# OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
# DESIGN CALCULATION OR ANALYSIS COVER SHEET

1. QA: QA
2. Page: 1 of: 365

| 3. System: | 4. Document Identifier: | 5. Verified: |
|---|---|---|
| DOE SNF Disposal Container | CAL-EDC-NU-000008 | ☑ Yes ☐ No |

| 6. Title: | 7. Group: |
|---|---|
| Benchmark and Critical Limit Calculation for DOE SNF | Risk and Criticality |

8. Document Status Designation:
☐ Preliminary ☑ Final ☐ Superseded ☐ Voided

9. Notes/Comments:
Attachment I is a compact disc attachment
Attachment II contains a hard-copy listing of the files and directory structure contained on Attachment I (2 pages)

Checker responsibilities were delineated as follows:

J.M. Scaglione checked the following portions of the report: Sections 5.7, 5.8, 5.9, 5.10, 6.0, 6.1, 6.2, 7, 8, the DIRs report, Attachment II, and the Pu&Mix and U233 folders of Attachment I

S.F. Deng checked the following portions: Section 6.3

H.R. Radulescu checked the following portions of the report: Sections 1, 2, 3, 4, 5.0, 5.1, 5.2, 5.3, 5.4, 5.5, 5.6, and the HEU and IEU&LEU folders of Attachment I

## RECORD OF REVISIONS

| 10. No. | 11. Reason for Revision | 12. Total No. of Pages | 13. Last Page No. | 14. Originator | 15. Checker | 16. Quality Engineering Representative | 17. Approved/ Accepted | 18. Date |
|---|---|---|---|---|---|---|---|---|
| 00 | Initial Issue | 367 | II-2 | M.S. 03 for MS 3/29/02 | JMS 3/29/02 SFD 03/29/02 HRR 03/29/2002 | for DT 3/29/02 | 03 3/29/02 | 03/29/2002 |

## CONTENTS

**Page**

1. PURPOSE ..... 15

2. METHOD ..... 15

3. ASSUMPTIONS ..... 16

4. USE OF COMPUTER SOFTWARE AND MODELS ..... 16
   4.1 SOFTWARE ..... 16
      4.1.1 MCNP ..... 16

5. CALCULATION ..... 17
   5.1 PREVIOUSLY EVALUATED HEU BENCHMARK CASES ..... 18
      5.1.1 HEU Benchmarks Related to Intact Waste Package Configurations ..... 18
      5.1.2 HEU Benchmarks Related to Degraded Waste Package Configurations ..... 35
   5.2 HEU BENCHMARK CASES ..... 48
      5.2.1 HEU Benchmarks Related to Intact Waste Package Configurations ..... 48
      5.2.2 HEU Benchmarks Related to Degraded Waste Package Configurations ..... 73
   5.3 PREVIOUSLY EVALUATED LEU BENCHMARK CASES ..... 102
      5.3.1 LEU Benchmarks Related to intact Waste Package Configurations ..... 102
      5.3.2 Benchmarks Related to Degraded Waste Package Configurations ..... 110
   5.4 LEU BENCHMARK CASES ..... 113
      5.4.1 LEU Benchmarks Related to intact Waste Package Configurations ..... 113
      5.4.2 LEU Benchmarks related to Degraded Waste Package Configurations ..... 116
   5.5 PREVIOUSLY EVALUATED IEU BENCHMARK CASES ..... 119
      5.5.1 IEU Benchmarks Related to Intact Waste Package Configurations ..... 119
      5.5.2 IEU Benchmarks Related to Degraded Waste Package Configurations ..... 119
   5.6 IEU BENCHMARK CASES ..... 131
      5.6.1 IEU Benchmarks Related to Intact Waste Package Configurations ..... 131
      5.6.2 IEU Benchmarks Related to Degraded Waste Package Configurations ..... 133
   5.7 PREVIOUSLY EVALUATED U-233 BENCHMARK CASES ..... 140
      5.7.1 U-233 Benchmarks related to Intact Waste Package Configurations ..... 140
      5.7.2 U-233 Benchmarks related to degraded Waste Package Configurations ..... 143
   5.8 U-233 BENCHMARK CASES ..... 143
      5.8.1 U-233 Benchmarks related to Intact Waste Package Configurations ..... 143
      5.8.2 U-233 Benchmarks related to degraded Waste Package Configurations ..... 143
   5.9 PREVIOUSLY EVALUATED PU AND MIXED BENCHMARK CASES ..... 154
      5.9.1 Pu and Mixed Benchmarks Related to Intact Waste Package Configurations. 154
      5.9.2 Pu and Mixed Benchmarks Related to Degraded Waste Package Configurations ..... 159
   5.10 PU AND MIXED BENCHMARK CASES ..... 172

**CONTENTS (Continued)**

**Page**

5.10.1 Pu and Mixed Benchmarks Related to Intact Waste Package Configurations. 172
5.10.2 Pu and Mixed Benchmarks Related to Degraded Waste Package Configurations ..... 187

6. BENCHMARK SELECTION ..... 231
6.1 IDENTIFICATION OF BENCHMARKS APPLICABLE TO THE DOE-OWNED SNF GROUPS ..... 231
6.1.1 HEU Oxide (Shippingport-PWR) DOE-Owned Fuel ..... 231
6.1.2 UZrH (Training, Research, Isotopes, General Atomics [TRIGA]) DOE-Owned Fuel ..... 233
6.1.3 Melt and Dilute Waste Form ..... 235
6.1.4 U Alloy (Enrico Fermi) DOE-Owned Fuel ..... 236
6.1.5 MOX (FFTF) DOE-Owned Fuel ..... 238
6.1.6 Graphite/Carbide (Fort Saint Vrain HTGR) DOE-Owned Fuel ..... 240
6.1.7 Th/U (Shippingport) LWBR DOE-Owned SNF ..... 242
6.1.8 U-metal (N Reactor) DOE-Owned Fuel ..... 244
6.1.9 LEU Oxide (Three Mile Island [TMI]) Core Debris Waste Form ..... 245
6.2 RANGE OF APPLICABILITY OF BENCHMARK CASES ..... 246
6.3 ESTIMATE AND PLOTS OF BIAS FOR DOE SNF TYPES ..... 341
6.3.1 Plots of bias for the DOE SNF Groupings in Section 6.2 ..... 346

7. REFERENCES ..... 363
7.1 DOCUMENTS CITED ..... 363
7.2 CODES, STANDARDS, REGULATIONS, AND PROCEDURES ..... 364

8. ATTACHMENTS ..... 365

## TABLES

Page

Table 5-1 SPERT-D Fuel Element Critical Experiments ..... 19
Table 5-2. LCE Reactivity Calculation Results for ..... 20
Table 5-3 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for HEU-MET-THERM-006 Experiments ..... 21
Table 5-4 AENCF for HEU-MET-THERM-006 Experiments ..... 22
Table 5-5 Cross-Shaped Fuel Rods Critical Experiments ..... 24
Table 5-6 LCE Reactivity Calculation Results for HEU-COMP-THERM -003, 004, 005, 006, 007, and 008 Experiments ..... 26
Table 5-7 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for HEU-COMP-THERM -003, 004, 005 006, 007, and 008 Experiments ..... 27
Table 5-8 AENCF For HEU-COMP-THERM -003, 004, 005, 006, 007, and 008 Experiments (MeV) ..... 28
Table 5-9 Configurations of HEU-COMP-THERM-010 Experiments ..... 30
Table 5-10 LCE Reactivity Calculation Results for HCT-010 Experiments ..... 31
Table 5-11 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for HEU-COMP-THERM-010 Experiments ..... 32
Table 5-12 AENCF for HEU-COMP-THERM-010 Experiments (MeV) ..... 32
Table 5-13 Benchmarks for HEU-COMP-INTER -002 Class of Experiments ..... 33
Table 5-14 Criticality Calculation Results for HEU-COMP-INTER -002 Experiments ..... 34
Table 5-15 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for HEU-COMP-INTER -002 Experiments ..... 35
Table 5-16 AENCF for HEU-COMP-INTER -002 Experiments ..... 35
Table 5-17 Benchmark Problem Summary for Configurations Incorporating Highly Enriched Uranium Nitrate Solutions ..... 35
Table 5-18 Results of Criticality Calculations for HEU Solutions ..... 41
Table 5-19 Russian Criticality Safety Benchmark Experiments ..... 46
Table 5-20 Criticality Calculation Results for HEU-METAL-FAST Experiments ..... 46
Table 5-21 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for HEU-METAL-FAST Experiments ..... 47
Table 5-22 AENCF for HEU-METAL-FAST Experiments (MeV) ..... 47
Table 5-23 HEU-MET-FAST-007 Critical Parameters ..... 48
Table 5-24 Results for HEU-MET-FAST-007 ..... 50
Table 5-25 Results for HEU-MET-FAST-030 ..... 52
Table 5-26 Results for HEU-MET-FAST-033 ..... 52
Table 5-27 Results for HEU-MET-FAST-034 ..... 53
Table 5-28 Results for HEU-MET-FAST-036 ..... 54
Table 5-29 Results for HEU-MET-FAST-037 ..... 55
Table 5-30 Results for HEU-MET-FAST-038 ..... 55
Table 5-31 Results for HEU-MET-INTER-006 ..... 56
Table 5-32 Results for HEU-MET-MIXED-001 ..... 56
Table 5-33 Results for HEU-MET-MIXED-004 ..... 57
Table 5-34 Results for HEU-COMP-INTER-003 ..... 58

**TABLES (Continued)**

**Page**

Table 5-35 Results for HEU-MET-FAST-023 ..... 58
Table 5-36 Results for HEU-MET-FAST-026 ..... 60
Table 5-37 Results for HEU-COMP-MIXED-001 ..... 66
Table 5-38 Results for HEU-COMP-MIXED-002 ..... 67
Table 5-39 Results for HEU-COMP-INTER-001 ..... 68
Table 5-40 Results for HEU-COMP-THERM-002 ..... 69
Table 5-41 Results for HEU-COMP-THERM-011, HEU-COMP-THERM-012, HEU-COMP-THERM-013, HEU-COMP-THERM-014 ..... 73
Table 5-42 Results for HEU-SOL-THERM-005 ..... 74
Table 5-43 Results for HEU-SOL-THERM-006 ..... 75
Table 5-44 Results for HEU-SOL-THERM-009 ..... 76
Table 5-45 Results for HEU-SOL-THERM-010 ..... 76
Table 5-46 Results for HEU-SOL-THERM-011 ..... 77
Table 5-47 MCNP Results for HEU-SOL-THERM-012 ..... 78
Table 5-48 Results for HEU-SOL-THERM-021 ..... 79
Table 5-49 Results for HEU-SOL-THERM-025 ..... 80
Table 5-50 Results for HEU-SOL-THERM-027 ..... 82
Table 5-51 Results for HEU-SOL-THERM-028 ..... 83
Table 5-52 Results for HEU-SOL-THERM-029 ..... 84
Table 5-53 Results for HEU-SOL-THERM-030 ..... 84
Table 5-54 Results for HEU-SOL-THERM-031 ..... 85
Table 5-55 Results for HEU-SOL-THERM-031 ..... 85
Table 5-56 Results for HEU-SOL-THERM-033 ..... 86
Table 5-57 Results for HEU-SOL-THERM-035 ..... 87
Table 5-58 Results for HEU-SOL-THERM-036 ..... 88
Table 5-59 Results for HEU-SOL-THERM-037 ..... 88
Table 5-60 Results for HEU-SOL-THERM-043 ..... 89
Table 5-61 Results for HEU-SOL-THERM-044 ..... 91
Table 5-62 Critical Mass / Density Comparisons for HEU-MET-FAST-002 ..... 92
Table 5-63 Results for HEU-MET-FAST-002 ..... 92
Table 5-64 Results for HEU-MET-FAST-004 ..... 93
Table 5-65 Results for HEU-MET-FAST-027 ..... 94
Table 5-66 Results for HEU-MET-FAST-031 ..... 96
Table 5-67 Results for HEU-MET-MIXED-002 ..... 96
Table 5-68 Results for HEU-MET-MIXED-003 ..... 97
Table 5-69 Results for HEU-MET-FAST-032 ..... 97
Table 5-70 Results for HEU-MET-FAST-041 ..... 98
Table 5-71 Results for HEU-MET-FAST-055 ..... 99
Table 5-72 Results for HEU-MET-THERM-003 ..... 100
Table 5-73 Results for HEU-MET-MIXED-005 ..... 102

**TABLES (Continued)**

**Page**

Table 5-74 Water-Reflected Fuel Rod Cluster Critical Experiments ..... 102
Table 5-75 LCE Reactivity Calculation Results for LEU-COMP-THERM-001 Experiments .. 103
Table 5-76 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for LEU-COMP-THERM -001 Experiments ..... 103
Table 5-77 AENCF for LEU-COMP-THERM –001 Experiments (MeV) ..... 103
Table 5-78 Differences in Absorber Plates used for Clusters of 2.35 wt.% $UO_2$ Fuel Rods ..... 104
Table 5-79 Results for Critical Experiments with Clusters of 2.35 wt.% $UO_2$ Fuel Rods ..... 105
Table 5-80 Differences in Experimental Configurations for Clusters of 4.31 wt% $UO_2$ Fuel Rods ..... 105
Table 5-81 Results for Critical Experiments with Clusters of 4.31 wt% $UO_2$ Fuel Rods ..... 105
Table 5-82 Results for LEU-COMP-THERM-027 ..... 106
Table 5-83 Results for Critical Configurations with 4.31 wt% $^{235}U$ Enriched $UO_2$ Rods in Highly Borated Water Lattices ..... 106
Table 5-84 Urania-Gadolinia Critical Experiment Descriptions ..... 107
Table 5-85 Results for Urania-Gadolinia Critical Experiments ..... 107
Table 5-86 MKIA Fuel Assembly Experiments (CRWMS M&O [1999a]) ..... 108
Table 5-87 MKIA Fuel Element Parameters ..... 108
Table 5-88 Criticality Calculation Results for MKIA Experiments ..... 109
Table 5-89 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for MKIA Experiments 109
Table 5-90 AENCF for MKIA Experiments (MeV) ..... 109
Table 5-91 $UO_3$-$H_2O$ Solution Experiments ..... 110
Table 5-92 Criticality Calculation Results for $UO_3$-$H_2O$ Solution Experiments ..... 110
Table 5-93 Δk Between Experimental $k_\infty$ and MCNP Calculated $k_\infty$ Results for $UO_3$-$H_2O$ Solution Experiments ..... 111
Table 5-94 AENCF for $UO_3$-$H_2O$ Solution Experiments (MeV) ..... 111
Table 5-95 Benchmark Problem Summary for Configurations Incorporating Low-Enrichment Uranium Solutions ..... 112
Table 5-96 Homogenous Critical Experiments Using Low Enriched Uranium Nitrate Solutions (Thermal) ..... 112
Table 5-97 Results for Homogenous Critical Experiments Using Low Enriched Uranyl Fluoride Solutions ..... 113
Table 5-98 Critical Loadings of the Driver Core for the Investigated Storage Configurations.. 114
Table 5-99 Results for LEU-COMP-THERM-041 ..... 115
Table 5-100 Critical Configurations for LEU-COMP-THERM-042 ..... 115
Table 5-101 Results for LEU-COMP-THERM-042 ..... 116
Table 5-102 Critical Separation Distance For LEU-COMP-THERM-049 ..... 117
Table 5-103 Results for LEU-COMP-THERM-049 ..... 118
Table 5-104 Benchmark $k_{eff} \pm \sigma$ values for IMF-001 Experiments ..... 119
Table 5-105 Criticality Calculation Results for IEU-MET-FAST-001 Experiments ..... 120

**TABLES (Continued)**

**Page**

Table 5-106 Δk Between Experimental $k_{eff}$ Results and MCNP Calculated $k_{eff}$ Results for the IEU-MET-FAST -001 Experiments ..... 120
Table 5-107 AENCF for the IEU-MET-FAST -001 Experiments (MeV) ..... 120
Table 5-108 Criticality Calculation Results for the IEU-MET-FAST-002 Experiment ..... 121
Table 5-109 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for IEU-MET-FAST -002 Experiment ..... 121
Table 5-110. AENCF for IEU-MET-FAST -002 Experiment (MeV) ..... 121
Table 5-111 Experimental Description for IEU-MET-FAST-003, 004, 005, 006 and 008 ..... 122
Table 5-112 Criticality Calculation Results for IEU-MET-FAST-003, 004, 005, 006, and 008 Experiments ..... 122
Table 5-113 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ Results for IEU-MET-FAST-003, 004, 005, 006, and 008 Experiments ..... 123
Table 5-114 AENCF for IEU-MET-FAST-003, 004, 005, 006, and 008 Experiments (MeV) ..... 123
Table 5-115 Configurations of IEU-SOL-THERM-001 Experiments ..... 124
Table 5-116 LCE Calculation Results for IEU-SOL-THERM -001 Experiments ..... 124
Table 5-117 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for IEU-SOL-THERM -001 Experiments ..... 125
Table 5-118 AENCF for IEU-SOL-THERM -001 Experiments (MeV) ..... 125
Table 5-119 LEU-SOL-THERM-003 Characteristics of Benchmark Models. ..... 125
Table 5-120 Criticality Calculations for LEU-SOL-THERM -003 Experiments ..... 126
Table 5-121 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ Results for LEU-SOL-THERM -003 Experiments ..... 126
Table 5-122 AENCF for LEU-SOL-THERM -003 Experiments (MeV) ..... 127
Table 5-123 Critical Core Configurations With One-Inch Cubes and Paraffin Reflector ..... 127
Table 5-124 Critical Configurations for Cores with Half-Cubes of Polyethylene, with Aluminum Spacers, or with Reflector Modifications ..... 128
Table 5-125 Critical Parameters for IEU-COMP-THERM-001 ..... 129
Table 5-126 Results of LCE Reactivity Calculations for IEU-COMP-THERM-001 ..... 130
Table 5-127 Results for IEU-COMP-THERM-002 ..... 132
Table 5-128 Results for IEU-COMP-THERM-003 ..... 133
Table 5-129 Results for IEU-MET-FAST-007 ..... 133
Table 5-130 Standard Test Assembly: Characteristics of Layers ..... 134
Table 5-131 Results for IEU-MET-FAST-009 ..... 134
Table 5-132 Results for IEU-MET-FAST-010 ..... 135
Table 5-133 Results for Stacy Experiments ..... 137
Table 5-134 Results for LEU-SOL-THERM-005 ..... 139
Table 5-135 Results for LEU-SOL-THERM-006 ..... 140
Table 5-136 Characteristics of U233-MET-FAST Experiments ..... 142
Table 5-137 Results of Criticality Calculations for U233-MET-FAST ..... 142

**TABLES (Continued)**

**Page**

Table 5-138 Summary of Material Data for Critical Conditions of 27.24-inch-diameter Spheres ..... 144
Table 5-139 Results for U233-SOL-THERM-001 ..... 144
Table 5-140 Simplified Model Dimensions ..... 145
Table 5-141 Benchmark-Model Solution Densities and Heights ..... 145
Table 5-142 Results for U233-SOL-THERM-002 ..... 146
Table 5-143 Solution Densities and Simplified Model Dimensions ..... 147
Table 5-144 Results for U233-SOL-THERM-003 ..... 147
Table 5-145 Benchmark-Model Solution Densities and Heights ..... 148
Table 5-146 Results for U233-SOL-THERM-004 ..... 149
Table 5-147 Results for U233-SOL-THERM-005 ..... 150
Table 5-148 Summary of Material Data for Critical Condition of the 48.04-inch-diameter Sphere ..... 150
Table 5-149 Results for U233-SOL-THERM-008 ..... 150
Table 5-150 Material Data for U-233-SOL-INTER-001 ..... 151
Table 5-151 Results for U233-SOL-INTER-001 ..... 152
Table 5-152 Results for U233-SOL-THERM-006 ..... 154
Table 5-153 FFTF Array Critical Experiments ..... 155
Table 5-154 LCE Reactivity Calculation Results for FFTF Experiments ..... 155
Table 5-155 $\Delta k$ Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for FFTF Experiments ..... 155
Table 5-156 AENCF for FFTF Experiments (MeV) ..... 155
Table 5-157 Saxton Critical Experiment Descriptions ..... 156
Table 5-158 Criticality Results for the Saxton Experiments ..... 157
Table 5-159 $\Delta k$ Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for Saxton Experiments ..... 158
Table 5-160 AENCF for Saxton Experiments (MeV) ..... 158
Table 5-161 Russian Pu Fast Metal Benchmark Experiments ..... 159
Table 5-162 Criticality Calculation Results for PMF Experiments ..... 160
Table 5-163 $\Delta k$ Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for PMF Experiments .. 160
Table 5-164 AENCF for PMF Experiments (MeV) ..... 161
Table 5-165 Benchmark Problem Summary for Configurations Incorporating Mixed Plutonium and Natural Uranium Nitrate Solutions ..... 161
Table 5-166 Benchmark Problem Summary for Configurations Incorporating Plutonium Nitrate Solutions ..... 164
Table 5-167 Results of MCNP Criticality Calculations ..... 168
Table 5-168 MIX-COMP-THERM-004 Core Dimensions of Benchmark Models ..... 173
Table 5-169 Results for MIX-COMP-THERM-004 ..... 173
Table 5-170 MIX-COMP-THERM-010 Criticality Measurements with Plutonium-Uranium Nitrate Solution in the Boiler Tube-Type Tank ..... 174

**TABLES (Continued)**

**Page**

Table 5-171 Results for MIX-COMP-THERM-010 ..... 175
Table 5-172 PU-MET-FAST-003 Summary of Critical Array Spacing ..... 176
Table 5-173. Results for PU-MET-FAST-003 ..... 176
Table 5-174 PU-MET-FAST-004 Experimental Center-to-Center (CTC) Critical Spacing ..... 177
Table 5-175 Results for PU-MET-FAST-004 ..... 177
Table 5-176 PU-MET-FAST-016 Center-to-Center Spacing and Critical Water Heights ..... 178
Table 5-177 Results for PU-MET-FAST-016 ..... 178
Table 5-178 PU-MET-FAST-017 Experimental Center-to-Center (CTC) Critical Spacing ..... 179
Table 5-179 PU-MET-FAST-017 Moderator Dimensions (cm) ..... 180
Table 5-180 Results for PU-MET-FAST-017 ..... 180
Table 5-181 PU-MET-FAST-037 Experimental Arrays ..... 181
Table 5-182 Results for PU-MET-FAST-037 ..... 182
Table 5-183 PU-COMP-MIXED-001 Experiment Parameters ..... 183
Table 5-184 Results for PU-COMP-MIXED-001 ..... 183
Table 5-185 PU-COMP-MIXED-002 Experiment Characteristics ..... 184
Table 5-186 Results for PU-COMP-MIXED-002 ..... 185
Table 5-187 Principal Mass Loadings in the ZPPR-21 Cores, kg ..... 187
Table 5-188 Results for MIX-MET-FAST-011 ..... 187
Table 5-189 MIX-SOL-THERM-005 Slab Thickness and Critical Heights ..... 188
Table 5-190 Results for MIX-SOL-THERM-005 ..... 188
Table 5-191 PU-SOL-THERM-002 Evaluated Case Data for 12-Inch Diameter, 15.19-Liter Spheres ..... 189
Table 5-192 Results for PU-SOL-THERM-002 ..... 189
Table 5-193 PU-SOL-THERM-006 Data for experiments with fully reflected 15-inch solution spheres ..... 190
Table 5-194 Results for PU-SOL-THERM-006 ..... 190
Table 5-195 PU-SOL-THERM-008 Data for Concrete-Reflected 14-inch Spheres ..... 191
Table 5-196 Results for PU-SOL-THERM-008 ..... 192
Table 5-197 PU-SOL-THERM-012 Configuration Descriptions for Benchmarks ..... 192
Table 5-198 Results for PU-SOL-THERM-012 ..... 193
Table 5-199 PU-SOL-THERM-013 Cylindrical Tank Separations of the Various Experimental Configurations and their Critical Heights ..... 195
Table 5-200 Results for PU-SOL-THERM-013 ..... 196
Table 5-201 PU-SOL-THERM-014 Cylindrical Tank Separations of the Various Experimental Configurations and their Critical Heights ..... 197
Table 5-202 Results for PU-SOL-THERM-014 ..... 198
Table 5-203 PU-SOL-THERM-015 Cylindrical Tank Separations (±0.1 cm ) of the Various Experimental Configurations and their Critical Heights ..... 200
Table 5-204 Results for PU-SOL-THERM-015 ..... 201

**TABLES (Continued)**

**Page**

Table 5-205 PU-SOL-THERM-016 Cylindrical Tank Separations of the Experimental Configurations and their Critical Heights ..... 202
Table 5-206 Results for PU-SOL-THERM-016 ..... 203
Table 5-207 PU-SOL-THERM-017 Cylindrical Tank Separations (±0.1 cm) of the Experimental Configurations and their Critical Heights ..... 204
Table 5-208 MCNP Results for PU-SOL-THERM-017 ..... 204
Table 5-209 PU-SOL-THERM-020 Data for Water-Reflected 14-Inch Spheres. ..... 205
Table 5-210 Results for PU-SOL-THERM-020 ..... 206
Table 5-211 PU-SOL-THERM-021 Geometry of Solutions for PU-SOL-THERM-021 ..... 207
Table 5-212 Results for PU-SOL-THERM-021 ..... 207
Table 5-213 PU-SOL-THERM-022 Characteristics of Configurations ..... 208
Table 5-214 Results for PU-SOL-THERM-022 ..... 208
Table 5-215 PU-SOL-THERM-023 Experimental Results: Inner Plutonium Nitrate Solution, $^{240}$Pu/ Pu = 33.89%. ..... 209
Table 5-216 PU-SOL-THERM-023 ..... 210
Table 5-217 PU-SOL-THERM-024 Critical Experiment Characteristics ..... 210
Table 5-218 Results for PU-SOL-THERM-024 ..... 211
Table 5-219 PU-SOL-THERM-025 Critical Experiment Characteristics ..... 212
Table 5-220 Results for PU-SOL-THERM-025 ..... 214
Table 5-221 PU-SOL-THERM-026 Critical Experiment Characteristics ..... 216
Table 5-222 Results for PU-SOL-THERM-026 ..... 216
Table 5-223 Results for MIX-MET-MIXED-001 ..... 218
Table 5-224 Results for MIX-MET-FAST-001 and -002 ..... 219
Table 5-225 Results for Russian VNIITF CTF Spherical MIX-MET-FAST Benchmarks ..... 221
Table 5-226 Results for Los Alamos Pu Spherical Benchmarks ..... 226
Table 5-227 Results for Russian VNIIEF CTF Pu Spherical Benchmarks ..... 228
Table 5-228 Results for PU-MET-MIXED-001 ..... 230
Table 6-1 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport PWR SNF (thermal spectrum) ..... 247
Table 6-2 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport PWR SNF (intermediate and fast spectrum) ..... 249
Table 6-3 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport PWR SNF ..... 251
Table 6-4 Area and Range of Applicability for Benchmark experiments for WP Configurations Containing Degraded Shippingport PWR SNF (intermediate and fast spectrum) ..... 255
Table 6-5 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact UZrH (TRIGA) SNF ..... 260
Table 6-6 Area and Range of Applicability of Benchmark experiments for WP Configurations Containing Intact UZrH(TRIGA) SNF (intermediate and fast spectrum) ..... 263

**TABLES (Continued)**

**Page**

Table 6-7 Area and Range of Applicability of Benchmark experiments for WP Configurations Containing Degraded UZrH(TRIGA) SNF (thermal spectrum) ..... 265
Table 6-8 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded UZrH (TRIGA) SNF (intermediate and fast spectrum) ..... 271
Table 6-9 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Melt and Dilute Waste Form (thermal spectrum) ..... 276
Table 6-10 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Melt and Dilute Waste Form (intermediate and fast spectrum) ... 277
Table 6-11 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Melt and Dilute Waste Form (thermal spectrum) ..... 279
Table 6-12 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Melt and Dilute Waste Form (intermediate and fast spectrum)283
Table 6-13 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Enrico Fermi SNF (thermal spectrum) ..... 285
Table 6-14 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Enrico Fermi Waste Form (intermediate and fast spectrum) ..... 287
Table 6-15 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Enrico Fermi SNF (thermal spectrum) ..... 289
Table 6-16 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Enrico Fermi SNF Waste Form (intermediate and fast spectrum) ..... 292
Table 6-17 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact FFTF SNF (thermal spectrum) ..... 295
Table 6-18 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact FFTF SNF (intermediate and fast spectrum) ..... 297
Table 6-19 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (thermal spectrum) ..... 300
Table 6-20 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (intermediate and fast spectrum) ..... 306
Table 6-21 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Fort Saint Vrain HTGR SNF (thermal spectrum) ..... 313
Table 6-22 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Fort Saint Vrain HTGR SNF (intermediate and fast spectrum) .. 314
Table 6-23 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Fort Saint Vrain HTGR SNF (thermal spectrum) ..... 316
Table 6-24 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Fort Saint Vrain HTGR SNF (intermediate and fast spectrum) ..... 319
Table 6-25 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport LWBR SNF (thermal spectrum) ..... 323

**TABLES (Continued)**

**Page**

Table 6-26 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport LWBR SNF (intermediate and fast spectrum) ..... 325
Table 6-27 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport LWBR SNF (thermal spectrum) ..... 327
Table 6-28 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport LWBR SNF (intermediate and fast spectrum). 329
Table 6-29 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact N-Reactor SNF (thermal and intermediate spectrum)..... 334
Table 6-30 Area and Range of Applicability Benchmark Experiments for WP Configurations Containing Degraded N-Reactor SNF (thermal and intermediate spectrum) ..... 335
Table 6-31 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact TMI Waste Form (thermal and intermediate spectrum)..... 336
Table 6-32 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded TMI Waste Form (thermal and intermediate spectrum) ..... 338
Table 6-33 Estimate of bias for DOE SNF groupings using method 1 and method 2..... 344

## FIGURES

**Page**

Figure 6-1 Bias for Benchmarks Applicable to HEU Oxide DOE-Owned Intact Fuel (Thermal Spectrum) ..... 346
Figure 6-2 Bias for Benchmarks Applicable to HEU Oxide DOE SNF ( Intermediate and Fast Spectrum) ..... 347
Figure 6-3 Bias for Benchmarks Applicable to HEU Oxide Degraded DOE-SNF (Thermal Spectrum) ..... 347
Figure 6-4 Bias for Benchmarks Applicable to HEU Oxide Degraded DOE-SNF (Intermediate and Fast Spectrum) ..... 348
Figure 6-5 Bias for Benchmarks Applicable to UZrH DOE-Owned Intact Fuel in Thermal Spectrum ..... 348
Figure 6-6 Bias for Benchmarks Applicable to UZrH DOE-Owned Intact Fuel in Intermediate and Fast Spectrum ..... 349
Figure 6-7 Bias for Benchmarks Applicable to UZrH DOE-Owned Degraded Fuel in Thermal Spectrum ..... 349
Figure 6-8 Bias for Benchmarks Applicable to UZrH DOE-Owned Degraded Fuel in Intermediate and Fast Spectrum ..... 350
Figure 6-9 Bias for Benchmarks Applicable to Melt and Dilute DOE-Owned Intact Fuel in Thermal Spectrum ..... 350
Figure 6-10 Bias for Benchmarks Applicable to Melt and Dilute DOE-Owned Intact Fuel in Intermediate and Fast Spectrum ..... 351
Figure 6-11 Bias for Benchmarks Applicable to Melt and Dilute DOE-Owned Degraded Fuel in Thermal Spectrum ..... 351
Figure 6-12 Bias for Benchmarks Applicable to Melt and Dilute DOE-Owned Degraded Fuel in Intermediate and Fast Spectrum ..... 352
Figure 6-13 Bias for Benchmarks Applicable to U Alloy DOE-Owned intact Fuel in Thermal Spectrum ..... 352
Figure 6-14 Bias for Benchmarks Applicable to U Alloy DOE-Owned intact Fuel in Intermediate and Fast Spectrum ..... 353
Figure 6-15 Bias for Benchmarks Applicable to U Alloy DOE-Owned Degraded Fuel in Thermal Spectrum. ..... 353
Figure 6-16 Bias for Benchmarks Applicable to U Alloy DOE-Owned Degraded Fuel in Intermediate and Fast Spectrum ..... 354
Figure 6-17 Bias for Benchmarks Applicable to MOX DOE-Owned intact Fuel in Thermal Spectrum ..... 354
Figure 6-18 Bias for Benchmarks Applicable to MOX DOE-Owned intact Fuel in Intermediate and Fast Spectrum ..... 355
Figure 6-19 Bias for Benchmarks Applicable to MOX DOE-Owned Degraded Fuel in Thermal Spectrum ..... 355
Figure 6-20 Bias for Benchmarks Applicable to MOX DOE-Owned Degraded Fuel in Intermediate and Fast Spectrum ..... 356
Figure 6-21 Bias for Benchmarks Applicable to Graphite/Carbide DOE-Owned intact Fuel in Thermal Spectrum. ..... 356

**FIGURES (Continued)**

**Page**

Figure 6-22 Bias for Benchmarks Applicable to Graphite/Carbide DOE-Owned intact Fuel in Intermediate and Fast Spectrum. ..... 357
Figure 6-23 Bias for Benchmarks Applicable to Graphite/Carbide DOE-Owned Degraded Fuel in Thermal Spectrum. ..... 357
Figure 6-24 Bias for Benchmarks Applicable to Graphite/Carbide DOE-Owned Degraded Fuel in Intermediate and Fast Spectrum. ..... 358
Figure 6-25 Bias for Benchmarks Applicable to Th/U DOE-Owned intact Fuel in Thermal Spectrum. ..... 358
Figure 6-26 Bias for Benchmarks Applicable to Th/U DOE-Owned intact Fuel in Intermediate and Fast Spectrum. ..... 359
Figure 6-27 Bias for Benchmarks Applicable to Th/U DOE-Owned Degraded Fuel in Thermal Spectrum. ..... 359
Figure 6-28 Bias for Benchmarks Applicable to Th/U DOE-Owned Degraded Fuel in Intermediate and Fast Spectrum. ..... 360
Figure 6-29 Bias for Benchmarks Applicable to U-metal DOE-Owned intact Fuel in Thermal and Intermediate Spectrum. ..... 360
Figure 6-30 Bias for Benchmarks Applicable to U-metal DOE-Owned Degraded Fuel in Thermal and Intermediate Spectrum. ..... 361
Figure 6-31 Bias for Benchmarks Applicable to LEU Oxide DOE-Owned intact Fuel in Thermal and Intermediate Spectrum. ..... 361
Figure 6-32 Bias for Benchmarks Applicable to LEU Oxide DOE-Owned intact Fuel in Thermal and Intermediate Spectrum. ..... 362

# 1. PURPOSE

The purpose of this document is to review the criticality benchmark experiments available for use in the evaluations for codisposal of the various types of U.S. Department of Energy (DOE) spent nuclear fuel (SNF) at the potential Yucca Mountain Monitored Geologic Repository. The critical experiments evaluated will be used to develop bias and uncertainty values to be utilized with the criticality evaluations of various codisposal waste package internal configurations. An allowance for calculational bias and experimental uncertainties in criticality benchmark calculations must be made per the requirements listed in the *Disposal Criticality Analysis Methodology Topical Report* (Section 3.5.3.2 of YMP [2000]). The results of the Monte Carlo N-Particle (MCNP) calculations include both previously evaluated criticality benchmarks (CRWMS M&O [1999a, 1999b, 1999c]) and the critical experiment updates to the latest edition of the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (NEA [2001]). A comprehensive list of the area and range of applicability for the criticality benchmarks associated with each spent fuel type is also given in this report.

The calculation is associated with the waste package design and was developed in accordance with the technical work plan, *Technical Work Plan For: Department of Energy Spent Nuclear Fuel Work Packages* (CRWMS M&O [2000e]). This calculation is subject to the Quality Assurance Requirements and Description (QARD) (DOE 2000) per the activity evaluation under work package number P6212310M2 in the technical work plan TWP-MGR-MD-000010 REV 01 (BSC [2002]).

# 2. METHOD

The evaluations of criticality benchmark experiments are performed as part of the Disposal Criticality Analysis Methodology program to quantify the MCNP code system's ability to accurately calculate the effective neutron multiplication factor ($k_{eff}$) for various configurations.

The MCNP code is first used to calculate the effective neutron multiplication factor ($k_{eff}$) for selected benchmark models of the critical experiments (NEA [2001]). The MCNP calculations used continuous-energy cross sections processed from the evaluated nuclear data files ENDF/B-V (Briesmeister 1997, App. G). These cross section libraries are part of the MCNP code system that has been obtained from the Software Configuration Management (SCM) in accordance with appropriate procedures. Each of the critical experiments is modeled, and the results reported from the MCNP calculations are the combined average values of $k_{eff}$ from the three estimates (collision, absorption, and track length) and the standard deviation of these results ($\sigma$) listed in the final generation summary in the MCNP output.

The existing MCNP results obtained with the same cross section libraries for the previously evaluated critical experiments (CRWMS M&O [1999a, 1999b and 1999c]) have been added to the general pool of data. The final sets of data have been grouped by associating the benchmark experiments with each DOE SNF fuel type, WP degradation and the specific neutron spectrum. The area and range of applicability for the grouped criticality benchmarks is individually presented in the last section of this calculation.

The MCNP bias and uncertainty associated with each set of benchmark experiments are assessed using classical statistical methods. The final uncertainty includes both calculational uncertainty and the uncertainties associated with experimental measurements and benchmark model simplifications.

## 3. ASSUMPTIONS

3.1 It is assumed that the results of the criticality benchmark experiments documented in CRWMS M&O (1999a, 1999b, and 1999c) would be the same as if they were performed as part of this calculation. The rationale for this assumption is that the MCNP inputs would be the same as they are in these references and since the version of the code and cross-section libraries would be the same the results should be the same (i.e., same code, same input, yields same output). This assumption was used thoughout Section 5.

## 4. USE OF COMPUTER SOFTWARE AND MODELS

### 4.1 SOFTWARE

#### 4.1.1 MCNP

The MCNP code (CRWMS M&O [1998b]) is used to calculate the $k_{eff}$ of the codisposal waste package. The software specifications are as follow:

- Status: Qualified
- Software name: MCNP
- Software version/revision number: Version 4B2
- Software tracking number (computer software configuration item [CSCI]): 30033 V4B2LV
- Computer type: Hewlett Packard (HP) 9000 Series Workstations
- Operating system: HP-UX 10.20
- Computer processing unit number: Software is installed on the Framatome ANP workstation "gr1" whose CPU number is E 9000/785 2008515632 and "gr0" whose CPU number is E 9000/782 2002611431.

The input and output files for the various MCNP calculations are documented in Attachment I, (Attachment II gives the list of the files on Attachment). The calculation files described in Sections 5 and 6 are such that an independent repetition of the software use may be performed.

The MCNP software used is: (a) appropriate for the application of $k_{eff}$ calculations, (b) used only within the range of validation as documented in CRWMS M&O (1998a) and Briesmeister

(1997), (c) obtained from the Software Configuration Management in accordance with AP-SI.1Q, *Software Management*.

## 5. CALCULATION

The purpose of this document is to review the criticality benchmarks available for use in the evaluations for co-disposal viability for the various types of DOE SNF at the Yucca Mountain Project Monitored Geologic Repository. The experiments evaluated will be used to determine the bias and uncertainty values to be used with the MCNP results for the criticality evaluations for these fuels. An allowance for calculational bias and experimental uncertainties in criticality benchmark calculations must be made per the requirements listed in the *Disposal Criticality Analysis Methodology Topical Report*. The results of the MCNP calculations for these experiments are presented with the corresponding fuel type, as are updates to the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (NEA [2001]). A comprehensive list of the benchmarks associated with each fuel type is given in this report.

Criticality benchmarks were considered for nine DOE-owned fuel types. These are Shippingport PWR (CRWMS M&O [2000c]), TRIGA (CRWMS M&O [2000a]), Melt and Dilute Waste Form (Radulescu 2001a), Enrico Fermi (CRWMS M&O [2000b]), Fast Flux Test Facility (FFTF) (CRWMS [M&O 1999d]), Shippingport LWBR (CRWMS M&O [2000d]), Ft. St. Vrain HTGR (Radulescu [2001b]), N-Reactor (CRWMS M&O [2001]) and TMI debris. The first two are primarily highly enriched uranium (HEU) fuels and the third and fourth can be considered intermediate enriched uranium (IEU) fuel. The fifth is a mixed oxide (MOX) fuel with both Pu and mixed isotope criticals being applicable. The sixth and seventh are $^{233}$U based fuels while N-Reactor is low enriched uranium (LEU). TMI debris can also be classified as of LEU type. The seventh is initially HEU. Both HEU and $^{233}$U benchmarks are applicable to Ft. St. Vrain HTGR fuel.

Concentrations of fissile isotopes and neutron absorbers are considered when evaluating whether the benchmark experiments are similar to or bound the cases documented in this report. In addition, two spectrum indices are used in evaluating whether the benchmark experiments are similar to or bound the cases in regards to the neutron spectrum. These two indices are the hydrogen to fissile isotope (H/X) ratio and the Average Energy of Neutron Causing Fission (AENCF) (CRWMS M&O [1999b, page 8]). The AENCF was calculated directly from values that are provided in the MCNP outputs. The values from the output files for calculating AENCF are the weight per source particle of a neutron lost to fission and the energy per source particle of a neutron lost to fission. The AENCF is then calculated by dividing the energy per source particle by the weight per source particle.

The calculation of the H/X ratio was performed by two methods based upon whether the critical configuration was symmetrical or not. For symmetric cases, the number of hydrogen atoms per unit cell was calculated and divided by the number of fissile atoms per unit cell. In the non-symmetric cases the moderator volume was determined by taking the volume of moderator from each unit cell plus additional moderator that neutrons would "see" in a path to another fuel rod. This volume was used to determine the total number of hydrogen atoms present in that volume

then dividing it by the total number of fissile atoms present in that volume. H/X data from Nuclear Energy Agency (NEA) 2001 was utilized when available.

All critical experiments are taken from NEA (2001) unless otherwise noted. The MCNP models have been constructed using the benchmark models provided in this reference. The experiments are presented using the names reported in NEA (2001) where the first part denotes enrichment (HEU, LEU, IEU, U-233, Pu, MIX), the second part represents the type of experiment (e.g, SOL for solution, MET for metal etc.) and the last part is part of the spectrum where most fissions occur (e.g., THERMAL, FAST, INTER). The difference between the experimental (benchmark model) $k_{eff}$ and the calculated $k_{eff}$ is designated as $\Delta k_{eff}$. In the following tables for previously evaluated cases $\Delta k_{eff}$ is reported as (benchmark model $k_{eff}$ – calculated $k_{eff}$), whereas in the other cases $\Delta k_{eff}$ is reported as (calculated $k_{eff}$ – benchmark model $k_{eff}$). This second definition is used in the analysis done in section 6.3 of this document. The combined uncertainty ($\sigma$) is calculated as the statistical sum of the experimental (or benchmark model) uncertainty reported in NEA (2001) and the standard deviation of the MCNP calculation.

A summary description of the critical experiments previously evaluated in CRWMS M&O (1999a, 1999b, and 1999c) is provided in the sections designated as "Previously Evaluated" for completeness.

## 5.1 PREVIOUSLY EVALUATED HEU BENCHMARK CASES

### 5.1.1 HEU Benchmarks Related to Intact Waste Package Configurations

The following summary descriptions of benchmarks already evaluated in CRWMS M&O (1999a) are applicable for HEU fuels.

#### 5.1.1.1 SPERT-D Fuel Experiments

Twenty-three critical experiments involving lattices of SPERT-D fuel elements were performed at Oak Ridge National Laboratories (ORNL). The fuel elements consisted of plates of uranium-aluminum alloy. Each fuel element contained approximately 300 grams (NEA [2001, HEU-MET-THERM-006]) of $^{235}U$ in 22 aluminum clad fuel plates. All 23 experiments were considered acceptable for use as benchmark experiments by the evaluator and detailed descriptions of each critical configuration is reported in NEA (2001). Table 5-1 provides a listing of the various SPERT-D cases, the benchmark $k_{eff}$ values, and the MCNP filenames. The critical experiments were represented on a 12-inch thick Plexiglas table (NEA [2001]). The reflector and moderator for cases 1 through 18 was demineralized water, and an aqueous solution of uranyl nitrate $U(92.6)O_2(NO_3)_2$ in cases 19 through 23 (NEA [2001]).

Table 5-1 SPERT-D Fuel Element Critical Experiments

| Handbook case name | Global description | Reflector above fuel (cm) | Benchmark $k_{eff} \pm \sigma$ | H/ $^{235}$U ratio* | MCNP case name |
|---|---|---|---|---|---|
| CASE1 | 4 x 3.77 lattice, 4.63 kg $^{235}$U, 0.0" spacing | 9.825 | 1.000 ± 0.004 | 193 | spert1 |
| CASE 2 | 4 x 3.16 lattice, 3.87 kg $^{235}$U, 0.25" spacing | 12.8982 | 1.000 ± 0.004 | 255 | spert2 |
| CASE 3 | 4 x 3.09 lattice, 3.79 kg $^{235}$U, 0.50" spacing | 9.8401 | 1.000 ± 0.004 | 264 | spert3 |
| CASE 4 | circular, 3.48 kg $^{235}$U, 0.50" spacing | 7.3136 | 1.000 ± 0.004 | 298 | spert4 |
| CASE 5 | 4 x 3.16 lattice, 3.87 kg $^{235}$U, 0.75" spacing | 17.21 | 1.000 ± 0.004 | 257 | spert5 |
| CASE 6 | 4 x 3.70 lattice, 4.54 kg $^{235}$U, 1.00" spacing | 13.853 | 1.000 ± 0.004 | 202 | spert6 |
| CASE 7 | 5 x 4.03 lattice, 6.16 kg $^{235}$U, 1.25" spacing | 13.0186 | 1.000 ± 0.004 | 253 | spert7 |
| CASE 8 | 6 x 5.34 lattice, 9.82 kg $^{235}$U, 1.50" spacing | 11.3984 | 1.000 ± 0.004 | 218 | spert8 |
| CASE 9 | 7 x 6.68 lattice, 6.16 kg $^{235}$U, 1.60" spacing | 12.0359 | 1.000 ± 0.004 | 195 | spert9 |
| CASE 10 | 4 x 3.2 x 3 lattice, 11.78 kg $^{235}$U, 0.0" spacing | 9.4309 | 1.000 ± 0.004 | 249 | spert10 |
| CASE 11 | 3 x 3.36 x 3 lattice, 9.28 kg $^{235}$U, 0.50" spacing | 12.0994 | 1.000 ± 0.004 | 233 | spert11 |
| CASE 12 | 4 x 4 x 3 lattice, 14.71 kg $^{235}$U, 1.25" spacing | 8.0 | 1.000 ± 0.004 | 179 | spert12 |
| CASE 13 | slab 16 x 2.32, 11.37 kg $^{235}$U, 1.25" spacing | 11.6162 | 1.000 ± 0.004 | 262 | spert13 |
| CASE 14 | slab 16 x 3, 14.71 kg $^{235}$U, 0.50"/2.19" spacing | 7.5728 | 1.000 ± 0.004 | 179 | spert14 |
| CASE 15 | slab 16 x 4, 19.62 kg $^{235}$U, 0.50"/2.56" spacing | 10.75 | 1.000 ± 0.004 | 180 | spert15 |
| CASE 16 | 2 slabs 16 x 2, 19.62 kg $^{235}$U, 0.50"/0.50"/6.37" spacing | 12.7351 | 1.000 ± 0.004 | 180 | spert16 |
| CASE 17 | slab 4 x 5.0 w/ Cd, 6.19 kg $^{235}$U, 0.0"/0.75" spacing | 10.7471 | 1.000 ± 0.004 | 234 | spert17 |
| CASE 18 | slab 4 x 7.04 w/ Cd, 8.64 kg $^{235}$U, 0.0"/0.75" spacing | 13.8573 | 1.000 ± 0.004 | 218 | spert18 |
| CASE 19 | U Nitrate (3.99 g $^{235}$U/liter) & 3 x 3.09, 2.86 kg $^{235}$U, 0.5" spacing, 0.0 g B/liter | 6.8208 | 1.000 ± 0.004 | 500 | spert19 |
| CASE 20 | U Nitrate (3.99 g $^{235}$U/liter) & 4 | 8.3311 | 1.000 ± 0.004 | 376 | spert20 |

| Table 5-1 SPERT-D Fuel Element Critical Experiments | | | | | |
|---|---|---|---|---|---|
| **Handbook case name** | **Global description** | **Reflector above fuel (cm)** | **Benchmark $k_{eff} \pm \sigma$** | **H/ $^{235}$U ratio*** | **MCNP case name** |
| | x 4.20, 5.15 kg $^{235}$U, 0.5" spacing, 0.389 g B/liter | | | | |
| CASE 21 | U Nitrate (3.99 g $^{235}$U/liter) & 5 x 4.41, 6.76 kg $^{235}$U, 0.5" spacing, 0.579 g B/liter | 4.6946 | 1.000 ± 0.004 | 213 | spert21 |
| CASE 22 | U Nitrate (3.99 g $^{235}$U/liter) & 6 x 4.96, 8.90 kg $^{235}$U, 0.5" spacing, 0.773 g B/liter | 5.5725 | 1.000 ± 0.004 | 158 | spert22 |
| CASE 23 | U Nitrate (3.99 g $^{235}$U/liter) & 6 x 5.55, 10.15 kg $^{235}$U, 0.5" spacing, 0.871 g B/liter | 7.7118 | 1.000 ± 0.004 | 134 | spert23 |

NOTE: [a] H/$^{235}$U denotes Hydrogen to $^{235}$U ratio and is reported in Table 5.1.3.1-1 of CRWMS M&O (1999a).

The following results are from CRWMS M&O (1999a, page 74). Table 5-2 presents the MCNP calculation results for ENDF/B-V and ENDF/B-VI cross section libraries for the SPERT-D Fuel set of critical experiments. Table 5-3 presents the differences between experimental and calculated $k_{eff}$ values. Table 5-4 presents the AENCF values for the SPERT-D Fuel set of critical experiments. The calculated results are scattered slightly above and below the benchmark $k_{eff}$ values. The calculated $k_{eff}$ for SPERT13 was significantly higher than the benchmark $k_{eff}$ for no apparent reason. This behavior was also observed in the sample calculations provided by the evaluator and stated in NEA (2001).

| Table 5-2. LCE Reactivity Calculation Results for HEU-MET-THERM-006 | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V $k_{eff} \pm \sigma$** | **ENDF/B-VI $k_{eff} \pm \sigma$** |
| spert1 | 0.99792 ± 0.00184 | 0.99727 ± 0.00176 |
| spert2 | 0.99952 ± 0.0019 | 0.99732 ± 0.00186 |
| spert3 | 1.00676 ± 0.00114 | 1.00384 ± 0.00095 |
| spert4 | 0.99542 ± 0.00173 | 0.98908 ± 0.00179 |
| spert5 | 1.00104 ± 0.00162 | 1.00072 ± 0.00183 |
| spert6 | 1.00133 ± 0.00168 | 0.99683 ± 0.00152 |
| spert7 | 0.99923 ± 0.00163 | 0.9952 ± 0.00147 |
| spert8 | 0.99843 ± 0.00154 | 0.99009 ± 0.0015 |
| spert9 | 1.00003 ± 0.00143 | 0.99697 ± 0.00149 |
| spert10 | 1.00608 ± 0.00177 | 1.00534 ± 0.0019 |

Table 5-2. LCE Reactivity Calculation Results for HEU-MET-THERM-006

| MCNP case name | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-VI $k_{eff} \pm \sigma$ |
|---|---|---|
| spert11 | 1.00565 ± 0.00163 | 1.00283 ± 0.00171 |
| spert12 | 1.00676 ± 0.00156 | 1.00466 ± 0.00175 |
| spert13 | 1.03289 ± 0.00181 | 1.02936 ± 0.00189 |
| spert14 | 0.99451 ± 0.00158 | 0.99427 ± 0.00159 |
| spert15 | 0.99355 ± 0.00107 | 0.99179 ± 0.00103 |
| spert16 | 1.00791 ± 0.00175 | 1.00549 ± 0.00183 |
| spert17 | 1.00569 ± 0.00188 | 1.00206 ± 0.00179 |
| spert18 | 1.00028 ± 0.00198 | 1.00615 ± 0.00181 |
| spert19 | 0.99482 ± 0.00148 | 0.98982 ± 0.00158 |
| spert20 | 0.99652 ± 0.00158 | 0.99315 ± 0.00166 |
| spert21 | 1.00113 ± 0.00186 | 1.00086 ± 0.00178 |
| spert22 | 1.00272 ± 0.00194 | 0.9984 ± 0.00175 |
| spert23 | 1.00695 ± 0.0011 | 1.00426 ± 0.00099 |

Table 5-3 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for HEU-MET-THERM-006 Experiments

| MCNP case name | ENDF/B-V $\Delta k_{eff} \pm 2\sigma$ | ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|
| spert1 | 0.00208 ± 0.00881 | 0.00273 ± 0.00874 |
| spert2 | 0.00048 ± 0.00886 | 0.00268 ± 0.00882 |
| spert3 | -0.00676 ± 0.00832 | -0.00384 ± 0.00822 |
| spert4 | 0.00458 ± 0.00872 | 0.01092 ± 0.00876 |
| spert5 | -0.00104 ± 0.00863 | -0.00072 ± 0.00880 |
| spert6 | -0.00133 ± 0.00868 | 0.00317 ± 0.00856 |
| spert7 | 0.00077 ± 0.00864 | 0.00480 ± 0.00852 |
| spert8 | 0.00157 ± 0.00857 | 0.00991 ± 0.00854 |
| spert9 | -0.00003 ± 0.00850 | 0.00303 ± 0.00854 |
| spert10 | -0.00608 ± 0.00875 | -0.00534 ± 0.00886 |
| spert11 | -0.00565 ± 0.00864 | -0.00283 ± 0.00870 |
| spert12 | -0.00676 ± 0.00859 | -0.00466 ± 0.00873 |
| spert13 | -0.03289 ± 0.00878 | -0.02936 ± 0.00885 |
| spert14 | 0.00549 ± 0.00860 | 0.00573 ± 0.00861 |

Table 5-3 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for HEU-MET-THERM-006 Experiments

| MCNP case name | ENDF/B-V $\Delta k_{eff} \pm 2\sigma$ | ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|
| spert15 | 0.00645 ± 0.00828 | 0.00821 ± 0.00826 |
| spert16 | -0.00791 ± 0.00873 | -0.00549 ± 0.00880 |
| spert17 | -0.00569 ± 0.00884 | -0.00206 ± 0.00413 |
| spert18 | -0.00028 ± 0.00893 | -0.00615 ± 0.00714 |
| spert19 | 0.00518 ± 0.00853 | 0.01018 ± 0.00860 |
| spert20 | 0.00348 ± 0.00860 | 0.00685 ± 0.00866 |
| spert21 | -0.00113 ± 0.00882 | -0.00086 ± 0.00876 |
| spert22 | -0.00272 ± 0.00889 | 0.00160 ± 0.00873 |
| spert23 | -0.00695 ± 0.00830 | -0.00426 ± 0.00824 |

Table 5-4 AENCF for HEU-MET-THERM-006 Experiments

| MCNP case name | ENDF/B-V | ENDF/B-VI |
|---|---|---|
| spert1 | 0.0147 | 0.0140 |
| spert2 | 0.0126 | 0.0125 |
| spert3 | 0.0117 | 0.0112 |
| spert4 | 0.0110 | 0.0113 |
| spert5 | 0.0105 | 0.0110 |
| spert6 | 0.0102 | 0.0098 |
| spert7 | 0.0097 | 0.0104 |
| spert8 | 0.0098 | 0.0096 |
| spert9 | 0.0099 | 0.0098 |
| spert10 | 0.0147 | 0.0140 |
| spert11 | 0.0115 | 0.0113 |
| spert12 | 0.0101 | 0.0102 |
| spert13 | 0.0143 | 0.0141 |
| spert14 | 0.0106 | 0.0107 |
| spert15 | 0.0106 | 0.0105 |
| spert16 | 0.0120 | 0.0118 |
| spert17 | 0.0131 | 0.0133 |
| spert18 | 0.0140 | 0.0135 |
| spert19 | 0.0097 | 0.0090 |

| Table 5-4 AENCF for HEU-MET-THERM-006 Experiments | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| spert20 | 0.0114 | 0.0115 |
| spert21 | 0.0126 | 0.0127 |
| spert22 | 0.0133 | 0.0131 |
| spert23 | 0.0132 | 0.0131 |

### 5.1.1.2 Hexagonally Pitched Lattices of Highly Enriched Fuel Rods of Cross-Shaped Cross Section

A series of critical experiments with water moderated hexagonally pitched lattices of highly enriched fuel rods of cross-shaped cross section was performed over several years in the Russian Research Center (RRC) "Kurchatov Institute". The 28 experiments analyzed under this category in this section consist of the following:

1) Fifteen critical two-zone lattice experiments corresponding to different combinations of inner and peripheral zones of cross-shaped fuel rods at two pitches. For detailed descriptions of these experimental configurations see NEA (2001, HEU-COMP-THERM-003).

2) Four critical configurations of hexagonal lattices of fuel rods with gadolinium (Gd) or samarium (Sm) rods. These experiments consisted of double lattices of fuel rods and absorber rods containing Gd or Sm. Detailed experimental configuration descriptions are available in NEA (2001, HEU-COMP-THERM-004).

3) One critical configuration of hexagonal pitched clusters of lattices of fuel rods with copper (Cu) rods. Detailed experimental configuration descriptions are available in NEA (2001, HEU-COMP-THERM-005). Two cases were used: one considers the fuel rod as cylinder and the other represents the fuel rod as it is in reality.

4) Three critical configurations with uniform hexagonal lattices with pitch values of 5.6, 10.0, and 21.13 mm. Detailed experimental configuration descriptions are in NEA (2001, HEU-COMP-THERM-006).

5) Three critical configurations with double hexagonal lattices of fuel rods and zirconium hydride rods. Detailed experimental configuration descriptions are in NEA (2001, HEU-COMP-THERM-007). Two cases were used: one considers the fuel rod as a cylinder and the other represents the fuel rod as it is in reality.

6) Two critical configurations with double hexagonal lattices of fuel rods and boron carbide rods. Detailed experimental configuration descriptions are available in NEA (2001, HEU-COMP-THERM-008).

Table 5-5 provides a listing of these various cases, the benchmark $k_{eff}$ values, and the MCNP filenames.

| Table 5-5 Cross-Shaped Fuel Rods Critical Experiments | | | | |
|---|---|---|---|---|
| **Handbook case name** | **Description** | **Benchmark $k_{eff} \pm \sigma$** | **H/$^{235}$U ratio[a]** | **MCNP case name** |
| HEU-COMP-THERM-003 2-Zone Critical Arrays with U(80 wt%)$O_2$+Cu Fuel and Lightweight Moderator | | | | |
| Array Number 1 | Center zone: 12.2 mm pitch, 19 rods<br>Outer zone: 6.1 mm pitch, 1390 rods | 1.000 ± 0.0044 | 51 | hct3-1 |
| Array Number 2 | Center zone: 12.2 mm pitch, 61 rods<br>Outer zone: 6.1 mm pitch, 1182 rods | 1.000 ± 0.0044 | 59 | hct3-2 |
| Array Number 3 | Center zone: 12.2 mm pitch, 121 rods<br>Outer zone: 6.1 mm pitch, 897 rods | 1.000 ± 0.0044 | 75 | hct3-3 |
| Array Number 4 | Center zone: 12.2 mm pitch, 199 rods<br>Outer zone: 6.1 mm pitch, 577 rods | 1.000 ± 0.0044 | 107 | hct3-4 |
| Array Number 5 | Center zone: 12.2 mm pitch, 271 rods<br>Outer zone: 6.1 mm pitch, 325 rods | 1.000 ± 0.0044 | 153 | hct3-5 |
| Array Number 6 | Center zone: 6.1 mm pitch, 1099 rods<br>Outer zone: 12.2 mm pitch, 167 rods | 1.000 ± 0.0044 | 73 | hct3-6 |
| Array Number 7 | Center zone: 6.1 mm pitch, 793 rods<br>Outer zone: 12.2 mm pitch, 250 rods | 1.000 ± 0.0044 | 103 | hct3-7 |
| Array Number 8 | Center zone: 6.1 mm pitch, 757 rods<br>Outer zone: 12.2 mm pitch, 249 rods | 1.000 ± 0.0044 | 105 | hct3-8 |
| Array Number 9 | Center zone: 6.1 mm pitch, 445 rods<br>Outer zone: 12.2 mm pitch, 319 rods | 1.000 ± 0.0044 | 144 | hct3-9 |
| Array Number 10 | Center zone: 6.1 mm pitch, 217 rods<br>Outer zone: 12.2 mm pitch, 372 rods | 1.000 ± 0.0044 | 193 | hct310 |
| Array Number 11 | Center zone: 6.1 mm pitch, 85 rods<br>Outer zone: 12.2 mm pitch, 415 rods | 1.000 ± 0.0044 | 239 | hct311 |
| Array Number 12 | Center zone: 18.3 mm pitch, 121 rods<br>Outer zone: 6.1 mm pitch, 985 rods | 1.000 ± 0.0044 | 115 | hct312 |
| Array Number 13 | Center zone: 18.3 mm pitch, 301 rods<br>Outer zone: 6.1 mm pitch, 426 rods | 1.000 ± 0.0044 | 302 | hct313 |
| Array Number 14 | Center zone: 6.1 mm pitch, 763 rods<br>Outer zone: 18.3 mm pitch, 186 rods | 1.000 ± 0.0044 | 168 | hct314 |
| Array Number 15 | Center zone: 6.1 mm pitch, 337 rods<br>Outer zone: 18.3mm pitch, 325 rods | 1.000 ± 0.0044 | 349 | hct315 |
| HEU-COMP-THERM -004 Water Moderator Hexagonally Pitched (5.3 mm) Lattices of U(90 wt%)$O_2$+Cu Fuel With Gd or Sm Rods | | | | |
| Case1 | 106 Gd rods on 27.54 mm pitch, 2760 fuel | 1.0000 ± 0.0038 | 35 | hct4-1 |

Table 5-5 Cross-Shaped Fuel Rods Critical Experiments

| Handbook case name | Description | Benchmark $k_{eff} \pm \sigma$ | $H/^{235}U$ ratio[a] | MCNP case name |
|---|---|---|---|---|
| | rods | | | |
| Case2 | 55 Gd rods on 36.72 mm pitch, 2520 fuel rods | 1.0000 ± 0.0039 | 35 | hct4-2 |
| Case3 | 121 Sm rods on 27.54 mm pitch, 3198 fuel rods | 1.0000 ± 0.0037 | 35 | hct4-3 |
| Case4 | 58 Gd rods on 36.72 mm pitch, 2727 fuel rods | 1.0000 ± 0.0038 | 35 | hct4-4 |
| HEU-COMP-THERM -005 Water Moderator Hexagonally Pitched (5.2 mm) Lattices of U(80 wt%)O2+Cu Fuel Rods | | | | |
| Case 1 | Fuel rod hexagonal-lattice pitch 5.2 mm, 2257 rods | 1.0000 ± 0.0094 | 23 | hct5-2 |
| HEU-COMP-THERM -006 Water Moderator Hexagonally Pitched Lattices of U(80 wt%)O2+Cu Fuel Rods | | | | |
| Case1 | 1819 fuel rods on a 5.6 mm pitch | 1.0000 ± 0.0058 | 30 | hct6-t1 |
| Case2 | 457 fuel rods on a 10.0 mm pitch | 1.0000 ± 0.0020 | 144 | hct6-t2 |
| Case3 | 554 fuel rods on a 21.13 mm pitch | 1.0000 ± 0.0048 | 716 | hct6-t3 |
| HEU-COMP-THERM -007 Water Moderator Hexagonally Pitched Lattices of U(80 wt%)O2+Cu Fuel Rods and Zirconium Hydride Rods | | | | |
| Case 1 | 523 Zr rods on 10.5655 mm pitch, 1064 fuel rods | 1.0000± 0.0035 | 91 | hct7-4 |
| Case 2 | 121 Zr rods on 21.1310 mm pitch, 1400 fuel rods | 1.0000± 0.0041 | 65 | hct7-5 |
| Case 3 | 31 Zr rods on 42.2620 mm pitch, 1484 fuel rods | 1.0000± 0.0043 | 60 | hct7-6 |
| HEU-COMP-THERM -008 Water Moderator Haxagonally Pitched (5.3 mm) Double Lattices of U(80 wt%)$O_2$+Cu Fuel and Boron Carbide Rods | | | | |
| Case1 | 217 $B_4C$ rods (1.0 g B/rod) on 21.2 mm pitch, 3460 fuel rods | 1.0000 ± 0.0055 | 25 | hct8-1 |
| Case2 | 169 $B_4C$ rods (3.5 g B/rod) on 26.5 mm pitch, 4130 fuel rods | 1.0000 ± 0.0056 | 25 | hct8-2 |

NOTE: [a] $H/^{235}U$ denotes Hydrogen to $^{235}U$ ratio and is reported in Table 5.1.3.2-1 of CRWMS M&O (1999a).

The following results are from CRWMS M&O (1999a, page 78). Table 5-6 presents the MCNP calculation results for ENDF/B-V and ENDF/B-VI cross section libraries for the HEU-COMP-THERM-003, 004, 005, 006, 007, and 008 set of critical experiments. Table 5-7 presents the difference between the experimental and calculated values of $k_{eff}$. Table 5-8 presents the AENCF for the HEU-COMP-THERM-003, 004, 005, 006, 007, and 008 set of critical experiments. The arrays with middle-spaced lattice in the peripheral zone (hct3-6 through hct3-11) gave high results. Arrays with a closely spaced lattice in the peripheral zone tended to give results less than the benchmark value (cases hct3-1 through hct3-4, hct3-12, and hct3-13). Two

cases were used to describe HEU-COMP-THERM -005 and HEU-COMP-THERM -007 experiments. In the first one (hct5-1, hct7-1, hct7-2, and hct7-3), the fuel rod was assumed to be cylindrical and in the second (hct5-2, hct7-4, hct7-5, and hct7-6) the cross-shaped fuel rod was represented. There is up to 0.9% difference between the $k_{eff}$ obtained using a cylindrical fuel rod and a cross-shaped fuel rod.

Table 5-6 LCE Reactivity Calculation Results for HEU-COMP-THERM -003, 004, 005, 006, 007, and 008 Experiments

| MCNP Case name | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-VI $k_{eff} \pm \sigma$ |
|---|---|---|
| hct3-1 | 0.99381 ± 0.00136 | 0.99473 ± 0.00138 |
| hct3-2 | 0.99746 ± 0.00144 | 0.9956 ± 0.00144 |
| hct3-3 | 0.99637 ± 0.00144 | 0.99473 ± 0.00149 |
| hct3-4 | 0.99796 ± 0.00148 | 0.99688 ± 0.00148 |
| hct3-5 | 1.00253 ± 0.00137 | 1.0006 ± 0.00151 |
| hct3-6 | 1.00616 ± 0.00152 | 1.00706 ± 0.0016 |
| hct3-7 | 1.01132 ± 0.00143 | 1.009 ± 0.00145 |
| hct3-8 | 1.00988 ± 0.00155 | 1.01081 ± 0.00143 |
| hct3-9 | 1.01309 ± 0.00154 | 1.01355 ± 0.00145 |
| hct3-10 | 1.00962 ± 0.0014 | 1.01004 ± 0.00149 |
| hct3-11 | 1.0133 ± 0.0014 | 1.01317 ± 0.00152 |
| hct3-12 | 0.98956 ± 0.00133 | 0.98921 ± 0.00135 |
| hct3-13 | 0.99677 ± 0.00119 | 0.99588 ± 0.00126 |
| hct3-14 | 1.005 ± 0.00157 | 1.00682 ± 0.0015 |
| hct3-15 | 1.00775 ± 0.00142 | 1.003 ± 0.00141 |
| hct4-1 | 0.98756 ± 0.00122 | 0.99143 ± 0.00122 |
| hct4-2 | 0.9889 ± 0.0012 | 0.99042 ± 0.00123 |
| hct4-3 | 0.99157 ± 0.00119 | 0.99272 ± 0.00121 |
| hct4-4 | 0.99116 ± 0.00114 | 0.98934 ± 0.00122 |
| hct5-1 | 0.98467 ± 0.00118 | 0.98829 ± 0.00117 |
| hct5-2 | 0.98455 ± 0.00128 | 0.988871 ± 0.00115 |
| hct6-t1 | 0.98952 ± 0.00137 | 0.99212 ± 0.00136 |
| hct6-t2 | 1.0104 ± 0.0013 | 1.00798 ± 0.00128 |
| hct6-t3 | 0.99557 ± 0.00095 | 0.99612 ± 0.00099 |
| hct7-1 | 0.99271 ± 0.00155 | 0.99909 ± 0.00149 |
| hct7-2 | 0.9947 ± 0.00153 | 0.99394 ± 0.00151 |

Table 5-6 LCE Reactivity Calculation Results for HEU-COMP-THERM -003, 004, 005, 006, 007, and 008 Experiments

| MCNP Case name | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-VI $k_{eff} \pm \sigma$ |
|---|---|---|
| hct7-3 | 0.99196 ± 0.00144 | 0.99499 ± 0.00154 |
| hct7-4 | 0.99932 ± 0.00164 | 0.99992 ± 0.00148 |
| hct7-5 | 0.99492 ± 0.00156 | 0.99969 ± 0.00152 |
| hct7-6 | 0.99487 ± 0.00154 | 0.99545 ± 0.00159 |
| hct8-1 | 0.99042 ± 0.00106 | 0.99281 ± 0.00108 |
| hct8-2 | 0.98954 ± 0.00112 | 0.99117 ± 0.00116 |

Table 5-7 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for HEU-COMP-THERM -003, 004, 005 006, 007, and 008 Experiments

| MCNP case name | ENDF/B-V $\Delta k_{eff} \pm 2\sigma$ | ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|
| hct3-1 | 0.00619 ± 0.00921 | 0.00527 ± 0.00922 |
| hct3-2 | 0.00254 ± 0.00926 | 0.00440 ± 0.00926 |
| hct3-3 | 0.00363 ± 0.00926 | 0.00527 ± 0.00929 |
| hct3-4 | 0.00204 ± 0.00928 | 0.00312 ± 0.00928 |
| hct3-5 | -0.00253 ± 0.00922 | -0.00060 ± 0.00930 |
| hct3-6 | -0.00616 ± 0.00931 | -0.00706 ± 0.00936 |
| hct3-7 | -0.01132 ± 0.00925 | -0.00900 ± 0.00927 |
| hct3-8 | -0.00988 ± 0.00933 | -0.01081 ± 0.00925 |
| hct3-9 | -0.01309 ± 0.00932 | -0.01355 ± 0.00927 |
| hct3-10 | -0.00962 ± 0.00923 | -0.01004 ± 0.00929 |
| hct3-11 | -0.01330 ± 0.00923 | -0.01317 ± 0.00931 |
| hct3-12 | 0.01044 ± 0.00919 | 0.01079 ± 0.00920 |
| hct3-13 | 0.00323 ± 0.00912 | 0.00412 ± 0.00915 |
| hct3-14 | -0.00500 ± 0.00934 | -0.00682 ± 0.00930 |
| hct3-15 | -0.00775 ± 0.00925 | -0.00300 ± 0.00924 |
| hct4-1 | 0.01244 ± 0.00798 | 0.00857 ± 0.00891 |
| hct4-2 | 0.01110 ± 0.00816 | 0.00958 ± 0.00818 |
| hct4-3 | 0.00843 ± 0.00777 | 0.00728 ± 0.00779 |
| hct4-4 | 0.00884 ± 0.00793 | 0.01066 ± 0.01094 |
| hct5-1 | 0.01533 ± 0.018948 | 0.01347 ± 0.0190 |

Table 5-7 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for HEU-COMP-THERM -003, 004, 005 006, 007, and 008 Experiments

| MCNP case name | ENDF/B-V $\Delta k_{eff} \pm 2\sigma$ | ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|
| hct5-2 | 0.01545 ± 0.018973 | 0.01507 ± 0.0191 |
| hct6-t1 | 0.01048 ± 0.01192 | 0.00788 ± 0.01191 |
| hct6-t2 | -0.01040 ± 0.00477 | -0.00798 ± 0.00475 |
| hct6-t3 | 0.00443 ± 0.00979 | 0.00388 ± 0.00980 |
| hct7-1 | 0.00729 ± 0.004675 | 0.00503 ± 0.0076 |
| hct7-2 | 0.0053 ± 0.005116 | 0.00367 ± 0.0087 |
| hct7-3 | 0.00804 ± 0.005175 | 0.00626 ± 0.0091 |
| hct7-4 | 0.00068 ± 0.004797 | -0.0009 ± 0.0076 |
| hct7-5 | 0.00508 ± 0.005152 | 0.00098 ± 0.0088 |
| hct7-6 | 0.00513 ± 0.005289 | 0.00444 ± 0.0091 |
| hct8-1 | 0.00958 ± 0.01120 | 0.00719 ± 0.01121 |
| hct8-2 | 0.01046 ± 0.01142 | 0.00883 ± 0.01144 |

Table 5-8 AENCF For HEU-COMP-THERM -003, 004, 005, 006, 007, and 008 Experiments (MeV)

| MCNP case name | ENDF/B-V | ENDF/B-VI |
|---|---|---|
| hct3-1 | 0.0467 | 0.0460 |
| hct3-2 | 0.0404 | 0.0405 |
| hct3-3 | 0.0337 | 0.0333 |
| hct3-4 | 0.0259 | 0.0261 |
| hct3-5 | 0.0202 | 0.0199 |
| hct3-6 | 0.0405 | 0.0398 |
| hct3-7 | 0.0339 | 0.0339 |
| hct3-8 | 0.0329 | 0.0334 |
| hct3-9 | 0.0263 | 0.0260 |
| hct-310 | 0.0209 | 0.0200 |
| hct3-11 | 0.0177 | 0.0173 |
| hct3-12 | 0.0265 | 0.0265 |
| hct3-13 | 0.0139 | 0.0133 |
| hct3-14 | 0.0300 | 0.0301 |
| hct3-15 | 0.0183 | 0.0179 |

Table 5-8 AENCF For HEU-COMP-THERM -003, 004, 005, 006, 007, and 008 Experiments (MeV)

| MCNP case name | ENDF/B-V | ENDF/B-VI |
|---|---|---|
| hct4-1 | 0.0744 | 0.0735 |
| hct4-2 | 0.0736 | 0.0716 |
| hct4-3 | 0.0756 | 0.0747 |
| hct4-4 | 0.0742 | 0.0729 |
| hct5-1 | 0.0779 | 0.0762 |
| hct5-2 | 0.0764 | 0.0765 |
| hct6-t1 | 0.0720 | 0.0718 |
| hct6-t2 | 0.0232 | 0.0229 |
| hct6-t3 | 0.0104 | 0.0106 |
| hct7-1 | 0.0347 | 0.0335 |
| hct7-2 | 0.0455 | 0.0451 |
| hct7-3 | 0.0485 | 0.0473 |
| hct7-4 | 0.0339 | 0.0334 |
| hct7-5 | 0.0458 | 0.0445 |
| hct7-6 | 0.0475 | 0.0470 |
| hct8-1 | 0.0882 | 0.0856 |
| hct8-2 | 0.0922 | 0.0912 |

### 5.1.1.3 Critical Experiments of EBOR Fuel Pins in Water

Twenty-one critical experiments involving lattices of EBOR (Experimental Beryllium Oxide Reactor) fuel pins were performed in 1967 at ORNL. The fuel pins consisted of compressed ceramic pellets contained in Hastelloy X-280 tubes. The pellets were a homogeneous mixture of $U(62.4)O_2$ and BeO. Two sets of experiments were conducted. The first set of experiments (total of 15 experiments) consisted of EBOR fuel pins arranged in various lattice configurations moderated and reflected by water (NEA [2001, HEU-COMP-THERM-010]). The second set (6 experiments) consisted of EBOR fuel pins arranged in various lattice configurations with boron and/or with uranyl nitrate in the water (NEA [2001, HEU-COMP-THERM-010]). All 21 fuel-pin experiments are considered acceptable for use as benchmark experiments by the evaluators. Detailed experimental configuration descriptions are available in Reference NEA (2001, HEU-COMP-THERM-010). Table 5-9 provides a listing of the 21 cases and the $k_{eff}$ values in the experimental configurations. The Plexiglas spacers and grid plates were omitted from the MCNP representation because they were considered to have a negligible effect on system reactivity by the evaluator (NEA [2001]).

Table 5-9 Configurations of HEU-COMP-THERM-010 Experiments

| Handbook case name | Surface separation (cm) | Critical number of pins | Critical height of solution (cm)[a] | Solution inside the core | Benchmark $k_{eff} \pm \sigma$ | $H/^{235}U$ ratio[b] | MCNP case name |
|---|---|---|---|---|---|---|---|
| 1 | 0.290 | 222 | 15.2000 | Water | 1.0000 ± 0.0050 | 49 | hct101 |
| 2 | 0.290 | 223 | -50.3000 | Water | 1.0000 ± 0.0050 | 36 | hct102 |
| 3 | 0.536 | 138 | 30.8000 | Water | 1.0000 ± 0.0050 | 72. | hct103 |
| 4 | 0.790 | 102 | -21.3000 | Water | 1.0000 ± 0.0050 | 101 | hct104 |
| 5 | 1.046 | 85 | 15.2000 | Water | 1.0000 ± 0.0050 | 148 | hct105 |
| 6 | 1.046 | 86 | -60.8000 | Water | 1.0000 ± 0.0050 | 100 | hct106 |
| 7 | 1.323 | 78 | 15.2000 | Water | 1.0000 ± 0.0050 | 170 | hct107 |
| 8 | 1.323 | 79 | -39.0000 | Water | 1.0000 ± 0.0050 | 134 | hct108 |
| 9 | 1.300 | 77 | -3.90000 | Water | 1.0000 ± 0.0050 | 204 | hct109 |
| 10 | 1.554 | 75 | 15.2000 | Water | 1.0000 ± 0.0050 | 265 | hct110 |
| 11 | 1.554 | 76 | -61.3000 | Water | 1.0000 ± 0.0050 | 178 | hct111 |
| 12 | 1.826 | 77 | -43.2000 | Water | 1.0000 ± 0.0050 | 247 | hct112 |
| 13 | 2.042 | 83 | -34.1000 | Water | 1.0000 ± 0.0050 | 282 | hct113 |
| 14 | 1.544 | 96 | -10.4000 | Water | 1.0000 ± 0.0050 | 189 | hct114 |
| 15 | 1.6477 | 75 | -12.2000 | Water | 1.0000 ± 0.0050 | 219 | hct115 |
| 16 | 1.5478 | 99 | 19.3675 | Water | 1.0001 ± 0.0044 | 204 | hct116 |
| 17 | 1.5478 | 114 | 19.3675 | Aqueous solution of boron | 1.0010 ± 0.0074 | 198 | hct117 |
| 18 | 1.5478 | 113 | 19.3675 | Aqueous solution of boron | 1.0000 ± 0.0074 | 200 | hct118 |
| 19 | 1.5478 | 133 | 19.3675 | Aqueous solution of boron | 1.0000 ± 0.0074 | 198 | hct119 |
| 20 | 1.5478 | 83 | 19.3675 | Aqueous solution of uranyl nitrate | 1.0001 ± 0.0047 | 302 | hct120 |
| 21 | 1.5478 | 133 | 19.3675 | Aqueous solution of uranyl nitrate and boron | 0.9997 ± 0.0076 | 192 | hct121 |

NOTE: [a] Negative water heights refer to distance below top of fuel.

[b] $H/^{235}U$ denotes Hydrogen to $^{235}U$ ratio and is reported in Table 5.1.3.3-1 of CRWMS M&O (1999a).

The following results are from CRWMS M&O (1999a, page 84). Table 5-10 presents the MCNP calculation results for ENDF/B-V and ENDF/B-VI cross section libraries for the HEU-

COMP-THERM-010 set of critical experiments. Table 5-11 shows the difference between experimental $k_{eff}$ results and MCNP calculated results and Table 5-12 shows the AENCF for the HEU-COMP-THERM-010 set of critical experiments. MCNP continuous energy simulations produced $k_{eff}$ values that were within 1% of 1.0, with the exception of cases 1 and 2 as the evaluators also noticed (NEA [2001]).

Table 5-10 LCE Reactivity Calculation Results for HCT-010 Experiments

| MCNP case Name | MCNP | MCNP | |
|---|---|---|---|
| | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-VI $k_{eff} \pm \sigma$ |
| Hct101 | 0.9858 ± 0.0013 | 0.98716 ± 0.00118 | 0.98784 ± 0.00112 |
| Hct102 | 0.9841 ± 0.0012 | 0.98361 ± 0.00115 | 0.98632 ± 0.00117 |
| Hct103 | 0.9927 ± 0.0011 | 0.99328 ± 0.00125 | 0.98998 ± 0.00117 |
| Hct104 | 0.9966 ± 0.0012 | 0.99468 ± 0.00125 | 0.99397 ± 0.00126 |
| Hct105 | 0.9983 ± 0.0012 | 0.99734 ± 0.00119 | 0.99568 ± 0.00113 |
| Hct106 | 0.9941 ± 0.0012 | 0.99239 ± 0.00121 | 0.99419 ± 0.00113 |
| Hct107 | 0.9992 ± 0.0011 | 0.99976 ± 0.00105 | 0.99841 ± 0.00113 |
| Hct108 | 1.0008 ± 0.0011 | 0.9991 ± 0.00122 | 0.99853 ± 0.00121 |
| Hct109 | 0.9976 ± 0.0012 | 0.99777 ± 0.00109 | 0.99798 ± 0.00116 |
| Hct110 | 1.0053 ± 0.0011 | 1.00198 ± 0.00104 | 1.00261 ± 0.00102 |
| Hct111 | 0.9999 ± 0.0011 | 0.99913 ± 0.00109 | 0.99882 ± 0.00111 |
| Hct112 | 0.9977 ± 0.0010 | 1.00052 ± 0.00105 | 0.99642 ± 0.00107 |
| Hct113 | 1.0037 ± 0.0011 | 1.00237 ± 0.001 | 1.0006 ± 0.0010 |
| Hct114 | 1.0008 ± 0.0012 | 1.00112 ± 0.00112 | 0.99823 ± 0.00113 |
| Hct115 | 1.0017 ± 0.0011 | 1.00059 ± 0.00113 | 0.99844 ± 0.00115 |
| Hct116 | 1.0044 ± 0.0011 | 1.00378 ± 0.00114 | 0.99892 ± 0.00115 |
| Hct117 | 1.0038 ± 0.0011 | 1.00377 ± 0.0011 | 1.00117 ± 0.00124 |
| Hct118 | 1.0039 ± 0.0011 | 1.0018 ± 0.00117 | 0.99989 ± 0.0012 |
| Hct119 | 1.0028 ± 0.0012 | 1.00151 ± 0.00105 | 0.99935 ± 0.00126 |
| Hct120 | 1.0051 ± 0.0009 | 1.0024 ± 0.001 | 1.00295 ± 0.00096 |
| Hct121 | 1.0044 ± 0.0011 | 1.00416 ± 0.0011 | 1.00250 ± 0.00111 |

Table 5-11 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for HEU-COMP-THERM-010 Experiments

| MCNP case name | ENDF/B-V $\Delta k_{eff} \pm 2\sigma$ | ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|
| hct101 | 0.01284 ± 0.010275 | 0.01216 ± 0.0103 |
| hct102 | 0.01639 ± 0.010261 | 0.01368 ± 0.0103 |
| hct103 | 0.00672 ± 0.010308 | 0.01002 ± 0.0103 |
| hct104 | 0.00532 ± 0.010308 | 0.00603 ± 0.0103 |
| hct105 | 0.00266 ± 0.010279 | 0.00432 ± 0.0103 |
| hct106 | 0.00761 ± 0.010289 | 0.00581 ± 0.0103 |
| hct107 | 0.00024 ± 0.010218 | 0.00159 ± 0.0103 |
| hct108 | 0.0009 ± 0.010293 | 0.00147 ± 0.0103 |
| hct109 | 0.00223 ± 0.010235 | 0.00202 ± 0.0103 |
| hct110 | -0.00198 ± 0.010214 | -0.0026 ± 0.0102 |
| hct111 | 0.00087 ± 0.010235 | 0.00118 ± 0.0102 |
| hct112 | -0.00052 ± 0.010218 | 0.00358 ± 0.0102 |
| hct113 | -0.00237 ± 0.010198 | -0.0006 ± 0.0102 |
| hct114 | -0.00112 ± 0.010248 | 0.00177 ± 0.0103 |
| hct115 | -0.00059 ± 0.010252 | 0.00156 ± 0.0103 |
| hct116 | -0.00378 ± 0.009091 | 0.00108 ± 0.0091 |
| hct117 | -0.00377 ± 0.014963 | -0.0012 ± 0.0150 |
| hct118 | -0.0018 ± 0.014984 | 0.00011 ± 0.0150 |
| hct119 | -0.00151 ± 0.014948 | 0.00065 ± 0.0150 |
| hct120 | -0.0024 ± 0.00961 | -0.00295 ± 0.00959 |
| hct121 | -0.00416 ± 0.015358 | -0.0025 ± 0.0097 |

Table 5-12 AENCF for HEU-COMP-THERM-010 Experiments (MeV)

| MCNP case name | ENDF/B-V | ENDF/B-VI |
|---|---|---|
| hct101 | 0.0797 | 0.0783 |
| hct102 | 0.0797 | 0.0800 |
| hct103 | 0.0557 | 0.0551 |
| hct104 | 0.0431 | 0.0423 |
| hct105 | 0.0360 | 0.0358 |
| hct106 | 0.0361 | 0.0358 |
| hct107 | 0.0308 | 0.0307 |

| Table 5-12 AENCF for HEU-COMP-THERM-010 Experiments (MeV) | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| hct108 | 0.0315 | 0.0309 |
| hct109 | 0.0314 | 0.0310 |
| hct110 | 0.0287 | 0.0282 |
| hct111 | 0.0291 | 0.0283 |
| hct112 | 0.0262 | 0.0257 |
| hct113 | 0.0250 | 0.0244 |
| hct114 | 0.0281 | 0.0280 |
| hct115 | 0.0289 | 0.0286 |
| hct116 | 0.0289 | 0.0281 |
| hct117 | 0.0293 | 0.0281 |
| hct118 | 0.0287 | 0.0283 |
| hct119 | 0.0295 | 0.0294 |
| hct120 | 0.0234 | 0.0228 |
| hct121 | 0.0278 | 0.0272 |

#### 5.1.1.4 Intermediate Heterogeneous Assembly with Highly Enriched Uranium Dioxide and Sand/Water Radial Reflector

An experiment was performed at the RRC "Kurchatov Institute" to investigate accidental sand and water immersion criticality safety of the thermionic intermediate reactor-converter with highly enriched fuel (approximately 96% $^{235}U$), zirconium hydride moderator, and end beryllium reflectors (NEA [2001, HEU-COMP-INTER-002]). Described in this category are five configurations of the critical assemblies simulating water ingress into different reactor cavities, as well as surrounding the reactor with sand and water. The experimental descriptions of each configuration are presented in Table 5-13. In configurations 1 and 2, wet sand is the reflector material and the rotating drums are filled with wet sand. All cavities are filled with water. The number of drums and the position of the control drums vary. In configurations 3, 4, and 5, the drum channels were removed, water is the radial reflector, and the number of drums and their positions vary.

| Table 5-13 Benchmarks for HEU-COMP-INTER -002 Class of Experiments | | | | | |
|---|---|---|---|---|---|
| **Experiment identifier** | **Material** | | | **Rotating drums** | |
| | **Radial reflector** | **Core cavities** | **$H/^{235}U$ ratio[a]** | **Total number** | **Position in assemblies[b]** |
| hci2-1 | $SiO_2$ $+H_2O$ | $H_2O$ | 14 | 12 | CD-5, φ=36° CD-1 ↑,CD-3 ↓, CD-6 ↓, the other CD and all SD ↑ |

Table 5-13 Benchmarks for HEU-COMP-INTER -002 Class of Experiments

| Experiment identifier | Material | | | Rotating drums | |
|---|---|---|---|---|---|
| | Radial reflector | Core cavities | $H/^{235}U$ ratio[a] | Total number | Position in assemblies[b] |
| hci2-2 | $SiO_2$ $+H_2O$ | $H_2O$ | 14 | 11 | CD-5, φ=121° CD-3 removed, CD-1, 2, 4, 6 ↓, CD-6 ↓, all SD ↑ |
| hci2-3 | $H_2O$ | $H_2O$ | 14 | 12 | CD-5, φ=119.5° the other CD and all SD ↑ |
| hci2-4 | $H_2O$ | $H_2O$ | 14 | 11 | CD-5, φ=95.5° CD-3 removed, the other CD and all SD ↑ |
| hci2-5 | $H_2O$ | $H_2O$ | 14 | 10 | CD-5, φ=76° CD-3 and SD-3 removed, the other CD and all SD ↑ |

NOTES: [a] $H/^{235}U$ denotes Hydrogen to $^{235}U$ ratio and is reported in Table 5.1.4.1-1 of CRWMS M&O (1999a).
[b] ↑ -Control drums turned out (φ=180°),
↓ -Control drums turned in (φ=0°),

The following results are from CRWMS M&O (1999a, page 83). Table 5-14 presents the MCNP calculation results for ENDF/B-V and ENDF/B-VI cross section libraries for the HEU-COMP-INTER-002 set of critical experiments. Table 5-15 shows the difference between referenced $k_{eff}$ results and the MCNP calculated results and Table 5-16 shows the AENCF for the HEU-COMP-INTER-002 set of critical experiments. The experimental benchmark shows that the MCNP code is able to predict the criticality of this experiment with a precision not worse than 0.68% (Table 5-15). This result is independent of the cross section library used.

Table 5-14 Criticality Calculation Results for HEU-COMP-INTER -002 Experiments

| MCNP Case name | MCNP | |
|---|---|---|
| | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-VI $k_{eff} \pm \sigma$ |
| hci2-1 | 0.99236 ± 0.00081 | 0.99933 ± 0.00081 |
| hci2-2 | 0.99339 ± 0.00107 | 1.00136 ± 0.00085 |
| hci2-3 | 0.99603 ± 0.00101 | 1.00422 ± 0.00087 |
| hci2-4 | 0.99676 ± 0.00101 | 1.00469 ± 0.00081 |
| hci2-5 | 0.99736 ± 0.001 | 1.00278 ± 0.00081 |

| Table 5-15 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for HEU-COMP-INTER -002 Experiments | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** $\Delta k_{eff} \pm 2\sigma$ | **ENDF/B-VI** $\Delta k_{eff} \pm 2\sigma$ |
| hci2-1 | 0.00764 ± 0.001807 | 0.00067 ± 0.0023 |
| hci2-2 | 0.00661 ± 0.002285 | -0.00140 ± 0.0023 |
| hci2-3 | 0.00397 ± 0.002173 | -0.00420 ± 0.0024 |
| hci2-4 | 0.00324 ± 0.002173 | -0.00470 ± 0.0023 |
| hci2-5 | 0.00264 ± 0.002154 | -0.00280 ± 0.0023 |

| Table 5-16 AENCF for HEU-COMP-INTER -002 Experiments | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| Hci2-1 | 0.2430 | 0.2354 |
| Hci2-2 | 0.2439 | 0.2357 |
| Hci2-3 | 0.2377 | 0.2313 |
| Hci2-4 | 0.2394 | 0.2312 |
| Hci2-5 | 0.2391 | 0.2313 |

### 5.1.2 HEU Benchmarks Related to Degraded Waste Package Configurations

The following summary descriptions of benchmarks previously evaluated in CRWMS M&O (1997) with results from CRWMS M&O (1999a) are applicable for HEU fuels.

#### 5.1.2.1 HEU Solution Configurations

A number of critical experiments involving solutions containing HEU are summarized in the following table. The concentration of fissile element in the solution, enrichment, reflector type and thickness, tank diameter, and solution height were among the parameters that were varied. The evaluation of these cases is documented in CRWMS M&O (1997 and 1999b). Table 5-17 shows basic characteristics of the experiments as reported in the relevant section of NEA (2001).

| Table 5-17 Benchmark Problem Summary for Configurations Incorporating Highly Enriched Uranium Nitrate Solutions | | | |
|---|---|---|---|
| **Reference Identifier** | **Experiment Identifier** | **Description**[a] | **Experimental** $k_{eff} \pm \sigma$ |
| HEU-SOL-THERM-001 | hest1-1 | 145.68 g U/liter, ID = 27.92 cm, SS-304 shell, Critical Height: 31.20, Critical Mass: 2782.8 gm. $H/^{235}U$ = 182 | 1.0000 ± 0.0025 |
| | hest1-2 | 346.73 g U/liter, ID = 27.92 cm, SS-304 shell, Critical Height: 31.20, Critical Mass: 2782.8 gm, $H/^{235}U$ = 182 | 1.0000 ± 0.0025 |

Table 5-17 Benchmark Problem Summary for Configurations Incorporating Highly Enriched Uranium Nitrate Solutions

| Reference Identifier | Experiment Identifier | Description[a] | Experimental $k_{eff} \pm \sigma$ |
|---|---|---|---|
| HEU-SOL-THERM-001 Minimally reflected cylinder with variable radius, 0.32 cm thick shell of type 304 stainless steel or type 6061 aluminum, 93.172 wt% $^{235}U$ | hest1-3 | 142.92 g U/liter, ID = 28.01 cm: , Al-6061 shell, Critical Height: 33.55, Critical Mass: 2954.6 gm, H/$^{235}U$ = 186 | 1.0000 ± 0.0025 |
| | hest1-4 | 357.71 g U/liter, ID = 28.01 cm:, Al-6061 shell, Critical Height: 30.91, Critical Mass: 6813.1 gm H/$^{235}U$ = 68 | 1.0000 ± 0.0025 |
| | hest1-5 | 54.89 g U/liter, ID = 33.01 cm:, Al-6061 shell, Critical Height: 39.48, Critical Mass: 1854.6 gm, H/$^{235}U$ =499 | 1.0000 ± 0.0025 |
| | hest1-6 | 59.65 g U/liter, ID = 33.01 cm:, Al-6061 shell Critical Height: 36.67, Critical Mass: 1872.0 gm, H/$^{235}U$ = 459 | 1.0000 ± 0.0025 |
| | hest1-7 | 137.40 g U/liter, ID = 33.01 cm:, Al-6061 shell, Critical Height: 23.96, Critical Mass: 2817.4 gm, H/$^{235}U$ = 193 | 1.0000 ± 0.0025 |
| | hest1-8 | 145.68 g U/liter, ID = 33.01 cm:, Al-6061 shell, Critical Height: 22.53, Critical Mass: 2951.1 gm, H/$^{235}U$ = 182 | 1.0000 ± 0.0025 |
| | hest1-9 | 357.71 g U/liter, ID = 33.01 cm:, Al-6061 shell, Critical Height: 22.53, Critical Mass: 6897.2 gm, H/$^{235}U$ = 68 | 1.0000 ± 0.0025 |
| | hest110 | 63.95 g U/liter, ID = 50.69 cm:, Al-6061 shell, Critical Height: 20.48, Critical Mass: 2643.0 gm, H/$^{235}U$ = 427 | 1.0000 ± 0.0025 |
| HEU-SOL-THERM-002 Concrete reflected cylinder with variable radius, 0.32 cm thick shell of type 304 stainless steel or type 6061 aluminum, 93.172 wt% $^{235}U$ | hest2-1 | 144.38 g U/liter, ID = 27.92 cm:, SS-304 shell, Critical Height: 29.79 Critical Mass:, cm: 2633.3 gm, H/$^{235}U$ = 184, 57.4 Critical Mass: North (N) X 64.6 cm East (E) X 82.0 cm Up (Position in Concrete Box) | 1.0000 ± 0.0020 |
| | hest2-2 | 144.38 g U/liter, ID = 27.92 cm, SS-304 shell, Critical Height: 24.19 cm, Critical Mass: 2138.3 gm, H/$^{235}U$ = 184, 1676 cm N X 104.49 cm E X 123.7 cm Up | 1.0000 ± 0.0020 |
| | hest2-3 | 334.77 g U/liter, ID = 27.92 cm, SS-304 shell, Critical Height: 27.23 Critical Mass: 5581.0 gm, H/$^{235}U$ = 74, 57.4 cm N X 64.6 cm E X 82.0 cm UP | 1.0000 ± 0.0020 |
| | hest2-4 | 334.77 g U/liter, ID = 27.92 cm, SS-304 shell, Critical Height: 21.79 cm, Critical Mass: 4466.1 gm, H/$^{235}U$ = 74, 16.76 cm N X 104.49 cm E X 123.7 cm Up | 1.0000 ± 0.0020 |
| | hest2-5 | 144.38 g U/liter, 28.01 cm, Al-6061 shell, Critical Height: 31..37 cm, Critical Mass: 2790.0 gm, H/$^{235}U$ = 184, 57.4 cm N X 64.6 cm E X 82.0 cm Up | 1.0000 ± 0.0020 |
| | hest2-6 | 144.38 g U/liter, ID = 18.01 cm, Al-6061 shell, Critical Height: 24.70 cm, Critical Mass: 2197.5 gm, H/$^{235}U$ = 184, 16.50 cm N X 104.0 cm E X 123.7 cm UP | 1.0000 ± 0.0020 |
| | hest2-7 | 334.77 g U/liter, ID = 28.01 cm, Al-6061 shell, Critical Height: 28.60 cm, Critical Mass: 5899.7 gm, H/$^{235}U$ = 74, 57.4 cm N X 64.6 cm E X 82.0 cm UP | 1.0000 ± 0.0020 |
| | hest2-8 | 334.77 g U/liter, ID = 28.01 cm Al-6061 shell,-6061 shell, Critical Height: 22.33 cm, Critical Mass: 4606.3 gm, H/$^{235}U$ = 74, 16.65 cm N X 103.75 cm E X 123.7 cm Up | 1.0000 ± 0.0020 |
| | hest2-9 | 59.65 g U/liter, ID = 33.01 cm, Al-6061 shell, Critical Height: 34.10 cm, Critical Mass: 1653.2 gm, H/$^{235}U$ = 460, 57.4 cm N X 64.6 cm E X 82.0 cm Up | 1.0000 ± 0.0020 |

Table 5-17 Benchmark Problem Summary for Configurations Incorporating Highly Enriched Uranium Nitrate Solutions

| Reference Identifier | Experiment Identifier | Description[a] | Experimental $k_{eff} \pm \sigma$ |
|---|---|---|---|
| HEU-SOL-THERM-002 | Hest2-10 | 59.65 g U/liter, ID = 33.01 cm, Al-6061 shell, Critical Height: 27.27 cm, Critical Mass: 1392.1 gm, H/$^{235}$U = 460, 16.825 cm N X 103.88 cm E X 123.7 cm Up | 1.0000 ± 0.0020 |
| | Hest2-11 | 144.38 g U/liter, ID = 33.01 cm, Al-6061 shell, Critical Height: 22.85 cm, Critical Mass: 2823.4 gm, H/$^{235}$U = 184, 57.4 cm N X 64.6 cm E X 82.0 cm UP | 1.0000 ± 0.0020 |
| | Hest2-12 | 144.38 g U/liter, ID = 33.01 cm, Al-6061 shell, Critical Height: 18.24 cm, Critical Mass: 2253.8 grm, H/$^{235}$U = 184, 16.825 cm N X !03.88 cm E X 123.7 cm Up | 1.0000 ± 0.0020 |
| | Hest2-13 | 334.77 g U/liter, ID = 33/01 cm, Al-6061 shell, Critical Height: 21.50 cm, Critical Mass: 6159.8 gm, H/$^{235}$U = 74, 57.4 cm N X 64.6 cm E X 82.0 cm Up | 1.0000 ± 0.0020 |
| | Hest2-14 | 334.77 g U/liter, ID = 33.01 cm, Al-6061 shell, Critical Height: 16.78 cm, Critical Mass: 4807.5 gm, H/$^{235}$U = 74, 16.825 cm N X !04.55 cm E X 123.7 cm Up | 1.0000 ± 0.0020 |
| HEU-SOL-THERM-003 Plexiglas reflected cylinder with variable radius, 0.32 cm thick shell of type 304 stainless steel or type 6061 aluminum, 93.172 wt% $^{235}$U | heust31 | 60.32 g U/liter, ID = 27.93 cm, SS-304 shell, Critical Height: 50.52 cm, Critical Mass: 1867.1 gm, H/$^{235}$U = 454, 17.71 cm N X 104.64 cm E X 122.9 cm UP | 1.0000 ± 0.0050 |
| | heust32 | 60.32 g U/liter, ID = 27.93 cm, SS-304 shell, Critical Height: 67.48 cm, Critical Mass: 2493.8 gm, H/$^{235}$U = 454, 61.1 cm N X 61.3 cm E X 122.9 cm Up | 1.0000 ± 0.0050 |
| | heust33 | 147.66 g U/liter, ID = 27.92 cm, SS-304 shell, Critical Height: 29.71 cm, Critical Mass: 2685.9 gm, H/$^{235}$U = 180, 60.4 cm N X 61.2 cm E X 81.4 cm Up | 1.0000 ± 0.0050 |
| | heust34 | 147.66 g U/liter, ID = cm, SS-304 shell, Critical Height: 25.03 cm, Critical Mass: 2262.8 gm H/$^{235}$U = 180, 17.06 cm N X 104.99 cm E X 122.9 cm UP | 1.0000 ± 0.0050 |
| | heust35 | 345.33 g U/liter, ID = 27.92 cm, SS-304 shell, Critical Height: 27.60 cm, Critical Mass: 5835.3 gm, H/$^{235}$U = 71, 60.4 cm N X 61.2 cm E X 81.4 cm UP | 1.0000 ± 0.0050 |
| | heust36 | 345.33 g U/liter, ID = 27.92 cm, SS-304 shell, Critical Height: 22.75 cm, Critical Mass: 4809.9 gm, H/$^{235}$U = 71, 16.81 cm N X 104.84 cm E X 122.9 cm Up | 1.0000 ± 0.0050 |
| | heust37 | 60.32 g U/liter, ID = 27.88 cm, Al-6061 shell, Critical Height: 51.67 cm, Critical Mass: 1902.7 gm, H/$^{235}$U = 454, 17.51 cm N X 105.15 cm E X 122.9 cm Up | 1.0000 ± 0.0050 |
| | heust38 | 147.66 g U/liter, ID = 28.01 cm, Al-6061 shell, Critical Height: 31.26 cm, Critical Mass: 2844.3 gm H/$^{235}$U = 180, 60.4 cm N X 61.2 cm E X 81.4 cm Up | 1.0000 ± 0.0050 |
| | heust39 | 147.66 g U/liter, ID = 28.01 cm, Al-6061 shell, Critical Height: 25.26 cm, Critical Mass: 2298.3 gm H/$^{235}$U = 180, 17.52 cm N X 104.95 cm E X 122.9 cm Up | 1.0000 ± 0.0050 |
| | hest310 | 345.33 g U/liter, ID = 28.01 cm, Al-6061 shell, Critical Height: 28.84 cm, Critical Mass: 6136.9 gm H/$^{235}$U = 71, 60.4 cm N X 61.2 E X 81.4 cm Up | 1.0000 ± 0.0050 |

Table 5-17 Benchmark Problem Summary for Configurations Incorporating Highly Enriched Uranium Nitrate Solutions

| Reference Identifier | Experiment Identifier | Description[a] | Experimental $k_{eff} \pm \sigma$ |
|---|---|---|---|
| HEU-SOL-THERM-003 | hest311 | 345.33 g U/liter, ID = 28.01 cm, Al-6061 shell, Critical Height: 22.87 cm, Critical Mass: 4866.5 gm, H/$^{235}$U = 71, 17.50 cm N X 105.1 cm E X 122.9 cm Up | 1.0000 ± 0.0050 |
| | hest312 | 60.32 g U/liter, ID = 33.01 cm, Al-6061 shell, Critical Height: 34.33 cm, Critical Mass: 1772.2 gm, H/$^{235}$U = 454, 60.4 cm N X 61.2 cm E X 81.4 cm Up | 1.0000 ± 0.0050 |
| | hest313 | 60.32 g U/liter, ID = 33.01 cm, Al-6061 shell, Critical Height: 27.70 cm, Critical Mass: 1430.0 gm, H/$^{235}$U = 454, 17.10 cm N X 105.18 cm E X 122.9 cm Up | 1.0000 ± 0.0050 |
| | hest314 | 60.32 g U/liter, ID = 33.01 cm, Al-6061 shell, Critical Height: 31.75 cm, Critical Mass: 1639.0 gm, H/$^{235}$U = 454, 61.1 cm N X 61.3 cm E X 122.9 cm Up | 1.0000 ± 0.0050 |
| | hest315 | 66.33 g U/liter, ID = 33.01 cm, Al-6061 shell, Critical Height: 25.10 cm, Critical Mass: 1424.8 gm, H/$^{235}$U = 412, 17.10 cm N X 105.18 cm E X 122.9 cm Up | 1.0000 ± 0.0050 |
| | hest316 | 147.66 g U/liter, ID = 33.01 cm, Al-6061 shell, Critical Height: 22.78 cm, Critical Mass: 2878.7 gm, H/$^{235}$U = 180, 60.4 cm N X 61.2 cm E X 81.4 cm Up | 1.0000 ± 0.0050 |
| | hest317 | 147.66 g U/liter, ID = 33.01 cm, Al-6061 shell, Critical Height: 18.49 cm, Critical Mass: 2336.6 gm, H/$^{235}$U = 180, 17.10 cm N X 105.18 cm E X 122.9 cm Up | 1.0000 ± 0.0050 |
| | hest318 | 345.33 g U/liter, ID = 33.01 cm, Al-6061 shell, Critical Height: 21.76 cm, Critical Mass: 6430.9 gm, H/$^{235}$U = 71, 60.4 cm N X 61.2 cm E X 81.4 cm Up | 1.0000 ± 0.0050 |
| | hest319 | 345.33 g U/liter, 33.01 cm, Al-6061 shell, Critical Height: 17.20 cm, Critical Mass: 5083.3 gm, H/$^{235}$U = 71, 17.30 cm N X 105.2 cm E X 122.9 cm Up | 1.0000 ± 0.0050 |
| HEU-SOL-THERM-008 Plexiglas reflected arrays of cylinders (30.48 cm pitch), 93.172 wt% $^{235}$U | heust81 | 4 X 4 Array, 60.32 g U/liter, ID = 21.12 cm, 0.40 cm thick shell of Type 6061 Aluminum, 0.32 cm thick sleeve of Type 304 Stainless Steel, Critical Height: 34.82 cm, Critical Mass: 735.8 gm X 16, H/$^{235}$U = 454 | 1.0000 ± 0.0030 |
| | heust83 | 4 X 4 Array, 60.32 g U/liter, ID = 21/12 cm, 0.40 cm thick shell of Type 6061 Aluminum, 0.0 cm thick sleeve of Type 304 Stainless Steel, Critical Height: 31.76 cm, Critical Mass: 671.1 gm X 16, H/$^{235}$U = 454 | 1.0000 ± 0.0030 |
| | heust86 | 2 X 2 Array, 355.94 g U/liter, ID = 21.12 cm, 0.40 cm thick shell of Type 6061 Aluminum, 0.32 cm thick sleeve of Type 304 Stainless Steel, Critical Height: 31.93 cm, Critical Mass: 3981.6 gm X 4, H/$^{235}$U = 69 | 1.0000 ± 0.0030 |
| | heust89 | 4 X 4 Array, 60.32 g U/liter, ID = 16.12 cm, 0.40 cm thick shell of Type 6061 Aluminum, 0.31 cm thick sleeve of Type 304 Stainless Steel, Critical Height: 105.85 cm, Critical Mass: 1303.1 gm X 16, H/$^{235}$U = 454 | 1.0000 ± 0.0030 |
| | hest813 | 2 X 3 Array, 355.94 g U/liter, ID = 16.12 cm, 0.40 cm thick shell of Type 6061 Aluminum, 0.31 cm thick sleeve of Type 304 Stainless Steel, Critical Height: 95.20 cm, Critical Mass: 1303.1 gm X 6, H/$^{235}$U = 69 | 1.0000 ± 0.0030 |

Table 5-17 Benchmark Problem Summary for Configurations Incorporating Highly Enriched Uranium Nitrate Solutions

| Reference Identifier | Experiment Identifier | Description[a] | Experimental $k_{eff} \pm \sigma$ |
|---|---|---|---|
| HEU-SOL-THERM-013 Unreflected 174-liter sphere poisoned with boric acid, 34.6 cm radius, 93.2 wt% $^{235}U$ | hest131 | 20.12 g U/liter, 0.0 g B/liter, Critical Mass = 3489.4 g, H/$^{235}U$ = 1375 | 1.0012 ± 0.0026 |
| | hest132 | 25.53 g U/liter, 0.0935 g B/liter, Critical Mass = 4427.6 g, H/$^{235}U$ = 1173 | 1.0007 ± 0.0036 |
| | hest133 | 26.77 g U/liter, 0.187 g B/liter, Critical Mass = 4642.7 g, H/$^{235}U$ = 1030 | 1.0009 ± 0.0036 |
| | hest134 | 28.45 g U/liter, 0.230 g B/liter, Critical Mass = 4934.0 g, H/$^{235}U$ = 971 | 1.0003 ± 0.0036 |
| HEU-SOL-THERM-014 Water reflected cylinders poisoned with Gd, 20 cm radius, 89.04 wt% $^{235}U$ | hest141 | 70.0 g U/liter, 0.0 g Gd/liter, Critical Height: 19.3 cm, Critical Mass: 1697.7 g, H/$^{235}U$ = 405 | 1.0000 ± 0.0028 |
| | hest142 | 68.1 g U/liter, 0.100 g Gd/liter, Critical Height: 27.4 cm, Critical Mass: 2344.8 g, H/$^{235}U$ = 418, Isotopic Gd Cross Sections | 1.0000 ± 0.0052 |
| | hest143 | 67.7 g U/liter, 0.193 g Gd/liter, Critical Height: 44.6 cm, Critical Mass: 3794.3 g, H/$^{235}U$ = 421, Isotopic Gd Cross Sections | 1.0000 ± 0.0087 |
| HEU-SOL-THERM-015 Water reflected cylinders poisoned with Gd, 20 cm radius, 89.04 wt% $^{235}U$ | hest151 | 100.5 g U/liter, 0.0 g Gd/liter, Critical Height: 18.7 cm, Critical Mass: 2361.7 g, H/$^{235}U$ = 278, 15.1 cm High Reflector | 1.0000 ± 0.0032 |
| | hest152 | 100.5 g U/liter, 0.0 g Gd/liter, Critical Height: 16.6 cm, Critical Mass: 2096.4 g, H/$^{235}U$ = 278, 45.0 cm High Reflector | 1.0000 ± 0.0034 |
| | hest153 | 98.8 g U/liter, 0.197 g Gd/liter, Critical Height: 29.9 cm, Critical Mass: 3712.3 g, H/$^{235}U$ = 283, 17.0 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0068 |
| | hest154 | 98.8 g U/liter, 0.197 g Gd/liter, Critical Height: 25.8 cm, Critical Mass: 3203.2 g, H/$^{235}U$ = 283, 45.0 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0069 |
| | hest155 | 95.2 g U/liter, 0.400 g Gd/liter, Critical Height: 51.2 cm, Critical Mass: 6125.2 g, H/$^{235}U$ = 295, 47.0 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0089 |
| HEU-SOL-THERM-016 Water reflected cylinders poisoned with Gd, 20 cm radius, 89.04 wt% $^{235}U$ | hest161 | 156.5 g U/liter, 0.0 g Gd/liter, Critical Height: 15.1 cm, Critical Mass: 2969.6 g, H/$^{235}U$ = 175 | 1.0000 ± 0.0036 |
| | hest162 | 143.9 g U/liter, 0.300 g Gd/liter, Critical Height: 22.1 cm, Critical Mass: 3988.0 g, H/$^{235}U$ = 192, Isotopic Gd Cross Sections | 1.0000 ± 0.0069 |
| | hest163 | 144.2 g U/liter, 0.525 g Gd/liter, Critical Height: 33.0 cm, Critical Mass: 5979.8 g, H/$^{235}U$ = 191, Isotopic Gd Cross Sections | 1.0000 ± 0.0079 |
| HEU-SOL-THERM-017 | hest171 | 202.4 g U/liter, 0.0 g Gd/liter, 12.4 cm Inner Radius, Critical Height: 22.6 cm, Critical Mass: 2776.7 g, H/$^{235}U$ = 133, 21.6 cm High Reflector | 1.0000 ± 0.0028 |
| | hest172 | 202.4 g U/liter, 0.0 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 15.6 cm, Critical Mass: 3967.8 g, H/$^{235}U$ = 133, 0.0 cm High Reflector | 1.0000 ± 0.0040 |
| | hest173 | 202.4 g U/liter, 0.0 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 14.3 cm, Critical Mass: 3637.1 g, H/$^{235}U$ = 133, 16.3 cm High Reflector | 1.0000 ± 0.0036 |

Table 5-17 Benchmark Problem Summary for Configurations Incorporating Highly Enriched Uranium Nitrate Solutions

| Reference Identifier | Experiment Identifier | Description[a] | Experimental $k_{eff} \pm \sigma$ |
|---|---|---|---|
| HEU-SOL-THERM-017 Water reflected cylinders poisoned with Gd, 89.04 wt% $^{235}U$ | hest174 | 196.2 g U/liter, 0.298 g Gd/liter, 12.4 cm cm Inner Radius, Critical Height: 34.8 cm, Critical Mass: 3298.2 g, H/$^{235}U$ = 137, 45.4 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0047 |
| | hest175 | 192.0 g U/liter, 0.497 g Gd/liter, 12.4 cm Inner Radius, Critical Height: 64.2 cm, Critical Mass: 5954.3 g, H/$^{235}U$ = 141, 71.5 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0058 |
| | hest176 | 192.0 g U/liter, 0.497 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 25.7 cm, Critical Mass: 6200.7 g, H/$^{235}U$ = 141, 0.0 cm High Reflector Isotopic Gd Cross Sections | 1.0000 ± 0.0055 |
| | hest177 | 192.0 g U/liter, 0.497 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 22.1 cm, Critical Mass: 5332.20 g, H/$^{235}U$ = 141, 46.4 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0057 |
| | hest178 | 186.2 g U/liter, 0.790 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 37.7 cm, Critical Mass: 8821.3 g, H/$^{235}U$ = 147, 0.0 cm High Reflector Isotopic Gd Cross Sections | 1.0000 ± 0.0067 |
| HEU-SOL-THERM-018 Water reflected cylinders poisoned with Gd, 89.04 wt% $^{235}U$ | hest181 | 300.0 g U/liter, 0.0 g Gd/liter, 12.4 cm Inner Radius, Critical Height: 21.7 cm, Critical Mass: 3144.7 g, H/$^{235}U$ = 86, 21.6 cm High Reflector | 1.0000 ± 0.0034 |
| | hest182 | 300.0 g U/liter, 0.0 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 15.5 cm, Critical Mass: 5843.4 g, H/$^{235}U$ = 86, 0.0 cm High Reflector | 1.0000 ± 0.0046 |
| | hest183 | 300.0 g U/liter, 0.0 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 14.3 cm, Critical Mass: 5291.0 g, H/$^{235}U$ = 86, 14.3 cm High Reflector | 1.0000 ± 0.0042 |
| | hest184 | 291.3 g U/liter, 0.497 g Gd/liter, 12.4 cm Inner Radius, Critical Height: 34.8 cm, Critical Mass: 3298.2 H/$^{235}U$ = 89, 45.4 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0044 |
| | hest185 | 291.3 g U/liter, 0.497 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 20.2 cm, Critical Mass: 7394.4 g, H/$^{235}U$ = 89, 0.0 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0046 |
| | hest186 | 291.3 g U/liter, 0.497 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 17.9 cm, Critical Mass: 6552.4 g, H/$^{235}U$ = 89, 21.4 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0045 |
| | hest187 | 283.3 g U/liter, 0.497 g Gd/liter, 12.4 cm Inner Radius, Critical Height: 83.2 cm, Critical Mass: 11385.8 g, H/$^{235}U$ = 92, 81.1 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0058 |
| | hest188 | 283.3 g U/liter, 0.977 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 28.1 cm, Critical Mass: 10003.7 g, H/$^{235}U$ = 92, 0.0 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0056 |
| | hest189 | 283.3 g U/liter, 0.977 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 23.3 cm, Critical Mass: 8294.9 g, H/$^{235}U$ = 91, 31.2 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0056 |
| | hst1810 | 285.3 g U/liter, 1.400 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 41.4 cm, Critical Mass: 14842.7 g, H/$^{235}U$ = 91, 0.0 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0057 |

Table 5-17 Benchmark Problem Summary for Configurations Incorporating Highly Enriched Uranium Nitrate Solutions

| Reference Identifier | Experiment Identifier | Description[a] | Experimental $k_{eff} \pm \sigma$ |
|---|---|---|---|
| HEU-SOL-THERM-018 | hst1811 | 285.3 g U/liter, 1.400 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 31.5 cm, Critical Mass: 11293.3 g, $H/^{235}U$ = 91, 55.4 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0059 |
| | hst1812 | 279.6 g U/liter, 1.943 g Gd/liter, 20.0 cm Inner Radius, Critical Height: 48.4 cm, Critical Mass: 17146.2 g, $H/^{235}U$ = 94, 59.4 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0065 |
| HEU-SOL-THERM-019 Water reflected cylinders poisoned with Gd, 89.04 wt% $^{235}U$ | hest191 | 447.3 g U/liter, 0.0 g Gd/liter, 12.4 cm Inner Radius, Critical Height: 22.6 cm, Critical Mass: 4883.2 g, $H/^{235}U$ = 55, 21.6 cm High Reflector | 1.0000 ± 0.0041 |
| | hest192 | 393.6 g U/liter, 0.647 g Gd/liter, 12.4 cm Inner Radius, Critical Height: 31.5 cm, Critical Mass: 5989.1 g, $H/^{235}U$ = 63, 42.4 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0041 |
| | hest193 | 400.0 g U/liter, 1.160 g Gd/liter, 12.4 cm Inner Radius, Critical Height: 45.5 cm, Critical Mass: 8791.5 g, $H/^{235}U$ = 61, 59.4 cm High Reflector, Isotopic Gd Cross Sections | 1.0000 ± 0.0067 |

NOTE: [a] $H/^{235}U$ denotes Hydrogen to $^{235}U$ ratio and is reported in the description of the experiments provided in NEA (2001).

