the fission product content of these waste are nearly identical. The resulting decay heat of borosilicate glass from a given reprocessing method is not significantly different for the higher burn-up fuel. Using the higher burn-up decay heat will provide slightly conservative values.

The Co-Extraction and New Extraction borosilicate glasses are nearly identical in decay heat until these glasses have cooled for between 100 and 750 years. At these longer time periods the New Extraction glasses become significantly cooler since the New Extraction process recovers the TRU elements (Np, Am and Cm) which contribute to the decay heat at long time periods.

The UREX borosilicate glasses are lower in initial decay heat since this process also separates and generates a Cs/Sr/Ba/Rb ceramic waste which contains most of the short lived fission products.

Figure 4-6 provides the effect of fuel age at the time of reprocessing on borosilicate glasses generated by the Co-Extraction, New Extraction and UREX processes. The initial decay heat is lower for older fuels at the time of reprocessing due to a reduction in the short lived fission products but this difference is minimal after 100 years after waste production.

Figure 4-7 provides the effect of LWR reactor type (PWR vs. BWR) on the borosilicate glasses generated by the Co-Extraction, New Extraction and UREX processes. The decay heat from the maximum burn-up PWR fuel (60 GWd/MT) is essentially identical to the maximum burn-up BWR fuel (50 GWd/MT).

Figure 4-8 provides the decay heat of both UREX process heat generating wastes (borosilicate glass and Cs/Sr/Ba/Rb ceramic).

Figure 4-9 provides the decay heat of both Electro-Chemical process heat generating wastes (Glass bonded Zeolite and Lanthanide glass).

Figure 4-10 provides a comparison of the ceramic waste forms generated by the UREX and Electro-Chemical processes (Cs/Sr/Ba/Rb ceramic and glass bonded zeolite). These wastes have similar decay heat attributes.

Appendix L contains additional data for the decay heat of waste forms resulting from the reprocessing of LWR UOX fuel. Isotopic composition data as a function of time is not reported in Appendix L due to space considerations but is available in electronic format from the lead author.

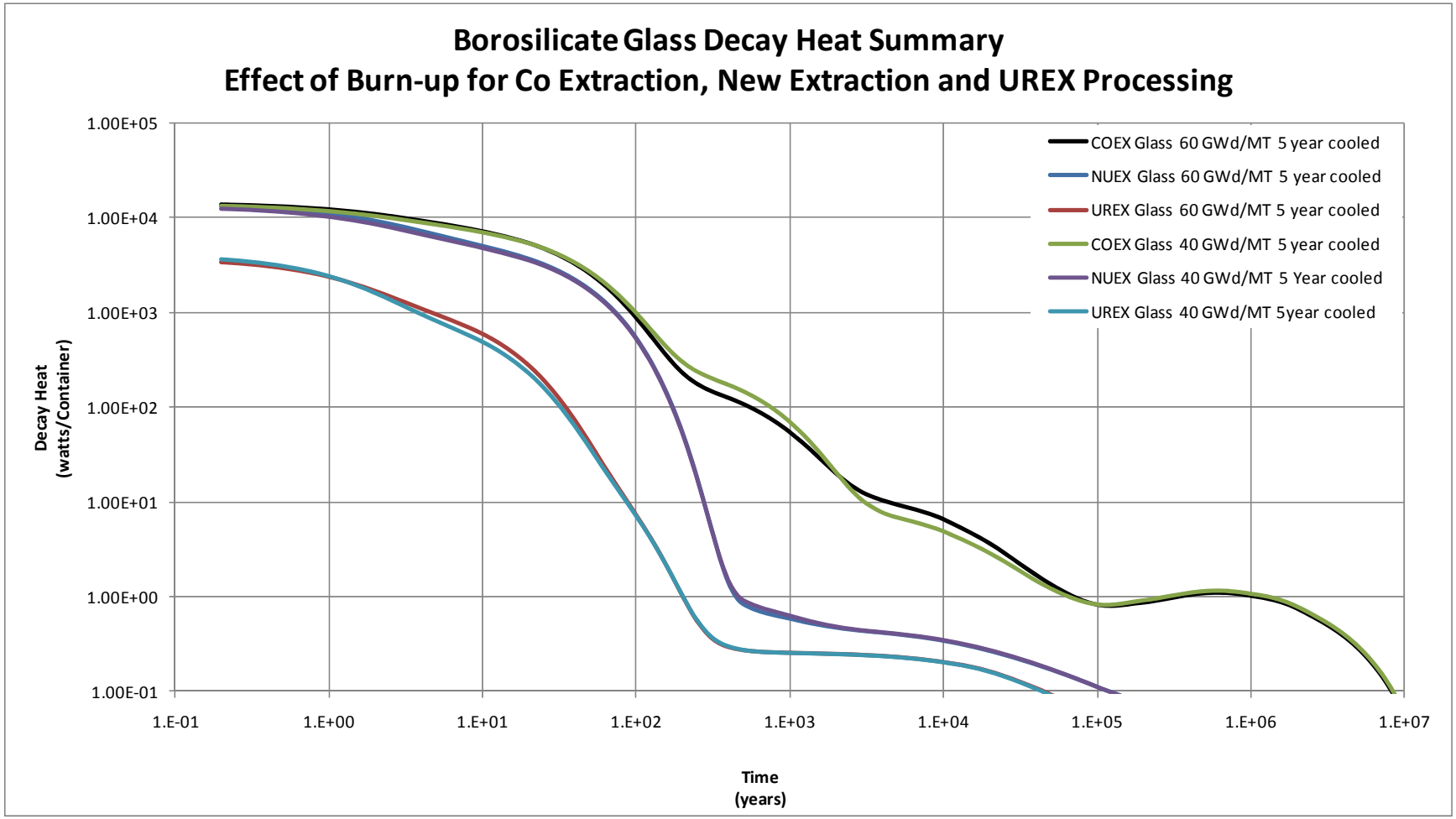


Figure 4-5 Effect of Burn-up on Borosilicate Glass Decay Heat Generated by Co-Extraction, New Extraction, and UREX Processing

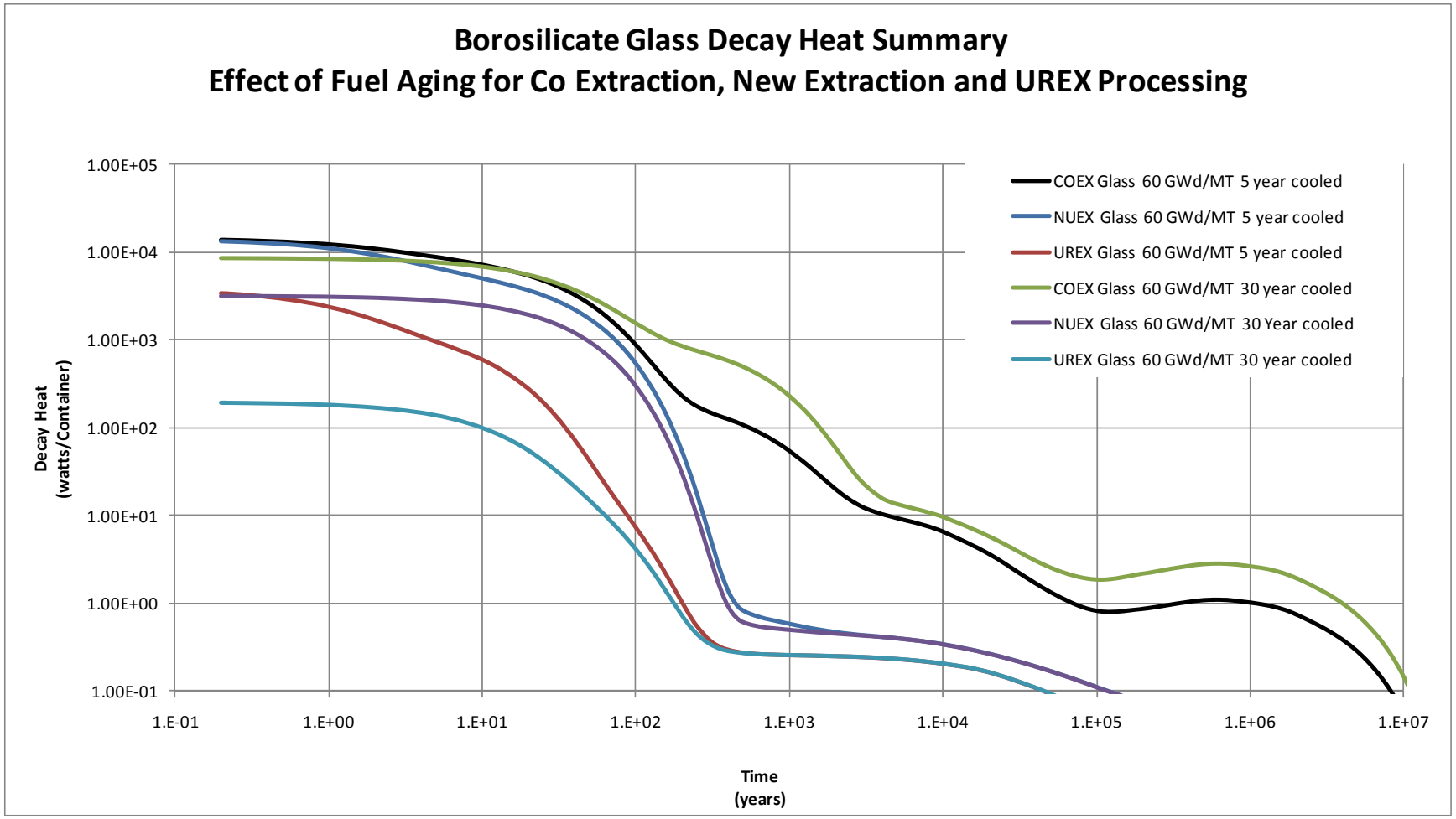


Figure 4-6 Effect of Fuel Age on Borosilicate Glass Decay Heat Generated by Co-Extraction, New Extraction, and UREX Processing

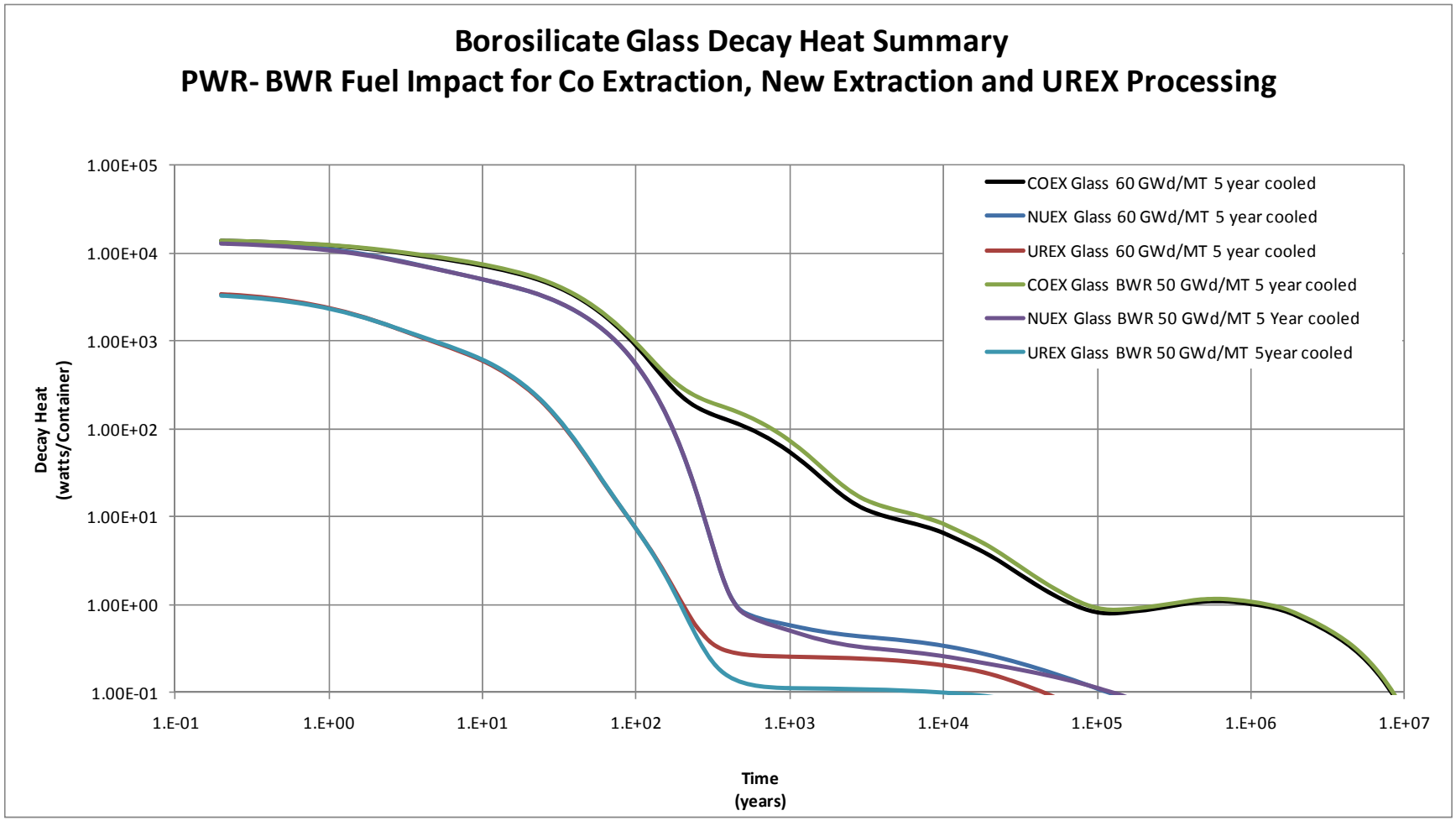


Figure 4-7 Effect of Reactor Type on Borosilicate Glass Decay Heat Generated by Co-Extraction, New Extraction, and UREX Processing

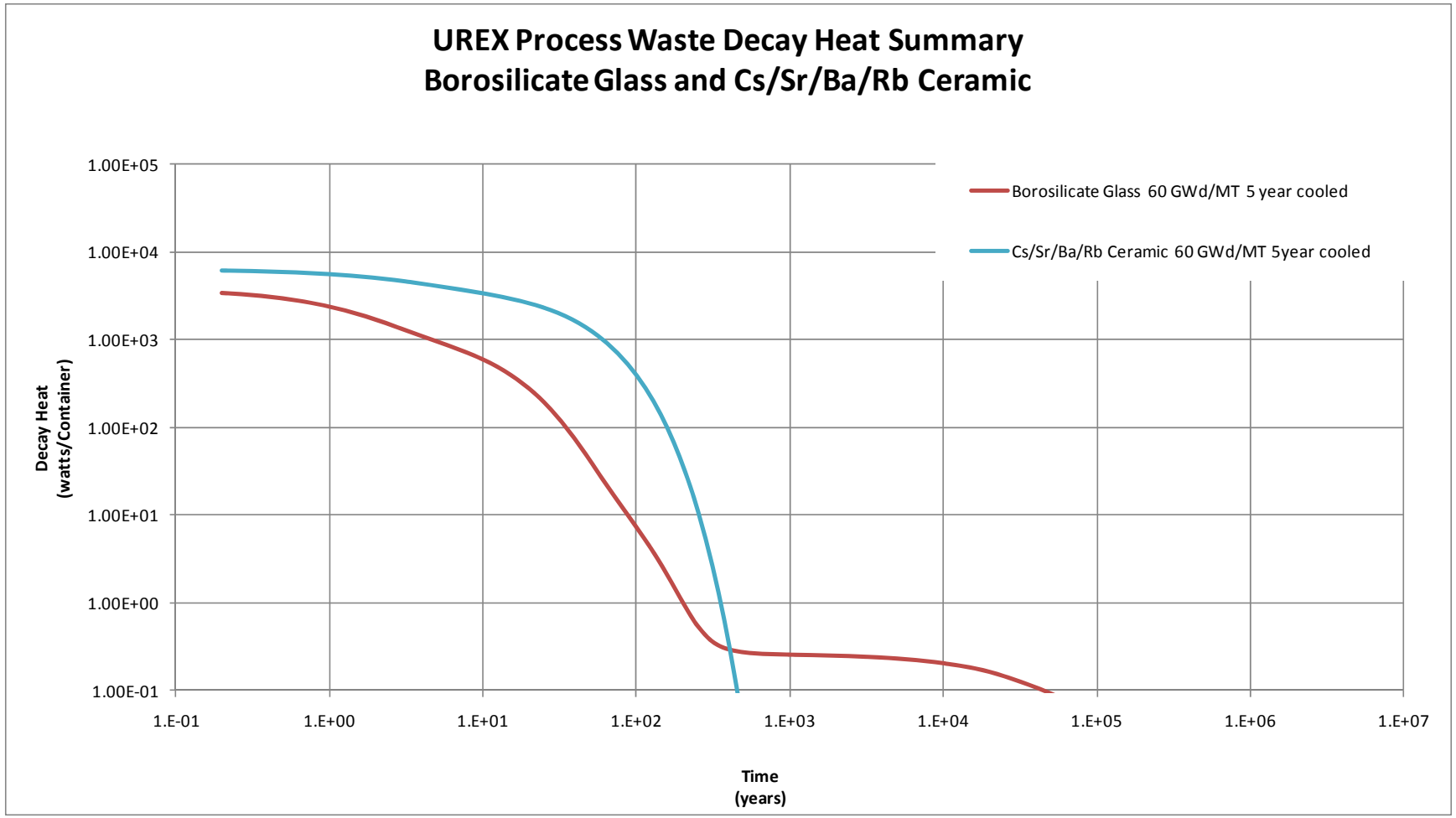


Figure 4-8 UREX Process Waste Decay Heat Summary

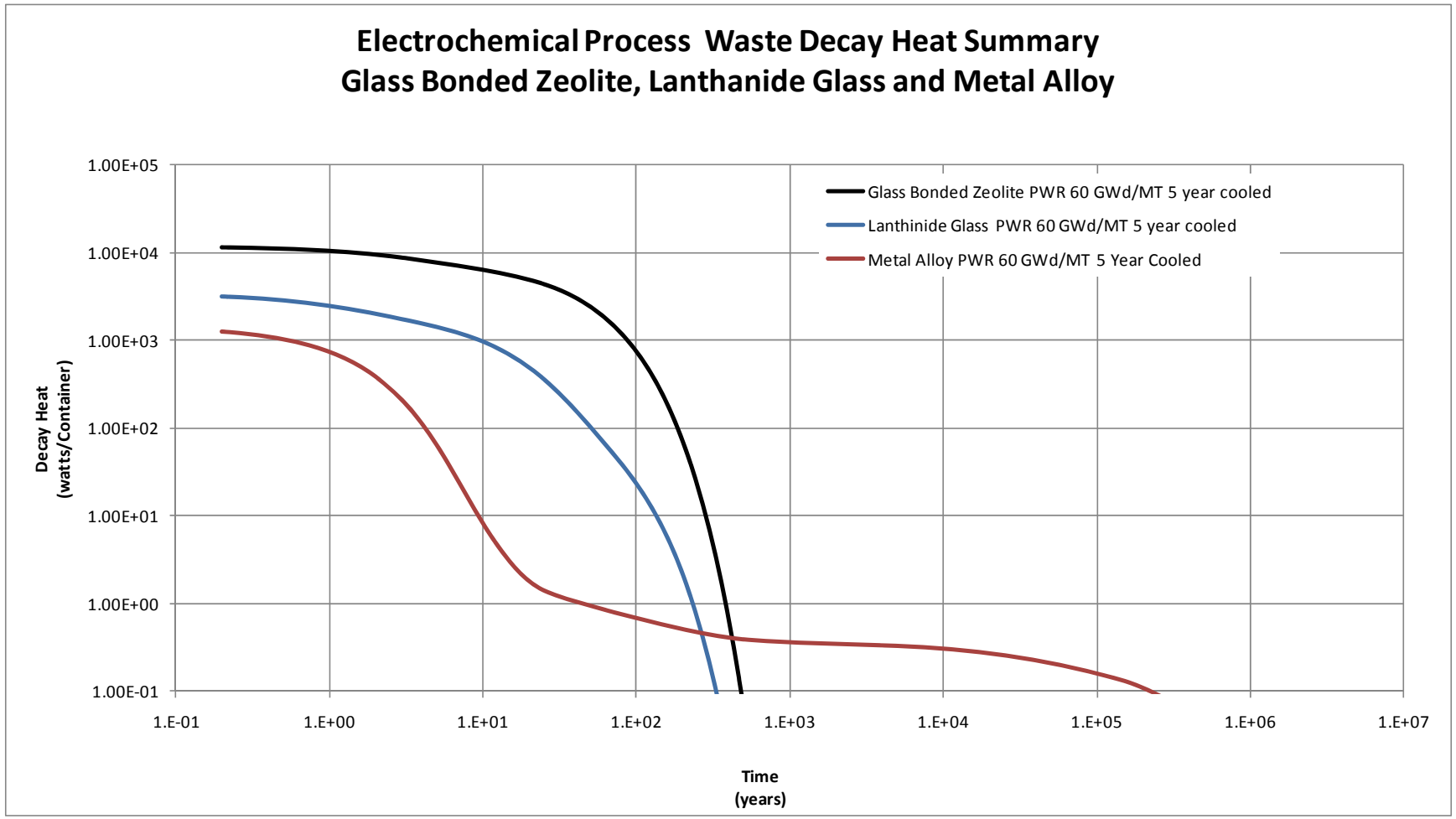


Figure 4-9 Electro-Chemical Process Waste Decay Heat Summary

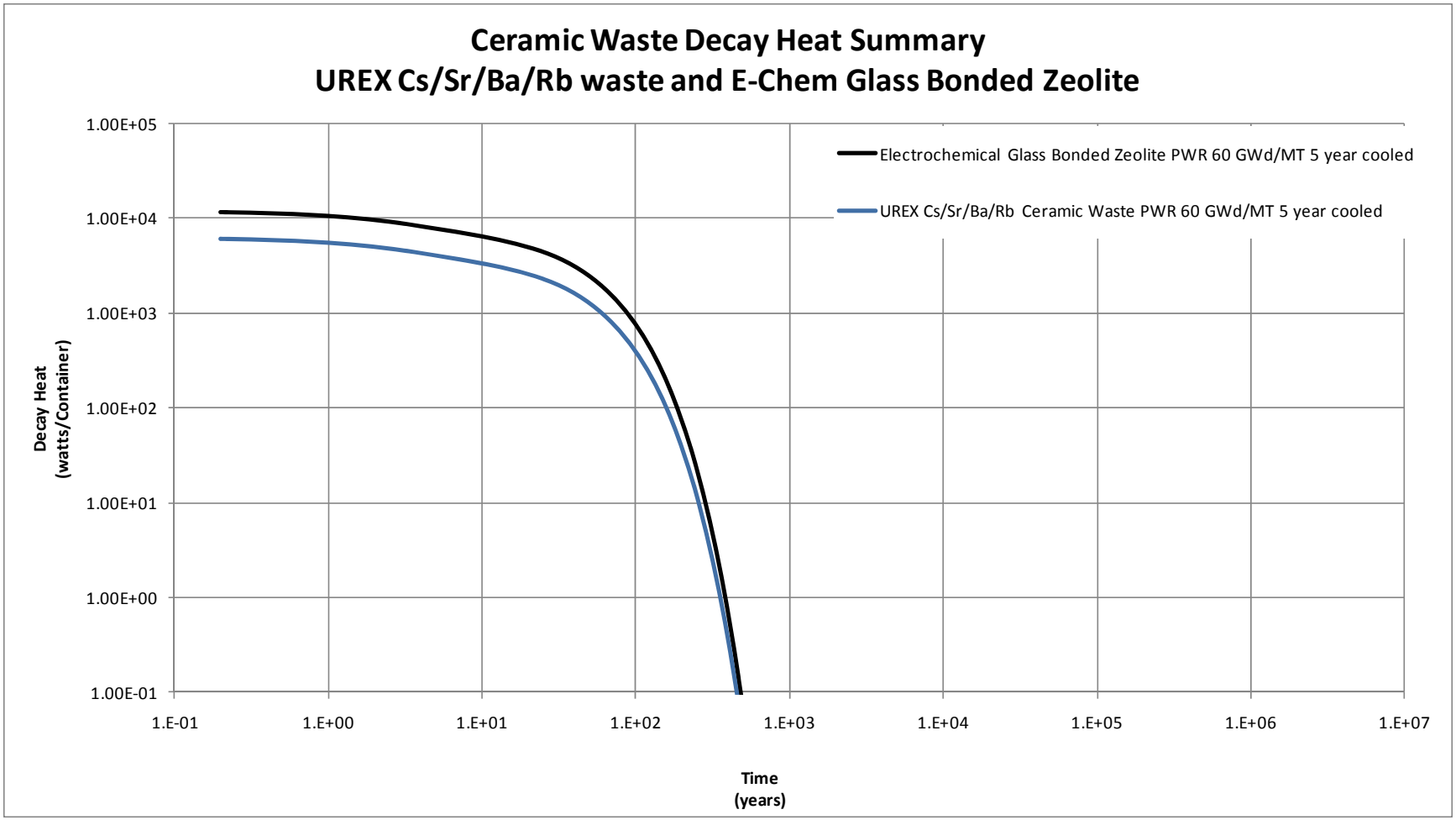


Figure 4-10 Comparison of Ceramic Waste Decay Heat Generated by UREX and Electro-Chemical Processing

4.5 Alternative Light Water Reactor Reprocessing Waste forms

The FCR&D program is also supporting the development of alternative fuel cycle reprocessing waste forms. The Separations and Waste forms Campaign's objectives in this area are to improve the durability while reducing the volume and or mass of the waste generated. The campaign is currently in the exploratory phase for development of these advanced waste forms and the information in this section is preliminary and is based on scientific and engineering judgment rather than experimentally determined results. The results in this section should be used as a general indication of expected trends as the development work progresses.

In consultation with the Separations and Waste Forms Campaign, estimates of the quantities and inventory for nine alternative waste forms have been developed. Four of the alternatives have application for the capture and disposal of volatile radionuclides; one is an alternative for technetium disposal and four are alternative fission product waste forms.

4.5.1 Alternative Off-Gas Waste Forms

The long half-life of I-129 (1.6×10^7 y) coupled with a high human dose conversion impact requires both high capture efficiency during reprocessing and long term waste durability to minimize I-129 release from the repository. While gaseous iodine waste streams can be captured on silver loaded zeolite; alternatives exist which may be more efficient or cost-effective. Explained in detail in Appendix G, this section focuses on three alternatives including:

- Glass encapsulated silver zeolite as an alternative to encapsulation in grout. A commercial BiZnB oxide glass powder is mixed with up to 20 wt% of silver mordenite loaded with iodine, pressed and densified at 500˚C, resulting in minimal AgI loss due to volatilization. Initial leach testing for iodine solubility indicated that levels below 1 micromole/liter can be achieved.
- Functionalized Silica Aerogels can capture up to 20% iodine by mass. References indicate a chemical durability two orders of magnitude higher than borosilicate glass. High chemical durability and structural stability is achieved by high silica content and low content of modifier oxides.
- Chalcogenide Aerogels can incorporate up to twice its weight in iodine. Performance testing has not been performed but is expected to have a stronger chemical affinity for iodine than the silver mordenite.

Kr-85 is captured, separated, and stored as a compressed gas in the baseline studies. Storage to allow the Kr-85 to decay (10.7 y) would be required for over one hundred years (10 half-lives) prior to release to the environment and is considered problematic. Studies have shown that Kr has extremely low mobility in silicon carbide at low temperatures and only becomes mobile above 1200°C. To date testing has demonstrated waste loadings of up to 6.7 mass%. Performance testing has not been conducted.

4.5.1.1 Alternative Off-Gas Waste Form Trends

Table 4-4 provides a summary of the alternative off-gas waste forms mass, volume, container count and the decay heat per container. The summary is normalized per metric ton of an LWR SNF with a burn-up of 60 GWd/MT being reprocessed after 5 years of cooling. For comparison the baseline waste form information from Table 4-1 is included.

In the baseline waste form, Kr is captured as 98% pure gas so that any sorbents will add to the mass in order to achieve a more durable waste form. Due to the high density of the alternative waste form the volume is decreased. The gas cylinder used as a baseline would not be used for the alternative waste form, and so 55 gallon drums were selected to be consistent with other solidified off-gas waste forms. This change in container size does not allow a meaningful comparison of the container count, but it should be noted that the increased container size and waste loading increases the decay heat by over a factor of 5.

Containers (55 gal drums) for the captured iodine waste were selected to use the maximum allowable container diameter (2 feet, see Appendix H) and provide for comparison of the waste forms. In practice these containers could be longer. Encapsulating the silver mordenite in glass instead of grout increases the mass, volume, and container count by 2.5 to 5x. The Captured I on Functionalized Silica Aerogels or Chalcogenide Aerogels decrease the mass and volume by about an order of magnitude compared to capture on silver mordenite.

4.5.2 Alternative Fission Product Waste Forms

The diverse nature of fission products and fission product chemistry has led to several waste form alternatives for encapsulation and disposal. These include: iron phosphate glass, glass ceramics, HLW Cermet, and epsilon metal. Iron Phosphate Glass has long been seen as a potential option for the vitrification of legacy wastes. Recent scoping tests analyzed 28 separate compositions and developed several promising glass waste forms. Glasses with 30 mass% waste loading were produced through melting at or below 1200 °C. The chemical durability of these partially crystalline waste forms also appeared to exceed the DOE requirements for PCT.

Glass Ceramics from a melt process can be used as an alternative to borosilicate glass. Glass ceramics were developed to provide, at a minimum, a durable crystalline phase for MoO_3 which forms a separated phase because of it limited solubility in silicate glasses. The target phase for the molybdenum is powellite ($(Ca, Nd)_1MoO_3$). Initial glass ceramic development resulted in a glass ceramic that contains powellite, oxyapatite, and lanthanide borosilicate phases. The glass phase separates upon cooling into both molybdate rich and silicate rich liquids. The molybdate rich phase then crystallizes into powellite. Oxyapatite and lanthanide borosilicate phases crystallize upon heat treatment of the glass at temperatures of 800–1100 °C. The waste loading can be up to 42% with a density of 3 kg/l. If the transition metals are targeted towards another waste form, waste loading can increase upwards of 50%. This study includes the disposition of the transition metals with the lanthanides in all the included reprocessing methods.

HLW Cermet waste form will improve the heat transfer characteristics and reduce the centerline temperature of the waste monolith and thus allow for increased waste loading compared with that for a glass.The ceramic phase of the cermet waste form provides a matrix that sequesters the short-lived, high-heat-generating fission product ^{137}Cs/Ba and ^{90}Sr/Y oxide components of used fuels. The metal phase of the cermet provides improved heat transfer and a matrix that will isolate the long-lived metallic fission products (e.g., Tc, Mo, Ru, Rh, Pd, and Ag) and cladding and hardware materials (e.g., Fe, Cr, Ni, and Sn). We believe that the cermet waste form may result in significant cost benefits through the improvements to the heat transfer from the waste during decay heat storage. The targeted waste loading is 50% with a density of 5.0 kg/l. The durability of this waste form has not been tested. Heat Tolerant Cermets are targeted towards the alkali and alkaline earth elements by adding copper to the baseline Cs/Sr bentonite clay materials to improve heat transfer. The baseline Cs/Sr/Ba/Rb clay makes up 30% and Cu makes up 70 % of the final waste form. The density is 3.5 kg/l. The durability is expected to be the same as the baseline waste form.

Epsilon Metal is a targeted waste form for Tc-99 and UDS. Tc-99 is a long lived ($2.13x10^5$ y) and highly mobile radionuclide when the wasteform contains the isotope as high valance state form e.g. Tc (IV). Tc is one of the five metals, Mo-Tc-Ru-Pd-Rh, which remain as undissolved solids during aqueous processing. Some of the Tc is soluble and must be separated from the uranium during solvent extraction. The UDS and the soluble TC are the targets towards an epsilon metal phase which are known to be very stable as natural analogues of the ε-metal phase exist or, more appropriately, existed in the remains of the natural nuclear reactors at Okolo, Gabon. The waste loading is 100%.

4.5.2.1 Alternative Fission Product Waste Form Trends

Table 4-5 provides a summary of the alternative fission waste forms mass, volume, container count, and the decay heat per container. The summary is normalized per metric ton of an LWR SNF with a burn-up

of 60 GWd/MT being reprocessed after 5 years of cooling. For comparison the baseline waste form information from Table 4-3 is included.

These waste forms all include high decay heat at the waste loadings proposed. Waste containers were sized using the constraints and methods described in Appendix H and the resulting sizes are included in Table 4-5.

The higher waste loadings assumed for the iron phosphate glass and glass ceramic waste forms reduce the mass and volume compared to the baseline borosilicate glass by less than a factor of 5. The canister count is higher than the baseline for the highest heat wastes e.g. the Co-Extraction waste, but can be less than the baseline if the decay heat is sufficiently reduced such that the maximum container size is allowed e.g. the UREX waste stream.

The HLW cermet is included as an alternative in the Co-Extraction reprocessing method and is compared to the combined baseline waste in the compressed hulls and hardware waste stream and the borosilicate glass. The alternative reduces the mass by 3x and the volume by nearly 20x.

The Heat tolerant cermets are used as alternatives to the Cs/Sr/Ba/Rb waste stream from the UREX process. Inclusion of the Cu to improve the heat transfer allows a 2 feet diameter canister to be used but the increased mass from the Cu addition is sufficient to increase the volume and container count.

The epsilon metal waste is an alternative to the metal alloy in the UREX process. The waste loading is 100% and the density and thermal conductivity are sufficiently high to allow the maximum diameter can. In fact the container will be mass limited and be only five feet tall. These properties result in the mass, volumes and container count all being significantly reduced as compared to the baseline but also results in an extremely high decay heat of over 54,000 watts per container. However the decay heat in the waste form is predominately from Ru-106 with a 369 day half-live. The decay heat is reduced by over 99% after only 5 years of storage.

Table 4-4 Summary of Alternative Waste Forms for Captured Off-gas

Alternative Waste Forms for Captured Off-gas				
	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)
Captured Kr				
Captured Kr in High Pressure Cylinders Containers: Standard Type 1 A high pressure cylinders containing 43.8 liters at 50 atm pressure.	8.16E-01	4.37	9.97E-02	162
Captured Kr on Silicon Carbide Containers: 55 gal drums containing 525kg of waste form	93	1.32	1.79E-01	904
Captured Iodine				
Captured I on Silver Mordenite Grouted Containers: 55 gallon drum. Each drum contains 460 kg of cured grout	13	2.14E-01	2.91E-02	0
Captured I on Silver Mordenite Bi Glass Encapsulated Containers: 55 gallon drum. Each drum contains 920 kg of Bi glass	67	5.36E-01	7.29E-02	0
Captured I on Functionalized Silica Aerogels Containers: 55 gallon drum. Each drum contains 440 kg of waste form	2.0	3.41E-02	4.63E-03	0
Captured I on Chalcogenide Aerogels Containers: 55 gallon drum. Each drum contains 625 kg of waste form	8.10E-01	9.54E-03	1.30E-03	0
Basis: LWR SNF 60 GWd/MTU burn-up and 5 year cooled				

Table 4-5 Summary of Alternative Fission Product Waste Forms

Alternative Waste Forms for Fission Product				
	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)
Co-Extraction Wastes				
Borosilicate Glass (Baseline) Containers: 2ft dia x 15ft tall	658	10.69	0.23	14000
Iron Phosphate Glass Containers: 10in dia x 8'-5" tall	211	3.19	0.70	4629
Glass Ceramic Containers: 13in dia x 11ft tall	151	2.28	0.23	14264
Total Borosilicate Glass and Compacted Metal (baseline)	963	13.35	0.31	n/a
HLW Cermet Containers: 2ft dia x 15ft tall	368	0.70	0.07	23185
NEW Extraction Wastes				
Borosilicate Glass (Baseline) Containers: 2ft dia x 15ft tall	366	5.95	0.13	13851
Iron Phosphate Glass Containers: 12in dia x 10ft tall	168	2.19	0.28	6529
Glass Ceramic Containers: 18in dia x 15ft tall	120	1.80	0.07	26808
UREX Wastes				
Borosilicate Glass (Baseline) Containers: 2ft dia x 15ft tall	307	4.99	0.11	3671
Iron Phosphate Glass Containers: 2ft dia x 10ft tall	131	1.76	0.04	10714
Glass Ceramic Containers: 2ft dia x 15ft tall	94	1.40	0.03	13432
Cs/Sr Bentonite Clay Containers: 22cm dia x 220cm tall containing 120kg of wasteform	44	1.07	0.36	6314
Cs/Sr Bentonite Clay with Cu Containers: 2ft dia x 15ft tall containing 3,300kg of wasteform	1477	21.07	0.45	5269
Metal Alloy (UDS+Tc+portion of hulls) Container: 2ft dia x 10ft tall Containing 3,600kg of waste form	21	0.18	0.01	8077
Epsilon Metal (UDS+Tc) Container: 2ft dia x 5ft tall Containing 3,600kg of waste form	3.3	0.01	9.19E-04	54523

5. MIXED OXIDE FUELS

Mixed oxide (MOX) fuel, contains more than one oxide of fissile or fertile materials. Itusually refers to a blend of oxides of plutonium and natural uranium, reprocessed uranium, or depleted uranium. One attraction of MOX fuel is that it is a way to destroy surplus weapons-grade plutonium, which otherwise would have to be disposed as nuclear waste. This study examines two potential sources of fissile material. Section 5.1 examines MOX fuels containing recovered plutonium and uranium derived from reprocessing commercial LWR fuel. Section 5.2 examines the MOX fuel produced from excess weapons grade plutonium.

5.1 MOX Fuel Derived from Reprocessing LWR UOX fuel

As described in Section 4.1.1 a Co-Extraction process can recover plutonium and uranium which can then be fabricated into a MOX fuel. The FCR&D has previously studied[18] various MOX fuel alternatives. Specifically they studied the scenario in which LWR UOX used nuclear fuel is burned to 51 GWd/MT, allowed to cool for 5 years post-irradiation and is then partitioned to separate the plutonium from the minor actinides, other heavy metal nuclides, and fission products. Because the Co-Extraction partitioning strategy is assumed, the spent fuel uranium in the LWR UOX SNF is assumed to be the uranium base of the MOX fuel (instead of natural or depleted uranium). This MOX fuel is stored for 2 years prior to introduction into the full MOX core. The delay time results in the build-up of Am-241 in the MOX fuel, from the decay of Pu-241.

The full MOX fuel core is subsequently burned to an average value of 50GWd/MT. The burn-up of the MOX core is limited to 50 GWd/MT because of a constraint on the plutonium content in the MOX fuel. Previous studies[19] have shown that plutonium content less than 12% (Pu in heavy metal) is necessary to ensure a negative void coefficient in a full MOX core; the specific value is plutonium isotopic vector dependent, but that dependence was not investigated in reference 18.[18]

Table 5-1 provides a summary of the LWR derived MOX fuel parameters. The average plutonium enrichment is 10.74%. Therefore, each metric ton of LWR fuel which is reprocessed allows fabrication of 108.9 kg of MOX fuel. Table 5-2 and Figure 5-1 provide the decay heat of the MOX fuel as a function of time. Appendix I Table I-1 provides the detailed isotopic composition of the discharged MOX fuel after 5, 30, 100 and 500 years of cooling. The discharged MOX fuel is assumed to not be recycled and is considered potential waste requiring disposition.

Table 5-1 includes only the heavy metal portion of the MOX fuel assembly. The hardware (cladding, spacers, etc.) are not included. Estimates of the hardware mass are estimated based on the historical mass of a PWR assembly of 158kg per assembly. The hardware mass and decay heat estimates are included in Table I-1.

Table 5-1 LWR Derived MOX Fuel Summary

LWR UOX fuel burn-up (GWd/t)	50
LWR MOX fuel burn-up (GWd/t)	50
LWR UOX core	
Uranium enrichment (%U-235)	4.21
Pu-239 in 5-yr cooled fuel (% total Pu)	52.7
Fissile Pu (239 & 241) in 5-yr cooled fuel (% total Pu)	64.7
Total Pu in 5-yr cooled fuel (% initial HM))	1.17
Total MA in 5-yr cooled fuel (% initial HM)	0.14
Total Pu in 5-yr cooled (kg/GWt-d)	0.234
Total MA in 5-yr cooled (kg /GWt-d)	0.027
Cycle length (Days)	495
LWR MOX core[a]	
Pu content in initial MOX fuel (%Pu/HM)	10.74
Uranium consumption (%)	4
Pu consumption (%)	25
Pu-239 consumption (%)	42
Pu fissile consumption (%)	37
Am production (%)	450
Np content in 5-yr cooled fuel (kg per initial ton MOX fuel)	0.89
Cm content in 5-yr cooled fuel (kg per initial ton MOX fuel)	0.99
Cycle length (Days)	495
[a]Consumption, production, and content data are differences between charge and 5-year post-irradiation states.	

Table 5-2 MOX Fuel 50 GWd/MT Used Fuel Decay Heat

Time (years) / Decay Heat (Watts/MT)	Discharge	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	5,737	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	13,829	991	561	352	221	110	1	0
Noble Metals Ag, Pd, Ru, Rh	23,181	12	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	46,102	110	21	4	1	0	0	0
Actinides Ac, Th, Pa, U	38,779	0	0	0	0	0	1	1
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	76,896	4,878	4,062	3,504	3,110	2,697	1,517	1,068
Others	19,517	13	2	1	0	0	0	0
Total	**224,040**	**6,004**	**4,647**	**3,860**	**3,332**	**2,807**	**1,519**	**1,068**

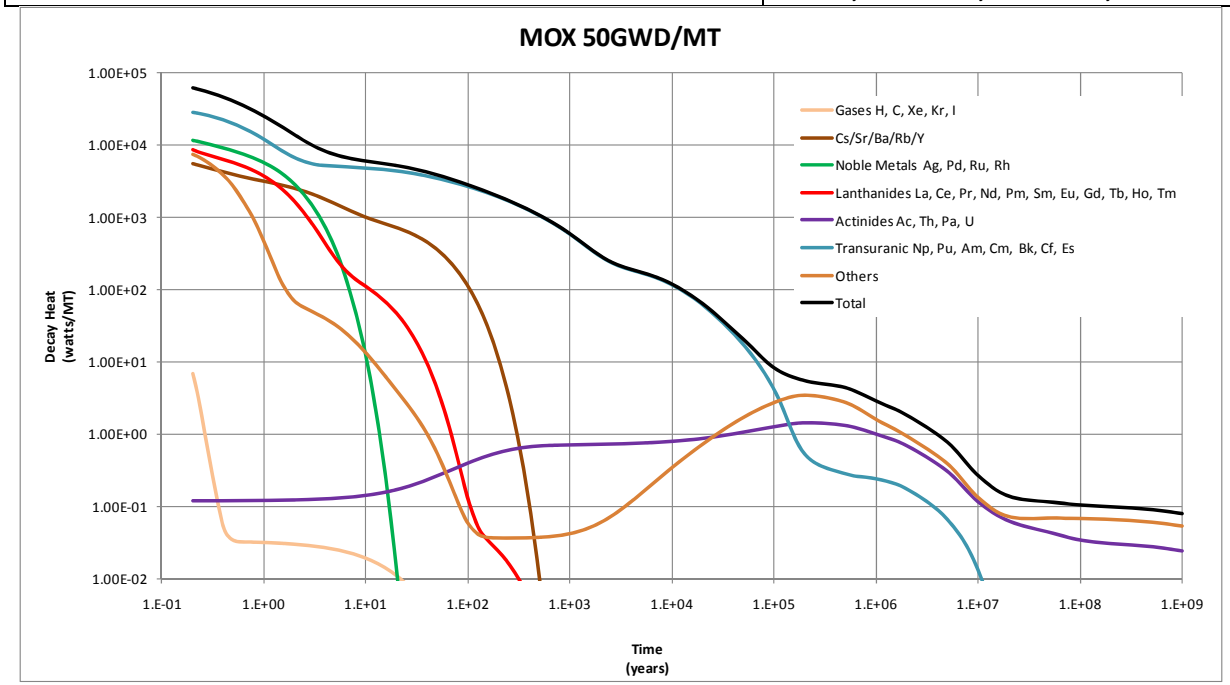


Figure 5-1 MO Fuel 50 GWd/MT Used Fuel Decay Heat

5.2 Weapons Grade Plutonium Derived MOX

Recognizing the threat of surplus weapons grade Pu to global security, the governments' of the U. S. of America and the Russian Federation agreed to pursue a long-term disposition initiative limiting the availability of this material for weapons use. The DOE selected the MOX fabrication process, successfully implemented in Europe, as the preferred nuclear non-proliferation activity to convert this excess plutonium into nuclear fuel for commercial nuclear power reactors. The MOX Fuel Fabrication Facility is currently under construction at the SRS and is expected to begin fuel production in 2017.

The MOX fuel is planned to be irradiated in PWR reactors that utilize 193 fuel assemblies containing 264 fuel rods in a square 17x17 array. The fuel assembly design is designated as the Framatome MK-BW/MOX1. The current plan is to irradiate the assemblies for two eighteen month cycles and then discharge them. The maximum fuel rod burn-up is limited to 50GWd/MT which is significantly less than current uranium fuel burn-up limits. The maximum plutonium content is limited to 6 weight percent (5% average) and the plutonium isotopic distribution may vary in the unirradiated fuel between 90 and 95% Pu-239, and between 5 and 9% Pu-240, and less than 1% Pu-241. The remainder of the fuel will be depleted uranium with a U-235 content between 0.2 and 0.3 weight percent.

The US MOX fuel program will dispose of 34 MT of weapons grade plutonium. To dispose of this quantity of plutonium the program will produce approximately 1684 fuel assemblies[23]. Each assembly will contain 462.2 kg HM per unirradiated fuel assembly (ref 23). A total of approximately 77.8MT HM of MOX fuel will be generated.

Reference 23 provides the isotopic content (Curies) of a typical MOX irradiated fuel assembly. The activity values have been converted to mass per assembly in Appendix J Table J-1. It should be noted that using this approach results in an underestimation of the total mass of material in each assembly since stable isotopes are reported in the reference. The authors are continuing to work with the MOX Fuel Fabrication Facility contractor to obtain a more complete data set.

6. POTENTIAL WASTE DERIVED FROM FULL RECYCLING ALTERNATIVES

A key attribute of the "fully closed" nuclear fuel cycle is that no UNF is disposed, only UNF reprocessing wastes are disposed. Power reactor systems have been previously studied with the majority of such studies utilizing fast spectrum reactors. These prior studies include numerous variations related to:

1. The start-up core, which can be produced from low enriched (<20%) uranium, weapons grade plutonium, or recovered TRU materials from existing LWR UNF (as described in Section 4).
2. The "equilibrium" core, which can have design and operating parameters specified to result in a TRU conversion ratios of:
    - less than 1 for TRU burning modes, these cases require additional TRU materials to produce the next reactor fuel charge,
    - equal to 1.0 for breakeven reactor operation such that the TRU production and consumption are balanced over each reactor cycle or
    - greater than 1 for systems which have a net production of TRU elements over each reactor cycle
3. Fuel type, which is typically an oxide, metal alloy or carbon based materials.
4. The reactor coolant, which is typically sodium, lead mixtures or gasses to maintain the fast reactor spectrum.

To investigate reducing the long term (vs. transitional) TRU disposal burden on the repository, this study selected sodium cooled FR with design features and operating parameters such that a burning TRU conversion ratio (< 1.0) is achieved. Both oxide and metal fuel forms are used and these fuel types are "associated" with aqueous and electro-chemical reprocessing technologies respectively. While this association is not technically mandated, using this association does allow the differences in the reprocessing methods to be examined.

6.1 Advanced Burner Reactor Design and Operating Parameters

Advanced Burner Reactor (ABR) core designs have been investigated and documented in references 20, 21 and 22.[20,21,22] These studies document the basic design and operating parameters for a 1000MWt sodium cooled reactor using both U-TRU-Zr metal alloy fuel and U-TRU oxide fuel. Table 6-1 summarizes key parameters for TRU conversion ratios of 0.5 and 0.75 for both fuel types. Some parameters (e.g. fuel mass per assembly) were obtained from the referenced author's working papers.

The discharged fuel isotopic concentrations associated with these studies were obtained from the System Analysis transmutation library. Figure 6-1 provides the decay heat of these fuels which are all similar. The parameters in Table 6-1 and the UNF isotopic data were combined to generate an overall reactor, fuel recycling, and fuel fabrication material balance for the four reactor configurations. These material balances are documented in Appendix K Tables K-1, K-2, K-3 and K-4 and are summarized in Table 6-2. Since the reactors operate with a TRU conversion rate of less than 1.0, additional TRU must be supplied to the reactor system each year. The TRU source from the reference documents is LWR UOX fuel with a burn-up of 50 GWd/MT cooled for 5 years. Both the quantity and quantity of LWR fuel which must be reprocessed annually is provided in Table 6.2

Table 6-1 Reactor Parameter Summary

	Oxide Fuel Core 1000 MWt CR=0.75	**Oxide Fuel Core** 1000 MWt CR= 0.5	**Metal Fuel Core** 1000 MWt CR = 0.75	**Metal Fuel Core** 1000 MWt CR =0.5
Power, (MWt)	1000	1000	1000	1000
Cycle Length (Effective Full Power Days)	353	326	232	221
Number of Batches (IC/MC/OC)	6 / 6 / 7	6 / 6 / 7	6 / 6 / 6.5	6 / 6 / 7
Fuel Form	U-TRU Oxide	U-TRU Oxide	U-TRU-10%Zr	U-TRU-10%Zr*
TRU Feed	Recycled ABR fuel (2 yr cooled)+ LWR 50 GWd/MT 5Yr cool	Recycled ABR fuel (2 yr cooled)+ LWR 50 GWd/MT 5Yr cool	Recycled ABR fuel (2 yr cooled)+ LWR 50 GWd/MT 5Yr cool	Recycled ABR fuel (2 yr cooled)+ LWR 50 GWd/MT 5Yr cool
TRU Enrichment (IC/MC/OC)	21.3 / 26.6 / 31.9	32.2 / 40.2 / 48.3	16.1 / 20.1 / 24.2	27.2 / 34.1 / 40.9
TRU Enrichment (avg)	25.3	38.2	21.3	33.8
Number of Batches (IC/MC/OC)	6 / 6 / 7	6 / 6 / 7	6 / 6 / 6.5	6 / 6 / 7
Conversion Ratio (TRU)	0.75	0.5	0.75	0.5
BOEC Core Loading (HM/TRU, MT)	15.3 / 3.88	10.9 / 4.07	13.4 / 2.85	9.45 / 3.08
Discharge Burn-up (Avg/Peak, GWd/MT)	131 / 192	166 / 257	99.6 / 127	132 / 177
Total Assemblies	325	325	313	313
Drivers (IC/MC/OC)	72 / 36 / 36	72 / 36 / 36	30 / 42 / 72	42 / 66 / 36
Control Rods (Primary/Secondary)	16 / 3	16 / 3	16 / 3	16 / 3
Reflector	102	102	90	90
Shield	60	60	60	60
Mass HM per Assembly(IC/MC/OC,kg)	112.5 / 112.7 / 113.0	82.0 / 82.3 / 82.6	97.6 / 97.7 / 97.8	73.7 / 70.2 / 64.9
Mass Zr per Assembly (IC/MC/OC, kg)	0 / 0 / 0	0 / 0 / 0	10.8 / 10.9 / 10.9	8.2 / 9.4 / 11.3
Mass Bond Na (kg)	0	0	2.34	1.77
Mass HT-9 Hardware (kg/assembly)	325.6	325.6	359.9	359.9

*Zr fraction is 10 wt % when the TRU fraction is less than 30 wt % (TRU/HM*100) and increases to 40 % Zr at 100% TRU

Table 6-2 Overall Reactor Material Balance Result

	Oxide Fuel Core 1000 MWt CR=0.75	**Oxide Fuel Core** 1000 MWt CR= 0.5	**Metal Fuel Core** 1000 MWt CR = 0.75	**Metal Fuel Core** 1000 MWt CR =0.5
Initial Core Charge (HM/TRU/Zr, MT)	16.23 / 4.04	11.84 / 4.46	14.07 / 2.98 / 1.57	10.07 / 3.34 / 1.37
Annual Fuel Requirements (HM/TRU/Zr, MT)	2.70 / 0.67	2.13 / 0.80	3.55 / 0.75 / 0.25	2.68 / 0.89 / 0.22
Annual LWR to Supply TRU (MT/yr)	6.72	14.72	5.78	14.00

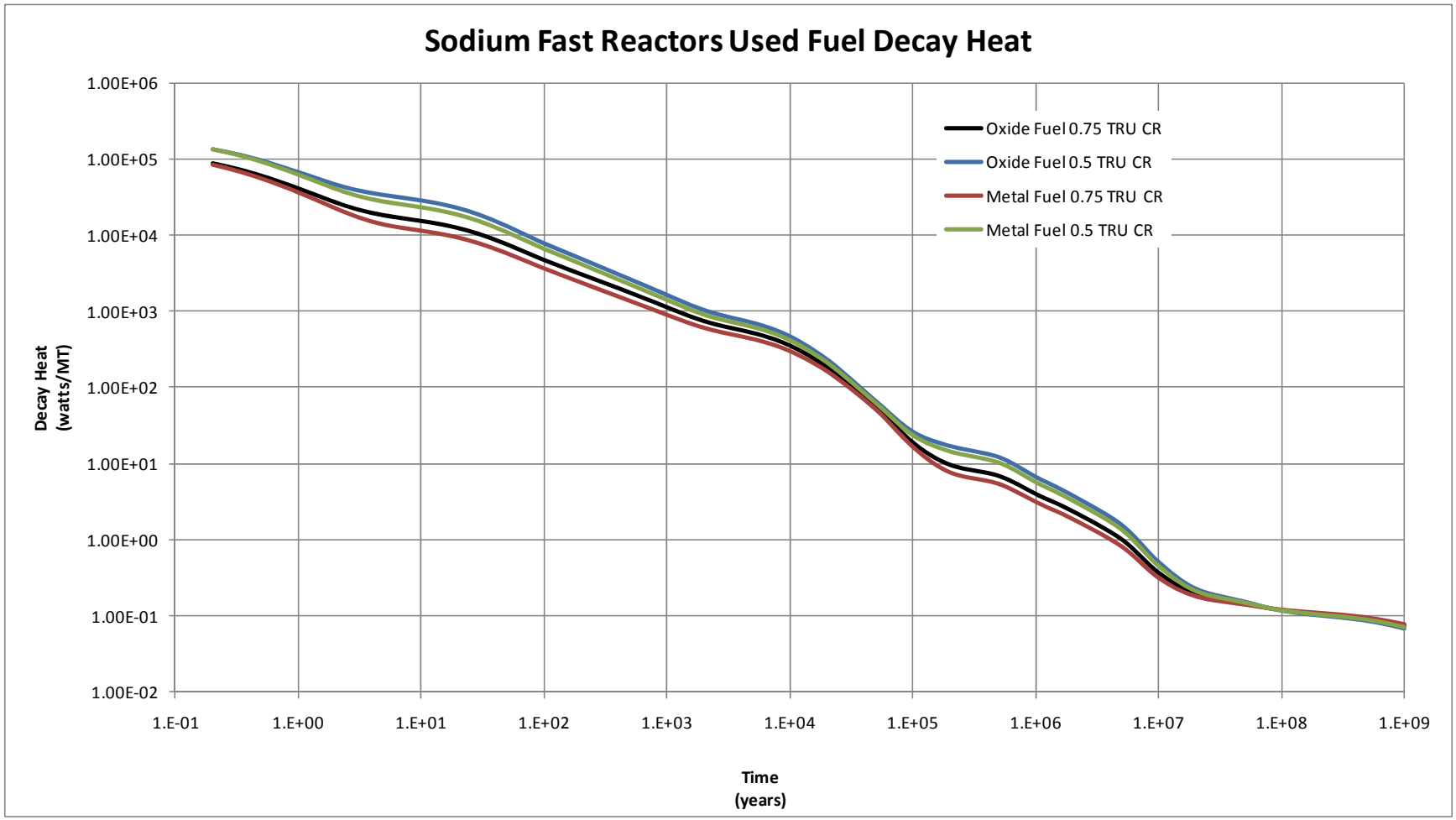


Figure 6-1 Sodium FR Used Fuel Decay Heat

The waste unit quantities resulting from reprocessing the ABR fuel (Section 6.1.3) are determined per MT of fuel recycled. However the repository system analyst will likely need to know the total quantities of waste to be disposed. In order to determine the total quantities several additional parameters will need to be considered. These include the thermal efficiency and overall utility of the power plant if such studies are related to net power generation. The total quantities must also include the waste generated from re-processing the LWR fuel as discussed in Section 4.[4]

6.1.1 Recycling ABR Oxide Fuel

To provide a tool for evaluation of the ABR reactor concepts, the potential waste from reprocessing oxide fuels has been determined using the new extraction method described in Section 4.1.2. The same material balance parameters and assumptions have been used (reference 15) and the block flow diagram in Figure 4.2 is applicable. The baseline waste forms described in Section 4.3 have been maintained to allow direct comparison with the waste generated by recycling LWR UOX fuel.

6.1.2 Recycling ABR Metal Fuel

The potential waste from reprocessing metal fuels has been determined using the electro-chemical process described in section 4.1.4 using the same material balance parameters and assumptions from reference 15. However, the electro-reduction process is not required since the fuel is introduced to the reprocessing facility as a metal. A revised electrochemical block follow diagram for metal fuel is given in Appendix K Figure K-1.

6.1.3 Potential Waste Summary from Recycling ABR fuels

The potential waste from reprocessing the metal and oxide advanced burner reactor fuels are provided in Table 6-3, 6-4 and 6-5.

Table 6-3 summarizes the projected off-gas waste forms. Trends include:

- The quantity of the grouted tritium waste form is constant regardless of the fuel type or burn-up. Although the increasing burn-up increases the decay heat per container, the mass and volume of the waste form is dominated by the water (from humid air in-leakage) captured concurrently with the tritium. The waste form unit mass, volume and containers per metric ton of fuel is the same as the LWR UOX (Table 4-1) since the material balance parameters associated with this unit operations are identical.
- The quantity of the grouted silver mordenite is proportional to the burn-up and enrichment for the aqueous process. The quantity for the electro-chemical process is assumed to be zero since the iodine is converted to a chloride salt and is included in the glass bonded zeolite waste form.
- The captured Kr mass, volume, container count and decay heat is proportional to the burn-up.
- The quantity of grouted carbonate projected for electrochemical reprocessing is about 2x the quantity for aqueous reprocessing for any given representative fuel. The aqueous processes are assumed to release about 50% of the carbon during voloxidation which is captured and treated for disposal. (The remaining carbon is likely released to the environment via atmospheric releases throughout the reprocessing facility and will likely be extremely difficult to capture. These losses are not included in the waste estimates for the aqueous reprocessing methods.)
- The electrochemical process includes an electro-refining step in which the remaining carbon is assumed to be released.
- The estimates assume a small contribution of carbon from the CO_2 in the off-gas system air in leakage.

Table 6-4 summarizes the projected metal waste forms.

Trends include:

- The unit quantity of the metal hulls and hardware are about 10x more than reprocessing LWR UOX fuel due to the higher HT-9 (stainless steel) in the advanced burner reactor fuel assembly design. This increase is due to both the smaller fuel pin diameter and the outer duct surrounding the 271 fuel pins in each assembly.
- The unit quantity of metal waste increases per metric ton of ABR fuel reprocessed as the enrichment and burn-up increases. This is due to the assumed constant mass of the hardware per assembly while the quantity of heavy metal decreases by approximately 25% per assembly (see Table 6-1).
- The quantity of the metal alloy waste projected from electro-chemical processing is nearly 2x the quantity projected from aqueous reprocessing. This is due to inclusion of discarded electrode baskets and process crucibles in the metal waste stream. Although the baskets and crucibles are used multiple (~10x) times the mass of this additional waste is significant.

Table 6-5 summarizes the projected fission product waste forms. Trends include:

- The quantity of the borosilicate glass waste form is limited by the decay heat to 14,000 watts per canister. Combined with the high burn-up, the quantity of borosilicate glass is nearly 10x the quantity resulting from reprocessing LWR UOX fuel (see Table 4-3).

The quantity of the glass bonded zeolite includes the waste from the capture of the chlorine gas released during the electrolysis step (see Figure K-1). The released chlorine is assumed to be captured as iron chloride and incorporated in the salt bearing waste. This increases the waste form mass by approximately 800 kg/MT compared to capturing and re-using the chlorine.

Table 6-3 Advanced Burner Reactor Fuel Reprocessing Off-Gas Waste Summary

		Captured Tritium Grouted				Captured I on Silver Mordenite Grouted			Captured C-14 as Carbonate Grouted			Captured Kr in High Pressure Cylinders			
		Containers: 10 liter poly bottle contained within a double steel box. Each bottle contains 23 kg of cured grout				Containers: 55 gallon drum. Each drum contains 460 kg of cured grout			Containers: 55 gallon drum. Each drum contains 460 kg of cured grout			Containers: Standard Type 1 A high pressure cylinders containing 43.8 liters at 50 atm pressure.			
Burn-up (GWd/MT)	Conversion Ratio	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)
Metal Based Fuel															
99.6 GWd/MT	0.75	2.10	0.09	0.09	0.65	0.00	0.00	0.000	19.62	0.31	0.043	0.93	4.89	0.112	201
132 GWd/MT	0.50	2.10	0.09	0.09	0.82	0.00	0.00	0.000	19.62	0.31	0.043	1.16	6.11	0.139	202
Oxide Based Fuel															
131 GWd/MT	0.75	2.10	0.09	0.09	0.80	43.07	0.69	0.094	9.81	0.16	0.021	1.21	6.38	0.146	188
166 GWd/MT	0.50	2.10	0.09	0.09	0.97	52.45	0.84	0.114	9.81	0.16	0.021	1.45	7.63	0.174	190

Table 6-4 Advanced Burner Reactor Fuel Reprocessing Metal Waste Summary

		Aqueous			Electrochemical			
		Compacted Hulls and Hardware			Metal Alloy			
		Containers: 2 ft diameter x 10 ft tall canisters. Each Canister Contains 3,600 kg.			Containers: 2 ft diameter x 10 ft tall canisters. Each Canister Contains 3,600 kg.			
Burn-up (GWd/MT)	Conversion Ratio	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)
Metal Based Fuel								
99.6 GWd/MT	0.75	n/a	n/a	n/a	4,403	38.41	1.22	3,838
132 GWd/MT	0.50	n/a	n/a	n/a	6,008	52.40	1.67	3,630
Oxide Based Fuel								
131 GWd/MT	0.75	2,890	25.21	0.80	n/a	n/a	n/a	n/a
166 GWd/MT	0.50	3,960	34.54	1.10	n/a	n/a	n/a	n/a

Table 6-5 Advanced Burner Reactor Fuel Reprocessing Fission Product Waste Summary

		Fast Reactor Fission Product Waste Summary											
		Aqueous				Electrochemical							
		Borosilicate Glass				Glass Bonded Zeolite				Lanthanide Glass			
		Containers: 2 ft diameter x 15 ft tall canisters. Each Canister Contains 2,900 kg.				Containers: 2 ft diameter x 15 ft tall canisters. Each Canister Contains 2,900 kg.				Containers: 6in diameter x 60in tall canisters. Each Canister Contains 500 kg.			
Burn-up (GWd/MT)	Conversion Ratio	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)	Mass (kg/MT)	Volume (ft^3/MT)	Containers per MT	Decay Heat (W/container)
Metal Based Fuel													
99.6 GWD/MT	0.75	n/a	n/a	n/a	n/a	2641	42.77	0.91	2900	58.39	0.11	0.12	21175
132 GWD/MT	0.5	n/a	n/a	n/a	n/a	3368	54.53	1.16	3368	73.14	0.14	0.15	21574
Oxide Based Fuel													
131 GWD/MT	0.75	1690.04	27.45	0.58	14000	n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a
166 GWD/MT	0.5	2090.08	33.95	0.72	14000	n/a	n/a	n/a	n/a	n/a	n/a	n/a	n/a

6.1.4 Characteristics of the Heat Generating Wastes from SFR Processes

A consideration of any repository design is the decay heat being generated by the emplaced waste. To assess the potential repository thermal loads, decay heat as a function of time was calculated for the heat generating wastes resulting from the New Extraction and Electro-Chemical reprocessing methods.

Figure 6-2 provides the decay heat of the waste forms resulting from the reprocessing of the advanced burner reactor fuels. The decay heat for a given waste form is essentially identical for either the metal or oxide fuels with varying burn-ups.

Appendix M contains additional data for the decay heat of waste forms resulting from the reprocessing of the advanced burner reactor fuels. Isotopic composition data as a function of time is not reported in Appendix M due to space considerations but is available in electronic format
from the lead author.

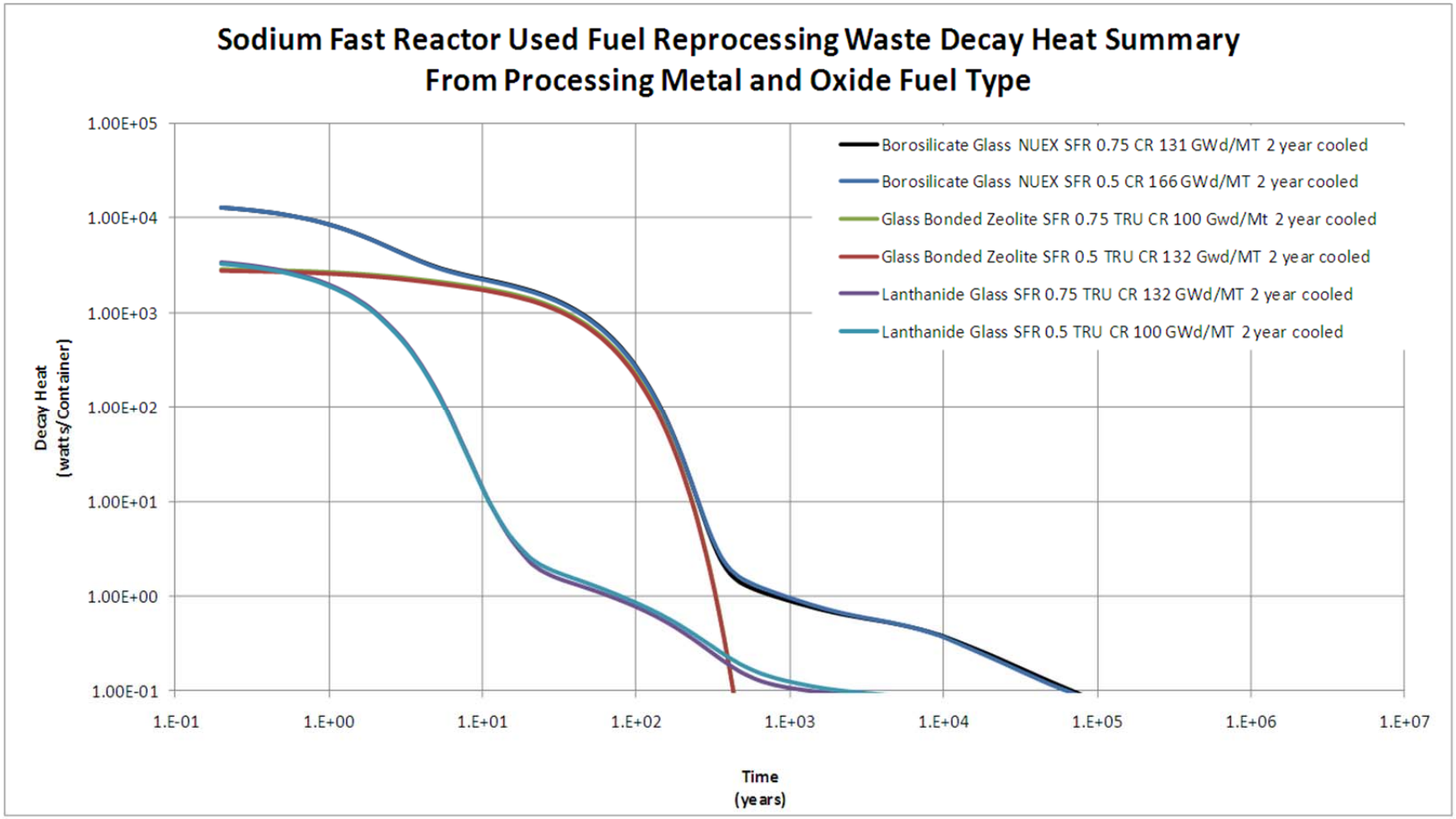


Figure 6-2 Sodium FR Used Fuel Reprocessing Waste Decay Heat Summary

7. ADVANCED ONCE-THROUGH REACTOR POTENTIAL WASTE

The potential waste inventory for advanced once-through reactor concepts have been investigated and documented in reference 25. This study evaluated the waste inventory for a high burn-up (100 GWd/t) light water reactor system and several advanced reactor concepts. The goal of these advanced systems is to achieve long fuel residence time and high uranium utilization. Several examples of these advanced systems include the CANDLE reactor, the sustainable sodium-cooled fast reactor (SSFR) the fast mixed spectrum reactor (FMSR) the ultra-long life fast reactor (ULFR) the Energy Multiplier Module (EM^2) and the travelling wave reactor (TWR).

These new once-through reactor concepts are being proposed to meet various objectives for advanced nuclear fuel cycle studies and by industrial and venture capital groups. The objectives of these systems include long life cores for remote use or foreign export, support proliferation risk reduction, and the effective utilization of fuel resources. To meet these objectives, fast spectrum systems have adopted a design concept quite different from that used in LWRs. A breed and burnup concept with a propagating burn zone has been utilized with low power density and in some cases a multi-batch fuel management scheme.

For comparison purposes, a medium burnup (50 GWd/t) LWR system was used as a reference system. For consistent comparison, most of the fuel cycle parameters have been normalized to the electricity generation in one year (i.e. per GWe-yr).

7.1 Advanced Once-Through Reactor Design and Core Performance Parameters

Table 7-1 provides a summary of the design and core performance parameters that were used in the fuel cycle performance analysis of the once-through nuclear systems. The PWR-50 and PWR-100 cases represent the PWR fuel cycles with fuel discharge burnups of 50 GWd/t and 100 GWd/t, respectively. In Table 7-1 the average uranium enrichment of the feed fuels represents the U-235 mass fraction per total heavy metal mass in the core including the axial and radial blankets. Thus, these values are smaller than the enrichments of the driver fuels that are required for igniting (driving) the reactors. Similarly, the average burnup is the homogenized value over all fuels.

Except for the SSFR and FMSR systems, the once-through fast spectrum systems employ a one-batch fuel management scheme in which whole nuclear fuel assemblies are charged and discharged at the beginning of life (BOL) and end of life (EOL), respectively. Consequently, for consistent comparison to the multi-batch nuclear systems, the annual UNF production rate is evaluated by dividing the heavy metal inventory by the reactor lifetime. It is noted that the TWR is considered a one-batch system although fuel shuffling occurs 15 times; the fuel shuffling is between the active core zone and fixed core zone, and the UNF discharge occurs at EOL. The SSFR and FMSR systems utilize a multi-batch fuel management scheme as in the once-through LWR system, but for these fast spectrum systems, LEU fuel is only required for the initial core because the core is sustained by feeding depleted uranium (DU) fuel in subsequent cycles. For these two fast spectrum systems, the burnup values that are indicated are those for the equilibrium state.

Table 7-1 Summary of Design and Core Performance Parameters of Once-Through Nuclear Systems

	PWR-50	PWR-100	CANDLE	SSFR	FMSR	ULFR	EM^2	TWR
Reactor Power, MW-thermal	3000	3000	3000	3000	3000	3000	500	3000
Reactor Power, MW-electric	1000	1000	1200	1200	1200	1200	238	1135
Thermal efficiency, %	33.3	33.3	40.0	40.0	40.0	40.0	47.6	37.8
Reactor Capacity Factor, %	90	90	90	90	90	90	90	93
Neutron spectrum	thermal	thermal	fast	fast	fast	fast	fast	fast
Fuel form	UOX	UOX	U-Zr	U-Zr	U-Zr	U-Mo	UC	U-Zr
Uranium enrichment, %	4.21	8.5	1.2	[a] 6.2/0.25	[a] 3.8/0.25	4.1	6.1	2.5
Tail uranium enrichment, %	0.25	0.25	0.25	0.25	0.25	0.25	0.35	0.30
Number of batches	3	3	1	34	34	1	1	1
Average burnup, GWd/t	50	100	258	277	257	166	136	93
Specific power density, MW/t	33.7	33.7	3.7	16.9	15.7	9.4	11.8	7.5
Cycle length per batch, year	1.5	3.0	[b] 200.0	1.5	1.5	54.0	37.0	38.0
HM inventory, ton	89.0	89.0	823.7	177.6	191.4	319.6	42.5	399.3
HM charge per batch, ton	29.7	29.7	823.7	5.22	5.6	319.6	42.5	399.2
HM discharge per batch, ton	28.1	26.6	621.1	3.7	4.1	263.4	36.6	360.1
HM fission, ton/year	1.03	1.02	1.03	1.02	1.02	1.04	0.16	1.03

a) First core and charge fuel of equilibrium core

b) Reactor operation time with 8 m core active height.

7.2 Advanced Once-Through UNF Masses and Characteristics

The specific power densities of the once-through fast spectrum systems are significantly derated, implying higher heavy metal inventories than for the LWR system. However, the high burnup with a long fuel residence time for these systems results in lower UNF mass per unit electricity generation (UNF production rate). Figure 7-1 shows the trend of the normalized UNF production rate versus average burnup. Generally, the normalized UNF production rate is inversely proportional to the average burnup; the minor variations were caused by the different thermal efficiency and capacity factor. The PWR-50 system has the highest UNF mass (~20 t/GWe-yr), while the SSFR has the lowest UNF mass (~3t/GWe-yr). The UNF production rate of the TWR is comparable to that of the PWR-100.

Figure 7-1 Comparison of Discharge UNF Masses

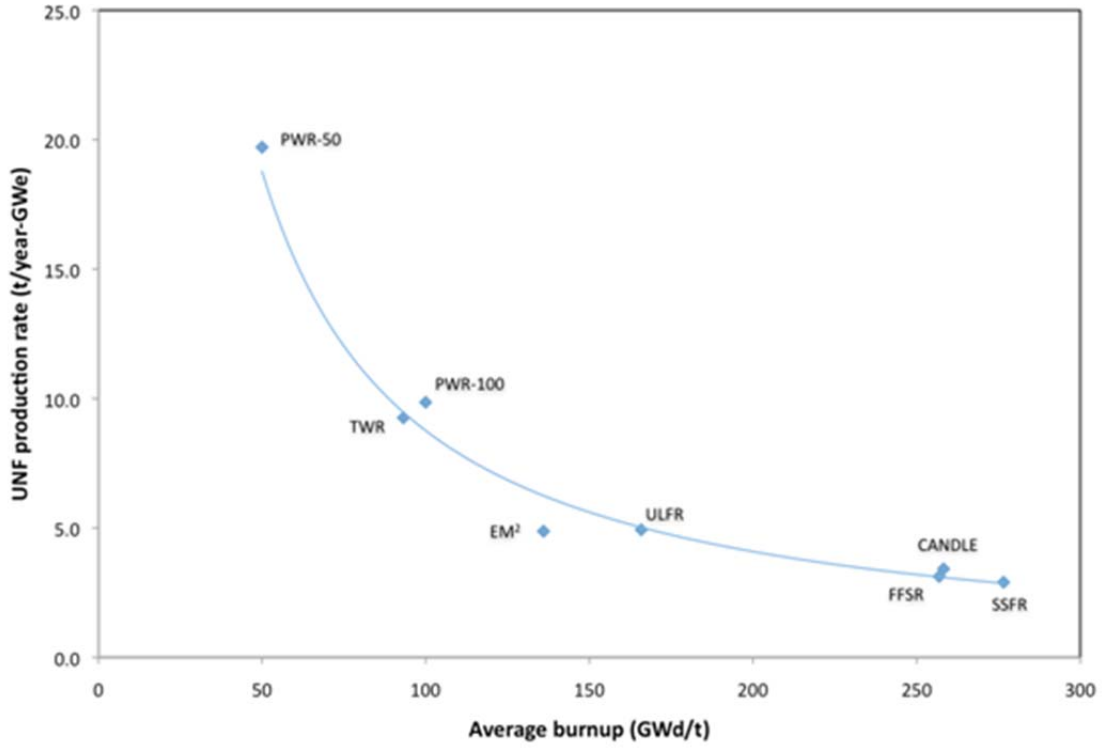


Table 7-2 provides the discharged UNF masses for each of the once through nuclear systems. It is noted that values for multi-batch systems (such as PWR, SSFR, and FMSR) are those for the equilibrium cycle.

Table 7-2 Normalized UNF Production Rates (at Discharge State)

	PWR-50	PWR-100	CANDLE	SSFR	FMSR	ULFR	EM^2	TWR
Normalized UNF production rate (t/GWe-yr)								
Total used nuclear fuel	19.71	9.86	3.42	2.90	3.13	4.93	4.87	9.26
Heavy metal	18.68	8.83	2.58	2.05	2.28	4.06	4.20	8.35
Uranium	18.43	8.64	2.33	1.77	1.98	3.68	3.79	7.81
Plutonium	0.24	0.17	0.24	0.28	0.29	0.37	0.39	0.53
Minor actinides	0.02	0.03	0.01	0.01	0.01	0.01	0.01	0.01
Fission products	1.03	1.02	0.84	0.85	0.85	0.87	0.68	0.91

7.3 Advanced Once-Through UNF Decay Heat

The decay heat values at the discharge state are provided in Table 7-3, including the values of the specific power density and the normalized UNF production rate per unit electricity generation. In this table, two decay heat values are provided: the decay heat per one metric ton of the UNF (MW/t) that was obtained from the ORIGEN-2 calculations and the normalized decay heat per unit electricity generation (MW/GWe-yr).

The leading contributors to the decay heat are provided in Table 7-4 for the discharge state and at 10,000 years after discharge. At the discharge state, very short-lived nuclides such as U-239, Np-239, I-134, etc are the leading contributors. These nuclides quickly saturate during core irradiation and are proportional to the neutron flux level. However, they quickly decay-out following neutron irradiation and discharge. Since the neutron flux level is proportional to the power density, the decay heat level of the UNF at the discharge state is roughly proportional to the power density of each once-through nuclear system.

The normalized decay heat curves of the once-through nuclear systems after 10-year post irradiation cooling are plotted in Figure 7-2.

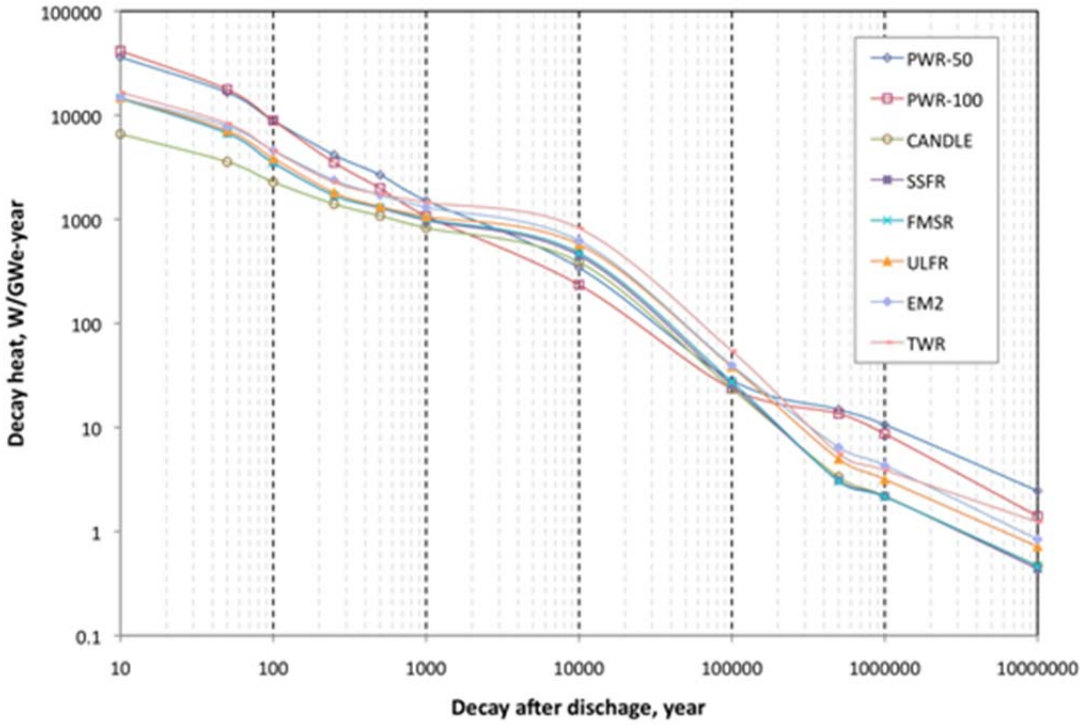
Table 7-3 Comparison of UNF Decay Heat at Discharge

Parameter	PWR-50	PWR-100	CANDLE	SSFR	FMSR	ULFR	EM^2	TWR
Specific power density, MW/t	33.70	33.70	3.66	16.89	15.67	9.39	11.76	7.51
UNF production rate, t/GWe-yr	19.71	9.86	3.42	2.90	3.13	4.93	4.87	9.26
Decay heat per unit UNF mass, MW/t	1.99	2.00	0.24	0.76	0.74	0.63	0.68	0.43
Normalized decay heat per unit electricity generation, MW/GWe-yr	39.14	19.74	0.83	2.20	2.30	3.11	3.32	4.02

Table 7-4 Decay Heat of Leading Contributors (W/t-UNF)

PWR-50		PWR-100		CANDLE		SSFR		FMSR		ULFR		EM^2		TWR	
At Discharge															
Total	$2.0x10^6$	Total	$2.0x10^6$	Total	$0.2x10^6$	Total	$0.8x10^6$	Total	$0.7x10^6$	Total	$0.6x10^6$	Total	$0.7x10^6$	Total	$0.4x10^6$
U239	55,420	U239	50,780	U239	9,199	U239	23,000	U239	22,810	U239	21,890	U239	23,780	U239	15,800
Np239	49,700	Np239	45,540	Np239	8,261	Np239	20,650	Np239	20,480	Np239	19,650	Np239	21,350	Np239	14,190
I134	38,540	I134	38,300	I134	4,434	Tc104	14,140	Tc104	13,740	I134	11,720	I134	12,500	I134	8,239
Cs138	34,500	Cs138	34,340	Tc104	4,408	I134	14,090	I134	13,710	Tc104	11,580	Cs138	11,830	Tc104	8,041
Cs140	33,510	Cs140	33,340	Cs138	4,218	Cs138	13,540	Cs138	13,160	Cs138	11,180	Tc104	11,560	Cs138	7,836
Nb102	31,150	Nb102	31,010	Nb102	4,065	Nb102	12,910	Nb102	12,560	Nb102	10,740	Nb102	11,350	Nb102	7,462
Y 96	30,180	Y 96	29,960	Cs140	3,788	Cs140	12,020	Cs140	11,690	Cs140	10,020	Cs140	10,770	Cs140	7,047
La142	29,550	Tc104	29,300	La142	3,345	La142	10,610	La142	10,330	La142	8,852	Y 96	9,579	La142	6,241
Tc104	29,330	La142	29,020	Y 96	3,291	Y 96	10,390	Y 96	10,120	Y 96	8,730	La142	9,485	Y 96	6,156
La140	27,100	La140	27,970	La140	3,071	La140	9,831	La140	9,519	La140	7,994	La140	8,410	La140	5,472
10,000 Years After Discharge															
Total	17	Total	24	Total	114	Total	154	Total	152	Total	116	Total	128	Total	89
Pu239	9.07	Pu239	11.79	Pu239	76.62	Pu239	101.40	Pu239	103.10	Pu239	88.14	Pu239	89.21	Pu239	69.64
Pu240	7.42	Pu240	10.29	Pu240	35.99	Pu240	52.07	Pu240	48.13	Pu240	27.38	Pu240	37.61	Pu240	18.64
Am243	0.49	Am243	1.22	Am243	0.24	Am243	0.20	Am243	0.16	U234	0.16	Am243	0.20	U234	0.08
Pu242	0.10	U234	0.21	U234	0.14	Sb126m	0.17	Sb126m	0.15	Sb126m	0.10	U234	0.18	Sb126m	0.05
U234	0.09	Pu242	0.18	Sb126m	0.14	U234	0.12	U234	0.11	Am243	0.06	Np237	0.09	Np237	0.04
Np237	0.05	Np237	0.09	Pu242	0.07	Np237	0.08	Np237	0.07	Np237	0.06	Sb126m	0.08	Am243	0.02
Np239	0.04	Np239	0.09	Np237	0.06	Pu242	0.07	Pu242	0.06	Tc 99	0.03	U236	0.03	Tc 99	0.02
Sb126m	0.01	Am241	0.09	Tc 99	0.04	Tc 99	0.05	Tc 99	0.04	Pu242	0.02	Pu242	0.03	U236	0.01
Am241	0.01	Cm245	0.09	Sb126	0.03	Sb126	0.03	Sb126	0.03	Sb126	0.02	Tc 99	0.03	Sb126	0.01
Cm245	0.01	Sb126m	0.03	U236	0.02	U236	0.02	U236	0.02	U236	0.02	Po214	0.02	Pu242	0.01

Figure 7-2 Normalized Decay Heat per Unit Electricity Generation



8. High Temperature Gas Reactor Potential Waste

This section provides estimates for the amount of waste generated by recycling used uranium oxide fuel from a representative prismatic block high temperature gas reactor (HTGR). Because much of the data is notional, the data and discussion here are order of magnitude type estimates.

This section begins with a general description of HTGRs, summarizes the different applications of HTGRs, describes HTGR fuel and core designs, describes the head-end processes unique to HTGR fuel, describes three standard aqueous processes used to separate fuel component streams, presents the resulting waste estimates, and concludes with a summary of trends in the results.

8.1 HTGR Description

HTGRs are a class of thermal spectrum, graphite moderated, helium cooled reactors capable of producing outlet temperatures on the order of 1000 degrees Celsius.[32] The higher operating temperature means that in addition to more efficient production of electricity, HTGRs can be used as a source of process heat for industrial applications, including the production of hydrogen. Because of their robust fuel design, HTGRs are also capable of significantly higher fuel burn-up than LWRs, and can be used to transmute actinides and fission products into forms desirable for disposal.[26] The unique fuel design which allows for this flexibility is also the cause of the HTGR's largest drawback: the large mass of carbon cladding associated with the fuel. For the reference fuel used in this estimate, the mass ratio of carbon and silicon carbide to heavy metal is 17 to 1. For a typical LWR, the mass ratio of Zircaloy to heavy metal ranges from only 0.3 to 1 for a PWR and 0.6 to 1 for a BWR (Table 4-2).

8.2 HTGR Applications

Since the 1960s, several different HTGR applications have been considered. In the early "breed and burn" application, fuel consisted of highly enriched uranium contained in fissile fuel particles and thorium contained in fertile fuel particles. The reactor would burn both U-235 from the fissile particles and U233 generated via neutron capture of thorium in the fertile particles[35]. The main advantage of this concept was the reduced natural uranium consumption when U233 was recovered from the fertile particles. This design was employed in both the Peach Bottom-1 and Fort Saint Vrain reactors [32].

In the more recent "deep burn" application, the HTGR's high burn-up capability is leveraged to destroy transuranic waste and weapons usable material. In this application, driver fuel composed of thermally fissile nuclear material converts non-fissile transmutation fuel into fissile isotopes that are destroyed in the reactor.[37] According to Shropshire and Herring, nearly complete destruction of weapons usable material and 75% destruction of transuranic waste are achievable with a single cycle. To achieve up to 95% destruction of transuranic waste, the reactor-irradiated transmutation fuel can be further irradiated in a second, subcritical system.

The current HTGR application, "power and process heat", is under development in DOE's Next Generation Nuclear Plant (NGNP) project.[27,28] In this application, the HTGR provides power and process heat for industrial applications such as refining or hydrogen generation. Unlike the previous concepts, HTGR fuel in this application consists exclusively of low-enriched uranium. "Power and process heat" is the HTGR application on which the waste estimates in this report are based.

8.3 HTGR Core Designs

HTGR fuel is unique in that the heavy metal is embedded within tristructural-isotropic (TRISO) fuel particles. Each TRISO particle is approximately one millimeter in diameter and consists of a fuel kernel of heavy metal (as an oxide, carbide, or oxycarbide) encapsulated in four successive layers of carbon and silicon carbide. The layer immediately encapsulating the fuel kernel is composed of porous carbon and acts as a buffer to accommodate fuel kernel swelling and to absorb fission product gasses. The second layer is composed of pyrolytic carbon (PyC) and provides structural strength and protects the fuel kernel

from chlorine used during the manufacturing process. The third layer is composed of silicon carbide (SiC) and provides primary containment of fission products and structural integrity. The fourth layer is also composed of pyrolytic carbon and serves as a containment barrier in case the silicon carbide layer is compromised. It also provides a bonding surface to make fuel pebbles or fuel compacts.

There are two generally accepted HTGR core designs, both of which use TRISO fuel particles : prismatic and pebble bed. In the pebble bed design, tens of thousands of TRISO particles are mixed with filler carbon to form spherical fuel pebbles approximately 60 mm in diameter. Thousands of the fuel pebbles are then loaded into a hopper to form the reactor core. The German AVR Juelich reactor was of the pebble bed design[35]..

In the prismatic block design, thousands of TRISO particles are mixed with filler carbon to form small cylindrical fuel compacts. Thousands of fuel compacts are then loaded into machined graphite blocks, in the shape of right hexagonal prisms, to form fuel elements. Hundreds of fuel elements are then assembled to form the reactor core. The prismatic block is currently being pursued by the NGNP and is the design on which the waste estimates in this report were made[27, 28].

8.4 Reprocessing

8.4.1 Head-End Process Description

Review of the literature indicates that once the fuel kernels from used HTGR fuel are exposed they can be recycled using traditional aqueous processes[36]. Exposing the fuel kernels, however, requires several head-end processing steps not necessary in LWR fuel reprocessing. Whereas LWR fuel pellets can be exposed by simply chopping the cladding, for HTGR fuel additional processing steps are necessary to extract the fuel kernel, which is bound successively within the TRISO coating, fuel compact, and fuel element.

The reference head-end process used as the basis for the waste estimates in this report, depicted in Figure 8-1, consists of the following five steps: 1. removing the fuel compacts from the fuel elements; 2. crushing the fuel compacts to facilitate burning; 3. burning the crushed compacts to expose the silicon carbide hulls; 4. secondary crushing to break the silicon carbide hulls; and 5. secondary burning of the inner carbon layers. These steps are based on the head-end process documented by Oak Ridge National Laboratory[26, 35].