The keff results shown in Table 5-18 are reported from CRWMS M&O (1999b, Table 6-1). The difference between the experimental and calculated keff values shown in Table 5-18 are calculated as the difference between the relevant values in Table 5-17 and Table 5-18.

Table 5-18 Results of Criticality Calculations for HEU Solutions

| Case | ENDF/B-VI | | | ENDF/B-V | | | ENDF/B-VI | | | ENDF/B-V | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | AENCF | $\Delta k_{eff} \pm 2\sigma$ | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| hest1-1 | 0.99683 | 0.00138 | 0.01567 | 1.00241 | 0.00131 | 0.01582 | 0.0032 | ± | 0.0057 | -0.0024 | ± | 0.0056 |
| hest110 | 0.98954 | 0.0018 | 0.00758 | 0.99468 | 0.00178 | 0.00757 | 0.0105 | ± | 0.0062 | 0.0053 | ± | 0.0061 |
| hest1-2 | 0.99891 | 0.00208 | 0.03779 | 0.99816 | 0.00209 | 0.03873 | 0.0011 | ± | 0.0065 | 0.0018 | ± | 0.0065 |
| hest1-3 | 1.00161 | 0.00194 | 0.01565 | 1.00453 | 0.00199 | 0.01546 | -0.0016 | ± | 0.0063 | -0.0045 | ± | 0.0064 |
| hest131 | 0.99705 | 0.00055 | 0.00260 | 1.00135 | 0.0005 | 0.00267 | 0.0042 | ± | 0.0053 | -0.0001 | ± | 0.0053 |
| hest132 | 0.99678 | 0.00056 | 0.00307 | 1.0002 | 0.00057 | 0.00307 | 0.0039 | ± | 0.0073 | 0.0005 | ± | 0.0073 |
| hest133 | 0.99282 | 0.0006 | 0.00354 | 0.99689 | 0.00065 | 0.00378 | 0.0081 | ± | 0.0073 | 0.0040 | ± | 0.0073 |
| hest134 | 0.99519 | 0.00062 | 0.00381 | 0.99777 | 0.00061 | 0.00374 | 0.0051 | ± | 0.0073 | 0.0025 | ± | 0.0073 |
| hest1-4 | 1.00107 | 0.00216 | 0.03945 | 1.0013 | 0.00203 | 0.04050 | -0.0011 | ± | 0.0066 | -0.0013 | ± | 0.0064 |

Table 5-18 Results of Criticality Calculations for HEU Solutions

| Case | ENDF/B-VI | | | ENDF/B-V | | | ENDF/B-VI | | | ENDF/B-V | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF | $\Delta k_{eff} \pm 2\sigma$ | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| hest141 | 0.99204 | 0.00126 | 0.00708 | 0.99854 | 0.00119 | 0.00716 | 0.0080 | ± | 0.0061 | 0.0015 | ± | 0.0061 |
| hest142 | 1.00965 | 0.00106 | 0.00710 | 1.01307 | 0.00115 | 0.00730 | -0.0096 | ± | 0.0106 | -0.0131 | ± | 0.0107 |
| hest143 | 1.01882 | 0.00109 | 0.00756 | 1.02137 | 0.00109 | 0.00775 | -0.0188 | ± | 0.0175 | -0.0214 | ± | 0.0175 |
| hest1-5 | 0.99849 | 0.00159 | 0.00619 | 1.00361 | 0.00166 | 0.00651 | 0.0015 | ± | 0.0059 | -0.0036 | ± | 0.0060 |
| hest151 | 0.99834 | 0.00128 | 0.01048 | 1.0058 | 0.00123 | 0.01056 | 0.0017 | ± | 0.0069 | -0.0058 | ± | 0.0069 |
| hest152 | 0.98944 | 0.00128 | 0.00992 | 0.99295 | 0.00125 | 0.00998 | 0.0106 | ± | 0.0073 | 0.0071 | ± | 0.0072 |
| hest153 | 1.00941 | 0.00121 | 0.01122 | 1.00874 | 0.0012 | 0.01127 | -0.0094 | ± | 0.0138 | -0.0087 | ± | 0.0138 |
| hest154 | 1.01185 | 0.00122 | 0.01032 | 1.01579 | 0.00115 | 0.01040 | -0.0118 | ± | 0.0140 | -0.0158 | ± | 0.0140 |
| hest155 | 1.01178 | 0.00112 | 0.01077 | 1.01277 | 0.00116 | 0.01133 | -0.0118 | ± | 0.0179 | -0.0128 | ± | 0.0180 |
| hest1-6 | 1.00073 | 0.00177 | 0.00712 | 1.01038 | 0.00187 | 0.00678 | -0.0007 | ± | 0.0061 | -0.0104 | ± | 0.0062 |
| hest161 | 0.98628 | 0.00129 | 0.01531 | 0.9922 | 0.00137 | 0.01513 | 0.0137 | ± | 0.0076 | 0.0078 | ± | 0.0077 |
| hest162 | 1.01039 | 0.0013 | 0.01487 | 1.01098 | 0.00122 | 0.01539 | -0.0104 | ± | 0.0140 | -0.0110 | ± | 0.0140 |
| hest163 | 1.01723 | 0.00122 | 0.01593 | 1.02046 | 0.00119 | 0.01608 | -0.0172 | ± | 0.0160 | -0.0205 | ± | 0.0160 |
| hest1-7 | 0.99985 | 0.002 | 0.01526 | 1.0023 | 0.00201 | 0.01501 | 0.0001 | ± | 0.0064 | -0.0023 | ± | 0.0064 |
| hest171 | 0.9909 | 0.00118 | 0.01885 | 0.99352 | 0.00125 | 0.01892 | 0.0091 | ± | 0.0061 | 0.0065 | ± | 0.0061 |
| hest172 | 0.97993 | 0.00126 | 0.02095 | 0.98387 | 0.0015 | 0.02097 | 0.0201 | ± | 0.0084 | 0.0161 | ± | 0.0085 |
| hest173 | 0.97985 | 0.00141 | 0.01969 | 0.98507 | 0.00134 | 0.01969 | 0.0202 | ± | 0.0077 | 0.0149 | ± | 0.0077 |
| hest174 | 1.0005 | 0.00134 | 0.01877 | 1.0036 | 0.00131 | 0.01947 | -0.0005 | ± | 0.0098 | -0.0036 | ± | 0.0098 |
| hest175 | 1.00821 | 0.00124 | 0.01927 | 1.01001 | 0.00121 | 0.01973 | -0.0082 | ± | 0.0119 | -0.0100 | ± | 0.0118 |
| hest176 | 1.00388 | 0.00138 | 0.02132 | 1.00653 | 0.00109 | 0.02183 | -0.0039 | ± | 0.0113 | -0.0065 | ± | 0.0112 |
| hest177 | 1.00984 | 0.00129 | 0.02007 | 1.01111 | 0.00121 | 0.02087 | -0.0098 | ± | 0.0117 | -0.0111 | ± | 0.0117 |
| hest178 | 1.00346 | 0.00122 | 0.02201 | 1.00409 | 0.00112 | 0.02210 | -0.0035 | ± | 0.0136 | -0.0041 | ± | 0.0136 |
| hest1-8 | 0.99318 | 0.0021 | 0.01608 | 1.00505 | 0.00213 | 0.01610 | 0.0068 | ± | 0.0065 | -0.0051 | ± | 0.0066 |
| hest181 | 0.9897 | 0.00125 | 0.02783 | 0.99364 | 0.00143 | 0.02846 | 0.0103 | ± | 0.0072 | 0.0064 | ± | 0.0074 |
| hest182 | 0.99139 | 0.00122 | 0.03053 | 0.99331 | 0.00142 | 0.03121 | 0.0086 | ± | 0.0095 | 0.0067 | ± | 0.0096 |

Table 5-18 Results of Criticality Calculations for HEU Solutions

| Case | ENDF/B-VI | | | ENDF/B-V | | | ENDF/B-VI | | | ENDF/B-V | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF | $\Delta k_{eff} \pm 2\sigma$ | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| hest183 | 0.98874 | 0.00132 | 0.02927 | 0.99025 | 0.00136 | 0.02976 | 0.0113 | ± | 0.0088 | 0.0098 | ± | 0.0088 |
| hest184 | 0.99834 | 0.00125 | 0.02829 | 1.00177 | 0.00133 | 0.02919 | 0.0017 | ± | 0.0091 | -0.0018 | ± | 0.0092 |
| hest185 | 0.9937 | 0.00142 | 0.03222 | 0.99727 | 0.00141 | 0.03274 | 0.0063 | ± | 0.0096 | 0.0027 | ± | 0.0096 |
| hest186 | 0.99258 | 0.00135 | 0.03063 | 0.99699 | 0.00129 | 0.03086 | 0.0074 | ± | 0.0094 | 0.0030 | ± | 0.0094 |
| hest187 | 1.00895 | 0.00119 | 0.02915 | 1.01204 | 0.00118 | 0.02988 | -0.0090 | ± | 0.0118 | -0.0120 | ± | 0.0118 |
| hest188 | 1.01154 | 0.00133 | 0.03256 | 1.01266 | 0.00117 | 0.03286 | -0.0115 | ± | 0.0115 | -0.0127 | ± | 0.0114 |
| hest189 | 1.00848 | 0.00127 | 0.03024 | 1.00855 | 0.00114 | 0.03109 | -0.0085 | ± | 0.0115 | -0.0086 | ± | 0.0114 |
| hest1-9 | 0.99817 | 0.00193 | 0.03977 | 0.99973 | 0.00212 | 0.04099 | 0.0018 | ± | 0.0063 | 0.0003 | ± | 0.0066 |
| hest191 | 1.00146 | 0.00133 | 0.04157 | 1.00102 | 0.00129 | 0.04249 | -0.0015 | ± | 0.0086 | -0.0010 | ± | 0.0086 |
| hest192 | 1.0043 | 0.00132 | 0.03851 | 1.00497 | 0.00116 | 0.03928 | -0.0043 | ± | 0.0086 | -0.0050 | ± | 0.0085 |
| hest193 | 0.99996 | 0.0012 | 0.04099 | 1.00044 | 0.00122 | 0.04166 | 0.0000 | ± | 0.0136 | -0.0004 | ± | 0.0136 |
| hest2-1 | 1.00211 | 0.00144 | 0.01534 | 1.00548 | 0.00148 | 0.01558 | -0.0021 | ± | 0.0049 | -0.0055 | ± | 0.0050 |
| hest210 | 0.99938 | 0.00196 | 0.00674 | 1.00937 | 0.00202 | 0.00663 | 0.0006 | ± | 0.0056 | -0.0094 | ± | 0.0057 |
| hest211 | 1.00142 | 0.00216 | 0.01577 | 1.00875 | 0.00211 | 0.01595 | -0.0014 | ± | 0.0059 | -0.0088 | ± | 0.0058 |
| hest212 | 1.00489 | 0.00211 | 0.01480 | 1.0127 | 0.00209 | 0.01487 | -0.0049 | ± | 0.0058 | -0.0127 | ± | 0.0058 |
| hest213 | 1.00223 | 0.0025 | 0.03659 | 0.99869 | 0.00232 | 0.03676 | -0.0022 | ± | 0.0064 | 0.0013 | ± | 0.0061 |
| hest214 | 1.0051 | 0.00232 | 0.03363 | 1.01062 | 0.00238 | 0.03377 | -0.0051 | ± | 0.0061 | -0.0106 | ± | 0.0062 |
| hest2-2 | 1.00627 | 0.00237 | 0.01434 | 1.00773 | 0.00235 | 0.01516 | -0.0063 | ± | 0.0062 | -0.0077 | ± | 0.0062 |
| hest2-3 | 0.99713 | 0.00256 | 0.03639 | 1.00219 | 0.0022 | 0.03740 | 0.0029 | ± | 0.0065 | -0.0022 | ± | 0.0059 |
| hest2-4 | 1.00836 | 0.00213 | 0.03443 | 1.00809 | 0.00242 | 0.03541 | -0.0084 | ± | 0.0058 | -0.0081 | ± | 0.0063 |
| hest2-5 | 1.00531 | 0.00217 | 0.01588 | 1.01049 | 0.0023 | 0.01622 | -0.0053 | ± | 0.0059 | -0.0105 | ± | 0.0061 |
| hest2-6 | 1.00976 | 0.00233 | 0.01493 | 1.00968 | 0.00215 | 0.01496 | -0.0098 | ± | 0.0061 | -0.0097 | ± | 0.0059 |
| hest2-7 | 0.99994 | 0.00248 | 0.03548 | 1.00691 | 0.00224 | 0.03747 | 0.0001 | ± | 0.0064 | -0.0069 | ± | 0.0060 |
| hest2-8 | 1.00552 | 0.0026 | 0.03454 | 1.01131 | 0.00206 | 0.03511 | -0.0055 | ± | 0.0066 | -0.0113 | ± | 0.0057 |
| hest2-9 | 0.9992 | 0.00201 | 0.00658 | 1.00348 | 0.00209 | 0.00654 | 0.0008 | ± | 0.0057 | -0.0035 | ± | 0.0058 |

Table 5-18 Results of Criticality Calculations for HEU Solutions

| Case | ENDF/B-VI | | | ENDF/B-V | | | ENDF/B-VI | | | ENDF/B-V | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF | $\Delta k_{eff} \pm 2\sigma$ | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| hest310 | 0.99709 | 0.00238 | 0.03810 | 1.00102 | 0.00243 | 0.03817 | 0.0029 | ± | 0.0111 | -0.0010 | ± | 0.0111 |
| hest311 | 1.00244 | 0.00231 | 0.03476 | 1.00606 | 0.00232 | 0.03566 | -0.0024 | ± | 0.0110 | -0.0061 | ± | 0.0110 |
| hest312 | 1.00061 | 0.00233 | 0.00715 | 1.0074 | 0.00204 | 0.00651 | -0.0006 | ± | 0.0110 | -0.0074 | ± | 0.0108 |
| hest313 | 1.00074 | 0.00183 | 0.00634 | 1.00045 | 0.00185 | 0.00654 | -0.0007 | ± | 0.0106 | -0.0005 | ± | 0.0107 |
| hest314 | 0.99809 | 0.00207 | 0.00680 | 1.00822 | 0.00205 | 0.00704 | 0.0019 | ± | 0.0108 | -0.0082 | ± | 0.0108 |
| hest315 | 0.9936 | 0.00143 | 0.00712 | 0.99675 | 0.0015 | 0.00704 | 0.0064 | ± | 0.0104 | 0.0032 | ± | 0.0104 |
| hest316 | 0.99192 | 0.00238 | 0.01574 | 1.00356 | 0.00241 | 0.01593 | 0.0081 | ± | 0.0111 | -0.0036 | ± | 0.0111 |
| hest317 | 1.00364 | 0.00237 | 0.01490 | 1.00604 | 0.00213 | 0.01498 | -0.0036 | ± | 0.0111 | -0.0060 | ± | 0.0109 |
| hest318 | 0.99356 | 0.00225 | 0.03806 | 1.00007 | 0.00225 | 0.03842 | 0.0064 | ± | 0.0110 | -0.0001 | ± | 0.0110 |
| hest319 | 1.00837 | 0.00213 | 0.03313 | 1.01306 | 0.00225 | 0.03414 | -0.0084 | ± | 0.0109 | -0.0131 | ± | 0.0110 |
| hest813 | 1.00096 | 0.00098 | 0.03511 | 1.00331 | 0.002 | 0.03616 | -0.0010 | ± | 0.0063 | -0.0033 | ± | 0.0072 |
| heust31 | 0.99825 | 0.00196 | 0.00682 | 1.00638 | 0.00189 | 0.00676 | 0.0018 | ± | 0.0107 | -0.0064 | ± | 0.0107 |
| heust32 | 0.99973 | 0.00215 | 0.00702 | 1.00635 | 0.002 | 0.00688 | 0.0003 | ± | 0.0109 | -0.0064 | ± | 0.0108 |
| heust33 | 1.00172 | 0.00241 | 0.01538 | 1.00317 | 0.00235 | 0.01580 | -0.0017 | ± | 0.0111 | -0.0032 | ± | 0.0110 |
| heust34 | 1.0022 | 0.00221 | 0.01494 | 1.00447 | 0.00249 | 0.01541 | -0.0022 | ± | 0.0109 | -0.0045 | ± | 0.0112 |
| heust35 | 0.99927 | 0.00243 | 0.03670 | 1.00436 | 0.00221 | 0.03818 | 0.0007 | ± | 0.0111 | -0.0044 | ± | 0.0109 |
| heust36 | 1.00274 | 0.00239 | 0.03419 | 1.00427 | 0.00241 | 0.03573 | -0.0027 | ± | 0.0111 | -0.0043 | ± | 0.0111 |
| heust37 | 0.99568 | 0.00184 | 0.00651 | 1.00585 | 0.00173 | 0.00685 | 0.0043 | ± | 0.0107 | -0.0058 | ± | 0.0106 |
| heust38 | 1.00251 | 0.00227 | 0.01597 | 1.01086 | 0.00202 | 0.01639 | -0.0025 | ± | 0.0110 | -0.0109 | ± | 0.0108 |
| heust39 | 0.99982 | 0.00216 | 0.01501 | 1.01063 | 0.00204 | 0.01512 | 0.0002 | ± | 0.0109 | -0.0106 | ± | 0.0108 |
| heust81 | 0.99928 | 0.0012 | 0.00651 | 1.00316 | 0.00134 | 0.00661 | 0.0007 | ± | 0.0065 | -0.0032 | ± | 0.0066 |
| heust83 | 0.99185 | 0.00116 | 0.00642 | 0.9973 | 0.0019 | 0.00644 | 0.0081 | ± | 0.0064 | 0.0027 | ± | 0.0071 |
| heust86 | 1.00546 | 0.00215 | 0.03642 | 1.00969 | 0.0023 | 0.03669 | -0.0055 | ± | 0.0074 | -0.0097 | ± | 0.0076 |
| heust89 | 0.9977 | 0.00115 | 0.00610 | 1.00373 | 0.00116 | 0.00660 | 0.0023 | ± | 0.0064 | -0.0037 | ± | 0.0064 |
| hst1810 | 1.02652 | 0.00132 | 0.03392 | 1.02526 | 0.00114 | 0.03464 | -0.0265 | ± | 0.0117 | -0.0253 | ± | 0.0116 |

Table 5-18 Results of Criticality Calculations for HEU Solutions

| Case | ENDF/B-VI | | | ENDF/B-V | | | ENDF/B-VI | | | ENDF/B-V | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF | $\Delta k_{eff} \pm 2\sigma$ | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| hst1811 | 1.02612 | 0.00102 | 0.03183 | 1.02814 | 0.00115 | 0.03203 | -0.0261 | ± | 0.0120 | -0.0281 | ± | 0.0120 |
| hst1812 | 1.02152 | 0.00115 | 0.03246 | 1.01861 | 0.00101 | 0.03353 | -0.0215 | ± | 0.0132 | -0.0186 | ± | 0.0132 |
| hxst142 | 1.00955 | 0.00106 | 0.00713 | 1.01307 | 0.00115 | 0.00730 | -0.0095 | ± | 0.0106 | -0.0131 | ± | 0.0107 |
| hxst143 | 1.02047 | 0.00113 | 0.00746 | 1.01953 | 0.00105 | 0.00781 | -0.0205 | ± | 0.0175 | -0.0195 | ± | 0.0175 |
| hxst153 | 1.00941 | 0.00121 | 0.01122 | 1.00874 | 0.0012 | 0.01127 | -0.0094 | ± | 0.0138 | -0.0087 | ± | 0.0138 |
| hxst154 | 1.01193 | 0.00107 | 0.01030 | 1.01453 | 0.00121 | 0.01056 | -0.0119 | ± | 0.0140 | -0.0145 | ± | 0.0140 |
| hxst155a | 1.00996 | 0.00121 | 0.01100 | 1.00925 | 0.00103 | 0.01112 | -0.0100 | ± | 0.0180 | -0.0092 | ± | 0.0179 |
| hxst162 | 1.00878 | 0.00128 | 0.01471 | 1.00991 | 0.00126 | 0.01520 | -0.0088 | ± | 0.0140 | -0.0099 | ± | 0.0140 |
| hxst163 | 1.01933 | 0.00114 | 0.01609 | 1.01915 | 0.00126 | 0.01625 | -0.0193 | ± | 0.0160 | -0.0192 | ± | 0.0160 |
| hxst174 | 0.99792 | 0.00125 | 0.01915 | 1.0036 | 0.00131 | 0.01947 | 0.0021 | ± | 0.0097 | -0.0036 | ± | 0.0098 |
| hxst175 | 1.00629 | 0.00117 | 0.01927 | 1.01033 | 0.00117 | 0.01940 | -0.0063 | ± | 0.0118 | -0.0103 | ± | 0.0118 |
| hxst178 | 1.00357 | 0.0011 | 0.02196 | 1.00537 | 0.00115 | 0.02178 | -0.0036 | ± | 0.0136 | -0.0054 | ± | 0.0136 |
| hxst184 | 0.99742 | 0.00122 | 0.02839 | 1.00149 | 0.00127 | 0.02925 | 0.0026 | ± | 0.0091 | -0.0015 | ± | 0.0092 |
| hxst187 | 1.00895 | 0.00119 | 0.02915 | 1.01204 | 0.00118 | 0.02988 | -0.0090 | ± | 0.0118 | -0.0120 | ± | 0.0118 |
| hxst192 | 1.00305 | 0.00134 | 0.03842 | 1.00187 | 0.00126 | 0.03923 | -0.0031 | ± | 0.0086 | -0.0019 | ± | 0.0086 |
| hxst193 | 0.99891 | 0.0013 | 0.04065 | 1.00049 | 0.00117 | 0.04126 | 0.0011 | ± | 0.0136 | -0.0005 | ± | 0.0136 |
| hxt1810 | 1.02416 | 0.00129 | 0.03394 | 1.02528 | 0.00126 | 0.03491 | -0.0242 | ± | 0.0117 | -0.0253 | ± | 0.0117 |
| hxt1812 | 1.02061 | 0.00114 | 0.03201 | 1.0221 | 0.00118 | 0.03353 | -0.0206 | ± | 0.0132 | -0.0221 | ± | 0.0132 |

### 5.1.2.2 HEU Fast Metal Fuel experiments

Reactivity calculations involving fast metal fuel evaluated in CRWMS M&O (1999a) are presented in this section. A series of critical experiments were performed by the Institute for Experimental Physics of the Russian Federal Nuclear Center (VNIIEF) at Arzamas-16 and the Institute for Technical Physics of the Russian Federal Nuclear Center (VNIITF) at Chelyabinsk-70. Detailed experimental configurations and material compositions for these experiments are included in NEA (2001). The series of experiments studied in NEA (2001) include uranium metal systems of high enrichment reflected by the following materials: depleted uranium, steel, aluminum, graphite, and polyethylene. Table 5-19 provides a listing of the Reference NEA

(2001) identifier numbers, some information about the experiment, the benchmark $k_{eff}$ value, and the MCNP case names for these experiments.

Table 5-19 Russian Criticality Safety Benchmark Experiments

| Handbook identifier | $^{235}U$ wt% or $^{239}Pu$ at% | Core dimensions (cm) | | Reflector material | Reflector thickness (cm) | Benchmark $k_{eff}$ | MCNP case name |
|---|---|---|---|---|---|---|---|
| | | Radius | Height | | | | |
| HEU-MET-FAST-008 | 90 | 10.150 | N/A | None | -- | 0.9989 ± 0.0016 | HMF8 |
| HEU-MET-FAST-011 | 90 | 7.550 | -- | Polyethylene | 13.230 | 0.9989 ± 0.0015 | HMF11 |
| HEU-MET-FAST-012 | 90 | 9.150 | -- | Aluminum | 0.850, 2.850 | 0.9992 ± 0.0018 | HMF12 |
| HEU-MET-FAST-013 | 90 | 8.350 | -- | Steel | 3.650 | 0.9990 ± 0.0015 | HMF13 |
| HEU-MET-FAST-014 | 90 | 7.750 | -- | Depleted U | 4.650 | 0.9989 ± 0.0017 | HMF14 |
| HEU-MET-FAST-015 | 96 | 9.995 | 11.130 | None | -- | 0.9996 ± 0.0017 | HMF15 |
| HEU-MET-FAST-018 | 90 | 9.150 | -- | None | -- | 1.0000 ± 0.0014 | HMF18 |
| HEU-MET-FAST-019 | 90 | 9.150 | -- | Graphite | 3.450 | 1.0000 ± 0.0028 | HMF19 |
| HEU-MET-FAST-020 | 90 | 8.350 | -- | Polyethylene | 1.450 | 1.0000 ± 0.0028 | HMF20 |
| HEU-MET-FAST-021 | 90 | 7.550 | -- | Steel | 9.700 | 1.0000 ± 0.0024 | HMF21 |
| HEU-MET-FAST-022 | 90 | 8.350 | -- | Duralumin | 3.900 | 1.0000 ± 0.0019 | HMF22 |
| HEU-MET-FAST-024 | 90 | 7.550 | -- | Steel, polyethylene | 0.850, 2.850 | 0.9990 ± 0.0015 | HMF24 |

The results presented in the following tables are from CRWMS M&O (1999a, page 98). Table 5-20 presents the MCNP calculation results for ENDF/B-V and ENDF/B-VI cross section libraries for the HEU Fast Metal Criticality set of experiments. Table 5-21 presents the differences between the experimental and calculated values of $k_{eff}$. Table 5-22 presents the AENCF values for this set of experiments. All of the results are within 1.5% of the benchmark values.

Table 5-20 Criticality Calculation Results for HEU-METAL-FAST Experiments

| MCNP case name | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-VI $k_{eff} \pm \sigma$ |
|---|---|---|
| HMF8 | 0.9951 ± 0.00058 | 0.99281 ± 0.00059 |
| HMF11 | 0.99348 ± 0.00079 | 0.99717 ± 0.00081 |
| HMF12 | 0.99399 ± 0.00043 | 0.9944 ± 0.00043 |
| HMF13 | 0.99997 ± 0.00061 | 0.99478 ± 0.00063 |
| HMF14 | 0.99831 ± 0.00062 | 0.99501 ± 0.00063 |
| HMF15 | 0.99263 ± 0.00056 | 0.99172 ± 0.00061 |

Table 5-20 Criticality Calculation Results for HEU-METAL-FAST Experiments

| MCNP case name | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-VI $k_{eff} \pm \sigma$ |
|---|---|---|
| HMF18 | 0.99805 ± 0.00056 | 0.99837 ± 0.00058 |
| HMF19 | 1.00277 ± 0.0006 | 1.00328 ± 0.0006 |
| HMF20 | 0.99574 ± 0.00061 | 0.99839 ± 0.00069 |
| HMF21 | 1.00192 ± 0.0006 | 0.99463 ± 0.00061 |
| HMF22 | 0.99272 ± 0.0006 | 0.99188 ± 0.0006 |
| HMF24 | 0.99521 ± 0.00109 | 0.99443 ± 0.00109 |

Table 5-21 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for HEU-METAL-FAST Experiments

| MCNP case name | ENDF/B-V $\Delta k_{eff} \pm 2\sigma$ | ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|
| HMF8 | 0.00490 ± 0.00340 | 0.00719 ± 0.00341 |
| HMF11 | 0.00542 ± 0.00339 | 0.00173 ± 0.00341 |
| HMF12 | 0.00601 ± 0.00370 | 0.00560 ± 0.00370 |
| HMF13 | -0.00037 ± 0.00324 | 0.00482 ± 0.00325 |
| HMF14 | 0.00169 ± 0.00362 | 0.00499 ± 0.00363 |
| HMF15 | 0.00737 ±0.00358 | 0.00828 ± 0.00361 |
| HMF18 | 0.00085 ± 0.00302 | 0.00053 ± 0.00303 |
| HMF19 | -0.00407 ± 0.00573 | -0.00458 ± 0.00573 |
| HMF20 | 0.00426 ±0 .00573 | 0.00161 ± 0.00577 |
| HMF21 | -0.00292 ± 0.00495 | 0.00437 ± 0.00495 |
| HMF22 | 0.00728 ± 0.00398 | 0.00812 ± 0.00398 |
| HMF24 | 0.00479 ± 0.00371 | 0.00557 ± 0.00371 |

Table 5-22 AENCF for HEU-METAL-FAST Experiments (MeV)

| MCNP case name | ENDF/B-V | ENDF/B-VI |
|---|---|---|
| HMF8 | 1.5501 | 1.4766 |
| HMF11 | 1.1607 | 1.1026 |
| HMF12 | 1.5237 | 1.4486 |
| HMF13 | 1.4838 | 1.4155 |
| HMF14 | 1.5465 | 1.4893 |

| Table 5-22 AENCF for HEU-METAL-FAST Experiments (MeV) | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| HMF15 | 1.5813 | 1.5000 |
| HMF18 | 1.5515 | 1.4753 |
| HMF19 | 1.4793 | 1.4018 |
| HMF20 | 1.4341 | 1.3635 |
| HMF21 | 1.4431 | 1.3807 |
| HMF22 | 1.5049 | 1.4299 |
| HMF24 | 1.2553 | 1.1917 |

## 5.2 HEU BENCHMARK CASES

### 5.2.1 HEU Benchmarks Related to Intact Waste Package Configurations

#### 5.2.1.1 HEU Plates, Discs, and Slabs

##### 5.2.1.1.1 HEU-MET-FAST-007

Forty-three critical experiments involving moderated slabs of highly enriched uranium metal (93.15 wt.% $^{235}U$) were performed at the ORNL, which was at that time operated by Union Carbide Corporation. The unreflected experiments consisted of uranium metal slabs interleaved with polyethylene, Plexiglas, or Teflon. Some of the experiments moderated by polyethylene also had a six-inch polyethylene reflector. Critical configurations had base dimensions of 5 x 10 and 10 x 10 inches. The uranium mass ranged from 15.712 to 120.709 kg. All forty-three experiments presented are considered acceptable for use as benchmark experiments (NEA [2001, HEU-MET-FAST-007]). Table 5-23 presents the critical parameters for this experiment and Table 5-24 provides the results.

| Table 5-23 HEU-MET-FAST-007 Critical Parameters | | | | | |
|---|---|---|---|---|---|
| **Case #** | **U Metal (kg)** | **Plastic (kg)** | **Measured Height (cm)** | **Measured $k_{eff}$ ± Uncertainty** | **Basic Assembly Features** |
| 1 | 104.001 | - | 8.722 | 0.9971 ± 0.0001 | Unreflected, Polyethyle ne Moderated, Base Dimensions 25.4 x 25.4 cm |
| 2 | 99.660 | 0.7142 | 9.545 | 0.9986 ± 0.0001 | |
| 3 | 97.756 | 0.9384 | 9.771 | 1.0012 ± 0.0001 | |
| 4 | 95.929 | 1.1177 | 9.898 | 0.9970 ± 0.0001 | |
| 5 | 93.975 | 1.4073 | 10.251 | 1.0000 ± 0.0001 | |
| 6 | 92.089 | 1.6070 | 10.401 | 1.0028 ± 0.0001 | |
| 7 | 92.089 | 1.6090 | 10.401 | 0.9996 ± 0.0001 | |

Table 5-23 HEU-MET-FAST-007 Critical Parameters

| Case # | U Metal (kg) | Plastic (kg) | Measured Height (cm) | Measured $k_{eff}$ ± Uncertainty | Basic Assembly Features |
|---|---|---|---|---|---|
| 8 | 92.055 | 1.6083 | 10.424 | 0.9992 ± 0.0001 | Unreflected, Polyethylene Moderated, Base Dimensions 25.4 x 25.4 cm |
| 9 | 92.021 | 1.6183 | 10.427 | 1.0017 ± 0.0008 | |
| 10 | 76.687 | 3.0944 | 11.689 | 1.0000 ± 0.0001 | |
| 11 | 64.300 | 4.4132 | 12.921 | 0.9982 ± 0.0001 | |
| 12 | 61.444 | 4.5931 | 13.007 | 0.9951 ± 0.0001 | |
| 13 | 61.444 | 4.5931 | 13.000 | 1.0009 ± 0.0001 | |
| 14 | 61.444 | 4.6831 | 13.165 | 0.9983 ± 0.0001 | |
| 15 | 61.333 | 4.4733 | 12.715 | 0.9978 ± 0.0001 | |
| 16 | 61.333 | 4.4953 | 12.758 | 0.9988 ± 0.0001 | |
| 17 | 46.058 | 6.083 | 14.211 | 0.9972 ± 0.0001 | |
| 18 | 46.058 | 6.1478 | 14.328 | 0.9991 ± 0.0001 | |
| 19 | 74.095 | - | 12.355 | 0.9983 ± 0.0001 | Unreflected, Polyethylene Moderated, Base Dimensions 12.7 x 25.4 cm |
| 20 | 71.721 | 1.1121 | 15.730 | 0.9981 ± 0.0001 | |
| 21 | 71.721 | 1.1699 | 15.931 | 0.9987 ± 0.0001 | |
| 22 | 71.721 | 1.2277 | 16.132 | 0.9994 ± 0.0001 | |
| 23 | 68.879 | 1.8390 | 17.750 | 0.9993 ± 0.0001 | |
| 24 | 68.879 | 1.8852 | 17.915 | 1.0001 ± 0.0001 | |
| 25 | 66.959 | 2.5234 | 19.759 | 0.9990 ± 0.0001 | |
| 26 | 66.959 | 2.5812 | 19.964 | 0.9997 ± 0.0001 | |
| 27 | 99.712 | 1.3717 | 10.140 | 0.9965 ± 0.0002 | Unreflected, Plexiglas Moderated, Base Dimensions 25.4 x 25.4 cm |
| 28 | 95.921 | 2.2778 | 10.996 | 0.9987 ± 0.0002 | |
| 29 | 92.063 | 3.1701 | 11.831 | 0.9978 ± 0.0002 | |
| 30 | 76.692 | 6.1834 | 14.514 | 0.9981 ± 0.0002 | |
| 31 | 61.403 | 10.0897 | 18.395 | 1.0013 ± 0.0002 | |
| 32 | 109.189 | 3.881 | 11.908 | 0.9959 ± 0.0001 | Unreflected, Teflon Moderated, Base Dimensions 25.4 x 25.4 cm |
| 33 | 114.991 | 6.963 | 14.709 | 0.9995 ± 0.0001 | |
| 34 | 120.709 | 10.460 | 17.770 | 0.9977 ± 0.0001 | |
| 35 | 31.771 | - | 5.362 | 1.0011 ± 0.0001 | Polyethylene Moderated & Reflected, Base Dimensions 12.7 x 25.4 cm |
| 36 | 30.664 | 0.3553 | 6.307 | 0.9999 ± 0.0001 | |
| 37 | 28.728 | 0.6716 | 7.120 | 0.9988 ± 0.0001 | |
| 38 | 28.646 | 0.7410 | 7.351 | 1.0000 ± 0.0001 | |
| 39 | 28.646 | 0.7640 | 7.427 | 1.0018 ± 0.0001 | |
| 40 | 28.641 | 0.7240 | 7.242 | 1.0013 ± 0.0001 | |
| 41 | 22.968 | 1.4141 | 8.674 | 0.9994 ± 0.0001 | |

Table 5-23 HEU-MET-FAST-007 Critical Parameters

| Case # | U Metal (kg) | Plastic (kg) | Measured Height (cm) | Measured $k_{eff}$ ± Uncertainty | Basic Assembly Features |
|---|---|---|---|---|---|
| 42 | 22.968 | 1.4488 | 8.809 | 1.0016 ± 0.0001 | |
| 43 | 15.712 | 2.3558 | 10.688 | 0.9998 ± 0.0001 | |

Table 5-24 Results for HEU-MET-FAST-007

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| Hmf7_01 | 0.9971 | 0.0001 | 1.5855 | 0.9941 | 0.0013 | -0.0031 | ± | 0.0027 |
| Hmf7_02 | 0.9986 | 0.0001 | 1.4036 | 0.9943 | 0.0013 | -0.0043 | ± | 0.0027 |
| Hmf7_03 | 1.0012 | 0.0001 | 1.3617 | 0.9929 | 0.0015 | -0.0083 | ± | 0.0030 |
| Hmf7_04 | 0.9970 | 0.0001 | 1.3232 | 0.9916 | 0.0015 | -0.0054 | ± | 0.0030 |
| Hmf7_05 | 1.0000 | 0.0001 | 1.2530 | 0.9909 | 0.0016 | -0.0091 | ± | 0.0032 |
| Hmf7_06 | 1.0028 | 0.0001 | 1.2230 | 0.9995 | 0.0014 | -0.0033 | ± | 0.0029 |
| Hmf7_07 | 0.9996 | 0.0001 | 1.2195 | 0.9945 | 0.0014 | -0.0051 | ± | 0.0028 |
| Hmf7_08 | 0.9992 | 0.0001 | 1.2156 | 0.9898 | 0.0014 | -0.0094 | ± | 0.0028 |
| Hmf7_09 | 1.0017 | 0.0008 | 1.2192 | 1.0003 | 0.0016 | -0.0014 | ± | 0.0036 |
| Hmf7_10 | 1.0000 | 0.0001 | 0.9503 | 0.9908 | 0.0019 | -0.0092 | ± | 0.0039 |
| Hmf7_11 | 0.9982 | 0.0001 | 0.7392 | 0.9887 | 0.0021 | -0.0095 | ± | 0.0042 |
| Hmf7_12 | 0.9951 | 0.0001 | 0.7036 | 0.9865 | 0.0015 | -0.0086 | ± | 0.0031 |
| Hmf7_13 | 1.0009 | 0.0001 | 0.7430 | 0.9956 | 0.0018 | -0.0053 | ± | 0.0036 |
| Hmf7_14 | 0.9983 | 0.0001 | 0.7021 | 0.9891 | 0.0018 | -0.0092 | ± | 0.0037 |
| Hmf7_15 | 0.9978 | 0.0001 | 0.7483 | 0.9924 | 0.0019 | -0.0055 | ± | 0.0038 |
| Hmf7_16 | 0.9988 | 0.0001 | 0.7497 | 0.9911 | 0.0019 | -0.0077 | ± | 0.0038 |
| Hmf7_17 | 0.9972 | 0.0001 | 0.5210 | 0.9886 | 0.0021 | -0.0086 | ± | 0.0043 |
| hmf7_18 | 0.9991 | 0.0001 | 0.5176 | 0.9936 | 0.0020 | -0.0055 | ± | 0.0040 |
| hmf7_19 | 0.9983 | 0.0001 | 1.5781 | 0.9971 | 0.0014 | -0.0012 | ± | 0.0028 |
| hmf7_20 | 0.9981 | 0.0001 | 1.2640 | 0.9890 | 0.0015 | -0.0091 | ± | 0.0030 |
| hmf7_21 | 0.9987 | 0.0001 | 1.2465 | 0.9938 | 0.0014 | -0.0049 | ± | 0.0028 |
| hmf7_22 | 0.9994 | 0.0001 | 1.2323 | 0.9935 | 0.0016 | -0.0059 | ± | 0.0032 |
| hmf7_23 | 0.9993 | 0.0001 | 1.1298 | 0.9937 | 0.0014 | -0.0056 | ± | 0.0029 |
| hmf7_24 | 1.0001 | 0.0001 | 1.1177 | 0.9950 | 0.0017 | -0.0051 | ± | 0.0033 |
| hmf7_25 | 0.9990 | 0.0001 | 1.0153 | 0.9921 | 0.0018 | -0.0069 | ± | 0.0036 |
| hmf7_26 | 0.9997 | 0.0001 | 1.0067 | 0.9894 | 0.0016 | -0.0103 | ± | 0.0033 |

Table 5-24 Results for HEU-MET-FAST-007

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hmf7_27 | 0.9965 | 0.0002 | 1.3907 | 0.9916 | 0.0014 | -0.0049 | ± | 0.0028 |
| hmf7_28 | 0.9987 | 0.0002 | 1.2784 | 0.9913 | 0.0016 | -0.0074 | ± | 0.0032 |
| hmf7_29 | 0.9978 | 0.0002 | 1.1714 | 0.9934 | 0.0015 | -0.0045 | ± | 0.0030 |
| hmf7_30 | 0.9981 | 0.0002 | 0.8973 | 0.9936 | 0.0018 | -0.0045 | ± | 0.0035 |
| hmf7_31 | 1.0013 | 0.0002 | 0.6162 | 0.9967 | 0.0020 | -0.0046 | ± | 0.0039 |
| hmf7_32 | 0.9959 | 0.0001 | 1.4869 | 0.9968 | 0.0013 | 0.0009 | ± | 0.0026 |
| hmf7_33 | 0.9995 | 0.0001 | 1.4324 | 1.0061 | 0.0015 | 0.0066 | ± | 0.0029 |
| hmf7_34 | 0.9977 | 0.0001 | 1.3814 | 1.0058 | 0.0014 | 0.0081 | ± | 0.0027 |
| hmf7_35 | 1.0011 | 0.0001 | 1.0130 | 0.9894 | 0.0019 | -0.0117 | ± | 0.0037 |
| hmf7_36 | 0.9999 | 0.0001 | 0.8594 | 0.9976 | 0.0017 | -0.0023 | ± | 0.0034 |
| hmf7_37 | 0.9988 | 0.0001 | 0.7462 | 0.9959 | 0.0020 | -0.0029 | ± | 0.0039 |
| hmf7_38 | 1.0000 | 0.0001 | 0.7181 | 0.9996 | 0.0017 | -0.0004 | ± | 0.0034 |
| hmf7_39 | 1.0018 | 0.0001 | 0.7182 | 1.0009 | 0.0016 | -0.0009 | ± | 0.0033 |
| hmf7_40 | 1.0013 | 0.0001 | 0.7339 | 0.9976 | 0.0019 | -0.0037 | ± | 0.0037 |
| hmf7_41 | 0.9994 | 0.0001 | 0.5383 | 0.9968 | 0.0019 | -0.0026 | ± | 0.0037 |
| hmf7_42 | 1.0016 | 0.0001 | 0.5269 | 0.9979 | 0.0017 | -0.0037 | ± | 0.0035 |
| hmf7_43 | 0.9998 | 0.0001 | 0.3145 | 0.9990 | 0.0018 | -0.0008 | ± | 0.0035 |

#### 5.2.1.1.2 HEU-MET-FAST-030

The experiment presented in this section was performed in 1989 at VNIITF using the criticality test facility FKBN (Vertical Lift Machine). The critical assembly core is a cylinder consisting of alternating discs of highly enriched uranium and beryllium. Depleted uranium was used for end and side reflectors. The end reflector is 20 cm thick, and the side reflector is 25 cm thick. Seven similar critical assemblies having different thicknesses of the material layers were performed. These experiments were used for neutron data validation in the range of fission and resonance neutron energy. Only one of these critical assemblies is chosen for consideration in this evaluation. The configuration of the system is simple enough and measurements were made with sufficient precision for the experiment to be considered as a criticality safety benchmark (NEA [2001]). Although categorized as "fast," a significant fraction of the total fissions in this assembly occurs in the intermediate energy range.

The critical assembly central part (core and end reflectors) is a cylinder, 19.99±0.01 cm in diameter, divided by a gap into two parts: a bottom and a top. The critical separation between the bottom and the top is 2.52±0.02 cm. The assembly core is built of alternating layers of HEU, nominally 96 wt.% $^{235}U$, and beryllium. The average disc thicknesses were measured for stacks

and, therefore, include possible clearances between them. Table 5-25 presents the results for this experiment.

Table 5-25 Results for HEU-MET-FAST-030

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hmf030 | 1.0000 | 0.0009 | 1.129 | 0.9999 | 0.0005 | -0.0001 | ± | 0.0021 |

### 5.2.1.1.3 HEU-MET-FAST-033

Two experiments presented in NEA (2001, HEU-MET-FAST-033) were performed at the Critical Test Facility of VNIITF in 1986. The critical assemblies are cylinders consisting of alternating discs of highly enriched uranium, polyethylene, and steel. Ten similar critical experiments having different thicknesses of the material layers were performed. These experiments were used for neutron data validation in the range of resonance and thermal neutron energy. For the detailed description two of these critical assemblies were chosen. The configurations of the systems are simple enough and measurements were made with sufficient precision for the experiments to be considered as criticality safety benchmarks (NEA [2001]). Although categorized as "fast," a significant fraction of the fissions in these two assemblies occurred in the intermediate energy range. Although categorized as "FAST," both configurations could be classified as "MIXED". This determination was not made until after formal publication of HEU-MET-FAST-033. Each critical assembly is a cylinder, 19.99±0.01 cm in diameter, divided by a gap into two parts: a bottom and a top. The critical separation, $H_{cr}$, between the bottom and the top is 0.30±0.02 cm in Assembly 1 and 0.36±0.02 cm in Assembly 2. The critical assemblies were built of alternating discs of HEU, nominally 96 wt.% $^{235}U$; polyethylene; and steel. The HEU discs are 0.995±0.002 cm thick, the polyethylene discs are 0.990±0.003 and 0.490±0.002 cm thick, and the steel discs are 0.490±0.002 cm thick. The average disc thicknesses were measured for stacks and, therefore, include possible gaps between the discs. Table 5-26 presents the results for this experiment.

Table 5-26 Results for HEU-MET-FAST-033

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumf33c1 | 0.9991 | 0.0014 | 0.850 | 0.9940 | 0.0004 | -0.0051 | ± | 0.0029 |
| heumf33c2 | 0.9991 | 0.0014 | 0.717 | 0.9924 | 0.0004 | -0.0067 | ± | 0.0029 |

### 5.2.1.1.4 HEU-MET-FAST-034

Three experiments were performed in 1986 at VNIITF using the criticality test facility FKBN (Vertical Lift Machine) and presented in NEA (2001, HEU-MET-FAST-034). The critical assemblies are cylinders consisting of alternating discs of highly enriched uranium, polyethylene,

and metal (Ti, Al, or steel). Twelve similar critical experiments having different thicknesses of the material layers were performed. These experiments were used for neutron-data validation in the range of fast and intermediate neutron energy. Only three of these critical assemblies were chosen for consideration in the evaluation.

The configuration of the systems is simple enough and measurements were made with sufficient precision for the experiments to be considered as criticality safety benchmarks (NEA [2001]). Although categorized as "fast," a significant fraction of the total fissions in these assemblies take place in the intermediate energy range. Each critical assembly is a cylinder, 19.9±0.01 cm in diameter, divided by a gap into two parts: a bottom and a top. The critical assemblies are built of alternating discs of HEU, nominally 96 wt.% $^{235}U$, polyethylene, and metal: Ti, Al, or steel. The HEU discs are 0.995±0.002 cm thick, the polyethylene discs are 0.990±0.003 and 0.490±0.002 cm thick, and the metal discs are 0.490±0.002 cm thick. The average disc thicknesses were measured for stacks and, therefore, include possible gaps between the discs. Table 5-27 shows the results for this experiment.

Table 5-27 Results for HEU-MET-FAST-034

| Experiment/Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumf34c1 | 0.9990 | 0.0012 | 0.958 | 0.9913 | 0.0004 | -0.0077 | ± | 0.0025 |
| heumf34c2 | 0.9990 | 0.0012 | 0.958 | 0.9910 | 0.0004 | -0.0080 | ± | 0.0025 |
| heumf34c3 | 0.9990 | 0.0012 | 0.953 | 0.9929 | 0.0004 | -0.0061 | ± | 0.0025 |

#### 5.2.1.1.5 HEU-MET-FAST-036

The experiments, designated as HEU-MET-FAST-036 in NEA (2001), were performed in 1986 at VNIITF using the criticality test facility FKBN (Vertical Lift Machine). The critical assemblies were cylinders consisting of alternating layers of highly enriched uranium, polyethylene, and depleted uranium. Six similar critical experiments having different thicknesses of the material layers were performed. These experiments were used for neutron-data validation in the range of resonance neutron energy. Two of these critical assemblies were chosen for evaluation. In the subsequent discussion, these assemblies are referred to as "Assembly 1" and "Assembly 2." The configurations of the systems are simple enough and measurements were made with sufficient precision for the experiments to be considered as criticality safety benchmarks (NEA [2001]). Although categorized as "FAST," Case 1 could be classified as "MIXED" or "FAST" depending on the cross section data used in the calculations. Case 2 is clearly "FAST".

Each critical assembly was a cylinder, 19.99±0.01 cm in diameter, divided by a gap into two parts: a movable bottom and a stationary top. The critical separation between the bottom and the top was 0.23±0.02 cm for Assembly 1 and 0.365±0.020 cm for Assembly 2. The critical assemblies were built of alternating layers of HEU, polyethylene, and depleted uranium (DU). On average, the individual HEU discs were 0.995±0.003 cm thick, the polyethylene discs were

0.990±0.005 and 0.490±0.005 cm thick, and the depleted uranium discs were 0.990±0.005 cm thick. The average disc thicknesses were measured for stacks and, therefore, include possible gaps (clearances) between the discs. Table 5-28 shows the results for this experiment.

Table 5-28 Results for HEU-MET-FAST-036

| Experiment /Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumf36c1 | 0.9993 | 0.0015 | 0.866 | 0.9932 | 0.0004 | -0.0061 | ± | 0.0031 |
| heumf36c2 | 0.9993 | 0.0013 | 1.097 | 0.9938 | 0.0004 | -0.0055 | ± | 0.0027 |

### 5.2.1.1.6 HEU-MET-FAST-037

The experiments designated as HEU-MET-FAST-037 were performed in 1988 at VNIITF using the criticality test facility FKBN (Vertical Lift Machine). The cores of the critical assemblies were cylinders consisting of alternating discs of highly enriched uranium, polyethylene, and depleted uranium. End and side polyethylene reflectors were 10 cm thick. Seven similar critical experiments having different thicknesses of the material layers were performed. These experiments were used for neutron-data validation in the range of resonance neutron energy. The two critical assemblies chosen for evaluation were very similar. They differ by the presence of a thin cadmium layer between the core and the side and end reflectors in one assembly and by the core loading required to bring each assembly to or near criticality. In the subsequent discussion, these assemblies are referred to as "Assembly 1" and "Assembly 2."

The configurations of the systems are simple enough and measurements were made with sufficient precision for the experiments to be considered as criticality safety benchmarks (NEA [2001]). Although categorized as "FAST," Case 1 could be classified as "MIXED" or "FAST" depending on the cross section data used in the calculations. Case 2 is clearly "FAST".

Each assembly core is a cylinder, 19.99±0.01 cm in diameter, divided by a gap into two parts: a movable bottom and a stationary top. The critical separation between the bottom and the top was 6.66±0.02 cm for Assembly 1 and 2.28±0.02 cm for Assembly 2. The critical assemblies were built of alternating discs of HEU, polyethylene, and DU. On average, in the active core region the individual HEU discs were 0.995±0.003-cm thick, the polyethylene discs were 0.990±0.005-cm thick, and the depleted uranium discs were 0.990±0.005-cm thick. The average disc thicknesses were measured for stacks and, therefore, include possible gaps (clearances) between the discs. Each layer of material consisted of one or more discs. Each axial reflector in the two assemblies comprised two 4.98-cm-thick polyethylene discs having the same diameter as the active core. In Assembly 2, in addition to the cadmium liner of the side reflector, 0.050±0.005-cm-thick cadmium discs, 19.60±0.04 cm in diameter, were placed between the axial reflectors and the active core region. A more precise thickness of the cadmium discs, 0.052±0.001 cm, was calculated from the reported disc mass and the reported density. Each HEU disc had a central hole, filled with a ring plug and a cylinder plug, both of the same material. Results for these experiments are presented in Table 5-29.

Table 5-29 Results for HEU-MET-FAST-037

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumf37c1 | 0.9997 | 0.0011 | 0.896 | 0.9966 | 0.0004 | -0.0031 | ± | 0.0023 |
| heumf37c2 | 0.9997 | 0.0011 | 1.020 | 0.9931 | 0.0004 | -0.0066 | ± | 0.0023 |

### 5.2.1.1.7 HEU-MET-FAST-038

The first configuration, "Heterogeneous Cylinder of Highly Enriched Uranium with Beryllium Moderator and Depleted Uranium Reflector," was described in NEA (2001, HEU-MET-FAST-030). The second is described in NEA (2001, HEU-MET-FAST-038).

The critical assembly central part (core and end reflectors) is a cylinder, 19.99±0.01 cm in diameter, divided by a gap into two parts: a movable bottom and a stationary top. The critical separation between the bottom and the top was 2.26±0.02 cm. The assembly core was built of alternating layers of HEU, beryllium (Be), and beryllium oxide (BeO). depleted uranium was used for end and side reflectors. The depleted uranium end reflector was 20-cm thick and the side reflector was 25-cm thick. This experiment is acceptable as a benchmark experiment (NEA [2001]). Table 5-30 shows the result for this experiment.

Table 5-30 Results for HEU-MET-FAST-038

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumf38c1 | 0.9999 | 0.0015 | 1.139 | 1.0012 | 0.0004 | 0.0013 | ± | 0.0031 |

### 5.2.1.1.8 HEU-MET-INTER-006

The Zeus experiment was assembled at the Los Alamos Critical Experiment Facility (LACEF) at Los Alamos National Laboratory (LANL). After assembly of two slightly subcritical configurations, criticality was achieved on April 26, 1999. Zeus was designed to encompass a series of critical experiments that produce intermediate spectra. The first set of experiments was intended to test the adequacy of $^{235}U$ cross sections in the intermediate-energy range. Plates of HEU metal were interspersed with graphite plates in a cylindrical stack that is completely surrounded by a copper reflector. The evaluation in NEA (2001, HEU-MET-INTER-006) describes the initial Zeus critical experiment in which HEU and graphite plates are arranged to produce a $C/^{235}U$ ratio of approximately 52:1. Based on this evaluation, the critical Zeus configuration was judged to be acceptable for use as a criticality-safety benchmark experiment.

The Zeus experiment was assembled on the general-purpose Comet vertical assembly machine at LACEF. The current assembly consists of a cylindrical core region containing interspersed plates of HEU metal and graphite that is surrounded on all sides by a metallic copper reflector. The assembly is divided into two portions. The upper portion, which contains the top segment of the

core and the top, corner, and side reflectors, rests on a stationary aluminum support plate attached to the Comet machine. The top segment of the core did rest on a thin, square stainless steel plate, called the diaphragm, which was supported by the corner copper reflectors. The remainder of the core and the bottom reflector were seated on a circular aluminum platen adapter plate connected to the vertical drive (ram) of the assembly machine. The lower core plates and the bottom reflector were held in position by a central, hollow aluminum alignment tube. The experiment was assembled by raising the bottom portion into the reflector until it fully closed to the steel diaphragm that supports the top segment of the core. There were no other control, safety, or shim rods inside the assembly. Three safety monitors were placed at varying distances from the assembly. Any one of these monitors would initiate separation of the materials by dropping the ram if the measured neutron leakage exceeded a preset level. Table 5-31 shows results for this experiment.

Table 5-31 Results for HEU-MET-INTER-006

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hmi06-01 | 0.9976 | 0.0008 | 0.342 | 0.9967 | 0.0003 | -0.0009 | ± | 0.0017 |
| hmi06-02 | 0.9997 | 0.0008 | 0.386 | 0.9984 | 0.0003 | -0.0013 | ± | 0.0017 |

### 5.2.1.1.9 HEU-MET-MIXED-001

The experiment presented in NEA (2001, HEU-MET-MIXED-001) was performed in 1988 at VNIITF using the criticality test facility FKBN (Vertical Lift Machine). The critical assembly core is a cylinder consisting of alternating discs of highly enriched uranium, polyethylene, and titanium. End and side polyethylene reflectors are 10.0 cm thick. Six similar critical assemblies having different thicknesses of the material layers were performed. These experiments were used for neutron data validation in the range of resonance and thermal neutron energy. Only one of these critical assemblies is chosen for consideration in this evaluation.

The critical assembly core is a cylinder, 19.99±0.01 cm in diameter, divided by a gap into two parts: a bottom and a top. The critical separation, $H_{cr}$, between the bottom and the top is 6.45±0.02 cm. The core is built of alternating discs of HEU, nominally 96 wt.% $^{235}U$, polyethylene, and titanium. The average disc thicknesses were measured for stacks and, therefore, include possible clearances between the discs. This experiment is acceptable as a benchmark experiment (NEA [2001]). Table 5-32 shows the calculated result for this experiment.

Table 5-32 Results for HEU-MET-MIXED-001

| Experiment/Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| Heumm1c1 | 0.9995 | 0.0013 | 0.762 | 1.0044 | 0.0004 | 0.0049 | ± | 0.0027 |

### 5.2.1.1.10 HEU-MET-MIXED-004

The experiment presented in NEA (2001, HEU-MET-MIXED-001) was performed in 1989 at VNIITF using the criticality test facility FKBN (Vertical Lift Machine). The core of the critical assembly is a cylinder consisting of alternating layers of highly enriched uranium, polyethylene, and depleted uranium. The end reflectors of depleted uranium have thicknesses of 6 cm (bottom) and 5 cm (top). The side reflector of depleted uranium has a thickness of 25 cm. Four similar critical experiments having different thicknesses of the material layers were performed. These experiments were used for neutron-data validation in the range of thermal and resonance neutron energy. One of these critical assemblies was chosen for evaluation. The configuration of the system is simple enough and measurements were made with sufficient precision for the experiment to be considered as a criticality safety benchmark. Although categorized as "mixed," a significant fraction of the total fissions in this assembly takes place in the intermediate energy range.

The assembly central part (core and end reflectors) was a cylinder, 19.99±0.01 cm in diameter, divided by a gap into two parts: a movable bottom and a stationary top. The critical separation etween the bottom and the top was 0.02±0.02 cm. The assembly core was built of alternating layers of HEU, polyethylene, and DU. On average, the individual HEU discs were 0.995±0.003-cm thick; the polyethylene discs were 0.990±0.005, 0.490±0.005 and 4.98±0.01-cm thick; and the depleted uranium discs were 0.990±0.005, 1.490±0.006, 1.990±0.007, and 4.97±0.01-cm thick. The average disc thicknesses were measured for stacks and, therefore, include possible gaps (clearances) between the discs. Each layer of material can consist of one or more discs. Result of this experiment is presented in Table 5-33.

Table 5-33 Results for HEU-MET-MIXED-004

| Experiment /Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumm4c1 | 0.9999 | 0.0009 | 0.914 | 1.0048 | 0.0004 | 0.0049 | ± | 0.0019 |

### 5.2.1.1.11 HEU-COMP-INTER-003

The COMET universal critical assembly machine was used to perform a series of seven uranium-hydride critical experiments in 1987-88 at LANL. The cylindrical assemblies were disks of canned $UH_3$ powder, reflected by depleted uranium (D38), beryllium, and iron. Two 2-cm and two 3-cm-thick disks of canned $UH_3$ compressed powder (6-in. diameter) were used in four of the experiments. These experiments had outer reflectors of depleted uranium (D38) and inner reflectors of D38, Be, and Fe. The difference in the two experiments with Be inner reflectors was that in one experiment one of the 2-cm cans of $UH_3$ was inverted. The total mass of the $UH_3$ for each of these experiments was 17.7 kg. Four 3-cm disks were used for two of the three experiments with the outer reflector removed. The total $UH_3$ mass for the Be and one D38 experiment was 21.3 kg. The total mass for the third D38 experiment was 23.1 kg. This was achieved by using two 2-cm disks and three 3-cm disks. All seven of these experiments are

considered to be acceptable as benchmark critical experiments (NEA [2001, HEU-COMP-INTER-003]). Table 5-34 shows the result for this experiment.

Table 5-34 Results for HEU-COMP-INTER-003

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hci3-1 | 1.0000 | 0.0057 | 0.824 | 1.0097 | 0.0016 | 0.0097 | ± | 0.0126 |
| hci3-2 | 1.0000 | 0.0061 | 0.675 | 1.0061 | 0.0017 | 0.0061 | ± | 0.0117 |
| hci3-3 | 1.0000 | 0.0056 | 0.679 | 1.0048 | 0.0018 | 0.0048 | ± | 0.0116 |
| hci3-4 | 1.0000 | 0.0055 | 0.702 | 1.0035 | 0.0017 | 0.0035 | ± | 0.0100 |
| hci3-5 | 1.0000 | 0.0047 | 0.564 | 0.9930 | 0.0018 | -0.0070 | ± | 0.0101 |
| hci3-6 | 1.0000 | 0.0047 | 0.706 | 0.9963 | 0.0017 | -0.0037 | ± | 0.0105 |
| hci3-7 | 1.0000 | 0.0050 | 0.705 | 0.9976 | 0.0019 | -0.0057 | ± | 0.0113 |

### 5.2.1.2 HEU Arrays of Cans or Cylinders

#### 5.2.1.2.1 HEU-MET-FAST-023

In 1962-63 a set of detailed criticality experiments utilizing enriched uranium metal cylinders was conducted at the Oak Ridge Critical Experimental Facility (CEF). Critical arrays of uranium metal cylinders were constructed on a split table apparatus and surface-to-surface spacings (STS) between the units were measured as a function of the number of units and the thickness of a paraffin reflector. Boron proportional counters were used to determine the critical state. Experiments were performed using cylinders in four weight groups and with five shapes. The experiments reported in NEA (2001, HEU-MET-FAST-023) all involve cylinders of approximately 10.5-kg mass. 22 experimental configurations are deemed acceptable as benchmarks.

Enriched uranium metal cylinders with a nominal mass of 10.5 kg were supported vertically on stainless steel rods passing through 0.508-cm diameter holes in the cylinders. Two rods with their centers 8.547 cm apart on a diameter passed through each cylinder. Vertical spacing was accomplished by sections of close fitting Inconel metal tubes slipped over the outside of these rods and between the cylinders. Results for HEU-MET-FAST-023 are shown in Table 5-35.

Table 5-35 Results for HEU-MET-FAST-023

| Experiment/Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hmf023_02 | 1.0000 | 0.0052 | 1.452 | 0.9932 | 0.0011 | -0.0068 | ± | 0.0106 |
| hmf023_03 | 1.0000 | 0.0052 | 1.185 | 1.0052 | 0.0012 | 0.0051 | ± | 0.0107 |
| hmf023_04 | 1.0000 | 0.0052 | 1.032 | 1.0251 | 0.0013 | 0.0251 | ± | 0.0107 |

Table 5-35 Results for HEU-MET-FAST-023

| Experiment/Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hmf023_05 | 1.0000 | 0.0052 | 1.005 | 1.0294 | 0.0013 | 0.0294 | ± | 0.0107 |
| hmf023_06 | 1.0000 | 0.0052 | 1.581 | 0.9935 | 0.0010 | -0.0065 | ± | 0.0106 |
| hmf023_07 | 1.0000 | 0.0052 | 1.461 | 0.9927 | 0.0011 | -0.0073 | ± | 0.0106 |
| hmf023_08 | 1.0000 | 0.0052 | 1.172 | 1.0004 | 0.0012 | 0.0004 | ± | 0.0107 |
| hmf023_09 | 1.0000 | 0.0052 | 1.015 | 1.0215 | 0.0011 | 0.0215 | ± | 0.0106 |
| hmf023_10 | 1.0000 | 0.0052 | 0.993 | 1.0270 | 0.0013 | 0.0270 | ± | 0.0107 |
| hmf023_12 | 1.0000 | 0.0052 | 1.460 | 0.9933 | 0.0011 | -0.0067 | ± | 0.0106 |
| hmf023_13 | 1.0000 | 0.0052 | 1.193 | 1.0119 | 0.0011 | 0.0119 | ± | 0.0106 |
| hmf023_14 | 1.0000 | 0.0052 | 1.044 | 1.0258 | 0.0013 | 0.0258 | ± | 0.0107 |
| hmf023_15 | 1.0000 | 0.0052 | 1.017 | 1.0296 | 0.0013 | 0.0296 | ± | 0.0107 |
| hmf023_16 | 1.0000 | 0.0052 | 1.581 | 0.9975 | 0.0010 | -0.0025 | ± | 0.0106 |
| hmf023_17 | 1.0000 | 0.0052 | 1.469 | 0.9961 | 0.0011 | -0.0039 | ± | 0.0106 |
| hmf023_18 | 1.0000 | 0.0052 | 1.181 | 0.9999 | 0.0012 | -0.0001 | ± | 0.0107 |
| hmf023_19 | 1.0000 | 0.0052 | 1.034 | 1.0185 | 0.0012 | 0.0185 | ± | 0.0107 |
| hmf023_20 | 1.0000 | 0.0052 | 1.009 | 1.0246 | 0.0012 | 0.0246 | ± | 0.0107 |
| hmf023_21 | 1.0000 | 0.0057 | 1.585 | 0.9934 | 0.0010 | -0.0066 | ± | 0.0116 |
| hmf023_22 | 1.0000 | 0.0052 | 0.983 | 1.0252 | 0.0011 | 0.0252 | ± | 0.0106 |
| hmf023_28 | 1.0000 | 0.0052 | 1.580 | 0.9948 | 0.0009 | -0.0052 | ± | 0.0106 |
| hmf023_29 | 1.0000 | 0.0052 | 1.582 | 0.9947 | 0.0010 | -0.0053 | ± | 0.0106 |

#### 5.2.1.2.2 HEU-MET-FAST-026

Critical arrays of uranium metal cylinders were constructed on a split-table apparatus with varying STS spacings between the units and varying paraffin reflector thickness. Experiments were performed using cylinders in four mass groups and with different height-to-diameter ratios. The experiments reported in this evaluation include cylinders with approximate masses of 15, 20, and 25 kg. This section provides results for 32 experimental configurations, all of which are acceptable as benchmarks (NEA [2001]).

Enriched uranium cylinders were supported with their axes vertical on stainless steel rods passing through 0.508-cm-diameter holes in the cylinders. Two rods with their centers 8.547 cm apart on a diameter passed through the cylinders. Vertical spacing was accomplished by sections of close fitting Inconel metal tubes placed on the outside of these rods and between the cylinders. The deviation in the heights of the cylinders from the nominal values was no greater than 0.003 cm, the diameters varied by as much as 1.5%, and the mass about twice that amount. Uncertainties in cylinder heights of 0.003 cm are listed. The units in a particular array were

made as similar as possible by careful selection of the individual cylinders. A uranium density of 18.76 g/cm$^3$ is reported (NEA [2001]). Results for HEU-MET-FAST-026 are shown in Table 5-36.

Table 5-36 Results for HEU-MET-FAST-026

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hmf26b1 | 0.9982 | 0.0042 | 1.5756 | 1.0001 | 0.0014 | 0.0019 | ± | 0.0089 |
| hmf26b2 | 0.9982 | 0.0042 | 1.4694 | 0.9982 | 0.0014 | -0.0000 | ± | 0.0089 |
| hmf26b3 | 1.0000 | 0.0038 | 1.2064 | 0.9980 | 0.0018 | -0.0020 | ± | 0.0084 |
| hmf26b4 | 1.0000 | 0.0038 | 1.1093 | 0.9970 | 0.0017 | -0.0030 | ± | 0.0083 |
| hmf26b5 | 1.0000 | 0.0038 | 1.0961 | 0.9977 | 0.0018 | -0.0023 | ± | 0.0084 |
| hmf26b6 | 0.9982 | 0.0042 | 1.5835 | 0.9963 | 0.0014 | -0.0019 | ± | 0.0089 |
| hmf26b7 | 0.9982 | 0.0042 | 1.4657 | 0.9970 | 0.0016 | -0.0012 | ± | 0.0090 |
| hmf26b8 | 1.0000 | 0.0038 | 1.2118 | 0.9975 | 0.0015 | -0.0025 | ± | 0.0082 |
| hmf26b9 | 1.0000 | 0.0038 | 1.1028 | 0.9975 | 0.0017 | -0.0025 | ± | 0.0083 |
| hmf26b10 | 1.0000 | 0.0038 | 1.0979 | 0.9975 | 0.0016 | -0.0025 | ± | 0.0082 |
| hmf26c1 | 0.9982 | 0.0042 | 1.5784 | 0.9989 | 0.0014 | 0.0007 | ± | 0.0088 |
| hmf26c2 | 0.9982 | 0.0042 | 1.5803 | 0.9982 | 0.0014 | -0.0000 | ± | 0.0088 |
| hmf26c3 | 0.9982 | 0.0042 | 1.4701 | 0.9987 | 0.0016 | 0.0005 | ± | 0.0090 |
| hmf26c4 | 1.0000 | 0.0038 | 1.3449 | 1.0023 | 0.0017 | 0.0023 | ± | 0.0083 |
| hmf26c5 | 1.0000 | 0.0038 | 1.2330 | 0.9985 | 0.0017 | -0.0015 | ± | 0.0083 |
| hmf26c6 | 1.0000 | 0.0038 | 1.1473 | 0.9993 | 0.0018 | -0.0007 | ± | 0.0084 |
| hmf26c7 | 1.0000 | 0.0038 | 1.1437 | 0.9955 | 0.0017 | -0.0045 | ± | 0.0084 |
| hmf26c8 | 0.9982 | 0.0042 | 1.5745 | 0.9949 | 0.0015 | -0.0033 | ± | 0.0089 |
| hmf26c9 | 0.9982 | 0.0042 | 1.4747 | 0.9944 | 0.0015 | -0.0038 | ± | 0.0089 |
| hmf26c10 | 1.0000 | 0.0038 | 1.2326 | 0.9969 | 0.0016 | -0.0031 | ± | 0.0083 |
| hmf26c11 | 1.0000 | 0.0038 | 1.1529 | 1.0026 | 0.0017 | 0.0026 | ± | 0.0083 |
| hmf26c12 | 1.0000 | 0.0038 | 1.1412 | 1.0039 | 0.0016 | 0.0039 | ± | 0.0082 |
| hmf26d1 | 0.9982 | 0.0042 | 1.5776 | 0.9964 | 0.0014 | -0.0018 | ± | 0.0089 |
| hmf26d2 | 0.9982 | 0.0042 | 1.4797 | 0.9998 | 0.0014 | 0.0016 | ± | 0.0089 |
| hmf26d3 | 1.0000 | 0.0038 | 1.2620 | 0.9980 | 0.0016 | -0.0021 | ± | 0.0083 |
| hmf26d4 | 1.0000 | 0.0038 | 1.1935 | 0.9989 | 0.0016 | -0.0011 | ± | 0.0082 |
| hmf26d5 | 1.0000 | 0.0038 | 1.1859 | 0.9956 | 0.0017 | -0.0044 | ± | 0.0083 |
| hmf26d6 | 0.9982 | 0.0042 | 1.5785 | 0.9989 | 0.0014 | 0.0007 | ± | 0.0089 |

| Table 5-36 Results for HEU-MET-FAST-026 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hmf26d7 | 0.9982 | 0.0042 | 1.4784 | 0.9959 | 0.0015 | -0.0023 | ± | 0.0089 |
| hmf26d8 | 1.0000 | 0.0038 | 1.2730 | 0.9982 | 0.0017 | -0.0018 | ± | 0.0083 |
| hmf26d9 | 1.0000 | 0.0038 | 1.2025 | 1.0015 | 0.0018 | 0.0015 | ± | 0.0084 |
| hmf26d10 | 1.0000 | 0.0038 | 1.1907 | 0.9961 | 0.0019 | -0.0039 | ± | 0.0085 |

#### 5.2.1.2.3 HEU-COMP-MIXED-001

Critical experiments with cans containing enriched uranium dioxide were performed in 1973 at the ORNL Y-12 Plant, which was operated by Union Carbide Corporation. The critical experiments were performed on the specifically designed Criticality Testing Unit, Horizontal Displacement, formerly known as the Split Table Apparatus. It was located in the South Experimental Area of the Critical Experiment Facility. Critical experiments consisted of cans in variously sized arrays containing enriched uranium dioxide ($UO_2$). Cans used in the experiments were described as "tins" and "similar to paint cans". Enrichment of the $UO_2$ material used in these critical experiments was 93.15 wt.% $^{235}U$. The array units consisted of large or small cans of $UO_2$, with various interspersed moderation schemes. Some experiments involved arrays of alcohol-water ($C_2H_6O$ - 5% $H_2O$) moderated $UO_2$ fissile units to determine effects of hydrogenous moderation. A polyethylene reflector surrounded all experiments on all six sides with a thickness of 15.24 cm such that the effects of room reflection could be ignored.

Twenty-nine experiments were evaluated in NEA (2001), HEU-COMP-MIXED-001. Three were hypothetical experiments that were not actually performed. Of the remaining 26 experiments, five were critical with a small separation between the two sides of the split table, and 15 were supercritical and six were subcritical with complete closure of the split table. These 26 experiments were considered acceptable as benchmark experiments (NEA [2001]).

Although categorized as "MIXED," the majority of the fissions occur in the fast-energy range for Experiments 5, 6, and 7 and in the thermal-energy range for Experiments 1, 2, 15, 16, 17, and 18.

The twenty-six critical configurations, with and without interspersed moderating material between the cans, consisted of three main types: two arrays of smaller cans (Type I), 14 arrays of larger cans (Type II), and 10 arrays of alcohol-water-moderated cans (Type III & IV). These four can types were arranged into various arrays.

The first group of critical experiments were arrays comprised of the smaller cans (Type I) containing unmoderated enriched $UO_2$. Two critical experimental results were reported involving arrays of Type I cans. Type I cans had an inner radius of 3.735 cm and an inside height of 11.59 cm. The average mass of the Type I can was 128 g with a wall thickness of 0.055 cm. The average $UO_2$ mass loading per Type I can was 421 ± 1 g. The height of $UO_2$ in the cans averaged

4.48 cm from height measurements that varied between 3.6 cm and 5.3 cm (±20%). Therefore, the average density of 2.1442 g/cm$^3$ was also uncertain by ± 20%. One hundred ninety-six Type I cans were filled with $UO_2$ for the purposes of these experiments.

Experiment 1 consisted of a 7x7x4 array of Type I cans. Each Type I can was centered at the bottom of a space with dimensions of 8.05 cm x 8.05 cm x 11.71 cm. Methyl methacrylate (Plexiglas) 2.34 cm thick separated the units. The array of Type I cans, were surrounded by a polyethylene reflector with a thickness of 15.24 cm. Delayed critical was achieved with the split table separated by a 0.92-in. (2.3368-cm) gap with 112 Type I cans on the movable half of the split-table assembly and 84 Type I cans were on the fixed half. The reactivity of this configuration at full closure of the split table was given as $k_{eff}$= 1.0167.

Experiment 2 consisted of the same parameters as described in Experiment 1, except 18 units were removed from the top layer. Seven units were removed from each outside top layer row parallel with the split-table separation plane. Four additional units were removed from the next row from one outside edge of the array. The experimental multiplication factor was determined to be $k_{eff}$= 1.0012 with the split-table apparatus at full closure.

Experiments 3 and 4 were hypothetical experiments and were not actually performed.

The next set of experiments consisted of arrays of Type II cans. The Type II cans had an inner radius of 7.65 cm and an inside height of 21.76 cm. The average mass of the Type II can was 360 g with a wall thickness of 0.03 cm. The average $UO_2$ mass loading per Type II can was 20,000 g. Eighteen (18) Type II cans were filled with $UO_2$ for the purposes of these experiments. Fourteen experiments, involving the Type II cans, were reported. The first three were conducted with no Plexiglas between the Type II cans. The next six were conducted with different thicknesses of Plexiglas between the top and bottom units. The last five experiments involving Type II cans were conducted with various thicknesses of Plexiglas in all three dimensions.

Experiment 5 consisted of a 4x4x1 array of Type II cans. The horizontal space between cans was 0.34 cm. A space between the top and bottom of the cans and the polyethylene reflector of 0.24 cm was created by the rolled edges of the Type II cans. Delayed critical was achieved with the split table separated by a 0.64-in. (1.6256-cm) gap. Twelve Type II cans were on the movable half of the split-table assembly and 4 Type II cans were on the fixed half of the split-table assembly. The reactivity in this configuration at full closure of the split table was given as $k_{eff}$= 1.0094.

The next six experiments were arrays of Type II cans with a sheet of Plexiglas between the top and bottom layers of units.

Experiment 8 consisted of a 3x3x2 array of Type II cans. The horizontal space between the units was 5.08 cm. A Plexiglas sheet, 2.36 cm thick, was present between the two layers of stacked cans. A 0.24-cm space between the top and bottom of the cans and the polyethylene reflector as well as the Plexiglas was created by the rolled edges of the Type II cans. The experimental

multiplication factor was determined to be $k_{eff}$ = 1.0001 with the split-table apparatus at full closure.

Experiment nine was exactly like Experiment 8 except the Plexiglas sheet between the two layers was 3.62 cm thick. Delayed critical was achieved with the split table separated by a 1.01-in. (2.5654-cm) gap with 12 Type II cans on the movable half of the split-table assembly and six Type II cans on the fixed half. The reactivity in this configuration at full closure of the split table was given as $k_{eff}$= 1.0137.

Experiment 10 was exactly like Experiment #9 except that one corner unit in the top layer was removed and the split-table apparatus was fully closed. The experimental multiplication factor was determined to be $k_{eff}$= 0.9996.

Experiment 11 consisted of a 3x3x2 array of Type II cans. The horizontal space between the cans was 5.08 cm. A Plexiglas sheet, 4.82 cm thick, was between the two stacked layers of cans. A space between the top and bottom of the cans and the polyethylene reflector as well as the Plexiglas sheet was 0.24 cm. Delayed critical was achieved with the split table not fully closed. The two halves of the split table were separated by a 0.97-in. (2.4638-cm) gap. Twelve Type II cans were on the movable half of the split-table assembly and 6 Type II cans were on the fixed half of the split-table assembly. The reactivity in this configuration at full closure of the split table was given as $k_{eff}$= 1.0129.

Experiment 12 was exactly like Experiment #11 except one corner can in the top layer was removed and the split-table apparatus was fully closed. The experimental multiplication factor was determined to be $k_{eff}$= 0.9989 with the split-table apparatus at full closure.

Experiment 13 was exactly like Experiment 8 except that the Plexiglas sheet was 6.08-cm-thick between the two layers of cans. The experimental multiplication factor was determined to be $k_{eff}$ = 1.0049 with the split-table apparatus at full closure.

Supplemental information indicates that Experiment 14 was not actually performed; therefore, it is not included.

The next five experiments consisted of arrays of Type II cans with Plexiglas separating the units. Experiment 15 consisted of a 3x3x2 array of Type II cans with three cans removed from one of the top edges of the array. Each Type II can was in the center of a space 20.54-cm square in the horizontal directions and separated from an adjacent can by Plexiglas 4.86 cm thick. A Plexiglas sheet, 4.82 cm thick, was between the two stacked layers of cans. A 0.24-cm space between the top and bottom of the cans and the Plexiglas sheet was created by the rolled edges of the can. The experimental multiplication factor was determined to be $k_{eff}$ = 1.0100 with the split-table apparatus at full closure.

Experiment 16 was exactly like Experiment 15 except that two cans from opposite corners and a third can from the middle of an edge were removed. The experimental multiplication factor was determined to be $k_{eff}$ = 1.0018 with the split-table apparatus at full closure.

Experiment 17 consisted of a 3x3x2 array of Type II cans with two cans removed from opposite corners of the top layer of the array. Each Type II can was in the center of a space 20.54-cm square in the horizontal directions and separated from an adjacent can by Plexiglas 4.86 cm thick. A space of 4.37 cm was above each can and 0.24 cm below each can. A 4.82-cm-thick sheet of Plexiglas was placed between the two layers of cans. The experimental multiplication factor was determined to be $k_{eff}$ = 1.0012 with the split-table apparatus at full closure.

Experiment 18 was exactly as described in Experiment 17 except that all 18 cans were present and the space was 7.86 cm above the cans and 0.24 cm below. The experimental multiplication factor was determined to be $k_{eff}$ = 1.009 with the split-table apparatus at full closure.

Experiment 19 consisted of a 3x3x2 array of Type II cans. Each can was centered in a space 30.6 cm wide in the x direction and 20.44 cm wide in the y direction. A space between the top and bottom of the cans and the polyethylene reflector was 0.24 cm created by the rolled edges of the can. A 2.41-cm-thick Plexiglas sheet was between the two layers and a 2.43-cm-thick Plexiglas sheet between adjacent cans. The experimental multiplication factor was determined to be $k_{eff}$ =1.0054 with the split-table apparatus at full closure.

The next eight experiments used Type III cans. Each Type III can contained a material mixture of $UO_2$ and alcohol. The alcohol was actually a solution of 95% $C_2H_6O$ (alcohol) and 5% water. Type III cans had an inner radius of 6.76 cm and an inside height of 25.18 cm. The average mass of a Type III can was 341 g with a wall thickness of 0.03 cm. Extra steel in the friction-closure lid was estimated to increase the effective thickness of the top of the can to 0.05 cm. The average $UO_2$ mass loading per Type III can was 17,000 g. The average $UO_2$ height inside each can was 17.87 cm.

Experiment 20 consisted of a 2x2x1 array of Type III cans. Each can was centered in a space 13.97-cm square in the horizontal directions. There was a 0.15-cm space above and below each can. The experimental multiplication factor was determined to be $k_{eff}$ = 1.0065 with the split-table apparatus at full closure.

Experiment 21 consisted of a 2x2x1 array of Type III cans. Each can was centered in a space 14.05-cm square in the horizontal directions. There was a 0.15-cm space above and below each can. The experimental multiplication factor was determined to be $k_{eff}$ = 1.0031 with the split-table apparatus at full closure.

Experiment 22 consisted of a 2x2x1 array of Type III cans. Each can was centered in a space 14.128-cm square in the horizontal directions. There was a 0.15-cm space above and below each can. The experimental multiplication factor was determined to be $k_{eff}$ = 1.0018 with the split-table apparatus at full closure.

Experiment 23 consisted of a 2x3x1 array of Type III cans. Each can was centered in a space 15.24-cm square in the horizontal directions. There was a 0.15-cm space above and below each can. Delayed critical was achieved with the split table not fully closed. The two halves of the split table were separated by a 0.60-in. (1.524-cm) gap with 3 Type III cans on the movable half of the split-table assembly and 3 Type III cans placed on the fixed half. The reactivity of this configuration at full closure of the split table was given as $k_{eff}$= 1.0146.

Experiment 24 consisted of a 2x3x1 array of Type III cans. Each can was centered in a space 15.876-cm square in the horizontal directions. There was a 0.15-cm space above and below each can. The experimental multiplication factor was determined to be $k_{eff}$ = 1.0025 with the split-table apparatus at full closure.

Experiment 25 consisted of a 2x3x1 array of Type III cans. Each can was centered in a space 16.034-cm square in the horizontal directions. There was a 0.15-cm space above and below each can. The experimental multiplication factor was determined to be $k_{eff}$ = 0.9988 with the split-table apparatus at full closure.

Experiment 26 consisted of a 2x4x1 array of Type III cans. Each can was centered in a space 16.986-cm square in the horizontal directions. There was a 0.15-cm space above and below each can. The experimental multiplication factor was determined to be $k_{eff}$ = 1.0003 with the split-table apparatus at full closure.

Experiment 27 consisted of a 3x3x1 array of Type III cans. Each can was centered in a space 18.098-cm square in the horizontal directions. There was a 0.15-cm space above and below each can. The experimental multiplication factor was determined to be $k_{eff}$ = 0.9996 with the split-table apparatus at full closure.

The last two experiments conducted used Type IV cans. The nine (9) Type IV cans were actually the nine Type III cans with 300 g of the alcohol layer above the $UO_2$-alcohol region removed. The layer of alcohol above the $UO_2$-alcohol region was reported to have a height of 0.53 cm, a mass of 61.5 g, and a density of 0.8020 g/cm$^3$. Experiment 28 consisted of a 3x3x1 array of Type IV cans. Each can was centered in a space 17.146-cm square in the horizontal directions. There was a 0.15-cm space above and below each can. The experimental multiplication factor was determined to be $k_{eff}$= 1.0042 with the split-table apparatus at full closure.

Experiment 29 consisted of a 3x3x1 array of Type IV cans. Each can was centered in a space 17.304-cm square in the horizontal directions. There was a 0.15-cm space above and below each can. The experimental multiplication factor was determined to be $k_{eff}$ = 0.9997 with the split-table apparatus at full closure. Results for these experiments are shown in Table 5-37.

Table 5-37 Results for HEU-COMP-MIXED-001

| Experiment/Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hcm-1 | 1.0000 | 0.0059 | 0.1045 | 1.0027 | 0.0010 | 0.0027 | ± | 0.0120 |
| hcm-2 | 1.0012 | 0.0059 | 0.1053 | 1.0059 | 0.0011 | 0.0047 | ± | 0.0120 |
| hcm-5 | 0.9985 | 0.0056 | 0.7833 | 0.9963 | 0.0010 | -0.0022 | ± | 0.0114 |
| hcm-6 | 0.9953 | 0.0056 | 0.7962 | 0.9899 | 0.0010 | -0.0054 | ± | 0.0114 |
| hcm-7 | 0.9997 | 0.0038 | 0.8015 | 0.9949 | 0.0010 | -0.0048 | ± | 0.0079 |
| hcm-8 | 0.9984 | 0.0052 | 0.6872 | 0.9915 | 0.0011 | -0.0069 | ± | 0.0106 |
| hcm-9 | 0.9983 | 0.0052 | 0.6536 | 0.9931 | 0.0011 | -0.0052 | ± | 0.0106 |
| hcm-10 | 0.9979 | 0.0052 | 0.6494 | 0.9941 | 0.0010 | -0.0038 | ± | 0.0106 |
| hcm-11 | 0.9983 | 0.0052 | 0.6385 | 0.9934 | 0.0011 | -0.0049 | ± | 0.0106 |
| hcm-12 | 0.9972 | 0.0052 | 0.6358 | 0.9960 | 0.0011 | -0.0012 | ± | 0.0106 |
| hcm-13 | 1.0032 | 0.0053 | 0.6309 | 0.9977 | 0.0011 | -0.0055 | ± | 0.0108 |
| hcm-15 | 1.0083 | 0.0050 | 0.4671 | 1.0063 | 0.0011 | -0.002 | ± | 0.0102 |
| hcm-16 | 1.0001 | 0.0046 | 0.4692 | 0.9949 | 0.0011 | -0.0075 | ± | 0.0095 |
| hcm-17 | 0.9997 | 0.0046 | 0.4647 | 0.9926 | 0.0010 | -0.0071 | ± | 0.0094 |
| hcm-18 | 1.0075 | 0.0046 | 0.4625 | 1.0000 | 0.0010 | -0.0075 | ± | 0.0094 |
| hcm-19 | 1.0039 | 0.0047 | 0.5191 | 1.0000 | 0.0011 | -0.0039 | ± | 0.0096 |
| hcm-20 | 1.0060 | 0.0065 | 0.5357 | 1.0051 | 0.0015 | -0.0009 | ± | 0.0133 |
| hcm-21 | 1.0026 | 0.0064 | 0.5378 | 1.0046 | 0.0016 | 0.0020 | ± | 0.0132 |
| hcm-22 | 1.0013 | 0.0064 | 0.5371 | 0.9995 | 0.0016 | -0.0018 | ± | 0.0132 |
| hcm-23 | 0.9995 | 0.0053 | 0.5350 | 1.0056 | 0.0015 | 0.0061 | ± | 0.0110 |
| hcm-24 | 1.0020 | 0.0053 | 0.5352 | 1.0003 | 0.0016 | -0.0017 | ± | 0.0111 |
| hcm-25 | 0.9983 | 0.0053 | 0.5333 | 0.9970 | 0.0014 | -0.0013 | ± | 0.0110 |
| hcm-26 | 0.9998 | 0.0053 | 0.5283 | 1.0001 | 0.0015 | 0.0003 | ± | 0.0110 |
| hcm-27 | 0.9991 | 0.0053 | 0.5302 | 0.9978 | 0.0016 | -0.0013 | ± | 0.0111 |
| hcm-28 | 1.0037 | 0.0053 | 0.5410 | 1.0033 | 0.0015 | -0.0005 | ± | 0.0110 |
| hcm-29 | 0.9992 | 0.0052 | 0.5401 | 0.9998 | 0.0014 | 0.0006 | ± | 0.0108 |

#### 5.2.1.2.4 HEU-COMP-MIXED-002

The twenty-three measurements that are reported in NEA (2001), HEU-COMP-MIXED-002, are part of a series performed in 1985 at the Solution Physics Facility of the Institute of Physics and Power Engineering (IPPE), Obninsk, Russia with highly enriched uranium dioxide. Critical experiment measurements were made with 19 cylinders of dry uranium dioxide in stainless steel cans immersed in mixtures of heavy and light water. The outer diameter of each of the 19 fuel

cylinders was 8.3 cm. Fuel cylinders were arranged inside of a hexagonal tank in triangular lattices with pitches 8.32 cm (tightly packed fuel cylinders), 9.32 cm, and 11.33 cm. Criticality was achieved by filling the space between cylinders with moderator, which was light water or one of three different mixtures of light and heavy water. All experiments were done with a thick bottom light-water reflector. Fourteen of the experiments also had a lateral light-water reflector. Although experimental uncertainties are higher than typically encountered (>1%), all 23 experiments are acceptable benchmark experiments (NEA [2001]).

Although these experiments are identified as "MIXED," the calculated spectrum of Case 6 is intermediate and the calculated spectra of Cases 1, 2, 21, 22, and 23 are fast.

The core was inside an exchangeable stainless steel hexagonal tank installed on the support inside the square reflector tank. A hexagonal stainless steel grid plate, 1.00±0.01 cm thick, was laid on the bottom of the hexagonal tank. The grid plate had 19 holes 0.92 cm in diameter drilled at the nodes of the triangular lattice at the required pitch. Fuel cylinders were made from stainless steel tubes with outer diameter 8.3 cm filled with uranium dioxide. The bottom plug of each tube had a cylindrical tip 0.90±0.01 cm in diameter and 1.0 cm long. The fuel cylinders were fixed in the holes of the bottom grid plate by these tips. At the top, fuel cylinders were fixed in the required triangular lattice by means of five aluminum slats attached to the flange of the hexagonal tank. The approximate thickness and width of these slats were 0.3 cm and 3.0 cm respectively. The cylinders were bolted to the slats through holes drilled at the proper distance from each other. The support structure was made of stainless steel channels. The mass of the channel beams was 5.9 kg/meter. Results for HEU-COMP-MIXED-002 are given in Table 5-38.

Table 5-38 Results for HEU-COMP-MIXED-002

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| Hcm02_1 | 1.0000 | 0.0085 | 0.868 | 0.9866 | 0.0017 | -0.0134 | ± | 0.0173 |
| Hcm02_2 | 1.0000 | 0.0088 | 0.865 | 0.9907 | 0.0017 | -0.0093 | ± | 0.0179 |
| Hcm02_3 | 1.0000 | 0.0093 | 0.724 | 0.9914 | 0.0016 | -0.0086 | ± | 0.0189 |
| Hcm02_4 | 1.0000 | 0.0087 | 0.716 | 0.9923 | 0.0017 | -0.0077 | ± | 0.0177 |
| Hcm02_5 | 1.0000 | 0.0089 | 0.722 | 0.9933 | 0.0017 | -0.0067 | ± | 0.0181 |
| hcm02_6 | 1.0000 | 0.0093 | 0.574 | 0.9852 | 0.0018 | -0.0148 | ± | 0.0189 |
| hcm02_7 | 1.0000 | 0.0086 | 0.578 | 0.9813 | 0.0019 | -0.0187 | ± | 0.0176 |
| hcm02_8 | 1.0000 | 0.0068 | 0.537 | 0.9943 | 0.0018 | -0.0057 | ± | 0.0141 |
| hcm02_9 | 1.0000 | 0.0076 | 0.541 | 0.9913 | 0.0018 | -0.0087 | ± | 0.0156 |
| hcm02_10 | 1.0000 | 0.0081 | 0.570 | 0.9856 | 0.0019 | -0.0144 | ± | 0.0167 |
| hcm02_11 | 1.0000 | 0.0088 | 0.568 | 0.9829 | 0.0019 | -0.0171 | ± | 0.0180 |
| hcm02_12 | 1.0000 | 0.0078 | 0.556 | 0.9900 | 0.0019 | -0.0100 | ± | 0.0160 |
| hcm02_13 | 1.0000 | 0.0083 | 0.559 | 0.9874 | 0.0017 | -0.0126 | ± | 0.0169 |

| Table 5-38 Results for HEU-COMP-MIXED-002 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hcm02_14 | 1.0000 | 0.0112 | 0.735 | 0.9880 | 0.0017 | -0.0120 | ± | 0.0227 |
| hcm02_15 | 1.0000 | 0.0111 | 0.730 | 0.9850 | 0.0017 | -0.0150 | ± | 0.0225 |
| hcm02_16 | 1.0000 | 0.0108 | 0.735 | 0.9861 | 0.0017 | -0.0139 | ± | 0.0219 |
| hcm02_17 | 1.0000 | 0.0112 | 0.732 | 0.9861 | 0.0016 | -0.0139 | ± | 0.0226 |
| hcm02_18 | 1.0000 | 0.0111 | 0.727 | 0.9902 | 0.0017 | -0.0098 | ± | 0.0225 |
| hcm02_19 | 1.0000 | 0.0107 | 0.712 | 0.9910 | 0.0017 | -0.0090 | ± | 0.0217 |
| hcm02_20 | 1.0000 | 0.0108 | 0.735 | 0.9824 | 0.0018 | -0.0176 | ± | 0.0219 |
| hcm02_21 | 1.0000 | 0.0092 | 0.902 | 0.9843 | 0.0016 | -0.0157 | ± | 0.0187 |
| hcm02_22 | 1.0000 | 0.0090 | 0.899 | 0.9879 | 0.0019 | -0.0121 | ± | 0.0184 |
| hcm02_23 | 1.0000 | 0.0093 | 0.896 | 0.9866 | 0.0016 | -0.0134 | ± | 0.0189 |

### 5.2.1.3 HEU Rodded Arrays

#### 5.2.1.3.1 HEU-COMP-INTER-001

During 1992-1993, an experiment was performed at the RRC Kurchatov Institute to investigate the accidental water immersion criticality safety of the thermionic intermediate reactor-converter with highly enriched fuel (approximately 96% $^{235}U$), zirconium hydride moderator, and beryllium reflector. Described in this evaluation are six configurations of critical assemblies simulating water ingress into different reactor cavities, as well as the complete water immersion of the reactor. All six configurations are considered acceptable for use as benchmark critical experiments. Table 5-39 shows results for configuration 5 of the experiment.

| Table 5-39 Results for HEU-COMP-INTER-001 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heuci1c5 | 1.0000 | 0.0000 | 0.220 | 0.9951 | 0.0008 | -0.0049 | ± | 0.0016 |

#### 5.2.1.3.2 HEU-COMP-THERM-002

Critical experiments of hexagonal graphite rods containing highly enriched uranium moderated and reflected by water were performed at ORNL from November 1965 to February 1966. The experiments were performed with Westinghouse type NRX-A3 and NRX-A4 fuel elements to provide data for criticality safety analyses of Rover fuel element production, storage, and transportation.

The Rover program was started in the late 1950s to provide a means of propulsion for space missions. Criticality was achieved as liquid hydrogen traveled down holes within the elements. Thermal energy produced from criticality converted the liquid hydrogen into a high-pressure gas, which exited the spacecraft at a high velocity providing propulsion.

Each NRX-A fuel element was a nominally 52-in.-long hexagonal graphite cylinder with a 0.870-in. diagonal (0.435-in. sides), containing 19 uniformly spaced longitudinal holes. Pyrolytic graphite-coated uranium dicarbide beads were uniformly distributed in the graphite. Three experiments were performed with shorter, 15.3125-in.-length fuel elements. Most experiments were performed with elements encased in sealed aluminum tubes which prevented water from coming in contact with the element, while other experiments were performed with elements in water. Vertically oriented arrays of both square-pitched and triangular-pitched configurations were constructed. A total of 25 critical assemblies were constructed. All 25 experiments are considered acceptable for use as benchmark experiments (NEA [2001]).

Four types of configurations were constructed:

1. Square-pitched lattices, elements in tubes
2. Triangular-pitched lattices, elements in tubes
3. Triangular-pitched lattices, bare elements
4. Triangular-pitched lattices, 15.3125-in. long bare elements.

Two fuel element types were used in the experiments, one designated as NRX-A3 and the other as NRX-A4. Except for differences in the uranium content, both types had the same composition – a dispersion of pyrolitic-graphite-coated uranium-dicarbide spherical beads in a graphite matrix. The uranium was enriched to 93.15 wt.% $^{235}U$. The average bead diameter was 0.003 in. and the average thickness of the pyrolytic graphite coating was 0.001 in. (no associated standard deviations were reported). The fuel elements were hexagonal cylinders with an "across-the-flats" measurement of 0.754 in. (0.435-in. sides, 0.870-in. diagonal) containing 19 uniformly spaced longitudinal holes. Results for HEU-COMP-THERM-002 are presented in Table 5-40.

Table 5-40 Results for HEU-COMP-THERM-002

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hct2_1 | 1.0011 | 0.0069 | 0.02439 | 1.0095 | 0.0017 | 0.0083 | ± | 0.0142 |
| hct2_2 | 1.0011 | 0.0069 | 0.01948 | 1.0157 | 0.0018 | 0.0146 | ± | 0.0142 |
| hct2_3 | 1.0011 | 0.0069 | 0.01643 | 1.0161 | 0.0017 | 0.0150 | ± | 0.0142 |
| hct2_4 | 1.0011 | 0.0069 | 0.01449 | 1.0186 | 0.0019 | 0.0175 | ± | 0.0143 |
| hct2_5 | 1.0011 | 0.0069 | 0.01295 | 1.0181 | 0.0016 | 0.0170 | ± | 0.0141 |
| hct2_6 | 1.0011 | 0.0069 | 0.01208 | 1.0182 | 0.0016 | 0.0171 | ± | 0.0141 |
| hct2_7 | 1.0011 | 0.0069 | 0.01181 | 1.0171 | 0.0014 | 0.0160 | ± | 0.0141 |

Table 5-40 Results for HEU-COMP-THERM-002

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hct2_8 | 1.0011 | 0.0069 | 0.01054 | 1.0178 | 0.0016 | 0.0167 | ± | 0.0141 |
| hct2_9 | 1.0011 | 0.0069 | 0.00994 | 1.0173 | 0.0013 | 0.0162 | ± | 0.0140 |
| hct2_10 | 1.0011 | 0.0069 | 0.00986 | 1.0160 | 0.0013 | 0.0149 | ± | 0.0140 |
| hct2_11 | 1.0011 | 0.0053 | 0.02013 | 1.0155 | 0.0018 | 0.0144 | ± | 0.0112 |
| hct2_12 | 1.0011 | 0.0053 | 0.01411 | 1.0171 | 0.0016 | 0.0159 | ± | 0.0111 |
| hct2_13 | 1.0011 | 0.0053 | 0.01249 | 1.0219 | 0.0017 | 0.0208 | ± | 0.0111 |
| hct2_14 | 1.0011 | 0.0055 | 0.01201 | 1.0179 | 0.0017 | 0.0168 | ± | 0.0115 |
| hct2_15 | 1.0011 | 0.0055 | 0.01122 | 1.0181 | 0.0014 | 0.0170 | ± | 0.0114 |
| hct2_16 | 1.0011 | 0.0055 | 0.01024 | 1.0161 | 0.0016 | 0.0150 | ± | 0.0114 |
| hct2_17 | 1.0011 | 0.0055 | 0.00978 | 1.0196 | 0.0013 | 0.0185 | ± | 0.0113 |
| hct2_18 | 1.002 | 0.0043 | 0.02096 | 1.0190 | 0.0018 | 0.0170 | ± | 0.0093 |
| hct2_19 | 1.002 | 0.0043 | 0.01522 | 1.0155 | 0.0016 | 0.0135 | ± | 0.0092 |
| hct2_20 | 1.002 | 0.0043 | 0.01223 | 1.0135 | 0.0017 | 0.0115 | ± | 0.0093 |
| hct2_21 | 1.002 | 0.0043 | 0.01125 | 1.0178 | 0.0014 | 0.0157 | ± | 0.0091 |
| hct2_22 | 1.002 | 0.0043 | 0.00936 | 1.0183 | 0.0015 | 0.0163 | ± | 0.0091 |
| hct2_23 | 1.0008 | 0.0085 | 0.01542 | 1.0196 | 0.0018 | 0.0188 | ± | 0.0174 |
| hct2_24 | 1.0008 | 0.0085 | 0.01212 | 1.0185 | 0.0016 | 0.0177 | ± | 0.0173 |
| hct2_25 | 1.0008 | 0.0085 | 0.0112 | 1.0136 | 0.0017 | 0.0128 | ± | 0.0173 |

#### 5.2.1.3.3 Kurchatov Institute Steel-Clad $UO_2$ Rod Experiments

A series of critical and subcritical experiments (reported in NEA (2001) as HEU-COMP-THERM-011, HEU-COMP-THERM-012, HEU-COMP-THERM-013, and HEU-COMP-THERM-014) with water-moderated clusters of highly enriched (~ 80wt.% $^{235}U$ ) fuel rods with steel clad was carried out at he RRC «Kurchatov Institute» in 1997. The fuel material was a mixture of $UO_2$ in an aluminum atrix. The four or nine square clusters consisted of uniform square-pitched fuel-rod lattices. The clusters were sized from 20x20 cm to 30x30 cm. The $\rho^{H}/\rho^{U235}$ ratio was 35 to 40 for the 14-mm-pitch lattices and ~170 for the 19.8-mm-pitch lattices (HEU-COMP-THERM-014). All critical conditions were achieved with a partially flooded core.

HEU-COMP-THERM-011 describes a series of five experiments with cores consisting of 1764 fuel rods assembled into four clusters of 21x21 fuel rods (30x30 cm) with ratio $\rho^{H}/\rho^{U235} \approx 40$. The fuel-rod lattice pitch value was 14 mm. The distance between clusters varied from 0 to 5.6 cm. Three of these experiments were critical with partially flooded cores, while the two remaining were fully flooded subcritical experiments. All five experiments were performed under the same

carefully controlled and monitored conditions. The three critical experiments are considered to be acceptable for use as benchmark critical experiments.

Experiments were performed in an open-top, steel tank, which was 2000 mm in diameter and 3000 mm high. The upper support ring of the tank was fixed on the floor of the experimental room. The support ring of the core basket sat on the tank support ring. The core basket consisted of two support plates and three lattice plates. All details of the basket were fixed to the upper support ring by six 60-mm-diameter stainless-steel tubes of wall thickness 4.5 mm.

The fuel-rod bottom caps rested upon a support plate made of 10-mm-thick D1 aluminum alloy, which, in turn, lay on a 30-mm-thick stainless-steel support plate. Both support plates were 1365 mm in diameter. In the bottom stainless-steel support plate were 162 holes 30 mm in diameter (6.0-cm hexagonal pitch).

Lattice plates maintained the 14-mm square lattice pitch of the fuel rods. According to the technical specifications for the lattice plate, the dimension between the centers of the first row of holes and 52$^{nd}$ (last) row is 714.0±0.1 mm. The diameter of the holes in the lattice plates was 13.15±0.05 mm. All three lattice plates (upper, middle, and bottom) have the same configuration and are made of 10-mm-thick D1 aluminum alloy sheet. The heights of the lattice plates were 60, 740, and 1420 mm above the support plates.

The side water reflector was over 500 mm thick. The bottom water reflector was more than 900 mm thick. For the three critical experiments, the water level was within the fuel region and there was no upper water reflector. For the two subcritical experiments the upper water reflector was more than 250 mm above the top level of the fuel.

The control rods were 8-mm-diameter boron-carbide rods positioned in aluminum guide tubes with thin perforated walls. The rods were situated either at the core periphery or in gaps between clusters. During the experiments all rods were withdrawn from the core to a height of at least 500 mm above the core and had no influence on critical conditions.

The fuel rods (fuel pins) were steel tubes filled with fuel. The outer diameter of the steel tubes was 13.04±0.04 mm and the inner diameter was 10.66±0.06 mm. The fuel unit had a 0.5-mm-thick leaktight zirconium clad, 10 mm in outer diameter and 61.5 mm long. The fuel cylinder within the clad was 7 mm in diameter and 60 mm long. Eight fuel units were placed in each fuel-rod tube. Above and below the fuel units were placed 10-mm-diameter stainless-steel cylindrical plugs. Above the top plug was a 56 mm air gap followed by a steel washer. Above the air gap was the top cap with a 25-mm-long pin for handling the fuel rod. Including the handling pin, the top cap was 58 mm long. The top cap was solid steel. Below the bottom plug, the steel bottom cap screwed onto the bottom of the fuel pin and compressed a rubber washer, which was sandwiched between two 1-mm-thick stainless-steel washers. This threaded portion of the fuel pin had a slightly larger inside diameter than the rest of the rod, and formed a lip that the steel washers, also having the same larger diameter, were compressed against, which expanded the rubber washer forming a waterproof seal. Thus, except for the rubber washer that was 3 mm

thick, the bottom end cap together with the fuel-pin clad formed an essentially solid piece when assembled. The fuel rod's total length was 1517 mm.

Each core configuration consisted of four square clusters made up of square-pitched fuel-rod lattices. In this series, five cores were assembled. The cores differed by the distance between clusters, which was created by vacant fuel-rod rows. Beginning with no separation between the four clusters, for each experiment the distance between clusters was enlarged by one row (1.4 cm) relative to the preceding experiment. All configurations were assembled dry and then were flooded with moderator (water) either until criticality was reached, or until they were completely flooded with an upper reflector thickness >250 mm above the fuel level for the subcritical experiments. The assembly's subcriticality was then assessed. The water temperature was measured during the course of each experiment. The water level was measured by two float-type water-level meters. Measurements were taken after surface waves subsided (~10 minutes). At the time of the measurements the heights indicated by the two meters differed by no more than 0.05-0.07 mm.

HEU-COMP-THERM-012 describes a series of five experiments with cores consisting of 1296 fuel rods of two types assembled into four clusters of 18x18 fuel rods (25x25 cm). Half the rods were the same type used in the other experiments of this series. The other half were rods with 30% more $UO_2$ in the $UO_2$-Al fuel mixture. The two types alternated in the lattice positions. The extra $UO_2$ in half the rods reduced the ratio $\rho^H/\rho^{U235}$ from 40 to $\approx 35$. The fuel-rod lattice-pitch value was 14 mm. The distance between clusters varied from 0 to 5.6 cm. Two of these experiments were critical with partially flooded cores, while the three remaining were fully flooded subcritical experiments. All five experiments were performed under the same carefully controlled and monitored conditions. The two critical experiments are considered to be acceptable for use as benchmark critical experiments.

HEU-COMP-THERM-013 describes a series of four experiments with cores consisting of 1764 fuel rods assembled into nine clusters of 14x14 fuel rods (20x20 cm) with ratio $\rho^H/\rho^{U235} \approx 40$. The fuel-rod lattice-pitch value was 14 mm. The distance between clusters varied from 1.4 to 5.6 cm. (Case 1 of HEU-COMP-THERM-011 is the same lattice with no separation between clusters.) Two of these experiments were critical with partially flooded cores, while the two remaining were fully flooded subcritical experiments. All four experiments were performed under the same carefully controlled and monitored conditions. The two critical experiments are considered to be acceptable for use as benchmark critical experiments.

HEU-COMP-THERM-014 describes a series of four experiments with cores consisting of 900 fuel rods assembled into nine clusters of 10x10 fuel rods (20x20 cm) with ratio $\rho^H/\rho^{U235} \approx 170$. The fuel-rod lattice-pitch value was 19.8 mm. The distance between clusters varied from 0 to 5.94 cm. Two of these experiments were critical with partially flooded cores, while the two remaining were fully flooded subcritical experiments. All four experiments were performed under the same carefully controlled and monitored conditions. The two critical experiments are considered to be acceptable for use as benchmark critical experiments. Results for these experiments are given in Table 5-41.

Table 5-41 Results for HEU-COMP-THERM-011, HEU-COMP-THERM-012, HEU-COMP-THERM-013, HEU-COMP-THERM-014

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| HEU-COMP-THERM-011 | | | | | | | | |
| hct11_1 | 0.9988 | 0.0042 | 0.0533 | 0.9868 | 0.0009 | -0.0120 | ± | 0.0086 |
| hct11_2 | 0.9988 | 0.0042 | 0.0503 | 0.9875 | 0.0009 | -0.0113 | ± | 0.0088 |
| hct11_3 | 0.9988 | 0.0042 | 0.0470 | 0.9881 | 0.0010 | -0.0107 | ± | 0.0090 |
| HEU-COMP-THERM-012 | | | | | | | | |
| hct12_1 | 0.9987 | 0.0032 | 0.0551 | 0.9858 | 0.0010 | -0.0129 | ± | 0.0067 |
| hct12_2 | 0.9987 | 0.0034 | 0.0509 | 0.9895 | 0.0009 | -0.0092 | ± | 0.0070 |
| HEU-COMP-THERM-013 | | | | | | | | |
| hct13_1 | 0.9988 | 0.0042 | 0.0481 | 0.9883 | 0.0008 | -0.0105 | ± | 0.0086 |
| hct13_2 | 0.9988 | 0.0043 | 0.0432 | 0.9902 | 0.0009 | -0.0086 | ± | 0.0088 |
| HEU-COMP-THERM-014 | | | | | | | | |
| hct14_1 | 0.9986 | 0.0048 | 0.0259 | 0.9976 | 0.0009 | -0.0010 | ± | 0.0098 |
| hct14_2 | 0.9986 | 0.0049 | 0.0227 | 0.9970 | 0.0008 | -0.0016 | ± | 0.0099 |

### 5.2.2 HEU Benchmarks Related to Degraded Waste Package Configurations

#### 5.2.2.1 HEU Solution Configurations

##### 5.2.2.1.1 HEU-SOL-THERM-005

Various critical experiments of highly enriched (87.4 wt% $^{235}U$) solutions of uranyl nitrate [$UO_2(NO_3)_2$] and uranyl fluoride [$UO_2F_2$] poisoned with commercially available Pyrex glass pipes were performed at ORNL in 1958 (NEA [2001, HEU-SOL-THERM-005]). Solutions and glass pipes were contained in a cylindrical stainless steel tank with a 20-inch inner diameter. The configurations were vertically oriented borated glass pipes in hexagonal arrays. Critical heights of the highly enriched solutions were measured. A few experiments were also performed without glass (clean criticals). Some experiments had full water reflection on sides and bottom. Seventeen of the 27 Pyrex glass pipe experiments, including five without glass, are considered acceptable for use as benchmark experiments. The other ten experiments are rejected for lack of important experimental details or inconsistencies of those given. in open-topped cylindrical tank containing fissile solution and Pyrex glass pipes was inside a 9-foot-diameter cylindrical steel tank which contained water for reflected experiments. The large reflector tank sat on a steel grate elevated at least 8 ft above the floor. The grate had a density equivalent to 1/2 in. of solid plate. Results for HEU-SOL-THERM-005 are given in Table 5-42.

Table 5-42 Results for HEU-SOL-THERM-005

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U^a$ |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| heust5c1 | 1.0000 | 0.0124 | 0.035 | 1.0007 | 0.0011 | 0.0007 | ± | 0.0249 | 80.3 |
| heust5c2 | 1.0000 | 0.0124 | 0.041 | 0.9880 | 0.0012 | -0.0120 | ± | 0.0249 | 80.3 |
| heust5c3 | 1.0000 | 0.0124 | 0.036 | 1.0128 | 0.0012 | 0.0128 | ± | 0.0249 | 80.3 |
| heust5c4 | 1.0000 | 0.0126 | 0.036 | 1.0128 | 0.0012 | 0.0128 | ± | 0.0253 | 80.3 |
| heust5c5 | 1.0000 | 0.0124 | 0.039 | 0.9937 | 0.0012 | -0.0063 | ± | 0.0249 | 80.3 |
| heust5c6 | 1.0000 | 0.0124 | 0.036 | 1.0082 | 0.0012 | 0.0082 | ± | 0.0249 | 80.3 |
| heust5c7 | 1.0000 | 0.0124 | 0.033 | 1.0048 | 0.0012 | 0.0048 | ± | 0.0249 | 80.3 |
| heust5c8 | 1.0000 | 0.0124 | 0.040 | 0.9914 | 0.0013 | -0.0086 | ± | 0.0249 | 80.3 |
| heust5c9 | 1.0000 | 0.0125 | 0.035 | 1.0070 | 0.0011 | 0.0070 | ± | 0.0251 | 80.3 |
| heust5c10 | 1.0000 | 0.0124 | 0.040 | 0.9947 | 0.0013 | -0.0053 | ± | 0.0249 | 80.3 |
| heust5c11 | 1.0000 | 0.0083 | 0.032 | 0.9885 | 0.0012 | -0.0115 | ± | 0.0168 | 80.3 |
| heust5c12 | 1.0000 | 0.0084 | 0.034 | 1.0201 | 0.0013 | 0.0201 | ± | 0.0170 | 80.3 |
| heust5c13 | 1.0000 | 0.0119 | 0.021 | 1.0028 | 0.0012 | 0.0028 | ± | 0.0239 | 141.0 |
| heust5c14 | 1.0000 | 0.0058 | 0.019 | 0.9910 | 0.0012 | -0.0090 | ± | 0.0118 | 141.0 |
| heust5c15 | 1.0000 | 0.0059 | 0.020 | 1.0115 | 0.0014 | 0.0115 | ± | 0.0121 | 141.0 |
| heust5c16 | 1.0000 | 0.0121 | 0.011 | 1.0068 | 0.0011 | 0.0068 | ± | 0.0243 | 276.0 |
| heust5c17 | 1.0000 | 0.0057 | 0.011 | 0.9926 | 0.0013 | -0.0074 | ± | 0.0117 | 276.0 |

NOTE: [a] $H/^{235}U$ denotes Hydrogen to $^{235}U$ ratio and is reported in NEA (2001), HEU-SOL-THERM-005.

#### 5.2.2.1.2 HEU-SOL-THERM-006

Twenty-nine critical experiments of highly enriched (93.6 wt.% $^{235}U$) uranyl nitrate [$UO_2(NO_3)_2$] poisoned with enriched boron in boric acid [$H_3BO_3$] were performed in 1985 at LANL (NEA [2001, HEU-SOL-THERM-006]). The solution was contained in a cylindrical tank surrounded by two annular regions, which provided for various radial reflector combinations. The inner annular region contained either air or a thick nickel sleeve. The outer annular region contained air, water, or borated water. Critical configurations had height to diameter ratios that ranged from 0.19 to 1.04. Uranium concentrations varied between 293.4 and 297.8 g/liter. The boron in the uranyl nitrate solution was enriched to 50.55 wt.% $^{10}B$, and concentrations varied between 0.0 and 2.64 g enriched B/liter. The $H/^{235}U$ ratio for the solution is calculated from Table 11 of NEA (2001), HEU-SOL-THERM-006 as between 84 and 85. All twenty-nine experiments are considered acceptable for use as benchmark experiments. Results for HEU-SOL-THERM-006 are shown in Table 5-43.

Table 5-43 Results for HEU-SOL-THERM-006

| Experiment /Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| heust6c1 | 0.9973 | 0.0050 | 0.033 | 0.9870 | 0.0015 | -0.0103 | ± | 0.0104 | 84.5 |
| heust6c2 | 0.9986 | 0.0054 | 0.036 | 0.9907 | 0.0014 | -0.0079 | ± | 0.0112 | 84.3 |
| heust6c3 | 1.0000 | 0.0065 | 0.037 | 1.0020 | 0.0013 | 0.0020 | ± | 0.0133 | 84.5 |
| heust6c4 | 1.0000 | 0.0078 | 0.040 | 1.0033 | 0.0012 | 0.0033 | ± | 0.0158 | 85.0 |
| heust6c5 | 1.0000 | 0.0091 | 0.041 | 1.0115 | 0.0011 | 0.0115 | ± | 0.0183 | 84.7 |
| heust6c6 | 1.0000 | 0.0087 | 0.042 | 1.0010 | 0.0011 | 0.0010 | ± | 0.0175 | 84.8 |
| heust6c7 | 1.0000 | 0.0088 | 0.043 | 1.0041 | 0.0011 | 0.0041 | ± | 0.0177 | 84.2 |
| heust6c8 | 0.9973 | 0.0050 | 0.032 | 0.9832 | 0.0014 | -0.0141 | ± | 0.0104 | 84.5 |
| heust6c9 | 0.9986 | 0.0054 | 0.035 | 0.9904 | 0.0013 | -0.0082 | ± | 0.0111 | 84.3 |
| heust6c10 | 1.0000 | 0.0065 | 0.037 | 1.0020 | 0.0013 | 0.0020 | ± | 0.0133 | 84.5 |
| heust6c11 | 1.0000 | 0.0088 | 0.042 | 1.0023 | 0.0011 | 0.0023 | ± | 0.0177 | 84.2 |
| heust6c12 | 0.9973 | 0.0050 | 0.032 | 0.9857 | 0.0014 | -0.0116 | ± | 0.0104 | 84.5 |
| heust6c13 | 0.9980 | 0.0054 | 0.035 | 0.9904 | 0.0014 | -0.0076 | ± | 0.0112 | 84.3 |
| heust6c14 | 1.0000 | 0.0078 | 0.039 | 1.0027 | 0.0013 | 0.0027 | ± | 0.0158 | 85.0 |
| heust6c15 | 1.0000 | 0.0091 | 0.041 | 1.0093 | 0.0011 | 0.0093 | ± | 0.0183 | 84.7 |
| heust6c16 | 1.0000 | 0.0087 | 0.042 | 1.0032 | 0.0012 | 0.0032 | ± | 0.0176 | 84.8 |
| heust6c17 | 1.0000 | 0.0088 | 0.042 | 1.0034 | 0.0012 | 0.0034 | ± | 0.0177 | 84.2 |
| heust6c18 | 1.0000 | 0.0078 | 0.039 | 1.0047 | 0.0012 | 0.0047 | ± | 0.0158 | 85.0 |
| heust6c19 | 1.0000 | 0.0091 | 0.040 | 1.0115 | 0.0011 | 0.0115 | ± | 0.0183 | 84.7 |
| heust6c20 | 1.0000 | 0.0087 | 0.041 | 1.0027 | 0.0012 | 0.0027 | ± | 0.0176 | 84.8 |
| heust6c21 | 1.0000 | 0.0088 | 0.042 | 1.0009 | 0.0011 | 0.0009 | ± | 0.0177 | 84.2 |
| heust6c22 | 1.0000 | 0.0065 | 0.038 | 1.0027 | 0.0014 | 0.0027 | ± | 0.0133 | 84.5 |
| heust6c23 | 1.0000 | 0.0078 | 0.039 | 1.0032 | 0.0013 | 0.0032 | ± | 0.0158 | 85.0 |
| heust6c24 | 1.0000 | 0.0091 | 0.040 | 1.0089 | 0.0012 | 0.0089 | ± | 0.0184 | 84.7 |
| heust6c25 | 1.0000 | 0.0087 | 0.042 | 1.0010 | 0.0011 | 0.0010 | ± | 0.0175 | 84.8 |
| heust6c26 | 1.0000 | 0.0088 | 0.042 | 1.0019 | 0.0011 | 0.0019 | ± | 0.0177 | 84.2 |
| heust6c27 | 0.9973 | 0.0050 | 0.032 | 0.9853 | 0.0014 | -0.0120 | ± | 0.0104 | 84.5 |
| heust6c28 | 0.9986 | 0.0054 | 0.036 | 0.9895 | 0.0014 | -0.0091 | ± | 0.0111 | 84.3 |
| heust6c29 | 1.0000 | 0.0088 | 0.042 | 1.0038 | 0.0011 | 0.0038 | ± | 0.0177 | 84.2 |

### 5.2.2.1.3 HEU-SOL-THERM-009

The four water-reflected spheres included reported in NEA [2001, HEU-SOL-THERM-009] are part of a series of experiments performed in the 1950's at the ORNL with highly enriched (93.18 wt.% $^{235}U$) uranium. Critical experiment measurements were made with uranium oxyfluoride ($UO_2F_2$) solutions at various uranium concentrations in two water-reflected spheres nominally nine in. in diameter (6.4 liters). Spherical reactors with nominal inner diameter of nine in. were fabricated of aluminum and surrounded by an effectively infinite water reflector. The spheres were supported in the water reflector only by the top and bottom overflow and feed tubes, respectively. Four critical configurations with different fuel concentrations were measured and are considered as acceptable for use as criticality safety benchmark experiments (NEA [2001]). Results for HEU-SOL-THERM-009 are given in Table 5-44.

Table 5-44 Results for HEU-SOL-THERM-009

| Experiment /Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ ratio[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| heust9c1 | 1.0000 | 0.0057 | 0.058 | 1.0051 | 0.0006 | 0.0051 | ± | 0.0115 | 35.8 |
| heust9c2 | 1.0000 | 0.0057 | 0.045 | 1.0045 | 0.0006 | 0.0045 | ± | 0.0115 | 47.3 |
| heust9c3 | 1.0000 | 0.0057 | 0.029 | 1.0047 | 0.0007 | 0.0047 | ± | 0.0115 | 76.1 |
| heust9c4 | 1.0000 | 0.0057 | 0.018 | 0.9994 | 0.0007 | -0.0006 | ± | 0.0115 | 126.5 |

NOTE: [a] The $H/^{235}U$ ratio are from page 3 of NEA (2001, HEU-SOL-THERM-009).

### 5.2.2.1.4 HEU-SOL-THERM-010

The four water-reflected sphere measurements reported in NEA (2001, HEU-SOL-THERM-010) are part of a series of experiments performed in the 1950's at the ORNL with highly enriched (93.13 % $^{235}U$ ) uranium. Critical experiment measurements were made with enriched uranium oxyfluoride solutions at various temperatures and uranium concentrations. A spherical reactor with nominal inner diameter of 26.4 cm (9.7 liters) was fabricated of aluminum and surrounded by an effectively infinite water reflector. The sphere was supported in the water reflector only by the top and bottom overflow and feed tubes, respectively. Four critical configurations at different temperatures and with different fuel concentrations were measured. All four configurations are considered acceptable for use as criticality safety benchmark experiments (NEA [2001]). Results for HEU-SOL-THERM-010 are shown in Table 5-45.

Table 5-45 Results for HEU-SOL-THERM-010

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ ratio[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $K_{eff}$ | $\sigma$ | | | | |
| heust10c1 | 1.0000 | 0.0018 | 0.009 | 1.0050 | 0.0007 | 0.0050 | ± | 0.0038 | 268.8 |
| heust10c2 | 1.0000 | 0.0018 | 0.009 | 1.0049 | 0.0006 | 0.0049 | ± | 0.0038 | 263.3 |
| heust10c3 | 1.0000 | 0.0018 | 0.010 | 1.0026 | 0.0006 | 0.0026 | ± | 0.0038 | 245.4 |

| Table 5-45 Results for HEU-SOL-THERM-010 | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ ratio[a] |
| | $k_{eff}$ | $\sigma$ | AENCF | $K_{eff}$ | $\sigma$ | | | | |
| heust10c4 | 1.0000 | 0.0018 | 0.010 | 0.9999 | 0.0007 | -0.0001 | ± | 0.0039 | 239.3 |

NOTE: [a] The $H/^{235}U$ ratio are from page 3 of NEA (2001, HEU-SOL-THERM-010).

### 5.2.2.1.5 HEU-SOL-THERM-011

The two water-reflected experiments reported in NEA (2001, HEU-SOL-THERM-011) are part of a series of measurements performed in the 1950s at the ORNL with highly-enriched uranium (93.2wt.% $^{235}U$). Critical experiment measurements were made with uranium oxyfluoride ($UO_2F_2$) solutions in a water-reflected 32-cm-inner-diameter (17-liter) sphere with an aluminum wall 1.27 mm thick. To provide 19 cm of water as an effectively infinite neutron reflector, the sphere was mounted in a cylinder of appropriate dimensions. The sphere was supported in the water reflector only by the top and bottom overflow and feed tubes, respectively. Two critical measurements are evaluated, one made in September 1954 and the other in September 1957. One measurement was part of a set of experiments to compare the properties of $^{235}U$ and $^{233}U$, and the other is from an unpublished set of experiments in which fissile solution was added to a critical system and the reactor period measured. Both configurations are considered as acceptable for use as criticality safety benchmark experiments (NEA [2001]). An additional measurement with a Unichrome liner in the sphere is not considered acceptable as a criticality safety benchmark. Results for HEU-SOL-THERM-011 are given in Table 5-46.

| Table 5-46 Results for HEU-SOL-THERM-011 | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$[a] |
| | $k_{eff}$ | $\sigma$ | AENCF | $K_{eff}$ | $\sigma$ | | | | |
| heust11c1 | 1.0000 | 0.0018 | 0.005 | 1.0086 | 0.0005 | 0.0086 | ± | 0.0038 | 523.0 |
| heust11c2 | 1.0000 | 0.0020 | 0.005 | 1.0044 | 0.0005 | 0.0044 | ± | 0.0041 | 533.0 |

NOTE: [a] The $H/^{235}U$ ratio are from page 3 and 4 of NEA (2001, HEU-SOL-THERM-011).

### 5.2.2.1.6 HEU-SOL-THERM-012

This water-reflected sphere is part of a series of experiments performed in the 1950's at the ORNL with highly enriched uranium (93.2 wt.% $^{235}U$). This measurement was made with a uranium oxyfluoride ($UO_2F_2$) solution in a 27.9-cm inner radius (91 liters) water-reflected sphere. The sphere was fabricated of 0.20-cm-thick 1100 aluminum and surrounded by an effectively infinite water reflector. This configuration is considered acceptable for use as a criticality safety benchmark experiment (NEA [2001]). Table 5-47 shows the MCNP result for this experiment.

Table 5-47 MCNP Results for HEU-SOL-THERM-012

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ ratio[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| Heust12c1 | 0.9999 | 0.0058 | 1.5237 | 0.9951 | 0.0004 | -0.0048 | ± | 0.0116 | 1270 |

NOTE: [a] The $H/^{235}U$ ratio is from page 3 of NEA (2001, HEU-SOL-THERM-012).

### 5.2.2.1.7 HEU-SOL-THERM-021

Thirty-four critical assemblies involving bare and hydrogenous-reflected arrays of cylinders of highly enriched uranyl nitrate solutions, reported in NEA (2001, HEU-SOL-THERM-021), are based on experiments performed at the Oak Ridge National Laboratory between May of 1961 and December of 1962, which was at that time operated by Union Carbide Corporation. The experiments consisted of cubic arrays of 8, 27, 64, or 125 cylindrical units with height-to-diameter ratios near unity. Each unit of the array held five liters of uranyl nitrate solution at concentrations of 415, 279, or 63.3 grams of uranium per liter. The uranium was enriched to 92.6 weight percent $^{235}U$. The arrays were either bare or reflected by paraffin and/or Plexiglas at thicknesses varying between 1.27 and 15.24 cm. These experiments were specifically designed and controlled to be amenable to calculational or broad empirical generalization. This evaluation reviews the 34 reported assemblies, one of which was subcritical by an unacceptably large margin and one of which was not based on an actual experiment as described. The remaining reported configurations were based on actual critical assemblies or were derived from small interpolations or extrapolation of critical assemblies. The assemblies were considered critical when a slight positive period was achieved. Therefore, 32 of the experimental configurations are deemed acceptable as benchmarks (NEA [2001]).

The containers used in the experiment were right circular cylinders of methyl methacrylate plastic (Plexiglas) having nominal dimensions of 20.32 cm outer diameter x 19.05 cm outer height with 0.635 cm thick walls. There were 120 units, which were pre-filled to a volume of 5.000 liters and are referred to as standard units. Additionally, there were five units, referred to as control units, which were of the same dimensions but were filled with solution by remote operation during the course of the experiment. The control units had nominal half-inch plastic fill tubes and vent tubes penetrating the bottom and top of the container, respectively. A unit cell consists of a fuel cylinder centered in a rectangular parallelepiped. The outer dimensions of this

cell are equal to the cylinder dimensions plus one-half of the reported STS distance in each direction. The term unit cell will be used in the remainder of the report. The unit cells were assembled into square-pitch arrays. Reflector material was placed at the outer bounds of these arrays. Results for HEU-SOL-THERM-021 are presented in Table 5-48.

Table 5-48 Results for HEU-SOL-THERM-021

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ ratio[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| hst021_01 | 0.9975 | 0.0054 | 0.0425 | 1.0050 | 0.0011 | 0.0074 | ± | 0.0110 | 58.78 |
| hst021_02 | 0.9975 | 0.0054 | 0.0403 | 1.0057 | 0.0011 | 0.0082 | ± | 0.0110 | 58.78 |
| hst021_03 | 0.9975 | 0.0054 | 0.0393 | 1.0099 | 0.0011 | 0.0124 | ± | 0.0110 | 58.78 |
| hst021_04 | 0.9975 | 0.0054 | 0.0370 | 1.0124 | 0.0009 | 0.0149 | ± | 0.0110 | 58.78 |
| hst021_05 | 0.9975 | 0.0054 | 0.0365 | 1.0118 | 0.0010 | 0.0142 | ± | 0.0110 | 58.78 |
| hst021_06 | 0.9975 | 0.0054 | 0.0365 | 1.0114 | 0.0010 | 0.0139 | ± | 0.0110 | 58.78 |
| hst021_07 | 0.9975 | 0.0054 | 0.0363 | 1.0088 | 0.0009 | 0.0113 | ± | 0.0110 | 58.78 |
| hst021_08 | 0.9975 | 0.0054 | 0.0362 | 1.0052 | 0.0010 | 0.0077 | ± | 0.0110 | 58.78 |
| hst021_09 | 0.9975 | 0.0054 | 0.0404 | 1.0086 | 0.0010 | 0.0111 | ± | 0.0110 | 58.78 |
| hst021_10 | 0.9975 | 0.0054 | 0.0399 | 1.0088 | 0.0010 | 0.0113 | ± | 0.0110 | 58.78 |
| hst021_11 | 0.9975 | 0.0054 | 0.0381 | 1.0084 | 0.0009 | 0.0109 | ± | 0.0110 | 58.78 |
| hst021_12 | 0.9975 | 0.0054 | 0.0369 | 1.0140 | 0.0010 | 0.0165 | ± | 0.0110 | 58.78 |
| hst021_13 | 0.9975 | 0.0054 | 0.0364 | 1.0125 | 0.0010 | 0.0150 | ± | 0.0110 | 58.78 |
| hst021_14 | 0.9975 | 0.0054 | 0.0363 | 1.0129 | 0.0009 | 0.0154 | ± | 0.0109 | 58.78 |
| hst021_15 | 0.9975 | 0.0054 | 0.0360 | 1.0133 | 0.0009 | 0.0158 | ± | 0.0110 | 58.78 |
| hst021_16 | 0.9978 | 0.0054 | 0.0437 | 0.9955 | 0.0010 | -0.0023 | ± | 0.0110 | 58.78 |
| hst021_17 | 0.9978 | 0.0054 | 0.0400 | 1.0141 | 0.0010 | 0.0163 | ± | 0.0110 | 58.78 |
| hst021_18 | 0.9978 | 0.0054 | 0.0403 | 1.0023 | 0.0010 | 0.0045 | ± | 0.0110 | 58.78 |
| hst021_19 | 0.9978 | 0.0054 | 0.0373 | 1.0145 | 0.0010 | 0.0167 | ± | 0.0110 | 58.78 |
| hst021_20 | 0.9978 | 0.0054 | 0.0373 | 1.0107 | 0.0009 | 0.0129 | ± | 0.0109 | 58.78 |
| hst021_21 | 0.9978 | 0.0054 | 0.0361 | 1.0133 | 0.0009 | 0.0155 | ± | 0.0110 | 58.78 |
| hst021_22 | 0.9978 | 0.0054 | 0.0368 | 1.0026 | 0.0010 | 0.0048 | ± | 0.0110 | 58.78 |
| hst021_23 | 0.9978 | 0.0054 | 0.0413 | 1.0015 | 0.0010 | 0.0037 | ± | 0.0110 | 58.78 |
| hst021_24 | 0.9978 | 0.0054 | 0.0407 | 1.0028 | 0.0010 | 0.0050 | ± | 0.0110 | 58.78 |
| hst021_25 | 0.9978 | 0.0054 | 0.0388 | 1.0060 | 0.0010 | 0.0082 | ± | 0.0110 | 58.78 |
| hst021_26 | 0.9978 | 0.0054 | 0.0432 | 0.9950 | 0.0010 | -0.0028 | ± | 0.0110 | 58.78 |
| hst021_27 | 0.9978 | 0.0054 | 0.0432 | 0.9995 | 0.0012 | -0.0017 | ± | 0.0111 | 58.78 |
| hst021_28 | 0.9975 | 0.0054 | 0.0283 | 1.0036 | 0.0010 | 0.0061 | ± | 0.0110 | 91.76 |
| hst021_29 | 0.9975 | 0.0054 | 0.0242 | 1.0081 | 0.0010 | 0.0106 | ± | 0.0110 | 91.76 |

| Table 5-48 Results for HEU-SOL-THERM-021 | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ ratio[a] |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| hst021_30 | 0.9978 | 0.0054 | 0.0390 | 1.0005 | 0.0010 | 0.0027 | ± | 0.0110 | 58.78 |
| hst021_31 | 0.9978 | 0.0054 | 0.0283 | 0.9976 | 0.0009 | -0.0002 | ± | 0.0110 | 91.76 |
| hst021_32 | 0.9978 | 0.0054 | 0.0067 | 1.0062 | 0.0009 | 0.0084 | ± | 0.0110 | 435.08 |

NOTE: [a] The $H/^{235}U$ ratio is calculated from the H and $^{235}U$ values provided in page 30 of HEU-SOL-THERM-021.

### 5.2.2.1.8 HEU-SOL-THERM-025

The 18 measurements included in this section are part of a series of experiments performed in 1987 at the Solution Physical Facility of the Institute of Physics and Power Engineering (IPPE), Obninsk, Russia with highly enriched (89.0 wt.% $^{235}U$) (NEA [2001, HEU-SOL-THERM-025]). Critical experiment measurements were made with uranyl nitrate solutions poisoned with gadolinium nitrate in a cylindrical tank with inner diameter 40 cm inserted in another cylindrical tank with inner diameter 59.4 cm with uranyl nitrate solutions without gadolinium. Natural gadolinium was used in the experiments. On the bottom and side thick water reflectors surrounded the cores. Eighteen configurations are considered to be acceptable for use as criticality safety benchmark experiments (NEA [2001]). Thirty-five similar configurations are reported in evaluations HEU-SOL-THERM-014, 015, 016, 017, 018, and 019.

The critical assembly consisted of three open-topped coaxial cylindrical tanks. The inner tank (1) was filled with an aqueous uranyl nitrate solution of uranyl nitrate ($UO_2(NO_3)_2$) with some excess of nitric acid ($HNO_3$) and also gadolinium nitrate ($Gd(NO_3)_3$) in most cases. The middle tank (2) was filled with distilled water in the first five experiments and with water solution of uranyl nitrate with some excess of nitric acid in the other experiments. The outer tank (3) was empty in Experiments 2, 3, 4, and 5, and was filled with distilled water in the other experiments. The cylindrical tanks were suspended in one another by the tank flanges and angle bars. In the first three experiments, the inner tank was suspended above the bottom of the middle tank by 15.1 cm; in all other experiments, the inner tank stood on the bottom of the middle tank. The middle tank always stood on the bottom of the outer tank. The average inside diameter of each tank was measured by filling the tank with a known volume of water and then measuring the water level. Results for HEU-SOL-THERM-025 are shown in Table 5-49.

| Table 5-49 Results for HEU-SOL-THERM-025 | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ ratio[a] |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| hst25_1 | 1.0002 | 0.0025 | 0.005 | 1.00384 | 0.00083 | 0.0036 | ± | 0.0053 | 555.8 |
| hst25_2 | 1.0007 | 0.0025 | 0.005 | 1.00515 | 0.00080 | 0.0045 | ± | 0.0052 | 555.8 |
| hst25_3 | 1.0002 | 0.0064 | 0.006 | 0.99868 | 0.00079 | -0.0015 | ± | 0.0129 | 563.5 |
| hst25_4 | 1.0003 | 0.0027 | 0.006 | 1.00609 | 0.00080 | 0.0058 | ± | 0.0056 | 534.6 |

Table 5-49 Results for HEU-SOL-THERM-025

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ ratio[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| hst25_5 | 1.0013 | 0.0030 | 0.008 | 1.00929 | 0.00091 | 0.0080 | ± | 0.0063 | 367.8 |
| hst25_6 | 1.0002 | 0.0067 | 0.006 | 1.00997 | 0.00075 | 0.0098 | ± | 0.0135 | 586.1 |
| hst25_7 | 1.0009 | 0.0073 | 0.007 | 1.01327 | 0.00078 | 0.0124 | ± | 0.0147 | 420.0 |
| hst25_8 | 1.0000 | 0.0067 | 0.007 | 1.01046 | 0.00077 | 0.0105 | ± | 0.0135 | 404.6 |
| hst25_9 | 1.0002 | 0.0065 | 0.009 | 1.00682 | 0.00081 | 0.0066 | ± | 0.0131 | 294.8 |
| hst25_10 | 1.0003 | 0.0043 | 0.012 | 1.01198 | 0.00079 | 0.0117 | ± | 0.0087 | 194.3 |
| hst25_11 | 1.0002 | 0.0045 | 0.012 | 1.01281 | 0.00080 | 0.0126 | ± | 0.0091 | 193.0 |
| hst25_12 | 1.0002 | 0.0045 | 0.015 | 1.01092 | 0.00078 | 0.0107 | ± | 0.0091 | 146.6 |
| hst25_13 | 1.0009 | 0.0047 | 0.015 | 1.01804 | 0.00074 | 0.0171 | ± | 0.0095 | 143.0 |
| hst25_14 | 1.0008 | 0.0053 | 0.020 | 1.01041 | 0.00081 | 0.0096 | ± | 0.0107 | 96.3 |
| hst25_15 | 1.0002 | 0.0058 | 0.019 | 1.00440 | 0.00080 | 0.0042 | ± | 0.0117 | 98.2 |
| hst25_16 | 1.0000 | 0.0049 | 0.028 | 1.01620 | 0.00079 | 0.0162 | ± | 0.0099 | 61.8 |
| hst25_17 | 1.0000 | 0.0055 | 0.027 | 1.00831 | 0.00077 | 0.0083 | ± | 0.0111 | 62.7 |
| hst25_18 | 1.0000 | 0.0061 | 0.026 | 1.00518 | 0.00077 | 0.0052 | ± | 0.0123 | 62.5 |

NOTE: [a] The $H/^{235}U$ ratio is calculated from the number densities provided in page 21 of HEU-SOL-THERM-025.

### 5.2.2.1.9 HEU-SOL-THERM-027

A large number of critical experiments with absorber elements of different types in uranium nitrate solution of different enrichments and concentrations were performed in 1961 - 1963 at the Solution Physical Facility of the Institute of Physics and Power Engineering (IPPE), Obninsk, Russia (NEA [2001, HEU-SOL-THERM-027]). The purpose of these experiments was to determine the effects of enrichment, concentration, geometry, neutron reflection, and type, diameter, number, and arrangement of absorber rods on the critical mass of light-water-moderated homogeneous uranyl nitrate solutions. The experiments included ones with a central boron carbide or cadmium rod, clusters of boron carbide rods, and triangular lattices of boron carbide rods in cylindrical tanks of different dimensions filled with solutions of uranyl nitrate. The nine experiments included in this evaluation were performed with uranium enriched to 89 wt.% $^{235}U$. Uranium nitrate solution was pumped into the core or inner tank, a stainless steel cylindrical tank with inner diameter 40.07 cm. In eight of the nine experiments, an absorber rod of various diameters with boron or cadmium was inserted in the center of the core tank. There was no outer reflector in these experiments. All nine configurations are considered to be acceptable for use as criticality safety benchmark experiments (NEA [2001]).

The critical assembly consisted of two open-topped coaxial cylindrical tanks. The inner tank was 40.07 cm inner diameter, 130 cm tall, with wall thickness 0.3 cm, and bottom thickness 0.6 cm. The outer tank was 198.4 cm inner diameter, 300 cm tall, with wall thickness 0.8 cm, and bottom

thickness 1.0 cm. In all experiments the inner tank was suspended in the outer tank with the tank flanges and angle bars. When mounted, the distance between the bottom (outer surface) of the inner tank and the bottom of the outer tank (inner surface) was 170 cm. The outer tank always stood on the floor. The inner tank (1) was partially filled with the aqueous solution of uranyl nitrate $UO_2(NO_3)_2$ with some excess of nitric acid ($HNO_3$). The outer tank (2) was empty in all cases. The average inside diameter of the inner tank was measured by filling the tank with a known volume of water and then measuring the water level. Several volumes were used to establish the inside diameter of the tank. The uncertainty of the core tank average inside diameter is ±0.2 cm. The absorber rod was inserted in the center of the inner tank in Cases 2 - 9. The absorber rod was a stainless steel tube 140 cm long, with wall thickness 0.1 cm, and bottom thickness 0.8 cm. The reported uncertainty of the tube outer diameter is ±1%. This tube was filled with two types of absorbers. In Cases 2 - 5 it was a powder of boron carbide. In Cases 6 - 9 it was a cadmium tube with outer diameter less than the inner diameter of the stainless steel tube by 0.1 cm, wall thickness 0.05 cm, and bottom thickness 0.05 cm, filled with water. The space between the cadmium tube and the stainless steel tube was empty. Results for HEU-SOL-THERM-027 are given in Table 5-50.

Table 5-50 Results for HEU-SOL-THERM-027

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ ratio[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| Hst27_1 | 1.0000 | 0.0046 | 0.015 | 1.00009 | 0.00099 | 0.0001 | ± | 0.0094 | 203.6 |
| Hst27_2 | 1.0000 | 0.0043 | 0.015 | 1.00204 | 0.00106 | 0.0020 | ± | 0.0089 | 203.6 |
| Hst27_3 | 1.0000 | 0.0037 | 0.015 | 1.00171 | 0.00101 | 0.0017 | ± | 0.0077 | 203.6 |
| Hst27_4 | 1.0000 | 0.0037 | 0.015 | 1.00394 | 0.00097 | 0.0039 | ± | 0.0077 | 203.6 |
| Hst27_5 | 1.0000 | 0.0044 | 0.015 | 1.00241 | 0.00100 | 0.0024 | ± | 0.0090 | 203.6 |
| Hst27_6 | 1.0000 | 0.0043 | 0.014 | 0.99517 | 0.00100 | -0.0048 | ± | 0.0088 | 203.6 |
| Hst27_7 | 1.0000 | 0.0038 | 0.015 | 1.00003 | 0.00102 | 0.0000 | ± | 0.0079 | 203.6 |
| Hst27_8 | 1.0000 | 0.0035 | 0.014 | 1.00260 | 0.00103 | 0.0026 | ± | 0.0073 | 203.6 |
| Hst27_9 | 1.0000 | 0.0039 | 0.015 | 1.00098 | 0.00103 | 0.0010 | ± | 0.0081 | 203.6 |

NOTE: [a] The $H/^{235}U$ ratio is calculated from the number densities provided in page 18 of HEU-SOL-THERM-027.

### 5.2.2.1.10 HEU-SOL-THERM-028

The 18 experiments reported in NEA (2001, HEU-SOL-THERM-028) were performed with uranium enriched to 89 wt.% $^{235}U$. Uranium nitrate solution with uranium concentration of 76 or 286 g/l was pumped into stainless steel cylindrical tanks of various diameters. In 9 of the 18 experiments, a boron carbide absorber rod was inserted in the center of the tank. The critical conditions were determined with and without the absorber rod for each case. There was a thick side and bottom water reflector in these experiments. All 18 configurations are considered to be acceptable for use as criticality safety benchmark experiments. Results for HEU-SOL-THERM-028 are shown in Table 5-51.

Table 5-51 Results for HEU-SOL-THERM-028

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ ratio[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| Hst28_1 | 1.0000 | 0.0023 | 0.007 | 1.00234 | 0.00086 | 0.0023 | ± | 0.0049 | 374.6 |
| Hst28_2 | 1.0000 | 0.0034 | 0.008 | 1.00156 | 0.00082 | 0.0016 | ± | 0.0070 | 374.6 |
| Hst28_3 | 1.0000 | 0.0026 | 0.007 | 1.00102 | 0.00085 | 0.0010 | ± | 0.0055 | 374.6 |
| Hst28_4 | 1.0000 | 0.0028 | 0.008 | 1.00236 | 0.00087 | 0.0024 | ± | 0.0059 | 374.6 |
| Hst28_5 | 1.0000 | 0.0031 | 0.008 | 0.99733 | 0.00092 | -0.0027 | ± | 0.0065 | 374.6 |
| Hst28_6 | 1.0000 | 0.0023 | 0.008 | 1.00141 | 0.00086 | 0.0014 | ± | 0.0049 | 374.6 |
| Hst28_7 | 1.0000 | 0.0038 | 0.007 | 1.00259 | 0.00090 | 0.0026 | ± | 0.0078 | 374.6 |
| Hst28_8 | 1.0000 | 0.0027 | 0.008 | 1.00111 | 0.00086 | 0.0011 | ± | 0.0057 | 374.6 |
| Hst28_9 | 1.0000 | 0.0049 | 0.026 | 1.00008 | 0.00095 | 0.0001 | ± | 0.0100 | 91.5 |
| Hst28_10 | 1.0000 | 0.0053 | 0.026 | 0.99653 | 0.00091 | -0.0035 | ± | 0.0108 | 91.5 |
| hst28_11 | 1.0000 | 0.0052 | 0.026 | 1.00188 | 0.00102 | 0.0019 | ± | 0.0106 | 91.5 |
| hst28_12 | 1.0000 | 0.0046 | 0.027 | 0.99633 | 0.00096 | -0.0037 | ± | 0.0094 | 91.5 |
| hst28_13 | 1.0000 | 0.0058 | 0.026 | 0.99826 | 0.00097 | -0.0017 | ± | 0.0118 | 91.5 |
| hst28_14 | 1.0000 | 0.0046 | 0.027 | 0.99993 | 0.00092 | -0.0001 | ± | 0.0094 | 91.5 |
| hst28_15 | 1.0000 | 0.0064 | 0.027 | 1.00725 | 0.00095 | 0.0072 | ± | 0.0129 | 91.5 |
| hst28_16 | 1.0000 | 0.0052 | 0.027 | 1.00292 | 0.00096 | 0.0029 | ± | 0.0106 | 91.5 |
| hst28_17 | 1.0000 | 0.0066 | 0.027 | 0.99727 | 0.00095 | -0.0027 | ± | 0.0133 | 91.5 |
| hst28_18 | 1.0000 | 0.0060 | 0.027 | 0.99794 | 0.00096 | -0.0021 | ± | 0.0122 | 91.5 |

NOTE: [a] The $H/^{235}U$ ratio is calculated from the number densities provided in page 20 of HEU-SOL-THERM-028.

### 5.2.2.1.11 HEU-SOL-THERM-029

The seven experiments reported in NEA (2001, HEU-SOL-THERM-029) were performed with uranium enriched to 89 wt.% $^{235}U$. Uranium nitrate solution with uranium concentration of 286 g/l was pumped into the core or inner tank, a stainless steel cylindrical tank with inner diameter 40.07 cm. One experiment was performed without absorber rods. In six of the seven experiments, a cluster of seven boron carbide absorber rods was inserted in the core tank. The cluster was an arrangement of one central absorber rod and six absorber rods placed around the central one. A different pitch between rods was used for each of the six experiments. There was a thick side and bottom water reflector in these experiments. The $H/^{235}U$ ratio is calculated from the number densities in page 22 of HEU-SOL-THERM-029 as 91.5. All seven configurations are considered to be acceptable for use as criticality safety benchmark experiments. Results for HEU-SOL-THERM-029 are presented in Table 5-52.

| Table 5-52 Results for HEU-SOL-THERM-029 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $K_{eff}$ | $\sigma$ | | | |
| Hst29_1 | 1.0000 | 0.0066 | 0.028 | 1.00095 | 0.00100 | 0.0010 | ± | 0.0134 |
| Hst29_2 | 1.0000 | 0.0058 | 0.027 | 1.00562 | 0.00097 | 0.0056 | ± | 0.0118 |
| Hst29_3 | 1.0000 | 0.0068 | 0.028 | 0.99616 | 0.00094 | -0.0038 | ± | 0.0137 |
| Hst29_4 | 1.0000 | 0.0074 | 0.028 | 0.99727 | 0.00097 | -0.0027 | ± | 0.0149 |
| Hst29_5 | 1.0000 | 0.0067 | 0.029 | 1.00111 | 0.00098 | 0.0011 | ± | 0.0135 |
| Hst29_6 | 1.0000 | 0.0065 | 0.029 | 1.00159 | 0.00097 | 0.0016 | ± | 0.0131 |
| Hst29_7 | 1.0000 | 0.0063 | 0.028 | 1.00199 | 0.00100 | 0.0020 | ± | 0.0128 |

### 5.2.2.1.12 HEU-SOL-THERM-030

The seven experiments reported in NEA (2001, HEU-SOL-THERM-030) were performed with uranium enriched to 89 wt.% $^{235}U$. Uranium nitrate solution with uranium concentration of 76 or 289 g/l was pumped into the core or inner tank, a stainless steel cylindrical tank with inner diameter 40.07 cm. Two experiments were performed without absorber rods. In five of the seven experiments, a cluster of three, four, or six boron carbide absorber rods was inserted in the center of the core tank. There was a thick side and bottom water reflector in these experiments. All seven configurations are considered to be acceptable for use as criticality safety benchmark experiments. Results for HEU-SOL-THERM-030 are given in Table 5-53.

| Table 5-53 Results for HEU-SOL-THERM-030 | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Experiment/Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ ratio[a] |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| Hst30_1 | 1.0000 | 0.0039 | 0.008 | 1.00043 | 0.00090 | 0.0004 | ± | 0.0080 | 374.6 |
| Hst30_2 | 1.0000 | 0.0032 | 0.008 | 1.00205 | 0.00087 | 0.0021 | ± | 0.0066 | 374.6 |
| Hst30_3 | 1.0000 | 0.0031 | 0.008 | 0.99961 | 0.00083 | -0.0004 | ± | 0.0064 | 374.6 |
| Hst30_4 | 1.0000 | 0.0064 | 0.028 | 1.00322 | 0.00099 | 0.0032 | ± | 0.0130 | 91.1 |
| Hst30_5 | 1.0000 | 0.0058 | 0.027 | 0.99887 | 0.00095 | -0.0011 | ± | 0.0118 | 91.1 |
| Hst30_6 | 1.0000 | 0.0059 | 0.028 | 1.00262 | 0.00100 | 0.0026 | ± | 0.0120 | 91.1 |
| Hst30_7 | 1.0000 | 0.0064 | 0.028 | 1.00028 | 0.00097 | 0.0003 | ± | 0.0129 | 91.1 |

NOTE: [a] The $H/^{235}U$ ratio is calculated from the number densities provided in page 22 and 23 of HEU-SOL-THERM-029.

### 5.2.2.1.13 HEU-SOL-THERM-031

The four experiments reported in NEA (2001, HEU-SOL-THERM-031) were performed with uranium enriched to 89 wt.% $^{235}U$. Uranium nitrate solution with uranium concentration of 289 g/l was pumped into the core or inner tank, a stainless steel cylindrical tank with inner diameter

40.07 cm. Eighteen or 36 boron carbide absorber rods were inserted in the center of the core tank. The rods were arranged in hexagonal lattices with pitches of 4.0 and 6.0 cm. There was a thick side and bottom water reflector in these experiments. The H/$^{235}$U ratio is calculated from the number densities in page 23 of HEU-SOL-THERM-031 as 91.1. The four configurations are considered to be acceptable for use as criticality safety benchmark experiments. Results for HEU-SOL-THERM-031 are given in Table 5-54.

Table 5-54 Results for HEU-SOL-THERM-031

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hst31_1 | 1.0000 | 0.0046 | 0.028 | 1.01650 | 0.00093 | 0.0165 | ± | 0.0094 |
| hst31_2 | 1.0000 | 0.0058 | 0.029 | 1.04116 | 0.00091 | 0.0412 | ± | 0.0117 |
| hst31_3 | 1.0000 | 0.0058 | 0.029 | 1.00109 | 0.00092 | 0.0011 | ± | 0.0117 |
| hst31_4 | 1.0000 | 0.0068 | 0.031 | 1.00745 | 0.00098 | 0.0074 | ± | 0.0137 |

### 5.2.2.1.14 HEU-SOL-THERM-032

In the 1950s a series of experiments was performed at the ORNL to investigate the critical conditions of unreflected homogeneous aqueous uranyl nitrate solutions, and to experimentally determine the thermal value of eta (h), the number of fission neutrons produced per thermal neutron absorbed, for the $^{235}$U and $^{233}$U isotopes. The measurements were made on both bare spherical and cylindrical configurations. The enrichment was 93.21 wt.% $^{235}$U. HEU-SOL-THERM-032 evaluates the 48.04-inch-diameter sphere of $^{235}$U solution, thereby completing the evaluation of all similar spherical configurations (HEU-SOL-THERM-013, U233-SOL-THERM-001 and U233-SOL-THERM-008). The H/$^{235}$U ratio in the solution was approximately 1835. The experiment was judged to be acceptable for use as a criticality-safety benchmark experiment (NEA [2001]). Table 5-55 shows results for HEU-SOL-THERM-031.

Table 5-55 Results for HEU-SOL-THERM-031

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| Heust032 | 1.0015 | 0.0026 | 0.0021 | 1.0001 | 0.0004 | -0.0014 | ± | 0.0053 |

### 5.2.2.1.15 HEU-SOL-THERM-033

A series of 12 experiments were performed at the Rocky Flats Critical Mass Laboratory beginning in August of 1980 (NEA [2001, HEU-SOL-THERM-033]). Highly enriched (93.22 wt.% $^{235}$U) uranyl nitrate with a H/$^{235}$U ratio of 68.1 was introduced into four nested stainless steel annular tanks. The tanks were inside a concrete enclosure, with various materials placed in the central region. These moderators and absorbers included boron-free concrete, borated concrete, borated plaster, cadmium, water, and borated water. Thirty-seven critical

configurations were attempted during the 12 experiments. Nine configurations were subcritical. Of the 28 critical configurations, 26 are judged acceptable benchmarks (NEA [2001]).

The experiments used four annular tanks in a nested configuration. Each tank consisted of two cylinders rolled from 6.4-mm-thick sheet metal. The cylinders were joined at the bottoms by 6.4-mm-thick steel to form an annular solution region thickness of design specification 38.1 ±0.8 mm. Each tank had 18 small-diameter spacing rods in evenly spaced sets of six to maintain the annular thickness. The two larger top rods, 19 mm in diameter, were used for lifting the tank. The other sixteen rods had 6.4-mm diameters. Using the top rod orientations, sets of the smaller-diameter rods were placed at one-third and two-thirds the height of the tank. All tanks had a maximum height of 2.134 m. Tank short heights were 2.097 m, 2.101 m, 2.105 m, and 2.110 m for tanks 1 through 4, respectively. The tank low points are aligned at the same elevation. The tank tops are horizontal and even with each other. Table 5-56 provides results for the detailed model.

Table 5-56 Results for HEU-SOL-THERM-033

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hst33d_02a | 1.0000 | 0.0111 | 0.036 | 1.00007 | 0.00128 | 0.0001 | ± | 0.0223 |
| hst33d_02b | 1.0000 | 0.0108 | 0.036 | 0.99792 | 0.00113 | -0.0021 | ± | 0.0217 |
| hst33d_02c | 1.0000 | 0.0065 | 0.036 | 0.99796 | 0.00119 | -0.0020 | ± | 0.0132 |
| hst33d_03a | 1.0000 | 0.0114 | 0.033 | 1.00634 | 0.00108 | 0.0063 | ± | 0.0229 |
| hst33d_03b | 1.0000 | 0.0111 | 0.034 | 1.00608 | 0.00115 | 0.0061 | ± | 0.0223 |
| hst33d_03c | 1.0000 | 0.0070 | 0.032 | 1.01079 | 0.00118 | 0.0108 | ± | 0.0142 |
| hst33d_04a | 1.0000 | 0.0114 | 0.035 | 1.00570 | 0.00109 | 0.0057 | ± | 0.0229 |
| hst33d_04b | 1.0000 | 0.0111 | 0.035 | 1.01160 | 0.00117 | 0.0116 | ± | 0.0223 |
| hst33d_05a | 1.0000 | 0.0111 | 0.035 | 1.01126 | 0.00114 | 0.0113 | ± | 0.0223 |
| hst33d_05b | 1.0000 | 0.0108 | 0.035 | 1.00608 | 0.00128 | 0.0061 | ± | 0.0218 |
| hst33d_06a | 1.0000 | 0.0111 | 0.035 | 1.00936 | 0.00112 | 0.0094 | ± | 0.0223 |
| hst33d_06b | 1.0000 | 0.0108 | 0.034 | 1.00915 | 0.00114 | 0.0091 | ± | 0.0217 |
| hst33d_07a | 1.0000 | 0.0111 | 0.035 | 1.00453 | 0.00107 | 0.0045 | ± | 0.0223 |
| hst33d_07b | 1.0000 | 0.0108 | 0.035 | 1.00406 | 0.00109 | 0.0041 | ± | 0.0217 |
| hst33d_08a | 1.0000 | 0.0111 | 0.034 | 1.00558 | 0.00113 | 0.0056 | ± | 0.0223 |
| hst33d_08b | 1.0000 | 0.0108 | 0.035 | 1.00213 | 0.00111 | 0.0021 | ± | 0.0217 |
| hst33d_09a | 1.0000 | 0.0111 | 0.036 | 1.00228 | 0.00115 | 0.0023 | ± | 0.0223 |
| hst33d_09b | 1.0000 | 0.0108 | 0.035 | 0.99359 | 0.00113 | -0.0064 | ± | 0.0217 |
| hst33d_09c | 1.0000 | 0.0104 | 0.036 | 0.99619 | 0.00116 | -0.0038 | ± | 0.0209 |
| hst33d_10a | 1.0000 | 0.0114 | 0.034 | 1.00267 | 0.00113 | 0.0027 | ± | 0.0229 |

| Table 5-56 Results for HEU-SOL-THERM-033 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| hst33d_10c | 1.0000 | 0.0070 | 0.032 | 1.00333 | 0.00103 | 0.0033 | ± | 0.0142 |
| hst33d_10d | 1.0000 | 0.0104 | 0.033 | 0.99286 | 0.00111 | -0.0071 | ± | 0.0209 |
| hst33d_11a | 1.0000 | 0.0111 | 0.035 | 1.00669 | 0.00110 | 0.0067 | ± | 0.0223 |
| hst33d_11b | 1.0000 | 0.0108 | 0.034 | 1.00176 | 0.00097 | 0.0018 | ± | 0.0217 |
| hst33d_12a | 1.0000 | 0.0111 | 0.036 | 1.00386 | 0.00112 | 0.0039 | ± | 0.0223 |
| hst33d_12b | 1.0000 | 0.0108 | 0.035 | 1.00165 | 0.00107 | 0.0016 | ± | 0.0217 |

### 5.2.2.1.16 HEU-SOL-THERM-035

The nine experiments reported in NEA (2001, HEU-SOL-THERM-035) were performed with uranium enriched to 89 wt.% $^{235}U$. Uranium nitrate solution with uranium concentration of 37.51 g/l, 74.87 g/l, or 152.3 g/l was pumped into the core or inner tank, a stainless steel cylindrical tank with inner diameter 110 cm. Three experiments were performed without absorber rods. In six experiments different numbers of boron carbide absorber rods were inserted in the core tank. The absorber rods were arranged in a hexagonal lattice with different pitches. There was a thick side and bottom water reflector in these experiments. All nine configurations are considered to be acceptable for use as criticality safety benchmark experiments. Experiments performed using the same tank and absorber rods with uranium enrichments of 5.64wt.% and 10wt.% are reported in evaluations LEU-SOL-THERM-005 and LEU-SOL-THERM-006. Table 5-57 provides results for HEU-SOL-THERM-035.

| Table 5-57 Results for HEU-SOL-THERM-035 | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$ ratio[a] |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| hst35_1 | 1.0000 | 0.0031 | 0.004 | 1.00490 | 0.00073 | 0.0049 | ± | 0.0064 | 766.6 |
| hst35_2 | 1.0000 | 0.0032 | 0.004 | 1.00859 | 0.00068 | 0.0086 | ± | 0.0065 | 766.6 |
| hst35_3 | 1.0000 | 0.0030 | 0.004 | 1.00832 | 0.00072 | 0.0083 | ± | 0.0062 | 766.6 |
| hst35_4 | 1.0000 | 0.0030 | 0.004 | 1.00757 | 0.00070 | 0.0076 | ± | 0.0062 | 766.6 |
| hst35_5 | 1.0000 | 0.0033 | 0.008 | 1.00565 | 0.00087 | 0.0056 | ± | 0.0068 | 378.6 |
| hst35_6 | 1.0000 | 0.0029 | 0.008 | 1.00828 | 0.00089 | 0.0083 | ± | 0.0061 | 378.6 |
| hst35_7 | 1.0000 | 0.0035 | 0.015 | 1.00811 | 0.00095 | 0.0081 | ± | 0.0073 | 180.7 |
| hst35_8 | 1.0000 | 0.0038 | 0.016 | 1.00480 | 0.00095 | 0.0048 | ± | 0.0078 | 180.7 |
| hst35_9 | 1.0000 | 0.0041 | 0.016 | 1.00486 | 0.00095 | 0.0049 | ± | 0.0084 | 180.7 |

NOTE: [a] The $H/^{235}U$ ratio is calculated from the number densities provided in page 34 and 35 of HEU-SOL-THERM-035.

### 5.2.2.1.17 HEU-SOL-THERM-036

The experiments reported in NEA (2001, HEU-SOL-THERM-036) were performed with uranium enriched to 89 wt.% $^{235}U$. Uranium nitrate solution with uranium concentration of 92.7 g/l was pumped into a stainless steel square tank with inner dimensions 72.6x2.6 cm. One experiment was performed without absorber rods, and three experiments were performed with different numbers of absorber rods arranged in a square uniform lattice with different pitches, inserted in the tank. There was no reflector in these experiments. The H/$^{235}U$ ratio is calculated as 302.5 from the number densities in page 23 of HEU-SOL-THERM-036. All four configurations are considered to be acceptable for use as criticality safety benchmark experiments (NEA [2001]). Table 5-58 shows results for HEU-SOL-THERM-036.

Table 5-58 Results for HEU-SOL-THERM-036

| Experiment /Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hst36_1 | 0.9974 | 0.0045 | 0.010 | 0.9989 | 0.0009 | 0.0015 | ± | 0.0092 |
| hst36_2 | 0.9979 | 0.0039 | 0.011 | 1.0044 | 0.0010 | 0.0065 | ± | 0.0080 |
| hst36_3 | 0.9993 | 0.0044 | 0.012 | 0.9995 | 0.0009 | 0.0002 | ± | 0.0090 |
| hst36_4 | 1.0000 | 0.0062 | 0.012 | 1.0008 | 0.0009 | 0.0008 | ± | 0.0125 |

### 5.2.2.1.18 HEU-SOL-THERM-037

The nine experiments reported in NEA (2001) were performed with uranium enriched to 89 wt.% $^{235}U$. Uranium nitrate solution with uranium concentration of 41.9 g/l, 61.4 g/l, or 83.0 g/l was pumped into the core or inner tank, a stainless steel cylindrical tank with inner diameter 160 cm. Three experiments were performed without absorber rods. In six experiments different numbers of boron carbide absorber rods were inserted in the core tank. The absorber rods were arranged in a hexagonal lattice with pitch 6.0 cm. There was a thin side and bottom water reflector in these experiments. All nine configurations are considered to be acceptable for use as criticality safety benchmark experiments. Table 5-59 presents results for HEU-SOL-THERM-037.

Table 5-59 Results for HEU-SOL-THERM-037

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/$^{235}U$ ratio[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| hst37_1 | 0.9980 | 0.0034 | 0.005 | 1.0129 | 0.0011 | 0.0149 | ± | 0.0071 | 684.5 |
| hst37_2 | 0.9990 | 0.0035 | 0.005 | 1.0048 | 0.0011 | 0.0058 | ± | 0.0073 | 684.5 |
| hst37_3 | 0.9970 | 0.0042 | 0.007 | 1.0096 | 0.0012 | 0.0126 | ± | 0.0087 | 463.1 |
| hst37_4 | 0.9980 | 0.0035 | 0.007 | 1.0141 | 0.0012 | 0.0161 | ± | 0.0074 | 463.1 |
| hst37_5 | 0.9980 | 0.0042 | 0.007 | 1.0089 | 0.0011 | 0.0109 | ± | 0.0087 | 463.1 |

| | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| hst37_6 | 0.9960 | 0.0051 | 0.009 | 1.0172 | 0.0013 | 0.0212 | ± | 0.0105 | 339.7 |
| hst37_7 | 0.9980 | 0.0034 | 0.009 | 1.0162 | 0.0012 | 0.0182 | ± | 0.0072 | 339.7 |
| hst37_8 | 0.9980 | 0.0040 | 0.009 | 1.0125 | 0.0013 | 0.0145 | ± | 0.0084 | 339.7 |
| hst37_9 | 0.9980 | 0.0047 | 0.009 | 1.0080 | 0.0012 | 0.0100 | ± | 0.0097 | 339.7 |

NOTE: [a] The H/$^{235}$U ratio is calculated from the number densities provided in page 24 and 25 of HEU-SOL-THERM-037.

### 5.2.2.1.19 HEU-SOL-THERM-043

In the late 1950's and early 1960's a series of critical experiments was done at the ORNL with highly enriched (93.2 wt.% $^{235}$U) uranium solutions in simple geometries. Three measurements in large unreflected spheres are the subject of this evaluation. These spheres are characterized as 17-liter, 91-liter and 174-liter spheres. Similar water-reflected critical experiments with oxyfluoride solutions are documented in HEU-SOL-THERM-011 (17 liter) and HEU-SOL-THERM-012 (91 liter). Unreflected measurements with uranium nitrate solutions are documented in HEU-SOL-THERM-013 (174 liter). Three experiments are considered acceptable as criticality benchmarks (NEA [2001]). Table 5-60 shows results for HEU-SOL-THERM-043.

Table 5-60 Results for HEU-SOL-THERM-043

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/$^{235}$U ratio[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| Heust43c1 | 0.9986 | 0.0017 | 0.014 | 0.9995 | 0.0007 | 0.0009 | ± | 0.0037 | 203.5 |
| Heust43c2 | 0.9995 | 0.0041 | 0.003 | 1.0082 | 0.0004 | 0.0087 | ± | 0.0082 | 1111.0 |
| Heust43c3 | 0.9990 | 0.0044 | 0.003 | 1.0033 | 0.0004 | 0.0043 | ± | 0.0088 | 1392.0 |

NOTE: [a] H/$^{235}$U ratio is from page 14 of HEU-SOL-THERM-043.

### 5.2.2.1.20 HEU-SOL-THERM-044

Sixty-two critical and critical-approach experiments were performed at the Rocky Flats Critical Mass Laboratory in 1983 and 1984 to study the effects of periodically distributed absorber rods containing boron, cadmium, gadolinium, and chlorine on the critical height of uranyl nitrate solution (93.58 wt.% $^{235}$U) in a concrete-reflected cylindrical tank. The uranium concentration was fixed at 362.5 grams of uranium per liter for all experiments. Critical heights were measured for eight different absorber materials with several different lattice patterns and grid spacings. The critical height for each experiment was determined from a linear interpolation between slightly supercritical and slightly subcritical states. Forty-five experiments achieved criticality. Thirty-one of the experiments that achieved criticality were accepted for this evaluation. Table 5-61 shows results for 16 cases that have been modeled.

The unique feature of the experiments covered here is the presence of absorber rods in the experimental tank. These experiments are most suitable for the evaluation or validation of

properties of the neutron absorbers in the absorber rods. There are more suitable evaluations available if the principal interest relates to the properties of uranyl nitrate solution.

The stainless steel tank used in these experiments was an open-top right circular cylinder with an inner diameter of 106.6 cm and an interior height of 151.0 cm. The bottom of the tank was 0.95 cm thick; the side of the tank was 0.64 cm thick. The top of the tank was stiffened by a stainless steel flange. The radial width of this flange was 5.0 cm, the vertical surface welded to the tank was 3.8 cm high, and the flange was 0.6 cm thick. The tank was supported by three stainless steel legs. The nominal length of each leg was 25.0 cm, although the legs were adjustable to level the tank.

For each experiment the tank held several hundred stainless steel tubes, and each tube generally contained a poison absorber rod. The steel tubes prevented contamination of the absorber rods by fissile solution in the tank. Each tube was open at the top for insertion and removal of absorber rods and seal-welded to a stainless steel disk at the bottom to exclude solution. The dimensions of a sample of these tubes were measured. The average inner and outer diameters were 2.609 cm and 2.845 cm, respectively. The tubes measured between 157.5 cm and 157.6 cm in length; the thickness of the bottom disk was 0.16 cm. The tubes were held in place by two stainless steel spacer plates. The lower spacer plate was supported about 2.5 cm above the tank bottom by four machine screws; the upper spacer plate rested on several small plastic blocks about 2.5 cm above the top of the tank. Each steel spacer plate was nominally 0.152 cm thick; the outer diameters of the lower and upper spacer plates were 106.0 cm and 111.8 cm respectively. Each spacer plate had a set of 2.9-cm square holes punched in a regular pattern to establish a lattice for the absorber rods. Several sets of spacer plates with different grid patterns were used during these experiments.

The tank was surrounded on four sides by a vertical concrete reflector, but there was no such reflector above or below the tank. Each side of the concrete reflector consisted of a lower segment and an upper segment. The four lower segments, cast in 1976 for an earlier program, were 122 cm high. The four upper segments, cast specifically for these experiments, were 30.4 cm high, so the total height of the concrete reflector was 152.4 cm. The concrete reflector was not aligned precisely with either the top or bottom of the tank.

Several different types of absorber rods were used during the experimental program. PVC rods were solid rods of chlorinated polyvinyl chloride. The PVC rods were machined to an outer diameter of 2.54 ± 0.008 cm and cut to a length of 152.4 cm. The borosilicate glass rods were ordinary Pyrex tubes (Pyrex is a registered trademark of Corning, Incorporated of Corning, New York) purchased from a commercial scientific supply house. The dimensions and masses of a sample of the rods were measured. The rod length was 121.9 ± 0.25 cm. The average outer diameter was 2.557 ± 0.014 cm. The average inner diameter was 1.788 ± 0.014 cm, although the inner diameter measurements were confined to the ends of the rods. The average weight of a rod was 702.6 ± 1.1 g. Boroflex rods were produced by wrapping Boroflex rubber (Boroflex is a registered trademark of Bisco Products which is now owned by Dow Corning) around mild steel tubes. The rubber was then glued to the tubes. After the rubber was glued to a tube, the tube was encased in a thin overlay of heat-shrinkable PVC tubing. The mild steel tubes were 157.4 cm

long based on one measurement. Measurement of a sample Boroflex rod determined that the correct outer diameter is 2.486 cm. The Boroflex rubber was manufactured 0.1024 ± 0.002 cm thick. The nominal thickness of the mild steel tubing was 0.124 cm. The thickness of the PVC overlay is difficult to measure, and the manufacturer cannot locate the relevant records.

Currently this tubing is available in 0.010, 0.015, and 0.020 cm thicknesses. The manufacturer suggests that the 0.015 cm stock is the most likely candidate, and this thickness was assumed for the calculational models. Based on these thicknesses and an outer rod diameter of 2.486 cm, the inner radius of the mild steel tube is 1.0016 cm. The outer radii of the mild steel tube, Boroflex rubber sheet, and PVC wrap are then 1.1256 cm, 1.2280 cm, and 1.2430 cm respectively. The Boroflex rubber wrap did not fit perfectly around the inner steel tube, so a small gap was present in each Boroflex rod where the edges of the rubber failed to meet. This gap was measured for three finished rods that were selected at random as being representative of all Boroflex rods. Table 5-61 shows calculated cases for this benchmark .

Table 5-61 Results for HEU-SOL-THERM-044

| Experiment/ Case[a] | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hst4410 | 0.9944 | 0.0077 | 0.039 | 0.9909 | 0.0018 | -0.0035 | ± | 0.0158 |
| hst4411 | 0.9944 | 0.0078 | 0.041 | 0.9847 | 0.0020 | -0.0097 | ± | 0.0161 |
| hst4412 | 0.9944 | 0.0078 | 0.040 | 0.9872 | 0.0017 | -0.0072 | ± | 0.0160 |
| hst4413 | 0.9964 | 0.0067 | 0.042 | 1.0000 | 0.0018 | 0.0036 | ± | 0.0139 |
| hst4416 | 0.9974 | 0.0062 | 0.043 | 1.0178 | 0.0018 | 0.0204 | ± | 0.0129 |
| hst4417 | 0.9964 | 0.0057 | 0.044 | 0.9987 | 0.0017 | 0.0023 | ± | 0.0119 |
| hst4419 | 0.9974 | 0.0063 | 0.045 | 1.0079 | 0.0018 | 0.0105 | ± | 0.0131 |
| hst4444 | 0.9984 | 0.0057 | 0.045 | 1.0004 | 0.0017 | 0.0020 | ± | 0.0119 |
| hst4449 | 0.9964 | 0.0047 | 0.034 | 1.0116 | 0.0017 | 0.0152 | ± | 0.0100 |
| hst4450 | 0.9946 | 0.0047 | 0.038 | 0.9881 | 0.0018 | -0.0065 | ± | 0.0101 |
| hst4451 | 0.9984 | 0.0057 | 0.046 | 1.0047 | 0.0017 | 0.0063 | ± | 0.0119 |
| hst4453 | 0.9984 | 0.0064 | 0.047 | 1.0189 | 0.0018 | 0.0205 | ± | 0.0133 |
| hst4454 | 0.9984 | 0.0065 | 0.046 | 1.0142 | 0.0015 | 0.0158 | ± | 0.0133 |
| hst4455 | 0.9984 | 0.0065 | 0.046 | 1.0196 | 0.0017 | 0.0212 | ± | 0.0134 |
| hst447 | 0.9944 | 0.0097 | 0.037 | 0.9948 | 0.0018 | 0.0004 | ± | 0.0197 |
| hst448 | 0.9946 | 0.0083 | 0.042 | 0.9955 | 0.0021 | 0.0009 | ± | 0.0171 |

NOTE: [a] The digits after hst44 refer to the experiment number given in Table 21 of HEU-SOL-THERM-044.

### 5.2.2.2 HEU Metal Fuel Experiments

#### 5.2.2.2.1 HEU-MET-FAST-002

The experiments included in this evaluation are six critical assemblies run on the Topsy machine in 1950. The six critical assemblies include a pseudosphere, a pseudocylinder, and four parallelepipeds. All are oralloy (93.5 wt.% $^{235}U$ enriched uranium), three to five inches wide, and reflected by eight inches of tuballoy (natural uranium). Although a number of experiments are described, the six tuballoy-reflected (tuballoy-tamper) assemblies chosen for analysis here seem to be the most carefully performed. They were taken to critical and were used to demonstrate the effect of geometry on critical mass. All six of these experiments are considered to be acceptable for use as criticality safety benchmark experiments (NEA [2001]).

The critical configurations include a 4.5-inch-diameter pseudosphere, a 4-inch-diameter x 4.5-inch-long pseudocylinder, a 4 x 4 x 3.5 inch parallelepiped, a 5 x 5 x 2.5 inch parallelepiped, a 3 x 3 x 7.5 inch parallelepiped, and a 3 x 3.5 x 6 inch parallelepiped. All are oralloy reflected by 8 inches of tuballoy (natural uranium). The oralloy pseudosphere, pseudocylinder, and parallelepipeds were built using ½-inch cubes. The reflectors were built up using 1-inch tuballoy cubes. The first two configurations are called pseudosphere and pseudocylinder because they were built with these small cubes; therefore, the external surfaces could not be perfectly rounded.

The reported critical mass for the pseudosphere was 17.4 kg. The reported critical masses and number of oralloy cubes for the other five configurations are given in Table 5-62. Results for this experiment are given in Table 5-63.

Table 5-62 Critical Mass / Density Comparisons for HEU-MET-FAST-002

| Shape of Configuration | Number of 0.5 Inch Cubes | Critical Mass (kg) |
|---|---|---|
| 4 inch diameter x 4.5 inch cylinder | 468 | 18.0 |
| 4 x 4 x 3.5 inch parallelepiped | 468 | 18.0 |
| 5 x 5 x 2.5 inch parallelepiped | 506 | 19.4 |
| 3 x 3 x 7.5 inch parallelepiped | 545 | 20.9 |
| 3 x 3.5 x 6 inch parallelepiped | 504 | 19.35 |

Table 5-63 Results for HEU-MET-FAST-002

| Experiment/Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumf2-1 | 1.0000 | 0.0030 | 1.599 | 1.0024 | 0.0011 | 0.0024 | ± | 0.0064 |
| heumf2-2 | 1.0000 | 0.0030 | 1.597 | 1.0034 | 0.0011 | 0.0034 | ± | 0.0064 |

| Table 5-63 Results for HEU-MET-FAST-002 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumf2-3 | 1.0000 | 0.0030 | 1.592 | 1.0020 | 0.0010 | 0.0020 | ± | 0.0063 |
| heumf2-4 | 1.0000 | 0.0030 | 1.583 | 1.0041 | 0.0010 | 0.0041 | ± | 0.0063 |
| heumf2-5 | 1.0000 | 0.0030 | 1.579 | 1.0039 | 0.0010 | 0.0038 | ± | 0.0064 |
| heumf2-6 | 1.0000 | 0.0030 | 1.583 | 1.0028 | 0.0011 | 0.0028 | ± | 0.0064 |

#### 5.2.2.2.2 HEU-MET-FAST-004

The objective of this experiment, conducted at the Pajarito Site at LANL in December 1976, was to provide a very accurate estimate of the critical mass of a sphere of highly enriched uranium (95.96 wt.% $^{235}U$) ) with an effectively infinite water reflector. The sphere was placed on top of a Plexiglas stand inside an aluminum tank containing water. The water level was raised until the configuration was slightly supercritical, and the excess reactivity was measured as a function of the height of the water above the sphere. The experimental results then were extrapolated to the condition of the same sphere in an infinite sea of water based on calculations. An outstanding feature of this experiment is that the uncertainties associated with it are quite small, and the results are remarkably insensitive to those uncertainties. This experiment is considered to be acceptable for use as a benchmark experiment for criticality safety (NEA [2001]).

Two hemispheres of HEU were joined together using a small pin of HEU to form a very nearly uniform sphere with a radius of 6.5537 cm. The sphere was placed on a three-legged Plexiglas stand inside an aluminum tank. The tank was between 60 and 70 cm high and had a diameter of 60 cm. In addition to the stand and the sphere, the tank contained two horizontal dry-well ports, approximately 10 cm below the sphere, to accommodate BF neutron detectors. A Pu-Be source was placed near the sphere. Water was slowly added to the tank until the configuration became slightly supercritical (but still substantially less than prompt supercritical). The excess reactivity then was measured as a function of the water height above the sphere. Table 5-64 lists the results for HEU-MET-FAST-004.

| Table 5-64 Results for HEU-MET-FAST-004 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumf4c1 | 1.0020 | 0.0007 | 1.170 | 0.9957 | 0.0007 | -0.0063 | ± | 0.0020 |
| heumf4c2 | 0.9985 | 0.0008 | 1.166 | 1.0017 | 0.0007 | 0.0032 | ± | 0.0022 |

#### 5.2.2.2.3 HEU-MET-FAST-027

Criticality measurements of a lead-reflected $^{235}U$(90 wt.%) assembly were conducted by VNIIEF in 1962 at its criticality test facility (CTF). The core pieces were manufactured also in 1962.

These experiments were computationally and theoretically re-evaluated in 1997. These efforts resulted in an acceptable spherical benchmark model of the critical assembly (NEA [2001]).

The lead-reflected core of $^{235}$U(90 wt.%) had a central cavity of 1.0-cm radius and comprised nine spherical layers of fissile material that extended to an outermost radius of 8.35 cm. The total value of the uncertainty in core radius, averaged over nine layers, was 0.03%. The lead reflector consisted of one spherical layer that extended to an outermost radius of 11.6 cm. The total value of the uncertainty in reflector radius was 0.15%. Each spherical core and reflector layer consisted of two hemispherical pieces. All hemispherical parts of uranium from layers two through nine had cylindrical threaded pole holes of 3.6-cm diameter. During measurements, these pole holes were filled with threaded plugs of core material. Additionally, the uranium layers had side cylindrical holes 2.2 cm in diameter. These side holes of the core layers also were plugged with suitable plugs of core material. Table 5-65 shows result for this experiment.

Table 5-65 Results for HEU-MET-FAST-027

| Experiment /Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| Heumf27c1 | 1.0000 | 0.0025 | 1.512 | 1.0098 | 0.0006 | -0.0098 | ± | 0.0051 |

### 5.2.2.2.4 HEU-MET-FAST-029

Criticality measurements of depleted-uranium-reflected $^{235}$U(90 wt.%) were conducted by VNIIEF in 1989 at its criticality test facility (CTF) (NEA [2001, HEU-MET-FAST-029]). These experiments were re-evaluated in 1996. These efforts resulted in a spherical benchmark critical-assembly model. A single experiment is evaluated and is considered acceptable for use as a criticality safety benchmark experiment.

The depleted-uranium-reflected assembly of $^{235}$U (90 wt.%) had a central cavity of 2.0-cm radius and incorporated seven spherical layers of fissile material that extended to an outermost radius of 7.55 cm. The total value of the uncertainty in core radius, averaged over seven layers, was 0.04%. The depleted-uranium reflector incorporated 4 spherical layers that extended to an outermost radius of 12.25 cm. The total value of the uncertainty in reflector radius, averaged over 4 layers, was 0.04%. Each spherical core and reflector layer consisted of two hemispherical pieces. All hemispherical parts of the core had cylindrical pole holes of 2.2-cm diameter. During measurements, these pole holes were filled with plugs of core material. All hemispherical parts of layer one and layer two of the reflector had threaded cylindrical pole holes of 3.6-cm diameter. During the measurements, threaded depleted-uranium plugs were screwed into these pole holes. Upper hemispherical pieces of layer three and layer four of the reflector had cylindrical pole holes of 6.2 cm diameter, and lower hemispherical pieces of the same layers had cylindrical pole holes of 16.0 cm diameter. During measurements, all these pole holes were filled with plugs of reflector material. The assembly was composed of two separate units. The upper unit, comprising four hemispherical pieces of depleted uranium, which were within inner and outer radius limits of 7.55 cm and 12.25 cm, was laid on a steel diaphragm 0.2 cm thick. The

lower (movable) unit incorporated the remainder of the parts. Table 5-65 shows results for this experiment.

Table 5-65 Results for HEU-MET-FAST-029

| Experiment /Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumf29c1 | 1.0000 | 0.0020 | 1.547 | 1.0038 | 0.0006 | 0.0038 | ± | 0.0042 |

### 5.2.2.2.5 HEU-MET-FAST-031

Measurements of critical parameters of an assembly of $^{235}U$ (90 wt.%) with central area of polyethylene and with a 17.45-cm polyethylene reflector were conducted by VNIIEF in 1962 at its criticality test facility (CTF) (NEA [2001, HEU-MET-FAST-031]). The core pieces were manufactured also in 1962. These experiments were computationally and theoretically reevaluated in 1997. These efforts resulted in an acceptable benchmark spherical critical-assembly model.

Although categorized as "fast," a significant fraction of the fissions in the assembly occur in the intermediate energy range.

The assembly of $^{235}U$ (90 wt.%) had a central cavity containing polyethylene and a 17.45-cm-thick polyethylene reflector on the CTF. The assembly core had a central cavity of 4.02-cm radius and incorporated 5 spherical layers of fissile material that extended to an outer radius of 7.55 cm with 27.9 kg of U. The total value of the uncertainty in core radius, averaged over 5 layers, was 0.03%. In the central cavity was placed a one-layer spherical shell of polyethylene, with internal and external radii of 1.4 and 4.02 cm, respectively. The polyethylene reflector incorporated 2 spherical layers with layer outer radii of 11.0 and 25.0 cm. The total value of the uncertainty in radii of the polyethylene pieces was 0.25%. Each spherical core and polyethylene layer consisted of two hemispherical pieces. All hemispherical parts of uranium had cylindrical threaded pole holes of 3.6-cm diameter. During measurements almost all of these pole holes were filled with threaded plugs of core material. More specifically, the upper pole plug in each of layers one and five and the lower pole plug of layer four were removed in the standard test assembly. Additionally, the uranium layers had side cylindrical holes 2.2 cm in diameter. These side holes of the core layers were all plugged with suitable plugs of core material. All upper hemispherical parts of the polyethylene reflector layers had threaded cylindrical pole holes of 3.6-cm diameter. During the measurements, threaded polyethylene plugs were screwed into these pole holes. The assembly was composed of two separate units. The upper unit, comprising one hemispherical piece of polyethylene, with inner and outer radii 11.0 and 25.0 cm, was laid on a steel diaphragm 0.3 cm thick. The lower (movable) unit incorporated the remainder of the parts. Table 5-66 shows results for HEU-MET-FAST-031.

| Table 5-66 Results for HEU-MET-FAST-031 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumf31c1 | 1.0000 | 0.0059 | 0.966 | 0.9964 | 0.0007 | -0.0036 | ± | 0.0119 |

### 5.2.2.2.6 HEU-MET-MIXED-002

Measurements of critical parameters of an assembly of $^{235}U$(90 wt.%) with central area of polyethylene and with a 12.85-cm polyethylene reflector were conducted by VNIIEF in 1962 at its criticality test facility (CTF). The core pieces were manufactured also in 1962. These experiments were computationally and theoretically reevaluated in 1997. These efforts resulted in an acceptable benchmark spherical critical-assembly model.

An assembly of $^{235}U$(90 wt.%) with a central cavity containing polyethylene and with a 12.85-cm-thick polyethylene reflector was configured on the CTF. The assembly core incorporated three spherical layers of fissile material that extended to an outer radius of 9.15 cm with 35.3 kg of U. The total value of the uncertainty in core radius, averaged over three layers, was 0.03%. In the central cavity was placed a spherical layer of polyethylene, with internal and external radii of 1.0 and 6.75 cm, respectively. The polyethylene reflector incorporated two spherical layers that extended outward to a radius of 22.0 cm. The total value of the uncertainty in radii of the polyethylene pieces was 0.25%. Table 5-67 shows results for HEU-MET-MIXED-002.

| Table 5-67 Results for HEU-MET-MIXED-002 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumm2c1 | 1.0000 | 0.0037 | 0.788 | 1.0032 | 0.0007 | 0.0032 | ± | 0.0075 |

### 5.2.2.2.7 HEU-MET-MIXED-003

Measurements of critical parameters of an assembly of $^{235}U$(90 wt.%) with central area of polyethylene and with a 15.85-cm polyethylene reflector were conducted by VNIIEF in 1962 at its criticality test facility (CTF). The core pieces were manufactured also in 1962. These experiments were computationally and theoretically reevaluated in 1997. These efforts resulted in an acceptable benchmark spherical critical-assembly mode (NEA [2001]).

An assembly of $^{235}U$(90 wt.%) with a central cavity containing polyethylene and with a 15.85-cm-thick polyethylene reflector was configured on the CTF. The assembly core incorporated three spherical layers of fissile material that extended to an outer radius of 9.15 cm with 35.3 kg of U. The total value of the uncertainty in core radius, averaged over three layers, was 0.03%. In the central cavity was placed the layer of polyethylene, with internal and external radii of 1.4 and 6.75 cm, respectively. The polyethylene reflector incorporated two spherical layers that

extended outward to a radius of 25.0 cm. The total value of the uncertainty in radii of the polyethylene pieces was 0.25%. Table 5-68 shows the result for this experiment.

Table 5-68 Results for HEU-MET-MIXED-003

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumm3c1 | 1.0000 | 0.0038 | 0.784 | 1.0039 | 0.0008 | 0.0039 | ± | 0.0078 |

### 5.2.2.2.8 HEU-MET-FAST-032

In the early 1950s a series of experiments was performed at the Los Alamos critical assembly facility to determine the critical mass of highly enriched uranium spheres surrounded by thin natural uranium reflectors. The objective of these experiments was to obtain a precision graph of the critical mass of highly enriched uranium metal spheres as a function of natural-uranium reflector thickness and to generate transport cross sections for the reflector material. Four configurations were measured with the thickness of the natural-uranium reflectors ranging from approximately 0.7 in. to 3.9 in. All four configurations were subcritical with measured maximum multiplications ranging from 53 to 167. Based on this evaluation all four configurations were judged to be acceptable for use as criticality-safety benchmark experiments.

All the experiments were performed on the Comet assembly machine at the Los Alamos critical assembly facility. The upper reflector hemisphere was screwed to a hydraulic piston attached to an "A" frame secured to the Comet vertical support structure. The lower reflector hemisphere containing the enriched-uranium sphere was seated on a hollow aluminum cylinder attached to the top of a hydraulic ram. During assembly the upper reflector hemisphere remained stationary while the lower reflector and core were raised. Note that this experiment did not contain any midplane diaphragm often used in similar experiments. The counting system consisted of three boron-lined neutron chambers placed in long-counter geometry and mounted on a lift situated several feet from the reflector surface. Any one of these monitors would initiate separation of the materials by dropping the ram and raising the hydraulic piston if the measured neutron leakage exceeded a preset level. Table 5-69 shows results for HEU-MET-FAST-032.

Table 5-69 Results for HEU-MET-FAST-032

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumf32c1 | 1.0000 | 0.0016 | 1.592 | 1.0054 | 0.0007 | 0.0054 | ± | 0.0035 |
| heumf32c2 | 1.0000 | 0.0027 | 1.590 | 1.0057 | 0.0006 | 0.0057 | ± | 0.0055 |
| heumf32c3 | 1.0000 | 0.0017 | 1.575 | 0.9992 | 0.0006 | -0.0008 | ± | 0.0036 |
| heumf32c4 | 1.0000 | 0.0017 | 1.565 | 0.9994 | 0.0006 | -0.0006 | ± | 0.0036 |

### 5.2.2.2.9 HEU-MET-FAST-041

In the early 1950s a series of experiments was performed at the Los Alamos critical assembly facility to determine the critical mass of highly enriched uranium (93.9 wt.% $^{235}U$) spheres surrounded by thin reflectors of various materials ranging in atomic weight from beryllium to natural uranium. The objective of these experiments was to obtain a precision graph of the critical mass of highly enriched uranium metal spheres as a function of reflector thickness and to generate transport cross sections for the reflector material. Six configurations are described in this evaluation; two with 1.85-inch- and 4.64-inch-thick beryllium reflectors, and four with 2-, 4-, 6-, and 8-inch-thick graphite reflectors. All six configurations were subcritical with measured maximum multiplications ranging from 42 to 150. In all cases the measured configurations were idealized by the experimenters and reported as homogeneous concentric spheres of adjacent core and reflector materials. A previous evaluation (HEU-MET-FAST-032) described four configurations with natural uranium reflectors performed in the same series. Based on this evaluation all six configurations were judged to be acceptable for use as criticality safety benchmark experiments (NEA [2001]).

All the experiments were performed on the Comet assembly machine at the Los Alamos critical assembly facility. The upper reflector hemisphere was screwed to a hydraulic piston attached to an "A" frame secured to the Comet vertical support structure. The lower reflector hemisphere containing the enriched uranium sphere was seated on a hollow aluminum cylinder attached to the top of a hydraulic ram. Note that this experiment did not contain any midplane diaphragm often used in similar configurations. Three safety monitors were placed at varying distances from the assembly. Any one of these monitors would initiate separation of the materials by dropping the ram and raising the hydraulic piston if the measured neutron leakage exceeded a preset level. Table 5-70 shows the results for this experiment.

Table 5-70 Results for HEU-MET-FAST-041

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumf41c1 | 1.0013 | 0.0030 | 1.331 | 0.9991 | 0.0007 | -0.0022 | ± | 0.0062 |
| heumf41c2 | 1.0022 | 0.0043 | 1.128 | 0.9973 | 0.0007 | -0.0049 | ± | 0.0087 |
| heumf41c3 | 1.0006 | 0.0029 | 1.466 | 0.9984 | 0.0006 | -0.0022 | ± | 0.0059 |
| heumf41c4 | 1.0006 | 0.0025 | 1.402 | 1.0009 | 0.0007 | 0.0003 | ± | 0.0052 |
| heumf41c5 | 1.0006 | 0.0031 | 1.354 | 0.9966 | 0.0008 | -0.0040 | ± | 0.0064 |
| heumf41c6 | 1.0006 | 0.0045 | 1.315 | 0.9993 | 0.0007 | -0.0013 | ± | 0.0091 |

### 5.2.2.2.10 HEU-MET-FAST-055

The fast zero power reactor experiment (ZPR-3) was built at the site of the National Reactor Testing Station in Idaho in 1955 to obtain neutron physics information necessary for the design of fast power breeder reactors. It was used to study blanketed cores having a range of

compositions in which the volume fractions of $^{235}U$, $^{238}U$, iron and sodium (simulated by reduced-density aluminum) were varied. Assembly 23 was planned to provide more information regarding the effects of heterogeneity and streaming in large dilute fast cores. Experiments were performed in Assembly 23 between May 6 and July 20, 1959. The ZPR-3 Assembly 23 was designed to have a very clean and simple benchmark configuration. In fact, the entire core of this assembly was built with essentially a single unit-cell using only plates of enriched U and aluminum. This assembly provides a useful benchmark for testing criticality calculations. Two nuclides, $^{235}U$ and Al, dominated the neutronic behavior in the core region. Because this assembly was fully reflected by depleted U and had high leakage from the core, the system reactivity was also sensitive to the $^{238}U$ data. The $^{235}U$ / Al ratio is such that the bulk of the neutron spectrum is in the 100-keV to 3-MeV energy range and the peak is at about 700 keV. As would be expected, the neutron balance for this assembly is dominated by $^{235}U$ and Al. This assembly is also closely related to the ICSBEP benchmark evaluations for ZPR-9 Assembly 34, the Uranium/Iron Benchmark Assembly and ZPR-9 Assembly 36, the U9 Benchmark Assembly (see IEU-MET-FAST-010). Like the Uranium/Iron Benchmark Assembly, this assembly addresses for the criticality safety community the perceived lack of metal/$^{235}U$ systems. A very accurate transformation to a simplified model is needed to make any of the ZPR assemblies a practical criticality-safety benchmark. Table 5-71 shows results for HEU-MET-FAST-055.

Table 5-71 Results for HEU-MET-FAST-055

| Experiment /Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hmf055 | 0.9955 | 0.0028 | 1.077 | 0.9938 | 0.0003 | -0.0017 | ± | 0.0056 |

### 5.2.2.2.11 HEU-MET-THERM-003

During or shortly prior to 1955, a series of experiments was conducted at the Pajarito site at LANL using cubic or nearly cubic lattices of oralloy (Oy) cubes immersed inside a tank of water. (Oralloy denotes uranium enriched to approximately 93 wt.% $^{235}U$.) The objective of these experiments was to help bridge the gap between critical mass data for solid units of Oy (water tamped) and for immersed Oy lathe turnings or solutions with molecular units. Although none of the configurations was taken to an actual critical condition, the results obtained were interpolated to estimate the optimum separation and minimum mass for criticality to occur. A total of 12 experiments were performed, with different sizes of cubes and different spacings between them. Six of these experiments, as well as a seventh configuration for a single cube with an extrapolated critical mass, are considered to be acceptable for use as benchmark experiments for criticality safety (NEA [2001]). The remaining six experiments are not considered acceptable, either because the measured multiplication was too low or because there was insufficient documentation.

Although most of the benchmark models that were developed from this series of experiments involve arrays of small oralloy cubes immersed in water, one of the benchmark models involves a single parallelepiped reflected by water and another involves a single cube reflected by water.

In Cases 5, 6, and 7 with lattices of small cubes, most of the fissions occur in the thermal-energy range. However, most of the fissions occur in the fast range for Cases 1 and 2 with a single oralloy unit. The fission spectra for Cases 3 and 4, with lattices of larger size cubes, is mixed. Therefore, Cases 1 and 2 are appropriately identified as HEU-MET-FAST-006, and Cases 3 and 4 are appropriately identified as HEU-MET-MIXED-010, but complete documentation and evaluation of those configurations are included with the evaluation of the other experiments in the series, HEU-MET-THERM-003.

The arrays of oralloy cubes were placed on Lucite plates and then lowered into a tank containing water. The experiments were performed on the Comet assembly machine, which consists of a hydraulic lift (ram) which rises vertically, and an air piston which extends downward from an A-frame superstructure over the. The Lucite plates were clamped to four brass rods, 0.5 in. by 24 in. long, and the brass rods were attached to the piston. Proper spacing of the Lucite plates was achieved by means of aluminum cylinders and clamps, while proper spacing of the oralloy cubes upon the Lucite plates was achieved by countersunk holes or grooves in the plates. The ram supported a 28 in. high x 35-1/2 in. diameter steel tank filled to within 5 in. from the top with water. The plates and the cubes they supported were inserted into the tank by raising the ram. One of the configurations, consisting of a single parallelepiped of oralloy, was supported further by a Lucite box with variable side dimensions that was fastened to the bottom support. Twelve experiments were performed with oralloy cubes and a parallelepiped:

- One experiment with a single parallelepiped, 4.5 in. by 4 in. by 4 in.
- Four experiments with 1-in. cubes at uniform center-to-center spacings of 1.25 in., 1.50 in., 1.75 in., and 2.00 in.
- Five experiments with 0.5-in. cubes at uniform center-to-center spacings of 0.75 in., 1.00 in., 1.17 in., 1.50 in., and 2.25 in.
- Two experiments with body-centered cubic configurations of 0.5-in. cubes. One of these experiments had a minimum corner-to-center spacing of 1.30 in. (which corresponds to corner-to-corner spacing of 1.50 in.), while the other had a vertical spacing of 0.75 in. and a horizontal spacing of 1.50 in.

Results for HEU-MET-THERM-003 are given in Table 5-72.

Table 5-72 Results for HEU-MET-THERM-003

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumt3c1 | 1.0000 | 0.0010 | 1.094 | 1.0039 | 0.0008 | 0.0039 | ± | 0.0025 |
| heumt3c2 | 0.9910 | 0.0030 | 1.091 | 0.9832 | 0.0008 | -0.0078 | ± | 0.0062 |
| heumt3c3 | 0.9826 | 0.0060 | 0.548 | 0.9766 | 0.0008 | -0.0060 | ± | 0.0121 |

| Table 5-72 Results for HEU-MET-THERM-003 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| heumt3c4 | 0.9876 | 0.0040 | 0.444 | 0.9842 | 0.0009 | -0.0034 | ± | 0.0082 |
| heumt3c5 | 0.9930 | 0.0030 | 0.2220 | 1.0014 | 0.0009 | 0.0084 | ± | 0.0063 |
| heumt3c6 | 0.9889 | 0.0030 | 0.291 | 0.9719 | 0.0009 | -0.0171 | ± | 0.0063 |
| heumt3c7 | 0.9919 | 0.0030 | 0.231 | 0.9887 | 0.0008 | -0.0032 | ± | 0.0062 |

### 5.2.2.3 HEU Si-Containing Experiments

#### 5.2.2.3.1 HEU-MET-MIXED-005

A series of five critical configurations with heterogeneous compositions of highly enriched uranium, silicon dioxide, and polyethylene was performed at the Institute of Physics and Power Engineering (IPPE), Obninsk, Russia at the Big Physical Stand (BFS) facility in August – October 1999. The U.S. National Spent Nuclear Fuel Program (NSNFP) at the Idaho National Engineering and Environmental Laboratory (INEEL) sponsored these experiments. The purpose of these experiments was to obtain data that are applicable to a wide range of criticality safety operations involving fissile contaminated waste streams. In particular, the data were intended for NSNFP validations of criticality safety calculations performed in support of storage of highly enriched spent nuclear fuels in the United States geologic repository at Yucca Mountain, Nevada. The experiments were performed at the BFS-1 facility and the critical configurations were designated as BFS-79/1, -79/2, -79/3, -79/4, and -79/5, according to the rules of the BFS facility. In October 1999 – February 2000 an analogous series of six experiments with plutonium fuel was performed (BFS-81 experiments). The plutonium-fueled experiments are evaluated in PU-MET-MIXED-001.

The critical assemblies were composed of sets of aluminum tubes (5.0 cm in outer diameter, 0.1 cm wall thickness, and about 2 m long) filled with pellets of highly enriched uranium metal in aluminum cans, silicon dioxide, and polyethylene. Also, quartz (silicon dioxide) sand was used to fill tubes in the reflector at the core periphery. Thin polyethylene dowels were inserted into the space between the tubes in some cases, to provide well-defined moderation similar to water. The pitch of the hexagonal lattice of the tubes was 5.1 cm. All five uranium-fueled experiments are evaluated here and are considered to be acceptable as criticality safety benchmark experiments. While the experiments of this series cover a wide range of spectra of neutrons causing fission (one thermal, two mixed, and two intermediate cases), the experiments are categorized as having mixed spectra. Table 5-73 shows results for HEU-MET-MIXED-005.

| Table 5-73 Results for HEU-MET-MIXED-005 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| hmm5_1 | 1.0007 | 0.0027 | 0.307 | 1.01308 | 0.00057 | 0.0124 | ± | 0.0055 |
| hmm5_2 | 1.0003 | 0.0028 | 0.247 | 1.02170 | 0.00055 | 0.0214 | ± | 0.0057 |
| hmm5_3 | 1.0012 | 0.0029 | 0.212 | 1.01904 | 0.00052 | 0.0178 | ± | 0.0059 |
| hmm5_4 | 1.0016 | 0.0030 | 0.3175 | 1.0145 | 0.0006 | 0.0129 | ± | 0.0061 |
| hmm5_5 | 1.0005 | 0.0040 | 0.377 | 1.00682 | 0.00052 | 0.0063 | ± | 0.0081 |

## 5.3 PREVIOUSLY EVALUATED LEU BENCHMARK CASES

### 5.3.1 LEU Benchmarks Related to intact Waste Package Configurations

#### 5.3.1.1 LEU Water-Reflected Fuel Rod Clusters in Square Pitched Arrays

A series of critical experiments with clusters of aluminum clad $UO_2$ fuel rods in a large water-filled tank was performed over a period of several years at the Critical Mass Laboratory at Pacific Northwest Laboratories (PNL). Eight cases were analyzed under this category in CRWMS M&O (1999a), that correspond to water-reflected clusters at 2.032 cm square pitch with no absorber plates, reflecting walls, dissolved poison, or gadolinium impurity. A detailed description of the experimental configuration is in NEA (2001, LEU-COMP-THERM-001). The benchmark $k_{eff} \pm$ standard deviation ($\sigma$) for this set of experiments is 0.9998 ± 0.0031. Each of the experiments used 2.35 wt% $^{235}U$ enriched $UO_2$ fuel with an average loading of 17.08 g of $^{235}U$ per rod. Table 5-74 gives brief descriptions of these experiments.

| Table 5-74 Water-Reflected Fuel Rod Cluster Critical Experiments | | | |
|---|---|---|---|
| Handbook case name | Description<br>Number of rods (X x Y), number of clusters, cluster separation | $H/^{235}U$[a] | MCNP case name |
| Case 1 | 20 x 18.08, 1 cluster | 459.0644 | Case1 |
| Case 2 | 20 x 17, 3 clusters, 11.92 ± 0.04 cm separation | 486.8467 | Case2 |
| Case 3 | 20 x 16, 3 clusters, 8.41 ± 0.05 cm separation | 469.1267 | Case3 |
| Case 4 | 20 x 16 (center), 22 x 16 (two outer), 3 clusters, 10.05 ± 0.05 cm separation | 474.2351 | Case4 |
| Case 5 | 20 x 15, 3 clusters, 6.39 ± 0.05 cm separation | 458.9289 | Case5 |
| Case 6 | 20 x 15 (center), 24 x 15 (two outer), 3 clusters, 8.01 ± 0.06 cm separation | 462.3500 | Case6 |
| Case 7 | 20 x 14, 3 clusters, 4.46 ± 0.10 cm separation | 449.1855 | Case7 |
| Case 8 | 19 x 6, 3 clusters, 7.57 ± 0.04 cm separation | 466.8974 | Case8 |

Note: a $H/^{235}U$ ratio is from section 5.1.1.1 of CRWMS M&O (1999a).

The following results are from CRWMS M&O (1999a, page 69). Table 5-75 presents the MCNP calculation results for ENDF/B-V nd ENDF/B-VI cross section libraries for the LEU-COMP-THERM -001 set of critical experiments. Table 5-76 shows $\Delta k_{eff}$ and Table 5-77 presents the AENCF values for the LEU-COMP-THERM-001 set of critical experiments. It can be seen that all of the calculated results are slightly less than the benchmark $k_{eff}$ value. All of the cross section sets under-predict the $k_{eff}$ value, with the ENDF/B-VI library being the least conservative.

Table 5-75 LCE Reactivity Calculation Results for LEU-COMP-THERM-001 Experiments

| MCNP case name | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-VI $k_{eff} \pm \sigma$ |
|---|---|---|
| Case_1 | 0.9987 ± 0.00191 | 0.99248 ± 0.00153 |
| Case_2 | 0.99557 ± 0.00185 | 0.9923 ± 0.00161 |
| Case_3 | 0.9957 ± 0.00175 | 0.99637 ± 0.00172 |
| Case_4 | 0.99907 ± 0.00144 | 0.98949 ± 0.00156 |
| Case_5 | 0.99313 ± 0.00157 | 0.99429 ± 0.00172 |
| Case_6 | 0.99882 ± 0.00143 | 0.99161 ± 0.00158 |
| Case_7 | 0.99677 ± 0.00143 | 0.99578 ± 0.00155 |
| Case_8 | 1.00246 ± 0.00162 | 0.99597 ± 0.00149 |

Table 5-76 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for LEU-COMP-THERM -001 Experiments

| MCNP case name | ENDF/B-V $\Delta k_{eff} \pm 2\sigma$ | ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|
| Case_1 | 0.00110 ± 0.00728 | 0.00732 ± 0.00691 |
| Case_2 | 0.00423 ± 0.00722 | 0.00750 ± 0.00699 |
| Case_3 | 0.00410 ± 0.00712 | 0.00343 ± 0.00486 |
| Case_4 | 0.00073 ± 0.00684 | 0.01031 ± 0.00694 |
| Case_5 | 0.00667 ± 0.00695 | 0.00551 ± 0.00709 |
| Case_6 | 0.00098 ± 0.00683 | 0.00819 ± 0.00696 |
| Case_7 | 0.00303 ± 0.00683 | 0.00402 ± 0.00693 |
| Case_8 | -0.00266 ± 0.00700 | 0.00383 ± 0.00485 |

Table 5-77 AENCF for LEU-COMP-THERM –001 Experiments (MeV)

| MCNP case name | ENDF/B-V | ENDF/B-VI |
|---|---|---|
| Case1 | 0.1239 | 0.1227 |

| Table 5-77 AENCF for LEU-COMP-THERM –001 Experiments (MeV) | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| Case2 | 0.1208 | 0.1210 |
| Case3 | 0.1220 | 0.1221 |
| Case4 | 0.1208 | 0.1207 |
| Case5 | 0.1218 | 0.1187 |
| Case6 | 0.1208 | 0.1219 |
| Case7 | 0.1186 | 0.1176 |
| Case8 | 0.1201 | 0.1214 |

### 5.3.1.2 Critical Configurations of Subcritical Clusters of 2.35 wt% Enriched $UO_2$ Rods in Water with Fixed Neutron Absorber Plates

Experiments with subcritical clusters of low-enrichment $UO_2$ fuel rods were performed at the PNL in 1977. The four experiments modeled with MCNP in CRWMS M&O (1999c) consisted of three rectangular arrays of aluminum-clad fuel rods. The fuel rods comprising the arrays had a uniform enrichment of 2.35 wt.% $^{235}U$. The three arrays of fuel were arranged in a row and, in three of the experiments, sheets of neutron poison were interposed between adjacent arrays. The structure of the experimental assembly was provided by aluminum structural members on the margins of the fuel arrays. Axial support for the fuel rods was provided by an acrylic base plate. The lateral alignment of the fuel rods was provided by another acrylic plate. The experimental apparatus was closely reflected by full-density water.

The pertinent differences among these four experiments are shown in Table 5-78. These critical experiments help demonstrate the ability of MCNP to accurately predict the critical multiplication factor for configurations containing light-water reactor fuel separated by absorber plates. These experiments are designated as LEU-COMP-THERM-016 in NEA-2001 and their results are presented in Table 5-79.

| Table 5-78 Differences in Absorber Plates used for Clusters of 2.35 wt.% $UO_2$ Fuel Rods | |
|---|---|
| **Experiment identifier** | **Interposed plate** |
| exp1 | none |
| exp2 | Boral™ |
| exp3 | Type 6061 Aluminum |
| exp4 | Type 304 Stainless Steel |

Table 5-79 Results for Critical Experiments with Clusters of 2.35 wt.% $UO_2$ Fuel Rods

| Case | ENDF/B-V | | | ENDF/B-VI | | |
|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF |
| exp1 | 0.99958 | 0.00087 | 0.12037 | 0.99657 | 0.00093 | 0.12075 |
| exp2 | 0.99762 | 0.00086 | 0.12291 | 0.99351 | 0.00097 | 0.12334 |
| exp3 | 1.00103 | 0.00091 | 0.12005 | 0.99613 | 0.00095 | 0.12083 |
| exp4 | 1.00241 | 0.00091 | 0.12057 | 0.99582 | 0.0009 | 0.12186 |

### 5.3.1.3 Critical Configurations with Subcritical Clusters of 4.31 wt% Enriched $UO_2$ Rods in Water with Reflecting Walls

The pertinent differences among the three experiments performed at PNL are shown in Table 5-80. These three experiments were modeled in CRWMS M&O (1999c) and results provided in Table 5-81. In these experiments three similar fuel assemblies were laterally surrounded by reflectors of different compositions. The fuel lattices in each critical experiment contained 4.31 wt.% $^{235}U$ enriched $UO_2$ fuel rods on a square pitch of 1.892 cm. These critical experiments demonstrate the ability of MCNP to accurately predict the critical multiplication factor for configurations with different shielding materials used for reflectors.

Table 5-80 Differences in Experimental Configurations for Clusters of 4.31 wt% $UO_2$ Fuel Rods

| Experiment identifier | Reflector |
|---|---|
| exp5 | uranium |
| exp6 | lead |
| exp7 | stainless steel |

Table 5-81 Results for Critical Experiments with Clusters of 4.31 wt% $UO_2$ Fuel Rods

| Case | ENDF/B-V | | | ENDF/B-VI | | |
|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF |
| exp5 | 0.9969 | 0.00103 | 0.28386 | 0.99295 | 0.00104 | 0.28018 |
| exp6 | 0.99772 | 0.00105 | 0.17775 | 0.9944 | 0.00111 | 0.17643 |
| exp7 | 0.99821 | 0.0011 | 0.17822 | 0.99144 | 0.00105 | 0.1773 |

### 5.3.1.4 Water-Moderated, Lead-Reflected Uranium Dioxide Rod Array

NEA (2001, LEU-COMP-THERM-027) documents a series of four experiments involving lead-reflected, water-moderated arrays of low-enriched $UO_2$ fuel rods. The experiments were subcritical approaches extrapolated to critical; the multiplication factor reached was very close to 1.000 (within 0.1%). The experiments were tests of the lead reflector effect. Only the first case was evaluated in CRWMS M&O (1999c). This case consisted of a 14 x 14 array of 4.74 wt% enriched $UO_2$ fuel rods reflected on four sides by 30-cm-thick lead reflectors with no water gap

between the array and the lead reflectors. This experiment was denoted as lct27-1 and the MCNP result is presented in Table 5-82.

Table 5-82 Results for LEU-COMP-THERM-027

| Case | ENDF/B-V | | | ENDF/B-VI | | |
|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF |
| Lct27-1 | 1.0139 | 0.0005 | 0.1024 | 1.0102 | 0.0006 | 0.1023 |

### 5.3.1.5 Critical Configurations with 4.31 wt% $^{235}U$ Enriched $UO_2$ Rods in Highly Borated Water Lattices

This set of four experiments was performed at the PNL and modeled in CRWMS M&O (1999c). These experiments used 4.31 wt% $^{235}U$ uniformly enriched $UO_2$ fuel rods arranged in square-pitch, water-moderated lattices of different size with various amounts of boric acid in the moderator. The fuel rods were clad with aluminum and were loaded into polypropylene lattice templates fastened inside a Plexiglas tank. The Plexiglas tank was surrounded on all four sides by an unborated water reflector and was positioned on top of a Plexiglas slab. The borated water was restricted to the water volume inside the Plexiglas tank.

Rectangular critical arrays were constructed by sequentially filling rows of the lattice template starting at the Plexiglas tank wall. The water level in the tank was held constant by removing an appropriate volume of water as each fuel rod was loaded. These experiments were denoted as "exp8" through "exp11" in page 41 of CRWMS M&O (1999c). The results are presented in Table 5-83.

Table 5-83 Results for Critical Configurations with 4.31 wt% $^{235}U$ Enriched $UO_2$ Rods in Highly Borated Water Lattices

| Case | ENDF/B-V | | | ENDF/B-VI | | |
|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF |
| Exp8 | 1.00700 | 0.00102 | 0.17807 | 1.0033 | 0.00105 | 0.17769 |
| Exp9 | 1.00687 | 0.00109 | 0.22393 | 1.00496 | 0.00105 | 0.22242 |
| Exp10 | 1.00499 | 0.00173 | 0.2225 | 1.00454 | 0.00182 | 0.22149 |
| Exp11 | 1.00046 | 0.00177 | 0.26803 | 0.99995 | 0.00162 | 0.26652 |

### 5.3.1.6 Laboratory Critical Experiments from the Urania-Gadolinia: Nuclear Model Development and Critical Experiment Benchmark Report

A number of critical experiments were performed by Babcock and Wilcox for urania fuel incorporating gadolinia as an integral burnable absorber. These experiments were evaluated with MCNP in CRWMS M&O (1999c, page 41). The configurations modeled with MCNP included critical configurations containing arrangements of 2.46 wt% $^{235}U$ enriched $UO_2$ fuel rods, 4.02 wt% $^{235}U$ enriched $UO_2$ fuel rods, combination 4 wt% $Gd_2O_3$ and 96 wt% (1.944 wt% $^{235}U$ enriched) $UO_2$ fuel rods, Ag-In-Cd absorber rods, and $B_4C$ absorber rods. The fuel rods were

supported by a top and bottom aluminum "eggcrate" type grid plate. The fuel rods rested on an aluminum base plate. The central 45 x 45 array of rod lattice cells was separated into nine 15 x 15 arrays of rod lattice cells. These arrays were intended to simulate pressurized water reactor fuel assembly lattices.

Descriptions of the experimental configurations are shown in Table 5-84 and MCNP results are included in Table 5-85 (CRWMS M&O [1999c, page 10 and 41]).