Several alternative processes are cited in the literature. Of particular interest is a process in which the carbon cladding is dissolved along with the fuel to be used as the primary constituent in a carbon-based waste form to immobilize fission products[26]. While this process has the potential to significantly reduce the amount of carbon-based waste, it is significantly less mature than the traditional aqueous processes on which this estimate is based.

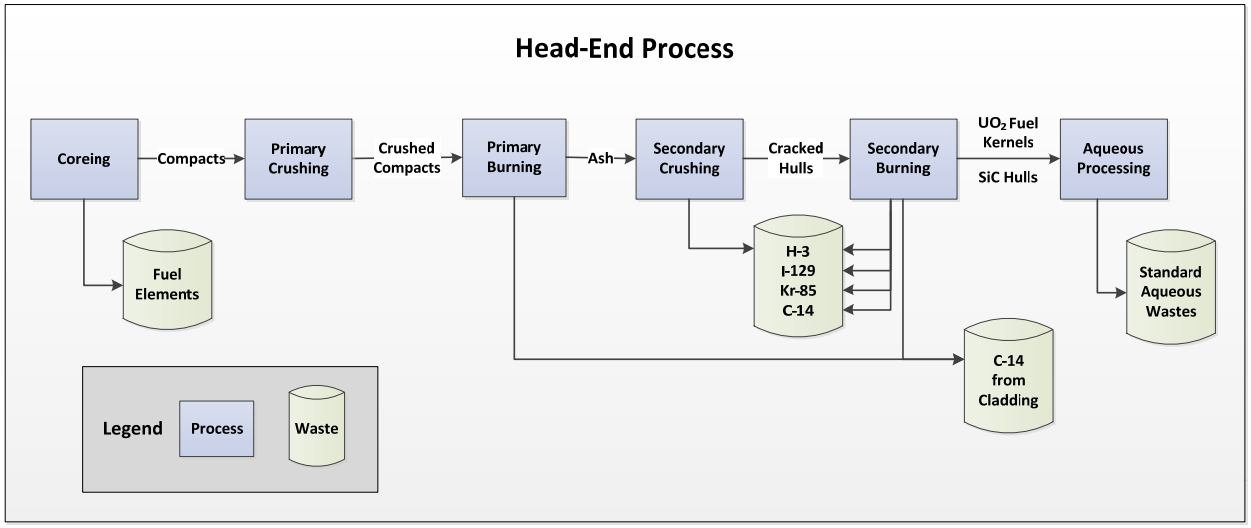


Figure 8-1 Head-End Process

The parameters required for waste estimates of the head-end process are summarized in Appendix N. Tables N-1 and N-3 in Appendix N document parameters at the fuel particle, compact, and element level and contain items including dimensions, density, and counts. While HTGR fuel parameters vary by design, the selected reference parameters are typical of designs published in the literature.

8.4.1.1 Removing Fuel Compacts

In the first step of head-end processing, fuel compacts are extracted from the fuel element by core drilling. This approach is likely to be necessary due to dimensional changes in the graphite under core irradiation conditions[36].. If no dimensional changes occur, it may be possible to simply slice the ends of the fuel channels and push the fuel compact columns out of the fuel element[26]. The extracted fuel compacts proceed to primary crushing and the empty fuel elements are discarded as waste.

Shropshire and Herring indicate that acceptable waste forms may be achieved by over-packing the fuel elements or encapsulating the fuel elements in a protective coating[37]. This estimate assumes an encapsulating treatment is sufficient to dispose of cored fuel elements without further treatment.

If the fuel compacts cannot be separated from the fuel element via either of the methods discussed above, the entire fuel element could be crushed and burned[35]. While this alternative is feasible, whole-element burning substantially increases the amount of carbonate-based waste generated.

8.4.1.2 Primary Crushing

Once the fuel compacts are extracted they are crushed to facilitate burning. All fission products are assumed to be contained within the silicon carbide hull, and the failure rate of the hulls is assumed to be minimal (0.01 to 0.0001% of the particles) [35]. Consequently, no fission products are assumed to be released during primary crushing.

8.4.1.3 Primary Burning

After crushing, the fuel compacts are burned to remove the filler carbon from the compact as well as the outer layer of carbon from the TRISO coating. The exposed silicon carbide hulls are then sent to secondary crushing.

The carbon dioxide (CO2) generated during oxidation of the carbon cladding contains C-14 due to irradiation of nitrogen impurities in the carbon cladding and activation of C-13[36]. The CO2 is scrubbed

from the off-gas by reacting it with calcium hydroxide (CaOH) to produce calcium carbonate (CaCO3). The calcium carbonate is grouted and sent to waste in 55-gallon drums.

8.4.1.4 Secondary Crushing

To penetrate the silicon carbide hull, the fuel particles are crushed in a roll crusher or jet mill to expose the inner carbon layer of the TRISO coating. While the efficiency of secondary crushing is not fully determined, for the purposes of this estimate it is assumed to be 100% efficient. The crushed hulls are sent to secondary burning.

The tritium (H-3) gas released during this step is captured with a molecular sieve in an off-gas system and is grouted and sent to waste. Secondary crushing is assumed to yield the same fraction of tritium as estimated in the LWR chopping step in the aqueous processes described in Section 4.

8.4.1.5 Secondary Burning

In the secondary burning step, the cracked silicon carbide hulls are burned to remove the inner two layers of carbon encapsulating the fuel kernel. The exposed fuel kernels and silicon carbide hulls are then sent on to a nitric acid dissolver for traditional aqueous processing.

Secondary burning is assumed to yield the same fractions of fission product gasses as estimated for the LWR voloxidation step in Section 4. This assumption is based on similar temperatures achieved during head-end burning (875°C) as in voloxidation.

An off-gas treatment system uses a molecular sieve to remove tritiated water, silver zeolite to remove iodine, and the Krypton Absorption by Liquid Carbon Dioxide (KALC) process to remove krypton. Carbon dioxide from the burned carbon cladding is fixated as CaCO3 using a calcium hydroxide scrubber[35]. While carbon dioxide is also released from the fuel itself during this step, it's mass is negligible in comparison.

8.4.2 Aqueous Processes

Once the fuel kernels are exposed, they can be processed using traditional aqueous methods. This analysis estimates the waste generated from each of the three aqueous processes identified in Section 4: Co-Extraction, New Extraction, and Uranium Extraction. Each of these processes is described in detail in Section 4 of the report and is briefly described below.

Co-Extraction is the simplest and most mature aqueous process evaluated. While it is similar to AREVA's COEX process, the two assume different processing methods and cannot be directly compared. In the Co-Extraction process, uranium and plutonium are recovered together. Fission product wastes and technetium are combined into a single borosilicate waste form. Un-dissolved solids, including silicon carbide hulls, are encapsulated and sent to waste as a separate waste form. Fission product gasses containing H-3 and I-129 are captured and sent to waste. Krypton released during dissolution is assumed to be released to the atmosphere.

New Extraction recovers all transuranic elements for transmutation. While it is similar to Energy Solutions' New Extraction process, the two assume different processing methods and cannot be directly compared. In the NUEX process, fission product wastes and technetium are combined into a single borosilicate waste form. Un-dissolved solids, including silicon carbide hulls, are encapsulated and sent to waste as a separate waste form. Fission product gasses containing H-3, I-129, and Kr-85 are captured and sent to waste.

UREX recovers all transuranic elements for transmutation. Fission products cesium, strontium, barium, and rubidium are captured as a solid ceramic waste form, technetium and UDS (except for silicon carbide hulls) are combined as an alloy waste form, which requires the addition of zircoloy and stainless steel, and the remaining fission products are combined into a borosilicate waste form. Silicon carbide hulls are

separated from the rest of the UDS and are encapsulated and sent to waste as a separate waste form. Fission product gasses containing H-3, I-129, and Kr-85 are captured and sent to waste.

8.4.3 Waste Estimation Methodology

Estimates of the head-end specific wastes were calculated using the data provided in Table N-1 to N-3. To calculate the waste masses from each of the aqueous processes, the estimate used the Excel-based flowsheets developed for aqueous reprocessing described in section four. The flowsheets calculate the mass of each waste form based on the elemental distribution of the used fuel and split fractions associated with each processing step.

The isotopic distribution of HTGR fuel was provided by the Systems Analysis campaign and is presented in Table N-4. Note that the data provided represents uncooled used fuel. Cooled isotopic distributions were calculated at 5, 30, 100, and 500 years using the "Isotope Tool". The isotopic distribution of the cooled fuel is presented in Table N-5. Also note that the resulting decayed isotopic distributions include isotopes of elements that are not considered in the aqueous processing flowsheets. As these isotopes represent less than one percent of the total mass, they were not considered in the analysis.

8.4.4 Waste Summary

The amounts of waste generated per metric ton of initial heavy metal (MTIHM) from processes described in Section 8.4.1 and Section 8.4.2 are presented in the following tables. Head-end specific wastes independent of the aqueous process are presented in Table 8-2. These wastes include cored fuel elements and carbon cladding from the TRISO particles and fuel compacts.

Fuel elements are assumed to be an acceptable waste form if treated with a protective coating and are not over-packed. The volume presented represents the external volume of the fuel elements. Carbon dioxide is fixated as calcium carbonate, grouted, and sent to waste in 55-gallon drums. The volume of the grouted waste form reflects the total container volume. The containers were mass constrained, each containing 460 kg of cured grout. A volumetric encapsulation ratio of 30% calcium carbonate to 70% concrete (density = 65.13 kg/ft3) was assumed for these calculations.

Table 8-2 Head-End Wastes

	Head-End Waste				
Waste	**Waste Form**	**Mass (kg/MTIHM)**	**Volume (ft^3/MTIHM)**	**Containers per MTIHM**	**Count per MTIHM**
Fuel Elements	Fuel Elements with Protective Coating	1.25E+04	4.35E+02	-	138
Carbon from TRISO Particle and Compact	Carbon as graphite	4.32E+03	8.03E+01	-	-
	Carbon as $CaCO_3$	3.60E+04	4.69E+02	-	-
	Grouted $CaCO_3$	1.07E+05	1.71E+03	233	-

Gasses containing H-3, I-129, and Kr-85 are captured during both head-end and aqueous processing. The resulting waste streams are presented in Table 8-3, Table 8-4, and Table 8-5. Tritium is grouted and sent to waste in 10 liter poly bottles within double steel boxes. Iodine, captured on silver mordenite, is grouted and sent to waste in 55-gallon drums. Krypton is sent to waste in standard Type 1 A high pressure cylinders.

Borosilicate glass, cesium/strontium ceramic, and metal alloy wastes are presented in Table 8-6, Table 8-7 and Table 8-8, respectively. In Co-extraction and New-Extraction, silicon carbide hulls (698.69 kg per MTIHM) are combined with UDS and encapsulated as a separate waste form. This waste's volume is calculated assuming a volumetric encapsulation ratio of 30% hulls and UDS to 70% grout. In UREX, UDS and technetium are sent to waste as a metal alloy, and silicon carbide hulls are encapsulated and sent to waste as a separate waste form. Zircoloy and stainless steel are assumed to be added to the UREX process to create the metal alloy waste since they are not available from HTGR cladding.

Table 8-10 shows the amount of uranium trioxide recovered via each aqueous process. Note that each process does not generate every waste form.

Table 8-3Tritium Waste

Tritium Waste				
Containers: 10 liter poly bottle contained within a double steel box. Each bottle contains 23 kg of cured grout				
Cooling Time (Yrs)	Aqueous Process	Mass (kg/MTIHM)	Volume (ft^3/ MTIHM)	Containers per MTIHM
5, 30, 100, 500	Co-Extraction, New-Extraction, UREX	2.10E+00	9.14E-02	9.14E-02

Table 8-4 Iodine Waste

Iodine Waste				
Containers: 55 gallon drum. Each drum contains 460 kg of cured grout				
Cooling Time (Yrs)	Aqueous Process	Mass (kg/ MTIHM)	Volume (ft^3/ MTIHM)	Containers per MTIHM
5, 30, 100, 500	Co-Extraction, New-Extraction, UREX	2.42E+01	3.87E-01	5.26E-02

Table 8-5 Krypton Waste

Krypton Waste					
Containers: Standard Type 1 A high pressure cylinders containing 43.8 liters at 50 atm pressure					
Cooling Time	Aqueous Process	Mass (kg/ MTIHM)	Volume (ft^3/ MTIHM)	Containers per MTIHM	Decay Heat (W/container)
5	Co-Extraction	1.13E-01	6.24E-01	1.43E-02	1.21E+02
	New- Extraction, UREX	2.26E+00	1.25E+01	2.85E-01	1.21E+02
30	Co-Extraction	1.11E-01	6.11E-01	1.39E-02	2.46E+01
	New- Extraction, UREX	2.22E+00	1.22E+01	2.79E-01	2.46E+01
100	Co-Extraction	1.10E-01	6.07E-01	1.39E-02	2.68E-01
	New- Extraction, UREX	2.20E+00	1.21E+01	2.77E-01	2.68E-01
500	Co-Extraction	1.10E-01	6.07E-01	1.39E-02	1.57E-12
	New- Extraction, UREX	2.20E+00	1.21E+01	2.77E-01	1.57E-12

Table 8-6 Borosilicate Glass

Borosilicate Glass								
Containers: 2 ft diameter x 15 ft tall canisters. Each Canister Contains 2,900 kg								
Cooling Time	Aqueous Process	Mass (kg/ MTIHM)	Volume (ft^3/ MTIHM)	Containers per MTIHM	Waste Loading (% Oxide)	Noble Metals (% Oxide)	Mo Loading (%MoO_3)	Decay Heat (W/container)
5	Co-Extraction	1.21E+03	1.96E+01	4.17E-01	6.63E+00	1.92E+00	4.43E-02	1.40E+04
	New-Extraction	1.19E+03	1.93E+01	4.10E-01	5.69E+00	2.00E+00	4.39E-02	8.64E+03
	UREX	1.14E+03	1.86E+01	3.94E-01	5.81E+00	2.00E+00	4.39E-02	2.86E+03
30	Co-Extraction	1.21E+03	1.96E+01	4.17E-01	7.16E+00	2.00E+00	4.61E-02	5.77E+03
	New-Extraction	1.19E+03	1.93E+01	4.10E-01	5.64E+00	2.00E+00	4.39E-02	1.96E+03
	UREX	1.14E+03	1.85E+01	3.94E-01	5.90E+00	2.00E+00	4.39E-02	5.59E+01
100	Co-Extraction	1.21E+03	1.96E+01	4.17E-01	7.26E+00	2.00E+00	4.61E-02	2.18E+03
	New-Extraction	1.19E+03	1.93E+01	4.10E-01	5.59E+00	2.00E+00	4.39E-02	3.79E+02
	UREX	1.14E+03	1.85E+01	3.94E-01	5.99E+00	2.00E+00	4.39E-02	4.07E+00
500	Co-Extraction	1.21E+03	1.96E+01	4.17E-01	7.26E+00	2.00E+00	4.61E-02	7.00E+02
	New-Extraction	1.19E+03	1.93E+01	4.10E-01	5.58E+00	2.00E+00	4.39E-02	3.59E-02
	UREX	1.14E+03	1.86E+01	3.94E-01	6.01E+00	2.00E+00	4.39E-02	3.81E-04

Table 8-7 Silicon Carbide Hulls and UDS

Silicon Carbide Hulls and UDS					
Containers: 55 gallon drum. Each Drum Contains 460 kg					
Cooling Time	Aqueous Process	Mass (kg/ MTIHM)	Volume (ft^3/ MTIHM)	Containers per MTIHM	Decay Heat (W/container)
5	Co-Extraction, New-Extraction	1.88E+03	3.01E+01	4.09E+00	3.37E+01
	UREX	1.88E+03	3.01E+01	4.09E+00	3.37E+01
30	Co-Extraction, New-Extraction	1.88E+03	3.00E+01	4.08E+00	3.38E+01
	UREX	1.88E+03	3.01E+01	4.09E+00	2.83E-02
100	Co-Extraction, New-Extraction	1.88E+03	3.01E+01	4.09E+00	2.83E-02
	UREX	1.88E+03	3.00E+01	4.08E+00	2.83E-02
500	Co-Extraction, New-Extraction	1.88E+03	3.01E+01	4.09E+00	1.61E-02
	UREX	1.88E+03	3.01E+01	4.09E+00	1.61E-02

Table 8-8 Cesium/Strontium Ceramic

Cesium/Strontium Ceramic					
Containers: 22cm diameter x 220cm tall canisters. Each Canister Contains 120 kg					
Cooling Time	Aqueous Process	Mass (kg/ MTIHM)	Volume (ft^3/ MTIHM)	Containers per MTIHM	Decay Heat (W/container)
5	Co-Extraction, New-Extraction	-	-	-	-
	UREX	8.40E+01	8.26E+00	7.00E-01	6.35E+03
30	Co-Extraction, New-Extraction	-	-	-	-
	UREX	8.13E+01	8.00E+00	6.77E-01	2.62E+03
100	Co-Extraction, New-Extraction	-	-	-	-
	UREX	7.84E+01	7.71E+00	6.53E-01	5.25E+02
500	Co-Extraction, New-Extraction	-	-	-	-
	UREX	7.77E+01	7.65E+00	6.48E-01	4.52E-02

Table 8-9 Metal Alloy

Metal Alloy					
Containers: 2 ft diameter x 10 ft tall canisters. Each Canister Contains 3,600 kg					
Cooling Time	Aqueous Process	Mass (kg/ MTIHM)	Volume (ft^3/ MTIHM)	Containers per MTIHM	Decay Heat (W/container)
5	Co-Extraction, New-Extraction	-	-	-	-
	UREX	1.45E+01	1.27E-01	4.04E-03	3.42E+04
30	Co-Extraction, New-Extraction	-	-	-	-
	UREX	1.44E+01	1.26E-01	4.00E-03	2.89E+01
100	Co-Extraction, New-Extraction	-	-	-	-
	UREX	1.45E+01	1.26E-01	4.02E-03	1.64E+01
500	Co-Extraction, New-Extraction	-	-	-	-
	UREX	1.45E+01	1.27E-01	4.03E-03	6.92E-01

Table 8-10 Recovered Uranium

Recovered Uranium				
Containers: 55 gallon drum. Each Drum Contains 400 kg				
Cooling Time	Aqueous Process	Mass (kg/ MTIHM)	Volume (ft^3/ MTIHM)	Containers per MTIHM
5	Co-Extraction	8.17E+02	1.81E+01	2.45E+00
	New-Extraction, UREX	8.11E+02	1.79E+01	2.44E+00
30	Co-Extraction	8.23E+02	1.82E+01	2.47E+00
	New Extraction, UREX	8.11E+02	1.79E+01	2.44E+00
100	Co-Extraction	8.26E+02	1.82E+01	2.48E+00
	New Extraction, UREX	8.12E+02	1.80E+01	2.44E+00
500	Co-Extraction	8.28E+02	1.83E+01	2.49E+00
	New Extraction, UREX	8.14E+02	1.80E+01	2.45E+00

8.5 Waste Observations and Trends

- The amount of carbon-based waste from an HTGR is substantially higher than for an LWR.
- The amount of borosilicate glass is greater than that from LWR, but still within an order of magnitude. The distribution of elements in the HTGR waste causes the noble oxide loading to be constraining, resulting in a lower waste oxide loading than in LWR waste.
- Fission product gasses containing H-3, I-129, and Kr-85 are comparable to wastes from an LWR.
- The amount of carbon-based waste from fission product gas is substantially less than from an LWR because the amount of carbon present in used HTGR fuel is approximately five orders of magnitude less.
- Decay heat was the constraining factor for only two waste forms, both generated from fuel cooled only five years: borosilicate glasses produced using the Co-Extraction process, and metal alloy produced using the UREX process. The decay heat limit for waste containers was calculated using the methodology presented in (Appendix H). For borosilicate glass canisters the decay heat limit was 14,000 watts, for cesium/strontium canisters the decay heat limit was 7,390 watts, and for metal alloy canisters the decay heat limit was 34,140 watts.

9. Thorium Based Fuel Cycle Potential Waste

This section proceeds through a general background on the use of thorium in nuclear reactors including different fuel characteristics and management strategies. This is followed by a summary of the previously performed neutron physics calculations and the decay calculations performed here. The section ends with observations on the isotopic trends.

This section presents the isotopic distributions of discharged thorium-TRU fuel that has been recycled up to eight times. The initial neutron sources in these calculations are separated transuranics from advanced light water reactor (ALWR) used fuel. The isotopic distribution at de-fueling is further aged here, to timescales that are relevant to both storage and disposal calculations.

Waste resulting from reprocessing thorium based fuels have not been estimated but, will be considered for future revisions.

9.1 Background

Natural thorium exists almost exclusively as Th-232. This isotope is not fissionable, but is fertile as the capture of a neutron and two beta decays creates fissionable U-233. Because of the fertile nature of Th-232, an initial neutron source is needed to create a chain reaction. The major options include: U-233, U-235, and Pu-239. U-235 can come from used fuel or natural sources, whereas the other two major options must be separated from used fuel. Thorium reactors have long been considered as an option for nuclear power production. In the initial stages of the nuclear age, there were significant concerns about the availability of uranium, and thus considerable attention was focused on the far more abundant thorium to the point that many pilot scale and a few commercial scale thorium reactors were built in the 1960's and 70's, with many of them operating into the 1980's.[40] However, the perceived lack and expense of uranium resources was generally invalid, and the vast majority of reactors currently in use are based on a uranium fuel cycle. Renewed interest in thorium originates with the apparent benefits of thorium over uranium as a fuel choice; these benefits most strongly relate to proliferation and waste disposal concerns.

In terms of proliferation, one of the major apparent benefits of thorium based fuel cycles is that the reactor can act as a plutonium sink. Separated plutonium can be added to the initial fuel matrix as a neutron source and is subsequently consumed in the nuclear reactions. Also, as is shown in detail below, the produced plutonium in a thorium fueled reactor is of lower quality, as the ratio of Pu-239:Pu-242 is smaller than in uranium fueled reactors. Further minimizing the proliferation risks, when a Th-U fuel is used U232 is produced. The half-life of U-232is 73.6 years and several of its daughters emit strong gamma radiation. Thus, the discharged fuel is inherently difficult to process and store as it requires significant shielding and remote handling.

Thorium also has many advantages from a waste perspective. The first advantage is the plutonium sink already mentioned. Plutonium is especially problematic from an ecological exposure perspective as it has fairly long lived alpha emitting isotopes. Besides acting as a plutonium sink, the lack of fertile U-238 in thorium fueled reactors produces much smaller total amounts of transuranics and minor actinides (TRU; Pu, Np, Am, and Cm). However, a thorium based fuel does produce isotopes which may have long term radiological impacts, such as Pa-231, and Th-229. Another advantage of thorium based fuels is that Th-oxides are more stable under a wider range of chemical conditions potentially found at a waste disposal site. In general they are less soluble and less prone to oxidation relative to U-oxides. Despite these advantages, there are several limitations to thorium based fuels as well.

Many of the disadvantages are related to the advantages just mentioned. While the increase in gamma emitters decreases proliferation risks, the added buildup of potential radiation dose from the discharged fuel makes handling and reprocessing more difficult due to increased shielding requirements and remote handling. Also the lack of chemical reactivity decreases negative consequences in waste disposal, but it makes reprocessing more difficult. In the thorium extraction (THOREX) process, a boiling mixture of 13M HNO3+0.05M HF+0.1M Al(NO3)3 and long dissolution periods are required to dissolve the fuel for further separation. And finally, the melting point of ThO2 is significantly higher, meaning that much higher temperatures are required to make thorium based fuels.

9.2 Thorium as a Fuel

Several reactor designs exist for the use of thorium as a nuclear energy source. One is the molten salt reactor covered in the following section of this report as well as a previous DOE report.[39]. Another major design, and the focus of this section, is to simply replace current LWR fuel with a thorium based fuel. Several variations on this theme have been recently considered, including: a once-through thorium fuel cycle to replace non-fissile uranium to extend fuel burnup, the use of thorium as fertile material with fissile plutonium burning, and the combination of thorium with U233 to limit transuranics production.[43,46] For uranium/thorium fuel mixtures, a critical component is how the fissile and fertile materials are combined; there are macro and micro combinations of fertile/fissile materials delineated at the scale of the

fuel rod. At the sub-fuel rod scale, the fissile and fertile material can either be located in a single fuel pellet with a uranium core surrounded by a thorium shell, or there can be distinct uranium and thorium fuel pellets which alternate in the fuel rod. Above the scale of the fuel rod, there are two different designs based on how the fuel assemblies are arranged. The first is the seed and blanket design, where the central fuel rods in an assembly consist of the fissile material, and the outer fuel rods consist of the fertile materials. This fuel assembly is designed to be a one-to-one replacement for the fuel assemblies in current LWRs. The alternative design is to fill whole assemblies with either fertile or fissile material, and then alternate the assemblies when placed in the core[44]. When compared to current LWR, the different uranium/thorium fuel designs produce significantly smaller volumes of total waste, as well as a degradation in weapons related quality and total mass of plutonium. These same fuels also have similar neutronics and thermal safety performance relative to LWRs[45].

When considering a thorium/plutonium/minor actinide (Np, Am, Cm) fuel, the spatial distributions of the fertile/fissile material in the core is generally not considered. Studies of this kind assume a homogenous distribution of the fertile/fissile material in the fuel pellets themselves. One such study found that up to 75% of the initially added plutonium could be destroyed in a single pass through the reactor with a pure Pu/Th fuel. This destruction efficiency declines to 50% when other minor actinides are added.[42] More recently, the plutonium and minor actinide destruction rates have been considered in a multi-recycle scenario.[41]

9.3 Modeling and Calculation Summary

The data for this report was produced in (Reference 41).[41] Raitses et al., perform neutronic calculations as well as present the decay heat and total radioactivity of the waste produced for several different thorium based fuel scenarios. The focus here is the 'self re-cycle' U-TRU-Th Oxide scenario. In this scenario, the initial fuel is Th232, mixed with TRU isotopes (Pu, Am, Np) which have been recovered from re-processing advanced light water reactor (ALWR) fuel. The discharge burnup of the ALWR fuel is 50 GWd/T. It is assumed that the used fuel is cooled for 5-years prior to re-processing, and that it requires two years to separate the TRU from the used fuel, fabricate new fuel and re-fuel the reactor. The initial re-processing is completed through the UREX+4 process. The initial charging isotopes are found in Table 9-1.

Table 9-1 Initial isotope distribution of fuel inserted into the reactor. Adapted from Raitses et al., 2009.

Isotopes	Initial Charge (kg/assembly)	% Total Mass
Th232	3.6731E+02	85.86
Np237	3.0537E+00	0.71
Pu238	1.4538E+00	0.34
Pu239	3.0634E+01	7.16
Pu240	1.4402E+01	3.37
Pu241	6.1535E+00	1.44
Pu242	4.1762E+00	0.98
Am241	6.2080E-01	0.15
Total Pu	5.6819E+01	13.28

The cycle length for this fuel is 18 months, with a capacity factor of 0.9. All 264 fuel rods contain the same fuel matrix. The fuel is subjected to a burnup of 50 GWd/T. After having passed through the reactor once, it is assumed that the fuel is cooled for 5 years then processed (UREX+4/THOREX) to extract the Th, U, Np and Pu with the Am and Cm going to storage. The U, Np and Pu are re-fabricated with Th232 into new fuel, with additional makeup from the ALWR used fuel as needed. This process is repeated seven times. The fuel makeup after each recycle is summarized by Figure 9-1. By the eighth

cycle, the change in k-effectives/k-infinities is still positive but approaching zero implying a 'pseudo-equilibrium'.

The discharge isotopic distribution from reference 41 is presented in Appendix O. The initial discharge values were subsequently decayed for periods of 5, 30, 100, and 500 years, showing the change in the isotopic distribution over the time frame relevant to above ground storage and early geologic disposal. Only isotopes with specific identities were decayed and tracked in the calculations. Isotopes with generic headings (e.g., TM-other (Be, Li, Co-Se, Nb, Ag-Te)) or elements that do not have an isotope distinction (e.g., Rb) were not aged, as the generic classification precludes the ability to assign a half-life to those components. Also, the generic headings cannot be assigned to a decay chain, so in-growth of these isotopes/elements and their daughters was not considered. The output and aging calculations were performed for the end of the first, fourth, and eighth cycles; these results are respectively presented in Appendix O Tables O-1 to O-3. All of the values are in weight percent (mass isotope/mass fuel).

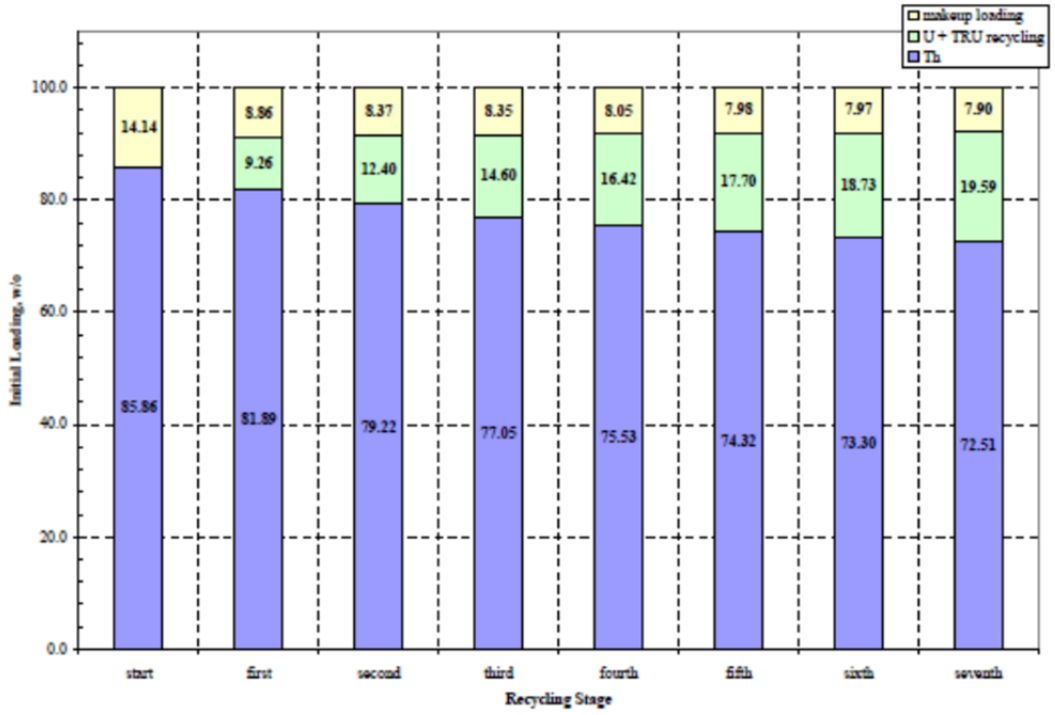


Figure 9-1 Fuel loading by percent for each cycle. From Raitses et al., 2009.

9.4 Isotope Trends and Summary

Previously identified isotopes of long-term ecological importance are variably affected by the multi-recycle scenario. Pa-231 decreases by about 2.5% from the first cycle to the last cycle both at time 0 and at 500 years. Th-229, however, is not produced at time 0, but increases in weight percent at the 500 year time point for increasing number of cycles. The increase from cycle 1 to cycle 4 is 64%, while the increase from cycle 4 to 8 is only 5%. This increase stems directly from the increase in U-233 as the number of cycles increases. Thus in the long term the additional recycling of fuel leads to marginal decreases in one isotope, but large increases in another.

The plutonium distributions in the fuel are shown as a percent in Table 9-2 after having been normalized to the total plutonium content. The major difference between the cycles is that of increased Pu242 and decreased Pu-239. The ratio is nearly 1:1 where one percent decrease in Pu-239 leads to a one percent

increase in Pu-242. This is advantageous from a proliferation perspective as Pu-239 is the weapons grade isotope.

Table 9-2 Plutonium isotope distribution normalized to the total plutonium present as a function of cycles and decay time.

Cycles	Years Decaying:	0	5	30	100	500
1	Pu238	7.9	8.3	7.6	4.8	0.2
	Pu239	25.7	26.7	30.1	32.7	34.8
	Pu240	33.9	35.4	40.8	44.6	45.9
	Pu241	18.5	15.1	5.1	0.2	0.0
	Pu242	14.0	14.5	16.4	17.8	19.1
	Pu244	0.0	0.0	0.0	0.0	0.0
4	Pu238	13.0	13.4	12.2	7.7	0.4
	Pu239	20.9	21.5	23.7	25.9	28.5
	Pu240	33.1	34.4	39.3	43.5	46.1
	Pu241	14.6	11.8	3.9	0.1	0.0
	Pu242	18.3	18.8	20.8	22.7	25.1
	Pu244	0.0	0.0	0.0	0.0	0.0
8	Pu238	14.2	14.4	13.0	8.2	0.4
	Pu239	20.9	21.4	23.5	25.6	28.3
	Pu240	32.9	34.1	38.6	42.6	45.3
	Pu241	12.9	10.4	3.4	0.1	0.0
	Pu242	19.2	19.6	21.5	23.5	26.0
	Pu244	0.0	0.0	0.0	0.0	0.0

The data as presented is limited both by the aforementioned limitations of generic element/isotope classifications as well as by the fact that the isotopes presented do not account for isotopes separated in the UREX+4/THOREX processes. There are waste streams unaccounted for in this analysis. However, the numbers that are shown describe the composition of the spent fuel, and thus can inform both fuel management during surface storage and can also act as a source term for specific waste components in a performance assessment type calculation for long-term geologic disposal.

10. Potential Waste From Molten Salt Reactors

This section summarizes available information useful in estimating reprocessing wastes for Molten Salt Reactors (MSR).

10.1 Description of Molten Salt Reactors

The MSR is a nuclear reactor design characterized by fissile and/or fertile fuel dissolved in molten salt that also serves as the reactor coolant. In a typical design, the molten salt and fuel mixture (fuel salt) flows through a graphite moderated core where nuclear reactions heat the fuel salt. The salt exits the core

to a heat exchanger in which heat is transferred to a secondary salt, which subsequently transfers heat to a power conversion system.
Most MSR designs use a single fluid in which both the fissile and fertile fuel is dissolved. Prior to the development of a method to separate fission products from thorium, two-fluid designs were considered. MSR designs typically utilize the Th-U233 fuel cycle, in which fissile U233 is bred from fertile Th232 However, other fuel cycles, including the U235 fuel cycle, are also possible in an MSR.
Use of a liquid fuel permits design features infeasible with a solid fuel, including on-line fueling and removal of fission products. Additionally, the flexibility afforded by a liquid fuel means MSRs can be designed for electricity generation, providing heat for industrial applications, producing additional fissile fuel, or burning actinides.

The unique design of the MSR provides several benefits compared to conventional light water reactors. Initial excess reactivity is minimized in an MSR as fuel can be added online to compensate for burnup and neutron poisons can be continuously removed. Since there are no fuel assemblies, fuel fabrication is not required, potentially reducing the cost of fueling the reactor. Also, the higher outlet temperatures achievable with a molten salt coolant result in greater thermal efficiencies.

There are several disadvantages of the MSR design. Since there are no assemblies to contain the fuel, the entire primary system is contaminated. This means that some maintenance functions have to be conducted remotely and decommissioning will be more costly. Also, most MSR designs require processing capabilities to remove fission products, which can increase reactor cost and create waste streams that must be dispositioned.

10.2 MSR Experience

10.2.1 Aircraft Reactor Experiment (ARE)

The MSR concept was first proposed by Oak Ridge National Laboratory (ORNL) in the late 1940s as a nuclear propulsion system to allow aircraft to remain airborne for extended periods of time. The MSR was selected for this application because of its much lighter weight than the pressurized water design developed for nuclear submarines, which would have required a substantially heavier pressure vessel. The 2.5 MW(th) reactor constructed for the Aircraft Reactor Experiment (ARE) operated for nine days in 1954, using NaF-ZrF_4-UF_4 as the fuel salt and BeO as a moderator. Although the aircraft propulsion program ended in 1956, a civilian program adapted from the ARE subsequently won the support of the Atomic Energy Commission (AEC).[56]

10.2.2 Molten Salt Reactor Experiment (MSRE)

Due to concerns that the commercial nuclear industry would be constrained by limited uranium resources, the emphasis of the AEC reactor development program shifted toward breeder reactors in the 1960s. While not a breeder reactor itself, the 8-MW(th) Molten Salt Reactor Experiment (MSRE) demonstrated many features of a breeder reactor, including the ability to use ^{233}U as fissile fuel. This was the first use of ^{233}U as a fuel in a reactor. The MSRE also demonstrated that the fluorination process could remove uranium from fuel salt, which is necessary to hold-up protactinium in a breeder reactor. Other studies were conducted with the MSRE, including experiments on xenon removal through helium sparging, fission product deposition within the primary circuit, and tritium migration behavior.[53]

When the MSRE program was terminated, the fuel salt, charged withU233, was drained into two drain tanks and left to solidify. From 2004 to 2008, uranium was recovered from the fuel salt using the same fluorination process used to remove the initial U235 charge, discussed in detail in Section 10.4.2.[47] The de-fueled salt is currently stored in the drain tanks awaiting a disposition decision.

10.3 Reference Reactors

Two MSR designs developed by ORNL will serve as illustrative concepts in this study: the Molten Salt Breeder Reactor (MSBR) and the Denatured Molten Salt Reactor (DMSR). While these are two of the most mature MSR designs available, neither reactor was ever constructed. They are described in greater detail in the following sections.

10.3.1 Molten Salt Breeder Reactor (MSBR)

In the 1970s ORNL proposed the construction of a demonstration breeder reactor, the MSBR. However, as the AEC hadalready committed substantial funding to the development of the liquid metal fast breeder reactor, the MSBR was never constructed.[53]

The MSBR design is a 1000-MW(e) thermal breeder reactor operating on the Th-U-233 fuel cycle. In this fuel cycle, Th-232 is transmuted into Pa-235 via neutron capture, which then decays into fissile U-233. Because of the large neutron cross-section of Pa-233 for thermal neutrons, Pa-233 is temporarily held-up outside of the reactor until it decays to U-235.[50] This is accomplished using fluorination and reductive-extraction processes in an on-line chemical processing plant housed within the reactor facility.[55]

The MSBR uses a single fluid LiF-BeF_2-ThF_4-UF_4 (71.7-16.0-12.0-0.3 mole %) fuel salt. Volatile fission products, primarily xenon and krypton, are removed by sparging the fuel with helium. Rare-earth and alkaline-earth fission products are removed with a metal-transfer process in which bismuth is used to transfer fission products into an acceptor salt. Based on the irradiation tolerance of graphite available at the time, the power density constrained the moderator life to approximately four years.[55]

10.3.2 Denatured Molten Salt Reactor (DMSR)

Because of the push for proliferation-resistant reactors in the late 1970s, ORNL subsequently produced a conceptual design for a denatured molten salt reactor. The DMSR is a 1000-MW(e) thermal converter reactor operating primarily on the Th-U233 fuel cycle, but also deriving fission energy from U-235.[49] The intent of this design was to inhibit the potential for spent MSR fuel to be used in a weapon by denaturing U-235 fuel with U-238.

The DMSR design is similar to the MSBR, except that the on-line processing plant is not required because Pa-233 is allowed to remain in the fuel salt to further inhibit proliferation risk. Because of the reduced neutron economy, the DMSR functions as a converter reactor.

The DMSR uses a single fluid LiF-BeF_2-ThF_4-UF_4 (71.5-16-11-1.5 mole %) fuel salt. The core design was modified such that the neutron flux and power density extend the life expectancy of the graphite moderator to 30 years, the same as the life of the reactor. In addition to a Th-232 charge, the reactor requires an initial charge of approximately 3,450 kg of 20% enriched U-235 to achieve criticality, and routine additions of denatured U-235 over its lifetime to maintain criticality. Unlike the MSBR, only volatile fission products and tritium are removed from the fuel salt. Soluble fission products and actinides are left in the fuel salt.[49]

10.4 MSR Waste

10.4.1 Operation Wastes

By design, MSRs generate fission product waste streams during routine operations. Both the MSBR and DMSR generate waste streams containing xenon, krypton, and tritium. The MSBR generates an additional rare-earth/alkaline-earth waste stream. Each of these wastes is discussed in more detail below. Because of its large thermal neutron absorption cross-section, Xe135 must be removed continuously from the fuel salt in both the MSBR and DMSR. Since it has a low solubility in the fuel salt, xenon (along with krypton) is removed relatively easily by sparging the fuel salt with helium. In this process, helium bubbles are injected into the flowing fuel salt stream. Xenon and krypton migrate from the salt into the bubbles through the physical process of mass transfer, and the bubbles are subsequently removed from the fuel salt. This process is controlled by the surface area of the bubbles and the mass transfer coefficient between the fluid and bubbles.[54] The process removed over 80% of the ^{135}Xe in the MSRE. While not quantified, a more efficient sparging system designed for the MSBR referenced by Engle would improve on this removal efficiency.[49]

Tritium must also be captured during operation of a MSBR and a DMSR, or it will migrate through the heat exchanger and eventually escape to the environment. Along with xenon and krypton, helium sparging captures 18-19% of the total tritium produced. The remaining tritium is expected to migrate through the heat exchanger into the secondary salt. In addition to acting as a heat-transfer fluid, the secondary salt functions to transform tritium into a less mobile, but still volatile, chemical form that can be condensed out of the secondary salt cover-gas. Over 80% of the total tritium produced is captured from this process, resulting in an aggregate tritium recovery rate of 98-99%.[49]

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In the MSBR, rare-earth and alkaline-earth fission products must also be removed. The MSBR uses a metal-transfer process in which the fission products are transported into a lithium chloride carrier salt using bismuth containing lithium and thorium. This process exploits the "relatively large differences in the extent to which rare earths and thorium distribute between bismuth containing a reductant and lithium chloride".[55] The carrier salt is then passed through a series of extractors to remove the fission products. No recovery rate is specified.

10.4.2 Fuel Recovery

In a modified open fuel cycle, the fissile and fertile fuel contained in the fuel salt would be separated and re-used. Uranium recovery capability was demonstrated successfully during the MSRE. However, a review of the literature did not identify a process to recover thorium from fluoride salts. A modified open fuel cycle would likely seek to recover thorium as well as uranium, as it constitutes the majority of the fuel. In the DMSR, the end-of-life estimate of ^{232}Th mass is 92,900 kg, compared to only 3,160 kg of fissile uranium.[49]

Uranium recovery in the MSRE used the fluorination process. In this process, the molten salt is sparged with either pure fluorine or a fluorine-helium mixture to convert UF_4 to gaseous UF_6. The resulting gas stream is passed through a 750°C sodium fluoride (NaF) absorber to retain unwanted volatilized fission products. The stream is then passed through a series of 250°C NaF absorbers to capture the uranium.[52]. As the uranium is in a form not suitable for long-term storage (radiolysis will separate fluorine from UF_6), it is converted to the more stable U_3O_8 using processes standard in the fuel fabrication industry. The primary waste generated from the fluorination process consists of the spent NaF uranium absorbers, which are expected to contain less than 1% of the captured uranium after conversion of UF_6 to U_3O_8.[51]

In 1968, the fluorination process recovered about 216 kg of uranium from the MSRE, leaving only 130 g of uranium in the salt; for a recovery rate of 99%.[51] A similar process was performed from 2005 to 2008 to recover the ^{233}U charge that was left in the fuel when the MSRE was terminated. While not quantified, the process recovered "most of the fuel" from the carrier salt.[47]

10.4.3 Disposition of the Balance of MSR Fuel Salt

Given the processing capabilities discussed above, it can be assumed that xenon, krypton, tritium, and uranium will be removed from DMSR and MSBR fuel salt prior to final disposition. In addition, rare-earth and alkaline-earth fission products will be removed from MSBR fuel salt. However, thorium is assumed to remain in the waste salt.

Disposition of the waste salt is unknown at this time. One of the disposal options being considered for the de-fueled salt from the MSRE is entombment at the Waste Isolation Pilot Plant (WIPP) in New Mexico. While this may be a disposition option for de-fueled MSRE salt, it is not feasible for commercial MSR waste salt as WIPP only accepts defense related waste.[48]

A technical problem associated with the waste salt is that radiolysis, caused by fission products and actinides retained in the salt, generates fluorine gas which will pressurize the waste container. ORNL has conducted research to identify fluorine-trapping materials that can be included in the waste package to prevent pressurizing salt storage containers.[58]

10.5 Next Steps

A more detailed estimate of MSR reprocessing waste will require additional information. First, the estimate of the isotopic distribution of spent MSR fuel should be revised. An isotopic dataset was created by the Systems Analysis campaign, but the results suggest that on-line removal of fission products may not have been accounted for in the analysis. Isotopic distribution of the fuel will be necessary to determine container loading limits based on decay heat. Second, the disposition of thorium must be determined. As thorium constitutes a large fraction of the fuel salt, the amount of fuel salt that must be disposed of will be reduced substantially if thorium is recovered along with uranium. This will require identification or development of a method for thorium removal from fuel salt. Third, the acceptability of de-fueled MSR salt as a waste form for a notional repository must be determined. Concerns over the solubility of the waste salt and its potential to release gas from radiolysis may require it to be disposed of in an alternative waste form. If de-fueled salt is not acceptable, an appropriate waste form must be identified along with processing technologies.

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Appendix A
Commercial Nuclear Fuel Characteristics

Table A-1 Physical characteristics of pressurized water reactor assembly classes.

Assembly Class	Array Size	Manufacturer Code	Version	Assembly Code	Length (in.)	Width (in.)	Clad Material
B&W 15 × 15	15 × 15	B&W	B&W Mark B	B1515B	165.7	8.54	Zircaloy-4
			B&W Mark B10	B1515B10	165.7	8.54	Zircaloy-4
			B&W Mark B3	B1515B3	165.7	8.54	Zircaloy-4
			B&W Mark B4	B1515B4	165.7	8.54	Zircaloy-4
			B&W Mark B4Z	B1515B4Z	165.7	8.54	Zircaloy-4
			B&W Mark B5	B1515B5	165.7	8.54	Zircaloy-4
			B&W Mark B5Z	B1515B5Z	165.7	8.54	Zircaloy-4
			B&W Mark B6	B1515B6	165.7	8.54	Zircaloy-4
			B&W Mark B7	B1515B7	165.7	8.54	Zircaloy-4
			B&W Mark B8	B1515B8	165.7	8.54	Zircaloy-4
			B&W Mark B9	B1515B9	165.7	8.54	Zircaloy-4
			B&W Mark BGD	B1515BGD	165.7	8.54	Zircaloy-4
			B&W Mark BZ	B1515BZ	165.7	8.54	Zircaloy-4
		WE	WE	B1515W	165.7	8.54	not available
B&W 17 × 17	17 × 17	B&W	B&W Mark C	B1717B	165.7	8.54	Zircaloy-4
CE 14 × 14	14 × 14	ANF	ANF	C1414A	157.0	8.10	Zircaloy-4
		CE	CE	C1414C	157.0	8.10	Zircaloy-4
		WE	WE	C1414W	157.0	8.10	Zircaloy-4
CE 16 × 16	16 × 16	CE	CE	C1616CSD	176.8	8.10	Zircaloy-4
CE System 80	16 × 16	CE	CE System 80	C8016C	178.3	8.10	Zircaloy-4
WE 14 × 14	14 × 14	ANF	ANF	W1414A	159.8	7.76	Zircaloy-4
		ANF	ANF Top Rod	W1414ATR	159.8	7.76	Zircaloy-4
		B&W	B&W	W1414B	159.8	7.76	not available
		WE	WE LOPAR	W1414WL	159.8	7.76	Zircaloy-4
		WE	WE OFA	W1414WO	159.8	7.76	Zircaloy-4
		WE	WE Std	W1414W	159.8	7.76	Zircaloy-4
WE 15 × 15	15 × 15	ANF	ANF	W1515A	159.8	8.44	Zircaloy-4
			ANF HT	W1515AHT	159.8	8.44	not available
			ANF Part Length	W1515APL	159.8	8.44	not available
		WE	LOPAR	W1515WL	159.8	8.44	Zircaloy-4
			OFA	W1515WO	159.8	8.44	Zircaloy-4
			WE Std	W1515W	159.8	8.44	Zircaloy
			WE Vantage 5	W1515WV5	159.8	8.44	not available

Assembly Class	Array Size	Manufacturer Code	Version	Assembly Code	Length (in.)	Width (in.)	Clad Material
WE 17 × 17	17 × 17	ANF	ANF	W1717A	159.8	8.44	Zircaloy-4
		B&W	B&W Mark B	W1717B	159.8	8.44	not available
		WE	WE	W1717WRF	159.8	8.44	not available
			WE	W1717WVJ	159.8	8.44	not available
			WE LOPAR	W1717WL	159.8	8.44	Zircaloy-4
			WE OFA	W1717WO	159.8	8.44	Zircaloy-4
			WE Pressurized	W1717WP	159.8	8.44	not available
			WE Vantage	W1717WV	159.8	8.44	not available
			WE Vantage +	W1717WV+	159.8	8.44	ZIRLO
			WE Vantage 5	W1717WV5	159.8	8.44	Zircaloy-4
			WE Vantage 5H	W1717WVH	159.8	8.44	not available
South Texas	17 × 17	WE	WE	WST17W	199.0	8.43	Zircaloy-4
Ft. Calhoun	14 × 14	ANF	ANF	XFC14A	146.0	8.10	not available
		CE	CE	XFC14C	146.0	8.10	Zircaloy-4
		WE	WE	XFC14W	146.0	8.10	not available
Haddam Neck	15 × 15	B&W	B&W SS	XHN15B	137.1	8.42	SS-304
			B&W Zir	XHN15BZ	137.1	8.42	Zircaloy
		GA	Gulf SS	XHN15HS	137.1	8.42	SS
			Gulf Zir	XHN15HZ	137.1	8.42	Zircaloy
		NU	NUM SS	XHN15MS	137.1	8.42	SS
			NUM Zir	XHN15MZ	137.1	8.42	Zircaloy
		WE	WE	XHN15W	137.1	8.42	SS-304
			WE Zir	XHN15WZ	137.1	8.42	not available
Indian Point-1	13 × 14	WE	WE	XIP14W	138.8	6.27	SS
Palisades	15 × 15	ANF	ANF	XPA15A	147.5	8.20	Zircaloy-4
		CE	CE	XPA15C	147.5	8.20	Zircaloy-4
St. Lucie-2	16 × 16	CE	CE	XSL16C	158.2	8.10	Zircaloy-4
San Onofre-1	14 × 14	WE	WE	XSO14W	137.1	7.76	SS-304
			WE D	XSO14WD	137.1	7.76	not available
			WE M	XSO14WM	137.1	7.76	not available
Yankee Rowe	15 × 16	ANF	ANF	XYR16A	111.8	7.62	Zircaloy-4
		CE	CE	XYR16C	111.8	7.62	Zircaloy-4
		UNC	UNC	XYR16U	111.8	7.62	not available
	17 × 18	WE	WE	XYR18W	111.8	7.62	SS
NOTE: Some characteristics of more recently discharged UNF (post-1999) have not yet been provided							

Table A-2 (continued).

TableA-2 Physical characteristics of boiling water reactor assembly classes.

Assembly Class	Array Size	Manufacturer Code	Version	Assembly Code	Length (in.)	Width (in.)	Clad Material
GE BWR/ 2,3	7 × 7	ANF	ANF	G2307A	171.2	5.44	Zircaloy-2
	8 × 8	ANF	ANF	G2308A	171.2	5.44	Zircaloy-2
	9 × 9	ANF	ANF	G2309A	171.2	5.44	Zircaloy-2
			ANF IX	G2309AIX	171.2	5.44	Zircaloy-2
	8 × 8	ANF	ANF Pressurized	G2308AP	171.2	5.44	Zircaloy-2
		GE	GE-10	G2308G10	171.2	5.44	Zircaloy-2
	9 × 9	GE	GE-11	G2309G11	171.2	5.44	Zircaloy-2
	7 × 7	GE	GE-2a	G2307G2A	171.2	5.44	Zircaloy-2
			GE-2b	G2307G2B	171.2	5.44	Zircaloy-2
			GE-3	G2307G3	171.2	5.44	Zircaloy-2
	8 × 8	GE	GE-4	G2308G4	171.2	5.44	Zircaloy-2
			GE-5	G2308G5	171.2	5.44	Zircaloy-2
			GE-7	G2308G7	171.2	5.44	NA
			GE-8a	G2308G8A	171.2	5.44	Zircaloy-2
			GE-8b	G2308G8B	171.2	5.44	Zircaloy-2
			GE-9	G2308G9	171.2	5.44	Zircaloy-2
			GE-Barrier	G2308GB	171.2	5.44	Zircaloy-2
			GE-Pressurized	G2308GP	171.2	5.44	Zircaloy-2
	NA	NA	NA	9X9IXQFA	171.2	5.44	NA
GE BWR/ 4-6	9 × 9	ANF	ANF	G4609A	176.2	5.44	Zircaloy-2
	10 × 10	ANF	ANF	G4610A	176.2	5.44	NA
	9 × 9	ANF	ANF 9-5	G4609A5	176.2	5.44	Zircaloy-2
			ANF 9X	G4609A9X	176.2	5.44	Zircaloy-2
			ANF IX	G4609AIX	176.2	5.44	Zircaloy-2
	10 × 10	ANF	ANF IX	G4610AIX	176.2	5.44	NA
	9 × 9	ANF	ANF X+	G4609AX+	176.2	5.44	NA
	8 × 8	ANF	ANF-Pressurized	G4608AP	176.2	5.44	Zircaloy-2
	NA	AREVA	NA	ATRIUM10	176.2	5.44	Zircaloy-2[a]

Table A-2 (continued).

Assembly Class	Array Size	Manufacturer Code	Version	Assembly Code	Length (in.)	Width (in.)	Clad Material
GE BWR/ 4-6 (Continued)	10 × 10	ABB	CE	G4610C	176.2	5.44	NA
	8 × 8	GE	GE-10	G4608G10	176.2	5.44	Zircaloy-2
			GE-11	G4608G11	176.2	5.44	NA
	9 × 9	GE	GE-11	G4609G11	176.2	5.44	Zircaloy-2
	8 × 8	GE	GE-12	G4608G12	176.2	5.44	NA
	10 × 10	GE	GE-12	G4610G12	176.2	5.44	Zircaloy-2
	9 × 9	GE	GE-13	G4609G13	176.2	5.44	Zircaloy-2
	10 × 10	GE	GE-14	G4610G14	176.2	5.44	NA
	7 × 7	GE	GE-2	G4607G2	176.2	5.44	Zircaloy-2
			GE-3a	G4607G3A	176.2	5.44	Zircaloy-2
			GE-3b	G4607G3B	176.2	5.44	Zircaloy-2
	8 × 8	GE	GE-4a	G4608G4A	176.2	5.44	Zircaloy-2
			GE-4b	G4608G4B	176.2	5.44	Zircaloy-2
			GE-5	G4608G5	176.2	5.44	Zircaloy-2
			GE-8	G4608G8	176.2	5.44	Zircaloy-2
			GE-9	G4608G9	176.2	5.44	Zircaloy-2
			GE-Barrier	G4608GB	176.2	5.44	Zircaloy-2
			GE-Pressurized	G4608GP	176.2	5.44	Zircaloy-2
		WE	WE	G4608W	176.2	5.44	Zircaloy-2
Big Rock Point	9 × 9	ANF	ANF	XBR09A	84	6.52	Zircaloy-2
	11 × 11	ANF	ANF	XBR11A	84	6.52	Zircaloy-2
	7 × 7	GE	GE	XBR07G	84	6.52	NA
	8 × 8	GE	GE	XBR08G	84	6.52	NA
	9 × 9	GE	GE	XBR09G	84	6.52	Zircaloy-2
	11 × 11	GE	GE	XBR11G	84	6.52	Zircaloy-2
		NFS	NFS	XBR11N	84	6.52	NA
Dresden-1	6 × 6	ANF	ANF	XDR06A	134.4	4.28	Zircaloy-2
		GE	GE	XDR06G	134.4	4.28	Zircaloy-2
	7 × 7	GE	GE SA-1	XDR07GS	134.4	4.28	NA
	8 × 8	GE	GE PF Fuels	XDR08G	134.4	4.28	NA
	6 × 6	GE	GE Type III-B	XDR06G3B	134.4	4.28	NA
			GE Type III-F	XDR06G3F	134.4	4.28	NA
			GE Type V	XDR06G5	134.4	4.28	NA
		UNC	UNC	XDR06U	134.4	4.28	NA

Table A-2 (continued).

Assembly Class	Array Size	Manufacturer Code	Version	Assembly Code	Length (in.)	Width (in.)	Clad Material
Humboldt Bay	6 × 6	ANF	6 × 6 ANF	XHB06A	95	4.67	Zircaloy
		GE	GE	XHB06G	95	4.67	Zircaloy-2
	7 × 7	GE	GE Type II	XHB07G2	95	4.67	Zircaloy
LaCrosse	10 × 10	AC	AC	XLC10L	102.5	5.62	SS348H
		ANF	ANF	XLC10A	102.5	5.62	SS348H
NOTE: Some characteristics of more recently discharged UNF (post-1999) have not yet been provided.							

Table A-3 Assembly types and their main characteristics as of December 31, 2002.

Reactor Type	Manufacturer Code	Assembly Code	Initial Uranium Loading (kg/assembly)		Enrichment (U^{235} wt %)			Burn-up (M/MTU)	
			Avg.	Max.	Min.	Avg.	Max.	Avg.	Max.
BWR	not available	9X9IXQFA	170.713	170.800	3.25	3.25	3.25	39,166	39,248
BWR	AC	XLC10L	120.160	121.034	3.63	3.77	3.94	14,419	21,532
BWR	ANF	G2307A	181.574	183.797	2.56	2.64	2.65	24,256	27,826
BWR	ANF	G2308A	174.624	184.355	2.39	2.66	3.13	28,814	36,826
BWR	ANF	G2308AP	172.753	173.132	2.82	2.83	2.83	34,366	34,826
BWR	ANF	G2309A	168.097	169.520	2.78	3.10	3.15	35,941	40,818
BWR	ANF	G2309AIX	169.185	170.059	3.25	3.31	3.82	39,151	43,778
BWR	ANF	G4608AP	176.175	176.800	2.62	2.88	3.40	31,248	35,518
BWR	ANF	G4609A	172.970	174.700	0.72	3.42	3.73	36,933	47,000
BWR	ANF	G4609A5	176.147	177.000	2.90	3.28	3.55	36,536	43,555
BWR	ANF	G4609A9X	169.155	176.800	2.53	2.87	3.11	36,880	43,330
BWR	ANF	G4609AIX	174.788	177.000	3.00	3.58	3.94	24,156	36,777
BWR	ANF	G4609AX+	167.264	167.277	3.13	3.14	3.15	39,239	40,457
BWR	ANF	G4610A	176.900	176.900	3.94	3.94	3.94	38,207	39,000
BWR	ANF	G4610AIX	175.000	175.000	3.39	3.39	3.39	37,706	38,009
BWR	ANF	XBR09A	127.687	131.406	3.45	3.48	3.52	20,981	22,811
BWR	ANF	XBR11A	130.237	133.174	3.13	3.42	3.82	22,716	34,212
BWR	ANF	XDR06A	95.206	95.478	2.23	2.23	2.24	4,907	5,742
BWR	ANF	XHB06A	69.734	73.800	2.35	2.40	2.41	9,037	22,377
BWR	ANF	XLC10A	108.657	109.609	3.68	3.69	3.71	15,017	20,126
BWR	AREVA	ATRIUM10	176.900	176.900	3.94	3.94	3.94	38,406	39,000
BWR	ABB	G4610C	175.683	176.300	2.51	3.29	3.62	38,133	42,640
BWR	GE	G2307G2A	194.902	197.604	2.07	2.10	2.11	16,775	24,902
BWR	GE	G2307G2B	193.203	197.400	1.65	2.15	2.62	16,384	29,728

Table A-3 (continued).

Reactor Type	Manufacturer Code	Assembly Code	Initial Uranium Loading (kg/assembly)		Enrichment (U^{235} wt %)			Burn-up (M/MTU)	
			Avg.	Max.	Min.	Avg.	Max.	Avg.	Max.
BWR	GE	G2307G3	187.419	189.105	1.96	2.41	2.60	25,420	38,861
BWR	GE	G2308G10	172.225	173.512	3.10	3.25	3.56	33,988	43,977
BWR	GE	G2308G4	183.991	185.496	2.19	2.51	2.76	26,087	40,523
BWR	GE	G2308G5	176.971	177.628	2.39	2.66	2.82	29,009	33,597
BWR	GE	G2308G7	178.520	179.400	2.96	2.97	2.99	31,570	35,894
BWR	GE	G2308G8A	175.695	179.584	2.55	3.09	3.40	34,848	44,933
BWR	GE	G2308G8B	172.590	178.000	2.96	3.19	3.39	36,400	42,518
BWR	GE	G2308G9	172.017	173.108	2.85	3.18	3.48	37,268	42,295
BWR	GE	G2308GB	177.983	180.060	2.62	2.80	3.39	32,014	43,381
BWR	GE	G2308GP	177.145	179.200	2.08	2.77	3.01	29,317	38,139
BWR	GE	G2309G11	165.650	169.500	3.10	3.56	3.78	40,522	45,117
BWR	GE	G4607G2	194.729	197.334	1.09	1.56	2.50	9,362	11,829
BWR	GE	G4607G3A	187.455	189.141	1.10	2.33	2.51	21,058	32,188
BWR	GE	G4607G3B	189.925	191.542	1.10	2.31	2.51	21,948	30,831
BWR	GE	G4608G10	177.778	186.094	2.63	3.24	3.70	36,695	44,343
BWR	GE	G4608G11	170.786	171.000	3.38	3.38	3.38	35,194	42,551
BWR	GE	G4608G12	180.873	181.484	3.69	3.71	3.99	32,069	34,462
BWR	GE	G4608G4A	183.931	185.221	2.19	2.62	2.99	24,931	43,430
BWR	GE	G4608G4B	186.709	187.900	2.10	2.31	2.76	21,362	32,941
BWR	GE	G4608G5	183.007	185.366	0.70	2.36	3.01	23,964	38,224
BWR	GE	G4608G8	179.801	185.854	2.95	3.19	3.40	34,905	44,640
BWR	GE	G4608G9	177.738	185.789	1.51	3.23	3.88	36,492	47,062
BWR	GE	G4608GB	184.636	186.653	0.71	2.53	3.25	26,297	45,986
BWR	GE	G4608GP	183.195	186.888	0.70	2.38	3.27	23,112	42,428
BWR	GE	G4609G11	170.123	178.136	1.46	3.56	4.14	40,351	65,149
BWR	GE	G4609G13	171.417	172.912	3.24	3.85	4.17	42,045	53,636

Table A-3 (continued).

Reactor Type	Manufacturer Code	Assembly Code	Initial Uranium Loading (kg/assembly)		Enrichment (U^{235} wt %)			Burn-up (M/MTU)	
			Avg.	Max.	Min.	Avg.	Max.	Avg.	Max.
BWR	GE	G4610G12	176.100	182.141	3.12	3.98	4.20	44,175	52,735
BWR	GE	G4610G14	179.127	180.402	4.01	4.11	4.24	5,868	8,915
BWR	GE	XBR07G	131.500	133.000	2.88	2.88	2.88	1,643	1,690
BWR	GE	XBR08G	112.500	113.000	2.85	2.85	2.85	4,546	7,027
BWR	GE	XBR09G	137.088	141.000	3.51	3.58	3.62	15,092	22,083
BWR	GE	XBR11G	124.500	132.000	3.11	3.46	3.63	22,802	24,997
BWR	GE	XDR06G	111.352	111.352	1.47	1.47	1.47	23,522	23,522
BWR	GE	XDR06G3B	101.610	102.520	1.83	1.83	1.83	18,632	27,106
BWR	GE	XDR06G3F	102.049	102.876	2.25	2.25	2.25	22,132	28,138
BWR	GE	XDR06G5	105.857	112.257	2.26	2.26	2.26	21,095	25,886
BWR	GE	XDR07GS	59.000	59.000	3.10	3.10	3.10	29,000	29,000
BWR	GE	XDR08G	99.714	99.714	1.95	1.95	1.95	25,287	25,287
BWR	GE	XHB06G	76.355	77.000	2.35	2.43	2.52	17,170	22,876
BWR	GE	XHB07G2	76.325	77.100	2.08	2.11	2.31	18,187	20,770
BWR	NFS	XBR11N	128.991	134.414	2.16	2.83	3.51	18,940	21,850
BWR	UNC	XDR06U	102.021	103.441	1.83	2.24	2.26	17,685	26,396
BWR	WE	G4608W	156.696	171.403	2.69	2.85	3.01	28,041	33,140
PWR	ANF	C1414A	380.870	400.000	0.30	3.50	4.32	38,899	50,871
PWR	ANF	W1414A	378.274	406.840	0.71	3.42	4.50	37,500	56,328
PWR	ANF	W1414ATR	362.788	368.011	2.39	3.38	3.57	38,168	46,000
PWR	ANF	W1515A	428.888	434.792	2.01	3.00	3.60	33,344	49,859
PWR	ANF	W1515AHT	434.546	438.074	3.51	4.08	4.59	45,441	56,922
PWR	ANF	W1515APL	307.361	310.073	1.23	1.55	1.88	27,971	37,770
PWR	ANF	W1717A	413.845	460.540	2.43	4.19	4.77	45,291	53,958
PWR	ANF	XFC14A	353.345	358.811	3.50	3.57	3.80	37,205	46,048
PWR	ANF	XPA15A	396.674	408.040	1.50	3.17	4.05	34,362	51,486

Table A-3 (continued).

Reactor Type	Manufacturer Code	Assembly Code	Initial Uranium Loading (kg/assembly)		Enrichment (U^{235} wt %)			Burn-up (M/MTU)	
			Avg.	Max.	Min.	Avg.	Max.	Avg.	Max.
PWR	ANF	XYR16A	233.555	237.300	3.49	3.78	4.02	29,034	35,088
PWR	B&W	B1515B	463.398	465.480	2.74	3.57	3.62	40,407	50,128
PWR	B&W	B1515B10	476.778	489.299	3.24	3.90	4.73	44,417	56,880
PWR	B&W	B1515B3	463.845	465.830	1.08	2.42	2.84	21,036	32,267
PWR	B&W	B1515B4	464.285	474.853	0.90	2.91	4.06	29,534	57,000
PWR	B&W	B1515B4Z	463.735	466.305	3.22	3.84	3.95	39,253	51,660
PWR	B&W	B1515B5	468.250	468.250	3.13	3.13	3.13	38,017	39,000
PWR	B&W	B1515B5Z	464.421	465.176	3.20	3.22	3.23	36,016	42,328
PWR	B&W	B1515B6	462.495	464.403	3.22	3.47	3.66	41,790	49,383
PWR	B&W	B1515B7	463.244	464.513	3.48	3.51	3.55	42,059	48,738
PWR	B&W	B1515B8	464.864	468.560	3.29	3.65	4.01	42,692	54,000
PWR	B&W	B1515B9	463.566	467.566	3.29	3.96	4.76	44,097	53,952
PWR	B&W	B1515BGD	429.552	430.255	3.92	3.92	3.92	49,027	58,310
PWR	B&W	B1515BZ	463.410	466.279	3.05	3.47	4.68	37,441	54,023
PWR	B&W	B1717B	456.722	457.929	2.64	2.84	3.04	29,517	33,904
PWR	B&W	W1414B	383.157	383.157	3.22	3.22	3.22	24,398	24,465
PWR	B&W	W1717B	455.799	466.688	2.00	3.84	4.60	40,741	54,014
PWR	B&W	XHN15B	409.913	415.060	3.00	3.99	4.02	33,776	37,833
PWR	B&W	XHN15BZ	363.921	368.072	3.40	3.80	3.91	34,278	42,956
PWR	CE	C1414C	382.437	408.508	1.03	3.20	4.48	33,597	56,000
PWR	CE	C1616CSD	413.912	442.986	1.87	3.62	4.63	37,916	63,328
PWR	CE	C8016C	421.468	442.000	1.92	3.57	4.27	38,490	56,312
PWR	CE	XFC14C	362.313	376.842	1.39	2.96	3.95	32,130	52,125
PWR	CE	XPA15C	412.442	416.780	1.65	2.47	3.06	16,020	33,630
PWR	CE	XSL16C	381.018	394.400	1.72	3.44	4.28	38,807	54,838
PWR	CE	XYR16C	228.766	233.400	3.51	3.80	3.92	24,282	35,999

Table A-3 (continued).

Reactor Type	Manufacturer Code	Assembly Code	Initial Uranium Loading (kg/assembly)		Enrichment (U^{235} wt %)			Burn-up (M/MTU)	
			Avg.	Max.	Min.	Avg.	Max.	Avg.	Max.
PWR	GA	XHN15HS	406.163	406.163	3.99	3.99	3.99	32,151	32,151
PWR	GA	XHN15HZ	362.863	362.863	3.26	3.26	3.26	18,546	18,546
PWR	NU	XHN15MS	405.979	406.992	3.66	3.66	3.66	28,324	28,324
PWR	NU	XHN15MZ	370.776	371.039	2.95	2.95	2.95	25,643	25,643
PWR	UNC	XYR16U	238.573	241.300	3.96	3.99	4.02	27,461	31,986
PWR	WE	B1515W	461.819	464.763	3.90	4.06	4.22	36,993	49,075
PWR	WE	C1414W	403.483	411.719	2.70	3.15	3.76	30,039	37,781
PWR	WE	W1414W	393.896	403.683	2.26	3.04	3.47	27,315	39,723
PWR	WE	W1414WL	399.092	405.809	2.27	3.07	3.41	31,940	47,932
PWR	WE	W1414WO	355.724	369.265	0.99	3.92	4.95	44,730	69,452
PWR	WE	W1515W	451.193	458.091	2.21	3.00	3.35	29,324	41,806
PWR	WE	W1515WL	455.236	465.600	1.85	2.98	3.80	30,874	55,385
PWR	WE	W1515WO	460.764	465.747	1.91	3.53	4.60	39,071	56,138
PWR	WE	W1515WV5	457.793	462.934	2.99	3.92	4.80	37,556	53,056
PWR	WE	W1717WL	461.323	469.200	1.60	3.12	4.40	32,340	58,417
PWR	WE	W1717WO	425.107	459.433	1.60	3.05	4.02	32,690	53,000
PWR	WE	W1717WP	417.069	417.878	3.73	4.59	4.81	50,707	58,237
PWR	WE	W1717WRF	455.497	456.735	4.00	4.18	4.42	45,530	48,037
PWR	WE	W1717WV	425.399	426.042	4.21	4.38	4.41	44,263	48,385
PWR	WE	W1717WV+	424.010	465.469	1.61	4.16	4.66	45,430	61,685
PWR	WE	W1717WV5	424.269	430.925	1.49	4.01	4.95	43,872	56,570
PWR	WE	W1717WVH	461.954	473.962	2.11	3.87	4.95	41,081	55,496
PWR	WE	W1717WVJ	461.518	465.200	3.71	3.99	4.40	43,922	46,847
PWR	WE	WST17W	540.480	546.600	1.51	3.38	4.41	35,926	54,399
PWR	WE	XFC14W	374.055	376.000	0.27	3.75	4.25	38,521	51,971
PWR	WE	XHN15W	415.557	421.227	3.02	3.59	4.00	27,922	35,196

Table A-3 (continued).

Reactor Type	Manufacturer Code	Assembly Code	Initial Uranium Loading (kg/assembly)		Enrichment (U^{235} wt %)			Burn-up (M/MTU)	
			Avg.	Max.	Min.	Avg.	Max.	Avg.	Max.
PWR	WE	XHN15WZ	384.894	386.689	4.20	4.39	4.60	14,321	19,376
PWR	WE	XIP14W	191.152	200.467	2.83	4.12	4.36	16,471	27,048
PWR	WE	XSO14W	368.153	374.885	3.16	3.87	4.02	27,232	39,275
PWR	WE	XSO14WD	373.323	373.643	4.01	4.01	4.02	18,259	18,424
PWR	WE	XSO14WM	311.225	311.225	0.71	0.71	0.71	19,307	19,636
PWR	WE	XYR18W	273.350	274.100	4.94	4.94	4.94	25,484	31,755

Appendix B
Commercial Used Nuclear Fuel Projections

Table B-1 Scenario 1 – no replacement nuclear generation.

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
1968	0	5	5	0	1	1	N/A	2.87	N/A	1,727
1969	0	96	96	0	10	10	N/A	2.00	N/A	15,267
1970	99	29	128	39	6	45	3.11	2.13	18,422	329
1971	113	413	526	44	65	109	3.32	2.10	23,818	7,819
1972	282	801	1,083	100	146	246	3.36	2.12	22,064	7,172
1973	165	564	729	67	93	161	3.05	2.38	24,159	12,965
1974	575	1,290	1,865	208	242	449	3.02	2.27	18,430	12,900
1975	797	1,223	2,020	322	226	548	2.68	2.19	18,260	16,882
1976	932	1,666	2,598	401	298	700	2.61	1.96	22,411	13,322
1977	1,106	2,047	3,153	466	383	850	2.71	2.06	25,224	16,704
1978	1,665	2,239	3,904	699	384	1,082	2.77	2.21	26,385	19,273
1979	1,656	2,131	3,787	718	400	1,118	2.73	2.32	27,201	22,307
1980	1,456	3,330	4,786	618	620	1,238	2.91	2.37	29,839	22,372
1981	1,626	2,467	4,093	694	459	1,153	2.96	2.47	30,319	23,860
1982	1,491	1,951	3,442	640	357	998	2.93	2.51	29,851	24,616
1983	1,787	2,646	4,433	775	482	1,257	2.96	2.63	30,230	26,659
1984	1,933	2,735	4,668	839	498	1,337	2.93	2.70	29,493	25,740
1985	2,034	2,989	5,023	860	543	1,403	3.05	2.52	31,970	23,450
1986	2,254	2,551	4,805	979	458	1,437	2.97	2.23	30,666	21,237
1987	2,567	3,393	5,960	1,097	611	1,708	3.03	2.38	31,400	21,961
1988	2,583	2,956	5,539	1,098	536	1,633	3.18	2.41	33,724	24,204
1989	2,742	3,803	6,545	1,195	693	1,888	3.18	2.33	32,728	22,505
1990	3,474	3,487	6,961	1,500	633	2,133	3.24	2.43	34,422	25,074
1991	2,810	3,192	6,002	1,222	576	1,798	3.38	2.67	35,446	28,329
1992	3,629	3,808	7,437	1,567	690	2,257	3.49	2.68	36,689	29,247

Table B-1 (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
1993	3,423	3,883	7,306	1,486	700	2,186	3.63	2.83	39,122	30,646
1994	2,800	3,776	6,576	1,199	676	1,874	3.69	3.08	40,222	33,404
1995	3,808	4,425	8,233	1,660	787	2,447	3.72	3.03	40,615	33,066
1996	3,594	4,823	8,417	1,540	856	2,396	3.76	3.11	38,884	35,085
1997	3,532	3,896	7,428	1,563	682	2,244	3.78	3.18	40,331	35,820
1998	2,322	3,880	6,202	1,002	677	1,679	3.96	3.23	43,303	36,419
1999	3,677	4,184	7,861	1,610	723	2,333	4.04	3.25	43,554	37,228
2000	3,154	4,405	7,559	1,385	764	2,149	4.13	3.36	44,968	38,138
2001	3,190	4,585	7,775	1,402	792	2,194	4.19	3.48	45,242	40,335
2002	3,498	3,966	7,464	1,521	688	2,208	4.29	3.67	46,770	40,960
2003	3,504	4,661	8,165	1,547	806	2,353	4.31	3.69	46,350	42,627
2004	3,201	4,105	7,306	1,401	712	2,113	4.33	3.85	46,950	43,447
2005	3,692	4,182	7,874	1,632	736	2,368	4.37	3.94	47,472	44,481
2006	3,536	4,222	7,758	1,564	747	2,311	4.42	4.04	47,999	45,540
2007	2,722	4,708	7,430	1,206	830	2,036	4.47	4.13	48,531	46,624
2008	3,653	4,268	7,921	1,606	768	2,374	4.52	4.23	49,070	47,733
2009	3,631	4,262	7,893	1,594	779	2,373	4.57	4.33	49,615	48,869
2010	2,685	4,720	7,405	1,183	846	2,029	4.62	4.43	50,165	50,032
2011	3,730	4,266	7,996	1,641	766	2,407	4.67	4.54	50,722	51,223
2012	3,554	4,356	7,910	1,559	782	2,341	4.73	4.65	51,285	52,442
2013	2,762	4,706	7,468	1,218	843	2,061	4.78	4.76	51,855	53,690
2014	3,653	4,280	7,933	1,569	768	2,337	4.83	4.87	52,430	54,968
2015	3,631	4,342	7,973	1,631	780	2,411	4.89	4.99	53,012	56,276
2016	2,685	4,720	7,405	1,183	846	2,029	4.94	4.99	53,601	56,276
2017	3,730	4,266	7,996	1,604	765	2,369	4.99	4.99	54,196	56,276
2018	3,554	4,356	7,910	1,596	782	2,378	4.99	4.99	54,196	56,276
2019	2,762	4,706	7,468	1,218	844	2,062	4.99	4.99	54,196	56,276
2020	3,653	4,280	7,933	1,569	768	2,337	4.99	4.99	54,196	56,276

Table B-1 (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2021	3,631	4,342	7,973	1,631	779	2,410	4.99	4.99	54,196	56,276
2022	2,685	4,720	7,405	1,183	846	2,029	4.99	4.99	54,196	56,276
2023	3,730	4,266	7,996	1,604	766	2,370	4.99	4.99	54,196	56,276
2024	3,554	4,356	7,910	1,596	782	2,378	4.99	4.99	54,196	56,276
2025	2,762	4,806	7,568	1,218	843	2,061	4.99	4.99	54,196	56,276
2026	3,653	4,280	7,933	1,569	768	2,337	4.99	4.99	54,196	56,276
2027	3,631	4,342	7,973	1,631	780	2,411	4.99	4.99	54,196	56,276
2028	2,685	4,720	7,405	1,183	846	2,029	4.99	4.99	54,196	56,276
2029	3,805	5,646	9,451	1,632	1,001	2,633	4.99	4.99	54,196	56,276
2030	3,786	4,652	8,438	1,699	833	2,532	4.99	4.99	54,196	56,276
2031	2,809	4,842	7,651	1,237	867	2,104	4.99	4.99	54,196	56,276
2032	3,824	5,856	9,680	1,655	1,035	2,690	4.99	4.99	54,196	56,276
2033	4,417	4,874	9,291	1,983	870	2,853	4.99	4.99	54,196	56,276
2034	3,023	6,820	9,843	1,334	1,217	2,551	4.99	4.99	54,196	56,276
2035	3,252	2,166	5,418	1,405	390	1,795	4.99	4.99	54,196	56,276
2036	3,445	3,236	6,681	1,531	579	2,110	4.99	4.99	54,196	56,276
2037	2,052	2,606	4,658	904	468	1,372	4.99	4.99	54,196	56,276
2038	2,769	2,268	5,037	1,218	407	1,625	4.99	4.99	54,196	56,276
2039	2,108	2,138	4,246	931	385	1,316	4.99	4.99	54,196	56,276
2040	1,856	2,148	4,004	830	384	1,214	4.99	4.99	54,196	56,276
2041	2,717	1,934	4,651	1,127	348	1,475	4.99	4.99	54,196	56,276
2042	2,197	2,388	4,585	959	425	1,384	4.99	4.99	54,196	56,276
2043	1,471	4,136	5,607	567	734	1,301	4.99	4.99	54,196	56,276
2044	2,113	3,376	5,489	917	595	1,512	4.99	4.99	54,196	56,276
2045	1,820	2,208	4,028	822	390	1,212	4.99	4.99	54,196	56,276
2046	1,291	2,136	3,427	578	380	958	4.99	4.99	54,196	56,276
2047	1,403	280	1,683	634	49	683	4.99	4.99	54,196	56,276
2048	361	0	361	177	0	177	4.99	4.99	54,196	56,276

Table B-1 (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2049	285	764	1,049	119	134	253	4.99	4.99	54,196	56,276
2050	361	0	361	152	0	152	4.99	4.99	54,196	56,276
2051	168	0	168	73	0	73	4.99	4.99	54,196	56,276
2052	0	0	0	0	0	0	4.99	4.99	54,196	56,276
2053	269	0	269	114	0	114	4.99	4.99	54,196	56,276
2054	76	0	76	35	0	35	4.99	4.99	54,196	56,276
2055	193	0	193	89	0	89	4.99	4.99	54,196	56,276
	209,314	273,347	482,661	91,375	48,819	140,195				

Table B-2 Scenario 2 – maintain current nuclear generation.

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
1968	0	5	5	0	1	1	N/A	2.87	N/A	1,727
1969	0	96	96	0	10	10	N/A	2.00	N/A	15,267
1970	99	29	128	39	6	45	3.11	2.13	18,422	329
1971	113	413	526	44	65	109	3.32	2.10	23,818	7,819
1972	282	801	1,083	100	146	246	3.36	2.12	22,064	7,172
1973	165	564	729	67	93	161	3.05	2.38	24,159	12,965
1974	575	1,290	1,865	208	242	449	3.02	2.27	18,430	12,900
1975	797	1,223	2,020	322	226	548	2.68	2.19	18,260	16,882
1976	932	1,666	2,598	401	298	700	2.61	1.96	22,411	13,322
1977	1,106	2,047	3,153	466	383	850	2.71	2.06	25,224	16,704
1978	1,665	2,239	3,904	699	384	1,082	2.77	2.21	26,385	19,273
1979	1,656	2,131	3,787	718	400	1,118	2.73	2.32	27,201	22,307
1980	1,456	3,330	4,786	618	620	1,238	2.91	2.37	29,839	22,372
1981	1,626	2,467	4,093	694	459	1,153	2.96	2.47	30,319	23,860
1982	1,491	1,951	3,442	640	357	998	2.93	2.51	29,851	24,616
1983	1,787	2,646	4,433	775	482	1,257	2.96	2.63	30,230	26,659
1984	1,933	2,735	4,668	839	498	1,337	2.93	2.70	29,493	25,740
1985	2,034	2,989	5,023	860	543	1,403	3.05	2.52	31,970	23,450
1986	2,254	2,551	4,805	979	458	1,437	2.97	2.23	30,666	21,237
1987	2,567	3,393	5,960	1,097	611	1,708	3.03	2.38	31,400	21,961
1988	2,583	2,956	5,539	1,098	536	1,633	3.18	2.41	33,724	24,204
1989	2,742	3,803	6,545	1,195	693	1,888	3.18	2.33	32,728	22,505
1990	3,474	3,487	6,961	1,500	633	2,133	3.24	2.43	34,422	25,074
1991	2,810	3,192	6,002	1,222	576	1,798	3.38	2.67	35,446	28,329
1992	3,629	3,808	7,437	1,567	690	2,257	3.49	2.68	36,689	29,247
1993	3,423	3,883	7,306	1,486	700	2,186	3.63	2.83	39,122	30,646
1994	2,800	3,776	6,576	1,199	676	1,874	3.69	3.08	40,222	33,404

Table B-2. (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
1995	3,808	4,425	8,233	1,660	787	2,447	3.72	3.03	40,615	33,066
1996	3,594	4,823	8,417	1,540	856	2,396	3.76	3.11	38,884	35,085
1997	3,532	3,896	7,428	1,563	682	2,244	3.78	3.18	40,331	35,820
1998	2,322	3,880	6,202	1,002	677	1,679	3.96	3.23	43,303	36,419
1999	3,677	4,184	7,861	1,610	723	2,333	4.04	3.25	43,554	37,228
2000	3,154	4,405	7,559	1,385	764	2,149	4.13	3.36	44,968	38,138
2001	3,190	4,585	7,775	1,402	792	2,194	4.19	3.48	45,242	40,335
2002	3,498	3,966	7,464	1,521	688	2,208	4.29	3.67	46,770	40,960
2003	3,504	4,661	8,165	1,547	806	2,353	4.31	3.69	46,350	42,627
2004	3,201	4,105	7,306	1,401	712	2,113	4.33	3.85	46,950	43,447
2005	3,692	4,182	7,874	1,632	736	2,368	4.37	3.94	47,472	44,481
2006	3,536	4,222	7,758	1,564	747	2,311	4.42	4.04	47,999	45,540
2007	2,722	4,708	7,430	1,206	830	2,036	4.47	4.13	48,531	46,624
2008	3,653	4,268	7,921	1,606	768	2,374	4.52	4.23	49,070	47,733
2009	3,631	4,262	7,893	1,594	779	2,373	4.57	4.33	49,615	48,869
2010	2,685	4,720	7,405	1,183	846	2,029	4.62	4.43	50,165	50,032
2011	3,730	4,266	7,996	1,641	766	2,407	4.67	4.54	50,722	51,223
2012	3,554	4,356	7,910	1,559	782	2,341	4.73	4.65	51,285	52,442
2013	2,762	4,706	7,468	1,218	843	2,061	4.78	4.76	51,855	53,690
2014	3,653	4,280	7,933	1,569	768	2,337	4.83	4.87	52,430	54,968
2015	3,631	4,342	7,973	1,631	780	2,411	4.89	4.99	53,012	56,276
2016	2,685	4,720	7,405	1,183	846	2,029	4.94	4.99	53,601	56,276
2017	3,730	4,266	7,996	1,604	765	2,369	4.99	4.99	54,196	56,276
2018	3,554	4,356	7,910	1,596	782	2,378	4.99	4.99	54,196	56,276
2019	2,762	4,706	7,468	1,218	844	2,062	4.99	4.99	54,196	56,276
2020	3,653	4,280	7,933	1,569	768	2,337	4.99	4.99	54,196	56,276
2021	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2022	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276

Table B-2. (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2023	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2024	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2025	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2026	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2027	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2028	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2029	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2030	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2031	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2032	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2033	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2034	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2035	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2036	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2037	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2038	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2039	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2040	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2041	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2042	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2043	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2044	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2045	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2046	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2047	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2048	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2049	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2050	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276

Table B-2. (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2051	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2052	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2053	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2054	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2055	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2056	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2057	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2058	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2059	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2060	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2061	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2062	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2063	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2064	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2065	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2066	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2067	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2068	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2069	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2070	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2071	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2072	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2073	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2074	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2075	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2076	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2077	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2078	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276

Table B-2. (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2079	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2080	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2081	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2082	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2083	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2084	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2085	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2086	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2087	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2088	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2089	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2090	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2091	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2092	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2093	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2094	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2095	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2096	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2097	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2098	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2099	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
2100	3,371	4,428	7,799	1,479	794	2,273	4.99	4.99	54,196	56,276
Total	400,824	527,265	928,089	175,342	94,470	269,813	4.68	4.53	50,579	50,593

Table B-3 Scenario 3 – 200 GWe/yr projected nuclear growth.

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
1968	0	5	5	0	1	1	N/A	2.87	N/A	1,727
1969	0	96	96	0	10	10	N/A	2.00	N/A	15,267
1970	99	29	128	39	6	45	3.11	2.13	18,422	329
1971	113	413	526	44	65	109	3.32	2.10	23,818	7,819
1972	282	801	1,083	100	146	246	3.36	2.12	22,064	7,172
1973	165	564	729	67	93	161	3.05	2.38	24,159	12,965
1974	575	1,290	1,865	208	242	449	3.02	2.27	18,430	12,900
1975	797	1,223	2,020	322	226	548	2.68	2.19	18,260	16,882
1976	932	1,666	2,598	401	298	700	2.61	1.96	22,411	13,322
1977	1,106	2,047	3,153	466	383	850	2.71	2.06	25,224	16,704
1978	1,665	2,239	3,904	699	384	1,082	2.77	2.21	26,385	19,273
1979	1,656	2,131	3,787	718	400	1,118	2.73	2.32	27,201	22,307
1980	1,456	3,330	4,786	618	620	1,238	2.91	2.37	29,839	22,372
1981	1,626	2,467	4,093	694	459	1,153	2.96	2.47	30,319	23,860
1982	1,491	1,951	3,442	640	357	998	2.93	2.51	29,851	24,616
1983	1,787	2,646	4,433	775	482	1,257	2.96	2.63	30,230	26,659
1984	1,933	2,735	4,668	839	498	1,337	2.93	2.70	29,493	25,740
1985	2,034	2,989	5,023	860	543	1,403	3.05	2.52	31,970	23,450
1986	2,254	2,551	4,805	979	458	1,437	2.97	2.23	30,666	21,237
1987	2,567	3,393	5,960	1,097	611	1,708	3.03	2.38	31,400	21,961
1988	2,583	2,956	5,539	1,098	536	1,633	3.18	2.41	33,724	24,204
1989	2,742	3,803	6,545	1,195	693	1,888	3.18	2.33	32,728	22,505
1990	3,474	3,487	6,961	1,500	633	2,133	3.24	2.43	34,422	25,074
1991	2,810	3,192	6,002	1,222	576	1,798	3.38	2.67	35,446	28,329
1992	3,629	3,808	7,437	1,567	690	2,257	3.49	2.68	36,689	29,247
1993	3,423	3,883	7,306	1,486	700	2,186	3.63	2.83	39,122	30,646
1994	2,800	3,776	6,576	1,199	676	1,874	3.69	3.08	40,222	33,404

Table B-3. (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
1995	3,808	4,425	8,233	1,660	787	2,447	3.72	3.03	40,615	33,066
1996	3,594	4,823	8,417	1,540	856	2,396	3.76	3.11	38,884	35,085
1997	3,532	3,896	7,428	1,563	682	2,244	3.78	3.18	40,331	35,820
1998	2,322	3,880	6,202	1,002	677	1,679	3.96	3.23	43,303	36,419
1999	3,677	4,184	7,861	1,610	723	2,333	4.04	3.25	43,554	37,228
2000	3,154	4,405	7,559	1,385	764	2,149	4.13	3.36	44,968	38,138
2001	3,190	4,585	7,775	1,402	792	2,194	4.19	3.48	45,242	40,335
2002	3,498	3,966	7,464	1,521	688	2,208	4.29	3.67	46,770	40,960
2003	3,504	4,661	8,165	1,547	806	2,353	4.31	3.69	46,350	42,627
2004	3,201	4,105	7,306	1,401	712	2,113	4.33	3.85	46,950	43,447
2005	3,692	4,182	7,874	1,632	736	2,368	4.37	3.94	47,472	44,481
2006	3,536	4,222	7,758	1,564	747	2,311	4.42	4.04	47,999	45,540
2007	2,722	4,708	7,430	1,206	830	2,036	4.47	4.13	48,531	46,624
2008	3,653	4,268	7,921	1,606	768	2,374	4.52	4.23	49,070	47,733
2009	3,631	4,262	7,893	1,594	779	2,373	4.57	4.33	49,615	48,869
2010	2,685	4,720	7,405	1,183	846	2,029	4.62	4.43	50,165	50,032
2011	3,730	4,266	7,996	1,641	766	2,407	4.67	4.54	50,722	51,223
2012	3,554	4,356	7,910	1,559	782	2,341	4.73	4.65	51,285	52,442
2013	2,762	4,706	7,468	1,218	843	2,061	4.78	4.76	51,855	53,690
2014	3,653	4,280	7,933	1,569	768	2,337	4.83	4.87	52,430	54,968
2015	3,631	4,342	7,973	1,631	780	2,411	4.89	4.99	53,012	56,276
2016	2,685	4,720	7,405	1,183	846	2,029	4.94	4.99	53,601	56,276
2017	3,730	4,266	7,996	1,604	765	2,369	4.99	4.99	54,196	56,276
2018	3,554	4,356	7,910	1,596	782	2,378	4.99	4.99	54,196	56,276
2019	2,762	4,706	7,468	1,218	844	2,062	4.99	4.99	54,196	56,276
2020	3,653	4,280	7,933	1,569	768	2,337	4.99	4.99	54,196	56,276
2021	3,456	4,529	7985	1,516	814	2,330	4.99	4.99	54,196	56,276

Table B-3. (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2022	3,540	4,640	8180	1,553	834	2,387	4.99	4.99	54,196	56,276
2023	3,624	4,750	8374	1,590	854	2,444	4.99	4.99	54,196	56,276
2024	3,709	4,860	8569	1,627	874	2,500	4.99	4.99	54,196	56,276
2025	3,793	4,971	8764	1,664	894	2,557	4.99	4.99	54,196	56,276
2026	3,877	5,081	8959	1,701	913	2,614	4.99	4.99	54,196	56,276
2027	3,961	5,192	9153	1,738	933	2,671	4.99	4.99	54,196	56,276
2028	4,046	5,302	9348	1,775	953	2,728	4.99	4.99	54,196	56,276
2029	4,130	5,413	9543	1,812	973	2,785	4.99	4.99	54,196	56,276
2030	4,214	5,523	9738	1,849	993	2,841	4.99	4.99	54,196	56,276
2031	4,299	5,634	9932	1,885	1,013	2,898	4.99	4.99	54,196	56,276
2032	4,383	5,744	10127	1,922	1,033	2,955	4.99	4.99	54,196	56,276
2033	4,467	5,855	10322	1,959	1,052	3,012	4.99	4.99	54,196	56,276
2034	4,551	5,965	10517	1,996	1,072	3,069	4.99	4.99	54,196	56,276
2035	4,636	6,076	10711	2,033	1,092	3,126	4.99	4.99	54,196	56,276
2036	4,720	6,186	10906	2,070	1,112	3,182	4.99	4.99	54,196	56,276
2037	4,804	6,297	11101	2,107	1,132	3,239	4.99	4.99	54,196	56,276
2038	4,889	6,407	11296	2,144	1,152	3,296	4.99	4.99	54,196	56,276
2039	4,973	6,517	11490	2,181	1,172	3,353	4.99	4.99	54,196	56,276
2040	5,057	6,628	11685	2,218	1,191	3,410	4.99	4.99	54,196	56,276
2041	5,141	6,738	11880	2,255	1,211	3,466	4.99	4.99	54,196	56,276
2042	5,226	6,849	12075	2,292	1,231	3,523	4.99	4.99	54,196	56,276
2043	5,310	6,959	12269	2,329	1,251	3,580	4.99	4.99	54,196	56,276
2044	5,394	7,070	12464	2,366	1,271	3,637	4.99	4.99	54,196	56,276
2045	5,479	7,180	12659	2,403	1,291	3,694	4.99	4.99	54,196	56,276
2046	5,563	7,291	12854	2,440	1,311	3,751	4.99	4.99	54,196	56,276
2047	5,647	7,401	13048	2,477	1,330	3,807	4.99	4.99	54,196	56,276
2048	5,731	7,512	13243	2,514	1,350	3,864	4.99	4.99	54,196	56,276

Table B-3. (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2049	5,816	7,622	13438	2,551	1,370	3,921	4.99	4.99	54,196	56,276
2050	5,900	7,733	13633	2,588	1,390	3,978	4.99	4.99	54,196	56,276
2051	5,984	7,843	13827	2,625	1,410	4,035	4.99	4.99	54,196	56,276
2052	6,069	7,953	14022	2,662	1,430	4,092	4.99	4.99	54,196	56,276
2053	6,153	8,064	14217	2,699	1,450	4,148	4.99	4.99	54,196	56,276
2054	6,237	8,174	14412	2,736	1,469	4,205	4.99	4.99	54,196	56,276
2055	6,321	8,285	14606	2,773	1,489	4,262	4.99	4.99	54,196	56,276
2056	6,406	8,395	14801	2,810	1,509	4,319	4.99	4.99	54,196	56,276
2057	6,490	8,506	14996	2,847	1,529	4,376	4.99	4.99	54,196	56,276
2058	6,574	8,616	15191	2,884	1,549	4,433	4.99	4.99	54,196	56,276
2059	6,659	8,727	15385	2,921	1,569	4,489	4.99	4.99	54,196	56,276
2060	6,743	8,837	15580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2061	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2062	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2063	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2064	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2065	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2066	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2067	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2068	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2069	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2070	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2071	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2072	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2073	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2074	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2075	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276

Table B-3. (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2076	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2077	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2078	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2079	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2080	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2081	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2082	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2083	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2084	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2085	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2086	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2087	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2088	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2089	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2090	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2091	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2092	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2093	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2094	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2095	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2096	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2097	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2098	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2099	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
2100	6,743	8,837	15,580	2,958	1,589	4,546	4.99	4.99	54,196	56,276
	604,794	793,854	1,398,648	264,810	142,517	407,327				

Table B-4 Scenario 4 400 GWe/yr projected nuclear growth.

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
1968	0	5	5	0	1	1	N/A	2.87	N/A	1,727
1969	0	96	96	0	10	10	N/A	2.00	N/A	15,267
1970	99	29	128	39	6	45	3.11	2.13	18,422	329
1971	113	413	526	44	65	109	3.32	2.10	23,818	7,819
1972	282	801	1,083	100	146	246	3.36	2.12	22,064	7,172
1973	165	564	729	67	93	161	3.05	2.38	24,159	12,965
1974	575	1,290	1,865	208	242	449	3.02	2.27	18,430	12,900
1975	797	1,223	2,020	322	226	548	2.68	2.19	18,260	16,882
1976	932	1,666	2,598	401	298	700	2.61	1.96	22,411	13,322
1977	1,106	2,047	3,153	466	383	850	2.71	2.06	25,224	16,704
1978	1,665	2,239	3,904	699	384	1,082	2.77	2.21	26,385	19,273
1979	1,656	2,131	3,787	718	400	1,118	2.73	2.32	27,201	22,307
1980	1,456	3,330	4,786	618	620	1,238	2.91	2.37	29,839	22,372
1981	1,626	2,467	4,093	694	459	1,153	2.96	2.47	30,319	23,860
1982	1,491	1,951	3,442	640	357	998	2.93	2.51	29,851	24,616
1983	1,787	2,646	4,433	775	482	1,257	2.96	2.63	30,230	26,659
1984	1,933	2,735	4,668	839	498	1,337	2.93	2.70	29,493	25,740
1985	2,034	2,989	5,023	860	543	1,403	3.05	2.52	31,970	23,450
1986	2,254	2,551	4,805	979	458	1,437	2.97	2.23	30,666	21,237
1987	2,567	3,393	5,960	1,097	611	1,708	3.03	2.38	31,400	21,961
1988	2,583	2,956	5,539	1,098	536	1,633	3.18	2.41	33,724	24,204
1989	2,742	3,803	6,545	1,195	693	1,888	3.18	2.33	32,728	22,505
1990	3,474	3,487	6,961	1,500	633	2,133	3.24	2.43	34,422	25,074
1991	2,810	3,192	6,002	1,222	576	1,798	3.38	2.67	35,446	28,329
1992	3,629	3,808	7,437	1,567	690	2,257	3.49	2.68	36,689	29,247
1993	3,423	3,883	7,306	1,486	700	2,186	3.63	2.83	39,122	30,646
1994	2,800	3,776	6,576	1,199	676	1,874	3.69	3.08	40,222	33,404

Table B-4. (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
1995	3,808	4,425	8,233	1,660	787	2,447	3.72	3.03	40,615	33,066
1996	3,594	4,823	8,417	1,540	856	2,396	3.76	3.11	38,884	35,085
1997	3,532	3,896	7,428	1,563	682	2,244	3.78	3.18	40,331	35,820
1998	2,322	3,880	6,202	1,002	677	1,679	3.96	3.23	43,303	36,419
1999	3,677	4,184	7,861	1,610	723	2,333	4.04	3.25	43,554	37,228
2000	3,154	4,405	7,559	1,385	764	2,149	4.13	3.36	44,968	38,138
2001	3,190	4,585	7,775	1,402	792	2,194	4.19	3.48	45,242	40,335
2002	3,498	3,966	7,464	1,521	688	2,208	4.29	3.67	46,770	40,960
2003	3,504	4,661	8,165	1,547	806	2,353	4.31	3.69	46,350	42,627
2004	3,201	4,105	7,306	1,401	712	2,113	4.33	3.85	46,950	43,447
2005	3,692	4,182	7,874	1,632	736	2,368	4.37	3.94	47,472	44,481
2006	3,536	4,222	7,758	1,564	747	2,311	4.42	4.04	47,999	45,540
2007	2,722	4,708	7,430	1,206	830	2,036	4.47	4.13	48,531	46,624
2008	3,653	4,268	7,921	1,606	768	2,374	4.52	4.23	49,070	47,733
2009	3,631	4,262	7,893	1,594	779	2,373	4.57	4.33	49,615	48,869
2010	2,685	4,720	7,405	1,183	846	2,029	4.62	4.43	50,165	50,032
2011	3,730	4,266	7,996	1,641	766	2,407	4.67	4.54	50,722	51,223
2012	3,554	4,356	7,910	1,559	782	2,341	4.73	4.65	51,285	52,442
2013	2,762	4,706	7,468	1,218	843	2,061	4.78	4.76	51,855	53,690
2014	3,653	4,280	7,933	1,569	768	2,337	4.83	4.87	52,430	54,968
2015	3,631	4,342	7,973	1,631	780	2,411	4.89	4.99	53,012	56,276
2016	2,685	4,720	7,405	1,183	846	2,029	4.94	4.99	53,601	56,276
2017	3,730	4,266	7,996	1,604	765	2,369	4.99	4.99	54,196	56,276
2018	3,554	4,356	7,910	1,596	782	2,378	4.99	4.99	54,196	56,276
2019	2,762	4,706	7,468	1,218	844	2,062	4.99	4.99	54,196	56,276
2020	3,653	4,280	7,933	1,569	768	2,337	4.99	4.99	54,196	56,276
2021	3,624	4,750	8374	1,590	854	2,444	4.99	4.99	54,196	56,276
2022	3,877	5,081	8959	1,701	913	2,614	4.99	4.99	54,196	56,276

Table B-4. (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2023	4,130	5,413	9543	1,812	973	2,785	4.99	4.99	54,196	56,276
2024	4,383	5,744	10127	1,922	1,033	2,955	4.99	4.99	54,196	56,276
2025	4,636	6,076	10711	2,033	1,092	3,126	4.99	4.99	54,196	56,276
2026	4,889	6,407	11296	2,144	1,152	3,296	4.99	4.99	54,196	56,276
2027	5,141	6,738	11880	2,255	1,211	3,466	4.99	4.99	54,196	56,276
2028	5,394	7,070	12464	2,366	1,271	3,637	4.99	4.99	54,196	56,276
2029	5,647	7,401	13048	2,477	1,330	3,807	4.99	4.99	54,196	56,276
2030	5,900	7,733	13633	2,588	1,390	3,978	4.99	4.99	54,196	56,276
2031	6,153	8,064	14217	2,699	1,450	4,148	4.99	4.99	54,196	56,276
2032	6,406	8,395	14801	2,810	1,509	4,319	4.99	4.99	54,196	56,276
2033	6,659	8,727	15385	2,921	1,569	4,489	4.99	4.99	54,196	56,276
2034	6,911	9,058	15970	3,032	1,628	4,660	4.99	4.99	54,196	56,276
2035	7,164	9,390	16554	3,142	1,688	4,830	4.99	4.99	54,196	56,276
2036	7,417	9,721	17138	3,253	1,747	5,001	4.99	4.99	54,196	56,276
2037	7,670	10,052	17722	3,364	1,807	5,171	4.99	4.99	54,196	56,276
2038	7,923	10,384	18307	3,475	1,867	5,342	4.99	4.99	54,196	56,276
2039	8,176	10,715	18891	3,586	1,926	5,512	4.99	4.99	54,196	56,276
2040	8,428	11,047	19475	3,697	1,986	5,683	4.99	4.99	54,196	56,276
2041	8,681	11,378	20059	3,808	2,045	5,853	4.99	4.99	54,196	56,276
2042	8,934	11,709	20644	3,919	2,105	6,024	4.99	4.99	54,196	56,276
2043	9,187	12,041	21228	4,030	2,164	6,194	4.99	4.99	54,196	56,276
2044	9,440	12,372	21812	4,141	2,224	6,365	4.99	4.99	54,196	56,276
2045	9,693	12,703	22396	4,252	2,284	6,535	4.99	4.99	54,196	56,276
2046	9,946	13,035	22981	4,362	2,343	6,706	4.99	4.99	54,196	56,276
2047	10,198	13,366	23565	4,473	2,403	6,876	4.99	4.99	54,196	56,276
2048	10,451	13,698	24149	4,584	2,462	7,047	4.99	4.99	54,196	56,276
2049	10,704	14,029	24733	4,695	2,522	7,217	4.99	4.99	54,196	56,276
2050	10,957	14,360	25318	4,806	2,581	7,388	4.99	4.99	54,196	56,276

Table B-4. (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2051	11,210	14,692	25902	4,917	2,641	7,558	4.99	4.99	54,196	56,276
2052	11,463	15,023	26486	5,028	2,701	7,729	4.99	4.99	54,196	56,276
2053	11,716	15,355	27070	5,139	2,760	7,899	4.99	4.99	54,196	56,276
2054	11,968	15,686	27655	5,250	2,820	8,070	4.99	4.99	54,196	56,276
2055	12,221	16,017	28239	5,361	2,879	8,240	4.99	4.99	54,196	56,276
2056	12,474	16,349	28823	5,472	2,939	8,410	4.99	4.99	54,196	56,276
2057	12,727	16,680	29407	5,582	2,998	8,581	4.99	4.99	54,196	56,276
2058	12,980	17,012	29992	5,693	3,058	8,751	4.99	4.99	54,196	56,276
2059	13,233	17,343	30576	5,804	3,118	8,922	4.99	4.99	54,196	56,276
2060	13,486	17,674	31160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2061	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2062	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2063	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2064	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2065	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2066	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2067	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2068	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2069	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2070	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2071	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2072	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2073	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2074	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2075	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2076	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2077	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2078	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276

Table B-4. (continued).

	Number of Assemblies			Total Initial Uranium (MTU)			Average Enrichment		Average Burn-up	
Year	PWR	BWR	Totals	PWR	BWR	Totals	PWR	BWR	PWR	BWR
2079	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2080	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2081	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2082	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2083	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2084	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2085	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2086	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2087	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2088	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2089	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2090	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2091	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2092	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2093	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2094	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2095	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2096	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2097	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2098	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2099	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
2100	13,486	17,674	31,160	5,915	3,177	9,092	4.99	4.99	54,196	56,276
	1,012,733	1,328,505	2,341,238	443,745	238,628	682,372				

Appendix C
Commercial Used Fuel Radionuclide Characteristics

Table C-1 Pressurized water reactor radionuclide inventory at 5 and 30 year decay.

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000		20000	40000	60000
Enrichment (%)	2.44	3.72	4.73		2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM		GM/MTIHM	GM/MTIHM	GM/MTIHM
H 1	3.39E+00	3.44E+00	3.48E+00		3.39E+00	3.44E+00	3.48E+00
H 2	4.05E-03	5.70E-03	7.92E-03		4.05E-03	5.70E-03	7.92E-03
H 3	4.39E-02	7.10E-02	9.54E-02		1.08E-02	1.75E-02	2.34E-02
HE 3	5.87E-03	6.92E-03	7.52E-03		1.90E-02	2.23E-02	2.43E-02
HE 4	1.82E+00	4.36E+00	7.30E+00		2.50E+00	6.34E+00	1.14E+01
LI 6	1.83E-02	9.65E-03	4.08E-03		1.83E-02	9.65E-03	4.08E-03
LI 7	1.08E+00	1.08E+00	1.08E+00		1.08E+00	1.08E+00	1.08E+00
BE 9	5.62E-04	1.13E-03	1.66E-03		5.62E-04	1.13E-03	1.66E-03
BE 10	1.08E-04	2.49E-04	4.14E-04		1.08E-04	2.49E-04	4.14E-04
B 10	1.78E-03	5.25E-04	2.20E-04		1.78E-03	5.25E-04	2.20E-04
B 11	9.59E-01	9.81E-01	1.00E+00		9.59E-01	9.81E-01	1.00E+00
C 12	1.55E+02	1.55E+02	1.55E+02		1.55E+02	1.55E+02	1.55E+02
C 13	6.50E+00	1.12E+01	1.56E+01		6.50E+00	1.12E+01	1.56E+01
C 14	1.99E-01	3.12E-01	4.55E-01		1.98E-01	3.11E-01	4.54E-01
N 14	1.06E+02	1.06E+02	1.06E+02		1.06E+02	1.06E+02	1.06E+02
N 15	4.28E-01	4.33E-01	4.40E-01		4.28E-01	4.33E-01	4.40E-01
O 16	1.34E+05	1.34E+05	1.34E+05		1.34E+05	1.34E+05	1.34E+05
O 17	5.44E+01	5.44E+01	5.44E+01		5.44E+01	5.44E+01	5.44E+01
O 18	3.09E+02	3.09E+02	3.09E+02		3.09E+02	3.09E+02	3.09E+02
F 19	1.07E+01	1.07E+01	1.07E+01		1.07E+01	1.07E+01	1.07E+01
NE 20	2.41E-04	4.08E-04	5.97E-04		2.41E-04	4.08E-04	5.97E-04
NE 21	7.35E-06	1.49E-05	2.19E-05		7.35E-06	1.49E-05	2.19E-05
NE 22	1.10E-05	2.22E-05	3.25E-05		1.10E-05	2.22E-05	3.25E-05

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	**20000**	**40000**	**60000**		**20000**	**40000**	**60000**
Enrichment (%)	**2.44**	**3.72**	**4.73**		**2.44**	**3.72**	**4.73**
ISOTOPE	**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**		**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**
NA 23	**1.50E+01**	**1.50E+01**	**1.50E+01**		**1.50E+01**	**1.50E+01**	**1.50E+01**
MG 24	**1.58E+00**	**1.59E+00**	**1.61E+00**		**1.58E+00**	**1.59E+00**	**1.61E+00**
MG 25	**2.06E-01**	**2.06E-01**	**2.06E-01**		**2.06E-01**	**2.06E-01**	**2.06E-01**
MG 26	**2.36E-01**	**2.36E-01**	**2.36E-01**		**2.36E-01**	**2.36E-01**	**2.36E-01**
AL 27	**1.02E+02**	**1.01E+02**	**1.01E+02**		**1.02E+02**	**1.01E+02**	**1.01E+02**
SI 28	**3.48E+02**	**3.48E+02**	**3.48E+02**		**3.48E+02**	**3.48E+02**	**3.48E+02**
SI 29	**1.83E+01**	**1.83E+01**	**1.83E+01**		**1.83E+01**	**1.83E+01**	**1.83E+01**
SI 30	**1.25E+01**	**1.25E+01**	**1.25E+01**		**1.25E+01**	**1.25E+01**	**1.25E+01**
SI 32	**3.10E-09**	**4.84E-09**	**7.14E-09**		**3.02E-09**	**4.71E-09**	**6.95E-09**
P 31	**1.85E+02**	**1.85E+02**	**1.84E+02**		**1.85E+02**	**1.85E+02**	**1.84E+02**
P 32	**1.87E-13**	**2.91E-13**	**4.30E-13**		**1.82E-13**	**2.84E-13**	**4.19E-13**
S 32	**1.92E+01**	**1.92E+01**	**1.92E+01**		**1.92E+01**	**1.92E+01**	**1.92E+01**
S 33	**1.64E-01**	**1.69E-01**	**1.75E-01**		**1.64E-01**	**1.69E-01**	**1.75E-01**
S 34	**9.03E-01**	**9.03E-01**	**9.03E-01**		**9.03E-01**	**9.03E-01**	**9.03E-01**
S 35	**3.82E-10**	**3.43E-10**	**3.75E-10**		**2.22E-41**	**1.99E-41**	**2.18E-41**
S 36	**3.86E-03**	**3.86E-03**	**3.86E-03**		**3.86E-03**	**3.86E-03**	**3.86E-03**
CL 35	**3.73E+00**	**3.62E+00**	**3.47E+00**		**3.73E+00**	**3.62E+00**	**3.47E+00**
CL 36	**2.35E-01**	**3.52E-01**	**5.01E-01**		**2.35E-01**	**3.52E-01**	**5.01E-01**
CL 37	**1.34E+00**	**1.34E+00**	**1.35E+00**		**1.34E+00**	**1.34E+00**	**1.35E+00**
AR 36	**3.14E-06**	**5.45E-06**	**8.48E-06**		**1.64E-05**	**2.53E-05**	**3.68E-05**
AR 37	**8.75E-22**	**9.78E-22**	**1.08E-21**		**0.00E+00**	**0.00E+00**	**0.00E+00**
AR 38	**8.91E-04**	**1.39E-03**	**2.04E-03**		**8.91E-04**	**1.39E-03**	**2.04E-03**
AR 39	**1.11E-06**	**2.03E-06**	**2.94E-06**		**1.04E-06**	**1.91E-06**	**2.75E-06**
AR 40	**1.55E-06**	**4.58E-06**	**9.59E-06**		**1.55E-06**	**4.58E-06**	**9.59E-06**
K 39	**0.00E+00**	**3.62E-08**	**5.70E-08**		**0.00E+00**	**1.63E-07**	**2.40E-07**
K 40	**3.16E-04**	**6.31E-04**	**9.07E-04**		**3.16E-04**	**6.31E-04**	**9.07E-04**

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000		20000	40000	60000
Enrichment (%)	2.44	3.72	4.73		2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM		GM/MTIHM	GM/MTIHM	GM/MTIHM
K 41	6.98E-06	2.13E-05	4.50E-05		7.22E-06	2.17E-05	4.56E-05
CA 40	1.93E+00	1.93E+00	1.93E+00		1.93E+00	1.93E+00	1.93E+00
CA 41	1.12E-03	1.71E-03	2.51E-03		1.12E-03	1.71E-03	2.51E-03
CA 42	1.35E-02	1.35E-02	1.35E-02		1.35E-02	1.35E-02	1.35E-02
CA 43	2.79E-03	2.79E-03	2.80E-03		2.79E-03	2.79E-03	2.80E-03
CA 44	4.59E-02	4.60E-02	4.61E-02		4.59E-02	4.60E-02	4.61E-02
CA 45	1.04E-08	9.66E-09	1.13E-08		1.41E-25	1.31E-25	1.54E-25
CA 46	9.40E-05	1.09E-04	1.24E-04		9.40E-05	1.09E-04	1.24E-04
CA 48	4.54E-03	4.54E-03	4.53E-03		4.54E-03	4.54E-03	4.53E-03
SC 45	6.98E-05	1.08E-04	1.56E-04		6.98E-05	1.08E-04	1.56E-04
SC 46	1.73E-11	2.03E-11	2.32E-11		2.72E-44	3.20E-44	3.64E-44
TI 46	8.78E+00	8.78E+00	8.77E+00		8.78E+00	8.78E+00	8.77E+00
TI 47	8.10E+00	8.09E+00	8.08E+00		8.10E+00	8.09E+00	8.08E+00
TI 48	8.12E+01	8.09E+01	8.04E+01		8.12E+01	8.09E+01	8.04E+01
TI 49	6.83E+00	7.20E+00	7.68E+00		6.83E+00	7.20E+00	7.68E+00
TI 50	6.04E+00	6.05E+00	6.07E+00		6.04E+00	6.05E+00	6.07E+00
V 50	3.07E-02	4.19E-02	5.10E-02		3.07E-02	4.19E-02	5.10E-02
V 51	1.15E+01	1.32E+01	1.55E+01		1.15E+01	1.32E+01	1.55E+01
CR 50	3.90E+02	3.88E+02	3.86E+02		3.90E+02	3.88E+02	3.86E+02
CR 51	3.44E-21	2.99E-21	3.27E-21		0.00E+00	0.00E+00	0.00E+00
CR 52	7.87E+03	7.87E+03	7.86E+03		7.87E+03	7.87E+03	7.86E+03
CR 53	9.11E+02	9.11E+02	9.12E+02		9.11E+02	9.11E+02	9.12E+02
CR 54	2.40E+02	2.45E+02	2.51E+02		2.40E+02	2.45E+02	2.51E+02
MN 54	8.20E-04	1.08E-03	1.29E-03		1.31E-12	1.72E-12	2.07E-12
MN 55	7.11E+02	7.09E+02	7.07E+02		7.11E+02	7.09E+02	7.08E+02
FE 54	1.49E+03	1.49E+03	1.49E+03		1.49E+03	1.49E+03	1.49E+03

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	**20000**	**40000**	**60000**		**20000**	**40000**	**60000**
Enrichment (%)	**2.44**	**3.72**	**4.73**		**2.44**	**3.72**	**4.73**
ISOTOPE	**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**		**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**
FE 55	**2.81E-01**	**3.60E-01**	**4.69E-01**		**3.59E-04**	**4.59E-04**	**5.97E-04**
FE 56	**2.45E+04**	**2.45E+04**	**2.45E+04**		**2.45E+04**	**2.45E+04**	**2.45E+04**
FE 57	**6.08E+02**	**6.20E+02**	**6.37E+02**		**6.08E+02**	**6.20E+02**	**6.37E+02**
FE 58	**8.17E+01**	**8.31E+01**	**8.44E+01**		**8.17E+01**	**8.31E+01**	**8.44E+01**
FE 59	**2.81E-15**	**2.69E-15**	**3.12E-15**		**0.00E+00**	**0.00E+00**	**0.00E+00**
CO 58	**2.70E-09**	**3.15E-09**	**3.34E-09**		**3.92E-48**	**4.57E-48**	**4.85E-48**
CO 59	**8.69E+01**	**8.34E+01**	**7.97E+01**		**8.69E+01**	**8.34E+01**	**7.97E+01**
CO 60	**3.02E+00**	**4.38E+00**	**5.73E+00**		**1.13E-01**	**1.63E-01**	**2.14E-01**
NI 58	**7.28E+03**	**7.26E+03**	**7.24E+03**		**7.28E+03**	**7.26E+03**	**7.24E+03**
NI 59	**2.51E+01**	**3.69E+01**	**5.09E+01**		**2.51E+01**	**3.69E+01**	**5.09E+01**
NI 60	**2.89E+03**	**2.89E+03**	**2.89E+03**		**2.89E+03**	**2.89E+03**	**2.90E+03**
NI 61	**1.34E+02**	**1.38E+02**	**1.43E+02**		**1.34E+02**	**1.38E+02**	**1.43E+02**
NI 62	**4.06E+02**	**4.03E+02**	**4.00E+02**		**4.06E+02**	**4.03E+02**	**4.00E+02**
NI 63	**4.83E+00**	**7.27E+00**	**1.04E+01**		**4.00E+00**	**6.02E+00**	**8.63E+00**
NI 64	**1.07E+02**	**1.07E+02**	**1.08E+02**		**1.07E+02**	**1.07E+02**	**1.08E+02**
CU 63	**1.33E+01**	**1.34E+01**	**1.36E+01**		**1.41E+01**	**1.47E+01**	**1.54E+01**
CU 65	**6.21E+00**	**6.30E+00**	**6.42E+00**		**6.21E+00**	**6.30E+00**	**6.42E+00**
ZN 64	**1.92E+01**	**1.92E+01**	**1.92E+01**		**1.92E+01**	**1.92E+01**	**1.92E+01**
ZN 65	**7.43E-05**	**7.59E-05**	**9.04E-05**		**3.97E-16**	**4.06E-16**	**4.84E-16**
ZN 66	**1.13E+01**	**1.13E+01**	**1.13E+01**		**1.13E+01**	**1.13E+01**	**1.13E+01**
ZN 67	**1.68E+00**	**1.67E+00**	**1.66E+00**		**1.68E+00**	**1.67E+00**	**1.66E+00**
ZN 68	**7.87E+00**	**7.88E+00**	**7.88E+00**		**7.87E+00**	**7.88E+00**	**7.88E+00**
ZN 70	**2.70E-01**	**2.73E-01**	**2.76E-01**		**2.70E-01**	**2.73E-01**	**2.76E-01**
GA 69	**2.04E-02**	**3.46E-02**	**5.05E-02**		**2.04E-02**	**3.46E-02**	**5.05E-02**
GA 71	**3.57E-05**	**5.53E-05**	**8.20E-05**		**3.57E-05**	**5.53E-05**	**8.20E-05**
GE 70	**8.22E-05**	**2.53E-04**	**5.40E-04**		**8.22E-05**	**2.53E-04**	**5.40E-04**

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	**20000**	**40000**	**60000**		**20000**	**40000**	**60000**
Enrichment (%)	**2.44**	**3.72**	**4.73**		**2.44**	**3.72**	**4.73**
ISOTOPE	**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**		**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**
GE 72	**1.32E-02**	**2.64E-02**	**3.89E-02**		**1.32E-02**	**2.64E-02**	**3.89E-02**
GE 73	**2.77E-02**	**5.39E-02**	**7.75E-02**		**2.77E-02**	**5.39E-02**	**7.75E-02**
GE 74	**5.89E-02**	**1.18E-01**	**1.76E-01**		**5.89E-02**	**1.18E-01**	**1.76E-01**
GE 76	**3.06E-01**	**6.01E-01**	**8.86E-01**		**3.06E-01**	**6.01E-01**	**8.86E-01**
AS 75	**1.24E-01**	**2.41E-01**	**3.49E-01**		**1.24E-01**	**2.41E-01**	**3.49E-01**
SE 76	**2.28E-03**	**8.48E-03**	**1.81E-02**		**2.28E-03**	**8.48E-03**	**1.81E-02**
SE 77	**6.40E-01**	**1.23E+00**	**1.76E+00**		**6.40E-01**	**1.23E+00**	**1.76E+00**
SE 78	**1.46E+00**	**2.92E+00**	**4.39E+00**		**1.46E+00**	**2.92E+00**	**4.39E+00**
SE 79	**3.60E+00**	**7.08E+00**	**1.05E+01**		**3.60E+00**	**7.08E+00**	**1.05E+01**
SE 80	**8.18E+00**	**1.61E+01**	**2.39E+01**		**8.18E+00**	**1.61E+01**	**2.39E+01**
SE 82	**2.07E+01**	**4.04E+01**	**5.98E+01**		**2.07E+01**	**4.04E+01**	**5.98E+01**
BR 79	**2.33E-04**	**5.32E-04**	**8.67E-04**		**1.19E-03**	**2.42E-03**	**3.66E-03**
BR 81	**1.35E+01**	**2.58E+01**	**3.72E+01**		**1.35E+01**	**2.58E+01**	**3.72E+01**
KR 80	**1.34E-04**	**2.75E-04**	**4.25E-04**		**1.34E-04**	**2.75E-04**	**4.25E-04**
KR 81	**1.08E-05**	**2.94E-05**	**5.85E-05**		**1.08E-05**	**2.94E-05**	**5.85E-05**
KR 82	**4.19E-01**	**1.42E+00**	**2.92E+00**		**4.19E-01**	**1.42E+00**	**2.92E+00**
KR 83	**2.79E+01**	**4.88E+01**	**6.21E+01**		**2.79E+01**	**4.88E+01**	**6.21E+01**
KR 84	**6.69E+01**	**1.37E+02**	**2.12E+02**		**6.69E+01**	**1.37E+02**	**2.12E+02**
KR 85	**1.09E+01**	**1.99E+01**	**2.79E+01**		**2.17E+00**	**3.96E+00**	**5.55E+00**
KR 86	**1.17E+02**	**2.28E+02**	**3.36E+02**		**1.17E+02**	**2.28E+02**	**3.36E+02**
RB 85	**6.39E+01**	**1.26E+02**	**1.87E+02**		**7.26E+01**	**1.42E+02**	**2.09E+02**
RB 86	**4.66E-32**	**1.01E-31**	**1.65E-31**		**0.00E+00**	**0.00E+00**	**0.00E+00**
RB 87	**1.50E+02**	**2.92E+02**	**4.30E+02**		**1.50E+02**	**2.92E+02**	**4.30E+02**
SR 86	**1.66E-01**	**6.42E-01**	**1.45E+00**		**1.66E-01**	**6.42E-01**	**1.45E+00**
SR 87	**2.76E-03**	**7.88E-03**	**1.78E-02**		**2.76E-03**	**7.88E-03**	**1.78E-02**
SR 88	**2.16E+02**	**4.19E+02**	**6.18E+02**		**2.16E+02**	**4.19E+02**	**6.18E+02**

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	**20000**	**40000**	**60000**		**20000**	**40000**	**60000**
Enrichment (%)	**2.44**	**3.72**	**4.73**		**2.44**	**3.72**	**4.73**
ISOTOPE	**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**		**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**
SR 89	**3.69E-10**	**3.33E-10**	**2.96E-10**		**0.00E+00**	**0.00E+00**	**0.00E+00**
SR 90	**2.96E+02**	**5.59E+02**	**8.05E+02**		**1.63E+02**	**3.08E+02**	**4.44E+02**
Y 89	**2.82E+02**	**5.46E+02**	**8.04E+02**		**2.82E+02**	**5.46E+02**	**8.04E+02**
Y 90	**7.42E-02**	**1.40E-01**	**2.02E-01**		**4.09E-02**	**7.73E-02**	**1.11E-01**
Y 91	**1.75E-08**	**1.61E-08**	**1.47E-08**		**1.83E-55**	**1.68E-55**	**1.53E-55**
ZR 90	**1.28E+05**	**1.28E+05**	**1.28E+05**		**1.28E+05**	**1.28E+05**	**1.28E+05**
ZR 91	**2.84E+04**	**2.86E+04**	**2.89E+04**		**2.84E+04**	**2.86E+04**	**2.89E+04**
ZR 92	**4.38E+04**	**4.42E+04**	**4.46E+04**		**4.38E+04**	**4.42E+04**	**4.46E+04**
ZR 93	**5.19E+02**	**1.00E+03**	**1.47E+03**		**5.19E+02**	**1.00E+03**	**1.47E+03**
ZR 94	**4.55E+04**	**4.59E+04**	**4.63E+04**		**4.55E+04**	**4.59E+04**	**4.63E+04**
ZR 95	**1.80E-07**	**1.75E-07**	**1.70E-07**		**1.96E-50**	**1.91E-50**	**1.85E-50**
ZR 96	**7.87E+03**	**8.32E+03**	**8.77E+03**		**7.87E+03**	**8.32E+03**	**8.77E+03**
NB 93	**7.22E+02**	**7.18E+02**	**7.14E+02**		**7.22E+02**	**7.18E+02**	**7.14E+02**
NB 93M	**1.15E-03**	**2.56E-03**	**4.11E-03**		**3.48E-03**	**6.81E-03**	**1.01E-02**
NB 94	**3.68E+00**	**6.75E+00**	**9.58E+00**		**3.68E+00**	**6.75E+00**	**9.57E+00**
NB 95	**2.19E-07**	**2.14E-07**	**2.07E-07**		**2.39E-50**	**2.33E-50**	**2.26E-50**
NB 95M	**7.52E-11**	**7.33E-11**	**7.11E-11**		**8.21E-54**	**8.00E-54**	**7.76E-54**
MO 92	**5.71E+01**	**5.71E+01**	**5.70E+01**		**5.71E+01**	**5.71E+01**	**5.70E+01**
MO 93	**1.15E-02**	**2.10E-02**	**3.07E-02**		**1.15E-02**	**2.09E-02**	**3.06E-02**
MO 94	**3.66E+01**	**3.66E+01**	**3.66E+01**		**3.66E+01**	**3.66E+01**	**3.66E+01**
MO 95	**5.45E+02**	**9.81E+02**	**1.38E+03**		**5.45E+02**	**9.81E+02**	**1.38E+03**
MO 96	**8.23E+01**	**1.28E+02**	**2.07E+02**		**8.23E+01**	**1.28E+02**	**2.07E+02**
MO 97	**5.43E+02**	**1.04E+03**	**1.52E+03**		**5.43E+02**	**1.04E+03**	**1.52E+03**
MO 98	**5.93E+02**	**1.08E+03**	**1.55E+03**		**5.93E+02**	**1.08E+03**	**1.55E+03**
MO100	**6.04E+02**	**1.16E+03**	**1.72E+03**		**6.04E+02**	**1.16E+03**	**1.72E+03**
TC 98	**1.79E-03**	**7.32E-03**	**1.63E-02**		**1.79E-03**	**7.32E-03**	**1.63E-02**

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000		20000	40000	60000
Enrichment (%)	2.44	3.72	4.73		2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM		GM/MTIHM	GM/MTIHM	GM/MTIHM
TC 99	4.90E+02	9.16E+02	1.28E+03		4.90E+02	9.16E+02	1.28E+03
RU 99	9.45E-03	2.08E-02	3.25E-02		4.93E-02	9.53E-02	1.37E-01
RU100	3.94E+01	1.48E+02	3.29E+02		3.94E+01	1.48E+02	3.29E+02
RU101	4.72E+02	9.24E+02	1.35E+03		4.72E+02	9.24E+02	1.35E+03
RU102	4.58E+02	9.45E+02	1.45E+03		4.58E+02	9.45E+02	1.45E+03
RU103	4.36E-13	4.58E-13	4.78E-13		0.00E+00	0.00E+00	0.00E+00
RU104	3.20E+02	6.61E+02	1.01E+03		3.20E+02	6.61E+02	1.01E+03
RU106	3.63E+00	5.37E+00	6.70E+00		1.24E-07	1.84E-07	2.29E-07
RH102	9.28E-05	3.27E-04	6.03E-04		2.36E-07	8.31E-07	1.53E-06
RH103	3.18E+02	5.09E+02	6.02E+02		3.18E+02	5.09E+02	6.02E+02
RH103M	3.90E-16	4.10E-16	4.28E-16		0.00E+00	0.00E+00	0.00E+00
RH106	3.41E-06	5.05E-06	6.29E-06		1.17E-13	1.73E-13	2.15E-13
PD104	9.14E+01	3.24E+02	6.57E+02		9.14E+01	3.24E+02	6.57E+02
PD105	2.24E+02	4.66E+02	7.01E+02		2.24E+02	4.66E+02	7.01E+02
PD106	2.01E+02	4.35E+02	6.94E+02		2.04E+02	4.41E+02	7.01E+02
PD107	1.26E+02	2.69E+02	4.13E+02		1.26E+02	2.69E+02	4.13E+02
PD108	8.61E+01	1.85E+02	2.83E+02		8.61E+01	1.85E+02	2.83E+02
PD110	2.81E+01	6.12E+01	9.51E+01		2.81E+01	6.12E+01	9.51E+01
AG107	4.66E-02	4.34E-02	4.02E-02		4.70E-02	4.41E-02	4.13E-02
AG108	9.93E-13	1.60E-12	2.25E-12		8.67E-13	1.40E-12	1.96E-12
AG108M	3.15E-04	5.06E-04	7.11E-04		2.75E-04	4.42E-04	6.20E-04
AG109	4.71E+01	8.87E+01	1.21E+02		4.71E+01	8.87E+01	1.21E+02
AG109M	2.76E-11	3.41E-11	3.87E-11		3.28E-17	4.07E-17	4.62E-17
AG110	3.76E-11	1.03E-10	1.80E-10		3.76E-22	1.03E-21	1.80E-21
AG110M	2.48E-03	6.81E-03	1.19E-02		2.48E-14	6.81E-14	1.19E-13
CD106	3.07E-01	3.06E-01	3.06E-01		3.07E-01	3.06E-01	3.06E-01

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	**20000**	**40000**	**60000**		**20000**	**40000**	**60000**
Enrichment (%)	**2.44**	**3.72**	**4.73**		**2.44**	**3.72**	**4.73**
ISOTOPE	**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**		**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**
CD108	**2.18E-01**	**2.21E-01**	**2.24E-01**		**2.18E-01**	**2.21E-01**	**2.24E-01**
CD109	**2.79E-05**	**3.46E-05**	**3.92E-05**		**3.32E-11**	**4.12E-11**	**4.67E-11**
CD110	**1.65E+01**	**5.41E+01**	**1.13E+02**		**1.65E+01**	**5.42E+01**	**1.13E+02**
CD111	**1.91E+01**	**3.74E+01**	**5.69E+01**		**1.91E+01**	**3.74E+01**	**5.69E+01**
CD112	**1.56E+01**	**2.67E+01**	**3.84E+01**		**1.56E+01**	**2.67E+01**	**3.84E+01**
CD113	**1.45E-01**	**1.88E-01**	**1.86E-01**		**1.45E-01**	**1.89E-01**	**1.86E-01**
CD113M	**1.13E-01**	**2.60E-01**	**4.36E-01**		**3.45E-02**	**7.93E-02**	**1.33E-01**
CD114	**2.36E+01**	**3.77E+01**	**5.13E+01**		**2.36E+01**	**3.77E+01**	**5.13E+01**
CD115M	**3.01E-14**	**3.40E-14**	**3.71E-14**		**0.00E+00**	**0.00E+00**	**0.00E+00**
CD116	**6.91E+00**	**1.20E+01**	**1.67E+01**		**6.91E+00**	**1.20E+01**	**1.67E+01**
IN113	**5.11E-01**	**8.50E-01**	**1.16E+00**		**5.89E-01**	**1.03E+00**	**1.46E+00**
IN113M	**7.99E-10**	**8.73E-10**	**9.98E-10**		**1.05E-33**	**1.15E-33**	**1.31E-33**
IN114	**4.50E-19**	**8.83E-19**	**1.35E-18**		**0.00E+00**	**0.00E+00**	**0.00E+00**
IN114M	**2.80E-14**	**5.49E-14**	**8.39E-14**		**0.00E+00**	**0.00E+00**	**0.00E+00**
IN115	**2.23E+00**	**2.47E+00**	**2.47E+00**		**2.23E+00**	**2.47E+00**	**2.47E+00**
IN115M	**8.35E-21**	**9.45E-21**	**1.03E-20**		**0.00E+00**	**0.00E+00**	**0.00E+00**
SN112	**3.84E+01**	**3.80E+01**	**3.76E+01**		**3.84E+01**	**3.80E+01**	**3.76E+01**
SN113	**1.33E-06**	**1.45E-06**	**1.66E-06**		**1.75E-30**	**1.91E-30**	**2.19E-30**
SN114	**2.65E+01**	**2.66E+01**	**2.67E+01**		**2.65E+01**	**2.66E+01**	**2.67E+01**
SN115	**1.43E+01**	**1.40E+01**	**1.36E+01**		**1.43E+01**	**1.40E+01**	**1.36E+01**
SN116	**5.91E+02**	**5.92E+02**	**5.94E+02**		**5.91E+02**	**5.92E+02**	**5.94E+02**
SN117	**3.21E+02**	**3.28E+02**	**3.35E+02**		**3.21E+02**	**3.28E+02**	**3.35E+02**
SN117M	**4.73E-41**	**5.23E-41**	**5.80E-41**		**0.00E+00**	**0.00E+00**	**0.00E+00**
SN118	**9.96E+02**	**9.98E+02**	**1.00E+03**		**9.96E+02**	**9.98E+02**	**1.00E+03**
SN119	**3.64E+02**	**3.74E+02**	**3.83E+02**		**3.64E+02**	**3.74E+02**	**3.83E+02**
SN119M	**6.94E-03**	**8.40E-03**	**1.00E-02**		**4.20E-14**	**5.08E-14**	**6.06E-14**

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000		20000	40000	60000
Enrichment (%)	2.44	3.72	4.73		2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM		GM/MTIHM	GM/MTIHM	GM/MTIHM
SN120	1.35E+03	1.36E+03	1.36E+03		1.35E+03	1.36E+03	1.36E+03
SN121M	8.29E-03	1.54E-02	2.26E-02		5.86E-03	1.09E-02	1.60E-02
SN122	2.00E+02	2.06E+02	2.11E+02		2.00E+02	2.06E+02	2.11E+02
SN123	2.32E-05	2.57E-05	2.84E-05		1.22E-26	1.35E-26	1.49E-26
SN124	2.47E+02	2.53E+02	2.59E+02		2.47E+02	2.53E+02	2.59E+02
SN126	1.66E+01	3.34E+01	4.99E+01		1.66E+01	3.34E+01	4.99E+01
SB121	6.37E+00	1.22E+01	1.72E+01		6.37E+00	1.22E+01	1.72E+01
SB123	6.43E+00	1.25E+01	1.80E+01		6.43E+00	1.25E+01	1.80E+01
SB124	3.18E-11	7.54E-11	1.25E-10		6.90E-57	1.64E-56	2.71E-56
SB125	3.02E+00	4.95E+00	6.45E+00		5.80E-03	9.49E-03	1.24E-02
SB126	7.88E-07	1.59E-06	2.37E-06		7.88E-07	1.59E-06	2.37E-06
SB126M	5.99E-09	1.21E-08	1.80E-08		5.99E-09	1.21E-08	1.80E-08
TE122	2.85E-01	1.06E+00	2.23E+00		2.85E-01	1.06E+00	2.23E+00
TE123	3.18E-03	1.69E-02	4.15E-02		3.18E-03	1.69E-02	4.15E-02
TE123M	1.63E-08	8.18E-08	2.11E-07		1.75E-31	8.79E-31	2.27E-30
TE124	1.67E-01	6.74E-01	1.51E+00		1.67E-01	6.74E-01	1.51E+00
TE125	9.91E+00	2.09E+01	3.17E+01		1.30E+01	2.59E+01	3.83E+01
TE125M	4.23E-02	6.92E-02	9.02E-02		8.11E-05	1.33E-04	1.73E-04
TE126	4.31E-01	1.03E+00	1.78E+00		4.33E-01	1.03E+00	1.79E+00
TE127	4.27E-08	4.70E-08	5.15E-08		2.59E-33	2.85E-33	3.13E-33
TE127M	1.22E-05	1.34E-05	1.47E-05		7.40E-31	8.15E-31	8.93E-31
TE128	6.66E+01	1.34E+02	2.01E+02		6.66E+01	1.34E+02	2.01E+02
TE129	5.85E-20	6.01E-20	6.20E-20		0.00E+00	0.00E+00	0.00E+00
TE129M	6.25E-17	6.41E-17	6.62E-17		0.00E+00	0.00E+00	0.00E+00
TE130	2.14E+02	4.29E+02	6.44E+02		2.14E+02	4.29E+02	6.44E+02
I127	3.41E+01	6.63E+01	9.59E+01		3.41E+01	6.63E+01	9.59E+01

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	**20000**	**40000**	**60000**		**20000**	**40000**	**60000**
Enrichment (%)	**2.44**	**3.72**	**4.73**		**2.44**	**3.72**	**4.73**
ISOTOPE	**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**		**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**
I129	**1.10E+02**	**2.16E+02**	**3.13E+02**		**1.10E+02**	**2.16E+02**	**3.13E+02**
XE127	**4.03E-22**	**2.06E-21**	**5.27E-21**		**0.00E+00**	**0.00E+00**	**0.00E+00**
XE128	**1.17E+00**	**4.75E+00**	**1.07E+01**		**1.17E+00**	**4.75E+00**	**1.07E+01**
XE129	**4.26E-03**	**3.03E-02**	**9.97E-02**		**4.38E-03**	**3.05E-02**	**1.00E-01**
XE130	**4.64E+00**	**1.56E+01**	**3.48E+01**		**4.64E+00**	**1.56E+01**	**3.48E+01**
XE131	**2.95E+02**	**4.77E+02**	**5.73E+02**		**2.95E+02**	**4.77E+02**	**5.73E+02**
XE131M	**2.33E-47**	**2.33E-47**	**2.34E-47**		**0.00E+00**	**0.00E+00**	**0.00E+00**
XE132	**6.18E+02**	**1.35E+03**	**2.16E+03**		**6.18E+02**	**1.35E+03**	**2.16E+03**
XE134	**8.90E+02**	**1.77E+03**	**2.65E+03**		**8.90E+02**	**1.77E+03**	**2.65E+03**
XE136	**1.40E+03**	**2.69E+03**	**4.01E+03**		**1.40E+03**	**2.69E+03**	**4.01E+03**
CS133	**7.21E+02**	**1.31E+03**	**1.78E+03**		**7.21E+02**	**1.31E+03**	**1.78E+03**
CS134	**1.03E+01**	**3.08E+01**	**5.71E+01**		**2.30E-03**	**6.90E-03**	**1.28E-02**
CS135	**1.89E+02**	**4.86E+02**	**7.72E+02**		**1.89E+02**	**4.86E+02**	**7.72E+02**
CS136	**5.07E-43**	**9.94E-43**	**1.59E-42**		**0.00E+00**	**0.00E+00**	**0.00E+00**
CS137	**6.50E+02**	**1.27E+03**	**1.86E+03**		**3.65E+02**	**7.11E+02**	**1.05E+03**
BA132	**5.38E-04**	**2.15E-03**	**4.70E-03**		**5.38E-04**	**2.15E-03**	**4.70E-03**
BA134	**5.64E+01**	**2.08E+02**	**4.44E+02**		**6.66E+01**	**2.39E+02**	**5.01E+02**
BA135	**5.10E-02**	**6.57E-01**	**2.67E+00**		**5.25E-02**	**6.60E-01**	**2.67E+00**
BA136	**8.77E+00**	**2.98E+01**	**6.49E+01**		**8.77E+00**	**2.98E+01**	**6.49E+01**
BA136M	**2.28E-50**	**4.46E-50**	**7.13E-50**		**0.00E+00**	**0.00E+00**	**0.00E+00**
BA137	**9.55E+01**	**2.21E+02**	**3.65E+02**		**3.81E+02**	**7.77E+02**	**1.18E+03**
BA137M	**9.95E-05**	**1.94E-04**	**2.85E-04**		**5.58E-05**	**1.09E-04**	**1.60E-04**
BA138	**7.74E+02**	**1.54E+03**	**2.30E+03**		**7.74E+02**	**1.54E+03**	**2.30E+03**
BA140	**0.00E+00**	**2.14E-42**	**2.07E-42**		**0.00E+00**	**0.00E+00**	**0.00E+00**
LA138	**3.97E-03**	**6.60E-03**	**8.07E-03**		**3.97E-03**	**6.60E-03**	**8.07E-03**
LA139	**7.42E+02**	**1.47E+03**	**2.17E+03**		**7.42E+02**	**1.47E+03**	**2.17E+03**

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000		20000	40000	60000
Enrichment (%)	2.44	3.72	4.73		2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM		GM/MTIHM	GM/MTIHM	GM/MTIHM
LA140	0.00E+00	3.22E-43	3.13E-43		0.00E+00	0.00E+00	0.00E+00
CE140	7.46E+02	1.49E+03	2.23E+03		7.46E+02	1.49E+03	2.23E+03
CE141	6.43E-16	6.28E-16	6.09E-16		0.00E+00	0.00E+00	0.00E+00
CE142	6.85E+02	1.36E+03	2.02E+03		6.85E+02	1.36E+03	2.02E+03
CE144	3.40E+00	3.82E+00	4.00E+00		7.27E-10	8.17E-10	8.56E-10
PR141	6.84E+02	1.34E+03	1.98E+03		6.84E+02	1.34E+03	1.98E+03
PR143	6.57E-40	6.34E-40	6.08E-40		0.00E+00	0.00E+00	0.00E+00
PR144	1.44E-04	1.61E-04	1.69E-04		3.07E-14	3.45E-14	3.61E-14
PR144M	7.18E-07	8.06E-07	8.45E-07		1.54E-16	1.72E-16	1.81E-16
ND142	9.40E+00	3.47E+01	8.08E+01		9.40E+00	3.47E+01	8.08E+01
ND143	5.44E+02	9.40E+02	1.16E+03		5.44E+02	9.40E+02	1.16E+03
ND144	7.37E+02	1.59E+03	2.60E+03		7.40E+02	1.59E+03	2.60E+03
ND145	4.26E+02	7.93E+02	1.10E+03		4.26E+02	7.93E+02	1.10E+03
ND146	3.98E+02	8.45E+02	1.35E+03		3.98E+02	8.45E+02	1.35E+03
ND147	1.43E-49	1.41E-49	1.39E-49		0.00E+00	0.00E+00	0.00E+00
ND148	2.25E+02	4.45E+02	6.61E+02		2.25E+02	4.45E+02	6.61E+02
ND150	1.06E+02	2.14E+02	3.20E+02		1.06E+02	2.14E+02	3.20E+02
PM146	1.40E-03	3.60E-03	5.61E-03		5.99E-05	1.54E-04	2.40E-04
PM147	3.35E+01	3.69E+01	3.47E+01		4.54E-02	4.99E-02	4.70E-02
PM148	3.80E-16	4.96E-16	4.72E-16		0.00E+00	0.00E+00	0.00E+00
PM148M	5.19E-14	6.78E-14	6.45E-14		0.00E+00	0.00E+00	0.00E+00
SM146	1.78E-03	9.41E-03	2.15E-02		2.28E-03	1.07E-02	2.35E-02
SM147	1.26E+02	1.86E+02	2.07E+02		1.60E+02	2.23E+02	2.42E+02
SM148	8.13E+01	2.32E+02	4.09E+02		8.13E+01	2.32E+02	4.09E+02
SM149	3.37E+00	4.11E+00	4.06E+00		3.37E+00	4.11E+00	4.06E+00
SM150	1.86E+02	3.69E+02	5.26E+02		1.86E+02	3.69E+02	5.26E+02

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	**20000**	**40000**	**60000**		**20000**	**40000**	**60000**
Enrichment (%)	**2.44**	**3.72**	**4.73**		**2.44**	**3.72**	**4.73**
ISOTOPE	**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**		**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**
SM151	**1.10E+01**	**1.80E+01**	**2.17E+01**		**9.03E+00**	**1.49E+01**	**1.79E+01**
SM152	**8.80E+01**	**1.45E+02**	**1.83E+02**		**8.80E+01**	**1.45E+02**	**1.83E+02**
SM154	**2.19E+01**	**4.55E+01**	**6.96E+01**		**2.19E+01**	**4.55E+01**	**6.96E+01**
EU150	**8.11E-08**	**3.68E-07**	**6.60E-07**		**5.01E-08**	**2.27E-07**	**4.08E-07**
EU151	**4.38E-01**	**7.23E-01**	**8.67E-01**		**2.36E+00**	**3.88E+00**	**4.66E+00**
EU152	**2.24E-02**	**4.49E-02**	**4.91E-02**		**6.25E-03**	**1.26E-02**	**1.37E-02**
EU153	**6.50E+01**	**1.50E+02**	**2.27E+02**		**6.50E+01**	**1.50E+02**	**2.27E+02**
EU154	**9.65E+00**	**3.16E+01**	**5.67E+01**		**1.29E+00**	**4.21E+00**	**7.56E+00**
EU155	**3.08E+00**	**9.16E+00**	**1.65E+01**		**9.35E-02**	**2.78E-01**	**5.01E-01**
EU156	**1.11E-36**	**2.48E-36**	**4.60E-36**		**0.00E+00**	**0.00E+00**	**0.00E+00**
GD152	**2.67E-02**	**7.44E-02**	**1.19E-01**		**3.12E-02**	**8.34E-02**	**1.29E-01**
GD153	**8.20E-06**	**3.26E-05**	**6.90E-05**		**3.60E-17**	**1.43E-16**	**3.03E-16**
GD154	**5.46E+00**	**2.00E+01**	**3.90E+01**		**1.38E+01**	**4.73E+01**	**8.81E+01**
GD155	**3.15E+00**	**9.42E+00**	**1.69E+01**		**6.14E+00**	**1.83E+01**	**3.29E+01**
GD156	**2.57E+01**	**8.67E+01**	**2.12E+02**		**2.57E+01**	**8.67E+01**	**2.12E+02**
GD157	**7.62E-02**	**1.51E-01**	**2.50E-01**		**7.62E-02**	**1.51E-01**	**2.50E-01**
GD158	**1.05E+01**	**2.45E+01**	**4.74E+01**		**1.05E+01**	**2.45E+01**	**4.74E+01**
GD160	**1.27E+00**	**2.11E+00**	**2.97E+00**		**1.27E+00**	**2.11E+00**	**2.97E+00**
TB159	**1.50E+00**	**3.25E+00**	**5.17E+00**		**1.50E+00**	**3.25E+00**	**5.17E+00**
TB160	**1.11E-09**	**2.93E-09**	**5.26E-09**		**1.07E-47**	**2.81E-47**	**5.05E-47**
DY160	**1.18E-01**	**4.24E-01**	**8.55E-01**		**1.18E-01**	**4.24E-01**	**8.55E-01**
DY161	**2.64E-01**	**5.36E-01**	**8.34E-01**		**2.64E-01**	**5.36E-01**	**8.34E-01**
DY162	**2.08E-01**	**4.31E-01**	**6.74E-01**		**2.08E-01**	**4.31E-01**	**6.74E-01**
DY163	**1.42E-01**	**3.85E-01**	**6.98E-01**		**1.42E-01**	**3.85E-01**	**6.98E-01**
DY164	**3.17E-02**	**9.49E-02**	**1.76E-01**		**3.17E-02**	**9.49E-02**	**1.76E-01**
HO165	**5.64E-02**	**1.88E-01**	**4.39E-01**		**5.64E-02**	**1.88E-01**	**4.39E-01**

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000		20000	40000	60000
Enrichment (%)	2.44	3.72	4.73		2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM		GM/MTIHM	GM/MTIHM	GM/MTIHM
HO166M	3.66E-04	2.04E-03	6.68E-03		3.61E-04	2.01E-03	6.58E-03
ER166	1.66E-02	5.65E-02	1.47E-01		1.66E-02	5.65E-02	1.47E-01
ER167	2.58E-03	4.38E-03	6.95E-03		2.58E-03	4.38E-03	6.95E-03
ER168	2.60E-03	7.87E-03	1.62E-02		2.60E-03	7.87E-03	1.62E-02
TM169	1.22E-05	6.04E-05	1.57E-04		1.22E-05	6.04E-05	1.57E-04
TM170	6.96E-11	4.96E-10	1.58E-09		2.92E-32	2.08E-31	6.62E-31
TM171	1.21E-08	1.43E-07	6.57E-07		1.46E-12	1.72E-11	7.91E-11
YB170	2.20E-06	2.14E-05	8.10E-05		2.20E-06	2.14E-05	8.10E-05
YB171	0.00E+00	1.46E-06	7.84E-06		0.00E+00	1.60E-06	8.50E-06
LU175	1.27E-02	1.43E-02	1.41E-02		1.27E-02	1.43E-02	1.41E-02
LU176	2.77E-04	5.12E-04	5.76E-04		2.77E-04	5.12E-04	5.76E-04
LU177	1.73E-12	3.88E-12	5.64E-12		3.19E-30	7.15E-30	1.04E-29
LU177M	1.74E-10	3.89E-10	5.67E-10		3.21E-28	7.18E-28	1.05E-27
HF174	1.66E-02	1.15E-02	7.33E-03		1.66E-02	1.15E-02	7.33E-03
HF175	2.78E-11	1.73E-11	1.23E-11		1.50E-50	9.33E-51	6.62E-51
HF176	7.63E-01	5.93E-01	4.63E-01		7.63E-01	5.93E-01	4.63E-01
HF177	3.03E-01	1.17E-01	7.63E-02		3.03E-01	1.17E-01	7.63E-02
HF178	4.53E+00	2.51E+00	1.43E+00		4.53E+00	2.51E+00	1.43E+00
HF179	6.09E+00	6.91E+00	6.58E+00		6.09E+00	6.91E+00	6.58E+00
HF180	8.15E+00	9.52E+00	1.09E+01		8.15E+00	9.52E+00	1.09E+01
HF181	2.85E-15	3.27E-15	4.16E-15		0.00E+00	0.00E+00	0.00E+00
HF182	1.10E-03	1.67E-03	2.85E-03		1.10E-03	1.67E-03	2.85E-03
TA181	2.09E-01	3.46E-01	4.98E-01		2.09E-01	3.46E-01	4.98E-01
TA182	5.84E-08	1.27E-07	2.03E-07		3.85E-11	5.83E-11	9.97E-11
W180	8.56E-03	7.98E-03	7.45E-03		8.56E-03	7.98E-03	7.45E-03
W181	5.41E-09	5.67E-09	6.15E-09		1.12E-31	1.18E-31	1.28E-31

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	**20000**	**40000**	**60000**		**20000**	**40000**	**60000**
Enrichment (%)	**2.44**	**3.72**	**4.73**		**2.44**	**3.72**	**4.73**
ISOTOPE	**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**		**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**
W182	**1.50E+00**	**1.23E+00**	**1.03E+00**		**1.50E+00**	**1.23E+00**	**1.03E+00**
W183	**1.30E+00**	**1.45E+00**	**1.57E+00**		**1.30E+00**	**1.45E+00**	**1.57E+00**
W184	**2.38E+00**	**2.54E+00**	**2.71E+00**		**2.38E+00**	**2.54E+00**	**2.71E+00**
W185	**1.50E-10**	**1.67E-10**	**2.01E-10**		**3.76E-47**	**4.19E-47**	**5.04E-47**
W186	**1.67E+00**	**1.38E+00**	**1.14E+00**		**1.67E+00**	**1.38E+00**	**1.14E+00**
W188	**2.23E-12**	**2.13E-12**	**2.20E-12**		**5.48E-52**	**5.25E-52**	**5.43E-52**
RE185	**1.31E-02**	**1.87E-02**	**2.31E-02**		**1.31E-02**	**1.87E-02**	**2.31E-02**
RE187	**3.88E-01**	**6.02E-01**	**7.36E-01**		**3.88E-01**	**6.02E-01**	**7.36E-01**
RE188	**2.29E-14**	**2.20E-14**	**2.27E-14**		**5.65E-54**	**5.41E-54**	**5.59E-54**
OS186	**3.66E-03**	**1.20E-02**	**2.31E-02**		**3.66E-03**	**1.20E-02**	**2.31E-02**
OS187	**3.09E-11**	**5.30E-11**	**6.67E-11**		**1.65E-10**	**2.62E-10**	**3.22E-10**
OS188	**4.30E-02**	**1.19E-01**	**2.22E-01**		**4.30E-02**	**1.19E-01**	**2.22E-01**
OS189	**7.76E-04**	**3.94E-03**	**1.04E-02**		**7.76E-04**	**3.94E-03**	**1.04E-02**
OS190	**6.28E-05**	**6.23E-04**	**2.47E-03**		**6.28E-05**	**6.23E-04**	**2.47E-03**
IR191	**0.00E+00**	**3.66E-06**	**1.75E-05**		**0.00E+00**	**3.66E-06**	**1.75E-05**
IR192	**3.25E-14**	**9.37E-13**	**6.79E-12**		**2.88E-14**	**8.43E-13**	**6.15E-12**
IR192M	**0.00E+00**	**0.00E+00**	**7.85E-09**		**0.00E+00**	**0.00E+00**	**7.31E-09**
IR193	**0.00E+00**	**0.00E+00**	**2.46E-06**		**0.00E+00**	**0.00E+00**	**2.46E-06**
PT192	**0.00E+00**	**2.19E-06**	**1.54E-05**		**0.00E+00**	**2.19E-06**	**1.54E-05**
PT193	**0.00E+00**	**1.84E-08**	**2.17E-07**		**0.00E+00**	**1.77E-08**	**2.10E-07**
TL206	**4.97E-17**	**8.79E-17**	**1.29E-16**		**4.97E-17**	**8.79E-17**	**1.29E-16**
TL207	**6.51E-15**	**2.39E-14**	**4.10E-14**		**4.56E-14**	**1.03E-13**	**1.50E-13**
TL208	**6.05E-12**	**3.56E-11**	**8.96E-11**		**8.94E-12**	**4.98E-11**	**1.22E-10**
TL209	**1.38E-18**	**1.53E-17**	**6.07E-17**		**4.00E-18**	**2.21E-17**	**7.16E-17**
PB204	**1.38E-02**	**1.38E-02**	**1.37E-02**		**1.38E-02**	**1.38E-02**	**1.37E-02**
PB205	**2.02E-05**	**3.39E-05**	**4.96E-05**		**2.02E-05**	**3.39E-05**	**4.96E-05**

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000		20000	40000	60000
Enrichment (%)	2.44	3.72	4.73		2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM		GM/MTIHM	GM/MTIHM	GM/MTIHM
PB206	2.31E-01	2.27E-01	2.21E-01		2.31E-01	2.27E-01	2.21E-01
PB207	2.30E-01	2.34E-01	2.39E-01		2.30E-01	2.34E-01	2.39E-01
PB208	5.26E-01	5.26E-01	5.27E-01		5.26E-01	5.27E-01	5.28E-01
PB209	5.73E-15	6.35E-14	2.53E-13		1.67E-14	9.20E-14	2.98E-13
PB210	8.25E-11	2.51E-10	5.40E-10		6.68E-09	9.60E-09	1.13E-08
PB211	5.04E-14	1.85E-13	3.17E-13		3.52E-13	7.97E-13	1.16E-12
PB212	3.57E-09	2.10E-08	5.29E-08		5.27E-09	2.94E-08	7.20E-08
PB214	2.72E-15	5.40E-15	7.48E-15		5.96E-14	8.29E-14	9.59E-14
BI208	2.69E-06	5.44E-06	7.95E-06		2.69E-06	5.43E-06	7.95E-06
BI209	4.00E-01	4.00E-01	4.00E-01		4.00E-01	4.00E-01	4.00E-01
BI210	5.08E-14	1.55E-13	3.32E-13		4.11E-12	5.91E-12	6.95E-12
BI210M	1.92E-05	3.39E-05	4.96E-05		1.92E-05	3.39E-05	4.96E-05
BI211	2.97E-15	1.09E-14	1.87E-14		2.08E-14	4.71E-14	6.85E-14
BI212	3.39E-10	1.99E-09	5.01E-09		5.00E-10	2.79E-09	6.83E-09
BI213	1.35E-15	1.49E-14	5.95E-14		3.92E-15	2.16E-14	7.01E-14
BI214	2.02E-15	4.01E-15	5.55E-15		4.43E-14	6.15E-14	7.12E-14
PO210	8.70E-10	9.12E-10	1.07E-09		1.14E-10	1.63E-10	1.92E-10
PO211	3.65E-20	1.34E-19	2.29E-19		2.55E-19	5.77E-19	8.40E-19
PO212	1.79E-20	1.05E-19	2.65E-19		2.65E-20	1.48E-19	3.62E-19
PO213	2.02E-24	2.24E-23	8.92E-23		5.87E-24	3.24E-23	1.05E-22
PO214	2.78E-22	5.51E-22	7.64E-22		6.09E-21	8.47E-21	9.80E-21
PO215	4.22E-20	1.55E-19	2.65E-19		2.95E-19	6.68E-19	9.72E-19
PO216	1.43E-14	8.37E-14	2.11E-13		2.11E-14	1.17E-13	2.88E-13
PO218	3.16E-16	6.26E-16	8.67E-16		6.92E-15	9.61E-15	1.11E-14
AT217	1.62E-20	1.79E-19	7.14E-19		4.70E-20	2.60E-19	8.42E-19
RN219	9.55E-17	3.51E-16	6.01E-16		6.69E-16	1.51E-15	2.20E-15

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	**20000**	**40000**	**60000**		**20000**	**40000**	**60000**
Enrichment (%)	**2.44**	**3.72**	**4.73**		**2.44**	**3.72**	**4.73**
ISOTOPE	**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**		**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**
RN220	**5.38E-12**	**3.16E-11**	**7.96E-11**		**7.95E-12**	**4.43E-11**	**1.09E-10**
RN222	**5.80E-13**	**1.15E-12**	**1.59E-12**		**1.27E-11**	**1.77E-11**	**2.04E-11**
FR221	**1.47E-16**	**1.63E-15**	**6.49E-15**		**4.27E-16**	**2.36E-15**	**7.65E-15**
FR223	**4.43E-16**	**1.63E-15**	**2.79E-15**		**3.10E-15**	**7.02E-15**	**1.02E-14**
RA223	**2.43E-11**	**8.92E-11**	**1.53E-10**		**1.70E-10**	**3.84E-10**	**5.60E-10**
RA224	**3.12E-08**	**1.83E-07**	**4.61E-07**		**4.60E-08**	**2.56E-07**	**6.28E-07**
RA225	**6.64E-13**	**7.36E-12**	**2.93E-11**		**1.93E-12**	**1.07E-11**	**3.46E-11**
RA226	**9.03E-08**	**1.79E-07**	**2.48E-07**		**1.98E-06**	**2.75E-06**	**3.18E-06**
RA228	**5.53E-14**	**1.46E-13**	**2.54E-13**		**7.64E-13**	**1.48E-12**	**2.07E-12**
AC225	**4.49E-13**	**4.97E-12**	**1.98E-11**		**1.30E-12**	**7.20E-12**	**2.34E-11**
AC227	**1.72E-08**	**6.30E-08**	**1.08E-07**		**1.20E-07**	**2.72E-07**	**3.96E-07**
AC228	**5.77E-18**	**1.52E-17**	**2.65E-17**		**7.98E-17**	**1.55E-16**	**2.16E-16**
TH227	**3.99E-11**	**1.47E-10**	**2.51E-10**		**2.79E-10**	**6.32E-10**	**9.19E-10**
TH228	**6.04E-06**	**3.55E-05**	**8.94E-05**		**8.94E-06**	**4.98E-05**	**1.22E-04**
TH229	**1.22E-07**	**1.36E-06**	**5.41E-06**		**3.56E-07**	**1.96E-06**	**6.37E-06**
TH230	**2.98E-03**	**4.46E-03**	**5.08E-03**		**1.43E-02**	**1.94E-02**	**2.28E-02**
TH231	**3.81E-08**	**3.41E-08**	**2.18E-08**		**3.81E-08**	**3.41E-08**	**2.19E-08**
TH232	**4.55E-04**	**1.02E-03**	**1.57E-03**		**2.33E-03**	**4.44E-03**	**6.11E-03**
TH234	**1.39E-05**	**1.36E-05**	**1.32E-05**		**1.39E-05**	**1.36E-05**	**1.32E-05**
PA231	**1.85E-04**	**5.64E-04**	**8.95E-04**		**4.12E-04**	**7.67E-04**	**1.02E-03**
PA233	**7.98E-06**	**2.33E-05**	**4.08E-05**		**8.69E-06**	**2.45E-05**	**4.20E-05**
PA234	**2.10E-10**	**2.04E-10**	**1.99E-10**		**2.10E-10**	**2.04E-10**	**1.99E-10**
PA234M	**4.70E-10**	**4.57E-10**	**4.46E-10**		**4.70E-10**	**4.57E-10**	**4.46E-10**
U232	**3.27E-04**	**1.86E-03**	**4.62E-03**		**3.34E-04**	**1.86E-03**	**4.56E-03**
U233	**1.22E-03**	**2.91E-03**	**4.30E-03**		**3.16E-03**	**8.48E-03**	**1.40E-02**
U234	**1.58E+02**	**1.91E+02**	**2.00E+02**		**1.67E+02**	**2.36E+02**	**3.06E+02**

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000		20000	40000	60000
Enrichment (%)	2.44	3.72	4.73		2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM		GM/MTIHM	GM/MTIHM	GM/MTIHM
U235	9.36E+03	8.39E+03	5.37E+03		9.36E+03	8.39E+03	5.38E+03
U236	2.57E+03	4.71E+03	6.23E+03		2.57E+03	4.71E+03	6.24E+03
U237	2.32E-05	3.66E-05	3.89E-05		6.97E-06	1.10E-05	1.17E-05
U238	9.60E+05	9.33E+05	9.10E+05		9.60E+05	9.33E+05	9.10E+05
U240	1.16E-13	8.15E-13	2.41E-12		1.16E-13	8.15E-13	2.41E-12
NP235	5.96E-08	2.64E-07	5.34E-07		6.84E-15	3.03E-14	6.13E-14
NP236	1.39E-04	6.37E-04	1.25E-03		1.39E-04	6.37E-04	1.25E-03
NP237	2.35E+02	6.86E+02	1.20E+03		2.56E+02	7.20E+02	1.24E+03
NP238	1.41E-07	4.93E-07	6.11E-07		1.25E-07	4.40E-07	5.45E-07
NP239	2.57E-05	1.19E-04	2.33E-04		2.57E-05	1.19E-04	2.33E-04
NP240M	1.02E-15	7.13E-15	2.11E-14		1.02E-15	7.13E-15	2.11E-14
PU236	9.53E-05	4.91E-04	1.15E-03		2.19E-07	1.13E-06	2.65E-06
PU237	4.70E-17	2.98E-16	8.38E-16		0.00E+00	0.00E+00	0.00E+00
PU238	5.09E+01	2.55E+02	5.99E+02		4.18E+01	2.10E+02	4.92E+02
PU239	4.70E+03	6.39E+03	7.42E+03		4.69E+03	6.39E+03	7.42E+03
PU240	1.45E+03	2.55E+03	4.02E+03		1.45E+03	2.57E+03	4.09E+03
PU241	7.50E+02	1.18E+03	1.26E+03		2.25E+02	3.55E+02	3.77E+02
PU242	2.28E+02	5.65E+02	8.17E+02		2.28E+02	5.65E+02	8.17E+02
PU243	1.94E-15	1.17E-13	1.16E-12		1.94E-15	1.17E-13	1.16E-12
PU244	6.08E-03	4.26E-02	1.26E-01		6.08E-03	4.26E-02	1.26E-01
AM241	2.24E+02	3.78E+02	4.10E+02		7.27E+02	1.17E+03	1.25E+03
AM242	8.97E-06	3.15E-05	3.90E-05		8.00E-06	2.81E-05	3.48E-05
AM242M	7.50E-01	2.63E+00	3.26E+00		6.69E-01	2.35E+00	2.91E+00
AM243	2.99E+01	1.39E+02	2.71E+02		2.99E+01	1.38E+02	2.71E+02
AM244	0.00E+00	5.94E-17	1.76E-16		0.00E+00	5.94E-17	1.76E-16
CM241	1.73E-22	8.19E-22	1.29E-21		0.00E+00	0.00E+00	0.00E+00

Table C-1 (continued).

	DECAY TIME 5 YEARS				DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	**20000**	**40000**	**60000**		**20000**	**40000**	**60000**
Enrichment (%)	**2.44**	**3.72**	**4.73**		**2.44**	**3.72**	**4.73**
ISOTOPE	**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**		**GM/MTIHM**	**GM/MTIHM**	**GM/MTIHM**
CM242	**4.04E-03**	**1.43E-02**	**1.86E-02**		**1.62E-03**	**5.68E-03**	**7.03E-03**
CM243	**9.03E-02**	**5.93E-01**	**1.15E+00**		**4.92E-02**	**3.23E-01**	**6.25E-01**
CM244	**4.38E+00**	**4.31E+01**	**1.37E+02**		**1.68E+00**	**1.66E+01**	**5.26E+01**
CM245	**1.35E-01**	**2.37E+00**	**9.56E+00**		**1.35E-01**	**2.37E+00**	**9.54E+00**
CM246	**9.38E-03**	**2.81E-01**	**1.88E+00**		**9.35E-03**	**2.80E-01**	**1.87E+00**
CM247	**5.43E-05**	**3.27E-03**	**3.26E-02**		**5.43E-05**	**3.27E-03**	**3.26E-02**
CM248	**1.79E-06**	**2.10E-04**	**3.32E-03**		**1.79E-06**	**2.10E-04**	**3.32E-03**
BK249	**2.33E-10**	**4.43E-08**	**8.69E-07**		**6.01E-19**	**1.14E-16**	**2.24E-15**
BK250	**0.00E+00**	**0.00E+00**	**1.42E-14**		**0.00E+00**	**0.00E+00**	**1.00E-19**
CF249	**1.30E-08**	**2.60E-06**	**5.22E-05**		**1.26E-08**	**2.51E-06**	**5.05E-05**
CF250	**1.63E-09**	**3.76E-07**	**1.04E-05**		**4.33E-10**	**9.99E-08**	**2.76E-06**
CF251	**0.00E+00**	**2.21E-07**	**6.77E-06**		**0.00E+00**	**2.17E-07**	**6.65E-06**
CF252	**6.19E-11**	**3.33E-08**	**1.88E-06**		**8.69E-14**	**4.67E-11**	**2.64E-09**
CF254	**0.00E+00**	**0.00E+00**	**6.52E-19**		**0.00E+00**	**0.00E+00**	**2.40E-64**
ES254	**0.00E+00**	**0.00E+00**	**2.95E-11**		**0.00E+00**	**0.00E+00**	**3.16E-21**
TOTAL	**1.44E+06**	**1.44E+06**	**1.44E+06**		**1.44E+06**	**1.44E+06**	**1.44E+06**
Decay Heat (Watts/MT)	**994**	**2,136**	**3,532**		**420**	**895**	**1,438**

Table C-2 Pressurized water reactor radionuclide inventory at 100 and 500 year decay.

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000	20000	40000	60000
Enrichment (%)	2.44	3.72	4.73	2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
H 1	3.39E+00	3.44E+00	3.48E+00	3.39E+00	3.44E+00	3.48E+00
H 2	4.05E-03	5.70E-03	7.92E-03	4.05E-03	5.70E-03	7.92E-03
H 3	2.12E-04	3.43E-04	4.61E-04	3.76E-14	6.09E-14	8.18E-14
HE 3	2.32E-02	2.72E-02	2.97E-02	2.33E-02	2.73E-02	2.98E-02
HE 4	4.74E+00	1.12E+01	1.92E+01	1.38E+01	2.71E+01	3.99E+01
LI 6	1.83E-02	9.65E-03	4.08E-03	1.83E-02	9.65E-03	4.08E-03
LI 7	1.08E+00	1.08E+00	1.08E+00	1.08E+00	1.08E+00	1.08E+00
BE 9	5.62E-04	1.13E-03	1.66E-03	5.62E-04	1.13E-03	1.66E-03
BE 10	1.08E-04	2.49E-04	4.14E-04	1.08E-04	2.49E-04	4.14E-04
B 10	1.78E-03	5.25E-04	2.20E-04	1.78E-03	5.25E-04	2.20E-04
B 11	9.59E-01	9.81E-01	1.00E+00	9.59E-01	9.81E-01	1.00E+00
C 12	1.55E+02	1.55E+02	1.55E+02	1.55E+02	1.55E+02	1.55E+02
C 13	6.50E+00	1.12E+01	1.56E+01	6.50E+00	1.12E+01	1.56E+01
C 14	1.96E-01	3.08E-01	4.50E-01	1.87E-01	2.93E-01	4.29E-01
N 14	1.06E+02	1.06E+02	1.06E+02	1.06E+02	1.06E+02	1.06E+02
N 15	4.28E-01	4.33E-01	4.40E-01	4.28E-01	4.33E-01	4.40E-01
O 16	1.34E+05	1.34E+05	1.34E+05	1.34E+05	1.34E+05	1.34E+05
O 17	5.44E+01	5.44E+01	5.44E+01	5.44E+01	5.44E+01	5.44E+01
O 18	3.09E+02	3.09E+02	3.09E+02	3.09E+02	3.09E+02	3.09E+02
F 19	1.07E+01	1.07E+01	1.07E+01	1.07E+01	1.07E+01	1.07E+01
NE 20	2.41E-04	4.08E-04	5.97E-04	2.41E-04	4.08E-04	5.97E-04
NE 21	7.35E-06	1.49E-05	2.19E-05	7.35E-06	1.49E-05	2.19E-05
NE 22	1.10E-05	2.22E-05	3.25E-05	1.10E-05	2.22E-05	3.25E-05
NA 23	1.50E+01	1.50E+01	1.50E+01	1.50E+01	1.50E+01	1.50E+01
MG 24	1.58E+00	1.59E+00	1.61E+00	1.58E+00	1.59E+00	1.61E+00
MG 25	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01
MG 26	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01
AL 27	1.02E+02	1.01E+02	1.01E+02	1.02E+02	1.01E+02	1.01E+02
SI 28	3.48E+02	3.48E+02	3.48E+02	3.48E+02	3.48E+02	3.48E+02
SI 29	1.83E+01	1.83E+01	1.83E+01	1.83E+01	1.83E+01	1.83E+01
SI 30	1.25E+01	1.25E+01	1.25E+01	1.25E+01	1.25E+01	1.25E+01
SI 32	2.80E-09	4.37E-09	6.45E-09	1.83E-09	2.85E-09	4.21E-09
P 31	1.85E+02	1.85E+02	1.84E+02	1.85E+02	1.85E+02	1.84E+02
P 32	1.69E-13	2.63E-13	3.89E-13	1.10E-13	1.72E-13	2.54E-13

Table C-2 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000	20000	40000	60000
Enrichment (%)	2.44	3.72	4.73	2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
S 32	1.92E+01	1.92E+01	1.92E+01	1.92E+01	1.92E+01	1.92E+01
S 33	1.64E-01	1.69E-01	1.75E-01	1.64E-01	1.69E-01	1.75E-01
S 34	9.03E-01	9.03E-01	9.03E-01	9.03E-01	9.03E-01	9.03E-01
S 36	3.86E-03	3.86E-03	3.86E-03	3.86E-03	3.87E-03	3.87E-03
CL 35	3.73E+00	3.62E+00	3.47E+00	3.73E+00	3.62E+00	3.47E+00
CL 36	2.35E-01	3.52E-01	5.01E-01	2.35E-01	3.51E-01	5.01E-01
CL 37	1.34E+00	1.34E+00	1.35E+00	1.34E+00	1.34E+00	1.35E+00
AR 36	5.36E-05	8.09E-05	1.16E-04	2.66E-04	3.99E-04	5.69E-04
AR 38	8.91E-04	1.39E-03	2.04E-03	8.91E-04	1.39E-03	2.04E-03
AR 39	8.70E-07	1.59E-06	2.30E-06	3.10E-07	5.68E-07	8.20E-07
AR 40	1.55E-06	4.58E-06	9.59E-06	1.55E-06	4.58E-06	9.59E-06
K 39	0.00E+00	4.78E-07	6.95E-07	0.00E+00	1.50E-06	2.17E-06
K 40	3.16E-04	6.31E-04	9.07E-04	3.16E-04	6.31E-04	9.07E-04
K 41	7.89E-06	2.27E-05	4.71E-05	1.17E-05	2.86E-05	5.56E-05
CA 40	1.93E+00	1.93E+00	1.93E+00	1.93E+00	1.93E+00	1.93E+00
CA 41	1.12E-03	1.71E-03	2.50E-03	1.11E-03	1.71E-03	2.50E-03
CA 42	1.35E-02	1.35E-02	1.35E-02	1.35E-02	1.35E-02	1.35E-02
CA 43	2.79E-03	2.79E-03	2.80E-03	2.79E-03	2.79E-03	2.80E-03
CA 44	4.59E-02	4.60E-02	4.61E-02	4.59E-02	4.60E-02	4.61E-02
CA 46	9.40E-05	1.09E-04	1.24E-04	9.40E-05	1.09E-04	1.24E-04
CA 48	4.54E-03	4.54E-03	4.53E-03	4.54E-03	4.54E-03	4.53E-03
SC 45	6.98E-05	1.08E-04	1.56E-04	6.98E-05	1.08E-04	1.56E-04
TI 46	8.78E+00	8.78E+00	8.77E+00	8.78E+00	8.78E+00	8.77E+00
TI 47	8.10E+00	8.09E+00	8.08E+00	8.10E+00	8.09E+00	8.08E+00
TI 48	8.12E+01	8.09E+01	8.04E+01	8.12E+01	8.09E+01	8.04E+01
TI 49	6.83E+00	7.20E+00	7.68E+00	6.83E+00	7.20E+00	7.68E+00
TI 50	6.04E+00	6.05E+00	6.07E+00	6.04E+00	6.05E+00	6.07E+00
V 50	3.07E-02	4.19E-02	5.10E-02	3.07E-02	4.19E-02	5.10E-02
V 51	1.15E+01	1.32E+01	1.55E+01	1.15E+01	1.32E+01	1.55E+01
CR 50	3.90E+02	3.88E+02	3.86E+02	3.90E+02	3.88E+02	3.86E+02
CR 52	7.87E+03	7.87E+03	7.86E+03	7.87E+03	7.87E+03	7.86E+03
CR 53	9.11E+02	9.11E+02	9.12E+02	9.11E+02	9.11E+02	9.12E+02
CR 54	2.40E+02	2.45E+02	2.51E+02	2.40E+02	2.45E+02	2.51E+02
MN 54	3.08E-37	4.05E-37	4.86E-37	0.00E+00	0.00E+00	0.00E+00
MN 55	7.11E+02	7.09E+02	7.08E+02	7.11E+02	7.09E+02	7.08E+02

Table C-2 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000	20000	40000	60000
Enrichment (%)	2.44	3.72	4.73	2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
FE 54	1.49E+03	1.49E+03	1.49E+03	1.49E+03	1.49E+03	1.49E+03
FE 55	2.82E-12	3.61E-12	4.69E-12	2.45E+04	2.45E+04	2.45E+04
FE 56	2.45E+04	2.45E+04	2.45E+04	0.00E+00	0.00E+00	0.00E+00
FE 57	6.08E+02	6.20E+02	6.37E+02	6.08E+02	6.20E+02	6.37E+02
FE 58	8.17E+01	8.31E+01	8.44E+01	8.17E+01	8.31E+01	8.44E+01
CO 59	8.69E+01	8.34E+01	7.97E+01	8.70E+01	8.35E+01	7.99E+01
CO 60	1.13E-05	1.64E-05	2.14E-05	1.60E-28	2.32E-28	3.03E-28
NI 58	7.28E+03	7.26E+03	7.24E+03	7.28E+03	7.26E+03	7.24E+03
NI 59	2.51E+01	3.69E+01	5.08E+01	2.50E+01	3.67E+01	5.06E+01
NI 60	2.89E+03	2.89E+03	2.90E+03	2.89E+03	2.89E+03	2.90E+03
NI 61	1.34E+02	1.38E+02	1.43E+02	1.34E+02	1.38E+02	1.43E+02
NI 62	4.06E+02	4.03E+02	4.00E+02	4.06E+02	4.03E+02	4.00E+02
NI 63	2.36E+00	3.55E+00	5.09E+00	1.16E-01	1.74E-01	2.50E-01
NI 64	1.07E+02	1.07E+02	1.08E+02	1.07E+02	1.07E+02	1.08E+02
CU 63	1.58E+01	1.71E+01	1.89E+01	1.80E+01	2.05E+01	2.38E+01
CU 65	6.21E+00	6.30E+00	6.42E+00	6.21E+00	6.30E+00	6.42E+00
ZN 64	1.92E+01	1.92E+01	1.92E+01	1.92E+01	1.92E+01	1.92E+01
ZN 66	1.13E+01	1.13E+01	1.13E+01	1.13E+01	1.13E+01	1.13E+01
ZN 67	1.68E+00	1.67E+00	1.66E+00	1.68E+00	1.67E+00	1.66E+00
ZN 68	7.87E+00	7.88E+00	7.88E+00	7.87E+00	7.88E+00	7.88E+00
ZN 70	2.70E-01	2.73E-01	2.76E-01	2.70E-01	2.73E-01	2.76E-01
GA 69	2.04E-02	3.46E-02	5.05E-02	2.04E-02	3.46E-02	5.05E-02
GA 71	3.57E-05	5.53E-05	8.20E-05	3.57E-05	5.53E-05	8.20E-05
GE 70	8.22E-05	2.53E-04	5.40E-04	8.22E-05	2.53E-04	5.40E-04
GE 72	1.32E-02	2.64E-02	3.89E-02	1.32E-02	2.64E-02	3.89E-02
GE 73	2.77E-02	5.39E-02	7.75E-02	2.77E-02	5.39E-02	7.75E-02
GE 74	5.89E-02	1.18E-01	1.76E-01	5.89E-02	1.18E-01	1.76E-01
GE 76	3.06E-01	6.01E-01	8.86E-01	3.06E-01	6.01E-01	8.86E-01
AS 75	1.24E-01	2.41E-01	3.49E-01	1.24E-01	2.41E-01	3.49E-01
SE 76	2.28E-03	8.48E-03	1.81E-02	2.28E-03	8.48E-03	1.81E-02
SE 77	6.40E-01	1.23E+00	1.76E+00	6.40E-01	1.23E+00	1.76E+00
SE 78	1.46E+00	2.92E+00	4.39E+00	1.46E+00	2.92E+00	4.39E+00
SE 79	3.59E+00	7.07E+00	1.05E+01	3.58E+00	7.04E+00	1.04E+01
SE 80	8.18E+00	1.61E+01	2.39E+01	8.18E+00	1.61E+01	2.39E+01
SE 82	2.07E+01	4.04E+01	5.98E+01	2.07E+01	4.04E+01	5.98E+01