Table 5-84 Urania-Gadolinia Critical Experiment Descriptions

| Exp. ID[a] | Number of 2.46 wt% $^{235}U$ fuel rods | Number of 4.02 wt% $^{235}U$ fuel rods | Number of $Gd_2O_3$ fuel rods | Number of $B_4C$ rods | Number of Ag-In-Cd rods | Number of void rods | Number of water holes |
|---|---|---|---|---|---|---|---|
| ugd1 | 4808 | 0 | 0 | 0 | 0 | 0 | 153 |
| ugd2 | 4808 | 0 | 0 | 0 | 16 | 0 | 137 |
| ugd3 | 4788 | 0 | 20 | 0 | 0 | 0 | 153 |
| ugd4 | 4788 | 0 | 20 | 0 | 16 | 0 | 137 |
| ugd5 | 4780 | 0 | 28 | 0 | 0 | 0 | 153 |
| ugd6 | 4780 | 0 | 28 | 0 | 16 | 0 | 137 |
| ugd7 | 4780 | 0 | 28 (Annular) | 0 | 0 | 0 | 153 |
| ugd8 | 4772 | 0 | 36 | 0 | 0 | 0 | 153 |
| ugd9 | 4772 | 0 | 36 | 0 | 16 | 0 | 137 |
| ugd10 | 4772 | 0 | 36 | 0 | 0 | 16 | 137 |
| ugd12 | 3920 | 888 | 0 | 0 | 0 | 0 | 153 |
| ugd13 | 3920 | 888 | 0 | 16 | 0 | 0 | 137 |
| ugd14 | 3920 | 860 | 28 | 0 | 0 | 0 | 153 |
| ugd15 | 3920 | 860 | 28 | 16 | 0 | 0 | 137 |
| ugd16 | 3920 | 852 | 36 | 0 | 0 | 0 | 153 |
| ugd17 | 3920 | 852 | 36 | 16 | 0 | 0 | 137 |
| ugd18 | 3676 | 944 | 0 | 0 | 0 | 0 | 180 |
| ugd19 | 3676 | 928 | 16 | 0 | 0 | 0 | 180 |
| ugd20 | 3676 | 912 | 32 | 0 | 0 | 0 | 180 |

NOTE: [a] ID = Identifier

Table 5-85 Results for Urania-Gadolinia Critical Experiments

| Case | ENDF/B-V | | | ENDF/B-VI | | |
|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF |
| Ugd1 | 0.99717 | 0.0014 | 0.20069 | 0.99688 | 0.00149 | 0.19672 |
| ugd20 | 1.00179 | 0.00159 | 0.20624 | 0.9971 | 0.00155 | 0.20429 |
| Ugd2 | 0.99892 | 0.0015 | 0.20018 | 0.9947 | 0.00151 | 0.19858 |
| Ugd3 | 0.99846 | 0.00145 | 0.19736 | 0.99449 | 0.00151 | 0.19988 |
| Ugd4 | 0.99911 | 0.00144 | 0.19744 | 1.00031 | 0.00149 | 0.1993 |
| Ugd5 | 0.99958 | 0.00147 | 0.20041 | 0.99686 | 0.0015 | 0.19842 |
| Ugd6 | 1.00229 | 0.00149 | 0.19897 | 0.9952 | 0.00154 | 0.19894 |
| Ugd7 | 0.99807 | 0.00144 | 0.19896 | 0.99287 | 0.00147 | 0.19687 |
| Ugd8 | 0.99904 | 0.00151 | 0.19942 | 0.99722 | 0.00147 | 0.19833 |
| Ugd9 | 1.00042 | 0.00159 | 0.19963 | 0.99743 | 0.00148 | 0.19651 |
| ugd10 | 1.00115 | 0.00145 | 0.20029 | 0.99696 | 0.00151 | 0.19695 |
| ugd12 | 1.00178 | 0.00146 | 0.20785 | 1.00134 | 0.00149 | 0.2088 |
| ugd13 | 0.99887 | 0.00166 | 0.20937 | 1.00037 | 0.00148 | 0.2066 |

Table 5-85 Results for Urania-Gadolinia Critical Experiments

| Case | ENDF/B-V | | | ENDF/B-VI | | |
|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF |
| ugd14 | 1.00069 | 0.00144 | 0.20585 | 0.99761 | 0.00153 | 0.20711 |
| ugd15 | 0.99927 | 0.00147 | 0.20333 | 0.9983 | 0.00156 | 0.20342 |
| ugd16 | 0.99904 | 0.0015 | 0.20947 | 0.99903 | 0.00158 | 0.20601 |
| ugd17 | 0.99909 | 0.00153 | 0.20572 | 0.998 | 0.00149 | 0.2024 |
| ugd18 | 0.99741 | 0.00156 | 0.21013 | 0.99819 | 0.00155 | 0.20862 |
| ugd19 | 0.99908 | 0.00155 | 0.20757 | 1.00098 | 0.00151 | 0.20506 |

### 5.3.1.7 LEU Systems Typical of N Reactor Fuel in the K Basin

Three cases, which analyze a lattice of actual N Reactor MKIA fuel elements were evaluated in CRWMS M&O (1999a, page 70). The MKIA cases analyzed 101.2 fuel elements, 67.4 fuel elements, and 90.3 fuel elements with corresponding pitches of 7.112 cm, 7.874 cm, and 8.636 cm, respectively. The MKIA experiments were performed for three different lattice pitches resulting in three experimental values to achieve a $k_{eff}$ of unity. The lattice pitches and corresponding critical number of fuel elements are listed in Table 5-86, and Table 5-87 provides the physical dimensions for the MKIA fuel rods. It should be noted that the MKIA experiments used 26.2 in. (66.548 cm) long fuel elements stacked two high for a 52.4 in. (133.096 cm) fuel column per lattice) with 121 filled fuel lattices. In this set of experiments the number of fuel elements was extrapolated to critical.

Table 5-86 MKIA Fuel Assembly Experiments (CRWMS M&O [1999a])

| Lattice pitch | Critical number of fuel elements experimentally determined | Benchmark $k_{eff} \pm \sigma$ | H/X ratio | MCNP case name |
|---|---|---|---|---|
| 7.112 cm | 101.2 | 1.000 ± 0.005 | 993.7259 | SUBC2P8H |
| 7.874 cm | 67.4 | 1.000 ± 0.005 | 1413.6232 | SUBC3P1H |
| 8.636 cm | 90.3 | 1.000 ± 0.005 | 1876.2220 | SUBC3P4H |

Table 5-87 MKIA Fuel Element Parameters

| Parameter | Outer tube | Inner tube |
|---|---|---|
| Zirconium clad OD cm (in.) | 6.106 (2.404) | 3.165 (1.246) |
| uranium OD cm (in.) | 5.979 (2.354) | 2.962 (1.166) |
| uranium ID cm (in.) | 4.592 (1.808) | 1.245 (0.490) |
| zirconium clad ID cm (in.) | 4.481 (1.764) | 1.118 (0.440) |
| wt% $U^{235}$ | 1.25 | 0.947 |
| wt% $U^{236}$ | 0.0392 | 0.0392 |
| wt% $U^{238}$ | 98.7108 | 99.0138 |
| end cap thickness cm (in.) | 0.483 (0.190) | 0.483 (0.190) |

| Table 5-87 MKIA Fuel Element Parameters | | |
|---|---|---|
| **Parameter** | **Outer tube** | **Inner tube** |
| fuel density (g/cm$^3$) | 18.82 | 18.82 |
| clad density (g/cm$^3$) | 6.55 | 6.55 |
| moderator density (g/cm$^3$) | 1.00 | 1.00 |

The following results are from CRWMS M&O (1999a, page 70). Table 5-88 presents the MCNP calculation results for ENDF/B-V, and ENDF/B-VI cross section libraries for the MKIA set of critical experiments and Table 5-89 shows Δk. Table 5-90 presents the AENCF values for the MKIA set of critical experiments.

| Table 5-88 Criticality Calculation Results for MKIA Experiments | | | |
|---|---|---|---|
| **MCNP case name** | **Reference MCNP $k_{eff}$ ± σ** | **ENDF/B-V $k_{eff}$ ± σ** | **ENDF/B-VI $k_{eff}$ ± σ** |
| subc2p8h | 1.0088 ± 0.0021 | 1.0048 ± 0.00294 | 1.01057 ± 0.00335 |
| subc3p1h | 1.0392 ± 0.0018 | 1.03902 ± 0.00248 | 1.04014 ± 0.00310 |
| subc3p4h | 1.0232 ± 0.0019 | 1.02088 ± 0.00277 | 1.01971 ± 0.00266 |

| Table 5-89 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for MKIA Experiments | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V $\Delta k_{eff}$ ± 2σ** | **ENDF/B-VI $\Delta k_{eff}$ ± 2σ** |
| subc2p8h | -0.00480 ± 0.01160 | -0.01057 ± 0.01251 |
| subc3p1h | -0.03902 ± 0.01116 | -0.04014 ± 0.04062 |
| subc3p4h | -0.02088 ± 0.01143 | -0.01971 ± 0.02042 |

| Table 5-90 AENCF for MKIA Experiments (MeV) | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| SUBC2P8H | 0.4085 | 0.4005 |
| SUBC3P1H | 0.3516 | 0.3447 |
| SUBC3P4H | 0.3145 | 0.3165 |

### 5.3.2 Benchmarks Related to Degraded Waste Package Configurations

#### 5.3.2.1 LEU Solution Experiments

This set of experiments is comprised of twelve cases that look at $UO_3$-$H_2O$ critical solutions. The $UO_3$-$H_2O$ experiments differed by varying enrichment and the (hydrogen to uranium) H/U ratio. The $UO_3$-$H_2O$ sets of experiments were simulated as homogeneous spheres and $k_\infty$ was calculated with MCNP. Table 5-91 contains a description of the $UO_3$-$H_2O$ experiments.

Table 5-91 $UO_3$-$H_2O$ Solution Experiments

| Enrichment (wt% $^{235}U$) | H/U | $K_\infty \pm$ experimental uncertainty | H/X ratio | MCNP case name |
|---|---|---|---|---|
| 1.0059 | 3.772 | 0.9920 ± 0.0060 | 370.2986 | SPHU9A |
| 1.0059 | 4.999 | 0.9925 ± 0.0050 | 490.7555 | SPHU9B |
| 1.0059 | 6.164 | 0.9875 ± 0.0058 | 605.1229 | SPHU9C |
| 1.0059 | 6.881 | 0.9821 ± 0.0054 | 675.5099 | SPHU9D |
| 1.0059 | 7.449 | 0.9702 ± 0.0070 | 731.2706 | SPHU9E |
| 1.0704 | 3.728 | 1.0063 ± 0.0073 | 343.9305 | SPHU9F |
| 1.0704 | 5.778 | 1.0064 ± 0.0078 | 533.0517 | SPHU9G |
| 1.0704 | 7.075 | 0.9957 ± 0.0061 | 652.7077 | SPHU9H |
| 1.1586 | 3.728 | 1.0298 ± 0.0056 | 317.7489 | SPHU9I |
| 1.1586 | 5.926 | 1.0330 ± 0.0051 | 475.6274 | SPHU9J |
| 1.1586 | 6.838 | 1.0313 ± 0.0032 | 582.8262 | SPHU9K |
| 1.1586 | 7.449 | 1.0209 ± 0.0051 | 634.9047 | SPHU9L |

The following results are from CRWMS M&O (1999a, page 90 and 91). Table 5-92 presents the MCNP calculation results for ENDF/B-V, and ENDF/B-VI cross section libraries for the $UO_3$-$H_2O$ solution set of critical experiments. Table 5-94 presents the AENCF values for the $UO_3$-$H_2O$ solution set of critical experiments.

Table 5-92 Criticality Calculation Results for $UO_3$-$H_2O$ Solution Experiments

| MCNP Case name | Reference MCNP $k_\infty \pm \sigma$ | ENDF/B-V $k_\infty \pm \sigma$ | ENDF/B-VI $k_\infty \pm \sigma$ |
|---|---|---|---|
| Sphu9a | 0.9898 ± 0.0015 | 0.99004 ± 0.00249 | 0.98665 ± 0.00293 |
| Sphu9b | 0.9945 ± 0.0015 | 0.99269 ± 0.00249 | 0.98518 ± 0.00271 |
| Sphu9c | 0.9830 ± 0.0017 | 0.97871 ± 0.00256 | 0.97958 ± 0.00190 |
| Sphu9d | 0.9761 ± 0.0015 | 0.97914 ± 0.00242 | 0.97132 ± 0.00233 |
| Sphu9e | 0.9680 ± 0.0014 | 0.96607 ± 0.00163 | 0.96429 ± 0.00221 |

Table 5-92 Criticality Calculation Results for $UO_3$-$H_2O$ Solution Experiments

| MCNP Case name | Reference MCNP $k_\infty \pm \sigma$ | ENDF/B-V $k_\infty \pm \sigma$ | ENDF/B-VI $k_\infty \pm \sigma$ |
|---|---|---|---|
| Sphu9f | 1.0125 ± 0.0017 | 1.00952 ± 0.00261 | 1.00009 ± 0.00306 |
| Sphu9g | 1.0103 ± 0.0017 | 1.0136 ± 0.00246 | 1.00763 ± 0.00235 |
| Sphu9h | 0.9964 ± 0.0016 | 0.99713 ± 0.00198 | 0.99364 ± 0.00224 |
| Sphu9i | 1.0358 ± 0.0014 | 1.03372 ± 0.00274 | 1.03156 ± 0.00233 |
| Sphu9j | 1.0412 ± 0.0014 | 1.04207 ± 0.00224 | 1.03178 ± 0.00240 |
| Sphu9k | 1.0311 ± 0.0016 | 1.02951 ± 0.00216 | 1.02122 ± 0.00206 |
| Sphu9l | 1.0240 ± 0.0015 | 1.02281 ± 0.0021 | 1.01910 ± 0.00237 |

Table 5-93 Δk Between Experimental $k_\infty$ and MCNP Calculated $k_\infty$ Results for $UO_3$-$H_2O$ Solution Experiments

| MCNP case name | ENDF/B-V $\Delta k_{eff} \pm 2\sigma$ | ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|
| sphu9a | 0.00196 ± 0.01299 | 0.00535 ± 0.00793 |
| sphu9b | -0.00019 ± 0.01117 | 0.00732 ± 0.00911 |
| sphu9c | 0.00879 ± 0.01268 | 0.00792 ± 0.00878 |
| sphu9d | 0.00296 ± 0.01183 | 0.01078 ± 0.01174 |
| sphu9e | 0.00413 ± 0.01437 | 0.00591 ± 0.00738 |
| sphu9f | -0.00322 ± 0.01551 | 0.00621 ± 0.00872 |
| sphu9g | -0.00720 ± 0.01636 | -0.00123 ± 0.00486 |
| sphu9h | -0.00143 ± 0.01283 | 0.00206 ± 0.00493 |
| sphu9l | -0.00392 ± 0.01247 | -0.00176 ± 0.00498 |
| sphu9j | -0.00907 ± 0.01114 | 0.00122 ± 0.00495 |
| sphu9k | 0.00179 ± 0.00772 | 0.01008 ± 0.01089 |
| sphu9l | -0.00191 ± 0.01103 | 0.00180 ± 0.00507 |

Table 5-94 AENCF for $UO_3$-$H_2O$ Solution Experiments (MeV)

| MCNP case name | ENDF/B-V | ENDF/B-VI |
|---|---|---|
| sphu9a | 0.2541 | 0.2562 |
| sphu9b | 0.2163 | 0.2155 |
| sphu9c | 0.1883 | 0.1839 |
| sphu9d | 0.1737 | 0.1740 |
| sphu9e | 0.1591 | 0.1660 |

Table 5-94 AENCF for $UO_3$-$H_2O$ Solution Experiments (MeV)

| MCNP case name | ENDF/B-V | ENDF/B-VI |
|---|---|---|
| sphu9f | 0.2511 | 0.2543 |
| sphu9g | 0.1839 | 0.1906 |
| sphu9h | 0.1651 | 0.1642 |
| sphu9i | 0.2495 | 0.2499 |
| sphu9j | 0.1783 | 0.1760 |
| sphu9k | 0.1661 | 0.1670 |
| sphu9l | 0.1549 | 0.1545 |

### 5.3.2.2 Low-Enrichment Uranium Solutions

The first set of experiments involving low-enrichment uranium are described in NEA (2001, LEU-SOL-THERM-002) and have been modeled in CRWMS M&O (1999c). These experiments are listed in Table 5-95. Results from page 56 of CRWMS M&O (1999c) are presented in Table 5-96.

Table 5-95 Benchmark Problem Summary for Configurations Incorporating Low-Enrichment Uranium Solutions

| Reference Identifier | Experiment Identifier |
|---|---|
| LEU-SOL-THERM-002 | LEUST21 |
| | LEUST22 |
| | LEUST23 |

Table 5-96 Homogenous Critical Experiments Using Low Enriched Uranium Nitrate Solutions (Thermal)

| Case | ENDF/B-V | | | ENDF/B-VI | | |
|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF |
| Leust21 | 0.99855 | 0.00058 | 0.02513 | 0.99532 | 0.00056 | 0.02492 |
| Leust22 | 0.99659 | 0.00064 | 0.0283 | 0.991 | 0.00067 | 0.02802 |
| Leust23 | 1.0009 | 0.0006 | 0.02684 | 0.99624 | 0.00058 | 0.02615 |

### 5.3.2.3 Low Enriched Uranyl Flouride Solutions

This experiment involved an aqueous solution of about 5 wt% enriched uranyl fluoride and is taken from NEA (2001, LEU-SOL-THERM-001). This experiment used the SHEBA-II (Solution High Energy Burst Assembly-II), which is a critical assembly experiment that was operated at the Los Alamos Critical Experiments Facility. This experiment is identified as Leust11 and was evaluated in Section 6.5.2 of CRWMS M&O (1999a). Results are shown in Table 5-97.

Table 5-97 Results for Homogenous Critical Experiments Using Low Enriched Uranyl Fluoride Solutions

| Case | ENDF/B-V | | | ENDF/B-VI | | |
|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF |
| Leust11 | 1.01182 | 0.00101 | 0.1884 | 1.00714 | 0.00092 | 0.0514 |

## 5.4 LEU BENCHMARK CASES

### 5.4.1 LEU Benchmarks Related to intact Waste Package Configurations

#### 5.4.1.1 LEU Rodded Arrays

##### 5.4.1.1.1 LEU-COMP-THERM-041

The CRISTO II experiment was performed in the EOLE reactor (CEA-Cadarache, France) in 1980- 1981 (NEA [2001, LEU-COMP-THERM-041]). The aim of this experiment was the study of high density storage: water gap $e_{H2O}$ = 4.5 cm. The experiment allows the comparison of the negative reactivity introduced by various kinds of neutron absorber plates (stainless steel, borated steel, and cadmium). CRISTO II provides the infinite multiplication factor of pressurized water reactor (PWR) assembly lattices for which a traditional critical experiment is not feasible; therefore, the $k_\infty$ curve and, consequently, the saturation point (the point at which the absorber becomes "black") are supplied as functions of the increase of the absorber efficiency (from 0 to 10 mg/cm$^2$ of $^{10}$B). Five configurations corresponding to four different absorber plates and one without absorber are considered acceptable for use as benchmark experiments. The enrichment was 3 wt.% $^{235}$U.

The core of the EOLE reactor is divided into two parts:

- A central part, called the experimental cavity (or test zone), consists of an aluminum vessel which contains the lattice to be studied (either a regular lattice with borated light-water moderator or the water-moderated PWR test lattice, with or without absorber plates)

- A driver core, composed of elements with $^{235}$U-enriched (90% and 93% by weight) U-Al plates, which surrounds the cavity. These Material Testing Reactor (MTR) elements (EOLE or MINERVE bundles) are moderated with light water in normal conditions of temperature and pressure.

The driver core enables the reactor to become critical. The reactor is controlled by four silver-indium-cadmium (AIC) blades, which slide into the special driver core MTR elements (BS1, BS2, BC1, BC2), and an automated cadmium control bar.

Each EOLE MTR bundle is constituted of two grooved blades which hold fuel plates. The whole is surrounded by 2-mm-thick AG3 clad. Each bundle can be dismantled and can contain 21

plates. In order to adjust the critical loading, some fuel plates can be replaced by AG3 plates (aluminum-magnesium alloy). When the bundle is positioned on the grid, the fuel plates are parallel to the cavity wall.

The MINERVE MTR bundle is also constituted of two grooved blades which hold fuel plates, but without surrounding clad. The whole can not be dismantled. Each bundle contains 18 fuel plates perpendicular to the cavity wall.

Two different types of experiments were described in this evaluation: storage array configurations and regular lattices. The regular lattices were not chosen for benchmark models. For the storage configurations, the lattice fuel pins (the pin characteristics are similar to those of a 17x17 PWR) are held in baskets. These baskets are composed of two 35-mm thick aluminum (AG3) plates linked with four empty tubular Zircaloy braces (inner/outer diameters 8.2 mm x 9.4 mm), simulating PWR guide tubes. The square lattice pitch is 12.6 mm ±0.01 mm. The array is constituted of 16 baskets containing 45 pins each. There are five kinds of storage configurations introduced into the cavity: without absorber plates, stainless steel absorber plates, borated-steel plates poisoned with 2.6 mg/cm$^2$ $^{10}$B, borated-steel plates poisoned with 9.6 mg/cm$^2$ $^{10}$B, and cadmium plates. The critical loading of the driver core as reported in Table 1 of LEU-COMP-THERM-041 is given in Table 5-98 and MCNP results are listed in Table 5-99.

Table 5-98 Critical Loadings of the Driver Core for the Investigated Storage Configurations.

| Conf. | Loading of Driver Core | | | Excess Reactivity ($10^{-5}$ Δk/k) | Position of the control bar [a] | Moderator Temperature (°C) |
|---|---|---|---|---|---|---|
| | Elements | Position | Number of U-Al plates | | | |
| A | 4 control | 1st row | 60 | +24 ±2 | 160 mm | 20.9 ± 0.1 |
| | 20 EOLE | 1st row | 256 | | | |
| B | 4 control | 1st row | 60 | +130 ±2 | 260 mm | 21.2 ± 0.1 |
| | 20 EOLE | 1st row | 428 | | | |
| C | 4 control | 1st row | 60 | +60 ±1.5 | 130 mm | 20.9 ± 0.1 |
| | 24 MINERVE | 1st row | 432 | | | |
| | 20 EOLE | 2nd row | 224 | | | |
| D | 4 control | 1st row | 60 | +32 ±2 | 90 mm | 21.0 ± 0.1 |
| | 24 MINERVE | 1st row | 432 | | | |
| | 24 EOLE | 2nd row | 264 | | | |
| E | 4 control | 1st row | 60 | +283 ±5 | 370 mm | 20.9 ± 0.1 |
| | 24 MINERVE | 1st row | 432 | | | |
| | 24 EOLE | 2nd row | 272 | | | |

NOTE: [a] Insertion depth (i.e., height between the lower absorber of the automated control bar and the upper fuel of the driver core elements).

| Table 5-99 Results for LEU-COMP-THERM-041 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{\infty,exp}$ [a] | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| lct41-01 | 1.1990 | 0.0062 | 0.118 | 1.2048 | 0.0006 | 0.0058 | ± | 0.0125 |
| lct41-02 | 1.1060 | 0.0062 | 0.131 | 1.1045 | 0.0006 | -0.0015 | ± | 0.0125 |
| lct41-03 | 0.8820 | 0.0079 | 0.168 | 0.8749 | 0.0008 | -0.0071 | ± | 0.0159 |
| lct41-04 | 0.8360 | 0.0091 | 0.180 | 0.8256 | 0.0008 | -0.0104 | ± | 0.0183 |
| lct41-05 | 0.8140 | 0.0103 | 0.185 | 0.7999 | 0.0007 | -0.0141 | ± | 0.0206 |

NOTE: [a] $k_{\infty,exp}$ is described on page 26 of LEU-COMP-THERM-041.

### 5.4.1.1.2 LEU-COMP-THERM-042

A series of critical approach experiments with clusters of 36-inch-long aluminum-clad U(2.35 wt.% enriched)$O_2$ fuel rods in a large water-filled tank was performed over the course of several years at the Critical Mass Laboratory at the PNL. The evaluation in NEA (2001) documents seven water-reflected experiments, performed in late 1979 or early 1980, with 3 rectangular clusters of 1.684-cm-pitched rods with absorber plates between clusters and steel walls on either side of the line of fuel clusters. The seven absorber-plate types were stainless steel, borated stainless steel, Boral, Boroflex, cadmium, copper, and copper with 1% cadmium. All of these experiments are judged to be acceptable as benchmark data.

Experiments were performed in a 0.952-cm-thick, open, carbon-steel tank. The minimum water-reflector thickness was 30.5 cm from the ends of the steel walls and from the sides of the clusters (including the steel walls) and 15 cm above and below the fuel. One of the three clusters was 25 rods long by 18 rods wide while the two outer clusters were 20 rods long by 18 wide. The clusters were arranged in a line with absorber plates between them. The absorber plates are positioned at the outer cell boundary of the center fuel cluster. Dimensions of the cluster separation and absorber plates as reported inTable 23 of LEU-COMP-THERM-042 are given in Table 5-100 and MCNP results are listed in Table 5-101.

| Table 5-100 Critical Configurations for LEU-COMP-THERM-042 | | | |
|---|---|---|---|
| Case | Separaton of Clusters, $S_c$ (cm) [a] | Absorber-Plate Material | Absorber-Plate Thickness (cm) |
| 1 | 7.866 | 304-L Steel | 0.302 |
| 2 | 4.386 | 304-L Steel with 1.1 wt% B | 0.298 |
| 3 | 2.276 | Boral B | 0.216 [b] |
| 4 | 2.566 | Boroflex | 0.226 [c] |
| 5 | 3.446 | Cadmium | 0.061 [d] |
| 6 | 7.376 | Copper | 0.337 |

| Table 5-100 Critical Configurations for LEU-COMP-THERM-042 | | | |
|---|---|---|---|
| **Case** | **Separaton of Clusters, $S_c$ (cm) [a]** | **Absorber-Plate Material** | **Absorber-Plate Thickness (cm)** |
| 7 | 5.016 | Copper-Cadmium | 0.357 |

NOTES: [a] Distance between outer cell boundaries of one cluster and outer cell boundaries of the next.

[b] Also 0.038 cm of Type 1100 aluminum on either side of the 0.216-cm-thick B4C-Al absorber material.

[c] Also 0.16-cm-thick Plexiglas on either side of the 0.226-mm-thick Boroflex.

[d] The 0.061-cm-thick cadmium is mounted on 0.296-cm-thick Plexiglas, with the cadmium closer to the center cluster.

| Table 5-101 Results for LEU-COMP-THERM-042 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **Experiment/ Case** | **Benchmark-Model** | | **Calculated** | | | **$\Delta k_{eff} \pm 2\sigma$** | | |
| | **$k_{eff}$** | **$\sigma$** | **AENCF** | **$k_{eff}$** | **$\sigma$** | | | |
| Lct42-01 | 1.0000 | 0.0016 | 0.169 | 0.9944 | 0.0008 | -0.0056 | ± | 0.0036 |
| Lct42-02 | 1.0000 | 0.0016 | 0.172 | 0.9927 | 0.0008 | -0.0073 | ± | 0.0036 |
| Lct42-03 | 1.0000 | 0.0016 | 0.175 | 0.9933 | 0.0008 | -0.0067 | ± | 0.0036 |
| Lct42-04 | 1.0000 | 0.0017 | 0.174 | 0.9950 | 0.0008 | -0.0050 | ± | 0.0038 |
| Lct42-05 | 1.0000 | 0.0033 | 0.173 | 0.9948 | 0.0008 | -0.0052 | ± | 0.0068 |
| Lct42-06 | 1.0000 | 0.0016 | 0.169 | 0.9947 | 0.0008 | -0.0053 | ± | 0.0035 |
| Lct42-07 | 1.0000 | 0.0018 | 0.171 | 0.9938 | 0.0008 | -0.0062 | ± | 0.0039 |

### 5.4.2 LEU Benchmarks related to Degraded Waste Package Configurations

#### 5.4.2.1 LEU Powder Experiments

##### 5.4.2.1.1 LEU-COMP-THERM-049

The experiments considered in NEA (2001, LEU-COMP-THERM-049) were low-water-moderated uranium dioxide (5wt.% enrichment) powder assemblies, with 'polythene' (polyethylene) reflection. Experiments were carried out using the split-table testing equipment called "MARACAS" in the experimental criticality facility at Valduc, near Dijon, France, in 1983 - 1987. This facility was run by S.R.S.C. (Safety and Criticality Research Service) of I.P.S.N. (Institute for Nuclear Protection Safety).

Experiments were subcritical approaches extrapolated to critical, with the multiplication factor reached being very close to 1.000 (within 0.1%). The subcritical approach parameter was the distance between the two half tables. Uranium dioxide powder was apportioned into boxes each containing 24 kg of dry oxide. The powder was moistened and the boxes were piled on a split table. The parallelepiped assembly was reflected by a 20-cm-thick polythene reflector.

A total oxide mass of 5,535 kg was available and H/U ratios of between 0.0 and 3.0 (9 wt.% water) were studied. The neutron spectrum is thermal but with a strong intermediate energy component. Eighteen accepted benchmark configurations are given in (NEA [2001, LEU-COMP-THERM-049]).

Critical configurations of three different types were assembled: (1) 9 single H/U configurations (Cases 1 to 12) comprised boxes with H/U ratio of either 2.0, 2.5, or 3.0, (2) 9 mixed H/U configurations (Cases 13 to 17) comprised boxes with H/U ratio of 2.0 and 3.0, arranged in concentric layers or as a checkerboard, and (3) 9 module configuration (Case 18) comprised boxes with H/U ratios of 2.0 and 3.0, empty boxes, and boxes with absorber crossing blades (borated stainless steel). The critical separation distances reported in Table 16 of LEU-COMP-THERM-049 are given in Table 5-102 and MCNP results are listed in Table 5-103.

Table 5-102 Critical Separation Distance For LEU-COMP-THERM-049

| Case | H/U | Configuration | Critical Distance "d" (cm) |
|---|---|---|---|
| 1 | 2 | R2 – 2R (6,6) | 2.6 |
| 2 | 2 | R1 – 3R (6,6) | 3.3 |
| 3 | 2 | R2 – 3R (6,6) | 9.83 |
| 4 | 2 | R3 – 3R (6,6) | 16.27 |
| 5 | 2.5 | R2 – 2R (4,5) | 4.23 |
| 6 | 2.5 | R1 – 3R (4,5) | 5.8 |
| 7 | 2.5 | R1 – 2R (5,5) | 1.33 |
| 8 | 2.5 | R2 – 2R (5,5) | 8.92 |
| 9 | 3 | R2 – 2R (4,3) | 3.23 |
| 10 | 3 | R1 – 3R (4,3) | 4.47 |
| 11 | 3 | R1 – 2R (4,4) | 3.52 |
| 12 | 3 | R2 – 2R (4,4) | 10.21 |
| 13 | 2.0 / 3.0 | R2 – 3R (4,4) concentric | 0.73 |
| 14 | 2.0 / 3.0 | R3 – 3R (4,4) concentric | 4.33 |
| 15 | 2.0 / 3.0 | R2 – 4R (4,4) concentric | 5.43 |
| 16 | 2.0 / 3.0 | R1 – 2R (5,5) checkerboard | 1.35 |
| 17 | 2.0 / 3.0 | R2 – 2R (5,5) checkerboard | 8.9 |
| 18 | 2.0 / 3.0 | R1 – 2R (3,3) module | 5.63 |

Table 5-103 Results for LEU-COMP-THERM-049

| Experiment/ Case | Benchmark Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/X |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| lct49-01 | 1.0000 | 0.0034 | 0.294 | 0.9923 | 0.0006 | -0.0077 | ± | 0.0069 | 2 |
| lct49-02 | 1.0000 | 0.0034 | 0.293 | 0.9937 | 0.0006 | -0.0063 | ± | 0.0069 | 2 |
| lct49-03 | 1.0000 | 0.0034 | 0.297 | 0.9929 | 0.0006 | -0.0071 | ± | 0.0069 | 2 |
| lct49-04 | 1.0000 | 0.0034 | 0.300 | 0.9931 | 0.0006 | -0.0069 | ± | 0.0069 | 2 |
| lct49-05 | 1.0000 | 0.0042 | 0.255 | 0.9944 | 0.0007 | -0.0056 | ± | 0.0085 | 2.5 |
| lct49-06 | 1.0000 | 0.0042 | 0.256 | 0.9946 | 0.0007 | -0.0054 | ± | 0.0085 | 2.5 |
| lct49-07 | 1.0000 | 0.0042 | 0.253 | 0.9932 | 0.0007 | -0.0068 | ± | 0.0085 | 2.5 |
| lct49-08 | 1.0000 | 0.0042 | 0.258 | 0.9921 | 0.0007 | -0.0079 | ± | 0.0085 | 2.5 |
| lct49-09 | 1.0000 | 0.0037 | 0.227 | 0.9933 | 0.0006 | -0.0067 | ± | 0.0075 | 3 |
| lct49-10 | 1.0000 | 0.0037 | 0.227 | 0.9946 | 0.0007 | -0.0054 | ± | 0.0075 | 3 |
| lct49-11 | 1.0000 | 0.0037 | 0.227 | 0.9933 | 0.0006 | -0.0067 | ± | 0.0075 | 3 |
| lct49-12 | 1.0000 | 0.0037 | 0.231 | 0.9924 | 0.0007 | -0.0076 | ± | 0.0075 | 3 |
| lct49-13 | 1.0000 | 0.0036 | 0.271 | 0.9935 | 0.0006 | -0.0065 | ± | 0.0073 | [a] |
| lct49-14 | 1.0000 | 0.0036 | 0.272 | 0.9941 | 0.0006 | -0.0059 | ± | 0.0073 | [a] |
| lct49-15 | 1.0000 | 0.0036 | 0.271 | 0.9937 | 0.0006 | -0.0063 | ± | 0.0073 | [a] |
| lct49-16 | 1.0000 | 0.0036 | 0.254 | 0.9938 | 0.0007 | -0.0062 | ± | 0.0073 | [a] |
| lct49-17 | 1.0000 | 0.0036 | 0.258 | 0.9929 | 0.0007 | -0.0071 | ± | 0.0073 | [a] |
| lct49-18 | 1.0000 | 0.0030 | 0.251 | 0.9970 | 0.0006 | -0.0030 | ± | 0.0061 | [a] |

NOTE: [a] Cases 13-18 had mixed H/U configurations where fissile boxes with H/X = 2 or 3 were arranged in concentric patterns (Cases 13-15), checkerboard patterns (Cases 16-17) or in a module configuration (Case 18).

## 5.5 PREVIOUSLY EVALUATED IEU BENCHMARK CASES

### 5.5.1 IEU Benchmarks Related to Intact Waste Package Configurations

None.

### 5.5.2 IEU Benchmarks Related to Degraded Waste Package Configurations

#### 5.5.2.1 IEU Fast Metal Configurations

##### 5.5.2.1.1 Early Jemima Experiments

The purpose of the early Aunt Jemima experiments, performed at the Pajarito critical assembly facility at Los Alamos (1952-1954), was to determine critical conditions for bare uranium cylinders of intermediate enrichment. Vertical cylindrical columns were constructed by stacking thin disks of enriched uranium (oralloy, or Oy, 93.4 wt% $^{235}U$) and natural uranium (tuballoy, or Tu) in different orders. A total of five critical cylindrical configurations of uranium disks, partial disks (in the shape of 45° circular sectors), and layers of rectangular blocks were assembled. The evaluators (NEA [2001, IEU-MET-FAST-001]) considered that four of the experiments (involving only the disks and sectors) qualified as benchmark experiments. The specifications for components cases is presented in reference NEA (2001, IEU-MET-FAST-001). The benchmark $k_{eff} \pm \sigma$ values for the cases are presented in Table 5-104.

Table 5-104 Benchmark $k_{eff} \pm \sigma$ values for IMF-001 Experiments

| Configuration No. | $k_{eff} \pm \sigma$ |
|---|---|
| 1 | 0.9988 ± 0.0012 |
| 2 | 0.9988 ± 0.0012 |
| 3 | 0.9990 ± 0.0010 |
| 4 | 0.9990 ± 0.0010 |

The following results are taken from CRWMS M&O (1999c, page 55). Table 5-105 presents the MCNP calculation results for ENDF/B-V and ENDF/B-VI cross section libraries Table 5-106 shows the difference between experimental $k_{eff}$ results and MCNP calculated results, and Table 5-107 shows the AENCF values. Overall, the experimental benchmark shows that the MCNP code is able to predict the criticality of this experiment with a precision not worse than 0.57% (Table 5-106). This result is independent of the cross section library used. However, $k_{eff}$ obtained using the ENDF/B-VI cross section library under-estimates the experimental $k_{eff}$. The $k_{eff}$ obtained using ENDF/B-V libraries over-estimates the experimental $k_{eff}$.

| Table 5-105 Criticality Calculation Results for IEU-MET-FAST-001 Experiments | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** $k_{eff} \pm \sigma$ | **ENDF/B-VI** $k_{eff} \pm \sigma$ |
| imf1-1 | 1.00224 ± 0.00029 | 0.99714 ± 0.00028 |
| imf1-2 | 1.00225 ± 0.00028 | 0.99666 ± 0.00027 |
| imf1-3 | 1.00401 ± 0.00029 | 0.9986 ± 0.00028 |
| imf1-4 | 1.00477 ± 0.00028 | 0.999 ± 0.00026 |

| Table 5-106 Δk Between Experimental $k_{eff}$ Results and MCNP Calculated $k_{eff}$ Results for the IEU-MET-FAST -001 Experiments | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** $\Delta k_{eff} \pm 2\sigma$ | **ENDF/B-VI** $\Delta k_{eff} \pm 2\sigma$ |
| imf1-1 | -0.00344 ± 0.002469 | 0.00166 ± 0.00246 |
| imf1-2 | -0.00345 ± 0.002464 | 0.00214 ± 0.00246 |
| imf1-3 | -0.00501 ± 0.002082 | 0.0004 ± 0.00208 |
| imf1-4 | -0.00577 ± 0.002077 | 0 ± 0.00207 |

| Table 5-107 AENCF for the IEU-MET-FAST -001 Experiments (MeV) | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| imf1-1 | 1.4394 | 1.3910 |
| imf1-2 | 1.4398 | 1.3919 |
| imf1-3 | 1.3860 | 1.3515 |
| imf1-4 | 1.3859 | 1.3495 |

#### 5.5.2.1.2 IEU-MET-FAST-002

This critical experiment reported in NEA (2001, IEU-MET-FAST-002) was a cylindrical assembly with a core of alternating plates of enriched and natural uranium surrounded by a natural uranium reflector. The core average enrichment was 16 wt% $^{235}U$ (NEA [2001]). This experiment was performed at the Los Alamos Pajarito critical assembly facility and can be considered an extension of the earlier "Jemima" experiments that determined the critical conditions of bare natural and enriched uranium disks of combined intermediate enrichments (29 - 94 wt% $^{235}U$; NEA [2001]). This experiment was judged to be an acceptable criticality-safety benchmark by the evaluators. Detailed descriptions of the experiment are given in Reference NEA (2001). The benchmark $k_{eff}$ value was 1.000 ± 0.003 (NEA [2001, IEU-MET-FAST-002]) for this experiment.

The following results are taken from CRWMS M&O (1999c). Table 5-108 presents the MCNP calculation results for ENDF/B-V, and ENDF/B-VI cross section libraries. Table 5-109 shows the difference between experimental $k_{eff}$ results and MCNP calculated results, and Table 5-110 shows the AENCF values. In this experiment, the only material used for both the core and the reflector was uranium. Overall, the experimental benchmark shows that the MCNP code is able to predict the criticality of this experiment with a precision not worse than 0.6% (Table 5-109). This result is independent of the cross section library used. However, calculations based on the ENDF/B-VI cross section library resulted in a lower $k_{eff}$ than the calculation based on the ENDF/B-V libraries.

Table 5-108 Criticality Calculation Results for the IEU-MET-FAST-002 Experiment

| MCNP case Name | MCNP | |
|---|---|---|
| | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-VI $k_{eff} \pm \sigma$ |
| imf2-1 | 1.00594 ± 0.0006 | 1.00327 ± 0.00056 |

Table 5-109 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for IEU-MET-FAST -002 Experiment

| MCNP case name | ENDF/B-V $\Delta k_{eff} \pm 2\sigma$ | ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|
| imf2-1 | -0.0059 ± 0.0061 | -0.00327 ± 0.0061 |

Table 5-110. AENCF for IEU-MET-FAST -002 Experiment (MeV)

| MCNP case name | ENDF/B-V | ENDF/B-VI |
|---|---|---|
| imf2-1 | 1.2784 | 1.2650 |

### 5.5.2.1.3 Spherical Assembly of $^{235}U$ (36 wt%)

A series of critical experiments with a spherical assembly of $^{235}U$ (36 wt%) was performed over the course of several years (1993-1996) by the Institute for Experimental Physics of the Russian Federal Nuclear Center (VNIIEF) (NEA [2001]). All of these experiments were consider acceptable for use as benchmark critical experiments by the evaluators. The assembly core included different layers of fissile material (lower core limit) and could be covered by different layers of graphite, steel or duralumin (upper core limit). The upper core layers are the reflector layers. Table 5-111 shows the description of each experiment and the benchmark $k_{eff}$ value for the experimental configuration as reported in NEA (2001).

| Table 5-111 Experimental Description for IEU-MET-FAST-003, 004, 005, 006 and 008. | | | |
|---|---|---|---|
| **NEA (2001) case name** | **Description** | **Benchmark $k_{eff} \pm \sigma$** | **MCNP case name** |
| IEU-MET-FAST-003 | Bare Spherical Assembly (detailed) | 1.0000 ± 0.0017 | imf3-1 |
| IEU-MET-FAST-004 | Graphite Reflected Spherical Assembly (detailed) | 1.0000 ± 0.003 | imf4-1 |
| IEU-MET-FAST-005 | Steel Reflected Spherical Assembly (detailed) | 1.0000 ± 0.0021 | imf5-1 |
| IEU-MET-FAST-006 | Duralumin Reflected Spherical Assembly (detailed) | 1.0000 ± 0.0023 | imf6-1 |
| IEU-MET-FAST-008 | Uranium Depleted Reflected Spherical Assembly | 1.0000 ± 0.0018 | imf8 |

The following results are taken from CRWMS M&O (1999c, page 55). Table 5-112 presents the MCNP calculation results for ENDF/B-V and ENDF/B-VI cross section libraries. Table 5-113 shows the difference between experimental $k_{eff}$ results and MCNP calculated results, and Table 5-114 shows the AENCF values. The use of a simplified case results in a slightly different $k_{eff}$ than the one calculated using the detailed case (the difference is less than 0.4%). Overall, the experimental benchmark calculation shows that the MCNP code is able to predict the criticality of this experiment with a precision not worse than 1.3% (Table 5-113). This result is independent of the cross section library used. However, calculations based on the ENDF/B-VI cross section library resulted in a more non-conservative $k_{eff}$ than the calculations based on the ENDF/B-V libraries.

| Table 5-112 Criticality Calculation Results for IEU-MET-FAST-003, 004, 005, 006, and 008 Experiments | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V $k_{eff} \pm \sigma$** | **ENDF/B-VI $k_{eff} \pm \sigma$** |
| Bare sphere | | |
| imf3-1 | 0.99925 ± 0.00076 | 0.99539 ± 0.00076 |
| Graphite reflected sphere | | |
| imf4-1 | 1.01048 ± 0.00084 | 1.00446 ± 0.00081 |
| Steel reflected sphere | | |
| imf5-1 | 1.00881 ± 0.00079 | 1.00053 ± 0.00081 |
| Duralumin reflected sphere | | |
| imf6-1 | 0.99597 ± 0.00081 | 0.99173 ± 0.00083 |
| Uranium depleted reflected sphere | | |
| imf8-1 | 1.0087 ± 0.00084 | 1.0038 ± 0.00081 |

| Table 5-113 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ Results for IEU-MET-FAST-003, 004, 005, 006, and 008 Experiments | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V $\Delta k_{eff} \pm 2\sigma$** | **ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$** |
| Bare sphere | | |
| imf3-1 | -0.00559 ± 0.003741 | 0.00461 ± 0.00372 |
| Graphite reflected sphere | | |
| imf4-1 | -0.00999 ± 0.006253 | -0.00446 ± 0.00621 |
| Steel reflected sphere | | |
| imf5-1 | -0.00951 ± 0.004531 | -0.00053 ± 0.00428 |
| Duralumin reflected sphere | | |
| imf6-1 | 0.00387 ± 0.004911 | 0.00827 ± 0.00467 |
| Uranium depleted reflected sphere | | |
| imf8-1 | -0.0087 ± 0.003973 | -0.0038 ± 0.00369 |

| Table 5-114 AENCF for IEU-MET-FAST-003, 004, 005, 006, and 008 Experiments (MeV) | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| imf3-2 | 1.3535 | 1.3162 |
| imf3-1 | 1.3502 | 1.3242 |
| imf4-2 | 1.3062 | 1.2777 |
| imf4-1 | 1.3071 | 1.2768 |
| imf5-2 | 1.2822 | 1.2583 |
| imf5-1 | 1.2852 | 1.2597 |
| imf6-2 | 1.2905 | 1.2568 |
| imf6-1 | 1.2892 | 1.2610 |
| imf8-1 | 1.3650 | 1.3337 |

### 5.5.2.2 IEU Homogeneous Configurations

#### 5.5.2.2.1 IEU-SOL-THERM-001

The purpose of the experiments reported in NEA (2001, IEU-SOL-THERM-001) and performed at the RRC "Kurchatov Institute" in 1980-1981, was to investigate nuclear safety issues for a special-purpose compact reactor with an aqueous solution of uranyl-sulphate (~20.9 at% $^{235}U$) and graphite reflector. Four configurations of critical assemblies with different concentrations of uranium in the solution were involved in this work. These configurations are described in sufficient detail and are considered acceptable for use as benchmark critical experiments by the

evaluators in NEA (2001, IEU-SOL-THERM-001). Table 5-115 provides a listing of the four cases and the benchmark $k_{eff}$ values in the experimental configurations. The spiral was simulated using two half torus.

Table 5-115 Configurations of IEU-SOL-THERM-001 Experiments

| Handbook case name | Description | Benchmark $k_{eff} \pm \sigma$ (p. 29, IEU-SOL-THERM-001) | H/X ratio | MCNP case name |
|---|---|---|---|---|
| 1 | Core height 49.24 cm<br>Core thickness 0.5cm | 1.000 ± 0.0052 | 444.0300 | ist1-1 |
| 2 | Core height 33.498 cm<br>Core thickness 0.5cm | 1.000 ± 0.0052 | 297.3818 | ist1-2 |
| 3 | Core height 32.251 cm<br>Core thickness 0.3cm | 1.000 ± 0.0052 | 297.3818 | ist1-3 |
| 4 | Core height 29.056 cm<br>Core thickness 0.5cm | 1.000 ± 0.0052 | 217.3670 | ist1-4 |

The following results are taken from CRWMS M&O (1999c, page 54). Table 5-116 presents the MCNP calculation results for ENDF/B-V and ENDF/B-VI cross section libraries for the IEU-SOL-THERM -001 set of critical experiments. Table 5-117 shows the difference between referenced $k_{eff}$ results and MCNP calculated results, and Table 5-118 shows the AENCF values for the IEU-SOL-THERM -001 set of critical experiments. It should be noted that the calculational results for all four cases were considerably lower than the benchmark $k_{eff}$ values. The MCNP calculated $k_{eff}$ values range from about 1.5% to 3.2% below the benchmark $k_{eff}$ values. The evaluator also observed similar behavior of under-predicting $k_{eff}$ (NEA [2001]). The reason for the differences could not be determined with the existing information.

Table 5-116 LCE Calculation Results for IEU-SOL-THERM -001 Experiments

| MCNP case name | MCNP | |
|---|---|---|
| | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-VI $k_{eff} \pm \sigma$ |
| ist1-1 | 0.98074 ± 0.00108 | 0.97915 ± 0.00106 |
| ist1-2 | 0.97596 ± 0.00121 | 0.97348 ± 0.00118 |
| ist1-3 | 0.97222 ± 0.00119 | 0.97079 ± 0.00117 |
| ist1-4 | 0.97105 ± 0.00124 | 0.96757 ± 0.00126 |

| Table 5-117 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for IEU-SOL-THERM -001 Experiments | | |
|---|---|---|
| **MCNP Case name** | **ENDF/B-V $\Delta k_{eff} \pm 2\sigma$** | **ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$** |
| ist1-1 | 0.01926 ± 0.01062 | 0.02085 ± 0.010614 |
| ist1-2 | 0.02404 ± 0.01068 | 0.02652 ± 0.010664 |
| ist1-3 | 0.02778 ± 0.01067 | 0.02921 ± 0.01066 |
| ist1-4 | 0.02895 ± 0.01069 | 0.03243 ± 0.010701 |

| Table 5-118 AENCF for IEU-SOL-THERM -001 Experiments (MeV) | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| ist1-1 | 0.0149 | 0.0146 |
| ist1-2 | 0.0207 | 0.0208 |
| ist1-3 | 0.0205 | 0.0208 |
| ist1-4 | 0.0275 | 0.0271 |

### 5.5.2.2.2 LEU-SOL-THERM-003

A series of critical experiments with aqueous uranyl nitrate solutions with uranium enriched to 10 wt% $^{235}U$ was performed at the Solution Critical Facility of the Institute of Physics and Power Engineering, Obninsk, Russia. These experiments are described in NEA (2001, LEU-SOL-THERM-003). Spheres with outer diameters of 66 cm, 88 cm, and 120 cm were used. Experiments differed from one another in geometry, size, and in uranium concentration in the solution. These experiments are denoted as lst3-1 through lst3-9 in CRWMS M&O (1999c) and characteristics are listed in Table 5-119.

| Table 5-119 LEU-SOL-THERM-003 Characteristics of Benchmark Models. | | | | | | |
|---|---|---|---|---|---|---|
| **Case Name** | **Inner Radius (cm)** | **Shell Thickness (cm)** | **Solution Height (cm)** | **$k_{eff}$** | **σ** | **$H/^{235}U$ ratio[a]** |
| lst3-1 | 32.95 | 0.15 | 16.41 | 0.9997 | 0.0039 | 770 |
| lst3-2 | 32.95 | 0.15 | 25.15 | 0.9993 | 0.0042 | 878 |
| lst3-3 | 32.95 | 0.15 | 32.95 | 0.9995 | 0.0042 | 897 |
| lst3-4 | 43.63 | 0.19 | 3.97 | 0.9995 | 0.0042 | 913 |
| lst3-5 | 43.63 | 0.19 | 28.05 | 0.9997 | 0.0048 | 1173 |
| lst3-6 | 43.63 | 0.19 | 43.63 | 0.9999 | 0.0049 | 1213 |
| lst3-7 | 59.89 | 0.21 | 4.61 | 0.9994 | 0.0049 | 1240 |
| lst3-8 | 59.89 | 0.21 | 41.93 | 0.9993 | 0.0052 | 1412 |
| lst3-9 | 59.89 | 0.21 | 59.89 | 0.9996 | 0.0052 | 1438 |

NOTE: [a] $H/^{235}U$ ratio is from Table 12 of LEU-SOL-THERM-003.

The following results are taken from CRWMS M&O (1999c, page 56). Table 5-120 presents the MCNP calculation results for ENDF/B-V and ENDF/B-VI cross section libraries for the LEU-SOL-THERM -003 set of critical experiments. Table 5-121 shows the differences between experimental $k_{eff}$ results and the MCNP calculated results, and Table 5-122 shows the AENCF values for the LEU-SOL-THERM -003 set of critical experiments. The MCNP continuous energy representations produced $k_{eff}$ values that were within 0.7% of 1.0.

Table 5-120 Criticality Calculations for LEU-SOL-THERM -003 Experiments

| MCNP case name | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-VI $k_{eff} \pm \sigma$ |
|---|---|---|
| lst3-1 | 0.9993 ± 0.0004 | 0.99274 ± 0.00067 |
| lst3-2 | 0.99705 ± 0.00038 | 0.99335 ± 0.00066 |
| lst3-3 | 1.00151 ± 0.00037 | 0.99701 ± 0.00065 |
| lst3-4 | 0.99536 ± 0.00038 | 0.98964 ± 0.00063 |
| lst3-5 | 0.99897 ± 0.00031 | 0.99468 ± 0.00056 |
| lst3-6 | 0.99924 ± 0.0003 | 0.99507 ± 0.00053 |
| lst3-7 | 0.99716 ± 0.0003 | 0.99397 ± 0.00051 |
| lst3-8 | 1.00079 ± 0.00025 | 0.99637 ± 0.00045 |
| lst3-9 | 0.99732 ± 0.00025 | 0.99603 ± 0.00041 |

Table 5-121 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ Results for LEU-SOL-THERM -003 Experiments

| MCNP case name | ENDF/B-V $\Delta k_{eff} \pm 2\sigma$ | ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|
| lst3-1 | 0.0004 ± 0.00784 | 0.00696 ± 0.00791 |
| lst3-2 | 0.00225 ± 0.00843 | 0.00595 ± 0.00850 |
| lst3-3 | -0.00201 ± 0.00843 | 0.00249 ± 0.00850 |
| lst3-4 | 0.00414 ± 0.00843 | 0.00986 ± 0.00849 |
| lst3-5 | 0.00073 ± 0.00962 | 0.00502 ± 0.00967 |
| lst3-6 | 0.00066 ± 0.00982 | 0.00483 ± 0.00986 |
| lst3-7 | 0.00224 ± 0.00982 | 0.00543 ± 0.00985 |
| lst3-8 | -0.00149 ± 0.01041 | 0.00293 ± 0.01044 |
| lst3-9 | 0.00228 ± 0.01041 | 0.00357 ± 0.01043 |

| Table 5-122 AENCF for LEU-SOL-THERM -003 Experiments (MeV) | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| lst3-1 | 0.0186 | 0.0186 |
| lst3-2 | 0.0166 | 0.0165 |
| lst3-3 | 0.0164 | 0.0161 |
| lst3-4 | 0.0162 | 0.0161 |
| lst3-5 | 0.0133 | 0.0128 |
| lst3-6 | 0.0129 | 0.0126 |
| lst3-7 | 0.0127 | 0.0124 |
| lst3-8 | 0.0114 | 0.0113 |
| lst3-9 | 0.0114 | 0.0112 |

### 5.5.2.2.3 IEU-COMP-THERM-001

One-inch cubes of $U(30)F_4$ -polytetrafluoroethylene $[(CF_2)_n]$ ("U-cubes") were stacked with one-inch cubes and half-cubes of polyethylene ("H-cubes") into cuboid shapes on two aluminum platforms, one movable (NEA [2001, IEU-COMP-THERM-001]). Blocks were added until criticality was achieved when the two cuboids were brought together. Most critical cores were reflected by paraffin. Sheets of cadmium or boron surrounded the core in a few cases. Twenty-nine ratios and patterns of "U-cubes" and "H-cubes" were reported in sufficient detail to qualify as acceptable benchmark experiments. Characteristics of these experiments are listed in Table 5-123.

The core was built in two sections on two aluminum platforms, one that was movable. The platforms were each 30 inches square by 5/8 inch thick. The joined platforms were centered about 4.5 inches above the top of the table and were approximately 3 feet from the two closest walls. The core was centered on the joined platforms.

For paraffin-reflected cores, commonly available paraffin slabs were stacked tightly beneath, around, and above the core. The paraffin reflector was approximately seven inches thick. This seven inch paraffin thickness was experimentally determined to be effectively infinite. For some critical configurations, sheets of cadmium or boron were placed between the core and the paraffin reflector. The critical number of cubes was determined by measuring the critical separation of two core sections for arrays that contained greater than the minimum critical.

| Table 5-123 Critical Core Configurations With One-Inch Cubes and Paraffin Reflector | | | | | |
|---|---|---|---|---|---|
| **Case Number** | **H-cubes: U-cubes[a]** | **Dimensions of complete layers (cubes)** | **H-cubes** | **U-cubes** | **Total** |
| 1 | 1:4 | 15x14x14 | 598 | 2392 | 2990 |
| 2 | 1:2 | 12x12x11 | 570 | 1140 | 1710 |

Table 5-123 Critical Core Configurations With One-Inch Cubes and Paraffin Reflector

| Case Number | H-cubes: U-cubes[a] | Dimensions of complete layers (cubes) | H-cubes | U-cubes | Total |
|---|---|---|---|---|---|
| 3 | 1:1 | 10x10x9 | 498 | 498 | 996 |
| 4 | 2:1 | 10x10x8 | 596 | 298 | 894 |
| 5 | 7:1 | 16x14x14 | 2800 | 400 | 3200 |
| 6 | 2:2 | 10x10x10 | 527 | 527 | 1054 |
| 7 | 4:4 | 11x10x10 | 558 | 558 | 1116 |
| 8 | 8:8 | 11x11x10 | 646 | 646 | 1292 |
| 9 | 2:4 | 13x12x12 | 645 | 1290 | 1935 |
| 10 | 1:2 | 11x11x14 | 578 | 1155 | 1733 |
| 11 | 1:2 | 10x10x19 | 648 | 1295 | 1943 |
| 12 | 1:2 | 9x9x39 | 1075 | 2150 | 3225 |
| 13 | 2:1 | 9x9x11 | 604 | 302 | 906 |
| 14 | 2:1 | 8x8x16 | 712 | 356 | 1068 |
| 15 | 2:1 | 8x7x26 | 976 | 488 | 1464 |

NOTE: [a] The 2:2, 4:4, and 8:8 cube ratios indicate numbers of contiguous U-cubes and H-cubes in the pattern. These experiments were studies of the effect of heterogeneity on criticality.

Table 5-124 Critical Configurations for Cores with Half-Cubes of Polyethylene, with Aluminum Spacers, or with Reflector Modifications

| Case Number | H-cubes: U-cubes | Dimensions of complete layers (cubes) | Reflector | H-cubes | U-cubes | Total |
|---|---|---|---|---|---|---|
| 16 | 4:01 | 11x11x10 | Paraffin | 1032 | 258 | 1290 |
| 17 | 1:02 | 16x16x16 | None | 1405 | 2810 | 4215 |
| 18 | 1:01 | 13x14x13 | None | 1232 | 1232 | 2464 |
| 19 | 4:01 | 14x13x12 | None | 1856 | 464 | 2320 |
| 20 | ½:1 | 12x13x11 | Paraffin | 585 | 1170 | 1755 |
| 21 | ½:2 | 15x15x13 | Paraffin | 612 | 2450 | 3062 |
| 22 | 1:02 | 15x15x14 | Cadmium-and-paraffin | 1117 | 2235 | 3352 |
| 23 | 1:01 | 12x13x12 | Cadmium-and-paraffin | 965 | 965 | 1930 |
| 24 | 1:01 | 13x13x13 | boron-and-paraffin | 1118 | 1118 | 2236 |
| 25 | 4:01 | 13x13x11 | Cadmium-and-paraffin | 1600 | 400 | 2000 |
| 26 | 4:1:½ | 12x13x12 | Paraffin | 1364 | 341 | 1875 |
| 27 | 4:01:01 | 14x13x12 | Paraffin | 1512 | 378 | 2268 |
| 28 | 1:2:½ | 14x14x13 | Paraffin | 755 | 1510 | 2642 |
| 29 | 1:02:01 | 16x15x15 | Paraffin | 903 | 1807 | 3614 |

The experimental and benchmark-model $k_{eff}$ are both 1.000 and the uncertainty in the benchmark-model $k_{eff}$ is estimated to be ±0.004.

The critical mass of $^{235}U$ and the $H/^{235}U$ ratio for each experiment is listed in Table 5-125 (CRWMS M&O [1997, page 48 through 51]).

Table 5-125 Critical Parameters for IEU-COMP-THERM-001

| Case Name | Critical Mass of $^{235}U$ (g) | $H/^{235}U$ Ratio |
|---|---|---|
| IECT101 | 36789 | 7.9 |
| IECT102 | 17533 | 15.9 |
| IECT103 | 7659 | 31.8 |
| IECT104 | 4583 | 63.5 |
| IECT105 | 6152 | 222 |
| IECT106 | 8105 | 31.8 |
| IECT107 | 4583 | 31.8 |
| IECT108 | 9936 | 31.8 |
| IECT109 | 19840 | 15.9 |
| IECT110 | 17764 | 15.9 |
| IECT111 | 19917 | 15.9 |
| IECT112 | 33067 | 15.9 |
| IECT113 | 4645 | 63.5 |
| IECT114 | 5475 | 63.5 |
| IECT115 | 7505 | 63.5 |
| IECT116 | 3968 | 63.5 |
| IECT117 | 43218 | 7.9 |
| IECT118 | 18948 | 15.9 |
| IECT119 | 7136 | 63.5 |
| IECT120 | 17995 | 7.9 |
| IECT121 | 37681 | 4.0 |
| IECT122 | 34374 | 7.9 |
| IECT123 | 14842 | 15.9 |
| iect124a | 17195 | 15.9 |
| IECT125 | 6152 | 63.5 |
| IECT126 | 5245 | 63.5 |
| IECT127 | 5814 | 63.5 |

| Table 5-125 Critical Parameters for IEU-COMP-THERM-001 | | |
|---|---|---|
| **Case Name** | **Critical Mass of $^{235}U$ (g)** | **$H/^{235}U$ Ratio** |
| IECT128 | 23224 | 7.9 |
| IECT129 | 27792 | 7.9 |

The following results are taken from CRWMS M&O (1999c, page 54). The $\Delta k_{eff}$ values are calculated as the difference between the experimental value and the calculated value in Table 5-126.

| Table 5-126 Results of LCE Reactivity Calculations for IEU-COMP-THERM-001 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **Case** | **ENDF/B-V** | | | **ENDF/B-VI** | | | **ENDF/B-V** | **ENDF/B-VI** |
| | **$k_{eff}$** | **σ** | **AENCF** | **$k_{eff}$** | **σ** | **AENCF** | **$\Delta k_{eff} \pm 2\sigma$** | **$\Delta k_{eff} \pm 2\sigma$** |
| IECT101 | 0.99735 | 0.00092 | 0.216791 | 1.01062 | 0.00088 | 0.21085 | 0.0068 | -0.0065 |
| IECT102 | 0.99601 | 0.00092 | 0.15817 | 1.00748 | 0.00097 | 0.15427 | 0.0081 | -0.0034 |
| IECT103 | 0.99305 | 0.00104 | 0.10412 | 1.0024 | 0.00103 | 0.10243 | 0.0111 | 0.0017 |
| IECT104 | 0.99735 | 0.00105 | 0.07405 | 1.00316 | 0.00102 | 0.07270 | 0.0068 | 0.0010 |
| IECT105 | 1.00847 | 0.00091 | 0.04552 | 1.00785 | 0.00082 | 0.04520 | -0.0044 | -0.0038 |
| IECT106 | 1.00026 | 0.00103 | 0.10793 | 1.00871 | 0.00099 | 0.10467 | 0.0039 | -0.0046 |
| IECT107 | 0.998 | 0.00101 | 0.11064 | 1.00645 | 0.00099 | 0.10997 | 0.0061 | -0.0023 |
| IECT108 | 0.99604 | 0.00101 | 0.11867 | 1.00358 | 0.00088 | 0.11611 | 0.0081 | 0.0005 |
| IECT109 | 1.00043 | 0.00084 | 0.16790 | 1.01287 | 0.00094 | 0.16326 | 0.0037 | -0.0088 |
| IECT110 | 0.99672 | 0.00099 | 0.15756 | 1.00845 | 0.00097 | 0.15359 | 0.0074 | -0.0043 |
| IECT111 | 0.99579 | 0.00096 | 0.15732 | 1.00818 | 0.00105 | 0.15255 | 0.0083 | -0.0040 |
| IECT112 | 0.99642 | 0.00101 | 0.15568 | 1.00601 | 0.00095 | 0.15347 | 0.0077 | -0.0019 |
| IECT113 | 0.99674 | 0.00103 | 0.07430 | 1.00295 | 0.00096 | 0.07350 | 0.0074 | 0.0012 |
| IECT114 | 0.99787 | 0.00094 | 0.07375 | 1.00358 | 0.00098 | 0.07261 | 0.0062 | 0.0005 |
| IECT115 | 0.99811 | 0.00099 | 0.07400 | 1.00365 | 0.001 | 0.07309 | 0.0060 | 0.0005 |
| IECT116 | 1.0021 | 0.00093 | 0.05547 | 1.00414 | 0.00092 | 0.05435 | 0.0020 | 0.0000 |
| IECT117 | 0.99651 | 0.00102 | 0.20814 | 1.01029 | 0.00102 | 0.20117 | 0.0076 | -0.0062 |
| IECT118 | 0.99756 | 0.00113 | 0.13428 | 1.00986 | 0.00106 | 0.13125 | 0.0066 | -0.0057 |

Table 5-126 Results of LCE Reactivity Calculations for IEU-COMP-THERM-001

| Case | ENDF/B-V | | | ENDF/B-VI | | | ENDF/B-V | ENDF/B-VI |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | AENCF | $\Delta k_{eff} \pm 2\sigma$ | $\Delta k_{eff} \pm 2\sigma$ |
| IECT119 | 1.0045 | 0.00101 | 0.06114 | 1.00479 | 0.00096 | 0.06006 | -0.0004 | -0.0007 |
| IECT120 | 1.0005 | 0.00094 | 0.15539 | 1.01303 | 0.00101 | 0.15118 | 0.0036 | -0.0089 |
| IECT121 | 0.99884 | 0.00087 | 0.21334 | 1.01392 | 0.00088 | 0.20757 | 0.0053 | -0.0098 |
| IECT122 | 0.999 | 0.00109 | 0.19772 | 1.01168 | 0.00102 | 0.19125 | 0.0051 | -0.0076 |
| IECT123 | 0.99516 | 0.00107 | 0.12826 | 1.00429 | 0.00111 | 0.12582 | 0.0090 | -0.0001 |
| IECT124A | 1.00036 | 0.00105 | 0.13305 | 1.0123 | 0.001 | 0.12898 | 0.0038 | -0.0082 |
| IECT125 | 0.99874 | 0.0009 | 0.05992 | 1.00527 | 0.00096 | 0.05891 | 0.0054 | -0.0012 |
| IECT126 | 1.00443 | 0.00095 | 0.05663 | 1.00838 | 0.00089 | 0.05559 | -0.0003 | -0.0043 |
| IECT127 | 1.0032 | 0.00086 | 0.05633 | 1.00637 | 0.00094 | 0.05607 | 0.0009 | -0.0023 |
| IECT128 | 1.00506 | 0.00091 | 0.15824 | 1.01497 | 0.00092 | 0.15360 | -0.0010 | -0.0109 |
| IECT129 | 1.00123 | 0.00099 | 0.15184 | 1.01219 | 0.00098 | 0.14720 | 0.0029 | -0.0081 |

## 5.6 IEU BENCHMARK CASES

### 5.6.1 IEU Benchmarks Related to Intact Waste Package Configurations

#### 5.6.1.1 IEU Rodded Arrays

##### 5.6.1.1.1 IEU-COMP-THERM-002

Critical approach experiments with stainless steel clad $UO_2$ fuel rods (17 wt.% $^{235}U$) in a water-filled tank were performed in 1970 - 1973 in the MATR facility at the Institute of Physics and Power Engineering, Obninsk, Russia (NEA [2001, IEU-COMP-THERM-002]). The fuel rods were arranged in hexagonal lattices with a pitch of 6.8 cm. Each lattice comprised one of three forms of the fuel rod: without absorber element, with gadolinium absorber element, or with cadmium absorber element in the center of each fuel rod. The lattices were fully reflected on all sides with water. The critical mass was defined for "cold" (~ 20°C) and "hot" (~ 200°C) assemblies. The six assemblies were zero-power experiments. Water for the "hot" assemblies was heated by an external electric heater. All six configurations presented in NEA (2001) are considered to be acceptable for use as benchmark critical experiments.

The experiments were performed in a stainless steel tank with outer diameter 1.5 m and inside height 2.2 m. The wall and bottom thickness of the tank was 15 cm. The top of the tank during the room-temperature experiment was open, but it was closed by a 22-cm-thick stainless steel

cover for the high temperature experiments. A stainless steel cylindrical annulus with outer diameter of 88.0 cm, wall thickness of 2.0 cm, and height of 125.0 cm was placed inside the tank and enveloped the assemblies.

The absorber elements, which were inserted in the central tube of the fuel rod, were stainless steel tubes with outer diameter of 1.10 cm and wall thickness 0.03 cm filled by a mixture of $Gd_2O_3$ or CdO with $Al_2O_3$. The total length of the mixture was 48 cm. The top and bottom of the absorber element tubes were closed by stainless steel plugs. Additional aluminum plugs were used to position the absorber element inside the fuel rod.

Three fuel rod lattices were investigated at both room temperature ("cold" assemblies) and at raised temperatures ("hot" assemblies). Critical conditions were achieved with the same loading at two different temperatures for each of three assemblies. MCNP results are shown in Table 5-127.

Table 5-127 Results for IEU-COMP-THERM-002

| Experiment/ Case | Benchmark Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| leuct2_1 | 1.0014 | 0.0039 | 0.044 | 0.9994 | 0.0008 | -0.0021 | ± | 0.0079 |
| leuct2_2 [a] | 1.0019 | 0.0040 | 0.047 | 1.0012 | 0.0008 | -0.0007 | ± | 0.0081 |
| leuct2_3 | 1.0017 | 0.0044 | 0.047 | 1.0026 | 0.0007 | 0.0009 | ± | 0.0089 |
| leuct2_4 [a] | 1.0019 | 0.0044 | 0.049 | 1.0017 | 0.0007 | -0.0002 | ± | 0.0089 |
| leuct2_5 | 1.0014 | 0.0043 | 0.047 | 0.9950 | 0.0007 | -0.0064 | ± | 0.0087 |
| leuct2_6 [a] | 1.0016 | 0.0044 | 0.049 | 0.9941 | 0.0007 | -0.0075 | ± | 0.0089 |

NOTE: [a] Experiments 2, 4, and 6 were run at 500 K, 400 K, and 400 K respectively. The MCNP models were all run at 300 K, except for the scattering treatment.

### 5.6.1.1.2 IEU-COMP-THERM-003

The benchmark experiments described in NEA (2001, IEU-COMP-THERM-003) were performed as a part of startup test of the 250-kW TRIGA Mark II reactor in Ljubljana, Slovenia. The experiments in steady-state operation were performed with completely fresh fuel (including instrumented elements and fueled followers of control rods) in a compact and uniform core (i.e., all elements including the fueled followers of control rods were of the same type with no nonfuel components in the critical core configuration) at well-controlled operating conditions. Standard commercial TRIGA fuel elements of 20 wt.% enrichment and 12 wt.% uranium concentration were used. It should be noted that the core lattice does not have a periodic structure. For this reason the benchmark can not be treated as a unit-cell problem with effective buckling but has to be modeled explicitly in 3D geometry. Commercial TRIGA fuel elements and fueled-follower control rods were used in the experiment.The $H/^{235}U$ ratio is 150 when calculated using the number densities provided in Table 21 of IEU-COMP-THERM-003. The two experiments are

considered to be acceptable as benchmark critical experiments. MCNP results for IEU-COMP-THERM-003 are shown in Table 5-128.

Table 5-128 Results for IEU-COMP-THERM-003

| Experiment /Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| leuct3/1 | 1.0006 | 0.0056 | 0.024 | 0.9975 | 0.0003 | -0.0031 | ± | 0.0112 |
| leuct3/2 | 1.0046 | 0.0056 | 0.024 | 1.0017 | 0.0003 | -0.0029 | ± | 0.0112 |

## 5.6.2 IEU Benchmarks Related to Degraded Waste Package Configurations

### 5.6.2.1 IEU Fast Metal Configurations

#### 5.6.2.1.1 IEU-MET-FAST-007

NEA (2001, IEU-MET-FAST-007) evaluates the initial Big Ten loading used for kinetic studies. This acceptable benchmark experiment represented the early Big Ten assembly in its simplest closed (supercritical) configuration. In this configuration all control and safety rods in the annular reflector were fully inserted, making the reflector nearly solid and homogeneous. This evaluation also specifies three benchmark models for the benchmark experiment. One model is a two-zone model with a homogenized cylindrical core and depleted-uranium reflector with a significant calculated bias (~1%). The other two models are referred to as the detailed and the simplified models. Both accurately specify all zones containing the four types of uranium used in the Big Ten assembly: 93 wt.% $^{235}U$, 10 wt.% $^{235}U$ , natural uranium , and depleted uranium. They differ only in the representation of six reflector control rods and their associated void gaps and liners. MCNP results for IEU-MET-FAST-007 are shown in Table 5-129.

Table 5-129 Results for IEU-MET-FAST-007

| Experiment/ Case | Benchmark Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| leumf7 | 1.0045 | 0.0007 | 1.253 | 1.0125 | 0.0002 | 0.0080 | ± | 0.0015 |

#### 5.6.2.1.2 IEU-MET-FAST-009

Criticality measurements of a polyethylene-reflected assembly of $^{235}U$(36%) were conducted by VNIIEF in 1977 at its criticality test facility (CTF). These experiments were computationally and theoretically reevaluated in 1997 and reported in NEA (2001, IEU-MET-FAST-009). These efforts resulted in an acceptable spherical benchmark model of the critical assembly.

The assembly core had a central cavity of 2.0-cm radius and included five spherical layers of fissile material. The outermost layer had an outer radius of 12.25 cm. The total value of the uncertainty in core radius, averaged over five layers, was 0.05 %. The polyethylene reflector

incorporated two spherical layers with total thickness 5.75 cm. A complete description is given in Table 5-130. The value of the uncertainty in reflector radius was 0.25 %. Each core and reflector layer consisted of two hemispherical pieces. All hemispherical parts of the core had cylindrical pole holes 2.2 cm in diameter. These holes were plugged during measurements with specialized plugs of the core material. MCNP results are shown in Table 5-131.

Table 5-130 Standard Test Assembly: Characteristics of Layers

| Layer No. | Radius, cm | | Layer mass, g |
|---|---|---|---|
| | Inner | outer | |
| $^{235}$U(36%) Core | | | |
| 1 | 2.000 | 6.000 | 16339 |
| 2 | 6.000 | 7.550 | 16791 |
| 3 | 7.550 | 9.150 | 26169 |
| 4 | 9.15 | 11.00 | 44087 |
| 5 | 11.00 | 12.25 | 39542 |
| Polyethylene Reflector | | | |
| 1 | 12.25 | 15.15 | 6040 |
| 2 | 15.15 | 18.00 | 8520 |

Table 5-131 Results for IEU-MET-FAST-009

| Experiment/ Case | Benchmark $k_{eff}$ | σ | AENCF | Calculated $k_{eff}$ | σ | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| ieumf09-01 | 1.0000 | 0.0053 | 1.014 | 1.0081 | 0.0008 | 0.0081 | ± | 0.0107 |

### 5.6.2.1.3 IEU-MET-FAST-010

The U9 Benchmark Assembly reported in NEA (2001, IEU-MET-FAST-010), was part of the Diagnostic Cores Program planned for the Argonne National Laboratory (ANL) ZPR-6 and ZPR-9 critical facilities. The objective of the Diagnostic Cores Program was to resolve longstanding discrepancies between calculated and measured values for small sample central worths, $\beta_{eff}$, and the $^{238}$U capture-to- $^{239}$Pu fission reaction rate ratio. The Diagnostic Cores Program consisted of two phases. Phase I was a definitive investigation of one cause of the central worth discrepancy, namely, the effects of heterogeneities in the vicinity of the samples which are not treated by the calculations. Phase II was planned to find the dependence on composition and spectrum of these three discrepancies. Because the Phase II Program encompassed a wide range of spectra and emphasized strongly the reactor materials of greatest importance to the discrepant parameters, these assemblies provide excellent systems for criticality-safety validation. The first assembly of the Phase II Diagnostic Cores Program was the U/Fe Benchmark Assembly (ZPR-9/34), which has been included in NEA (2001) as benchmark HEU-MET-FAST-035.

This assembly provides a useful benchmark for testing criticality calculations. Two nuclides, $^{235}U$ and $^{238}U$, dominated the neutronic behavior in the core region. The $^{235}U$ / $^{238}U$ ratio is such that the bulk of the neutron spectrum is in the 20-keV to 4-MeV energy range and the peak is at about 300 keV. As would be expected, the neutron balance for this assembly is dominated by $^{235}U$ and $^{238}U$. MCNP results are shown in Table 5-132.

Table 5-132 Results for IEU-MET-FAST-010

| Experiment/ Case | Benchmark $k_{eff}$ | $\sigma$ | AENCF | Calculated $k_{eff}$ | $\sigma$ | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| ieumf10 | 0.9954 | 0.0024 | 1.149 | 0.9982 | 0.0002 | 0.0028 | ± | 0.0048 |

### 5.6.2.2 IEU Solution Experiments

#### 5.6.2.2.1 STACY Experiments

Several sets of experiments involving 10% enriched uranyl nitrate solutions have been performed at the Static Experiment Critical Facility (STACY) at the Nuclear Fuel Cycle Safety Engineering Research Facility in the Tokai Research Establishment of the Japan Atomic Energy Research Institute. They have been incorporated and evaluated as acceptable for use as critical benchmark data in NEA (2001).

LEU-SOL-THERM-004 documents the series of experiments using the water-reflected 60-cm-diameter and 150-cm-high cylindrical tank performed in 1995. Seven sets of critical data were obtained. The uranium concentration of the fuel solution ranged from 225 to 310 gU/liter. On the bottom, side, and top of the core tank was a thick water reflector. The seven critical configurations are considered to be acceptable for use as critical benchmark data.

LEU-SOL-THERM-007 documents the series of experiments using the unreflected 60-cm-diameter and 150-cm-high cylindrical tank performed in 1996. Five sets of critical data were obtained. The uranium concentration of the fuel solution ranged from 242 to 313 gU/liter. The core tank was unreflected. The five critical configurations are considered to be acceptable for use as critical benchmark data.

LEU-SOL-THERM-008 documents the series of experiments employing the 60-cm-diameter cylindrical core tank performed in 1996 with concrete reflectors. The uranium concentration and the free nitric-acid concentration were adjusted to approximately 240 g/l and 2.1 mol/l, respectively. Four concrete reflectors of different thicknesses, packed in annular tube-shaped containers, were prepared and arranged against the outer wall of the core tank. All four experiments are acceptable benchmark experiments.

LEU-SOL-THERM-009 documents the series of experiments employing the 60-cm-diameter cylindrical core tank in 1996 with borated-concrete reflectors. The uranium concentration and

the free nitric-acid concentration were adjusted to approximately 240 g/l and 2.1 mol/l, respectively. Three borated-concrete reflectors of different boron content, packed in annular tube-shaped containers, were prepared and arranged against the outer wall of the core tank. All three experiments are acceptable benchmark experiments.

LEU-SOL-THERM-010 documents the series of experiments employing the 60-cm-diameter cylindrical core tank with polyethylene reflectors in 1996. The uranium concentration and the free nitric-acid concentration were adjusted to approximately 240 g/l and 2.1 mol/l, respectively. Four thicknesses of reflectors, polyethylene blocks packed in annular tube-shaped containers, were prepared and arranged next to the outer wall of the core tank. All four experiments are acceptable benchmark experiments.

LEU-SOL-THERM-016 documents the series of experiments employing the water-reflected 28-cm-thick, 69-cm-wide slab core tank in 1997 and 1998. The uranium concentration was adjusted, in stages, to values in the range of approximately 464 gU/l to 300 gU/l. The free nitric acid concentration ranged from approximately 0.8 mol/l to 1.0 mol/l. All seven critical configurations are accepted as benchmark experiments.

LEU-SOL-THERM-017 documents the series of experiments employing the unreflected 28-cm-thick, 69-cm-wide slab core tank in 1997. The uranium concentration was adjusted, in stages, to values in the range of approximately 464 gU/l to 315 gU/l. The free nitric acid concentration ranged from approximately 0.8 mol/l to 1.0 mol/l. All six critical configurations are accepted as benchmark experiments.

LEU-SOL-THERM-018 documents the series of experiments employing the concrete-reflected 28-cm-thick, 69-cm-wide slab core tank in 1997. The uranium concentration and the free nitric-acid concentration were adjusted to approximately 310-315 g/l and 0.8-1.0 mol/l, respectively. Six thicknesses of reflectors, concrete blocks packed in slab-shaped containers, were prepared and arranged symmetrically on the large side walls of the core tank. All six experiments are acceptable benchmark experiments.

LEU-SOL-THERM-019 documents the series of experiments employing the polyethylene-reflected 28-cm-thick, 69-cm-wide slab core tank in 1997 and 1998. The uranium concentration and the free nitric-acid concentration were adjusted to approximately 320 g/l and 1.0 mol/l, respectively. Six thicknesses of reflectors, polyethylene blocks packed in slab-shaped containers, were prepared and arranged symmetrically on the large side walls of the core tank. All six experiments are acceptable benchmark experiments.

LEU-SOL-THERM-020 documents the series of experiments employing the water-reflected, 80-cm-diameter cylindrical core tank in 1998 and 1999. The uranium concentration was adjusted, in stages, to values in the range of approximately 243 gU/l to 194 gU/l. The free nitric-acid concentration was kept at 1.0 mol/l, approximately. All four experiments are acceptable benchmark experiments.

LEU-SOL-THERM-021 documents the series of experiments employing the unreflected 80-cm-diameter cylindrical core tank in 1998 and 1999. The uranium concentration was adjusted, in stages, to values in the range of approximately 243 gU/l to 194 gU/l. The free nitric-acid concentration was kept at 1.0 mol/l, approximately. All four experiments are acceptable benchmark experiments. MCNP results are shown in Table 5-133.

Table 5-133 Results for Stacy Experiments

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| **LEU-SOL-THERM-004** | | | | | | | | | |
| lst4_1 | 0.9994 | 0.0008 | 0.0188 | 1.0029 | 0.0007 | 0.0035 | ± | 0.0021 | 719.02 |
| lst4_29 | 0.9999 | 0.0009 | 0.0179 | 1.0034 | 0.0006 | 0.0035 | ± | 0.0022 | 771.30 |
| lst4_33 | 0.9999 | 0.0009 | 0.0170 | 1.0013 | 0.0007 | 0.0014 | ± | 0.0023 | 842.18 |
| lst4_34 | 0.9999 | 0.0010 | 0.0157 | 1.0037 | 0.0006 | 0.0038 | ± | 0.0023 | 895.83 |
| lst4_46 | 0.9999 | 0.0010 | 0.0154 | 1.0032 | 0.0006 | 0.0033 | ± | 0.0023 | 941.69 |
| lst4_51 | 0.9994 | 0.0011 | 0.0148 | 1.0023 | 0.0005 | 0.0029 | ± | 0.0024 | 982.52 |
| lst4_54 | 0.9996 | 0.0011 | 0.0142 | 1.0026 | 0.0005 | 0.0030 | ± | 0.0024 | 1017.55 |
| **LEU-SOL-THERM-007** | | | | | | | | | |
| leust7_1 | 0.9961 | 0.0009 | 0.0200 | 0.9966 | 0.0002 | 0.0005 | ± | 0.0018 | 709.2 |
| leust7_2 | 0.9973 | 0.0009 | 0.0187 | 0.9995 | 0.0002 | 0.0022 | ± | 0.0018 | 770.0 |
| leust7_3 | 0.9985 | 0.0010 | 0.0173 | 0.9979 | 0.0002 | -0.0006 | ± | 0.0020 | 842.2 |
| leust7_4 | 0.9988 | 0.0011 | 0.0166 | 1.0005 | 0.0002 | 0.0017 | ± | 0.0022 | 896.0 |
| leust7_5 | 0.9983 | 0.0011 | 0.0159 | 0.9989 | 0.0002 | 0.0006 | ± | 0.0022 | 942.2 |
| **LEU-SOL-THERM-008** | | | | | | | | | |
| lst8_72 | 0.9999 | 0.0014 | 0.0152 | 1.0038 | 0.0002 | 0.0039 | ± | 0.0028 | 954.8 |
| lst8_74 | 1.0002 | 0.0015 | 0.0154 | 1.0023 | 0.0002 | 0.0021 | ± | 0.0030 | 954.8 |
| lst8_76 | 0.9999 | 0.0014 | 0.0153 | 1.0028 | 0.0002 | 0.0029 | ± | 0.0028 | 950.7 |
| lst8_78 | 0.9999 | 0.0014 | 0.0153 | 1.0040 | 0.0002 | 0.0041 | ± | 0.0028 | 956.4 |
| **LEU-SOL-THERM-009** | | | | | | | | | |
| lst9_92 | 0.9998 | 0.0014 | 0.0155 | 1.0018 | 0.0005 | 0.0020 | ± | 0.0030 | 935.8 |
| lst9_93 | 0.9999 | 0.0014 | 0.0157 | 1.0021 | 0.0002 | 0.0022 | ± | 0.0028 | 934.1 |
| lst9_94 | 0.9999 | 0.0014 | 0.0158 | 1.0022 | 0.0002 | 0.0023 | ± | 0.0028 | 933.5 |
| **LEU-SOL-THERM-010** | | | | | | | | | |
| lst10_83 | 0.9999 | 0.0015 | 0.0153 | 1.0023 | 0.0003 | 0.0024 | ± | 0.0031 | 946.2 |
| lst10_85 | 0.9999 | 0.0014 | 0.0154 | 1.0019 | 0.0003 | 0.0020 | ± | 0.0029 | 944.8 |
| lst10_86 | 1.0000 | 0.0014 | 0.0153 | 1.0032 | 0.0003 | 0.0032 | ± | 0.0029 | 943.6 |
| lst10_88 | 1.0001 | 0.0014 | 0.0154 | 1.0026 | 0.0003 | 0.0025 | ± | 0.0029 | 941.7 |

Table 5-133 Results for Stacy Experiments

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U^a$ |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| **LEU-SOL-THERM-016** | | | | | | | | | |
| lst16_05 | 0.9996 | 0.0013 | 0.0267 | 1.0093 | 0.0007 | 0.0097 | ± | 0.0030 | 468.7 |
| lst16_13 | 0.9999 | 0.0013 | 0.0248 | 1.0080 | 0.0006 | 0.0081 | ± | 0.0029 | 514.2 |
| lst16_25 | 0.9994 | 0.0014 | 0.0216 | 1.0075 | 0.0006 | 0.0081 | ± | 0.0030 | 608.4 |
| lst16_29 | 0.9996 | 0.0014 | 0.0209 | 1.0068 | 0.0006 | 0.0072 | ± | 0.0030 | 650.2 |
| lst16_31 | 0.9995 | 0.0014 | 0.0195 | 1.0059 | 0.0005 | 0.0064 | ± | 0.0030 | 699.1 |
| lst16_40 | 0.9992 | 0.0015 | 0.0186 | 1.0043 | 0.0005 | 0.0051 | ± | 0.0032 | 738.9 |
| lst16_96 | 0.9994 | 0.0015 | 0.0180 | 1.0047 | 0.0005 | 0.0053 | ± | 0.0032 | 771.8 |
| **LEU-SOL-THERM-017** | | | | | | | | | |
| lst17_04 | 0.9981 | 0.0013 | 0.0275 | 1.0051 | 0.0007 | 0.0070 | ± | 0.0030 | 468.7 |
| lst17_22 | 0.9986 | 0.0013 | 0.0258 | 1.0049 | 0.0006 | 0.0063 | ± | 0.0029 | 510.8 |
| lst17_23 | 0.9989 | 0.0014 | 0.0224 | 1.0052 | 0.0006 | 0.0063 | ± | 0.0030 | 610.9 |
| lst17_26 | 0.9992 | 0.0014 | 0.0212 | 1.0043 | 0.0006 | 0.0051 | ± | 0.0030 | 650.1 |
| lst17_30 | 0.9987 | 0.0015 | 0.0200 | 1.0043 | 0.0005 | 0.0056 | ± | 0.0032 | 699.2 |
| lst17_47 | 0.9996 | 0.0015 | 0.0192 | 1.0042 | 0.0006 | 0.0046 | ± | 0.0032 | 729.0 |
| **LEU-SOL-THERM-018** | | | | | | | | | |
| RUN133 | 0.9992 | 0.0010 | 0.0183 | 1.0033 | 0.0002 | 0.0041 | ± | 0.0020 | 751.4 |
| RUN142 | 0.9996 | 0.0010 | 0.0187 | 1.0042 | 0.0003 | 0.0046 | ± | 0.0021 | 736.8 |
| RUN143 | 0.9996 | 0.0010 | 0.0188 | 1.0045 | 0.0003 | 0.0049 | ± | 0.0021 | 735.7 |
| RUN144 | 0.9997 | 0.0010 | 0.0187 | 1.0033 | 0.0003 | 0.0036 | ± | 0.0021 | 734.3 |
| RUN145 | 0.9992 | 0.0010 | 0.0187 | 1.0038 | 0.0003 | 0.0046 | ± | 0.0021 | 732.9 |
| RUN146 | 0.9996 | 0.0010 | 0.0186 | 1.0037 | 0.0003 | 0.0041 | ± | 0.0021 | 730.8 |
| **LEU-SOL-THERM-019** | | | | | | | | | |
| RUN149 | 0.9997 | 0.0009 | 0.0190 | 1.0043 | 0.0003 | 0.0046 | ± | 0.0019 | 725.4 |
| RUN150 | 0.9995 | 0.0009 | 0.0190 | 1.0043 | 0.0003 | 0.0048 | ± | 0.0019 | 727.8 |
| RUN151 | 0.9999 | 0.0009 | 0.0191 | 1.0049 | 0.0002 | 0.0050 | ± | 0.0018 | 726.5 |
| RUN152 | 0.9996 | 0.0009 | 0.0191 | 1.0054 | 0.0003 | 0.0057 | ± | 0.0019 | 724.5 |
| RUN153 | 0.9998 | 0.0009 | 0.0191 | 1.0050 | 0.0003 | 0.0052 | ± | 0.0019 | 723.1 |
| RUN183 | 0.9994 | 0.0009 | 0.0189 | 1.0036 | 0.0003 | 0.0042 | ± | 0.0019 | 721.3 |
| **LEU-SOL-THERM-020** | | | | | | | | | |
| LST20C1 | 0.9995 | 0.0010 | 0.0150 | 1.0014 | 0.0003 | 0.0019 | ± | 0.0021 | 971.0 |
| LST20C2 | 0.9996 | 0.0010 | 0.0143 | 1.0000 | 0.0003 | 0.0004 | ± | 0.0021 | 1053.9 |
| LST20C3 | 0.9997 | 0.0012 | 0.0131 | 0.9993 | 0.0003 | -0.0004 | ± | 0.0025 | 1168.0 |

| Table 5-133 Results for Stacy Experiments | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{235}U$[a] |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| LST20C4 | 0.9998 | 0.0012 | 0.0125 | 1.0004 | 0.0003 | 0.0006 | ± | 0.0025 | 1239.3 |
| **LEU-SOL-THERM-021** | | | | | | | | | |
| LST21C1 | 0.9983 | 0.0009 | 0.0154 | 0.9991 | 0.0003 | 0.0008 | ± | 0.0019 | 971.0 |
| LST21C2 | 0.9985 | 0.0010 | 0.0144 | 0.9996 | 0.0003 | 0.0011 | ± | 0.0021 | 1052.7 |
| LST21C3 | 0.9989 | 0.0011 | 0.0135 | 0.9976 | 0.0003 | -0.0013 | ± | 0.0023 | 1168.0 |
| LST21C4 | 0.9993 | 0.0012 | 0.0127 | 0.9999 | 0.0003 | 0.0006 | ± | 0.0025 | 1238.9 |

NOTE: [a] The $H/^{235}U$ ratio is calculated from the number densities reported in section 3 of the respective NEA (2001), LEU-SOL-THERM chapter.

### 5.6.2.2.2 LEU-SOL-THERM-005

A large number of critical experiments with absorber elements of different types in uranium nitrate solution of different enrichments and concentrations were performed in 1961 - 1963 at the Solution Physical Facility of the Institute of Physics and Power Engineering (IPPE), Obninsk, Russia. The purpose of these experiments was to determine the effects of enrichment, concentration, geometry, neutron reflection, and type, diameter, number, and arrangement of absorber rods on the critical mass of light-water-moderated homogeneous uranyl nitrate solutions. The experiments included ones with a central boron carbide or cadmium rod, clusters of boron carbide rods, and triangular lattices of boron carbide rods in cylindrical tanks of different dimensions filled with solutions of uranyl nitrate. The three experiments included in this evaluation (NEA [2001]) were performed with uranium enriched to 5.64 wt.% $^{235}U$. Uranium nitrate solution with uranium concentration of 400.2 g/l was pumped into the core or inner tank, a stainless steel cylindrical tank with inner diameter 110 cm. One experiment as performed without absorber rods, another one with a central rod, and another one with a cluster of seven absorber rods arranged at the corners and center of a hexagon with a pitch of 31.8 cm, inserted in the center of the core tank. There was a thick side and bottom water reflector in these experiments. The $H/^{235}U$ ratio for these experiments is 972.2 (calculated from solution data in LEU-SOL-THERM-005). All three configurations are considered to be acceptable for use as criticality safety benchmark experiments (NEA [2001]). MCNP results are shown in Table 5-134.

| Table 5-134 Results for LEU-SOL-THERM-005 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **LEU-SOL-THERM-005** | | | | | | | | |
| Experiment /Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| CASE_1 | 1.0000 | 0.0042 | 0.0255 | 0.9990 | 0.0005 | -0.0010 | ± | 0.0085 |
| CASE_2 | 1.0000 | 0.0051 | 0.0254 | 0.9985 | 0.0005 | -0.0015 | ± | 0.0102 |
| CASE_3 | 1.0000 | 0.0064 | 0.0260 | 0.9983 | 0.0005 | -0.0017 | ± | 0.0128 |

#### 5.6.2.2.3 LEU-SOL-THERM-006

Description for this set is same as LEU-SOL-THERM-005 except for enrichment which is 10.07 wt.%. The H/$^{235}$U ratio for these experiments is 531.5 (calculated from solution data in LEU-SOL-THERM-005). MCNP results are shown in Table 5-135.

Table 5-135 Results for LEU-SOL-THERM-006

| LEU-SOL-THERM-006 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment /Case | Benchmark-Model $k_{eff}$ | σ | AENCF | Calculated $k_{eff}$ | σ | $\Delta k_{eff} \pm 2\sigma$ | | |
| leust6_1 | 1.0000 | 0.0037 | 0.0245 | 0.9996 | 0.0007 | -0.0004 | ± | 0.0075 |
| leust6_2 | 1.0000 | 0.0038 | 0.0251 | 1.0075 | 0.0007 | 0.0075 | ± | 0.0077 |
| leust6_3 | 1.0000 | 0.0041 | 0.0253 | 1.0014 | 0.0007 | 0.0014 | ± | 0.0083 |
| leust6_4 | 1.0000 | 0.0041 | 0.0254 | 1.0020 | 0.0007 | 0.0020 | ± | 0.0083 |
| leust6_5 | 1.0000 | 0.0047 | 0.0257 | 1.0039 | 0.0007 | 0.0039 | ± | 0.0095 |

## 5.7 PREVIOUSLY EVALUATED U-233 BENCHMARK CASES

### 5.7.1 U-233 Benchmarks related to Intact Waste Package Configurations

#### 5.7.1.1 U-233 Metal Spheres

The following is a listing of $^{233}$U experiments previously evaluated in CRWMS M&O (1999b) along with a brief description as reported in NEA (2001).

Table 5-136 lists the characteristics of the U233-MET-FAST experiments and MCNP results from CRWMS M&O (1999b) are shown in Table 5-137.

U233-MET-FAST-001

In 1961, the $^{233}$U Jezebel critical assembly was fabricated and operated at Los Alamos Scientific Laboratory. This experiment was an unreflected, or bare, $^{233}$U metal critical assembly, nearly spherical in shape. The uranium was 98 at.% $^{233}$U and was unalloyed. The $^{233}$U Jezebel is considered to be acceptable for use as a benchmark critical experiment (NEA [2001]).

U233-MET-FAST-002

Two critical experiments were performed at Los Alamos Scientific Laboratory using spherical masses of uranium highly enriched in $^{233}$U surrounded by uranium highly enriched in $^{235}$U. The first experiment was performed using a core mass of 10.012 kg and

will be referred to as case u2332a. The second experiment was performed using a core mass of 7.601 kg and will be referred to as the case u2332b. These experiments are considered to be acceptable as benchmark critical experiments (NEA [2001]).

U233-MET-FAST-003

Two critical experiments were performed using spherical masses of U reflected by natural uranium at Los Alamos Scientific Laboratory. The first experiment was performed using a core mass of 10.012 kg and will be referred to as case u2333a. The second experiment was performed using a core mass of 7.601 kg and will be referred to as case u2333b. These experiments are considered to be acceptable as benchmark critical experiments (NEA [2001]).

U233-MET-FAST-004

Two critical experiments were performed at Los Alamos Scientific Laboratory using spherical masses of uranium highly enriched in U reflected by tungsten. The first experiment was 233 performed using a core mass of 10.012 kg and will be referred to as case u2334a1. The second experiment was performed using a core mass of 7.601 kg and will be referred to as case u2334b1. These experiments are considered to be acceptable as benchmark critical experiments (NEA [2001]).

U233-MET-FAST-005

Two critical were performed using spherical masses of $^{233}$U reflected by beryllium at Los Alamos Scientific Laboratory. The first experiment was performed using a core mass of 10.012 kg and will be referred to as case u2335a. The second experiment was performed using a core mass of 7.601 and will be referred to as u2335b. These experiments are considered to be acceptable as benchmark critical experiments (NEA [2001]).

U233-MET-FAST-006

In 1964 a critical experiment was performed at Los Alamos Scientific Laboratory using a spherical $^{233}$U core reflected by natural uranium. The average of two critical experimental configurations was used to determine the final critical mass of a $^{233}$U metal sphere intimately reflected by thick natural uranium. Delayed critical was achieved with both configurations. The results of this experiment are considered to be acceptable as a benchmark critical experiment (NEA [2001]).

Table 5-136 Characteristics of U233-MET-FAST Experiments

| Case | Density (g U/cm$^3$) | Critical Mass (kg) | Diameter (cm) | $^{233}$U (wt%) | $^{235}$U (wt%) | Reflector |
|---|---|---|---|---|---|---|
| u2331a | 18.424 | 16.535 | 11.9676 | 98.13 | 0.03 | Bare |
| u2332a | 18.621 | 10.012 | 10.0888 | 98.20 | 0.00 | U(93.2 wt% $^{235}$U); 18.8 g/cm$^3$; 1.2217 cm |
| u2332b | 18.644 | 7.601 | 9.1998 | 98.20 | 0.00 | U(93.2 wt% $^{235}$U); 18.8 g/cm$^3$; 1.9888 cm |
| u2333a | 18.621 | 10.012 | 10.0888 | 98.20 | 0.00 | U(0.72 wt% $^{235}$U); 18.92 g/cm$^3$; 2.3012 cm |
| u2333b | 18.644 | 7.601 | 9.1998 | 98.20 | 0.00 | U(0.72 wt% $^{235}$U); 18.92 g/cm$^3$; 5.3086 cm |
| u2334a1 | 18.621 | 10.012 | 10.0888 | 98.20 | 0.00 | W alloy; 17.21 g/cm$^3$; 2.4384 cm |
| u2334b1 | 18.644 | 7.601 | 9.1998 | 98.20 | 0.00 | W alloy; 17.21 g/cm$^3$; 5.7912 cm |
| u2335a | 18.621 | 10.012 | 10.0888 | 98.20 | 0.00 | BeO; 1.83 g/cm$^3$; 2.0447 cm |
| u2335b | 18.644 | 7.601 | 9.1998 | 98.20 | 0.00 | BeO; 1.83 g/cm$^3$; 4.1961 cm |
| u2336a | 18.420 | 5.740 | 8.4116 | 98.13 | 0.03 | U(0.72 wt% $^{235}$U); 19.00 g/cm$^3$; 19.9136 cm |

Table 5-137 Results of Criticality Calculations for U233-MET-FAST

| Case | ENDF/B-V | | | ENDF/B-VI | | |
|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF |
| U2331a | 0.99297 | 0.00038 | 1.77385 | 0.99354 | 0.00038 | 1.77170 |
| U2332a | 0.99547 | 0.00038 | 1.73702 | 0.99609 | 0.0004 | 1.72533 |
| U2332b | 0.99807 | 0.00039 | 1.70789 | 0.99764 | 0.00039 | 1.68809 |
| U2333a | 0.99583 | 0.00041 | 1.74832 | 0.99645 | 0.0004 | 1.74824 |
| U2333b | 0.99771 | 0.00041 | 1.76231 | 0.99738 | 0.00042 | 1.76086 |
| U2334a1 | 1.00356 | 0.00041 | 1.61336 | 1.00278 | 0.00043 | 1.61505 |
| U2334b1 | 1.00637 | 0.00043 | 1.51775 | 1.00564 | 0.00044 | 1.52050 |

| Table 5-137 Results of Criticality Calculations for U233-MET-FAST | | | | | | |
|---|---|---|---|---|---|---|
| Case | ENDF/B-V | | | ENDF/B-VI | | |
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF |
| U2335a | 0.99351 | 0.00043 | 1.61950 | 0.99452 | 0.00042 | 1.61670 |
| U2335b | 0.99681 | 0.00045 | 1.51871 | 0.99707 | 0.00045 | 1.51252 |
| U2336a | 1.00057 | 0.00045 | 1.77403 | 1.00122 | 0.00047 | 1.77535 |

### 5.7.2 U-233 Benchmarks related to degraded Waste Package Configurations

None.

## 5.8 U-233 BENCHMARK CASES

### 5.8.1 U-233 Benchmarks related to Intact Waste Package Configurations

None.

### 5.8.2 U-233 Benchmarks related to degraded Waste Package Configurations

#### 5.8.2.1 U-233 Solution in Spheres

##### 5.8.2.1.1 U-233-SOL-THERM-001

In the 1950s a series of measurements was performed at the Oak Ridge National Laboratory to investigate the critical conditions of unreflected homogeneous aqueous uranyl-nitrate solutions, and to experimentally determine the thermal value of η (the number of fission neutrons produced per thermal neutron absorbed in the fissile isotope) for the $^{235}U$ and $^{233}U$ isotopes. The measurements were made on both bare spherical and cylindrical configurations. This section discusses the spherical configuration only. The sphere was constructed by welding two 0.32-cm-thick hemispherical shells of type 1100 aluminum. The solutions in these spheres differed primarily by the amount of boron poison present in the solution and the corresponding critical concentration of uranium. The H/ $^{233}U$ ratio in these solutions range from 1324 to 1533. In the $^{233}U$ experiments the chambers were surrounded with lead to reduce the effects of gamma rays arising from decay products of $^{232}U$, a contaminant of $^{233}U$. All five experiments were judged to be acceptable for use as criticality safety benchmark experiments (NEA [2001]). Table 5-138 provides the summary of material data for critical conditions of 27.24-inch-diameter Spheres and Table 5-139 shows the MCNP results.

Table 5-138 Summary of Material Data for Critical Conditions of 27.24-inch-diameter Spheres

| Exp. No. | Isotopic Composition (wt.%) | | | | Solution Density (g/ml) | Total Uranium (mg/g) | Total Boron (mg/g) | Total Nitrate (mg/g) | Total Thorium (mg/g) |
|---|---|---|---|---|---|---|---|---|---|
| | $^{233}U$ | $^{234}U$ | $^{235}U$ | $^{238}U$ | | | | | |
| 1 | 97.7 | 1.62 | 0.04 | 0.64 | 1.0226 | 16.76 | 0 | 11.9 | 0.074 |
| 2 | 97.7 | 1.62 | 0.04 | 0.64 | 1.0253 | 17.42 | 0.0233 | 12.3 | 0.077 |
| 3 | 97.7 | 1.62 | 0.04 | 0.64 | 1.0274 | 18.03 | 0.0453 | 12.8 | 0.08 |
| 4 | 97.7 | 1.62 | 0.04 | 0.64 | 1.0275 | 18.67 | 0.067 | 13.2 | 0.083 |
| 5 | 97.7 | 1.62 | 0.04 | 0.64 | 1.0286 | 19.27 | 0.0887 | 13.6 | 0.085 |

Table 5-139 Results for U233-SOL-THERM-001

| Experiment /Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{233}U$[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| ust001-1 | 1.0000 | 0.0031 | 0.0038 | 1.0018 | 0.0005 | 0.0018 | ± | 0.0063 | 1531.53 |
| ust001-2 | 1.0005 | 0.0033 | 0.0041 | 1.0004 | 0.0006 | -0.0001 | ± | 0.0067 | 1471.67 |
| ust001-3 | 1.0006 | 0.0033 | 0.0043 | 0.9994 | 0.0006 | -0.0012 | ± | 0.0067 | 1420.05 |
| ust001-4 | 0.9998 | 0.0033 | 0.0043 | 0.9989 | 0.0006 | -0.0009 | ± | 0.0067 | 1369.67 |
| ust001-5 | 0.9999 | 0.0033 | 0.0043 | 0.9987 | 0.0006 | -0.0012 | ± | 0.0067 | 1325.44 |

NOTE: [a] $H/^{233}U$ is calculated using number densities given in Table 4 of U233-SOL-THERM-001.

### 5.8.2.1.2 U233-SOL-THERM-002

A number of critical experiments involving $^{233}U$ were performed at ORNL during the year 1952 in order to determine the minimum critical mass, minmum critical volume, and maximum safe pipe size of water moderated solutions of $^{233}U$. Nineteen of these utilizing uranyl nitrate ($UO_2(NO_3)_2$) solutions in 8-, 8.5-, 9-, 10-, and 12-inch-diameter cylinders are evaluated in NEA (2001). The cylinders were reflected with six inches of paraffin in the radial direction and on the top and bottom. Two experiments performed in the eight-inch-diameter cylinder were large extrapolations to critical and are not considered acceptable as benchmark experiments. Therefore, 17 experiments were judged acceptable for use as criticality-safety benchmark experiments, although the evaluated uncertainty is somewhat large (NEA [2001]).

Each different-diameter vessel was fabricated as a separate assembly and was approximately equilateral (a cylinder where the height equals the diameter). The benchmark model specifications are given in Table 5-140 and Table 5-141. The vessel wall had a thickness of 0.1291 cm for the sides, top, and bottom. MCNP results are shown in Table 5-142.

Table 5-140 Simplified Model Dimensions

| Geometry Region | 8 inch | 8.5 inch | 9 inch | 10 inch | 12 inch |
|---|---|---|---|---|---|
| vessel inside radius (cm) | 10.2645 | 10.7641 | 11.4351 | 12.7287 | 15.2571 |
| vessel inside straight height (cm) | 20.529 | 21.5282 | 22.8702 | 25.4574 | 30.5142 |
| volume equivalent cylinder height of conical bottom (cm) | 0.1195 | 0.1253 | 0.1331 | 0.1482 | 0.1776 |

Table 5-141 Benchmark-Model Solution Densities and Heights

| Experiment # | Solution Density (g/cc) | Solution Height (cm) |
|---|---|---|
| 8-inch-diameter | | |
| 4 | 1.2866 | 16.5195 |
| 5 | 1.2322 | 16.8195 |
| 8 | 1.1854 | 18.9195 |
| 22 | 1.3935 | 14.8195 |
| 24 | 1.5432 | 14.5195 |
| 34 | 1.2379 | 16.8195 |
| 35 | 1.1650 | 19.4195 |
| 8.5-inch-diameter | | |
| 10 | 1.1451 | 19.5253 |
| 11 | 1.1207 | 21.6253 |
| 9-inch-diameter | | |
| 12 | 1.1009 | 21.4331 |
| 36 | 1.0932 | 23.0331 |
| 10-inch-diameter | | |
| 14 | 1.0899 | 19.4482 |
| 15 | 1.0776 | 22.6482 |
| 38 | 1.0686 | 25.3482 |
| 12-inch-diameter | | |
| 17 | 1.0608 | 21.2776 |
| 18 | 1.0556 | 23.9776 |
| 19 | 1.0458 | 30.5776 |

Table 5-142 Results for U233-SOL-THERM-002

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{233}U$[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| Ust02-04 | 1.0040 | 0.0087 | 0.0260 | 1.0103 | 0.0011 | 0.0063 | ± | 0.0175 | 119.0 |
| Ust02-05 | 1.0040 | 0.0087 | 0.0214 | 0.9973 | 0.0011 | -0.0068 | ± | 0.0175 | 149.2 |
| Ust02-08 | 1.0040 | 0.0087 | 0.0173 | 1.0113 | 0.0010 | 0.0073 | ± | 0.0175 | 192.2 |
| Ust02-10 | 1.0040 | 0.0087 | 0.0138 | 1.0096 | 0.0011 | 0.0056 | ± | 0.0175 | 246.0 |
| Ust02-11 | 1.0040 | 0.0087 | 0.0115 | 1.0126 | 0.0010 | 0.0086 | ± | 0.0175 | 295.7 |
| Ust02-12 | 1.0040 | 0.0087 | 0.0100 | 1.0006 | 0.0010 | -0.0034 | ± | 0.0175 | 354.7 |
| Ust02-14 | 1.0040 | 0.0087 | 0.0098 | 0.9875 | 0.0009 | -0.0165 | ± | 0.0175 | 393.6 |
| Ust02-15 | 1.0040 | 0.0087 | 0.0083 | 1.0026 | 0.0010 | -0.0014 | ± | 0.0175 | 458.9 |
| Ust02-17 | 1.0040 | 0.0087 | 0.0072 | 0.9897 | 0.0009 | -0.0143 | ± | 0.0175 | 579.5 |
| Ust02-18 | 1.0040 | 0.0087 | 0.0066 | 1.0029 | 0.0008 | -0.0011 | ± | 0.0175 | 628.3 |
| Ust02-19 | 1.0040 | 0.0087 | 0.0056 | 1.0102 | 0.0008 | 0.0062 | ± | 0.0175 | 752.6 |
| Ust02-22 | 1.0040 | 0.0087 | 0.0356 | 0.9967 | 0.0011 | -0.0073 | ± | 0.0175 | 83.9 |
| Ust02-24 | 1.0040 | 0.0087 | 0.0490 | 0.9976 | 0.0012 | -0.0064 | ± | 0.0176 | 57.1 |
| Ust02-34 | 1.0040 | 0.0087 | 0.0223 | 1.0038 | 0.0011 | -0.0002 | ± | 0.0175 | 144.2 |
| Ust02-35 | 1.0040 | 0.0087 | 0.0155 | 1.0103 | 0.0009 | 0.0063 | ± | 0.0175 | 212.1 |
| Ust02-36 | 1.0040 | 0.0087 | 0.0096 | 1.0115 | 0.0009 | 0.0075 | ± | 0.0175 | 377.9 |
| Ust02-38 | 1.0040 | 0.0087 | 0.0075 | 1.0097 | 0.0008 | 0.0056 | ± | 0.0175 | 512.6 |

NOTE: [a] $H/^{233}U$ is calculated using number densities given on page 25 of U233-SOL-THERM-002.

### 5.8.2.1.3 U233-SOL-THERM-003

A number of critical experiments involving $^{233}U$ were performed at ORNL during the years 1952 and 1953 in order to determine the minimum critical mass, minmum critical volume, and maximum safe pipe size of water moderated solutions of $^{233}U$. Eighteen of these utilizing uranyl fluoride ($UO_2F_2$) solutions in 5-, 5.4-, 6-, 6.6-, 7.5-, 8-, 8.5-, 9-, and 12-inch-diameter cylinders are evaluated in NEA (2001). The 5-, 5.4-, and 6-inch-diameter cylinders were 36 inches tall, and were reflected by at least six inches of paraffin in the radial direction and on the bottom. The larger cylinders were approximately equilateral (H/D=1), and were reflected by six inches of paraffin in the radial direction and on the top and bottom. Eight experiments were large extrapolations to critical and are not considered acceptable as benchmark experiments. Therefore, 10 experiments were judged acceptable for use as criticality-safety benchmark experiments, although the evaluated uncertainties are somewhat large (NEA [2001]).

The vessel wall had a thickness of 0.1291 cm for the sides, top, and bottom. All inside surfaces of the 5-, 6-, 7.5-, 8-, 8.5-, 9-, and 12-inch-diameter cylinders were coated with Heresite. The inside surfaces of the 5.4- and 6.6-inch-diameter cylinders were coated with Unichrome. The benchmark model specifications are given in Table 5-143. MCNP results are shown in Table 5-144.