Table C-2 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000	20000	40000	60000
Enrichment (%)	2.44	3.72	4.73	2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
BR 79	3.88E-03	7.70E-03	1.15E-02	1.92E-02	3.78E-02	5.61E-02
BR 81	1.35E+01	2.58E+01	3.72E+01	1.35E+01	2.58E+01	3.72E+01
KR 80	1.34E-04	2.75E-04	4.25E-04	1.34E-04	2.75E-04	4.25E-04
KR 81	1.08E-05	2.94E-05	5.85E-05	1.07E-05	2.93E-05	5.84E-05
KR 82	4.19E-01	1.42E+00	2.92E+00	4.19E-01	1.42E+00	2.92E+00
KR 83	2.79E+01	4.88E+01	6.21E+01	2.79E+01	4.88E+01	6.21E+01
KR 84	6.69E+01	1.37E+02	2.12E+02	6.69E+01	1.37E+02	2.12E+02
KR 85	2.35E-02	4.28E-02	6.01E-02	1.38E-13	2.51E-13	3.52E-13
KR 86	1.17E+02	2.28E+02	3.36E+02	1.17E+02	2.28E+02	3.36E+02
RB 85	7.48E+01	1.46E+02	2.15E+02	7.48E+01	1.46E+02	2.15E+02
RB 87	1.50E+02	2.92E+02	4.30E+02	1.50E+02	2.92E+02	4.30E+02
SR 86	1.66E-01	6.42E-01	1.45E+00	1.66E-01	6.42E-01	1.45E+00
SR 87	2.76E-03	7.88E-03	1.78E-02	2.76E-03	7.88E-03	1.78E-02
SR 88	2.16E+02	4.19E+02	6.18E+02	2.16E+02	4.19E+02	6.18E+02
SR 90	3.08E+01	5.82E+01	8.39E+01	2.26E-03	4.27E-03	6.15E-03
Y 89	2.82E+02	5.46E+02	8.04E+02	2.82E+02	5.46E+02	8.04E+02
Y 90	7.73E-03	1.46E-02	2.10E-02	5.67E-07	1.07E-06	1.54E-06
ZR 90	1.28E+05	1.28E+05	1.28E+05	1.28E+05	1.28E+05	1.29E+05
ZR 91	2.84E+04	2.86E+04	2.89E+04	2.84E+04	2.86E+04	2.89E+04
ZR 92	4.38E+04	4.42E+04	4.46E+04	4.38E+04	4.42E+04	4.46E+04
ZR 93	5.19E+02	1.00E+03	1.47E+03	5.19E+02	1.00E+03	1.47E+03
ZR 94	4.55E+04	4.59E+04	4.63E+04	4.55E+04	4.59E+04	4.63E+04
ZR 96	7.87E+03	8.32E+03	8.77E+03	7.87E+03	8.32E+03	8.77E+03
NB 93	7.22E+02	7.18E+02	7.14E+02	7.22E+02	7.18E+02	7.15E+02
NB 93M	4.36E-03	8.41E-03	1.23E-02	4.39E-03	8.46E-03	1.24E-02
NB 94	3.67E+00	6.73E+00	9.55E+00	3.62E+00	6.64E+00	9.42E+00
MO 92	5.71E+01	5.71E+01	5.70E+01	5.71E+01	5.71E+01	5.70E+01
MO 93	1.13E-02	2.06E-02	3.02E-02	1.04E-02	1.91E-02	2.79E-02
MO 94	3.66E+01	3.66E+01	3.66E+01	3.67E+01	3.67E+01	3.68E+01
MO 95	5.45E+02	9.81E+02	1.38E+03	5.45E+02	9.81E+02	1.38E+03
MO 96	8.23E+01	1.28E+02	2.07E+02	8.23E+01	1.28E+02	2.07E+02
MO 97	5.43E+02	1.04E+03	1.52E+03	5.43E+02	1.04E+03	1.52E+03
MO 98	5.93E+02	1.08E+03	1.55E+03	5.93E+02	1.08E+03	1.55E+03
MO100	6.04E+02	1.16E+03	1.72E+03	6.04E+02	1.16E+03	1.72E+03

Table C-2 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000	20000	40000	60000
Enrichment (%)	2.44	3.72	4.73	2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
TC 98	1.79E-03	7.32E-03	1.63E-02	1.79E-03	7.31E-03	1.63E-02
TC 99	4.90E+02	9.15E+02	1.28E+03	4.89E+02	9.14E+02	1.28E+03
RU 99	1.61E-01	3.04E-01	4.28E-01	7.98E-01	1.50E+00	2.09E+00
RU100	3.94E+01	1.48E+02	3.29E+02	3.94E+01	1.48E+02	3.29E+02
RU101	4.72E+02	9.24E+02	1.35E+03	4.72E+02	9.24E+02	1.35E+03
RU102	4.58E+02	9.45E+02	1.45E+03	4.58E+02	9.45E+02	1.45E+03
RU104	3.20E+02	6.61E+02	1.01E+03	3.20E+02	6.61E+02	1.01E+03
RU106	1.55E-28	2.29E-28	2.85E-28	0.00E+00	0.00E+00	0.00E+00
RH102	1.28E-14	4.50E-14	8.29E-14	3.84E-56	1.36E-55	2.50E-55
RH103	3.18E+02	5.09E+02	6.02E+02	3.18E+02	5.09E+02	6.02E+02
RH106	1.45E-34	2.15E-34	2.68E-34	0.00E+00	0.00E+00	0.00E+00
PD104	9.14E+01	3.24E+02	6.57E+02	9.14E+01	3.24E+02	6.57E+02
PD105	2.24E+02	4.66E+02	7.01E+02	2.24E+02	4.66E+02	7.01E+02
PD106	2.04E+02	4.41E+02	7.01E+02	2.04E+02	4.41E+02	7.01E+02
PD107	1.26E+02	2.69E+02	4.13E+02	1.26E+02	2.69E+02	4.13E+02
PD108	8.61E+01	1.85E+02	2.83E+02	8.61E+01	1.85E+02	2.83E+02
PD110	2.81E+01	6.12E+01	9.51E+01	2.81E+01	6.12E+01	9.51E+01
AG107	4.79E-02	4.61E-02	4.44E-02	5.33E-02	5.76E-02	6.20E-02
AG108	5.92E-13	9.52E-13	1.34E-12	6.67E-14	1.07E-13	1.51E-13
AG108M	1.87E-04	3.01E-04	4.23E-04	2.11E-05	3.40E-05	4.77E-05
AG109	4.71E+01	8.87E+01	1.21E+02	4.71E+01	8.87E+01	1.21E+02
AG109M	8.49E-34	1.05E-33	1.19E-33	0.00E+00	0.00E+00	0.00E+00
CD106	3.07E-01	3.06E-01	3.06E-01	3.07E-01	3.06E-01	3.06E-01
CD108	2.18E-01	2.21E-01	2.24E-01	2.18E-01	2.21E-01	2.24E-01
CD109	8.60E-28	1.06E-27	1.21E-27	0.00E+00	0.00E+00	0.00E+00
CD110	1.65E+01	5.42E+01	1.13E+02	1.65E+01	5.42E+01	1.13E+02
CD111	1.91E+01	3.74E+01	5.69E+01	1.91E+01	3.74E+01	5.69E+01
CD112	1.56E+01	2.67E+01	3.84E+01	1.56E+01	2.67E+01	3.84E+01
CD113	1.45E-01	1.89E-01	1.87E-01	1.45E-01	1.89E-01	1.87E-01
CD113M	1.24E-03	2.85E-03	4.78E-03	6.91E-12	1.59E-11	2.67E-11
CD114	2.36E+01	3.77E+01	5.13E+01	2.36E+01	3.77E+01	5.13E+01
CD116	6.91E+00	1.20E+01	1.67E+01	6.91E+00	1.20E+01	1.67E+01
IN113	6.22E-01	1.11E+00	1.59E+00	6.24E-01	1.11E+00	1.59E+00
IN115	2.23E+00	2.47E+00	2.47E+00	2.23E+00	2.47E+00	2.47E+00
SN112	3.84E+01	3.80E+01	3.76E+01	3.84E+01	3.80E+01	3.76E+01

Table C-2 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000	20000	40000	60000
Enrichment (%)	2.44	3.72	4.73	2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
SN114	2.65E+01	2.66E+01	2.67E+01	2.65E+01	2.66E+01	2.67E+01
SN115	1.43E+01	1.40E+01	1.36E+01	1.43E+01	1.40E+01	1.36E+01
SN116	5.91E+02	5.92E+02	5.94E+02	5.91E+02	5.92E+02	5.94E+02
SN117	3.21E+02	3.28E+02	3.35E+02	3.21E+02	3.28E+02	3.35E+02
SN118	9.96E+02	9.98E+02	1.00E+03	9.96E+02	9.98E+02	1.00E+03
SN119	3.64E+02	3.74E+02	3.83E+02	3.64E+02	3.74E+02	3.83E+02
SN120	1.35E+03	1.36E+03	1.36E+03	1.35E+03	1.36E+03	1.36E+03
SN121M	2.22E-03	4.12E-03	6.04E-03	8.64E-06	1.60E-05	2.35E-05
SN122	2.00E+02	2.06E+02	2.11E+02	2.00E+02	2.06E+02	2.11E+02
SN124	2.47E+02	2.53E+02	2.59E+02	2.47E+02	2.53E+02	2.59E+02
SN126	1.66E+01	3.34E+01	4.99E+01	1.65E+01	3.33E+01	4.98E+01
SB121	6.38E+00	1.22E+01	1.72E+01	6.38E+00	1.22E+01	1.72E+01
SB123	6.43E+00	1.25E+01	1.80E+01	6.43E+00	1.25E+01	1.80E+01
SB125	1.43E-10	2.34E-10	3.05E-10	0.00E+00	0.00E+00	0.00E+00
SB126	7.88E-07	1.59E-06	2.37E-06	7.86E-07	1.58E-06	2.36E-06
SB126M	5.99E-09	1.21E-08	1.80E-08	5.97E-09	1.20E-08	1.80E-08
TE122	2.85E-01	1.06E+00	2.23E+00	2.85E-01	1.06E+00	2.23E+00
TE123	3.18E-03	1.69E-02	4.15E-02	3.18E-03	1.69E-02	4.15E-02
TE124	1.67E-01	6.74E-01	1.51E+00	1.67E-01	6.74E-01	1.51E+00
TE125	1.30E+01	2.59E+01	3.83E+01	1.30E+01	2.59E+01	3.83E+01
TE125M	2.00E-12	3.28E-12	4.27E-12	0.00E+00	0.00E+00	0.00E+00
TE126	4.42E-01	1.05E+00	1.82E+00	4.87E-01	1.14E+00	1.95E+00
TE128	6.66E+01	1.34E+02	2.01E+02	6.66E+01	1.34E+02	2.01E+02
TE130	2.14E+02	4.29E+02	6.44E+02	2.14E+02	4.29E+02	6.44E+02
I127	3.41E+01	6.63E+01	9.59E+01	3.41E+01	6.63E+01	9.59E+01
I129	1.10E+02	2.16E+02	3.13E+02	1.10E+02	2.16E+02	3.13E+02
XE128	1.17E+00	4.75E+00	1.07E+01	1.17E+00	4.75E+00	1.07E+01
XE129	4.72E-03	3.12E-02	1.01E-01	6.67E-03	3.50E-02	1.07E-01
XE130	4.64E+00	1.56E+01	3.48E+01	4.64E+00	1.56E+01	3.48E+01
XE131	2.95E+02	4.77E+02	5.73E+02	2.95E+02	4.77E+02	5.73E+02
XE132	6.18E+02	1.35E+03	2.16E+03	6.18E+02	1.35E+03	2.16E+03
XE134	8.90E+02	1.77E+03	2.65E+03	8.90E+02	1.77E+03	2.65E+03
XE136	1.40E+03	2.69E+03	4.01E+03	1.40E+03	2.69E+03	4.01E+03
CS133	7.21E+02	1.31E+03	1.78E+03	7.21E+02	1.31E+03	1.78E+03
CS134	1.38E-13	4.16E-13	7.72E-13	0.00E+00	0.00E+00	0.00E+00

Table C-2 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000	20000	40000	60000
Enrichment (%)	2.44	3.72	4.73	2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
CS135	1.89E+02	4.86E+02	7.72E+02	1.89E+02	4.86E+02	7.71E+02
CS137	7.24E+01	1.41E+02	2.07E+02	7.01E-03	1.37E-02	2.01E-02
BA132	5.38E-04	2.15E-03	4.70E-03	5.38E-04	2.15E-03	4.70E-03
BA134	6.66E+01	2.39E+02	5.01E+02	6.66E+01	2.39E+02	5.01E+02
BA135	5.64E-02	6.71E-01	2.69E+00	7.92E-02	7.29E-01	2.78E+00
BA136	8.77E+00	2.98E+01	6.49E+01	8.77E+00	2.98E+01	6.49E+01
BA137	6.73E+02	1.35E+03	2.02E+03	7.46E+02	1.49E+03	2.23E+03
BA137M	1.11E-05	2.16E-05	3.17E-05	1.07E-09	2.09E-09	3.07E-09
BA138	7.74E+02	1.54E+03	2.30E+03	7.74E+02	1.54E+03	2.30E+03
LA138	3.97E-03	6.60E-03	8.07E-03	3.97E-03	6.60E-03	8.07E-03
LA139	7.42E+02	1.47E+03	2.17E+03	7.42E+02	1.47E+03	2.17E+03
CE140	7.46E+02	1.49E+03	2.23E+03	7.46E+02	1.49E+03	2.23E+03
CE142	6.85E+02	1.36E+03	2.02E+03	6.85E+02	1.36E+03	2.02E+03
CE144	6.11E-37	6.86E-37	7.19E-37	0.00E+00	0.00E+00	0.00E+00
PR141	6.84E+02	1.34E+03	1.98E+03	6.84E+02	1.34E+03	1.98E+03
PR144	2.58E-41	2.90E-41	3.03E-41	0.00E+00	0.00E+00	0.00E+00
PR144M	1.29E-43	1.45E-43	1.52E-43	0.00E+00	0.00E+00	0.00E+00
ND142	9.40E+00	3.47E+01	8.08E+01	9.40E+00	3.47E+01	8.08E+01
ND143	5.44E+02	9.40E+02	1.16E+03	5.44E+02	9.40E+02	1.16E+03
ND144	7.40E+02	1.59E+03	2.60E+03	7.40E+02	1.59E+03	2.60E+03
ND145	4.26E+02	7.93E+02	1.10E+03	4.26E+02	7.93E+02	1.10E+03
ND146	3.98E+02	8.45E+02	1.35E+03	3.98E+02	8.45E+02	1.35E+03
ND148	2.25E+02	4.45E+02	6.61E+02	2.25E+02	4.45E+02	6.61E+02
ND150	1.06E+02	2.14E+02	3.20E+02	1.06E+02	2.14E+02	3.20E+02
PM146	8.84E-09	2.27E-08	3.54E-08	1.13E-30	2.91E-30	4.53E-30
PM147	4.21E-10	4.63E-10	4.37E-10	0.00E+00	0.00E+00	0.00E+00
SM146	2.30E-03	1.07E-02	2.36E-02	2.30E-03	1.07E-02	2.36E-02
SM147	1.60E+02	2.23E+02	2.42E+02	1.60E+02	2.23E+02	2.42E+02
SM148	8.13E+01	2.32E+02	4.09E+02	8.13E+01	2.32E+02	4.09E+02
SM149	3.37E+00	4.11E+00	4.06E+00	3.37E+00	4.11E+00	4.06E+00
SM150	1.86E+02	3.69E+02	5.26E+02	1.86E+02	3.69E+02	5.26E+02
SM151	5.27E+00	8.67E+00	1.04E+01	2.42E-01	3.98E-01	4.79E-01
SM152	8.80E+01	1.45E+02	1.83E+02	8.80E+01	1.45E+02	1.83E+02
SM154	2.19E+01	4.55E+01	6.96E+01	2.19E+01	4.55E+01	6.96E+01
EU150	1.30E-08	5.91E-08	1.06E-07	5.89E-12	2.67E-11	4.79E-11

Table C-2 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000	20000	40000	60000
Enrichment (%)	2.44	3.72	4.73	2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
EU151	6.12E+00	1.01E+01	1.21E+01	1.12E+01	1.84E+01	2.21E+01
EU152	1.76E-04	3.55E-04	3.88E-04	2.47E-13	4.97E-13	5.44E-13
EU153	6.50E+01	1.50E+02	2.27E+02	6.50E+01	1.50E+02	2.27E+02
EU154	4.56E-03	1.49E-02	2.68E-02	4.55E-17	1.49E-16	2.67E-16
EU155	5.27E-06	1.57E-05	2.82E-05	2.76E-30	8.22E-30	1.48E-29
GD152	3.29E-02	8.68E-02	1.33E-01	3.30E-02	8.69E-02	1.33E-01
GD154	1.51E+01	5.15E+01	9.56E+01	1.51E+01	5.15E+01	9.57E+01
GD155	6.23E+00	1.86E+01	3.34E+01	6.23E+00	1.86E+01	3.34E+01
GD156	2.57E+01	8.67E+01	2.12E+02	2.57E+01	8.67E+01	2.12E+02
GD157	7.62E-02	1.51E-01	2.50E-01	7.62E-02	1.51E-01	2.50E-01
GD158	1.05E+01	2.45E+01	4.74E+01	1.05E+01	2.45E+01	4.74E+01
GD160	1.27E+00	2.11E+00	2.97E+00	1.27E+00	2.11E+00	2.97E+00
TB159	1.50E+00	3.25E+00	5.17E+00	1.50E+00	3.25E+00	5.17E+00
DY160	1.18E-01	4.24E-01	8.55E-01	1.18E-01	4.24E-01	8.55E-01
DY161	2.64E-01	5.36E-01	8.34E-01	2.64E-01	5.36E-01	8.34E-01
DY162	2.08E-01	4.31E-01	6.74E-01	2.08E-01	4.31E-01	6.74E-01
DY163	1.42E-01	3.85E-01	6.98E-01	1.42E-01	3.85E-01	6.98E-01
DY164	3.17E-02	9.49E-02	1.76E-01	3.17E-02	9.49E-02	1.76E-01
HO165	5.64E-02	1.88E-01	4.39E-01	5.64E-02	1.88E-01	4.39E-01
HO166M	3.47E-04	1.93E-03	6.32E-03	2.75E-04	1.53E-03	5.02E-03
ER166	1.66E-02	5.66E-02	1.47E-01	1.67E-02	5.70E-02	1.49E-01
ER167	2.58E-03	4.38E-03	6.95E-03	2.58E-03	4.38E-03	6.95E-03
ER168	2.60E-03	7.87E-03	1.62E-02	2.60E-03	7.87E-03	1.62E-02
TM169	1.22E-05	6.04E-05	1.57E-04	1.22E-05	6.04E-05	1.57E-04
TM171	1.55E-23	1.82E-22	8.37E-22	0.00E+00	0.00E+00	0.00E+00
YB170	2.20E-06	2.14E-05	8.10E-05	2.20E-06	2.14E-05	8.10E-05
YB171	0.00E+00	1.60E-06	8.50E-06	0.00E+00	1.60E-06	8.50E-06
LU175	1.27E-02	1.43E-02	1.41E-02	1.27E-02	1.43E-02	1.41E-02
LU176	2.77E-04	5.12E-04	5.76E-04	2.77E-04	5.12E-04	5.76E-04
HF174	1.66E-02	1.15E-02	7.33E-03	1.66E-02	1.15E-02	7.33E-03
HF176	7.63E-01	5.93E-01	4.63E-01	7.63E-01	5.93E-01	4.63E-01
HF177	3.03E-01	1.17E-01	7.63E-02	3.03E-01	1.17E-01	7.63E-02
HF178	4.53E+00	2.51E+00	1.43E+00	4.53E+00	2.51E+00	1.43E+00
HF179	6.09E+00	6.91E+00	6.58E+00	6.09E+00	6.91E+00	6.58E+00
HF180	8.15E+00	9.52E+00	1.09E+01	8.15E+00	9.52E+00	1.09E+01

Table C-2 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000	20000	40000	60000
Enrichment (%)	2.44	3.72	4.73	2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
HF182	1.10E-03	1.67E-03	2.85E-03	1.10E-03	1.67E-03	2.85E-03
TA181	2.09E-01	3.46E-01	4.98E-01	2.09E-01	3.46E-01	4.98E-01
TA182	3.85E-11	5.83E-11	9.97E-11	3.85E-11	5.83E-11	9.97E-11
W180	8.56E-03	7.98E-03	7.45E-03	8.56E-03	7.98E-03	7.45E-03
W182	1.50E+00	1.23E+00	1.03E+00	1.50E+00	1.23E+00	1.03E+00
W183	1.30E+00	1.45E+00	1.57E+00	1.30E+00	1.45E+00	1.57E+00
W184	2.38E+00	2.54E+00	2.71E+00	2.38E+00	2.54E+00	2.71E+00
W186	1.67E+00	1.38E+00	1.14E+00	1.67E+00	1.38E+00	1.14E+00
RE185	1.31E-02	1.87E-02	2.31E-02	1.31E-02	1.87E-02	2.31E-02
RE187	3.88E-01	6.02E-01	7.36E-01	3.88E-01	6.02E-01	7.36E-01
OS186	3.66E-03	1.20E-02	2.31E-02	3.66E-03	1.20E-02	2.31E-02
OS187	5.41E-10	8.46E-10	1.04E-09	2.69E-09	4.18E-09	5.11E-09
OS188	4.30E-02	1.19E-01	2.22E-01	4.30E-02	1.19E-01	2.22E-01
OS189	7.76E-04	3.94E-03	1.04E-02	7.76E-04	3.94E-03	1.04E-02
OS190	6.28E-05	6.23E-04	2.47E-03	6.28E-05	6.23E-04	2.47E-03
IR191	0.00E+00	3.66E-06	1.75E-05	0.00E+00	3.66E-06	1.75E-05
IR192	2.35E-14	6.89E-13	5.03E-12	7.44E-15	2.18E-13	1.59E-12
IR192M	0.00E+00	0.00E+00	5.98E-09	0.00E+00	0.00E+00	1.89E-09
IR193	0.00E+00	0.00E+00	2.48E-06	0.00E+00	0.00E+00	2.56E-06
PT192	0.00E+00	2.19E-06	1.54E-05	0.00E+00	2.19E-06	1.54E-05
PT193	0.00E+00	1.61E-08	1.91E-07	0.00E+00	9.25E-09	1.09E-07
TL206	4.97E-17	8.79E-17	1.29E-16	4.97E-17	8.79E-17	1.29E-16
TL207	1.90E-13	2.65E-13	2.96E-13	1.08E-12	1.07E-12	8.18E-13
TL208	4.57E-12	2.54E-11	6.24E-11	9.73E-14	5.42E-13	1.33E-12
TL209	2.54E-17	7.94E-17	1.67E-16	8.23E-16	1.93E-15	3.00E-15
PB204	1.38E-02	1.38E-02	1.37E-02	1.38E-02	1.38E-02	1.37E-02
PB205	2.02E-05	3.39E-05	4.96E-05	2.02E-05	3.39E-05	4.96E-05
PB206	2.31E-01	2.27E-01	2.21E-01	2.31E-01	2.27E-01	2.21E-01
PB207	2.30E-01	2.34E-01	2.39E-01	2.30E-01	2.34E-01	2.39E-01
PB208	5.26E-01	5.28E-01	5.30E-01	5.27E-01	5.29E-01	5.32E-01
PB209	1.06E-13	3.31E-13	6.95E-13	3.43E-12	8.04E-12	1.25E-11
PB210	1.51E-07	2.16E-07	2.75E-07	6.18E-06	1.13E-05	1.81E-05
PB211	1.47E-12	2.05E-12	2.29E-12	8.38E-12	8.28E-12	6.33E-12
PB212	2.69E-09	1.50E-08	3.68E-08	5.74E-11	3.19E-10	7.83E-10
PB214	6.33E-13	9.23E-13	1.20E-12	1.64E-11	3.00E-11	4.87E-11

Table C-2 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000	20000	40000	60000
Enrichment (%)	2.44	3.72	4.73	2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
BI208	2.69E-06	5.43E-06	7.95E-06	2.69E-06	5.43E-06	7.94E-06
BI209	4.00E-01	4.00E-01	4.00E-01	4.00E-01	4.00E-01	4.00E-01
BI210	9.28E-11	1.33E-10	1.69E-10	3.80E-09	6.92E-09	1.12E-08
BI210M	1.92E-05	3.39E-05	4.96E-05	1.91E-05	3.39E-05	4.96E-05
BI211	8.67E-14	1.21E-13	1.35E-13	4.94E-13	4.88E-13	3.73E-13
BI212	2.56E-10	1.42E-09	3.49E-09	5.44E-12	3.03E-11	7.43E-11
BI213	2.48E-14	7.78E-14	1.63E-13	8.06E-13	1.89E-12	2.94E-12
BI214	4.70E-13	6.86E-13	8.94E-13	1.22E-11	2.23E-11	3.62E-11
PO210	2.56E-09	3.68E-09	4.67E-09	1.05E-07	1.91E-07	3.08E-07
PO211	1.06E-18	1.48E-18	1.66E-18	6.07E-18	5.99E-18	4.58E-18
PO212	1.35E-20	7.53E-20	1.85E-19	2.88E-22	1.60E-21	3.93E-21
PO213	3.73E-23	1.17E-22	2.45E-22	1.21E-21	2.83E-21	4.40E-21
PO214	6.46E-20	9.43E-20	1.23E-19	1.67E-18	3.07E-18	4.98E-18
PO215	1.23E-18	1.72E-18	1.92E-18	7.02E-18	6.93E-18	5.30E-18
PO216	1.08E-14	5.99E-14	1.47E-13	2.29E-16	1.28E-15	3.12E-15
PO218	7.34E-14	1.07E-13	1.40E-13	1.90E-12	3.48E-12	5.65E-12
AT217	2.98E-19	9.34E-19	1.96E-18	9.68E-18	2.27E-17	3.53E-17
RN219	2.79E-15	3.89E-15	4.35E-15	1.59E-14	1.57E-14	1.20E-14
RN220	4.06E-12	2.26E-11	5.54E-11	8.64E-14	4.81E-13	1.18E-12
RN222	1.35E-10	1.97E-10	2.57E-10	3.49E-09	6.40E-09	1.04E-08
FR221	2.71E-15	8.48E-15	1.78E-14	8.79E-14	2.06E-13	3.20E-13
FR223	1.29E-14	1.80E-14	2.02E-14	7.38E-14	7.29E-14	5.57E-14
RA223	7.08E-10	9.87E-10	1.10E-09	4.04E-09	3.99E-09	3.05E-09
RA224	2.35E-08	1.31E-07	3.21E-07	5.00E-10	2.79E-09	6.83E-09
RA225	1.23E-11	3.84E-11	8.05E-11	3.98E-10	9.32E-10	1.45E-09
RA226	2.10E-05	3.06E-05	3.99E-05	5.43E-04	9.96E-04	1.62E-03
RA228	3.21E-12	5.97E-12	8.01E-12	1.78E-11	3.27E-11	4.35E-11
AC225	8.27E-12	2.59E-11	5.44E-11	2.69E-10	6.29E-10	9.78E-10
AC227	5.01E-07	6.99E-07	7.82E-07	2.86E-06	2.82E-06	2.16E-06
AC228	3.35E-16	6.23E-16	8.36E-16	1.86E-15	3.41E-15	4.54E-15
TH227	1.16E-09	1.62E-09	1.82E-09	6.64E-09	6.55E-09	5.01E-09
TH228	4.57E-06	2.54E-05	6.24E-05	9.73E-08	5.42E-07	1.33E-06
TH229	2.26E-06	7.07E-06	1.48E-05	7.33E-05	1.72E-04	2.67E-04
TH230	4.86E-02	7.48E-02	1.04E-01	2.72E-01	5.28E-01	8.89E-01
TH231	3.81E-08	3.42E-08	2.19E-08	3.83E-08	3.45E-08	2.23E-08

Table C-2 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000	20000	40000	60000
Enrichment (%)	2.44	3.72	4.73	2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
TH232	7.58E-03	1.41E-02	1.89E-02	3.80E-02	6.98E-02	9.29E-02
TH234	1.39E-05	1.36E-05	1.32E-05	1.39E-05	1.36E-05	1.32E-05
PA231	1.05E-03	1.34E-03	1.39E-03	4.66E-03	4.58E-03	3.47E-03
PA233	1.18E-05	2.95E-05	4.74E-05	2.54E-05	5.12E-05	7.06E-05
PA234	2.10E-10	2.04E-10	1.99E-10	2.10E-10	2.04E-10	1.99E-10
PA234M	4.70E-10	4.57E-10	4.46E-10	4.70E-10	4.57E-10	4.46E-10
U232	1.70E-04	9.48E-04	2.32E-03	3.63E-06	2.02E-05	4.95E-05
U233	9.87E-03	2.62E-02	4.33E-02	8.23E-02	1.82E-01	2.69E-01
U234	1.84E+02	3.24E+02	5.11E+02	2.07E+02	4.38E+02	7.79E+02
U235	9.37E+03	8.41E+03	5.39E+03	9.43E+03	8.48E+03	5.47E+03
U236	2.58E+03	4.73E+03	6.27E+03	2.64E+03	4.84E+03	6.44E+03
U237	2.40E-07	3.78E-07	4.02E-07	6.72E-12	1.18E-10	4.74E-10
U238	9.60E+05	9.33E+05	9.10E+05	9.60E+05	9.33E+05	9.10E+05
U240	1.16E-13	8.15E-13	2.41E-12	1.16E-13	8.15E-13	2.41E-12
NP235	2.53E-34	1.12E-33	2.26E-33	0.00E+00	0.00E+00	0.00E+00
NP236	1.38E-04	6.36E-04	1.25E-03	1.38E-04	6.35E-04	1.25E-03
NP237	3.49E+02	8.69E+02	1.40E+03	7.48E+02	1.51E+03	2.08E+03
NP238	9.11E-08	3.20E-07	3.96E-07	1.47E-08	5.16E-08	6.39E-08
NP239	2.55E-05	1.18E-04	2.31E-04	2.46E-05	1.14E-04	2.23E-04
NP240M	1.02E-15	7.13E-15	2.11E-14	1.02E-15	7.13E-15	2.11E-14
PU236	3.09E-10	1.42E-09	2.78E-09	3.08E-10	1.42E-09	2.78E-09
PU238	2.42E+01	1.21E+02	2.84E+02	1.09E+00	5.36E+00	1.23E+01
PU239	4.68E+03	6.38E+03	7.40E+03	4.63E+03	6.31E+03	7.33E+03
PU240	1.44E+03	2.56E+03	4.11E+03	1.38E+03	2.46E+03	3.94E+03
PU241	7.75E+00	1.22E+01	1.30E+01	2.17E-04	3.80E-03	1.53E-02
PU242	2.28E+02	5.65E+02	8.17E+02	2.28E+02	5.65E+02	8.17E+02
PU243	1.94E-15	1.17E-13	1.16E-12	1.94E-15	1.17E-13	1.16E-12
PU244	6.08E-03	4.26E-02	1.26E-01	6.08E-03	4.26E-02	1.26E-01
AM241	8.50E+02	1.36E+03	1.46E+03	4.52E+02	7.24E+02	7.74E+02
AM242	5.81E-06	2.04E-05	2.53E-05	9.38E-07	3.29E-06	4.08E-06
AM242M	4.86E-01	1.71E+00	2.11E+00	7.84E-02	2.75E-01	3.41E-01
AM243	2.97E+01	1.38E+02	2.69E+02	2.86E+01	1.32E+02	2.59E+02
AM244	0.00E+00	5.94E-17	1.76E-16	0.00E+00	5.94E-17	1.76E-16
CM242	1.18E-03	4.13E-03	5.11E-03	1.90E-04	6.66E-04	8.25E-04
CM243	8.96E-03	5.88E-02	1.14E-01	5.34E-07	3.51E-06	6.78E-06

Table C-2 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	20000	40000	60000	20000	40000	60000
Enrichment (%)	2.44	3.72	4.73	2.44	3.72	4.73
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
CM244	1.16E-01	1.14E+00	3.61E+00	2.59E-08	2.55E-07	8.10E-07
CM245	1.34E-01	2.35E+00	9.48E+00	1.30E-01	2.28E+00	9.18E+00
CM246	9.25E-03	2.77E-01	1.85E+00	8.72E-03	2.61E-01	1.75E+00
CM247	5.43E-05	3.27E-03	3.26E-02	5.43E-05	3.27E-03	3.26E-02
CM248	1.79E-06	2.10E-04	3.32E-03	1.79E-06	2.10E-04	3.32E-03
BK249	5.35E-43	1.02E-40	1.99E-39	0.00E+00	0.00E+00	0.00E+00
BK250	0.00E+00	0.00E+00	1.00E-19	0.00E+00	0.00E+00	9.86E-20
CF249	1.09E-08	2.19E-06	4.40E-05	4.96E-09	9.92E-07	1.99E-05
CF250	1.06E-11	2.45E-09	6.77E-08	4.98E-19	1.11E-16	3.55E-15
CF251	0.00E+00	2.05E-07	6.30E-06	0.00E+00	1.51E-07	4.62E-06
CF252	8.93E-22	4.80E-19	2.71E-17	0.00E+00	0.00E+00	0.00E+00
ES254	0.00E+00	0.00E+00	3.83E-49	0.00E+00	0.00E+00	0.00E+00
TOTAL	1.44E+06	1.44E+06	1.44E+06	1.44E+06	1.44E+06	1.44E+06
Decay Heat (Watts/MT)	170	339	499	52	85	95

Table C-3 Boiling water reactor radionuclide inventory at 5 and 30 year decay.

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
H 1	6.99E+00	7.01E+00	7.07E+00	6.99E+00	7.01E+00	7.07E+00
H 2	7.36E-03	1.06E-02	1.43E-02	7.36E-03	1.06E-02	1.43E-02
H 3	3.65E-02	5.76E-02	8.19E-02	8.97E-03	1.42E-02	2.01E-02
HE 3	5.42E-03	6.72E-03	7.33E-03	1.74E-02	2.16E-02	2.35E-02
HE 4	1.61E+00	3.08E+00	7.46E+00	2.22E+00	4.41E+00	1.14E+01
LI 6	2.21E-02	1.16E-02	5.72E-03	2.21E-02	1.16E-02	5.72E-03
LI 7	1.09E+00	1.09E+00	1.09E+00	1.09E+00	1.09E+00	1.09E+00
BE 9	5.67E-04	9.14E-04	1.72E-03	5.67E-04	9.14E-04	1.72E-03
BE 10	8.78E-05	1.77E-04	3.66E-04	8.78E-05	1.77E-04	3.66E-04
B 10	2.97E-03	6.30E-04	2.67E-04	2.97E-03	6.30E-04	2.67E-04
B 11	9.81E-01	9.95E-01	1.03E+00	9.81E-01	9.95E-01	1.03E+00
C 12	1.88E+02	1.88E+02	1.88E+02	1.88E+02	1.88E+02	1.88E+02
C 13	6.28E+00	8.70E+00	1.44E+01	6.28E+00	8.70E+00	1.44E+01
C 14	2.00E-01	3.21E-01	4.73E-01	2.00E-01	3.20E-01	4.72E-01
N 14	1.38E+02	1.38E+02	1.38E+02	1.38E+02	1.38E+02	1.38E+02
N 15	5.54E-01	5.59E-01	5.66E-01	5.54E-01	5.59E-01	5.66E-01
O 16	1.35E+05	1.35E+05	1.35E+05	1.35E+05	1.35E+05	1.35E+05
O 17	5.45E+01	5.45E+01	5.45E+01	5.45E+01	5.45E+01	5.45E+01
O 18	3.10E+02	3.10E+02	3.10E+02	3.10E+02	3.10E+02	3.10E+02
F 19	1.07E+01	1.07E+01	1.07E+01	1.07E+01	1.07E+01	1.07E+01
NE 20	2.14E-04	3.43E-04	5.48E-04	2.14E-04	3.43E-04	5.48E-04
NE 21	7.19E-06	1.16E-05	2.19E-05	7.19E-06	1.16E-05	2.19E-05
NE 22	1.07E-05	1.72E-05	3.25E-05	1.07E-05	1.72E-05	3.25E-05
NA 23	1.50E+01	1.50E+01	1.50E+01	1.50E+01	1.50E+01	1.50E+01
MG 24	1.58E+00	1.59E+00	1.60E+00	1.58E+00	1.59E+00	1.60E+00
MG 25	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01
MG 26	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01
AL 27	1.71E+02	1.71E+02	1.71E+02	1.71E+02	1.71E+02	1.71E+02
SI 28	4.01E+02	4.01E+02	4.01E+02	4.01E+02	4.01E+02	4.01E+02
SI 29	2.10E+01	2.10E+01	2.10E+01	2.10E+01	2.10E+01	2.10E+01
SI 30	1.44E+01	1.44E+01	1.44E+01	1.44E+01	1.44E+01	1.44E+01
SI 32	6.20E-10	8.04E-10	1.32E-09	6.04E-10	7.83E-10	1.28E-09
P 31	5.17E+01	5.17E+01	5.17E+01	5.17E+01	5.17E+01	5.17E+01
P 32	3.74E-14	4.84E-14	7.92E-14	3.64E-14	4.72E-14	7.72E-14
S 32	2.94E+01	2.94E+01	2.94E+01	2.94E+01	2.94E+01	2.94E+01

Table C-3 (continued).

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
S 33	2.52E-01	2.59E-01	2.68E-01	2.52E-01	2.59E-01	2.68E-01
S 34	1.39E+00	1.39E+00	1.39E+00	1.39E+00	1.39E+00	1.39E+00
S 35	3.28E-10	3.00E-10	2.94E-10	1.90E-41	1.74E-41	1.71E-41
S 36	5.92E-03	5.92E-03	5.92E-03	5.92E-03	5.92E-03	5.92E-03
CL 35	3.76E+00	3.65E+00	3.52E+00	3.76E+00	3.65E+00	3.52E+00
CL 36	2.02E-01	3.18E-01	4.45E-01	2.02E-01	3.18E-01	4.45E-01
CL 37	1.34E+00	1.34E+00	1.34E+00	1.34E+00	1.34E+00	1.34E+00
AR 36	2.72E-06	4.95E-06	8.24E-06	1.41E-05	2.29E-05	3.34E-05
AR 37	7.82E-22	7.47E-22	8.89E-22	0.00E+00	0.00E+00	0.00E+00
AR 38	7.72E-04	1.24E-03	1.82E-03	7.72E-04	1.24E-03	1.82E-03
AR 39	1.04E-06	1.67E-06	2.80E-06	9.77E-07	1.57E-06	2.62E-06
AR 40	0.00E+00	3.26E-06	8.43E-06	0.00E+00	3.26E-06	8.43E-06
K 39	0.00E+00	0.00E+00	5.99E-08	0.00E+00	0.00E+00	2.34E-07
K 40	3.10E-04	4.92E-04	9.13E-04	3.10E-04	4.92E-04	9.13E-04
K 41	5.86E-06	1.49E-05	4.00E-05	6.07E-06	1.53E-05	4.04E-05
CA 40	1.93E+00	1.93E+00	1.93E+00	1.93E+00	1.93E+00	1.93E+00
CA 41	9.58E-04	1.54E-03	2.21E-03	9.58E-04	1.54E-03	2.21E-03
CA 42	1.35E-02	1.35E-02	1.35E-02	1.35E-02	1.35E-02	1.35E-02
CA 43	2.80E-03	2.80E-03	2.82E-03	2.80E-03	2.80E-03	2.82E-03
CA 44	4.60E-02	4.61E-02	4.63E-02	4.60E-02	4.61E-02	4.63E-02
CA 45	8.76E-09	8.33E-09	8.44E-09	1.19E-25	1.13E-25	1.14E-25
CA 46	1.00E-04	1.13E-04	1.44E-04	1.00E-04	1.13E-04	1.44E-04
CA 48	4.54E-03	4.54E-03	4.53E-03	4.54E-03	4.54E-03	4.53E-03
SC 45	6.17E-05	9.81E-05	1.43E-04	6.17E-05	9.81E-05	1.43E-04
SC 46	2.45E-11	2.32E-11	2.94E-11	3.86E-44	3.65E-44	4.63E-44
TI 46	3.59E+01	3.59E+01	3.59E+01	3.59E+01	3.59E+01	3.59E+01
TI 47	3.31E+01	3.31E+01	3.31E+01	3.31E+01	3.31E+01	3.31E+01
TI 48	3.34E+02	3.33E+02	3.33E+02	3.34E+02	3.33E+02	3.33E+02
TI 49	2.60E+01	2.65E+01	2.72E+01	2.60E+01	2.65E+01	2.72E+01
TI 50	2.46E+01	2.46E+01	2.47E+01	2.46E+01	2.46E+01	2.47E+01
V 50	3.86E-02	4.17E-02	5.04E-02	3.86E-02	4.17E-02	5.04E-02
V 51	1.54E+01	1.65E+01	1.77E+01	1.54E+01	1.65E+01	1.77E+01
CR 50	4.29E+02	4.28E+02	4.26E+02	4.29E+02	4.28E+02	4.26E+02
CR 51	1.74E-21	1.65E-21	1.54E-21	0.00E+00	0.00E+00	0.00E+00
CR 52	8.62E+03	8.61E+03	8.61E+03	8.62E+03	8.61E+03	8.61E+03
CR 53	9.97E+02	9.97E+02	9.97E+02	9.97E+02	9.97E+02	9.97E+02
CR 54	2.57E+02	2.60E+02	2.64E+02	2.57E+02	2.60E+02	2.64E+02

Table C-3 (continued).

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
MN 54	5.54E-04	5.75E-04	7.46E-04	8.85E-13	9.19E-13	1.19E-12
MN 55	8.76E+02	8.74E+02	8.72E+02	8.76E+02	8.74E+02	8.72E+02
FE 54	1.58E+03	1.57E+03	1.57E+03	1.58E+03	1.57E+03	1.57E+03
FE 55	1.68E-01	2.19E-01	2.54E-01	2.14E-04	2.79E-04	3.24E-04
FE 56	2.58E+04	2.58E+04	2.58E+04	2.58E+04	2.58E+04	2.58E+04
FE 57	6.30E+02	6.38E+02	6.49E+02	6.30E+02	6.38E+02	6.49E+02
FE 58	8.55E+01	8.61E+01	8.74E+01	8.55E+01	8.61E+01	8.74E+01
FE 59	1.58E-15	1.54E-15	1.55E-15	0.00E+00	0.00E+00	0.00E+00
CO 58	1.83E-09	1.76E-09	2.16E-09	2.66E-48	2.55E-48	3.14E-48
CO 59	1.47E+02	1.44E+02	1.41E+02	1.47E+02	1.44E+02	1.41E+02
CO 60	2.06E+00	2.91E+00	3.88E+00	7.70E-02	1.08E-01	1.45E-01
NI 58	1.09E+04	1.09E+04	1.09E+04	1.09E+04	1.09E+04	1.09E+04
NI 59	1.41E+01	2.21E+01	3.11E+01	1.41E+01	2.21E+01	3.11E+01
NI 60	4.33E+03	4.33E+03	4.33E+03	4.33E+03	4.33E+03	4.34E+03
NI 61	1.94E+02	1.97E+02	1.99E+02	1.94E+02	1.97E+02	1.99E+02
NI 62	6.13E+02	6.12E+02	6.10E+02	6.13E+02	6.12E+02	6.10E+02
NI 63	2.61E+00	4.14E+00	5.86E+00	2.16E+00	3.43E+00	4.86E+00
NI 64	1.61E+02	1.61E+02	1.61E+02	1.61E+02	1.61E+02	1.61E+02
CU 63	1.41E+01	1.42E+01	1.43E+01	1.46E+01	1.49E+01	1.53E+01
CU 65	6.55E+00	6.61E+00	6.68E+00	6.55E+00	6.61E+00	6.68E+00
ZN 64	1.91E+01	1.91E+01	1.91E+01	1.91E+01	1.91E+01	1.91E+01
ZN 65	6.43E-05	6.39E-05	6.78E-05	3.44E-16	3.42E-16	3.63E-16
ZN 66	1.13E+01	1.13E+01	1.13E+01	1.13E+01	1.13E+01	1.13E+01
ZN 67	1.68E+00	1.67E+00	1.66E+00	1.68E+00	1.67E+00	1.66E+00
ZN 68	7.87E+00	7.87E+00	7.88E+00	7.87E+00	7.87E+00	7.88E+00
ZN 70	2.70E-01	2.71E-01	2.75E-01	2.70E-01	2.71E-01	2.75E-01
GA 69	1.87E-02	2.99E-02	4.80E-02	1.87E-02	2.99E-02	4.80E-02
GA 71	3.04E-05	4.92E-05	7.19E-05	3.04E-05	4.92E-05	7.19E-05
GE 70	7.19E-05	1.85E-04	5.08E-04	7.19E-05	1.85E-04	5.08E-04
GE 72	1.08E-02	1.88E-02	3.39E-02	1.08E-02	1.88E-02	3.39E-02
GE 73	2.20E-02	3.90E-02	6.61E-02	2.20E-02	3.90E-02	6.61E-02
GE 74	4.58E-02	8.53E-02	1.48E-01	4.58E-02	8.53E-02	1.48E-01
GE 76	2.28E-01	4.44E-01	7.27E-01	2.28E-01	4.44E-01	7.27E-01
AS 75	9.52E-02	1.77E-01	2.91E-01	9.52E-02	1.77E-01	2.91E-01
SE 76	1.69E-03	5.12E-03	1.50E-02	1.69E-03	5.12E-03	1.50E-02
SE 77	4.77E-01	9.17E-01	1.45E+00	4.77E-01	9.17E-01	1.45E+00
SE 78	1.10E+00	2.17E+00	3.62E+00	1.10E+00	2.17E+00	3.62E+00

Table C-3 (continued).

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
SE 79	2.69E+00	5.29E+00	8.65E+00	2.69E+00	5.29E+00	8.65E+00
SE 80	6.03E+00	1.20E+01	1.95E+01	6.03E+00	1.20E+01	1.95E+01
SE 82	1.51E+01	3.03E+01	4.86E+01	1.51E+01	3.03E+01	4.86E+01
BR 79	1.75E-04	3.99E-04	7.74E-04	8.92E-04	1.81E-03	3.08E-03
BR 81	9.96E+00	1.95E+01	3.04E+01	9.96E+00	1.95E+01	3.04E+01
KR 80	1.12E-04	2.09E-04	3.83E-04	1.12E-04	2.09E-04	3.83E-04
KR 81	8.84E-06	2.01E-05	5.02E-05	8.84E-06	2.01E-05	5.02E-05
KR 82	3.08E-01	9.07E-01	2.39E+00	3.08E-01	9.07E-01	2.39E+00
KR 83	2.08E+01	3.79E+01	5.32E+01	2.08E+01	3.79E+01	5.32E+01
KR 84	4.77E+01	1.01E+02	1.68E+02	4.77E+01	1.01E+02	1.68E+02
KR 85	7.83E+00	1.49E+01	2.15E+01	1.56E+00	2.96E+00	4.27E+00
KR 86	8.33E+01	1.70E+02	2.68E+02	8.33E+01	1.70E+02	2.68E+02
RB 85	4.57E+01	9.41E+01	1.51E+02	5.20E+01	1.06E+02	1.68E+02
RB 86	3.11E-32	6.00E-32	1.13E-31	0.00E+00	0.00E+00	0.00E+00
RB 87	1.07E+02	2.19E+02	3.43E+02	1.07E+02	2.19E+02	3.43E+02
SR 86	1.16E-01	3.88E-01	1.16E+00	1.16E-01	3.88E-01	1.16E+00
SR 87	3.93E-03	7.57E-03	1.91E-02	3.93E-03	7.57E-03	1.91E-02
SR 88	1.54E+02	3.14E+02	4.93E+02	1.54E+02	3.14E+02	4.93E+02
SR 89	2.62E-10	2.53E-10	2.20E-10	0.00E+00	0.00E+00	0.00E+00
SR 90	2.10E+02	4.19E+02	6.30E+02	1.16E+02	2.31E+02	3.47E+02
Y 89	2.00E+02	4.10E+02	6.40E+02	2.00E+02	4.10E+02	6.40E+02
Y 90	5.26E-02	1.05E-01	1.58E-01	2.90E-02	5.79E-02	8.71E-02
Y 91	1.26E-08	1.22E-08	1.09E-08	1.31E-55	1.27E-55	1.14E-55
ZR 90	2.65E+05	2.66E+05	2.65E+05	2.66E+05	2.66E+05	2.66E+05
ZR 91	5.85E+04	5.87E+04	5.88E+04	5.85E+04	5.87E+04	5.88E+04
ZR 92	9.05E+04	9.09E+04	9.13E+04	9.05E+04	9.09E+04	9.13E+04
ZR 93	4.72E+02	8.88E+02	1.44E+03	4.72E+02	8.88E+02	1.44E+03
ZR 94	9.40E+04	9.43E+04	9.47E+04	9.40E+04	9.43E+04	9.47E+04
ZR 95	1.39E-07	1.36E-07	1.34E-07	1.52E-50	1.49E-50	1.47E-50
ZR 96	1.57E+04	1.61E+04	1.65E+04	1.57E+04	1.61E+04	1.65E+04
NB 93	1.59E+02	1.58E+02	1.58E+02	1.59E+02	1.58E+02	1.58E+02
NB 93M	1.05E-03	2.26E-03	4.27E-03	3.17E-03	6.03E-03	9.97E-03
NB 94	2.90E-01	4.60E-01	8.04E-01	2.90E-01	4.60E-01	8.04E-01
NB 95	1.70E-07	1.66E-07	1.64E-07	1.85E-50	1.81E-50	1.79E-50
NB 95M	5.83E-11	5.70E-11	5.62E-11	6.36E-54	6.22E-54	6.13E-54
MO 92	1.42E+00	1.42E+00	1.42E+00	1.42E+00	1.42E+00	1.42E+00
MO 93	3.56E-04	5.72E-04	9.84E-04	3.55E-04	5.69E-04	9.79E-04

Table C-3 (continued).

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
MO 94	9.10E-01	9.10E-01	9.09E-01	9.10E-01	9.10E-01	9.09E-01
MO 95	3.76E+02	7.25E+02	1.14E+03	3.76E+02	7.25E+02	1.14E+03
MO 96	1.05E+01	3.74E+01	1.11E+02	1.05E+01	3.74E+01	1.11E+02
MO 97	4.04E+02	7.86E+02	1.30E+03	4.04E+02	7.86E+02	1.30E+03
MO 98	3.73E+02	7.40E+02	1.22E+03	3.73E+02	7.40E+02	1.22E+03
MO100	4.25E+02	8.44E+02	1.40E+03	4.25E+02	8.44E+02	1.40E+03
TC 98	1.27E-03	4.27E-03	1.32E-02	1.27E-03	4.27E-03	1.32E-02
TC 99	3.70E+02	7.07E+02	1.08E+03	3.70E+02	7.07E+02	1.08E+03
RU 99	7.13E-03	1.60E-02	2.97E-02	3.72E-02	7.35E-02	1.18E-01
RU100	2.65E+01	8.88E+01	2.55E+02	2.65E+01	8.88E+01	2.55E+02
RU101	3.58E+02	6.98E+02	1.14E+03	3.58E+02	6.98E+02	1.14E+03
RU102	3.52E+02	7.04E+02	1.24E+03	3.52E+02	7.04E+02	1.24E+03
RU103	3.41E-13	3.43E-13	3.67E-13	0.00E+00	0.00E+00	0.00E+00
RU104	2.58E+02	4.97E+02	8.87E+02	2.58E+02	4.97E+02	8.87E+02
RU106	3.04E+00	3.99E+00	5.18E+00	1.04E-07	1.36E-07	1.77E-07
RH102	6.95E-05	1.97E-04	4.75E-04	1.77E-07	5.01E-07	1.21E-06
RH103	2.55E+02	4.10E+02	5.36E+02	2.55E+02	4.10E+02	5.36E+02
RH103M	3.05E-16	3.07E-16	3.28E-16	0.00E+00	0.00E+00	0.00E+00
RH106	2.85E-06	3.75E-06	4.87E-06	9.76E-14	1.28E-13	1.67E-13
PD104	6.67E+01	2.17E+02	5.54E+02	6.67E+01	2.17E+02	5.54E+02
PD105	1.88E+02	3.57E+02	6.39E+02	1.88E+02	3.57E+02	6.39E+02
PD106	1.67E+02	3.23E+02	6.26E+02	1.70E+02	3.27E+02	6.31E+02
PD107	1.07E+02	2.04E+02	3.83E+02	1.07E+02	2.04E+02	3.83E+02
PD108	7.40E+01	1.41E+02	2.63E+02	7.40E+01	1.41E+02	2.63E+02
PD110	2.41E+01	4.59E+01	8.83E+01	2.41E+01	4.59E+01	8.83E+01
AG107	4.69E-02	4.44E-02	4.04E-02	4.71E-02	4.49E-02	4.14E-02
AG108	9.03E-13	1.40E-12	2.11E-12	7.88E-13	1.23E-12	1.84E-12
AG108M	2.86E-04	4.45E-04	6.68E-04	2.50E-04	3.88E-04	5.83E-04
AG109	4.12E+01	7.11E+01	1.16E+02	4.12E+01	7.11E+01	1.16E+02
AG109M	2.71E-11	2.88E-11	3.55E-11	3.23E-17	3.43E-17	4.22E-17
AG110	3.10E-11	6.66E-11	1.48E-10	3.10E-22	6.65E-22	1.48E-21
AG110M	2.05E-03	4.39E-03	9.79E-03	2.04E-14	4.39E-14	9.78E-14
CD106	3.08E-01	3.07E-01	3.07E-01	3.08E-01	3.07E-01	3.07E-01
CD108	2.19E-01	2.21E-01	2.24E-01	2.19E-01	2.21E-01	2.24E-01
CD109	2.74E-05	2.91E-05	3.59E-05	3.27E-11	3.47E-11	4.27E-11
CD110	1.38E+01	3.52E+01	1.02E+02	1.38E+01	3.52E+01	1.02E+02
CD111	1.67E+01	2.85E+01	5.26E+01	1.67E+01	2.85E+01	5.26E+01

Table C-3 (continued).

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
CD112	1.41E+01	2.11E+01	3.53E+01	1.41E+01	2.11E+01	3.53E+01
CD113	1.46E-01	1.53E-01	1.98E-01	1.47E-01	1.54E-01	1.99E-01
CD113M	9.44E-02	1.81E-01	3.85E-01	2.88E-02	5.51E-02	1.17E-01
CD114	2.14E+01	3.03E+01	4.72E+01	2.14E+01	3.03E+01	4.72E+01
CD115M	2.41E-14	2.40E-14	2.84E-14	0.00E+00	0.00E+00	0.00E+00
CD116	6.05E+00	9.14E+00	1.49E+01	6.05E+00	9.14E+00	1.49E+01
IN113	9.06E-01	1.35E+00	2.13E+00	9.72E-01	1.47E+00	2.39E+00
IN113M	1.51E-09	1.41E-09	1.68E-09	1.98E-33	1.85E-33	2.21E-33
IN114	7.41E-19	1.18E-18	2.24E-18	0.00E+00	0.00E+00	0.00E+00
IN114M	4.61E-14	7.32E-14	1.39E-13	0.00E+00	0.00E+00	0.00E+00
IN115	1.95E+00	2.09E+00	2.22E+00	1.95E+00	2.09E+00	2.22E+00
IN115M	6.69E-21	6.65E-21	7.88E-21	0.00E+00	0.00E+00	0.00E+00
SN112	7.98E+01	7.93E+01	7.82E+01	7.98E+01	7.93E+01	7.82E+01
SN113	2.51E-06	2.34E-06	2.79E-06	3.30E-30	3.08E-30	3.67E-30
SN114	5.50E+01	5.51E+01	5.54E+01	5.50E+01	5.51E+01	5.54E+01
SN115	2.99E+01	2.90E+01	2.81E+01	2.99E+01	2.90E+01	2.81E+01
SN116	1.22E+03	1.22E+03	1.22E+03	1.22E+03	1.22E+03	1.22E+03
SN117	6.61E+02	6.66E+02	6.78E+02	6.61E+02	6.66E+02	6.78E+02
SN117M	8.84E-41	8.45E-41	9.85E-41	0.00E+00	0.00E+00	0.00E+00
SN118	2.06E+03	2.06E+03	2.06E+03	2.06E+03	2.06E+03	2.06E+03
SN119	7.51E+02	7.59E+02	7.79E+02	7.51E+02	7.59E+02	7.79E+02
SN119M	1.34E-02	1.33E-02	1.65E-02	8.08E-14	8.04E-14	9.99E-14
SN120	2.80E+03	2.81E+03	2.81E+03	2.80E+03	2.81E+03	2.81E+03
SN121M	1.39E-02	2.25E-02	3.83E-02	9.85E-03	1.59E-02	2.71E-02
SN122	4.09E+02	4.12E+02	4.18E+02	4.09E+02	4.12E+02	4.18E+02
SN123	2.05E-05	2.00E-05	2.32E-05	1.08E-26	1.05E-26	1.22E-26
SN124	5.03E+02	5.06E+02	5.11E+02	5.03E+02	5.06E+02	5.11E+02
SN126	1.35E+01	2.46E+01	4.35E+01	1.35E+01	2.46E+01	4.35E+01
SB121	6.69E+00	1.11E+01	1.88E+01	6.70E+00	1.11E+01	1.88E+01
SB123	5.43E+00	9.38E+00	1.62E+01	5.43E+00	9.38E+00	1.62E+01
SB124	2.32E-11	4.21E-11	8.93E-11	5.04E-57	9.14E-57	1.94E-56
SB125	2.90E+00	4.13E+00	5.81E+00	5.56E-03	7.92E-03	1.12E-02
SB126	6.39E-07	1.17E-06	2.07E-06	6.39E-07	1.17E-06	2.07E-06
SB126M	4.86E-09	8.90E-09	1.57E-08	4.86E-09	8.90E-09	1.57E-08
TE122	2.88E-01	7.77E-01	2.39E+00	2.88E-01	7.77E-01	2.39E+00
TE123	3.25E-03	1.12E-02	4.47E-02	3.25E-03	1.12E-02	4.47E-02
TE123M	1.55E-08	4.81E-08	1.95E-07	1.67E-31	5.17E-31	2.10E-30

Table C-3 (continued).

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
TE124	1.28E-01	3.87E-01	1.29E+00	1.28E-01	3.87E-01	1.29E+00
TE125	9.59E+00	1.76E+01	3.29E+01	1.25E+01	2.18E+01	3.88E+01
TE125M	4.06E-02	5.78E-02	8.13E-02	7.78E-05	1.11E-04	1.56E-04
TE126	3.57E-01	7.41E-01	1.65E+00	3.60E-01	7.46E-01	1.66E+00
TE127	3.39E-08	3.39E-08	3.86E-08	2.06E-33	2.06E-33	2.34E-33
TE127M	9.69E-06	9.69E-06	1.10E-05	5.88E-31	5.88E-31	6.69E-31
TE128	5.25E+01	9.99E+01	1.72E+02	5.25E+01	9.99E+01	1.72E+02
TE129	4.55E-20	4.48E-20	4.71E-20	0.00E+00	0.00E+00	0.00E+00
TE129M	4.86E-17	4.78E-17	5.03E-17	0.00E+00	0.00E+00	0.00E+00
TE130	1.65E+02	3.21E+02	5.47E+02	1.65E+02	3.21E+02	5.47E+02
I127	2.76E+01	5.00E+01	8.38E+01	2.76E+01	5.00E+01	8.38E+01
I129	8.70E+01	1.63E+02	2.72E+02	8.70E+01	1.63E+02	2.72E+02
XE127	2.85E-22	9.38E-22	3.81E-21	0.00E+00	0.00E+00	0.00E+00
XE128	9.03E-01	2.87E+00	9.18E+00	9.03E-01	2.87E+00	9.18E+00
XE129	3.02E-03	1.55E-02	8.15E-02	3.11E-03	1.56E-02	8.18E-02
XE130	3.19E+00	1.02E+01	2.71E+01	3.19E+00	1.02E+01	2.71E+01
XE131	2.27E+02	3.81E+02	4.86E+02	2.27E+02	3.81E+02	4.86E+02
XE131M	1.79E-47	1.76E-47	1.79E-47	0.00E+00	0.00E+00	0.00E+00
XE132	4.67E+02	9.87E+02	1.81E+03	4.67E+02	9.87E+02	1.81E+03
XE134	6.65E+02	1.33E+03	2.21E+03	6.65E+02	1.33E+03	2.21E+03
XE136	1.03E+03	1.99E+03	3.22E+03	1.03E+03	1.99E+03	3.22E+03
CS133	5.45E+02	1.02E+03	1.50E+03	5.45E+02	1.02E+03	1.50E+03
CS134	7.26E+00	1.96E+01	4.31E+01	1.63E-03	4.38E-03	9.65E-03
CS135	1.67E+02	3.90E+02	7.60E+02	1.67E+02	3.90E+02	7.60E+02
CS136	4.01E-43	6.73E-43	1.27E-42	0.00E+00	0.00E+00	0.00E+00
CS137	4.88E+02	9.52E+02	1.53E+03	2.74E+02	5.34E+02	8.61E+02
BA132	3.82E-04	1.26E-03	3.78E-03	3.82E-04	1.26E-03	3.78E-03
BA134	4.02E+01	1.33E+02	3.74E+02	4.75E+01	1.53E+02	4.17E+02
BA135	3.37E-02	3.29E-01	2.42E+00	3.50E-02	3.32E-01	2.43E+00
BA136	7.19E+00	2.07E+01	5.98E+01	7.19E+00	2.07E+01	5.98E+01
BA136M	1.80E-50	3.02E-50	5.69E-50	0.00E+00	0.00E+00	0.00E+00
BA137	7.17E+01	1.66E+02	3.27E+02	2.86E+02	5.84E+02	0.00E+00
BA137M	7.47E-05	1.46E-04	2.35E-04	4.19E-05	8.17E-05	1.32E-04
BA138	5.74E+02	1.15E+03	1.90E+03	5.74E+02	1.15E+03	1.90E+03
BA140	0.00E+00	1.63E-42	1.59E-42	0.00E+00	0.00E+00	0.00E+00
LA138	2.81E-03	5.20E-03	6.60E-03	2.81E-03	5.20E-03	6.60E-03
LA139	5.52E+02	1.10E+03	1.80E+03	5.52E+02	1.10E+03	1.80E+03

Table C-3 (continued).

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
LA140	0.00E+00	2.46E-43	2.40E-43	0.00E+00	0.00E+00	0.00E+00
CE140	5.55E+02	1.11E+03	1.85E+03	5.55E+02	1.11E+03	1.85E+03
CE141	4.82E-16	4.76E-16	4.64E-16	0.00E+00	0.00E+00	0.00E+00
CE142	5.07E+02	1.02E+03	1.67E+03	5.07E+02	1.02E+03	1.67E+03
CE144	2.50E+00	2.85E+00	2.81E+00	5.34E-10	6.09E-10	6.00E-10
PR141	5.10E+02	1.01E+03	1.64E+03	5.10E+02	1.01E+03	1.64E+03
PR143	4.93E-40	4.84E-40	4.66E-40	0.00E+00	0.00E+00	0.00E+00
PR144	1.06E-04	1.20E-04	1.19E-04	2.26E-14	2.57E-14	2.54E-14
PR144M	5.28E-07	6.01E-07	5.93E-07	1.13E-16	1.29E-16	1.27E-16
ND142	6.07E+00	2.24E+01	6.08E+01	6.07E+00	2.24E+01	6.08E+01
ND143	4.16E+02	7.34E+02	1.04E+03	4.16E+02	7.34E+02	1.04E+03
ND144	5.26E+02	1.17E+03	2.05E+03	5.28E+02	1.17E+03	2.06E+03
ND145	3.17E+02	6.09E+02	9.19E+02	3.17E+02	6.09E+02	9.19E+02
ND146	2.94E+02	6.19E+02	1.10E+03	2.94E+02	6.19E+02	1.10E+03
ND147	1.09E-49	1.07E-49	1.07E-49	0.00E+00	0.00E+00	0.00E+00
ND148	1.69E+02	3.34E+02	5.52E+02	1.69E+02	3.34E+02	5.52E+02
ND150	8.23E+01	1.61E+02	2.74E+02	8.23E+01	1.61E+02	2.74E+02
PM146	1.02E-03	2.30E-03	4.18E-03	4.35E-05	9.86E-05	1.79E-04
PM147	2.65E+01	3.22E+01	3.00E+01	3.59E-02	4.36E-02	4.07E-02
PM148	3.25E-16	3.97E-16	4.42E-16	0.00E+00	0.00E+00	0.00E+00
PM148M	4.43E-14	5.41E-14	6.03E-14	0.00E+00	0.00E+00	0.00E+00
SM146	1.15E-03	5.28E-03	1.78E-02	1.51E-03	6.10E-03	1.93E-02
SM147	9.93E+01	1.63E+02	1.97E+02	1.26E+02	1.95E+02	2.27E+02
SM148	5.76E+01	1.60E+02	3.37E+02	5.76E+01	1.60E+02	3.37E+02
SM149	2.98E+00	3.25E+00	3.83E+00	2.98E+00	3.25E+00	3.83E+00
SM150	1.40E+02	2.73E+02	4.38E+02	1.40E+02	2.73E+02	4.38E+02
SM151	1.03E+01	1.39E+01	2.11E+01	8.46E+00	1.15E+01	1.74E+01
SM152	6.89E+01	1.17E+02	1.57E+02	6.89E+01	1.17E+02	1.57E+02
SM154	1.77E+01	3.40E+01	6.13E+01	1.77E+01	3.40E+01	6.13E+01
EU150	7.91E-08	2.45E-07	8.01E-07	4.89E-08	1.52E-07	4.95E-07
EU151	4.12E-01	5.59E-01	8.48E-01	2.21E+00	2.99E+00	4.54E+00
EU152	2.31E-02	4.25E-02	6.30E-02	6.47E-03	1.19E-02	1.76E-02
EU153	4.95E+01	1.09E+02	1.92E+02	4.95E+01	1.09E+02	1.92E+02
EU154	7.03E+00	2.07E+01	4.75E+01	9.38E-01	2.76E+00	6.34E+00
EU155	2.53E+00	5.94E+00	1.45E+01	7.69E-02	1.80E-01	4.42E-01
EU156	7.68E-37	1.48E-36	3.07E-36	0.00E+00	0.00E+00	0.00E+00
GD152	2.52E+00	2.28E+00	1.86E+00	2.52E+00	2.29E+00	1.87E+00

Table C-3 (continued).

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
GD153	1.25E-03	1.12E-03	1.14E-03	5.48E-15	4.93E-15	5.01E-15
GD154	3.21E+01	3.90E+01	5.74E+01	3.82E+01	5.70E+01	9.86E+01
GD155	5.56E+00	7.49E+00	1.73E+01	8.02E+00	1.33E+01	3.14E+01
GD156	5.44E+02	5.65E+02	6.43E+02	5.44E+02	5.65E+02	6.43E+02
GD157	2.11E+00	2.26E+00	3.34E+00	2.11E+00	2.26E+00	3.34E+00
GD158	6.59E+02	6.75E+02	7.13E+02	6.59E+02	6.75E+02	7.13E+02
GD160	3.47E+02	3.47E+02	3.45E+02	3.47E+02	3.47E+02	3.45E+02
TB159	1.52E+01	2.36E+01	3.90E+01	1.52E+01	2.36E+01	3.90E+01
TB160	1.14E-08	1.81E-08	3.63E-08	1.10E-46	1.74E-46	3.48E-46
DY160	1.42E+00	3.15E+00	7.49E+00	1.42E+00	3.15E+00	7.49E+00
DY161	1.73E+00	2.83E+00	6.10E+00	1.73E+00	2.83E+00	6.10E+00
DY162	7.63E-01	1.57E+00	3.68E+00	7.63E-01	1.57E+00	3.68E+00
DY163	3.28E-01	9.21E-01	2.93E+00	3.28E-01	9.21E-01	2.93E+00
DY164	5.60E-02	1.70E-01	7.17E-01	5.60E-02	1.70E-01	7.17E-01
HO165	6.26E-02	2.38E-01	1.19E+00	6.26E-02	2.38E-01	1.19E+00
HO166M	3.40E-04	1.79E-03	1.42E-02	3.35E-04	1.76E-03	1.40E-02
ER166	1.48E-02	4.71E-02	2.78E-01	1.48E-02	4.72E-02	2.79E-01
ER167	2.30E-03	3.50E-03	9.06E-03	2.30E-03	3.50E-03	9.06E-03
ER168	2.13E-03	5.23E-03	1.63E-02	2.13E-03	5.23E-03	1.63E-02
TM169	9.66E-06	3.47E-05	1.47E-04	9.66E-06	3.47E-05	1.47E-04
TM170	5.22E-11	2.32E-10	1.29E-09	2.19E-32	9.73E-32	5.41E-31
TM171	8.43E-09	5.50E-08	4.61E-07	1.01E-12	6.62E-12	5.55E-11
YB170	0.00E+00	1.03E-05	7.44E-05	0.00E+00	1.03E-05	7.44E-05
YB171	0.00E+00	0.00E+00	6.74E-06	0.00E+00	0.00E+00	7.20E-06
LU175	2.37E-02	2.86E-02	2.77E-02	2.37E-02	2.86E-02	2.77E-02
LU176	5.50E-04	9.17E-04	1.28E-03	5.50E-04	9.17E-04	1.28E-03
LU177	2.78E-12	5.79E-12	9.06E-12	5.12E-30	1.07E-29	1.67E-29
LU177M	2.79E-10	5.81E-10	9.11E-10	5.14E-28	1.07E-27	1.68E-27
HF174	3.79E-02	2.75E-02	1.82E-02	3.79E-02	2.75E-02	1.82E-02
HF175	4.95E-11	3.30E-11	2.11E-11	2.67E-50	1.78E-50	1.14E-50
HF176	1.61E+00	1.36E+00	9.82E-01	1.61E+00	1.36E+00	9.82E-01
HF177	7.62E-01	3.85E-01	1.96E-01	7.62E-01	3.85E-01	1.96E-01
HF178	9.64E+00	6.63E+00	3.20E+00	9.64E+00	6.63E+00	3.20E+00
HF179	1.24E+01	1.40E+01	1.36E+01	1.24E+01	1.40E+01	1.36E+01
HF180	1.68E+01	1.87E+01	2.25E+01	1.68E+01	1.87E+01	2.25E+01
HF181	4.93E-15	5.24E-15	6.73E-15	0.00E+00	0.00E+00	0.00E+00
HF182	1.74E-03	2.44E-03	4.13E-03	1.74E-03	2.44E-03	4.13E-03

Table C-3 (continued).

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
TA181	3.89E-01	6.09E-01	9.61E-01	3.89E-01	6.09E-01	9.61E-01
TA182	1.19E-07	1.98E-07	4.00E-07	6.09E-11	8.54E-11	1.45E-10
W180	1.50E-02	1.43E-02	1.30E-02	1.50E-02	1.43E-02	1.30E-02
W181	8.69E-09	7.79E-09	8.69E-09	1.80E-31	1.62E-31	1.80E-31
W182	2.64E+00	2.33E+00	1.85E+00	2.64E+00	2.33E+00	1.85E+00
W183	2.25E+00	2.45E+00	2.74E+00	2.25E+00	2.45E+00	2.74E+00
W184	4.14E+00	4.31E+00	4.72E+00	4.14E+00	4.31E+00	4.72E+00
W185	2.29E-10	2.26E-10	2.81E-10	5.73E-47	5.65E-47	7.04E-47
W186	2.96E+00	2.60E+00	2.02E+00	2.96E+00	2.60E+00	2.02E+00
W188	3.18E-12	2.48E-12	2.59E-12	7.84E-52	6.11E-52	6.38E-52
RE185	2.13E-02	2.87E-02	3.81E-02	2.13E-02	2.87E-02	3.81E-02
RE187	6.47E-01	9.18E-01	1.27E+00	6.47E-01	9.18E-01	1.27E+00
RE188	3.28E-14	2.56E-14	2.67E-14	8.07E-54	6.29E-54	6.57E-54
OS186	5.80E-03	1.54E-02	3.96E-02	5.80E-03	1.54E-02	3.96E-02
OS187	5.21E-11	8.19E-11	1.25E-10	2.77E-10	4.00E-10	5.66E-10
OS188	6.59E-02	1.53E-01	3.64E-01	6.59E-02	1.53E-01	3.64E-01
OS189	1.17E-03	4.26E-03	1.72E-02	1.17E-03	4.26E-03	1.72E-02
OS190	9.32E-05	5.55E-04	4.16E-03	9.32E-05	5.55E-04	4.16E-03
IR191	0.00E+00	2.89E-06	2.75E-05	0.00E+00	2.89E-06	2.75E-05
IR192	4.16E-14	5.99E-13	1.06E-11	3.68E-14	5.38E-13	9.58E-12
IR192M	0.00E+00	0.00E+00	1.22E-08	0.00E+00	0.00E+00	1.14E-08
IR193	0.00E+00	0.00E+00	2.98E-06	0.00E+00	0.00E+00	2.99E-06
PT192	0.00E+00	0.00E+00	2.50E-05	0.00E+00	0.00E+00	2.50E-05
PT193	0.00E+00	1.00E-08	3.65E-07	0.00E+00	9.67E-09	3.53E-07
TL206	4.67E-17	7.49E-17	1.25E-16	4.67E-17	7.49E-17	1.25E-16
TL207	4.88E-15	1.87E-14	4.37E-14	3.59E-14	8.31E-14	1.51E-13
TL208	4.29E-12	1.95E-11	7.71E-11	6.31E-12	2.62E-11	9.96E-11
TL209	9.28E-19	8.37E-18	6.71E-17	2.76E-18	1.27E-17	7.57E-17
PB204	1.38E-02	1.38E-02	1.37E-02	1.38E-02	1.38E-02	1.37E-02
PB205	1.83E-05	2.94E-05	4.67E-05	1.83E-05	2.94E-05	4.67E-05
PB206	2.32E-01	2.28E-01	2.23E-01	2.32E-01	2.28E-01	2.23E-01
PB207	2.28E-01	2.32E-01	2.37E-01	2.28E-01	2.32E-01	2.37E-01
PB208	5.26E-01	5.26E-01	5.27E-01	5.26E-01	5.26E-01	5.27E-01
PB209	3.87E-15	3.49E-14	2.80E-13	1.15E-14	5.28E-14	3.15E-13
PB210	6.14E-11	2.00E-10	5.75E-10	5.03E-09	8.29E-09	1.02E-08
PB211	3.78E-14	1.45E-13	3.38E-13	2.78E-13	6.43E-13	1.17E-12
PB212	2.53E-09	1.15E-08	4.55E-08	3.72E-09	1.55E-08	5.88E-08

Table C-3 (continued).

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
PB214	2.04E-15	4.72E-15	7.63E-15	4.49E-14	7.11E-14	8.48E-14
BI208	2.63E-06	4.22E-06	7.96E-06	2.63E-06	4.22E-06	7.96E-06
BI209	4.00E-01	4.00E-01	4.00E-01	4.00E-01	4.00E-01	4.00E-01
BI210	3.78E-14	1.23E-13	3.54E-13	3.09E-12	5.10E-12	6.28E-12
BI210M	1.80E-05	2.89E-05	4.83E-05	1.80E-05	2.89E-05	4.83E-05
BI211	2.23E-15	8.53E-15	1.99E-14	1.64E-14	3.79E-14	6.91E-14
BI212	2.40E-10	1.09E-09	4.31E-09	3.53E-10	1.47E-09	5.58E-09
BI213	9.09E-16	8.19E-15	6.57E-14	2.71E-15	1.24E-14	7.41E-14
BI214	1.51E-15	3.50E-15	5.67E-15	3.34E-14	5.28E-14	6.30E-14
PO210	7.79E-10	7.37E-10	8.50E-10	8.55E-11	1.41E-10	1.74E-10
PO211	2.73E-20	1.05E-19	2.45E-19	2.01E-19	4.65E-19	8.48E-19
PO212	1.27E-20	5.76E-20	2.28E-19	1.87E-20	7.77E-20	2.95E-19
PO213	1.36E-24	1.23E-23	9.85E-23	4.06E-24	1.86E-23	1.11E-22
PO214	2.08E-22	4.82E-22	7.80E-22	4.59E-21	7.27E-21	8.66E-21
PO215	3.16E-20	1.21E-19	2.83E-19	2.33E-19	5.38E-19	9.81E-19
PO216	1.01E-14	4.58E-14	1.81E-13	1.48E-14	6.17E-14	2.35E-13
PO218	2.37E-16	5.47E-16	8.85E-16	5.21E-15	8.25E-15	9.83E-15
AT217	1.09E-20	9.84E-20	7.89E-19	3.25E-20	1.49E-19	8.90E-19
RN219	7.17E-17	2.74E-16	6.41E-16	5.27E-16	1.22E-15	2.22E-15
RN220	3.81E-12	1.73E-11	6.85E-11	5.60E-12	2.33E-11	8.85E-11
RN222	4.35E-13	1.01E-12	1.63E-12	9.58E-12	1.52E-11	1.81E-11
FR221	9.91E-17	8.94E-16	7.16E-15	2.95E-16	1.35E-15	8.08E-15
FR223	3.32E-16	1.27E-15	2.97E-15	2.44E-15	5.65E-15	1.03E-14
RA223	1.82E-11	6.97E-11	1.63E-10	1.34E-10	3.10E-10	5.65E-10
RA224	2.21E-08	1.00E-07	3.97E-07	3.25E-08	1.35E-07	5.13E-07
RA225	4.48E-13	4.04E-12	3.24E-11	1.33E-12	6.12E-12	3.65E-11
RA226	6.77E-08	1.56E-07	2.53E-07	1.49E-06	2.36E-06	2.81E-06
RA228	3.84E-14	1.11E-13	2.39E-13	5.30E-13	1.13E-12	1.70E-12
AC225	3.03E-13	2.73E-12	2.19E-11	9.02E-13	4.14E-12	2.47E-11
AC227	1.29E-08	4.93E-08	1.15E-07	9.47E-08	2.19E-07	3.99E-07
AC228	4.01E-18	1.15E-17	2.49E-17	5.53E-17	1.18E-16	1.78E-16
TH227	2.99E-11	1.15E-10	2.68E-10	2.20E-10	5.09E-10	9.28E-10
TH228	4.01E-18	1.15E-17	2.49E-17	6.31E-06	2.62E-05	9.96E-05
TH229	8.26E-08	7.45E-07	5.97E-06	2.46E-07	1.13E-06	6.73E-06
TH230	2.24E-03	3.92E-03	4.63E-03	1.08E-02	1.64E-02	1.97E-02
TH231	3.19E-08	3.16E-08	2.23E-08	3.19E-08	3.17E-08	2.23E-08
TH232	3.16E-04	7.74E-04	1.37E-03	1.61E-03	3.40E-03	5.00E-03

Table C-3 (continued).

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
TH234	1.41E-05	1.38E-05	1.34E-05	1.41E-05	1.38E-05	1.34E-05
PA231	1.41E-04	4.43E-04	8.94E-04	3.30E-04	6.31E-04	1.03E-03
PA233	5.62E-06	1.46E-05	3.16E-05	6.32E-06	1.56E-05	3.30E-05
PA234	2.12E-10	2.07E-10	2.02E-10	2.12E-10	2.07E-10	2.02E-10
PA234M	4.74E-10	4.64E-10	4.52E-10	4.74E-10	4.64E-10	4.52E-10
U232	2.31E-04	9.99E-04	3.86E-03	2.35E-04	9.79E-04	3.72E-03
U233	8.42E-04	1.84E-03	3.48E-03	2.23E-03	5.36E-03	1.10E-02
U234	1.19E+02	1.66E+02	1.69E+02	1.26E+02	1.92E+02	2.61E+02
U235	7.83E+03	7.78E+03	5.47E+03	7.84E+03	7.79E+03	5.48E+03
U236	1.78E+03	3.61E+03	4.97E+03	1.78E+03	3.61E+03	4.98E+03
U237	2.31E-05	3.06E-05	4.20E-05	6.94E-06	9.19E-06	1.26E-05
U238	9.68E+05	9.48E+05	9.24E+05	9.68E+05	9.48E+05	9.24E+05
U240	9.18E-14	4.53E-13	2.35E-12	9.18E-14	4.53E-13	2.35E-12
NP235	4.33E-08	1.32E-07	4.16E-07	4.97E-15	1.52E-14	4.77E-14
NP236	1.05E-04	3.32E-04	1.07E-03	1.05E-04	3.32E-04	1.07E-03
NP237	1.65E+02	4.31E+02	9.32E+02	1.86E+02	4.59E+02	9.71E+02
NP238	1.53E-07	4.34E-07	9.13E-07	1.37E-07	3.87E-07	8.15E-07
NP239	2.25E-05	8.34E-05	2.46E-04	2.25E-05	8.32E-05	2.45E-04
NP240M	8.03E-16	3.96E-15	2.06E-14	8.03E-16	3.96E-15	2.06E-14
PU236	6.67E-05	2.41E-04	8.54E-04	1.53E-07	5.53E-07	1.96E-06
PU237	2.96E-17	1.18E-16	5.41E-16	0.00E+00	0.00E+00	0.00E+00
PU238	3.78E+01	1.50E+02	5.22E+02	3.11E+01	1.23E+02	4.29E+02
PU239	4.23E+03	4.83E+03	6.47E+03	4.22E+03	4.83E+03	6.47E+03
PU240	1.23E+03	1.92E+03	3.17E+03	1.23E+03	1.93E+03	3.25E+03
PU241	7.46E+02	9.89E+02	1.36E+03	2.24E+02	2.97E+02	4.07E+02
PU242	2.02E+02	4.60E+02	8.32E+02	2.02E+02	4.60E+02	8.32E+02
PU243	1.34E-15	3.90E-14	1.14E-12	1.34E-15	3.90E-14	1.14E-12
PU244	4.80E-03	2.37E-02	1.23E-01	4.80E-03	2.37E-02	1.23E-01
AM241	2.23E+02	3.19E+02	4.60E+02	7.24E+02	9.82E+02	1.37E+03
AM242	9.77E-06	2.77E-05	5.83E-05	8.71E-06	2.47E-05	5.20E-05
AM242M	8.16E-01	2.31E+00	4.87E+00	7.29E-01	2.06E+00	4.35E+00
AM243	2.62E+01	9.70E+01	2.86E+02	2.62E+01	9.68E+01	2.86E+02
AM244	0.00E+00	3.30E-17	1.72E-16	0.00E+00	3.30E-17	1.72E-16
CM241	1.32E-22	4.14E-22	1.12E-21	0.00E+00	0.00E+00	0.00E+00
CM242	4.15E-03	1.19E-02	2.48E-02	1.76E-03	4.99E-03	1.05E-02
CM243	8.42E-02	3.96E-01	1.36E+00	4.58E-02	2.16E-01	7.38E-01
CM244	3.70E+00	2.45E+01	1.44E+02	1.42E+00	9.42E+00	5.53E+01

Table C-3 (continued).

	DECAY TIME 5 YEARS			DECAY TIME 30 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
CM245	1.15E-01	1.11E+00	1.08E+01	1.14E-01	1.11E+00	1.08E+01
CM246	6.54E-03	1.14E-01	1.80E+00	6.52E-03	1.13E-01	1.79E+00
CM247	3.76E-05	1.09E-03	3.21E-02	3.76E-05	1.09E-03	3.21E-02
CM248	1.16E-06	5.88E-05	3.18E-03	1.16E-06	5.88E-05	3.18E-03
BK249	1.49E-10	1.04E-08	8.29E-07	3.85E-19	2.68E-17	2.14E-15
CF249	8.36E-09	6.15E-07	5.17E-05	8.10E-09	5.95E-07	5.00E-05
CF250	9.61E-10	8.49E-08	7.56E-06	2.55E-10	2.26E-08	2.01E-06
CF251	0.00E+00	4.62E-08	5.44E-06	0.00E+00	4.54E-08	5.34E-06
CF252	3.18E-11	6.04E-09	1.09E-06	4.46E-14	8.48E-12	1.53E-09
CF254	0.00E+00	0.00E+00	2.35E-19	0.00E+00	0.00E+00	8.65E-65
ES254	0.00E+00	0.00E+00	1.34E-11	0.00E+00	0.00E+00	1.44E-21
TOTAL	1.73E+06	1.73E+06	1.73E+06	1.73E+06	1.73E+06	1.73E+06
Decay Heat (Watts/MT)	745	1,508	2,292	330	655	1,253

Table C-4 Boiling water reactor radionuclide inventory at 100 and 500 year decay.