Table 5-143 Solution Densities and Simplified Model Dimensions

| Experiment | Solution Density ($g/cm^3$) | Solution Radius (cm) | Outer Radius of Unichrome (cm) | Outer Radius of Aluminum (cm) | Outer Radius of Paraffin (cm) | Adjusted Solution Height (cm) | Vessel Inside Height (cm) |
|---|---|---|---|---|---|---|---|
| 40 | 1.388 | 6.3230 | N/A | 6.4521 | 21.6921 [a] | 56.3528 | 91.2928 |
| 41 | 1.388 | 6.8265 | 7.0045 | 7.1336 | 22.3736 [a] | 48.5411 | 91.2811 |
| 42 | 1.388 | 7.5589 | N/A | 7.6880 | 22.9280 [a] | 23.8240 | 91.2640 |
| 45 | 1.604 | 8.3302 | 8.5082 | 8.6373 | 23.8773 | 16.7061 | 16.7061 |
| 55 | 1.707 | 8.3302 | 8.5082 | 8.6373 | 23.8773 | 16.5061 | 16.7061 |
| 57 | 1.198 | 9.5246 | N/A | 9.6537 | 24.8937 | 18.1783 | 18.8283 |
| 58 | 1.121 | 10.2645 | N/A | 10.3936 | 25.6336 | 19.9610 | 20.2900 |
| 61 [b] | 1.090 | 10.7641 | N/A | 10.8932 | 26.1332 | 21.9494 | 21.9494 [b] |
| 62 [b] | 1.075 | 11.4351 | N/A | 11.5642 | 26.8042 | 22.8338 | 22.8338 [b] |
| 65 | 1.035 | 15.2571 | N/A | 15.3862 | 30.6262 | 30.1448 | 30.1590 |

NOTE: [a] No paraffin above top surface of lid.

[b] Experiment extrapolated the critical height past the actual height of the vessel. The height of the vessel in the simplified model has been increased to accommodate the extrapolated critical height.

Table 5-144 Results for U233-SOL-THERM-003

| Experiment/ Case | Benchmark-Model | | | Calculated | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{233}U$ [a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| Ust03-40 | 0.9995 | 0.0087 | 0.0387 | 1.0080 | 0.0010 | 0.0085 | ± | 0.0175 | 74.1 |
| Ust03-41 | 0.9991 | 0.0151 | 0.0397 | 1.0260 | 0.0011 | 0.0269 | ± | 0.0303 | 74.1 |
| Ust03-42 | 1.0007 | 0.0087 | 0.0400 | 1.0044 | 0.0011 | 0.0037 | ± | 0.0176 | 74.1 |
| Ust03-45 | 1.0015 | 0.0126 | 0.0610 | 1.0140 | 0.0011 | 0.0124 | ± | 0.0253 | 45.9 |
| Ust03-55 | 1.0006 | 0.0122 | 0.0693 | 1.0197 | 0.0011 | 0.0191 | ± | 0.0245 | 39.4 |
| Ust03-57 | 1.0012 | 0.0087 | 0.0209 | 1.0244 | 0.0010 | 0.0232 | ± | 0.0175 | 154.0 |
| Ust03-58 | 1.0016 | 0.0087 | 0.0138 | 1.0167 | 0.0010 | 0.0151 | ± | 0.0175 | 250.1 |
| Ust03-61 | 1.0016 | 0.0087 | 0.0108 | 1.0133 | 0.0010 | 0.0117 | ± | 0.0175 | 328.7 |

| Table 5-144 Results for U233-SOL-THERM-003 | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | | Calculated | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{233}U$[a] |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| Ust03-62 | 1.0018 | 0.0087 | 0.0095 | 1.0107 | 0.0010 | 0.0089 | ± | 0.0175 | 395.3 |
| Ust03-65 | 1.0008 | 0.0087 | 0.0056 | 1.0073 | 0.0008 | 0.0065 | ± | 0.0175 | 774.7 |

NOTE: [a] $H/^{233}U$ is from page 29 of U233-SOL-THERM-003.

### 5.8.2.1.4 U233-SOL-THERM-004

A number of critical experiments involving $^{233}U$ were performed at ORNL during the year 1952 in order to determine the minimum critical mass, minmum critical volume, and maximum safe pipe size of water moderated solutions of $^{233}U$. Thirteen of these utilizing uranyl nitrate ($UO_2(NO_3)_2$) solutions in five-, six-, and seven and one half-inch-diameter cylinders are evaluated in NEA (2001). The five- and six-inch-diameter cylinders were reflected by at least six inches of paraffin in the radial direction and on the bottom. The seven and one half-inch-diameter cylinders were reflected by six inches of paraffin in the radial direction and on the top and bottom. None of the experiments performed with uranyl nitrate in the five-inch-diameter cylinders attained criticality, thus they are not considered acceptable as benchmark experiments. One experiment performed in the six-inch-diameter cylinder was extrapolated to critical based on measurements at 75% of the estimated critical height and is not considered acceptable as a benchmark experiment. Therefore eight experiments were judged acceptable for use as criticality-safety benchmark experiments.

Each different-diameter vessel was fabricated as a separate assembly. The five- and six-inch-diameter vessels were 36 inches tall while the seven and one half-inch-diameter vessel was approximately equilateral. For experiments in the 6-inch-diameter vessel the benchmark model is a right circular aluminum cylinder with an inside radius of 7.5589 cm and an inside height of 91.5280 cm. The aluminum wall thickness for the sides, top, and bottom is 0.1291 cm. The vessel is reflected by 15.24 cm of paraffin radially and on the bottom. For experiments in the 7.5-inch-diameter vessel the benchmark model is a right circular aluminum cylinder with an inside radius of 9.5246 cm and an inside height of 19.1609 cm. The aluminum wall thickness for the sides, top, and bottom is 0.1291 cm. The vessel is reflected by 15.24 cm of paraffin radially and on the top and bottom. All benchmark model solution densities and heights are given in Table 5-145. MCNP results are shown in Table 5-146.

| Table 5-145 Benchmark-Model Solution Densities and Heights. | | |
|---|---|---|
| Experiment # | Solution Density (g/cc) | Solution Height (cm) |
| 6-inch-diameter | | |
| 3 | 1.2866 | 38.5880 |

| Table 5-145 Benchmark-Model Solution Densities and Heights. | | |
|---|---|---|
| **Experiment #** | **Solution Density (g/cc)** | **Solution Height (cm)** |
| 6 | 1.2322 | 46.8880 |
| 20 | 1.3935 | 30.7880 |
| 25 | 1.5432 | 27.9880 |
| 30 | 1.48 | 29.0880 |
| 7.5-inch-diameter | | |
| 27 | 1.5432 | 16.4109 |
| 28 | 1.48 | 16.3109 |
| 33 | 1.2379 | 18.7109 |

Table 5-146 Results for U233-SOL-THERM-004

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{233}$ [b] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma^{a}$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| ust04-03 | 1.0039 | 0.0088 | 0.0257 | 1.0086 | 0.0011 | 0.0047 | ± | 0.0177 | 119.00 |
| ust04-06 | 1.0034 | 0.0086 | 0.0208 | 1.0113 | 0.0010 | 0.0079 | ± | 0.0173 | 149.20 |
| ust04-20 | 1.0041 | 0.0089 | 0.0353 | 1.0006 | 0.0011 | -0.0035 | ± | 0.0179 | 83.90 |
| ust04-25 | 1.0051 | 0.0089 | 0.0493 | 0.9936 | 0.0011 | -0.0115 | ± | 0.0179 | 57.14 |
| ust04-27 | 1.0020 | 0.0105 | 0.0479 | 1.0119 | 0.0011 | 0.0099 | ± | 0.0211 | 57.14 |
| ust04-28 | 1.0020 | 0.0104 | 0.0425 | 1.0063 | 0.0011 | 0.0043 | ± | 0.0209 | 66.42 |
| ust04-30 | 1.0037 | 0.0090 | 0.0430 | 0.9988 | 0.0011 | -0.0049 | ± | 0.0181 | 66.42 |
| ust04-33 | 1.0020 | 0.0102 | 0.0215 | 1.0087 | 0.0011 | 0.0067 | ± | 0.0205 | 144.23 |

NOTE: [a] These uncertainties are relatively high, > 1% in some cases.
[b] $H/^{233}U$ is from page 23 of U233-SOL-THERM-004.

### 5.8.2.1.5 U233-SOL-THERM-005

A number of critical experiments involving $^{233}U$ were performed at ORNL during the years 1952 and 1953 in order to determine the minimum critical mass, minmum critical volume, and maximum safe pipe size of water moderated solutions of $^{233}U$. Two of these experiments utilizing uranyl nitrate ($UO_2(NO_3)_2$) solutions in simple geometry are evaluated in this report. Both experiments were judged acceptable for use as criticality-safety benchmark experiments.

Experiment 37 is in a 10.4-inch-diameter sphere, and Experiment 39 is in a 10-inch-diameter cylinder (which was approximately equilateral based on internal dimensions). The reaction vessels were constructed of 3S aluminum with a Heresite coating to reduce corrosion. The $^{233}U$

concentration ranges from 49 to 62 g 233U/l. The solution densities were 1.087 g/cm$^3$ for the sphere and 1.069 g/cm$^3$ for the cylinder. Both experiments were reflected by at least six inches of water in all directions.

The simplified benchmark model (NEA [2001]) for Case 1 (Experiment 37) is a sphere of solution with a radius of 13.2118 cm, surrounded by an aluminum wall that is 0.1291 cm thick and a 15.24-cm-thick water reflector. The simplified benchmark model for Case 2 (Experiment 39) is a right circular cylinder of solution with a radius of 12.7287 cm. The inner height of the cylinder is 25.6482 cm. This solution cylinder is surrounded on all sides by 0.1291 cm of aluminum and then by 15.24 cm of water reflector. MCNP results are shown in Table 5-147.

Table 5-147 Results for U233-SOL-THERM-005

| Experiment/ Case | Benchmark Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{233}U^a$ |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| ust05-01 | 1.0000 | 0.0040 | 0.0094 | 1.0054 | 0.0009 | 0.0054 | ± | 0.0082 | 406 |
| ust05-02 | 1.0000 | 0.0049 | 0.0078 | 1.0075 | 0.0009 | 0.0075 | ± | 0.0100 | 515 |

NOTE: [a] $H/^{233}U$ is calculated from the number densities given on page 21of U233-SOL-THERM-005.

### 5.8.2.1.6 U233-SOL-THERM-008

In the 1950's a series of experiments was performed at ORNL to investigate the critical conditions of unreflected homogeneous aqueous uranyl nitrate solutions, and to experimentally determine the thermal value of $\eta$ (the number of fission neutrons produced per thermal neutron absorbed) for the $^{235}U$ and $^{233}U$ isotopes. The measurements were made on both bare spherical and cylindrical configurations. This section describes one configuration of a $^{233}U$ solution contained in a 48.04-inch-diameter sphere. The $H/^{233}U$ ratio in the solution was approximately 1985. The experiment is judged to be acceptable for use as a criticality safety benchmark experiment. (NEA [2001]). Summary of material data is given in Table 5-148 and MCNP results are shown in Table 5-149.

Table 5-148 Summary of Material Data for Critical Condition of the 48.04-inch-diameter Sphere

| Isotopic Composition (wt.%) | | | | Solution Density (g/ml) | Total Uranium (mg/g) | Total Nitrate (mg/g) | Total Thorium (mg/g) |
|---|---|---|---|---|---|---|---|
| $^{233}U$ | $^{234}U$ | $^{235}U$ | $^{236}U$ | | | | |
| 97.67 | 1.54 | 0.03 | 0.76 | 1.0153 | 13.05 | 7.6 | 0.056 |

Table 5-149 Results for U233-SOL-THERM-008

| Experiment/ Case | Benchmark-Model $k_{eff}$ | $\sigma$ | AENCF | Calculated $k_{eff}$ | $\sigma$ | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| ust008 | 1.0006 | 0.0029 | 0.0030 | 0.9986 | 0.0004 | -0.0020 | ± | 0.0059 |

### 5.8.2.1.7 U233-SOL-INTER-001

A series of criticality studies were performed at Lawrence Livermore National Laboratory in the late 1950's using aqueous solutions of $^{233}U$ in the form of $UO_2F_2$ stabilized with 0.3% by weight of HF. The $^{233}U$ concentration in these experiments ranged from 0.13 to 0.87 kg/l. Eight type 347 stainless steel spheres ranging in inner radius from 7.87 to 12.45 cm were available for use as containers for the solutions. The seven smallest spheres were used in these experiments with 0.87, 0.75 and 0.57 kg/l $^{233}U$ solutions. Reflectors of beryllium, polyethylene and beryllium-polyethylene composites were used. Thirty-three configurations were evaluated and accepted as criticality-safety benchmark models (NEA [2001]). Table 5-150 shows material data for this experiment and MCNP results are shown in Table 5-151.

Table 5-150 Material Data for U-233-SOL-INTER-001

| Solution No | Sphere No | Critical Reflector Thickness | | | |
|---|---|---|---|---|---|
| | | U-233 Mass (kg) | Be (cm) | $CH_2$ (cm) | Be+$CH_2$ (cm) |
| 1 | 1 | 1.77 | 8.00±1.5% | --- | --- |
| | 2 | 2.25 | 5.82±2.3% | --- | --- |
| | 3 | 2.66 | 4.67±1.0% | --- | 1.14+4.11 |
| | 4 | 3.28 | 3.50±3.0% | --- | 0.51+3.31 |
| | 5 | 3.82 | 2.69±2.0% | 3.05 | 1.27+1.47 |
| | 6 | 4.58 | 1.83±2.0% | --- | 0.64+1.35 |
| 2 | 1 | 1.54 | 8.00±2.0% | --- | --- |
| | 2 | 1.94 | 5.94±1.0% | --- | 1.65+5.18 |
| | 3 | 2.30 | 4.70±2.0% | 11.68 | 1.14+4.06 |
| | 4 | 2.84 | 3.43±1.0% | --- | --- |
| | 5 | 3.30 | 2.62±4.0% | 3.07 | --- |
| | 6 | 3.96 | 1.78±2.0% | --- | --- |
| | 7 | 4.68 | 1.18±3.0% | 1.52 | --- |
| 3 | 1 | 1.16 | 8.69±1.5% | --- | --- |
| | 2 | 1.47 | 6.20±2.0% | --- | --- |
| | 3 | 1.74 | 5.03±1.0% | --- | 1.14+4.34 |
| | 4 | 2.15 | 3.61±1.5% | --- | --- |
| | 5 | 2.50 | 2.72±5.0% | 3.10 | --- |
| | 6 | 3.00 | 2.08±4.0% | --- | --- |
| | 7 | 3.55 | 1.19±4.0% | 1.60 | --- |

Table 5-151 Results for U233-SOL-INTER-001

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{233}U^a$ |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| usi01-01m | 1.0000 | 0.0083 | 0.1068 | 0.9860 | 0.0003 | -0.0140 | ± | 0.0166 | 24.66 |
| usi01-02m | 1.0000 | 0.0085 | 0.1136 | 0.9814 | 0.0004 | -0.0186 | ± | 0.0170 | 24.66 |
| usi01-03m | 1.0000 | 0.0066 | 0.1185 | 0.9838 | 0.0005 | -0.0162 | ± | 0.0132 | 24.66 |
| usi01-04m | 1.0000 | 0.0061 | 0.1092 | 0.9952 | 0.0006 | -0.0048 | ± | 0.0123 | 24.66 |
| usi01-05m | 1.0000 | 0.0082 | 0.1236 | 0.9868 | 0.0007 | -0.0132 | ± | 0.0165 | 24.66 |
| usi01-06m | 1.0000 | 0.0061 | 0.1146 | 0.9895 | 0.0007 | -0.0105 | ± | 0.0123 | 24.66 |
| usi01-07m | 1.0000 | 0.0059 | 0.1269 | 0.9851 | 0.0007 | -0.0149 | ± | 0.0119 | 24.66 |
| usi01-08m | 1.0000 | 0.0056 | 0.1180 | 0.9830 | 0.0007 | -0.0170 | ± | 0.0113 | 24.66 |
| usi01-09m | 1.0000 | 0.0068 | 0.1246 | 0.9821 | 0.0007 | -0.0179 | ± | 0.0137 | 24.66 |
| usi01-10m | 1.0000 | 0.0053 | 0.1325 | 0.9822 | 0.0007 | -0.0178 | ± | 0.0107 | 24.66 |
| usi01-11m | 1.0000 | 0.0057 | 0.1291 | 0.9814 | 0.0007 | -0.0186 | ± | 0.0115 | 24.66 |
| usi01-12m | 1.0000 | 0.0091 | 0.0922 | 0.9835 | 0.0007 | -0.0165 | ± | 0.0183 | 29.27 |
| usi01-13m | 1.0000 | 0.0071 | 0.0973 | 0.9835 | 0.0007 | -0.0165 | ± | 0.0143 | 29.27 |
| usi01-14m | 1.0000 | 0.0052 | 0.0914 | 0.9929 | 0.0007 | -0.0071 | ± | 0.0105 | 29.27 |
| usi01-15m | 1.0000 | 0.0075 | 0.1018 | 0.9811 | 0.0007 | -0.0189 | ± | 0.0151 | 29.27 |
| usi01-16m | 1.0000 | 0.0028 | 0.0906 | 0.9847 | 0.0007 | -0.0153 | ± | 0.0058 | 29.27 |
| usi01-17m | 1.0000 | 0.0055 | 0.0947 | 0.9907 | 0.0007 | -0.0093 | ± | 0.0111 | 29.27 |
| usi01-18m | 1.0000 | 0.0083 | 0.1069 | 0.9794 | 0.0007 | -0.0206 | ± | 0.0167 | 29.27 |
| usi01-19m | 1.0000 | 0.0057 | 0.1104 | 0.9782 | 0.0003 | -0.0218 | ± | 0.0114 | 29.27 |
| usi01-20m | 1.0000 | 0.0056 | 0.1018 | 0.9818 | 0.0008 | -0.0182 | ± | 0.0113 | 29.27 |
| usi01-21m | 1.0000 | 0.0050 | 0.1147 | 0.9773 | 0.0007 | -0.0227 | ± | 0.0101 | 29.27 |
| usi01-22m | 1.0000 | 0.0049 | 0.1181 | 0.9802 | 0.0007 | -0.0198 | ± | 0.0099 | 29.27 |
| usi01-23m | 1.0000 | 0.0047 | 0.1118 | 0.9939 | 0.0008 | -0.0061 | ± | 0.0095 | 29.27 |
| usi01-24m | 1.0000 | 0.0081 | 0.0685 | 0.9932 | 0.0008 | -0.0068 | ± | 0.0163 | 40.35 |
| usi01-25m | 1.0000 | 0.0081 | 0.0733 | 0.9855 | 0.0007 | -0.0145 | ± | 0.0163 | 40.35 |
| usi01-26m | 1.0000 | 0.0065 | 0.0759 | 0.9897 | 0.0007 | -0.0103 | ± | 0.0131 | 40.35 |
| usi01-27m | 1.0000 | 0.0051 | 0.0713 | 0.9931 | 0.0007 | -0.0069 | ± | 0.0103 | 40.35 |
| usi01-28m | 1.0000 | 0.0061 | 0.0798 | 0.9856 | 0.0007 | -0.0144 | ± | 0.0123 | 40.35 |
| usi01-29m | 1.0000 | 0.0098 | 0.0829 | 0.9784 | 0.0007 | -0.0216 | ± | 0.0197 | 40.35 |
| usi01-30m | 1.0000 | 0.0053 | 0.0766 | 0.9795 | 0.0007 | -0.0205 | ± | 0.0107 | 40.35 |
| usi01-31m | 1.0000 | 0.0071 | 0.0852 | 0.9923 | 0.0007 | -0.0078 | ± | 0.0143 | 40.35 |
| usi01-32m | 1.0000 | 0.0053 | 0.0888 | 0.9794 | 0.0008 | -0.0206 | ± | 0.0107 | 40.35 |

Table 5-151 Results for U233-SOL-INTER-001

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{233}U$[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| Usi01-33m | 1.0000 | 0.0046 | 0.0836 | 0.9953 | 0.0008 | -0.0047 | ± | 0.0093 | 40.35 |

NOTE: [a] $H/^{233}U$ is calculated from the number densities given on page 36 of U233-SOL-INTER-001.

### 5.8.2.2 U-233 Solutions Contained Within Arrays of Cans

#### 5.8.2.2.1 U233-SOL-THERM-006

Critical experiments using aqueous solutions of uranyl nitrate, having a high $^{233}U$ content in arrays of cylindrical cans, were performed between August of 1966 and April of 1968 at ORNL. The purpose of the experiments was to establish polyethylene-reflected and unreflected critical configurations for multiple "tightly" and "loosely" neutronically coupled subcritical cylindrical solution units within cubic unit cells forming cubic arrays. Results from the experiments were to have potential use in the processing, storage, and transportation of $^{233}U$ solutions.

During the experimental campaign it was necessary to "purify" the solution to remove the highly radioactive daughter products of the trace quantities of $^{232}U$ in the solution. The first series of experiments were performed with "Pre-purified" solutions identified as Sample 2. Following the first series of experiments the solution was passed through a solvent extraction process to remove the highly radioactive daughter products. The extraction process was not specific to the trace quantity of thorium that was present in the original solution. Therefore, the returned "Purified" solution also contained trace quantities of thorium. The "Purified" solution used for the second series of array experiments was identified as Sample 11. Sample 2 had a density of approximately 1.4812 g/ml and a uranium density of approximately 342.7 mgU/ml, resulting in approximately 1462 g U for each array unit can. As used in the 2 x 2 x 2 and 3 x 3 x 3 arrays, the total mass of U in the arrays was about 11696 g U and 39474 g U, respectively. Sample 11 solution had a density of approximately 1.2781 g/ml and a uranium density of approximately 203.8 mgU/ml, resulting in approximately 885 g U for each array unit can. As used in the 2 x 2 x 2 and 3 x 3 x 3 arrays, the total mass of U in the arrays was about 7080 g U and 23895 g U, respectively.

The experiments involved a total of 51 configurations. Of the 51 configurations, 18 were configured as an array of 2x2x2 cans and 33 were configured as an array of 3x3x3 cans. The spacing between cans was varied to obtain the different configurations using the same solution. 25 out of 51 evaluated configurations were judged to be acceptable as benchmarks (NEA [2001]), of which 12 unreflected experiment configurations were analyzed in this section. Table 5-152 presents the results for the 12 MCNP cases.

Table 5-152 Results for U233-SOL-THERM-006

| Experiment /Case | Benchmark-Model | | Calculated[a] | | | $\Delta k_{eff} \pm 2\sigma$ | | | $H/^{233}U$[b] ratio |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| m35 | 1.0000 | 0.0035 | 0.0576 | 1.0230 | 0.0008 | 0.0230 | ± | 0.0072 | 68.11 |
| m36 | 1.0000 | 0.0035 | 0.0583 | 1.0113 | 0.0008 | 0.0113 | ± | 0.0072 | 68.11 |
| m37[c] | 1.0000 | 0.0035 | 0.0588 | 0.9996 | 0.0008 | -0.0004 | ± | 0.0072 | 68.11 |
| m38 | 1.0000 | 0.0035 | 0.0584 | 1.0011 | 0.0008 | 0.0011 | ± | 0.0072 | 68.11 |
| m45 | 1.0000 | 0.0035 | 0.0590 | 1.0186 | 0.0011 | 0.0186 | ± | 0.0073 | 68.11 |
| m61 | 1.0000 | 0.0035 | 0.0345 | 1.0228 | 0.0008 | 0.0228 | ± | 0.0072 | 119.78 |
| m62 | 1.0000 | 0.0035 | 0.0346 | 1.0132 | 0.0008 | 0.0132 | ± | 0.0072 | 119.78 |
| m63 | 1.0000 | 0.0035 | 0.0352 | 0.9988 | 0.0008 | -0.0012 | ± | 0.0072 | 119.78 |
| m65 | 1.0000 | 0.0035 | 0.0350 | 0.9939 | 0.0008 | -0.0061 | ± | 0.0072 | 119.78 |
| m77 | 1.0000 | 0.0035 | 0.0358 | 0.9939 | 0.0011 | -0.0061 | ± | 0.0073 | 119.78 |
| m78 | 1.0000 | 0.0035 | 0.0358 | 0.9932 | 0.0011 | -0.0068 | ± | 0.0074 | 119.78 |
| m79 | 1.0000 | 0.0035 | 0.0355 | 0.9929 | 0.0011 | -0.0071 | ± | 0.0074 | 119.78 |

NOTES: [a] $^{232}U$ was left out of the fuel composition since it has very small (negligible) number density.
[b] $H/^{233}U$ is calculated using number densities on page 78 of U233-SOL-THERM-006.

## 5.9 PREVIOUSLY EVALUATED PU AND MIXED BENCHMARK CASES

### 5.9.1 Pu and Mixed Benchmarks Related to Intact Waste Package Configurations

#### 5.9.1.1 FFTF Fuel Pin Array Experiments

This section describes an experimental program, conducted at PNL, that predicted critical configurations extrapolated from near critical experiments with plutonium oxide-uranium oxide fuel pins containing about 20 wt% plutonium with light water moderation and reflection (MIX-COMP-THERM-001 [NEA (2001)]). Four experiments were evaluated and are considered acceptable for use as benchmark experiments (NEA [2001]). These experiments were performed at the PNL Critical Mass Laboratory. The experimental configuration was comprised of an array of fast test reactor fuel pins within a large tank containing water. Various lattice pitches were used in the array, resulting in different numbers of fuel rods being required to obtain criticality. The fuel used for the experimental program was a mixture of $PuO_2$ and $UO_2$, with the pins comprised of 19.7 wt% plutonium. The plutonium contained 11.5 wt% $^{240}Pu$, and the uranium in the $PuO_2$-$UO_2$ mixture was natural uranium. The remainder of the pin consisted of end-caps, plenum, and other types of hardware (e.g., natural $UO_2$ insulators and Inconel reflectors). The benchmark $k_{eff}$ for these experiments was experimentally determined to be unity. These experiments were previously evaluated in section 5.1.5.1 of CRWMS M&O (1999a). Table 5-153 provides a brief description of these experiments along with their MCNP case names as given in CRWMS M&O (1999a).

| Table 5-153 FFTF Array Critical Experiments | | | | |
|---|---|---|---|---|
| **Experiment Number** | **Description (p. 15, NEA [2001])** | **H/X ratio** | **σ** | **MCNP case name** |
| 001 | 18 pin lattice width, 1.2588 cm lattice spacing, 279 total pins | 1742.38 | 0.0026 | fftf001 |
| 004 | 18 pin lattice width, 1.5342 cm lattice spacing, 205 total pins | 11273.00 | 0.0032 | fftf004 |
| 005 | 28 pin lattice width, 0.9525 cm lattice spacing, 605 total pins | 835.37 | 0.0025 | fftf005 |
| 006 | 14 pin lattice width, 1.9050 cm lattice spacing, 162 total pins | 17210.22 | 0.0039 | fftf006 |

The results that follow are from CRWMS M&O (1999a, page 86). Table 5-154 presents the MCNP calculation results for ENDF/B-V and ENDF/B-VI cross section libraries for the FFTF set of critical experiments. Table 5-156 presents the AENCF.

| Table 5-154 LCE Reactivity Calculation Results for FFTF Experiments | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V $k_{eff} \pm \sigma$** | **ENDF/B-VI $k_{eff} \pm \sigma$** |
| fftf001 | 1.00567±0.00342 | 0.99485 ± 0.00344 |
| fftf004 | 0.99333±0.00348 | 1.00137 ± 0.00337 |
| fftf005 | 0.99582±0.00361 | 0.99835 ± 0.00316 |
| fftf006 | 1.01291±0.00285 | 0.99947 ± 0.00313 |

| Table 5-155 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for FFTF Experiments[a] | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V $\Delta k_{eff} \pm 2\sigma$** | **ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$** |
| fftf001 | -0.00567 ± 0.02102 | 0.00515 ± 0.00862 |
| fftf004 | 0.00667 ± 0.02106 | -0.00137 ± 0.00929 |
| fftf005 | 0.00418 ± 0.02115 | 0.00165 ± 0.00806 |
| fftf006 | -0.01291 ± 0.02068 | 0.00053 ± 0.01000 |

NOTE: [a] $\Delta k_{eff}$ is defined as benchmark $k_{eff}$ – calculated $k_{eff}$.

| Table 5-156 AENCF for FFTF Experiments (MeV) | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| Fftf001 | 0.1044 | 0.1043 |
| Fftf004 | 0.0816 | 0.0779 |

| Table 5-156 AENCF for FFTF Experiments (MeV) | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| Fttf005 | 0.1717 | 0.1727 |
| Fttf006 | 0.0635 | 0.0646 |

### 5.9.1.2 Saxton Plutonium Program Experiment

Single and multi-region uranium and plutonium oxide fueled cores, water moderated, clean, and borated, have been used in a series of critical experiments at the Westinghouse Reactor Evaluation Center in support of the Saxton Plutonium Program. In this series of experiments, criticality was achieved entirely by varying the water level inside the core tank. The fuel used in the experiments were $UO_2$ fuel with 5.74 wt% $^{235}U$ enrichment, and MOX fuel containing 6.6 wt% $PuO_2$ and natural enriched $UO_2$. These experiments were previously evaluated in section 5.1.5.2 of CRWMS M&O (1999a). A description of the experiments analyzed in this calculation, and the MCNP case names is presented in Table 5-157. A detailed description of the experimental configuration for the MOX single-region experiments is provided in NEA (2001), MIX-COMP-THERM-003 while the descriptions for the multiregion experiments is given in CRWMS M&O (1999a). The benchmark $k_{eff}$ values for the single-region experiments were 1.000 with an uncertainty of ± 0.0071, ± 0.0057, ± 0.0052, ± 0.0028, ± 0.0024, and ± 0.0020 for cases 3, 4, 5, 6, 7, and 8 (NEA [2001]). Page 42 of CRWMS M&O (1999a) states that a benchmark $k_{eff}$ value and associated uncertainty values for the multi-region experiments were not available and were assumed to be 1.000 and ± 0.0042, respectively.

Table 5-157 Saxton Critical Experiment Descriptions

**Single-Region Critical Experiment Description**

| Reference case name | Fuel | Pitch (cm) | Core configuration | Critical water height (cm) | Boron concentration (ppmB[a]) | MCNP case name |
|---|---|---|---|---|---|---|
| 1 | $UO_2$ | 1.3208 | 449 cylindrical | 95.25 | 0 | ssr83 |
| 2 | $UO_2$ | 1.4224 | 19 x 19 square | 83.71 | 0 | ssr48 |
| 3 | MOX | 1.3208 | 22 x 23 square | 84.56 | 0 | ssr70 |
| 4 | MOX | 1.4224 | 19 x 19 square | 82.46 | 0 | ssr57 |
| 5 | MOX | 1.4224 | 21 x 21 square | 89.70 | 337 | ssr27 |
| 6 | MOX | 1.8669 | 13 x 13 square | 70.11 | 0 | ssr66 |
| 7 | MOX | 2.0117 | 12 x 12 square | 78.43 | 0 | ssr53 |
| 8 | MOX | 2.6416 | 11 x 11 square | 81.17 | 0 | ssr74 |

**Multi-region Critical Experiment Description**

| Reference case name | Core configuration | Critical water height (cm) | Boron concentration (ppmB) | MCNP case name |
|---|---|---|---|---|
| 1 | 19 x 19 square: 11 x 11 MOX center region; $UO_2$ outer region | 91.07 | 0 | smr1 |

| Table 5-157 Saxton Critical Experiment Descriptions | | | | |
|---|---|---|---|---|
| 2 | 19 x 19 square: 11 x 11 MOX center region; $UO_2$ outer region; Al plate at the fuel interface | 92.07 | 0 | smr9 |
| 3 | 27 x 27 square: 19 x 19 $UO_2$ center region; MOX outer region | 86.70 | 1252 | smr5 |
| 4 | 27 x 27 square: 19 x 19 MOX center region; $UO_2$ outer region; water slot at the region boundary | 99.80 | 1453 | Smr11 |
| 5 | 27 x 27 square: 19 x 19 MOX center region; $UO_2$ outer region; Al slab at the interface | 106.35 | 1453 | Smr12 |
| 8 | 27 x 27 square: 19 x 19 MOX center region; $UO_2$ outer region; L shaped $UO_2$ insert in MOX region | 92.19 | 1425 | smr8 |

NOTE: [a] ppmB = parts per million boron by mass

The results that follow are from CRWMS M&O (1999a, page 88). Table 5-158 presents the MCNP calculation results for ENDF/B-V and ENDF/B-VI cross section libraries for the Saxton set of critical experiments. Table 5-160 presents the AENCF values for the Saxton set of critical experiments. Table 5-159 presents Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for these experiments.

| Table 5-158 Criticality Results for the Saxton Experiments | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V $k_{eff} \pm \sigma$** | **ENDF/B-VI $k_{eff} \pm \sigma$** |
| smr1 | 0.99627 ± 0.00072 | 0.99302 ± 0.00072 |
| smr5 | 0.99442 ± 0.00072 | 0.99087 ± 0.00074 |
| smr8 | 0.99861 ± 0.00064 | 0.99444 ± 0.00058 |
| smr9 | 0.99468 ± 0.00072 | 0.99151 ± 0.00074 |
| smr11 | 0.99833 ± 0.00072 | 0.99247 ± 0.00079 |
| smr12 | 0.99869 ± 0.00073 | 0.99358 ± 0.00077 |
| ssr27 | 0.99728 ± 0.00077 | 0.99195 ± 0.00077 |
| ssr48 | 0.99396 ± 0.00074 | 0.99136 ± 0.00072 |
| ssr53 | 1.0034 ± 0.00065 | 0.99503 ± 0.00074 |
| ssr57 | 0.99673 ± 0.00076 | 0.99009 ± 0.00072 |
| ssr66 | 1.00272 ± 0.00074 | 0.99383 ± 0.00073 |
| ssr70 | 0.99369 ± 0.00074 | 0.98936 ± 0.00071 |
| ssr74 | 1.00551 ± 0.00066 | 0.99858 ± 0.00072 |
| ssr83 | 0.9928 ± 0.00069 | 0.98821 ± 0.00073 |

Table 5-159 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for Saxton Experiments[a]

| MCNP case name | ENDF/B-V $\Delta k_{eff} \pm 2\sigma$ | ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|
| smr1 | 0.00373 ± 0.00852 | 0.00698 ± 0.00852 |
| smr5 | 0.00558 ± 0.00852 | 0.00913 ± 0.00853 |
| smr8 | 0.00139 ± 0.00850 | 0.00556 ± 0.00848 |
| smr9 | 0.00532 ± 0.00852 | 0.00849 ± 0.00853 |
| smr11 | 0.00167 ± 0.00852 | 0.00753 ± 0.00855 |
| smr12 | 0.00131 ± 0.00853 | 0.00642 ± 0.00854 |
| ssr27 | 0.00272 ± 0.01051 | 0.00805 ± 0.01051 |
| ssr48 | 0.00604 ± 0.00853 | 0.00864 ± 0.00852 |
| ssr53 | -0.00340 ± 0.00497 | 0.00497 ± 0.00502 |
| ssr57 | 0.00327 ± 0.01150 | 0.00991 ± 0.01149 |
| ssr66 | -0.00272 ± 0.00579 | 0.00617 ± 0.00579 |
| ssr70 | 0.00631 ± 0.01428 | 0.01064 ± 0.01427 |
| ssr74 | -0.00551 ± 0.00421 | 0.00142 ± 0.00425 |
| ssr83 | 0.00720 ± 0.00851 | 0.01179 ± 0.00853 |

NOTE: [a] $\Delta k_{eff}$ is defined as experimental $k_{eff}$ – MCNP calculated $k_{eff}$.

Table 5-160 AENCF for Saxton Experiments (MeV)

| MCNP case name | ENDF/B-V | ENDF/B-VI |
|---|---|---|
| smr1 | 0.1707 | 0.1705 |
| smr5 | 0.1928 | 0.1919 |
| smr8 | 0.2045 | 0.2037 |
| smr9 | 0.1683 | 0.1667 |
| smr11 | 0.2049 | 0.2027 |
| smr12 | 0.2041 | 0.2024 |
| ssr27 | 0.2013 | 0.2017 |
| ssr48 | 0.1564 | 0.1546 |
| ssr53 | 0.1063 | 0.1064 |
| ssr57 | 0.1933 | 0.1928 |
| ssr66 | 0.1182 | 0.1189 |
| ssr70 | 0.2294 | 0.2290 |
| ssr74 | 0.0799 | 0.0800 |
| ssr83 | 0.1821 | 0.1811 |

### 5.9.2 Pu and Mixed Benchmarks Related to Degraded Waste Package Configurations

#### 5.9.2.1 Plutonium Fueled Fast Metal Systems

Reactivity calculations involving Fast Metal Fuel are presented in this section. A series of critical experiments performed by the Institute for Experimental Physics of the Russian Federal Nuclear Center (VNIIEF) at Arzamas-16 and the Institute for Technical Physics of the Russian Federal Nuclear Center (VNIITF) at Chelyabinsk-70 are described. Detailed experimental configurations and material compositions for these experiments are included in NEA (2001). Table 5-161 provides a listing of NEA (2001) identifier numbers, some information about the experiment, the benchmark $k_{eff}$ value, and the MCNP case names.

Table 5-161 Russian Pu Fast Metal Benchmark Experiments

| Handbook identifier | $^{235}$U wt% or $^{239}$Pu at% | Core dimensions (cm) Radius | Reflector material | Reflector thickness (cm) | Benchmark $k_{eff}$ | MCNP case name |
|---|---|---|---|---|---|---|
| PU-MET-FAST-020 | 89 | 5.350 | Depleted U | 7.650 | 0.9993 ± 0.0017 | PMF20 |
| PU-MET-FAST-022 | 98 | 6.670 | None | -- | 1.0000 ± 0.0021 | PMF22 |
| PU-MET-FAST-023 | 98 | 6.000 | Graphite | 2.350 | 1.0000 ± 0.0020 | PMF23 |
| PU-MET-FAST-024 | 98 | 6.000 | Polyethylene | 1.550 | 1.0000 ± 0.0020 | PMF24 |
| PU-MET-FAST-025 | 98 | 6.000 | Steel | 1.550 | 1.0000 ± 0.0020 | PMF25 |
| PU-MET-FAST-026 | 98 | 5.350 | Steel | 11.900 | 1.0000 ± 0.0024 | PMF26 |
| PU-MET-FAST-027 | 89 | 5.350 | Polyethylene | 5.580 | 1.0000 ± 0.0022 | PMF27 |
| PU-MET-FAST-028 | 89 | 5.350 | Steel | 19.650 | 1.0000 ± 0.0022 | PMF28 |
| PU-MET-FAST-029 | 88 | 5.350 | None | -- | 1.0000 ± 0.0020 | PMF29 |
| PU-MET-FAST-030 | 88 | 4.660 | Graphite | 4.490 | 1.0000 ± 0.0021 | PMF30 |
| PU-MET-FAST-031 | 88 | 4.660 | Polyethylene | 3.690 | 1.0000 ± 0.0021 | PMF31 |
| PU-MET-FAST-032 | 88 | 4.660 | Steel | 4.490 | 1.0000 ± 0.0020 | PMF32 |

The results that follow are from CRWMS M&O (1999a, pages 104 and 105). Table 5-162 presents the MCNP calculation results for ENDF/B-V and ENDF/B-VI cross section libraries for the Pu fueled Russian Criticality set of experiments. Table 5-164 presents the AENCF values for the Pu fueled Russian Criticality set of experiments. As presented in Table 5-163, it can be seen that all of the results are within 1.5% of the benchmark values.

Table 5-162 Criticality Calculation Results for PMF Experiments

| MCNP Case name | ENDF/B-V $k_{eff} \pm \sigma$ | ENDF/B-VI $k_{eff} \pm \sigma$ |
|---|---|---|
| PMF20 | 1.00085 ± 0.00064 | 0.99878 ± 0.00062 |
| PMF22 | 0.99627 ± 0.00057 | 0.99707 ± 0.00051 |
| PMF23 | 0.99742 ± 0.00059 | 0.99869 ± 0.00061 |
| PMF24 | 0.99781 ± 0.00065 | 1.00122 ± 0.00064 |
| PMF25 | 0.99835 ± 0.00062 | 0.9971 ± 0.00064 |
| PMF26 | 1.00167 ± 0.00059 | 0.99754 ± 0.00057 |
| PMF27 | 1.00178 ± 0.00078 | 1.00258 ± 0.00073 |
| PMF28 | 1.00324 ± 0.00063 | 0.99891 ± 0.0006 |
| PMF29 | 0.99309 ± 0.00056 | 0.99463 ± 0.00058 |
| PMF30 | 1.00078 ± 0.00066 | 1.00176 ± 0.00057 |
| PMF31 | 1.00208 ± 0.00073 | 1.00305 ± 0.00074 |
| PMF32 | 0.99938 ± 0.0006 | 0.9964 ± 0.00065 |

Table 5-163 Δk Between Experimental $k_{eff}$ and MCNP Calculated $k_{eff}$ for PMF Experiments

| MCNP case name | ENDF/B-V $\Delta k_{eff} \pm 2\sigma$ | ENDF/B-VI $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|
| PMF20 | -0.00165 ± 0.00363 | 0.00042 ± 0.00131 |
| PMF22 | 0.00373 ± 0.00416 | 0.00293 ± 0.00310 |
| PMF23 | 0.00258 ± 0.00417 | 0.00131 ± 0.00179 |
| PMF24 | 0.00219 ± 0.00421 | -0.00122 ± 0.00177 |
| PMF25 | 0.00165 ± 0.00419 | 0.00290 ± 0.00317 |
| PMF26 | -0.00167 ± 0.00494 | 0.00246 ± 0.00271 |
| PMF27 | -0.00278 ± 0.00467 | -0.00358 ± 0.00387 |
| PMF28 | -0.00434 ± 0.00458 | -0.00001 ± 0.00120 |
| PMF29 | 0.00691 ± 0.00396 | 0.00537 ± 0.00549 |
| PMF30 | -0.00078 ± 0.00440 | -0.00176 ± 0.00210 |
| PMF31 | -0.00208 ± 0.00445 | -0.00305 ± 0.00339 |
| PMF32 | 0.00062 ± 0.00418 | 0.00360 ± 0.00383 |

| Table 5-164 AENCF for PMF Experiments (MeV) | | |
|---|---|---|
| **MCNP case name** | **ENDF/B-V** | **ENDF/B-VI** |
| PMF20 | 1.8895 | 1.8960 |
| PMF22 | 1.8914 | 1.9004 |
| PMF23 | 1.7988 | 1.8070 |
| PMF24 | 1.7418 | 1.7414 |
| PMF25 | 1.8293 | 1.8431 |
| PMF26 | 1.7250 | 1.7407 |
| PMF27 | 1.4726 | 1.4768 |
| PMF28 | 1.7100 | 1.7229 |
| PMF29 | 1.9154 | 1.9254 |
| PMF30 | 1.8138 | 1.8213 |
| PMF31 | 1.6057 | 1.6082 |
| PMF32 | 1.8113 | 1.8263 |

### 5.9.2.2 Pu and Mixed Solution Systems

The experiments involving plutonium and uranium with naturally occurring isotopic ratios are described in NEA (2001) and are listed in Table 5-165 with a brief description of each experiment.

| Table 5-165 Benchmark Problem Summary for Configurations Incorporating Mixed Plutonium and Natural Uranium Nitrate Solutions | | | |
|---|---|---|---|
| **Reference Identifier** | **Experiment Identifier** | **Description[a]** | **Experimental $k_{eff} \pm \sigma$** |
| MIX-SOL-THERM-001<br>Water reflected annular cylindrical tank with central bottle and various annular inserts including B4C concrete and polyethylene, 91-92 wt% $^{239}$Pu- in Pu | PNL3187 (087)[b] | 102.19 g Pu/liter, 365.20 g U/liter, 2% $B_4C$ Concrete Annulus, No Bottle, Critical Height: 48.55 cm, H/$^{239}$Pu (annular tank) = 234, 91.118 wt% $^{239}$Pu in Pu | 1.0000 ± 0.0016 |
| | PNL3391 (091) | 103.37 g Pu/liter, 363.66 g U/liter, 0% $B_4C$ Concrete Annulus, Bottle 2, Critical Height: 27.67 cm, H/$^{239}$Pu (annular tank) = 231, H/$^{239}$Pu (bottle) = 231, 91.118 wt% $^{239}$Pu in Pu | 1.0000 ± 0.0016 |
| | PNL3492 (092) | 106.30 g Pu/liter, 373.33 g U/liter, 1% $B_4C$ Concrete Annulus, Bottle 2, Critical Height: 37.19 cm, H/$^{239}$Pu (annular tank) = 225, cm, H/$^{239}$Pu (bottle) = 231, 91.117 wt% $^{239}$Pu in Pu | 1.0000 ± 0.0016 |
| | PNL3593 (093) | 107.91 g Pu/liter, 379.55 g U/liter, 6% $B_4C$ Concrete Annulus, Bottle 2, Critical Height: 51.10 cm, H/$^{239}$Pu (annular tank) = 220, H/p$^{239}$Pu (bottle) =231, 91.117 wt% $^{239}$Pu in Pu | 1.0000 ± 0.0016 |

Table 5-165 Benchmark Problem Summary for Configurations Incorporating Mixed Plutonium and Natural Uranium Nitrate Solutions

| Reference Identifier | Experiment Identifier | Description[a] | Experimental $k_{eff} \pm \sigma$ |
|---|---|---|---|
| MIX-SOL-THERM-001 | PNL3694 (094) | 108.27 g Pu/liter, 380.41 g U/liter, No Concrete Annulus, Bottle 2, Critical Height: 32.86 cm, H/$^{239}$Pu (annular tank) = 219 H/$^{239}$Pu (bottle) = 231, 91.117 wt% $^{239}$Pu in Pu | 1.0000 ± 0.0016 |
| | PNL3795 (095) | 195.61 g Pu/liter, 6.5 g U/liter, 2% $B_4C$ Concrete Annulus, Bottle 3, Critical Height: 27.51 cm, H/$^{239}$Pu (annular tank) = 125, H/$^{239}$Pu (bottle) =126, 91.572 wt% $^{239}$Pu in Pu | 1.0000 ± 0.0016 |
| | PNL3896 (096) | 110.13 g Pu/liter, 3.8 g U/liter, 2% $B_4C$ Concrete Annulus, Bottle 3, Critical Height: 25.69 cm, H/$^{239}$Pu (annular tank) = 242, H/$^{239}$Pu (bottle) = 126, 91.572 wt% $^{239}$Pu in Pu | 1.0000 ± 0.0016 |
| | PNL3897 (097) | 58.30 g Pu/liter, 2.3 g U/liter, 2% $B_4C$ Concrete Annulus, Bottle 3, Critical Height: 28.94 cm, H/$^{239}$Pu (annular tank) = 477, H/$^{239}$Pu (bottle) = 126, 91.572 wt% $^{239}$Pu in Pu | 1.0000 ± 0.0016 |
| | PNL3898 (098) | 72.74 g Pu/liter, 247.33 g U/liter, 2% $B_4C$ Concrete Annulus, Bottle 2, Critical Height: 39.58 cm, H/$^{239}$Pu (annular tank) = 354, H/$^{239}$Pu (bottle) = 231, 91.117 wt% $^{239}$Pu in Pu | 1.0000 ± 0.0016 |
| | PNL3808 (108) | 47.08 g Pu/liter, 161.72 g U/liter, 2% $B_4C$ Concrete Annulus, Bottle 2, Critical Height: 45.09 cm, H/$^{239}$Pu (annular tank) = 569, H/$^{239}$Pu (bottle) = 231, 91.117 wt% $^{239}$Pu in Pu | 1.0000 ± 0.0016 |
| | PNL3999 (099) | 73.64 g Pu/liter, 250.30 g U/liter, Polyethylene with Cd Cover Annulus, Bottle 2, Critical Height: 79.18 cm, H/$^{239}$Pu (annular tank) = 349, H/$^{239}$Pu (bottle) = 231, 91.117 wt% $^{239}$Pu in Pu | 1.0000 ± 0.0052 |
| | PNL5300 (100) | 74.25 g Pu/liter, 251.64 g U/liter, Solid Polyethylene with Cd Cover, Critical Height: 104.62 cm, H/$^{239}$Pu (annular tank) = 346, 91.117 wt% $^{239}$Pu in Pu | 1.0000 ± 0.0052 |
| MIX-SOL-THERM-002 Water reflected cylindrical tank with 68.68 cm ID, 91.102 wt% $^{239}$Pu in Pu | PNL1158 (058) | 11.88 g Pu/liter, 11.05 g U/liter, Critical Height: 76.80 cm, H/$^{239}$Pu = 2,403 | 1.0000 ± 0.0024 |
| | PNL1159 (059) | 11.73 g Pu/liter, 10.78 g U/liter, Critical Height: 83.14 cm, H/$^{239}$Pu = 2,435 | 1.0000 ± 0.0024 |
| | PNL1161 (061) | 12.19 g Pu/liter, 41.04 g U/liter, Critical Height: 81.72 cm, H/$^{239}$Pu = 2,317 | 1.0000 ± 0.0024 |
| MIX-SOL-THERM-003 | Awre1 (1) | 101.3 g Pu/liter, 228.5 g U/liter, ID = 25.425 cm, Critical Height: 56.31 cm, H/$^{239}$Pu = 239 | 0.9985 ± 0.0020 |
| | Awre2 (2) | 101.3 g Pu/liter, 228.5 g U/liter, ID = 30.62 cm, Critical Height: 29.89 cm, H/$^{239}$Pu = 239 | 0.9960 ± 0.0020 |

Table 5-165 Benchmark Problem Summary for Configurations Incorporating Mixed Plutonium and Natural Uranium Nitrate Solutions

| Reference Identifier | Experiment Identifier | Description[a] | Experimental $k_{eff} \pm \sigma$ |
|---|---|---|---|
| MIX-SOL-THERM-003<br>Water/polyethylene reflected cylindrical tank with various diameters, 93.95 wt% $^{239}$Pu in Pu | Awre3 (3) | 101.3 g Pu/liter, 228.5 g U/liter ID = 37.99 cm, Critical Height: 21.17 cm, H/$^{239}$Pu = 239 | 0.9935 ± 0.0020 |
| | Awre4 (4) | 101.3 g Pu/liter, 228.5 g U/liter, ID = 50.72 cm, Critical Height: 16.05 cm, H/$^{239}$Pu = 239 | 0.9909 ± 0.0020 |
| | Awre5 (5) | 31.58 g Pu/liter, 71.3 g U/liter, ID = 30.62 cm, Critical Height: 46.18 cm, H/$^{239}$Pu = 847 | 0.9981 ± 0.0022 |
| | Awre6 (6) | 31.58 g Pu/liter, 71.3 g U/liter, ID = 37.99 cm, Critical Height: 28.24 cm, H/$^{239}$Pu = 847 | 0.9959 ± 0.0022 |
| | Awre7 (7) | 31.58 g Pu/liter, 71.3 g U/liter, ID = 50.72 cm, Critical Height: 20.39 cm, H/$^{239}$Pu = 847 | 0.9935 ± 0.0022 |
| | Awre8 (8) | 18.61 g Pu/liter, 42.2 g U/liter, ID = 37.99 cm, Critical Height: 72.86 cm, H/$^{239}$Pu = 1461 | 0.9988 ± 0.0025 |
| | Awre9 (9) | 18.61 g Pu/liter, 42.2 g U/liter, ID = 50.72 cm, Critical Height: 33.59 cm, H/$^{239}$Pu = 1461 | 0.9958 ± 0.0025 |
| | Awre10 (10) | 17.50 g Pu/liter, 39.6 g U/liter, ID = 50.72 cm, Critical Height: 37.16 cm, H/$^{239}$Pu = 1556 | 0.9964 ± 0.0025 |
| MIX-SOL-THERM-004<br>Cylindrical tank with 35.39 cm ID and either water reflector, concrete reflector, or no reflector, 91.118 wt% $^{239}$Pu in Pu | PNL1577 (077) | 172.56 g Pu/liter, 262.79 g U/liter, No Reflector, Critical Height: 57.97 cm, H/$^{239}$Pu = 137 | 1.0000 ± 0.0033 |
| | PNL1678 (078) | 172.82 g Pu/liter, 262.55 g U/liter, Water Reflector, Critical Height: 28.93 cm, H/$^{239}$Pu = 136 | 1.0000 ± 0.0033 |
| | PNL1783 (083) | 173.22 g Pu/liter, 262.88 g U/liter, Concrete Reflector, Critical Height: 30.60 cm, H/$^{239}$Pu = 136 | 1.0000 ± 0.0078 |
| | PNL1868 (068) | 118.71 g Pu/liter, 173.98 g U/liter, Concrete Reflector, Critical Height: 27.03 cm, H/$^{239}$Pu = 214 | 1.0000 ± 0.0078 |
| | PNL1969 (069) | 119.04 g Pu/liter, 174.67 g U/liter, Water Reflector, Critical Height: 25.26 cm, H/$^{239}$Pu = 213 | 1.0000 ± 0.0078 |
| | PNL2070 (070) | 118.90 g Pu/liter, 174.53 g U/liter, No Reflector, Critical Height: 41.08 cm, H/$^{239}$Pu = 214 | 1.0000 ± 0.0078 |
| | PNL2565 (065) | 41.69 g Pu/liter, 63.38 g U/liter, No Reflector, Critical Height: 44.46 cm, H/$^{239}$Pu = 664 | 1.0000 ± 0.0033 |
| | PNL2666 (066) | 41.89 g Pu/liter, 63.65 g U/liter, Water Reflector, Critical Height: 28.11 cm, H/$^{239}$Pu = 660 | 1.0000 ± 0.0033 |
| | PNL2767 (067) | 41.83 g Pu/liter, 63.55 g U/liter, Concrete Reflector, Critical Height: 29.36 cm, H/$^{239}$Pu = 661 | 1.0000 ± 0.0078 |

NOTES: [a] H/$^{239}$Pu ratio is calculated from the atom density ratio given in NEA (2001).
[b] NEA (2001) experiment identifiers are in ( ).

The experiments involving plutonium are from NEA (2001),and are listed in Table 5-166.

Table 5-166 Benchmark Problem Summary for Configurations Incorporating Plutonium Nitrate Solutions

| Reference Identifier | Experiment Identifier | Description[a] | Experimental $k_{eff} \pm \sigma$ |
|---|---|---|---|
| PU-SOL-THERM-001 Water reflected 11.5-inch diameter spheres, 0.049-inch thick shell of Type 304L stainless steel, 4.57 wt% $^{240}$Pu in Pu | pust1t1 | 73.0 g Pu/liter, Critical Mass: 945 gm, H/$^{239}$Pu = 371 | 1.0000 ± 0.0050 |
| | pust1t2 | 96.0 g Pu/liter, Critical Mass: 1243 gm, H/$^{239}$Pu = 272 | 1.0000 ± 0.0050 |
| | pust1t3 | 119.0 g Pu/liter, Critical Mass: 1541 gm, cm, H/$^{239}$Pu = 216 | 1.0000 ± 0.0050 |
| | pust1t4 | 132.0 g Pu/liter, Critical Mass: 1709 gm, H/$^{239}$Pu = 190 | 1.0000 ± 0.0050 |
| | pust1t5 | 140.0 g Pu/liter, Critical Mass: 1813 gm, H/$^{239}$Pu = 180 | 1.0000 ± 0.0050 |
| | pust1t6 | 268.7 g Pu/liter, Critical Mass: 3480 gm, H/$^{239}$Pu = 91 | 1.0000 ± 0.0050 |
| PU-SOL-THERM-003 Water reflected 13-inch diameter spheres, 0.050-inch thick shell of Type 347 stainless steel or 20 gauge 2S Al shell | pu003-1 | 33.32 g Pu/liter, Critical Mass: 631 gm, H/$^{239}$Pu = 788, 1.76 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu003-2 | 34.32 g Pu/liter, Critical Mass: 650 gm, H/$^{239}$Pu = 756, 1.76 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu003-3 | 37.43 g Pu/liter, Critical Mass: 709 gm, H/$^{239}$Pu = 699, 3.12 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu003-4 | 38.12 g Pu/liter, Critical Mass: 722 gm, H/$^{239}$Pu = 682, 3.12 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu003-5 | 40.65 g Pu/liter, Critical Mass: 770 gm, H/$^{239}$Pu = 627, 3.12 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu003-6 | 44.09 g Pu/liter, Critical Mass: 835 gm, H/$^{239}$Pu = 563, 3.12 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu003-7 | 35.98 g Pu/liter, Critical Mass: 649 gm, H/$^{239}$Pu = 738, 3.12 wt% $^{240}$Pu, 20 Gauge 2S Al Shell | 1.0000 ± 0.0047 |
| | pu003-8 | 36.81 g Pu/liter, Critical Mass: 664 gm, H/$^{239}$Pu = 714, 3.12 wt% $^{240}$Pu, 20 Gauge 2S Al Shell | 1.0000 ± 0.0047 |
| PU-SOL-THERM-004 Water reflected 14-inch diameter spheres, 0.50-inch thick shell of Type 347 stainless steel | pu004-1 | 26.27 g Pu/liter, Critical Mass: 621 gm, H/$^{239}$Pu = 987, 0.54 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu004-2 | 26.31 g Pu/liter, Critical Mass: 622 gm, H/$^{239}$Pu = 977, 0.54 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu004-3 | 27.20 g Pu/liter, Critical Mass: 643 gm, H/$^{239}$Pu = 935, 0.54 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu004-4 | 28.09 g Pu/liter, Critical Mass: 664 gm, H/$^{239}$Pu = 889, 0.54 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu004-5 | 27.58 g Pu/liter, Critical Mass: 652 gm, H/$^{239}$Pu = 942, 1.76 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu004-6 | 28.60 g Pu/liter, Critical Mass: 676 gm, H/$^{239}$Pu = 927, 3.12 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu004-7 | 29.57 g Pu/liter, Critical Mass: 699 gm, H/$^{239}$Pu = 892, 3.12 wt% $^{240}$Pu | 1.0000 ± 0.0047 |

Table 5-166 Benchmark Problem Summary for Configurations Incorporating Plutonium Nitrate Solutions

| Reference Identifier | Experiment Identifier | Description[a] | Experimental $k_{eff} \pm \sigma$ |
|---|---|---|---|
| PU-SOL-THERM-004 | pu004-8 | 29.95 g Pu/liter, Critical Mass: 708 gm, H/$^{239}$Pu = 869, 3.12 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu004-9 | 31.60 g Pu/liter, Critical Mass: 747 gm, H/$^{239}$Pu = 805, 3.12 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu04-10 | 35.36 Pu/liter, Critical Mass: 836 gm, H/$^{239}$Pu = 689, 3.12 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu04-11 | 39.38 g Pu/liter, Critical Mass: 931 gm, H/$^{239}$Pu = 592, 3.12 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu04-12 | 29.44 g Pu/liter, Critical Mass: 696 gm, H/$^{239}$Pu = 892, 3.12 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu04-13 | 29.27 g Pu/liter, Critical Mass: 692 gm, H/$^{239}$Pu = 903, 3.43 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| PU-SOL-THERM-005<br>Water reflected 14-inch diameter spheres, 0.050-inch thick shell of Type 347 stainless steel | pu005-1 | 29.65 g Pu/liter, Critical Mass: 701 gm, H/$^{239}$Pu = 903, 4.05 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu005-2 | 30.54 g Pu/liter, Critical Mass: 722 gm, H/$^{239}$Pu = 868, 4.05 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu005-3 | 31.43 g Pu/liter, Critical Mass: 743 gm, H/$^{239}$Pu = 834, 4.05 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu005-4 | 33.54 g Pu/liter, Critical Mass: 973 gm, H/$^{239}$Pu = 765, 4.05 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu005-5 | 36.04 g Pu/liter, Critical Mass: 852 gm, H/$^{239}$Pu = 694, 4.05 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu005-6 | 38.49 g Pu/liter, Critical Mass: 910 gm, H/$^{239}$Pu = 633, 4.05 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu005-7 | 40.91 g Pu/liter, Critical Mass: 967 gm, H/$^{239}$Pu = 581, 4.05 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu005-8 | 30.58 g Pu/liter, Critical Mass: 723 gm, H/$^{239}$Pu = 869, 4.40 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| | pu005-9 | 31.85 g Pu/liter, Critical Mass: 753 gm, H/$^{239}$Pu = 825, 4.40 wt% $^{240}$Pu | 1.0000 ± 0.0047 |
| PU-SOL-THERM-007[b] | pu007-2 | 232 g Pu/liter, Critical Volume: 12.35 liters, Height Above Sphere Center: 10.8373 cm, H/$^{239}$Pu =110 | 1.0000 ± 0.0047 |
| | pu007-3 | 221 g Pu/liter, Critical Volume: 12.35 liters, Height Above Sphere Center: 10.8373 cm, H/$^{239}$Pu = 114 | 1.0000 ± 0.0047 |
| | pu007-5 | 100.2 g Pu/liter, Critical Volume: 12.39 liters, Height Above Sphere Center: 10.9741 cm, H/$^{239}$Pu = 268 | 1.0000 ± 0.0047 |
| | pu007-6 | 101.5 g Pu/liter, Critical Volume: 12.30 liters, Height Above Sphere Center: 10.6720 cm, H/$^{239}$Pu = 262 | 1.0000 ± 0.0047 |
| | pu007-7 | 100.1 g Pu/liter, Critical Volume: 12.39 liters, Height Above Sphere Center: 10.9741 cm, H/$^{239}$Pu = 266 | 1.0000 ± 0.0047 |

Table 5-166 Benchmark Problem Summary for Configurations Incorporating Plutonium Nitrate Solutions

| Reference Identifier | Experiment Identifier | Description[a] | Experimental $k_{eff} \pm \sigma$ |
|---|---|---|---|
| PU-SOL-THERM-007[b] Water reflected partly filled 11.5-inch diameter spheres, 0.049-inch thick shell of Type 304L stainless steel, 4.57 wt% $^{240}$Pu in Pu | pu007-8 | 101.6 g Pu/liter, Critical Volume: 12.37 liters, Height Above Sphere Center: 10.9051 cm, H/$^{239}$Pu = 258 | 1.0000 ± 0.0047 |
| | pu007-9 | 101.6 g Pu/liter, Critical Volume: 12.23 liters, Height Above Sphere Center: 10.4503 cm, H/$^{239}$Pu = 260 | 1.0000 ± 0.0047 |
| | pu07-10 | 93.5 g Pu/liter, Critical Volume: 12.35 liters, Height Above Sphere Center: 10.8373 cm, H/$^{239}$Pu =285 | 1.0000 ± 0.0047 |
| PU-SOL-THERM-009 Unreflected 48-inch diameter sphere, 0.303-inch thick shell of type 1100 aluminum, 97.386 wt% $^{239}$Pu and 2.521 wt% $^{240}$Pu in Pu | pust9-1 | 10.02 g Pu/liter, Critical Volume: 656.6 liters, Height Above Sphere Center: 15.9558 cm, H/$^{239}$Pu = 2648 | 1.0000 ± 0.0033 |
| | pust9-2 | 9.539 g Pu/liter, Critical Volume: 906.5 liters, Height Above Sphere Center: 45.3705 cm, H/$^{239}$Pu = 2779 | 1.0000 ± 0.0033 |
| | pust9-3 | 9.457 g Pu/liter, Critical Volume: 949.1 liters, Full Sphere, H/$^{239}$Pu = 2803 | 1.0000 ± 0.0033 |
| PU-SOL-THERM-0010 Water reflected cylinders, 0.062-inch thick shell of type 347 stainless steel, 2.9 wt% $^{240}$Pu in Pu | pu10091 | 99.09 g Pu/liter, cm Inner Radius (IR) = 11.4264 cm, Critical Height: 30.7086 cm, Critical Mass: 1249 g, H/$^{239}$Pu = 267 | 1.0000 ± 0.0048 |
| | pu10092 | 73.92 g Pu/liter, IR = 11.4264 cm, Critical Height: 35.4076 cm, Critical Mass: 1073 g, H/$^{239}$Pu = 357 | 1.0000 ± 0.0048 |
| | pu10093 | 54.53 g Pu/liter, IR = 11.4264 cm, Critical Height: 44.5770 cm, Critical Mass: 997 g, H/$^{239}$Pu = 484 | 1.0000 ± 0.0048 |
| | pu10111 | 54.53 g Pu/liter, IR = 13.9684 cm, Critical Height: 25.6032 cm, Critical Mass: 856 g, H/$^{239}$Pu = 485, Extra 0.065 inch layer of stainless steel placed around cylinder | 1.0000 ± 0.0048 |
| | pu10112 | 47.21 g Pu/liter, IR = 13.9684 cm, Critical Height: 28.1686 cm, Critical Mass: 815 g, H/$^{239}$Pu = 558, Extra 0.065 inch layer of stainless steel placed around cylinder | 1.0000 ± 0.0048 |
| | pu10113 | 47.21 g Pu/liter, IR = 13.9684 cm, Critical Height: 27.0764 cm, Critical Mass: 784 g, H/$^{239}$Pu = 558 | 1.0000 ± 0.0048 |
| | pu10114 | 41.73 g Pu/liter, IR = 13.9684 cm, Critical Height: 32.6390 cm, Critical Mass: 835 g, H/$^{239}$Pu = 606 | 1.0000 ± 0.0048 |
| | pu10115 | 36.90 g Pu/liter, IR = 13.9684 cm, Critical Height: 43.0022 cm, Critical Mass: 973 g, H/$^{239}$Pu = 665 | 1.0000 ± 0.0048 |
| | pu10116 | 63.99 g Pu/liter, IR = 13.9684 cm, Critical Height: 22.8092 cm, Critical Mass: 895 g, H/$^{239}$Pu = 414 | 1.0000 ± 0.0048 |

Table 5-166 Benchmark Problem Summary for Configurations Incorporating Plutonium Nitrate Solutions

| Reference Identifier | Experiment Identifier | Description[a] | Experimental $k_{eff} \pm \sigma$ |
|---|---|---|---|
| PU-SOL-THERM-0010 | pu10117 | 48.98 g Pu/liter, IR = 13.9684 cm, Critical Height: 25.9588 cm, Critical Mass: 780 g, H/$^{239}$Pu = 535 | 1.0000 ± 0.0048 |
| | pu10121 | 48.75 g Pu/liter, IR = 15.2390 cm, Critical Height: 22.3520 cm, Critical Mass: 799 g, H/$^{239}$Pu = 543 | 1.0000 ± 0.0048 |
| | pu10122 | 42.29 g Pu/liter, IR = 15.2390 cm, Critical Height: 25.2476 cm, Critical Mass: 779 g, H/$^{239}$Pu = 618 | 1.0000 ± 0.0048 |
| | pu10123 | 36.52 g Pu/liter, IR = 15.2390 cm, Critical Height: 28.4734 cm, Critical Mass: 758 g, H/$^{239}$Pu = 728 | 1.0000 ± 0.0048 |
| | pu10124 | 31.14 g Pu/liter, IR = 15.2390 cm, Critical Height: 33.4264 cm, Critical Mass: 759 g, H/$^{239}$Pu = 850 | 1.0000 ± 0.0048 |
| PU-SOL-THERM-0011 Unreflected 16- & 18-inch diameter spheres, 0.050 inch thick shell of type 347 stainless steel, 0.020 inch thick Cd cover on the 18-inch sphere, 4.2 wt% $^{240}$Pu in Pu 4.17wt% $^{240}$Pu for 16-inch spheres | pu11161 | 34.96 g Pu/liter, IR = 20.1206 cm, Critical Mass: 1194 g, H/$^{239}$Pu = 765 | 1.0000 ± 0.0052 |
| | pu11162 | 36.22 g Pu/liter, Ir = 20.1206 cm, Critical Mass: 1237 g, H/$^{239}$Pu = 736 | 1.0000 ± 0.0052 |
| | pu11163 | 38.13 g Pu/liter, IR = 20.1206 cm, Critical Mass: 1302 g, H/$^{239}$Pu = 691 | 1.0000 ± 0.0052 |
| | pu11164 | 38.16 g Pu/liter, IR = 20.1206 cm, Critical Mass: 1303 g, H/$^{239}$Pu = 682 | 1.0000 ± 0.0052 |
| | pu11165 | 43.43 g Pu/liter, IR = 20.1206 cm, Critical Mass: 1483 g, H/$^{239}$Pu = 575 | 1.0000 ± 0.0052 |
| | pu11181 | 22.35 g Pu/liter, IR = 22.6974 cm, Critical Mass: 1095 g, H/$^{239}$Pu = 1208 | 1.0000 ± 0.0052 |
| | pu11182 | 23.27 g Pu/liter, IR = 22.6974 cm, Critical Mass: 1140 g, H/$^{239}$Pu = 1151 | 1.0000 ± 0.0052 |
| | pu11183 | 23.10 g Pu/liter, IR = 22.6974 cm, Critical Mass: 1132 g, H/$^{239}$Pu = 1158 | 1.0000 ± 0.0052 |
| | pu11184 | 23.82 g Pu/liter, IR = 22.6974 cm, Critical Mass: 1167 g, H/$^{239}$Pu = 1100 | 1.0000 ± 0.0052 |
| | pu11185 | 25.20 g Pu/liter, IR = 22.6974 cm, Critical Mass: 1235 g, H/$^{239}$Pu = 1039 | 1.0000 ± 0.0052 |
| | pu11186 | 27.49 g Pu/liter, IR = 22.6974 cm, Critical Mass: 1347 g, H/$^{239}$Pu = 908 | 1.0000 ± 0.0052 |
| | pu11187 | 23.94 g Pu/liter, IR = 22.6974 cm, Critical Mass: 1173 g, H/$^{239}$Pu = 1103 | 1.0000 ± 0.0052 |

NOTE: [a] H/$^{239}$Pu ratio is calculated from the atom density ratio given in NEA (2001).
[b] Case 1,4, and 7 left out intentionally.

The results of MCNP criticality calculations from CRWMS M&O (1999c, pages 44-47) are presented in Table 5-167.

Table 5-167 Results of MCNP Criticality Calculations

| Case | ENDF/B-VI | | | ENDF/B-V | | | ENDF/B-VI | | | ENDF/B-V | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF | $\Delta k_{eff} \pm 2\sigma$ | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| pnl3187 | 0.98843 | 0.00105 | 0.04181 | 0.99762 | 0.00116 | 0.04174 | 0.01157 | ± | 0.00383 | 0.00238 | ± | 0.00395 |
| pnl3391 | 0.98609 | 0.00104 | 0.04130 | 0.99425 | 0.00116 | 0.04105 | 0.01391 | ± | 0.00382 | 0.00575 | ± | 0.00395 |
| pnl3492 | 0.98923 | 0.00119 | 0.04361 | 0.99754 | 0.00116 | 0.04313 | 0.01077 | ± | 0.00399 | 0.00246 | ± | 0.00395 |
| pnl3593 | 0.99095 | 0.00108 | 0.04649 | 0.99727 | 0.00111 | 0.04586 | 0.00905 | ± | 0.00386 | 0.00273 | ± | 0.00389 |
| pnl3694 | 0.99364 | 0.00116 | 0.04427 | 1.00255 | 0.00118 | 0.04452 | 0.00636 | ± | 0.00395 | -0.00255 | ± | 0.00398 |
| pnl3795 | 0.99703 | 0.00119 | 0.03983 | 1.00166 | 0.00116 | 0.04000 | 0.00297 | ± | 0.00399 | -0.00166 | ± | 0.00395 |
| pnl3808 | 0.9928 | 0.00107 | 0.02061 | 1.00195 | 0.00113 | 0.02130 | 0.00720 | ± | 0.00385 | -0.00195 | ± | 0.00392 |
| pnl3896 | 0.99527 | 0.00113 | 0.02353 | 1.00235 | 0.00124 | 0.02319 | 0.00473 | ± | 0.00392 | -0.00235 | ± | 0.00405 |
| pnl3897 | 0.99639 | 0.00104 | 0.01426 | 1.00449 | 0.00105 | 0.01420 | 0.00361 | ± | 0.00382 | -0.00449 | ± | 0.00383 |
| pnl3898 | 0.99728 | 0.00116 | 0.03000 | 1.00287 | 0.00104 | 0.02994 | 0.00272 | ± | 0.00395 | -0.00287 | ± | 0.00382 |
| pnl3999 | 1.00133 | 0.00117 | 0.02906 | 1.00919 | 0.00107 | 0.02959 | -0.00133 | ± | 0.01066 | -0.00919 | ± | 0.01062 |
| pnl5300 | 1.00106 | 0.00102 | 0.02891 | 1.008 | 0.0011 | 0.02884 | -0.00106 | ± | 0.01060 | -0.00800 | ± | 0.01063 |
| pnl1158 | 1.00301 | 0.00065 | 0.00386 | 1.00691 | 0.00065 | 0.00384 | -0.00301 | ± | 0.00497 | -0.00691 | ± | 0.00497 |
| pnl1159 | 1.00244 | 0.00069 | 0.00390 | 1.00735 | 0.00062 | 0.00370 | -0.00244 | ± | 0.00499 | -0.00735 | ± | 0.00496 |
| pnl1161 | 1.00033 | 0.0006 | 0.00605 | 1.0079 | 0.00067 | 0.00607 | -0.00033 | ± | 0.00495 | -0.00790 | ± | 0.00498 |
| awre1 | 1.00516 | 0.00127 | 0.03133 | 1.01469 | 0.00103 | 0.03153 | -0.00666 | ± | 0.00474 | -0.01619 | ± | 0.00450 |
| awre2 | 1.00467 | 0.00115 | 0.03199 | 1.01568 | 0.00115 | 0.03149 | -0.00867 | ± | 0.00461 | -0.01968 | ± | 0.00461 |
| awre3 | 1.00181 | 0.00117 | 0.03253 | 1.01203 | 0.0012 | 0.03197 | -0.00831 | ± | 0.00463 | -0.01853 | ± | 0.00466 |
| awre4 | 0.9984 | 0.0011 | 0.03207 | 1.0051 | 0.00118 | 0.03189 | -0.00750 | ± | 0.00457 | -0.01420 | ± | 0.00464 |
| awre5 | 1.00019 | 0.00107 | 0.01028 | 1.00847 | 0.001 | 0.01043 | -0.00209 | ± | 0.00489 | -0.01037 | ± | 0.00483 |
| awre6 | 1.00588 | 0.00107 | 0.01041 | 1.01067 | 0.00103 | 0.01035 | -0.00998 | ± | 0.00489 | -0.01477 | ± | 0.00486 |
| awre7 | 0.99924 | 0.00101 | 0.01046 | 1.00796 | 0.00099 | 0.01050 | -0.00574 | ± | 0.00484 | -0.01446 | ± | 0.00482 |
| awre8 | 1.00379 | 0.00079 | 0.00682 | 1.01284 | 0.00083 | 0.00686 | -0.00499 | ± | 0.05002 | -0.01404 | ± | 0.05003 |
| awre9 | 1.00207 | 0.00094 | 0.00692 | 1.0094 | 0.00092 | 0.00661 | -0.00627 | ± | 0.00534 | -0.01360 | ± | 0.00533 |
| awre10 | 1.00221 | 0.00088 | 0.00652 | 1.01024 | 0.00084 | 0.00662 | -0.00581 | ± | 0.00530 | -0.01384 | ± | 0.00527 |
| pnl1577 | 0.98992 | 0.00125 | 0.05858 | 0.99578 | 0.00122 | 0.05887 | 0.01008 | ± | 0.00706 | 0.00422 | ± | 0.00704 |

Table 5-167 Results of MCNP Criticality Calculations

| Case | ENDF/B-VI | | | ENDF/B-V | | | ENDF/B-VI | | | ENDF/B-V | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF | $\Delta k_{eff} \pm 2\sigma$ | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| pnl1678 | 0.98991 | 0.00125 | 0.05119 | 0.99735 | 0.00115 | 0.05044 | 0.01009 | ± | 0.00706 | 0.00265 | ± | 0.00699 |
| pnl1783 | 0.99258 | 0.00124 | 0.05330 | 0.99922 | 0.00118 | 0.05337 | 0.00742 | ± | 0.01382 | 0.00078 | ± | 0.01380 |
| pnl1868 | 0.99364 | 0.00127 | 0.03450 | 1.0039 | 0.00127 | 0.03434 | 0.00636 | ± | 0.00633 | -0.00390 | ± | 0.00633 |
| pnl1969 | 0.98934 | 0.00128 | 0.03334 | 1.00003 | 0.0012 | 0.03340 | 0.01066 | ± | 0.00634 | -0.00003 | ± | 0.00628 |
| pnl2070 | 0.99312 | 0.00133 | 0.03853 | 0.99956 | 0.00143 | 0.03770 | 0.00688 | ± | 0.01583 | 0.00044 | ± | 0.01586 |
| pnl2565 | 0.99786 | 0.00119 | 0.01298 | 1.00148 | 0.00123 | 0.01294 | 0.00214 | ± | 0.00572 | -0.00148 | ± | 0.00575 |
| pnl2666 | 0.99694 | 0.00109 | 0.01169 | 1.00179 | 0.00112 | 0.01169 | 0.00306 | ± | 0.00564 | -0.00179 | ± | 0.00566 |
| pnl2767 | 0.99664 | 0.00113 | 0.01216 | 1.00607 | 0.00113 | 0.01230 | 0.00336 | ± | 0.01556 | -0.00607 | ± | 0.01556 |
| pust1t1 | 1.00058 | 0.00102 | 0.01301 | 1.01064 | 0.00101 | 0.01260 | -0.00058 | ± | 0.01021 | -0.01064 | ± | 0.01020 |
| pust1t2 | 1.00408 | 0.00104 | 0.01715 | 1.01278 | 0.001 | 0.01722 | -0.00408 | ± | 0.01021 | -0.01278 | ± | 0.01020 |
| pust1t3 | 1.00692 | 0.00106 | 0.02172 | 1.01447 | 0.00105 | 0.02086 | -0.00692 | ± | 0.01022 | -0.01447 | ± | 0.01022 |
| pust1t4 | 0.99921 | 0.00107 | 0.02405 | 1.00766 | 0.00105 | 0.02374 | 0.00079 | ± | 0.01023 | -0.00766 | ± | 0.01022 |
| pust1t5 | 1.00222 | 0.00103 | 0.02514 | 1.01107 | 0.00099 | 0.02464 | -0.00222 | ± | 0.01021 | -0.01107 | ± | 0.01019 |
| pust1t6 | 1.00585 | 0.00105 | 0.04810 | 1.01003 | 0.00104 | 0.04883 | -0.00585 | ± | 0.01022 | -0.01003 | ± | 0.01021 |
| pu003-1 | 1.00208 | 0.00095 | 0.00631 | 1.0089 | 0.0009 | 0.00634 | -0.00208 | ± | 0.01018 | -0.00890 | ± | 0.01016 |
| pu003-2 | 1.00029 | 0.00091 | 0.00664 | 1.00764 | 0.00091 | 0.00655 | -0.00029 | ± | 0.01016 | -0.00764 | ± | 0.01016 |
| pu003-3 | 1.00324 | 0.00097 | 0.00711 | 1.0116 | 0.00095 | 0.00696 | -0.00324 | ± | 0.01019 | -0.01160 | ± | 0.01018 |
| pu003-4 | 1.00401 | 0.00092 | 0.00731 | 1.01162 | 0.00097 | 0.00723 | -0.00401 | ± | 0.01017 | -0.01162 | ± | 0.01019 |
| pu003-5 | 1.00371 | 0.0009 | 0.00736 | 1.01162 | 0.00091 | 0.00772 | -0.00371 | ± | 0.01016 | -0.01162 | ± | 0.01016 |
| pu003-6 | 1.00425 | 0.00094 | 0.00863 | 1.01275 | 0.00097 | 0.00858 | -0.00425 | ± | 0.01018 | -0.01275 | ± | 0.01019 |
| pu003-7 | 1.00545 | 0.00092 | 0.00667 | 1.01438 | 0.00092 | 0.00669 | -0.00545 | ± | 0.01017 | -0.01438 | ± | 0.01017 |
| pu003-8 | 1.00196 | 0.00096 | 0.00700 | 1.01194 | 0.00093 | 0.00681 | -0.00196 | ± | 0.01018 | -0.01194 | ± | 0.01017 |
| pu004-1 | 1.00302 | 0.00087 | 0.00521 | 1.01168 | 0.00086 | 0.00515 | -0.00302 | ± | 0.00956 | -0.01168 | ± | 0.00956 |
| pu004-2 | 0.99646 | 0.00095 | 0.00517 | 1.00671 | 0.00085 | 0.00534 | 0.00354 | ± | 0.00959 | -0.00671 | ± | 0.00955 |
| pu004-3 | 0.9994 | 0.00085 | 0.00542 | 1.0087 | 0.00091 | 0.00555 | 0.00060 | ± | 0.00955 | -0.00870 | ± | 0.00957 |
| pu004-4 | 0.99678 | 0.00094 | 0.00569 | 1.00559 | 0.0009 | 0.00563 | 0.00322 | ± | 0.00959 | -0.00559 | ± | 0.00957 |

Table 5-167 Results of MCNP Criticality Calculations

| Case | ENDF/B-VI | | | ENDF/B-V | | | ENDF/B-VI | | | ENDF/B-V | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF | $\Delta k_{eff} \pm 2\sigma$ | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| pu004-5 | 1.00082 | 0.00087 | 0.00538 | 1.00684 | 0.00088 | 0.00544 | -0.00082 | ± | 0.00956 | -0.00684 | ± | 0.00956 |
| pu004-6 | 1.00092 | 0.00088 | 0.00549 | 1.00867 | 0.00089 | 0.00572 | -0.00092 | ± | 0.00956 | -0.00867 | ± | 0.00957 |
| pu004-7 | 1.00513 | 0.00087 | 0.00582 | 1.01179 | 0.00086 | 0.00582 | -0.00513 | ± | 0.00956 | -0.01179 | ± | 0.00956 |
| pu004-8 | 0.99815 | 0.00089 | 0.00599 | 1.00786 | 0.00088 | 0.00593 | 0.00185 | ± | 0.00957 | -0.00786 | ± | 0.00956 |
| pu004-9 | 0.99749 | 0.00089 | 0.00643 | 1.0091 | 0.00087 | 0.00615 | 0.00251 | ± | 0.00957 | -0.00910 | ± | 0.00956 |
| pu005-1 | 1.00322 | 0.00087 | 0.00583 | 1.00871 | 0.00089 | 0.00568 | -0.00322 | ± | 0.00956 | -0.00871 | ± | 0.00957 |
| pu005-2 | 0.99995 | 0.00088 | 0.00587 | 1.01075 | 0.00091 | 0.00579 | 0.00005 | ± | 0.00956 | -0.01075 | ± | 0.00957 |
| pu005-3 | 1.00236 | 0.00092 | 0.00614 | 1.01227 | 0.00087 | 0.00622 | -0.00236 | ± | 0.00958 | -0.01227 | ± | 0.00956 |
| pu005-4 | 1.00279 | 0.00088 | 0.00665 | 1.01183 | 0.00088 | 0.00678 | -0.00279 | ± | 0.00956 | -0.01183 | ± | 0.00956 |
| pu005-5 | 1.00389 | 0.00093 | 0.00758 | 1.01254 | 0.00088 | 0.00725 | -0.00389 | ± | 0.00958 | -0.01254 | ± | 0.00956 |
| pu005-6 | 1.00338 | 0.00092 | 0.00771 | 1.01279 | 0.00092 | 0.00780 | -0.00338 | ± | 0.00958 | -0.01279 | ± | 0.00958 |
| pu005-7 | 1.00194 | 0.00093 | 0.00845 | 1.01031 | 0.00094 | 0.00837 | -0.00194 | ± | 0.00958 | -0.01031 | ± | 0.00959 |
| pu005-8 | 0.99915 | 0.00089 | 0.00600 | 1.00559 | 0.0009 | 0.00608 | 0.00085 | ± | 0.00957 | -0.00559 | ± | 0.00957 |
| pu005-9 | 0.99937 | 0.00088 | 0.00618 | 1.00929 | 0.00086 | 0.00638 | 0.00063 | ± | 0.00956 | -0.00929 | ± | 0.00956 |
| pu007-2 | 1.00561 | 0.00106 | 0.04120 | 1.01358 | 0.00109 | 0.04098 | -0.00561 | ± | 0.00964 | -0.01358 | ± | 0.00965 |
| pu007-3 | 0.99948 | 0.00105 | 0.03932 | 1.00535 | 0.00104 | 0.03969 | 0.00052 | ± | 0.00963 | -0.00535 | ± | 0.00963 |
| pu007-5 | 1.00665 | 0.00104 | 0.01758 | 1.01438 | 0.00102 | 0.01754 | -0.00665 | ± | 0.00963 | -0.01438 | ± | 0.00962 |
| pu007-6 | 1.00134 | 0.001 | 0.01757 | 1.00808 | 0.00106 | 0.01773 | -0.00134 | ± | 0.00961 | -0.00808 | ± | 0.00964 |
| pu007-7 | 1.00333 | 0.00101 | 0.01747 | 1.01039 | 0.00105 | 0.01770 | -0.00333 | ± | 0.00961 | -0.01039 | ± | 0.00963 |
| pu007-8 | 0.99316 | 0.00105 | 0.01783 | 1.00512 | 0.00102 | 0.01796 | 0.00684 | ± | 0.00963 | -0.00512 | ± | 0.00962 |
| pu007-9 | 0.99427 | 0.00106 | 0.01774 | 1.00267 | 0.00108 | 0.01795 | 0.00573 | ± | 0.00964 | -0.00267 | ± | 0.00964 |
| pu04-10 | 1.00196 | 0.00093 | 0.00734 | 1.00979 | 0.00092 | 0.00720 | -0.00196 | ± | 0.00958 | -0.00979 | ± | 0.00958 |
| pu04-11 | 0.99716 | 0.00092 | 0.00836 | 1.00799 | 0.00096 | 0.00849 | 0.00284 | ± | 0.00958 | -0.00799 | ± | 0.00959 |
| pu04-12 | 1.0028 | 0.00087 | 0.00572 | 1.01028 | 0.00092 | 0.00573 | -0.00280 | ± | 0.00956 | -0.01028 | ± | 0.00958 |
| pu04-13 | 0.99942 | 0.00091 | 0.00585 | 1.00634 | 0.00089 | 0.00574 | 0.00058 | ± | 0.00957 | -0.00634 | ± | 0.00957 |
| pu07-10 | 0.99653 | 0.00106 | 0.01620 | 1.00643 | 0.00101 | 0.01711 | 0.00347 | ± | 0.00964 | -0.00643 | ± | 0.00961 |

Table 5-167 Results of MCNP Criticality Calculations

| Case | ENDF/B-VI | | | ENDF/B-V | | | ENDF/B-VI | | | ENDF/B-V | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | AENCF | $\Delta k_{eff} \pm 2\sigma$ | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| pu10091 | 1.01378 | 0.00102 | 0.01672 | 1.02188 | 0.00105 | 0.01658 | -0.01378 | ± | 0.00981 | -0.02188 | ± | 0.00983 |
| pu10092 | 1.01053 | 0.00097 | 0.01272 | 1.01932 | 0.00103 | 0.01270 | -0.01053 | ± | 0.00979 | -0.01932 | ± | 0.00982 |
| pu10093 | 1.00474 | 0.00096 | 0.00969 | 1.01428 | 0.00099 | 0.00930 | -0.00474 | ± | 0.00979 | -0.01428 | ± | 0.00980 |
| pu10111 | 1.01211 | 0.00102 | 0.00985 | 1.01833 | 0.00095 | 0.01026 | -0.01211 | ± | 0.00981 | -0.01833 | ± | 0.00979 |
| pu10112 | 1.00964 | 0.00097 | 0.00862 | 1.01691 | 0.00096 | 0.00888 | -0.00964 | ± | 0.00979 | -0.01691 | ± | 0.00979 |
| pu10113 | 1.00701 | 0.00097 | 0.00876 | 1.01378 | 0.00097 | 0.00867 | -0.00701 | ± | 0.00979 | -0.01378 | ± | 0.00979 |
| pu10114 | 0.99799 | 0.00096 | 0.00810 | 1.00854 | 0.00093 | 0.00798 | 0.00201 | ± | 0.00979 | -0.00854 | ± | 0.00978 |
| pu10115 | 1.00213 | 0.00093 | 0.00734 | 1.01003 | 0.0009 | 0.00753 | -0.00213 | ± | 0.00978 | -0.01003 | ± | 0.00977 |
| pu10116 | 1.01027 | 0.00102 | 0.01128 | 1.01922 | 0.001 | 0.01109 | -0.01027 | ± | 0.00981 | -0.01922 | ± | 0.00981 |
| pu10117 | 1.0005 | 0.00094 | 0.00920 | 1.00973 | 0.001 | 0.00901 | -0.00050 | ± | 0.00978 | -0.00973 | ± | 0.00981 |
| pu10121 | 1.00876 | 0.00094 | 0.00896 | 1.01726 | 0.00096 | 0.00892 | -0.00876 | ± | 0.00978 | -0.01726 | ± | 0.00979 |
| pu10122 | 1.0067 | 0.00096 | 0.00776 | 1.0153 | 0.00098 | 0.00795 | -0.00670 | ± | 0.00979 | -0.01530 | ± | 0.00980 |
| pu10123 | 1.01591 | 0.00093 | 0.00699 | 1.02207 | 0.00095 | 0.00671 | -0.01591 | ± | 0.00978 | -0.02207 | ± | 0.00979 |
| pu10124 | 1.00717 | 0.0009 | 0.00599 | 1.01695 | 0.00089 | 0.00594 | -0.00717 | ± | 0.00977 | -0.01695 | ± | 0.00976 |
| pu11161 | 1.0111 | 0.001 | 0.00744 | 1.01669 | 0.00101 | 0.00737 | -0.01110 | ± | 0.01059 | -0.01669 | ± | 0.01059 |
| pu11162 | 1.01343 | 0.00098 | 0.00773 | 1.02064 | 0.00103 | 0.00759 | -0.01343 | ± | 0.01058 | -0.02064 | ± | 0.01060 |
| pu11163 | 1.01656 | 0.00104 | 0.00794 | 1.02545 | 0.001 | 0.00829 | -0.01656 | ± | 0.01061 | -0.02545 | ± | 0.01059 |
| pu11164 | 1.00895 | 0.00101 | 0.00808 | 1.01704 | 0.00101 | 0.00851 | -0.00895 | ± | 0.01059 | -0.01704 | ± | 0.01059 |
| pu11165 | 1.00576 | 0.00102 | 0.00988 | 1.01318 | 0.00104 | 0.00955 | -0.00576 | ± | 0.01060 | -0.01318 | ± | 0.01061 |
| pu11181 | 0.99468 | 0.00091 | 0.00503 | 1.00317 | 0.00088 | 0.00501 | 0.00532 | ± | 0.01056 | -0.00317 | ± | 0.01055 |
| pu11182 | 1.00073 | 0.00088 | 0.00510 | 1.00894 | 0.00091 | 0.00527 | -0.00073 | ± | 0.01055 | -0.00894 | ± | 0.01056 |
| pu11183 | 0.99776 | 0.00092 | 0.00512 | 1.0051 | 0.00091 | 0.00528 | 0.00224 | ± | 0.01056 | -0.00510 | ± | 0.01056 |
| pu11184 | 0.99428 | 0.00087 | 0.00545 | 1.00198 | 0.00088 | 0.00525 | 0.00572 | ± | 0.01054 | -0.00198 | ± | 0.01055 |
| pu11185 | 1.0044 | 0.00089 | 0.00574 | 1.00964 | 0.00095 | 0.00588 | -0.00440 | ± | 0.01055 | -0.00964 | ± | 0.01057 |
| pu11186 | 1.00011 | 0.00093 | 0.00620 | 1.00717 | 0.00094 | 0.00634 | -0.00011 | ± | 0.01057 | -0.00717 | ± | 0.01057 |
| pu11187 | 0.99943 | 0.00087 | 0.00533 | 1.00729 | 0.00091 | 0.00540 | 0.00057 | ± | 0.01054 | -0.00729 | ± | 0.01056 |

Table 5-167 Results of MCNP Criticality Calculations

| Case | ENDF/B-VI | | | ENDF/B-V | | | ENDF/B-VI | | | ENDF/B-V | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | AENCF | $\Delta k_{eff} \pm 2\sigma$ | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| pust9-1 | 1.01103 | 0.00074 | 0.00259 | 1.01992 | 0.00082 | 0.00294 | -0.01103 | ± | 0.00676 | -0.01992 | ± | 0.00680 |
| pust9-2 | 1.01723 | 0.00078 | 0.00259 | 1.02331 | 0.00083 | 0.00262 | -0.01723 | ± | 0.00678 | -0.02331 | ± | 0.00681 |
| pust9-3 | 1.01755 | 0.00076 | 0.00245 | 1.0209 | 0.00083 | 0.00251 | -0.01755 | ± | 0.00677 | -0.02090 | ± | 0.00681 |

## 5.10 PU AND MIXED BENCHMARK CASES

### 5.10.1 Pu and Mixed Benchmarks Related to Intact Waste Package Configurations

#### 5.10.1.1 Arrays

##### 5.10.1.1.1 MIX-COMP-THERM-004

Experimental critical sizes are reported for light-water-moderated lattices with 3.01 wt.% $PuO_2$ - natural $UO_2$ fuel rods. The experiments were performed between 1972 and 1975 at Tokai Research Establishment of the Japan Atomic Energy Research Institute (JAERI). The experimental cores selected as basic benchmark experiments had fuel rods arranged in a square array. Critical sizes were determined by adjusting the water height. The water-to-fuel volume ratio in the lattice cells ranged from 2.42 to 5.55. Eleven configurations are judged to be acceptable as benchmark experiments in NEA (2001).

The Tank-Type Critical Assembly (TCA) is a light-water-moderated critical assembly in which the core is composed of a fuel rod array supported by upper, middle, and lower grid plates. The reactor is operated by raising the water level from the bottom of the core tank by a feed water pump. The experimental core lattice is constructed in an open-top cylindrical core tank having a diameter of 1.832 m and a height of 2.078 m by holding the fuel rods vertically with a set of grid plates. They are set at upper, central, and lower positions in the core tank. The six vertical grid supports are approximately 70 cm from the center of the tank. They are made of stainless steel. Light water serves as both moderator and reflector, and there are no control rods. Reactivity is controlled by adjusting the water level.

The 1.065-cm-diameter pellets are clad by a Zircaloy-2 tube. The length of the fuel region is 70.6 cm. The water-to-fuel volume ratio is changed by replacing the grid plates with another set, which has a different lattice pitch. The center-to-center rod spacing of the square lattices ranges from 1.825 to 2.474 cm. Brief descriptions of the cases are given in Table 5-168 and MCNP results are shown in Table 5-169.

Table 5-168 MIX-COMP-THERM-004 Core Dimensions of Benchmark Models

| Lattice name | Case no. | Lattice pitch (cm) | Number of rods in one side | Water level (cm)[a] | H/Pu[b] |
|---|---|---|---|---|---|
| 2.42PU | 1 | 1.825 | 23 | 59.55 | 402 |
| | 2 | | | 61.90 | |
| | 3 | | | 64.06 | |
| 2.98PU | 4 | 1.956 | 21 | 61.50 | 494 |
| | 5 | | | 64.40 | |
| | 6 | | | 69.40 | |
| 4.24PU | 7 | 2.225 | 20 | 60.32 | |
| | 8 | | | 62.99 | |
| | 9 | | | 65.63 | 703 |
| 5.55PU | 10 | 2.474 | 21 | 62.05 | |
| | 11 | | | 64.53 | 921 |

NOTES: [a] Water level measured from the bottom of the fuel pellets.
[b] H/Pu ratio is the Hydrogen to Fuel ratio as reported on page 4 of MIX-COMP-THERM-004.

Table 5-169 Results for MIX-COMP-THERM-004

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| Mixct-4-1 | 1.0000 | 0.0046 | 0.1218 | 0.9957 | 0.0011 | -0.0043 | ± | 0.0094 |
| mixct-4-2 | 1.0000 | 0.0046 | 0.1208 | 0.9987 | 0.0011 | -0.0013 | ± | 0.0094 |
| mixct-4-3 | 1.0000 | 0.0046 | 0.1208 | 0.9970 | 0.0011 | -0.0031 | ± | 0.0094 |
| mixct-4-4 | 1.0000 | 0.0039 | 0.1066 | 0.9957 | 0.0010 | -0.0043 | ± | 0.0081 |
| mixct-4-5 | 1.0000 | 0.0039 | 0.1059 | 0.9986 | 0.0010 | -0.0014 | ± | 0.0080 |
| mixct-4-6 | 1.0000 | 0.0039 | 0.1061 | 0.9992 | 0.0010 | -0.0008 | ± | 0.0081 |
| mixct-4-7 | 1.0000 | 0.004 | 0.0875 | 0.9994 | 0.0010 | -0.0006 | ± | 0.0082 |
| mixct-4-8 | 1.0000 | 0.004 | 0.0869 | 1.0030 | 0.0010 | 0.0030 | ± | 0.0082 |
| mixct-4-9 | 1.0000 | 0.004 | 0.0861 | 1.0002 | 0.0011 | 0.0002 | ± | 0.0083 |
| mixct-4-10 | 1.0000 | 0.0051 | 0.0751 | 1.0011 | 0.0009 | 0.0011 | ± | 0.0104 |
| mixct-4-11 | 1.0000 | 0.0051 | 0.0747 | 1.0015 | 0.0009 | 0.0015 | ± | 0.0104 |

### 5.10.1.1.2 MIX-COMP-THERM-010

From November 4, 1987 through April 4, 1988 11 subcritical approach experiments were performed with plutonium-uranium nitrate solutions having a Pu/(Pu + U) weight ratio of approximately 0.22 in a boiler tube-type assembly at the Critical Approach Facility at Battelle-PNL. These experiments were performed with an array of mixed oxide fuel pins in aqueous plutonium-uranium nitrate solutions. The concentration of the solutions in the boiler tube-type tank was varied from 4 to 468 g (Pu + U)/liter. The fuel pins in guide tubes were contained in a

boiler tube-type tank and arranged in a 1.4-cm square-pitch lattice, which resembled cylindrical geometry. The last of the 11 experiments was performed with the fuel pins removed from the dry guide tubes. The experiments were performed with a water reflector.

This set of experiments was designed to determine the critical height of the plutonium-uranium nitrate solution for the lattice assembly to be just critical and to determine the effectiveness of gadolinium, which was added to the solution for five of the experiments, as a neutron poison. Eleven experimental cases are acceptable as criticality benchmarks (NEA [2001]).

The lattice and moderator configuration is a carbon steel dual-cylinder tank with the boiler tube-type tank and an annular cylinder mounted inside the tank. The boiler tube-type tank was fabricated to provide a fixed array of guide tubes. During these critical experiments, 996 Type 3.2 Fast Flux Test Facility (FFTF) fuel pins were placed inside the tubes, and the tubes were surrounded by plutonium-uranium nitrate solutions at varying heights and compositions. The vessel has an outside diameter of 53.188 cm with a wall thickness of 0.091 cm. The guide tubes, which perforated the top lattice plate to allow loading and unloading of fuel pins, were attached to the top lattice plate and sealed at the bottom by 0.635-cm-long plugs, laser-welded into the ends of the tubes to prevent the plutonium-uranium nitrate solution from entering the tubes. The thickness of the top plate is 1.930 cm. Brief details (from NEA [2001, MIX-MISC-THERM-001]) are in Table 5-170 and MCNP results are in Table 5-171.

Table 5-170 MIX-COMP-THERM-010 Criticality Measurements with Plutonium-Uranium Nitrate Solution in the Boiler Tube-Type Tank [a]

| Experiment Number | Pu (g/liter) | U (g/liter) | Gd (g/liter) | Density ($g/cm^3$) [b] | Critical Height (cm) [c] |
|---|---|---|---|---|---|
| 106 | 0.88 | 2.7 | 0.0 | 1.0170 | 18.41 |
| 107 | 73.95 | 254.09 | 0.0 | 1.4833 | 21.34 |
| 109 | 47.50 | 163.06 | 0.0 | 1.3168 | 20.01 |
| 110 | 22.63 | 77.84 | 0.0 | 1.1577 | 18.90 |
| 111 | 103.70 | 363.98 | 0.0 | 1.6784 | 23.87 |
| 112 | 103.18 | 360.62 | 0.49 | 1.6758 | 30.28 |
| 113 | 102.23 | 359.69 | 0.98 | 1.6746 | 38.78 |
| 114 | 102.65 | 359.59 | 1.47 | 0.6745 | 51.06 |
| 115 | 102.68 | 359.55 | 1.97 | 1.6748 | 73.08 |
| 116 | 103.61 | 362.45 | 2.16 | 1.6790 | 90.27 |
| 117 | 83.30 | 286.57 | 0.0 | 1.5507 | 27.42 |

NOTES: [a] Boiler tube-type tank contained 996 FFTF Type 3.2 pins except for experiment 117 where the fuel pins were removed from the guide tubes. The vessel was water reflected.
[b] Density measured at 23C.
[c] Zero reference is the top of the polyethylene spacer and the bottom of the MOX fuel region.

| Table 5-171 Results for MIX-COMP-THERM-010 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment /Case | Benchmark Model | | Calculated[a] | | | $\Delta k_{eff}$ ± 2σ | | |
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | | | |
| mct10-06 | 1.0000 | 0.0044 | 0.1390 | 0.9954 | 0.0006 | -0.0046 | ± | 0.0089 |
| mct10-07 | 1.0000 | 0.0044 | 0.1453 | 0.9953 | 0.0006 | -0.0047 | ± | 0.0089 |
| mct10-09 | 1.0000 | 0.0044 | 0.1397 | 0.9976 | 0.0006 | -0.0024 | ± | 0.0089 |
| mct10-10 | 1.0000 | 0.0044 | 0.1382 | 0.9954 | 0.0006 | -0.0046 | ± | 0.0089 |
| mct10-11 | 1.0000 | 0.0044 | 0.1536 | 0.9987 | 0.0006 | -0.0013 | ± | 0.0089 |
| mct10-12 | 1.0000 | 0.0032 | 0.1514 | 1.0033 | 0.0006 | 0.0033 | ± | 0.0065 |
| mct10-13 | 1.0000 | 0.0032 | 0.1509 | 1.0041 | 0.0006 | 0.0041 | ± | 0.0065 |
| mct10-14 | 1.0000 | 0.0032 | 0.1477 | 1.0033 | 0.0006 | 0.0033 | ± | 0.0065 |
| mct10-15 | 1.0000 | 0.0066 | 0.1470 | 1.0041 | 0.0006 | 0.0041 | ± | 0.0132 |
| mct10-16 | 1.0000 | 0.0066 | 0.1455 | 1.0060 | 0.0006 | 0.0060 | ± | 0.0132 |
| mct10-17.b[b] | 1.0000 | 0.0025 | 0.0326 | 1.0051 | 0.0006 | 0.0051 | ± | 0.0052 |

NOTES: [a] The boron in the solution is modeled as $^{10}B$ while reference indicates natural boron. It is believed that since the concentration is very low (impurity level) the effect of this discrepancy on $k_{eff}$ is negligible.

[b] The concentrations supplied in section 3.3 of MIX-COMP-THERM-010 are not realistic (too high). Thus Boron concentration for this case is taken from the input supplied in the inputs directory of NEA (2001), MIX-COMP-THERM-010.

### 5.10.1.1.3 PU-MET-FAST-003

Between 1965 and 1969 at the Lawrence Livermore Laboratory, plutonium metal "parts" weighing 3 kg and 6 kg were used to form reflected and unreflected arrays, of various sizes, on an aluminum table. Five experiments were reported in sufficient detail to qualify as benchmark experiments. The basic geometrical unit for the experiments was a "Pu part." This part consisted of a 3-kg plutonium billet and its container, as well as a top and bottom heat sink made of aluminum. Some of the experiments utilized a 6-kg "double Pu part," which consisted of a bottom heat sink, a 3-kg Pu billet, a middle aluminum heat sink, another 3 kg Pu billet, and an upper aluminum heat sink. In both cases, the plutonium billet is a solid metal right circular cylinder of alpha-phase Pu surrounded by a close fitting, deep-drawn, seamless aluminum can with a steel lid.

The plutonium billet is a right circular cylinder of 19.53 ±0.08 g/cm$^3$ alpha-phase plutonium (93.56 wt.% $^{239}$Pu and 5.97 wt. % $^{240}$Pu) with a height of 4.633 ± 0.015 cm and a diameter of 6.525 ± 0.005 cm. The average weight of this billet is 3.026 ± 0.008 kg with a variation of 3.004 to 3.043 kg. The critical array spacing and brief descriptions of the experiments are provided in Table 5-172, and the results given in Table 5-173.

| Table 5-172 PU-MET-FAST-003 Summary of Critical[a] Array Spacing | | | | |
|---|---|---|---|---|
| Case Number | Description | Critical center to center separation (cm) | | Can Temp (K) |
| | | Vertical | Lateral | |
| 101 | 2x2x2 bare, 3kg part | 5.40±0.03 | 7.30±0.03 | 315 |
| 102 | 2x2x2 one side reflected, 3kg part | 5.74±0.03 | 7.64±0.03 | 309 |
| 103 | 3x3x3 bare, 3kg part | 7.71±0.03 | 9.60±0.03 | 306 |
| 104 | 3x3x3 one side reflected, 3kg part | 8.24±0.03 | 10.15±0.03 | 315 |
| 105 | 2x2x1 bare, 6kg part | n/a | 7.59±0.03 | 308 |

NOTE: [a] These experiments were taken to delayed critical.

| Table 5-173. Results for PU-MET-FAST-003 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model $k_{eff}$ | σ | AENCF | Calculated $k_{eff}$ | σ | $\Delta k_{eff} \pm 2\sigma$ | | |
| PMF003_101 | 1.0000 | 0.0030 | 1.8996 | 0.9985 | 0.0006 | -0.0015 | ± | 0.0061 |
| PMF003_102 | 1.0000 | 0.0030 | 1.8231 | 0.9924 | 0.0006 | -0.0076 | ± | 0.0061 |
| PMF003_103 | 1.0000 | 0.0030 | 1.9008 | 0.9938 | 0.0005 | -0.0062 | ± | 0.0061 |
| PMF003_104 | 1.0000 | 0.0030 | 1.8053 | 0.9946 | 0.0006 | -0.0054 | ± | 0.0061 |
| PMF003_105 | 1.0000 | 0.0030 | 1.9023 | 0.9939 | 0.0006 | -0.0061 | ± | 0.0061 |

### 5.10.1.1.4 PU-MET-FAST-004

In the late 1960s, a detailed set of plutonium metal experiments was conducted at the Lawrence Livermore Laboratory. Critical arrays of plutonium metal cylinders were built utilizing 130 3-kg plutonium metal cylinders supported above a split table apparatus. Nine unmoderated experiments were reported in sufficient detail to permit the development of benchmark specifications in NEA (2001). The building block for all of the experiments was a right circular cylinder of plutonium metal sealed in an aluminum can with a steel lid. The average height and radius of these cylinders were 4.633 cm and 3.2625 cm, respectively. The can containing the plutonium resembles a food can, with outer radius of 3.2995 cm and bottom thickness of 0.087 cm. The thickness of the steel lid was 0.021 cm. The plutonium metal cylinders were fabricated from alpha phase plutonium (93.56 wt.% $^{239}$Pu and 5.97 wt. % $^{240}$Pu) with an average weight of 3.026 ± 0.008 kg and a density of 19.53 ± 0.08 g/cm$^3$. Brief descriptions of the experiments are provided in Table 5-174 and the results are given in Table 5-175. Note that in the benchmark model Cu was replaced for Zn since Cross sections for zinc (Zn) were unavailable in the MCNP libraries. Since zinc was present only as a trace impurity, it was ommitted from components in which it occured with the exception of the bottom shoe (where it was 39 wt.%). In the bottom shoe, zinc was replaced, atom-for-atom, with copper (Cu).

| Table 5-174 PU-MET-FAST-004 Experimental Center-to-Center (CTC) Critical Spacing | | | | |
|---|---|---|---|---|
| **Case Number** | **Description** | **Vertical CTC** | **Lateral-CTC** | **Can-Temp.(K)** |
| 207 | 4x4x4 bare 3kg | 7.86 ± 0.01 | 12.51 ± 0.01 | 304 |
| 208 | 4x4x4 bare 6kg | 17.12 ± 0.02 | 17.28 ± 0.01 | 307 |
| 209 | 4x4x4 bare 6kg | 32.12 ± 0.02 | 13.09 ± 0.01 | 304 |
| 210 | 4x4x4 bare 6kg | 47.12 ± 0.02 | 11.93 ± 0.01 | 305 |
| 211 | 4x4x4 bare 6kg | 22.12 ± 0.02 | 15.23 ± 0.01 | 306 |
| 212 | 4x4x4 bare 6kg | 13.12 ± 0.02 | 20.19 ± 0.02 | 314 |
| 213 | 4x4x1 bare 6kg | - | 10.91 ± 0.02 | 304 |
| 214 | 3x3x3 bare 6kg | 13.68 ± 0.02 | 14.51 ± 0.03 | 308 |
| 215 | 2x2x2 bare 6kg | 11.98 ± 0.02 | 9.76 ± 0.03 | 303 |

| Table 5-175 Results for PU-MET-FAST-004 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| **Experiment/ Case** | **Benchmark Model** | | **Calculated** | | | **$\Delta k_{eff} \pm 2\sigma$** | | |
| | **$k_{eff}$** | **$\sigma$** | **AENCF** | **$k_{eff}$** | **$\sigma$** | | | |
| PMF004_207 | 1.0000 | 0.003 | 1.8968 | 0.9954 | 0.0011 | -0.0047 | ± | 0.0064 |
| PMF004_208 | 1.0000 | 0.003 | 1.8996 | 0.9912 | 0.0013 | -0.0088 | ± | 0.0065 |
| PMF004_209 | 1.0000 | 0.003 | 1.9040 | 0.9944 | 0.0011 | -0.0056 | ± | 0.0064 |
| PMF004_210 | 1.0000 | 0.003 | 1.8916 | 0.9949 | 0.0012 | -0.0051 | ± | 0.0065 |
| PMF004_211 | 1.0000 | 0.003 | 1.9049 | 0.9921 | 0.0013 | -0.0079 | ± | 0.0066 |
| PMF004_212 | 1.0000 | 0.003 | 1.9057 | 0.9927 | 0.0012 | -0.0073 | ± | 0.0064 |
| PMF004_213 | 1.0000 | 0.003 | 1.9060 | 0.9952 | 0.0012 | -0.0049 | ± | 0.0065 |
| PMF004_214 | 1.0000 | 0.003 | 1.9087 | 0.9930 | 0.0012 | -0.0070 | ± | 0.0065 |
| PMF004_215 | 1.0000 | 0.003 | 1.9137 | 0.9943 | 0.0012 | -0.0057 | ± | 0.0064 |

### 5.10.1.1.5 PU-MET-FAST-016

In 1982, a set of critical experiments was conducted at the Rocky Flats Critical Mass Laboratory (CML). Plutonium metal cylinders were assembled into a 3x3x3 array and flooded with water. For each of the thirteen experiments, horizontal and vertical spacings were chosen and water was added until the array was critical. Three of the measurements remained subcritical on full reflection and are not considered valid as benchmark experiments (NEA [2001]). Ten of the configurations were critical; however, four of these were repeated measurements. Therefore, six experiments are considered valid as benchmark experiments.

Twenty-seven 3-kg plutonium metal parts were used in these experiments. The plutonium part consisted of a machined right circular cylinder of plutonium metal contained in a seamless

aluminum can with a mild steel lid. The average height and diameter of the set of 130 plutonium cylinders from which these cylinders were taken was 46.33±0.15 mm and 65.25 ± 0.05 mm, respectively. The aluminum can fit closely around the plutonium and had a radial thickness of 0.37 mm. The bottom of the aluminum can was 0.87 mm thick with an irregular shape. The top was a mild steel disk 0.21 mm thick and crimped in place. This can was then placed in a steel can with an outer diameter of 76.2 a mm and a height of 69.47 mm. The thickness of the steel can sides, top, and bottom were 3.4 mm, 6.7 mm, and 9.1 mm, respectively.