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
H1	6.99E+00	7.01E+00	7.07E+00	6.99E+00	7.01E+00	7.07E+00
H2	7.36E-03	1.06E-02	1.43E-02	7.36E-03	1.06E-02	1.43E-02
H3	1.76E-04	2.78E-04	3.96E-04	3.13E-14	4.94E-14	7.02E-14
HE3	2.13E-02	2.63E-02	2.86E-02	2.14E-02	2.64E-02	2.87E-02
HE4	4.32E+00	7.97E+00	1.89E+01	1.30E+01	2.05E+01	3.93E+01
LI6	2.21E-02	1.16E-02	5.72E-03	2.21E-02	1.16E-02	5.72E-03
LI7	1.09E+00	1.09E+00	1.09E+00	1.09E+00	1.09E+00	1.09E+00
BE9	5.67E-04	9.14E-04	1.72E-03	5.67E-04	9.14E-04	1.72E-03
BE10	8.78E-05	1.77E-04	3.66E-04	8.77E-05	1.76E-04	3.65E-04
B10	2.97E-03	6.30E-04	2.67E-04	2.97E-03	6.30E-04	2.67E-04
B11	9.81E-01	9.95E-01	1.03E+00	9.81E-01	9.95E-01	1.03E+00
C12	1.88E+02	1.88E+02	1.88E+02	1.88E+02	1.88E+02	1.88E+02
C13	6.28E+00	8.70E+00	1.44E+01	6.28E+00	8.70E+00	1.44E+01
C14	1.98E-01	3.18E-01	4.68E-01	1.89E-01	3.03E-01	4.46E-01
N14	1.38E+02	1.38E+02	1.38E+02	1.38E+02	1.38E+02	1.38E+02
N15	5.54E-01	5.59E-01	5.66E-01	5.54E-01	5.59E-01	5.66E-01
O16	1.35E+05	1.35E+05	1.35E+05	1.35E+05	1.35E+05	1.35E+05
O17	5.45E+01	5.45E+01	5.45E+01	5.45E+01	5.45E+01	5.45E+01
O18	3.10E+02	3.10E+02	3.10E+02	3.10E+02	3.10E+02	3.10E+02
F19	1.07E+01	1.07E+01	1.07E+01	1.07E+01	1.07E+01	1.07E+01
NE20	2.14E-04	3.43E-04	5.48E-04	2.14E-04	3.43E-04	5.48E-04
NE21	7.19E-06	1.16E-05	2.19E-05	7.19E-06	1.16E-05	2.19E-05
NE22	1.07E-05	1.72E-05	3.25E-05	1.07E-05	1.72E-05	3.25E-05
NA23	1.50E+01	1.50E+01	1.50E+01	1.50E+01	1.50E+01	1.50E+01
MG24	1.58E+00	1.59E+00	1.60E+00	1.58E+00	1.59E+00	1.60E+00
MG25	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01	2.06E-01
MG26	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01	2.36E-01
AL27	1.71E+02	1.71E+02	1.71E+02	1.71E+02	1.71E+02	1.71E+02
SI28	4.01E+02	4.01E+02	4.01E+02	4.01E+02	4.01E+02	4.01E+02
SI29	2.10E+01	2.10E+01	2.10E+01	2.10E+01	2.10E+01	2.10E+01
SI30	1.44E+01	1.44E+01	1.44E+01	1.44E+01	1.44E+01	1.44E+01
SI32	5.61E-10	7.27E-10	1.19E-09	3.66E-10	4.74E-10	7.76E-10
P31	5.17E+01	5.17E+01	5.17E+01	5.17E+01	5.17E+01	5.17E+01
P32	3.38E-14	4.38E-14	7.16E-14	2.20E-14	2.86E-14	4.67E-14
S32	2.94E+01	2.94E+01	2.94E+01	2.94E+01	2.94E+01	2.94E+01
S33	2.52E-01	2.59E-01	2.68E-01	2.52E-01	2.59E-01	2.68E-01
S34	1.39E+00	1.39E+00	1.39E+00	1.39E+00	1.39E+00	1.39E+00
S36	5.92E-03	5.93E-03	5.93E-03	5.93E-03	5.93E-03	5.93E-03
CL35	3.76E+00	3.65E+00	3.52E+00	3.76E+00	3.65E+00	3.52E+00
CL36	2.02E-01	3.18E-01	4.45E-01	2.02E-01	3.18E-01	4.45E-01
CL37	1.34E+00	1.34E+00	1.34E+00	1.34E+00	1.34E+00	1.34E+00
AR36	4.60E-05	7.32E-05	1.04E-04	2.28E-04	3.60E-04	5.06E-04

Table C-4 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
AR38	7.72E-04	1.24E-03	1.82E-03	7.72E-04	1.24E-03	1.82E-03
AR39	8.16E-07	1.31E-06	2.19E-06	2.91E-07	4.68E-07	7.82E-07
AR40	0.00E+00	3.26E-06	8.43E-06	0.00E+00	3.26E-06	8.43E-06
K39	0.00E+00	0.00E+00	6.68E-07	0.00E+00	0.00E+00	2.08E-06
K40	3.10E-04	4.92E-04	9.13E-04	3.10E-04	4.92E-04	9.13E-04
K41	6.64E-06	1.62E-05	4.18E-05	9.91E-06	2.14E-05	4.93E-05
CA40	1.93E+00	1.93E+00	1.93E+00	1.93E+00	1.93E+00	1.93E+00
CA41	9.57E-04	1.54E-03	2.21E-03	9.54E-04	1.53E-03	2.20E-03
CA42	1.35E-02	1.35E-02	1.35E-02	1.35E-02	1.35E-02	1.35E-02
CA43	2.80E-03	2.80E-03	2.82E-03	2.80E-03	2.80E-03	2.82E-03
CA44	4.60E-02	4.61E-02	4.63E-02	4.60E-02	4.61E-02	4.63E-02
CA46	1.00E-04	1.13E-04	1.44E-04	1.00E-04	1.13E-04	1.44E-04
CA48	4.54E-03	4.54E-03	4.53E-03	4.54E-03	4.54E-03	4.53E-03
SC45	6.17E-05	9.81E-05	1.43E-04	6.17E-05	9.81E-05	1.43E-04
TI46	3.59E+01	3.59E+01	3.59E+01	3.59E+01	3.59E+01	3.59E+01
TI47	3.31E+01	3.31E+01	3.31E+01	3.31E+01	3.31E+01	3.31E+01
TI48	3.34E+02	3.33E+02	3.33E+02	3.34E+02	3.33E+02	3.33E+02
TI49	2.60E+01	2.65E+01	2.72E+01	2.60E+01	2.65E+01	2.72E+01
TI50	2.46E+01	2.46E+01	2.47E+01	2.46E+01	2.46E+01	2.47E+01
V50	3.86E-02	4.17E-02	5.04E-02	3.86E-02	4.17E-02	5.04E-02
V51	1.54E+01	1.65E+01	1.77E+01	1.54E+01	1.65E+01	1.77E+01
CR50	4.29E+02	4.28E+02	4.26E+02	4.29E+02	4.28E+02	4.26E+02
CR52	8.62E+03	8.61E+03	8.61E+03	8.62E+03	8.61E+03	8.61E+03
CR53	9.97E+02	9.97E+02	9.97E+02	9.97E+02	9.97E+02	9.97E+02
CR54	2.57E+02	2.60E+02	2.64E+02	2.57E+02	2.60E+02	2.64E+02
MN54	2.08E-37	2.16E-37	2.80E-37	0.00E+00	0.00E+00	0.00E+00
MN55	8.76E+02	8.74E+02	8.72E+02	8.76E+02	8.74E+02	8.72E+02
FE54	1.58E+03	1.57E+03	1.57E+03	1.58E+03	1.57E+03	1.57E+03
FE55	1.68E-12	2.20E-12	2.55E-12	0.00E+00	0.00E+00	0.00E+00
FE56	2.58E+04	2.58E+04	2.58E+04	2.58E+04	2.58E+04	2.58E+04
FE57	6.30E+02	6.38E+02	6.49E+02	6.30E+02	6.38E+02	6.49E+02
FE58	8.55E+01	8.61E+01	8.74E+01	8.55E+01	8.61E+01	8.74E+01
CO59	1.47E+02	1.44E+02	1.41E+02	1.47E+02	1.44E+02	1.41E+02
CO60	7.72E-06	1.09E-05	1.45E-05	1.09E-28	1.54E-28	2.05E-28
NI58	1.09E+04	1.09E+04	1.09E+04	1.09E+04	1.09E+04	1.09E+04
NI59	1.41E+01	2.21E+01	3.11E+01	1.40E+01	2.20E+01	3.10E+01
NI60	4.33E+03	4.33E+03	4.34E+03	4.33E+03	4.33E+03	4.34E+03
NI61	1.94E+02	1.97E+02	1.99E+02	1.94E+02	1.97E+02	1.99E+02
NI62	6.13E+02	6.12E+02	6.10E+02	6.13E+02	6.12E+02	6.10E+02
NI63	1.28E+00	2.02E+00	2.87E+00	6.27E-02	9.93E-02	1.41E-01
NI64	1.61E+02	1.61E+02	1.61E+02	1.61E+02	1.61E+02	1.61E+02
CU63	1.55E+01	1.63E+01	1.73E+01	1.67E+01	1.82E+01	2.00E+01

Table C-4 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
CU65	6.55E+00	6.61E+00	6.68E+00	6.55E+00	6.61E+00	6.68E+00
ZN64	1.91E+01	1.91E+01	1.91E+01	1.91E+01	1.91E+01	1.91E+01
ZN66	1.13E+01	1.13E+01	1.13E+01	1.13E+01	1.13E+01	1.13E+01
ZN67	1.68E+00	1.67E+00	1.66E+00	1.68E+00	1.67E+00	1.66E+00
ZN68	7.87E+00	7.87E+00	7.88E+00	7.87E+00	7.87E+00	7.88E+00
ZN70	2.70E-01	2.71E-01	2.75E-01	2.70E-01	2.71E-01	2.75E-01
GA69	1.87E-02	2.99E-02	4.80E-02	1.87E-02	2.99E-02	4.80E-02
GA71	3.04E-05	4.92E-05	7.19E-05	3.04E-05	4.92E-05	7.19E-05
GE70	7.19E-05	1.85E-04	5.08E-04	7.19E-05	1.85E-04	5.08E-04
GE72	1.08E-02	1.88E-02	3.39E-02	1.08E-02	1.88E-02	3.39E-02
GE73	2.20E-02	3.90E-02	6.61E-02	2.20E-02	3.90E-02	6.61E-02
GE74	4.58E-02	8.53E-02	1.48E-01	4.58E-02	8.53E-02	1.48E-01
GE76	2.28E-01	4.44E-01	7.27E-01	2.28E-01	4.44E-01	7.27E-01
AS75	9.52E-02	1.77E-01	2.91E-01	9.52E-02	1.77E-01	2.91E-01
SE76	1.69E-03	5.12E-03	1.50E-02	1.69E-03	5.12E-03	1.50E-02
SE77	4.77E-01	9.17E-01	1.45E+00	4.77E-01	9.17E-01	1.45E+00
SE78	1.10E+00	2.17E+00	3.62E+00	1.10E+00	2.17E+00	3.62E+00
SE79	2.68E+00	5.28E+00	8.65E+00	2.67E+00	5.26E+00	8.61E+00
SE80	6.03E+00	1.20E+01	1.95E+01	6.03E+00	1.20E+01	1.95E+01
SE82	1.51E+01	3.03E+01	4.86E+01	1.51E+01	3.03E+01	4.86E+01
BR79	2.90E-03	5.76E-03	9.54E-03	1.43E-02	2.83E-02	4.64E-02
BR81	9.96E+00	1.95E+01	3.04E+01	9.96E+00	1.95E+01	3.04E+01
KR80	1.12E-04	2.09E-04	3.83E-04	1.12E-04	2.09E-04	3.83E-04
KR81	8.84E-06	2.01E-05	5.01E-05	8.83E-06	2.00E-05	5.01E-05
KR82	3.08E-01	9.07E-01	2.39E+00	3.08E-01	9.07E-01	2.39E+00
KR83	2.08E+01	3.79E+01	5.32E+01	2.08E+01	3.79E+01	5.32E+01
KR84	4.77E+01	1.01E+02	1.68E+02	4.77E+01	1.01E+02	1.68E+02
KR85	1.68E-02	3.20E-02	4.62E-02	9.86E-14	1.87E-13	2.71E-13
KR86	8.33E+01	1.70E+02	2.68E+02	8.33E+01	1.70E+02	2.68E+02
RB85	5.35E+01	1.09E+02	1.72E+02	5.35E+01	1.09E+02	1.72E+02
RB87	1.07E+02	2.19E+02	3.43E+02	1.07E+02	2.19E+02	3.43E+02
SR86	1.16E-01	3.88E-01	1.16E+00	1.16E-01	3.88E-01	1.16E+00
SR87	3.93E-03	7.57E-03	1.91E-02	3.93E-03	7.57E-03	1.91E-02
SR88	1.54E+02	3.14E+02	4.93E+02	1.54E+02	3.14E+02	4.93E+02
SR90	2.19E+01	4.36E+01	6.56E+01	1.60E-03	3.20E-03	4.81E-03
Y89	2.00E+02	4.10E+02	6.40E+02	2.00E+02	4.10E+02	6.40E+02
Y90	5.48E-03	1.09E-02	1.65E-02	4.02E-07	8.02E-07	1.21E-06
ZR90	2.66E+05	2.66E+05	2.66E+05	2.66E+05	2.66E+05	2.66E+05
ZR91	5.85E+04	5.87E+04	5.88E+04	5.85E+04	5.87E+04	5.88E+04
ZR92	9.05E+04	9.09E+04	9.13E+04	9.05E+04	9.09E+04	9.13E+04

Table C-4 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
ZR93	4.72E+02	8.88E+02	1.44E+03	4.72E+02	8.88E+02	1.44E+03
ZR94	9.40E+04	9.43E+04	9.47E+04	9.40E+04	9.43E+04	9.47E+04
ZR96	1.57E+04	1.61E+04	1.65E+04	1.57E+04	1.61E+04	1.65E+04
NB93	1.59E+02	1.58E+02	1.58E+02	1.59E+02	1.59E+02	1.58E+02
NB93M	3.96E-03	7.46E-03	1.21E-02	3.99E-03	7.50E-03	1.22E-02
NB94	2.89E-01	4.59E-01	8.02E-01	2.85E-01	4.53E-01	7.91E-01
MO92	1.42E+00	1.42E+00	1.42E+00	1.42E+00	1.42E+00	1.42E+00
MO93	3.50E-04	5.61E-04	9.66E-04	3.23E-04	5.19E-04	8.92E-04
MO94	9.11E-01	9.11E-01	9.11E-01	9.15E-01	9.17E-01	9.22E-01
MO95	3.76E+02	7.25E+02	1.14E+03	3.76E+02	7.25E+02	1.14E+03
MO96	1.05E+01	3.74E+01	1.11E+02	1.05E+01	3.74E+01	1.11E+02
MO97	4.04E+02	7.86E+02	1.30E+03	4.04E+02	7.86E+02	1.30E+03
MO98	3.73E+02	7.40E+02	1.22E+03	3.73E+02	7.40E+02	1.22E+03
MO100	4.25E+02	8.44E+02	1.40E+03	4.25E+02	8.44E+02	1.40E+03
TC98	1.27E-03	4.27E-03	1.32E-02	1.27E-03	4.27E-03	1.32E-02
TC99	3.70E+02	7.06E+02	1.08E+03	3.69E+02	7.06E+02	1.08E+03
RU99	1.21E-01	2.34E-01	3.65E-01	6.02E-01	1.15E+00	1.77E+00
RU100	2.65E+01	8.88E+01	2.55E+02	2.65E+01	8.88E+01	2.55E+02
RU101	3.58E+02	6.98E+02	1.14E+03	3.58E+02	6.98E+02	1.14E+03
RU102	3.52E+02	7.04E+02	1.24E+03	3.52E+02	7.04E+02	1.24E+03
RU104	2.58E+02	4.97E+02	8.87E+02	2.58E+02	4.97E+02	8.87E+02
RU106	1.29E-28	1.70E-28	2.21E-28	0.00E+00	0.00E+00	0.00E+00
RH102	9.56E-15	2.72E-14	6.54E-14	2.88E-56	8.18E-56	1.97E-55
RH103	2.55E+02	4.10E+02	5.36E+02	2.55E+02	4.10E+02	5.36E+02
RH106	1.21E-34	1.60E-34	2.07E-34	0.00E+00	0.00E+00	0.00E+00
PD104	6.67E+01	2.17E+02	5.54E+02	6.67E+01	2.17E+02	5.54E+02
PD105	1.88E+02	3.57E+02	6.39E+02	1.88E+02	3.57E+02	6.39E+02
PD106	1.70E+02	3.27E+02	6.31E+02	1.70E+02	3.27E+02	6.31E+02
PD107	1.07E+02	2.04E+02	3.83E+02	1.07E+02	2.04E+02	3.83E+02
PD108	7.40E+01	1.41E+02	2.63E+02	7.40E+01	1.41E+02	2.63E+02
PD110	2.41E+01	4.59E+01	8.83E+01	2.41E+01	4.59E+01	8.83E+01
AG107	4.80E-02	4.65E-02	4.43E-02	5.25E-02	5.52E-02	6.06E-02
AG108	5.38E-13	8.36E-13	1.26E-12	6.06E-14	9.42E-14	1.42E-13
AG108M	1.70E-04	2.65E-04	3.98E-04	1.92E-05	2.98E-05	4.48E-05
AG109	4.12E+01	7.11E+01	1.16E+02	4.12E+01	7.11E+01	1.16E+02
AG109M	8.35E-34	8.87E-34	1.09E-33	0.00E+00	0.00E+00	0.00E+00
CD106	3.08E-01	3.07E-01	3.07E-01	3.08E-01	3.07E-01	3.07E-01
CD108	2.19E-01	2.21E-01	2.24E-01	2.19E-01	2.21E-01	2.25E-01
CD109	8.45E-28	8.98E-28	1.11E-27	0.00E+00	0.00E+00	0.00E+00
CD110	1.38E+01	3.52E+01	1.02E+02	1.38E+01	3.52E+01	1.02E+02
CD111	1.67E+01	2.85E+01	5.26E+01	1.67E+01	2.85E+01	5.26E+01
CD112	1.41E+01	2.11E+01	3.53E+01	1.41E+01	2.11E+01	3.53E+01

Table C-4 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
CD113	1.47E-01	1.54E-01	1.99E-01	1.47E-01	1.54E-01	1.99E-01
CD113M	1.03E-03	1.98E-03	4.22E-03	5.77E-12	1.10E-11	2.35E-11
CD114	2.14E+01	3.03E+01	4.72E+01	2.14E+01	3.03E+01	4.72E+01
CD116	6.05E+00	9.14E+00	1.49E+01	6.05E+00	9.14E+00	1.49E+01
IN113	1.00E+00	1.52E+00	2.51E+00	1.00E+00	1.53E+00	2.51E+00
IN115	1.95E+00	2.09E+00	2.22E+00	1.95E+00	2.09E+00	2.22E+00
SN112	7.98E+01	7.93E+01	7.82E+01	7.98E+01	7.93E+01	7.82E+01
SN114	5.50E+01	5.51E+01	5.54E+01	5.50E+01	5.51E+01	5.54E+01
SN115	2.99E+01	2.90E+01	2.81E+01	2.99E+01	2.90E+01	2.81E+01
SN116	1.22E+03	1.22E+03	1.22E+03	1.22E+03	1.22E+03	1.22E+03
SN117	6.61E+02	6.66E+02	6.78E+02	6.61E+02	6.66E+02	6.78E+02
SN118	2.06E+03	2.06E+03	2.06E+03	2.06E+03	2.06E+03	2.06E+03
SN119	7.51E+02	7.59E+02	7.79E+02	7.51E+02	7.59E+02	7.79E+02
SN120	2.80E+03	2.81E+03	2.81E+03	2.80E+03	2.81E+03	2.81E+03
SN121M	3.73E-03	6.03E-03	1.03E-02	1.45E-05	2.35E-05	4.00E-05
SN122	4.09E+02	4.12E+02	4.18E+02	4.09E+02	4.12E+02	4.18E+02
SN124	5.03E+02	5.06E+02	5.11E+02	5.03E+02	5.06E+02	5.11E+02
SN126	1.35E+01	2.46E+01	4.35E+01	1.34E+01	2.46E+01	4.34E+01
SB121	6.70E+00	1.11E+01	1.89E+01	6.71E+00	1.11E+01	1.89E+01
SB123	5.43E+00	9.38E+00	1.62E+01	5.43E+00	9.38E+00	1.62E+01
SB125	1.37E-10	1.96E-10	2.75E-10	0.00E+00	0.00E+00	0.00E+00
SB126	6.39E-07	1.17E-06	2.07E-06	6.37E-07	1.17E-06	2.06E-06
SB126M	4.86E-09	8.89E-09	1.57E-08	4.85E-09	8.87E-09	1.57E-08
TE122	2.88E-01	7.77E-01	2.39E+00	2.88E-01	7.77E-01	2.39E+00
TE123	3.25E-03	1.12E-02	4.47E-02	3.25E-03	1.12E-02	4.47E-02
TE124	1.28E-01	3.87E-01	1.29E+00	1.28E-01	3.87E-01	1.29E+00
TE125	1.25E+01	2.18E+01	3.88E+01	1.25E+01	2.18E+01	3.88E+01
TE125M	1.92E-12	2.73E-12	3.85E-12	0.00E+00	0.00E+00	0.00E+00
TE126	3.66E-01	7.58E-01	1.68E+00	4.03E-01	8.26E-01	1.80E+00
TE128	5.25E+01	9.99E+01	1.72E+02	5.25E+01	9.99E+01	1.72E+02
TE130	1.65E+02	3.21E+02	5.47E+02	1.65E+02	3.21E+02	5.47E+02
I127	2.76E+01	5.00E+01	8.38E+01	2.76E+01	5.00E+01	8.38E+01
I129	8.70E+01	1.63E+02	2.72E+02	8.70E+01	1.63E+02	2.72E+02
XE128	9.03E-01	2.87E+00	9.18E+00	9.03E-01	2.87E+00	9.18E+00
XE129	3.38E-03	1.61E-02	8.26E-02	4.92E-03	1.90E-02	8.74E-02
XE130	3.19E+00	1.02E+01	2.71E+01	3.19E+00	1.02E+01	2.71E+01
XE131	2.27E+02	3.81E+02	4.86E+02	2.27E+02	3.81E+02	4.86E+02
XE132	4.67E+02	9.87E+02	1.81E+03	4.67E+02	9.87E+02	1.81E+03
XE134	6.65E+02	1.33E+03	2.21E+03	6.65E+02	1.33E+03	2.21E+03
XE136	1.03E+03	1.99E+03	3.22E+03	1.03E+03	1.99E+03	3.22E+03
CS133	5.45E+02	1.02E+03	1.50E+03	5.45E+02	1.02E+03	1.50E+03

Table C-4 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
CS134	9.81E-14	2.64E-13	5.82E-13	0.00E+00	0.00E+00	0.00E+00
CS135	1.67E+02	3.90E+02	7.59E+02	1.67E+02	3.90E+02	7.59E+02
CS137	5.44E+01	1.06E+02	1.71E+02	5.27E-03	1.03E-02	1.65E-02
BA132	3.82E-04	1.26E-03	3.78E-03	3.82E-04	1.26E-03	3.78E-03
BA134	4.75E+01	1.53E+02	4.17E+02	4.75E+01	1.53E+02	4.17E+02
BA135	3.85E-02	3.41E-01	2.44E+00	5.86E-02	3.87E-01	2.53E+00
BA136	7.19E+00	2.07E+01	5.98E+01	7.19E+00	2.07E+01	5.98E+01
BA137	5.06E+02	1.01E+03	1.69E+03	5.60E+02	1.12E+03	1.86E+03
BA137M	8.32E-06	1.62E-05	2.61E-05	8.06E-10	1.57E-09	2.53E-09
BA138	5.74E+02	1.15E+03	1.90E+03	5.74E+02	1.15E+03	1.90E+03
LA138	2.81E-03	5.20E-03	6.60E-03	2.81E-03	5.20E-03	6.60E-03
LA139	5.52E+02	1.10E+03	1.80E+03	5.52E+02	1.10E+03	1.80E+03
CE140	5.55E+02	1.11E+03	1.85E+03	5.55E+02	1.11E+03	1.85E+03
CE142	5.07E+02	1.02E+03	1.67E+03	5.07E+02	1.02E+03	1.67E+03
CE144	4.49E-37	5.11E-37	5.04E-37	0.00E+00	0.00E+00	0.00E+00
PR141	5.10E+02	1.01E+03	1.64E+03	5.10E+02	1.01E+03	1.64E+03
PR144	1.89E-41	2.16E-41	2.13E-41	0.00E+00	0.00E+00	0.00E+00
PR144M	9.47E-44	1.08E-43	1.06E-43	0.00E+00	0.00E+00	0.00E+00
ND142	6.07E+00	2.24E+01	6.08E+01	6.07E+00	2.24E+01	6.08E+01
ND143	4.16E+02	7.34E+02	1.04E+03	4.16E+02	7.34E+02	1.04E+03
ND144	5.28E+02	1.17E+03	2.06E+03	5.28E+02	1.17E+03	2.06E+03
ND145	3.17E+02	6.09E+02	9.19E+02	3.17E+02	6.09E+02	9.19E+02
ND146	2.94E+02	6.19E+02	1.10E+03	2.94E+02	6.19E+02	1.10E+03
ND148	1.69E+02	3.34E+02	5.52E+02	1.69E+02	3.34E+02	5.52E+02
ND150	8.23E+01	1.61E+02	2.74E+02	8.23E+01	1.61E+02	2.74E+02
PM146	6.42E-09	1.45E-08	2.64E-08	8.21E-31	1.86E-30	3.38E-30
PM147	3.33E-10	4.05E-10	3.78E-10	0.00E+00	0.00E+00	0.00E+00
SM146	1.53E-03	6.13E-03	1.93E-02	1.53E-03	6.13E-03	1.93E-02
SM147	1.26E+02	1.95E+02	2.27E+02	1.26E+02	1.95E+02	2.27E+02
SM148	5.76E+01	1.60E+02	3.37E+02	5.76E+01	1.60E+02	3.37E+02
SM149	2.98E+00	3.25E+00	3.83E+00	2.98E+00	3.25E+00	3.83E+00
SM150	1.40E+02	2.73E+02	4.38E+02	1.40E+02	2.73E+02	4.38E+02
SM151	4.94E+00	6.69E+00	1.01E+01	2.27E-01	3.07E-01	4.66E-01
SM152	6.89E+01	1.17E+02	1.57E+02	6.89E+01	1.17E+02	1.57E+02
SM154	1.77E+01	3.40E+01	6.13E+01	1.77E+01	3.40E+01	6.13E+01
EU150	1.27E-08	3.94E-08	1.29E-07	5.74E-12	1.78E-11	5.82E-11
EU151	5.74E+00	7.77E+00	1.18E+01	1.04E+01	1.42E+01	2.15E+01
EU152	1.83E-04	3.36E-04	4.97E-04	2.56E-13	4.70E-13	6.97E-13
EU153	4.95E+01	1.09E+02	1.92E+02	4.95E+01	1.09E+02	1.92E+02
EU154	3.33E-03	9.78E-03	2.25E-02	3.32E-17	9.75E-17	2.24E-16
EU155	4.34E-06	1.02E-05	2.49E-05	2.27E-30	5.32E-30	1.30E-29

Table C-4 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
GD152	2.53E+00	2.29E+00	1.87E+00	2.53E+00	2.29E+00	1.87E+00
GD154	3.92E+01	5.97E+01	1.05E+02	3.92E+01	5.97E+01	1.05E+02
GD155	8.10E+00	1.34E+01	3.19E+01	8.10E+00	1.34E+01	3.19E+01
GD156	5.44E+02	5.65E+02	6.43E+02	5.44E+02	5.65E+02	6.43E+02
GD157	2.11E+00	2.26E+00	3.34E+00	2.11E+00	2.26E+00	3.34E+00
GD158	6.59E+02	6.75E+02	7.13E+02	6.59E+02	6.75E+02	7.13E+02
GD160	3.47E+02	3.47E+02	3.45E+02	3.47E+02	3.47E+02	3.45E+02
TB159	1.52E+01	2.36E+01	3.90E+01	1.52E+01	2.36E+01	3.90E+01
DY160	1.42E+00	3.15E+00	7.49E+00	1.42E+00	3.15E+00	7.49E+00
DY161	1.73E+00	2.83E+00	6.10E+00	1.73E+00	2.83E+00	6.10E+00
DY162	7.63E-01	1.57E+00	3.68E+00	7.63E-01	1.57E+00	3.68E+00
DY163	3.28E-01	9.21E-01	2.93E+00	3.28E-01	9.21E-01	2.93E+00
DY164	5.60E-02	1.70E-01	7.17E-01	5.60E-02	1.70E-01	7.17E-01
HO165	6.26E-02	2.38E-01	1.19E+00	6.26E-02	2.38E-01	1.19E+00
HO166M	3.22E-04	1.69E-03	1.34E-02	2.55E-04	1.34E-03	1.06E-02
ER166	1.48E-02	4.72E-02	2.79E-01	1.49E-02	4.76E-02	2.82E-01
ER167	2.30E-03	3.50E-03	9.06E-03	2.30E-03	3.50E-03	9.06E-03
ER168	2.13E-03	5.23E-03	1.63E-02	2.13E-03	5.23E-03	1.63E-02
TM169	9.66E-06	3.47E-05	1.47E-04	9.66E-06	3.47E-05	1.47E-04
TM171	1.07E-23	7.01E-23	5.88E-22	0.00E+00	1.03E-05	7.44E-05
YB170	0.00E+00	1.03E-05	7.44E-05	0.00E+00	0.00E+00	7.20E-06
YB171	0.00E+00	0.00E+00	7.20E-06	0.00E+00	0.00E+00	0.00E+00
LU175	2.37E-02	2.86E-02	2.77E-02	2.37E-02	2.86E-02	2.77E-02
LU176	5.50E-04	9.17E-04	1.28E-03	5.50E-04	9.17E-04	1.28E-03
HF174	3.79E-02	2.75E-02	1.82E-02	3.79E-02	2.75E-02	1.82E-02
HF176	1.61E+00	1.36E+00	9.82E-01	1.61E+00	1.36E+00	9.82E-01
HF177	7.62E-01	3.85E-01	1.96E-01	7.62E-01	3.85E-01	1.96E-01
HF178	9.64E+00	6.63E+00	3.20E+00	9.64E+00	6.63E+00	3.20E+00
HF179	1.24E+01	1.40E+01	1.36E+01	1.24E+01	1.40E+01	1.36E+01
HF180	1.68E+01	1.87E+01	2.25E+01	1.68E+01	1.87E+01	2.25E+01
HF182	1.74E-03	2.44E-03	4.13E-03	1.74E-03	2.44E-03	4.13E-03
TA181	3.89E-01	6.09E-01	9.61E-01	3.89E-01	6.09E-01	9.61E-01
TA182	6.09E-11	8.54E-11	1.45E-10	6.09E-11	8.54E-11	1.44E-10
W180	1.50E-02	1.43E-02	1.30E-02	1.50E-02	1.43E-02	1.30E-02
W182	2.64E+00	2.33E+00	1.85E+00	2.64E+00	2.33E+00	1.85E+00
W183	2.25E+00	2.45E+00	2.74E+00	2.25E+00	2.45E+00	2.74E+00
W184	4.14E+00	4.31E+00	4.72E+00	4.14E+00	4.31E+00	4.72E+00
W186	2.96E+00	2.60E+00	2.02E+00	2.96E+00	2.60E+00	2.02E+00
RE185	2.13E-02	2.87E-02	3.81E-02	2.13E-02	2.87E-02	3.81E-02
RE187	6.47E-01	9.18E-01	1.27E+00	6.47E-01	9.18E-01	1.27E+00
OS186	5.80E-03	1.54E-02	3.96E-02	5.80E-03	1.54E-02	3.96E-02

Table C-4 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
OS187	9.05E-10	1.29E-09	1.80E-09	4.50E-09	6.38E-09	8.86E-09
OS188	6.59E-02	1.53E-01	3.64E-01	6.59E-02	1.53E-01	3.64E-01
OS189	1.17E-03	4.26E-03	1.72E-02	1.17E-03	4.26E-03	1.72E-02
OS190	9.32E-05	5.55E-04	4.16E-03	9.32E-05	5.55E-04	4.16E-03
IR191	0.00E+00	2.89E-06	2.75E-05	0.00E+00	2.89E-06	2.75E-05
IR192	3.01E-14	4.40E-13	7.83E-12	9.52E-15	1.39E-13	2.48E-12
IR192M	0.00E+00	0.00E+00	9.31E-09	0.00E+00	0.00E+00	2.95E-09
IR193	0.00E+00	0.00E+00	3.02E-06	0.00E+00	0.00E+00	3.16E-06
PT192	0.00E+00	0.00E+00	2.50E-05	0.00E+00	0.00E+00	2.50E-05
PT193	0.00E+00	8.78E-09	3.20E-07	0.00E+00	5.04E-09	1.84E-07
TL206	4.67E-17	7.49E-17	1.25E-16	4.67E-17	7.49E-17	1.25E-16
TL207	1.56E-13	2.26E-13	2.98E-13	9.03E-13	9.72E-13	8.28E-13
TL208	3.22E-12	1.34E-11	5.09E-11	6.86E-14	2.85E-13	1.08E-12
TL209	1.84E-17	4.95E-17	1.51E-16	6.84E-16	1.33E-15	2.54E-15
PB204	1.38E-02	1.38E-02	1.37E-02	1.38E-02	1.38E-02	1.37E-02
PB205	1.83E-05	2.94E-05	4.67E-05	1.83E-05	2.94E-05	4.67E-05
PB206	2.32E-01	2.28E-01	2.23E-01	2.32E-01	2.28E-01	2.23E-01
PB207	2.28E-01	2.32E-01	2.37E-01	2.28E-01	2.32E-01	2.37E-01
PB208	5.26E-01	5.27E-01	5.29E-01	5.26E-01	5.27E-01	5.31E-01
PB209	7.66E-14	2.06E-13	6.29E-13	2.85E-12	5.55E-12	1.06E-11
PB210	1.14E-07	1.78E-07	2.37E-07	4.65E-06	8.39E-06	1.57E-05
PB211	1.20E-12	1.75E-12	2.30E-12	6.98E-12	7.52E-12	6.41E-12
PB212	1.90E-09	7.90E-09	3.00E-08	4.05E-11	1.68E-10	6.39E-10
PB214	4.77E-13	7.52E-13	1.04E-12	1.23E-11	2.23E-11	4.20E-11
BI208	2.63E-06	4.22E-06	7.95E-06	2.63E-06	4.22E-06	7.95E-06
BI209	4.00E-01	4.00E-01	4.00E-01	4.00E-01	4.00E-01	4.00E-01
BI210	6.99E-11	1.10E-10	1.46E-10	2.86E-09	5.16E-09	9.63E-09
BI210M	1.80E-05	2.89E-05	4.83E-05	1.80E-05	2.89E-05	4.83E-05
BI211	7.10E-14	1.03E-13	1.36E-13	4.12E-13	4.44E-13	3.78E-13
BI212	1.80E-10	7.49E-10	2.85E-09	3.84E-12	1.60E-11	6.06E-11
BI213	1.80E-14	4.84E-14	1.48E-13	6.69E-13	1.30E-12	2.49E-12
BI214	3.54E-13	5.59E-13	7.70E-13	9.15E-12	1.66E-11	3.12E-11
PO210	1.93E-09	3.03E-09	4.03E-09	7.91E-08	1.43E-07	2.66E-07
PO211	8.71E-19	1.27E-18	1.67E-18	5.06E-18	5.44E-18	4.64E-18
PO212	9.53E-21	3.96E-20	1.51E-19	2.03E-22	8.44E-22	3.21E-21
PO213	2.70E-23	7.27E-23	2.22E-22	1.00E-21	1.96E-21	3.73E-21
PO214	4.87E-20	7.69E-20	1.06E-19	1.26E-18	2.28E-18	4.29E-18
PO215	1.01E-18	1.47E-18	1.93E-18	5.85E-18	6.29E-18	5.37E-18
PO216	7.58E-15	3.15E-14	1.20E-13	1.62E-16	6.71E-16	2.55E-15
PO218	5.53E-14	8.72E-14	1.20E-13	1.43E-12	2.59E-12	4.87E-12
AT217	2.16E-19	5.82E-19	1.78E-18	8.04E-18	1.57E-17	2.99E-17
RN219	2.28E-15	3.32E-15	4.37E-15	1.33E-14	1.43E-14	1.22E-14

Table C-4 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
RN220	2.86E-12	1.19E-11	4.52E-11	6.10E-14	2.53E-13	9.62E-13
RN222	1.02E-10	1.60E-10	2.21E-10	2.63E-09	4.76E-09	8.96E-09
FR221	1.96E-15	5.28E-15	1.61E-14	7.30E-14	1.42E-13	2.71E-13
FR223	1.06E-14	1.54E-14	2.03E-14	6.15E-14	6.62E-14	5.64E-14
RA223	5.80E-10	8.44E-10	1.11E-09	3.37E-09	3.62E-09	3.09E-09
RA224	1.66E-08	6.89E-08	2.62E-07	3.53E-10	1.47E-09	5.57E-09
RA225	8.88E-12	2.39E-11	7.29E-11	3.30E-10	6.43E-10	1.23E-09
RA226	1.58E-05	2.50E-05	3.44E-05	4.09E-04	7.41E-04	1.39E-03
RA228	2.23E-12	4.57E-12	6.45E-12	1.24E-11	2.50E-11	3.48E-11
AC225	6.00E-12	1.61E-11	4.93E-11	2.23E-10	4.34E-10	8.29E-10
AC227	4.10E-07	5.97E-07	7.86E-07	2.38E-06	2.57E-06	2.19E-06
AC228	2.32E-16	4.77E-16	6.73E-16	1.29E-15	2.61E-15	3.63E-15
TH227	9.53E-10	1.39E-09	1.82E-09	5.53E-09	5.95E-09	5.08E-09
TH228	3.22E-06	1.34E-05	5.09E-05	6.86E-08	2.85E-07	1.08E-06
TH229	1.64E-06	4.40E-06	1.34E-05	6.08E-05	1.19E-04	2.26E-04
TH230	3.66E-02	5.94E-02	8.96E-02	2.05E-01	3.85E-01	7.68E-01
TH231	3.19E-08	3.17E-08	2.23E-08	3.21E-08	3.19E-08	2.26E-08
TH232	5.25E-03	1.08E-02	1.52E-02	2.64E-02	5.34E-02	7.42E-02
TH234	1.41E-05	1.38E-05	1.34E-05	1.41E-05	1.38E-05	1.34E-05
PA231	8.61E-04	1.16E-03	1.40E-03	3.89E-03	4.16E-03	3.52E-03
PA233	9.46E-06	1.98E-05	3.89E-05	2.30E-05	3.81E-05	6.42E-05
PA234	2.12E-10	2.07E-10	2.02E-10	2.12E-10	2.07E-10	2.02E-10
PA234M	4.74E-10	4.64E-10	4.52E-10	4.74E-10	4.64E-10	4.52E-10
U232	1.20E-04	4.99E-04	1.90E-03	2.56E-06	1.06E-05	4.04E-05
U233	7.39E-03	1.70E-02	3.46E-02	7.08E-02	1.29E-01	2.33E-01
U234	1.39E+02	2.44E+02	4.40E+02	1.56E+02	3.11E+02	6.74E+02
U235	7.84E+03	7.80E+03	5.49E+03	7.89E+03	7.85E+03	5.56E+03
U236	1.79E+03	3.63E+03	5.01E+03	1.84E+03	3.70E+03	5.14E+03
U237	2.39E-07	3.16E-07	4.34E-07	5.68E-12	5.50E-11	5.34E-10
U238	9.68E+05	9.48E+05	9.24E+05	9.68E+05	9.48E+05	9.24E+05
U240	9.18E-14	4.53E-13	2.35E-12	9.18E-14	4.53E-13	2.35E-12
NP235	1.84E-34	5.61E-34	1.76E-33	0.00E+00	0.00E+00	0.00E+00
NP236	1.05E-04	3.32E-04	1.07E-03	1.05E-04	3.31E-04	1.07E-03
NP237	2.79E+02	5.84E+02	1.15E+03	6.76E+02	1.12E+03	1.89E+03
NP238	9.93E-08	2.81E-07	5.92E-07	1.60E-08	4.54E-08	9.56E-08
NP239	2.23E-05	8.26E-05	2.44E-04	2.15E-05	7.96E-05	2.35E-04
NP240M	8.03E-16	3.96E-15	2.06E-14	8.03E-16	3.96E-15	2.06E-14
PU236	2.34E-10	7.41E-10	2.39E-09	2.33E-10	7.40E-10	2.39E-09
PU238	1.80E+01	7.13E+01	2.47E+02	8.34E-01	3.22E+00	1.09E+01
PU239	4.22E+03	4.82E+03	6.46E+03	4.17E+03	4.77E+03	6.39E+03
PU240	1.22E+03	1.92E+03	3.28E+03	1.17E+03	1.84E+03	3.14E+03

Table C-4 (continued).

	DECAY TIME 100 YEARS			DECAY TIME 500 YEARS		
Burn-up (MWd/MTIHM)	15000	30000	50000	15000	30000	50000
Enrichment (%)	1.79	2.93	3.74	1.79	2.93	3.74
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
PU241	7.71E+00	1.02E+01	1.40E+01	1.84E-04	1.78E-03	1.73E-02
PU242	2.02E+02	4.60E+02	8.32E+02	2.02E+02	4.60E+02	8.32E+02
PU243	1.34E-15	3.90E-14	1.14E-12	1.34E-15	3.90E-14	1.14E-12
PU244	4.80E-03	2.37E-02	1.23E-01	4.80E-03	2.37E-02	1.23E-01
AM241	8.47E+02	1.14E+03	1.59E+03	4.50E+02	6.07E+02	8.42E+02
AM242	6.33E-06	1.79E-05	3.78E-05	1.02E-06	2.90E-06	6.10E-06
AM242M	5.29E-01	1.50E+00	3.16E+00	8.54E-02	2.42E-01	5.10E-01
AM243	2.60E+01	9.61E+01	2.84E+02	2.50E+01	9.26E+01	2.73E+02
AM244	0.00E+00	3.30E-17	1.72E-16	0.00E+00	3.30E-17	1.72E-16
CM242	1.28E-03	3.63E-03	7.64E-03	2.07E-04	5.86E-04	1.23E-03
CM243	8.35E-03	3.93E-02	1.35E-01	4.98E-07	2.34E-06	8.01E-06
CM244	9.75E-02	6.46E-01	3.80E+00	2.19E-08	1.45E-07	8.52E-07
CM245	1.14E-01	1.10E+00	1.07E+01	1.10E-01	1.07E+00	1.04E+01
CM246	6.45E-03	1.12E-01	1.77E+00	6.08E-03	1.06E-01	1.67E+00
CM247	3.76E-05	1.09E-03	3.21E-02	3.76E-05	1.09E-03	3.21E-02
CM248	1.16E-06	5.87E-05	3.18E-03	1.16E-06	5.87E-05	3.18E-03
BK249	3.43E-43	2.39E-41	1.90E-39	0.00E+00	0.00E+00	0.00E+00
CF249	7.05E-09	5.18E-07	4.35E-05	3.20E-09	2.35E-07	1.97E-05
CF250	6.26E-12	5.53E-10	4.92E-08	2.49E-19	2.10E-17	2.27E-15
CF251	0.00E+00	4.30E-08	5.06E-06	0.00E+00	3.16E-08	3.71E-06
CF252	4.59E-22	8.72E-20	1.58E-17	0.00E+00	0.00E+00	0.00E+00
ES254	0.00E+00	0.00E+00	1.74E-49	0.00E+00	0.00E+00	0.00E+00
TOTAL	1.73E+06	1.73E+06	1.73E+06	1.73E+06	1.73E+06	1.73E+06
Decay Heat (Watts/MT)	150	257	462	52	71	102

Table C-5 100 GWd/MT PWR Fuel Radionuclide Inventory at 5, 30, 100 and 500 years

Burn-up (MWd/MTIHM)	**100,000**			
Enrichment (%)	8.32 % U-235			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
H 3	1.13E-01	2.77E-02	5.45E-04	9.63E-14
HE 4	0.00E+00	0.00E+00	0.00E+00	0.00E+00
LI 6	0.00E+00	0.00E+00	0.00E+00	0.00E+00
LI 7	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BE 9	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BE 10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
C 14	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NI 66	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CU 66	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZN 66	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZN 67	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZN 68	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZN 70	0.00E+00	0.00E+00	0.00E+00	0.00E+00
GA 69	0.00E+00	0.00E+00	0.00E+00	0.00E+00
GA 71	0.00E+00	0.00E+00	0.00E+00	0.00E+00
GE 70	0.00E+00	0.00E+00	0.00E+00	0.00E+00
GE 72	6.05E-02	6.05E-02	6.05E-02	6.05E-02
GE 73	1.20E-01	1.20E-01	1.20E-01	1.20E-01
GE 74	2.87E-01	2.87E-01	2.87E-01	2.87E-01
GE 76	1.47E+00	1.47E+00	1.47E+00	1.47E+00
AS 75	5.56E-01	5.56E-01	5.56E-01	5.56E-01
SE 76	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SE 77	2.83E+00	2.83E+00	2.83E+00	2.83E+00
SE 78	7.36E+00	7.36E+00	7.36E+00	7.36E+00
SE 79	1.74E+01	1.74E+01	1.74E+01	1.73E+01
SE 80	4.00E+01	4.00E+01	4.00E+01	4.00E+01
SE 82	1.01E+02	1.01E+02	1.01E+02	1.01E+02
BR 79	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BR 81	0.00E+00	0.00E+00	0.00E+00	0.00E+00
KR 80	6.87E-04	6.87E-04	6.87E-04	6.87E-04
KR 81	1.16E-04	1.16E-04	1.16E-04	1.16E-04
KR 82	0.00E+00	0.00E+00	0.00E+00	0.00E+00
KR 83	9.08E+01	9.08E+01	9.08E+01	9.08E+01
KR 84	3.75E+02	3.75E+02	3.75E+02	3.75E+02
KR 85	4.42E+01	8.78E+00	9.50E-02	5.56E-13
KR 86	5.75E+02	5.75E+02	5.75E+02	5.75E+02
RB 85	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RB 86	2.37E-31	1.10E-178	0.00E+00	0.00E+00
RB 87	7.37E+02	7.37E+02	7.37E+02	7.37E+02

Table C-5 (Continued)

Burn-up (MWd/MTIHM)	**100,000**			
Enrichment (%)	8.32 % U-235			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
SR 86	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SR 87	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SR 88	1.06E+03	1.06E+03	1.06E+03	1.06E+03
SR 89	3.09E-10	1.11E-64	3.99E-217	0.00E+00
SR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 89	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y 91	1.54E-08	1.63E-55	4.81E-187	0.00E+00
ZR 90	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 91	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 92	1.94E+03	1.94E+03	1.94E+03	1.94E+03
ZR 93	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 94	2.29E+03	2.29E+03	2.29E+03	2.29E+03
ZR 95	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ZR 96	2.40E+03	2.40E+03	2.40E+03	2.40E+03
NB 93	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NB 93M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NB 94	1.99E-03	1.98E-03	1.98E-03	1.95E-03
NB 95	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NB 95M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MO 95	0.00E+00	0.00E+00	0.00E+00	0.00E+00
MO 96	3.06E+02	3.06E+02	3.06E+02	3.06E+02
MO 97	2.36E+03	2.36E+03	2.36E+03	2.36E+03
MO 98	2.42E+03	2.42E+03	2.42E+03	2.42E+03
MO100	2.80E+03	2.80E+03	2.80E+03	2.80E+03
TC 98	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TC 99	1.99E+03	1.99E+03	1.99E+03	1.99E+03
RU 99	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RU100	7.24E+02	7.24E+02	7.24E+02	7.24E+02
RU101	2.20E+03	2.20E+03	2.20E+03	2.20E+03
RU102	2.47E+03	2.47E+03	2.47E+03	2.47E+03
RU103	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RU104	1.63E+03	1.63E+03	1.63E+03	1.63E+03
RU106	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RH102	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RH103	0.00E+00	0.00E+00	0.00E+00	0.00E+00
RH103M	4.37E-16	4.74E-86	5.91E-282	0.00E+00
RH106	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PD104	1.28E+03	1.28E+03	1.28E+03	1.28E+03

Table C-5 (continued).

Burn-up (MWd/MTIHM)	**100,000**			
Enrichment (%)	8.32 % U-235			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
PD105	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PD106	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PD107	6.54E+02	6.54E+02	6.54E+02	6.54E+02
PD108	4.46E+02	4.46E+02	4.46E+02	4.46E+02
PD110	1.53E+02	1.53E+02	1.53E+02	1.53E+02
AG107	0.00E+00	0.00E+00	0.00E+00	0.00E+00
AG108	1.32E-14	1.16E-14	7.89E-15	8.89E-16
AG108M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
AG109	0.00E+00	0.00E+00	0.00E+00	0.00E+00
AG109M	2.40E-13	2.86E-19	7.40E-36	1.21E-130
AG110	2.55E-10	2.53E-21	3.97E-52	3.70E-228
AG110M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CD108	1.96E-03	1.96E-03	1.96E-03	1.97E-03
CD109	2.43E-07	2.90E-13	7.49E-30	1.23E-124
CD110	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CD111	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CD112	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CD113	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CD113M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CD114	6.48E+01	6.48E+01	6.48E+01	6.48E+01
CD115M	3.55E-14	8.13E-76	2.09E-248	0.00E+00
CD116	2.31E+01	2.31E+01	2.31E+01	2.31E+01
IN113	0.00E+00	0.00E+00	0.00E+00	0.00E+00
IN113M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
IN114	0.00E+00	0.00E+00	0.00E+00	0.00E+00
IN114M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
IN115	2.84E+00	2.84E+00	2.84E+00	2.84E+00
IN115M	1.00E-20	2.30E-82	5.89E-255	0.00E+00
SN114	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SN115	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SN116	2.51E+01	2.51E+01	2.51E+01	2.51E+01
SN117	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SN117M	2.15E-42	1.18E-238	0.00E+00	0.00E+00
SN118	2.38E+01	2.38E+01	2.38E+01	2.38E+01
SN119	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SN119M	4.64E-04	2.81E-15	1.09E-46	3.58E-226
SN120	2.41E+01	2.41E+01	2.41E+01	2.41E+01
SN121M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SN122	2.63E+01	2.63E+01	2.63E+01	2.63E+01

Table C-5 (Continued)

Burn-up (MWd/MTIHM)	**100,000**			
Enrichment (%)	8.32 % U-235			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
SN123	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SN124	3.55E+01	3.55E+01	3.55E+01	3.55E+01
SN126	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SB121	2.08E+01	2.08E+01	2.08E+01	2.08E+01
SB123	2.68E+01	2.68E+01	2.68E+01	2.68E+01
SB124	1.61E-10	3.42E-56	4.47E-184	0.00E+00
SB125	0.00E+00	0.00E+00	0.00E+00	0.00E+00
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
TE122	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE123	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE123M	3.35E-07	3.59E-30	1.73E-94	0.00E+00
TE124	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE125	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE125M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE126	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE127	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE127M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE128	3.24E+02	3.24E+02	3.24E+02	3.24E+02
TE129	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE129M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TE130	1.06E+03	1.06E+03	1.06E+03	1.06E+03
I127	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I129	4.89E+02	4.89E+02	4.89E+02	4.89E+02
XE127	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE128	2.29E+01	2.29E+01	2.29E+01	2.29E+01
XE129	2.92E-01	2.93E-01	2.94E-01	3.03E-01
XE130	7.60E+01	7.60E+01	7.60E+01	7.60E+01
XE131	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE131M	2.36E-47	2.17E-278	0.00E+00	0.00E+00
XE132	3.76E+03	3.76E+03	3.76E+03	3.76E+03
XE134	4.45E+03	4.45E+03	4.45E+03	4.45E+03
XE136	6.59E+03	6.59E+03	6.59E+03	6.59E+03
CS133	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CS134	9.52E+01	2.14E-02	1.29E-12	5.34E-71
CS135	1.44E+03	1.44E+03	1.44E+03	1.44E+03
CS136	2.35E-42	3.78E-252	0.00E+00	0.00E+00
CS137	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BA132	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BA134	9.57E+02	1.05E+03	1.05E+03	1.05E+03

Table C-5 (continued).

Burn-up (MWd/MTIHM)	**100,000**			
Enrichment (%)	8.32 % U-235			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
BA135	1.01E+01	1.01E+01	1.01E+01	1.03E+01
BA136	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BA136M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BA137	6.97E+02	2.03E+03	3.39E+03	3.73E+03
BA137M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BA138	3.87E+03	3.87E+03	3.87E+03	3.87E+03
BA140	2.15E-42	2.51E-257	0.00E+00	0.00E+00
LA138	1.17E-02	1.17E-02	1.17E-02	1.17E-02
LA139	3.63E+03	3.63E+03	3.63E+03	3.63E+03
LA140	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CE140	3.83E+03	3.83E+03	3.83E+03	3.83E+03
CE141	6.27E-16	1.72E-100	0.00E+00	0.00E+00
CE142	3.40E+03	3.40E+03	3.40E+03	3.40E+03
CE144	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PR141	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PR143	6.30E-40	1.47E-242	0.00E+00	0.00E+00
PR144	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PR144M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ND142	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ND143	1.64E+03	1.64E+03	1.64E+03	1.64E+03
ND144	4.70E+03	4.70E+03	4.70E+03	4.70E+03
ND145	1.73E+03	1.73E+03	1.73E+03	1.73E+03
ND146	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ND147	1.45E-49	4.12E-298	0.00E+00	0.00E+00
ND148	1.10E+03	1.10E+03	1.10E+03	1.10E+03
ND150	5.26E+02	5.26E+02	5.26E+02	5.26E+02
PM146	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PM147	4.13E+01	5.59E-02	5.19E-10	6.59E-56
PM148	5.51E-16	1.47E-82	5.78E-269	0.00E+00
PM148M	7.52E-14	2.01E-80	7.89E-267	0.00E+00
SM146	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SM147	2.81E+02	3.22E+02	3.22E+02	3.22E+02
SM148	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SM149	4.68E+00	4.68E+00	4.68E+00	4.68E+00
SM150	8.19E+02	8.19E+02	8.19E+02	8.19E+02
SM151	3.11E+01	2.57E+01	1.50E+01	6.87E-01
SM152	2.65E+02	2.65E+02	2.65E+02	2.65E+02
SM154	1.12E+02	1.12E+02	1.12E+02	1.12E+02
EU150	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table C-5 (Continued)

Burn-up (MWd/MTIHM)	**100,000**	1.26E+01	4.48E-02	4.46E-16
Enrichment (%)	8.32 % U-235	8.39E-01	4.73E-05	2.48E-29
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
EU151	1.25E+00	6.70E+00	1.74E+01	3.17E+01
EU152	0.00E+00	0.00E+00	0.00E+00	0.00E+00
EU153	3.55E+02	3.55E+02	3.55E+02	3.55E+02
EU154	9.47E+01	1.26E+01	4.48E-02	4.46E-16
EU155	2.76E+01	8.39E-01	4.73E-05	2.48E-29
EU156	6.74E-36	6.55E-217	0.00E+00	0.00E+00
GD152	2.05E-01	2.21E-01	2.27E-01	2.27E-01
GD153	1.10E-04	4.84E-16	7.61E-48	1.41E-229
GD154	7.56E+01	1.58E+02	1.70E+02	1.70E+02
GD155	2.85E+01	5.53E+01	5.62E+01	5.62E+01
GD156	4.93E+02	4.93E+02	4.93E+02	4.93E+02
GD157	4.35E-01	4.35E-01	4.35E-01	4.35E-01
GD158	9.63E+01	9.63E+01	9.63E+01	9.63E+01
GD160	3.89E+00	3.89E+00	3.89E+00	3.89E+00
TB159	8.80E+00	8.80E+00	8.80E+00	8.80E+00
TB160	0.00E+00	0.00E+00	0.00E+00	0.00E+00
DY160	0.00E+00	0.00E+00	0.00E+00	0.00E+00
DY161	0.00E+00	0.00E+00	0.00E+00	0.00E+00
DY162	1.11E+00	1.11E+00	1.11E+00	1.11E+00
DY163	1.22E+00	1.22E+00	1.22E+00	1.22E+00
DY164	3.11E-01	3.11E-01	3.11E-01	3.11E-01
HO165	9.38E-01	9.38E-01	9.38E-01	9.38E-01
HO166M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
ER166	3.76E-01	3.76E-01	3.77E-01	3.80E-01
ER167	1.41E-02	1.41E-02	1.41E-02	1.41E-02
ER168	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TM169	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TM170	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TM171	0.00E+00	0.00E+00	0.00E+00	0.00E+00
YB170	0.00E+00	0.00E+00	0.00E+00	0.00E+00
YB171	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TL206	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TL207	4.17E-15	3.09E-14	1.54E-13	9.55E-13
TL208	3.42E-12	2.94E-12	1.51E-12	4.16E-14
TL209	1.25E-17	2.37E-17	1.36E-16	3.74E-15
PB206	9.08E-12	6.82E-10	1.36E-07	9.48E-05
PB207	1.84E-09	3.25E-08	5.13E-07	1.75E-05
PB208	7.35E-06	3.45E-05	8.41E-05	1.36E-04

Table C-5 (continued).

Burn-up (MWd/MTIHM)	**100,000**	1.26E+01	4.48E-02	4.46E-16
Enrichment (%)	8.32 % U-235	8.39E-01	4.73E-05	2.48E-29
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
PB209	5.40E-14	1.02E-13	5.84E-13	1.61E-11
PB210	4.79E-11	3.26E-09	2.04E-07	2.31E-05
PB211	3.23E-14	2.39E-13	1.19E-12	7.39E-12
PB212	2.02E-09	1.74E-09	8.88E-10	2.46E-11
PB214	5.43E-16	3.56E-14	9.79E-13	6.26E-11
BI208	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BI209	1.01E-09	4.36E-09	4.28E-08	4.71E-06
BI210	2.95E-14	2.00E-12	1.25E-10	1.42E-08
BI210M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
BI211	1.90E-15	1.41E-14	7.04E-14	4.36E-13
BI212	1.91E-10	1.65E-10	8.42E-11	2.33E-12
BI213	1.27E-14	2.40E-14	1.37E-13	3.78E-12
BI214	4.03E-16	2.64E-14	7.27E-13	4.65E-11
PO210	8.27E-13	5.63E-11	3.52E-09	3.99E-07
PO211	2.35E-20	1.74E-19	8.67E-19	5.37E-18
PO212	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PO213	1.90E-23	3.60E-23	2.06E-22	5.68E-21
PO214	5.55E-23	3.63E-21	1.00E-19	6.39E-18
PO215	2.70E-20	2.00E-19	9.99E-19	6.19E-18
PO216	8.04E-15	6.93E-15	3.54E-15	9.80E-17
PO218	6.30E-17	4.12E-15	1.14E-13	7.25E-12
AT217	1.52E-19	2.88E-19	1.65E-18	4.55E-17
RN219	6.13E-17	4.54E-16	2.26E-15	1.40E-14
RN220	3.03E-12	2.61E-12	1.34E-12	3.70E-14
RN222	1.16E-13	7.58E-12	2.09E-10	1.33E-08
FR221	1.38E-15	2.61E-15	1.50E-14	4.13E-13
FR223	2.84E-16	2.11E-15	1.05E-14	6.51E-14
RA223	1.56E-11	1.15E-10	5.76E-10	3.57E-09
RA224	1.77E-08	1.52E-08	7.78E-09	2.15E-10
RA225	6.26E-12	1.18E-11	6.77E-11	1.87E-09
RA226	1.80E-08	1.18E-06	3.25E-05	2.07E-03
RA228	6.20E-13	4.12E-12	1.54E-11	8.04E-11
AC225	4.23E-12	7.99E-12	4.57E-11	1.26E-09
AC227	1.10E-08	8.15E-08	4.07E-07	2.52E-06
AC228	6.47E-17	4.30E-16	1.60E-15	8.40E-15
TH227	2.56E-11	1.90E-10	9.47E-10	5.87E-09
TH228	3.42E-06	2.94E-06	1.50E-06	4.16E-08

Table C-5 (Continued)

Burn-up (MWd/MTIHM)	**100,000**	1.26E+01	4.48E-02	4.46E-16
Enrichment (%)	8.32 % U-235	8.39E-01	4.73E-05	2.48E-29
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
TH229	1.15E-06	2.18E-06	1.25E-05	3.44E-04
TH230	6.96E-04	1.21E-02	1.02E-01	1.20E+00
TH231	3.41E-08	3.41E-08	3.42E-08	3.45E-08
TH232	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TH234	1.25E-05	1.25E-05	1.25E-05	1.25E-05
PA231	9.90E-05	3.02E-04	8.70E-04	4.12E-03
PA233	5.20E-05	5.38E-05	6.21E-05	9.75E-05
PA234	1.88E-10	1.88E-10	1.88E-10	1.88E-10
PA234M	4.20E-10	4.20E-10	4.20E-10	4.20E-10
U232	0.00E+00	0.00E+00	0.00E+00	0.00E+00
U233	3.43E-03	1.58E-02	5.38E-02	3.60E-01
U234	6.36E+01	2.57E+02	6.34E+02	1.12E+03
U235	8.38E+03	8.39E+03	8.40E+03	8.48E+03
U236	1.18E+04	1.18E+04	1.19E+04	1.20E+04
U237	5.90E-05	1.77E-05	6.11E-07	1.59E-09
U238	8.58E+05	8.58E+05	8.58E+05	8.58E+05
U240	1.62E-12	1.62E-12	1.62E-12	1.62E-12
NP235	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NP236	6.81E-03	6.81E-03	6.81E-03	6.79E-03
NP237	1.53E+03	1.59E+03	1.83E+03	2.87E+03
NP238	3.33E-07	2.97E-07	2.16E-07	3.48E-08
NP239	4.21E-04	4.20E-04	4.17E-04	4.02E-04
NP240M	1.42E-14	1.42E-14	1.42E-14	1.42E-14
PU236	1.21E-08	1.52E-08	1.52E-08	1.52E-08
PU237	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PU238	1.10E+03	9.02E+02	5.19E+02	2.22E+01
PU239	7.03E+03	7.03E+03	7.02E+03	6.95E+03
PU240	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PU241	1.91E+03	5.72E+02	1.98E+01	5.15E-02
PU242	0.00E+00	0.00E+00	0.00E+00	0.00E+00
PU243	5.30E-12	5.30E-12	5.30E-12	5.30E-12
PU244	8.47E-02	8.47E-02	8.47E-02	8.47E-02
AM241	6.40E+02	1.92E+03	2.22E+03	1.18E+03
AM242	2.13E-05	1.90E-05	1.38E-05	2.22E-06
AM242M	1.78E+00	1.58E+00	1.15E+00	1.86E-01
AM243	4.90E+02	4.89E+02	4.85E+02	4.68E+02
AM244	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CM241	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CM242	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table C-5 (continued).

Enrichment (%)	8.32 % U-235			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
CM243	1.96E+00	1.07E+00	1.95E-01	1.16E-05
CM244	2.83E+02	1.09E+02	7.46E+00	1.67E-06
CM245	3.21E+01	3.21E+01	3.19E+01	3.09E+01
CM246	6.61E+00	6.59E+00	6.52E+00	6.15E+00
CM247	1.49E-01	1.49E-01	1.49E-01	1.49E-01
CM248	2.12E-02	2.12E-02	2.12E-02	2.12E-02
BK249	6.05E-06	1.56E-14	1.38E-38	5.06E-176
BK250	7.50E-19	7.49E-19	7.47E-19	7.35E-19
CF249	3.87E-04	3.74E-04	3.26E-04	1.48E-04
CF250	6.27E-05	1.67E-05	4.09E-07	2.64E-14
CF251	4.52E-05	4.44E-05	4.20E-05	3.09E-05
CF252	1.44E-05	2.02E-08	2.07E-16	4.70E-62
CF254	3.85E-18	1.42E-63	8.72E-191	0.00E+00
ES254	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL	9.68E+05	9.69E+05	9.71E+05	9.71E+05

Appendix D
DOE Used Nuclear Fuel Characteristics

The following discussions of each of the 34 groups provide a description of the fuel group and an example of fuel that makes up the group. When appropriate, a more detailed description of a fuel with the largest percentage of MTHM within each group is provided. This discussion is not intended to address each fuel in the group.

Intact fuel is made up of fuel from the good or fair cladding condition categories. Nonintact fuel is composed of fuel from the poor or none cladding categories

Group 1: U Metal, Zirc Clad, Low-Enriched Uranium

This group contains a low-enriched uranium-metal compound spent nuclear fuel (SNF) with zirconium cladding (accounting for approximately 86% of the DOE SNF inventory by mass). Greater than 99% of the MTHM of SNF in this group is N Reactor SNF. The N Reactor was used for both material and power production. N Reactor fuel consists of two concentric tubes about 2.4 in. in diameter and typically 2 feet long. N Reactor SNF has a nominal enrichment of about 1% and a typical burn-up of about 2.4 GWd/MTU. The cladding condition of the N Reactor SNF is fair to poor.

Group 2: U Metal, Nonzirc Clad, Low-Enriched Uranium

This group contains a low-enriched uranium-metal compound SNF with nonzirc cladding. The largest single source of SNF in this group (over 40% of the MTHM) is from the Single-Pass Reactor, which was used for material production. The Single-Pass Reactor SNF consists of circular tubes roughly 1.5 in. in diameter and 0.66 feet long. The Single-Pass Reactor SNF has a nominal enrichment of about 1% and an average burn-up of about 3 GWd/MTU. The cladding condition of the Single-Pass Reactor SNF is generally poor.

Group 3: U-Zirc

This group contains uranium-zirc compound SNF. Greater than 99% of the MTHM of fuel in this group is from the Heavy Water Components Test Reactor. Heavy Water Components Test Reactor semi-production run SNF is the dominant SNF in this group (67% of the MTHM). Heavy Water Components Test Reactor semi-production run SNF consists of circular tubes about 2.1 in. in diameter and 11 feet long. The Heavy Water Components Test Reactor semi-production run SNF is about 0.6% enriched. The condition of the Heavy Water Components Test Reactor semi-production run SNF cladding is fair.

Group 4: U-Mo

This group contains a uranium-molybdenum alloy compound SNF. More than 99% of the MTHM of the SNF in this group is from the Enrico Fermi Atomic Power Plant, and the majority (over 90% of the MTHM) of the SNF in this group consists of Fermi standard fuel subassemblies. Fermi was a sodium-cooled fast neutron spectrum power reactor. Fermi driver fuel consists of rods roughly 0.16 in. in diameter and 2.7 feet long. The Fermi standard fuel subassembly SNF has an enrichment of about 26% and an average burn-up of about 1.6 GWd/MTU. The condition of the cladding for the SNF in this group ranges from good to none.

Group 5: U Oxide, Zirc Clad, Intact, High-Enriched Uranium

This group contains a high-enriched uranium oxide SNF with intact zirc cladding. Greater than 90% of the MTHM of the SNF in this group consists of Shippingport PWR Core 2 blanket SNF, which is a uranium oxide compound dispersed in a zirconium-oxide (Seed 1) or zirconium-oxide calcium-oxide (Seed 2) matrix. Shippingport PWR was a light-water-moderated and cooled power reactor. Shippingport PWR fuel assemblies consist of 19 flat plates; the assemblies are 7.4 in. square and about 8.7 feet long. The Shippingport PWR Core 2 SNF has an enrichment of about 69% to 81% and a burn-up of roughly

38% of the initial fissile mass. The condition of the Shippingport PWR Core 2 blanket fuel cladding is good.

Group 6: U Oxide, Zirc Clad, Intact, Medium-Enriched Uranium

This group contains medium-enriched uranium oxide SNF with intact zirc cladding. Greater than 80% of the MTHM in this group consists of Experimental Boiling Water Reactor SNF. The Experimental Boiling Water Reactor was a DOE light-water-cooled and moderated experimental power reactor. Experimental Boiling Water Reactor SNF consists of plate-type assemblies, roughly 3.75 in. square and 5.2 feet long. Experimental Boiling Water Reactor SNF has an enrichment of 6% and a maximum burn-up of 1.6 GWd/MTU. The cladding condition of the Experimental Boiling Water Reactor SNF is fair.

Group 7: U Oxide, Zirc Clad, Intact, Low-Enriched Uranium

This group contains low-enriched uranium oxide with intact zirc cladding. The majority (75% of the MTHM) of the SNF in this group was generated by typical commercial power reactors, such as the Robert E. Ginna, Calvert Cliffs, Big Rock Point, Surry, and Turkey Point reactors. The commercial power reactor SNF configuration includes intact rod arrays. The commercial power reactor SNF in this group has enrichments ranging from 0.6% to 2.9%. The average burn-up of the commercial power reactor SNF in this group ranges from about 1.6 GWd/MTU for some Big Rock Point SNF to about 43 GWd/MTU for the Calvert Cliffs 1 SNF. The cladding condition of the commercial power reactor SNF in this group is good.

Group 8: U Oxide, SST/Hastelloy Clad, Intact, High-Enriched Uranium

This group contains high-enriched uranium oxide with intact stainless steel or Hastelloy cladding. About 40% of the MTHM of the SNF in this group was generated by superheaters for the Pathfinder Atomic Power Plant, a power reactor, and the Boiling Reactor Experiment V, a test, research, and education reactor. The Pathfinder SNF consists of rods 0.9 in. in diameter and 6.5 feet long. The Boiling Reactor Experiment V SNF consists of flat plate assemblies 3.7 in. wide and 2.1 feet long. The SNF in this group has an enrichment of roughly 93%. The Pathfinder and Boiling Reactor Experiment V SNF in this group have a burn-up of less than 6% of the initial fissile mass, and the cladding condition is good to fair.

Group 9: U Oxide, SST Clad, Intact, Medium-Enriched Uranium

This group contains medium-enriched uranium oxide SNF with intact stainless steel cladding. Greater than 80% of the MTHM of the SNF in this group was driver fuel for the Power Burst Facility, which was a test reactor designed to investigate fuel performance during accident conditions. Power Burst Facility SNF consists of rods measuring 0.75 in. in diameter and 4 feet long. Power Burst Facility SNF has an enrichment of about 18% and an average burn-up of about 0.5 GWd/MTU. The Power Burst Facility cladding condition is good.

Group 10: U Oxide, SST Clad, Intact, Low-Enriched Uranium

This group contains low-enriched uranium oxide SNF with intact stainless steel cladding. This group contains a small amount of material, over 40% of which by MTHM was generated by Connecticut Yankee reactors. The Connecticut Yankee SNF is typical commercial power reactor SNF, except that it has stainless steel cladding. The Connecticut Yankee SNF has an enrichment of 1.9%. The Connecticut Yankee SNF has a burn-up of about 32 GWd/MTU. The cladding condition of the Connecticut Yankee SNF is good.

Group 11: U Oxide, Nonalum Clad, Nonintact or Declad, High-Enriched Uranium

This group contains high-enriched uranium oxide SNF with nonaluminum cladding that is not intact or that has been removed. About 60% of the MTHM of the SNF in this group is generated from medical isotope production targets from foreign research reactors in Canada. The Canadian foreign research reactor targets have an enrichment of about 50%. As there is no cladding on the Canadian foreign research reactor targets, the fuel cladding is categorized as none.

Group 12: U Oxide, Nonalum Clad, Nonintact or Declad, Medium-Enriched Uranium

This group contains medium-enriched uranium oxide SNF with failed nonaluminum cladding or no cladding. Virtually all of this SNF was generated as a result of severe-condition fuel experiments. These experiments generally involved segments of previously irradiated fuel rods that were sectioned and placed into experiment capsules for further irradiation under extremely high temperatures. The SNF in this group has enrichments ranging from 5% to nearly 20%. The cladding condition of the SNF in this group is either poor or none (the cladding has been removed).

Group 13: U Oxide, Nonalum Clad, Nonintact or Declad, Low-Enriched Uranium

This group contains low-enriched uranium oxide SNF with failed nonaluminum cladding or no cladding. 99% of the MTHM of the SNF in this group is core debris from the Three Mile Island Unit 2 reactor accident. The Three Mile Island Unit 2 fuel has an enrichment of about 2.4% and a burn-up of about 3.2 GWd/MTU. The cladding condition of the Three Mile Island Unit 2 SNF is poor.

Group 14: U Oxide, Alum Clad, High-Enriched Uranium

This group contains high-enriched uranium oxide SNF with aluminum cladding. Greater than 80% of the MTHM of the SNF in this group is High-Flux Isotope Reactor SNF. The High-Flux Isotope Reactor is a DOE test reactor. High-Flux Isotope Reactor SNF consists of two concentric assemblies consisting of curved involute plates that are separated for disposal. The outer assemblies are about 17 in. in diameter and 2.6 feet long, and the inner assemblies are about 12 in. in diameter and 2.5 feet long. High-Flux Isotope Reactor SNF has an enrichment of about 87%. High-Flux Isotope Reactor SNF has an average burn-up of about 230 GWd/MTU. The cladding condition of High-Flux Isotope Reactor SNF is good.

Group 15: U Oxide, Alum Clad, Medium-Enriched Uranium, Low-Enriched Uranium

This group contains medium-enriched uranium oxide SNF with aluminum cladding. Nearly all of the SNF in this group was generated from a number of foreign research reactors. The largest single source (56% of the MTHM) is the G.A. Siwabessy RSG-GAS-30 reactor in Indonesia. This Indonesian foreign research reactor SNF consists of square assembly plate-type fuel with a typical width of 3 in. and a length of about 2.9 feet. This Indonesian research reactor SNF has an enrichment of about 10% and a burn-up of about 50% of the initial fissile mass. The cladding condition of most of the Indonesian research reactor SNF in this group is good.

Group 16: U-Alx, Al-Clad High-Enriched Uranium

This group contains high-enriched uranium aluminide SNF. The SNF in this group is generated from domestic and foreign test, research, and education reactors. The Advanced Test Reactor is the largest single source of SNF in this group, accounting for 67% of the MTHM. The Advanced Test Reactor SNF consists of curved plate assemblies about 4.2 in. wide, 2.6 in. high, and 5.5 feet long, before being cropped to about 4.1 feet for storage. The Advanced Test Reactor SNF has a typical enrichment of about 80% with an average burn-up of about 250 GWd/MTU. The cladding condition of Advanced Test Reactor SNF is good.

Group 17: U-Alx, Al-Clad Medium-Enriched Uranium

This group contains medium-enriched uranium aluminide SNF. The SNF in this group is generated from numerous domestic and foreign test, research, and education reactors. The largest single source of SNF in this group (30% of the MTHM) is the R-2 reactor in Sweden. The R-2 SNF is a square assembly of plate-type fuel about 3 in. wide and about 2.9 feet long. The R-2 SNF has an enrichment of about 9% and a burn-up of 60% of the initial fissile mass. The cladding condition of the SNF in this group is generally good.

Group 18: U3Si2

This group contains uranium-silicide SNF. The SNF in this group is generated from numerous domestic and foreign test, research, and education reactors. About 45% of the MTHM in this group consists of foreign research reactor multipin clusters generated by the National Research Universal reactor in

Canada. The National Research Universal reactor is heavy water moderated and cooled. National Research Universal SNF has a typical enrichment of about 5.6% and a burn-up of about 76% of the initial fissile mass. The cladding condition of National Research Universal SNF is good.

Group 19: Th/U Carbide, TRISO- or BISO-Coated Particles in Graphite

This group contains thorium-carbide and uranium-carbide SNF with tristructural isotropic- or buffered isotropic-coated particles embedded in a graphite matrix. About 95% of the MTHM of the SNF in this group was generated from the Fort St. Vrain reactor. The Fort St. Vrain SNF consists of hexagonal graphite blocks about 14 in. wide by 2.6 feet long, containing tristructural isotropic-coated (i.e., inner pyrocarbon, silicon carbide, and outer pyrocarbon coatings) particles. The Fort St. Vrain SNF has an enrichment of about 80% and burn-ups of about 45% of the initial fissile mass. The particle coating condition of the Fort St. Vrain SNF is good.

Group 20: Th/U Carbide, Monopyrolytic Carbon-Coated Particles in Graphite

This group contains thorium-carbide and uranium-carbide SNF with monopyrolytic carbon-coated particles in a graphite matrix. The coated particles are embedded in a graphite matrix. Nearly all (greater than 99%) of the SNF in this group is Peach Bottom Unit 1 reactor core 1 fuel. The Peach Bottom Unit 1 reactor was a helium-cooled, graphite-moderated, electric power reactor. The Peach Bottom Unit 1 SNF is about 3.5 in. wide and 12 feet long. The Peach Bottom Unit 1 core 1 SNF has a typical enrichment of about 86% and a burn-up of about 30% of the initial fissile mass. The particle coating condition of the Peach Bottom Unit 1 core 1 SNF is poor.

Group 21: Pu/U Carbide, Nongraphite Clad, Not Sodium Bonded

This group contains a small quantity of plutonium/uranium-carbide SNF with nongraphite cladding and no sodium bonding. This SNF was generated primarily by the Fast Flux Test Facility and has stainless steel cladding. The Fast Flux Test Facility reactor was a sodium-cooled DOE test and research reactor. About 56% of the MTHM in this group is the Fast Flux Test Facility test fuel assembly TFA-FC-1. The Fast Flux Test Facility TFA-FC-1 assembly cross section is a hexagon about 4.6 in. across the flats, 5.2 in. across the points, and the SNF is 12 feet long. The Fast Flux Test Facility TFA-FC-1 SNF is about 21% enriched and has a burn-up of about 60 GWd/MTU. The Fast Flux Test Facility TFA-FC-1 cladding condition is good.

Group 22: MO, Zirc Clad

This group contains a small quantity of MO, uranium-oxide, and plutonium-oxide SNF with zirconium cladding. About 60% of the MTHM in this group is Experimental Boiling Water Reactor SNF, which experimented with the recycling of plutonium. The Experimental Boiling Water Reactor SNF has an enrichment of 1.6% and a burn-up of 3% of the initial fissile mass. The Experimental Boiling Water Reactor SNF cladding condition is fair.

Group 23: MO, SST Clad

This group contains MO, uranium-oxide, and plutonium-oxide SNF with stainless steel cladding. About 80% of the MTHM of this group is Fast Flux Test Facility reactor driver fuel assemblies and test driver fuel assemblies. The Fast Flux Test Facility driver and test driver fuel assembly cross section is a hexagon about 4.6 in. across the flats and 5.2 in. across the points, and the SNF is 12 feet long. The Fast Flux Test Facility driver fuel assembly and test driver fuel assembly SNF have enrichments of about 24% and an average burn-up of about 70 GWd/MTU. The cladding condition of the SNF in this group is poor to good.

Group 24: MO, Non-SST/Nonzirc Clad

This group contains a small quantity of MO (uranium-oxide and plutonium-oxide, MO) SNF that does not have stainless steel or zirconium cladding. The SNF in this group is mostly the residue from hot cells and small experiments and does not have intact cladding. The majority of the SNF in this group (97% of the

MTHM) is mixed-oxide scrap with an enrichment of about 15%. The cladding condition of the SNF in this group is either poor or none.

Group 25: Th/U Oxide, Zirc Clad

This group contains thorium-oxide and uranium-oxide SNF with zirconium cladding. The SNF in this group was generated by the Shippingport Atomic Power Station with the Light Water Breeder Reactor core. The Shippingport Light Water Breeder Reactor was a power reactor that converted fertile 232Th to fissile 233U. About 27% of the MTHM in this group is Shippingport Light Water Breeder Reactor reflector IV SNF. Shippingport Light Water Breeder Reactor reflector IV assemblies are rods in a rectangular array about 17.1 in. by 13.8 in. and 11.8 feet long. The Shippingport Light Water Breeder Reactor reflector IV SNF has an enrichment of about 98% and a burn-up of about 2 GWd/MTU. The cladding condition of the Shippingport Light Water Breeder Reactor reflector IV SNF is generally good.

Group 26: Th/U Oxide, SST Clad

This group contains thorium-oxide and uranium-oxide SNF with stainless steel cladding. About 66% of the MTHM of the SNF in this group was generated from the Elk River Reactor, a light water power reactor. Elk River Reactor assemblies are rods in square arrays that are about 1.4 in. wide and high and 5.3 feet long. Elk River Reactor SNF has an enrichment of 96%. Elk River Reactor SNF has a typical burn-up of about 5.4 GWd/MTU. The cladding condition of the Elk River Reactor SNF is generally fair.

Group 27: U-Zirc Hydride, SST/Incoloy Clad, High-Enriched Uranium

This group contains high-enriched, uranium-zirc hydride SNF with stainless steel or Incoloy cladding. Most of the SNF in this group was generated from numerous domestic and foreign TRIGA research reactors, with no dominant single generator. The TRIGA SNF in this group is generally of the fuel life improvement program design. TRIGA fuel life improvement program rods are typically 1.5 in. in diameter and 2.4 feet long. The enrichment of the TRIGA fuel life improvement program SNF in this group has a range from about 60% to 70%, and the burn-up ranges from about 9,400 MWd/MTU to over 300 GWd/MTU. The cladding condition of the TRIGA fuel life improvement program SNF is generally good.

Group 28: U-Zirc Hydride, SST/Incoloy Clad, Medium-Enriched Uranium

This group contains medium-enriched uranium-zirconium hydride SNF with stainless steel or Incoloy cladding. The SNF in this group was generated from numerous domestic and foreign TRIGA research reactors, with no dominant single generator. TRIGA rods in this group are typically 1.5 in. in diameter and 2.4 to 3.8 feet long. The TRIGA SNF in this group has enrichments ranging from about 12% to 20% with burn-ups ranging from slight irradiation to nearly 95 GWd/MTU. The cladding condition of the SNF in this group is generally good.

Group 29: U-Zirc Hydride, Alum Clad, Medium-Enriched Uranium

This group contains medium-enriched uranium-zirconium hydride SNF with aluminum cladding. The SNF in this group was generated from numerous domestic and foreign TRIGA research reactors, with no dominant single generator. The TRIGA rods in this group are typically 1.5 in. in diameter and 2.4 feet long. The TRIGA SNF in this group has enrichments ranging from about 17% to 20%. The SNF in this group has highly variable burn-ups, ranging from slightly irradiated to about 37 GWd/MTU. The cladding condition of the SNF in this group is generally good.

Group 30: U-Zirc Hydride, Declad

This group contains uranium-zirconium hydride SNF that has been declad. The SNF in this group was generated from the System for Nuclear Auxiliary Power program, which was an experimental power program that involved five different reactors. The System for Nuclear Auxiliary Power rods are about 1.2

in. in diameter and 1.2 feet long. The System for Nuclear Auxiliary Power SNF has an enrichment of about 90%. The cladding has been removed, so the cladding condition is none.

Group 31: Metallic Sodium Bonded

This group contains a wide variety of SNF that has the common attribute of containing metallic-sodium bonding between the fuel matrix and the cladding. This group contains a wide variety of SNF that has the common attribute of containing metallic sodium bonding between the fuel matrix and the cladding. This group of fuel comprises approximately 59.9 MTHM.

Current plans are to treat a portion of this SNF and dispose of it as high-level waste. The SNF in this group was generated primarily from various breeder reactors. Over half of the MTHM from the Fermi reactor, nearly half of the MTHM from the Experimental Breeder Reactor II, and a small amount from the Fast Flux Test Facility reactor was generated. A limited amount of SNF was generated as a result of irradiation of sodium-containing experiments in thermal reactors. The SNF consists of a variety of compounds (uranium and plutonium oxides, carbides, metals, and alloys). Most of the SNF in this group has stainless steel cladding. The SNF includes rods, rod arrays, and experiment capsules that range from 0.2 to 9.1 inches in diameter/width and 1.8 to 12 feet long. The SNF in this group has enrichments that range from depleted uranium (0.2%) to 93%. The SNF in this group has variable burn-ups ranging from low to high (2,000 to 100,000 MWd/MTU). The condition of the cladding is variable and is considered good, fair, or poor

Group 32: Naval

Naval SNF is addressed in Section 4.0.

Group 33: Canyon Stabilization

This SNF is being treated in the SRS canyons and will be disposed of as HLW; therefore, this SNF is not addressed in this section.

Group 34: Miscellaneous

This group contains SNF that does not fit into other groups. The SNF in this group was generated from numerous reactors of different types. The dominant source is the Keuring van Electrotechnische Materialen SNF from the Aqueous Homogeneous Suspension Reactor, an experimental power reactor. Keuring van Electrotechnische Materialen SNF consists of canisters of thorium-oxide and uranium-oxide scrap. Keuring van Electrotechnische Materialen SNF has an enrichment of about 90%. Keuring van Electrotechnische Materialen SNF does not have cladding, so the condition is none.

Table D-1 Ranges of nominal properties for DOE spent nuclear fuel.

Fuel Group	MTHM[a]	EOL Effective Enrichment (%)	Cladding Composition	Cladding Condition	Fuel Compound Names	Fuel Matrix	Configuration	Length (ft)	Width/ Height/ Diameter (in.)
01. U metal, zirc clad, LEU	2103	1.7–0.5	Zirconium	Fair Poor	U metal	None	Plates Tubes	2.1–9.9	1.0–4.3
02. U metal, nonzirc clad, LEU	8	3.4–0.2	SST Aluminum	Poor Good Fair	U metal	None	Cans of scrap Tubes None	0.6–0.9	1.4–1.9
03. U-zirc	<1	92.9–0.5	Zirconium	Fair Good	U metal 2% Zr U-Zr	None	Tube Cylinders Plates	2.0–12.5	2.0–7.4
04. U-Mo	4	25.8–2.4	Zirconium Aluminum None	Good Poor Fair None	U-Mo	None	Rod Tube Plates in can	1.0–3.8	0.1–2.1
05. U oxide, zirc clad, intact, HEU	<1	92.5–23.1	Zirconium	Fair Good	UO_2	ZrO_2-CaO Graphite ZrO_2	Rod Assembly Plates	3.1–9.0	0.3–7.4
06. U oxide, zirc clad, intact, MEU	2	6.9–5	Zirconium	Fair Good	UO_2	None	Plates Rod Cans of rods Element	2.9–5.2	0.3–3.8
07. U oxide, zirc clad, intact, LEU	90	4.9%–0.6	Zirconium	Good Fair	UO_2	None	Tubes Rod Plates Assembly	0.8–14.7	0.4–8.5
08. U oxide, SST/hastelloy clad, intact, HEU	<1	93.2–91.0	SST Hastelloy	Good Fair	U oxide UO_2	SST SST (316L) SST 304B SST 304 None	Tubes Cans of scrap Rod Plates Rod assembly	2.1–6.6	0.9–3.7

Table D-1 (continued).

Fuel Group	MTHM[a]	EOL Effective Enrichment (%)	Cladding Composition	Cladding Condition	Fuel Compound Names	Fuel Matrix	Configuration	Length (ft)	Width/ Height/ Diameter (in.)
09. U oxide, SST clad, intact, MEU	<1	20.0–5.5	SST	Good Fair	UO_2-BeO_2 UO_2	ZrO_2-CaO None	Rod Element	2.4–4.0	0.3–1.5
10. U oxide, SST clad, intact, LEU	<1	1.9–0.2	SST	Good Fair	UO_2	None	Tube Rod	1.5–12.0	0.4–8.5
11. U oxide, nonalum clad, nonintact or declad, HEU	<1	93.3–21.0	Nichrome Hastelloy SST Zirconium None	Poor None	UO_2	BEO SST Nichrome None	Cans of scrap	0.2–2.8	2.8–5.6
12. U oxide, nonalum clad, nonintact or declad, MEU	<1	18.6–5.2	None Zirconium SST	Poor	UO_2	Gd_2O_3 None SST	Experiment capsule Scrap Cans of scrap	3.4–9.9	0.4–9.1
13. U oxide, nonalum clad, nonintact or declad, LEU	83	3.2–1.1	Zirconium SST	Poor	UO_2	None	Cans of scrap Scrap Rod	12.4–13.5	0.5–14.0
14. U oxide, alum clad, HEU	5	89.9–58.1	Aluminum	Good Fair	U_3O_8	Alum	Plates	2.0–3.6	2.8–17.2
15. U oxide, alum clad, MEU and LEU	<1	20.0–8.9	Aluminum	Good Fair	U_3O_8	Alum	Plates Assembly	2.2–3.3	3.0–4.8
16. U-ALx, HEU	8	93.3–21.9	Aluminum	Good Fair	U-ALx	Alum	Rods Tubes Plates Pin cluster Assemblies Elements	0.4–10.1	1.3–16.3

Table D-1 (continued).

Fuel Group	MTHM[a]	EOL Effective Enrichment (%)	Cladding Composition	Cladding Condition	Fuel Compound Names	Fuel Matrix	Configuration	Length (ft)	Width/ Height/ Diameter (in.)
17. U-ALx, MEU	3	20.0–9.0	Aluminum	Good Fair	U-ALX	Alum	Assembly Element Plates	2.0–3.4	2.1–4.1
18. U3Si2	8	22.0–5.2	Aluminum	Good Fair Poor	U_3SI_2	Alum	Tubes Multi-pin cluster Assembly Cans of Scrap	2.0–3.4	2.6–4.1
19. Th/U carbide, TRISO or BISO coated particles in graphite b	25	84.4–71.4	BISC TRISO	Good	ThC_2-UC_2 ThC-UC	Graphite	Tubes Cans of scrap	2.6–10.5	3.5–14.2
20. Th/U carbide, mono-pyrolytic carbon coated particles in graphite b	2	93.2–80.6	Mono-pyrolytic carbon	Poor	ThCO-UCO ThC_2-UC_2	Graphite	Element Carbon coated part Cans of Scrap	~12.0	~3.5
21. Pu/U carbide, nongraphite clad, not sodium bonded	<1	67.3–1	SST	Good Fair Poor	Pu/U carbide	None	Element Cans of scrap Rod	7.7–12.0	0.2–5.2
22. MOX, zirc clad	2	21.3–1.3	Zirconium	Poor Good Fair	PuO_2-UO_2	None	Rod Cans of Scrap Plates Element	3.3–7.1	0.3–6.6
23. MOX, SST clad	11	87.4–2.1	SST	Poor Good Fair	PuO_2-UO_2 PuO_2	None	Rod Plates Element Cans of Scrap Scrap	1.1–12.0	0.2–9.1

Table D-1 (continued).

Fuel Group	MTHM[a]	EOL Effective Enrichment (%)	Cladding Composition	Cladding Condition	Fuel Compound Names	Fuel Matrix	Configuration	Length (ft)	Width/ Height/ Diameter (in.)
24. MOX, non-SST/nonzirc clad	<1	54.3–5	Unknown	N/A Poor	PuO_2-UO_2	None Unknown	Scrap Cans of scrap	Unknown	Unknown
25. Th/U oxide, zirc clad	43	98.4–10.1	Zirconium	Good Poor N/A	ThO_2-UO_2 ceramic	None	Rod Assembly Cans of scrap	~11.8	9.0–22.3
26. Th/U oxide, SST clad	8	97.8–7.6	SST	Fair Good Poor	ThO_2-UO_2	None	Assembly Cans of scrap Rod	5.2–11.7	0.4–11.9
27. U-zirc hydride, SST/incoloy clad, HEU	<1	93.2–42.5	SST Incoloy	Good Fair	U-ZrHX-Er	None	Rod Element	2.4–3.8	0.5–3.2
28. U-zirc hydride, SST/incoloy clad, MEU	2	20.0–11.9	SST Incoloy	Good Poor	U-ZrHX U-ZrHX-Er	None	Element Canister of scrap	2.4–3.8	~1.5
29. U-zirc hydride, alum clad, MEU	<1	20.0–16.8	Aluminum	Good	U-ZrHX	None	Element	~2.4	~1.5
30. U-zirc hydride, declad	<1	~89.7	None	N/A	U-ZrHX	None	Declad rod	~1.2	~1.2
31. Metallic sodium bonded	60	93.2-<0.1	SST None Unknown	Poor Good N/A Fair	PuO_2-UO_2 U-10Zr U-Mo U-10Zr U metal U-Pu-Zr UO_2 U metal Pu/U alloy U-5 fissium Pu/U carbide	None	Fuel in sodium Rod Assembly Cans of Scrap Scrap	1.8–12.0	0.2–9.1

Table D-1 (continued).

Fuel Group	MTHM[a]	EOL Effective Enrichment (%)	Cladding Composition	Cladding Condition	Fuel Compound Names	Fuel Matrix	Configuration	Length (ft)	Width/ Height/ Diameter (in.)
32. Naval	65	—	—	—	—	—	—	—	—
33. Canyon stabilization	N/A	—	—	—	—	—	—	—	—
34. Misc (not previously listed)	<1	90.0–14.6	None Zirconium Unknown Aluminum SST	Fair Poor N/A Good	ThO_2-UO_2 U-Th metal U metal Am oxide Pu/U nitride	None Alum (1100) Unknown	Cans of scrap Tube Rod	0.3–9.9	0.5–2.6

a. MTHM are rounded to next higher whole number or reported as <1 MTHM, as applicable.

b. For fuel groups 19 and 20, cladding composition and cladding condition are reporting particle coating composition and condition.

Group 31 is sodium-bonded fuel. Some of this material has been or will be treated into HLW.

Group 32 information is provided in the Naval Nuclear Propulsion section.

Group 33 will be processed into HLW.

Table D-2 Total DOE SNF radionuclide inventory.

Radionuclide	2010	
	Nominal Fuel Inventories (Ci)	Bounding Fuel Inventories (Ci)
227Ac	4.98×10^{1}	1.01×10^{2}
110Ag	2.74	4.86
110mAg	2.06×10^{2}	3.65×10^{2}
111Ag	—	—
241Am	2.11×10^{6}	3.90×10^{6}
242Am	5.11×10^{3}	9.48×10^{3}
242mAm	5.13×10^{3}	9.53×10^{3}
243Am	4.06×10^{3}	7.59×10^{3}
136mBa	—	—
137mBa	3.60×10^{7}	6.64×10^{7}
140Ba	—	—
10Be	6.12×10^{-1}	1.29
211Bi	4.99×10^{1}	1.01×10^{2}
212Bi	2.48×10^{4}	5.06×10^{4}
14C	1.83×10^{4}	2.79×10^{4}
113Cd	—	—
113mCd	5.35×10^{3}	9.93×10^{3}
115mCd	7.78×10^{-8}	1.39×10^{-7}
141Ce	9.27×10^{-9}	1.63×10^{-8}
142Ce	1.44×10^{-2}	2.52×10^{-2}
144Ce	2.94×10^{6}	5.30×10^{6}
36Cl	2.98×10^{2}	4.67×10^{2}
242Cm	4.24×10^{3}	7.88×10^{3}
243Cm	1.69×10^{3}	3.23×10^{3}
244Cm	2.32×10^{5}	4.45×10^{5}
245Cm	7.14×10^{1}	1.39×10^{2}
246Cm	1.10×10^{1}	2.16×10^{1}
247Cm	4.37×10^{-5}	8.62×10^{-5}
60Co	7.49×10^{6}	9.79×10^{6}
51Cr	1.64×10^{-13}	2.88×10^{-13}
134Cs	1.85×10^{6}	3.20×10^{6}
135Cs	3.13×10^{2}	5.78×10^{2}
136Cs	—	—
137Cs	3.81×10^{7}	7.02×10^{7}
152Eu	4.67×10^{3}	8.22×10^{3}

Table D-2 (continued).

Radionuclide	2010	
	Nominal Fuel Inventories (Ci)	Bounding Fuel Inventories (Ci)
154Eu	8.37×10^{5}	1.51×10^{6}
155Eu	2.94×10^{5}	5.27×10^{5}
156Eu	—	—
55Fe	7.67×10^{5}	9.71×10^{5}
59Fe	6.83×10^{-8}	1.23×10^{-7}
223Fr	6.87×10^{-1}	1.40
153Gd	6.66	1.18×10^{1}
3H	2.45×10^{5}	4.21×10^{5}
129I	1.95×10^{1}	3.63×10^{1}
131I	—	—
114In	2.65×10^{-9}	4.38×10^{-9}
114mIn	2.77×10^{-9}	4.58×10^{-9}
115mIn	5.47×10^{-12}	9.77×10^{-12}
85Kr	1.86×10^{6}	3.42×10^{6}
140La	—	—
54Mn	1.56×10^{3}	2.96×10^{3}
93Mo	1.42×10^{2}	2.21×10^{2}
93mNb	1.31×10^{3}	2.18×10^{3}
94Nb	2.37×10^{2}	3.49×10^{2}
95Nb	4.04	7.18
95mNb	1.35×10^{-2}	2.40×10^{-2}
144Nd	6.78×10^{-7}	1.21×10^{-6}
147Nd	—	—
59Ni	4.56×10^{4}	7.13×10^{4}
63Ni	5.28×10^{6}	8.20×10^{6}
237Np	2.02×10^{2}	3.76×10^{2}
231Pa	7.04×10^{1}	1.43×10^{2}
233Pa	2.02×10^{2}	3.76×10^{2}
234Pa	6.44×10^{-1}	1.22
234mPa	4.95×10^{2}	9.41×10^{2}
210Pb	1.74×10^{-2}	2.74×10^{-2}
211Pb	4.99×10^{1}	1.01×10^{2}
212Pb	2.48×10^{4}	5.06×10^{4}
107Pd	4.50×10^{1}	8.55×10^{1}
145Pm	5.39×10^{2}	9.09×10^{2}
147Pm	7.51×10^{6}	1.38×10^{7}

Table D-2 (continued).

Radionuclide	2010	
	Nominal Fuel Inventories (Ci)	Bounding Fuel Inventories (Ci)
148Pm	1.38×10^{-8}	2.49×10^{-8}
148mPm	2.44×10^{-7}	4.42×10^{-7}
212Po	1.59×10^{4}	3.24×10^{4}
215Po	4.99×10^{1}	1.01×10^{2}
216Po	2.48×10^{4}	5.06×10^{4}
143Pr	—	—
144Pr	2.94×10^{6}	5.30×10^{6}
144mPr	3.52×10^{4}	6.36×10^{4}
236Pu	2.68	4.85
237Pu	9.77×10^{-11}	1.58×10^{-10}
238Pu	9.72×10^{5}	1.79×10^{6}
239Pu	4.75×10^{5}	7.71×10^{5}
240Pu	3.65×10^{5}	6.21×10^{5}
241Pu	1.54×10^{7}	3.21×10^{7}
242Pu	5.05×10^{2}	8.38×10^{2}
244Pu	8.54×10^{-5}	1.61×10^{-4}
223Ra	4.99×10^{1}	1.01×10^{2}
224Ra	2.48×10^{4}	5.06×10^{4}
226Ra	3.71×10^{-2}	5.39×10^{-2}
228Ra	3.39	6.94
87Rb	1.23×10^{-2}	2.19×10^{-2}
103mRh	3.67×10^{-6}	6.48×10^{-6}
106Rh	6.26×10^{5}	1.14×10^{6}
219Rn	4.99×10^{1}	1.01×10^{2}
220Rn	2.48×10^{4}	5.06×10^{4}
103Ru	4.07×10^{-6}	7.19×10^{-6}
106Ru	6.26×10^{5}	1.14×10^{6}
124Sb	6.47×10^{-5}	1.11×10^{-4}
125Sb	2.35×10^{5}	4.30×10^{5}
126Sb	3.93×10^{1}	7.21×10^{1}
126mSb	2.81×10^{2}	5.15×10^{2}
79Se	2.91×10^{2}	5.39×10^{2}
145Sm	9.93	1.97×10^{1}
147Sm	1.29×10^{-2}	2.26×10^{-2}
151Sm	5.95×10^{5}	1.08×10^{6}
119mSn	5.30×102	1.01×10^{3}

Table D-2 (continued).