The MCNP inputs for this case were taken directly from NEA (2001) and the following discrepancies with experimental data should have a negligible impact on the calculated $k_{eff}$. For the MCNP model representation the zinc impurity in the aluminum wall was not included in these computations because the zinc cross section is not available in the current MCNP cross section library. In addition the tin in the mild steel was neglegted. For these input files a slightly different plutonium density (19.52 g/cm$^3$) was used in calculating the atom densities. Thus the plutonium and americium atom densities differ slightly from the benchmark values. Also these input files have the lowest can being 25.0 cm from the bottom of the sleeve instead of 25.1 cm as indicated in the benchmark model description, so that the water height relative to the cylinders is 0.1 cm higher. The diameters of the stainless steel can and sleeve used in the sample calculations differed slightly than those found in NEA (2001). The inner and outer diameter used was 6.96 and 7.625 cm. The sleeve outer diameter was 7.945 cm. Brief descriptions of the cases are given in Table 5-176 and MCNP results are shown in Table 5-177.

Table 5-176 PU-MET-FAST-016 Center-to-Center Spacing and Critical Water Heights

| Benchmark Case # | H / $^{239}$Pu | Lateral Spacing ΔX = ΔY (cm) | Vertical Spacing ΔZ (cm) | Water Height (cm) |
|---|---|---|---|---|
| 1 | 14.2 | 12.00 | 12.75 | 48.30 |
| 2 | 16.3 | 12.70 | 12.75 | 58.40 |
| 3 | 16.9 | 12.90 | 12.75 | 60.60 |
| 4 | 17.2 | 13.00 | 12.75 | 62.78 |
| 5 | 17.3 | 13.05 | 12.75 | 65.80 |
| 6 | 17.1 | 12.80 | 13.10 | 65.60 |

Table 5-177 Results for PU-MET-FAST-016

| Experiment/ Case | Benchmark | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $K_{eff}$ | σ | | | |
| pmf016_1 | 0.9974 | 0.0042 | 1.2810 | 1.0121 | 0.0008 | 0.0147 | ± | 0.0085 |
| pmf016_2 | 1.0000 | 0.0038 | 1.2540 | 1.0022 | 0.0008 | 0.0022 | ± | 0.0078 |
| pmf016_3 | 1.0000 | 0.0033 | 1.2500 | 1.0019 | 0.0008 | 0.0019 | ± | 0.0068 |
| pmf016_4 | 1.0000 | 0.0030 | 1.2540 | 0.9987 | 0.0007 | -0.0013 | ± | 0.0062 |

| Table 5-177 Results for PU-MET-FAST-016 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $K_{eff}$ | $\sigma$ | | | |
| pmf016_5 | 1.0000 | 0.0034 | 1.2490 | 0.9987 | 0.0008 | -0.0013 | ± | 0.0070 |
| pmf016_6 | 1.0000 | 0.0032 | 1.2440 | 1.0027 | 0.0008 | 0.0027 | ± | 0.0066 |

### 5.10.1.1.6 PU-MET-FAST-017

In the late 1960s, a detailed set of plutonium metal experiments was conducted at the Lawrence Livermore Laboratory. Critical arrays of plutonium metal cylinders were built utilizing 130 3-kg plutonium metal cylinders supported above a split table apparatus. The experiments in this evaluation were moderated. Of the six moderated experiments evaluated, five moderated experiments were reported in sufficient detail to permit the development of benchmark specifications in NEA (2001). The building block for all of the experiments was a right circular cylinder of plutonium metal sealed in an aluminum can with a steel lid. The average height and radius of these cylinders was 4.633 cm and 3.2625 cm, respectively. The can containing the plutonium resembles a food can with outer radius of 3.2995 cm and bottom thickness of 0.087 cm. The thickness of the steel lid was 0.021 cm. A brief description of the experimental configurations is presented in Table 5-178. MCNP results are shown in Table 5-180.

| Table 5-178 PU-MET-FAST-017 Experimental Center-to-Center (CTC) Critical Spacing | | | | |
|---|---|---|---|---|
| Case Number | Description | Moderator Thickness | Vertical CTC | Lateral CTC |
| 201 | 4x4x4 bare 6kg | 1.27 | 25.79 ± 0.02 | 17.50 ± 0.01 |
| 202 | 4x4x4 bare 6kg | 2.54 | 25.82 ± 0.02 | 21.24 ± 0.01 |
| 203 | 4x4x4 bare 6kg | 3.81 | 25.82 ± 0.02 | 24.52 ± 0.02 |
| 204 | 4x4x4 bare 3kg | 1.27 | 9.63 ± 0.01 | 14.19 ± 0.01 |
| 205 | 4x4x4 bare 3kg | 2.54 | 13.64 ± 0.01 | 14.55 ± 0.01 |

Vertical moderation was introduced by mock-explosive moderator disks of varying thickness placed inside the support tube, above and below single and double plutonium parts. Lateral moderation was implemented by moderator jackets placed on the outside of the support tube surrounding each plutonium part. The dimensions of the various moderator units used are provided in Table 5-179. The bottom of each moderator jacket was positioned even with the bottom of the lower moderator disk for each individual plutonium part. Note that Cross sections for zinc (Zn) were unavailable in the MCNP libraries. Since zinc was present only as a trace impurity, it was ommitted from components in which it occured with the exception of the bottom shoe (where it was 39 wt.%). In the bottom shoe, zinc was replaced, atom-for-atom, with copper (Cu). Barium (Ba) and Strontium (Sr), which appear as trace elements in the moderator, were also ommitted. In addition the Tin (Sn) content in the moderator material specification is

omitted. It is believed that this will have a negligible effect on $k_{eff}$. The rationale for this is that the Sn concentration is very low and Sn is essentially transparent to neutrons (has very small cross sections).

Table 5-179 PU-MET-FAST-017 Moderator Dimensions (cm)

| Relative Thickness | Inner Disk | 3-Kg Part Jacket | 6-Kg Part Jacket |
|---|---|---|---|
| Small | Radius: 3.3975<br>Thickness: 1.278 | Height: 8.407<br>Inner Radius: 3.8100<br>Thickness: 1.267 | Height: 14.267<br>Inner Radius: 3.8125<br>Thickness: 1.267 |
| Medium | Radius: 3.3975<br>Thickness: 2.555 | Height: 10.932<br>Inner Radius: 3.8175<br>Thickness: 2.543 | Height: 16.746<br>Inner Radius: 3.8165<br>Thickness: 2.548 |
| Large | Radius: 3.3975<br>Thickness: 3.813 | Height: 13.480<br>Inner Radius: 3.8140<br>Thickness: 3.818 | Height: 19.334<br>Inner Radius: 3.8140<br>Thickness: 3.818 |

Table 5-180 Results for PU-MET-FAST-017

| Experiment/ Case | Benchmark Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| PMF017_201 | 1.0000 | 0.0030 | 1.7671 | 0.9938 | 0.0013 | -0.0062 | ± | 0.0065 |
| PMF017_202 | 1.0000 | 0.0030 | 1.6515 | 0.9955 | 0.0011 | -0.0045 | ± | 0.0064 |
| PMF017_203 | 1.0000 | 0.0030 | 1.5765 | 1.0022 | 0.0014 | 0.0022 | ± | 0.0066 |
| PMF017_204 | 1.0000 | 0.0030 | 1.6423 | 0.9974 | 0.0011 | -0.0026 | ± | 0.0064 |
| PMF017_205 | 1.0000 | 0.0030 | 1.4391 | 1.0044 | 0.0012 | 0.0044 | ± | 0.0065 |

### 5.10.1.1.7 PU-MET-FAST-037

Between late 1973 and early 1976, a set of critical experiments was conducted at the Rocky Flats Critical Mass Laboratory (CML). Plutonium metal cylinders were assembled into 2x2xN arrays, where the height N=2, 3, 4, or 5 cylinders, and flooded with water. The horizontal and vertical spacings were varied were between the experiments. The arrays were assembled and water was added until the critical water height could be determined. Results of 68 experiments were reported for this set. Forty-five measurements, which remained subcritical on full reflection and six measurements which had very large uncertainties in the extrapolation to critical water height are not considered valid as benchmark experiments. An additional experiment was questionable since the results lay between two subcritical configurations and is not considered valid for benchmarking. The remaining 16 experiments are considered valid for benchmark experiments (NEA [2001]).

A total of 20 3-kg plutonium metal parts were used in these experiments. The plutonium part consisted of a machined right circular cylinder of plutonium metal contained in a seamless aluminum can with a mild-steel lid. The average height and diameter of the set of 130 plutonium cylinders from which these cylinders were taken was 46.33 ± 0.15 mm and 65.25 ± 0.05 mm, respectively. The aluminum can fit closely around the plutonium and had a radial thickness of 0.37 mm. The bottom of the aluminum can was 0.87 mm thick with an irregular shape. The top was a mild-steel disk 0.21 mm thick and crimped in place. This can was then placed in a steel can with an outer diameter of 76.2 mm and a height of 69.47 mm. The thickness' of the steel can sides, top, and bottom were 3.4 mm, 6.7 mm, and 9.1 mm, respectively. The experimental parameters are shown in Table 5-181 below. Table 5-182 lists the results for this experiment set. Note that for the benchmark model the zinc impurity in the aluminum wall was not included in these computations because the zinc cross section is not available in the MCNP cross section library.

Table 5-181 PU-MET-FAST-037 Experimental Arrays

| Case Number | Horizontal Spacing (mm)[a] | Vertical Spacing (mm) | Between Trays (mm) | Height of Top of Tray (mm) | Water Height (mm)[b] |
|---|---|---|---|---|---|
| Array Size: 2 x 2 x 2 (24,231 g Pu) | | | | | |
| 1 | 76 | 133.35 | 50.8 | 316 | 202 ± 0.5 |
| 2 | 76 | 184.15 | 101.6 | 366.8 | 235 ± 1 |
| 3 | 76 | 214.65 | 132.1 | 397.2 | 246 ± 2 |
| 4 | 76 | 234.95 | 152.4 | 417.6 | 463 ± 3 |
| 5 | 90 | 107.95 | 25.4 | 290.6 | 235.5 ± 0.5 |
| 6 | 90 | 133.35 | 50.8 | 316 | 308.8 ± 0.5 |
| 7 | 100 | 100.58 | 18.03 | 283.2 | 263 ± 1 |
| 8 | 100 | 125.95 | 43.4 | 308.6 | 343 ± 1 |
| 9 | 100 | 128.25 | 45.7 | 310.9 | 351 ± 0.5 |
| 10 | 120 | 82.55 | 0 | 265.2 | 296 ± 1 |
| Array Size: 2 x 2 x 2 (36,309 g Pu) | | | | | |
| 11 | 76 | 387.35 | 304.8 | 957.3 | 255 ± 5 |
| 12 | 90 | 173.95 | 91.4 | 530.5 | 616 ± 2 |
| 13 | 120 | 118.15 | 35.6 | 418.8 | 460 ± 2 |
| Array Size: 2 x 2 x 4 (48,422 g Pu) | | | | | |
| 14 | 100 | 153.65 | 71.1 | 643.6 | 684 ± 2 |
| 15 | 141 | 95.25 | 12.7 | 468.3 | 421 ± 3 |
| Array Size: 2 x 2 x 5 (60,541 g Pu) | | | | | |
| 16 | 170 | 82.55 | 0 | 512.8 | 470.5 ± 1 |

NOTES: [a] Center-to-center spacing.
[b] Water heights are relative to the bottom of the tank.

Table 5-182 Results for PU-MET-FAST-037

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| pmf037_01 | 1.0000 | 0.0044 | 1.5348 | 1.0022 | 0.0007 | 0.0022 | ± | 0.0089 |
| pmf037_02 | 1.0000 | 0.0044 | 1.4853 | 0.9999 | 0.0008 | -0.0001 | ± | 0.0089 |
| pmf037_03 | 1.0000 | 0.0043 | 1.4798 | 0.9977 | 0.0008 | -0.0023 | ± | 0.0087 |
| pmf037_04 | 1.0000 | 0.0043 | 1.4556 | 1.0027 | 0.0007 | 0.0027 | ± | 0.0087 |
| pmf037_05 | 1.0000 | 0.0037 | 1.4200 | 1.0000 | 0.0008 | 0.0000 | ± | 0.0076 |
| pmf037_06 | 1.0000 | 0.0040 | 1.3708 | 0.9985 | 0.0007 | -0.0015 | ± | 0.0081 |
| pmf037_07 | 1.0000 | 0.0038 | 1.3642 | 1.0001 | 0.0008 | 0.0001 | ± | 0.0078 |
| pmf037_08 | 1.0000 | 0.0033 | 1.3188 | 1.0006 | 0.0008 | 0.0006 | ± | 0.0068 |
| pmf037_09 | 1.0000 | 0.0037 | 1.3182 | 0.9998 | 0.0007 | -0.0002 | ± | 0.0075 |
| pmf037_10 | 1.0000 | 0.0034 | 1.3436 | 1.0020 | 0.0008 | 0.0020 | ± | 0.0070 |
| pmf037_11 | 1.0000 | 0.0038 | 1.4671 | 0.9942 | 0.0008 | -0.0058 | ± | 0.0078 |
| pmf037_12 | 1.0000 | 0.0040 | 1.3496 | 0.9994 | 0.0007 | -0.0006 | ± | 0.0081 |
| pmf037_13 | 1.0000 | 0.0030 | 1.2816 | 1.0000 | 0.0008 | 0.0000 | ± | 0.0062 |
| pmf037_14 | 1.0000 | 0.0032 | 1.3015 | 1.0033 | 0.0008 | 0.0032 | ± | 0.0066 |
| pmf037_15 | 1.0000 | 0.0033 | 1.3163 | 1.0018 | 0.0008 | 0.0018 | ± | 0.0068 |
| pmf037_16 | 1.0000 | 0.0039 | 1.3649 | 1.0036 | 0.0008 | 0.0036 | ± | 0.0079 |

### 5.10.1.1.8 PU-COMP-MIXED-001

Five unreflected experiments involving arrays of plutonium-oxide-polystyrene cubes (compacts) are described in NEA (2001). The reported critical dimensions are based on experiments performed at the Hanford Plutonium Critical Mass Laboratory between May of 1963 and September of 1970. Experimental arrays were constructed from PuO2-polystyrene cubes with H/Pu ratios of 0.04, 5, 15, and 49.6. The $^{240}$Pu concentration in these cubes ranged from 2.2 - 18.35 wt.%. Although the exact cube configurations, which led to criticals are not available, the experimenters have provided simplified models for each of the experiments. All five of the experiments, which span the neutron energy spectrum from fast to intermediate to thermal, are deemed acceptable for use as criticality-safety benchmark experiments.

Although categorized as "MIXED," the majority of the fissions occur in the fast-energy range for Case 1 and in the thermal-energy range for Case 5.

These experiments were arrays of cubes or near cubes about two inches on a side, stacked on a split table. The cubes were encased in cladding material to aid in contamination control. Parameters that were reported for each experiment are listed in Table 5-183 and the results are in Table 5-184.

Table 5-183 PU-COMP-MIXED-001 Experiment Parameters

| Case | Corrected Critical Slab Dimensions (cm) | | | Experimental Data Sheet Parameters (cubes or inches) | | | | H/Pu | Pu Density ($g/cm^3$) | Wt.% $^{240}Pu$ | Critical Mass (kg) |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | Length | Width | Height | L. | W. | H. | Exp. No. | | | | |
| 1 | 30.78 | 30.78 | 20.99 ± 0.22 | 6 | 6 | 9.67 | 18-000-017 | 0.04 | 5.762 ± 0.039 | 18.35 | 114.6 ± 1.4 |
| 2 | 35.84 | 35.84 | 28.82 ± 0.05 | 7 | 7 | 11.27 | 12-005-005 | 5 | 2.302 | 11.46 | 85.03 ± 0.2 |
| 3 | 30.68 | 30.68 | 32.84 ± 0.52 | 6 | 6 | 6.25 | 2-015-017 | 15 | 1.12 ± 0.02 | 2.2 | 34.62 ± 0.55 |
| 4 | 35.63 | 35.63 | 36.04 ± 0.60 | 7 | 7 | 6.5 | 8-015-001 | 15 | 1.05 ± 0.02 | 8.06 | 48.04 ± 0.80 |
| 5 | 46.08 | 40.96 | 49.12 ± 0.50 | 9 | 8 | 9.426 | 18-050-009 | 49.6 | 0.367 ± 0.004 | 18.35 | 34.0 ± 0.3 |

Table 5-184 Results for PU-COMP-MIXED-001

| Experiment/ Case | Benchmark Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma^a$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| pcm001_1 | 0.9986 | 0.0041 | 1.7020 | 1.0286 | 0.0009 | 0.0300 | ± | 0.0084 |
| pcm001_2 | 1.0000 | 0.0068 | 0.6330 | 1.0188 | 0.0013 | 0.0188 | ± | 0.0138 |
| pcm001_3 | 0.9990 | 0.0067 | 0.2750 | 1.0150 | 0.0013 | 0.0160 | ± | 0.0137 |
| pcm001_4 | 1.0000 | 0.0066 | 0.2880 | 0.9853 | 0.0014 | -0.0147 | ± | 0.0135 |
| pcm001_5 | 0.9989 | 0.0072 | 0.0988 | 1.0101 | 0.0015 | 0.0112 | ± | 0.0147 |

NOTE: [a] The experimental uncertainty ($\sigma$) is the maximum deviation as given in page 24 of NEA (2001, PU-COMP-MIXED-001).

### 5.10.1.1.9 PU-COMP-MIXED-002

Twenty-nine Plexiglas reflected experiments involving arrays of plutonium-oxide-polystyrene cubes (compacts) are reported in NEA (2001, PU-COMP-MIXED-002). The reported critical dimensions are based on experiments performed at the Hanford Plutonium Critical Mass Laboratory between May of 1963 and September of 1970. Experimental slabs were constructed from $PuO_2$-polystyrene cubes with H/Pu ratios of 0.04, 5, 15, and 49.6. The $^{240}Pu$ concentration in these cubes ranged from 2.2 - 18.35 wt.%. Although the exact cube configurations, which led to critical are not available, the experimenters have provided simplified models for each of the experiments. All twenty-nine of the reflected experiments, which span the neutron energy spectrum from fast to intermediate to thermal, are deemed acceptable for use as criticality safety benchmark experiments in NEA (2001).

Although categorized as "MIXED," the majority of the fissions occur in the fast-energy range for Cases 1 – 5 and in the thermal-energy range for Cases 10 – 29.

These experiments were arrays of cubes or near cubes about two inches on a side, stacked on a split table. The cubes were encased in cladding material to aid in contamination control.

Parameters that were reported for each experiment are listed in Table 5-185 and the results are presented in Table 5-186.

Table 5-185 PU-COMP-MIXED-002 Experiment Characteristics

| Case | Corrected Critical Slab Dimensions (cm) | | | Experimental Data Sheet Parameters (cubes or inches) | | | | H/Pu | Pu Density ($g/cm^3$) | Wt. % $^{240}Pu$ | Critical Mass (kg) |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | Length | Width | Height | L. | W. | H. | Exp. No. | | | | |
| 1 | 25.65 | 25.65 | 10.03 ± 0.01 | 5 | 5 | 2.67 | 18-000-010 | 0.04 | 5.762 ± 0.039 | 18.35 | 38.3 ± 0.3 |
| 2 | 25.65 | 30.78 | 8.98 ± 0.01 | 5 | 6 | 2.33 | 18-000-011 | 0.04 | 5.762 ± 0.039 | 18.35 | 41.1 ± 0.3 |
| 3 | 30.78 | 30.78 | 7.97 ± 0.03 | 6 | 6 | 2 | 18-000-009 | 0.04 | 5.762 ± 0.039 | 18.35 | 43.8 ± 0.3 |
| 4 | 30.78 | 41.05 | 6.86 ± 0.08 | 6 | 8 | 1.67 | 18-000-013 | 0.04 | 5.762 ± 0.039 | 18.35 | 50.3 ± 0.7 |
| 5 | 41.05 | 41.05 | 5.95 | 8 | 8 | 1.33 | 18-000-012 | 0.04 | 5.762 ± 0.039 | 18.35 | 58.2 ± 0.3 |
| 6 | 25.6 | 25.6 | 18.33 ± 0.02 | 5 | 5 | 7.4967" | 12-005-004 | 5 | 2.302 | 11.46 | 27.66 ± 0.01 |
| 7 | 30.72 | 30.72 | 14.18 ± 0.04 | 6 | 6 | 5.815" | 12-005-023 | 5 | 2.302 | 11.46 | 30.94 ± 0.04 |
| 8 | 40.96 | 40.96 | 10.59 ± 0.02 | 8 | 8 | 4.344" | 12-005-024 | 5 | 2.302 | 11.46 | 41.20 ± 0.03 |
| 9 | 51.2 | 51.2 | 9.04 ± 0.01 | 10 | 10 | 3.692" | 12-005-025 | 5 | 2.302 | 11.46 | 54.78 ± 0.02 |
| 10 | 51.69 | 46.13 | 9.04 ± 0.06 | 10 | 8.94 | 1.75 | 2-015-067 | 15 | 1.12 ± 0.02 | 2.2 | 24.14 ± 0.18 |
| 11 | 41.35 | 38.46 | 10.34 ± 0.05 | 8 | 7.57 | 2 | 2-015-066 | 15 | 1.12 ± 0.02 | 2.2 | 18.42 ± 0.0 |
| 12 | 31.01 | 31.01 | 13.13 ± 0.02 | 6 | 6 | 2.55 | 2-015-065 | 15 | 1.12 ± 0.02 | 2.2 | 14.14 ± 0.02 |
| 13 | 25.86 | 25.86 | 16.43 ± 0.06 | 5 | 5 | 3.22 | 2-015-064 | 15 | 1.12 ± 0.02 | 2.2 | 12.30 ± 0.04 |
| 14 | 23.27 | 23.27 | 19.79 ± 0.05 | 4.5 | 4.5 | 4.17 | 2-015-020 | 15 | 1.12 ± 0.02 | 2.2 | 12.01 ± 0.03 |
| 15 | 20.68 | 20.68 | 24.87 ± 0.06 | 4 | 4 | 4.88 | 2-015-063 | 15 | 1.12 ± 0.02 | 2.2 | 11.91 ± 0.03 |
| 16 | 15.52 | 18.08 | 50.04 ± 0.13 | 3 | 3.5 | 12.79 | 2-015-054 | 15 | 1.12 ± 0.02 | 2.2 | 15.72 ± 0.04 |
| 17 | 51.31 | 68.25 | 10.36 ± 0.03 | 10 | 13.48 | 2 | 8-015-017 | 15 | 1.05 ± 0.02 | 8.06 | 38.09 ± 0.11 |
| 18 | 35.92 | 35.92 | 15.42 ± 0.03 | 7 | 7 | 3.04 | 8-015-012 | 15 | 1.05 ± 0.02 | 8.06 | 20.89 ± 0.04 |
| 19 | 30.78 | 30.78 | 18.56 ± 0.02 | 6 | 6 | 3.65 | 8-015-011 | 15 | 1.05 ± 0.02 | 8.06 | 18.46 ± 0.02 |
| 20 | 25.65 | 25.65 | 25.03 ± 0.05 | 5 | 5 | 4.93 | 8-015-016 | 15 | 1.05 ± 0.02 | 8.06 | 17.29 ± 0.03 |
| 21 | 25.65 | 25.65 | 25.13 ± 0.03 | 5 | 5 | 4.947 | 8-015-010 | 15 | 1.05 ± 0.02 | 8.06 | 17.36 ± 0.02 |
| 22 | 20.52 | 20.52 | 49.15 ± 0.01 | 4 | 4 | 9.79 | 8-015-013 | 15 | 1.05 ± 0.02 | 8.06 | 21.73 ± 0.01 |
| 23 | 61.08 | 61.08 | 16.35 ± 0.06 | 12 | 12 | 3.1325 | 18-050-001 | 49.6 | 0.367 ± 0.004 | 18.35 | 22.4 ± 0.3 |
| 24 | 50.9 | 61.08 | 17.48 ± 0.05 | 10 | 12 | 3.3975 | 18-050-002 | 49.6 | 0.367 ± 0.004 | 18.35 | 19.9 ± 0.2 |
| 25 | 50.9 | 50.9 | 18.68 ± 0.08 | 10 | 10 | 3.665 | 18-050-003 | 49.6 | 0.367 ± 0.004 | 18.35 | 17.7 ± 0.2 |
| 26 | 50.9 | 45.81 | 19.69 ± 0.07 | 10 | 9 | 3.7995 | 18-050-004 | 49.6 | 0.367 ± 0.004 | 18.35 | 16.8 ± 0.2 |
| 27 | 40.72 | 45.81 | 22.06 ± 0.06 | 8 | 9 | 4.265 | 18-050-005 | 49.6 | 0.367 ± 0.004 | 18.35 | 15.1 ± 0.2 |
| 28 | 40.72 | 40.72 | 23.58 ± 0.10 | 8 | 8 | 4.53 | 18-050-006 | 49.6 | 0.367 ± 0.004 | 18.35 | 14.3 ± 0.2 |
| 29 | 40.72 | 30.54 | 29.64 ± 0.12 | 8 | 6 | 5.7995 | 18-050-007 | 49.6 | 0.367 ± 0.004 | 18.35 | 13.5 ± 0.2 |

Table 5-186 Results for PU-COMP-MIXED-002

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma^a$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| Pcm002_01 | 0.9990 | 0.0046 | 1.0460 | 1.0318 | 0.0009 | 0.0328 | ± | 0.0094 |
| Pcm002_02 | 0.9990 | 0.0046 | 1.0300 | 1.0309 | 0.0009 | 0.0319 | ± | 0.0094 |
| Pcm002_03 | 0.9990 | 0.0046 | 1.0090 | 1.0253 | 0.0008 | 0.0263 | ± | 0.0093 |
| Pcm002_04 | 0.9990 | 0.0046 | 0.9810 | 1.0199 | 0.0009 | 0.0209 | ± | 0.0094 |
| Pcm002_05 | 0.9990 | 0.0046 | 0.9430 | 1.0139 | 0.0008 | 0.0149 | ± | 0.0093 |
| Pcm002_06 | 1.0000 | 0.0044 | 0.4390 | 1.0188 | 0.0009 | 0.0188 | ± | 0.0090 |
| Pcm002_07 | 1.0000 | 0.0044 | 0.4320 | 1.0185 | 0.0009 | 0.0185 | ± | 0.0090 |
| Pcm002_08 | 1.0000 | 0.0044 | 0.4220 | 1.0174 | 0.0008 | 0.0174 | ± | 0.0090 |
| Pcm002_09 | 1.0000 | 0.0044 | 0.4110 | 1.0192 | 0.0008 | 0.0192 | ± | 0.0090 |
| Pcm002_10 | 1.0000 | 0.0044 | 0.1840 | 1.0290 | 0.0009 | 0.0290 | ± | 0.0090 |
| Pcm002_11 | 1.0000 | 0.0044 | 0.1880 | 1.0257 | 0.0009 | 0.0257 | ± | 0.0090 |
| Pcm002_12 | 1.0000 | 0.0044 | 0.1900 | 1.0241 | 0.0009 | 0.0241 | ± | 0.0090 |
| Pcm002_13 | 1.0000 | 0.0044 | 0.1940 | 1.0228 | 0.0010 | 0.0228 | ± | 0.0090 |
| Pcm002_14 | 1.0000 | 0.0044 | 0.1940 | 1.0272 | 0.0009 | 0.0272 | ± | 0.0090 |
| Pcm002_15 | 1.0000 | 0.0044 | 0.1940 | 1.0241 | 0.0009 | 0.0241 | ± | 0.0090 |
| Pcm002_16 | 1.0000 | 0.0044 | 0.1910 | 1.0206 | 0.0009 | 0.0206 | ± | 0.0090 |
| Pcm002_17 | 0.9988 | 0.0055 | 0.1955 | 1.0060 | 0.0009 | 0.0072 | ± | 0.0111 |
| Pcm002_18 | 0.9988 | 0.0055 | 0.2037 | 1.0081 | 0.0009 | 0.0093 | ± | 0.0111 |
| Pcm002_19 | 0.9988 | 0.0055 | 0.2044 | 1.0064 | 0.0009 | 0.0076 | ± | 0.0111 |
| Pcm002_20 | 0.9988 | 0.0055 | 0.2050 | 1.0063 | 0.0009 | 0.0075 | ± | 0.0111 |
| Pcm002_21 | 0.9988 | 0.0055 | 0.2068 | 1.0084 | 0.0009 | 0.0096 | ± | 0.0111 |
| Pcm002_22 | 0.9988 | 0.0055 | 0.2054 | 1.0110 | 0.0009 | 0.0122 | ± | 0.0111 |
| Pcm002_23 | 1.0000 | 0.0046 | 0.0770 | 1.0058 | 0.0009 | 0.0058 | ± | 0.0094 |
| Pcm002_24 | 1.0000 | 0.0046 | 0.0770 | 1.0090 | 0.0009 | 0.0090 | ± | 0.0094 |
| Pcm002_25 | 1.0000 | 0.0046 | 0.0780 | 1.0081 | 0.0009 | 0.0081 | ± | 0.0094 |
| Pcm002_26 | 1.0000 | 0.0046 | 0.0770 | 1.0103 | 0.0010 | 0.0103 | ± | 0.0094 |
| Pcm002_27 | 1.0000 | 0.0046 | 0.0780 | 1.0090 | 0.0009 | 0.0090 | ± | 0.0094 |
| Pcm002_28 | 1.0000 | 0.0046 | 0.0780 | 1.0095 | 0.0009 | 0.0095 | ± | 0.0094 |
| Pcm002_29 | 1.0000 | 0.0046 | 0.0780 | 1.0104 | 0.0009 | 0.0104 | ± | 0.0094 |

NOTE: [a] The experimental uncertainty ($\sigma$) is the + deviation as given in page 25 of NEA (2001, PU-COMP- MIXED-002)

### 5.10.1.1.10 MIX-MET-FAST-011

The Zero Power Physics Reactor (ZPPR) fast critical facility was built at the Argonne National Laboratory-West (ANL-W) site in Idaho in 1969 to obtain neutron physics information necessary for the design of fast breeder reactors. ZPPR Assembly 21 consisted of a series of six criticality benchmark cores built in the ZPPR facility to provide data for validating criticality calculations for systems likely to arise in the Integral Fast Reactor (IFR) fuel processing operations. The assemblies were graphite reflected and had core fuel compositions containing different mixtures of plutonium and uranium in mock-up Pu/U/zirconium fuel. The first of these configurations was built with only plutonium fuel to take advantage of the inherent neutron source of $^{240}$Pu. Then enriched uranium was progressively substituted for plutonium in Phases 21B through 21E, which are evaluated in NEA (2001) and used in this calculation.

Each of the six phases of ZPPR Assembly 21 was designed to have a very clean and simple benchmark configuration. For any given phase the entire core was built with essentially a single unit cell. For the four mixed U- and Pu-fueled configurations of Phases 21B through 21E, the nuclides $^{239}$Pu, $^{235}$U and $^{238}$U dominated the neutronic behavior in the core region. The core isotopic composition was such that the bulk of the neutron spectrum is in the 100-keV to 4-MeV energy range and the peak is at about 600 keV. Thus, the spectrum is in the fast range. The four ZPPR Assembly 21 experiments (Phases 21B through 21E) have been determined to be acceptable criticality-safety benchmark experiments (NEA [2001]).

All configurations had the same core volume, which approximated a cylinder of about 40 cm height and a stepped radial boundary of about 19 cm equivalent radius. Six different plate-type cells were used to vary the Pu/U composition in the assemblies. The fuel compositions were varied by replacing the initial loading of Pu-U-Mo plates and high-$^{239}$Pu plates with combinations of 93%-enriched uranium plates and depleted uranium plus steel plates. The zirconium remained constant and the remainder of the cells were made of steel "void cans". In these variations, the $^{238}$U and steel content remained approximately constant. A fixed shield of natural lithium was used throughout the series of configurations to produce a calculable model without requiring significant corrections for "room-return" neutrons. The radial reflector thickness was adjusted to make each phase critical. Thus the gap between the radial reflector and the room-return shield varied among the different phases.

Each of the configurations selected for the simplified benchmark models contains 11 regions. For all four of the mixed-fissile configurations of ZPPR-21 the innermost region (1) is a cylindrical core with a radius of 18.95920 cm and half-height of 20.68936 cm. The graphite reflector region above the core is bounded radially by the same cylinder and extends axially to 30.84936 cm. Most dimensions of the remaining interior regions of the benchmark models vary between the four phases. Table 5-187 shows the principal mass loadings of the cores, and Table 5-188 gives the MCNP results.

Table 5-187 Principal Mass Loadings in the ZPPR-21 Cores, kg

| Material | 21B[a] (mmf11-01) | 21C (mmf11-02) | 21D (mmf11-03) | 21E (mmf11-04) |
|---|---|---|---|---|
| $^{239}Pu$ | 74.9 | 55.4 | 35.9 | 16.4 |
| $^{240}Pu$ | 4.9 | 4.0 | 3.1 | 2.2 |
| $^{241}Pu$ | 0.18 | 0.16 | 0.14 | 0.12 |
| $^{235}U$ | 20.5 | 61.1 | 81.4 | 122.0 |
| $^{238}U$ | 91.6 | 94.0 | 95.2 | 97.6 |
| Zr | 30.7 | 30.7 | 30.7 | 30.7 |
| Steel | 80.5 | 78.8 | 84.2 | 82.0 |

NOTE: [a] MCNP case name are given in paranthesis.

Table 5-188 Results for MIX-MET-FAST-011

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| mmf11-01 | 0.9897 | 0.0023 | 1.075 | 0.9926 | 0.0005 | 0.0029 | ± | 0.0047 |
| mmf11-02 | 0.9998 | 0.0023 | 1.064 | 0.9972 | 0.0005 | -0.0026 | ± | 0.0047 |
| mmf11-03 | 1.0018 | 0.0024 | 0.962 | 0.9999 | 0.0004 | -0.0019 | ± | 0.0049 |
| mmf11-04 | 1.0012 | 0.0024 | 0.971 | 1.0012 | 0.0005 | -0.0000 | ± | 0.0049 |

### 5.10.2 Pu and Mixed Benchmarks Related to Degraded Waste Package Configurations

#### 5.10.2.1 Pu and Mixed Solution Configurations

##### 5.10.2.1.1 MIX-SOL-THERM-005

The MIX-SOL-THERM-005 series of critical experiments was part of an extensive program jointly sponsored by the DOE and the Power Reactor and Nuclear Fuel Development Corporation (PNC) of Japan and was carried out in the mid- 1980s. The experiments evaluated in NEA (2001) were performed with mixed plutonium-uranium nitrate solution in a variable-thickness slab tank with two 106.7 cm square sides and a width that could be varied from 7.6 to 22.8 cm. The objective of these experiments was to obtain experimental data to permit the validation of computer codes for criticality calculations and of cross-section data to minimize the uncertainties inherent therein, so that facility safety, efficiency, and reliability could be enhanced. The concentrations of the solution were about 105, 293, and 435 g (Pu+U)/liter with a ratio of plutonium to total heavy metal (plutonium plus uranium) of about 0.40 for all eight experiments. Four measurements were made with a water reflector, and four with no reflector.

Seven of the eight experiments (063, 064, 071, 072, 074, 075, and 076) provide acceptable benchmark criticality data (NEA [2001]). Experiment 073, with no reflector, could not achieve criticality within vessel height limitations.

The solution slab thickness for each experiment, the measured critical heights, and the temperatures of the reflector water, where applicable, are provided in Table 5-189 and the results are given in Table 5-190. For the water-reflected experiments, the water level was even with the top of the slab tank (the midpoint of the upper bellows pleat). The empty reflector tank remained in place during the bare experiments. Note that the egg crate grids are modeled in the MCNP cases as 0.3174 versus reference value of 0.3175 cm.

Table 5-189 MIX-SOL-THERM-005 Slab Thickness and Critical Heights

| Experiment ID Number | Reflector (Temperature °C) | Slab Thickness (cm) | Critical Height (cm) [a] |
|---|---|---|---|
| 063 | bare | 19.81 | 54.70 |
| 064 | water[b] (17.3) | 13.97 | 80.14 |
| 071 | Water[b] (22.1) | 13.97 | 47.44 |
| 072 | water[b] (22.7) | 12.70 | 67.16 |
| 074 | Bare | 18.03 | 70.13 |
| 075 | Bare | 19.05 | 85.38 |
| 076 | water[b] (21.7) | 13.46 | 74.19 |

NOTES: [a] Measured from the midpoint of the bottom pleats
[b] Water height is 106.7 cm, measured from the midpoint of the bottom pleats of the slab tank.

Table 5-190 Results for MIX-SOL-THERM-005

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| Msl5-63 | 1.0000 | 0.0037 | 0.013 | 0.9877 | 0.0007 | -0.0123 | ± | 0.0076 |
| Msl5-64 | 1.0000 | 0.0037 | 0.012 | 1.0045 | 0.0007 | 0.0045 | ± | 0.0075 |
| Msl5-71 | 1.0000 | 0.0037 | 0.033 | 1.0032 | 0.0008 | 0.0032 | ± | 0.0076 |
| Msl5-72 | 1.0000 | 0.0037 | 0.032 | 1.0001 | 0.0008 | 0.0001 | ± | 0.0076 |
| Msl5-74 | 1.0000 | 0.0037 | 0.037 | 0.9922 | 0.0008 | -0.0078 | ± | 0.0076 |
| Msl5-75 | 1.0000 | 0.0037 | 0.059 | 0.9898 | 0.0009 | -0.0103 | ± | 0.0076 |
| Msl5-76 | 1.0000 | 0.0037 | 0.049 | 0.9974 | 0.0008 | -0.0026 | ± | 0.0075 |

### 5.10.2.1.2 PU-SOL-THERM-002

Seven experiments that were performed in the P-11 area of the Hanford Reservation in the early 1950s are evaluated in NEA (2001). These experiments are commonly referred to as the P-11 experiments. The experiments were directed toward determining the effect of geometry, concentration, foreign atoms, plutonium isotopic content, neutron reflection, and temperature on the critical mass of light water moderated and reflected homogeneous plutonium solutions. The series of experiments used a number of spherical containers and cylindrical containers. The reported plutonium compositions included only two plutonium isotopes, $^{239}$Pu at 96.88 wt.% and $^{240}$Pu at 3.12 wt.%. Seven plutonium nitrate solution experiments in a 12-inch sphere with a water reflector are evaluated in NEA (2001). The plutonium nitrate solution was contained in a nominally 12-inch diameter sphere, which was, in turn, inside a large cylindrical tank that provided at least 12 inches of water reflection around the sphere. All seven experiments are acceptable benchmark experiments (NEA [2001]). A brief description of the experiments is provided in Table 5-191 and the results are given in Table 5-192.

Table 5-191 PU-SOL-THERM-002 Evaluated Case Data for 12-Inch Diameter, 15.19-Liter Spheres

| Case Number | Critical Mass (g) | $\rho_{Pu}$ (g/l) | Solution Density (g/cm$^3$) | Total $\rho_{NO3}$ (g/l) | $\rho_{Fe}$ (g/l) | $\rho_{H2O}$ (g/l) |
|---|---|---|---|---|---|---|
| 1 | 757 | 49.84 | 1.1312 | 138.5 | 0.189 | 941.26 |
| 2 | 781 | 51.42 | 1.1482 | 163 | 0.272 | 931.72 |
| 3 | 852 | 56.09 | 1.1682 | 207 | 0.245 | 902.45 |
| 4 | 906 | 59.64 | 1.1898 | 237 | 0.192 | 890.12 |
| 5 | 962 | 63.33 | 1.2152 | 270 | 0.197 | 878.35 |
| 6 | 1065 | 70.11 | 1.2412 | 322 | 0.218 | 844.82 |
| 7 | 1171 | 77.09 | 1.2695 | 359 | 0.237 | 828.64 |

Table 5-192 Results for PU-SOL-THERM-002

| Experiment/ Case | Benchmark-Model $k_{eff}$ | σ | AENCF | Calculated $k_{eff}$ | σ | $\Delta k_{eff} \pm 2\sigma$ | | | H/X |
|---|---|---|---|---|---|---|---|---|---|
| pst002_1 | 1.0000 | 0.0047 | 0.009 | 1.0095 | 0.0010 | 0.0095 | ± | 0.0096 | 508.0 |
| pst002_2 | 1.0000 | 0.0047 | 0.010 | 1.0115 | 0.0010 | 0.0115 | ± | 0.0096 | 489.2 |
| pst002_3 | 1.0000 | 0.0047 | 0.010 | 1.0083 | 0.0010 | 0.0083 | ± | 0.0096 | 437.3 |
| pst002_4 | 1.0000 | 0.0047 | 0.011 | 1.0126 | 0.0010 | 0.0126 | ± | 0.0096 | 407.5 |
| pst002_5 | 1.0000 | 0.0047 | 0.012 | 1.0144 | 0.0010 | 0.0144 | ± | 0.0096 | 392.8 |
| pst002_6 | 1.0000 | 0.0047 | 0.013 | 1.0105 | 0.0010 | 0.0105 | ± | 0.0096 | 333.5 |
| pst002_7 | 1.0000 | 0.0047 | 0.014 | 1.0131 | 0.0010 | 0.0131 | ± | 0.0096 | 299.3 |

### 5.10.2.1.3 PU-SOL-THERM-006

Three experiments that were performed in the P-11 area of the Hanford Reservation in the early 1950's are evaluated in NEA (2001). These experiments came from a group of experiments commonly referred to as the P-11 experiments. The experiments were directed toward determining the effect of geometry, concentration, foreign atoms, plutonium isotopic content, neutron reflection, and temperature on the critical mass of light water moderated and reflected homogeneous plutonium solutions. The series of experiments used a number of spherical containers and cylindrical containers. Only the plutonium nitrate solution experiments in the 15-inch sphere with 3.12 weight percent $^{240}$Pu are discussed here. The reported plutonium compositions included only $^{239}$Pu and $^{240}$Pu. The plutonium nitrate solution was contained in a nominally 15-inch diameter sphere, which was inside a large cylindrical tank that provided effectively full water reflection around the sphere.

Table 5-193 shows the data for the experiments with fully reflected 15-inch solution spheres that were reported and Table 5-194 shows the results.

Table 5-193 PU-SOL-THERM-006 Data for experiments with fully reflected 15-inch solution spheres

| Case Number | Wt. % $^{240}$Pu | Critical Mass (g) | $\rho_{pu}$ (g/l) | Solution Density (g/cm$^3$) | Total $\rho_{NO3}$ (g/l) | $\rho_{Fe}$ (g/l) | $\rho_{H2O}$ (g/l) |
|---|---|---|---|---|---|---|---|
| 1 | 3.12 | 719 | 24.8 | 1.0899 | 116 | 0.088 | 947.54 |
| 2 | 3.12 | 741 | 25.56 | 1.1068 | 147 | 0.094 | 932.19 |
| 3 | 3.12 | 782 | 26.97 | 1.1409 | 212 | 0.096 | 898.84 |

Table 5-194 Results for PU-SOL-THERM-006

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/X |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| pst006_1 | 1.0000 | 0.0035 | 0.005 | 1.0080 | 0.0009 | 0.0080 | ± | 0.0072 | 1028.2 |
| pst006_2 | 1.0000 | 0.0035 | 0.005 | 1.0103 | 0.0009 | 0.0103 | ± | 0.0072 | 986.2 |
| pst006_3 | 1.0000 | 0.0035 | 0.006 | 1.0076 | 0.0009 | 0.0076 | ± | 0.0072 | 910.9 |

### 5.10.2.1.4 PU-SOL-THERM-008

Evaluated in NEA (2001) are plutonium solution experiments performed in 1961-1962 at the Battelle PNL Critical Mass Laboratory (CML), in 14-inch-diameter spherical steel shells with spherical or hemispherical concrete reflectors of various thicknesses. The solutions were plutonium nitrate having various plutonium concentrations and with the plutonium having a 240 Pu weight percent reported as 4.57. The experiment equipment consisted of a stainless steel spherical shell, nominally 14 inches in diameter, having as a reflector concrete of various

thicknesses and either fully enclosing the sphere or as a hemispherical shell. Some experiments had an air gap between the steel tank and the concrete. The 14-inch-diameter sphere was reported to have a measured volume of 23.22±0.02 liters when full. This volume requires an effective radius of 17.698 cm.

Table 5-195 lists configuration details for 14 acceptable full-sphere critical configurations and 15 acceptable critical experiments (NEA [2001]). One critical configuration was found to be unacceptable. One other experiment was found to be insufficiently described to use.

Table 5-195 PU-SOL-THERM-008 Data for Concrete-Reflected 14-inch Spheres.

| Case | Type-of-Refl.[a] | gPu/l | Spec.Grav.[b] | Acid-Mol. | Total-$gNO_3$/l | $gH_2O$/l | Volume (l) | Crit.Mass (kg)[c] |
|---|---|---|---|---|---|---|---|---|
| 1 | E | 35.5 | 1.143 | 2.89 | 216.02 | 888.57(889) | 23.22 | 0.82 |
| 2 | E | 45.2 | 1.253 | 5.86 | 410.23 | 791.66(792) | 23.22 | 1.05 |
| 3 | E-2 | 46.4 | 1.252 | 5.62 | 396.60 | 803.34 | 23.22 | 1.08 |
| 4 | E-1 | 46.9 | 1.258 | 5.62 | 397.12 | 808.32 | 23.22 | 1.09 |
| 5 | E | 32.8 | 1.093 | 1.1 | 102.23 | 956.86 | 23.22 | 0.76 |
| 6 | E-3 | 40 | (1.097)[f] | 1.1 | 109.70 | 946.19 | 23.22 | 0.93 |
| 7 | A | 29.6 | 1.085 | 1.07 | 97.05 | 957.27 | 23.22 | 0.69 |
| 8 | E-3 | 50.9 | 1.222 | 4.57 | 336.16 | 830.33 | 23.22 | 1.18 |
| 9 | E-3 | 232 | (1.578)[f] | 6.7 | 656.08 | 683.17 | 23.22 | 5.39 |
| 10 | A | 43.4 | 1.285 | 6.37 | 440.00 | 795.18 | 23.22 | 1.01 |
| 11 | A | 36.6 | 1.209 | 4.39 | 310.17 | 857.81 | 23.22 | 0.85 |
| 12 | C | 67.9 | 1.25 | 4.29 | 336.43 | 841.35 | 23.22 | 1.58 |
| 13 | C | 50.4 | 1.175 | 2.28 | 193.65 | 928.65 | 23.22 | 1.17 |
| 14 | D | 75 | 1.193 | 2.1 | 208.01 | 907.87 | 23.22 | 1.74 |
| 15 | B | 46.4 | 1.146 | 2.05 | 175.24 | 922.29 | 23.22 | 1.08 |

NOTES: [a] A = 10-inch-thick close-fitting concrete.
B = 10-inch-thick close-fitting concrete hemisphere.
C = 4-inch air gap + 6-inch-thick concrete spherical shell.
D = 4-inch air gap + 6-inch-thick concrete hemispherical shell.
E = 4-inch-thick close-fitting concrete.
E-1 = Additional 0.072 inch stainless steel on shell and 4-inch-thick close fitting concrete.
E-2 = Additional 0.036 inch stainless steel on shell and 4-inch-thick close fitting concrete.
E-3 = 0.030-inch cadmium layer on shell and 4-inch-thick close-fitting concrete.
[b] As reported, but actually solution density in grams per cubic centimeter.
[c] Calculated using data in Columns 3 and 8.

The models used in this evaluation are based on reported critical configurations as adjusted from experiments in NEA (2001) and have spheres completely full of solution and include the inlet and outlet tubes in the geometry. Note that case 6 shown in Table 5-195 is not modeled since it was determined to be unacceptable as a benchmark (NEA [2001]). The MCNP results for PU-SOL-THERM-008 are shown in Table 5-196.

Table 5-196 Results for PU-SOL-THERM-008

| Experiment/Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/Pu[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | | | | |
| pst008_1.10A | 1.0000 | 0.0033 | 0.0070 | 1.0098 | 0.0005 | 0.0098 | ± | 0.0067 | 686.2 |
| pst008_2.10A | 1.0000 | 0.0037 | 0.0100 | 1.0111 | 0.0005 | 0.0111 | ± | 0.0075 | 497.6 |
| pst008_3.10A | 1.0000 | 0.0040 | 0.0100 | 1.0193 | 0.0005 | 0.0193 | ± | 0.0081 | 490.2 |
| pst008_4.10A | 1.0000 | 0.0040 | 0.0100 | 1.0238 | 0.0005 | 0.0238 | ± | 0.0081 | 487.8 |
| pst008_5.10A | 1.0000 | 0.0028 | 0.0070 | 1.0232 | 0.0005 | 0.0232 | ± | 0.0057 | 785.0 |
| pst008_7.10A | 1.0000 | 0.0028 | 0.0060 | 1.0184 | 0.0005 | 0.0184 | ± | 0.0057 | 870.0 |
| pst008_8.10A | 1.0000 | 0.0040 | 0.0110 | 1.0029 | 0.0005 | 0.0029 | ± | 0.0081 | 456.0 |
| pst008_9.10A | 1.0000 | 0.0061 | 0.0540 | 1.0072 | 0.0006 | 0.0072 | ± | 0.0123 | 85.4 |
| pst008_10.10A | 1.0000 | 0.0037 | 0.0090 | 1.0291 | 0.0005 | 0.0291 | ± | 0.0075 | 523.1 |
| pst008_11.10A | 1.0000 | 0.0031 | 0.0080 | 1.0227 | 0.0005 | 0.0227 | ± | 0.0063 | 652.9 |
| pst008_12.10A | 1.0000 | 0.0041 | 0.0140 | 1.0215 | 0.0005 | 0.0215 | ± | 0.0083 | 345.1 |
| pst008_13.10A | 1.0000 | 0.0041 | 0.0100 | 1.0293 | 0.0005 | 0.0293 | ± | 0.0083 | 501.5 |
| pst008_14.10A | 1.0000 | 0.0042 | 0.0160 | 1.0114 | 0.0006 | 0.0114 | ± | 0.0085 | 329.1 |
| pst008_15.10A | 1.0000 | 0.0041 | 0.0100 | 1.0164 | 0.0005 | 0.0164 | ± | 0.0083 | 539.9 |

NOTE: [a] H/Pu is the Hydrogen to Plutonium (Pu isotopes) ratio as calculated from the number densities used in the benchmark models.

### 5.10.2.1.5 PU-SOL-THERM-012

The experiments described in this section considered plutonium nitrate solution in a large right parallelepiped tank, 130 x 130 x 100 cm. The tank was either fully reflected by water on six sides, water reflected on five sides, or not water reflected. A plutonium concentration range of 13.2 to 105 g/l was examined. The plutonium contained 18.88 wt.% $^{240}$Pu. The experimental configuration consisted of a right parallelepiped tank, with a flat bottom, having inside dimensions of 130 cm x 130 cm horizontally and 100 cm vertically, and a 0.5-cm wall thickness. This tank was inside a large pool and was supported by four support feet, 50 cm high. Configuration descriptions for benchmarks are shown in Table 5-197 and the results are presented in Table 5-198. All 23 experiments are considered acceptable for use as benchmark experiments (NEA [2001]).

Table 5-197 PU-SOL-THERM-012 Configuration Descriptions for Benchmarks

| Pu conc. | 5 Sides Water Reflection | | | 6 Sides Water Reflection | | | No Close Water Reflection | | |
|---|---|---|---|---|---|---|---|---|---|
| (g/l) | Cases | Critical Volume (l) | Critical height (cm)[a] | Cases | Critical Volume (l) | Critical height (cm)[a] | Cases | Critical Volume (l) | Critical height (cm)[a] |
| 105.0 | 6(b) | 275.8±0.7 | 16.32±0.04 | - | - | - | - | - | - |
| 84.0 | 7(b) | 275.8±0.7 | 16.32±0.04 | - | - | - | - | - | - |

Table 5-197 PU-SOL-THERM-012 Configuration Descriptions for Benchmarks

| Pu conc. | 5 Sides Water Reflection | | | 6 Sides Water Reflection | | | No Close Water Reflection | | |
|---|---|---|---|---|---|---|---|---|---|
| (g/l) | Cases | Critical Volume (l) | Critical height (cm)[a] | Cases | Critical Volume (l) | Critical height (cm)[a] | Cases | Critical Volume (l) | Critical height (cm)[a] |
| 52.7 | 8(b) | 297.9±0.7 | 17.63±0.04 | - | - | - | 14(b) | 328.3±0.7 | 19.43±0.04 |
| 31.9 | 9(b) | 360.1±0.7 | 21.31±0.04 | - | - | - | 15(b) | 388.3±0.7 | 22.98±0.04 |
| 26.9 | 10(b) | 390.5±0.7 | 23.11±0.04 | - | - | - | 16(b) | 418.4±0.7 | 24.76±0.04 |
| 21.7 | 11(b) | 458.8±0.7 | 27.15±0.04 | - | - | - | 17(b) | 486.7±0.8 | 28.80±0.05 |
| 19.7 | 12(b) | 505.4±0.8 | 29.90±0.05 | 1(c) | 476.0±1.0 | 28.17±0.06 | 18(b) | 532.6±0.8 | 31.52±0.05 |
| 17.7 | - | - | - | 2(c) | 545.0±1.0 | 32.25±0.06 | 19(d) | 600.4±1.1 | 35.53±0.07 |
| 16.7 | - | - | - | 3(c) | 599.0±1.0 | 35.44±0.06 | 20(d) | 652.4±1.2 | 38.60±0.07 |
| 14.7 | - | - | - | 4(c) | 772.0±1.0 | 45.68±0.06 | 21(d) | 829.6±1.3 | 49.09±0.08 |
| 13.2 | 13(b) | 1140.0±1.0 | 67.46±0.06 | 5(c) | 1118.0±1.0 | 66.15±0.06 | 22(d) | 1175.8±1.5 | 69.57±0.09 |
| 13.2 | | | | | | | 23(b) | 1183.0±1.0 | 70.00±0.06 |

NOTES: [a] True critical height deduced from the critical volume: = volume /(130 x 130 cm)
[b] Lucoflex box, empty, positioned at the top position
[c] Lucoflex box, full of water, floating on nitrate solution
[d] Lucoflex box, full of water, positioned at the top position.

Table 5-198 Results for PU-SOL-THERM-012

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/Pu[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| pst012_01 | 1.0000 | 0.0043 | 0.0048 | 1.0115 | 0.0009 | 0.0115 | + | 0.0088 | 1273.6 |
| pst012_02 | 1.0000 | 0.0043 | 0.0042 | 1.0118 | 0.0008 | 0.0118 | + | 0.0087 | 1420.4 |
| pst012_03 | 1.0000 | 0.0058 | 0.0040 | 1.0124 | 0.0007 | 0.0124 | + | 0.0117 | 1505.7 |
| pst012_04 | 1.0000 | 0.0058 | 0.0036 | 1.0117 | 0.0007 | 0.0117 | + | 0.0117 | 1710.0 |
| pst012_05 | 1.0000 | 0.0058 | 0.0035 | 1.0178 | 0.0006 | 0.0178 | + | 0.0117 | 1903.6 |
| pst012_06 | 1.0000 | 0.0007 | 0.0222 | 1.0075 | 0.0013 | 0.0075 | + | 0.0030 | 228.6 |
| pst012_07 | 1.0000 | 0.0013 | 0.0171 | 1.0080 | 0.0013 | 0.0080 | + | 0.0036 | 291.3 |
| pst012_08 | 1.0000 | 0.0013 | 0.0111 | 1.0121 | 0.0012 | 0.0121 | + | 0.0035 | 467.9 |
| pst012_09 | 1.0000 | 0.0043 | 0.0070 | 1.0157 | 0.0011 | 0.0157 | + | 0.0089 | 779.8 |
| pst012_10 | 1.0000 | 0.0043 | 0.0061 | 1.0111 | 0.0009 | 0.0111 | + | 0.0088 | 926.0 |
| pst012_11 | 1.0000 | 0.0043 | 0.0051 | 1.0149 | 0.0010 | 0.0149 | + | 0.0088 | 1159.0 |
| pst012_12 | 1.0000 | 0.0043 | 0.0047 | 1.0144 | 0.0008 | 0.0144 | + | 0.0088 | 1273.6 |
| pst012_13 | 1.0000 | 0.0058 | 0.0034 | 1.0141 | 0.0006 | 0.0141 | + | 0.0117 | 1903.6 |
| pst012_14 | 1.0000 | 0.0013 | 0.0118 | 1.0074 | 0.0011 | 0.0074 | + | 0.0034 | 467.9 |
| pst012_15 | 1.0000 | 0.0043 | 0.0074 | 1.0161 | 0.0011 | 0.0161 | + | 0.0089 | 779.8 |

Table 5-198 Results for PU-SOL-THERM-012

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/Pu[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| pst012_16 | 1.0000 | 0.0043 | 0.0063 | 1.0111 | 0.0010 | 0.0111 | ± | 0.0088 | 926.0 |
| pst012_17 | 1.0000 | 0.0043 | 0.0051 | 1.0144 | 0.0009 | 0.0144 | ± | 0.0088 | 1159.0 |
| pst012_18 | 1.0000 | 0.0043 | 0.0050 | 1.0149 | 0.0009 | 0.0149 | ± | 0.0088 | 1273.6 |
| pst012_19 | 1.0000 | 0.0043 | 0.0043 | 1.0153 | 0.0009 | 0.0153 | ± | 0.0088 | 1420.4 |
| pst012_20 | 1.0000 | 0.0058 | 0.0042 | 1.0149 | 0.0008 | 0.0149 | ± | 0.0117 | 1505.7 |
| pst012_21 | 1.0000 | 0.0058 | 0.0038 | 1.0141 | 0.0007 | 0.0141 | ± | 0.0117 | 1710.0 |
| pst012_22 | 1.0000 | 0.0058 | 0.0035 | 1.0155 | 0.0007 | 0.0155 | ± | 0.0117 | 1903.6 |
| pst012_23 | 1.0000 | 0.0058 | 0.0034 | 1.0151 | 0.0006 | 0.0151 | ± | 0.0117 | 1903.6 |

NOTE: [a] H/Pu is the Hydrogen to Plutonium (Fuel) ratio as calculated from the number densities for the solutions in Tables 9.a and 9.b of PU-SOL-THERM-012.

### 5.10.2.1.6 PU-SOL-THERM-013

The experiments in this section are a subset of criticality measurements involving interacting arrays which were conducted in 1968 at the Valduc facility near the city of Dijon, France by the Protection and Radiological Safety Service section of the French Commissariat a l'Energie Atomique.

These experiments involved about 100 configurations of between two and nine interacting cylindrical tanks of uranium and plutonium nitrate solutions of varying concentrations, suspended in air in a large concrete building. The plutonium contained 4.23 wt.% $^{240}$Pu. This section lists experiments dealing with plutonium nitrate solution of 115.1gPu/l in cylindrical tanks of 256-mm outer diameter in the following configurations:

i) Two cylinders
ii) Three cylinders, whose centers lie at the corners of an equilateral triangle
iii) Three cylinders, whose centers lie at the corners of a right-angled isosceles triangle
iv) Four cylinders, whose centers lie at the corners of a square
v) Six cylinders on two lines
vi) Seven cylinders; six forming a hexagon with the seventh occupying the center
vii) Nine cylinders on three lines, the centers forming a square array.

The inside and outside diameters of the solution-input pipe of the cylindrical tanks are 1.5 and 2.1 cm respectively. The cylindrical tank bottom has a conical shape with a maximum thickness of 1.5 cm at the sides. The inverted cone has a base angle of 2°. The cylinder tank wall is 0.3 cm thick with an outside diameter of 25.6 cm and an outside height of 103.7 cm. The cylinder tank top thickness is 1.2 cm. The temperature of the solution in the experiments was maintained in a range between 23 °C and 27 °C. Note that case 3 was not modeled. Table 5-199 gives a brief

description of the experiments, and Table 5-200 provides the MCNP results. These experiments are acceptable as benchmarks (NEA [2001]).

Table 5-199 PU-SOL-THERM-013 Cylindrical Tank Separations of the Various Experimental Configurations and their Critical Heights

| Geometric Configuration | Case Number | Surface-to-Surface Distance (cm) | Temperature (°C) | Critical Height (cm) | Critical Volume (l) |
|---|---|---|---|---|---|
| Two Cylinders | 1 | 0 | 25.4 | 52.96±0.02 | 51.93±0.08 |
| | 2 | 5 | 26.4 | 78.72±0.02 | 77.19±0.01 |
| | | | | | |
| | | | | | |
| Three Cylinders in Equilateral Triangle | 4 | 5 | 24.4 | 41.09±0.02 | 60.43±0.09 |
| | 5 | 10 | 25.7 | 53.14±0.02 | 78.2±0.1 |
| | 6 | 20 | 25 | 75.58±0.02 | 111.2±0.2 |
| | | | | | |
| Three Cylinders Right angled Isosceles Triangle | 7 | 0 | 23.3 | 36.05±0.02 | 53.02±0.09 |
| | 8 | 5 | 25 | 48.75±0.02 | 71.7±0.1 |
| | 9 | 10 | 24.5 | 60.99±0.02 | 89.7±0.2 |
| | 10 | 20 | 25.7 | 86.26±0.02 | 126.9±0.2 |
| | | | | | |
| Four Cylinders in a Square | 11 | 0 | 24.3 | 26.71±0.02 | 52.37±0.09 |
| | 12 | 5 | 24.9 | 36.32±0.02 | 71.2±0.1 |
| | 13 | 10 | 24.9 | 45.04±0.02 | 88.3±0.2 |
| | 14 | 20 | 24.7 | 63.22±0.02 | 124.0±0.2 |
| | | | | | |
| Six Cylinders in Two Lines | 15 | 0 | 24.1 | 24.01±0.02 | 70.6±0.20 |
| | 16 | 5 | 24.3 | 31.89±0.02 | 93.8±0.2 |
| | 17 | 10 | 24.8 | 39.25±0.02 | 115.5±.2 |
| | | | | | |
| Seven Cylinders in a Hexagon | 18 | 1.2 | 24.8 | 21.43±0.02 | 73.5±.02 |
| | 19 | 5 | 25.2 | 27.10±0.02 | 93.0±0.2 |
| | 20 | 10 | 25.6 | 33.69±0.02 | 115.6±.02 |
| | | | | | |
| Nine Cylinders in Three Lines | 21 | 0 | 25.3 | 21.48±0.02 | N/A |
| | 22 | 5 | 26.8 | 28.05±0.02 | N/A |

Table 5-200 Results for PU-SOL-THERM-013

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| pst13-01 | 0.9980 | 0.0040 | 0.0247 | 1.0088 | 0.0008 | 0.0108 | ± | 0.0082 |
| pst13-02 | 0.9980 | 0.0040 | 0.0251 | 1.0080 | 0.0008 | 0.0100 | ± | 0.0082 |
| pst13-04 | 0.9965 | 0.0052 | 0.0252 | 1.0060 | 0.0008 | 0.0095 | ± | 0.0105 |
| pst13-05 | 0.9965 | 0.0052 | 0.0251 | 1.0075 | 0.0008 | 0.0110 | ± | 0.0105 |
| pst13-06 | 0.9965 | 0.0052 | 0.0251 | 1.0062 | 0.0009 | 0.0097 | ± | 0.0105 |
| pst13-07 | 0.9965 | 0.0052 | 0.0246 | 1.0093 | 0.0008 | 0.0128 | ± | 0.0105 |
| pst13-08 | 0.9965 | 0.0052 | 0.0249 | 1.0095 | 0.0008 | 0.0130 | ± | 0.0105 |
| pst13-09 | 0.9965 | 0.0052 | 0.0247 | 1.0060 | 0.0009 | 0.0095 | ± | 0.0105 |
| pst13-10 | 0.9965 | 0.0052 | 0.0249 | 1.0060 | 0.0009 | 0.0095 | ± | 0.0105 |
| pst13-11 | 0.9965 | 0.0052 | 0.0251 | 1.0120 | 0.0008 | 0.0155 | ± | 0.0105 |
| pst13-12 | 0.9965 | 0.0052 | 0.0249 | 1.0078 | 0.0008 | 0.0113 | ± | 0.0105 |
| pst13-13 | 0.9965 | 0.0052 | 0.0250 | 1.0053 | 0.0008 | 0.0088 | ± | 0.0105 |
| pst13-14 | 0.9965 | 0.0052 | 0.0250 | 1.0080 | 0.0008 | 0.0114 | ± | 0.0105 |
| pst13-15 | 0.9959 | 0.0062 | 0.0243 | 1.0149 | 0.0008 | 0.0190 | ± | 0.0125 |
| pst13-16 | 0.9959 | 0.0062 | 0.0251 | 1.0081 | 0.0008 | 0.0122 | ± | 0.0125 |
| pst13-17 | 0.9959 | 0.0062 | 0.0251 | 1.0072 | 0.0008 | 0.0113 | ± | 0.0125 |
| pst13-18 | 0.9959 | 0.0062 | 0.0245 | 1.0063 | 0.0008 | 0.0104 | ± | 0.0125 |
| pst13-19 | 0.9959 | 0.0062 | 0.0248 | 1.0069 | 0.0008 | 0.0110 | ± | 0.0125 |
| pst13-20 | 0.9959 | 0.0062 | 0.0249 | 1.0059 | 0.0008 | 0.0100 | ± | 0.0125 |
| pst13-21 | 0.9966 | 0.0043 | 0.0245 | 1.0122 | 0.0008 | 0.0105 | ± | 0.0087 |
| pst13-22 | 0.9966 | 0.0043 | 0.0250 | 1.0070 | 0.0008 | 0.0104 | ± | 0.0087 |

### 5.10.2.1.7 PU-SOL-THERM-014

The experiments described in this section are a subset of criticality measurements involving interacting arrays which were conducted in 1968 at the Valduc facility near the city of Dijon, France by the Protection and Radiological Safety Service, section of the French Commissariat a l'Energie Atomique. This section lists the following configurations:

i) Two cylinders
ii) Three cylinders in a line
iii) Three cylinders, whose centers lie at the corners of an equilateral triangle
iv) Three cylinders, whose centers lie at the corners of a right-angled isosceles triangle
v) Four cylinders, whose centers lie at the corners of a square

vi) Four cylinders, whose centers lie at the corners of a rectangle of fixed width (15 cm).

Inside and outside diameters of the solution input pipe of the cylindrical tanks are 15 and 21-mm, respectively. The tank bottom has a conical shape with a maximum thickness of 1.5 cm at the sides. The inverted cone has a base angle of 2°. The cylinder wall is 3 mm thick with an outside diameter of 300 mm and an outside height of 1037 mm. The cylinder top thickness is 12 mm. The complete set of experiments involved about 100 cases and 35 experimental cases were judged to be acceptable benchmarks (NEA [2001]). The H/Fuel (All Pu isotopes plus Am) ratio as calculated from the solution data in section 3.3 of NEA (2001, PU-SOL-THERM-014) is approximately 210. Table 5-201 gives a brief description of the cases, and Table 5-202 provides the MCNP results.

Table 5-201 PU-SOL-THERM-014 Cylindrical Tank Separations of the Various Experimental Configurations and their Critical Heights

| Geometric Configuration | Case Number | Surface-to-Surface Distance (cm) | Temperature (°C) | Critical Height (cm) | Critical Volume (l) |
|---|---|---|---|---|---|
| | | | | | |
| Two Cylinders | 1 | 0 | 25.5 | 30.54±0.02 | 41.47±0.07 |
| | 2 | 5 | 25 | 33.80±0.02 | 45.90±0.08 |
| | 3 | 15 | 25.7 | 36.67±0.02 | 49.80±0.08 |
| | 4 | 30 | 25.5 | 38.39±0.02 | 52.13±0.08 |
| | 5 | 50 | 26.2 | 39.50±0.02 | 53.64±0.08 |
| | 6 | 145 | 25.5 | 40.37±0.02 | 54.82±0.09 |
| | | | | | |
| Three Cylinders in Line | 7 | 0 | 25 | 28.30±0.02 | 57.65±0.10 |
| | 8 | 5 | 24.2 | 31.84±0.02 | 64.85±0.10 |
| | 9 | 15 | 24.4 | 35.19±0.02 | 71.7±0.2 |
| | 10 | 30 | 24.9 | 37.47±0.02 | 76.3±0.2 |
| | 11 | 60 | 24.5 | 39.09±0.02 | 79.6±0.2 |
| | 12 | 120 | 24.8 | 40.03±0.02 | 81.5±0.2 |
| | | | | | |
| Three Cylinders in Equilateral Triangle | 13 | 0 | 24.5 | 23.12±0.03 | 47.1±0.1 |
| | 14 | 10 | 24.7 | 31.26±0.02 | 63.7±0.1 |
| | 15 | 30 | 25.6 | 36.28±0.02 | 73.9±0.2 |
| | 16 | 60 | 25.1 | 38.53±0.02 | 78.5±0.2 |
| | 17 | 120 | 24.6 | 39.77±0.02 | 81.0±0.2 |
| | | | | | |

Table 5-201 PU-SOL-THERM-014 Cylindrical Tank Separations of the Various Experimental Configurations and their Critical Heights

| Geometric Configuration | Case Number | Surface-to-Surface Distance (cm) | Temperature (°C) | Critical Height (cm) | Critical Volume (l) |
|---|---|---|---|---|---|
| Three Cylinders in Right Isosceles Triangle | | | | | |
| | 18 | 0 | 24.4 | 26.70±0.02 | 54.39±0.10 |
| | 19 | 10 | 25.2 | 32.71±0.02 | 66.6±0.1 |
| | 20 | 30 | 25.2 | 36.90±0.02 | 75.2±0.2 |
| | 21 | 60 | 25.1 | 38.88±0.02 | 79.2±0.2 |
| | 22 | 120 | 24.1 | 39.83±0.02 | 81.1±0.2 |
| | 23 | 145 | 23.6 | 39.86±0.02 | 81.2±0.2 |
| Four Cylinders in a square | | | | | |
| | 24 | 0 | 24.4 | 22.79±0.02 | 61.9±0.2 |
| | 25 | 10 | 24.6 | 29.75±0.02 | 80.8±0.2 |
| | 26 | 30 | 24.8 | 35.31±0.02 | 95.9±0.2 |
| | 27 | 60 | 24 | 37.90±0.02 | 102.9±0.2 |
| | 28 | 120 | 24.1 | 39.36±0.02 | 106.9±0.2 |
| | 29 | 145 | 24.1 | 39.57±0.02 | 107.5±0.2 |
| Four Cylinders in a Rectangle with fixed Width (15 cm) | | | | | |
| | 30 | 0 | 24.2 | 26.83±0.02 | 72.9±0.2 |
| | 31 | 5 | 23.9 | 29.26±0.02 | 79.5±0.2 |
| | 32 | 15 | 23.9 | 31.76±0.02 | 86.3±0.2 |
| | 33 | 30 | 23.8 | 33.54±0.02 | 91.1±0.2 |
| | 34 | 60 | 24.6 | 35.02±0.02 | 95.1±0.2 |
| | 35 | 120 | 23.5 | 35.78±0.02 | 97.2±0.2 |

Table 5-202 Results for PU-SOL-THERM-014

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| pst014_1 | 0.9980 | 0.0032 | 0.024 | 1.0104 | 0.0010 | 0.0124 | ± | 0.0067 |
| pst014_2 | 0.9980 | 0.0032 | 0.024 | 1.0094 | 0.0010 | 0.0114 | ± | 0.0067 |
| pst014_3 | 0.9980 | 0.0032 | 0.025 | 1.0100 | 0.0010 | 0.0119 | ± | 0.0067 |
| pst014_4 | 0.9980 | 0.0032 | 0.024 | 1.0103 | 0.0010 | 0.0123 | ± | 0.0067 |
| pst014_5 | 0.9980 | 0.0032 | 0.025 | 1.0110 | 0.0010 | 0.0130 | ± | 0.0067 |
| pst014_6 | 0.9980 | 0.0032 | 0.024 | 1.0088 | 0.0010 | 0.0108 | ± | 0.0067 |

Table 5-202 Results for PU-SOL-THERM-014

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| pst014_7 | 0.9980 | 0.0032 | 0.025 | 1.0104 | 0.0010 | 0.0124 | ± | 0.0067 |
| pst014_8 | 0.9980 | 0.0032 | 0.025 | 1.0090 | 0.0010 | 0.0110 | ± | 0.0067 |
| pst014_9 | 0.9980 | 0.0032 | 0.025 | 1.0110 | 0.0010 | 0.0130 | ± | 0.0067 |
| pst014_10 | 0.9980 | 0.0032 | 0.024 | 1.0072 | 0.0010 | 0.0092 | ± | 0.0067 |
| pst014_11 | 0.9980 | 0.0032 | 0.025 | 1.0109 | 0.0010 | 0.0129 | ± | 0.0067 |
| pst014_12 | 0.9980 | 0.0032 | 0.025 | 1.0115 | 0.0010 | 0.0135 | ± | 0.0067 |
| pst014_13 | 0.9980 | 0.0043 | 0.025 | 1.0118 | 0.0010 | 0.0138 | ± | 0.0088 |
| pst014_14 | 0.9980 | 0.0043 | 0.025 | 1.0084 | 0.0010 | 0.0104 | ± | 0.0088 |
| pst014_15 | 0.9980 | 0.0043 | 0.025 | 1.0087 | 0.0010 | 0.0107 | ± | 0.0088 |
| pst014_16 | 0.9980 | 0.0043 | 0.025 | 1.0086 | 0.0010 | 0.0106 | ± | 0.0088 |
| pst014_17 | 0.9980 | 0.0043 | 0.024 | 1.0100 | 0.0010 | 0.0120 | ± | 0.0088 |
| pst014_18 | 0.9980 | 0.0043 | 0.024 | 1.0116 | 0.0010 | 0.0136 | ± | 0.0088 |
| pst014_19 | 0.9980 | 0.0043 | 0.025 | 1.0092 | 0.0010 | 0.0112 | ± | 0.0088 |
| pst014_20 | 0.9980 | 0.0043 | 0.024 | 1.0095 | 0.0010 | 0.0115 | ± | 0.0088 |
| pst014_21 | 0.9980 | 0.0043 | 0.024 | 1.0091 | 0.0010 | 0.0111 | ± | 0.0088 |
| pst014_22 | 0.9980 | 0.0043 | 0.024 | 1.0070 | 0.0010 | 0.0090 | ± | 0.0088 |
| pst014_23 | 0.9980 | 0.0043 | 0.024 | 1.0080 | 0.0011 | 0.0100 | ± | 0.0089 |
| pst014_24 | 0.9980 | 0.0043 | 0.024 | 1.0114 | 0.0010 | 0.0134 | ± | 0.0088 |
| pst014_25 | 0.9980 | 0.0043 | 0.025 | 1.0083 | 0.0010 | 0.0103 | ± | 0.0088 |
| pst014_26 | 0.9980 | 0.0043 | 0.024 | 1.0096 | 0.0010 | 0.0116 | ± | 0.0088 |
| pst014_27 | 0.9980 | 0.0043 | 0.025 | 1.0079 | 0.0010 | 0.0098 | ± | 0.0088 |
| pst014_28 | 0.9980 | 0.0043 | 0.024 | 1.0099 | 0.0010 | 0.0119 | ± | 0.0088 |
| pst014_29 | 0.9980 | 0.0043 | 0.025 | 1.0059 | 0.0010 | 0.0078 | ± | 0.0088 |
| pst014_30 | 0.9980 | 0.0043 | 0.025 | 1.0091 | 0.0010 | 0.0111 | ± | 0.0088 |
| pst014_31 | 0.9980 | 0.0043 | 0.024 | 1.0084 | 0.0011 | 0.0104 | ± | 0.0089 |
| pst014_32 | 0.9980 | 0.0043 | 0.025 | 1.0084 | 0.0010 | 0.0104 | ± | 0.0088 |
| pst014_33 | 0.9980 | 0.0043 | 0.025 | 1.0075 | 0.0010 | 0.0095 | ± | 0.0088 |
| pst014_34 | 0.9980 | 0.0043 | 0.025 | 1.0095 | 0.0010 | 0.0115 | ± | 0.0088 |
| pst014_35 | 0.9980 | 0.0043 | 0.025 | 1.0084 | 0.0010 | 0.0104 | ± | 0.0088 |

### 5.10.2.1.8 PU-SOL-THERM-015

The experiments described in this section are a subset of criticality measurements involving interacting arrays which were conducted in 1968 at the Valduc facility near the city of Dijon, France by the Protection and Radiological Safety Service, section of the French Commissariat a l'Energie Atomique. The complete set of experiments involved about 100 experimental cases of various configurations between two and nine interacting cylindrical tanks of uranium or plutonium nitrate solutions of varying concentrations, suspended in air in a large concrete building. The plutonium contained 4.23 wt.% $^{240}$Pu. This evaluation documents experiments dealing with plutonium nitrate solution of 152.5 gPu/l in cylindrical tanks of 300 mm outer diameter in the following configurations:

i) Two cylinders
ii) Three cylinders, whose centers lie at the corners of an equilateral triangle
iii) Three cylinders, whose centers lie at the corners of a right-angled isosceles triangle
iv) Four cylinders, whose centers lie at the corners of a square.

Seventeen experimental cases were judged to be acceptable benchmarks (NEA [2001]). The H/Fuel (All Pu isotopes plus Am) ratio as calculated from the solution data in Section 3.3 of NEA (2001, PU-SOL-THERM-015) is approximately 155. Table 5-203 gives a brief description of the cases, and Table 5-204 provides the MCNP results.

Table 5-203 PU-SOL-THERM-015 Cylindrical Tank Separations (±0.1 cm ) of the Various Experimental Configurations and their Critical Heights

| Geometric Configuration | Case Number | Surface-to-Surface Distance (cm) | Temp. (°C) | Critical Height (cm) | Critical Volume (l) |
|---|---|---|---|---|---|
| Two-Cylinders | 1 | 0 | 25.6 | 31.07±0.02 | 42.19±0.07 |
| | 2 | 5 | 26 | 34.70±0.02 | 47.12±0.08 |
| | 3 | 15 | - | 37.73±0.02 | 51.24±0.08 |
| | 4 | 30 | - | 39.72±0.02 | 53.94±0.08 |
| | 5 | 50 | 26.4 | 40.86±0.02 | 55.49±0.09 |
| | 6 | N/A | 26.6 | 42.53±0.02 | 28.88±0.05 |
| Three-Cylinders in-Equilateral Triangle | 7 | 0 | 25.4 | 23.33±0.02 | 47.52±0.09 |
| | 8 | 10 | 25.9 | 31.90±0.02 | 64.98±0.10 |
| | 9 | 30 | 25.8 | 37.30±0.02 | 76.0±0.2 |
| | 10 | 60 | 26.1 | 39.90±0.02 | 81.3±0.2 |
| Three-Cylinders in-Right-angled Isosceles-Triangle | 11 | 0 | 25.7 | 26.58±0.02 | 54.14±0.10 |
| | 12 | 10 | 25.6 | 33.17±0.02 | 67.6±0.2 |
| | 13 | 30 | 25.6 | 37.92±0.02 | 77.2±0.2 |
| | 14 | 60 | 25.7 | 40.16±0.02 | 81.8±0.2 |

Table 5-203 PU-SOL-THERM-015 Cylindrical Tank Separations (±0.1 cm ) of the Various Experimental Configurations and their Critical Heights

| Geometric Configuration | Case Number | Surface-to-Surface Distance (cm) | Temp. (°C) | Critical Height (cm) | Critical Volume (l) |
|---|---|---|---|---|---|
| Four-Cylinders-in a-Square | 15 | 0 | 26 | 22.96±0.02 | 62.4±0.2 |
| | 16 | 10 | 25.6 | 30.36±0.02 | 82.5±0.2 |
| | 17 | 30 | 25.8 | 36.25±0.03 | 98.5±0.2 |

Table 5-204 Results for PU-SOL-THERM-015

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| pst015_1 | 0.9980 | 0.0038 | 0.033 | 1.0088 | 0.0010 | 0.0108 | ± | 0.0079 |
| pst015_2 | 0.9980 | 0.0038 | 0.033 | 1.0078 | 0.0010 | 0.0098 | ± | 0.0079 |
| pst015_3 | 0.9980 | 0.0038 | 0.033 | 1.0096 | 0.0010 | 0.0116 | ± | 0.0079 |
| pst015_4 | 0.9980 | 0.0038 | 0.033 | 1.0092 | 0.0010 | 0.0112 | ± | 0.0079 |
| pst015_5 | 0.9980 | 0.0038 | 0.033 | 1.0089 | 0.0010 | 0.0109 | ± | 0.0079 |
| pst015_6 | 0.9980 | 0.0038 | 0.033 | 1.0091 | 0.0010 | 0.0111 | ± | 0.0079 |
| pst015_7 | 0.9971 | 0.0047 | 0.033 | 1.0093 | 0.0010 | 0.0122 | ± | 0.0096 |
| pst015_8 | 0.9971 | 0.0047 | 0.033 | 1.0065 | 0.0010 | 0.0094 | ± | 0.0096 |
| pst015_9 | 0.9971 | 0.0047 | 0.033 | 1.0071 | 0.0010 | 0.0100 | ± | 0.0096 |
| pst015_10 | 0.9971 | 0.0047 | 0.033 | 1.0076 | 0.0010 | 0.0105 | ± | 0.0096 |
| pst015_11 | 0.9971 | 0.0047 | 0.033 | 1.0053 | 0.0010 | 0.0082 | ± | 0.0096 |
| pst015_12 | 0.9971 | 0.0047 | 0.033 | 1.0084 | 0.0010 | 0.0113 | ± | 0.0096 |
| pst015_13 | 0.9971 | 0.0047 | 0.033 | 1.0069 | 0.0010 | 0.0098 | ± | 0.0096 |
| pst015_14 | 0.9971 | 0.0047 | 0.033 | 1.0068 | 0.0010 | 0.0097 | ± | 0.0096 |
| pst015_15 | 0.9971 | 0.0047 | 0.033 | 1.0097 | 0.0010 | 0.0126 | ± | 0.0096 |
| pst015_16 | 0.9971 | 0.0047 | 0.033 | 1.0087 | 0.0010 | 0.0116 | ± | 0.0096 |
| pst015_17 | 0.9971 | 0.0047 | 0.033 | 1.0079 | 0.0010 | 0.0108 | ± | 0.0096 |

### 5.10.2.1.9 PU-SOL-THERM-016

The experiments described in this section are a subset of criticality measurements involving interacting arrays which were conducted in 1968 at the Valduc facility near the city of Dijon, France by the Protection and Radiological Safety Service, section of the French Commissariat a l'Energie Atomique (NEA [2001]).

The complete set of experiments involved about 100 configurations of between two and nine interacting cylindrical tanks of uranium or plutonium nitrate solutions of varying concentrations, suspended in air in a large concrete building. The plutonium contained 4.23 wt.% of $^{240}$Pu.

This evaluation documents experiments dealing with plutonium nitrate solution at two concentrations (115.1 gPu/l and 152.5 gPu/l) in two cylindrical tanks (300 mm and 256 mm outer diameter) in the following configurations (NEA [2001]):

i) Two tanks (300 mm and 256 mm) containing 152.5 gPu/l of plutonium nitrate solution with the solution at the same height in both cylinders

ii) Two tanks (300 mm and 256 mm) containing 115.1 gPu/l of plutonium nitrate solution with the solution at the same height in both cylinders

iii) A 300 mm tank containing 115.1 gPu/l of plutonium nitrate solution and a 256 mm tank containing a 2N nitric acid solution. The height of the nitric acid solution in the 256 mm tank was maintained at 41 cm.

The critical height of the plutonium nitrate solution was determined for various surface-to-surface distances between the cylinders. All 11 experimental cases listed in Table 5-205 were judged to be acceptable benchmarks (NEA [2001]). Results for this set of benchmarks is given in Table 5-206. Note that the 256 mm cylinder is modeled as 12.79 radius instead of reference of 12.8 cm.

Table 5-205 PU-SOL-THERM-016 Cylindrical Tank Separations of the Experimental Configurations and their Critical Heights

| Case Number | Critical Height in 30-cm Cyclinder (cm) | Critical Height in 25.6-cm Cyclinder (cm) | Spacing (D) Between Tanks (cm) |
|---|---|---|---|
| 1 | 34.86 | 34.534 | 0 |
| 2 | 40.15 | 39.824 | 10 |
| 3 | 42.09 | 41.764 | 30 |
| 4 | 42.43 | 42.104 | 60 |
| 5 | 34.09 | 33.764 | 0 |
| 6 | 39 | 38.674 | 10 |
| 7 | 40.66 | 40.334 | 30 |
| 8 | 40.9 | 40.574 | 60 |
| 9 | 38.6 | 41 | 0 |
| 10 | 40.37 | 41 | 10 |
| 11 | 40.89 | 41 | 30 |

Table 5-206 Results for PU-SOL-THERM-016

| Experiment/ Case | Benchmark-Model $k_{eff}$ | σ | AENCF | Calculated $k_{eff}$ | σ | $\Delta k_{eff} \pm 2\sigma$ | | | H/Pu[a] |
|---|---|---|---|---|---|---|---|---|---|
| pst016_1 | 0.9980 | 0.0043 | 0.0330 | 1.00763 | 0.0010 | 0.0096 | ± | 0.0088 | 155.2 |
| pst016_10 | 0.9963 | 0.0033 | 0.0242 | 1.01062 | 0.0011 | 0.0143 | ± | 0.0069 | 210.1 |
| pst016_11 | 0.9963 | 0.0033 | 0.0242 | 1.01008 | 0.0010 | 0.0138 | ± | 0.0069 | 210.1 |
| pst016_2 | 0.9980 | 0.0043 | 0.0330 | 1.00712 | 0.0010 | 0.0091 | ± | 0.0088 | 155.2 |
| pst016_3 | 0.9980 | 0.0043 | 0.0325 | 1.00886 | 0.0010 | 0.0109 | ± | 0.0088 | 155.2 |
| pst016_4 | 0.9980 | 0.0043 | 0.0328 | 1.01002 | 0.0010 | 0.0120 | ± | 0.0088 | 155.2 |
| pst016_5 | 0.9969 | 0.0038 | 0.0241 | 1.00862 | 0.0010 | 0.0117 | ± | 0.0079 | 210.1 |
| pst016_6 | 0.9969 | 0.0038 | 0.0246 | 1.00963 | 0.0010 | 0.0127 | ± | 0.0079 | 210.1 |
| pst016_7 | 0.9969 | 0.0038 | 0.0247 | 1.0103 | 0.0010 | 0.0134 | ± | 0.0079 | 210.1 |
| pst016_8 | 0.9969 | 0.0038 | 0.0244 | 1.00913 | 0.0010 | 0.0122 | ± | 0.0079 | 210.1 |
| pst016_9 | 0.9963 | 0.0033 | 0.0245 | 1.01067 | 0.0010 | 0.0144 | ± | 0.0069 | 210.1 |

NOTE: [a] H/Pu is the Hydrogen to Plutonium (Fuel) ratio as calculated from the number densities for the solutions in section 3.3 of PU-SOL-THERM-016.

### 5.10.2.1.10 PU-SOL-THERM-017

The experiments described in this section are a subset of criticality measurements involving interacting arrays which were conducted in 1968 at the Valduc facility near the city of Dijon, France by the Protection and Radiological Safety Service, section of the French Commissariat a l'Energie Atomique. This evaluation documents experiments dealing with plutonium nitrate solution at a concentration of 115.1 gPu/l in two cylindrical tanks in the following configurations (NEA [2001]):

i) Two tanks, one of 256 mm outer diameter and the other of 300 mm outer diameter, containing 15.1 gPu/l of plutonium nitrate solution with the solution at certain predetermined heights in the 256 mm cylinder for each given surface-to surface distance

ii) Two tanks of 300 mm outer diameter containing 115.1 gPu/l of plutonium nitrate solution with the solution at certain predetermined heights in one of the tanks for each given surface-to-surface distance.

Basic parameters for these experiments are given in Table 5-207 and results are listed in Table 5-208. The H/Fuel (Pu and Am) ratio for these cases as calculated from the number densities given in section 3.3 of NEA (2001, PU-SOL-THERM-017) is approximately 210. Note that in the MCNP models the 256mm cylinder is modeled as 12.79 radius instead of reference of 12.8 cm. These experiments are acceptable as benchmarks (NEA [2001]).

Table 5-207 PU-SOL-THERM-017 Cylindrical Tank Separations (±0.1 cm) of the Experimental Configurations and their Critical Heights

| Geometric Configuration | Case Number | Surface-to-Surface Distance (cm) | Temp. (°C) | Height of Solution in 256 or 300mm tank (cm) | Critical Height (cm) In 300mm tank (cm) | Critical Volume (l) |
|---|---|---|---|---|---|---|
| 256 and 300 mm Cylinders with-115.1 gPu/l | 1 | 30 | 24.1 | 0 | 40.66±0.02 | 27.61±0.05 |
| | 2 | 30 | 24.2 | 20 | 40.54±0.02 | 37.34±0.06 |
| | 3 | 30 | 24 | 40 | 40.27±0.02 | 46.96±0.07 |
| | 4 | 30 | 24.8 | 60 | 40.13±0.02 | 56.70±0.08 |
| | 5 | 30 | 26.1 | 80 | 40.12±0.02 | 66.46±0.09 |
| | 6 | 15 | 25 | 20 | 40.34±0.02 | 37.20±0.06 |
| | 7 | 15 | 25.4 | 80 | 38.74±0.02 | 65.52±0.09 |
| | 8 | 5 | 25.5 | 6 | 40.45±0.02 | 30.39±0.05 |
| | 9 | 5 | 24.4 | 10 | 40.17±0.02 | 32.18±0.05 |
| | 10 | 5 | 24.5 | 20 | 39.43±0.02 | 36.58±0.06 |
| Two Cylinders Of 300 mm OD-with 115.1gPu/l | 11 | 15 | 24.5 | 0 | 40.64±0.02 | 27.59±0.05 |
| | 12 | 15 | 26 | 10 | 40.70±0.02 | 34.42±0.06 |
| | 13 | 15 | 26.3 | 20 | 40.43±0.02 | 41.02±0.07 |
| | 14 | 15 | 26.2 | 30 | 39.57±0.02 | 47.24±0.07 |
| | 15 | 30 | 25.7 | 0 | 40.87±0.02 | 27.75±0.05 |
| | 16 | 30 | 26.8 | 10 | 41.00±0.02 | 34.64±0.06 |
| | 17 | 30 | 26.2 | 20 | 40.81±0.02 | 41.29±0.07 |
| | 18 | 30 | 26.7 | 30 | 40.59±0.02 | 47.94±0.07 |

Table 5-208 MCNP Results for PU-SOL-THERM-017

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| pst017_1 | 0.9969 | 0.0038 | 0.025 | 1.0089 | 0.0010 | 0.0120 | ± | 0.0078 |
| pst017_10 | 0.9969 | 0.0038 | 0.025 | 1.0082 | 0.0010 | 0.0113 | ± | 0.0079 |
| pst017_11 | 0.9969 | 0.0038 | 0.024 | 1.0100 | 0.0010 | 0.0131 | ± | 0.0079 |
| pst017_12 | 0.9969 | 0.0038 | 0.025 | 1.0081 | 0.0010 | 0.0112 | ± | 0.0079 |
| pst017_13 | 0.9969 | 0.0038 | 0.024 | 1.0084 | 0.0010 | 0.0115 | ± | 0.0079 |
| pst017_14 | 0.9969 | 0.0038 | 0.025 | 1.0081 | 0.0010 | 0.0112 | ± | 0.0079 |
| pst017_15 | 0.9969 | 0.0038 | 0.024 | 1.0102 | 0.0010 | 0.0133 | ± | 0.0079 |
| pst017_16 | 0.9969 | 0.0038 | 0.025 | 1.0108 | 0.0010 | 0.0139 | ± | 0.0079 |

| Table 5-208 MCNP Results for PU-SOL-THERM-017 | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| pst017_17 | 0.9969 | 0.0038 | 0.025 | 1.0088 | 0.0010 | 0.0119 | ± | 0.0079 |
| pst017_18 | 0.9969 | 0.0038 | 0.024 | 1.0088 | 0.0010 | 0.0119 | ± | 0.0079 |
| pst017_2 | 0.9969 | 0.0038 | 0.025 | 1.0109 | 0.0011 | 0.0140 | ± | 0.0079 |
| pst017_3 | 0.9969 | 0.0038 | 0.024 | 1.0080 | 0.0010 | 0.0110 | ± | 0.0078 |
| pst017_4 | 0.9969 | 0.0038 | 0.024 | 1.0067 | 0.0010 | 0.0098 | ± | 0.0079 |
| pst017_5 | 0.9969 | 0.0038 | 0.025 | 1.0102 | 0.0010 | 0.0133 | ± | 0.0079 |
| pst017_6 | 0.9969 | 0.0038 | 0.025 | 1.0088 | 0.0010 | 0.0119 | ± | 0.0079 |
| pst017_7 | 0.9969 | 0.0038 | 0.025 | 1.0076 | 0.0010 | 0.0107 | ± | 0.0079 |
| pst017_8 | 0.9969 | 0.0038 | 0.025 | 1.0101 | 0.0010 | 0.0132 | ± | 0.0079 |
| pst017_9 | 0.9969 | 0.0038 | 0.025 | 1.0088 | 0.0010 | 0.0119 | ± | 0.0078 |

### 5.10.2.1.11 PU-SOL-THERM-020

This section describes plutonium solution experiments performed at the Battelle PNL Critical Mass Laboratory, in a 14-inch-diameter spherical steel shell reflected by either water or water plus a cadmium interface. The solution was plutonium nitrate with the plutonium having 4.57 wt.% $^{240}$Pu. The 14-inch-diameter sphere was reported to have a measured volume of 23.22±0.02 liters when full. This volume requires an effective radius of 17.698 cm. These experiments are acceptable as benchmark experiments (NEA [2001]). Basic parameters for these experiments are given in Table 5-209 and results of MCNP models (designated as "A" in NEA [2001]) are listed in Table 5-210.

| Table 5-209 PU-SOL-THERM-020 Data for Water-Reflected 14-Inch Spheres. | | | | | | |
|---|---|---|---|---|---|---|
| Case Number | GPu/l. | Reported Volume (l) | Volume Less Solutin in Support Tube (l)[a] | Model Solution Volume(l)[b] | Void Volume ($cm^3$)[c] | H(cm)[d] |
| 1 | 39.2 | 23.1 | 23.1 | 22.92 | 290 | 15.3596 |
| 2 | 38.4 | 23.3 | 23.29 | 23.1 | 110 | 16.2696 |
| 3 | 33.5 | 22.96 | 22.96 | 22.78 | 430 | 14.8363 |
| 5 | 47.9 | 23.13 | 23.13 | 22.94 | 270 | 15.4434 |
| 6 | 49.5 | 23.2 | 23.19 | 23 | 210 | 15.7146 |
| 7 | 34.4 | 22.99 | 22.99 | 22.81 | 400 | 14.9407 |
| 8 | 69.4 | 23.1 | 23.1 | 22.92 | 290 | 15.3596 |
| 9 | 46.9 | 23.25 | 23.24 | 23.05 | 160 | 15.9708 |

NOTES: [a] Reported critical volume less 0.01 liter for volumes greater than 23.14 liters to allow for the solution in the support tube.
[b] Solution volume in the fourth column times (1.0 - 0.008) to compensate for the deletion of the support tubes in the water reflector.
[c] Full volume of 23.21 liters less solution volume in column five.
[d] H is the height from a horizontal plane through the center of the sphere to the surface of the solution.

Table 5-210 Results for PU-SOL-THERM-020

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/Pu[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $K_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| pst020_1.T8A | 1.0000 | 0.0059 | 0.0078 | 1.0105 | 0.0009 | 0.0105 | ± | 0.0119 | 603.6 |
| pst020_2.T8A | 1.0000 | 0.0059 | 0.0078 | 1.0144 | 0.0010 | 0.0144 | ± | 0.0120 | 621.1 |
| pst020_3.T8A | 1.0000 | 0.0059 | 0.0067 | 1.0077 | 0.0009 | 0.0077 | ± | 0.0119 | 747.2 |
| pst020_5.T8A | 1.0000 | 0.0059 | 0.0100 | 1.0080 | 0.0010 | 0.0080 | ± | 0.0120 | 462.5 |
| pst020_6.T8A | 1.0000 | 0.0059 | 0.0106 | 1.0103 | 0.0010 | 0.0103 | ± | 0.0120 | 451.9 |
| pst020_7.T8A | 1.0000 | 0.0059 | 0.0067 | 1.0046 | 0.0009 | 0.0046 | ± | 0.0119 | 723.8 |
| pst020_8.T8A | 1.0000 | 0.0059 | 0.0146 | 1.0109 | 0.0011 | 0.0109 | ± | 0.0120 | 343.0 |
| pst020_9.T8A | 1.0000 | 0.0059 | 0.0096 | 1.0044 | 0.0010 | 0.0044 | ± | 0.0120 | 540.1 |

NOTE: [a] . H/Pu (Pu isotopes) ratio is calculated from the number densities given in section 3.3 of NEA (2001, PU-SOL-THERM-020).

### 5.10.2.1.12 PU-SOL-THERM-021

Evaluated in this section are plutonium solution experiments performed in 1964 at the Battelle PNL Critical Mass Laboratory, in a 15.2-inch-diameter spherical steel shell either reflected by water or with no water reflection. The solutions were plutonium nitrate of varying plutonium concentration with the plutonium having 4.57 wt.% $^{240}$Pu. These experiments were accepted as benchmark experiments (NEA [2001]).

The experiment equipment consisted of a stainless steel spherical shell, nominally 15.2 inches in diameter, either surrounded by an effectively infinite water reflector or unreflected. A number of experiments with various plutonium concentrations were performed. The spherical shell of the 15.2-inch sphere was made of 304L stainless steel with a thickness of 0.048 inches (0.1219 cm). The sphere had several penetrations including offset control and safety channels on the top, a tube for exposing foils in the solution, and a bottom pipe for adding and removing the solution. The temperature was stated to be "maintained at 25 °C." Table 5-211 gives an overview of the experimental parameters while Table 5-212 lists the results for detailed MCNP models.

Table 5-211 PU-SOL-THERM-021 Geometry of Solutions for PU-SOL-THERM-021

| Case Number | Exper. No. | gPu/l | Reported Critical Volume (l) | Model Solution Volume (l) | Model Solution Radius (cm) | Solution Height above Centerline, H |
|---|---|---|---|---|---|---|
| Bare Experiments | | | | | | |
| 1 | 1258 | 39 | 30.266 | 30.256 | 19.3304 | Full |
| 2 | 1273 | 34.3 | 30.2 | 30.19 | 19.3163 | Full |
| 3 | 1288 | 172.3 | 30.2 | 30.19 | 19.3163 | Full |
| Water Reflected Experiments | | | | | | |
| 4 | 2325 | 25.2 | 30.181 | 30.171 | 19.3163 | 18.754 |
| 5 | 2330 | 24.4 | 30.154 | 30.144 | 19.3163 | 18.439 |
| 6 | 2346 | 38.7 | 30.221 | 30.211 | 19.3208 | Full |

Table 5-212 Results for PU-SOL-THERM-021

| Experiment/Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/Pu[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| pst021_1.T9A | 1.0000 | 0.0032 | 0.008 | 1.0123 | 0.0010 | 0.0123 | ± | 0.0067 | 667.0 |
| pst021_2.T9A | 1.0000 | 0.0032 | 0.007 | 1.0123 | 0.0010 | 0.0123 | ± | 0.0067 | 758.2 |
| pst021_3.T9A | 1.0000 | 0.0065 | 0.040 | 1.0095 | 0.0012 | 0.0095 | ± | 0.0132 | 125.3 |
| pst021_4.T9A | 1.0000 | 0.0025 | 0.005 | 1.0089 | 0.0009 | 0.0089 | ± | 0.0053 | 1031.8 |
| pst021_5.T9A | 1.0000 | 0.0025 | 0.005 | 1.0105 | 0.0009 | 0.0105 | ± | 0.0053 | 1068.6 |
| pst021_6.T9A | 1.0000 | 0.0044 | 0.009 | 1.0118 | 0.0009 | 0.0118 | ± | 0.0090 | 552.4 |

NOTE: [a] H/Pu ratio is calculated from the number densities given in section 3.3 of NEA (2001, PU-SOL-THERM-021).

### 5.10.2.1.13 PU-SOL-THERM-022

The experiments described in this section plutonium nitrate solution in an annular cylinder tank 50 x 20 cm dia. The tank was reflected by water on the side and at the bottom. A plutonium concentration range of 28.5 to 165 g/l was examined. The plutonium contained 18.88 wt.% $^{240}$Pu. These experiments, performed at VALDUC facility (CEA-FRANCE) in 1973, were subcritical approaches extrapolated to critical. The multiplication factor reached was very close to 1.000 (within 0.1%). Listed here are nine experiments with air in the central region and 8 with an insert of paraffin plus cadmium sheet. All 17 are acceptable benchmark experiments (NEA [2001]).

The experimental configuration consisted of an annular cylinder. It was made of two concentric tubes having inside diameters of 50 cm and 19.4 cm, 120.1 cm high, and a 0.3-cm wall thickness. These tubes were welded to a bottom plate (thickness varying from 1.8 cm to 1.2 cm) and to a

1.8-cm-thick top plate. This top plate had 8 holes (6 cm diameters), each at 45°, centered on a circle 35 cm in diameter. The central region was either empty (air) or accommodated a paraffin cylinder surrounded by an 0.08-cm-thick cadmium sheet. The tank was supported by support feet, 31.3 cm in height. Tubes, plates, and support feet were of stainless steel. Basic parameters for these experiments are given in Table 5-213 and results of simplified and detailed MCNP models (designated as "A" in NEA [2001]) are listed in Table 5-214.

Table 5-213 PU-SOL-THERM-022 Characteristics of Configurations

| Case | Experiment Date (1973) | Pu Conc. (g/l) | $(NO_3)$ [M] | Acidity [M] | Fe Conc. (g/l) | Solution Density (g/cm$^3$) | Temp. (°C) | Critical Height (cm) | |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | Air | Paraffin+Cd |
| 1 | Oct.15 | 152 | 6.68 | 3.85 | 3.4 | 1.386 | 21 | 36.92 | - |
| 2 | Oct.24 | 104 | 4.57 | 2.65 | 2.12 | 1.271 | 20.5 | 34.34 | - |
| 3 | Oct.25 | 62 | 3.02 | 1.88 | 1.28 | 1.171 | 20.5 | 37.15 | - |
| 4 | Oct.26 | 51 | 2.8 | 1.85 | 1.13 | 1.151 | 20.5 | 41.14 | - |
| 5 | Oct.29 | 40.9 | 2.63 | 1.87 | 0.92 | 1.135 | 20.5 | 49.19 | - |
| 6 | Oct.30 | 36 | 2.58 | 1.9 | 0.88 | 1.127 | 20.5 | 57.68 | - |
| 7 | Oct.31 | 33.1 | 2.51 | 1.9 | 0.71 | 1.121 | 20.5 | 66.66 | - |
| 8 | Nov.2 | 30.8 | 2.45 | 1.88 | 0.69 | 1.118 | 20.8 | 77.33 | - |
| 9 | Nov.5 | 28.7 | 2.47 | 1.94 | 0.65 | 1.114 | 20.3 | 93.36 | - |
| 10 | Nov.16 | 165 | 6.17 | 3.11 | 3.8 | 1.393 | 21.8 | - | 60.74 |
| 11 | Nov.19 | 136 | 5.3 | 2.76 | 3.2 | 1.329 | 21 | - | 54.35 |
| 12 | Nov.21 | 105 | 4.14 | 2.17 | 2.5 | 1.261 | 21.5 | - | 51.48 |
| 13 | Nov.22 | 86 | 3.82 | 2.21 | 1.87 | 1.226 | 21 | - | 54.27 |
| 14 | Nov.27 | 74 | 3.59 | 2.22 | 1.56 | 1.202 | 21.5 | - | 59.35 |
| 15 | Nov.29 | 61 | 3.34 | 2.19 | 1.41 | 1.18 | 21.5 | - | 70.48 |
| 16 | Nov.24 | 55.8 | 3.17 | 2.12 | 1.33 | 1.168 | 21 | - | 83.64 |
| 17 | Dec.3 | 53.7 | 3.11 | 2.11 | 1.25 | 1.164 | 21.5 | - | 92.13 |

Table 5-214 Results for PU-SOL-THERM-022

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/Pu[a] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| pst022_01 | 1.0000 | 0.002 | 0.032 | 1.0020 | 0.0011 | 0.0020 | ± | 0.0045 | 147.1 |
| pst022_02 | 1.0000 | 0.002 | 0.021 | 1.0071 | 0.0009 | 0.0071 | ± | 0.0044 | 229.3 |
| pst022_03 | 1.0000 | 0.002 | 0.013 | 1.0093 | 0.0010 | 0.0093 | ± | 0.0045 | 398.7 |
| pst022_04 | 1.0000 | 0.002 | 0.011 | 1.0083 | 0.0009 | 0.0083 | ± | 0.0044 | 487.3 |
| pst022_05 | 1.0000 | 0.002 | 0.009 | 1.0093 | 0.0009 | 0.0093 | ± | 0.0044 | 610.8 |

| Table 5-214 Results for PU-SOL-THERM-022 | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/Pu[a] |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| pst022_06 | 1.0000 | 0.003 | 0.008 | 1.0113 | 0.0008 | 0.0113 | ± | 0.0062 | 694.4 |
| pst022_07 | 1.0000 | 0.003 | 0.007 | 1.0117 | 0.0007 | 0.0117 | ± | 0.0062 | 756.1 |
| pst022_08 | 1.0000 | 0.003 | 0.007 | 1.0124 | 0.0008 | 0.0124 | ± | 0.0062 | 815.0 |
| pst022_09 | 1.0000 | 0.003 | 0.006 | 1.0103 | 0.0008 | 0.0103 | ± | 0.0062 | 872.2 |
| pst022_10 | 1.0000 | 0.002 | 0.034 | 1.0024 | 0.0010 | 0.0024 | ± | 0.0045 | 138.4 |
| pst022_11 | 1.0000 | 0.002 | 0.029 | 0.9969 | 0.0010 | -0.0031 | ± | 0.0044 | 171.4 |
| pst022_12 | 1.0000 | 0.002 | 0.021 | 1.0035 | 0.0010 | 0.0035 | ± | 0.0045 | 230.0 |
| pst022_13 | 1.0000 | 0.002 | 0.017 | 1.0014 | 0.0010 | 0.0014 | ± | 0.0045 | 282.6 |
| pst022_14 | 1.0000 | 0.002 | 0.015 | 1.0017 | 0.0010 | 0.0017 | ± | 0.0045 | 329.1 |
| pst022_15 | 1.0000 | 0.002 | 0.012 | 0.9968 | 0.0009 | -0.0032 | ± | 0.0044 | 402.4 |
| pst022_16 | 1.0000 | 0.002 | 0.011 | 1.0007 | 0.0009 | 0.0007 | ± | 0.0044 | 441.2 |
| pst022_17 | 1.0000 | 0.002 | 0.011 | 1.0024 | 0.0009 | 0.0024 | ± | 0.0044 | 459.2 |

NOTE: [a] H/Pu (Pu+$^{241}$Am) ratio is calculated from the number densities given in Tables 7.a and 7.b of Section 3.3 of NEA (2001, PU-SOL-THERM-022).

### 5.10.2.1.14 PU-SOL-THERM-023

The experiments described here considered two concentric cylinder tanks containing two different plutonium nitrate solutions. The large outer tank, or driver, always contained the same solution (116.9 g/l of plutonium, $^{240}$Pu/Pu Total = 4.23 wt.%). For the first series of experiments, the small inner tank (2.2 or 4.2 liters capacity) contained a nitrate solution of high $^{240}$Pu content (33.89 wt.% $^{240}$Pu), the plutonium concentration range being 0.0 to 139.6 g/l; the maximum plutonium mass used was 336 g. For the second series of experiments the inner tank contained a nitrate solution of low $^{240}$Pu content (4.23 wt.%), the plutonium concentration range being 6.67 to 124.6 g/l. These experiments, performed in a large tank of water at the VALDUC facility (CEA-France) in 1969-1970 were subcritical approaches extrapolated to critical. The multiplication factor reached was very close to 1 (within 0.1%). Descriptions of three water reflected experiments are given in Table 5-215. These experiments are considered acceptable for use as benchmark experiments (NEA [2001]) and results given in Table 5-216.

| Table 5-215 PU-SOL-THERM-023 Experimental Results: Inner Plutonium Nitrate Solution, $^{240}$Pu/ Pu = 33.89%. | | | | | |
|---|---|---|---|---|---|
| Water Reflected | | | | | |
| Inner Tank | C(Pu) (g/l) | Exper. Number | Case Number | Date (mm/dd/yy) | Hc[a] (cm) ± 0.03 |
| 4.209 liters | 79.8 | 141 | 1 | 12/23/69 | 25.02 |

| Table 5-215 PU-SOL-THERM-023 Experimental Results: Inner Plutonium Nitrate Solution, $^{240}$Pu/ Pu = 33.89%. | | | | | |
|---|---|---|---|---|---|
| Water Reflected | | | | | |
| **Inner Tank** | **C(Pu) (g/l)** | **Exper. Number** | **Case Number** | **Date (mm/dd/yy)** | **Hc[a] (cm) ± 0.03** |
| 2.155 liters | 139.6 | 126 | 7 | 09/18/69 | 22.74 |
| | 79.8 | 140 | 11 | 12/23/69 | 22.57 |

NOTE: [a] Hc denotes the critical height of the solution.

| Table 5-216 PU-SOL-THERM-023 | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| **Experiment/ Case** | **Benchmark-Model** | | **Calculated[a]** | | | **$\Delta k_{eff} \pm 2\sigma$** | | | **H/Pu[b]** |
| | **$k_{eff}$** | **σ** | **AENCF** | **$k_{eff}$** | **σ** | | | | |
| Pst23-1 | 1.0000 | 0.0038 | 0.0212 | 1.0011 | 0.0009 | 0.0011 | ± | 0.0078 | 468.73 |
| Pst23-7 | 1.0000 | 0.0038 | 0.0234 | 1.0036 | 0.0009 | 0.0036 | ± | 0.0078 | 259.18 |
| Pst23-11 | 1.0000 | 0.0038 | 0.0214 | 1.0012 | 0.0009 | 0.0012 | ± | 0.0078 | 468.73 |

NOTES: [a] Air Composition is different then reference information. It is believed that the difference will have a negligible effect on $k_{eff}$
[b] H/Pu ($^{239}$Pu + $^{240}$Pu) ratio is calculated from the number densities given in of Section 3.3 of NEA (2001, PU-SOL-THERM-023).

### 5.10.2.1.15 PU-SOL-THERM-024

During the late 1960s a number of critical experiments were performed on plutonium nitrate solutions in slab geometry at Battelle-PNL. This section lists the 23 Plexiglas-reflected experiments. These 23 experiments include five different solutions and span the range of experimental critical slab thicknesses. Table 5-217 provides some of the experimental details and the results are presented in Table 5-218. These 23 experiments are acceptable as criticality benchmarks (NEA [2001]).

| Table 5-217 PU-SOL-THERM-024 Critical Experiment Characteristics | | | | | | |
|---|---|---|---|---|---|---|
| **Case** | **Pu Conc. (g/l)** | **Density (g/cm$^3$)** | **Molarity** | **$^{240}$Pu wt.%** | **Thicknes s (in)** | **Height (in)** |
| 1 | 240.8 | 1.51 | 4.14 | 18.4 | 8.35 | 18.26 |
| 2 | 240.8 | 1.51 | 4.14 | 18.4 | 7.6 | 23.32 |
| 3 | 240.8 | 1.51 | 4.14 | 18.4 | 7.25 | 28.08 |
| 4 | 240.8 | 1.51 | 4.14 | 18.4 | 7. | 34.03 |
| 5 | 240.8 | 1.51 | 4.14 | 18.4 | 6.9 | 37.66 |
| 6 | 412.1 | 1.7643 | 4.16 | 18.4 | 8.9 | 22.13 |
| 7 | 160 | 1.3595 | 3.04 | 18.4 | 8.4 | 14.66 |
| 8 | 160 | 1.3595 | 3.04 | 18.4 | 7.9 | 16.24 |

Table 5-217 PU-SOL-THERM-024 Critical Experiment Characteristics

| Case | Pu Conc. (g/l) | Density ($g/cm^3$) | Molarity | $^{240}Pu$ wt.% | Thickness (in) | Height (in) |
|---|---|---|---|---|---|---|
| 9 | 160 | 1.3595 | 3.04 | 18.4 | 7.5 | 18.06 |
| 10 | 160 | 1.3595 | 3.04 | 18.4 | 6.9 | 23.05 |
| 11 | 160 | 1.3595 | 3.04 | 18.4 | 6.6 | 27.81 |
| 12 | 160 | 1.3595 | 3.04 | 18.4 | 6.4 | 33.27 |
| 13 | 160 | 1.3595 | 3.04 | 18.4 | 6.3 | 37.67 |
| 14 | 201.6 | 1.3775 | 1.6 | 23.2 | 8.9 | 15.49 |
| 15 | 201.6 | 1.3775 | 1.6 | 23.2 | 8.4 | 17.09 |
| 16 | 201.6 | 1.3775 | 1.6 | 23.2 | 7.9 | 19.55 |
| 17 | 201.6 | 1.3775 | 1.6 | 23.2 | 7.2 | 26.46 |
| 18 | 66.5 | 1.1848 | 2.36 | 18.4 | 8.4 | 14.292 |
| 19 | 66.5 | 1.1848 | 2.36 | 18.4 | 7.9 | 15.881 |
| 20 | 66.5 | 1.1848 | 2.36 | 18.4 | 7.4 | 18.275 |
| 21 | 66.5 | 1.1848 | 2.36 | 18.4 | 6.9 | 22.538 |
| 22 | 66.5 | 1.1848 | 2.36 | 18.4 | 6.4 | 33.234 |
| 23 | 66.5 | 1.1848 | 2.36 | 18.4 | 6.3 | 37.392 |

Table 5-218 Results for PU-SOL-THERM-024

| Experiment/ Case | Benchmark-Model | | Calculated[a] | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/Pu[b] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| pst024_1 | 1.0000 | 0.0062 | 0.056 | 1.0081 | 0.0020 | 0.0081 | ± | 0.0130 | 87.5 |
| pst024_10 | 1.0000 | 0.0053 | 0.034 | 1.0094 | 0.0019 | 0.0094 | ± | 0.0112 | 143.9 |
| pst024_11 | 1.0000 | 0.0053 | 0.035 | 1.0079 | 0.0019 | 0.0079 | ± | 0.0112 | 143.9 |
| pst024_12 | 1.0000 | 0.0053 | 0.034 | 1.0082 | 0.0020 | 0.0082 | ± | 0.0113 | 143.9 |
| pst024_13 | 1.0000 | 0.0053 | 0.034 | 1.0078 | 0.0019 | 0.0078 | ± | 0.0113 | 143.9 |
| pst024_14 | 1.0000 | 0.0053 | 0.044 | 1.0034 | 0.0021 | 0.0034 | ± | 0.0114 | 115.8 |
| pst024_15 | 1.0000 | 0.0053 | 0.045 | 1.0046 | 0.0018 | 0.0046 | ± | 0.0112 | 115.8 |
| pst024_16 | 1.0000 | 0.0053 | 0.044 | 1.0032 | 0.0019 | 0.0032 | ± | 0.0112 | 115.8 |
| pst024_17 | 1.0000 | 0.0053 | 0.043 | 1.0033 | 0.0018 | 0.0033 | ± | 0.0112 | 115.8 |
| pst024_18 | 1.0000 | 0.0051 | 0.014 | 1.0095 | 0.0019 | 0.0094 | ± | 0.0109 | 367.3 |
| pst024_19 | 1.0000 | 0.0051 | 0.014 | 1.0089 | 0.0018 | 0.0089 | ± | 0.0108 | 367.3 |
| pst024_2 | 1.0000 | 0.0062 | 0.056 | 0.9981 | 0.0019 | -0.0019 | ± | 0.0130 | 87.5 |
| pst024_20 | 1.0000 | 0.0051 | 0.014 | 1.0143 | 0.0018 | 0.0143 | ± | 0.0108 | 367.3 |
| pst024_21 | 1.0000 | 0.0051 | 0.014 | 1.0106 | 0.0018 | 0.0106 | ± | 0.0108 | 367.3 |

Table 5-218 Results for PU-SOL-THERM-024

| Experiment/ Case | Benchmark-Model | | Calculated[a] | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/Pu[b] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| pst024_22 | 1.0000 | 0.0051 | 0.014 | 1.0119 | 0.0017 | 0.0118 | ± | 0.0108 | 367.3 |
| pst024_23 | 1.0000 | 0.0051 | 0.015 | 1.0083 | 0.0017 | 0.0083 | ± | 0.0108 | 367.3 |
| pst024_3 | 1.0000 | 0.0062 | 0.056 | 1.0000 | 0.0020 | 0.0000 | ± | 0.0130 | 87.5 |
| pst024_4 | 1.0000 | 0.0062 | 0.056 | 1.0025 | 0.0019 | 0.0025 | ± | 0.0129 | 87.5 |
| pst024_5 | 1.0000 | 0.0062 | 0.056 | 0.9998 | 0.0019 | -0.0002 | ± | 0.0130 | 87.5 |
| pst024_6 | 1.0000 | 0.0077 | 0.105 | 0.9966 | 0.0019 | -0.0034 | ± | 0.0159 | 44.9 |
| pst024_7 | 1.0000 | 0.0053 | 0.035 | 1.0111 | 0.0018 | 0.0111 | ± | 0.0112 | 143.9 |
| pst024_8 | 1.0000 | 0.0053 | 0.034 | 1.0096 | 0.0018 | 0.0096 | ± | 0.0112 | 143.9 |
| pst024_9 | 1.0000 | 0.0053 | 0.034 | 1.0086 | 0.0021 | 0.0086 | ± | 0.0114 | 143.9 |

NOTES: [a] In the inputs, the slots of the egg-crate grid are 0.3174 cm thick rather than 0.3175 cm thick, and the reflector-tank bottom is 0.316 cm farther below the solution tank than in the benchmark model. These small differences are not expected to affect $k_{eff}$.

[b] H/Pu (all Pu isotopes and $^{241}$Am) ratio is calculated from the number densities provided in section 3.3 of NEA 2001, PU-SOL-THERM-024.

### 5.10.2.1.13 PU-SOL-THERM-025

During the late 1960s a large number of critical experiments were performed on plutonium nitrate solutions in slab geometry at Battelle-PNL. For the critical experiments the plutonium concentrations in the solutions ranged from 58 to 412 g/l. Three different plutonium isotopic compositions were used in these experiments: 4.6, 18.4 and 23.2 wt.% $^{240}$Pu. Experiments were performed with water-reflected, Plexiglas reflected (PU-SOL-THERM-024), and bare slabs (PU-SOL-THERM-026). Forty-four water-reflected cases are listed in Table 5-219. These cases encompass all of the plutonium solutions used and the full range of solution heights and thicknesses. These forty-four cases are acceptable as criticality benchmarks (NEA [2001]). Results for the MCNP models are provided in Table 5-220.

Table 5-219 PU-SOL-THERM-025 Critical Experiment Characteristics

| Case | Pu Conc. (g/l) | Molarity | Density (g/cm$^3$) | $^{240}$Pu wt.% | Thickness (in) | Height (in) |
|---|---|---|---|---|---|---|
| 1 | 58 | 2.34 | 1.1717 | 4.6 | 4.5 | 39.61 |
| 2 | 58 | 2.34 | 1.1717 | 4.6 | 4.6 | 31.46 |
| 3 | 58 | 2.34 | 1.1717 | 4.6 | 4.8 | 23.93 |
| 4 | 58 | 2.34 | 1.1717 | 4.6 | 5.3 | 16.47 |
| 5 | 58 | 2.34 | 1.1717 | 4.6 | 5.8 | 13.18 |
| 6 | 58 | 2.34 | 1.1717 | 4.6 | 6.3 | 11.34 |

Table 5-219 PU-SOL-THERM-025 Critical Experiment Characteristics

| Case | Pu Conc. (g/l) | Molarity | Density ($g/cm^3$) | $^{240}Pu$ wt.% | Thickness (in) | Height (in) |
|---|---|---|---|---|---|---|
| 7 | 66.5 | 2.36 | 1.1848 | 18.4 | 5.5 | 37.66 |
| 8 | 66.5 | 2.36 | 1.1848 | 18.4 | 5.6 | 32.521 |
| 9 | 66.5 | 2.36 | 1.1848 | 18.4 | 5.7 | 29.048 |
| 10 | 66.5 | 2.36 | 1.1848 | 18.4 | 5.9 | 24.373 |
| 11 | 66.5 | 2.36 | 1.1848 | 18.4 | 6.3 | 19.298 |
| 12 | 66.5 | 2.36 | 1.1848 | 18.4 | 6.9 | 15.447 |
| 13 | 66.5 | 2.36 | 1.1848 | 18.4 | 7.9 | 12.306 |
| 14 | 160 | 3.04 | 1.3595 | 18.4 | 5.3 | 33.49 |
| 15 | 160 | 3.04 | 1.3595 | 18.4 | 5.4 | 29.76 |
| 16 | 160 | 3.04 | 1.3595 | 18.4 | 5.5 | 26.81 |
| 17 | 160 | 3.04 | 1.3595 | 18.4 | 5.8 | 21.56 |
| 18 | 160 | 3.04 | 1.3595 | 18.4 | 6.2 | 17.75 |
| 19 | 160 | 3.04 | 1.3595 | 18.4 | 6.7 | 14.97 |
| 20 | 160 | 3.04 | 1.3595 | 18.4 | 6.8 | 14.45 |
| 21 | 201.6 | 1.6 | 1.3775 | 23.2 | 5.8 | 31.63 |
| 22 | 201.6 | 1.6 | 1.3775 | 23.2 | 5.9 | 28.88 |
| 23 | 201.6 | 1.6 | 1.3775 | 23.2 | 6.2 | 23.25 |
| 24 | 201.6 | 1.6 | 1.3775 | 23.2 | 6.7 | 18.39 |
| 25 | 201.6 | 1.6 | 1.3775 | 23.2 | 7.2 | 15.64 |
| 26 | 201.6 | 1.6 | 1.3775 | 23.2 | 7.9 | 13.23 |
| 27 | 201.6 | 1.6 | 1.3775 | 23.2 | 8.4 | 12.31 |
| 28 | 240.8 | 4.14 | 1.51 | 18.4 | 5.65 | 37.35 |
| 29 | 240.8 | 4.14 | 1.51 | 18.4 | 5.7 | 34.97 |
| 30 | 240.8 | 4.14 | 1.51 | 18.4 | 5.8 | 31.2 |
| 31 | 240.8 | 4.14 | 1.51 | 18.4 | 6 | 26.39 |
| 32 | 240.8 | 4.14 | 1.51 | 18.4 | 6.2 | 23.18 |
| 33 | 240.8 | 4.14 | 1.51 | 18.4 | 6.9 | 17.26 |
| 34 | 283.8 | 2.23 | 1.5306 | 23.2 | 7.2 | 18.92 |
| 35 | 283.8 | 2.23 | 1.5306 | 23.2 | 7.6 | 16.84 |
| 36 | 283.8 | 2.23 | 1.5306 | 23.2 | 7.9 | 15.6 |
| 37 | 283.8 | 2.23 | 1.5306 | 23.2 | 8.4 | 14.03 |
| 38 | 283.8 | 2.23 | 1.5306 | 23.2 | 8.9 | 12.987 |
| 39 | 412.1 | 4.16 | 1.7643 | 18.4 | 6.8 | 26.25 |

Table 5-219 PU-SOL-THERM-025 Critical Experiment Characteristics

| Case | Pu Conc. (g/l) | Molarity | Density ($g/cm^3$) | $^{240}Pu$ wt.% | Thickness (in) | Height (in) |
|---|---|---|---|---|---|---|
| 40 | 412.1 | 4.16 | 1.7643 | 18.4 | 7 | 23.67 |
| 41 | 412.1 | 4.16 | 1.7643 | 18.4 | 7.4 | 20.23 |
| 42 | 412.1 | 4.16 | 1.7643 | 18.4 | 7.9 | 17.46 |
| 43 | 412.1 | 4.16 | 1.7643 | 18.4 | 8.4 | 15.63 |
| 44 | 412.1 | 4.16 | 1.7643 | 18.4 | 8.9 | 14.32 |

Table 5-220 Results for PU-SOL-THERM-025

| Experiment /Case | Benchmark-Model | | Calculated[a] | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/Pu[b] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| pst025_1 | 1.0000 | 0.0039 | 0.011 | 1.0176 | 0.0011 | -0.0176 | ± | 0.0081 | 424.3 |
| pst025_2 | 1.0000 | 0.0039 | 0.011 | 1.0170 | 0.0012 | -0.0170 | ± | 0.0082 | 424.3 |
| pst025_3 | 1.0000 | 0.0039 | 0.011 | 1.0186 | 0.0012 | -0.0186 | ± | 0.0082 | 424.3 |
| pst025_4 | 1.0000 | 0.0039 | 0.011 | 1.0157 | 0.0013 | -0.0157 | ± | 0.0082 | 424.3 |
| pst025_5 | 1.0000 | 0.0039 | 0.011 | 1.0178 | 0.0012 | -0.0178 | ± | 0.0082 | 424.3 |
| pst025_6 | 1.0000 | 0.0039 | 0.011 | 1.0109 | 0.0012 | -0.0109 | ± | 0.0082 | 424.3 |
| pst025_7 | 1.0000 | 0.0038 | 0.014 | 1.0046 | 0.0013 | -0.0046 | ± | 0.0080 | 367.2 |
| pst025_8 | 1.0000 | 0.0038 | 0.013 | 1.0091 | 0.0012 | -0.0091 | ± | 0.0080 | 367.2 |
| pst025_9 | 1.0000 | 0.0038 | 0.014 | 1.0049 | 0.0012 | -0.0049 | ± | 0.0080 | 367.2 |
| pst025_10 | 1.0000 | 0.0038 | 0.013 | 1.0071 | 0.0012 | -0.0071 | ± | 0.0080 | 367.2 |
| pst025_11 | 1.0000 | 0.0038 | 0.013 | 1.0053 | 0.0012 | -0.0053 | ± | 0.0080 | 367.2 |
| pst025_12 | 1.0000 | 0.0038 | 0.014 | 1.0093 | 0.0012 | -0.0093 | ± | 0.0080 | 367.2 |
| pst025_13 | 1.0000 | 0.0038 | 0.013 | 1.0053 | 0.0013 | -0.0053 | ± | 0.0080 | 367.2 |
| pst025_14 | 1.0000 | 0.0040 | 0.032 | 1.0046 | 0.0013 | -0.0046 | ± | 0.0084 | 143.9 |
| pst025_15 | 1.0000 | 0.0040 | 0.031 | 1.0051 | 0.0013 | -0.0051 | ± | 0.0084 | 143.9 |
| pst025_16 | 1.0000 | 0.0040 | 0.031 | 1.0060 | 0.0013 | -0.0060 | ± | 0.0084 | 143.9 |
| pst025_17 | 1.0000 | 0.0040 | 0.031 | 1.0044 | 0.0013 | -0.0044 | ± | 0.0084 | 143.9 |
| pst025_18 | 1.0000 | 0.0040 | 0.031 | 1.0057 | 0.0013 | -0.0057 | ± | 0.0084 | 143.9 |
| pst025_19 | 1.0000 | 0.0040 | 0.032 | 1.0043 | 0.0013 | -0.0043 | ± | 0.0084 | 143.9 |
| pst025_20 | 1.0000 | 0.0040 | 0.032 | 1.0028 | 0.0014 | -0.0028 | ± | 0.0085 | 143.9 |
| pst025_21 | 1.0000 | 0.0044 | 0.039 | 0.9987 | 0.0012 | 0.0013 | ± | 0.0091 | 115.8 |
| pst025_22 | 1.0000 | 0.0044 | 0.040 | 0.9975 | 0.0013 | 0.0025 | ± | 0.0092 | 115.8 |
| pst025_23 | 1.0000 | 0.0044 | 0.040 | 1.0003 | 0.0013 | -0.0003 | ± | 0.0092 | 115.8 |
| pst025_24 | 1.0000 | 0.0044 | 0.041 | 0.9950 | 0.0014 | 0.0050 | ± | 0.0092 | 115.8 |

Table 5-220 Results for PU-SOL-THERM-025

| Experiment /Case | Benchmark-Model | | Calculated[a] | | | $\Delta k_{eff} \pm 2\sigma$ | | | H/Pu[b] |
|---|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | | |
| pst025_25 | 1.0000 | 0.0044 | 0.040 | 0.9970 | 0.0013 | 0.0030 | ± | 0.0092 | 115.8 |
| pst025_26 | 1.0000 | 0.0044 | 0.041 | 0.9956 | 0.0014 | 0.0044 | ± | 0.0092 | 115.8 |
| pst025_27 | 1.0000 | 0.0044 | 0.041 | 0.9935 | 0.0013 | 0.0065 | ± | 0.0092 | 115.8 |
| pst025_28 | 1.0000 | 0.0048 | 0.050 | 0.9954 | 0.0013 | 0.0046 | ± | 0.0099 | 87.5 |
| pst025_29 | 1.0000 | 0.0048 | 0.050 | 0.9938 | 0.0013 | 0.0062 | ± | 0.0099 | 87.5 |
| pst025_30 | 1.0000 | 0.0048 | 0.050 | 0.9957 | 0.0013 | 0.0043 | ± | 0.0099 | 87.5 |
| pst025_31 | 1.0000 | 0.0048 | 0.050 | 0.9991 | 0.0013 | 0.0009 | ± | 0.0099 | 87.5 |
| pst025_32 | 1.0000 | 0.0048 | 0.050 | 0.9958 | 0.0013 | 0.0042 | ± | 0.0099 | 87.5 |
| pst025_33 | 1.0000 | 0.0048 | 0.052 | 0.9943 | 0.0013 | 0.0057 | ± | 0.0099 | 87.5 |
| pst025_34 | 1.0000 | 0.0049 | 0.059 | 0.9962 | 0.0013 | 0.0038 | ± | 0.0101 | 77.7 |
| pst025_35 | 1.0000 | 0.0049 | 0.060 | 0.9940 | 0.0012 | 0.0060 | ± | 0.0101 | 77.7 |
| pst025_36 | 1.0000 | 0.0049 | 0.060 | 0.9956 | 0.0012 | 0.0044 | ± | 0.0101 | 77.7 |
| pst025_37 | 1.0000 | 0.0049 | 0.060 | 0.9949 | 0.0012 | 0.0051 | ± | 0.0101 | 77.7 |
| pst025_38 | 1.0000 | 0.0049 | 0.060 | 0.9940 | 0.0013 | 0.0060 | ± | 0.0101 | 77.7 |
| pst025_39 | 1.0000 | 0.0060 | 0.094 | 0.9961 | 0.0013 | 0.0039 | ± | 0.0123 | 44.9 |
| pst025_40 | 1.0000 | 0.0060 | 0.095 | 0.9927 | 0.0013 | 0.0073 | ± | 0.0123 | 44.9 |
| pst025_41 | 1.0000 | 0.0060 | 0.095 | 0.9929 | 0.0012 | 0.0071 | ± | 0.0122 | 44.9 |
| pst025_42 | 1.0000 | 0.0060 | 0.096 | 0.9936 | 0.0012 | 0.0064 | ± | 0.0122 | 44.9 |
| pst025_43 | 1.0000 | 0.0060 | 0.096 | 0.9894 | 0.0013 | 0.0106 | ± | 0.0123 | 44.9 |
| pst025_44 | 1.0000 | 0.0060 | 0.097 | 0.9904 | 0.0012 | 0.0096 | ± | 0.0122 | 44.9 |

NOTES: [a] In the inputs, the slots of the egg-crate grid are 0.3174 cm thick rather than 0.3175 cm thick. The $^{241}Am$ in cases 14-20 is modeled as having a number density of 1.1192e-8 vs. reference of 1.1192e-6. Also the Si in the Stainless Steel composition has density of 8.5983e-4 vs. the reference value of 8.5982e-4. These small differences from the benchmark model are not expected to affect $k_{eff}$ by more than two standard deviations.

[b] H/Pu (all Pu and Am isotopes) ratio is calculated from the number densities provided in section 3.3 of NEA 2001, PU-SOL-THERM-024.

### 5.10.2.1.14 PU-SOL-THERM-026

During the late 1960's a large number of critical experiments were performed on plutonium nitrate solutions in slab geometry at Battelle-PNL. For the critical experiments the plutonium concentrations in the solutions ranged from about 58 to 412 g/l. Three different plutonium isotopic compositions were used in these experiments – 4.6, 18.4, and 23.2 wt.% $^{240}Pu$. Twenty-one bare and unreflected cases are listed in Table 5-221, were derived from the total of thirty-five experiments. Fifteen cases are individual experiments. Six cases are composites formed by combining experiments with the same reported solution parameters and critical solution

thickness. An average critical solution height was used to characterize each composite case. The 21 cases are acceptable as criticality benchmarks (NEA [2001]). Results for the MCNP models are presented in Table 5-222.

Table 5-221 PU-SOL-THERM-026 Critical Experiment Characteristics

| Case | Pu Conc. (g/l) | Solution Density ($g/cm^3$) | Molarity | $^{240}Pu$ wt.% | Solution Thickness (cm) | Solution Height (cm) | $H/^{239}Pu$[a] |
|---|---|---|---|---|---|---|---|
| 1 | 4.6 | 57.3 | 2.29 | 1.1692 | 16.002 | 97.358 | 452.5 |
| 2 | 4.6 | 57.7 | 2.38 | 1.1711 | 16.256 | 81.509 | 447.6 |
| 3 | 4.6 | 57.7 | 2.41 | 1.1721 | 16.51 | 70.612 | 447.3 |
| 4 | 18.4 | 66.5 | 2.36 | 1.1848 | 19.05 | 93.754 | 484.5 |
| 5 | 18.4 | 66.5 | 2.36 | 1.1848 | 19.304 | 83.467 | 484.5 |
| 6 | 18.4 | 66.5 | 2.36 | 1.1848 | 20.066 | 64.851 | 484.5 |
| 7 | 18.4 | 66.5 | 2.36 | 1.1848 | 21.336 | 50.239 | 484.5 |
| 8 | 18.4 | 66.5 | 2.36 | 1.1848 | 22.606 | 42.857 | 484.5 |
| 9 | 18.4 | 160 | 3.04 | 1.3595 | 19.304 | 96.774 | 189.8 |
| 10 | 18.4 | 160 | 3.04 | 1.3595 | 19.558 | 85.242 | 189.8 |
| 11 | 18.4 | 160 | 3.04 | 1.3595 | 20.066 | 71.272 | 189.8 |
| 12 | 18.4 | 160 | 3.04 | 1.3595 | 20.574 | 62.433 | 189.8 |
| 13 | 18.4 | 160 | 3.04 | 1.3595 | 21.336 | 54 | 189.8 |
| 14 | 18.4 | 160 | 3.04 | 1.3595 | 22.606 | 45.364 | 189.8 |
| 15 | 23.2 | 201.6 | 1.6 | 1.3775 | 21.336 | 67.208 | 161.3 |
| 16 | 23.2 | 201.6 | 1.6 | 1.3775 | 22.606 | 55.651 | 161.3 |
| 17 | 18.4 | 240.8 | 4.14 | 1.51 | 21.209 | 89.51 | 115.4 |
| 18 | 18.4 | 240.8 | 4.14 | 1.51 | 21.59 | 78.892 | 115.4 |
| 19 | 18.4 | 240.8 | 4.14 | 1.51 | 22.098 | 69.342 | 115.4 |
| 20 | 18.4 | 240.8 | 4.14 | 1.51 | 22.543 | 63.17 | 115.4 |
| 21 | 18.4 | 240.8 | 4.14 | 1.51 | 22.606 | 62.408 | 115.4 |

NOTE: [a] $H/^{239}Pu$ is calculated from solution density data given in Section 3.3 of NEA (2001, PU-SOL-THERM-026).

Table 5-222 Results for PU-SOL-THERM-026

| Experiment/ Case | Benchmark-Model | | Calculated[a] | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| Pst026_01 | 1.0000 | 0.0052 | 0.013 | 1.00237 | 0.00097 | 0.0024 | ± | 0.0106 |
| Pst026_02 | 1.0000 | 0.0052 | 0.013 | 1.00011 | 0.00093 | 0.0001 | ± | 0.0106 |
| Pst026_03 | 1.0000 | 0.0051 | 0.013 | 1.00132 | 0.00091 | 0.0013 | ± | 0.0104 |

Table 5-222 Results for PU-SOL-THERM-026

| Experiment/ Case | Benchmark-Model | | Calculated[a] | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| Pst026_04 | 1.0000 | 0.0045 | 0.015 | 1.00241 | 0.00089 | 0.0024 | ± | 0.0092 |
| Pst026_05 | 1.0000 | 0.0044 | 0.015 | 1.00173 | 0.00093 | 0.0017 | ± | 0.0090 |
| Pst026_06 | 1.0000 | 0.0043 | 0.015 | 1.00079 | 0.00092 | 0.0008 | ± | 0.0088 |
| Pst026_07 | 1.0000 | 0.0041 | 0.015 | 1.00273 | 0.00092 | 0.0027 | ± | 0.0084 |
| Pst026_08 | 1.0000 | 0.0039 | 0.015 | 1.00325 | 0.00092 | 0.0032 | ± | 0.0080 |
| Pst026_09 | 1.0000 | 0.0049 | 0.037 | 0.99951 | 0.00096 | -0.0005 | ± | 0.0100 |
| Pst026_10 | 1.0000 | 0.0048 | 0.037 | 1.00211 | 0.00096 | 0.0021 | ± | 0.0098 |
| Pst026_11 | 1.0000 | 0.0047 | 0.037 | 1.00208 | 0.00099 | 0.0021 | ± | 0.0096 |
| Pst026_12 | 1.0000 | 0.0047 | 0.037 | 1.00227 | 0.00099 | 0.0023 | ± | 0.0096 |
| Pst026_13 | 1.0000 | 0.0045 | 0.037 | 1.00111 | 0.00097 | 0.0011 | ± | 0.0092 |
| Pst026_14 | 1.0000 | 0.0044 | 0.037 | 1.00536 | 0.00095 | 0.0054 | ± | 0.0090 |
| Pst026_15 | 1.0000 | 0.0046 | 0.047 | 0.99000 | 0.00096 | -0.0100 | ± | 0.0094 |
| Pst026_16 | 1.0000 | 0.0044 | 0.047 | 0.99609 | 0.00098 | -0.0039 | ± | 0.0090 |
| Pst026_17 | 1.0000 | 0.0050 | 0.059 | 0.99486 | 0.00094 | -0.0051 | ± | 0.0102 |
| Pst026_18 | 1.0000 | 0.0049 | 0.060 | 0.99501 | 0.00096 | -0.0050 | ± | 0.0100 |
| Pst026_19 | 1.0000 | 0.0049 | 0.059 | 0.99537 | 0.00092 | -0.0046 | ± | 0.0100 |
| Pst026_20 | 1.0000 | 0.0049 | 0.060 | 0.99477 | 0.00092 | -0.0052 | ± | 0.0100 |
| Pst026_21 | 1.0000 | 0.0049 | 0.060 | 0.99480 | 0.00095 | -0.0052 | ± | 0.0100 |

NOTE: [a] In the inputs, the slots of the egg-crate grid are 0.3174 cm thick rather than 0.3175 cm thick. This small difference from the benchmark model is not expected to affect keff.

### 5.10.2.2 Pu and Mixed Metal Configurations

#### 5.10.2.2.1 MIX-MET-MIXED-001

The experiment presented in this section was performed in 1987 at VNIITF using the criticality test facility FKBN (Vertical Lift Machine). The critical assembly was a cylinder consisting of alternating layers of plutonium, highly enriched uranium, and polyethylene. The experiment was used for neutron-data validation in the range of thermal and resonance neutron energies. The configuration of the system is simple enough and measurements were made with sufficient precision for the experiment to be considered as a criticality-safety benchmark.

The critical assembly was a cylinder, 19.99±0.01 cm in diameter, divided by a gap into two parts: a movable bottom and a stationary top. The critical separation between the bottom and the top was 1.12±0.02 cm. The critical assembly was built of alternating layers of plutonium, highly enriched uranium and polyethylene. Used in the assembly were 0.490±0.005 cm-thick Pu discs canned in steel foil, 0.995±0.003 cm-thick HEU discs, 0.990±0.005 and 0.490±0.005 cm-thick

polyethylene discs, and 0.490±0.005 cm-thick polyethylene rings (annular discs around the Pu discs). These average disc thicknesses were measured for stacks and, therefore, included possible clearances between the discs. Each layer of material consisted of one or more discs. Based on weighting the number densities by the relative thicknesses of the various discs, the H/ $^{239}$Pu and H/ $^{235}$U ratios were calculated to be 3.57 and 1.71, respectively. The H/Fissile atom ratio is calculated from the data given in Section 3.3 of MIX-MET-MIXED-001 as 2.31. The MCNP result for this experiment is given in Table 5-223.

Table 5-223 Results for MIX-MET-MIXED-001

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| mixmm1 | 0.9991 | 0.0013 | 0.777 | 0.9914 | 0.00042 | -0.0077 | ± | 0.0027 |

### 5.10.2.2.2 Los Alamos Spherical MIX-MET-FAST Benchmarks

The following is a listing of Los Alamos spherical experiments with brief descriptions. The detailed descriptions of the experiments are provided in NEA (2001). The results of these experiments are presented in Table 5-224.

MIX-MET-FAST-001

In October 1958, a critical experiment at Los Alamos Scientific Laboratory was performed using a spherical mass of delta-phase plutonium surrounded by highly enriched uranium. The experiment was performed using the Planet universal assembly machine. Two hemispheres, each with a 0.85-inch-diameter central source cavity and a diameter of 3.970 inches, were constructed of delta-phase plutonium alloy plated with 0.005-inch-thick nickel. Hemishells of various thicknesses of highly enriched uranium were constructed to enclose the plutonium hemispheres. The experiment is considered to be acceptable as a benchmark critical experiment (NEA [2001]) and identified as case mixmf1 in Table 5-224.

MIX-MET-FAST-002

In October 1960 a series of three critical experiments was performed at Los Alamos Scientific Laboratory using spherical composite cores of alpha-phase plutonium and highly enriched uranium reflected by natural uranium. The three experiments were performed using three distinct spherical masses of alpha-phase plutonium of varying $^{240}$Pu contents, surrounded by varying thicknesses of uranium highly enriched in $^{235}$U, and reflected by varying thicknesses of natural uranium. Of the three experiments, one was critical, another was slightly supercritical and the last one was subcritical. A curve fit to the data was used to obtain the critical mass of the supercritical experiment. The data was extrapolated to obtain the critical mass of the subcritical experiment. All three experiments are acceptable as benchmark critical experiments (NEA [2001]).

The three experiments studied in this section are composite cores of plutonium and highly enriched uranium reflected by natural uranium. The experiments are differentiated by their cores and the thicknesses of their reflectors. The first experimental assembly, identified as case mixmf2-1 in Table 5-224, is composed of a sphere of plutonium metal with a mass of 1615.45 g ($^{240}$Pu content was 2.34 at.%) surrounded by 0.974 inches of highly enriched uranium metal and reflected by 7.45 inches of natural uranium.. The second experimental assembly, identified as case mixmf2-1 in Table 5-224, is composed of a sphere of plutonium metal with a mass of 1610.30 g ($^{240}$Pu content was 4.73 at.%) surrounded by 0.988 inches of and highly enriched uranium metal and reflected by 7.43 inches of natural uranium. The third experimental assembly, identified as case mixmf2-1 in Table 5-224, is composed of a sphere of plutonium metal with a mass of 1611.19 g ($^{240}$Pu content was 16.1 at.%) surrounded by 1.039 inches of and highly enriched uranium metal and reflected by 7.38 inches of natural uranium.

Table 5-224 Results for MIX-MET-FAST-001 and -002

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ |
|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | |
| mixmf1 | 1.0000 | 0.0016 | 1.807 | 0.99601 | 0.0020 | 0.0000 ± 0.0000 |
| mixmf2-1 | 1.0000 | 0.0042 | 1.713 | 1.0061 | 0.0010 | 0.0061 ± 0.0086 |
| mixmf2-2 | 1.0000 | 0.0044 | 1.705 | 1.0080 | 0.0011 | 0.0061 ± 0.0086 |
| mixmf2-3 | 1.0000 | 0.0048 | 1.700 | 1.0067 | 0.0010 | 0.0067± 0.0098 |

### 5.10.2.2.3 VNIITF CTF Spherical MIX-MET-FAST Benchmarks

The following is a listing of VNIITF spherical experiments with brief descriptions. The detailed descriptions of the experiments are provided in NEA (2001). The results of these experiments are presented in Table 5-225.

MIX-MET-FAST-003

This experiment is a critical configuration that was assembled in 1992 using the VNIITF Criticality Test Facility (CTF). The assembly is a sphere of plutonium surrounded by highly enriched uranium Measurements were made with sufficient precision, and the configuration of the system is simple enough and defined well enough, for the experiment to be considered as a criticality safety benchmark. (NEA [2001]).

The VNIITF critical assemblies are "split," (i.e., they are divided into two parts by a gap). They are more complicated than classical "unsplit" critical assemblies. Their description includes two additional parameters: (1) the critical separation, (hcr) (i.e., the distance between flat surfaces of the facing hemispheres [including both the thickness of the diaphragm and the height of the void gap]); and (2) the separation radius (Rsep) describing the manner of the assembly separation. (Rsep) is the radius of the portion of the upper

hemisphere that remains with the lower, movable mass. The split assemblies have an important advantage: the extrapolation from a real configuration to an "unsplit" one and related errors are avoided.

A neutron source of $^{238}$Pu-Be, 1 cm in diameter and 1 cm in length, which produces $1.3 \times 10^6$ neutrons/s and does not disturb the criticality of the assembly, was placed inside the central cavity. In this experiment, neutron multiplication, (M), was measured as a function of the gap width, (h). Remote measurements of the gap width were performed with an accuracy of 0.01 cm. The maximum value M=540 was obtained, with a gap width of 1.29±0.01 cm. The critical gap width, hcr=1.225 cm, was determined by extrapolation of the function 1/M(h) to zero. In this extrapolation only measurements with M > 100 were used. The combined uncertainty in hcr from the measurements and from the extrapolation was less than 0.02 cm (NEA [2001]).

MIX-MET-FAST-004

The experiments described here are two critical configurations that were assembled in 1983 using the VNIITF Criticality Test Facility. Each assembly is a sphere of plutonium surrounded by spherical shells of highly enriched uranium and reflected by beryllium or beryllium oxide. These experiments compare the effects of the two reflective materials, because the cores and the reflector sizes are the same. Measurements were made with sufficient precision, and the configuration of the system is simple enough and defined well enough, for the experiments to be considered as criticality-safety benchmarks (NEA [2001]).

MIX-MET-FAST-005

The experiment described here is a critical configuration that was assembled in 1992 using the VNIITF Criticality Test Facility. The assembly is a sphere of plutonium surrounded by highly enriched uranium and reflected by aluminium. This experiment was used to validate neutron data of aluminium. Measurements were made with sufficient precision, and the configuration of the system is simple enough and defined well enough, for the experiment to be considered as a criticality safety benchmark (NEA [2001]).

MIX-MET-FAST-009

Criticality measurements of a δ-phase metal $^{239}$Pu(98 wt.%) assembly with a 0.75-cm-thick external shell of $^{235}$U (90 wt.%) were performed in 1990 at the VNIIEF criticality test facility. In 1997 the latest computational and theoretical re-evaluation of these experiments was done. These efforts resulted in an acceptable spherical benchmark model of the critical assembly.

The assembly core had a central cavity of 1.0-cm radius and included 6 spherical layers of $^{239}$Pu (δ, 98%) metal and an outer layer of $^{235}$U (90%) metal. The innermost layer of plutonium consisted of one hemispherical piece. The total value of the uncertainty in the plutonium layer radii, averaged over 6 layers, was 0.06%. The value of the uncertainty in the uranium layer radii was 0.04%. Each layer of $^{239}$Pu (δ, 98 wt.%) from layer two through layer six comprised two hemispherical pieces with cylindrical pole holes 2.2 cm in diameter that were plugged during measurements with specialized plugs of the appropriate layer material. The layer of $^{235}$U(90 wt.%) also comprised two hemispherical pieces with cylindrical pole holes 2.2 cm in diameter that were plugged during measurements with specialized plugs of this layer material. The assembly included two separate units. The upper unit, comprising one hemisphere of $^{235}$U (90 wt.%), was laid on a steel diaphragm 0.1 cm thick. The other pieces were all incorporated in the lower (movable) unit.

MIX-MET-FAST-010

Criticality measurements of a δ-phase $^{239}$Pu (98 wt.%) metal assembly with a 2.89-cm-thick external shell of $^{235}$U (90 wt.%) were performed in 1990 at the VNIIEF criticality test facility. In 1997 the latest computational and theoretical re-evaluation of these experiments was done. These efforts resulted in an acceptable spherical benchmark model of the critical assembly.

The experiment consisted of a $^{239}$Pu (δ, 98 wt.%) assembly with an external shell of $^{235}$U (90 wt.%) positioned on the CTF. The assembly core had a central cavity of 1.4-cm radius and included three spherical layers of $^{239}$Pu (δ, 98 wt.%) metal and 4 outer spherical layers of $^{235}$U (90 wt.%) metal. The total value of the uncertainty in the plutonium layer radii, averaged over three layers, was 0.06%. The value of the uncertainty in the uranium layer radii, averaged over four layers, was 0.04%. All layers of $^{239}$Pu(δ, 98 wt.%) and $^{235}$U(90 wt.%) were composed of two hemispherical pieces with cylindrical pole holes 2.2-cm in diameter that were plugged during measurements with specialized plugs of the appropriate layer material. The assembly included two separate units. The upper unit, comprising one hemisphere of $^{235}$U(90 wt.%) (with inner and outer radii of 6.75 cm and 7.55 cm, respectively), was laid on a steel diaphragm 0.1-cm thick. The other pieces were all incorporated in the lower (movable) unit.

Table 5-225 Results for Russian VNIITF CTF Spherical MIX-MET-FAST Benchmarks

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | σ | AENCF | $k_{eff}$ | σ | | | |
| mixmf3 | 0.9993 | 0.0016 | 1.774 | 0.99931 | 0.00061 | 0.0000 | ± | 0.0034 |
| mmf4-38 | 0.9993 | 0.0013 | 1.547 | 0.99862 | 0.00064 | -0.0007 | ± | 0.0029 |
| mmf4-39 | 0.9993 | 0.0013 | 1.541 | 1.00015 | 0.00063 | 0.0009 | ± | 0.0029 |
| mixmf5 | 0.9990 | 0.0017 | 1.740 | 1.00053 | 0.00031 | 0.0015 | ± | 0.0035 |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| mixmf9 | 1.0000 | 0.0010 | 1.851 | 0.99771 | 0.00059 | -0.0023 | ± | 0.0023 |
| mixmf10 | 1.0000 | 0.0009 | 1.723 | 0.99839 | 0.00061 | -0.0016 | ± | 0.0022 |

### 5.10.2.2.4 Los Alamos Pu Spherical Benchmarks

The following is a listing of Los Alamos Pu spherical experiments with brief descriptions. The detailed descriptions of the experiments are provided in NEA (2001). The results of these experiments are presented in Table 5-226.

PU-MET-FAST-001

In the early 1950s, the $^{239}$Pu Jezebel critical assembly was fabricated and operated at the Los Alamos Scientific Laboratory. There were actually three Jezebel assemblies built, one using Pu (4.5 at.% $^{240}$Pu) and referred to as the $^{239}$Pu Jezebel, one using Pu (20 at.% $^{240}$Pu) and referred to as the $^{240}$Pu Jezebel, and one using $^{233}$U and referred to as the $^{233}$U Jezebel. Only the $^{239}$Pu Jezebel is described in this section. The $^{239}$Pu Jezebel was an unreflected, or bare, delta-phase $^{239}$Pu critical assembly, nearly spherical in shape. It is considered to be acceptable for use as a benchmark critical experiment (NEA [2001]).

Two different configurations of $^{239}$Pu Jezebel were used as basic critical experiments. Configuration A contained 16.751 kg of Pu-alloy at an average density of 15.61 g/cm$^3$. It had no polar gross mass adjustment disks and was measured to be subcritical by 0.43 lower mass adjustment plug (equivalent to 10 g alloy distributed over the exterior surface), with all mass adjustment plugs in place and the control rod fully inserted. With this configuration, the critical mass was determined to be 16.761 kg delta-phase Pu alloy. Configuration B contained 16.909 kg alloy at an average density of 15.60 g/cm$^3$. It had two polar gross mass adjustment disks and went critical with six lower mass-adjustment plugs removed and the control rod extracted 1.375 in.

The average composition of the delta-phase plutonium alloy used in Jezebel was 98.98 wt.% plutonium and 1.02 wt.% gallium. The recommended benchmark critical experiment model is a 17,020 gram sphere of plutonium alloy with a density of 15.61 g/cm$^3$, and a radius of 6.3849 cm.

PU-MET-FAST-002

This experiment used the same assembly as for PU-MET-FAST-001. The difference is the composition of Pu. The recommended benchmark critical experiment model is an isolated, bare sphere of delta-phase plutonium alloy with a critical mass = 19.460 ± 0.156 kg alloy at a density of 15.73 g/cm$^3$ which has a radius of 6.6595 cm.

PU-MET-FAST-005

In 1958, an experiment was performed at Los Alamos Scientific Laboratory using a slightly subcritical spherical mass of delta-phase plutonium reflected by tungsten. The experiment is considered to be acceptable as a benchmark critical experiment. This experiment was performed as part of a series of experiments. The experiment was performed using the Planet universal assembly machine. The core was composed of two hemispheres of delta-phase plutonium alloy having a diameter of 3.970 inches and plated with 0.005-inch-thick nickel with a 0.85-inch-diameter source cavity in the center of the two hemispheres. Hemishells of various thicknesses of tungsten were constructed to enclose the plutonium hemispheres. The delta-phase plutonium alloy core consists of 99.00 wt.% plutonium (97.79 at.% $^{239}$Pu) and 1.00 wt.% gallium. The density of the core alloy is 15.778 g/cm$^3$. The idealized critical experiment consists of a 3.970-inch-diameter delta-phase plutonium spherical core (density = 15.778 g/cm$^3$ ) intimately surrounded by a spherical shell of tungsten 1.85-inches thick, with a density of 17.21 g/cm$^3$.

PU-MET-FAST-006

In the mid-1960s a critical experiment was performed at Los Alamos Scientific Laboratory using a spherical delta-phase plutonium core reflected by natural uranium. Delayed critical (DC) was achieved. The results of this experiment are considered to be acceptable as a benchmark critical experiment (NEA [2001]).

The Flattop assembly used in this experiment has a core of delta-phase plutonium metal alloy, enclosed in a thick natural uranium reflector. The core is composed of two hemispheres of plutonium metal. Both halves combined to form a sphere with an outside diameter of 3.586 inches, which includes the Ni coating. The benchmark model is a simple delta-phase Pu-alloy sphere with a density of 15.53 g/cm$^3$ and a mass of 6060 grams plutonium with 19.6088-cm natural uranium reflection at a density of 19.0 g/cm$^3$.

PU-MET-FAST-008

From December 1960 through November 1961, a critical assembly was operated at Los Alamos Scientific Laboratory using a spherical mass of delta-phase plutonium closely reflected by thorium. The experiment was designed to study the neutronic properties of thorium in support of a potential fast, $^{233}$U breeder reactor. Uncertainty in the critical mass of this original experiment prompted the assembly to be set up again and reactivity measurements taken. The average of the results for two critical experimental configurations was used to determine the final critical mass. The result of this experiment is considered to be acceptable as a benchmark critical experiment.

The experiment was performed using the Thor assembly machine, a modification of the Planet universal assembly machine. The modifications were to the control rod drive and safety mechanism. The plutonium core was constructed of three major parts; an upper

polar cap, a lower polar cap, and a central section. Together, the three parts approximated a sphere 10.59 cm in diameter when assembled. The upper polar cap and central section remained stationary while the lower polar cap was located on a pneumatic lift. The central section had a 1.27 cm-diameter glory hole. The glory hole was filled with plutonium as needed. All plutonium parts were coated with 0.013 cm-thick nickel.

The cylindrical thorium reflector was composed of the peripheral reflector, the upper safety block, the lower safety block, and a vernier control cylinder. These combined to form a 21.0-inch-diameter cylinder with a height of 21.0 inches. The first reflector part was the peripheral reflector, which was a fixed annular cylinder. The six-inch-diameter upper safety block surrounded the upper polar cap and the central section and fit inside of the upper portion of the peripheral reflector. The 5-inch-diameter lower safety block supported the lower polar cap and fit inside of the lower portion of the peripheral reflector. The three-inch-diameter control rod entered the bottom through the lower safety block directly under the lower polar cap. The control rod had a total travel distance of five inches. The upper safety block and the peripheral reflector had a 1.27-cm glory hole which lined up with the glory hole in the central section of the core. All thorium parts were coated with 0.0013 cm of chromium. The average composition of the plutonium core is: 93.59 wt.% $^{239}$Pu, 5.10 wt.% $^{240}$Pu and 0.30 wt% $^{241}$Pu. The plutonium also contains 1.01 wt.% gallium. The density of the plutonium core is 15.29 g/cm$^3$. The reflector is 100 wt.% $^{232}$Th at a density of 11.58 g/cm$^3$. A one-dimensional and a two dimensional model are reported in Section 3 of NEA (2001, PU-MET-FAST-008). For this study the two-dimensional model is chosen for the calculation.

PU-MET-FAST-009

In 1960 a critical experiment was performed at Los Alamos Scientific Laboratory using a spherical mass of delta-phase plutonium reflected by type 2014 aluminum. Measurements were taken with a single spherical mass of plutonium and three different reflector thicknesses. Measurements with the first two reflector thicknesses resulted in subcritical configurations, and measurements with the final reflector thickness yielded a slightly supercritical configuration. A curve was fitted to the three data points to derive the critical mass. This experiment is considered to be acceptable as a benchmark critical experiment (NEA [2001]).

The idealized one-dimensional sphere of 99.0 wt.% plutonium (94.8 wt.% $^{239}$Pu) and 1.0 wt.% gallium, reflected by a spherical shell of aluminum (type 2014) is what is accepted as the calculational benchmark model. The core consists of 11.154 kg delta-phase plutonium-gallium alloy (4.34 inches in diameter) at a density of 15.90 g/cm$^3$ reflected by 3.12 ± 0.03 inches of Al(2014) at a density of 2.82 g/cm$^3$.

PU-MET-FAST-010

In 1958, an experiment was performed at Los Alamos Scientific Laboratory using a slightly subcritical spherical mass of delta-phase plutonium reflected by natural uranium. The experiment is considered to be acceptable as a benchmark critical experiment (NEA [2001]).

This experiment was performed using the Planet universal assembly machine. Two hemispheres with a 0.85-inch-diameter central source cavity and a diameter of 3.970 inches were constructed of delta-phase plutonium alloy and plated with 0.005-inch-thick nickel. Hemishells of various thicknesses of natural uranium were constructed to enclose the plutonium hemispheres. The delta-phase plutonium alloy core consisted of 99.00 wt.% plutonium (97.79 at.% $^{239}$Pu) and 1.00 wt.% gallium. The density of the core alloy was 15.778 g/cm$^3$. The idealized critical experiment consists of a 3.970-inch-diameter delta-phase plutonium spherical core (density = 15.778 g/cm$^3$ ) intimately surrounded by a spherical shell of natural uranium 1.625 inches thick, with a density of 18.92 g/cm$^3$.

PU-MET-FAST-011

In 1968, an experiment was performed at Los Alamos Scientific Laboratory using two subcritical spherical masses of alpha-phase plutonium reflected by water. From the two subcritical inverse multiplication measurements, an accurate prediction was made of the critical mass of an alpha-phase plutonium sphere reflected by water. The result of this experiment is considered to be acceptable as a benchmark critical experiment. An idealized one dimensional, water-reflected 5790 ± 25 gram Pu (94.5 at.% $^{239}$Pu, 5.18 at.% $^{240}$Pu, 0.30 at.% $^{241}$Pu, 0.02 at.% $^{242}$Pu) sphere is what is accepted as the calculational benchmark model. The uncertainty in $k_{eff}$ is due to a 25 gram uncertainty in the critical mass reported by the experimenters (NEA [2001]).

PU-MET-FAST-018

In 1958 an experiment was performed at Los Alamos Scientific Laboratory using a slightly subcritical spherical mass of delta-phase plutonium reflected by beryllium. The experiment is considered to be acceptable as a benchmark critical experiment. The experiment was performed using the Planet universal assembly machine. Two hemispheres, each with a 0.85-inch-diameter central source cavity and a diameter of 3.970 inches, were constructed of delta-phase plutonium alloy plated with 0.005-inch-thick nickel. Hemishells of various thicknesses of beryllium were constructed to enclose the plutonium hemispheres. The delta-phase plutonium alloy core consists of 99.00 wt.% plutonium (94.79 at.% $^{239}$Pu) and 1.00 wt.% gallium. The density of the core alloy was 15.778 g/cm$^3$. The beryllium reflector was 98.0 wt.% beryllium with 2.0 wt.% oxygen at a density of 1.83 g/cm$^3$. The idealized critical experiment consists of a 3.970-inch-diameter delta-phase plutoniumspherical core (density = 15.778 g/cm$^3$) closely surrounded by a spherical shell of beryllium 1.452-inches thick, with a density of 1.83 g/cm$^3$.

| Table 5-226 Results for Los Alamos Pu Spherical Benchmarks | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| pumf01 | 1.0000 | 0.002 | 1.915 | 0.9987 | 0.00115 | -0.0013 | ± | 0.0046 |
| pumf02 | 1.0000 | 0.002 | 1.915 | 0.9992 | 0.00113 | -0.0008 | ± | 0.0046 |
| pumf05 | 1.0000 | 0.0013 | 1.661 | 1.0100 | 0.00133 | 0.0100 | ± | 0.0037 |
| pmf006 | 1.0000 | 0.003 | 1.901 | 1.0047 | 0.00137 | 0.0047 | ± | 0.0066 |
| pmf8-2[a] | 1.0000 | 0.0006 | 1.7677 | 1.0060 | 0.0009 | 0.0060 | ± | 0.0022 |
| pumf09 | 1.0000 | 0.0027 | 1.804 | 0.9999 | 0.00094 | -0.0001 | ± | 0.0057 |
| pumf10 | 1.0000 | 0.0018 | 1.901 | 1.0004 | 0.00127 | 0.0004 | ± | 0.0044 |
| pumf11 | 1.0000 | 0.001 | 1.518 | 0.9991 | 0.00139 | -0.0009 | ± | 0.0034 |
| pumf18 | 1.0000 | 0.0030 | 1.666 | 1.0002 | 0.00127 | 0.0002 | ± | 0.0065 |

NOTE: [a] One dimensional and two dimensional benchmark model available. Two dimensional benchmark model chosen for in evaluation of this experiment.

### 5.10.2.2.5 Russian VNIITF CTF Pu Spherical Benchmarks

The following is a listing of VNIITF CTF Pu spherical experiments with brief descriptions, and the results are presented Table 5-227. These experiments are acceptable as benechmarks (NEA [2001]).

PU-MET-FAST-019

The experiment described in this section is a critical configuration that was assembled in 1983 using the VNIITF Criticality Test Facility (NEA [2001]). Measurements were made with sufficient precision, and the configuration of the system is simple enough and defined well enough, for the experiment to be considered as a criticality-safety benchmark. The critical assembly is a plutonium sphere reflected by beryllium. The assembly is a spherical unit divided into two parts separated by a gap. The core inner and outer radii are 1.4 cm and 5.35 cm, respectively. The reflector inner and outer radii are 5.35 cm and 11.0 cm, respectively. The equatorial surfaces of the reflector hemispheres are flat and are 0.15 cm from a horizontal plane through the center of the sphere. The mass of the plutonium was 9748.7 g and the volume was 629.937 $cm^3$. The density of the plutonium was 5.4757g/$cm^3$ (NEA [2001]).

PU-MET-FAST-021

The experiments presented in this section are plutonium cylinders reflected by beryllium or beryllium oxide on their ends. Two such critical assemblies were built. These experiments compared the effects of the two reflective materials and were used for neutron data

validation, because the only difference between the two assemblies was in the end reflector material: Be or BeO. Each critical assembly has an identical cylindrical core of 95 at.% $^{239}$Pu and end reflectors of Be or BeO. Each assembly consists of two almost equal parts, divided by a gap. The bottom part is movable, and the top is immovable. The critical separation is 0.010±0.015 cm in the Be reflected assembly and 0.240±0.015 cm in the BeO-reflected assembly. The core is assembled of ten steel-covered plutonium discs 12.00 cm in nominal diameter. The end reflectors consist of beryllium or beryllium oxide discs, 19.99±0.01 cm in diameter. The bottom and top Be reflectors' measured heights are 14.94±0.01 cm and 14.85±0.01 cm, respectively. The Be bottom reflector consists of three nominally 5-cm thick discs. The Be top reflector consists of 20 nominally 0.5-cm-thick discs and one 5-cm-thick disc. The bottom and top BeO reflectors' measured heights are 14.94±0.01 cm. The plutonium bottom and top masses are 4685.5±0.7 g and 4731.7±0.7 g, respectively. The full plutonium mass is 9417.2 ±1 g. The beryllium bottom and top masses, are 8655.7±0.5 g and 8532.9±0.8 g, respectively. The full beryllium reflector mass is 17188.6±1 g. The beryllium oxide bottom and top masses are 13365.7±0.7 g and 13524.8±0.7 g, respectively. The full beryllium oxide reflector mass is 26890.5±1 g (NEA [2001]).

PU-MET-FAST-035

Criticality measurements of lead-reflected delta-phase $^{239}$Pu(98wt.%) metal were conducted by VNIIEF in 1965 at its criticality test facility (NEA [2001]). The assembly core had a central cavity and included 5 spherical layers of plutonium alloy. The lead reflector comprised one spherical layer with a thickness of 3.15 cm. Each core and reflector layer was formed by two hemispherical pieces. The critical-assembly model is a sphere with a core of $^{239}$ Pu(delta phase, 98wt.%) metal alloy having a central cavity of 1.2-cm radius, and with outer-layer radii layer of 3.15, 4.02, 4.66, 5.35, and 6.0 cm, and a single-layer spherical lead reflector with outer radius of 9.15 cm.

PU-MET-FAST-036

Criticality measurements of a spherical assembly of delta-phase $^{239}$Pu(98wt.%) metal with a compound reflector of 0.05-cm-thick cadmium and 2.3-cm thick polyethylene were conducted by VNIIEF in 1965 at its criticality test facility. The critical assembly model is a sphere with a core of $^{239}$Pu(delta-phase, 98wt.%) metal alloy . The critical assembly is represented by a six-layer plutonium spherical core with a central cavity of 1.16-cm radius, and with outer-layer radii of 1.4, 3.15, 4.02, 4.66, 5.35, and 6.00 cm, and a compound reflector, consisting of a contiguous cadmium layer with outer radius of 6.05 cm, and an adjacent three-layer shell of polyethylene with layer outer radii of 6.75, 7.55, and 8.35 cm. The critical mass of the benchmark spherical critical-assembly model was 13.693 kg.

PU-MET-FAST-039

Criticality measurements of a Duralumin-reflected delta-phase $^{239}$Pu(98wt.%) metal assembly were conducted by VNIIEF in 1964. The critical assembly is represented by a spherical plutonium core comprised of five layers with a central cavity of 3.112-cm radius, and outer-layer radii of 4.02, 4.66, 5.35, 6.00, and 6.75 cm. The plutonium core is encompassed by a three-layer Duralumin reflector with outer-layer radii of 8.35, 9.15, and 11.00 cm. The critical mass of the plutonium in the benchmark spherical critical-assembly model is 17.718 kg.

PU-MET-FAST-040

Criticality measurements of a copper-reflected delta-phase $^{239}$Pu(98wt.%) metal assembly were conducted by VNIIEF in 1964. The fissile material used in the test assembly was delta-phase plutonium metal with a density ranging from 15.21 to 15.31 g/cm$^3$. The mass proportion between the most abundant plutonium nuclides in the core layers, $^{239}$Pu and $^{240}$Pu, was approximately 98.2% and 1.8%, respectively. The critical assembly is represented by a spherical core comprising five layers of plutonium metal alloy and having a central cavity of 1.531-cm radius and outer-layer radii of 3.15, 4.02, 4.66, 5.35 and 6.00 cm. The plutonium sphere is surrounded by one spherical layer of copper reflector with an outer radius of 7.6 cm. The critical mass of the benchmark spherical critical-assembly model was 13.564 kg.

PU-MET-FAST-041

Criticality measurements of a depleted-uranium-reflected, alpha-phase $^{239}$Pu(88wt.%) metal assembly were conducted by VNIIEF in April 1965. The fissile material used in the test assembly was alpha-phase plutonium metal with a density ranging from 18.54 to 18.93 g/cm$^3$. The critical assembly is represented by a 4-layer plutonium core with a central cavity of 0.901-cm radius and with outer-layer radii of 1.0, 1.40, 3.15, and 4.02 cm. The core is surrounded by a 17- layer depleted uranium reflector with layer outer radii of 4.66, 5.35, 6.00, 6.75, 7.55, 8.35, 9.15, 11.0, 12.25, 13.25, 14.0, 15.0, 16.5, 18.0, 21.5, 23.0, and 25.0 cm . The resulting critical mass of the benchmark spherical critical assembly model was 5.070 kg.

Table 5-227 Results for Russian VNIIEF CTF Pu Spherical Benchmarks

| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| pumf019 | 0.9992 | 0.0015 | 1.6200 | 1.0005 | 0.0007 | 0.0013 | ± | 0.0033 |
| pumf021-1 | 1.0000 | 0.0026 | 1.6610 | 1.0032 | 0.0007 | 0.0032 | ± | 0.0054 |
| pumf021-2 | 1.0000 | 0.0026 | 1.667 | 0.9979 | 0.0006 | -0.0021 | ± | 0.0054 |
| pmf035 | 1.0000 | 0.0016 | 1.8240 | 1.0085 | 0.0007 | 0.0085 | ± | 0.0035 |

| Table 5-227 Results for Russian VNIIEF CTF Pu Spherical Benchmarks | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Experiment/ Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| pmf036 | 1.0000 | 0.0031 | 1.7160 | 1.0029 | 0.0007 | 0.0029 | ± | 0.0063 |
| pmf039 | 1.0000 | 0.0022 | 1.8110 | 0.9870 | 0.0006 | -0.0130 | ± | 0.0046 |
| pmf040 | 1.0000 | 0.0038 | 1.7940 | 0.9912 | 0.0006 | -0.0088 | ± | 0.0077 |
| pmf041 | 1.0000 | 0.0016 | 1.9440 | 1.0092 | 0.0007 | 0.0092 | ± | 0.0035 |

### 5.10.2.3 Pu Si-Containing Experiments

#### 5.10.2.3.1 PU-MET-MIXED-001

A series of six critical heterogeneous configurations of plutonium, silicon dioxide, and polyethylene was performed at the Institute of Physics and Power Engineering (IPPE), Obninsk, Russia at the Big Physical Stand (BFS) facility in October 1999 through February 2000. The U.S. National Spent Nuclear Fuel Program (NSNFP) at the Idaho National Engineering and Environmental Laboratory (INEEL) sponsored these experiments (NEA [2001]). The experiments were directed to obtaining data that are applicable to a wide range of criticality safety operations involving fissile contaminated waste streams. In particular, the data were intended for NSNFP validations of criticality safety calculations performed in support of storage of spent nuclear fuels in the United States geologic repository at Yucca Mountain, Nevada.

The experiments were performed at the BFS-1 facility and the critical configurations were designated as BFS-81/1, -81/1A, -81/2, -81/3, -81/4, and -81/5, according to the rules of the BFS facility.

The critical assemblies were composed of sets of aluminum tubes (5.0 cm in outer diameter, 0.1 cm wall thickness, and about 2 m long) filled with pellets of plutonium metal in stainless steel cans, silicon dioxide, and polyethylene. Also, quartz (silicon dioxide) sand was used to fill tubes in the reflector at the core periphery. Thin polyethylene dowels were inserted into the space between the tubes in some cases, to provide well-defined moderation similar to water. The pitch of the Hexagonal lattice of the tubes was 5.1 cm.

Six plutonium-fueled experiments are listed in NEA (2001) and are considered to be acceptable as criticality safety benchmark experiments. While the experiments of this series cover a wide range of spectra of neutrons causing fission (two thermals, one mixed, and three intermediate cases), the experiments are categorized as having mixed spectra. Five of the six cases (omitted case is BFS-81/1) have been modeled and the results are provided in Table 5-228.

Table 5-228 Results for PU-MET-MIXED-001

| Experiment /Case | Benchmark-Model | | Calculated | | | $\Delta k_{eff} \pm 2\sigma$ | | |
|---|---|---|---|---|---|---|---|---|
| | $k_{eff}$ | $\sigma$ | AENCF | $k_{eff}$ | $\sigma$ | | | |
| pmm1-2 | 1.0002 | 0.0025 | 0.3781 | 1.0051 | 0.00057 | 0.0049 | ± | 0.00513 |
| pmm1-3 | 1.0000 | 0.0025 | 0.3392 | 1.0088 | 0.00058 | 0.0088 | ± | 0.00513 |
| pmm1-4 | 1.0001 | 0.0025 | 0.2175 | 1.0165 | 0.00058 | 0.0164 | ± | 0.00513 |
| Pmm1-5 | 1.0003 | 0.0025 | 0.2157 | 1.0159 | 0.0006 | 0.0156 | | 0.0051 |
| Pmm1-a1[a] | 1.0002 | 0.0032 | 0.4510 | 0.9983 | 0.00056 | -0.0019 | ± | 0.00650 |

NOTE: [a] Configuration BFS-81/1A.

# 6. BENCHMARK SELECTION

## 6.1 IDENTIFICATION OF BENCHMARKS APPLICABLE TO THE DOE-OWNED SNF GROUPS

### 6.1.1 HEU Oxide (Shippingport-PWR) DOE-Owned Fuel

#### 6.1.1.1 Benchmarks Related to Intact Waste Package Configurations

The Shippingport PWR type fuel consists of HEU oxide plate fuels that have 93.2% enrichment and form an assembly. Several critical experiments were previously identified as relevant for calculation of an interim critical limit for the Shippingport-PWR DOE SNF (93% enriched) (Section 2.4.1 of CRWMS M&O [2000c]). Criteria in choosing these experiments is that they cover the same or similar fuel type (HEU) and fissile isotope enrichment. These were a series of critical experiments with water-moderated hexagonally-pitched lattices of highly enriched fuel rods of cross-shaped cross section. The designations for these experiments in NEA (2001) are: HEU-COMP-THERM-003 (15 cases), HEU-COMP-THERM-005 (1 case), HEU-COMP-THERM-006 (three cases), HEU-COMP-THERM-007 (three cases), HEU-MET-THERM-006 (23 Cases), In addition the following were previously evaluated in CRWMS M&O (1999a): HEU-COMP-THERM-008 (two Cases), HEU-COMP-THERM-010 (21 Cases), and HEU-COMP-INTER-002 (five Cases).

Brief descriptions for the above are provided in Section 5.1.1. Through a detailed search of the currently evaluated experiments documented in NEA (2001), the following additional experiments were added in order to demonstrate an adequate range of applicability considering materials and moderation variations:

| | |
|---|---|
| HEU-MET-FAST-007 – Section 5.2.1.1.1 | 43 Cases |
| HEU-MET-FAST-033 – Section 5.2.1.1.3 | 2 Cases |
| HEU-MET-FAST-034 – Section 5.2.1.1.4 | 3 Cases |
| HEU-MET-INTER-006 – Section 5.2.1.1.8 | 2 Cases |
| HEU-MET-MIXED-001 – Section 5.2.1.1.9 | 1 Case |
| HEU-MET-FAST-036 – Section 5.2.1.1.5 | 2 Cases |
| HEU-MET-FAST-037 – Section 5.2.1.1.6 | 2 Cases |
| HEU-MET-MIXED-004 – Section 5.2.1.1.10 | 1 Case |
| HEU-MET-FAST-030 – Section 5.2.1.1.2 | 1 Case |
| HEU-MET-FAST-038 – Section 5.2.1.1.7 | 1 Case |
| HEU-MET-MIXED-005 – Section 5.2.2.3.1 | 5 Cases |
| HEU-COMP-INTER-001 – Section 5.2.1.3.1 | 1 Case |
| HEU-COMP-THERM-011 – Section 5.2.1.3.3 | 3 Cases |
| HEU-COMP-THERM-012 – Section 5.2.1.3.3 | 2 Cases |
| HEU-COMP-THERM-013 – Section 5.2.1.3.3 | 2 Cases |
| HEU-COMP-THERM-014 – Section 5.2.1.3.3 | 2 Cases |

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-1 (thermal spectrum) and Table 6-2 (intermediate and fast spectrum). The spectrum categorizations are done according to the classification given in page 3 of the introduction of NEA (2001).

#### 6.1.1.2 Benchmarks Related to Degraded Waste Package Configurations

Previously evaluated critical experiments with highly enriched uranium nitrate solution (approximately 90%) which were used to calculate the interim critical limit (Section 2.4.1 of CRWMS [M&O 2000c]) are described in Section 5.1.2: HEU-SOL-THERM-001 (10 cases), HEU-SOL-THERM-002 (14 cases), HEU-SOL-THERM-003 (19 cases). The concentration of fissile element in the solution, enrichment, reflector type and thickness, tank diameter, and solution height were among the parameters that were varied in these experiments. Through a detailed search of the currently evaluated solution experiments documented in NEA (2001), the following additional experiments were added in order to demonstrate an adequate range of applicability considering materials and moderation variations:

HEU-SOL-THERM-012 – Section 5.2.2.1.6 1 Case
HEU-SOL-THERM-032 – Section 5.2.2.1.14 1 Case
HEU-SOL-THERM-005 – Section 5.2.2.1.1 17 Cases
HEU-SOL-THERM-006 – Section 5.2.2.1.2 29 Cases
HEU-SOL-THERM-009 – Section 5.2.2.1.3 4 Cases
HEU-SOL-THERM-010 – Section 5.2.2.1.4 4 Cases
HEU-SOL-THERM-011 – Section 5.2.2.1.5 2 Cases
HEU-SOL-THERM-027 – Section 5.2.2.1.9 9 Cases
HEU-SOL-THERM-028 – Section 5.2.2.1.10 18 Cases
HEU-SOL-THERM-029 – Section 5.2.2.1.11 7 Cases
HEU-SOL-THERM-030 – Section 5.2.2.1.12 7 Cases
HEU-SOL-THERM-031 – Section 5.2.2.1.13 4 Cases
HEU-SOL-THERM-035 – Section 5.2.2.1.16 9 Cases
HEU-SOL-THERM-036 – Section 5.2.2.1.17 4 Cases
HEU-SOL-THERM-037 – Section 5.2.2.1.18 9 Cases
HEU-SOL-THERM-033 – Section 5.2.2.1.15 26 Cases
HEU-SOL-THERM-043 – Section 5.2.2.1.19 3 Cases

The HEU fast metal cases listed in Section 5.1.2.2 need to be included in the list of applicable benchmarks. These were HEU-MET-FAST-008 (one case), 011 (one case), 012 (one case), 013 (one case), 014 (one case), 015 (one case), 018 (one case), 019 (one case), 020 (one case), 021 (one case), 022 (one case), 024 (one case). The following additional experiments will be added in order to demonstrate an adequate range of applicability considering materials and moderation variations:

HEU-MET-FAST-002 – Section 5.2.2.2.1 6 Cases
HEU-MET-FAST-004 – Section 0 2 Cases

| | |
|---|---|
| HEU-MET-FAST-027 – Section 5.2.2.2.3 | 1 Case |
| HEU-MET-FAST-029 – Section 5.2.2.2.4 | 1 Case |
| HEU-MET-FAST-031 – Section 5.2.2.2.5 | 1 Case |
| HEU-MET-MIXED-002 – Section 5.2.2.2.6 | 1 Case |
| HEU-MET-MIXED-003 – Section 5.2.2.2.7 | 1 Case |
| HEU-MET-FAST-032 – Section 5.2.2.2.8 | 4 Cases |
| HEU-MET-FAST-041 – Section 5.2.2.2.9 | 6 Cases |
| HEU-MET-FAST-055 – Section 5.2.2.2.10 | 1 Case |
| HEU-MET-THERM-003 – Section 5.2.2.2.11 | 7 Cases |

The Si-containing experiment listed below is also important for HEU fuels.

| | |
|---|---|
| HEU-MET-MIXED-005 – Section 5.2.2.3.1 | 5 Cases |

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-3 (thermal spectrum) and Table 6-4 (intermediate and fast spectrum).

### 6.1.2 UZrH (Training, Research, Isotopes, General Atomics [TRIGA]) DOE-Owned Fuel

#### 6.1.2.1 Benchmarks Related to Intact Waste Package Configurations

The TRIGA type fuel is unique in that it incorporates a moderator in the matrix as zirconium hydride and is in the form of individual fuel rods. Two sets of experiments were previously identified in Section 2.4.1 of CRWMS M&O (2000a) as relevant for calculation of an interim critical limit for the TRIGA SNF (70% enriched) with respect to intact criticality analyses and previously evaluated in CRWMS M&O (1999a, page 77): the TRIGA SNF Rod and a series of critical experiments with water moderated hexagonal pitched lattices with highly enriched fuel rods (HEU-COMP-THERM-003 (15 cases), HEU-COMP-THERM-004 (four cases), HEU-COMP-THERM-005 (one case), HEU-COMP-THERM-006 (three cases), HEU-COMP-THERM-007 (three cases), HEU-COMP-THERM-008 (two cases), HEU-COMP-THERM-010 (21 cases), and HEU-MET-THERM-006 (23 cases). Brief descriptions of these experiments taken from NEA (2001) are provided in Section 5.1.1. The TRIGA SNF Rod benchmarks (Section 5.6.1.1.2 IEU-COMP-THERM-003 [2 cases]) were recently included in NEA (2001), and thus the cases for these experiments were rerun with parameters provided in NEA (2001).

Through a detailed search of the currently evaluated experiments documented in NEA (2001), the following additional experiments were added in order to demonstrate an adequate range of applicability considering materials and moderation variations. Criteria in choosing these experiments (except for IEU-COMP-THERM-003) is that enrichment level is comparable to higher enrichment (70%) TRIGA fuel:

| | |
|---|---|
| HEU-MET-FAST-023 – Section 5.2.1.2.1 | 22 Cases |
| HEU-MET-FAST-026 – Section 5.2.1.2.2 | 32 Cases |
| HEU-COMP-MIXED-001 – Section 5.2.1.2.3 | 26 Cases |

| | |
|---|---|
| HEU-COMP-MIXED-002 – Section 5.2.1.2.4 | 23 Cases |
| HEU-COMP-INTER-001 – Section 5.2.1.3.1 | 1 Case |
| HEU-COMP-INTER-003 – Section 5.2.1.1.11 | 7 Cases |
| HEU-COMP-THERM-002 – Section 5.2.1.3.2 | 25 Cases |
| HEU-COMP-THERM-011 – Section 5.2.1.3.3 | 3 Cases |
| HEU-COMP-THERM-012 – Section 5.2.1.3.3 | 2 Cases |
| HEU-COMP-THERM-013 – Section 5.2.1.3.3 | 2 Cases |
| HEU-COMP-THERM-014 – Section 5.2.1.3.3 | 2 Cases |

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-5 (thermal spectrum) and Table 6-6 (intermediate and fast spectrum).

#### 6.1.2.2 Benchmarks Related to Degraded Waste Package Configurations

Critical experiments with highly enriched uranium nitrate solution (approximately 90%) which were used to calculate the interim critical limit are described in Section 5.1.2.1: HEU-SOL-THERM-001 (10 cases), HEU-SOL-THERM-008 (five cases), HEU-SOL-THERM-013 (four cases), HEU-SOL-THERM-014 (three cases), HEU-SOL-THERM-015 (five cases), HEU-SOL-THERM-016 (three cases), HEU-SOL-THERM-017 (eight cases), HEU-SOL-THERM-018 (12 cases), and HEU-SOL-THERM-019 (three cases). The concentration of fissile element in the solution, enrichment, amount and type of neutron absorber (Gd and Boron [B]), reflector type and thickness, tank diameter, and solution height were among the parameters that were varied. Through a detailed search of the currently evaluated solution experiments documented in NEA (2001), the following additional experiments were added in order to demonstrate an adequate range of applicability considering materials and moderation variations:

| | |
|---|---|
| HEU-SOL-THERM-012 – Section 5.2.2.1.6 | 1 Case |
| HEU-SOL-THERM-032 – Section 5.2.2.1.14 | 1 Case |
| HEU-SOL-THERM-005 – Section 5.2.2.1.1 | 17 Cases |
| HEU-SOL-THERM-006 – Section 5.2.2.1.2 | 29 Cases |
| HEU-SOL-THERM-009 – Section 5.2.2.1.3 | 4 Cases |
| HEU-SOL-THERM-010 – Section 5.2.2.1.4 | 4 Cases |
| HEU-SOL-THERM-011 – Section 5.2.2.1.5 | 2 Cases |
| HEU-SOL-THERM-021 – Section 5.2.2.1.7 | 32 Cases |
| HEU-SOL-THERM-025 – Section 5.2.2.1.8 | 18 Cases |
| HEU-SOL-THERM-027 – Section 5.2.2.1.9 | 9 Cases |
| HEU-SOL-THERM-028 – Section 5.2.2.1.10 | 18 Cases |
| HEU-SOL-THERM-029 – Section 5.2.2.1.11 | 7 Cases |
| HEU-SOL-THERM-030 – Section 5.2.2.1.12 | 7 Cases |
| HEU-SOL-THERM-031 – Section 5.2.2.1.13 | 4 Cases |
| HEU-SOL-THERM-035 – Section 5.2.2.1.16 | 9 Cases |
| HEU-SOL-THERM-036 – Section 5.2.2.1.17 | 4 Cases |
| HEU-SOL-THERM-037 – Section 5.2.2.1.18 | 9 Cases |
| HEU-SOL-THERM-033 – Section 5.2.2.1.15 | 26 Cases |

HEU-SOL-THERM-043 – Section 5.2.2.1.19 3 Cases
HEU-SOL-THERM-044 – Section 5.2.2.1.20 16 Cases

The HEU fast metal cases previously evaluated as listed in Section 5.1.2.2 need to be included in the list of applicable benchmarks. These were HEU-MET-FAST-008 (1 case), 011 (1 case), 012 (1 case), 013 (1 case), 014 (1 case), 015 (1 case), 018 (1 case), 019 (1 case), 020 (1 case), 021 (1 case), 022 (1 case), and 024 (1 case). The following additional experiments will be added in order to demonstrate an adequate range of applicability considering materials and moderation variations:

HEU-MET-FAST-002 – Section 5.2.2.2.1 6 Cases
HEU-MET-FAST-004 – Section 5.2.2.2.2 2 Cases
HEU-MET-FAST-027 – Section 5.2.2.2.3 1 Case
HEU-MET-FAST-029 – Section 5.2.2.2.4 1 Case
HEU-MET-FAST-031 – Section 5.2.2.2.5 1 Case
HEU-MET-MIXED-002 – Section 5.2.2.2.6 1 Case
HEU-MET-MIXED-003 – Section 5.2.2.2.7 1 Case
HEU-MET-FAST-032 – Section 5.2.2.2.8 4 Cases
HEU-MET-FAST-041 – Section 5.2.2.2.9 6 Cases
HEU-MET-FAST-055 – Section 5.2.2.2.10 1 Case
HEU-MET-THERM-003 – Section 5.2.2.2.11 7 Cases

The Si-containing experiment listed below is also important for HEU fuels.

HEU-MET-MIXED-005 – Section 5.2.2.3.1 5 Cases

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-7 (thermal spectrum) and Table 6-8 (intermediate and fast spectrum).

### 6.1.3 Melt and Dilute Waste Form

The Melt and Dilute fuelis in the form of a melt/dilute "Ingot" (with depleted uranium) with enrichment levels below 20%.Very few evaluated benchmarks are available for the intermediate enrichment of the Melt and Dilute waste form (20% enriched). The focus will be on extending the application of the HEU cases into the intermediate range with the use of the IEU cases available and/or interpolating between LEU and HEU cases. Because of the homogeneous nature of the MD waste form, the distinction between intact and degraded geometries and compositions is less clear than for other DOE SNF, and the cases will be used interchangeably.

#### 6.1.3.1 Benchmarks Related to Intact Waste Package Configurations

The experiments that were previously identified in Radulescu (2001a) as relevant for calculation of an interim critical limit for the MD ingots with respect to intact criticality analyses are: IEU-COMP-THERM-001 (29 cases), IEU-MET-FAST-001 (four cases), IEU-MET-FAST-002 (one case), IEU-MET-FAST-003 (one case), IEU-MET-FAST-004 (one case), IEU-MET-FAST-005 (one case), IEU-MET-FAST-006 (one case), and IEU-MET-FAST-008 (one case). Summary

descriptions of the previously evaluated cases are provided in Section 5.5.2. Through a detailed search of the currently evaluated experiments documented in NEA (2001), the following additional experiments were included in order to demonstrate an adequate range of applicability considering materials and moderation variations:

| | |
|---|---|
| IEU-COMP-THERM-002 – Section 5.6.1.1.1 | 6 Cases |
| IEU-COMP-THERM-003 – Section 5.6.1.1.2 | 2 Cases |
| IEU-MET-FAST-007 – Section 5.6.2.1.1 | 1 Case |
| IEU-MET-FAST-009 – Section 5.6.2.1.2 | 1 Case |
| IEU-MET-FAST-010 – Section 5.6.2.1.3 | 1 Case |

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-9 (thermal spectrum) and Table 6-10 (intermediate and fast spectrum).

#### 6.1.3.2 Benchmarks Related to Degraded Waste Package Configurations

The following solution experiments will be used for severely degraded cases. The evaluated cases (Section 5.5.2.2) include experiments IEU-SOL-THERM-001 (four cases), LEU-SOL-THERM-003 (nine cases) and IEU-COMP-THERM-001 (29 cases). The following LEU benchmarks from NEA (2001) are for 10% enrichment, which should be similar in behavior to 20% enrichment.

| | |
|---|---|
| LEU-SOL-THERM-004 – Section 5.6.2.2.1 | 7 Cases |
| LEU-SOL-THERM-007 – Section 5.6.2.2.1 | 5 Cases |
| LEU-SOL-THERM-008 – Section 5.6.2.2.1 | 4 Cases |
| LEU-SOL-THERM-009 – Section 5.6.2.2.1 | 3 Cases |
| LEU-SOL-THERM-010 – Section 5.6.2.2.1 | 4 Cases |
| LEU-SOL-THERM-016 – Section 5.6.2.2.1 | 7 Cases |
| LEU-SOL-THERM-017 – Section 5.6.2.2.1 | 6 Cases |
| LEU-SOL-THERM-006 – Section 5.6.2.2.1 | 5 Cases |
| LEU-SOL-THERM-018 – Section 5.6.2.2.1 | 6 Cases |
| LEU-SOL-THERM-019 – Section 5.6.2.2.1 | 6 Cases |
| LEU-SOL-THERM-020 – Section 5.6.2.2.1 | 4 Cases |
| LEU-SOL-THERM-021 – Section 5.6.2.2.1 | 4 Cases |

The intermediate and fast spectrum criticals used for the intact configuration can be used for the degraded configuration also.

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-11 (thermal spectrum) and Table 6-12 (intermediate and fast spectrum).

### 6.1.4 U Alloy (Enrico Fermi) DOE-Owned Fuel

The Enrico Fermi type fuel is composed of U-Alloy (U-Mo/U-Zr) material that is enriched to 26% $^{235}$U. The fuel pins are in stored in canisters. Very few evaluated benchmarks are available

for the intermediate enrichment of the Enrico Fermi SNF. The focus was on extending the application of the HEU cases into the intermediate range with the use of the IEU cases available and/or interpolating between LEU and HEU cases.

#### 6.1.4.1 Benchmarks Related to Intact Waste Package Configurations

Several critical experiments were previously identified in Section 2.4.1 of CRWMS M&O (2000b) as relevant for calculation of an interim critical limit for the Shippingport-PWR DOE SNF (93% enriched) and can be also utilized here. These were a series of critical experiments with water-moderated hexagonally-pitched lattices of highly enriched fuel rods of cross-shaped cross section. The designations for these experiments in NEA (2001) are HEU-COMP-THERM-003 (15 cases), HEU-COMP-THERM-004 (four cases), HEU-COMP-THERM-005 (one case), HEU-COMP-THERM-006 (three cases), HEU-COMP-THERM-007 (three cases). Another applicable experiment previously evaluated in Section 5.1.2 of CRWMS M&O (1999a) is IEU-COMP-THERM-001 (29 cases). Brief descriptions of these experiments are provided in Section 5.1.1.2 and 5.5.2.2. The IEU fast metal experiments utilized for the melt and dilute waste form are also applicable to U alloy (Enrico Fermi) SNF. Through a detailed search of the currently evaluated experiments documented in NEA (2001), the following additional experiments were added in order to demonstrate an adequate range of applicability considering materials and moderation variations:

IEU-COMP-THERM-002 – Section 5.6.1.1.1 6 Cases
IEU-COMP-THERM-003 – Section 5.6.1.1.2 2 Cases

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-13 (thermal spectrum) and Table 6-14 (intermediate and fast spectrum).

#### 6.1.4.2 Benchmarks Related to Degraded Waste Package Configurations

The following solution experiments will be used for severely degraded cases. The evaluated cases (Section 5.5.2) include experiments IEU-SOL-THERM-001 (four cases), LEU-SOL-THERM-003 (nine cases) and IEU-COMP-THERM-001 (29 cases). The following LEU benchmarks from NEA (2001) are for 10% enrichment, which should be similar in behavior to 26% enrichment.

LEU-SOL-THERM-004 – Section 5.6.2.2.1 7 Cases
LEU-SOL-THERM-007 – Section 5.6.2.2.1 5 Cases
LEU-SOL-THERM-008 – Section 5.6.2.2.1 4 Cases
LEU-SOL-THERM-009 – Section 5.6.2.2.1 3 Cases
LEU-SOL-THERM-010 – Section 5.6.2.2.1 4 Cases
LEU-SOL-THERM-016 – Section 5.6.2.2.1 7 Cases
LEU-SOL-THERM-017 – Section 5.6.2.2.1 6 Cases
LEU-SOL-THERM-006 – Section 5.6.2.2.1 5 Cases
LEU-SOL-THERM-018 – Section 5.6.2.2.1 6 Cases
LEU-SOL-THERM-019 – Section 5.6.2.2.1 6 Cases

| | |
|---|---|
| LEU-SOL-THERM-020 – Section 5.6.2.2.1 | 4 Cases |
| LEU-SOL-THERM-021 – Section 5.6.2.2.1 | 4 Cases |

The intermediate and fast spectrum criticals used for the intact configuration can be used for the degraded configuration also.

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-15 (thermal spectrum) and Table 6-16 (intermediate and fast spectrum).

### 6.1.5 MOX (FFTF) DOE-Owned Fuel

FFTF fuel has been designated as the representative fuel for the mixed-oxide (MOX) fuel group which is a mixture of uranium and plutonium oxides. This fuel has plutonium enrichments ranging from ~22% to ~30%. Part of the fuel is in hexagonal shaped assemblies while some fuel pins are in canisters. PU and MIXED benchmarks are applicable for this fuel.

#### 6.1.5.1 Benchmarks Related to Intact Waste Package Configurations for FFTF

Three sets of experiments were previously identified (Section 2.4.1 of CRWMS M&O (1999d) as relevant for calculation of the interim critical limit for FFTF fuel with respect to intact criticality analyses: the FFTF fuel pin array experiment (MIX-COMP-THERM-001 [four cases], the Saxton plutonium experiment (MIX-COMP-THERM-003 [six cases], a series of critical experiments with water moderated hexagonally pitched lattices with highly enriched fuel rods (HEU-COMP-THERM-003 [15 cases], HEU-COMP-THERM-004 [four cases], HEU-COMP-THERM-005 [one case], HEU-COMP-THERM-006 [three cases], HEU-COMP-THERM-007 [three cases], HEU-COMP-THERM-008 [two cases], and HEU-COMP-THERM-010 [21 cases], . The FFTF and Saxton experiments are described in Section 5.9.1 and the hexagonally pitched lattices are described in Section 5.1.1.2. Through a detailed search of the currently evaluated experiments documented in, the following additional experiments were added in order to demonstrate an adequate range of applicability considering materials and moderation variations. Because $^{239}$Pu- decays to $^{239}$U, HEU benchmarks also need to be considered.

| | |
|---|---|
| MIX-COMP-THERM-004 – Section 5.10.1.1.1 | 11 Cases |
| MIX-COMP-THERM-010 – Section 5.10.1.1.2 | 11 Cases |
| PU-MET-FAST-003 – Section 5.10.1.1.3 | 5 Cases |
| PU-MET-FAST-004 – Section 5.10.1.1.4 | 9 Cases |
| PU-MET-FAST-016 – Section 5.10.1.1.5 | 6 Cases |
| PU-MET-FAST-017 – Section 5.10.1.1.6 | 5 Cases |
| PU-MET-FAST-037 – Section 5.10.1.1.7 | 16 Cases |
| PU-COMP-MIXED-001 – Section 5.10.1.1.8 | 5 Cases |
| PU-COMP-MIXED-002 – Section 5.10.1.1.9 | 29 Cases |

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-17 (thermal spectrum) and Table 6-18.

### 6.1.5.2 Benchmarks Related to Degraded Waste Package Configurations

The critical experiments with mixed nitrate solution which were used to calculate the interim critical limit in Section 2.4.1 of CRWMS M&O (1999d) are described in Section 5.9.2: MIX-SOL-THERM-001 (12 cases), MIX-SOL-THERM-002 (three cases), MIX-SOL-THERM-003 (10 cases), and MIX-SOL-THERM-004 (nine cases). The concentration of fissile elements (Pu and U) in the solution, enrichment, amount of absorber ($B_4C$ concrete and polyethylene with Cd cover), tank diameter, and solution height were among the parameters that were varied. All other nitrate solution experiments previously evaluated and described in Section 5.9.2 should also be used: PU-SOL-THERM-001 (6 cases), 003 (8 cases), 004 (13 cases), 005 (nine cases), 007 (eight cases), 009 (three cases), 010 (14 cases), and 011 (12 cases). Through a detailed search of the currently evaluated solution experiments documented in NEA (2001), the following additional experiments were added in order to demonstrate an adequate range of applicability considering materials and moderation variations:

| | |
|---|---|
| MIX-SOL-THERM-005 – Section 5.10.2.1.1 | 7 Cases |
| PU-SOL-THERM-002 – Section 5.10.2.1.2 | 7 Cases |
| PU-SOL-THERM-006 – Section 5.10.2.1.3 | 3 Cases |
| PU-SOL-THERM-008 – Section 5.10.2.1.4 | 14 Cases |
| PU-SOL-THERM-012 – Section 5.10.2.1.5 | 23 Cases |
| PU-SOL-THERM-013 – Section 5.10.2.1.6 | 21 Cases |
| PU-SOL-THERM-014 – Section 5.10.2.1.7 | 35 Cases |
| PU-SOL-THERM-015 – Section 5.10.2.1.8 | 17 Cases |
| PU-SOL-THERM-016 – Section 5.10.2.1.9 | 11 Cases |
| PU-SOL-THERM-017 – Section 5.10.2.1.10 | 18 Cases |
| PU-SOL-THERM-020 – Section 5.10.2.1.11 | 8 Cases |
| PU-SOL-THERM-021 – Section 5.10.2.1.12 | 6 Cases |
| PU-SOL-THERM-022 – Section 5.10.2.1 | 17 Cases |
| PU-SOL-THERM-023 – Section 5.10.2.1 | 3 Cases |
| PU-SOL-THERM-024 – Section 5.10.2.1 | 23 Cases |
| PU-SOL-THERM-025 – Section 5.10.2.1.13 | 44 Cases |
| PU-SOL-THERM-026 – Section 5.10.2.1.14 | 21 Cases |

The PU and Mixed fast metal cases evaluated in Section 5.9.1 need to be included in the list of applicable benchmarks. These were PU-MET-FAST-020 (one case), 022 (one case), 023 (one case), 024 (one case), 025 (one case), 026 (one case), 027 (one case), 028 (one case), 029 (one case), 030 (one case), 031 (one case), and 032 (one case). The following additional experiments were added in order to demonstrate an adequate range of applicability considering materials and moderation variations:

| | |
|---|---|
| MIX-MET-MIXED-001 – Section 5.10.2.2.1 | 1 Case |
| MIX-MET-FAST-001 – Section 5.10.2.2.2 | 1 Case |
| MIX-MET-FAST-002 – Section 5.10.2.2.2 | 3 Cases |
| MIX-MET-FAST-003 – Section 5.10.2.2.3 | 1 Case |

| | |
|---|---|
| MIX-MET-FAST-004 – Section 5.10.2.2.3 | 2 Cases |
| MIX- MET-FAST-005 – Section 5.10.2.2.3 | 1 Case |
| MIX-MET-FAST-009 – Section 5.10.2.2.3 | 1 Case |
| MIX-MET-FAST-010 – Section 5.10.2.2.3 | 1 Case |
| MIX-MET-FAST-011 – Section 5.10.1.1.10 | 4 Cases |
| PU-MET-FAST-001 – Section 5.10.2.2.4 | 1 Case |
| PU-MET-FAST-002 – Section 5.10.2.2.4 | 1 Case |
| PU-MET-FAST-005 – Section 5.10.2.2.4 | 1 Case |
| PU-MET-FAST-006 – Section 5.10.2.2.4 | 1 Case |
| PU-MET-FAST-009 – Section 5.10.2.2.4 | 1 Case |
| PU-MET-FAST-010 – Section 5.10.2.2.4 | 1 Case |
| PU-MET-FAST-011 – Section 5.10.2.2.4 | 1 Case |
| PU-MET-FAST-018 – Section 5.10.2.2.4 | 1 Case |
| PU-MET-FAST-019 – Section 5.10.2.2.5 | 1 Case |
| PU-MET-FAST-021 – Section 5.10.2.2.5 | 2 Cases |
| PU-MET-FAST-035 – Section 5.10.2.2.5 | 1 Case |
| PU-MET-FAST-036 – Section 5.10.2.2.5 | 1 Case |
| PU-MET-FAST-039 – Section 5.10.2.2.5 | 1 Case |
| PU-MET-FAST-040 – Section 5.10.2.2.5 | 1 Case |
| PU-MET-FAST-041 – Section 5.10.2.2.5 | 1 Case |

The Si-containing experiment listed below is also important for MIXED and PU fuels.

| | |
|---|---|
| PU-MET-MIXED-001 – Section 5.10.2.3.1 | 5 Cases |

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-19 (thermal spectrum) and Table 6-20 (intermediate and fast spectrum).

### 6.1.6 Graphite/Carbide (Fort Saint Vrain HTGR) DOE-Owned Fuel

Fort Saint Vrain SNF has been designated as the representative fuel for the DOE-owned graphite/carbide fuel group. This fuel is a combination of UC2 (~93% $^{233}$U enriched) and ThC2 fuel particles embedded in a graphite matrix and is in the form of assemblies.. Relatively few benchmark experiments are available for the unique mixture of fissile isotopes ($^{233}$U and $^{235}$U) that characterizes this fuel. The focus was on extending the application of the existing HEU benchmark cases.

#### 6.1.6.1 Benchmarks Related to Intact Waste Package Configurations for Fort Saint Vrain SNF

The following critical experiments documented in NEA (2001) were included in order to demonstrate an adequate range of applicability considering materials and moderation variations:

| | |
|---|---|
| HEU-COMP-THERM-002 – Section 5.2.1.3.2 | 25 Cases |
| HEU-COMP-MIXED-001 – Section 5.2.1.2.3 | 26 Cases |
| HEU-MET-INTER-006 – Section 5.2.1.1.8 | 2 Cases |
| U233-MET-FAST-001 – Section 5.7.1.1 | 1 Case |
| U233-MET-FAST-002 – Section 5.7.1.1 | 2 Cases |
| U233-MET-FAST-003 – Section 5.7.1.1 | 2 Cases |
| U233-MET-FAST-004 – Section 5.7.1.1 | 2 Cases |
| U233-MET-FAST-005 – Section 5.7.1.1 | 2 Cases |
| U233-MET-FAST-006 – Section 5.7.1.1 | 1 Case |
| HEU-MET-FAST-007 – Section 5.2.1.1.1 | 43 Cases |
| HEU-MET-FAST-033 – Section 5.2.1.1.3 | 2 Cases |
| HEU-MET-FAST-034 – Section 5.2.1.1.4 | 3 Cases |
| HEU-MET-FAST-036 – Section 5.2.1.1.5 | 2 Cases |
| HEU-MET-MIXED-001 – Section 5.2.1.1.9 | 1 Case |
| HEU-MET-MIXED-004 – Section 5.2.1.1.10 | 1 Case |
| HEU-MET-MIXED-005 – Section 5.2.2.3.1 | 5 Cases |
| U233-SOL-THERM-006 – Section 5.2.2.3.1 | 12 Cases |

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-21 (thermal spectrum) and Table 6-22 (intermediate and fast spectrum).

### 6.1.6.2 Benchmarks Related to Degraded Waste Package Configurations

The following critical experiments with highly enriched uranium ($^{233}U$) nitrate solution (approximately 97 wt.%) documented in NEA (2001) were included:

| | |
|---|---|
| U233-SOL-THERM-001 – Section 5.8.2.1.1 | 5 Cases |
| U233-SOL-THERM-002 – Section 5.8.2.1.2 | 17 Cases |
| U233-SOL-THERM-003 – Section 5.8.2.1.3 | 10 Cases |
| U233-SOL-THERM-004 – Section 5.8.2.1.4 | 8 Cases |
| U233-SOL-THERM-005 – Section 5.8.2.1.5 | 2 Cases |
| U233-SOL-THERM-006 – Section 5.8.2.2.1 1 | 12 Cases |
| U233-SOL-THERM-008 – Section 5.8.2.1.6 | 1 Case |
| U233-SOL-INTER-001 – Section 5.8.2.1.7 | 33 Cases |

The fast metal experiments mentioned in Section 6.1.6.1 are also applicable for the degraded configurations. The following HEU critical experiments were added in order to demonstrate an adequate range of applicability considering materials and moderation variations:

| | |
|---|---|
| HEU-COMP-THERM-002 – Section 5.2.1.3.2 | 25 Cases |
| HEU-COMP-MIXED-001 – Section 5.2.1.2.3 | 26 Cases |
| HEU-MET-INTER-006 – Section 5.2.1.1.8 | 2 Cases |
| HEU-MET-MIXED-002 – Section 5.2.2.2.6 | 1 Case |
| HEU-MET-MIXED-003 – Section 5.2.2.2.7 | 1 Case |

| | |
|---|---|
| HEU-MET-FAST-011 – Section 5.1.2.2 | 1 Case |
| HEU-MET-FAST-019 – Section 5.1.2.2 | 1 Case |
| HEU-MET-FAST-020 – Section 5.1.2.2 | 1 Case |
| HEU-MET-FAST-024 – Section 5.1.2.2 | 1 Case |
| HEU-MET-FAST-031 – Section 5.2.2.2.5 | 1 Case |
| HEU-MET-FAST-041 – Section 5.2.2.2.9 | 6 Cases |

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-23 (thermal spectrum) and Table 6-24 (intermediate and fast spectrum).

### 6.1.7 Th/U (Shippingport) LWBR DOE-Owned SNF

Shippingport LWBR SNF has been designated as the representative fuel for the Th/U Oxide DOE-owned SNF group. This fuel is composed of high $^{233}$Uenrichments (~98%) mixed in binary fuel pellets of UO2 (~5.2 wt% U) with a thoria (ThO2) matrix and Zircalloy-4 cladding. The fuel is in the form of hexagonal assemblies (CRWMS M&O [2000d]). The application of the existing HEU benchmark cases and $^{233}$U experiments has been extended as surrogate critical experiments for the Shippingport LWBR SNF intact and degraded configurations. Critical experiments that used Shippingport LWBR rods are being evaluated by the NSNFP and will be used to calculate and validate the final critical limit for licensing.

#### 6.1.7.1 Benchmarks Related to Intact Waste Package Configurations for Shippingport LWBR SNF

Several surrogate experiments with highly enriched fuel rods were previously used in CRWMS M&O (2000d) for establishing an interim criticality limit: HEU-COMP-THERM-003 (15 cases), HEU-COMP-THERM-005 (1 case), HEU-COMP-THERM-006 (3 cases), and HEU-COMP-THERM-007 (3 cases) and HEU-COMP-INTER-002 (5 cases). The experiments are detailed in Section 5.1.1. The following additional experiments were added in order to demonstrate an adequate range of applicability considering materials and moderation variations.

| | |
|---|---|
| HEU-COMP-THERM-011– Section 5.2.1.3.3 | 3 Cases |
| HEU-COMP-THERM-013 – Section 5.2.1.3.3 | 2 Cases |
| HEU-COMP-MIXED-001 – Section 5.2.1.2.3 | 26 Cases |
| U233-MET-FAST-001 – Section 5.7.1.1 | 1 Case |
| U233-MET-FAST-002 – Section 5.7.1.1 | 2 Cases |
| U233-MET-FAST-003 – Section 5.7.1.1 | 2 Cases |
| U233-MET-FAST-004 – Section 5.7.1.1 | 2 Cases |
| U233-MET-FAST-005 – Section 5.7.1.1 | 2 Cases |
| U233-MET-FAST-006 – Section 5.7.1.1 | 1 Case |
| U233-SOL-THERM-006 – Section 5.8.2.2.1 | 12 Cases |
| HEU-MET-FAST-007 – Section 5.2.1.1.1 | 43 Cases |
| HEU-MET-FAST-033 – Section 5.2.1.1.3 | 2 Cases |
| HEU-MET-FAST-034 – Section 5.2.1.1.4 | 3 Cases |
| HEU-MET-FAST-036 – Section 5.2.1.1.3 | 2 Cases |

| | |
|---|---|
| HEU-MET-MIXED-001 – Section 5.2.1.1.9 | 1 Case |
| HEU-MET-MIXED-004 – Section 5.2.1.1.10 | 1 Case |
| HEU-MET-MIXED-005 – Section 5.2.2.3.1 | 5 Cases |
| HEU-COMP-THERM-012 – Section 5.2.1.3.3 | 2 Cases |

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-25 (thermal spectrum) and Table 6-26 (intermediate and fast spectrum).

#### 6.1.7.2 Benchmarks Related to Degraded Waste Package Configurations

The following critical experiments with highly enriched uranium (U233) nitrate solution (approximately 97%) documented in NEA (2001) were included in order to demonstrate an adequate range of applicability considering materials and moderation variations:

| | |
|---|---|
| U233-SOL-THERM-001 – Section 5.8.2.1.1 | 5 Cases |
| U233-SOL-THERM-002 – Section 5.8.2.1.2 | 17 Cases |
| U233-SOL-THERM-003 – Section 5.8.2.1.3 | 10 Cases |
| U233-SOL-THERM-004 – Section 5.8.2.1.4 | 8 Cases |
| U233-SOL-THERM-005 – Section 5.8.2.1.5 | 2 Cases |
| U233-SOL-THERM-006 – Section 5.8.2.2.1 | 12 Cases |
| U233-SOL-THERM-008 – Section 5.8.2.1.6 | 1 Case |
| U233-SOL-INTER-001 – Section 5.8.2.1.7 | 33 Cases |

The following fast and intermediate spectrum critical experiments with HEU and U233 were added in order to demonstrate an adequate range of applicability:

| | |
|---|---|
| U233-MET-FAST-001 – Section 5.7.1.1 | 1 Case |
| U233-MET-FAST-002 – Section 5.7.1.1 | 2 Cases |
| U233-MET-FAST-003 – Section 5.7.1.1 | 2 Cases |
| U233-MET-FAST-004 – Section 5.7.1.1 | 2 Cases |
| U233-MET-FAST-005 – Section 5.7.1.1 | 2 Cases |
| U233-MET-FAST-006 – Section 5.7.1.1 | 1 Case |
| HEU-MET-FAST-007 – Section 5.2.1.1.1 | 43 Cases |
| HEU-MET-FAST-011 – Section 5.1.2.2 | 1 Case |
| HEU-MET-FAST-019 – Section 5.1.2.2 | 1 Case |
| HEU-MET-FAST-020 – Section 5.1.2.2 | 1 Case |
| HEU-MET-FAST-024 – Section 5.1.2.2 | 1 Case |
| HEU-MET-FAST-031 – Section 5.2.2.2.5 | 1 Case |
| HEU-MET-FAST-033 – Section 5.2.1.1.3 | 2 Cases |
| HEU-MET-FAST-034 – Section 5.2.1.1.4 | 3 Cases |
| HEU-MET-FAST-036 – Section 5.2.1.1.3 | 2 Cases |
| HEU-MET-FAST-041 – Section 5.2.2.2.9 | 6 Cases |
| HEU-COMP-MIXED-001 – Section 5.2.1.2.3 | 26 Cases |
| HEU-MET-MIXED-001 – Section 5.2.1.1.9 | 1 Case |
| HEU-MET-MIXED-002 – Section 5.2.2.2.6 | 1 Case |

| | |
|---|---|
| HEU-MET-MIXED-003 – Section 5.2.2.2.7 | 1 Case |
| HEU-MET-MIXED-004 – Section 5.2.1.1.10 | 1 Case |
| HEU-MET-MIXED-005 – Section 5.2.2.3.1 | 5 Cases |
| PU-MET-FAST-008 – Section 5.10.2.2.4 | 1 Case (thorium reflected) |

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-27 (thermal spectrum) and Table 6-28 (intermediate and fast spectrum).

### 6.1.8 U-metal (N Reactor) DOE-Owned Fuel

N reactor SNF has been designated as the representative fuel for the DOE-owned U-metal fuel group. This U-metal fuel is very low enriched (~1% $^{235}$U) and had a graphite moderator. The fuel is in the form of of two concentric tubes of uranium metal co-extruded with Zircalloy-2 cladding (CRWMS M&O [2001]).

#### 6.1.8.1 Benchmarks Related to Intact Waste Package Configurations for N Reactor SNF

Several previously evaluated critical experiments with low enriched fuel elements are relevant for the N reactor fuel with respect to intact criticality analyses. Three cases presented in Section 5.3.1 describe an experiment performed with actual N reactor Mark IA fuel elements for three different lattice pitches. Other evaluated relevant experiments include LEU-COMP-THERM-001 (8 cases), LEU-COMP-THERM-016 (four cases) the experiment with 2.35 wt% 235U fuel rods clad in Al alloy with fixed neutron absorber plates (four cases) and the experiment with 4.31 wt% $^{235}$U subcritical clusters and reflecting walls (three cases) which are presented in Section 5.3.1.3. The unevaluated critical experiment LEU-COMP-THERM-042 (7 cases) in Section 5.4.1.1.2 was also added to the pool of experiments that are relevant for the intact configurations. The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-29.

#### 6.1.8.2 Benchmarks Related to Degraded Waste Package Configurations for N Reactor SNF

The evaluated experiment that looked at UO3-H2O solutions (12 cases) is described in Section 5.3.2. The cases differed by varying enrichment and hydrogen to fissile ratio. Two other low enriched solution experiments (LEU-SOL-THERM-001 [one case] and LEU-SOL-THERM-002 [three cases] have been previously evaluated and are presented in Section 5.3.2.2 and 5.3.2.3. The following additional experiments were added in order to demonstrate an adequate range of applicability considering materials and moderation variations.

| | |
|---|---|
| LEU-SOL-THERM-005 – Section 5.6.2.2.2 | 3 Cases |
| LEU-COMP-THERM-049 – Section 5.4.2.1.1 | 18 Cases |

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-30.

### 6.1.9 LEU Oxide (Three Mile Island [TMI]) Core Debris Waste Form

Part of the critical experiments with low enriched fuel elements evaluated for the commercial SNF (CRWMS M&O [1999]) are relevant for the TMI core debris waste package intact and degraded configurations. This low enriched (~5% $^{235}U$) UO2 fuel type consists of the intact and damaged fuel assemblies from the TMI-2 accident.

#### 6.1.9.1 Benchmarks Related to Intact Waste Package Configurations for TMI Core Debris Waste Form

The relevant experiments include LEU-COMP-THERM-001 (8 cases), LEU-COMP-THERM-027, the experiment with 2.35 wt% $^{235}U$ fuel rods clad in Al alloy with fixed neutron absorber plates (four cases), the 4.31 wt% $^{235}U$ subcritical clusters and reflecting walls experiments (three cases), the 4.31 wt% $^{235}U$ rods in highly borated water lattices critical experiments (four cases), and the critical experiments with urania-gadolinia (20 cases), which are described in Section 5.3.1. The unevaluated critical experiments LEU-COMP-THERM-042 (five cases) and LEU-COMP-THERM-041 (seven Cases) were also added to the pool of experiments that are appropriate for the intact configurations. The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-31.

Critical experiments with low enriched solutions that are relevant for the degraded configurations are the experiments with $UO_3$-$H_2O$ solutions (12 cases) which are described in Section 5.3.2. Two other low enriched solution experiments (LEU-SOL-THERM-001 [one case] and LEU-SOL-THERM-002 [three cases]) are presented in Section 5.3.2.2 and 5.3.2.3. The following additional experiments were added in order to demonstrate an adequate range of applicability considering materials and moderation variations.

| | |
|---|---|
| LEU-SOL-THERM-004 – Section 5.6.2.2.1 | 7 Cases |
| LEU-SOL-THERM-005 – Section 5.6.2.2.2 | 3 Cases |
| LEU-SOL-THERM-007 – Section 5.6.2.2.1 | 5 Cases |
| LEU-COMP-THERM-049 – Section 5.4.2.1.1 | 18 Cases |

The area and range of applicability of the above-mentioned critical experiments is summarized in Table 6-32.

## 6.2 RANGE OF APPLICABILITY OF BENCHMARK CASES

The tables in this section provide range of applicability for the benchmark cases as assigned in Section 6.1. The range of applicability consists of key physical parameters that define a particular fissile configuration. The physical parameters include material properties, geometry properties, and average energy of a neutron causing fission. An area or range of applicability is defined for the experiments used in bias determination. The relevant parameters for criticality experiments applicable for each DOE SNF type, based on the grouping presented in Section 6.1, are listed in the following tables. Information about the fissionable element, enrichment, physical form, neutron absorber (if present), non fissionable materials in the fuel region, moderator (if applicable), geometry of the fissionable material, reflector and AENCF range are provided.