Radionuclide	2010	
	Nominal Fuel Inventories (Ci)	Bounding Fuel Inventories (Ci)
121mSn	7.97×102	1.11×10^{3}
123Sn	2.53×101	4.54×10^{1}
125Sn	—	—
126Sn	2.81×10^{2}	5.15×10^{2}
89Sr	7.33×10^{-3}	1.30×10^{-2}
90Y	3.12×10^{7}	5.72×10^{7}
91Y	2.71×10^{-1}	4.81×10^{-1}
65Zn	5.97×10^{3}	1.07×10^{4}
93Zr	1.68×10^{3}	2.82×10^{3}
95Zr	1.82	3.23
TOTAL	1.91×10^{8}	3.48×10^{8}

Appendix E
Navy Used Nuclear Fuel Radionuclide Inventory

Table E–1 Radionuclide inventory for a representative Naval SNF canister 5 years after reactor shutdown.

Isotope	Activity (curies)
227Ac	2.12×10^{-4}
241Am	3.56×10^{1}
242Am	3.84×10^{-1}
242mAm	3.86×10^{-1}
243Am	4.66×10^{-1}
137mBa	2.93×10^{5}
14C	6.40×10^{0}
113mCd	2.33×10^{1}
144Ce	1.47×10^{4}
249Cf	1.04×10^{-6}
251Cf	7.15×10^{-8}
252Cf	8.08×10^{-6}
36Cl	1.36×10^{-1}
242Cm	9.70×10^{-1}
243Cm	4.68×10^{-1}
244Cm	4.40×10^{1}
245Cm	3.85×10^{-3}
246Cm	1.20×10^{-3}
247Cm	1.54×10^{-8}
248Cm	6.50×10^{-8}
60Co	1.18×10^{3}
134Cs	4.95×10^{4}
135Cs	3.68×10^{0}
137Cs	3.11×10^{5}
152Eu	3.71×10^{1}
154Eu	7.17×10^{3}
155Eu	2.12×10^{3}
55Fe	1.68×10^{3}
3H	1.15×10^{3}
129I	8.03×10^{-2}
85Kr	2.41×10^{4}
93mNb	2.27×10^{3}
94Nb	2.06×10^{2}
59Ni	1.34×10^{1}

Table E-1 (continued).

Isotope	Activity (curies)
63Ni	1.63×10^{3}
236Np	4.92×10^{-5}
237Np	1.17×10^{0}
238Np	1.74×10^{-3}
239Np	4.66×10^{-1}
231Pa	7.77×10^{-4}
210Pb	2.97×10^{-6}
107Pd	4.42×10^{-2}
147Pm	9.20×10^{4}
144Pr	1.47×10^{4}
236Pu	6.33×10^{-1}
237Pu	1.84×10^{-7}
238Pu	7.80×10^{3}
239Pu	9.87×10^{0}
240Pu	1.04×10^{1}
241Pu	2.56×10^{3}
242Pu	5.65×10^{-2}
244Pu	6.72×10^{-9}
226Ra	1.50×10^{-5}
228Ra	9.03×10^{-10}
102Rh	1.12×10^{-2}
106Rh	3.20×10^{3}
106Ru	3.20×10^{3}
125Sb	4.13×10^{3}
126Sb	1.34×10^{-1}
126mSb	9.55×10^{-1}
79Se	2.67×10^{-1}
147Sm	2.48×10^{-5}
151Sm	9.78×10^{2}
121mSn	2.58×10^{1}
126Sn	9.55×10^{-1}
90Sr	3.05×10^{5}
99Tc	5.11×10^{1}
125mTe	1.01×10^{3}
229Th	2.14×10^{-5}
230Th	3.22×10^{-3}
232Th	1.19×10^{-5}
208Tl	8.76×10^{-2}

Table E-1 (continued)

Isotope	Activity (curies)
232U	5.29×10^{-1}
233U	6.52×10^{-2}
234U	1.86×10^{1}
235U	2.65×10^{-1}
236U	1.84×10^{0}
237U	6.13×10^{-2}
238U	9.20×10^{-4}
90Y	3.05×10^{5}
93Zr	8.69×10^{0}

Appendix F
High Level Waste Radionuclide Inventory

Table F-1 Total radionuclide inventory for each HLW glass type at 2017.

Nuclide	Radioactivity (Ci)				
	HS	SRS	WVDP	INL[a]	Total[a]
^{225}Ac	2.31	9.37×10^{-1}	2.32×10^{-1}	9.35×10^{-15}	3.48
^{227}Ac	1.33×10^{2}	1.41×10^{-4}	1.22×10^{1}	1.66×10^{-14}	1.45×10^{2}
^{228}Ac	1.36×10^{1}	6.69	1.64	4.64×10^{-12}	2.19×10^{1}
^{241}Am	1.42×10^{5}	2.22×10^{6}	5.30×10^{4}	1.27×10^{4}	2.43×10^{6}
^{242}Am	—	4.97×10^{2}	2.59×10^{2}	1.01×10^{-2}	7.56×10^{2}
^{242m}Am	—	5.00×10^{2}	2.61×10^{2}	—	7.61×10^{2}
^{243}Am	1.50×10^{1}	9.24×10^{3}	3.46×10^{2}	1.39×10^{-2}	9.60×10^{3}
^{217}At	2.31	9.37×10^{-1}	2.32×10^{-1}	9.35×10^{-15}	3.48
^{137m}Ba	2.99×10^{7}	2.81×10^{8}	3.66×10^{6}	5.62×10^{6}	3.20×10^{8}
^{210}Bi	1.75×10^{-2}	4.06×10^{-5}	1.42×10^{-4}	1.59×10^{-10}	1.77×10^{-2}
^{211}Bi	1.33×10^{2}	1.41×10^{-4}	1.23×10^{1}	1.52×10^{-18}	1.45×10^{2}
^{212}Bi	4.95×10^{1}	7.26	7.53	6.93×10^{-9}	6.43×10^{1}
^{213}Bi	2.31	9.37×10^{-1}	2.32×10^{-1}	9.35×10^{-15}	3.48
^{214}Bi	8.97×10^{-2}	3.12×10^{-4}	5.36×10^{-4}	6.44×10^{-4}	9.12×10^{-2}
^{14}C	—	—	1.37×10^{2}	2.78×10^{-2}	1.37×10^{2}
^{113}Cd	6.37×10^{-15}	1.77×10^{-7}	2.19×10^{-15}	—	1.77×10^{-7}
^{113m}Cd	7.30×10^{3}	—	5.70×10^{2}	—	7.87×10^{3}
^{144}Ce	—	3.20	2.45×10^{-11}	—	3.20
^{249}Cf	—	1.55×10^{2}	—	—	1.55×10^{2}
^{251}Cf	—	1.24×10^{2}	—	—	1.24×10^{2}
^{242}Cm	—	4.12×10^{2}	2.15×10^{2}	1.24×10^{-2}	6.27×10^{2}
^{243}Cm	9.28	2.24×10^{3}	6.96×10^{1}	4.70×10^{-4}	2.32×10^{3}
^{244}Cm	1.60×10^{2}	2.00×10^{6}	2.72×10^{3}	1.03×10^{-2}	2.00×10^{6}
^{245}Cm	—	1.63×10^{2}	8.79×10^{-1}	3.69×10^{-6}	1.64×10^{2}
^{246}Cm	—	1.96×10^{2}	1.01×10^{-1}	8.66×10^{-8}	1.96×10^{2}
^{247}Cm	—	1.48×10^{2}	—	3.09×10^{-14}	1.48×10^{2}

Nuclide	Radioactivity (Ci)				
	HS	SRS	WVDP	INL[a]	Total[a]
^{248}Cm	—	—	—	9.35×10^{-15}	9.35×10^{-15}
^{60}Co	9.88×10^{2}	3.33×10^{5}	2.20×10^{1}	3.21×10^{1}	3.34×10^{5}
^{134}Cs	8.47×10^{1}	4.39×10^{4}	5.90×10^{-1}	3.28×10^{-2}	4.40×10^{4}
^{135}Cs	1.46×10^{3}	1.61×10^{2}	1.63×10^{2}	1.78×10^{3}	
^{137}Cs	3.16×10^{7}	2.98×10^{8}	3.87×10^{6}	5.95×10^{6}	3.39×10^{8}
^{152}Eu	7.16×10^{2}	—	9.02×10^{1}	—	8.06×10^{2}
^{154}Eu	3.80×10^{4}	1.25×10^{6}	1.09×10^{4}	5.98×10^{3}	1.30×10^{6}
^{155}Eu	8.58×10^{2}	1.03×10^{3}	4.59×10^{2}	7.55	2.35×10^{3}
^{55}Fe	—	—	6.86×10^{-1}	—	6.86×10^{-1}
^{221}Fr	2.31	9.37×10^{-1}	2.32×10^{-1}	9.35×10^{-15}	3.48
^{223}Fr	1.83	1.95×10^{-6}	1.69×10^{-1}	2.29×10^{-16}	2.00
^{152}Gd	1.08×10^{-11}	—	6.16×10^{-12}	—	1.70×10^{-11}
^{3}H	—	—	1.80×10^{1}	3.56×10^{3}	3.58×10^{3}
^{129}I	4.80×10^{1}	2.18	2.10×10^{-1}	5.64	5.60×10^{1}
^{40}K	—	—	—	—	—
^{138}La	—	—	—	—	—
^{93m}Nb	3.21×10^{3}	1.57×10^{3}	2.46×10^{2}	4.74×10^{2}	5.50×10^{3}
^{94}Nb	—	—	—	5.36×10^{-3}	5.36×10^{-3}
^{144}Nd	—	9.66×10^{-12}	—	—	9.66×10^{-12}
^{59}Ni	1.37×10^{3}	5.71×10^{3}	1.06×10^{2}	—	7.19×10^{3}
^{63}Ni	1.14×10^{5}	5.04×10^{5}	7.06×10^{3}	—	6.25×10^{5}
^{236}Np	—	—	9.47	—	9.47
^{237}Np	1.41×10^{2}	2.01×10^{2}	2.39×10^{1}	6.26	3.72×10^{2}
^{238}Np	—	2.25	1.17	—	3.42
^{239}Np	1.50×10^{1}	9.24×10^{3}	3.46×10^{2}	1.47×10^{-3}	9.60×10^{3}
^{231}Pa	2.72×10^{2}	9.69×10^{-4}	1.52×10^{1}	1.48×10^{-9}	2.87×10^{2}
^{233}Pa	1.41×10^{2}	2.01×10^{2}	2.39×10^{1}	6.06×10^{-2}	3.66×10^{2}
^{234}Pa	2.59×10^{-1}	4.19×10^{-1}	1.11×10^{-3}	1.46×10^{-7}	6.79×10^{-1}
^{234m}Pa	1.99×10^{2}	3.22×10^{2}	8.54×10^{-1}	3.19×10^{-4}	5.22×10^{2}
^{209}Pb	2.31	9.37×10^{-1}	2.32×10^{-1}	2.80×10^{-15}	3.48
^{210}Pb	1.75×10^{-2}	4.06×10^{-5}	1.42×10^{-4}	9.16×10^{-9}	1.77×10^{-2}

Table F-1 Continued

Nuclide	Radioactivity (Ci)				
	HS	SRS	WVDP	INL[a]	Total[a]
^{211}Pb	1.33×10^{2}	1.41×10^{-4}	1.23×10^{1}	1.52×10^{-18}	1.45×10^{2}
^{212}Pb	4.95×10^{1}	7.26	7.53	1.05×10^{-8}	6.43×10^{1}
^{214}Pb	8.97×10^{-2}	3.12×10^{-4}	5.36×10^{-4}	6.44×10^{-4}	9.12×10^{-2}
^{107}Pd	—	8.84	1.10×10^{1}	—	1.98×10^{1}
^{146}Pm	—	—	3.67×10^{-1}	—	3.67×10^{-1}
^{147}Pm	—	1.03×10^{6}	7.01×10^{1}	2.67×10^{1}	1.03×10^{6}
^{210}Po	1.61×10^{-2}	3.58×10^{-5}	1.34×10^{-4}	7.59×10^{-14}	1.63×10^{-2}
^{211}Po	3.66×10^{-1}	3.88×10^{-7}	3.37×10^{-2}	4.18×10^{-21}	4.00×10^{-1}
^{212}Po	3.17×10^{1}	4.65	4.83	4.44×10^{-9}	4.12×10^{1}
^{213}Po	2.26	9.17×10^{-1}	2.27×10^{-1}	9.15×10^{-15}	3.40
^{214}Po	8.97×10^{-2}	3.12×10^{-4}	5.36×10^{-4}	6.43×10^{-4}	9.12×10^{-2}
^{215}Po	1.33×10^{2}	1.41×10^{-4}	1.23×10^{1}	1.52×10^{-18}	1.45×10^{2}
^{216}Po	4.95×10^{1}	7.26	7.53	6.11×10^{-8}	6.43×10^{1}
^{218}Po	8.97×10^{-2}	3.12×10^{-4}	5.36×10^{-4}	6.44×10^{-4}	9.12×10^{-2}
^{144}Pr	—	3.20	2.45×10^{-11}	—	3.20
^{144m}Pr	—	4.49×10^{-2}	3.43×10^{-13}	—	4.49×10^{-2}
^{236}Pu	—	—	8.43×10^{-1}	—	8.43×10^{-1}
^{238}Pu	4.31×10^{3}	6.14×10^{6}	6.85×10^{3}	8.98×10^{4}	6.24×10^{6}
^{239}Pu	6.91×10^{4}	1.18×10^{5}	1.65×10^{3}	1.81×10^{3}	1.91×10^{5}
^{240}Pu	1.23×10^{4}	5.94×10^{4}	1.23×10^{3}	1.57×10^{3}	7.45×10^{4}
^{241}Pu	5.78×10^{4}	3.50×10^{5}	2.22×10^{4}	1.93×10^{4}	4.49×10^{5}
^{242}Pu	1.00	1.44×10^{2}	1.65	3.42	1.50×10^{2}
^{243}Pu	—	1.48×10^{2}	—	2.22×10^{-14}	1.48×10^{2}
^{223}Ra	1.33×10^{2}	1.41×10^{-4}	1.23×10^{1}	1.52×10^{-18}	1.45×10^{2}
^{224}Ra	4.95×10^{1}	7.26	7.53	6.11×10^{-8}	6.43×10^{1}
^{225}Ra	2.31	9.37×10^{-1}	2.32×10^{-1}	1.07×10^{-12}	3.48
^{226}Ra	8.97×10^{-2}	3.12×10^{-4}	5.36×10^{-4}	9.69×10^{-3}	1.00×10^{-1}
^{228}Ra	1.36×10^{1}	6.69	1.64	1.24×10^{-11}	2.19×10^{1}
^{102}Rh	—	—	—	1.99×10^{-5}	1.99×10^{-5}
^{106}Rh	1.70×10^{-2}	2.98×10^{1}	1.41×10^{-7}	—	2.98×10^{1}
^{219}Rn	1.33×10^{2}	1.41×10^{-4}	1.23×10^{1}	1.52×10^{-18}	1.45×10^{2}

Nuclide	Radioactivity (Ci)				
	HS	SRS	WVDP	INL[a]	Total[a]
^{220}Rn	4.95×10^{1}	7.26	7.53	6.11×10^{-8}	6.43×10^{1}
^{222}Rn	8.97×10^{-2}	3.12×10^{-4}	5.36×10^{-4}	6.44×10^{-4}	9.12×10^{-2}
^{106}Ru	1.70×10^{-2}	2.98×10^{1}	1.41×10^{-7}	—	2.98×10^{1}
^{125}Sb	4.18×10^{2}	6.22×10^{4}	7.83	1.03	6.26×10^{4}
^{126}Sb	8.11×10^{1}	7.42×10^{2}	1.46×10^{1}	2.62×10^{-1}	8.38×10^{2}
^{126m}Sb	5.79×10^{2}	5.30×10^{3}	1.04×10^{2}	8.91×10^{1}	6.07×10^{3}
^{79}Se	1.22×10^{2}	3.60×10^{3}	6.02×10^{1}	—	3.78×10^{3}
^{146}Sm	—	—	8.63×10^{-8}	—	8.63×10^{-8}
^{147}Sm	—	3.46×10^{-4}	4.44×10^{-7}	1.81×10^{-13}	3.46×10^{-4}
^{151}Sm	3.10×10^{6}	1.01×10^{6}	6.85×10^{4}	—	4.18×10^{6}
^{121}Sn	—	8.95×10^{3}	9.59	—	8.96×10^{3}
^{121m}Sn	—	1.15×10^{4}	1.24×10^{1}	—	1.15×10^{4}
^{126}Sn	5.79×10^{2}	5.30×10^{3}	1.04×10^{2}	8.91×10^{1}	6.07×10^{3}
^{90}Sr	3.43×10^{7}	1.80×10^{8}	3.46×10^{6}	7.04×10^{6}	2.25×10^{8}
^{99}Tc	2.97×10^{4}	6.19×10^{4}	1.70×10^{3}	3.41×10^{3}	9.67×10^{4}
^{125m}Te	1.02×10^{2}	1.52×10^{4}	1.91	1.07×10^{-3}	1.53×10^{4}
^{227}Th	1.31×10^{2}	1.39×10^{-4}	1.21×10^{1}	5.81×10^{-17}	1.43×10^{2}
^{228}Th	4.93×10^{1}	7.23	7.51	1.74×10^{-6}	6.40×10^{1}
^{229}Th	2.31	9.37×10^{-1}	2.32×10^{-1}	1.21×10^{-10}	3.48
^{230}Th	1.42×10^{-2}	9.12×10^{-2}	5.96×10^{-2}	9.50×10^{-7}	1.65×10^{-1}
^{231}Th	9.00	4.49	1.01×10^{-1}	1.29×10^{-1}	1.37×10^{1}
^{232}Th	8.00	9.49	1.64	9.89×10^{-8}	1.91×10^{1}
^{234}Th	1.99×10^{2}	3.22×10^{2}	8.54×10^{-1}	3.19×10^{-4}	5.22×10^{2}
^{206}Tl	2.32×10^{-8}	5.36×10^{-11}	1.88×10^{-10}	2.10×10^{-16}	2.34×10^{-8}
^{207}Tl	1.33×10^{2}	1.41×10^{-4}	1.22×10^{1}	1.52×10^{-18}	1.45×10^{2}
^{208}Tl	1.78×10^{1}	2.61	2.71	2.49×10^{-9}	2.31×10^{1}
^{209}Tl	4.86×10^{-2}	1.97×10^{-2}	4.88×10^{-3}	1.96×10^{-16}	7.32×10^{-2}
^{232}U	3.73×10^{1}	1.82	5.74	4.63×10^{-3}	4.49×10^{1}
^{233}U	5.10×10^{2}	3.79×10^{2}	9.53	1.33×10^{-3}	8.99×10^{2}
^{234}U	2.20×10^{2}	4.89×10^{2}	5.05	9.95×10^{1}	8.14×10^{2}
^{235}U	9.00	4.49	1.01×10^{-1}	5.90×10^{-1}	1.42×10^{1}

Table F-1 Continued

Nuclide	Radioactivity (Ci)				
	HS	SRS	WVDP	INL[a]	Total[a]
^{236}U	6.00	2.48×10^{1}	2.97×10^{-1}	1.54	3.26×10^{1}
^{237}U	1.38	8.38	5.32×10^{-1}	1.76×10^{-2}	1.03×10^{1}
^{238}U	1.99×10^{2}	3.22×10^{2}	8.54×10^{-1}	2.94×10^{-2}	5.22×10^{2}
^{90}Y	3.43×10^{7}	1.80×10^{8}	3.46×10^{6}	7.04×10^{6}	2.25×10^{8}
^{93}Zr	4.81×10^{3}	2.61×10^{3}	2.72×10^{2}	—	7.69×10^{3}
Total	1.34×10^{8}	9.54×10^{8}	1.46×10^{7}	2.58×10^{7}	1.13×10^{9}

NOTE: [a]Radionuclide inventory for Idaho National Laboratory HLW canister is provided for year 2035.HS = Hanford Site; INL = Idaho National Laboratory; SRS = Savannah River Site; WVDP = West Valley Demonstration Project.

Table F-2 Estimated total radionuclide inventory for SRS HLW as of 11/16/2006

Nuclide	Radioactivity (Ci)
Ni-59	2.96E+03
Ni-63	2.60E+05
Co-60	3.09E+05
Se-79	1.43E+03
Sr-90	8.84E+07
Y-90 (= Sr-90)	8.84E+07
Zr-93	4.68E+02
Nb-93m	1.19E+02
Tc-99	2.34E+04
Ru-106	5.76E+03
Pd.107	1.04E+00
Cd-113	1.86E-08
Sn-121m	1.41E+03
Sb-125	1.52E+05
Sn-126	1.78E+03
I-129	2.61E-01
Cs-134	4.17E+05
Cs-135	4.64E+02
Cs-137	1.19E+08
Ba-137m	1.12E+08
Ce-144	4.19E+03
Pr-144m	4.19E+03
Pm-147	3.64E+06
Sm-151	1.89E+05
Eu-154	8.20E+05
Eu-155	4.83E+02
Th-229 .	1.03E-01
Th-230	5.68E-03
Th-232	2.82E+00
U-232	4.70E-01
U-233	1.48E+02
U-234	1.18E+02
U-235	2.43E+00
U-236	7.23E+00
U-238	8.87E+01
Np-237	1.12E+02

Table F-2 Continued

Nuclide	Radioactivity (Ci)
Pu-238	2.01E+06
Pu-239	5.80E+04
Pu-240	2.36E+04
Pu-241	1.22E+06
Pu-242	3.53E+01
Am-241	3.74E+05
Am-242m	2.81E+02
Am-243	1.17E+03
Cm-243	1.53E+02
Cm-244	4.14E+05
Cm-245	3.79E+01
Cm-246	3.84E+01
Cm-247	1.60E+01
Cm-248	0.00E+00
Cf-249	1.84E+01
Cf-251	1.65E+01
Total	4.18E+08

Table F-3 Radionuclide Inventory for Electro-Chemical Processing of Sodium Bonded UNF.

	Ceramic	Metal
Radionuclides[c]	Total curies for the year 2000	Total curies for the year 2000
Hydrogen-3	--	--
Carbon-14	--	4.3
Chlorine-36	--	--
Cobalt-60	--	$3.2x10^{3}$
Nickel-59	--	$1.1x10^{1}$
Nickel-63	--	$4.1x10^{2}$
Selenium-79	--	--
Krypton-85	--	--
Strontium-90	$7.1x10^{5}$	--
Niobium-93	--	$2.9x10^{1}$
Niobium-94	--	2.7
Zirconium-93	--	--
Technetium-99	--	$1.3x10^{2}$
Rhodium-101	--	--
Rhodium-102	--	--
Ruthenium-106	--	$2.1x10^{4}$
Palladium-107	--	--
Tin-126	--	2.8
Iodine-129	$3.4x10^{-1}$	--
Cesium-134	$7.9x10^{3}$	--
Cesium-135	$1.6x10^{1}$	--
Cesium-137	$8.5x10^{5}$	--
Samarium-151	--	--
Lead-210	--	--
Radium-226	$3.0x10^{-5}$	--
Radium-228	--	--
Actinium-227	--	--
Thorium-229	--	--
Thorium-230	$4.7x10^{-3}$	--
Thorium-232	$2.3x10^{-9}$	--
Protactinium-231	--	--
Uranium-232	$2.6x10^{-3}$	$1.2x10^{-4}$
Uranium-233	$2.0x10^{-4}$	$5.8x10^{-5}$
Uranium-234	2.8	$7.7x10^{-1}$
Uranium-235	$8.8x10^{-2}$	$2.5x10^{-2}$
Uranium-236	$6.3x10^{-2}$	$1.8x10^{-2}$
Uranium-238	$2.8x10^{-1}$	$9.7x10^{-2}$
Neptunium-237	1.3	$2.4x10^{-5}$
Plutonium-238	$3.6x10^{2}$	$6.6x10^{-3}$

Table F-3 Continued

	Ceramic	Metal
Radionuclides[c]	Total curies for the year 2000	Total curies for the year 2000
Plutonium-239	1.7×10^{4}	3.3×10^{-1}
Plutonium-240	1.5×10^{3}	2.9×10^{-2}
Plutonium-241	1.1×10^{4}	1.9×10^{-1}
Plutonium-242	1.2×10^{-1}	2.0×10^{-6}
Americium-241	1.6×10^{3}	3.1×10^{-2}
Americium-242/242m	1.4×10^{1}	2.7×10^{-4}
Americium-243	2.8×10^{-1}	4.8×10^{-6}
Curium-242	1.2×10^{1}	2.3×10^{-4}
Curium-243	1.6×10^{-1}	3.0×10^{-6}
Curium-244	1.9	3.1×10^{-5}
Curium-245	6.8×10^{-5}	1.1×10^{-9}
Curium-246	4.2×10^{-7}	7.1×10^{-12}
Curium-247	2.4×10^{-13}	4.0×10^{-18}
Curium-248	2.6×10^{-14}	4.4×10^{-19}
Californium-252	6.5×10^{-19}	--

Appendix G
Alternative LWR Reprocessing Waste Forms

G-1 Low Temperature Glass Encapsulants for I-129.

One issue with using AgI to contain ^{129}I is its relatively high vapor pressure at moderate temperatures[1], which limits the thermal processing that can be done on a AgI-containing waste form without excessive loss of ^{129}I (>0.1%). Because of this, immobilization using borosilicate glass, for example, is not feasible due to excessive iodine loss at the required processing temperature. Other, low-melting, glasses such as vanadium and lead oxide glasses,[2,3] may not meet chemically stability requirements. Another possibility that has recently been investigated is forming a glass using AgI and $Ag_4P_2O_7$.[4-6] In this case, the glass was melted at 500˚C and was shown to have low solubility due to the formation a protective AgI surface layer after exposure to water. However, this approach may not be applicable to AgI-containing zeolites. The use of grout to contain AgI-MOR has also been investigated, but possibilities of carbonate release with time limit its effectiveness as a long term storage material.[7]

In this work, a new approach was investigated for forming ^{129}I-containing waste forms. This approach involved mixing either AgI or AgI-zeolite powder with a glass powder that can be sintered to high density at 500˚C. Since the glass is not melted, a more refractory and therefore more chemically stable glass can be used. A bismuth oxide-based glass was chosen due to the low solubility of bismuth oxide in aqueous solution at pH>7.[8,9] In this approach, waste forms would be produced by crushing the AgI or the AgI-zeolite, mixing it with the glass powder, pressing the mixture into billets and then heating the billets to 500˚C to densify the waste form. In this work, the feasibility of such a process was demonstrated and the materials produced were analyzed using powder X-ray diffraction, simultaneous Thermal Gravimetric and Differential Thermal Analyses (TGA/DTA) and Scanning Electron Microscopy (SEM). An aqueous leaching study was also performed at 90˚C on crushed samples of the AgI-zeolite and AgI mixed with the Bi-based glass.

Two types of samples were made using a mixture of the glass and AgI or AgI-MOR, respectively. The materials were then die pressed into a pellet and heated in air for up to 3 hr at 500˚C.

Results and Discussion

Thermal testing of the waste forms indicate significant stability to around 800°C, at ~1 torr vapor pressure (< 1 wt% loss occurs from RT to 550°C). To date, samples synthesized at 500°C have maximum loadings of 20wt%AgI-MOR/glass and 50wt%AgI/glass both of which are shown to be dense (<1% open porosity).

PCT-|B leaching studies indicated high stability and durabilty for these materials, especially in comparison to reference AgI-Ag phosphate glass.

Conclusion

The feasibility of using a low sintering temperature glass to encapsulate silver iodide, either alone or in a zeolite, has been demonstrated.[10,11] A commercial BiZnB oxide glass powder is mixed with up to 50 wt% AgI or 20 wt% of AgI-MOR, pressed and densified at 500˚C, resulting in minimal AgI loss due to volatilization. Initial leach testing for iodine solubility indicated that levels below 1 micromole/liter can be achieved. We have begun to make our own compositions of low temperature glasses based on some of the fundamental materials parameters we discovered. In particular we will pursue new Bi formulations and try to understand the effects of Bi glass stoichiometry on Iodine retention.

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G-2 Iodine Capture and Immobilization with Functionalized Silica Aerogels

Targeted Waste Stream(s): Iodine
Currently Achievable Waste Loading : Densified silica aerogels retained ~20 mass% of iodine.
Currently Achievable Density: ~2.1 g/cm^3
Durability: Woignier et al. (1998) indicated that the sintered aerogels had a chemical durability two orders of magnitude higher than that of borosilicate glass. High chemical durability and structural stability of the waste form is achieved by high content of silica and low content of modifier oxides. Moreover, silica glass exhibits good mechanical properties and high thermal shock resistance.
Potential Improvement: The optimization of surface-functionalization chemistries and sintering conditions can more than double the current waste loading.
Remaining Questions: Determine the iodine sorption performance including the breakthrough curves of functionalized silica aerogels in a gas stream simulating the fuel reprocessing off-gas. Perform the detailed study of the chemical durability of iodine loaded densified silica aerogels.
Woignier T., Reynes J., Phalippou J., Dussossoy J.L. and Jacquet-Francillon N. *Sintered silica aerogel: a host matrix for long life nuclear wastes*. Journal of Non-Crystalline Solids 225 353-357 (1998).

G-3 Chalcogenide Aerogel

Targeted Waste Stream(s): Iodine

Description: Owing to its long half-life (1.6×10^7 y) wastes containing ^{129}I must be immobilized for hundreds of thousands of years. The EPA regulation 40 CFR 190 requires an ^{129}I capture decontamination factor (DF) of roughly 200 (more than 99% captured and immobilized). While the iodine (I) waste stream can be currently captured on a silver-loaded zeolite (AgZ), this work is focused on a more stable and durable iodine waste form facilitated by chalcogels – aerogels containing metal and chalcogenide (i.e., S, Se, and/or Te) atoms. Better stability and efficiency of the chalcogel-based waste form are expected from stronger complex formations over AgZ: a stronger chemical affinity of chalcogenide atom with iodine gas. The stronger chemical affinity is due to the soft Lewis acid/soft Lewis base complex formation, according to the Hard/Soft Acid-Base (HSAB) principle. We have successfully synthesized Ge-S chalcogels, modified the gelation process to increase the effective surface area, and conducted iodine gas adsorption tests (in a saturation condition), as well as characterized the chalcogel (e.g., surface area, pore sizes, and pore distribution). Sorption tests are ongoing to determine capture efficiency at similar low iodine concentrations and high flow conditions to those expected in the recycle off-gas system.

Currently Achievable Waste Loading: The chalcogels have been shown to adsorb twice their weight in iodine under saturated conditions at room temperature. The uptake at low concentrations expected in the off-gas stream is currently under investigation.

Currently Achievable Density: For amorphous Ge_4S_{10}, $\rho \sim 3000$ kg/m^3, though with the addition of iodine and Pt in the structure, a collapsed chalcogel would have a significantly higher density.

Durability: Although performance assessment tests have not yet been performed, acceptable chemical durability is expected based on limited durability data on chalcohalides. If the durability of Ge-S aerogels is unacceptable, a large number of other potential chemistries are available to optimize the performance.

Potential Improvement: A precious metal (e.g., Pt) is currently being used in chalcogel to catalyze network formation. Investigations into the replacement of the Pt with a more cost-effective alternative are planned. In addition, higher selectivity to iodine may be possible through optimization of physical characteristics such as pore size.

Remaining Questions: The following issues are remaining:

- *Sintering to full density*. Chalcogels begin to densify at 150–300 °C. Therefore, by controlling chemistry and tailoring pores, one can design a chalcogel that will densify below

500 °C. It expected that low volatilization losses are expected with this low temperature process, but the experiments have yet to be performed.

- *Long-term corrosion of chalcogel*. No chemical durability data for chalcogels has been reported in the literature. Also, limited durability data exist for bulk chalcogenide glasses, but some compositions exhibit reasonable resistance to corrosion.

G-4 Krypton Implantation in Silicon Carbide

Targeted Waste Stream(s): Krypton

Description: Because of the low diffusivities for fission products and superior physical and chemical stability of silicon carbide (SiC), it is an ideal candidate for immobilization of C, Kr, and Xe (and perhaps other fission products). Silicon carbide possesses excellent physical and chemical properties that make it a promising material for many nuclear applications, including structural components in fusion reactors [1-5], a barrier for fission product diffusion in gas-cooled fission reactors [6], and an inert matrix for the transmutation of plutonium and other TRUs [7-8]. The high thermal conductivity of SiC also enhances homogeneous heat distribution in devices and rapid heat transfer. Unlike traditional semiconductor materials, thermal diffusion of dopants in SiC requires extremely high temperatures because of the extremely low diffusivities for impurities in SiC. This low diffusivity for impurities is one of the reasons SiC is used as the fission product barrier in tristructural-isotropic (TRISO) nuclear fuel and its use for this application has been demonstrated at temperatures over 2000 K [9-11]. Studies have also shown that Xe and Kr have extremely low mobilities in SiC at low temperatures and only become mobile above 1200˚C [12-14]. Since Xe and Kr diffusivities are extremely low at ambient temperatures, their release rates are expected to be negligible for millions of years, even along grain boundaries [12-13, 15-19]. One prior study indicated that up to 20 atomic % Ar could be entrained during PVD of Ti-films [20]. It is anticipated that equivalent amounts for Xe and Kr could be entrained within CVD or PVD SiC, resulting in waste loadings of >50 mass %. Because of the low diffusivities for fission products and superior physical and chemical stability of SiC, it is an ideal candidate for immobilization of C, Kr, and Xe (and perhaps other fission products).

Since industrial scale fabrication of SiC monoliths with physical vapor deposition methods is a well-established technology [21-25], there is little, if any, technology development needed to produce dense, pure SiC. We chose physical vapor deposition SiC as an industrially scaleable process from which to begin this development. As a demonstration, we used low energy ion guns to bombard krypton onto the surface of a SiC film during its deposition by magnetron sputtering. A depth profile study of three deposition conditions revealed bulk (non-surface) concentrations of 1.4, 5.7, and 6.7 mass % krypton. Although as yet significantly lower than theoretically possible, these loadings successfully demonstrate the potential of gaseous radionuclide immobilization in silicon carbide.

Currently Achievable Waste Loading: 6.7 mass%

Currently Achievable Density: 2.8 g/cm^3

Durability: Although performance assessment tests have not yet been performed, many literature studies have predicted negligible release rates for noble gasses through SiC over millions of years, even along grain boundaries [12-13, 15-19].

Potential Improvement: The use of a more-powerful ion implantation system would enable higher loadings and less unincorporated krypton during waste form production. Models predict the possible incorporation of over 50 mass% krypton, but these levels may be accompanied by clustering (bubbles) and decreased overall durability. Regardless, waste loadings of 3-5 times the currently demonstrated levels are expected to be achievable.

Remaining Questions: The percentage of krypton introduced into the waste form deposition chamber which is incorporated into the waste form has not yet been determined. Detailed krypton release studies have not yet been performed, although the concentration of krypton incorporated in the films has been observed to be stable in ambient conditions and in ultra-high vacuum.

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G-5 Iron Phosphate Based Glasses

Targeted Waste Stream(s): Mixed aqueous waste stream with high molybdenum
Description: Phosphate glasses have long been seen as a potential option for the vitrification of legacy wastes. For example, Russia is actively in production of alkali-alumino-phosphate waste glasses.[1] Phosphate glasses have several differences with borosilicate glasses, some beneficial and some posing operational challenges. Phosphate glasses have different solubilities for various elements than silicate glasses, including higher solubility for key radionuclides. Several of the limits currently known for alkali-boro-silicate waste glasses may be at least partially relieved by alkali-iron-phosphate glasses, due to higher solubility, including S, Cr, Fe, Mo, Cl, F, and lanthanide elements. On the other hand, they tend to be more corrosive to glass contact materials, (i.e. metals, electrodes, some crucible materials) and this is the subject of current studies. Although less technically mature than the borosilicate glasses currently produced in the U.S., a great deal of research has been performed illustrating the promise of the iron-phosphate system for legacy waste glasses.[2-11]
Based on a recent study [12], iron phosphate-based compositions are also a feasible alternative for the immobilization of a combined waste stream produced by the aqueous reprocessing method. These scoping tests analyzed 28 separate compositions and developed several promising glass waste forms. Glasses with 30 mass% waste loading were produced by melting at or below 1200 °C for 1.5 to 2 hours. Waste loadings up to 45 wt% were also achieved, albeit by melting at 1450 °C and producing a small amount of crystal formation in the waste forms. The chemical durability of these partially crystalline waste forms also appeared to exceed the DOE requirements for PCT.
Currently Achievable Waste Loading: 30 mass%
Currently Achievable Density: 3.35 g/cm^3

Durability: The chemical durability of vitrified iron phosphate-based waste forms containing waste loadings up to 30 wt% meets all current DOE requirements for PCT.
Potential Improvement: There is strong potential for improvement in several areas, including (a) increasing the waste loading, (b) lowering the melting temperature, (c) improving the vitrification tendency of the melt, and (d) improving the chemical durability of the waste forms by modifying the base iron phosphate composition. In addition, the use of cold-crucible induction melter technology could help mitigate the corrosiveness of phosphate glasses with regards to crucibles and electrical elements.
Remaining Questions: The processing and scale-up remain as issues, although the operational Russian vitrification process speaks to the potential of this system.

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G-6 Glass ceramic from a melt process

Targeted Waste Stream(s): 1) Combined Cs + LN, or 2) Combined Cs + TM + LN fission products from the UREX$^+$ process

Description: A glass ceramic waste from was used as an alternative to glass to immobilize two combined waste streams options generated by the uranium extraction plus (UREX$^+$) process: 1) combined cesium strontium (Cs) + lanthanide (LN) fission products, 2) combined Cs + transition metal (TM) + LN fission products.

Option 1 glass ceramics were developed to provide a crystalline phase for each of the major fission products types, being lanthanide, alkali, and alkaline earths. The lanthanides were accommodated by two crystalline phases, oxyapatite ($Ca_{2.2}Nd_{7.8}(SiO_4)_6O_{1.9}$ and a lanthanide borosilicate ($Nd_3BSi_2O_8$). The alkaline earths were immobilized in a barium alumino silicate ($BaAl_2Si_2O_8$) and oxyapatite. The vast majority of alkalis are in the glass phase. The glass ceramics have a maximum 80 mass% total crystallinity in the waste form.

Option 2 glass ceramics were developed to provide, at a minimum, a durable crystalline phase for MoO_3 which forms a separated phase because of it limited solubility in silicate glasses. The target phase for the molybdenum is powellite ($(Ca,Nd)_1MoO_3$). Initial glass ceramic development resulted in a glass ceramic that contains powellite, oxyapatite, and lanthanide borosilicate phases. The glass phase separates upon cooling into both molybdate rich and silicate rich liquids. The molybdate rich phase then crystallizes into powellite. Oxyapatite and lanthanide borosilicate phases crystallize upon heat treatment of the glass at temperatures of 800–1100 °C.

Currently Achievable Waste Loading: Option 1 59 mass%
Option 2 42 mass%

Currently Achievable Density: Option 1 ~4000 kg/m^3
Option 2 ~3000 kg/m^3

Durability: Durability of the glass ceramics has not been evaluated to date, however similar highly crystallized glasses were shown to be quite durable.[1]

Potential Improvement: The major improvement that can be made with the glass ceramics is the development of a glass ceramic that requires minimal deviation from the slow cooling temperature profile that would be expected based on the canister. The ultimate goal is to avoid any reheating of the glass ceramic after cooling of the canister which will allow the fabrication of glass ceramics with the same infrastructure as used for glass.

Remaining Questions: What is the durability of the glass ceramic waste forms for option 1 and 2? Fully characterize the elemental compositions of the crystalline phases. Is T_g still a concern for glass ceramics as it is for glass?

G-7 HLW Cermet

Targeted Waste Stream(s): Aqueous HLW Raffinate, UDS, Cladding waste

Description: This work is directed at providing a new option for HLW with the benefits of simplified separations technologies and improved waste forms, which, in turn, enables overall cost minimization. The cermet waste form will improve the heat transfer characteristics and reduce the centerline temperature of the waste monolith and thus allow for increased waste loading compared with that for a glass.

The ceramic phase of the cermet waste form provides a matrix that sequesters the short-lived, high-heat-generating fission product ^{137}Cs/Ba and ^{90}Sr/Y oxide components of used fuels. The metal phase of the cermet provides improved heat transfer and a matrix that will isolate the long-lived metallic fission products (e.g., Tc, Mo, Ru, Rh, Pd, and Ag) and cladding and hardware materials (e.g., Fe, Cr, Ni, and Sn). We believe that the cermet waste form may result in significant cost benefits through the improvements to the heat transfer from the waste during decay heat storage.

Currently Achievable Waste Loading: > 50 mass% (durability unknown).

Currently Achievable Density: > 5,000 kg/m^3 for density of waste-loaded CERMET.

Durability: Unknown at this time.

Thermal Diffusivity: Unknown at this time – To be measured in FY10. Estimated to be ~4μm^2/μs^2

Potential Improvement: The current research efforts explore alternate "cermet" waste forms with higher loading of high-heat components that would not require the segregation wastes. Cermets are a composite material that consists of ceramic phases dispersed in a continuous metal phase. This provides an opportunity to tailor the metal and oxide phases of the material to produce a waste form that possesses the advantages of both materials. Success in this work may also enable an important new option of using the decay heat emissions during storage for process and facility heating applications.

Remaining Questions: What is the leach performance of loaded cermets? Degree to which the ceramic phase can be tailored? What key additives can help to improve durability / performance? What is the optimum waste loading limit and how do changes in waste loading affect the performance?

G-8 Heat-tolerant Cermets

Targeted Waste Stream(s): Alkali and alkaline earth elements

Description: Jim Cunnane is working on adding copper to the baseline Cs/Sr bentonite clay materials to improve heat transfer. Basically take the maximum clay waste loading and dilute it with 70 mass% Cu. As I recall, the max waste loading for the bentonite was pretty high and caused the heat limit to be reached. The addition of copper here allows for efficient thermal transport and a higher heat load. Research is ongoing, so no hard data yet.

Currently Achievable Waste Loading: 30 mass% bentonite/waste, 70 mass% Cu

Currently Achievable Density: 3.15 g/cm3

Durability: Same as Kaminski's clay form.

Potential Improvement: Potential to reach limit of pure Cs pollucite (42 mass% Cs) as the clay phase

Remaining Questions: The processing and scale-up are relatively poorly developed, particularly for radioactive work.

G-9 Immobilization of 99Tc as epsilon metal

Targeted Waste Stream(s): ^{99}Tc
Description: In spent nuclear fuel, five metals, Mo-Tc-Ru-Pd-Rh, accumulate in small particles called the epsilon phase or ε-metal. The number and sizes of the ε-metal particles depends on fuel burn-up and fuel type (Kleykamp 1987; Kleykamp 1988). These phases have been of interest for some time because they constitute some of the sludge in the dissolver after spent fuel has been dissolved in acid (Kleykamp 1987; Schmid 1986). These particles were first characterized in fuel by Kleykamp and co-workers (Kleykamp 1985; Kleykamp 1988; Kleykamp et al. 1985; Kleykamp, and Pejsa 1984) and later by Thomas and coworkers (Thomas, Einziger, and Woodley 1989; Thomas, and Guenther 1989). Although there is some variation in the composition, the typical composition of the ε-metal phase is, in mass%, Mo (33.7), Ru (40.5), Tc (7.0), Pd (11.7), Rh (4.2), and Te (3.8) (Cui et al. 2004). Studies of the dissolution of the ε-metal phase, either synthetic or actual, come to the same conclusions that Mo dissolves more readily than Tc (Re) and Tc more readily than the remaining metals (Cui, Eriksen, and Eklund 2001; Cui et al. 2004; Finn, Hoh, and Wolf 1996; Röllin, Spahiu, and Eklund 2001; Wronkiewicz et al. 2002)
Natural analogues of the ε-metal phase exist or, more appropriately, existed in the remains of the natural nuclear reactors at Okolo, Gabon. The uranium ore for these natural reactors are located in four important deposits are named Okolo, Okélobondo, Mikouloungou, and Bangombé (Cui et al. 2004). These reactors operated about 2 Ga ago. The fate of fission products has been studied extensively (Cui, Eriksen, and Eklund 2001; Cui et al. 2004; Hidaka, and Holliger 1998; Utsunomiya, and Ewing 2006). These examinations come to the same conclusions as the studies of synthetic or actual ε-metal phases, namely that the Mo and Tc are preferentially dissolved from the ε-metal phase.
Currently Achievable Waste Loading: 100% based on ε-metal
Currently Achievable Density: 12×10^3 kg/m^3
Durability: Based on the results from the examination of the natural reactors at Okolo, Gabon, the long term behavior of the ε-metal is more than acceptable. The addition of the extra ^{99}Tc dissolved into the aqueous phase during fuel dissolution may change that durability and will be the subject of the testing program.
Potential Improvement: It is unlikely that there will be improvements to this waste form.
Remaining Questions: Some ^{99}Tc reports to the aqueous phase during fuel dissolution. If this ^{99}Tc is separated, reduced to metal, and combined with the ε-metal, there is a possibility that the durability of the metal phase could be degraded.

Appendix H
Constraints on Waste Packaging

Waste package counts and the resulting packaged waste volumes are dependent upon the size of the containers selected for each waste form. The approach used in this study is to select the maximum container size, resulting in the minimum number of containers and volume, based upon the constraints discussed below. The waste form properties such as waste loading (decay heat) and density are considered in applying these constraints.

H-1 Physical Constraints

Any repository will be designed to accept a variety of waste packages and is subject to physical size constraints resulting from different design attributes. This study assumes limits developed for implementation of the repository at Yucca Mountain are reasonable for other repository locations. This study assumes the diameter, length, and mass of the largest container is limited to 0.6m (2 ft) in diameter, 4.5m (15 feet) long and 4,200 kg as proposed by the waste acceptance documents for Yucca Mountain.[i] The largest empty container weighs about 600 kg (1300 lbs), so the maximum canister capacity is about 3600 kg. Also, containers are only filled about 90% full to prevent overflow, reducing the working volume to about 1.18 m^3. Thus waste forms with a density greater than 3600 kg / 1.18 m^3 ~3,000 kg/m^3 will be weight limited provided no other limits are more constraining.
Containers which have diameters less than 18 inches use a diameter to length ratio of 1 to 10 to calculate the volume of the container.

H-2 Thermal Constraints

Waste forms are subject to thermal stability limits. For example, the alkali/alkaline earth elements are very soluble in glass; it is possible to add so much radioactive Cs/Sr to glass that it causes it to exceed the glass transition (tolerance) temperature, allowing it to crystallize, thereby altering the waste form. Other radionuclides also contribute significant decay heat. Container size must therefore be limited based on waste loading, total thermal output rate (power), and the acceptable maximum allowable temperature to prevent waste form degradation. Thermal constraints are developed for waste form interim storage for the time between production and disposal.
Referring to Figure H-1, heat is conducted radially to the cylindrical surface based on the difference between the centerline temperature, T_c, and the waste container surface temperature, T_s, at a rate correlated to the waste form thermal conductivity, k (W/m/K). From the canister surface heat is lost to the environment primarily by convection as air flows up through the cylindrical vault in which the canister is placed. Heat loss is proportional to the temperature difference between the container surface, T_s, and the nominal air temperature, T_∞, at a rate correlated to the convective heat transfer coefficient, h ($W/m^2/K$). Heat can also be lost radiantly from the canister surface and radiated back by the vault, but for simplicity it is assumed here that the canister and wall surfaces are in near equilibrium and net radiant losses are negligible. Similarly, there are conductive and convective heat losses from the container bottom and top respectively, but these are assumed to be negligible in comparison to the convective losses with free flow of ventilation air.

A simplified analytical solution for a heat generating cylinder[ii] is used to calculate the heat generation rate, q (W/m^3). The simplified expression is shown below:

$$q = 4 \times [(T_c) - (T_\infty)] / [r^2/k + 2r/h]$$

where,

q is the heat generation rate in (W/m^3),
T_c is the centerline temperature in (C),
T_∞ is the air temperature in (C),
r is the radius in (m),
k is thermal conductivity in (W/m/K),
h is convective heat transfer coefficient in ($W/m^2/K$)

Table H-1 provides the heat generation rate per unit volume for various cylinder diameters and centerline temperatures for borosilicate glass (where k=1.0 W/m/K represents the thermal conductivity of borosilicate glass, h= 5.7 $W/m^2/K$ represents the heat transfer coefficient from a cast iron transmission surface though air, and T_∞=125 °F (51.7 C) represents the air temperature). Note that the table is specific to borosilicate glass. The thermal conductivity and tolerance temperature of other waste forms are summarized in Table H-3.

Table H-2 provides the maximum heat generation rate per container of borosilicate glass by multiplying the unit volume heat generation rate calculated in Table H-1 by the volume of each container. Container volumes were calculated using the diameter to length ratio of 1 to 10 for containers up to a maximum diameter of 18 inches. For containers with a diameter of 24 inches, a length of 15 feet was used. Additionally, containers were assumed to only be filled to 90% of capacity. Assuming the tolerance temperature for borosilicate glass is 650 C and a container diameter of 24 inches, the maximum allowable decay heat is approximately 14,382 watts. Using this methodology, the maximum allowable decay heat is approximately 7,390 watts for the Cs/Sr ceramic waste form (k=300 W/m/K, tolerance temperature=1,000 C, diameter=22 inches) and 34,140 watts for the metal alloy waste form (k=17 W/m/K, tolerance temperature=1,250 C, diameter=24 inches).

For the advanced waste forms in section 4.5, the maximum waste loading was applied and using the applicable thermal conductivity and tolerance temperature in Table H-3, the maximum diameter of the container was determined.

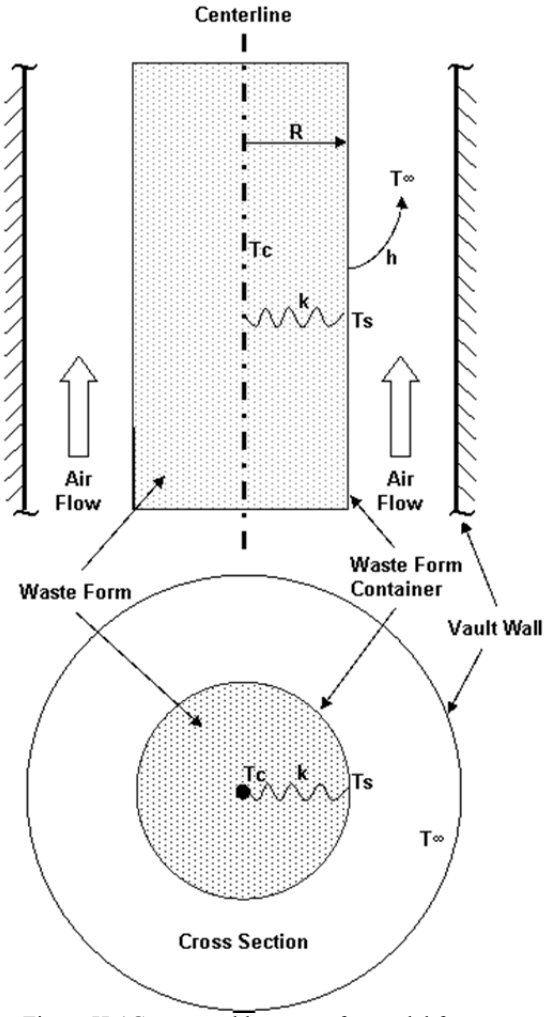

Figure H-1Conceptual heat transfer model for waste container in storage

Table H-1 Maximum Decay Heat per Unit Volume W/m3 for Borosilicate Glass

		Center Line Temp (C)							
Diameter (in)	Diameter (cm)	500	600	650	800	900	1000	1250	1500
3	7.6	121,007	147,998	161,493	201,979	228,969	255,960	323,436	390,912
6	15.2	55,106	67,397	73,543	91,980	104,271	116,563	147,291	178,019
9	22.9	33,728	41,252	45,013	56,298	63,821	71,344	90,152	108,959
12	30.5	23,381	28,596	31,204	39,027	44,242	49,457	62,495	75,533
18	45.7	13,538	16,557	18,067	22,597	25,616	28,636	36,185	43,734
24	61.0	8,973	10,975	11,976	14,978	16,979	18,981	23,985	28,988

Basis: Thermal conductivity k=1.0 W/m/K, Convective heat transfer coefficient h= 5.7 W/m2/K, Air temperature T_{∞}=125 °F

Table H-2 Maximum Decay Heat (W) per Container for Borosilicate Glass

		Center Line Temp (C)							
Diameter (in)	Diameter (cm)	500	600	650	800	900	1000	1250	1500
3	7.6	378	463	505	632	716	801	1,012	1,223
6	15.2	1,379	1,686	1,840	2,301	2,609	2,916	3,685	4,454
9	22.9	2,848	3,483	3,801	4,754	5,389	6,024	7,613	9,201
12	30.5	4,680	5,724	6,246	7,812	8,855	9,899	12,509	15,119
18	45.7	9,145	11,185	12,205	15,265	17,305	19,345	24,444	29,544
24	61.0	10,777	13,180	14,382	17,988	20,392	22,795	28,805	34,814

Basis: Thermal conductivity k=1.0 W/m/K, Convective heat transfer coefficient h= 5.7 W/m2/K, Air temperature T_{∞}=125 °F

[i] "Waste Acceptance System Requirements Documents" DOE/RW-0351 rev 5, DOC.20070522.0007, May 31, 2007

[ii]

Table H-3 Properties of High Decay Heat Waste Forms

Alternative Waste Form	Target Waste Stream	Waste Loading (mass%)	Density (g/cm3)	Temperature Tolerance (°C)	Thermal Conductivity (W/m*K)
Epsilon Metal	Tc, noble metals	100	12	1200	100
Bi-based Glasses	I	25	4.4	400	0.1
Silica Aerogels	I	20	2.1	1000	1.3
Chalcogenide Aerogels	I	67	3	150	0.3
Silicon Carbide Deposition	Kr	6.7	2.8	1500	20
Glass Ceramic	Cs/Sr + Lanth + Trans Metals	42	3	1250	0.6
Iron Phosphate Glass	Cs/Sr + Lanth + Trans Metals	30	3.4	650	0.6
Heat-tolerant Cermets	Cs/Sr	10	3.15	1000	300
HLW Cermet	Aqueous raffinate + UDS/H&H	50	5	reasonable	high

[i] "Waste Acceptance System Requirements Documents" DOE/RW-0351 rev 5, DOC.20070522.0007, May 31, 2007

[ii] Rohsenow, W. M., Hartnett, J. P., Handbook of Heat Transfer, McGraw Hill, 1973, p. 3-109

Appendix I
Commercial MOX Fuel Radionuclide Characteristics

Table I-1 Commercial MOX Fuel Isotopic Data at 5, 30, 100 and 500 years Decay Time

Burn-up (MWd/MTIHM)	**50,000**			
Enrichment (%)	6.83 Fissile Pu			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
H 3	7.68E-02	1.89E-02	3.71E-04	6.58E-14
HE 4	3.21E+01	5.82E+01	1.11E+02	2.65E+02
LI 6	3.54E-04	3.54E-04	3.54E-04	3.54E-04
LI 7	1.51E-05	1.51E-05	1.51E-05	1.51E-05
BE 9	2.90E-05	2.90E-05	2.90E-05	2.90E-05
BE 10	1.94E-04	1.94E-04	1.94E-04	1.94E-04
C 14	3.91E-05	3.90E-05	3.87E-05	3.69E-05
ZN 66	3.03E-08	3.03E-08	3.03E-08	3.03E-08
ZN 67	9.22E-10	9.22E-10	9.22E-10	9.22E-10
ZN 68	1.20E-11	1.20E-11	1.20E-11	1.20E-11
ZN 70	0.00E+00	0.00E+00	0.00E+00	0.00E+00
SE 76	1.12E-02	1.12E-02	1.12E-02	1.12E-02
SE 77	1.24E+00	1.24E+00	1.24E+00	1.24E+00
SE 78	3.35E+00	3.35E+00	3.35E+00	3.35E+00
SE 79	7.44E+00	7.44E+00	7.43E+00	7.40E+00
SE 80	1.61E+01	1.61E+01	1.61E+01	1.61E+01
SE 82	3.48E+01	3.48E+01	3.48E+01	3.48E+01
KR 80	5.73E-04	5.73E-04	5.73E-04	5.73E-04
KR 81	7.18E-05	7.18E-05	7.18E-05	7.17E-05
KR 82	1.57E+00	1.57E+00	1.57E+00	1.57E+00
KR 83	4.46E+01	4.46E+01	4.46E+01	4.46E+01
KR 84	1.04E+02	1.04E+02	1.04E+02	1.04E+02
KR 85	1.48E+01	2.94E+00	3.18E-02	1.86E-13
KR 86	1.57E+02	1.57E+02	1.57E+02	1.57E+02
RB 85	9.09E+01	1.03E+02	1.06E+02	1.06E+02
RB 87	1.96E+02	1.96E+02	1.96E+02	1.96E+02
SR 86	5.10E-01	5.10E-01	5.10E-01	5.10E-01
SR 87	5.98E-03	5.98E-03	5.98E-03	5.98E-03
SR 88	2.74E+02	2.74E+02	2.74E+02	2.74E+02
SR 89	2.16E-10	0.00E+00	0.00E+00	0.00E+00
SR 90	3.61E+02	1.99E+02	3.76E+01	2.76E-03
Y 89	3.47E+02	3.47E+02	3.47E+02	3.47E+02
Y 90	9.05E-02	4.99E-02	9.43E-03	6.91E-07
Y 91	1.13E-08	0.00E+00	0.00E+00	0.00E+00
ZR 90	6.63E+01	2.28E+02	3.90E+02	4.27E+02

Table I-1 Continued

Burn-up (MWd/MTIHM)	**50,000**			
Enrichment (%)	6.83 Fissile Pu			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
ZR 91	5.01E+02	5.01E+02	5.01E+02	5.01E+02
ZR 92	6.08E+02	6.08E+02	6.08E+02	6.08E+02
ZR 93	7.64E+02	7.64E+02	7.64E+02	7.64E+02
ZR 94	8.57E+02	8.57E+02	8.57E+02	8.57E+02
ZR 95	1.53E-07	0.00E+00	0.00E+00	0.00E+00
ZR 96	1.03E+03	1.03E+03	1.03E+03	1.03E+03
NB 93	5.43E-04	5.96E-03	2.90E-02	1.67E-01
NB 93M	1.96E-03	5.20E-03	6.42E-03	6.45E-03
NB 94	2.17E-03	2.17E-03	2.16E-03	2.14E-03
NB 95	1.93E-07	0.00E+00	0.00E+00	0.00E+00
NB 95M	6.38E-11	0.00E+00	0.00E+00	0.00E+00
MO 95	9.17E+02	9.17E+02	9.17E+02	9.17E+02
MO 96	5.71E+01	5.71E+01	5.71E+01	5.71E+01
MO 97	1.10E+03	1.10E+03	1.10E+03	1.10E+03
MO 98	1.19E+03	1.19E+03	1.19E+03	1.19E+03
MO100	1.42E+03	1.42E+03	1.42E+03	1.42E+03
TC 98	1.47E-02	1.47E-02	1.47E-02	1.47E-02
TC 99	1.11E+03	1.11E+03	1.11E+03	1.11E+03
RU 99	2.54E-02	1.16E-01	3.69E-01	1.81E+00
RU100	2.01E+02	2.01E+02	2.01E+02	2.01E+02
RU101	1.24E+03	1.24E+03	1.24E+03	1.24E+03
RU102	1.41E+03	1.41E+03	1.41E+03	1.41E+03
RU103	5.52E-13	0.00E+00	0.00E+00	0.00E+00
RU104	1.34E+03	1.34E+03	1.34E+03	1.34E+03
RU106	1.11E+01	3.79E-07	0.00E+00	0.00E+00
RH102	1.01E-03	2.56E-06	1.39E-13	0.00E+00
RH103	9.21E+02	9.21E+02	9.21E+02	9.21E+02
RH103M	4.95E-16	0.00E+00	0.00E+00	0.00E+00
RH106	1.04E-05	3.56E-13	0.00E+00	0.00E+00
PD104	5.18E+02	5.18E+02	5.18E+02	5.18E+02
PD105	1.08E+03	1.08E+03	1.08E+03	1.08E+03
PD106	1.12E+03	1.13E+03	1.13E+03	1.13E+03
PD107	7.68E+02	7.68E+02	7.68E+02	7.68E+02
PD108	5.41E+02	5.41E+02	5.41E+02	5.41E+02
PD110	1.77E+02	1.77E+02	1.77E+02	1.77E+02
AG107	5.67E-04	2.62E-03	8.36E-03	4.11E-02
AG108	1.25E-14	1.09E-14	7.44E-15	8.39E-16
AG108M	3.96E-06	3.45E-06	2.36E-06	2.66E-07
AG109	2.41E+02	2.41E+02	2.41E+02	2.41E+02
AG109M	2.73E-13	3.25E-19	0.00E+00	0.00E+00

Table I-1 Continued

Burn-up (MWd/MTIHM)	**50,000**			
Enrichment (%)	6.83 Fissile Pu			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
AG110	2.91E-10	2.91E-21	0.00E+00	0.00E+00
AG110M	1.92E-02	1.92E-13	0.00E+00	0.00E+00
CD108	2.70E-03	2.70E-03	2.70E-03	2.70E-03
CD109	2.76E-07	3.29E-13	0.00E+00	0.00E+00
CD110	1.97E+02	1.97E+02	1.97E+02	1.97E+02
CD111	9.27E+01	9.27E+01	9.27E+01	9.27E+01
CD112	4.91E+01	4.91E+01	4.91E+01	4.91E+01
CD113	5.51E-01	5.51E-01	5.52E-01	5.52E-01
CD113M	5.93E-01	1.81E-01	6.50E-03	3.63E-11
CD114	5.52E+01	5.52E+01	5.52E+01	5.52E+01
CD115M	4.09E-14	0.00E+00	0.00E+00	0.00E+00
CD116	1.62E+01	1.62E+01	1.62E+01	1.62E+01
IN113	2.16E-01	6.28E-01	8.02E-01	8.08E-01
IN113M	0.00E+00	0.00E+00	0.00E+00	0.00E+00
IN114	5.52E-20	0.00E+00	0.00E+00	0.00E+00
IN114M	3.43E-15	0.00E+00	0.00E+00	0.00E+00
IN115	4.04E+00	4.04E+00	4.04E+00	4.04E+00
IN115M	1.16E-20	0.00E+00	0.00E+00	0.00E+00
SN114	7.43E-03	7.43E-03	7.43E-03	7.43E-03
SN115	6.90E-01	6.90E-01	6.90E-01	6.90E-01
SN116	1.66E+01	1.66E+01	1.66E+01	1.66E+01
SN117	1.58E+01	1.58E+01	1.58E+01	1.58E+01
SN118	1.60E+01	1.60E+01	1.60E+01	1.60E+01
SN119	1.63E+01	1.63E+01	1.63E+01	1.63E+01
SN119M	4.20E-04	2.54E-15	0.00E+00	0.00E+00
SN120	1.62E+01	1.62E+01	1.62E+01	1.62E+01
SN121M	8.55E-03	6.05E-03	2.29E-03	8.92E-06
SN122	1.73E+01	1.73E+01	1.73E+01	1.73E+01
SN123	2.73E-05	1.47E-26	0.00E+00	0.00E+00
SN124	2.31E+01	2.31E+01	2.31E+01	2.31E+01
SN126	5.57E+01	5.57E+01	5.56E+01	5.55E+01
SB121	1.51E+01	1.51E+01	1.51E+01	1.51E+01
SB123	1.84E+01	1.84E+01	1.84E+01	1.84E+01
SB124	1.01E-10	0.00E+00	0.00E+00	0.00E+00
SB125	6.92E+00	1.33E-02	3.28E-10	0.00E+00
SB126	2.64E-06	2.64E-06	2.64E-06	2.64E-06
SB126M	2.01E-08	2.01E-08	2.01E-08	2.00E-08
TE122	1.54E+00	1.54E+00	1.54E+00	1.54E+00
TE123	2.72E-02	2.72E-02	2.72E-02	2.72E-02
TE123M	1.06E-07	0.00E+00	0.00E+00	0.00E+00

Table I-1 Continued

Burn-up (MWd/MTIHM)	**50,000**			
Enrichment (%)	6.83 Fissile Pu			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
TE124	1.23E+00	1.23E+00	1.23E+00	1.23E+00
TE125	3.13E+01	3.83E+01	3.84E+01	3.84E+01
TE125M	9.68E-02	1.86E-04	4.59E-12	0.00E+00
TE126	1.81E+00	1.82E+00	1.85E+00	2.00E+00
TE127	5.93E-08	1.50E-31	0.00E+00	0.00E+00
TE127M	1.69E-05	4.27E-29	0.00E+00	0.00E+00
TE128	2.10E+02	2.10E+02	2.10E+02	2.10E+02
TE129	6.77E-20	0.00E+00	0.00E+00	0.00E+00
TE129M	7.23E-17	0.00E+00	0.00E+00	0.00E+00
TE130	6.34E+02	6.34E+02	6.34E+02	6.34E+02
I127	1.16E+02	1.16E+02	1.16E+02	1.16E+02
I129	3.47E+02	3.47E+02	3.47E+02	3.47E+02
XE127	5.65E-21	0.00E+00	0.00E+00	0.00E+00
XE128	1.02E+01	1.02E+01	1.02E+01	1.02E+01
XE129	6.56E-02	6.60E-02	6.71E-02	7.32E-02
XE130	2.19E+01	2.19E+01	2.19E+01	2.19E+01
XE131	6.44E+02	6.44E+02	6.44E+02	6.44E+02
XE132	1.82E+03	1.82E+03	1.82E+03	1.82E+03
XE134	2.17E+03	2.17E+03	2.17E+03	2.17E+03
XE136	3.20E+03	3.20E+03	3.20E+03	3.20E+03
CS133	1.60E+03	1.60E+03	1.60E+03	1.60E+03
CS134	3.91E+01	8.75E-03	5.28E-13	0.00E+00
CS135	9.17E+02	9.17E+02	9.17E+02	9.17E+02
CS137	1.61E+03	9.06E+02	1.80E+02	1.74E-02
BA132	4.32E-03	4.32E-03	4.32E-03	4.32E-03
BA134	2.79E+02	3.18E+02	3.18E+02	3.18E+02
BA135	9.66E-01	9.73E-01	9.92E-01	1.10E+00
BA136	5.96E+01	5.96E+01	5.96E+01	5.96E+01
BA137	2.84E+02	9.93E+02	1.72E+03	1.90E+03
BA137M	2.47E-04	1.39E-04	2.75E-05	2.66E-09
BA138	1.78E+03	1.78E+03	1.78E+03	1.78E+03
LA138	3.28E-03	3.28E-03	3.28E-03	3.28E-03
LA139	1.72E+03	1.72E+03	1.72E+03	1.72E+03
CE140	1.72E+03	1.72E+03	1.72E+03	1.72E+03
CE142	1.51E+03	1.51E+03	1.51E+03	1.51E+03
CE144	3.85E+00	8.22E-10	6.90E-37	0.00E+00
PR141	1.56E+03	1.56E+03	1.56E+03	1.56E+03
PR144	1.62E-04	3.47E-14	4.04E-43	0.00E+00
PR144M	8.12E-07	1.74E-16	0.00E+00	0.00E+00
ND142	3.38E+01	3.38E+01	3.38E+01	3.38E+01

Table I-1 Continued

Burn-up (MWd/MTIHM)	**50,000**			
Enrichment (%)	6.83 Fissile Pu			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
ND143	1.10E+03	1.10E+03	1.10E+03	1.10E+03
ND144	1.53E+03	1.54E+03	1.54E+03	1.54E+03
ND145	8.63E+02	8.63E+02	8.63E+02	8.63E+02
ND146	9.55E+02	9.55E+02	9.55E+02	9.55E+02
ND148	5.63E+02	5.63E+02	5.63E+02	5.63E+02
ND150	3.33E+02	3.33E+02	3.33E+02	3.33E+02
PM146	6.73E-03	2.88E-04	4.25E-08	0.00E+00
PM147	4.26E+01	5.77E-02	5.36E-10	0.00E+00
PM148	7.89E-16	0.00E+00	0.00E+00	0.00E+00
PM148M	1.08E-13	0.00E+00	0.00E+00	0.00E+00
SM146	1.60E-02	1.84E-02	1.85E-02	1.85E-02
SM147	2.12E+02	2.55E+02	2.55E+02	2.55E+02
SM148	3.09E+02	3.09E+02	3.09E+02	3.09E+02
SM149	6.71E+00	6.71E+00	6.71E+00	6.71E+00
SM150	4.64E+02	4.64E+02	4.64E+02	4.64E+02
SM151	3.41E+01	2.82E+01	1.64E+01	7.54E-01
SM152	2.04E+02	2.04E+02	2.04E+02	2.04E+02
SM154	9.76E+01	9.76E+01	9.76E+01	9.76E+01
EU150	1.67E-06	1.03E-06	2.68E-07	1.21E-10
EU151	1.39E+00	7.37E+00	1.91E+01	3.48E+01
EU152	1.20E-01	3.35E-02	9.45E-04	1.32E-12
EU153	2.50E+02	2.50E+02	2.50E+02	2.50E+02
EU154	6.30E+01	8.40E+00	2.98E-02	2.97E-16
EU155	2.15E+01	6.52E-01	3.67E-05	1.92E-29
GD152	5.54E-02	7.94E-02	8.85E-02	8.87E-02
GD153	1.91E-04	8.39E-16	0.00E+00	0.00E+00
GD154	4.16E+01	9.61E+01	1.05E+02	1.05E+02
GD155	2.23E+01	4.31E+01	4.38E+01	4.38E+01
GD156	1.71E+02	1.71E+02	1.71E+02	1.71E+02
GD157	3.95E-01	3.95E-01	3.95E-01	3.95E-01
GD158	6.04E+01	6.04E+01	6.04E+01	6.04E+01
GD160	4.82E+00	4.82E+00	4.82E+00	4.82E+00
TB159	9.58E+00	9.58E+00	9.58E+00	9.58E+00
TB160	7.87E-09	0.00E+00	0.00E+00	0.00E+00
HO165	5.90E-01	5.90E-01	5.90E-01	5.90E-01
HO166M	7.22E-03	7.12E-03	6.84E-03	5.43E-03
TM169	2.80E-04	2.80E-04	2.80E-04	2.80E-04
TM170	2.13E-09	0.00E+00	0.00E+00	0.00E+00
TM171	6.44E-07	7.74E-11	8.25E-22	0.00E+00
TL206	4.24E-27	4.24E-27	4.24E-27	4.23E-27

Table I-1 Continued

Burn-up (MWd/MTIHM)	**50,000**			
Enrichment (%)	6.83 Fissile Pu			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
TL207	1.98E-14	7.74E-14	1.70E-13	5.82E-13
TL208	6.57E-12	5.90E-12	3.01E-12	7.20E-14
TL209	7.21E-18	1.79E-17	1.26E-16	7.26E-15
PB206	1.50E-11	3.78E-09	5.53E-07	3.19E-04
PB207	5.06E-09	1.02E-07	7.88E-07	1.23E-05
PB208	8.95E-06	6.33E-05	1.63E-04	2.65E-04
PB209	3.00E-14	7.46E-14	5.26E-13	3.02E-11
PB210	2.18E-10	1.69E-08	7.59E-07	7.55E-05
PB211	1.53E-13	5.99E-13	1.32E-12	4.50E-12
PB212	3.88E-09	3.48E-09	1.78E-09	4.25E-11
PB214	5.51E-15	1.69E-13	3.56E-12	2.04E-10
BI208	4.58E-16	4.58E-16	4.58E-16	4.58E-16
BI209	3.52E-10	2.60E-09	3.48E-08	7.62E-06
BI210	1.34E-13	1.04E-11	4.67E-10	4.64E-08
BI210M	1.63E-15	1.63E-15	1.63E-15	1.63E-15
BI211	9.03E-15	3.53E-14	7.77E-14	2.66E-13
BI212	3.68E-10	3.30E-10	1.68E-10	4.03E-12
BI213	7.06E-15	1.75E-14	1.24E-13	7.10E-12
BI214	4.09E-15	1.25E-13	2.64E-12	1.51E-10
PO210	3.12E-12	2.88E-10	1.29E-08	1.28E-06
PO211	1.11E-19	4.33E-19	9.53E-19	3.26E-18
PO212	1.95E-20	1.75E-20	8.91E-21	2.13E-22
PO213	1.06E-23	2.63E-23	1.86E-22	1.07E-20
PO214	5.63E-22	1.72E-20	3.64E-19	2.08E-17
PO215	1.28E-19	5.01E-19	1.10E-18	3.77E-18
PO216	1.55E-14	1.39E-14	7.09E-15	1.70E-16
PO218	6.38E-16	1.96E-14	4.13E-13	2.36E-11
AT217	8.48E-20	2.11E-19	1.49E-18	8.54E-17
RN219	2.90E-16	1.14E-15	2.50E-15	8.54E-15
RN220	5.84E-12	5.24E-12	2.68E-12	6.40E-14
RN222	1.17E-12	3.60E-11	7.59E-10	4.35E-08
FR221	7.70E-16	1.91E-15	1.35E-14	7.75E-13
FR223	1.35E-15	5.26E-15	1.16E-14	3.96E-14
RA223	7.38E-11	2.89E-10	6.35E-10	2.17E-09
RA224	3.38E-08	3.04E-08	1.55E-08	3.71E-10
RA225	3.48E-12	8.65E-12	6.10E-11	3.50E-09
RA226	1.83E-07	5.60E-06	1.18E-04	6.76E-03
RA228	2.14E-13	1.68E-12	6.54E-12	3.78E-11
AC225	2.35E-12	5.84E-12	4.12E-11	2.37E-09
AC227	5.21E-08	2.04E-07	4.49E-07	1.54E-06

Table I-1 Continued

Burn-up (MWd/MTIHM)	**50,000**			
Enrichment (%)	6.83 Fissile Pu			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
AC228	2.23E-17	1.75E-16	6.83E-16	3.94E-15
TH227	1.21E-10	4.74E-10	1.04E-09	3.57E-09
TH228	6.55E-06	5.90E-06	3.01E-06	7.20E-08
TH229	6.42E-07	1.59E-06	1.12E-05	6.46E-04
TH230	5.49E-03	5.05E-02	3.54E-01	3.88E+00
TH231	1.65E-08	1.66E-08	1.68E-08	1.84E-08
TH232	1.30E-03	4.95E-03	1.54E-02	8.23E-02
TH234	1.24E-05	1.24E-05	1.24E-05	1.24E-05
PA231	4.48E-04	5.46E-04	8.24E-04	2.49E-03
PA233	3.03E-05	4.18E-05	9.00E-05	2.95E-04
PA234	1.86E-10	1.86E-10	1.86E-10	1.86E-10
PA234M	4.17E-10	4.17E-10	4.17E-10	4.17E-10
U232	2.80E-04	2.20E-04	1.12E-04	2.69E-06
U233	4.84E-03	1.33E-02	5.63E-02	8.19E-01
U234	3.42E+02	9.32E+02	2.08E+03	3.59E+03
U235	4.05E+03	4.08E+03	4.14E+03	4.51E+03
U236	4.99E+03	5.05E+03	5.23E+03	6.25E+03
U237	3.18E-04	9.56E-05	3.30E-06	6.50E-09
U238	8.51E+05	8.51E+05	8.51E+05	8.51E+05
U240	4.14E-12	4.14E-12	4.14E-12	4.14E-12
NP235	0.00E+00	0.00E+00	0.00E+00	0.00E+00
NP236	5.69E-03	5.69E-03	5.68E-03	5.67E-03
NP237	8.91E+02	1.23E+03	2.65E+03	8.70E+03
NP238	7.18E-06	6.40E-06	4.65E-06	7.51E-07
NP239	1.69E-03	1.68E-03	1.67E-03	1.61E-03
NP240M	3.62E-14	3.62E-14	3.62E-14	3.62E-14
PU236	9.91E-09	1.27E-08	1.27E-08	1.27E-08
PU237	1.04E-15	0.00E+00	0.00E+00	0.00E+00
PU238	3.35E+03	2.75E+03	1.59E+03	7.06E+01
PU239	3.31E+04	3.31E+04	3.30E+04	3.27E+04
PU240	2.44E+04	2.48E+04	2.49E+04	2.39E+04
PU241	1.03E+04	3.09E+03	1.06E+02	2.10E-01
PU242	8.91E+03	8.91E+03	8.91E+03	8.90E+03
PU243	5.16E-12	5.16E-12	5.16E-12	5.16E-12
PU244	2.16E-01	2.16E-01	2.16E-01	2.16E-01
AM241	4.49E+03	1.13E+04	1.29E+04	6.84E+03
AM242	4.58E-04	4.09E-04	2.97E-04	4.79E-05
AM242M	3.83E+01	3.42E+01	2.48E+01	4.01E+00
AM243	1.96E+03	1.96E+03	1.94E+03	1.87E+03
AM244	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table I-1 Continued

Burn-up (MWd/MTIHM)	**50,000**			
Enrichment (%)	6.83 Fissile Pu			
Cooling Time (years)	5	30	100	500
ISOTOPE	GM/MTIHM	GM/MTIHM	GM/MTIHM	GM/MTIHM
CM241	7.43E-20	0.00E+00	0.00E+00	0.00E+00
CM242	2.05E-01	8.26E-02	6.00E-02	9.69E-03
CM243	1.07E+01	5.83E+00	1.06E+00	6.33E-05
CM244	8.51E+02	3.27E+02	2.24E+01	5.03E-06
CM245	1.31E+02	1.31E+02	1.30E+02	1.26E+02
CM246	7.32E+00	7.30E+00	7.22E+00	6.81E+00
CM247	1.45E-01	1.45E-01	1.45E-01	1.45E-01
CM248	1.10E-02	1.10E-02	1.10E-02	1.10E-02
BK249	3.25E-06	8.38E-15	7.41E-39	0.00E+00
CF249	2.13E-04	2.06E-04	1.79E-04	8.13E-05
CF250	2.74E-05	7.28E-06	1.78E-07	4.98E-15
CF251	1.75E-05	1.71E-05	1.62E-05	1.19E-05
CF252	2.46E-06	3.46E-09	3.56E-17	0.00E+00
Subtotal	1.00E+06	1.00E+06	1.00E+06	1.00E+06
O 16	1.27E+05	1.27E+05	1.27E+05	1.27E+05
TOTAL	1.13E+06	1.13E+06	1.13E+06	1.13E+06
Decay Heat (Watts/MT)	6,940	4,325	2,520	814
Hulls and Hardware (kg/MT)	344	344	344	344

Appendix J
Weapons Grade MOX Fuel Radionuclide Characteristics

Table J-1 Weapons Grade MOX Fuel Isotopic Data at Discharge

Isotope	Discharge g/assembly
H3	3.94E-02
Be10	9.09E-05
C14	1.83E-05
Ni66	2.07E-09
Ni72	4.26E-11
Ni73	5.61E-12
Ni74	5.06E-12
Ni75	5.90E-13
Ni76	3.69E-13
Ni77	4.17E-14
Ni78	1.37E-14
Cu66	3.24E-12
Cu67	2.54E-10
Cu72	3.18E-10
Cu73	3.63E-10
Cu74	5.64E-11
Cu75	8.11E-11
Cu76	1.64E-11
Cu77	1.49E-11
Cu78	3.23E-12
Cu79	9.96E-13
Cu80	8.27E-14
Cu81	7.11E-15
Zn69	1.81E-11
Zn69m	2.12E-10
Zn71	3.56E-11
Zn71m	1.49E-08
Zn72	1.16E-05
Zn73	3.92E-09
Zn74	3.45E-08
Zn75	6.58E-09
Zn76	7.06E-09
Zn77	2.39E-09
Zn78	6.40E-09
Zn79	4.20E-10
Zn80	4.01E-10
Zn81	1.77E-11
Zn82	3.37E-12
Zn83	1.50E-13
Ga70	2.74E-13
Ga72	3.53E-06
Ga73	3.00E-06
Ga74	5.84E-08
Ga75	1.07E-07
Ga76	5.84E-08
Ga77	6.37E-08
Ga78	6.30E-08
Ga79	2.64E-08
Ga80	1.78E-08
Ga81	4.49E-09
Ga82	3.51E-10
Ga83	8.58E-11
Ga84	3.22E-11
Ga85	9.08E-13
As76	9.64E-06
As77	8.43E-04
As78	1.30E-04
As79	2.21E-05
As80	1.62E-06
As80	2.42E-06
As82	2.64E-06
As82m	6.67E-07
As83	2.64E-06
As84	8.99E-07
As85	1.59E-07
As86	3.47E-08
As87	3.54E-09
As88	3.50E-10
As89	4.21E-11
As90	3.29E-13
Ge71	3.44E-12
Ge73m	8.95E-11
Ge75	4.73E-06
Ge75m	2.61E-09
Ge77	8.25E-05
Ge77m	2.74E-07
Ge78	1.23E-04
Ge79	1.60E-06
Ge80	1.89E-06
Ge81	8.49E-07
Ge82	3.44E-07
Ge83	7.48E-08
Ge84	3.15E-08
Ge85	1.13E-09
Ge86	2.71E-10
Ge87	6.56E-11
Ge88	1.06E-12
Co72	3.84E-14
Co73	1.63E-14
Co74	2.72E-15
Co75	4.29E-16
Br79m	1.13E-13
Br80	1.23E-09
Br80m	1.37E-08
Br82	2.02E-03
Br82m	4.95E-06
Br83	2.89E-03
Br84	1.14E-03
Br84m	6.57E-06
Br85	1.07E-04
Br86	3.96E-05
Br86m	7.45E-07
Br87	5.34E-05
Br88	1.39E-05
Br89	2.89E-06
Br90	6.31E-07
Br91	6.30E-08
Br92	6.51E-09
Br93	2.90E-09
Br94	6.18E-11
Br95	7.81E-13
Br96	1.26E-13
Se77m	5.51E-10
Se79	6.42E-01
Se79m	9.51E-06
Se81	1.75E-04
Se81m	4.20E-05
Se83	2.28E-04
Se83m	1.13E-05
Se84	1.14E-04
Se85	9.70E-06
Se85m	4.08E-06
Se86	9.25E-06
Se87	1.75E-06
Se88	2.90E-07
Se89	2.78E-08
Se90	9.47E-09
Se91	4.02E-10
Se92	5.07E-11
Se93	1.99E-13
Kr79	5.71E-13
Kr81m	1.37E-12

Table J-1 (Continued)

Isotope	Discharge g/assembly
Kr83m	2.24E-03
Kr85	9.38E+00
Kr85m	1.01E-02
Kr87	5.69E-03
Kr88	1.71E-02
Kr89	3.71E-04
Kr90	6.55E-05
Kr91	1.24E-05
Kr92	1.52E-06
Kr93	3.89E-07
Kr94	3.43E-08
Kr95	1.29E-08
Kr96	1.78E-09
Kr97	9.12E-12
Kr98	5.01E-12
Rb86	9.66E-03
Rb86m	3.02E-08
Rb87	8.95E+01
Rb88	1.84E-03
Rb89	1.99E-03
Rb90	2.96E-04
Rb90m	1.92E-04
Rb91	1.70E-04
Rb92	1.24E-05
Rb93	1.57E-05
Rb94	4.08E-06
Rb95	2.79E-07
Rb96	4.05E-08
Rb97	6.38E-09
Rb98	1.07E-09
Rb99	1.74E-11
Rb100	7.39E-12
Rb101	6.47E-14
Sr87m	3.58E-07
Sr89	9.67E+00
Sr90	1.99E+02
Sr91	1.12E-01
Sr92	3.81E-02
Sr93	2.20E-03
Sr94	3.91E-04
Sr95	1.23E-04
Sr96	1.44E-05
Sr97	3.61E-07
Sr98	7.13E-07

Isotope	Discharge g/assembly
Sr99	1.33E-07
Sr100	5.81E-08
Sr101	1.83E-09
Sr102	3.86E-10
Sr103	1.65E-12
Sr104	7.40E-11
Y89m	4.93E-08
Y90	5.15E-02
Y90m	1.46E-07
Y91	1.67E+01
Y91m	5.66E-03
Y92	5.00E-02
Y93	1.21E-01
Y94	6.49E-03
Y95	3.92E-03
Y96	8.66E-04
Y97	5.87E-06
Y98	1.20E-06
Y99	2.02E-06
Y100	7.85E-07
Y101	3.67E-07
Y102	2.64E-08
Y103	8.59E-09
Y104	3.63E-10
Y105	5.33E-12
Y106	5.27E-14
Y107	5.39E-16
Zr90m	2.03E-10
Zr93	2.40E+02
Zr95	3.54E+01
Zr97	4.22E-01
Zr98	2.37E-04
Zr99	1.87E-05
Zr100	5.26E-05
Zr101	1.57E-05
Zr102	8.90E-05
Zr103	2.41E-06
Zr104	1.81E-06
Zr105	4.20E-08
Zr106	7.89E-09
Zr107	2.00E-11
Zr108	1.30E-12
Zr109	2.67E-16
Nb93m	1.47E-04
Nb94	5.82E-04

Isotope	Discharge g/assembly
Nb94m	1.14E-09
Nb95	1.95E+01
Nb95m	2.31E-02
Nb96	1.55E-03
Nb97	3.03E-02
Nb97m	3.95E-04
Nb98	2.17E-05
Nb98m	2.42E-04
Nb99	7.23E-05
Nb99m	5.49E-04
Nb100	2.00E-05
Nb100m	2.20E-06
Nb101	5.59E-05
Nb102	2.15E-05
Nb103	9.27E-05
Nb104	3.21E-06
Nb105	2.45E-06
Nb106	1.80E-07
Nb107	4.37E-08
Nb108	4.74E-10
Nb109	2.30E-11
Nb110	3.64E-13
Nb111	2.29E-15
Nb112	3.59E-17
Mo99	2.06E+00
Mo101	7.46E-03
Mo102	5.80E-03
Mo103	5.65E-04
Mo104	8.04E-04
Mo105	3.57E-04
Mo106	3.76E-05
Mo107	1.19E-05
Mo108	4.74E-07
Mo109	3.44E-08
Mo110	6.46E-09
Mo111	1.01E-10
Mo112	1.75E-11
Mo113	7.80E-14
Mo114	6.01E-14
Mo115	8.95E-17
Tc98	7.58E-03
Tc99	5.28E+02
Tc99m	1.67E-01
Tc100	4.33E-05
Tc101	7.24E-03

Table J-1 (Continued).

Isotope	Discharge g/assembly
Tc103	4.40E-04
Tc102m	2.67E-06
Tc103	4.80E-04
Tc104	9.60E-03
Tc105	3.79E-03
Tc106	2.33E-04
Tc107	1.33E-04
Tc108	9.16E-06
Tc109	3.01E-05
Tc110	8.42E-08
Tc111	2.56E-08
Tc112	9.63E-10
Tc113	2.38E-10
Tc114	1.13E-11
Tc115	4.20E-13
Tc116	7.96E-14
Tc117	1.23E-15
Tc118	2.08E-17
Ru103	3.27E+01
Ru105	1.29E-01
Ru106	1.89E+02
Ru107	1.37E-03
Ru108	9.83E-04
Ru109	8.39E-05
Ru110	1.32E-05
Ru111	4.34E-06
Ru112	6.09E-08
Ru113	7.18E-08
Ru114	3.94E-08
Ru115	1.14E-09
Ru116	6.08E-10
Ru117	2.13E-11
Ru118	6.83E-12
Ru119	2.35E-13
Ru120	5.21E-14
Rh102	2.05E-03
Rh103m	3.26E-02
Rh104	2.72E-04
Rh104m	1.22E-04
Rh105	9.64E-01
Rh105m	1.03E-04
Rh106	1.87E-04
Rh106m	1.18E-03
Rh107	7.08E-03
Rh108	6.19E-05

Isotope	Discharge g/assembly
Rh108M	1.75E-05
Rh109	2.47E-04
Rh109m	6.86E-05
Rh110	3.45E-06
Rh110m	2.84E-06
Rh111	3.24E-05
Rh112	9.24E-07
Rh113	8.32E-08
Rh114	7.11E-08
Rh115	9.64E-08
Rh116	5.53E-09
Rh117	2.91E-09
Rh118	1.95E-10
Rh119	7.80E-11
Rh120	7.91E-12
Rh121	2.87E-12
Rh122	1.51E-13
Rh123	1.63E-14
Pd107	3.46E+02
Pd107m	1.77E-06
Pd109	1.85E-01
Pd109m	8.02E-06
Pd111	7.29E-04
Pd111m	4.35E-04
Pd112	1.72E-02
Pd113	1.20E-05
Pd114	1.36E-05
Pd115	2.07E-06
Pd116	7.29E-07
Pd117	1.87E-07
Pd118	5.74E-08
Pd119	1.13E-08
Pd120	2.04E-08
Pd121	1.21E-09
Pd122	8.01E-10
Pd123	3.48E-11
Pd124	3.28E-11
Pd125	2.44E-12
Pd126	9.46E-13
Ag106	1.97E-14
Ag108	2.19E-09
Ag108m	5.65E-04
Ag109m	1.52E-04
Ag110	6.72E-05
Ag110m	1.60E+00

Isotope	Discharge g/assembly
Ag111	3.62E-01
Ag111m	3.64E-05
Ag112	2.69E-03
Ag113	2.47E-03
Ag113m	1.71E-06
Ag114	4.43E-07
Ag115	5.19E-05
Ag115m	3.09E-07
Ag116	9.72E-06
Ag116m	8.55E-08
Ag117	2.54E-06
Ag117m	1.84E-07
Ag118	1.14E-07
Ag118m	6.12E-08
Ag119	1.80E-07
Ag120	2.71E-08
Ag121	4.86E-08
Ag122	8.31E-10
Ag123	2.94E-09
Ag124	6.49E-10
Ag125	3.56E-10
Ag126	7.97E-11
Ag127	6.17E-11
Ag128	1.31E-11
Cd109	1.01E-06
Cd111m	1.54E-05
Cd113m	1.94E-01
Cd115	1.25E-02
Cd115m	1.27E-02
Cd117	5.72E-04
Cd117m	1.73E-04
Cd118	1.64E-04
Cd119	2.04E-05
Cd119m	3.71E-06
Cd120	2.70E-06
Cd121	6.72E-07
Cd122	3.08E-07
Cd123	3.12E-07
Cd124	9.02E-07
Cd125	4.82E-08
Cd126	1.51E-07
Cd127	2.16E-08
Cd128	3.29E-08
Cd129	4.40E-09
Cd130	2.41E-09

Table J-1 (Continued)

Isotope	Discharge g/assembly
Cd131	9.06E-11
Cd132	1.32E-11
In113m	1.45E-10
In114	3.22E-09
In114m	1.39E-04
In115m	1.01E-03
In116	1.94E-07
In116m	1.69E-04
In117	1.21E-04
In117m	3.90E-04
In118	2.73E-07
In118m	2.08E-08
In119	3.80E-06
In119m	3.56E-05
In120	2.41E-06
In120m	3.21E-09
In121	1.53E-07
In121m	1.06E-05
In122	6.32E-07
In122m	1.07E-08
In123	2.92E-07
In123m	6.39E-07
In124	3.35E-07
In125	1.38E-07
In125m	5.67E-07
In126	2.32E-07
In127	2.72E-07
In127m	4.96E-07
In128	7.18E-07
In129	1.83E-07
In130	7.09E-08
In131	2.03E-08
In132	2.11E-09
In133	7.63E-11
In134	8.76E-12
Sn119m	2.37E-02
Sn121	5.60E-03
Sn121m	5.26E-02
Sn123	4.70E-02
Sn123m	1.49E-04
Sn125	3.16E-02
Sn125m	6.52E-05
Sn126	2.49E+01
Sn127	2.09E-03
Sn127m	9.10E-05
Sn128	3.38E-03
Sn129	4.00E-04
Sn129m	1.07E-04
Sn130	4.45E-04
Sn131	1.14E-04
Sn132	6.04E-05
Sn133	5.96E-07
Sn134	6.38E-08
Sn135	1.91E-09
Sn136	2.36E-10
Sb122	3.00E-03
Sb122m	3.24E-07
Sb124	5.29E-02
Sb124m	1.78E-08
Sb125	8.06E+00
Sb126	4.71E-06
Sb126m	1.57E-03
Sb127	2.23E-02
Sb128	3.62E-03
Sb128m	6.39E-04
Sb129	5.37E-03
Sb130	1.79E-03
Sb130m	1.02E-03
Sb131	6.56E-03
Sb132	5.31E-04
Sb132m	6.69E-04
Sb133	5.38E-04
Sb134	7.08E-06
Sb134m	6.15E-06
Sb135	6.23E-07
Sb136	1.39E-08
Sb137	3.55E-09
Sb138	1.16E-10
Sb139	1.12E-11
Te123m	1.64E-03
Te125m	1.04E-01
Te127	2.29E-02
Te127m	1.15E+00
Te129	8.84E-03
Te129m	1.27E+00
Te131	7.95E-03
Te131m	1.59E-01
Te132	2.58E+00
Te133	5.00E-03
Te133m	1.82E-02
Te134	2.56E-02
Te135	1.21E-04
Te136	5.55E-05
Te137	3.30E-06
Te138	4.27E-07
Te139	1.72E-08
Te140	4.10E-09
Te141	1.72E-11
Te142	1.69E-11
I128	2.30E-04
I129	1.66E+02
I130	8.87E-03
I130m	5.73E-05
I131	4.49E+00
I132	7.79E-02
I133	9.63E-01
I133m	9.86E-06
I134	4.39E-02
I134m	3.48E-04
I135	2.99E-01
I136	4.46E-04
I136m	1.26E-04
I137	1.50E-04
I138	2.05E-05
I139	2.86E-06
I140	3.46E-07
I141	2.34E-08
I142	4.17E-09
I143	6.24E-11
I144	1.04E-12
I145	4.54E-14
Xe127	5.03E-06
Xe129m	5.94E-04
Xe131m	9.86E-02
Xe133	5.67E+00
Xe133m	7.96E-02
Xe134m	1.29E-07
Xe135	1.62E-01
Xe135m	2.72E-03
Xe137	2.79E-03
Xe138	9.04E-03
Xe139	2.90E-04
Xe140	6.62E-05
Xe141	3.44E-06
Xe142	1.03E-06

Table J-1 (Continued).

Isotope	Discharge g/assembly
Xe143	4.26E-08
Xe144	3.21E-08
Xe145	3.23E-09
Xe146	2.85E-10
Xe147	2.52E-13
Cs132	2.30E-04
Cs134	1.01E+02
Cs134m	3.62E-03
Cs135	4.66E+02
Cs135m	1.13E-03
Cs136	8.04E-01
Cs137	8.91E+02
Cs138	2.28E-02
Cs138m	1.11E-04
Cs139	6.12E-03
Cs140	6.21E-04
Cs141	1.83E-04
Cs142	7.15E-06
Cs143	3.32E-06
Cs144	6.79E-07
Cs145	1.04E-07
Cs146	6.15E-09
Cs147	5.45E-10
Cs148	3.65E-11
Cs149	4.08E-13
Cs150	2.37E-14
Ba133	1.69E-07
Ba135m	3.80E-04
Ba136m	2.53E-08
Ba137m	1.37E-04
Ba139	5.57E-02
Ba140	1.27E+01
Ba141	1.12E-02
Ba142	6.10E-03
Ba143	1.06E-04
Ba144	6.18E-05
Ba145	1.68E-05
Ba146	2.79E-06
Ba147	6.59E-07
Ba148	3.68E-07
Ba149	6.58E-09
Ba150	1.18E-09
Ba151	2.82E-12
Ba152	6.48E-14
La140	1.74E+00

Isotope	Discharge g/assembly
La141	1.46E-01
La142	5.55E-02
La143	7.65E-03
La144	3.14E-04
La145	1.65E-04
La146	3.01E-05
La147	1.80E-05
La148	7.95E-07
La149	5.45E-07
La150	2.33E-08
La151	5.87E-09
La152	4.71E-11
La153	2.02E-11
La154	2.15E-13
La155	1.25E-15
Ce139	2.40E-05
Ce141	2.93E+01
Ce142	7.17E+02
Ce143	1.09E+00
Ce144	1.88E+02
Ce145	1.17E-03
Ce146	4.62E-03
Ce147	2.97E-04
Ce148	1.32E-04
Ce149	1.68E-06
Ce150	8.63E-07
Ce151	2.72E-07
Ce152	5.81E-07
Ce153	2.25E-08
Ce154	4.90E-09
Ce155	8.06E-11
Ce156	1.03E-11
Ce157	1.49E-13
Pr139	1.34E-09
Pr140	5.43E-08
Pr141	0.00E+00
Pr142	2.98E-02
Pr143	1.04E+01
Pr144	8.00E-03
Pr144m	4.65E-05
Pr145	1.40E-01
Pr146	7.95E-03
Pr147	3.25E-03
Pr148	5.05E-04
Pr149	3.72E-04

Isotope	Discharge g/assembly
Pr150	2.29E-05
Pr151	4.16E-06
Pr152	3.21E-06
Pr153	1.29E-06
Pr154	4.75E-08
Pr155	1.51E-08
Pr156	6.29E-10
Pr157	1.21E-10
Pr158	2.35E-12
Pr159	1.06E-13
Nd141	2.49E-09
Nd147	4.33E+00
Nd149	1.83E-02
Nd151	1.31E-03
Nd152	8.49E-04
Nd153	4.94E-05
Nd154	1.63E-05
Nd155	3.98E-06
Nd156	3.58E-06
Nd157	6.76E-08
Nd158	2.13E-08
Nd159	3.32E-10
Nd160	2.61E-11
Nd161	2.40E-13
Pm145	2.28E-05
Pm146	7.80E-03
Pm147	9.81E+01
Pm148	6.45E-01
Pm148m	8.43E-01
Pm149	8.84E-01
Pm150	3.61E-04
Pm151	1.82E-01
Pm152	3.14E-04
Pm152m	2.16E-05
Pm153	2.86E-04
Pm154	8.58E-05
Pm154m	1.10E-05
Pm155	1.24E-05
Pm156	2.46E-06
Pm157	6.34E-06
Pm158	1.03E-07
Pm159	2.81E-08
Pm160	7.42E-10
Pm161	9.77E-11
Pm162	1.45E-12

Table J-1 (Continued)

Isotope	Discharge g/assembly
Sm145	1.52E-06
Sm147	4.25E+01
Sm151	1.52E+01
Sm153	8.19E-01
Sm155	5.69E-04
Sm156	9.36E-03
Sm157	8.82E-05
Sm158	2.65E-04
Sm159	7.14E-06
Sm160	4.69E-06
Sm161	3.95E-08
Sm162	8.13E-09
Sm163	8.48E-11
Sm164	8.46E-12
Sm165	7.26E-14
Eu149	1.50E-10
Eu150	1.26E-06
Eu152	4.60E-02
Eu152m	6.83E-05
Eu154	3.65E+01
Eu155	6.78E+00
Eu156	3.70E+00
Eu157	1.73E-02
Eu158	3.06E-04
Eu159	6.34E-05
Eu160	1.20E-06
Eu161	3.91E-07
Eu162	6.08E-07
Eu163	6.88E-09
Eu164	1.22E-10
Eu165	1.56E-11
Gd153	7.86E-03
Gd155m	1.58E-12
Gd159	6.15E-03
Gd161	2.79E-06
Gd162	2.92E-06
Gd163	1.61E-07
Gd164	6.72E-07
Gd165	1.15E-08
TB160	1.58E-01
Tb161	1.02E-02
Tb162	2.21E-06
Tb162m	6.30E-07
Tb163	2.30E-06
Tb164	1.26E-07

Isotope	Discharge g/assembly
Tb165	7.72E-09
dy165	6.75E-05
Tl208	4.62E-13
Pb212	2.74E-10
Bi212	2.60E-11
Po212	1.37E-21
Po216	1.09E-15
Rn220	4.12E-13
Ra224	2.39E-09
Th228	4.60E-07
Th230	3.21E-05
Th231	4.31E-08
Th233	1.85E-11
Th234	6.13E-06
Pa231	1.27E-05
Pa232	3.49E-08
Pa233	2.74E-06
Pa234	5.80E-10
Pa234m	2.08E-10
U231	4.34E-11
U232	7.10E-05
U233	1.25E-04
U243	4.24E+00
U235	3.95E+02
U236	1.41E+02
U237	1.37E+00
U238	4.23E+05
U239	3.23E-01
Np235	4.52E-06
Np236	2.66E-04
Np236m	1.85E-06
Np237	8.49E+01
Np238	2.50E-01
Np239	4.66E+01
Np240m	2.79E-04
Pu236	2.97E-04
Pu237	3.72E-04
Pu238	8.77E+01
Pu239	7.35E+03
Pu240	4.25E+03
Pu241	2.62E+03
Pu242	9.14E+02
Pu243	2.40E-01
Am239	1.63E-08
Am240	7.16E-06

Isotope	Discharge g/assembly
Am241	1.28E+02
Am242	2.30E-01
Am242m	3.58E+00
Am243	2.47E+02
Am244	2.13E-01
Am245	3.12E-13
Cm241	4.53E-06
Cm242	3.48E+01
Cm243	1.38E+00
Cm244	1.05E+02
Cm245	6.29E+00
Cm246	8.56E-01
Cm247	2.23E-02
Cm248	3.36E-03
Cm249	6.65E-08
Bk249	8.12E-05
Bk250	5.97E-08
Bk251	1.29E-12
Cf249	1.21E-05
Cf250	1.04E-05
Cf251	8.83E-06
Cf252	5.51E-06
Cf253	1.12E-08
Es253	1.00E-08
Es254	2.92E-10
Total	443,525

Appendix K
Sodium Cooled Fast Reactor Material Balance

Table K-1 Metal Equilibrium LWR SNF CR=0.75

Mass Isotopes (kg)	Reactor Full Core Initial Charge	Reactor Annual Fresh Fuel	Reactor Annual SNF at Discharge	Reactor Annual SNF at 2yr Aged	Annual TRU Burned	Depleted U for FR Fuel
Primary Fuel Isotopes						
U234	8.26	2.08	2.00	2.26		
U235	6.00	1.51	0.92	0.94		0.62
U236	5.70	1.44	1.43	1.47		
U237	-	-	0.01	0.00		
U238	11,067.01	2,790.22	2,480.47	2,480.47		309.75
U239	-	-	0.00	-		
Np237	47.04	11.86	7.24	7.30	4.62	
Np239	-	-	0.51	0.00	(0.51)	
Pu238	72.92	18.38	16.68	17.22	1.71	
Pu239	1,455.99	367.09	332.53	332.99	34.56	
Pu240	865.63	218.24	204.75	205.47	13.49	
Pu241	151.16	38.11	31.51	28.66	6.60	
Pu242	193.00	48.66	43.78	43.78	4.88	
Pu243	-	-	0.00	0.00	(0.00)	
Am241	75.54	19.05	14.70	17.50	4.34	
Am242m	4.67	1.18	1.04	1.03	0.14	
Am243	61.03	15.39	14.30	14.30	1.09	
Cm242	0.50	0.13	0.84	0.04	(0.71)	
Cm243	0.27	0.07	0.07	0.07	(0.00)	
Cm244	38.32	9.66	10.00	9.26	(0.33)	
Cm245	9.77	2.46	2.56	2.56	(0.09)	
Total TRU	*2,976*	*750*	*680*	*680*	*69.78*	-
Total HM	*14,073*	*3,548.1*	*3,165.3*	*3,165.3*		*310.4*
Zr	*1,567*	*251.0*	*251.0*	*251.0*		
% Enrichment	*21.1*	*21.1*	*21.5*	*21.5*		
Fission Products						
I, C, H, Kr, Xe	-	-	43.84	43.84		
Cs, Sr, Ba, Rb	-	-	60.61	59.61		
Ag, Pd, Ru, Rh	-	-	66.26	66.26		
Total Lanthanides	-	-	94	93		
Y	-	-	2.64	2.57		
Tc	-	-	8.45	8.46		
Mo	-	-	31.63	32.17		
others	-	-	47.70	49.04		
Total Fission Products	-	-	*354.99*	*354.99*		
Total Fuel	**14,073**	**3,548**	**3,520**	**3,520**		
Hardware						
Assemblies per yr	144	23.08	23.08	23.08		
Mass HT-9 per year	51,825	8,300	8,300	8,300		

LWR = 50GW d/MT burn-up 5 year cooled
Hardware mass per values in "ABR-1000 Inventory" = 359.9 kg HT-9 per assembly

Table K-2 Oxide Equilibrium Core LWR SNF CR=0.75

Mass Isotopes (kg)	Reactor Full Core Initial Charge	Reactor Annual Fresh Fuel	Reactor Annual SNF at Discharge	Reactor Annual SNF at 2yr Aged	Annual TRU Burned	Depleted U for FR Fuel
Primary Fuel Isotopes						
U234	13.12	2.18	2.12	2.36		
U235	8.48	1.41	0.83	0.84		0.61
U236	8.33	1.38	1.38	1.42		
U237	-	-	0.01	0.00		
U238	12,134.83	2,016.32	1,711.95	1,711.95		304.38
U239	-	-	0.00	-		
Np237	54.08	8.99	4.59	4.65	4.40	
Np239	-	-	0.33	0.00	(0.33)	
Pu238	105.32	17.50	15.69	16.15	1.81	
Pu239	1,804.73	299.87	259.95	260.27	39.93	
Pu240	1,266.21	210.39	194.43	195.19	15.96	
Pu241	235.20	39.08	31.86	28.98	7.22	
Pu242	286.16	47.55	42.46	42.47	5.08	
Pu243	-	-	0.00	0.00	(0.00)	
Am241	112.77	18.74	14.59	17.42	4.15	
Am242m	7.05	1.17	1.02	1.01	0.16	
Am243	90.05	14.96	13.80	13.80	1.16	
Cm242	0.69	0.11	0.75	0.04	(0.63)	
Cm243	0.51	0.08	0.09	0.08	(0.00)	
Cm244	63.20	10.50	10.60	9.82	(0.10)	
Cm245	18.47	3.07	3.10	3.10	(0.03)	
Total TRU	*4,044*	*672*	*593*	*593*	*78.78*	
Total HM	*16,225*	*2,696.0*	*2,309.5*	*2,309.5*		*305.0*
% Enrichment	*24.9*	*24.9*	*25.7*	*25.7*		
Fission Products						
I, C, H, Kr, Xe	-	-	43.69	43.68		
Cs, Sr, Ba, Rb	-	-	59.96	59.04		
Ag, Pd, Ru, Rh	-	-	66.28	66.28		
Total	-	-	93	92		
Y	-	-	2.59	2.55		
Tc	-	-	8.38	8.40		
Mo	-	-	31.73	32.07		
others	-	-	48.43	49.71		
Total Fission	-	-	*354.14*	*354.14*		
Total Fuel	**16,225**	**2,696**	**2,664**	**2,664**		
Hardware						
Assemblies per	144	23.14	23.14	23.14		
Mass HT-9	46,890	7,535	7,535	7,535		

LWR = 50GW d/MT burn-up 5 year cooled
Hardware mass per values in "ABR-1000 Inventory" = 325.6 kg HT-9 per assembly

Table K-3 Oxide Equilibrium Core LWR SNF CR=0.5

Mass Isotopes (kg)	Reactor Full Core Initial Charge	Reactor Annual Fresh Fuel	Reactor Annual SNF at Discharge	Reactor Annual SNF at 2yr Aged	Annual TRU Burned	Depleted U for FR Fuel
Primary Fuel Isotopes						
U234	16.92	3.04	2.94	3.31		
U235	7.35	1.32	0.94	0.95		0.42
U236	8.03	1.44	1.43	1.47		
U237	-	-	0.00	0.00		
U238	7,328.02	1,318.39	1,110.27	1,110.27		208.12
U239	-	-	0.00	-		
Np237	87.77	15.79	6.76	6.83	9.04	
Np239	-	-	0.24	0.00	(0.24)	
Pu238	153.01	27.53	23.58	24.24	3.95	
Pu239	1,633.74	293.93	206.86	207.10	87.06	
Pu240	1,411.85	254.01	216.11	217.37	37.90	
Pu241	312.12	56.15	39.14	35.60	17.02	
Pu242	432.53	77.82	66.69	66.69	11.12	
Pu243	-	-	0.00	0.00	(0.00)	
Am241	146.73	26.40	18.61	22.10	7.78	
Am242m	8.72	1.57	1.35	1.34	0.22	
Am243	138.63	24.94	22.42	22.41	2.52	
Cm242	0.93	0.17	1.08	0.05	(0.91)	
Cm243	0.74	0.13	0.13	0.13	0.00	
Cm244	101.32	18.23	18.25	16.91	(0.02)	
Cm245	29.60	5.33	5.37	5.37	(0.05)	
Total TRU	*4,458*	*802*	*627*	*626*	*175.38*	
Total HM		*2,130.2*	*1,742.2*	*1,742.1*		*208.5*
% Enrichment		*37.6*	*36.0*	*35.9*		
Fission Products						
I, C, H, Kr, Xe	-	-	41.63	41.62		
Cs, Sr, Ba, Rb	-	-	57.11	56.23		
Ag, Pd, Ru, Rh	-	-	63.50	63.50		
Total Lanthanides	-	-	88	88		
Y	-	-	2.42	2.38		
Tc	-	-	7.95	7.96		
Mo	-	-	30.16	30.49		
others	-	-	45.97	47.22		
Total Fission	-	-	*337.00*	*337.02*		
Total Fuel		**2,130**	**2,079**	**2,079**		
Hardware						
Assemblies per yr	144	23.14	23.14	23.14		
Mass, HT-9	46,890	7,535	7,535	7,535		

LWR = 50GW d/MT burn-up 5 year cooled
Hardware mass per values in "ABR-1000 Inventory" = 325.6 kg HT-9 per assembly

Table K-4 Metal Equilibrium LWR SNF CR=0.5

Mass Isotopes (kg)	Reactor Full Core Initial Charge	Reactor Annual Fresh Fuel	Reactor Annual SNF at Discharge	Reactor Annual SNF at 2yr Aged	Annual TRU Burned	Depleted U for FR Fuel
Primary Fuel Isotopes						
U234		2.99	2.85	3.25		
U235	5.11	1.36	0.96	0.97		0.42
U236	5.50	1.46	1.45	1.50		
U237		-	0.01	0.00		
U238	6,694.23	1,781.57	1,570.23	1,570.23		211.35
U239	-	-	0.00	-		
Np237		19.06	10.22	10.30	8.85	
Np239	-	-	0.36	0.00		
Pu238		29.65	25.62	26.45	4.03	
Pu239	1,292.85	344.07	261.12	261.52	82.96	
Pu240	1,029.55	274.00	237.20	238.48	36.80	
Pu241		57.70	41.25	37.51	16.45	
Pu242	314.38	83.67	72.81	72.81	10.86	
Pu243	-	-	0.00	0.00		
Am241		28.14	20.32	23.98	7.83	
Am242m	6.33	1.68	1.47	1.45	0.22	
Am243		26.92	24.47	24.46	2.45	
Cm242	0.76	0.20	1.29	0.06		
Cm243		0.12	0.12	0.12	0.00	
Cm244	67.61	17.99	18.16	16.82		
Cm245	17.61	4.69	4.75	4.75		
Total TRU	*3,336*	*888*	*719*	*719*	*168.77*	-
Total HM	*10,065*	*2,678.7*	*2,294.6*	*2,294.7*		*211.8*
Zr	*1,372*	*218.9*	*218.9*	*218.9*		
% Enrichment	*33.1*	*33.1*	*31.3*	*31.3*		
Fission Products						
I, C, H, Kr, Xe	-	-	41.70	41.69		
Cs, Sr, Ba, Rb	-	-	57.60	56.64		
Ag, Pd, Ru, Rh	-	-	63.34	63.34		
Total	-	-	89	88		
Y	-	-	2.46	2.40		
Tc	-	-	7.99	8.01		
Mo	-	-	30.02	30.53		
others	-	-	45.23	46.51		
Total Fission	-	-	*337.19*	*337.19*		
Total Fuel	**10,065**	**2,679**	**2,632**	**2,632**		
HardWare						
Assemblies per yr	144	23.14	23.14	23.14		
Mass HT-9	51,825	8,330	8,330	8,330		

LWR = 50GW d/MT burn-up 5 year cooled
Hardware mass per values in "ABR-1000 Inventory" = 359.9 kg HT-9 per assembly

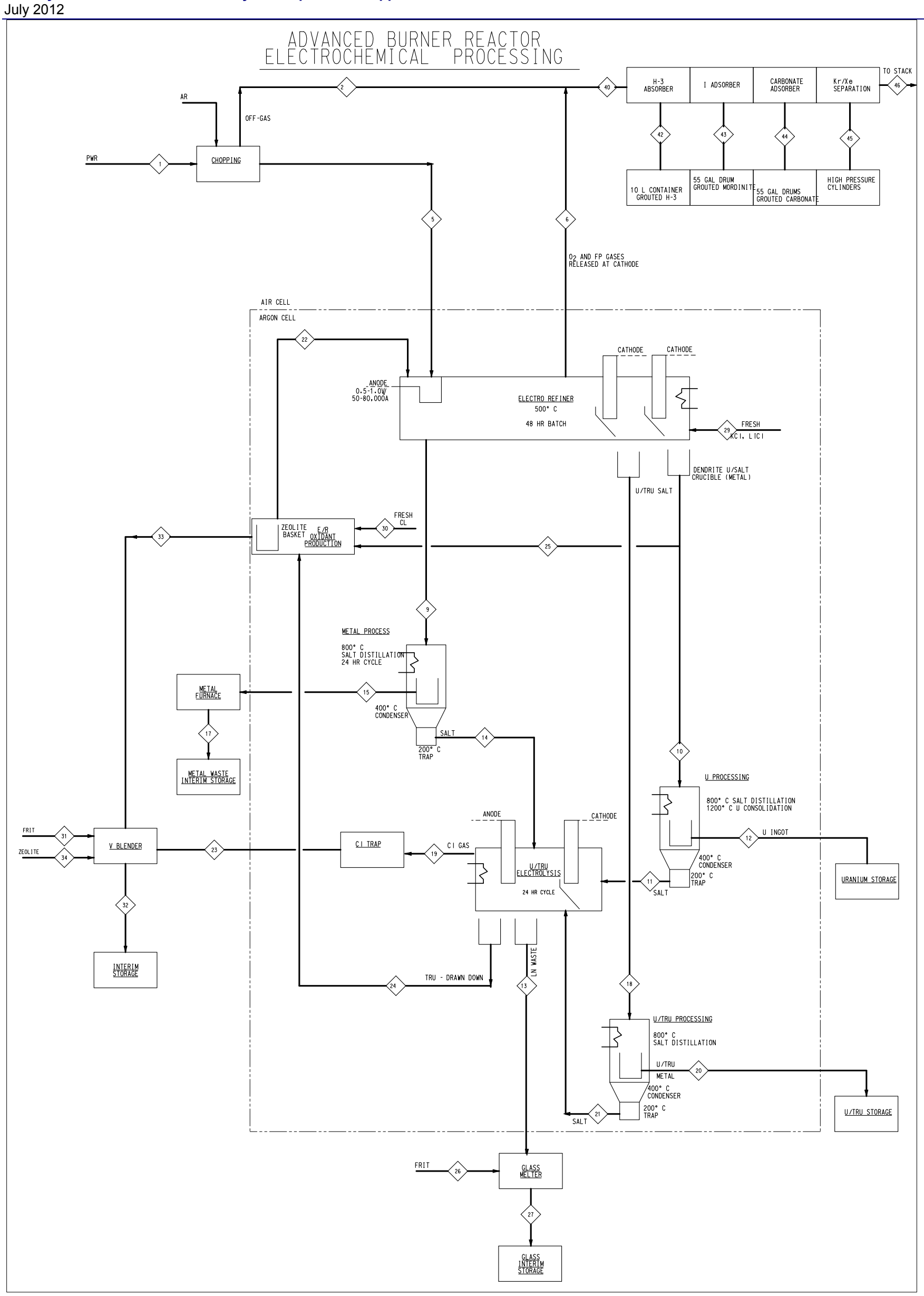

Figure K-1 Block Flow Diagram for the Electro-Chemical Reprocessing of Advanced Burner Reactor UNF

Appendix L
Decay Heat for Waste from Reprocessing LWR UOX Fuel

Table L-1 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 40 GWd/MT Burn-up 5 Year Cooled

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	10,230	5,921	3,642	2,279	1,426	706	7	0
Noble Metals Ag, Pd, Ru, Rh	1,229	1	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	1,339	253	49	10	2	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	1,196	909	583	427	349	295	201	148
Others	8	1	0	0	0	0	0	0
Total	**14,002**	**7,084**	**4,275**	**2,716**	**1,778**	**1,002**	**208**	**148**

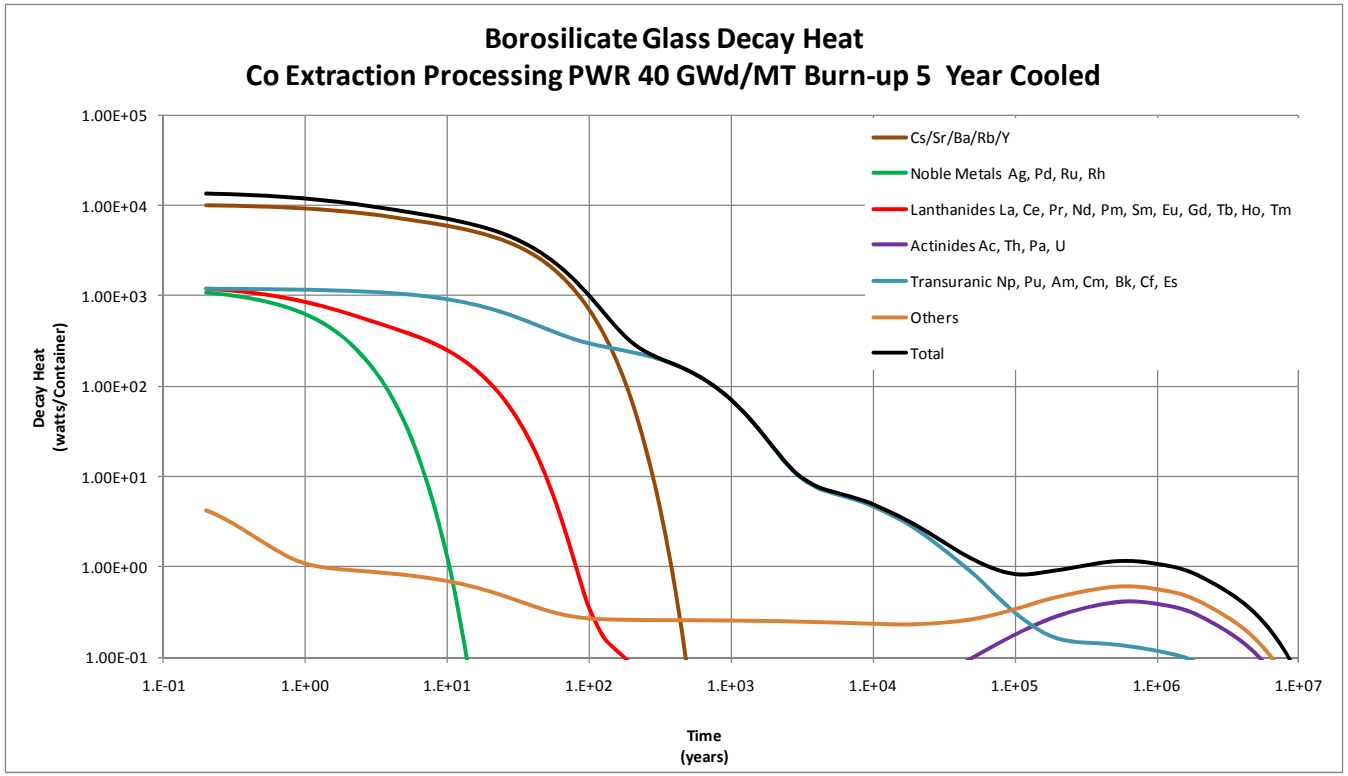

Figure L-1 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 40 GWd/MT Burn-up 5 Year Cooled

Table L-2 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 40 GWd/MT Burn-up 5 Year Cooled

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	9,721	4,355	2,659	1,673	1,053	525	5	0
Noble Metals Ag, Pd, Ru, Rh	1,982	2	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	2,165	409	79	16	3	1	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	1	1	1	1	1	1	1	1
Others	13	1	1	0	0	0	0	0
Total	**13,882**	**4,768**	**2,740**	**1,691**	**1,058**	**527**	**6**	**1**

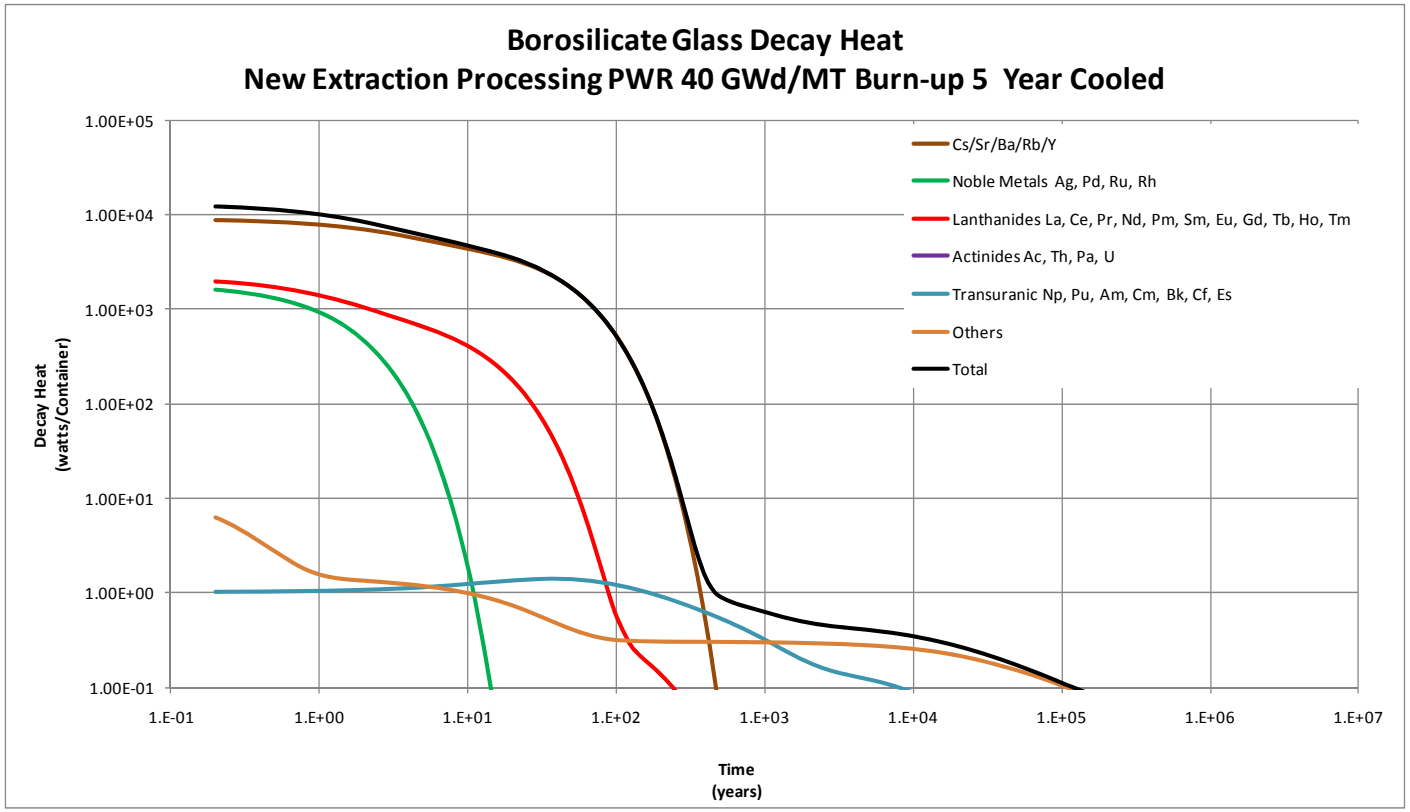

Figure L-2 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 40 GWd/MT Burn-up 5 Year Cooled

Table L-3 Borosilicate Glass Decay Heat Generated by UREX Processing of 40 GWd/MT 5 Year Cooled PWR Fuel

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	110	52	32	20	13	6	0	0
Noble Metals Ag, Pd, Ru, Rh	1,545	2	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	2,220	439	85	17	4	1	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	10	1	1	0	0	0	0	0
Total	**3,885**	**494**	**117**	**38**	**17**	**7**	**0**	**0**

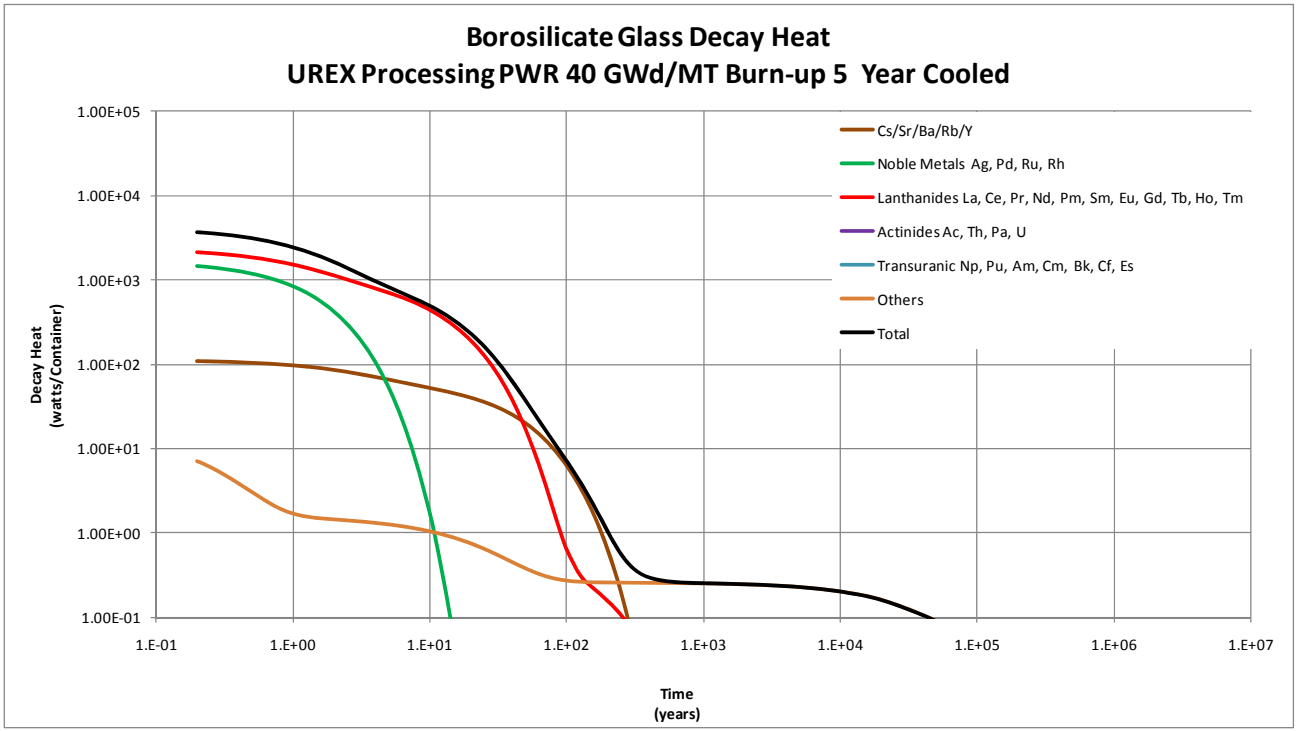

Figure L-3 Borosilicate Glass Decay Heat Generated by UREX Processing of 40 GWd/MT 5 Year Cooled PWR Fuel

Table L-4 Cs Sr Ceramic Decay heat Generated by UREX Processing of 40 GWd/MT 5 Year Cooled PWR Fuel

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	5,823	3,414	2,101	1,314	822	407	4	0
Noble Metals Ag, Pd, Ru, Rh	20	0	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	71	14	3	1	0	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	0	0	0	0	0	0	0	0
Total	**5,914**	**3,429**	**2,104**	**1,315**	**823**	**407**	**4**	**0**

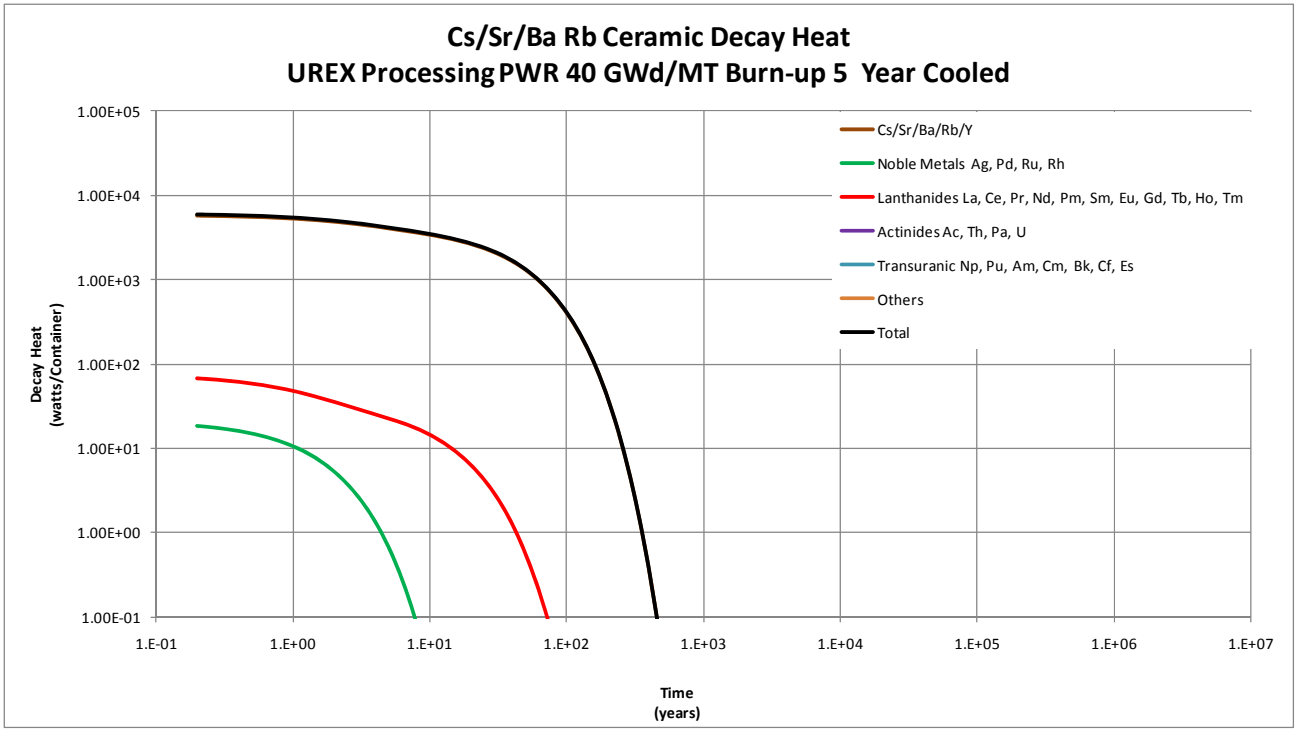

Figure L-4 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 40 GWd/MT 5 Year Cooled PWR Fuel

Table L-5 Glass Bonded Zeolite Decay Heat Generated by Electro-Chemical Processing of 40 GWd/MT 5 year Cooled PWR Fuel

Time (years) Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	9,822	5,752	3,539	2,214	1,385	686	6	0
Noble Metals Ag, Pd, Ru, Rh	-	-	-	-	-	-	-	-
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	-	-	-	-	-	-	-	-
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	-	-	-	-	-	-	-	-
Total	**9,822**	**5,752**	**3,539**	**2,214**	**1,385**	**686**	**6**	**0**

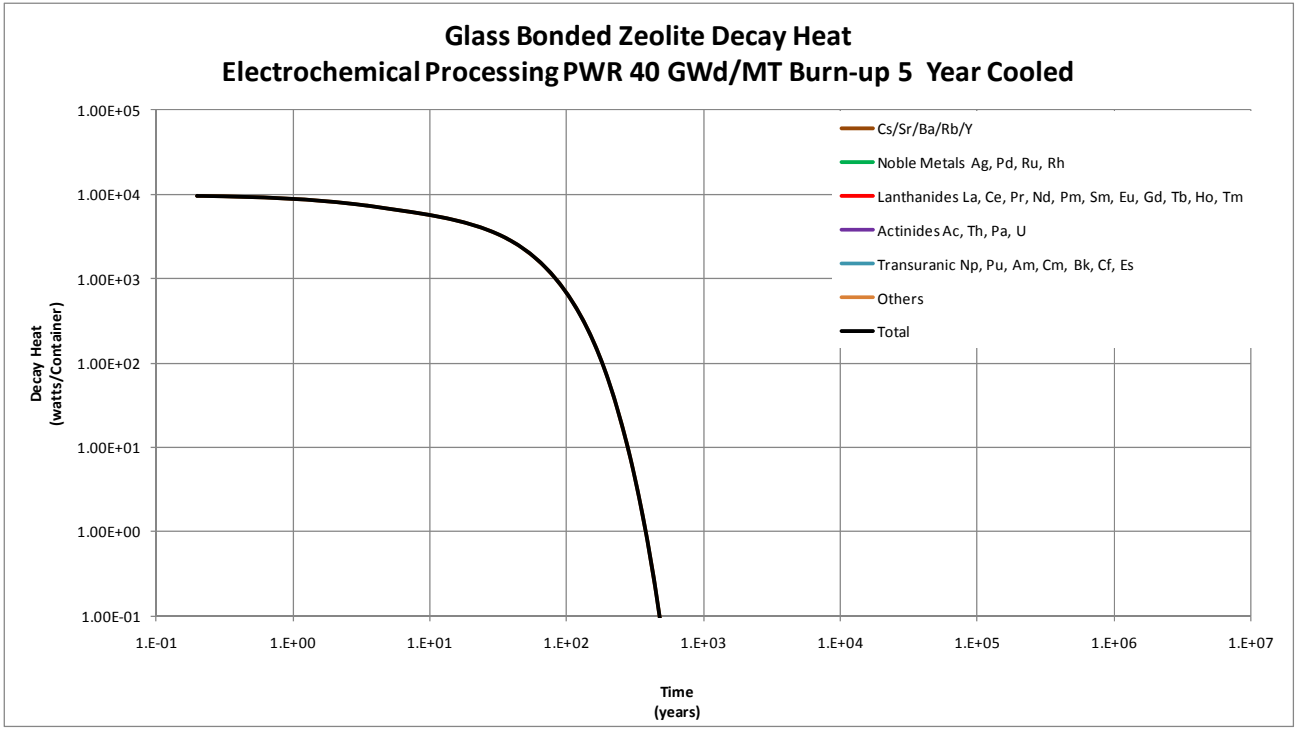

Figure L-5 Glass Bonded Zeolite Decay Heat Generated by Electro-Chemical Processing of 40 GWd/MT 5 year Cooled PWR Fuel

Table L-6 Lanthanide Glass Decay Heat Generated by Electro-Chemical Processing of 40 GWd/MT 5 Year Cooled PWR Fuel

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	202	163	100	63	39	19	0	0
Noble Metals Ag, Pd, Ru, Rh	-	-	-	-	-	-	-	-
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	3,262	642	126	25	5	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	-	-	-	-	-	-	-	-
Total	**3,464**	**805**	**226**	**88**	**44**	**20**	**0**	**0**

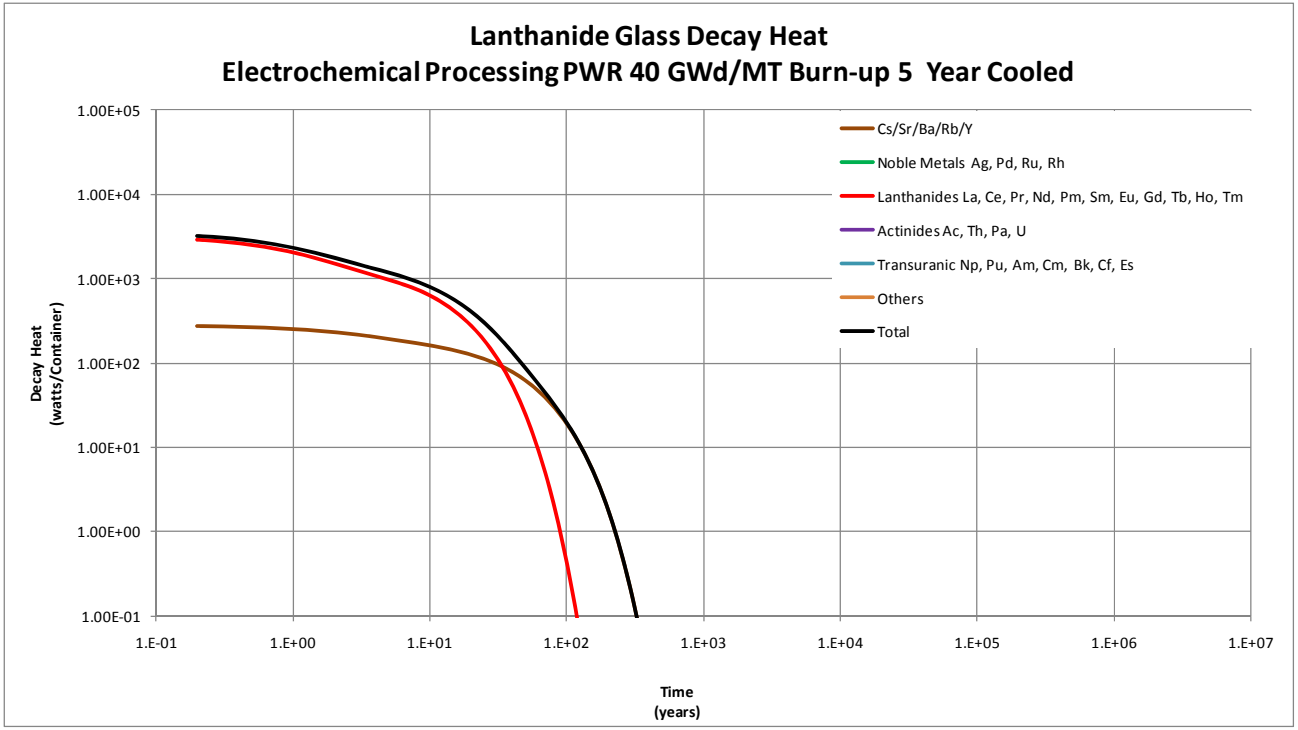

Figure L-6 Lanthanide Glass Decay Heat Generated by Electro-Chemical Processing of 40 GWd/MT 5 Year Cooled PWR Fuel

Table L-7 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 60 GWd/MT Burn-up 5 Year Cooled

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	9,987	5,391	3,302	2,066	1,293	640	6	0
Noble Metals Ag, Pd, Ru, Rh	956	1	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	1,130	279	54	11	2	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	1,943	1,390	760	463	321	232	143	106
Others	7	1	0	0	0	0	0	0
Total	**14,023**	**7,062**	**4,116**	**2,540**	**1,617**	**872**	**149**	**107**

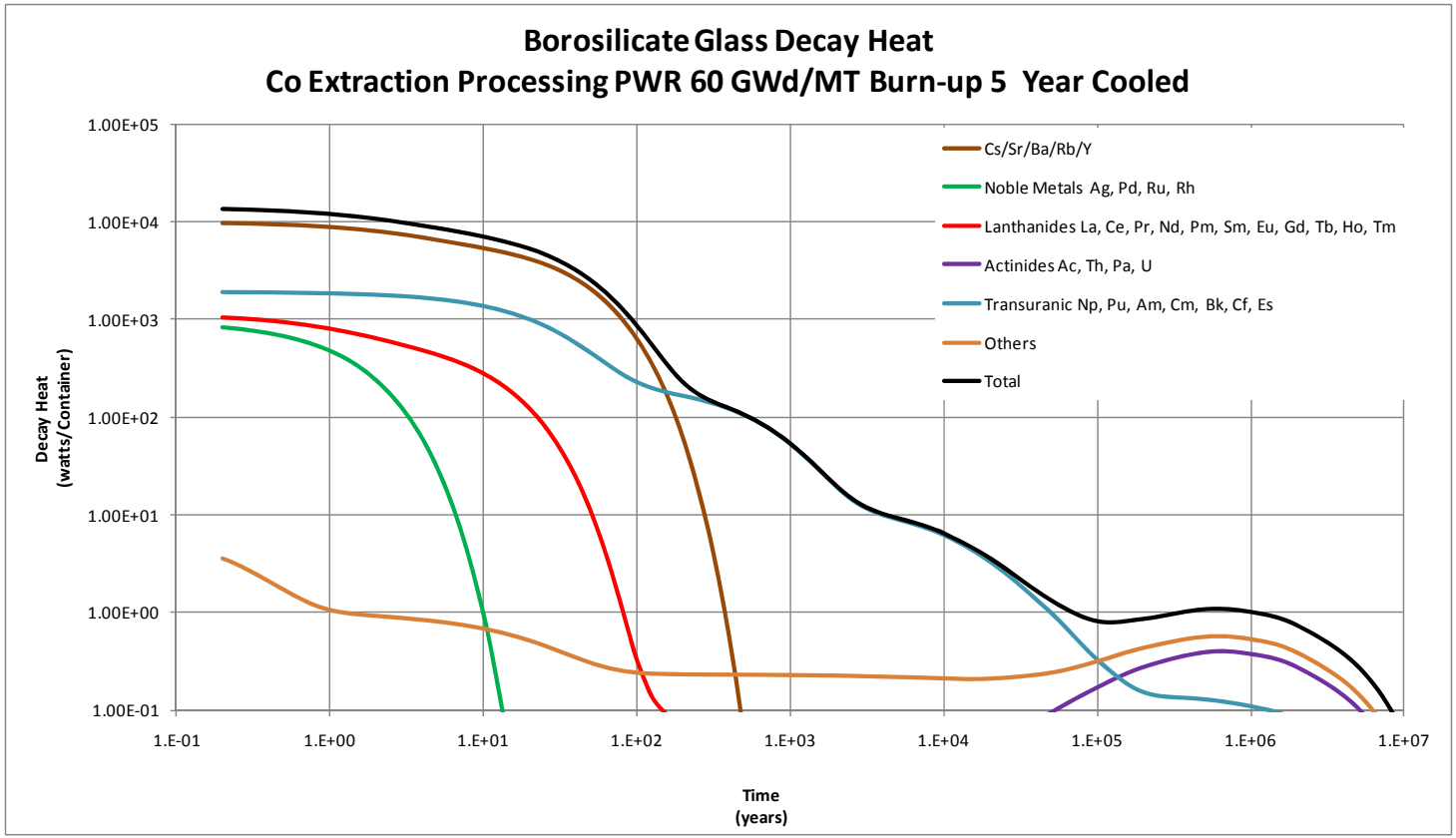

Figure L-7 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 60 GWd/MT Burn-up 5 Year Cooled

Table L-8 Borosilicate Glass Decay heat Generated by New Extraction Processing of 60 GWd/MT Burn-up 5 Year Cooled

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	10,776	4,455	2,699	1,698	1,068	533	5	0
Noble Metals Ag, Pd, Ru, Rh	1,713	2	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	2,030	502	98	20	4	1	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	2	2	2	2	1	1	1	0
Others	12	1	1	0	0	0	0	0
Total	**14,532**	**4,961**	**2,799**	**1,720**	**1,074**	**535**	**6**	**1**

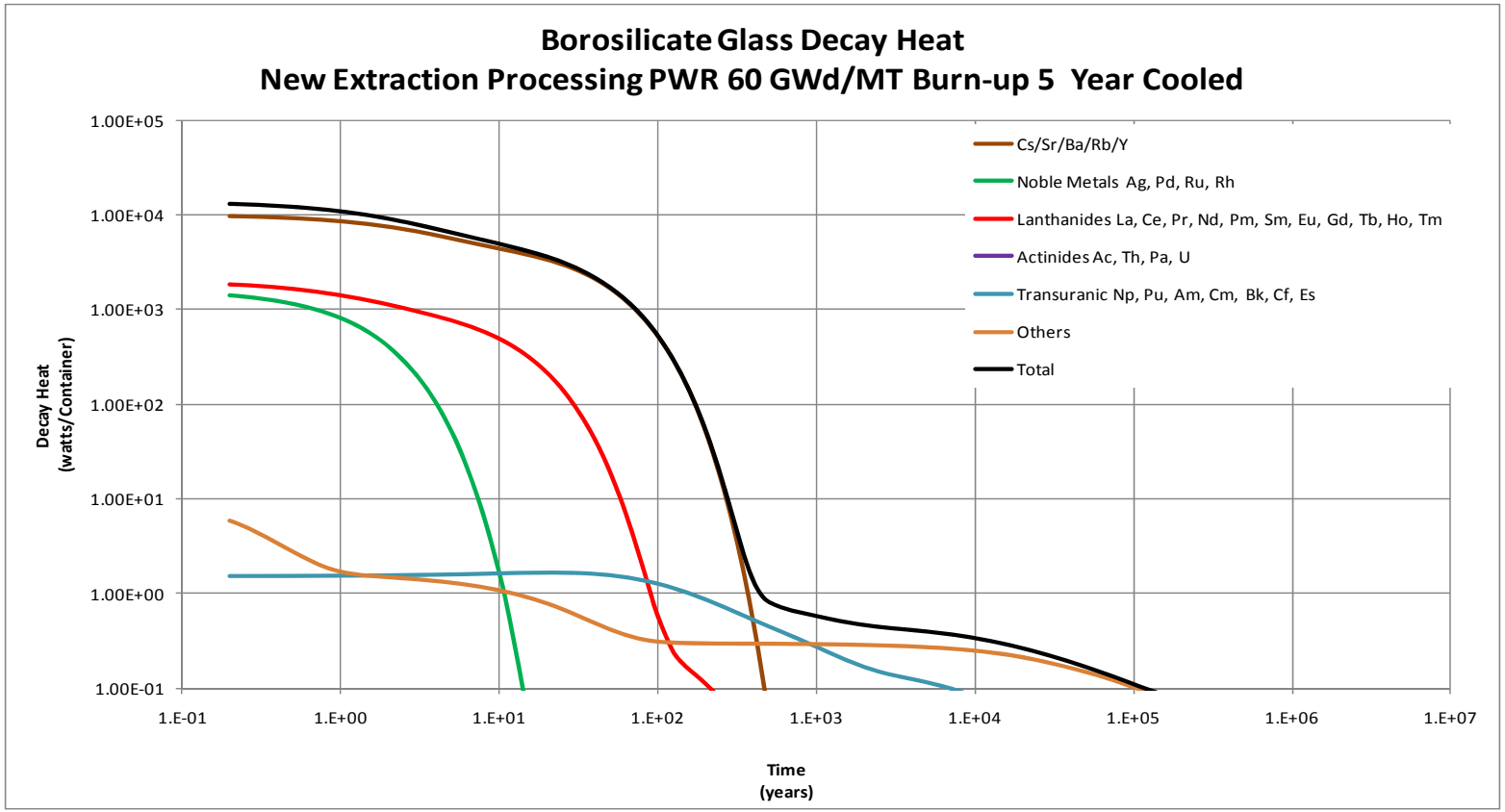

Figure L-8 Borosilicate Glass Decay heat Generated by New Extraction Processing of 60 GWd/MT Burn-up 5 Year Cooled

Table L-9 Borosilicate Glass Decay Heat Generated by UREX Processing of 60 GWd/MT 5 year Cooled PWR Fuel

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	125	53	32	20	13	6	0	0
Noble Metals Ag, Pd, Ru, Rh	1,336	1	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	2,098	539	105	21	4	1	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	9	1	1	0	0	0	0	0
Total	**3,569**	**595**	**138**	**42**	**18**	**7**	**0**	**0**

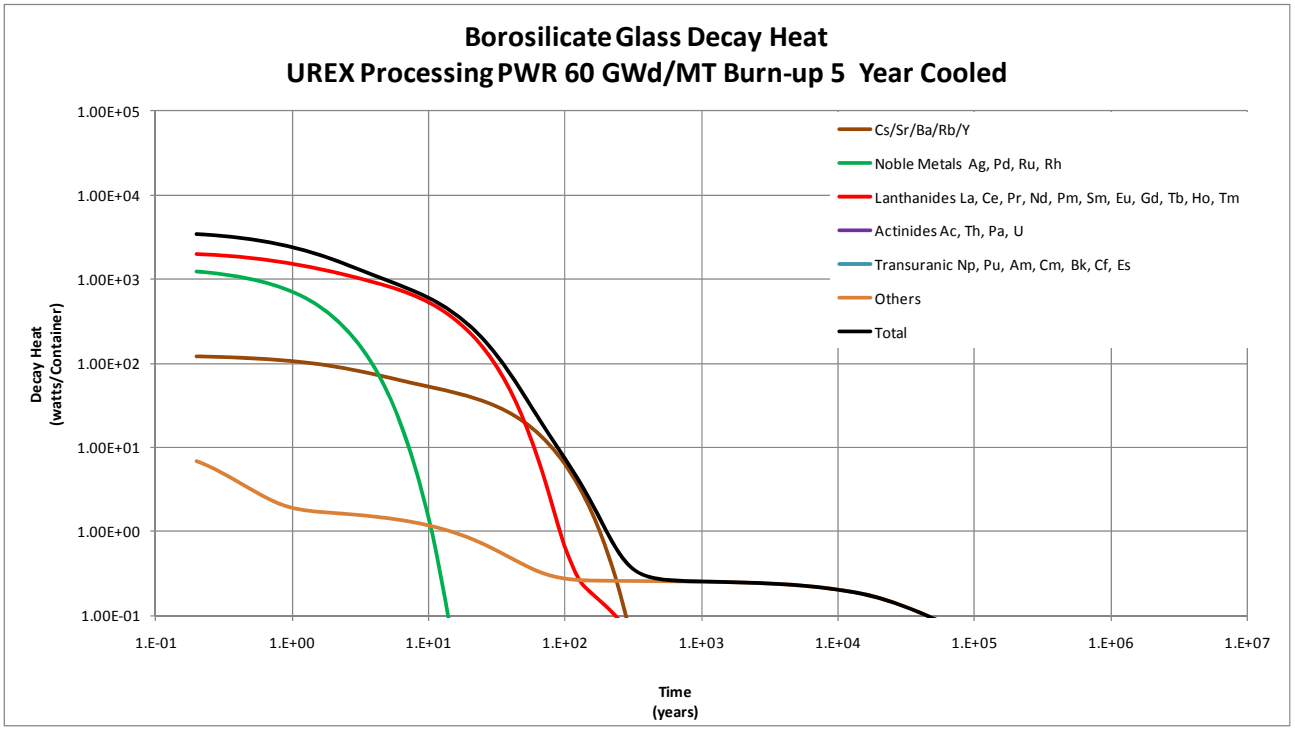

Figure L-9 Borosilicate Glass Decay Heat Generated by UREX Processing of 60 GWd/MT 5 Year Cooled PWR Fuel

Table L-10 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 60 GWd/MT 5 Year Cooled PWR Fuel

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	6,099	3,342	2,047	1,281	802	397	4	0
Noble Metals Ag, Pd, Ru, Rh	17	0	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	66	17	3	1	0	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	0	0	0	0	0	0	0	0
Total	**6,182**	**3,360**	**2,051**	**1,282**	**802**	**397**	**4**	**0**

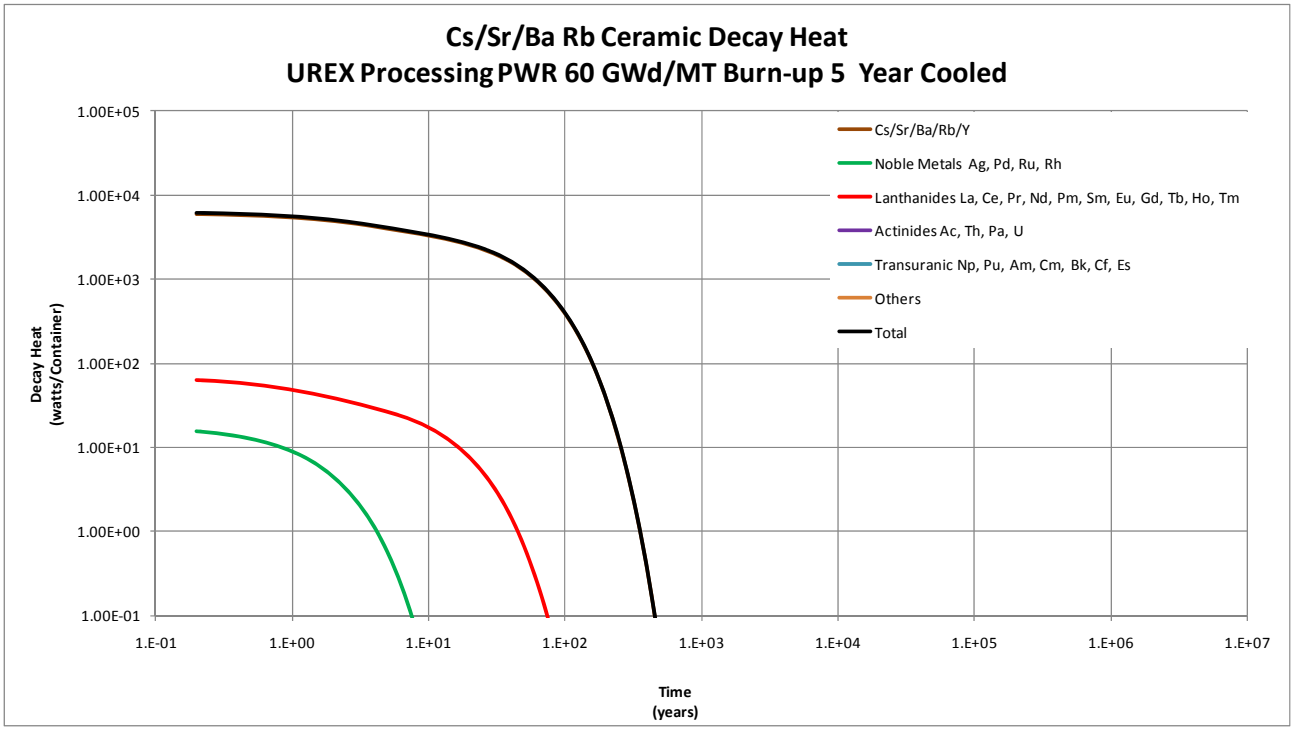

Figure L-10 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 60 GWd/MT 5 Year Cooled PWR Fuel

Table L-11 Glass Bonded Zeolite Decay Heat Generated by Electro-Chemical Processing of 60 GWd.MT 5 Year Cooled PRW Fuel

Time (years) Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	11,584	6,338	3,882	2,429	1,520	753	7	0
Noble Metals Ag, Pd, Ru, Rh	-	-	-	-	-	-	-	-
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	-	-	-	-	-	-	-	-
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	-	-	-	-	-	-	-	-
Total	**11,584**	**6,338**	**3,882**	**2,429**	**1,520**	**753**	**7**	**0**

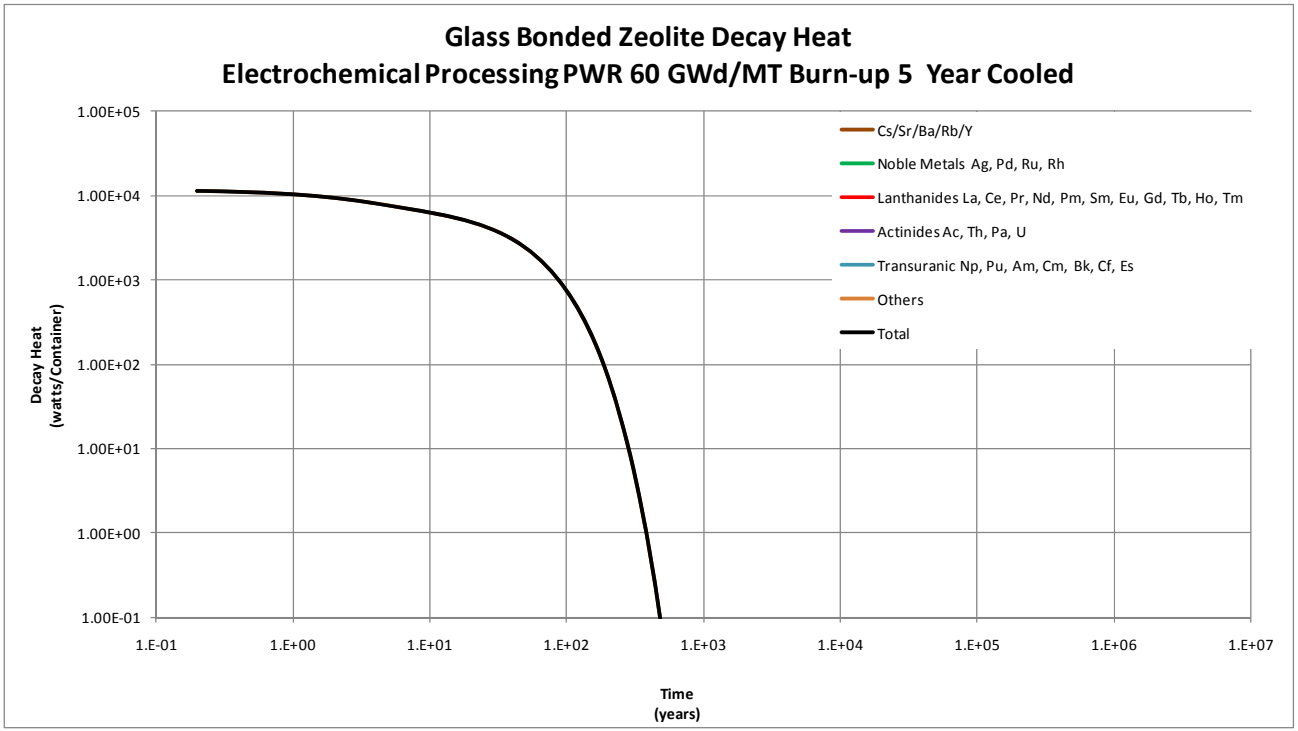

Figure L-11 Glass Bonded Zeolite Bonded Decay Het Generated by Electro-Chemical Processing of 60 GWd/MT 5 Year Cooled PWR Fuel

Table L-12 Lanthanide Glass Decay Heat Generated by Electro-Chemical Processing of 60 GWd/MT 5 yer Cooled PWR Fuel

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	355	192	118	74	46	23	0	0
Noble Metals Ag, Pd, Ru, Rh	-	-	-	-	-	-	-	-
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	2,903	773	152	30	6	1	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	-	-	-	-	-	-	-	-
Total	**3,257**	**965**	**269**	**104**	**52**	**23**	**0**	**0**

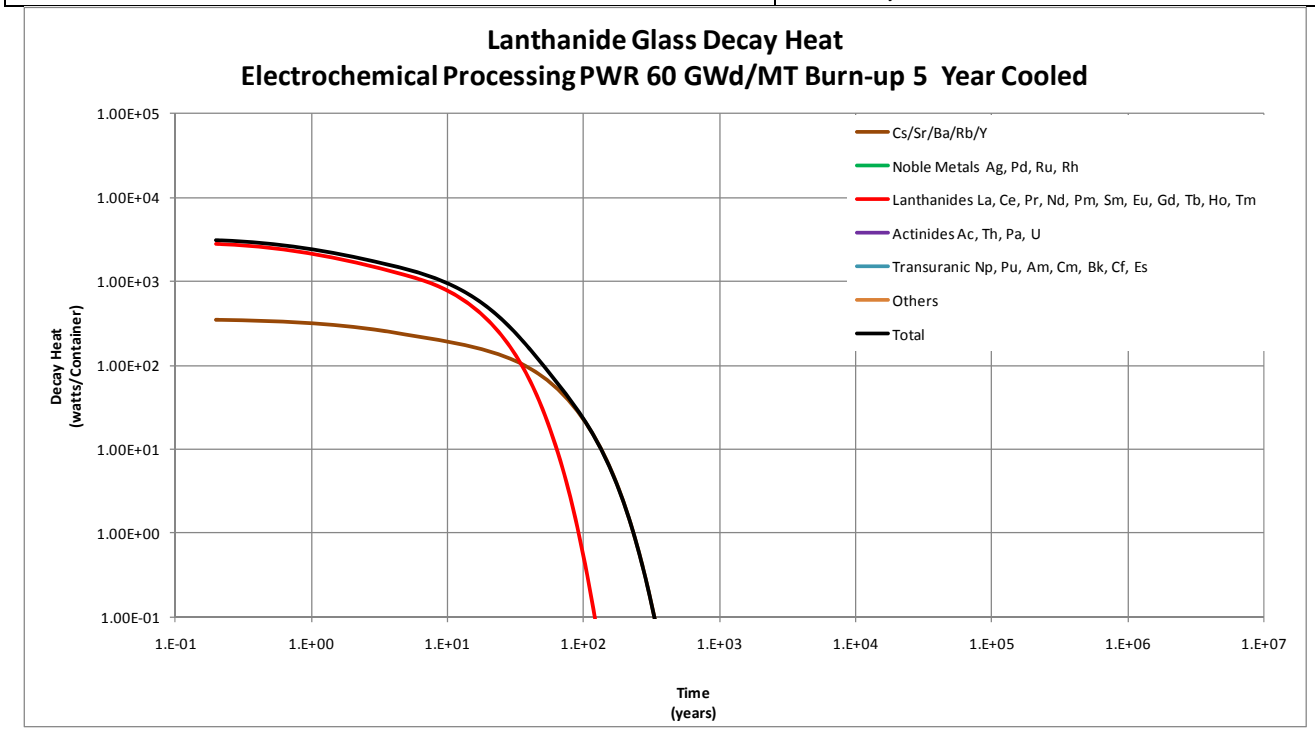

Figure L-12 Lanthanide Glass Decay Heat Generated by Electo-Chemical Processing of 60 GWd/MT 5 Year Cooled PWR Fuel

Table L-13 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 60 GWd/MT Burn-up 30 Year Cooled

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	6,310	4,984	3,119	1,952	1,221	605	6	0
Noble Metals Ag, Pd, Ru, Rh	0	0	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	138	62	13	3	1	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	2,214	1,841	1,400	1,178	1,058	960	682	498
Others	1	1	0	0	0	0	0	0
Total	**8,663**	**6,887**	**4,532**	**3,133**	**2,280**	**1,565**	**688**	**499**

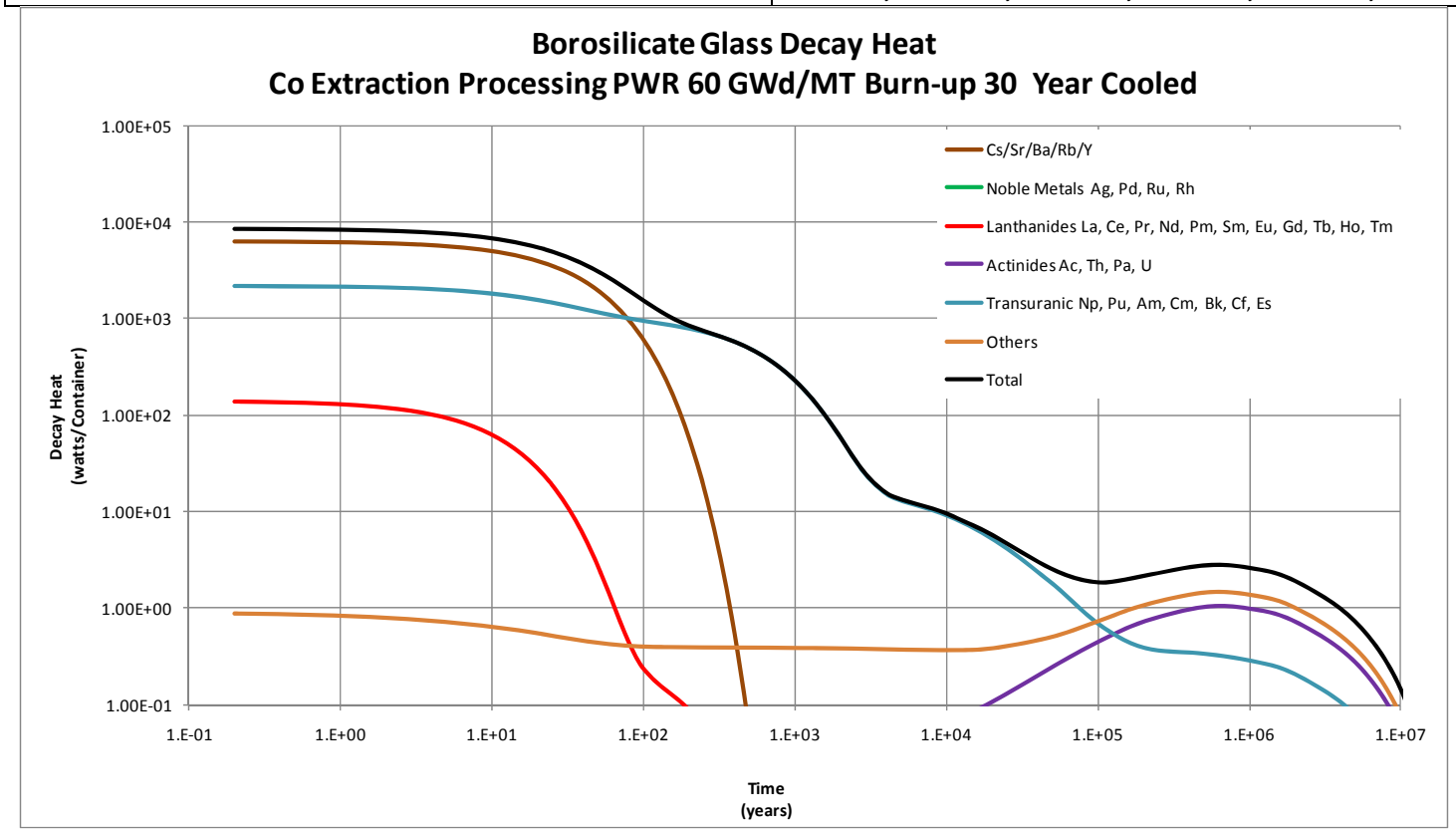

Figure L-13 Borosilicate Glass Decay Heat Generated by Co-Extraction Processing of 60 GWd/MT Burn-up 30 Year Cooled

Table L-14 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 60 GWd/MT Burn-up 30 Year Cooled

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	3,340	2,404	1,512	952	599	299	3	0
Noble Metals Ag, Pd, Ru, Rh	0	0	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	146	65	13	3	1	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	1	1	1	1	1	1	0	0
Others	1	1	0	0	0	0	0	0
Total	**3,489**	**2,471**	**1,527**	**956**	**601**	**300**	**4**	**1**

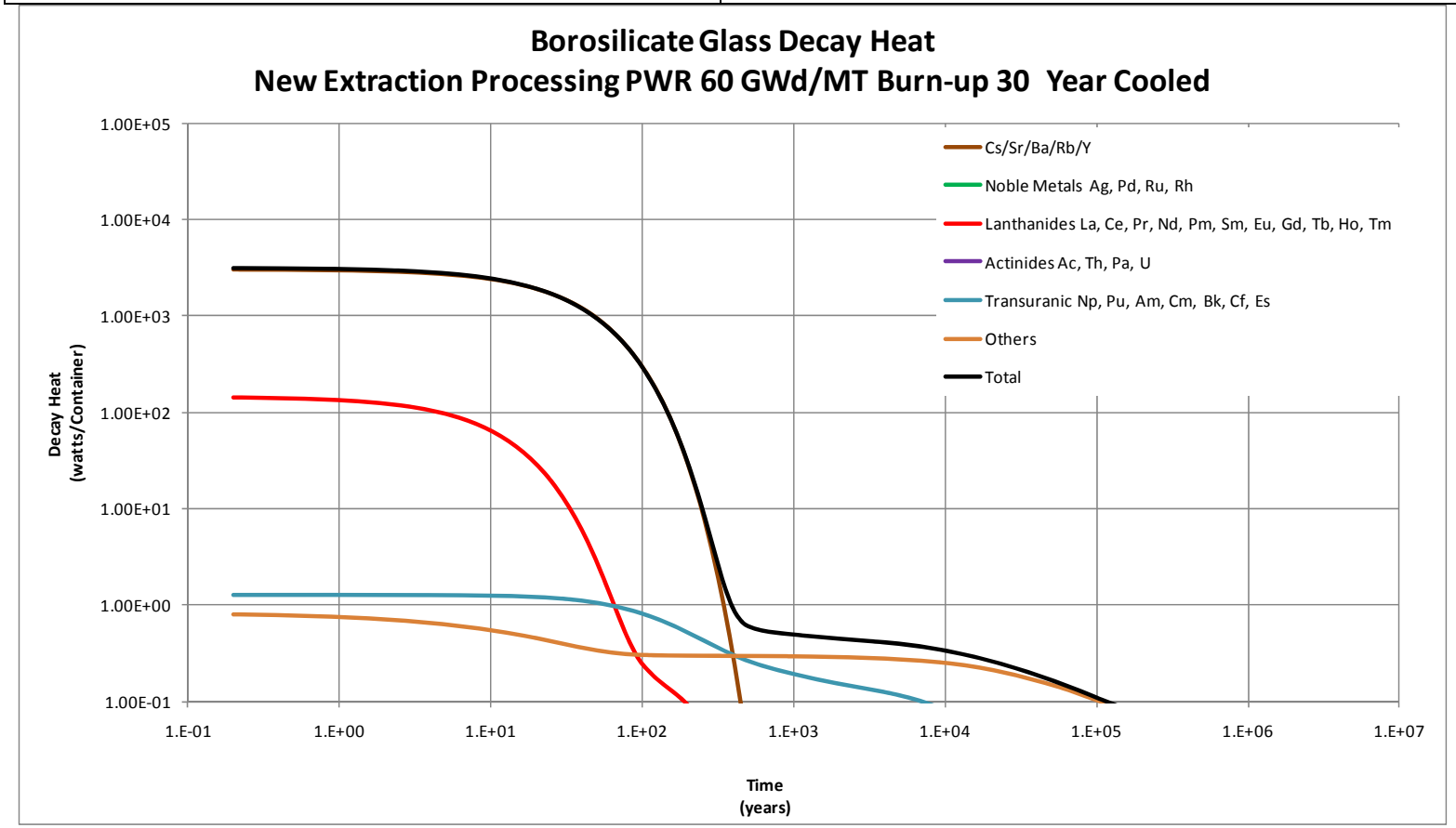

Figure L-14 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 60 GWd/MT Burn-up 30 Year Cooled

Table L-15 Borosilicate Glass Decay Heat Generated by UREX Processing of 60 GWd.MT 30 Yer Cooled PWR Fuel

Time (years) Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	3,054	29	18	11	7	4	0	0
Noble Metals Ag, Pd, Ru, Rh	0	0	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	157	70	14	3	1	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	1	1	0	0	0	0	0	0
Total	**3,212**	**99**	**33**	**15**	**8**	**4**	**0**	**0**

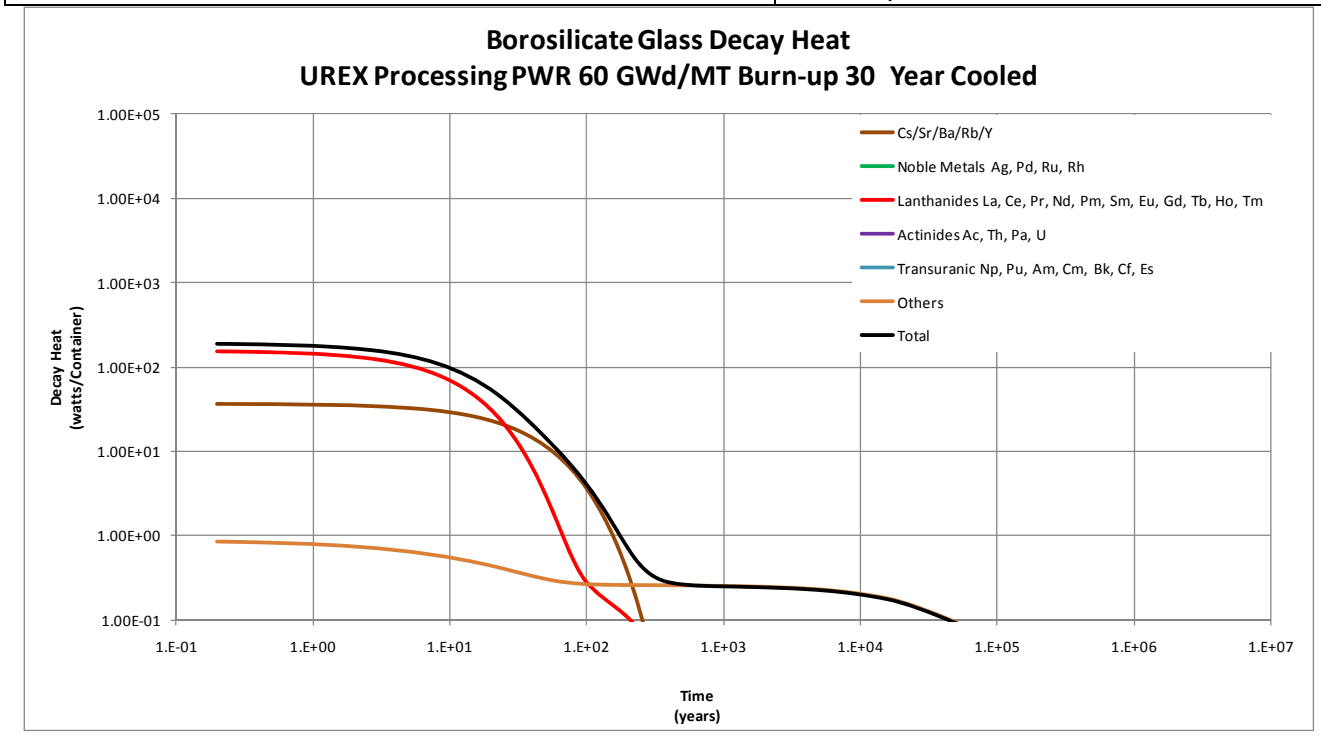

Figure L-15 Borosilicate Glass Decay Heat Generated by UREX Processing of 60 GWd/MT 30 Year PWR Fuel

Table L-16 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 60 GWd/MT 30 Yer Cooled PWR Fuel

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	-	-	-	-	-	-	-	-
Cs/Sr/Ba/Rb/Y	2,361	1,872	1,171	733	459	227	2	0
Noble Metals Ag, Pd, Ru, Rh	0	0	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	5	2	0	0	0	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	0	0	0	0	0	0	0	0
Total	**2,366**	**1,874**	**1,172**	**733**	**459**	**227**	**2**	**0**

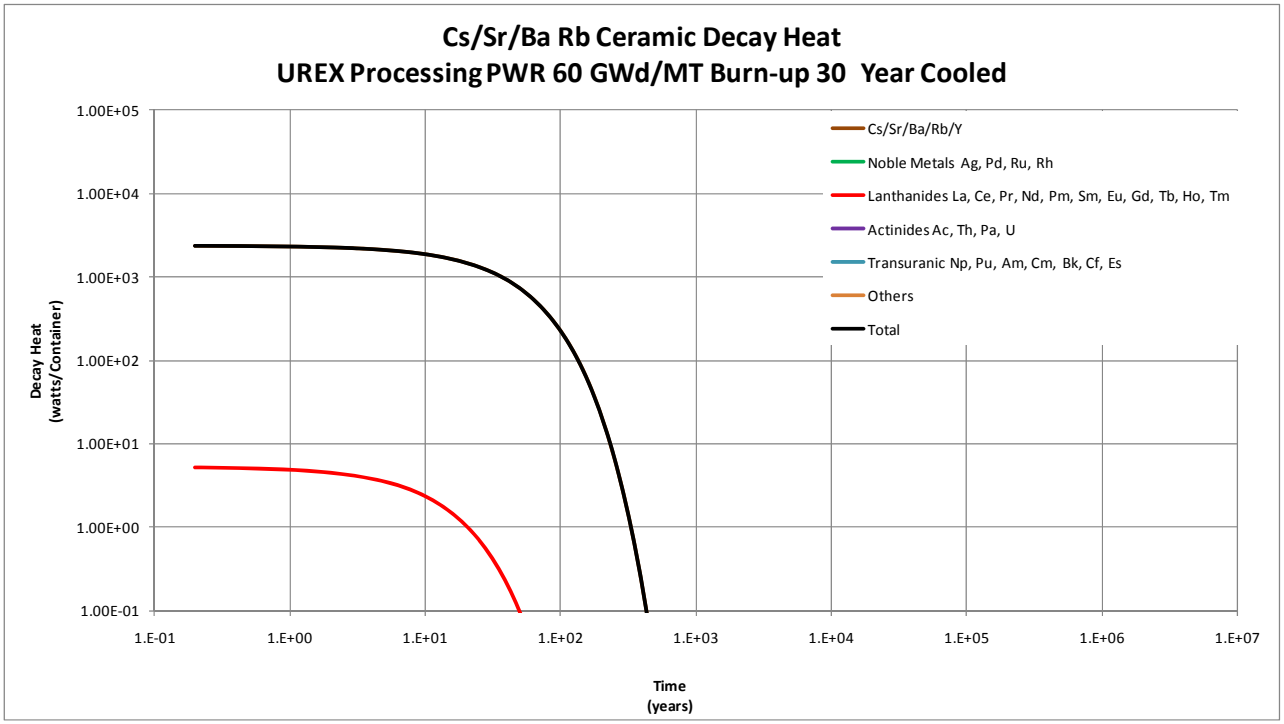

Figure L-16 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 60 GWd/MT 30 Year Cooled PWR Fuel

Table L-17 Glass Bonded Zeolite Decay Heat Generated by Electro- Chemical Processing of 60 GWd/MT 30 Year Cooled PWR Fuel

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	4,445	3,524	2,205	1,380	864	428	4	0
Noble Metals Ag, Pd, Ru, Rh	-	-	-	-	-	-	-	-
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	-	-	-	-	-	-	-	-
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	-	-	-	-	-	-	-	-
Total	**4,445**	**3,524**	**2,205**	**1,380**	**864**	**428**	**4**	**0**

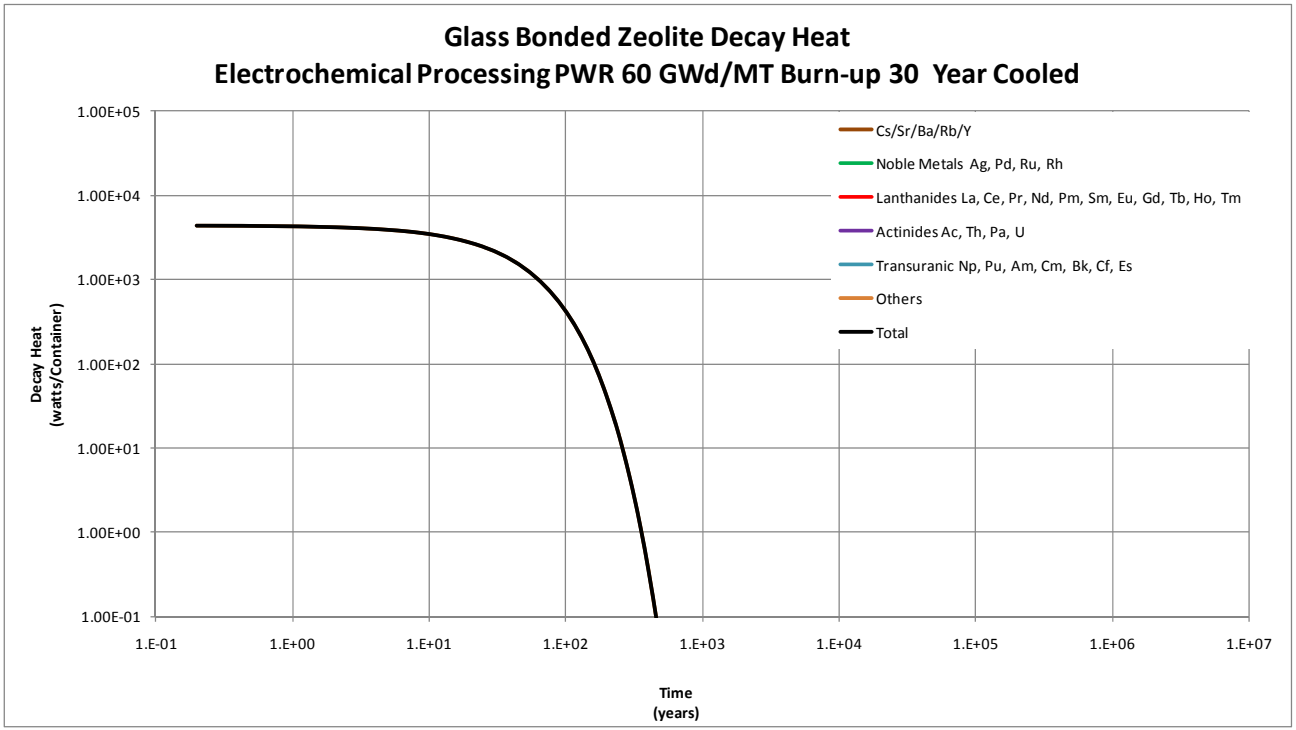

Figure L-17 Glass Bonded Zeolite Decay Heat Generated by Electo-Chemical Processing of 60 GWd/MT 30 Year Cooled PWR Fuel

Table L-18 Lanthanide Glass Decay Heat Generated by Electo-Chemical Processing of 60 GWd.MT 30 Year Cooled PWR Fuel

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	138	109	68	43	27	13	0	0
Noble Metals Ag, Pd, Ru, Rh	-	-	-	-	-	-	-	-
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	226	101	20	4	1	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	-	-	-	-	-	-	-	-
Total	**363**	**210**	**88**	**47**	**28**	**13**	**0**	**0**

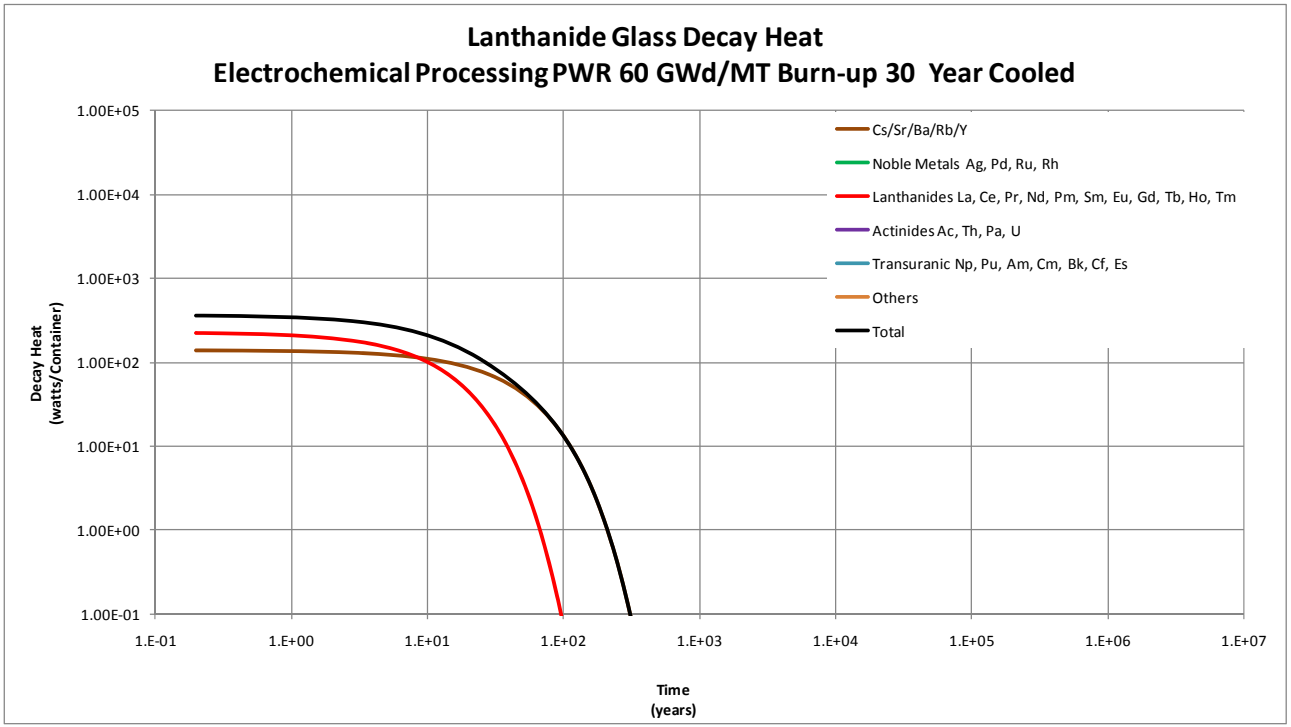

Figure L-18 Lanthanide Glass Decay Heat Generated by Electo-Chemical Processing of 60 GWd/MT 30 Year Cooled PWR Fuel

Table L-19 Borosilicate Glass Decay Het Generated by Co-Extraction Processing of BWR 50 GWd/MT Burn-up 5 Year Cooled

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	9,551	5,256	3,224	2,017	1,263	625	6	0
Noble Metals Ag, Pd, Ru, Rh	899	1	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	1,079	285	56	11	2	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	2,506	1,797	990	610	428	313	195	144
Others	8	1	0	0	0	0	0	0
Total	**14,043**	**7,340**	**4,270**	**2,639**	**1,694**	**939**	**201**	**145**

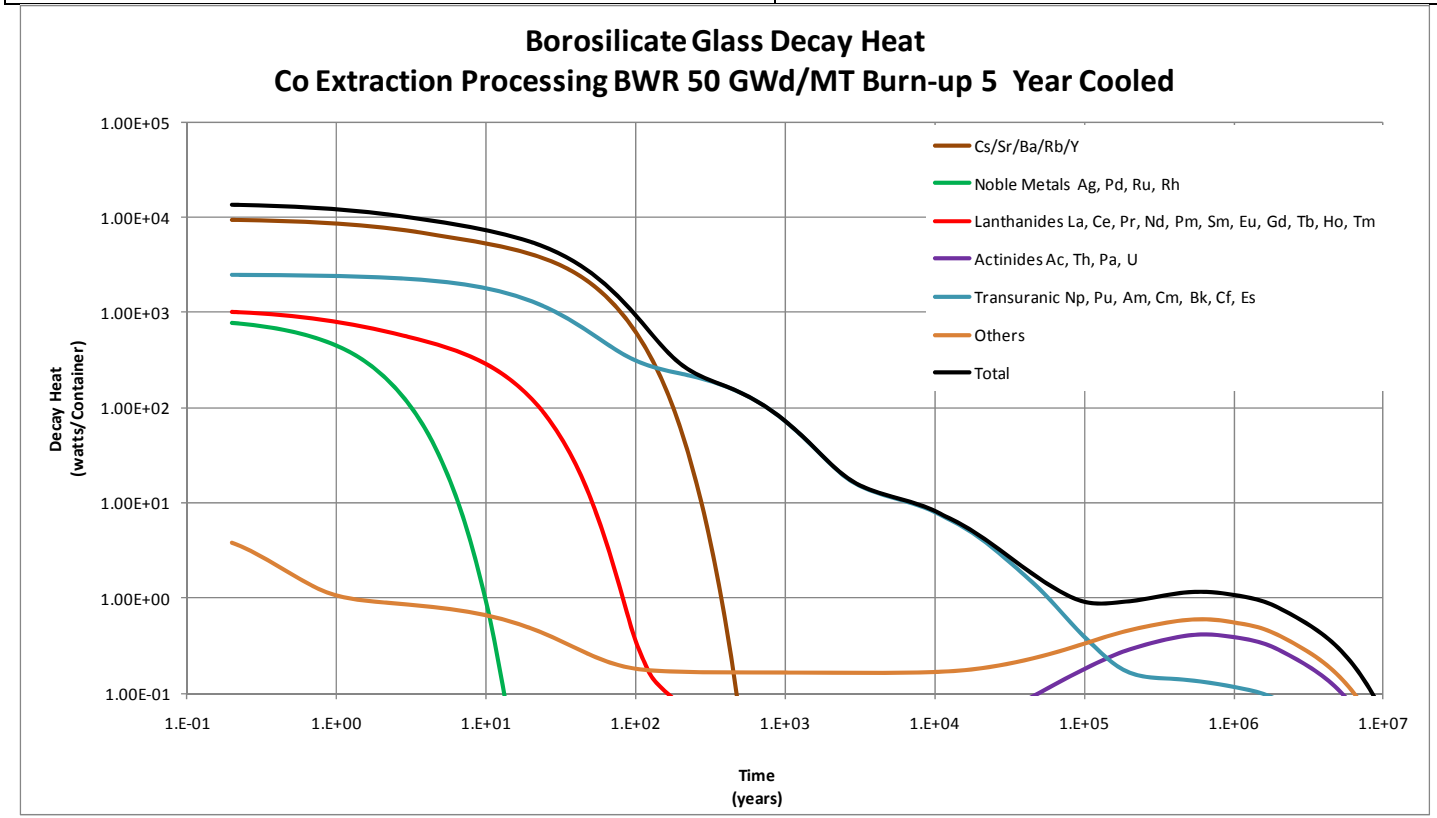

Figure L-19 Borosilicate Glass Decay Het Generated by Co-Extraction Processing of BWR 50 GWd/MT Burn-up 5 Year Cooled

Table L-20 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 50 GWd/MT Burn-up 5 Year Cooled

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	10,439	4,555	2,768	1,742	1,096	547	5	0
Noble Metals Ag, Pd, Ru, Rh	1,659	2	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	1,996	527	103	21	4	1	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	2	2	2	2	2	2	1	1
Others	13	1	1	0	0	0	0	0
Total	**14,109**	**5,086**	**2,873**	**1,764**	**1,102**	**549**	**6**	**1**

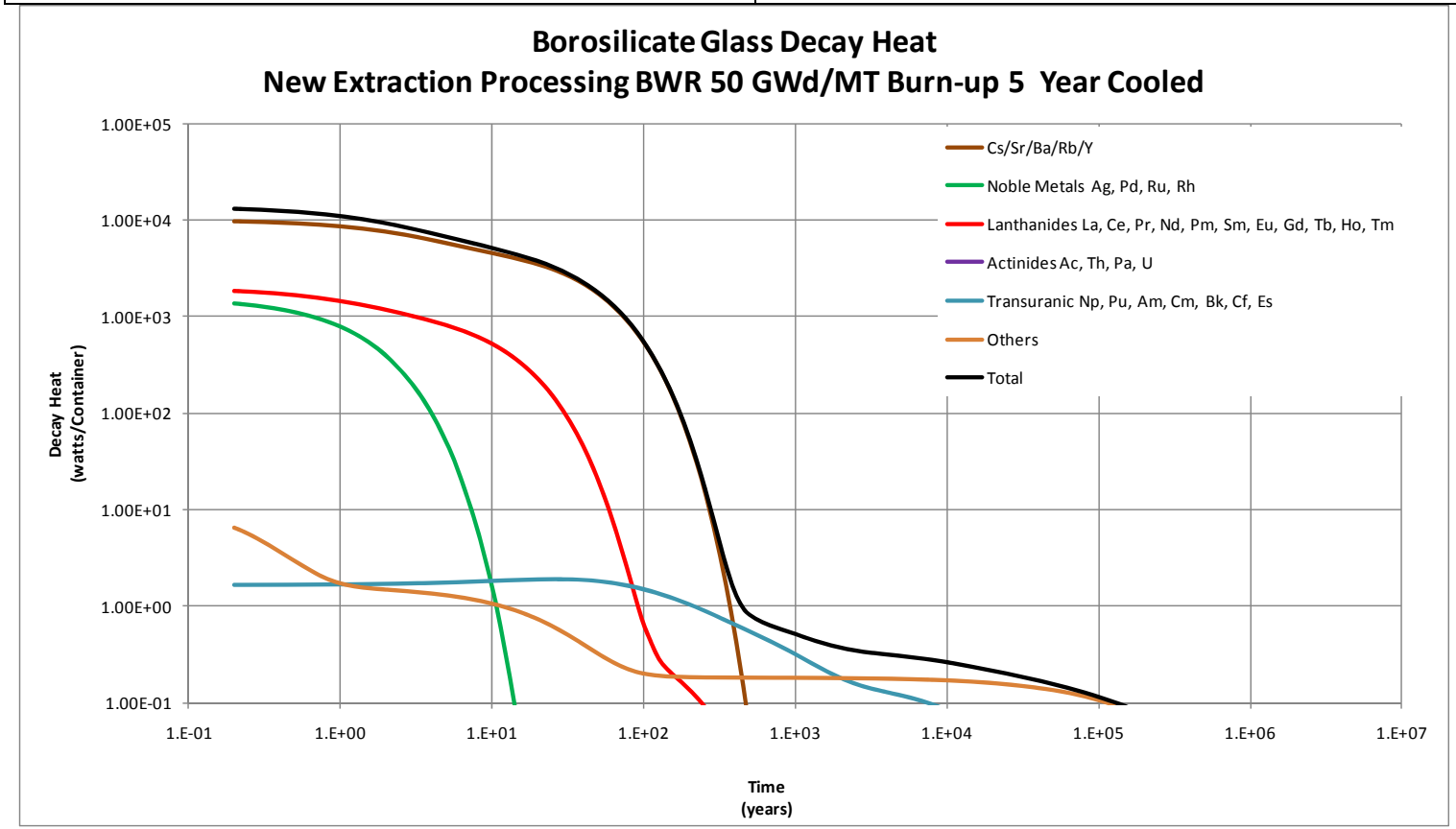

Figure L-20 Borosilicate Glass Decay Heat Generated by New Extraction Processing of 50 GWd/MT Burn-up 5 Year Cooled

Table L-21 Borosilicate Glass Decay Heat Generated by UREX Processing of 50 GWd.MT 5 Year Cooled BWR Fuel

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	123	55	33	21	13	7	0	0
Noble Metals Ag, Pd, Ru, Rh	1,294	1	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	2,071	566	110	22	5	1	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	10	1	1	0	0	0	0	0
Total	**3,499**	**624**	**144**	**43**	**18**	**7**	**0**	**0**

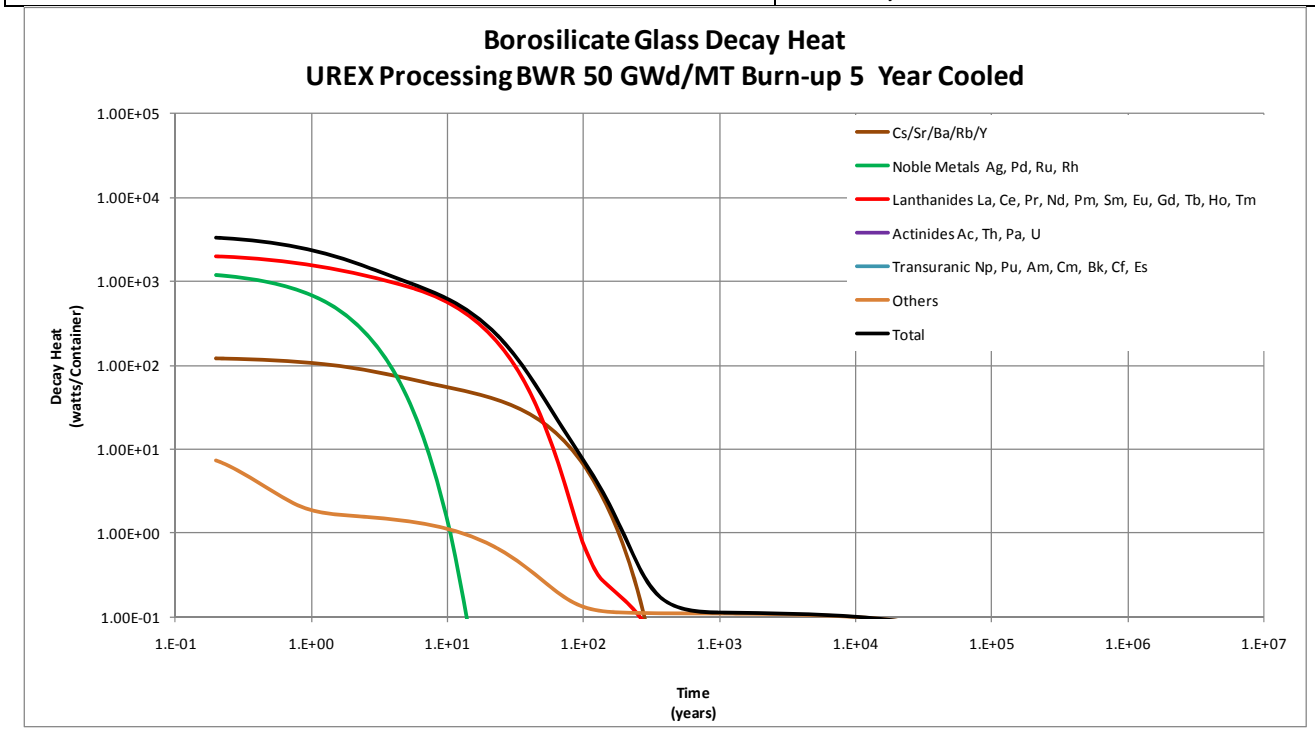

Figure L-21 Borosilicate Glass Decay Heat Generated by UREX Processing of 50 GWd.MT 5 year Cooled BWR Fuel

Table L-22 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 50 GWd/MT 5 Year Cooled BWR Fuel

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	5,694	3,180	1,950	1,221	764	378	4	0
Noble Metals Ag, Pd, Ru, Rh	15	0	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	61	17	3	1	0	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	0	0	0	0	0	0	0	0
Total	**5,771**	**3,197**	**1,954**	**1,221**	**764**	**378**	**4**	**0**

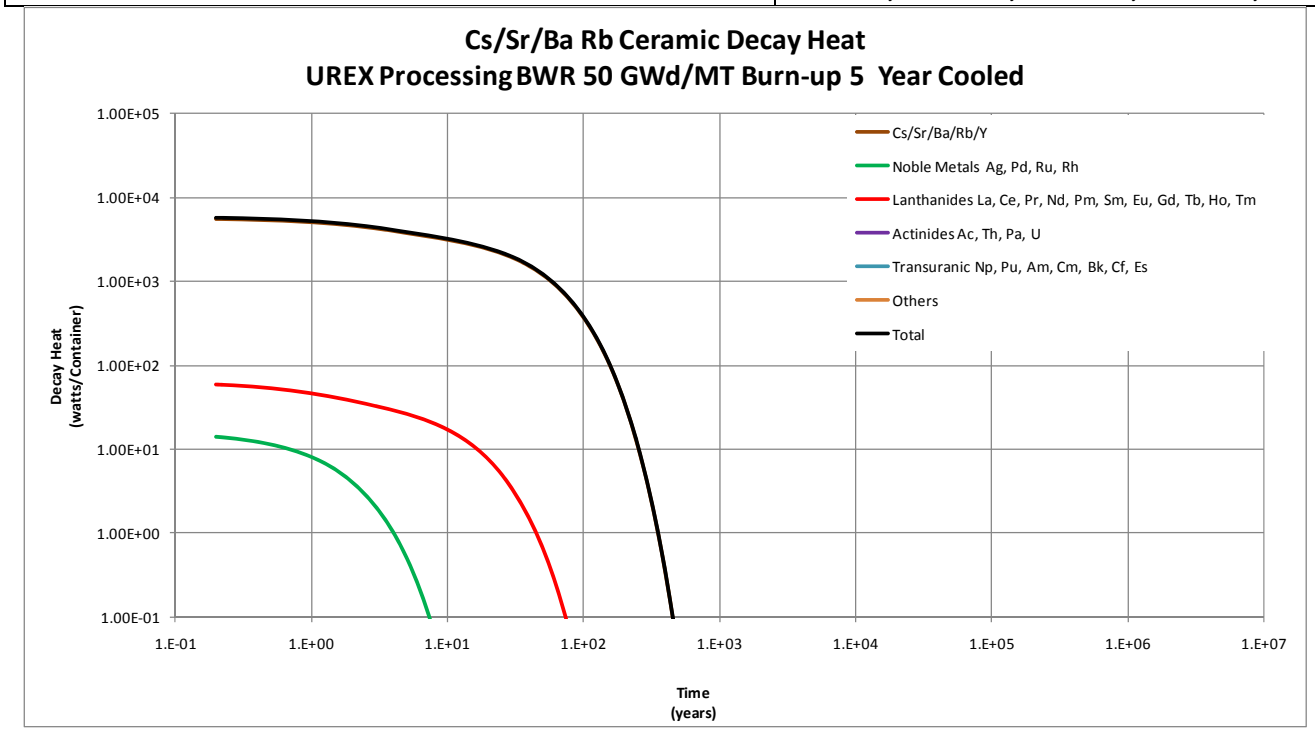

Figure L-22 Cs Sr Ceramic Decay Heat Generated by UREX Processing of 50 GWd/MT 5 Year Cooled BWR Fuel

Table L-23 Glass Bonded Zeolite Decay Heat Generated by Electro-Chemical Processing of 50 GWd/MT 5 year Cooled BWR Fuel

Time (years) Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	10,240	5,710	3,502	2,192	1,372	679	6	0
Noble Metals Ag, Pd, Ru, Rh	-	-	-	-	-	-	-	-
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	-	-	-	-	-	-	-	-
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	-	-	-	-	-	-	-	-
Total	**10,240**	**5,710**	**3,502**	**2,192**	**1,372**	**679**	**6**	**0**

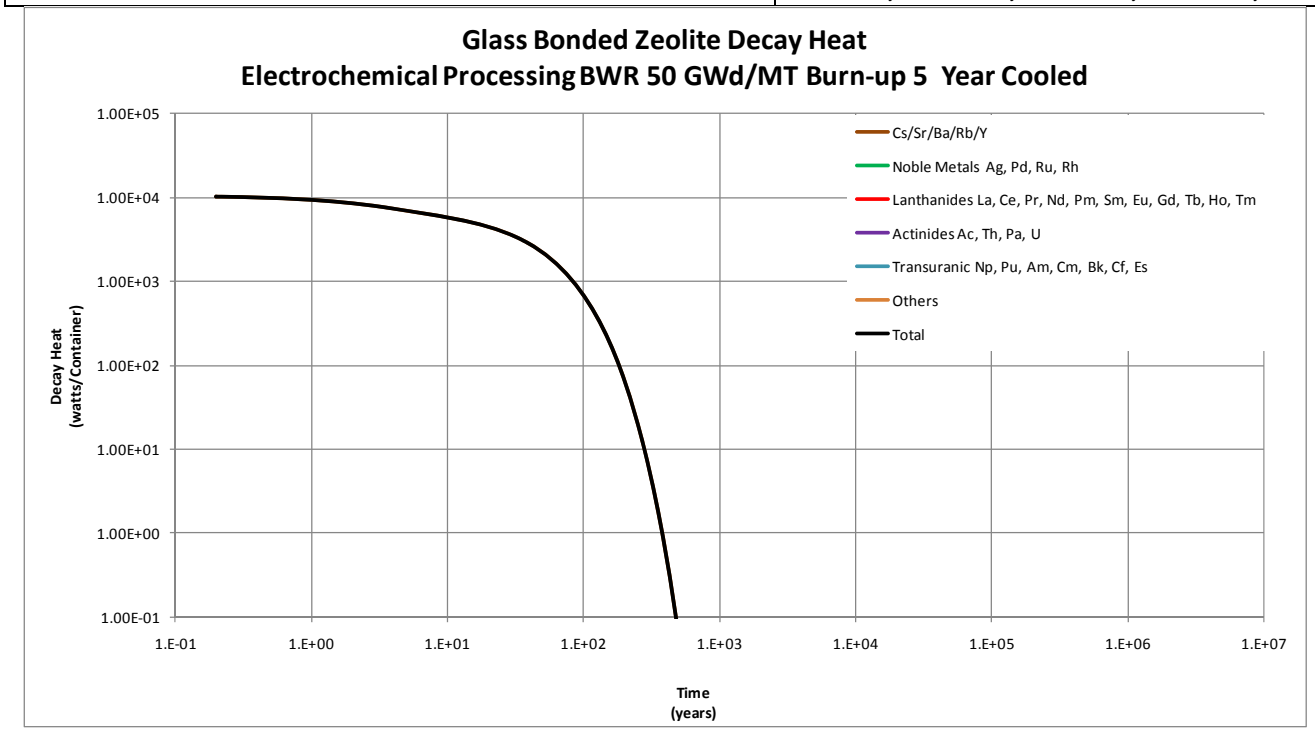

Figure L-23 Glass Bonded Zeolite Decay Heat Generated by Electro-Chemical Processing of 50 GWd.MT 5 year Cooled BWR Fuel

Table L-24 Lanthanide Glass Decay Heat Generated by Electro-Chemical Processing of 50 GWd/MT 5 year Cooled BWR Fuel

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	225	164	101	63	39	20	0	0
Noble Metals Ag, Pd, Ru, Rh	-	-	-	-	-	-	-	-
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	2,587	709	139	28	6	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	-	-	-	-	-	-	-	-
Total	**2,812**	**874**	**240**	**91**	**45**	**20**	**0**	**0**

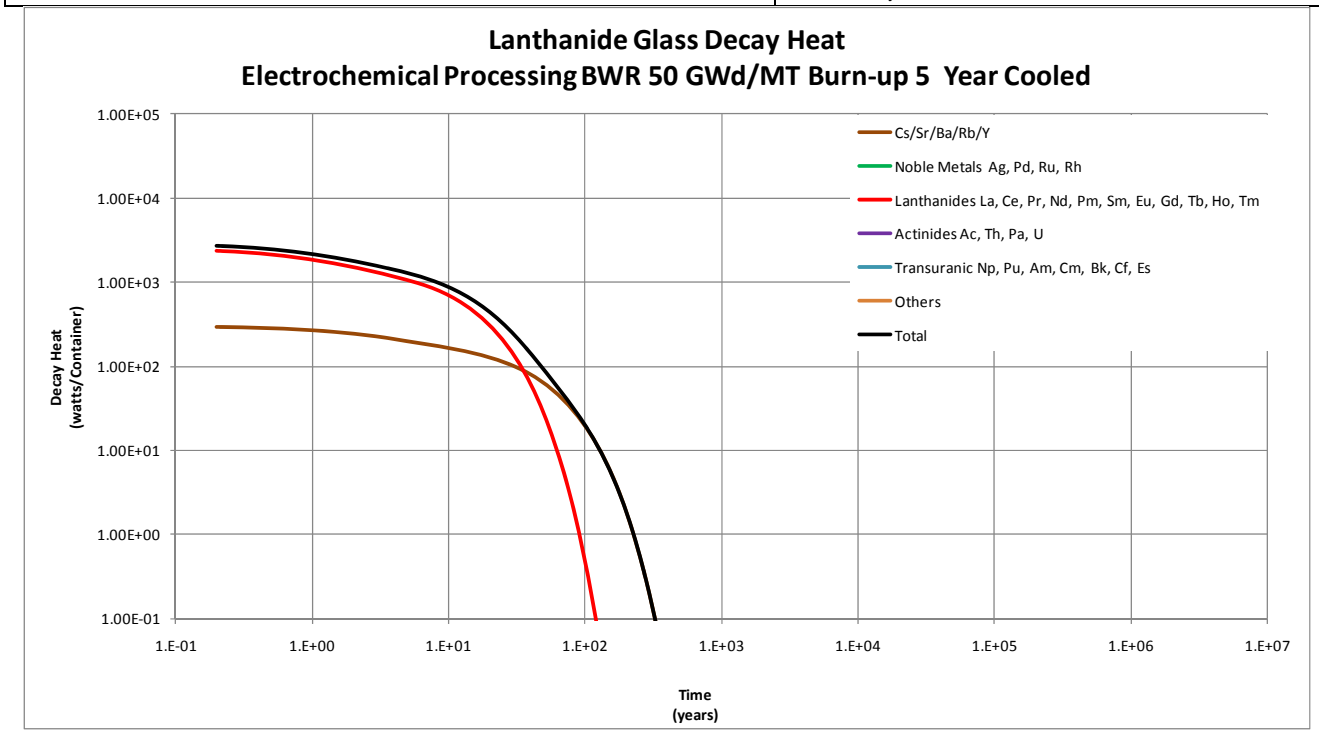

Figure L-24 Lanthanide Glass Decay Het Generated by Electro-Chemical Processing of 50 GWd/MT 5 Year Cooled BWR Fuel

Appendix M
Decay Heat for Waste from Reprocessing Advanced Burner Sodium Cooled Fast Reactor Fuel

Table M-1 New Extraction Glass Decay Heat Generated by Processing Sodium FR Oxide 0.75 CR 131 GWd/MT Burn-up 2 Year Cooled

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	-	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	2,908	2,131	1,319	830	523	261	3	0
Noble Metals Ag, Pd, Ru, Rh	7,002	7	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	3,787	98	16	5	2	1	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	4	4	4	3	3	3	2	1
Others	28	2	1	0	0	0	0	0
Total	**13,728**	**2,241**	**1,340**	**839**	**528**	**266**	**4**	**1**

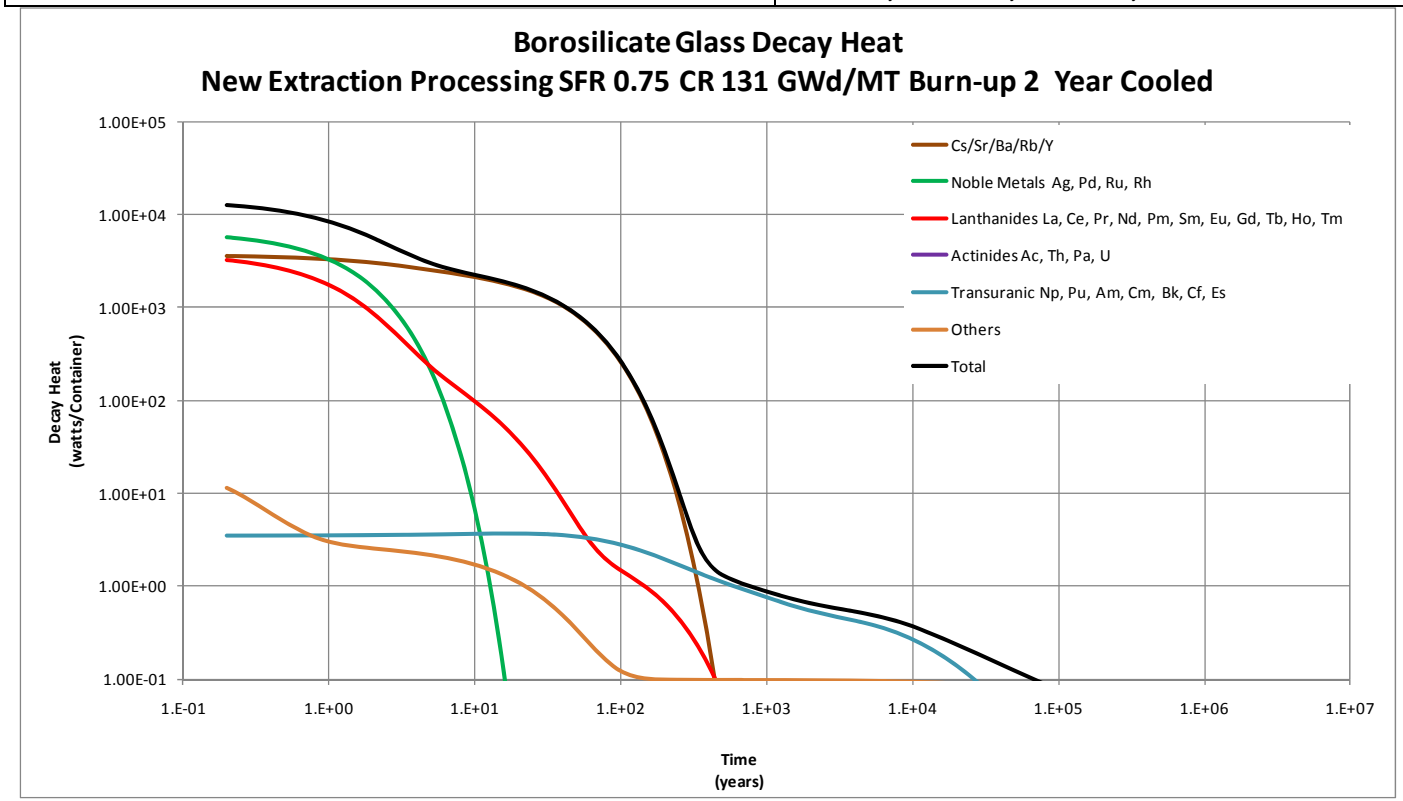

Figure M-1 New Extraction Glass Decay Heat Generated by Processing Sodium FR Oxide 0.75 CR 131 GWd/MT Burn-up 2 Year Cooled

Table M-2 New Extraction Glass Decay Heat Generated by Processing Sodium FR Oxide 0.5 CR 166 GWd/MT Burn-up 2 Year Cooled

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	-	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	2,888	2,088	1,291	813	512	256	3	0
Noble Metals Ag, Pd, Ru, Rh	7,071	7	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	3,735	97	16	5	2	1	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	5	5	5	5	4	4	2	1
Others	28	2	1	0	0	0	0	0
Total	**13,727**	**2,199**	**1,313**	**822**	**518**	**261**	**5**	**1**

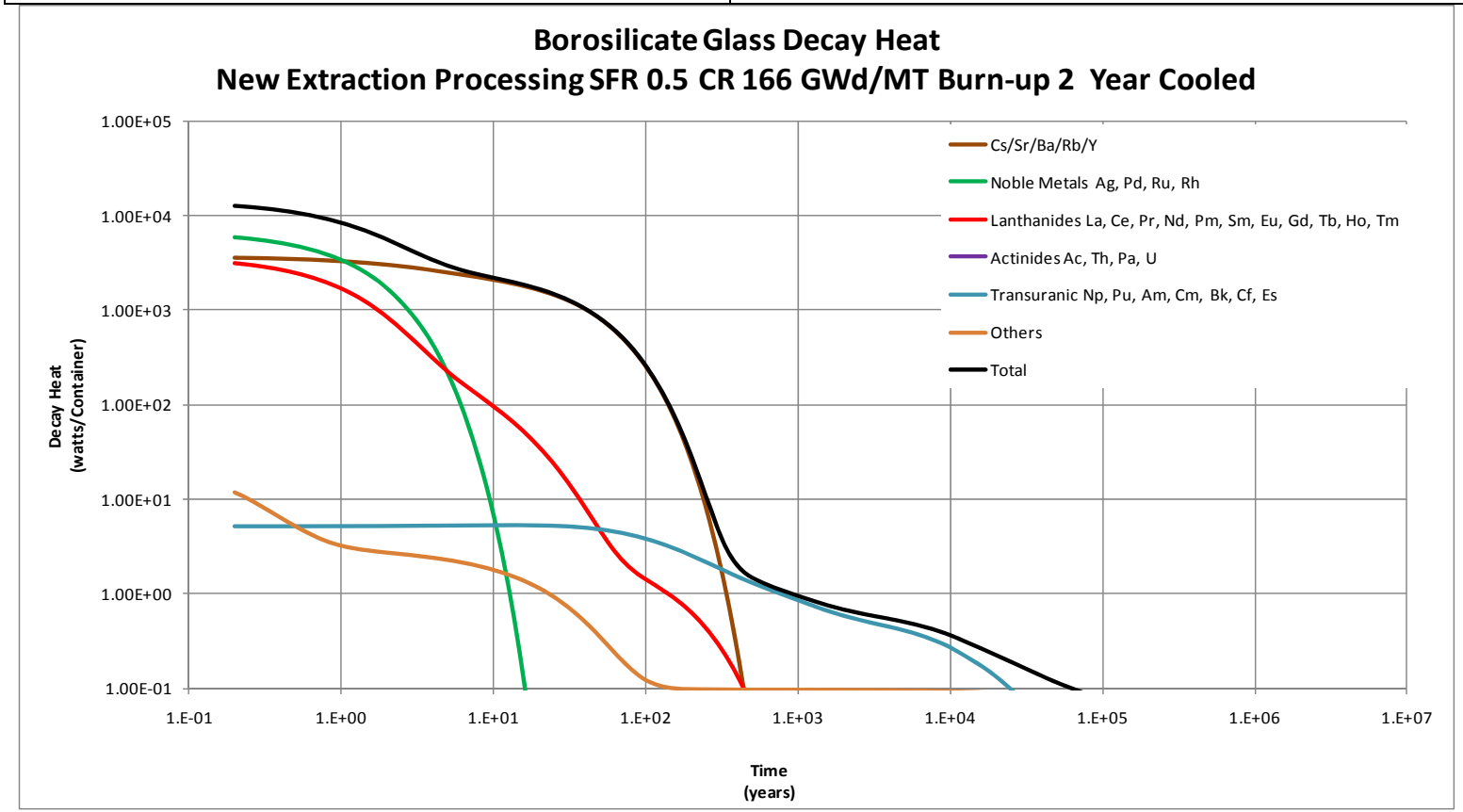

Figure M-2 New Extraction Glass Decay Heat Generated by Processing Sodium FR Oxide 0.5 CR 166 GWd/MT Burn-up 2 Year Cooled

Table M-3 Electro-Chemical Glass Bonded Zeolite Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ration of 0.75

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	2,255	1,785	1,106	693	435	216	2	0
Noble Metals Ag, Pd, Ru, Rh	-	-	-	-	-	-	-	-
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	-	-	-	-	-	-	-	-
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	-	-	-	-	-	-	-	-
Total	**2,255**	**1,785**	**1,106**	**693**	**435**	**216**	**2**	**0**

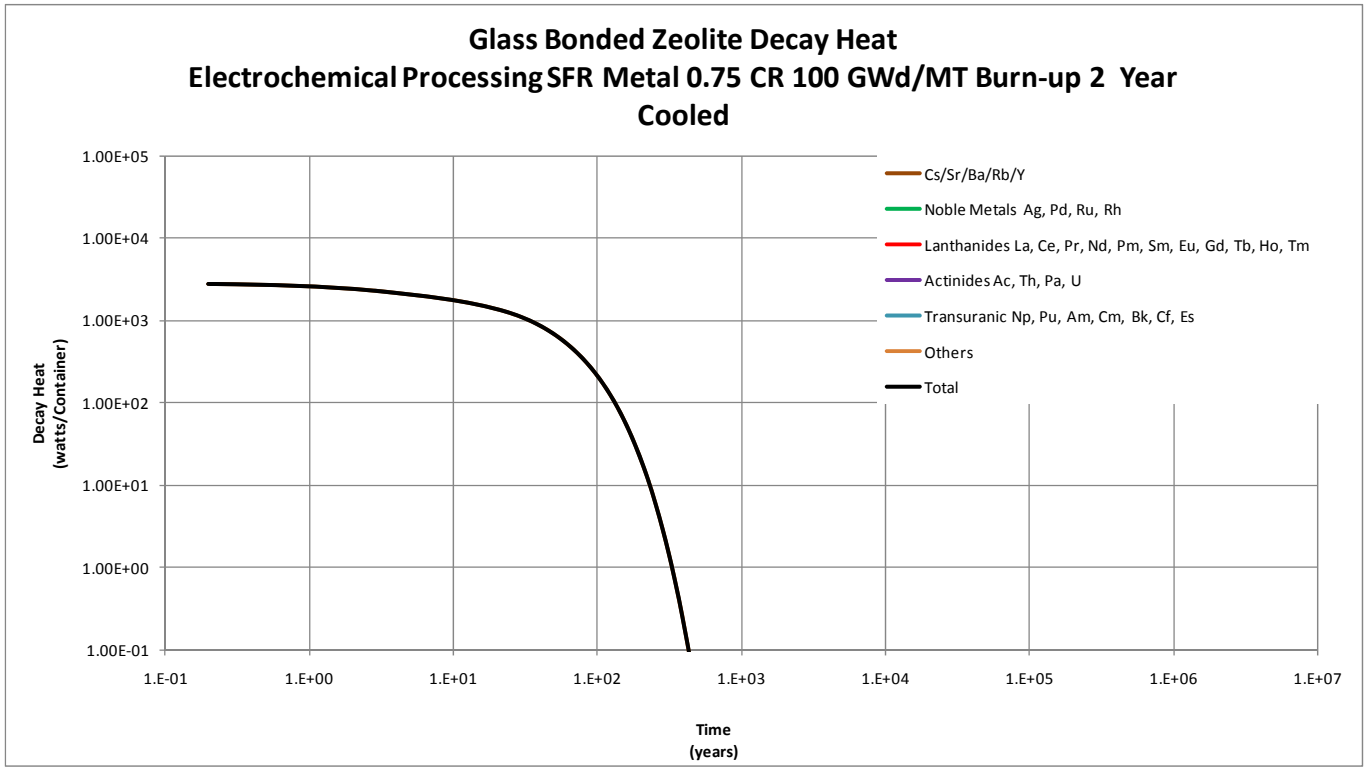

Figure M-3 Electro-Chemical Glass Bonded Zeolite Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.75

Table M-4 Electro-Chemical Lanthanide Glass Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.75

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	0	0	0	0	0	0	0	0
Noble Metals Ag, Pd, Ru, Rh	3,777	4	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	113	9	1	1	1	1	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	1	0	0	0	0	0	0	0
Others	15	1	0	0	0	0	0	0
Total	**3,905**	**14**	**2**	**1**	**1**	**1**	**0**	**0**

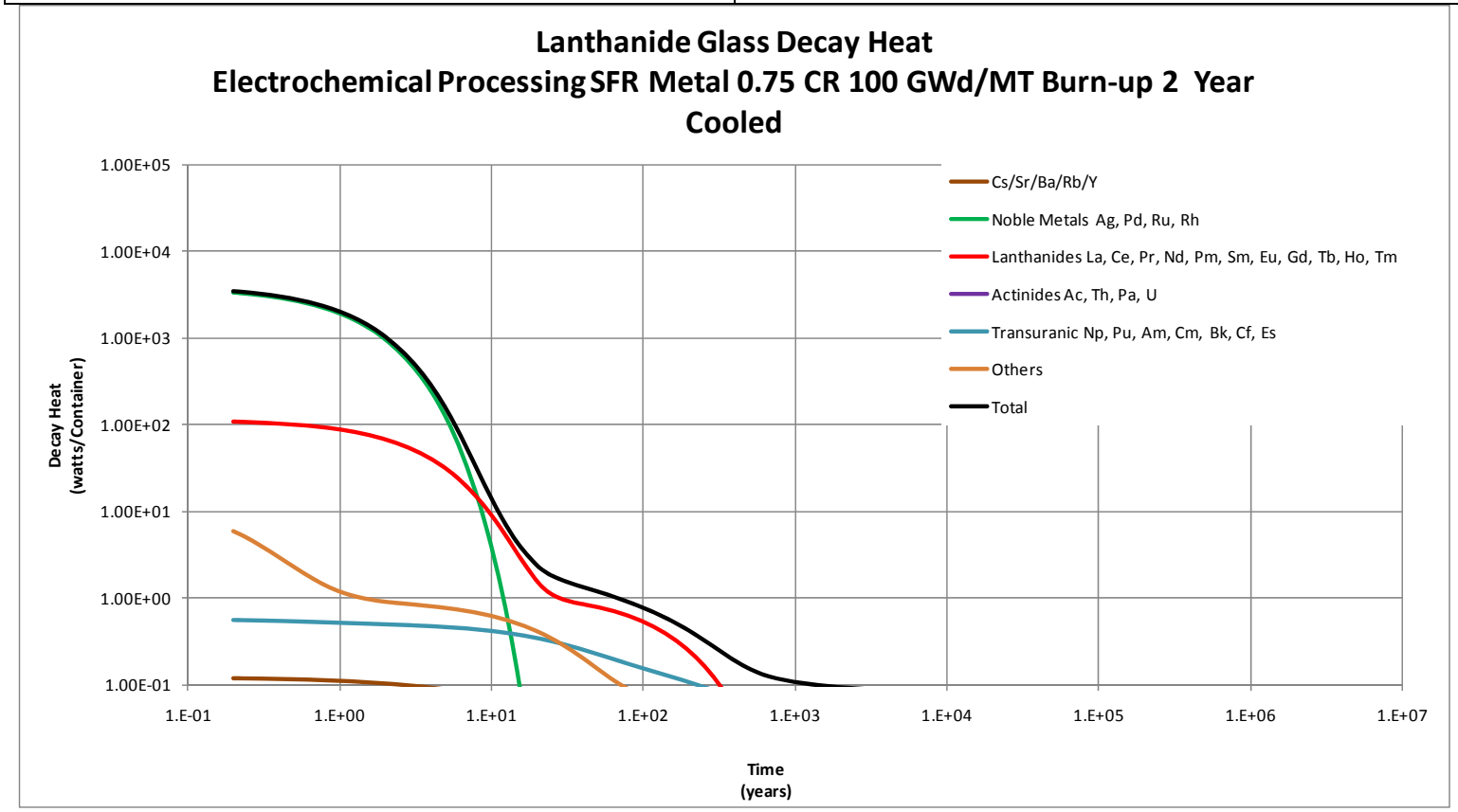

Figure M-4 Electro-Chemical Lanthanide Glass Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.75

Table M-5 Electro-Chemical Glass Bonded Zeolite Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.5

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	2,257	1,746	1,081	678	425	211	2	0
Noble Metals Ag, Pd, Ru, Rh	-	-	-	-	-	-	-	-
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	-	-	-	-	-	-	-	-
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	0	0	0	0	0	0	0	0
Others	-	-	-	-	-	-	-	-
Total	**2,257**	**1,746**	**1,081**	**678**	**425**	**211**	**2**	**0**

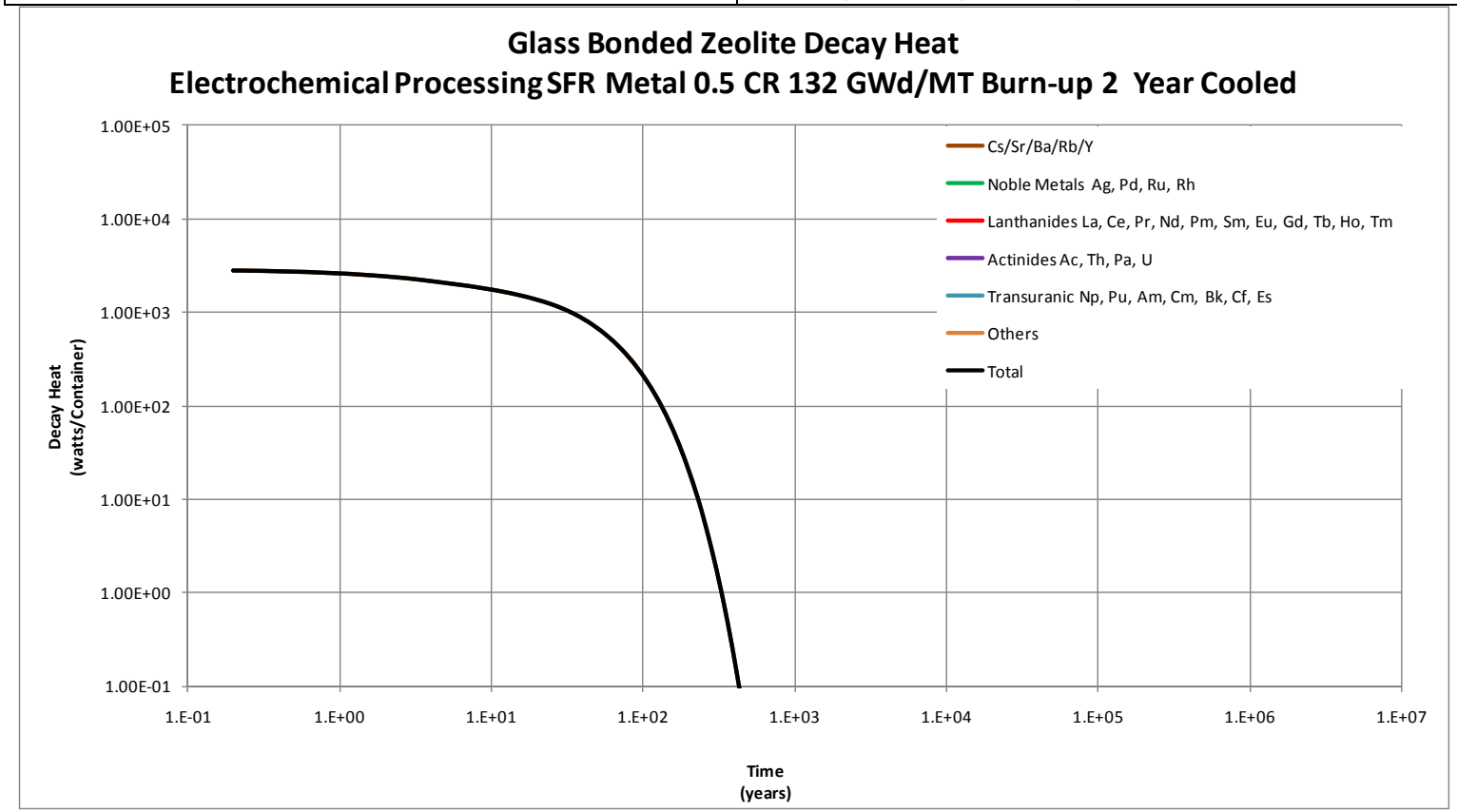

Figure M-5 Electro-Chemical Glass Bonded Zeolite Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.5

Table M-6 Electro-Chemical Lanthanide Glass Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.5

Time (years) / Decay Heat (Watts/Container)	Initial Production	10	30	50	70	100	300	500
Gases H, C, Xe, Kr, I	0	0	0	0	0	0	0	0
Cs/Sr/Ba/Rb/Y	0	0	0	0	0	0	0	0
Noble Metals Ag, Pd, Ru, Rh	3,575	4	0	0	0	0	0	0
Lanthanides La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Ho, Tm	103	8	1	1	1	0	0	0
Actinides Ac, Th, Pa, U	0	0	0	0	0	0	0	0
Transuranic Np, Pu, Am, Cm, Bk, Cf, Es	1	1	1	0	0	0	0	0
Others	14	1	0	0	0	0	0	0
Total	**3,694**	**14**	**2**	**1**	**1**	**1**	**0**	**0**

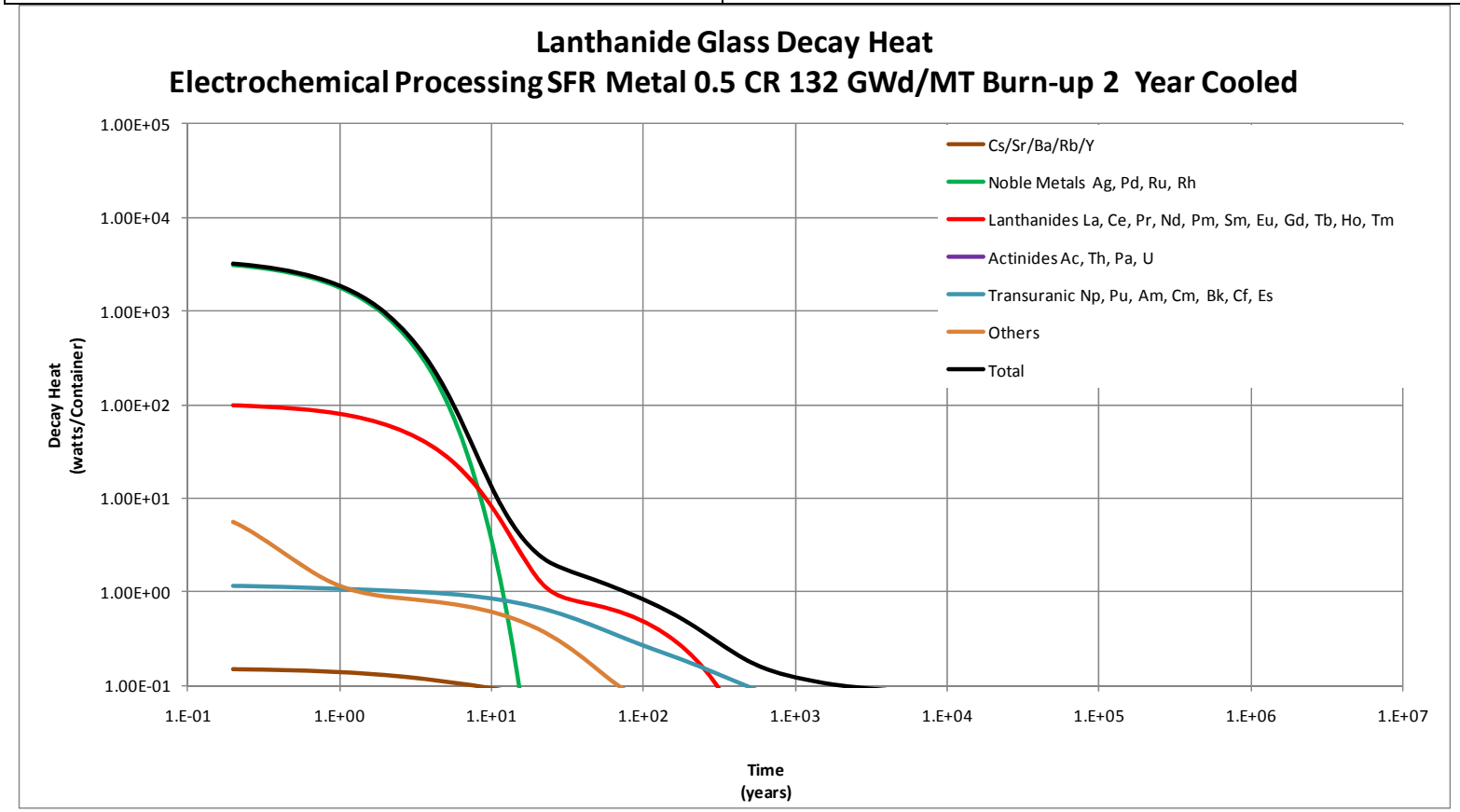

Figure M-6 Electro-Chemical Lanthanide Glass Decay Heat Generated by Processing Sodium FR Metal Fuel with a TRU Conversion Ratio of 0.5

Appendix N
High Temperature Gas Reactor Fuel Characteristic

Table N-1 TRISO Particle Parameters

TRISO Particle Parameters			
Item	Value	Units	Source
Kernel Diameter	350	µm	INL/EXT-10-19704, Pg 39
Buffer Layer Thickness	100	µm	INL/EXT-10-19704, Pg 39
Inner PyC Thickness	35	µm	INL/EXT-10-19704, Pg 39
SiC Thickness	35	µm	INL/EXT-10-19704, Pg 39
Outer PyC Thickness	40	µm	INL/EXT-10-19704, Pg 39
Kernel Density	10.86	g/cm^3	INL/EXT-10-19704, Pg 39
Buffer Layer Density	1.05	g/cm^3	INL/EXT-10-19704, Pg 39
Inner PyC Density	1.9	g/cm^3	INL/EXT-10-19704, Pg 39
SiC Density	3.18	g/cm^3	INL/EXT-10-19704, Pg 39
Outer PyC Density	1.9	g/cm^3	INL/EXT-10-19704, Pg 39

Table N-2 Fuel Compact Parameters

Fuel Compact Parameters			
Item	Value	Units	Source
Height	49.3	mm	NGNP-S00218 Rev. 0, Pg 29
Diameter	12.45	mm	NGNP-S00218 Rev. 0, Pg 29
Density	2.07	g/cm^3	Derived from INL/EXT-10-19704, Pg 39
Particles/Compact	10,753	count	INL/EXT-10-19704, Pg 40

Table N-3 Fuel Element Parameters

Fuel Element Parameters			
Item	**Value**	**Units**	**Source**
Height	793	mm	NGNP-S00218 Rev. 0, Pg 26
Flat to Flat Distance	360	mm	NGNP-S00218 Rev. 0, Pg 26
Compacts/Element	3,126	count	INL/EXT-10-19704, Pg 40
Unloaded Mass	90	kg	INL/EXT-10-19704, Pg 40

Table N-4 Isotopic Distribution of Spent HTGR Fuel

Isotope	Discharge
He4	1.03E-22
Pb206 (stable)	3.50E-17
Pb207 (stable)	1.53E-15
Pb208 (stable)	6.01E-13
Pb210	1.45E-18
Bi209 (stable)	0.00E+00
Ra226	2.69E-17
Ra228	2.59E-20
Ac227	0.00E+00
Th228	5.95E-13
Th229	1.06E-13
Th230	3.31E-12
Th232	4.68E-10
Pa231	6.24E-11
U232	1.31E-10
U233	6.89E-10
U234	2.41E-06
U235	5.68E-02
U236	1.82E-02
U238	7.90E-01
Np237	1.61E-03
Pu238	4.75E-04
Pu239	1.41E-02
Pu240	4.71E-03
Pu241	5.48E-03
Pu242	1.90E-03
Pu244	6.93E-08
Am241	8.40E-05
Am242m	4.44E-07
Am243	4.50E-04
Cm242	4.19E-05
Cm243	8.00E-07
Cm244	9.71E-05
Cm245	6.06E-06
Cm246	3.00E-07
Cm247	2.75E-09
Cm248	1.39E-10
Cm250	1.34E-17
Bk249	1.37E-12

Table N-4 Continued

Isotope	Discharge
Cf249	9.99E-14
Cf250	5.50E-13
Cf251	2.45E-13
Cf252	9.03E-14
H3	1.75E-07
C14	8.72E-09
C-other	0
Kr81	2.83E-11
Kr85	8.26E-05
Inert gas other (Kr, Xe)	1.90E-02
Rb	1.14E-03
Sr90 (include Y90 decay energy)	2.00E-03
Sr-other	1.41E-03
Zr93 (include Nb93m decay energy)	2.50E-03
Zr95 (include Nb95m decay energy)	4.22E-04
Zr-other	9.63E-03
Tc99	2.62E-03
Tc-other	1.13E-06
Ru106 (include Rh106 decay energy)	6.14E-04
Pd107	6.18E-04
Mo-Ru-Rh-Pd-other	1.95E-02
Se79	1.58E-05
Cd113m	3.92E-09
Sn126 (include Sb126m and Sb126 decay energy)	5.83E-05
Sb125 (include Te125m decay energy)	2.74E-05
TM-other (Be, Li, Co-Se, Nb, Ag-Te)	2.47E-03
I129	4.78E-04
Halogen-other (Br, I)	2.63E-04
Cs134	4.83E-04
Cs135	1.21E-03
Cs137 (include Ba137m decay energy)	4.18E-03
Cs-other	3.77E-03
Ba	4.81E-03
Ce144 (include Pr144m and Pr144 decay energy)	1.87E-03
Pm147	6.74E-04
Sm146	3.43E-09
Sm147	1.43E-04
Sm151	6.08E-05
Eu154	7.27E-05
Eu155	2.00E-05

Table N-4 Continued

Isotope	Discharge
Ho166m	3.09E-10
Lanthanide-other (including Y)	2.63E-02

Table N-5 Isotopic Distribution of Spent HTGR Fuel

Isotope	Discharge	5 Years	30 Years	100 Years	500 Years
H3	1.750E-07	1.322E-07	3.248E-08	6.381E-10	1.128E-19
H-other	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
He	1.034E-22	1.519E-06	6.328E-06	1.982E-05	6.844E-05
Li	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Be	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
B	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
C14	8.719E-09	8.714E-09	8.688E-09	8.614E-09	8.207E-09
C-other	1.913E-17	1.913E-17	1.913E-17	1.913E-17	1.913E-17
N	0.000E+00	5.273E-12	3.159E-11	1.049E-10	5.118E-10
O	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
F	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ne	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Na	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Mg	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Al	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Si	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
P	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
S	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cl	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ar	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
K	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ca	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Sc	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ti	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
V	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cr	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Mn54	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Mn56	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Mn-other	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Fe55	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Fe-other	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Co60	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Co-other	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ni	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cu	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Zn	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ga	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ge	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
As	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Se79	1.579E-05	1.579E-05	1.579E-05	1.578E-05	1.571E-05
Se-other	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Br	0.000E+00	8.425E-10	5.054E-09	1.684E-08	8.403E-08
Kr81	2.829E-11	2.829E-11	2.829E-11	2.828E-11	2.824E-11
Kr85	8.260E-05	5.978E-05	1.187E-05	1.285E-07	7.515E-19
Kr-other	1.901E-03	1.901E-03	1.901E-03	1.901E-03	1.901E-03

Table N-5 Continued

Isotope	Discharge	5 Years	30 Years	100 Years	500 Years
Rb	1.140E-03	1.163E-03	1.211E-03	1.223E-03	1.223E-03
Sr90 (include Y90 decay energy)	2.000E-03	1.776E-03	9.793E-04	1.851E-04	1.358E-08
Sr-other	1.410E-03	1.410E-03	1.410E-03	1.410E-03	1.410E-03
Y	0.000E+00	4.453E-07	2.456E-07	4.642E-08	3.405E-12
Zr93 (include Nb93m decay energy)	2.499E-03	2.499E-03	2.499E-03	2.499E-03	2.499E-03
Zr95 (include Nb95m decay energy)	4.220E-04	1.077E-12	1.164E-55	5.767E-176	0.000E+00
Zr-other	9.630E-03	9.854E-03	1.065E-02	1.144E-02	1.163E-02
Nb94	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Nb-other	0.000E+00	5.664E-09	3.398E-08	1.133E-07	5.663E-07
Mo	0.000E+00	4.220E-04	4.220E-04	4.220E-04	4.220E-04
Tc99	2.615E-03	2.615E-03	2.615E-03	2.614E-03	2.611E-03
Tc-other	1.126E-06	1.126E-06	1.126E-06	1.126E-06	1.126E-06
Ru106 (include Rh106 decay energy)	6.136E-04	1.971E-05	6.741E-13	8.395E-34	2.943E-153
Ru-other	0.000E+00	4.254E-08	2.552E-07	8.507E-07	4.251E-06
Rh	1.949E-02	1.949E-02	1.949E-02	1.949E-02	1.949E-02
Pd107	6.177E-04	6.177E-04	6.177E-04	6.177E-04	6.177E-04
Pd-other	0.000E+00	5.939E-04	6.136E-04	6.136E-04	6.136E-04
Ag	0.000E+00	3.296E-10	1.977E-09	6.591E-09	3.295E-08
Cd113m	3.924E-09	3.094E-09	9.430E-10	3.386E-11	1.878E-19
Cd-other	0.000E+00	1.162E-12	4.173E-12	5.446E-12	5.493E-12
In	0.000E+00	8.287E-10	2.977E-09	3.884E-09	3.918E-09
Sn126 (include Sb126m and Sb126 decay energy)	5.831E-05	5.830E-05	5.829E-05	5.827E-05	5.811E-05
Sn-other	2.470E-03	2.470E-03	2.470E-03	2.470E-03	2.470E-03
Sb125 (include Te125m decay energy)	2.737E-05	7.830E-06	1.502E-08	3.706E-16	1.247E-59
Sb-other	0.000E+00	2.791E-12	2.790E-12	2.789E-12	2.781E-12
Te	0.000E+00	1.954E-05	2.736E-05	2.741E-05	2.757E-05
I129	4.785E-04	4.785E-04	4.785E-04	4.785E-04	4.785E-04
I-other	2.628E-04	2.628E-04	2.628E-04	2.628E-04	2.628E-04
Xe	1.711E-02	1.711E-02	1.711E-02	1.711E-02	1.711E-02
Cs134	4.829E-04	8.997E-05	2.019E-08	1.224E-18	5.049E-77
Cs135	1.212E-03	1.212E-03	1.212E-03	1.212E-03	1.211E-03
Cs137 (include Ba137m decay energy)	4.184E-03	3.727E-03	2.092E-03	4.150E-04	4.018E-08
Cs-other	3.769E-03	3.769E-03	3.769E-03	3.769E-03	3.769E-03
Ba	4.812E-03	5.662E-03	7.387E-03	9.064E-03	9.479E-03
La	2.632E-03	2.632E-03	2.632E-03	2.632E-03	2.632E-03
Ce144 (include Pr144m and Pr144 decay energy)	1.872E-03	2.180E-05	4.671E-15	3.945E-42	7.774E-197
Ce-other	2.632E-03	2.632E-03	2.632E-03	2.632E-03	2.632E-03
Pr	2.632E-03	2.632E-03	2.632E-03	2.632E-03	2.632E-03
Nd	5.263E-03	7.114E-03	7.136E-03	7.136E-03	7.136E-03
Pm147	6.743E-04	1.799E-04	2.436E-07	2.263E-15	2.874E-61
Pm-other	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Sm146	3.426E-09	3.426E-09	3.426E-09	3.426E-09	3.426E-09
Sm147	1.430E-04	6.373E-04	8.170E-04	8.173E-04	8.173E-04
Sm151	6.082E-05	5.852E-05	4.827E-05	2.815E-05	1.292E-06
Sm-other	2.632E-03	2.632E-03	2.632E-03	2.632E-03	2.632E-03

Table N-5 Continued

Isotope	Discharge	5 Years	30 Years	100 Years	500 Years
Eu154	7.270E-05	4.859E-05	6.478E-06	2.297E-08	2.290E-22
Eu155	2.001E-05	9.946E-06	3.021E-07	1.702E-11	8.928E-36
Eu-other	2.632E-03	2.634E-03	2.644E-03	2.664E-03	2.691E-03
Gd	2.632E-03	2.666E-03	2.718E-03	2.724E-03	2.724E-03
Tb	2.632E-03	2.632E-03	2.632E-03	2.632E-03	2.632E-03
Dy	1.316E-03	1.316E-03	1.316E-03	1.316E-03	1.316E-03
Ho166m	3.087E-10	3.078E-10	3.034E-10	2.914E-10	2.313E-10
Ho-other	1.316E-03	1.316E-03	1.316E-03	1.316E-03	1.316E-03
Er	0.000E+00	8.901E-13	5.302E-12	1.732E-11	7.742E-11
Tm	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Yb	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Lu	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Hf	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ta	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
W	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Re	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Os	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Ir	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Pt	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Au	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Hg	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Tl	0.000E+00	2.880E-18	2.788E-18	2.308E-18	6.407E-18
Pb206 (stable)	3.504E-17	3.534E-17	2.451E-16	5.417E-14	4.203E-11
Pb207 (stable)	1.531E-15	2.638E-15	1.338E-13	3.010E-12	1.154E-10
Pb208 (stable)	6.013E-13	4.043E-12	2.820E-11	7.252E-11	1.176E-10
Pb210	1.451E-18	2.880E-18	1.156E-15	8.561E-14	1.022E-11
Bi209 (stable)	0.000E+00	5.274E-17	1.251E-15	3.595E-14	5.915E-12
Po	0.000E+00	5.652E-20	1.998E-17	1.479E-15	1.766E-13
At	0.000E+00	1.962E-26	1.587E-25	1.654E-24	6.015E-23
Rn	0.000E+00	2.570E-18	5.195E-18	8.992E-17	5.898E-15
Fr	0.000E+00	6.833E-22	1.199E-20	8.082E-20	9.805E-19
Ra226	2.688E-17	2.905E-15	4.439E-13	1.380E-11	9.175E-10
Ra228	2.589E-20	3.760E-19	5.366E-18	2.267E-17	1.227E-16
Ac227	0.000E+00	1.957E-14	4.087E-13	2.549E-12	1.682E-11
Th228	5.946E-13	2.872E-12	2.633E-12	1.342E-12	2.855E-14
Th229	1.061E-13	1.485E-13	1.201E-12	1.252E-11	4.552E-10
Th230	3.310E-12	1.724E-10	4.846E-09	4.431E-08	5.342E-07
Th232	4.676E-10	3.116E-09	1.636E-08	5.350E-08	2.671E-07
Pa231	6.239E-11	3.376E-10	1.713E-09	5.562E-09	2.748E-08
U232	1.310E-10	1.248E-10	9.813E-11	5.001E-11	1.063E-12
U233	6.892E-10	3.255E-09	1.651E-08	6.094E-08	5.094E-07
U234	2.412E-06	2.188E-05	1.094E-04	2.797E-04	5.004E-04
U235	5.685E-02	5.685E-02	5.686E-02	5.689E-02	5.705E-02
U236	1.820E-02	1.820E-02	1.821E-02	1.825E-02	1.844E-02
U238	7.896E-01	7.896E-01	7.896E-01	7.896E-01	7.896E-01
Np237	1.610E-03	1.616E-03	1.734E-03	2.264E-03	4.545E-03

Table N-5 Continued

Isotope	Discharge	5 Years	30 Years	100 Years	500 Years
Pu238	4.754E-04	4.969E-04	4.079E-04	2.347E-04	9.999E-06
Pu239	1.407E-02	1.407E-02	1.406E-02	1.404E-02	1.389E-02
Pu240	4.706E-03	4.720E-03	4.756E-03	4.749E-03	4.554E-03
Pu241	5.482E-03	4.310E-03	1.294E-03	4.457E-05	9.722E-09
Pu242	1.898E-03	1.898E-03	1.898E-03	1.897E-03	1.896E-03
Pu244	6.926E-08	6.926E-08	6.926E-08	6.926E-08	6.926E-08
Am241	8.395E-05	1.251E-03	4.146E-03	4.856E-03	2.581E-03
Am242m	4.443E-07	4.343E-07	3.875E-07	2.816E-07	4.545E-08
Am243	4.505E-04	4.503E-04	4.492E-04	4.463E-04	4.298E-04
Cm242	4.195E-05	1.900E-08	9.405E-10	6.835E-10	1.103E-10
Cm243	8.003E-07	7.086E-07	3.858E-07	7.030E-08	4.187E-12
Cm244	9.707E-05	8.016E-05	3.078E-05	2.112E-06	4.729E-13
Cm245	6.065E-06	6.062E-06	6.050E-06	6.016E-06	5.823E-06
Cm246	2.997E-07	2.995E-07	2.984E-07	2.954E-07	2.786E-07
Cm247	2.746E-09	2.746E-09	2.746E-09	2.746E-09	2.746E-09
Cm248	1.388E-10	1.389E-10	1.389E-10	1.389E-10	1.388E-10
Cm250	1.337E-17	1.337E-17	1.335E-17	1.332E-17	1.310E-17
Bk249	1.373E-12	2.628E-14	6.759E-23	6.005E-47	2.197E-184
Cf249	9.992E-14	1.435E-12	1.391E-12	1.211E-12	5.492E-13
Cf250	5.495E-13	4.216E-13	1.121E-13	2.747E-15	1.382E-21
Cf251	2.448E-13	2.438E-13	2.392E-13	2.266E-13	1.664E-13
Cf252	9.025E-14	2.426E-14	3.405E-17	3.503E-25	7.949E-71

Appendix O
Isotopic Distribution of Discharged Thorium Based Fuel

Table O-1 Isotopic Distribution at the end of the first cycle

Years of Decay:	0	5	30	100	500
Isotope					
H3	1.242E-07	9.377E-08	2.304E-08	4.527E-10	8.000E-20
He4	0.000E+00	1.410E-05	5.364E-05	1.310E-04	3.224E-04
C14	4.896E-11	4.893E-11	4.878E-11	4.837E-11	4.609E-11
C-other	0.000E+00				
TM-other (Be, Li, Co-Se, Nb, Ag-Te)	2.018E-03				
Se79	9.967E-06	9.967E-06	9.964E-06	9.957E-06	9.914E-06
Kr81	5.927E-11	5.927E-11	5.926E-11	5.925E-11	5.917E-11
Kr85	2.952E-05	2.136E-05	4.242E-06	4.590E-08	2.685E-19
Inert gas other (Kr, Xe)	9.813E-03				
Rb	4.046E-04				
Sr90	5.438E-04	4.828E-04	2.663E-04	5.033E-05	3.692E-09
Sr-other	4.183E-04				
Zr93	9.739E-04	9.739E-04	9.739E-04	9.738E-04	9.736E-04
Zr95	8.077E-05	2.061E-13	2.228E-56	1.104E-176	0.000E+00
Zr-other	3.736E-03				
Mo-Ru-Rh-Pd-other	1.558E-02				
Tc99	1.506E-03	1.506E-03	1.506E-03	1.506E-03	1.504E-03
Tc-other	3.944E-07				
Ru106	4.439E-04	1.426E-05	4.877E-13	6.074E-34	2.129E-153
Pd107	1.047E-03	1.047E-03	1.047E-03	1.047E-03	1.047E-03
Cd113m	6.039E-07	4.761E-07	1.451E-07	5.211E-09	2.890E-17
Sn126	5.728E-05	5.728E-05	5.727E-05	5.724E-05	5.708E-05
Sb125	2.219E-05	6.349E-06	1.218E-08	3.005E-16	1.011E-59
I129	4.288E-04	4.288E-04	4.288E-04	4.288E-04	4.288E-04
Halogen-other (Br, I)	1.852E-04				
Cs134	1.124E-04	2.093E-05	4.696E-09	2.847E-19	1.175E-77
Cs135	1.548E-03	1.548E-03	1.548E-03	1.548E-03	1.548E-03
Cs137	2.309E-03	2.057E-03	1.154E-03	2.290E-04	2.218E-08
Cs-other	2.245E-03				
Ba	2.475E-03				
Lanthanide-other (including Y)	1.681E-02				
Ce144	4.432E-04	5.162E-06	1.106E-15	9.339E-43	1.841E-197
Pm147	3.329E-04	8.885E-05	1.203E-07	1.118E-15	1.419E-61
Sm146	6.724E-09	6.724E-09	6.724E-09	6.724E-09	6.724E-09
Sm147	2.178E-04	4.619E-04	5.506E-04	5.507E-04	5.507E-04
Sm151	5.332E-05	5.131E-05	4.232E-05	2.468E-05	1.133E-06
Eu154	5.083E-05	3.397E-05	4.529E-06	1.606E-08	1.601E-22
Eu155	2.784E-05	1.384E-05	4.204E-07	2.369E-11	1.242E-35
Ho166m	1.438E-09	1.434E-09	1.413E-09	1.357E-09	1.077E-09
Pb206	0.000E+00	1.157E-17	1.439E-14	1.727E-12	7.302E-10
Pb207	0.000E+00	1.198E-09	3.386E-08	2.215E-07	1.479E-06

Table O-1 Continued

Years of Decay:	0	5	30	100	500
Pb208	0.000E+00	1.592E-06	1.441E-05	3.802E-05	6.205E-05
Pb210	0.000E+00	2.903E-16	6.351E-14	2.277E-12	1.742E-10
Bi209	0.000E+00	7.145E-11	2.570E-09	2.849E-08	7.030E-07
Ra226	0.000E+00	4.557E-13	2.070E-11	3.402E-10	1.552E-08
Ra228	0.000E+00	1.571E-10	3.715E-10	3.890E-10	3.890E-10
Ac227	0.000E+00	1.608E-08	6.719E-08	1.046E-07	1.082E-07
Th228	0.000E+00	1.478E-06	1.403E-06	7.149E-07	1.530E-08
Th229	0.000E+00	3.315E-07	1.987E-06	6.599E-06	3.235E-05
Th230	0.000E+00	2.120E-08	1.755E-07	9.403E-07	8.664E-06
Th232	8.302E-01	8.302E-01	8.302E-01	8.302E-01	8.302E-01
Pa231	1.673E-04	1.673E-04	1.672E-04	1.670E-04	1.656E-04
U232	6.977E-05	6.649E-05	5.227E-05	2.664E-05	5.662E-07
U233	1.543E-02	1.543E-02	1.542E-02	1.542E-02	1.539E-02
U234	1.398E-03	1.645E-03	2.749E-03	4.901E-03	7.714E-03
U235	2.625E-04	2.654E-04	2.797E-04	3.198E-04	5.483E-04
U236	1.459E-05	2.854E-05	9.943E-05	3.009E-04	1.430E-03
U238	4.694E-19	9.701E-08	5.821E-07	1.940E-06	9.704E-06
Np237	3.475E-03	3.503E-03	3.883E-03	5.462E-03	1.220E-02
Pu238	6.212E-03	6.265E-03	5.147E-03	2.971E-03	1.319E-04
Pu239	2.025E-02	2.025E-02	2.025E-02	2.023E-02	2.009E-02
Pu240	2.664E-02	2.685E-02	2.744E-02	2.762E-02	2.650E-02
Pu241	1.458E-02	1.146E-02	3.442E-03	1.188E-04	2.778E-07
Pu242	1.101E-02	1.101E-02	1.101E-02	1.101E-02	1.101E-02
Pu244	2.082E-07	2.082E-07	2.082E-07	2.082E-07	2.082E-07
Am241	1.913E-03	5.003E-03	1.264E-02	1.436E-02	7.628E-03
Am242m	6.889E-05	6.733E-05	6.008E-05	4.366E-05	7.046E-06
Am243	2.586E-03	2.585E-03	2.579E-03	2.562E-03	2.468E-03
Cm242	3.077E-04	2.943E-07	1.458E-07	1.060E-07	1.710E-08
Cm243	1.505E-05	1.333E-05	7.257E-06	1.322E-06	7.876E-11
Cm244	1.316E-03	1.087E-03	4.174E-04	2.863E-05	6.413E-12
Cm245	1.733E-04	1.733E-04	1.729E-04	1.719E-04	1.664E-04
Cm246	1.075E-05	1.075E-05	1.071E-05	1.060E-05	9.995E-06
Cm247	1.951E-07	1.951E-07	1.951E-07	1.951E-07	1.951E-07
Cm248	9.808E-09	9.808E-09	9.807E-09	9.806E-09	9.798E-09
Cm250	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Bk249	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cf249	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cf250	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cf251	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cf252	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Total	1.000E+00	9.449E-01	9.437E-01	9.425E-01	9.422E-01

Table O-2 Isotopic Distribution at the end of the fourth cycle

Years of decay:	0	5	30	100	500
Isotope					
H3	1.308E-07	9.875E-08	2.427E-08	4.768E-10	8.426E-20
He4	0.000E+00	3.492E-05	1.303E-04	3.016E-04	6.624E-04
C14	5.240E-11	5.237E-11	5.221E-11	5.177E-11	4.932E-11
C-other	0.000E+00				
TM-other (Be, Li, Co-Se, Nb, Ag-Te)	2.114E-03				
Se79	1.142E-05	1.142E-05	1.142E-05	1.141E-05	1.136E-05
Kr81	4.542E-11	4.542E-11	4.541E-11	4.540E-11	4.534E-11
Kr85	3.478E-05	2.517E-05	4.999E-06	5.409E-08	3.164E-19
Inert gas other (Kr, Xe)	1.031E-02				
Rb	4.876E-04				
Sr90	6.365E-04	5.651E-04	3.117E-04	5.890E-05	4.320E-09
Sr-other	4.991E-04				
Zr93	1.073E-03	1.073E-03	1.073E-03	1.073E-03	1.073E-03
Zr95	8.570E-05	2.187E-13	2.364E-56	1.171E-176	0.000E+00
Zr-other	4.097E-03				
Mo-Ru-Rh-Pd-other	1.619E-02				
Tc99	1.619E-03	1.619E-03	1.618E-03	1.618E-03	1.616E-03
Tc-other	4.255E-07				
Ru106	4.774E-04	1.533E-05	5.244E-13	6.532E-34	2.289E-153
Pd107	1.127E-03	1.127E-03	1.127E-03	1.127E-03	1.127E-03
Cd113m	6.439E-07	5.077E-07	1.547E-07	5.557E-09	3.081E-17
Sn126	5.850E-05	5.850E-05	5.849E-05	5.846E-05	5.830E-05
Sb125	2.208E-05	6.319E-06	1.212E-08	2.991E-16	1.006E-59
I129	4.434E-04	4.434E-04	4.434E-04	4.434E-04	4.434E-04
Halogen-other (Br, I)	1.961E-04				
Cs134	8.440E-05	1.572E-05	3.528E-09	2.139E-19	8.824E-78
Cs135	1.821E-03	1.821E-03	1.821E-03	1.821E-03	1.821E-03
Cs137	2.469E-03	2.200E-03	1.235E-03	2.450E-04	2.372E-08
Cs-other	2.427E-03				
Ba	2.650E-03				
Lanthanide-other (including Y)	1.811E-02				
Ce144	4.815E-04	5.608E-06	1.201E-15	1.015E-42	2.000E-197
Pm147	4.055E-04	1.082E-04	1.465E-07	1.361E-15	1.728E-61
Sm146	5.320E-09	5.320E-09	5.320E-09	5.320E-09	5.320E-09
Sm147	2.568E-04	5.541E-04	6.621E-04	6.623E-04	6.623E-04
Sm151	7.843E-05	7.546E-05	6.224E-05	3.630E-05	1.666E-06
Eu154	3.747E-05	2.504E-05	3.339E-06	1.184E-08	1.180E-22
Eu155	3.106E-05	1.544E-05	4.690E-07	2.643E-11	1.386E-35
Ho166m	8.098E-10	8.074E-10	7.959E-10	7.643E-10	6.067E-10
Pb206	0.000E+00	4.693E-17	5.632E-14	6.220E-12	2.330E-09
Pb207	0.000E+00	1.191E-09	3.367E-08	2.203E-07	1.471E-06
Pb208	0.000E+00	2.103E-06	1.905E-05	5.024E-05	8.199E-05
Pb210	0.000E+00	1.187E-15	2.447E-13	7.968E-12	5.433E-10
Bi209	0.000E+00	1.172E-10	4.214E-09	4.672E-08	1.153E-06
Ra226	0.000E+00	1.855E-12	7.849E-11	1.173E-09	4.830E-08

Table O-2 Continued

Years of decay:	0	5	30	100	500
Ra228	0.000E+00	1.407E-10	3.327E-10	3.484E-10	3.484E-10
Ac227	0.000E+00	1.599E-08	6.681E-08	1.040E-07	1.076E-07
Th228	0.000E+00	1.953E-06	1.853E-06	9.447E-07	2.017E-08
Th229	0.000E+00	5.436E-07	3.257E-06	1.082E-05	5.305E-05
Th230	0.000E+00	8.555E-08	6.458E-07	3.129E-06	2.656E-05
Th232	7.434E-01	7.434E-01	7.434E-01	7.434E-01	7.434E-01
Pa231	1.664E-04	1.664E-04	1.663E-04	1.660E-04	1.646E-04
U232	9.220E-05	8.787E-05	6.907E-05	3.520E-05	7.482E-07
U233	2.530E-02	2.530E-02	2.529E-02	2.528E-02	2.524E-02
U234	5.801E-03	6.478E-03	9.510E-03	1.543E-02	2.320E-02
U235	2.357E-03	2.361E-03	2.381E-03	2.435E-03	2.747E-03
U236	4.645E-04	4.874E-04	6.048E-04	9.416E-04	2.833E-03
U238	4.625E-19	2.128E-07	1.277E-06	4.258E-06	2.130E-05
Np237	4.718E-03	4.786E-03	5.443E-03	7.931E-03	1.844E-02
Pu238	1.717E-02	1.720E-02	1.414E-02	8.176E-03	3.716E-04
Pu239	2.753E-02	2.754E-02	2.755E-02	2.755E-02	2.746E-02
Pu240	4.362E-02	4.416E-02	4.566E-02	4.626E-02	4.441E-02
Pu241	1.928E-02	1.516E-02	4.552E-03	1.574E-04	7.233E-07
Pu242	2.416E-02	2.416E-02	2.416E-02	2.417E-02	2.418E-02
Pu244	2.030E-07	2.030E-07	2.030E-07	2.030E-07	2.030E-07
Am241	6.553E-03	1.061E-02	2.055E-02	2.241E-02	1.190E-02
Am242m	2.917E-04	2.851E-04	2.544E-04	1.849E-04	2.983E-05
Am243	6.234E-03	6.231E-03	6.216E-03	6.176E-03	5.948E-03
Cm242	7.235E-04	9.983E-07	6.173E-07	4.486E-07	7.240E-08
Cm243	4.574E-05	4.050E-05	2.205E-05	4.018E-06	2.393E-10
Cm244	3.247E-03	2.681E-03	1.030E-03	7.063E-05	1.582E-11
Cm245	4.512E-04	4.511E-04	4.501E-04	4.476E-04	4.332E-04
Cm246	2.444E-05	2.442E-05	2.433E-05	2.409E-05	2.271E-05
Cm247	4.772E-07	4.772E-07	4.772E-07	4.772E-07	4.771E-07
Cm248	1.960E-08	1.960E-08	1.960E-08	1.960E-08	1.958E-08
Cm250	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Bk249	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cf249	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cf250	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cf251	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cf252	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Total	1.000E+00	9.414E-01	9.401E-01	9.388E-01	9.385E-01

Table O-3 Isotopic Distribution at the end of the eight cycle

Years of decay:	0	5	30	100	500
Isotope					
H3	1.328E-07	1.003E-07	2.465E-08	4.843E-10	8.559E-20
He4	0.000E+00	4.261E-05	1.630E-04	3.797E-04	8.238E-04
C14	5.317E-11	5.313E-11	5.297E-11	5.253E-11	5.004E-11
C-other	0.000E+00				
TM-other (Be, Li, Co-Se, Nb, Ag-Te)	2.141E-03				
Se79	1.118E-05	1.118E-05	1.118E-05	1.117E-05	1.113E-05
Kr81	4.352E-11	4.352E-11	4.352E-11	4.351E-11	4.345E-11
Kr85	3.411E-05	2.468E-05	4.902E-06	5.304E-08	3.103E-19
Inert gas other (Kr, Xe)	1.037E-02				
Rb	4.802E-04				
Sr90	6.317E-04	5.608E-04	3.093E-04	5.846E-05	4.288E-09
Sr-other	4.911E-04				
Zr93	1.078E-03	1.078E-03	1.078E-03	1.078E-03	1.077E-03
Zr95	8.603E-05	2.195E-13	2.373E-56	1.176E-176	0.000E+00
Zr-other	4.121E-03				
Mo-Ru-Rh-Pd-other	1.653E-02				
Tc99	1.655E-03	1.655E-03	1.655E-03	1.654E-03	1.652E-03
Tc-other	4.346E-07				
Ru106	4.931E-04	1.584E-05	5.418E-13	6.747E-34	2.365E-153
Pd107	1.162E-03	1.162E-03	1.162E-03	1.162E-03	1.162E-03
Cd113m	6.534E-07	5.152E-07	1.570E-07	5.638E-09	3.127E-17
Sn126	5.833E-05	5.833E-05	5.832E-05	5.829E-05	5.813E-05
Sb125	2.205E-05	6.308E-06	1.210E-08	2.986E-16	1.004E-59
I129	4.455E-04	4.455E-04	4.455E-04	4.455E-04	4.455E-04
Halogen-other (Br, I)	1.959E-04				
Cs134	7.498E-05	1.397E-05	3.134E-09	1.900E-19	7.840E-78
Cs135	1.930E-03	1.930E-03	1.930E-03	1.930E-03	1.930E-03
Cs137	2.503E-03	2.230E-03	1.252E-03	2.483E-04	2.404E-08
Cs-other	2.481E-03				
Ba	2.687E-03				
Lanthanide-other (including Y)	1.833E-02				
Ce144	4.872E-04	5.674E-06	1.216E-15	1.027E-42	2.023E-197
Pm147	4.293E-04	1.146E-04	1.551E-07	1.441E-15	1.830E-61
Sm146	4.810E-09	4.810E-09	4.810E-09	4.810E-09	4.810E-09
Sm147	2.702E-04	5.849E-04	6.993E-04	6.995E-04	6.995E-04
Sm151	9.083E-05	8.740E-05	7.209E-05	4.204E-05	1.930E-06
Eu154	3.331E-05	2.226E-05	2.968E-06	1.053E-08	1.049E-22
Eu155	3.324E-05	1.653E-05	5.019E-07	2.829E-11	1.484E-35
Ho166m	6.833E-10	6.813E-10	6.715E-10	6.449E-10	5.119E-10
Pb206	0.000E+00	7.095E-17	8.365E-14	9.053E-12	3.278E-09
Pb207	0.000E+00	1.168E-09	3.301E-08	2.159E-07	1.442E-06
Pb208	0.000E+00	1.931E-06	1.749E-05	4.613E-05	7.528E-05
Pb210	0.000E+00	1.781E-15	3.619E-13	1.150E-11	7.573E-10
Bi209	0.000E+00	1.230E-10	4.425E-09	4.905E-08	1.210E-06

Years of decay:	0	5	30	100	500
Ra226	0.000E+00	2.780E-12	1.157E-10	1.684E-09	6.727E-08
Ra228	0.000E+00	1.323E-10	3.128E-10	3.275E-10	3.275E-10
Ac227	0.000E+00	1.567E-08	6.550E-08	1.019E-07	1.054E-07
Th228	0.000E+00	1.793E-06	1.702E-06	8.673E-07	1.852E-08
Th229	0.000E+00	5.708E-07	3.420E-06	1.136E-05	5.570E-05
Th230	0.000E+00	1.279E-07	9.443E-07	4.450E-06	3.681E-05
Th232	6.990E-01	6.990E-01	6.990E-01	6.990E-01	6.990E-01
Pa231	1.631E-04	1.631E-04	1.630E-04	1.628E-04	1.614E-04
U232	8.465E-05	8.067E-05	6.342E-05	3.232E-05	6.869E-07
U233	2.656E-02	2.656E-02	2.656E-02	2.655E-02	2.651E-02
U234	8.730E-03	9.634E-03	1.368E-02	2.157E-02	3.195E-02
U235	4.053E-03	4.057E-03	4.081E-03	4.149E-03	4.534E-03
U236	1.231E-03	1.259E-03	1.403E-03	1.815E-03	4.129E-03
U238	4.590E-19	2.748E-07	1.649E-06	5.497E-06	2.750E-05
Np237	5.497E-03	5.585E-03	6.370E-03	9.262E-03	2.144E-02
Pu238	2.304E-02	2.293E-02	1.886E-02	1.091E-02	4.979E-04
Pu239	3.401E-02	3.402E-02	3.403E-02	3.403E-02	3.393E-02
Pu240	5.350E-02	5.413E-02	5.590E-02	5.660E-02	5.433E-02
Pu241	2.104E-02	1.654E-02	4.968E-03	1.719E-04	8.116E-07
Pu242	3.119E-02	3.119E-02	3.120E-02	3.121E-02	3.123E-02
Pu244	1.977E-07	1.977E-07	1.977E-07	1.977E-07	1.977E-07
Am241	8.926E-03	1.334E-02	2.412E-02	2.597E-02	1.378E-02
Am242m	4.151E-04	4.057E-04	3.620E-04	2.631E-04	4.246E-05
Am243	7.977E-03	7.974E-03	7.955E-03	7.903E-03	7.611E-03
Cm242	8.227E-04	1.332E-06	8.786E-07	6.385E-07	1.030E-07
Cm243	5.283E-05	4.678E-05	2.547E-05	4.641E-06	2.764E-10
Cm244	3.839E-03	3.170E-03	1.217E-03	8.351E-05	1.870E-11
Cm245	5.063E-04	5.061E-04	5.051E-04	5.022E-04	4.861E-04
Cm246	2.411E-05	2.409E-05	2.400E-05	2.376E-05	2.241E-05
Cm247	4.521E-07	4.521E-07	4.521E-07	4.521E-07	4.521E-07
Cm248	1.609E-08	1.609E-08	1.609E-08	1.609E-08	1.608E-08
Cm250	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Bk249	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cf249	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cf250	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cf251	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Cf252	0.000E+00	0.000E+00	0.000E+00	0.000E+00	0.000E+00
Total	1.000E+00	9.407E-01	9.393E-01	9.380E-01	9.377E-01

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