Table 6-1 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport PWR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-THERM-006 (23 cases) | U | 93.17 wt% $^{235}U$ | U-Al alloy (fuel plates) | B (cases 20-23) | Al cladding | Water; dilute aqueous solution of uranyl nitrate (cases 19-23) H/X range (fuel region) =134 - 500 (calc) | Rectangular arrays of fuel elements with various spacing. Each fuel element is composed of up to a maximum of 22 fuel plates | Assembly reflected by water or dilute aqueous uranyl nitrate solutions | 0.0097-0.0147 |
| Experiment HEU-COMP-THERM-003 (15 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding Al-lattice plates | Water H/X range(fuel region) = 51-349 | Cylindrical two-zones hexagonally pitched lattices (active fuel length =69.2 cm) – varied pitches | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0139-0.0467 |
| Experiment HEU-COMP-THERM-005 (1 case) (detailed benchmark model evaluated) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding Al-lattice plates | Water H/X (fuel region) = 23 | Hexagonally pitched array of fuel rod clusters (each containing a hexagonally pitched lattice of fuel rods-hex. Lattice pitch = 0.52 cm) | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0764 |
| Experiment HEU-COMP-THERM-006 (3 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding Al-lattice plates | Water H/X range (fuel region) = 30 - 716 | Cylindrical hexagonally pitched lattice (pitches of 0.56, 1 and 2.113 cm) | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0104-0.0720 |
| Experiment HEU-COMP-THERM-007 (3 cases) (detailed benchmark models evaluated) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | ZrH rods, SS-cladding Al-lattice plates | Water +ZrH rods H/X range (fuel region) = 60 - 91 | Cylindrical hexagonally double lattice of fuel rods and ZrH rods. Fixed pitch for fuel rods (0.61 cm) and variable for ZrH rods | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0339-0.0475 |

Table 6-1 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport PWR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-COMP-THERM-008 (2 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | B | $B_4C$ rods, SS-cladding<br>Al-lattice plates | Water<br>H/X = 25 | Cylindrical hexagonally double lattice of fuel rods and $B_4C$ rods. Fixed pitch for fuel rods (0.53 cm) and variable for $B_4C$ rods (2.12 and 2.65 cm) | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0882-0.0922 |
| Experiment HEU-COMP-THERM-010 (21cases) | U | 62.4 wt% $^{235}U$ | $UO_2$+BeO | B as boric acid (few cases) | Hastelloy X-280 cladding | Water (15 cases); water +uranyl nitrate and/or boric acid (few cases)<br>H/X range =36 to 302 | Square or cylindrical assemblies with square/hexagona pitched lattices (lattice pitch between 1.24 to 3 cm; active fuel length=193 cm) | Some of the configurations are fully reflected by water; some are partially flooded and are not reflected at top | 0.0234-0.0797 |
| Experiment HEU-COMP-THERM-011 (3 cases) | U | 80 wt% $^{235}U$ | $UO_2$ +Al alloy (sintered) | None | SS cladding, Al lattice plate | Water; partially flooded cores<br>H/X=170 | Four square clusters of cylindrical fuel rods (21x21, pitch =1.4 cm) arranged in a square geometry (dist between clusters = 0 to 4.8 cm) | Assembly reflected by water (except top of assembly - partially flooded) | 0.0470-0.0533 |
| Experiment HEU-COMP-THERM-012 (2 cases) | U | 80 wt% $^{235}U$ | $UO_2$ +Al alloy (sintered) | None | SS cladding, Al lattice plate | Water; partially flooded cores<br>H/X=35 | Four square clusters of cylindrical fuel rods (18x18, pitch =1.4 cm) arranged in a square geometry (dist between clusters =1.4 and 2.8 cm) | Assembly reflected by water (except top of assembly - partially flooded) | 0.0509-0.0551 |
| Experiment HEU-COMP-THERM-013 (2 cases) | U | 80 wt% $^{235}U$ | $UO_2$ +Al alloy (sintered) | None | SS cladding, Al lattice plate | Water; partially flooded cores<br>H/X=40 | Nine square clusters of cylindrical fuel rods (14x14, pitch =1.4 cm)) arranged in a square geometry (dist between clusters =1.4 and 2.8 cm) | Assembly reflected by water (except top of assembly - partially flooded) | 0.0432-0.0481 |

Table 6-1 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport PWR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-COMP-THERM-014 (2 cases) | U | 79.4 at% $^{235}U$ | $UO_2$ +Al alloy (sintered) | None | SS cladding, Al lattice plate | Water; partially flooded cores<br>H/X=170 | Nine square clusters of cylindrical fuel rods (10x10, pitch =1.98 cm) arranged in a square geometry (dist between clusters is 0 and 1.98 cm) | Assembly reflected by water (except top of assembly - partially flooded) | 0.0227-0.0259 |

Table 6-2 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport PWR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-COMP-INTER-002 (also HEU-COMP-MIXED-004) (5 cases) | U | 96 wt% $^{235}U$ | $UO_2$ | None | Al foil cladding | ZrH + water in various reactor cavities<br>H/X = 14 (calc) | Cylindrical array of fuel rods with BeO axial reflector. | Assembly is radially reflected by sand and water. Axially is reflected by BeO | 0.2377-0.2439 |
| Experiment HEU-MET-FAST-007 (43 cases) | U | 93.15 wt% $^{235}U$ | U metal slabs | None | SS stringers | Polyethylene, Plexiglas or teflon plates<br>H/X range = 0 to 5 (exp) | Multiple layers of rectangular plates of uranium and moderator (base dimension 25.4 cm x 25.4 cm or 12.7cm x 25.4 cm) | Unreflected; some configurations are reflected by polyethylene | 0.3145-1.5855 |
| Experiment HEU-MET-FAST-030 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | SS diaphragm | Be discs | Cylindrical assembly of alternating U and Be discs (19.99 cm diameter) | Reflected by depleted U and Be (end reflector) | 1.1290 |
| Experiment HEU-MET-FAST-033 (2 cases) | U | 96 wt% $^{235}U$ | U metal discs | None | SS discs | Polyethylene discs | Cylindrical assembly of alternating U, Polyethylene, and SS discs (19.99 cm diameter) | Unreflected | 0.7170-0.8500 |

Table 6-2 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport PWR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-034 (3 cases) | U | 96 wt% $^{235}U$ | U metal discs | None | Ti, Al or Steel discs | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and metal discs (19.99 cm diameter) | Unreflected | 0.9530-0.9580 |
| Experiment HEU-MET-FAST-036 (2 cases) | U | 96 wt% $^{235}U$ | U metal discs (both HEU and Depleted Uranium | None | SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and depleted uranium discs (19.99 cm diameter) | Unreflected | 0.8660-1.0970 |
| Experiment HEU-MET-FAST-037 (2 cases) | U | 96 wt% $^{235}U$ | U metal discs (both HEU and Depleted Uranium | None | SS diaphragm; Polyethylene | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and depleted uranium discs (19.99 cm diameter) | Reflected by polyethylene (thickness 10cm) or Cd + polyethylene | 0.8960-1.0200 |
| Experiment HEU-MET-FAST-038 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | SS diaphragm | Be and BeO discs | Cylindrical assembly of alternating U, Be, and BeO discs (19.99 cm diameter) | Reflected by depleted uranium | 1.1390 |
| HEU-MET-INTER-006 (2 cases) | U | 93.2 wt% $^{235}U$ | U metal discs | None | SS diaphragm, Al | C discs $C/^{235}U$ =52 | Cylindrical assembly of alternating U and C discs (53.34 cm diameter) | Reflected by copper | 0.3423-0.3864 |
| HEU-MET-MIXED-001 (1 case) | U | 96 wt% $^{235}U$ (Average) | U metal discs | None | Ti discs, SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and Ti discs (19.99 cm diameter) | Reflected by polyethylene (thickness 10 cm) | 0.7620 |
| HEU-MET-MIXED-004 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, Depleted U, and polyethylene discs (19.99 cm diameter) | Reflected by depleted uranium (metal) | 0.9140 |

Table 6-2 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport PWR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| HEU-COMP-INTER-001 (HEU-COMP-MIXED-003) (6 cases – only case 5 was evaluated) | U | 96 wt% $^{235}U$ | $UO_2$ | None | Al foil cladding | ZrH + water in various reactor cavities | Cylindrical array of fuel rods in a ZrH block with BeO axial and radial reflector. | Reflected by BeO and water | 0.2200 |
| HEU-MET-MIXED-005 (5 cases) | U | 90 wt% $^{235}U$ | U metal pellets | B4C present in some rods | SiO2, Al tubes | Polyethylene, sand ($SiO_2$) | Hexagonally pitched lattice of tubes containing fuel and/or moderator | $SiO_2$, depleted uranium | 0.2120-0.3770 |

Table 6-3 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport PWR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-SOL-THERM-001 (10 cases) | U | 93.17 wt% $^{235}U$ | Aqueous solution of Uranyl nitrate | None | Stainless Steel, Al | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =68 - 499 (calc) | Single homogeneous unit contained in a cylindrical tank made of SS-304 or Al-6061(diameter between 27.9 to 50.7 cm) | Unreflected (minimally reflected by concrete walls) | 0.0065-0.0410 |
| Experiment HEU-SOL-THERM-002 (14 cases) | U | 93.17 wt% $^{235}U$ | Aqueous solution of Uranyl nitrate | None | Stainless Steel, Al | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =74-460 (calc) | Single homogeneous unit contained in a cylindrical tank of SS-304 or Al-6061(diameter between 27.9 to 33 cm) | Reflected by concrete walls | 0.0063-0.0374 |

Table 6-3 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport PWR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-SOL-THERM-003 (19 cases) | U | 93.17 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Stainless Steel, Al | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =71-454 (calc) | Single homogeneous unit contained in a cylindrical tank of SS-304 or Al-6061(diameter between 27.9 to 33 cm) | Reflected by Plexiglas | 0.0065-0.0384 |
| Experiment HEU-SOL-THERM-005 (17 cases) | U | 87.4 wt% $^{235}U$ and 93.2 wt% $^{235}U$ | Aqueous solution of uranyl nitrate and uranyl fluoride | Boron in borated glass (Pyrex pipes) | Pyrex glass | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =80 - 276 (reported) | Single homogeneous unit contained in a cylindrical tank of SS-316L (diameter 50.8 cm) | Unreflected or reflected (side and bottom) by light water | 0.0110-0.0410 |
| Experiment HEU-SOL-THERM-006 (29 cases) | U | 93.06 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Enriched boron in boric acid | Boric acid | Water in aqueous solution of uranyl nitrate<br>H/X =84 - 85 (calc) | Single homogeneous unit contained in a cylindrical tank of SS-304L (inner diameter 71.12 cm) | Unreflected or reflected (side) by light water, borated water, nickel, water + nickel, and borated water + nickel | 0.0320-0.0430 |
| Experiment HEU-SOL-THERM-009 (4 cases) | U | 93.18 wt% $^{235}U$ | Aqueous solution of uranium oxyfluoride | None | Al | Water in aqueous solution of uranyl fluoride<br>H/X =36 - 127 (reported; calc) | Single homogeneous unit contained in a spherical vessel made of Al (inner diameter ~23 cm) | Reflected by light water | 0.0180-0.0450 |
| Experiment HEU-SOL-THERM-010 (4 cases) | U | 93.12 wt% $^{235}U$ | Aqueous solution of uranium oxyfluoride (temp. from 27.5 to 85.5 °C) | None | Al | Water in aqueous solution of uranyl oxyfluoride<br>H/X =239 - 269 (reported) | Single homogeneous unit contained in a spherical vessel made of Al (inner diameter 26.4 cm) | Reflected by light water | 0.0090-0.0100 |

Table 6-3 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport PWR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-SOL-THERM-011 (2 cases) | U | 93.2 wt% $^{235}U$ | Aqueous solution of uranium oxyfluoride | None | | Water in aqueous solution of uranyl oxyfluoride H/X =523 - 533 (reported) | Single homogeneous unit contained in a spherical vessel made of Al (inner diameter 32 cm) | Reflected by light water | 0.0050 |
| Experiment HEU-SOL-THERM-012 (1 case) | U | 93.2 wt% $^{235}U$ | Aqueous solution of uranium oxyfluoride | None | Al | Water in aqueous solution of uranyl oxyfluoride H/X =1272 (reported) | Single homogeneous unit contained in a spherical vessel made of Al (inner diameter 27.9 cm) | Reflected by light water | 0.0027 |
| Experiment HEU-SOL-THERM-027 (9 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron or Cd | B4C or Cd absorber rods; nitric acid | Water in aqueous solution of uranyl nitrate H/X =203.6 (calc) | Single homogeneous unit contained in a cylindrical tank of SS (diameter 40.07 cm) | Unreflected | 0.0140-0.0150 |
| Experiment HEU-SOL-THERM-028 (18 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | One central $B_4C$ absorber rod (9 cases) with SS cladding; nitric acid | Water in aqueous solution of uranyl nitrate H/X = 91.5 and 374.5 (calc) | Single homogeneous unit contained in cylindrical tanks made of SS with various diameters (diameters between 22.04 to 40.07 cm) | Reflected by light water (side and bottom) | 0.0070-0.0270 |
| Experiment HEU-SOL-THERM-029 (7 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | Cluster of 7 $B_4C$ absorber rod (6 cases); nitric acid | Water in aqueous solution of uranyl nitrate H/X = 91.5 (calc) | Single homogeneous unit contained in a cylindrical tank made of SS (diameter 40.07 cm) | Reflected by light water (side and bottom) | 0.0270-0.0290 |
| Experiment HEU-SOL-THERM-030 (7 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | Cluster of 3 to 6 $B_4C$ absorber rods (5 cases); nitric acid | Water in aqueous solution of uranyl nitrate H/X = 91.1 and 374.6 (calc) | Single homogeneous unit contained in a cylindrical tank made of SS (diameter 40.07 cm) | Reflected by light water (side and bottom) | 0.0080-0.0280 |

Table 6-3 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport PWR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-SOL-THERM-031 (4 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | 18 or 36 $B_4C$ absorber rods ; nitric acid arranged in triangular lattices (4 to 6 cm pitch) | Water in aqueous solution of uranyl nitrate H/X = 91.1 (calc) | Single homogeneous unit contained in a cylindrical tank made of SS (diameter 40.07 cm) | Reflected by light water (side and bottom) | 0.0280-0.0310 |
| Experiment HEU-SOL-THERM-032 (1 case) | U | 93.2 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | | Water in aqueous uranyl nitrate H/X = 1835 (reported) | Single homogeneous unit contained in a spherical vessel made of Al (inner radius 61.01 cm) | Unreflected | 0.0021 |
| Experiment HEU-SOL-THERM-033 (26 cases) | U | 93.2 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron and Cadmium | Various absorbers including borated concrete plugs, plaster plugs, Cd lamination and water drums | Water in aqueous solution of uranyl nitrate H/X = 68.1 (calc) | Single homogeneous unit contained in a nested structure of 4 cylindrical tanks made of SS 304 (inner diameters of 58.7, 70.5, 82.2, 94 cm) | Reflected by concrete (parallelepiped) | 0.0320-0.0360 |
| Experiment HEU-SOL-THERM-035 (9 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | Various arrangements of $B_4C$ absorber rods ; 2 SS lattice plates; nitric acid | Water in aqueous solution of uranyl nitrate H/X = 181-767 (calc) | Single homogeneous unit contained in a cylindrical tank made of SS (inner diameter 110 cm) | Reflected by light water (side and bottom) | 0.0040-0.0160 |
| Experiment HEU-SOL-THERM-036 (4 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | Square pitched lattices of $B_4C$ absorber rods in 3 cases; nitric acid | Water in aqueous solution of uranyl nitrate H/X = 302.5 (calc) | Single homogeneous unit contained in a square tank made of SS + Cd wall (inner dimension 72.6 cm) | Unreflected | 0.0100-0.0120 |

Table 6-3 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport PWR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-SOL-THERM-037 (9 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | Hexagonally pitched lattice of $B_4C$ absorber rods in 6 cases; nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range = 339.7-684.5 (calc) | Single homogeneous unit contained in a cylindrical tank made of SS (inner diameter 160 cm) | Reflected by a thin layer (4 cm) of light water | 0.0050-0.0090 |
| Experiment HEU-SOL-THERM-043 (3 cases) | U | 93.2 wt% $^{235}U$ | Aqueous solution of uranium oxyfluoride | None | | Water in aqueous uranium oxyfluoride<br>H/X range = 203.5-1392.0 (calc; reported) | Single homogeneous unit contained in a spherical vessel made of Al (inner radius 16, 27.9 and 34.6 cm) | Unreflected | 0.0030-0.0140 |

Table 6-4 Area and Range of Applicability for Benchmark experiments for WP Configurations Containing Degraded Shippingport PWR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-THERM-003 (7 cases) | U | 94 to 94.5 wt% $^{235}U$ | U metal blocks (oralloy) | None | Lucite plates | Light water | Arrays of metal blocks (cubes and parallelepipeds) in a cubic geometry with various spacings | Reflected by light water | 0.2220-1.0940 |
| Experiment HEU-MET-MIXED-002 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | Polyethylene | Spherical core with polyethylene central area | Assembly core incorporates 3 spherical layers of fissile material( max radius 9.15 cm) | Reflected by polyethylene (thickness 12.85 cm) | 0.7880 |

Table 6-4 Area and Range of Applicability for Benchmark experiments for WP Configurations Containing Degraded Shippingport PWR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-MIXED-003 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | Polyethylene | Spherical core with polyethylene central area | Assembly core incorporates 3 spherical layers of fissile material( max radius 9.15 cm) | Reflected by polyethylene (thickness 15.85 cm) | 0.7840 |
| Experiment HEU-MET-FAST-002 (6 cases) | U | 93.5 wt% $^{235}U$ | U metal blocks (oralloy) | None | | None | Single homogeneous units in each experiment (1 pseudo sphere , 1 pseudo cylinder and 4 parallelepipeds) | Reflected by metallic natural U (tuballoy) | 1.5790-1.5990 |
| Experiment HEU-MET-FAST-004 (1 case)(2 benchmark models) | U | 97.67 wt% $^{235}U$ | U metal sphere | None | | None | Single homogeneous unit (sphere) with a radius of 6.5537 cm | Reflected by water | 1.1700 |
| Experiment HEU-MET-FAST-008 (1 case) | U | 89.77wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=10.15 cm) incorporates multiple (10) spherical layers of fissile material and has a central spherical cavity | Unreflected | 1.5501 |
| Experiment HEU-MET-FAST-011 (1 case) | U | 89.61wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=7.55 cm) incorporates multiple (7) spherical layers of fissile material and has a central spherical cavity (radius =2 cm) | Reflected by polyethylene (thickness 10.45 cm) | 1.1607 |

Table 6-4 Area and Range of Applicability for Benchmark experiments for WP Configurations Containing Degraded Shippingport PWR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-012 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=9.14 cm) incorporates multiple (12) spherical layers of fissile material and has a central cavity (radius = 0.6 cm) | Reflected by Al | 1.5237 |
| Experiment HEU-MET-FAST-013 (1 case) | U | 89.6 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=8.34 cm) incorporates multiple (9) spherical layers of fissile material and has a central cavity (radius = 1.4 cm) | Reflected by SS (thickness=3.65 cm) | 1.4838 |
| Experiment HEU-MET-FAST-014 (1 case) | U | 89.6 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=8.35 cm) incorporates multiple spherical layers of fissile material and has a central cavity (radius = 3.15) | Reflected by depleted uranium (thickness=4.65 cm) | 1.5465 |
| Experiment HEU-MET-FAST-015 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | | None | Cylindrical assembly of U metal discs (19.99 cm diameter) | Unreflected | 1.5813 |
| Experiment HEU-MET-FAST-018 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=9.15 cm) incorporates multiple (10) spherical layers of fissile material and has a central cavity (radius = 1 cm) | Unreflected | 1.5515 |

Table 6-4 Area and Range of Applicability for Benchmark experiments for WP Configurations Containing Degraded Shippingport PWR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-019 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=9.15 cm) incorporates multiple (7) spherical layers of fissile material and has a central cavity (radius = 4.029 cm) | Reflected by graphite (thickness=3.45 cm) | 1.4793 |
| Experiment HEU-MET-FAST-020 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=8.35 cm) incorporates multiple (9) spherical layers of fissile material | Reflected by polyethylene (thickness=1.45 cm) | 1.4341 |
| Experiment HEU-MET-FAST-021 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=7.55 cm) incorporates multiple (8) spherical layers of fissile material and has a central cavity (radius = 0.89 cm) | Reflected by 2 steel layers (thickness=9.7 cm) | 1.4431 |
| Experiment HEU-MET-FAST-022 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=8.35 cm) incorporates multiple (9) spherical layers of fissile material and has a central cavity (radius =1 cm) | Reflected by duralumin (thickness=3.9 cm) | 1.5049 |
| Experiment HEU-MET-FAST-024 (1 case) | U | 89.6 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=7.54 cm) incorporates multiple (10) spherical layers of fissile material and has a central cavity (radius = 0.6 cm) | Reflected by steel (thickness =0.8 cm) and polyethylene (thickness=9.65 cm) | 1.2553 |

Table 6-4 Area and Range of Applicability for Benchmark experiments for WP Configurations Containing Degraded Shippingport PWR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-027 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=8.35 cm) incorporates multiple (9) spherical layers of fissile material and has a central cavity (radius =1 cm) | Reflected by lead (thickness =3.25 cm) | 1.5120 |
| Experiment HEU-MET-FAST-029 (1 case) | U | 89.6 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=7.55 cm) incorporates multiple (7) spherical layers of fissile material and has a central cavity (radius = 2 cm) | Reflected by depleted uranium (thickness =4.7 cm) | 1.5470 |
| Experiment HEU-MET-FAST-031 (1 case) | U | 89.4 wt% $^{235}U$ (average) | U metal | None | Polyethylene | Polyethylene in the central cavity | The spherical assembly core (radius=7.55 cm) incorporates multiple spherical layers of fissile material. The central area contains a spherical block of polyethylene (radius= 4.02) with a central cavity (radius= 1.4 cm) | Reflected by polyethylene (thickness =17.45 cm) | 0.9660 |
| Experiment HEU-MET-FAST-032 (4 cases) | U | 94 wt% $^{235}U$ | U metal | None | | None | The spherical assembly core (radius between 6.326 and 7.755 cm) incorporates multiple spherical layers of fissile material and has a central cavity | Reflected by natural uranium (with thickness ranging from 9.982 cm in the first case to 1.735 cm in the last case) | 1.5920-1.5650 |

Table 6-4 Area and Range of Applicability for Benchmark experiments for WP Configurations Containing Degraded Shippingport PWR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-041 (6 cases) | U | 94 wt% $^{235}U$ | U metal | None | | None | Single homogeneous unit (sphere) with a radius between 5.64 and 7.38 cm | Reflected by Be(2 cases) and graphite (4 cases) | 1.1280-1.4660 |
| Experiment HEU-MET-FAST-055 (1 case) | U | 93.5 – 93.9 wt% $^{235}U$ | U metal | None | Al | None | Complex cylindrical geometry of the core (radius = 30.73 cm) | Reflected by depleted uranium | 1.0770 |
| HEU-MET-MIXED-005 (5 cases) | U | 90 wt% $^{235}U$ | U metal pellets | B as $B_4C$ present in some rods | $SiO_2$, Al tubes | Polyethylene, sand ($SiO_2$) | Hexagonally pitched lattice of tubes containing fuel and/or moderator | $SiO_2$, depleted uranium | 0.2120-0.3770 |

Table 6-5 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact UZrH (TRIGA) SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-COMP-THERM-003 (2 cases) | U | 19.9 wt% $^{235}U$ | U-ZrH | $B_4C$ in absorber rods | Stainless steel (SS304) cladding; Mo discs; central Zr rod; graphite inserts | ZrH + water<br>H/X = 150.1 | Complex cylindrical arrays of TRIGA fuel rods | The core is radially reflected by graphite and axially by water | 0.0240 |
| Experiment HEU-COMP-THERM-003 (15 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding<br>Al-lattice plates | Water<br>H/X range(fuel region) = 51 - 349 | Cylindrical two-zones hexagonally pitched lattices (active fuel length =69.2 cm). Varied pitches. | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0139-0.0467 |

Table 6-5 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact UZrH (TRIGA) SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-COMP-THERM-004 (4 cases) | U | 88.87 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | Gd or Sm rods | SS-cladding<br>Al-lattice plates; Bronze caps, $Al_2O_3$ in absorber rods | Water<br>H/X (fuel region) = 35 | Cylindrical hexagonally pitched lattices ; double lattices of fuel rods (0.53 cm) and absorber rods(2.75 and 3.67 cm | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0736-0.0756 |
| Experiment HEU-COMP-THERM-005 (1 case) (detailed benchmark model evaluated) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding<br>Al-lattice plates | Water<br>H/X (fuel region) = 23 | Hexagonally pitched array of fuel rod clusters (each containing a hexagonally pitched lattice of fuel rods); hex pitch = 0.52 cm | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0764 |
| Experiment HEU-COMP-THERM-006<br>(3 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding<br>Al-lattice plates | Water<br>H/X range (fuel region) = 30 - 716 | Cylindrical hexagonally pitched lattice (pitches of 0.56, 1 and 2.113 cm) | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0104-0.0720 |
| Experiment HEU-COMP-THERM-007<br>(3 cases) (detailed benchmark models evaluated) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | ZrH rods, SS-cladding<br>Al-lattice plates | Water +ZrH rods<br>H/X range (fuel region) = 60 - 91 | Cylindrical hexagonally double lattice of fuel rods and ZrH rods. Fixed pitch for fuel rods (0.61 cm) and variable for ZrH rods | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0339-0.0475 |
| Experiment HEU-COMP-THERM-008 (2 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | B | $B_4C$ rods, SS-cladding<br>Al-lattice plates | Water<br>H/X = 25 | Cylindrical hexagonally double lattice of fuel rods and $B_4C$ rods. Fixed pitch for fuel rods (0.53 cm) and variable for $B_4C$ rods 2.12 and 2.65 cm) | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0882-0.0922 |

Table 6-5 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact UZrH (TRIGA) SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-COMP-THERM-010 (21cases) | U | 62.4 wt% $^{235}U$ | $UO_2$+BeO mixture | B as boric acid (few cases) | Hastelloy X-280 cladding | Water (15 cases); water +uranyl nitrate and/or boric acid (few cases) H/X range =36 to 302 | Square or cylindrical assemblies with square/hexagona pitched lattices (lattice pitch between 1.24 to 3 cm; active fuel length=193 cm) | Some of the configurations are fully reflected by water; some are partially flooded and are not reflected at top | 0.0234-0.0797 |
| Experiment HEU-COMP-THERM-011 (3 cases) | U | 80 wt% $^{235}U$ | $UO_2$ +Al alloy (sintered) | None | SS cladding; Al lattice plate | Water; partially flooded cores H/X=170 | Four square clusters of cylindrical fuel rods (21x21, pitch =1.4 cm)) arranged in a square geometry (dist between clusters = 0 to 4.8 cm) | Assembly reflected by water (except top of assembly - partially flooded) | 0.0470-0.0533 |
| Experiment HEU-COMP-THERM-012 (2 cases) | U | 80 wt% $^{235}U$ | $UO_2$ +Al alloy (sintered) | None | SS cladding; Al lattice plate | Water; partially flooded cores H/X=35 | Four square clusters of cylindrical fuel rods (18x18, pitch =1.4 cm)) arranged in a square geometry (dist between clusters =1.4 and 2.8 cm) | Assembly reflected by water (except top of assembly - partially flooded) | 0.0509-0.0551 |
| Experiment HEU-COMP-THERM-013 (2 cases) | U | 80 wt% $^{235}U$ | $UO_2$ +Al alloy (sintered) | None | SS cladding; Al lattice plate | Water; partially flooded cores H/X=40 | Nine square clusters of cylindrical fuel rods (14x14, pitch =1.4 cm)) arranged in a square geometry (dist between clusters =1.4 and 2.8 cm) | Assembly reflected by water (except top of assembly - partially flooded) | 0.0432-0.0481 |

Table 6-5 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact UZrH (TRIGA) SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-COMP-THERM-014 (2 cases) | U | 79.4 at% $^{235}U$ | $UO_2$ +Al alloy (sintered) | None | SS cladding; Al lattice plate | Water; partially flooded cores<br>H/X=170 | Nine square clusters of cylindrical fuel rods (10x10, pitch =1.98 cm) arranged in a square geometry (dist between clusters is 0 and 1.98 cm) | Assembly reflected by water (except top of assembly - partially flooded) | 0.0227-0.0259 |
| Experiment HEU-MET-THERM-006 (23 cases) | U | 93.17 wt% $^{235}U$ | U-Al alloy (fuel plates) | B (cases 20-23) | Al cladding | Water; dilute aqueous solution of uranyl nitrate (cases 19 - 23)<br>H/X range (fuel region) =134 - 500 (calc) | Rectangular arrays of fuel elements with various spacing. Each fuel element is composed of up to a maximum of 22 fuel plates | Assembly reflected by water or dilute aqueous uranyl nitrate solutions | 0.0097-0.0147 |
| Experiment HEU-COMP-THERM-002 (25 cases) | U | 93.15 wt% $^{235}U$ | Uranium dicarbide | None | Al tubes; Plexiglas supports | Graphite and Water<br>C/X=87-88.9 | Various arrays (triangular or square pitched lattices) of Al tubed or bare fuel elements (hexagonal graphite blocks containing uranium dicarbide beads) surrounded by water | Assembly reflected by water | 0.0094-0.0244 |

Table 6-6 Area and Range of Applicability of Benchmark experiments for WP Configurations Containing Intact UZrH(TRIGA) SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-023 (22 cases) | U | 93.2 wt% $^{235}U$ | U metal (cylinders) | None | Supporting stainless steel rods | None | Different cubic arrays (2x2x2;3x3x3;4x4x4) of HEU cylinders (approx. 10.5 kg each) | Unreflected or reflected by paraffin | 0.9830-1.5850 |

Table 6-6 Area and Range of Applicability of Benchmark experiments for WP Configurations Containing Intact UZrH(TRIGA) SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-026 (32 cases) | U | 93.2 wt% $^{235}U$ | U metal (cylinders) | None | Supporting stainless steel rods | None | Different cubic arrays (2x2x2;3x3x3) of HEU cylinders (approx. 15, 20 or 26 kg each) | Unreflected or reflected by paraffin | 1.0961-1.5835 |
| Experiment HEU-COMP-INTER-003 (7 cases) | U | 90.7 wt% $^{235}U$ | U hydride ($UH_3$) compressed powder | None | Steel cans; steel fillers, Al spacers | None | Complex cylindrical geometry comprising interspersed discs (fuel radius=7.5 cm) | Single or two layers reflectors (Be, depleted uranium or iron) | 0.5640-0.8240 |
| Experiment HEU-COMP-INTER-001 (HEU-COMP-MIXED-003) (6 cases – only case 5 was evaluated)) | U | 96 wt% $^{235}U$ | $UO_2$ | None | Al foil cladding | ZrH + water in various reactor cavities | Cylindrical array of fuel rods in a ZrH block with BeO axial and radial reflector. | Reflected by BeO and water | 0.2200 |
| Experiment HEU-COMP-MIXED-001 (HEU-COMP-THERM-001) (26 cases) | U | 93.1 wt% $^{235}U$ | $UO_2$ | None | Carbon steel cans | Alcohol-water solution in some cases<br>H/X=0-49 (calc)<br>Plexiglas | Complex arrays of cans in rectangular geometry | Reflected by polyethylene (thickness 15.24 cm)<br>Plexiglas present | 0.1045-0.8015 |
| Experiment HEU-COMP-MIXED-002 (23 cases) | U | 89.4 wt% $^{235}U$ | $UO_2$ | None | SS cladding and plugs | Mixtures of light and heavy water | Hexagonal array of tubes with variable pitches (diameter= 8.3 cm) containing uranium dioxide powder | Partially reflected by light water (bottom and laterally) | 0.5370-0.9020 |

Table 6-7 Area and Range of Applicability of Benchmark experiments for WP Configurations Containing Degraded UZrH(TRIGA) SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-SOL-THERM-001 (10 cases) | U | 93.17 wt% $^{235}U$ | Aqueous solution of Uranyl nitrate | None | Stainless steel, Al | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =68 - 499 (calc) | Single homogeneous unit contained in a cylindrical tank made of SS-304 or Al-6061(diameter between 27.9 to 50.7 cm) | Unreflected (minimally reflected by concrete walls) | 0.0065-0.0410 |
| Experiment HEU-SOL-THERM-005 (17 cases) | U | 87.4 wt% $^{235}U$ and 93.2 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron in borated glass (Pyrex pipes) | Pyrex glass | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =80 - 276 (reported) | Single homogeneous unit contained in a cylindrical tank of SS-316L (diameter 50.8 cm) | Unreflected or reflected (side and bottom) by light water | 0.0110-0.0410 |
| Experiment HEU-SOL-THERM-006 (29 cases) | U | 93.06 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Enriched boron in boric acid | Boric acid | Water in aqueous solution of uranyl nitrate<br>H/X =84 - 85 (calc) | Single homogeneous unit contained in a cylindrical tank of SS-304L (inner diameter 71.12 cm) | Unreflected or reflected (side) by light water, borated water, Nickel, water + nickel, borated water + nickel | 0.0320-0.0430 |
| Experiment HEU-SOL-THERM-008 (14 cases, 5 evaluated) | U | 93.17 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Al tanks; SS sleeves, nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (in solution) =69 – 454 (calc) | Arrays (2x2, 2x3, 4x4) of cylindrical tanks placed in a rectangular geometry | Reflected by Plexiglas | 0.0064-0.0367 |
| Experiment HEU-SOL-THERM-009 (4 cases) | U | 93.18 wt% $^{235}U$ | Aqueous solution of uranium oxyfluoride | None | | Water in aqueous solution of uranyl fluoride<br>H/X =36 - 127 (reported; calc) | Single homogeneous unit contained in a spherical vessel made of Al (inner diameter ~23 cm) | Reflected by light water | 0.0180-0.0450 |

Table 6-7 Area and Range of Applicability of Benchmark experiments for WP Configurations Containing Degraded UZrH(TRIGA) SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-SOL-THERM-010 (4 cases) | U | 93.12 wt% $^{235}U$ | Aqueous solution of uranium oxyfluoride (temp. from 27.5 to 85.5 °C) | None | | Water in aqueous solution of uranyl oxyfluoride<br>H/X =239 - 269 (reported) | Single homogeneous unit contained in a spherical vessel made of Al (inner diameter 26.4 cm) | Reflected by light water | 0.0090-0.0100 |
| Experiment HEU-SOL-THERM-011 (2 cases) | U | 93.12 wt% $^{235}U$ | Aqueous solution of uranium oxyfluoride | None | | Water in aqueous solution of uranyl oxyfluoride<br>H/X =523 - 533 (reported) | Single homogeneous unit contained in a spherical vessel made of Al (inner diameter 32 cm) | Reflected by light water | 0.0050 |
| Experiment HEU-SOL-THERM-012 (1 case) | U | 93.2 wt% $^{235}U$ | Aqueous solution of uranium oxyfluoride | None | | Water in aqueous solution of uranyl oxyfluoride<br>H/X =1272 (reported) | Single homogeneous unit contained in a spherical vessel made of Al (inner diameter 27.9 cm) | Reflected by light water | 0.0027 |
| Experiment HEU-SOL-THERM-013 (4 cases) | U | 93.2 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | | Water in aqueous uranyl nitrate<br>H/X range = 971 – 1375 (calc) | Single homogeneous unit contained in an Al spherical vessel (inner radius 34.59 cm) | Unreflected | 0.0026-0.0038 |
| Experiment HEU-SOL-THERM-014 (3 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Gd | Nitric acid | Water in aqueous uranyl nitrate<br>H/X range = 405-421(calc) | Single homogeneous unit contained in a SS cylindrical tank (inner radius 20 cm) | Reflected by water (laterally and bottom) | 0.0071-0.0076 |

Table 6-7 Area and Range of Applicability of Benchmark experiments for WP Configurations Containing Degraded UZrH(TRIGA) SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-SOL-THERM-015 (5 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Gd | Nitric acid | Water in aqueous uranyl nitrate<br>H/X range = 278-295 (calc) | Single homogeneous unit contained in a SS cylindrical tank (inner radius 20 cm) | Reflected by water (laterally and bottom) | 0.0100-0.0113 |
| Experiment HEU-SOL-THERM-016 (3 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Gd | Nitric acid | Water in aqueous uranyl nitrate<br>H/X range = 175-192 (calc) | Single homogeneous unit contained in a SS cylindrical tank (inner radius 20 cm) | Reflected by water (laterally and bottom) | 0.0151-0.0161 |
| Experiment HEU-SOL-THERM-017 (8 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Gd | Nitric acid | Water in aqueous uranyl nitrate<br>H/X range = 133-147 (calc) | Single homogeneous unit contained in a SS cylindrical tank (inner radius 12.4 or 20 cm) | Reflected by water (laterally and bottom) | 0.0189-0.0221 |
| Experiment HEU-SOL-THERM-018 (12 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Gd | Nitric acid | Water in aqueous uranyl nitrate<br>H/X range = 86-94 (calc) | Single homogeneous unit contained in a SS cylindrical tank (inner radius 12.4 or 20 cm) | Reflected by water (laterally and bottom) | 0.0285-0.0329 |
| Experiment HEU-SOL-THERM-019 (3 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Gd | Nitric acid | Water in aqueous uranyl nitrate<br>H/X range = 55-63 (calc) | Single homogeneous unit contained in a SS cylindrical tank (inner radius 12.4 cm) | Reflected by water (laterally and bottom) | 0.0393-0.0425 |
| Experiment HEU-SOL-THERM-021 (32 cases) | U | 92.6 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Plexiglas containers; Al supporting structure; nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (in solution) =59 – 435 (calc) | Arrays (2x2x2, 3x3x3, 4x4x4, 5x5x5) of cylindrical containers placed in a rectangular geometry | Unreflected or reflected by Plexiglas and/or Paraffin | 0.0067-0.0437 |

Table 6-7 Area and Range of Applicability of Benchmark experiments for WP Configurations Containing Degraded UZrH(TRIGA) SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-SOL-THERM-025 (18 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Gd | Nitric acid | Water in aqueous uranyl nitrate<br>H/X range = 61.8-556 | SS cylindrical tank (inner diameter 40 cm) filled with uranyl nitrate poisoned with Gd inserted in another cylindrical tank (inner diameter 59.4 cm containing uranyl nitrate | Reflected by water (laterally and bottom) | 0.0050-0.0280 |
| Experiment HEU-SOL-THERM-027 (9 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron or Cd | B4C or Cd absorber rods; SS; nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X =203.6 (calc) | Single homogeneous unit contained in a cylindrical tank of SS (diameter 40.07 cm) | Unreflected | 0.0140-0.0150 |
| Experiment HEU-SOL-THERM-028 (18 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | One central $B_4C$ absorber rod (9 cases); nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X = 91.5 and 374.5 (calc) | Single homogeneous unit contained in cylindrical tanks made of SS with various diameters (diameters between 22.04 to 40.07 cm) | Reflected by light water (side and bottom) | 0.0070-0.0270 |
| Experiment HEU-SOL-THERM-029 (7 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | Cluster of 7 $B_4C$ absorber rod (6 cases); nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X = 91.5 (calc) | Single homogeneous unit contained in a cylindrical tank made of SS (diameter 40.07 cm) | Reflected by light water (side and bottom) | 0.0270-0.0290 |
| Experiment HEU-SOL-THERM-030 (7 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | Cluster of 3 to 6 $B_4C$ absorber rods (5 cases); nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X = 91.1 and 374.6 (calc) | Single homogeneous unit contained in a cylindrical tank made of SS (diameter 40.07 cm) | Reflected by light water (side and bottom) | 0.0080-0.0280 |

Table 6-7 Area and Range of Applicability of Benchmark experiments for WP Configurations Containing Degraded UZrH(TRIGA) SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-SOL-THERM-031 (4 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | 18 or 36 $B_4C$ absorber rods ; nitric acid | Water in aqueous solution of uranyl nitrate H/X = 91.1 (calc) | Single homogeneous unit contained in a cylindrical tank made of SS (diameter 40.07 cm) | Reflected by light water (side and bottom) | 0.0280-0.0310 |
| Experiment HEU-SOL-THERM-032 (1 case) | U | 93.2 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | | Water in aqueous uranyl nitrate H/X = 1835 (reported) | Single homogeneous unit contained in a spherical vessel made of Al (inner radius 61.01 cm) | Unreflected | 0.0021 |
| Experiment HEU-SOL-THERM-033 (26 cases) | U | 93.2 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron and Cadmium | Various absorbers including borated concrete plugs, plaster plugs, Cd lamination and water drums | Water in aqueous solution of uranyl nitrate H/X = 68.1 (calc) | Single homogeneous unit contained in a nested structure of 4 cylindrical tanks made of SS 304(inner diameters of 58.7, 70.5, 82.2, 94 cm) | Reflected by concrete (parallelepiped) | 0.0320-0.0360 |
| Experiment HEU-SOL-THERM-035 (9 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | Various arrangements of $B_4C$ absorber rods ; 2 SS lattice plates; nitric acid | Water in aqueous solution of uranyl nitrate H/X = 181-767 (calc) | Single homogeneous unit contained in a cylindrical tank made of SS (inner diameter 110 cm) | Reflected by light water (side and bottom) | 0.0040-0.0160 |
| Experiment HEU-SOL-THERM-036 (4 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | Square pitched lattices of $B_4C$ absorber rods in 3 cases; nitric acid | Water in aqueous solution of uranyl nitrate H/X = 302.5 (calc) | Single homogeneous unit contained in a square tank made of SS + Cd wall(inner dimension 72.6 cm) | Unreflected | 0.0100-0.0120 |

Table 6-7 Area and Range of Applicability of Benchmark experiments for WP Configurations Containing Degraded UZrH(TRIGA) SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-SOL-THERM-037 (9 cases) | U | 89 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | Boron | Hexagonally pitched lattice of $B_4C$ absorber rods in 6 cases; nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range = 339.7-684.5 (calc) | Single homogeneous unit contained in a cylindrical tank made of SS (inner diameter 160 cm) | Reflected by a thin layer (4 cm) of light water | 0.0050-0.0090 |
| Experiment HEU-SOL-THERM-043 (3 cases) | U | 93.2 wt% $^{235}U$ | Aqueous solution of uranium oxyfluoride | None | | Water in aqueous uranium oxyfluoride<br>H/X range = 203.5-1392.0 (calc; reported) | Single homogeneous unit contained in a spherical vessel made of Al (inner radius 16, 27.9 and 34.6 cm) | Unreflected | 0.0030-0.0140 |
| Experiment HEU-SOL-THERM-044 (16 cases) | U | 93.17 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | B, Cl, Cd, Gd | Absorber were placed in tubes filled with Pyrex glass, PVC, Boraflex rubber, Borax, Cd sleeves, and Cd or Gd oxides | Water in aqueous solution of uranyl nitrate<br>H/X = 67.2 (calc) | Solution in a SS cylindrical tank (diameter 106.6 cm) containing an array of identical absorber rods | Reflected by concrete | 0.0340-0.0470 |

Table 6-8 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded UZrH (TRIGA) SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-002 (6 cases) | U | 93.5 wt% $^{235}U$ | U metal blocks (oralloy) | None | | None | Single homogeneous units in each experiment (1 sphere , 1 cylinder and 4 parallelepipeds) | Reflected by metallic natural U (tuballoy) | 1.5790-1.5990 |
| Experiment HEU-MET-FAST-004 (1 case)(2 benchmark models) | U | 97.67 wt% $^{235}U$ | U metal sphere | None | | None | Single homogeneous unit (sphere) with a radius of 6.5537 cm | Reflected by water | 1.1700 |
| Experiment HEU-MET-FAST-008 (1 case) | U | 89.77 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=10.15 cm) incorporates multiple (10) spherical layers of fissile material and has a central spherical cavity (radius = 2 cm) | Unreflected | 1.5501 |
| Experiment HEU-MET-FAST-011 (1 case) | U | 89.61 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=7.55 cm) incorporates multiple (7) spherical layers of fissile material and has a central spherical cavity (radius =2 cm) | Reflected by polyethylene (thickness 10.45 cm) | 1.1607 |

Table 6-8 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded UZrH (TRIGA) SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-012 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=9.14 cm) incorporates multiple (12) spherical layers of fissile material and has a central cavity (radius = 0.6 cm) | Reflected by Al | 1.5237 |
| Experiment HEU-MET-FAST-013 (1 case) | U | 89.6 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=8.34 cm) incorporates multiple (9) spherical layers of fissile material and has a central cavity (radius = 1.4 cm) | Reflected by SS (thickness=3.65 cm) | 1.4838 |
| Experiment HEU-MET-FAST-014 (1 case) | U | 89.6 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=8.35 cm) incorporates multiple (7) spherical layers of fissile material and has a central cavity (radius = 3.15 cm) | Reflected by depleted uranium (thickness=4.65 cm) | 1.5465 |
| Experiment HEU-MET-FAST-015 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | | None | Cylindrical assembly of U metal discs (19.99 cm diameter) | Unreflected | 1.5813 |

Table 6-8 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded UZrH (TRIGA) SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-018 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=9.15 cm) incorporates multiple (10) spherical layers of fissile material and has a central cavity (radius = 1 cm) | Unreflected | 1.5515 |
| Experiment HEU-MET-FAST-019 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=9.15 cm) incorporates multiple (7) spherical layers of fissile material and has a central cavity (radius = 4.029 cm) | Reflected by graphite (thickness=3.45 cm) | 1.4793 |
| Experiment HEU-MET-FAST-020 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=8.35 cm) incorporates multiple (9) spherical layers of fissile material | Reflected by polyethylene (thickness=1.45 cm) | 1.4341 |
| Experiment HEU-MET-FAST-021 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=7.55 cm) incorporates multiple (8) spherical layers of fissile material and has a central cavity (radius = 0.89 cm) | Reflected by 2 steel layers (thickness=9.7 cm) | 1.4431 |

Table 6-8 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded UZrH (TRIGA) SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-022 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=8.35 cm) incorporates multiple (9) spherical layers of fissile material and has a central cavity (radius =1 cm) | Reflected by duralumin (thickness=3.9 cm) | 1.5049 |
| Experiment HEU-MET-FAST-024 (1 case) | U | 89.6 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=7.54 cm) incorporates multiple (10) spherical layers of fissile material and has a central cavity (radius = 0.6 cm) | Reflected by steel (thickness =0.8 cm) and polyethylene (thickness=9.65 cm) | 1.2553 |
| Experiment HEU-MET-FAST-027 (1 case) | U | 89.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=8.35 cm) incorporates multiple (9) spherical layers of fissile material and has a central cavity (radius =1 cm) | Reflected by lead (thickness =3.25 cm) | 1.5120 |
| Experiment HEU-MET-FAST-029 (1 case) | U | 89.6 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=7.55 cm) incorporates multiple (7) spherical layers of fissile material and has a central cavity (radius = 2 cm) | Reflected by depleted uranium (thickness =4.7 cm) | 1.5470 |

Table 6-8 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded UZrH (TRIGA) SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-031 (1 case) | U | 89.4 wt% $^{235}U$ (average) | U metal | None | Polyethylene | Polyethylene in the central cavity | The spherical assembly core (radius=7.55 cm) incorporates multiple spherical layers of fissile material. The central area contains a spherical block of polyethylene (radius= 4.02 cm) with a central cavity (radius = 1.4 cm) | Reflected by polyethylene (thickness =17.45 cm) | 0.9660 |
| Experiment HEU-MET-FAST-032 (4 cases) | U | 94 wt% $^{235}U$ | U metal | None | | None | The spherical assembly core (radius between 6.326 and 7.755 cm) incorporates multiple spherical layers of fissile material and has a central cavity | Reflected by natural uranium (with thickness ranging from 9.982 cm in the first case to 1.735 cm in the last case) | 1.5920-1.5650 |
| Experiment HEU-MET-FAST-041 (6 cases) | U | 93.5 – 93.9 wt% $^{235}U$ | U metal | None | | None | Single homogeneous unit (sphere) with a radius between 5.64 and 7.38 cm | Reflected by Be(2 cases) and graphite (4 cases) | 1.1280-1.4660 |
| Experiment HEU-MET-FAST-055 (1 case) | U | 93 wt% $^{235}U$ | U metal | None | Al | None | Complex cylindrical geometry of the core (radius = 30.73 cm) | Reflected by depleted uranium | 1.0770 |
| HEU-MET-MIXED-005 (5 cases) | U | 90 wt% $^{235}U$ | U metal pellets | B as $B_4C$ present in some rods | $SiO_2$, Al tubes | Polyethylene, sand ($SiO_2$) | Hexagonally pitched lattice of tubes containing fuel and/or moderator | $SiO_2$, depleted uranium | 0.2120-0.3770 |

Table 6-8 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded UZrH (TRIGA) SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-THERM-003 (7 cases) | U | 94 to 94.5 wt% $^{235}U$ | U metal blocks (oralloy) | None | Lucite plates | Light water | Arrays of metal blocks (cubes and parallelepipeds) in a cubic geometry with various spacings | Reflected by light water | 0.2220-1.0940 |
| Experiment HEU-MET-MIXED-002 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | Polyethylene | Spherical core with polyethylene central area | Assembly core incorporates 3 spherical layers of fissile material (max radius 9.15 cm) | Reflected by polyethylene (thickness 12.85 cm) | 0.7880 |
| Experiment HEU-MET-MIXED-003 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | Polyethylene | Spherical core with polyethylene central area | Assembly core incorporates 3 spherical layers of fissile material (max radius 9.15 cm) | Reflected by polyethylene (thickness 15.85 cm) | 0.7840 |

Table 6-9 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Melt and Dilute Waste Form (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-COMP-THERM-001 (29 cases) | U | 29.83 wt% $^{235}U$ | Compound containing $UF_4$ and polytetrafluo r-oethylene | Cd + B (few cases) | | Polyethylene<br>H/X range = 4 – 222 (calc) | Cuboid shapes constructed from small cubes of compound and moderator on two Al platforms | Unreflected or reflected by Paraffin (few cases have sheets of B or paraffin surrounding the core) | 0.0455-0.2168 |

Table 6-9 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Melt and Dilute Waste Form (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-COMP-THERM-002 (6 cases) | U | 17 wt% $^{235}U$ | $UO_2$ | Gd, Cd | Stainless steel cladding | Water | Cylindrical hexagonally pitched lattice (pitch =6.8 cm). Cold (~20 °C) and hot experiments (~200 °C) performed | Reflected by water | 0.0440-0.0490 |
| Experiment IEU-COMP-THERM-003 (2 cases) | U | 19.9 wt% $^{235}U$ | U-ZrH | $B_4C$ in absorber rods | Stainless steel (SS304) cladding; Mo discs; central Zr rod; graphite inserts | ZrH + water<br>H/X = 150.1 | Complex cylindrical arrays of TRIGA fuel rods | The core is radially reflected by graphite and axially by water | 0.0240 |

Table 6-10 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Melt and Dilute Waste Form (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-MET-FAST-001 (4 cases, detailed benchmark models evaluated) | U | 93.4 wt% $^{235}U$ + natural U | HEU metal discs interspersed with natural U | None | Al collar | None | Complex cylindrical geometry comprising interspersed U discs (fuel radius = 13.335 cm) | Unreflected | 1.3859-1.4398 |
| Experiment IEU-MET-FAST-002 (1 case) | U | Av. 16.19 wt% $^{235}U$ | HEU metal discs (93.4 wt% $^{235}U$ interspersed with natural U | None | | None | Complex cylindrical geometry comprising interspersed U discs (fuel radius = 19.05 cm, column height = 31.951 cm) | Reflected by natural U | 1.2784 |

Table 6-10 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Melt and Dilute Waste Form (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-MET-FAST-003 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 15.324 cm) incorporates multiple spherical layers of fissile material | Unreflected | 1.3502 |
| Experiment IEU-MET-FAST-004 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 14 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 2.788 cm) | Reflected by graphite (thickness = 3.2 cm) | 1.3071 |
| Experiment IEU-MET-FAST-005 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 13.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 2.686 cm) | Reflected by steel (thickness = 8.25 cm) | 1.2852 |
| Experiment IEU-MET-FAST-006 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 13.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 2.1 cm) | Reflected by duralumin (thickness = 11.75 cm) | 1.2892 |
| Experiment IEU-MET-FAST-007 (1 case) | U | 10 wt% $^{235}U$ | U metal | None | | None | Complex cylindrical geometry | Reflected by depleted uranium (thickness = 15.24 cm) | 1.2530 |

Table 6-10 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Melt and Dilute Waste Form (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-MET-FAST-008 (1 case) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 13.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 1.4 cm) | Reflected by depleted uranium (thickness=3.25 cm) | 1.3650 |
| Experiment IEU-MET-FAST-009 (1 case) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 12.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 3.445 cm) | Reflected by polyethylene (thickness = 5.75 cm) | 1.0140 |
| Experiment IEU-MET-FAST-010 (1 case) | U | 9 wt% $^{235}U$ (average) | U metal | None | Stainless steel, Al | None | Simplified cylindrical geometry (radius=40.996, height=76.36) | Reflected by depleted uranium (thickness > 24 cm) | 1.1490 |

Table 6-11 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Melt and Dilute Waste Form (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-SOL-THERM-001 (4 cases) | U | 20.9 at% $^{235}U$ | Aqueous solution of uranyl sulfate | None | SS core channels; SS coiled pipe | Water in aqueous solution of uranyl sulfate<br>H/X range (fuel region) =217 – 444 (calc) | The uranyl sulfate solution is contained in a cylindrical tank with hemispheric bottom( SS(diameter = 30.5 cm; height=65.6 cm) | Reflected by graphite (except top) | 0.0149-0.0275 |

Table 6-11 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Melt and Dilute Waste Form (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-COMP-THERM-001 (29 cases) | U | 29.83 wt% $^{235}U$ | Compound containing $UF_4$ and polytetrafluoroethylene | Cd + B (few cases) | | Polyethylene<br>H/X range = 4.0 – 222 (calc) | Cuboid shapes constructed from small cubes of compound and moderator on two Al platforms | Unreflected or reflected by Paraffin; Cd and/or B sheets placed arround the core in some cases | 0.0455-0.2168 |
| Experiment LEU-SOL-THERM-003 (9 cases) | U | 10 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Excess nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =770-1437 | The uranyl nitrate solution is contained in a spherical tank (radius=32.95, 43.63, and 59.89 cm) | Unreflected | 0.0114-0.0186 |
| Experiment LEU-SOL-THERM-004 (7 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Excess nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =719-1018 (calc) | The uranyl nitrate solution is contained in a cylindrical tank (diameter=59 cm) | Reflected by water (thickness > 20 cm) | 0.0142-0.0188 |
| Experiment LEU-SOL-THERM-006 (5 cases) | U | 10 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | B4C | Excess nitric acid; absorber rods (SS tubes filed with B4C powder) | Water in aqueous solution of uranyl nitrate<br>H/X (fuel region) =532 | The uranyl nitrate solution is contained in a cylindrical tank (diameter=110 cm) | Reflected by water (laterally and bottom) (thickness > 20 cm) | 0.0245-0.0257 |
| Experiment LEU-SOL-THERM-007 (5 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Excess nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =709-942 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter = 59 cm) | Unreflected | 0.0159-0.0200 |

Table 6-11 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Melt and Dilute Waste Form (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment LEU-SOL-THERM-008 (4 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Excess nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =951-956 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter = 59 cm) | Reflected by concrete (laterally) | 0.0152-0.0154 |
| Experiment LEU-SOL-THERM-009 (3 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Excess nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =934-936 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter = 59 cm) | Reflected by borated concrete (laterally) | 0.0155-0.0158 |
| Experiment LEU-SOL-THERM-010 (4 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Excess nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) = 942-946 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter = 59 cm) | Reflected by polyethylene (laterally) | 0.0153-0.0154 |
| Experiment LEU-SOL-THERM-016 (7 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Excess nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) = 469-772 (calc.) | The uranyl nitrate solution is contained in a rectangular tank-slab geometry-(dimensions = 28 x 69 cm) | Reflected by water (thickness > 20 cm) | 0.0180-0.0267 |
| Experiment LEU-SOL-THERM-017 (6 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Excess nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =469-729 (calc.) | The uranyl nitrate solution is contained in a rectangular tank-slab geometry-(dimensions = 28 x 69 cm) | Unreflected | 0.0192-0.0275 |

Table 6-11 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Melt and Dilute Waste Form (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment LEU-SOL-THERM-018 (6 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Excess nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) = 731-751 (calc.) | The uranyl nitrate solution is contained in a rectangular tank-slab geometry-(dimensions = 28 x 69 cm) | Reflected by concrete (laterally) | 0.0183-0.0188 |
| Experiment LEU-SOL-THERM-019 (6 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Excess nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) = 721-728 (calc.) | The uranyl nitrate solution is contained in a rectangular tank-slab geometry-(dimensions = 28 x 69 cm) | Reflected by polyethylene (laterally) | 0.0189-0.0191 |
| Experiment LEU-SOL-THERM-020 (4 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Excess nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =971-1239 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter = 79 cm) | Reflected by water (thickness > 20 cm) | 0.0125-0.0150 |
| Experiment LEU-SOL-THERM-021 (4 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Excess nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) = 971-1239 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter = 79 cm) | Unreflected | 0.0127-0.0154 |

Table 6-12 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Melt and Dilute Waste Form (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-MET-FAST-001 (4 cases, detailed benchmark models evaluated) | U | 93.4 wt% $^{235}U$ + natural U | HEU metal discs interspersed with natural U | None | Al collar | None | Complex cylindrical geometry comprising interspersed U discs (fuel radius=13.335 cm) | Unreflected | 1.3859-1.4398 |
| Experiment IEU-MET-FAST-002 (1 case) | U | Av. 16.19 wt% $^{235}U$ | HEU metal discs (93.4 wt% $^{235}U$ interspersed with natural U | None | | None | Complex cylindrical geometry comprising interspersed U discs (fuel radius=19.05 cm, column height=31.951 cm) | Reflected by natural U | 1.2784 |
| Experiment IEU-MET-FAST-003 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=15.324 cm) incorporates multiple spherical layers of fissile material | Unreflected | 1.3502 |
| Experiment IEU-MET-FAST-004 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=14 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 2.788 cm) | Reflected by graphite (thickness=3.2 cm) | 1.3071 |

Table 6-12 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Melt and Dilute Waste Form (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-MET-FAST-005 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=13.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 2.686 cm) | Reflected by steel (thickness=8.25 cm) | 1.2852 |
| Experiment IEU-MET-FAST-006 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=13.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 2.1 cm) | Reflected by duralumin (thickness=11.75 cm) | 1.2892 |
| Experiment IEU-MET-FAST-007 (1 case) | U | 10 wt% $^{235}U$ | U metal | None | | None | Complex cylindrical geometry | Reflected by depleted uranium (thickness = 15.24 cm) | 1.2530 |
| Experiment IEU-MET-FAST-008 (1 case) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 13.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 1.4 cm) | Reflected by depleted uranium (thickness=3.25 cm) | 1.3650 |
| Experiment IEU-MET-FAST-009 (1 case) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 12.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 3.445 cm) | Reflected by polyethylene (thickness=5.75 cm) | 1.0140 |

Table 6-12 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Melt and Dilute Waste Form (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-MET-FAST-010 (1 case) | U | 9 wt% $^{235}U$ (average) | U metal | None | Stainless steel, Al | None | Simplified cylindrical geometry (radius = 40.996, height = 76.36) | Reflected by depleted uranium (thickness > 24 cm) | 1.1490 |

Table 6-13 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Enrico Fermi SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-COMP-THERM-001 (29 cases) | U | 29.83 wt% $^{235}U$ | Compound containing UF4 and polytetrafluoroethylene | Cd + B (few cases) | | Polyethylene<br>H/X range = 4.0 – 222 (calc) | Cuboid shapes constructed from small cubes of compound and moderator on two Al platforms | Unreflected or reflected by Paraffin; Cd and/or B sheets placed arround the core in some cases | 0.0455-0.2168 |
| Experiment IEU-COMP-THERM-002 (6 cases) | U | 17 wt% $^{235}U$ | $UO_2$ | Gd, Cd | Stainless steel cladding | Water | Cylindrical hexagonally pitched lattice (pitch =6.8 cm). Cold and hot experiments(~200 °C) performed | Reflected by water | 0.0440-0.0490 |
| Experiment IEU-COMP-THERM-003 (2 cases) | U | 19.9 wt% $^{235}U$ | U-ZrH | B4C in absorber rods | Stainless steel (SS304) cladding; Mo discs; central Zr rod; graphite inserts | ZrH + water<br>H/X = 150.1 | Complex cylindrical arrays of TRIGA fuel rods | The core is radially reflected by graphite and axially by water | 0.0240 |

Table 6-13 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Enrico Fermi SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-COMP-THERM-003 (15 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding<br>Al-lattice plates | Water<br>H/X range(fuel region) = 51-349 | Cylindrical two-zones hexagonally pitched lattices (active fuel length =69.2 cm) Varied pitches. | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0139-0.0467 |
| Experiment HEU-COMP-THERM-004 (4 cases) | U | 88.87 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | Gd or Sm rods | SS-cladding; Al-lattice plates; Bronze caps; Gd or Sm rods | Water<br>H/X = 35 | Cylindrical hexagonally pitched lattices ; double lattices of fuel rods (0.53 cm) and absorber rods(2.75 and 3.67 cm | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0736-0.0756 |
| Experiment HEU-COMP-THERM-005)(1 case) (detailed benchmark model evaluated) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding<br>Al-lattice plates | Water<br>H/X (fuel region) = 23 | Hexagonally pitched array of fuel rod clusters (each containing a hexagonally pitched lattice of fuel rods) hex. pitch =0.52 cm | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0764 |
| Experiment HEU-COMP-THERM-006<br>(3 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding<br>Al-lattice plates | Water<br>H/X range (fuel region) = 30-716 | Cylindrical hexagonally pitched lattice (pitches of 0.56, 1, and 2.113 cm) | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0104-0.0720 |
| Experiment HEU-COMP-THERM-007<br>(3 cases) (detailed benchmark models evaluated) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | ZrH rods, SS-cladding<br>Al-lattice plates | Water +ZrH rods<br>H/X range (fuel region) = 60 - 91 | Cylindrical hexagonally double lattice of fuel rods and ZrH rods. Fixed pitch for fuel rods (0.61 cm) and variable for ZrH rods | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0339-0.0475 |

Table 6-14 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Enrico Fermi Waste Form (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-MET-FAST-001 (4 cases, detailed benchmark models evaluated) | U | 93.4 wt% $^{235}U$ + natural U | HEU metal discs interspersed with natural U | None | Al collar | None | Complex cylindrical geometry comprising interspersed U discs (fuel radius = 13.335 cm) | Unreflected | 1.3859-1.4398 |
| Experiment IEU-MET-FAST-002 (1 case) | U | Av. 16.19 wt% $^{235}U$ | HEU metal discs (93.4 wt% $^{235}U$ interspersed with natural U | None | | None | Complex cylindrical geometry comprising interspersed U discs (fuel radius = 19.05 cm, column height = 31.951 cm) | Reflected by natural U | 1.2784 |
| Experiment IEU-MET-FAST-003 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 15.324 cm) incorporates multiple spherical layers of fissile material | Unreflected | 1.3502 |
| Experiment IEU-MET-FAST-004 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 14 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 2.788 cm) | Reflected by graphite (thickness = 3.2 cm) | 1.3071 |

Table 6-14 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Enrico Fermi Waste Form (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-MET-FAST-005 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 13.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 2.686 cm) | Reflected by steel (thickness = 8.25 cm) | 1.2852 |
| Experiment IEU-MET-FAST-006 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 13.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 2.1 cm) | Reflected by duralumin (thickness = 11.75 cm) | 1.2892 |
| Experiment IEU-MET-FAST-007 (1 case) | U | 10 wt% $^{235}U$ | U metal | None | | None | Complex cylindrical geometry | Reflected by depleted uranium (thickness = 15.24 cm) | 1.2530 |
| Experiment IEU-MET-FAST-008 (1 case) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=13.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 1.4 cm) | Reflected by depleted uranium (thickness=3.25 cm) | 1.3650 |
| Experiment IEU-MET-FAST-009 (1 case) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 12.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 3.445 cm) | Reflected by polyethylene (thickness=5.75 cm) | 1.0140 |

Table 6-14 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Enrico Fermi Waste Form (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-MET-FAST-010 (1 case) | U | 9 wt% $^{235}U$ (average) | U metal | None | Stainless steel, Al | None | Simplified cylindrical geometry (radius = 40.996, height = 76.36) | Reflected by depleted uranium (thickness > 24 cm) | 1.1490 |

Table 6-15 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Enrico Fermi SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-SOL-THERM-001 (4 cases) | U | 20.9 at% $^{235}U$ | Aqueous solution of uranyl sulfate | None | SS core channels; SS coiled pipe | Water in aqueous solution of uranyl sulfate<br>H/X range (fuel region) =217 – 444 (calc) | The uranyl sulfate solution is contained in a cylindrical tank with hemispheric bottom( SS(diameter = 30.5 cm; height=65.6 cm) | Reflected by graphite (except top) | 0.0149-0.0275 |
| Experiment IEU-COMP-THERM-001 (29 cases) | U | 29.83 wt% $^{235}U$ | Compound containing $UF_4$ and polytetrafluoroethylene | Cd + B (few cases) | | Polyethylene<br>H/X range = 4.0 – 222 (calc) | Cuboid shapes constructed from small cubes of compound and moderator on two Al platforms | Unreflected or reflected by Paraffin; Cd and/or B sheets placed around the core in some cases | 0.0455-0.2168 |
| Experiment LEU-SOL-THERM-003 (9 cases) | U | 10 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =770-1437 | The uranyl nitrate solution is contained in a spherical tank (radius=32.95, 43.63, and 59.89 cm) | Unreflected | 0.0114-0.0186 |

Table 6-15 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Enrico Fermi SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment LEU-SOL-THERM-004 (7 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =719-1018 (calc) | The uranyl nitrate solution is contained in a cylindrical tank (diameter=59 cm) | Reflected by water (thickness > 20 cm) | 0.0142-0.0188 |
| Experiment LEU-SOL-THERM-006 (5 cases) | U | 10 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | B4C | Exces nitric acid; absorber rods (SS tubes filed with B4C powder) | Water in aqueous solution of uranyl nitrate<br>H/X (fuel region) =532 | The uranyl nitrate solution is contained in a cylindrical tank (diameter=110 cm) | Reflected by water (laterally and bottom) (thickness > 20 cm) | 0.0245-0.0257 |
| Experiment LEU-SOL-THERM-007 (5 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =709-942 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter=59 cm) | Unreflected | 0.0159-0.0200 |
| Experiment LEU-SOL-THERM-008 (4 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =951-956 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter=59 cm) | Reflected by concrete (laterally) | 0.0152-0.0154 |
| Experiment LEU-SOL-THERM-009 (3 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =934-936 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter =59 cm) | Reflected by borated concrete (laterally) | 0.0155-0.0158 |

Table 6-15 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Enrico Fermi SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment LEU-SOL-THERM-010 (4 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =942-946 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter =59 cm) | Reflected by polyethylene (laterally) | 0.0153-0.0154 |
| Experiment LEU-SOL-THERM-016 (7 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =469-772 (calc.) | The uranyl nitrate solution is contained in a rectangular tank-slab geometry-(dimensions=28x69 cm) | Reflected by water (thickness > 20 cm) | 0.0180-0.0267 |
| Experiment LEU-SOL-THERM-017 (6 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =469-729 (calc.) | The uranyl nitrate solution is contained in a rectangular tank-slab geometry-(dimensions=28x69 cm) | Unreflected | 0.0192-0.0275 |
| Experiment LEU-SOL-THERM-018 (6 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =731-751 (calc.) | The uranyl nitrate solution is contained in a rectangular tank-slab geometry-(dimensions=28x69 cm) | Reflected by concrete (laterally) | 0.0183-0.0188 |
| Experiment LEU-SOL-THERM-019 (6 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =721-728 (calc.) | The uranyl nitrate solution is contained in a rectangular tank-slab geometry-(dimensions=28x69 cm) | Reflected by polyethylene (laterally) | 0.0189-0.0191 |

Table 6-15 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Enrico Fermi SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment LEU-SOL-THERM-020 (4 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =971-1239 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter =79 cm) | Reflected by water (thickness > 20 cm) | 0.0125-0.0150 |
| Experiment LEU-SOL-THERM-021 (4 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =971-1239 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter =79 cm) | Unreflected | 0.0127-0.0154 |

Table 6-16 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Enrico Fermi SNF Waste Form (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-MET-FAST-001 (4 cases, detailed benchmark models evaluated) | U | 93.4 wt% $^{235}U$ + natural U | HEU metal discs interspersed with natural U | None | Al collar | None | Complex cylindrical geometry comprising interspersed U discs (fuel radius = 13.335 cm) | Unreflected | 1.3859-1.4398 |
| Experiment IEU-MET-FAST-002 (1 case) | U | Av. 16.19 wt% $^{235}U$ | HEU metal discs (93.4 wt% $^{235}U$ interspersed with natural U | None | | None | Complex cylindrical geometry comprising interspersed U discs (fuel radius = 19.05 cm, column height = 31.951 cm) | Reflected by natural U | 1.2784 |

Table 6-16 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Enrico Fermi SNF Waste Form (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-MET-FAST-003 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 15.324 cm) incorporates multiple spherical layers of fissile material | Unreflected | 1.3502 |
| Experiment IEU-MET-FAST-004 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 14 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 2.788 cm) | Reflected by graphite (thickness = 3.2 cm) | 1.3071 |
| Experiment IEU-MET-FAST-005 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 13.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 2.686 cm) | Reflected by steel (thickness = 8.25 cm) | 1.2852 |
| Experiment IEU-MET-FAST-006 (1 case) (detailed benchmark model evaluated) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 13.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 2.1 cm) | Reflected by duralumin (thickness = 11.75 cm) | 1.2892 |
| Experiment IEU-MET-FAST-007 (1 case) | U | 10 wt% $^{235}U$ | U metal | None | | None | Complex cylindrical geometry | Reflected by depleted uranium (thickness = 15.24 cm) | 1.2530 |

Table 6-16 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Enrico Fermi SNF Waste Form (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment IEU-MET-FAST-008 (1 case) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 13.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 1.4 cm) | Reflected by depleted uranium (thickness=3.25 cm) | 1.3650 |
| Experiment IEU-MET-FAST-009 (1 case) | U | 36.5 wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius = 12.25 cm) incorporates multiple spherical layers of fissile material and has a central hole (radius = 3.445 cm) | Reflected by polyethylene (thickness = 5.75 cm) | 1.0140 |
| Experiment IEU-MET-FAST-010 (1 case) | U | 9 wt% $^{235}U$ (average) | U metal | None | Stainless steel, Al | None | Simplified cylindrical geometry (radius=40.996, height=76.36) | Reflected by depleted uranium (thickness > 24 cm) | 1.1490 |

Table 6-17 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact FFTF SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment MIX-COMP-THERM-001 (4 cases) | Pu and U | 19.70 wt% Pu in fuel; 11.5 wt% Pu240 in Pu; natural U in UO2 | PuO2 + UO2 pellets in cylindrical fuel rods | None | SS cladding, polypropylene grid plate | Water<br>H/X range (fuel region) = 835-17210 | Fuel pins (diameter=0.5842) on square pitch (various pitches) (fuel length=91.44) | Assembly reflected by water (>15 cm on all sides); fuel pins are reflected axially by Inconel-600 inserts (height ~ 14 cm) | 0.0635-0.1717 |
| Experiment MIX-COMP-THERM-003 (6 cases) | Pu and U | natural UO2 6.6wt% $PuO_2$ in MOX | PuO2 + UO2 pellets in cylindrical fuel rods | None | Zircaloy-4 cladding,<br>Al grid plates | partially moderated with water | fuel pins on square pitch (various pitches between 1.32 and 2.64 cm) | Assembly reflected by water or by borated water (Case 3 only) (thickness > 30 cm) | 0.08-0.2294 |
| Experiment HEU-COMP-THERM-003 (15 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding<br>Al-lattice plates | Water<br>H/X range (fuel region) = 51-349 | Cylindrical two-zones hexagonally pitched lattices (active fuel length =69.2 cm) Varied pitches | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0139-0.0467 |
| Experiment HEU-COMP-THERM-004 (4 cases) | U | 88.87 at% $^{235}U$ | $UO_2$ +Cu (cross shaped fuel rods) | Gd or Sm | SS-cladding,<br>Al-lattice plates, Gd or Sm Bronze Caps | Water<br>H/X range (fuel region) = 35 | double lattices of two fuel rod lattice and absorber rod lattice – both hexagonal pitch (fuel rod pitch=5.3 mm, absorber rod pitch =27.5 and 36.7 mm) | Assembly reflected by water (thickness >20 cm on top and >50 cm radially and on bottom) | 0.0736-0.0756 |
| Experiment HEU-COMP-THERM-005 (1 case)(detailed model evaluated) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding<br>Al-lattice plates | Water<br>H/X (fuel region) = 23 | Hexagonally pitched array of fuel rod clusters (each containing a hexagonally pitched lattice of fuel rods-hex. Lattice pitch = 0.52 cm) | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0764 |

Table 6-17 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact FFTF SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-COMP-THERM-006 (3 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding Al-lattice plates | Water H/X range (fuel region) = 30-716 | Cylindrical hexagonally pitched lattice (pitches of 0.56, 1, and 2.113 cm) | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0104-0.0720 |
| Experiment HEU-COMP-THERM-007 (3 cases) (detailed benchmark models evaluated) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | ZrH rods, SS-cladding Al-lattice plates | Water +ZrH rods H/X range (fuel region) = 60 - 91 | Cylindrical hexagonally double lattice of fuel rods and ZrH rods. Fixed pitch for fuel rods (0.61 cm) and variable for ZrH rods | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0339-0.0475 |
| Experiment HEU-COMP-THERM-008 (2 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | B | $B_4C$ rods, SS-cladding Al-lattice plates | Water H/X = 25 | Cylindrical hexagonally double lattice of fuel rods and $B_4C$ rods. Fixed pitch for fuel rods (0.53 cm) and variable for $B_4C$ rods (2.12 and 2.65 cm) | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0882-0.0922 |
| Experiment HEU-COMP-THERM-010 (21cases) | U | 62.4 wt% $^{235}U$ | $UO_2$+BeO | B as boric acid (few cases) | Hastelloy X-280 cladding | Water (15 cases); water +uranyl nitrate and/or boric acid (few cases) H/X range =36 to 302 | Square or cylindrical assemblies with square/hexagona pitched lattices (lattice pitch between 1.24 to 3 cm; active fuel length=193 cm) | Some of the configurations are fully reflected by water; some are partially flooded and are not reflected at top; in cases with uranyl nitrate and boric acid solutions the assembly is reflected by solution | 0.0234-0.0797 |

Table 6-17 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact FFTF SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment MIX-COMP-THERM-004 (11 cases) | Pu+U | 3.01 wt% $PuO_2$, (68.18 wt% $^{239}Pu$); natural U | Sintered pellets in rods | None | Zircaloy-2 cladding | Water<br>H/Pu range = 402-921<br>Water/Fuel volume ratio (in lattice) = 2.42-5.55 | Square arrays of cylindrical fuel rods with various spacing. Lattice pitch = 1.825 cm to 2.474 cm) | Assembly partially reflected by water (thickness > 30 cm) | 0.0747-0.1218 |
| Experiment MIX-COMP-THERM-010 (MIX-MISC-THERM-001) (11 cases) | Pu+U | in fuel pellets mixture - 17.13 wt% $^{239}Pu$, 0.49 wt% $^{235}U$ | Pu-U-sintered pellets, Pu-U nitrate solution | Gd (in solution for 5 cases); B and Cd as impurities | Nitrate, Gd (in nitrate solution), SS cladding and guide tubes, polyethylene at bottom of rods | Water in solution | Cylindrical geometry of MOX fuel pins in square lattice (pitch = 1.4 cm) in aqueous Pu-U-nitrate solution in boiler-tube type assembly | Assembly partially reflected by water (thickness > 20 cm) | 0.033-0.153 |

Table 6-18 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact FFTF SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-MET-FAST-003 (5 cases) | Pu | 93.56 wt% $^{239}Pu$ | Pu metal as right circular cylinder | None | Al container with steel lid around Pu metal; Al tube spacers; Al support tube | None | Pu parts arranged in 2x2x2, 3x3x3 or 2x2x1 arrays with various vertical and horizontal spacing | Array reflected on one side with polyethylene (for 2 cases) | 1.81-1.905 |
| Experiment PU-MET-FAST-004 (9 cases) | Pu | 93.56 wt% $^{239}Pu$ | Pu metal as right circular cylinder | None | Al container with steel lid around Pu metal; Al tube spacers; Al support tube | None | Pu parts arranged in 4x4x4, 4x4x1, 3x3x3, or 2x2x2 arrays with various vertical and horizontal spacing | None | 1.900-1.912 |

Table 6-18 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact FFTF SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-MET-FAST-016 (6 cases) | Pu | 93.56 wt% $^{239}$Pu | Pu metal as right circular cylinder | None | Al container with mild steel lid around Pu metal; steel can around Al container, perforated Al sleeves | Water<br>H/$^{239}$Pu range = 14.2-17.3 | Pu parts arranged in 3x3x3 arrays with various vertical and horizontal spacing | Array reflected by water (Case 1 is only partially reflected by water) | 1.244-1.281 |
| Experiment PU-MET-FAST-017 (5 cases) | Pu | 93.56 wt% $^{239}$Pu | Pu metal as right circular cylinder | None | Al container with steel lid around Pu metal; Al tube spacers; Al support tube | Vertical and lateral moderation provided by mock-explosive moderator disks and jackets (generic C, H, N, and O)<br>H/$^{239}$Pu range = 1.08-5.80 | Pu parts arranged in 4x4x4 arrays with various vertical and horizontal spacing | None | 1.220-1.589 |
| Experiment PU-MET-FAST-037 (16 cases) | Pu | 93.56 wt% $^{239}$Pu | Pu metal as right circular cylinder | None | Al container with mild steel lid around Pu metal; steel can around Al container, Al plates forming support trays | Water<br>H/$^{239}$Pu range = 3.86-14.48 | Pu parts arranged in 2x2xN (N=2,3,4, or 5) arrays with various vertical and horizontal spacing | Arrays are partially or fully reflected by water | 1.2816-1.5348 |

Table 6-18 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact FFTF SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-COMP-MIXED-001 (5 cases) | Pu | 2.2-18.35 wt % $^{240}$Pu; 19.846-65.84 wt% $^{239}$Pu in cube (Cases 1-3,5); 91.16 wt% $^{239}$Pu in Pu (Case 4) | $PuO_2$ cubes with polystyrene | None | Various materials as contamination control (shrink wrap (Case 1), tape (Cases 2, 5), Al paint with rubberized plastic (Case 3-4)); Cladding contained C, H, Cl (Cases 2, 5); Cladding contained C, H, O (Case 1) | Polystyrene H/Pu range = 0.04-49.6 | Cubes stacked into various sized slab geometries | None | 0.099-1.702 |
| Experiment PU-COMP-MIXED-002 (29 cases | Pu | 2.2-18.35 wt $^{240}$Pu; 19.846-65.84 wt% $^{239}$Pu in cube (all cases except 17-22); 91.16 wt% $^{239}$Pu in Pu (Cases 17-22) | $PuO_2$ cubes with polystyrene | None | Various materials as contamination control (shrink wrap (Cases 1-5), tape (Cases 6-9,23-29), Al paint with rubberized plastic (Cases 10-22)); Cladding contained C, H, Cl (Cases 6-9, 23-29); Cladding contained C, H, O (Cases 1-5 | Polystyrene H/Pu range = 0.04-49.6 | Cubes stacked into various sized slab geometries | Assembly reflected by 6-in thick Plexiglas. | 0.077-1.046 |

Table 6-19 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment MIX-SOL-THERM-001 (12 cases) | Pu and U | 7.94 -8.34 wt% $^{240}$Pu in Pu (91.12-91.57 wt% $^{239}$Pu), 0.70-2.29 wt% $^{235}$U in U | Pu-U nitrate solution | B in concrete, Cd in inserts | Nitric acid, concrete with B4C insert, Cd-covered polyethylene inserts | Water H/$^{239}$Pu range (annular tank) = 125 – 569 | Solution contained in annular cylinder. Center of cylinder contained solution and/or inserts of various materials. | Assembly reflected by water | 0.014-0.046 |
| Experiment MIX-SOL-THERM-002 (3 cases) | Pu and U | 8.31 wt% $^{240}$Pu in Pu (91.10 wt% $^{239}$Pu), 0.44 - 0.71 wt% $^{235}$U in U | Pu-U nitrate solution | None | Nitric acid, | Water H/$^{239}$Pu range (annular tank) = 2317 - 2435 | Solution contained in cylinder. | Assembly reflected by water | 0.0039-0.0061 |
| Experiment MIX-SOL-THERM-003 (10 cases) | Pu and U | 5.63 wt% $^{240}$Pu in Pu (93.95 wt% $^{239}$Pu), 0.72 wt% $^{235}$U in U | Pu-U nitrate solution | None | Nitric acid, | Water H/$^{239}$Pu range = 239 - 1556 | Solution contained in cylindrical vessels of varying radii. | Assembly reflected by water and polyethylene (on top) | 0.0065-0.0320 |
| Experiment MIX-SOL-THERM-004 (9 cases) | Pu and U | 8.31 wt% $^{240}$Pu in Pu (91.12 wt% $^{239}$Pu), 0.56 wt% $^{235}$U in U | Pu-U nitrate solution | None | Nitric acid | Water H/$^{239}$Pu range = 126 – 664 | Solution contained in cylindrical tank with inner diameter = 35.39 cm. | Assembly reflected by SS walls of cylinder and water (Cases 2,5,8) or concrete (Cases 3,4,9). Case 1, 6, and 7 reflected only by SS walls of cylinder. | 0.0117-0.0589 |
| Experiment MIX-SOL-THERM-005 (7 cases) | Pu and U | 8.31 wt% $^{240}$Pu in Pu (91.12 wt% $^{239}$Pu), 0.564 wt% $^{235}$U in U | Pu-U nitrate solution | None | nitric acid | Water H/X range = 134-664 | slab tank with width varying from 7.6-22.8 cm | Assembly reflected by water for 4 cases (3 cases were not reflected) | 0.012-0.059 |

Table 6-19 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-SOL-THERM-001 (6 cases) | Pu | 4.57 wt% $^{240}$Pu in Pu (95.12 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid | Water H/$^{239}$Pu range = 91–371 | solution in 11.5-inch sphere | Assembly reflected by 0.05-in thick SS sphere and >36 cm of water | 0.0126-0.0488 |
| Experiment PU-SOL-THERM-002 (7 cases) | Pu | 3.12 wt% $^{240}$Pu in Pu (96.88 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid | Water H/X range = 299.3-508.0 | solution in 12-inch sphere | Assembly reflected by 0.05-in thick SS sphere and >12-in of water | 0.009-0.014 |
| Experiment PU-SOL-THERM-003 (8 cases) | Pu | 1.76 or 3.12 wt% $^{240}$Pu in Pu; $^{239}$Pu is balance | Pu nitrate solution | None | nitric acid | Water H/$^{239}$Pu range = 563-788 | solution in 13-inch sphere | Assembly reflected by SS or Al sphere (Cases 7-8) and >11.5 in. water | 0.0063-0.0086 |
| Experiment PU-SOL-THERM-004 (13 cases) | Pu | 0.54-3.43 wt% $^{240}$Pu in Pu; $^{239}$Pu is balance | Pu nitrate solution | None | nitric acid | Water H/$^{239}$Pu range = 592-903 | solution in 14-inch sphere | Assembly reflected by SS sphere and >11 in. water | 0.0052-0.0085 |
| Experiment PU-SOL-THERM-005 (9 cases) | Pu | 4.05 or 4.40 wt% $^{240}$Pu in Pu; $^{239}$Pu is balance | Pu nitrate solution | None | nitric acid | Water H/$^{239}$Pu range = 581-903 | solution in 14-inch sphere | Assembly reflected by SS sphere and >11 in. water | 0.0057-0.0084 |
| Experiment PU-SOL-THERM-006 (3 cases) | Pu | 3.12 wt% $^{240}$Pu in Pu (96.88 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid | Water H/X range = 910.9-1028.2 | solution in 15-inch sphere | Assembly reflected by 0.05-in thick SS sphere and >10.5-in of water | 0.005 |

Table 6-19 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-SOL-THERM-007 (8 cases) | Pu | 4.57 wt% $^{240}$Pu in Pu (95.12 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid | Water H/$^{239}$Pu range = 110-285 | 11.5-inch sphere partially filled with Pu nitrate solution | Assembly reflected by 0.05-in thick SS sphere and >21 cm of water | 0.0171-0.0410 |
| Experiment PU-SOL-THERM-008 (14 cases) | Pu | 4.57 wt% $^{240}$Pu in Pu (95.12 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid | Water H/Pu = 85-870 | solution in 14-inch sphere (completely or nearly full critical volumes) | Assembly reflected by SS sphere and by a full or half hemispherical concrete reflector of various thicknesses | 0.006-0.054 |
| Experiment PU-SOL-THERM-009 (3 cases) | Pu | 2.52 wt% $^{240}$Pu in Pu (97.39 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid | Water H/$^{239}$Pu range = 2648-2803 | solution in 48-inch sphere (filled or partially filled critical volume) | Assembly reflected by Al sphere only | 0.0025-0.0029 |
| Experiment PU-SOL-THERM-010 (14 cases) | Pu | 2.9 wt% $^{240}$Pu in Pu; $^{239}$Pu is balance | Pu nitrate solution | None | nitric acid | Water H/$^{239}$Pu range = 267-850 | solution in cylinders with diameter between 9 and 12 in. | Assembly reflected by SS cylinder and > 12 in water | 0.0060-0.0166 |
| Experiment PU-SOL-THERM-011 (12 cases) | Pu | 4.2 wt% $^{240}$Pu in Pu; $^{239}$Pu is balance | Pu nitrate solution | Cd in 18 in sphere | nitric acid | Water H/$^{239}$Pu range = 575-1208 | solution in 16 or 18-inch diameter spheres | Assembly reflected by SS sphere wall ; 18-in sphere had Cd layer | 0.0050-0.0096 |

Table 6-19 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-SOL-THERM-012 (23 cases) | Pu | 18.88 wt% $^{240}$Pu in Pu (74.20 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid | Water H/Pu range = 228.6-1903.6 | Solution in right parallelepiped tank (130cm x 130 cm x 100 cm) | Assembly reflected by 0.5 cm SS walls and by water on 5 or 6 sides or not reflected (10 cases) | 0.0030.022 |
| Experiment PU-SOL-THERM-013 (21cases) | Pu | 4.23 wt% $^{240}$Pu in Pu (95.47 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid, SS cylinder walls | Water in solution H/X = 210.1) | Solution-filled 256-mm diameter cylinders in array using between 2 and 9 cylinders in line or in triangular, square, or hexagonal geometry | Unreflected | 0.0243-0.0251 |
| Experiment PU-SOL-THERM-014 (35 cases) | Pu | 4.23 wt% $^{240}$Pu in Pu (95.47 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid, SS cylinder walls | Water in solution (H/X = 210.1) | Solution-filled 300-mm diameter cylinders in array using between 2 and 4 cylinders in line or in triangular, square, or rectangular geometry | Unreflected | 0.024 - 0.025 |
| Experiment PU-SOL-THERM-015 (17 cases) | Pu | 4.23 wt% $^{240}$Pu in Pu (95.47 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid, SS cylinder walls | Water in solution (H/X = 155.2) | Solution-filled 300-mm diameter cylinders in array using between 2 and 4 cylinders in line or in triangular or square geometry | Unreflected | 0.033 |
| Experiment PU-SOL-THERM-016 (11 cases) | Pu | 4.23 wt% $^{240}$Pu in Pu (95.47 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid; SS cylinder walls | Water in solution (H/X = 155.2 or 210.1) | Two solution-filled 256-mm and 300-mm diameter cylinders with different concentrations | Unreflected | 0.024-0.033 |

Table 6-19 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-SOL-THERM-017 (18 cases) | Pu | 4.23 wt% $^{240}$Pu in Pu (95.47 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid, SS cylinder walls | Water in solution (H/X = 210.1) | Two solution-filled 256-mm and 300-mm diameter cylinders with different concentrations | Unreflected | 0.024 - 0.025 |
| Experiment PU-SOL-THERM-020 (8 cases) | Pu | 4.57 wt% $^{240}$Pu in Pu (95.12 wt% $^{239}$Pu) | Pu nitrate solution | Cd in reflector interface (few cases) | nitric acid | Water in solution (H/X = 341.2-758.9) | Solution in 14-in diameter sphere (full or nearly full critical volume) | Assembly reflected by SS sphere and by water or by water + Cd interface (Cases 8,9,14,15) | 0.007-0.015 |
| Experiment PU-SOL-THERM-021 (6 cases) | Pu | 4.57 wt% $^{240}$Pu in Pu (95.12 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid | Water in solution (H/X = 124.8 – 1101.7) | Solution in 15.2-in diameter sphere (full or nearly full critical volume) | Assembly reflected by SS sphere and by water (6 cases had no water reflection) | 0.005-0.040 |
| Experiment PU-SOL-THERM-022 (17 cases) | Pu | 18.88 wt% $^{240}$Pu in Pu (74.2 wt% $^{239}$Pu) | Pu nitrate solution | Cd (in reflector interface) (few cases) | nitric acid; SS in inner annulus | Water in solution (H/X = 138.4 - 872.2) | Solution in annular cylinder tank (50cm x 20 cm diameter) | Assembly reflected by SS tank walls and by water on the side and bottom (air or paraffin+Cd (Cases 10-17) in central region) | 0.006-0.034 |
| Experiment PU-SOL-THERM-023 (3 cases) | Pu | 4.23 wt% $^{240}$Pu in Pu (outer tank) 33.89 or 4.23 wt% $^{240}$Pu (inner tank); $^{239}$Pu is balance | Pu nitrate solutions | None | nitric acid SS inner tank wall | Water in solution H/X = 259.18 or 468.73 | Solutions in two concentric cylinder tanks | Assembly reflected by SS tank walls and by water on bottom and radially outside outer tank | 0.0212-0.0234 |

Table 6-19 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-SOL-THERM-024 (23 cases) | Pu | 18.40 or 23.185 wt% $^{240}$Pu in Pu (75.92 or 71.821 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid | Water in solution H/X = 44.9 – 367.3 | Slab tank with varying thickness (16.0 - 22.6 cm) and solution height (36 – 96 cm) | Assembly reflected by SS tank walls and 1-in thick Plexiglas | 0.014-0.105 |
| Experiment PU-SOL-THERM-025 (44 cases) | Pu | 4.671, 18.40 or 23.185 wt% $^{240}$Pu in Pu (95.059, 75.92 or 71.821 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid | Water in solution H/X = 44.9-424.3 | Slab tank with varying thickness (11.4 – 22.6 cm) and solution height (28.0 – 100.0 cm) | Assembly reflected by SS tank walls and by water on the sides and bottom | 0.011-0.097 |
| Experiment PU-SOL-THERM-026 (21 cases) | Pu | 4.671, 18.40 or 23.185 wt% $^{240}$Pu in Pu (95.059, 75.92 or 71.821 wt% $^{239}$Pu) | Pu nitrate solution | None | nitric acid | Water in solution H/$^{239}$Pu = 115.4-484.5 | Slab tank with varying thickness (16.0 – 22.6 cm) and solution height (42.9 – 97.4 cm) | Assembly reflected by SS tank walls | 0.013-0.060 |
| Experiment MIX-MET-FAST-011 (4 cases) | Pu and U | Avg. 87.56 - 93.40 at% $^{239}$Pu in Pu; 18.43 to 55.4 at% $^{235}$U in U (depending on configuration) | Mixture of metallic plates: PuUMo alloy, PuAl alloy and U metal plates | None | SS, Zr, | None | Complex cylindrical geometry (fissile core radius ~18,96 cm, half-height ~20.96 cm) | Assembly reflected by graphite (thickness = 10.16 cm axially and between 9.95 cm and 12.79 cm radially) | 0.962-1.075 |

Table 6-20 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment MIX-MET-MIXED-001 (MIX-MET-INTER-002) (1 case) | Pu and U | 4.59 wt% $^{240}$Pu in Pu (95.21 wt% $^{239}$Pu), 95.98 wt% $^{235}$U in U | Pu and HEU metal discs | None | polyethylene, SS cover around Pu discs | polyethylene H/$^{239}$Pu=3.57 H/$^{235}$U=1.71 H/Fissile=2.31 | Cylinder of alternating layers of Pu, HEU, and polyethylene, divided by a gap into two parts (gap=1.12 cm thick) | Unreflected | 0.777 |
| Experiment MIX-MET-FAST-001 (1 case) | Pu and U | 4.90 at% $^{240}$Pu in Pu (94.79 at% $^{239}$Pu), 93.2 wt% $^{235}$U in U | delta-phase Pu alloy and HEU metal | None | Ni covering Pu, 1 wt% Gallium in Pu alloy | None | Pu sphere (radius = 5.4019 cm) | Surrounded by HEU (1.6637 cm thick) | 1.807 |
| Experiment MIX-MET-FAST-002 (3 cases) | Pu and U | 2.34-16.1 at% $^{240}$Pu in Pu, (97.56-80.47 at% $^{239}$Pu) 93.3 wt% $^{235}$U in U | alpha-phase Pu alloy and HEU metal | None | Ni covering Pu | None | Pu sphere (radius = 2.7051 cm) surrounded by HEU (varying thickness) | Core reflected by natural uranium (varying thickness) | 1.700-1.713 |
| Experiment MIX-MET-FAST-003 (1 case) | Pu and U | 9.10-9.56 wt% $^{240}$Pu in Pu, (89.19-89.92 wt% $^{239}$Pu) 89.56-89.66 wt% $^{235}$U in U | delta-phase Pu alloy and HEU metal | None | Ni plating around Pu, Cu and Ni plating around HEU, Duralumin tube inserted in one side of HEU, Ga in Pu alloy | None | Hemisperhical shells of Pu (split in center, critical gap width=1.225 cm). Central region was void. | Surrounded by hemispherical shells of HEU | 1.774 |
| Experiment MIX-MET-FAST-004 (2 cases) | Pu and U | 9.10-9.21 wt% $^{240}$Pu in Pu, (89.76-89.92 wt% $^{239}$Pu) 89.55-89.67 wt% $^{235}$U in U | delta-phase Pu alloy and HEU metal | None | Ni plating around Pu, Cu and Ni plating around HEU, Ga in Pu alloy | None | Hemisperhical shells of Pu surrounded by hemispherical shells of HEU (split in center, critical gap width=0.40 or 0.75 cm). Central region was void. | Assembly reflected by Beryllium or Beryllium oxide | 1.541 - 1.547 |

Table 6-20 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment MIX-MET-FAST-005 (1 case) | Pu and U | 9.10-9.56 wt% $^{240}$Pu in Pu, (89.19-90.08 wt% $^{239}$Pu) 89.56-89.66 wt% $^{235}$U in U | delta-phase Pu alloy and HEU metal | None | Ni plating around Pu, Cu and Ni plating around HEU, Ga in Pu alloy | None | Hemisperhical shells of Pu surrounded by hemispherical shells of HEU (split in center, critical gap width=0.40 cm). Central region was void. | Assembly reflected by aluminum | 1.740 |
| Experiment MIX-MET-FAST-009 (1 case) | Pu and U | 1.8 wt% $^{240}$Pu in Pu, (98.2 wt% $^{239}$Pu) 90.6 wt% $^{235}$U in U | delta-phase Pu alloy and HEU metal | None | Ni plating around Pu, Cu and Ni plating around HEU, Ga in Pu metal | None | Pu spherical shells (total outer radius = 6.00 cm) surrounded by HEU shell (0.75 cm thick). Central region was void (radius = 1.084 cm) | None | 1.851 |
| Experiment MIX-MET-FAST-010 (1 case) | Pu and U | 1.8 wt% $^{240}$Pu in Pu, (98.2 wt% $^{239}$Pu) 90.6 wt% $^{235}$U in U | delta-phase Pu alloy and HEU metal | None | Ni plating around Pu, Cu and Ni plating around HEU, Ga in Pu metal | None | Pu spherical shells (total outer radius = 4.66 cm) surrounded by HEU shell (2.89 cm thick). Central region was void (radius = 1.222 cm) | None | 1.723 |
| Experiment MIX-MET-FAST-011 (4 cases) | Pu and U | Avg. 87.56-93.40 at% $^{239}$Pu in Pu; 18.43 to 55.4 at% $^{235}$U in U (depending on configuration) | Mixture of metallic plates; PuUMo alloy, PuAl alloy, and U metal plates | None | SS, Zr | None | Complex cylindrical geometry (fissile core radius ~ 18.96 cm, half-height ~20.96 cm) | Assembly reflected by graphite (thickness = 10.16 cm axially and between 9.95 cm and 12.79 cm radially) | 0.962-1.075 |
| Experiment PU-MET-FAST-001 (1 case) | Pu | 4.5 at% $^{240}$Pu in Pu, (95.2 wt% $^{239}$Pu) | delta-phase Pu metal | None | Ni plating around Pu, 1.02 wt% Ga in Pu metal | None | Pu sphere with radius = 6.3849 cm | None | 1.915 |

Table 6-20 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-MET-FAST-002 (1 case) | Pu | 20.1 at% $^{240}$Pu in Pu, (76.4 wt% $^{239}$Pu) | delta-phase Pu metal | None | Ni plating around Pu, 1.01 wt% Ga in Pu metal | None | Pu sphere with radius = 6.6595 cm | None | 1.915 |
| Experiment PU-MET-FAST-005 (1 case) | Pu | 4.90 at% $^{240}$Pu in Pu (94.79 at% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, 1 wt% Ga in Pu alloy | None | Pu sphere with radius = 5.0419 cm | Pu sphere reflected by 4.699 cm of tungsten alloy | 1.661 |
| Experiment PU-MET-FAST-006 (1 case) | Pu | 4.80 wt% $^{240}$Pu in Pu alloy, (93.8 wt% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, 1.1 wt% Ga in Pu alloy | None | Pu sphere with radius = 4.5332 cm | Sphere reflected by natural uranium (19.6088 cm thick) | 1.901 |
| Experiment PU-MET-FAST-009 (1 case) | Pu | 4.9 wt% $^{240}$Pu in Pu, (94.8 wt% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, 1.0 wt% Ga in Pu alloy | None | Pu sphere with radius = 5.5118 cm | Sphere reflected by aluminum (7.9248 cm thick) | 1.803 |
| Experiment PU-MET-FAST-010 (1 case) | Pu | 4.9 at% $^{240}$Pu in Pu, (94.79 at% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, 1.0 wt% Ga in Pu alloy | None | Pu sphere with radius = 5.0419 cm | Sphere reflected by natural uranium (4.1275 cm thick) | 1.901 |
| Experiment PU-MET-FAST-011 (1 case) | Pu | 5.18 at% $^{240}$Pu in Pu, (94.5 at% $^{239}$Pu) | alpha-phase Pu | None | Cu and Lucite covering Pu | None | Pu sphere with radius = 4.1217 cm | Sphere reflected by water (25.4 cm thick) | 1.518 |
| Experiment PU-MET-FAST-018 (1 case) | Pu | 4.9 at% $^{240}$Pu in Pu, (94.79 at% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, 1.0 wt% Ga in Pu alloy | None | Pu sphere with radius = 5.0419 cm | Sphere reflected by beryllium (3.6881 cm thick) | 1.666 |

Table 6-20 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-MET-FAST-019 (1 case) | Pu | 9.1-9.6 wt% $^{240}$Pu in Pu, (89.2 - 90.1 wt% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, 1.55-1.74 wt% Ga in Pu alloy | None | Pu hemispheres with outer radius = 5.35 cm. Central region was void (radius = 1.4 cm) | Sphere reflected by beryllium (5.65 cm thick) | 1.620 |
| Experiment PU-MET-FAST-020 (1 case) | Pu | 9.3 wt% $^{240}$Pu in Pu, (89.66 wt% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, 1.55-1.74 wt% Ga in Pu alloy | None | Pu concentric spherical shells with outer radius = 5.35 cm. Central region was void (radius = 1.4 cm) | Sphere reflected by depleted uranium (7.65 cm thick) | 1.8895 |
| Experiment PU-MET-FAST-021 (2 cases) | Pu | 4.59 wt% $^{240}$Pu in Pu, (95.21 wt% $^{239}$Pu) | Pu metal discs | None | Steel coating over Pu discs | None | Pu discs layered into two cylinders separated by gap (0.01 or 0.24 cm) | Cylinders reflected on ends by Be or BeO | 1.661- |
| Experiment PU-MET-FAST-022 (1 case) (detailed model evaluated) | Pu | 1.8 wt% $^{240}$Pu in Pu, (98.2 wt% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, 1.69 wt% Ga in Pu alloy | None | Pu concentric spherical shells with outer radius =6.67 cm. Central region was void (radius = 1.4 cm) | None | 1.8914 |
| Experiment PU-MET-FAST-023 (1 case) (detailed model evaluated) | Pu | 1.8 wt% $^{240}$Pu in Pu, (98.2 wt% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, 1.69 wt% Ga in Pu alloy | None | Pu concentric spherical shells with outer radius =6.00 cm. Central region was void (radius = 1.715 cm) | Sphere reflected by graphite (2.35 cm thick) | 1.7988 |
| Experiment PU-MET-FAST-024 (1 case) (detailed model evaluated) | Pu | 1.8 wt% $^{240}$Pu in Pu, (98.2 wt% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, 1.69 wt% Ga in Pu alloy | None | Pu concentric spherical shells with outer radius =6.00 cm. Central void region homogenized in the model | Sphere reflected by polyethylene(1.55 cm thick) | 1.7418 |

Table 6-20 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-MET-FAST-025 (1 case) (detailed model evaluated) | Pu | 1.8 wt% $^{240}$Pu in Pu, (98.2 wt% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, Avg. 1.69 wt% Ga in Pu alloy | None | Pu concentric spherical shells with outer radius =6.00 cm. Central void region homogenized in the model | Sphere reflected by steel(1.55 cm thick) | 1.8293 |
| Experiment PU-MET-FAST-026 (1 case) (detailed model evaluated) | Pu | 1.8 wt% $^{240}$Pu in Pu, (98.2 wt% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, Avg. 1.69 wt% Ga in Pu alloy | None | Pu concentric spherical shells with outer radius =5.35 cm. Central region was void (radius = 0.77 cm in the model) | Sphere reflected by steel (11.9 cm thick) | 1.7250 |
| Experiment PU-MET-FAST-027 (1 case) (detailed model evaluated) | Pu | 9.3 wt% $^{240}$Pu in Pu, (89.7 wt% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, Avg. 1.69 wt% Ga in Pu alloy | None | Pu concentric spherical shells with outer radius =5.35 cm. Central region was void (radius = 0.985 cm in the model) | Sphere reflected by polyethylene (5.58 cm thick) | 1.4726 |
| Experiment PU-MET-FAST-028 (1 case) (detailed model evaluated) | Pu | 9.3 wt% $^{240}$Pu in Pu, (89.7 wt% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, Avg. 1.69 wt% Ga in Pu alloy | None | Pu concentric spheric shells with outer radius =5.35 cm. Central region shape was complex. The model homogenizes it. | Sphere reflected by steel (19.65 cm thick) | 1.7100 |
| Experiment PU-MET-FAST-029 (1 case) (detailed model evaluated) | Pu | 9.9 wt% $^{240}$Pu in Pu, (88.6 wt% $^{239}$Pu) | alfa-phase Pu alloy | None | Lacquer coating (74.8 wt% C, 5 wt% H) | None | Pu concentric spherical shells with outer radius =5.35 cm. Central region was void (radius =0.8 cm in the model) | None | 1.9154 |

Table 6-20 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-MET-FAST-030 (1 case) (detailed model evaluated) | Pu | 9.7 wt% $^{240}$Pu in Pu, (88.9 wt% $^{239}$Pu) | alfa-phase Pu alloy | None | Varnish-paint coating (74.8 wt% C, 5 wt% H) | None | Pu concentric spherical shells with outer radius =4.66 cm. | Sphere reflected by graphite (4.49 cm thick) | 1.8138 |
| Experiment PU-MET-FAST-031 (1 case) (detailed model evaluated) | Pu | 9.7 wt% $^{240}$Pu in Pu, (88.9 wt% $^{239}$Pu) | alfa-phase Pu alloy | None | Varnish-paint coating (74.8 wt% C, 5 wt% H) | None | Pu concentric spheric shells with outer radius =4.66 cm. Central region shape was complex. Model simplified to inner void radius of 1.165 cm. | Sphere reflected by polyethylene (3.69 cm thick) | 1.6057 |
| Experiment PU-MET-FAST-032 (1 case) (detailed model evaluated) | Pu | 9.7 wt% $^{240}$Pu in Pu, (88.9 wt% $^{239}$Pu) | alfa-phase Pu alloy | None | Varnish-paint coating (74.8 wt% C, 5 wt% H) | None | Pu concentric spherical shells with outer radius =4.66 cm. Central region was void (radius = 0.7 cm in the model) | Sphere reflected by steel (4.49 cm thick) | 1.8113 |
| Experiment PU-MET-FAST-035 (1 case) | Pu | 1.8 at% $^{240}$Pu in Pu, (98.2 at% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, Ga in Pu alloy | None | Pu hemispheres with outer radius = 6.0 cm. Central region was void (radius = 1.2 cm) | Sphere reflected by lead (3.15 cm thick) | 1.824 |
| Experiment PU-MET-FAST-036 (1 case) | Pu | 1.8 at% $^{240}$Pu in Pu, (98.2 at% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, Ga in Pu alloy | None | Pu hemispheres with outer radius = 6.0 cm. Central region was void (radius = 1.16 cm in the model) | Sphere reflected by Cadmium (0.05 cm thick) and polyethylene (2.3 cm thick) | 1.716 |

Table 6-20 Area and Range of Applicability for WP Configurations Containing Degraded FFTF SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment PU-MET-FAST-039 (1 case) | Pu | 1.8 at% $^{240}$Pu in Pu, (98.2 at% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, Ga in Pu alloy | None | Pu hemispheres with outer radius = 6.75 cm. Central region was void (radius = 3.112 cm in the model) | Sphere reflected by Duralumin (4.25 cm thick) | 1.811 |
| Experiment PU-MET-FAST-040 (1 case) | Pu | 1.8 at% $^{240}$Pu in Pu, (98.2 at% $^{239}$Pu) | delta-phase Pu alloy | None | Ni covering Pu, Ga in Pu alloy | None | Pu hemispheres with outer radius = 6.00 cm. Central region was void (radius = 1.531 cm in the model) | Sphere reflected by Copper (1.6 cm thick) | 1.7940 |
| Experiment PU-MET-FAST-041 (1 case) | Pu | 10.0 wt% $^{240}$Pu in Pu, (88.3 wt% $^{239}$Pu) | alpha-phase Pu alloy | None | Varnish coating around Pu (74.8 wt% C, 5 wt% H) | None | Pu hemispheres with outer radius = 4.02 cm. Central region was void (radius = 0.901 cm in the model) | Sphere reflected by depleted uranium (0.44 wt% $^{235}$U, 20.98 cm thick) | 1.947 |
| Experiment PU-MET-MIXED-001 (5 cases) | Pu | 4.58 wt% $^{240}$Pu in Pu, (95.17 wt% $^{239}$Pu) | Pu metal pellets | B in boron carbide rods | SS cans (contain Pu metal), Al clad rods, Silicon dioxide polyethylene, boron carbide in control-rod fuel tubes, Al markers in tubes, 1.65 wt% Ga in Pu metal | polyethylene | Fuel / control / moderator / reflector rods arranged in hexagonal lattice with rod pitch = 5.1 cm | Fuel rods reflected by SiO2. Outer reflector materials included polyethylene, depleted uranium, and boron carbide | 0.216-0.451 |

Table 6-21 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Fort Saint Vrain HTGR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-COMP-THERM-002 (25 cases) | U | 93.15 wt% $^{235}U$ | Uranium dicarbide | None | Al tubes; Plexiglas supports | Graphite and Water<br>C/X 87 – 88.9 | Various arrays (triangular or square pitched lattices) of Al tubed or bare fuel elements (hexagonal graphite blocks containing uranium dicarbide beads) surrounded by water | Assembly reflected by water | 0.0094-0.0244 |
| Experiment HEU-COMP-MIXED-001 (26 cases) | U | 93.15 wt% $^{235}U$ | $UO_2$ | None | Carbon steel cans | Alcohol-water solution in some cases; also some Plexiglas<br>H/X=0-49 (calc) | Complex arrays of cans in rectangular geometry | Reflected by polyethylene (thickness 15.24 cm) | 0.1045-0.8015 |
| Experiment HEU-MET-INTER-006 (2 cases) | U | 93.2 wt% $^{235}U$<br>(Average) | U metal discs | None | SS diaphragm; Al | C discs<br>C/$^{235}U$=52 | Cylindrical assembly of alternating U and C discs (53.34 cm diameter) | Reflected by copper | 0.3423-0.3864 |
| Experiment U233-SOL-THERM-006 (12 cases) | U | 97.56 or 97.54 wt% $^{233}U$; 0.025 or 0.026 wt% $^{235}U$ | Uranyl nitrate | None | SS cans, Al trays, SS rods, Th (trace) | Water in aqueous solution of uranyl nitrate<br>H/X=69-121 (calc) | Complex arrays of cans containing uranyl nitrate solution in rectangular geometry (2x2x2 or 3x3x3) | Unreflected | 0.0344-0.0599 |

Table 6-22 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Fort Saint Vrain HTGR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment U233-MET-FAST-001 (1 case) | U | 98.13 at% 233U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.9838 cm) | Unreflected | 1.7739 |
| Experiment U233-MET-FAST-002 (2 cases) | U | 98.2 wt% $^{233}$U(sphere); 93.2 wt% $^{235}$U in reflector) | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by enriched uranium (93.2 wt% $^{235}$U ) (thickness 1.2217 and 1.9888 cm) | 1.7079-1.7370 |
| Experiment U233-MET-FAST-003 (2 cases) | U | 98.2 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by natural uranium (thickness 2.3012 and 5.3086 cm) | 1.7483-1.7623 |
| Experiment U233-MET-FAST-004 (2 cases) | U | 98.2 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by tungsten (thickness 2.4384 and 5.7912 cm) | 1.5178-1.6134 |
| Experiment U233-MET-FAST-005 (2 cases) | U | 98.2 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by beryllium (thickness 2.0447 and 4.1961 cm) | 1.5187-1.6195 |
| Experiment U233-MET-FAST-006 (1 case) | U | 98.13 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 4.2058 cm cm) | Reflected by natural uranium (0.72 wt% $^{235}$U (thickness = 19.9136 cm) | 1.7740 |

Table 6-22 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Fort Saint Vrain HTGR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-007 (43 cases) | U | 93.15 wt% $^{235}U$ | U metal slabs | None | SS stringers | Polyethylene, Plexiglas or teflon plates<br>H/X range = 0 to 5 (exp) | Multiple layers of rectangular plates of uranium and moderator (base dimension 25.4 cm x 25.4 cm or 12.7cm x 25.4 cm) | Unreflected; some configurations are reflected by polyethylene | 0.3145-1.5855 |
| Experiment HEU-MET-FAST-033 (2 cases) | U | 96 wt% $^{235}U$ | U metal discs | None | SS discs | Polyethylene discs | Cylindrical assembly of alternating U, Polyethylene, and SS discs (19.99 cm diameter) | Unreflected | 0.7170-0.8500 |
| Experiment HEU-MET-FAST-034 (3 cases) | U | 96 wt% $^{235}U$ | U metal discs | None | Ti, Al or Steel discs | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and metal discs (19.99 cm diameter) | Unreflected | 0.9530-0.9580 |
| Experiment HEU-MET-FAST-036 (2 cases) | U | 96 wt% $^{235}U$ | U metal discs (both HEU and Depleted Uranium | None | SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and depleted uranium discs (19.99 cm diameter) | Unreflected | 0.8660-1.0970 |
| Experiment HEU-MET-MIXED-001 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | Ti discs; SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and Ti discs (19.99 cm diameter) | Reflected by polyethylene (thickness = 10 cm) | 0.7620 |
| Experiment HEU-MET-MIXED-004 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, depleted U, and polyethylene discs (19.99 cm diameter) | Reflected by depleted uranium (metal) | 0.9140 |

Table 6-22 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Fort Saint Vrain HTGR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-MIXED-005 (5 cases) | U | 90 wt% $^{235}U$ | U metal pellets | B as B4C in some rods | SiO2, Al tubes | Polyethylene, sand ($SiO_2$) | Hexagonally pitched lattice of tubes containing fuel and/or moderator | $SiO_2$, depleted uranium | 0.2120-0.3770 |
| Experiment HEU-COMP-MIXED-001 (26 cases) | U | 93.15 wt% $^{235}U$ | $UO_2$ | None | Carbon steel cans | Alcohol-water solution in some cases; Plexiglas also present H/X=0-49 (calc) | Complex arrays of cans in rectangular geometry | Reflected by polyethylene (thickness 15.24 cm); Plexiglas also present | 0.1045-0.8015 |

Table 6-23 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Fort Saint Vrain HTGR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment U233-SOL-THERM-001 (5 cases) | U | 97.7 wt% $^{233}U$; 0.04 wt% $^{235}U$ | Uranyl nitrate | B (4 cases) | Nitric acid, thorium (trace) | Water in aqueous solution of uranyl nitrate H/X range = 1324-1533 (reported) | Solution contained in an Al sphere (inner radius = 34.595 cm, thickness = 0.32 cm) | Unreflected | 0.0038-0.0043 |
| Experiment U233-SOL-THERM-002 (17 cases) | U | 98.7 wt% $^{233}U$; 0.01 wt% $^{235}U$ | Uranyl nitrate | None | Nitric acid | Water in aqueous solution of uranyl nitrate H/X range = 57.1-752.6 (reported) | Solution contained in single Al cylindrical vessel (radius = 10.2645, 10.7641, 11.4351, 12.7287, or 15.2571 cm) | Reflected by paraffin (thickness=15.24 cm) | 0.0056-0.0490 |

Table 6-23 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Fort Saint Vrain HTGR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment U233-SOL-THERM-003 (10 cases) | U | 98.7 wt% $^{233}$U; 0.04 wt% $^{235}$U | Uranyl fluoride | None | HF | Water in aqueous solution of uranyl fluoride<br>H/X range = 39.4-775 (reported) | Solution contained in single Al cylindrical vessel (inner radius =6.3230, 6.8265, 7.5589, 8.3302, 9.5246, 10.2645, 10.7641, 11.4351, or 15.2571 cm) | Reflected by paraffin (thickness=15.24 cm) | 0.0056-0.0693 |
| Experiment U233-SOL-THERM-004 (8 cases) | U | 98.7 wt% $^{233}$U; 0.01 wt% $^{235}$U | Uranyl nitrate | None | Nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range = 57.1-149.2(reported) | Solution contained in single Al cylindrical vessel (radius = 7.5589 or 9.5246cm) | Reflected (radial and bottom) by paraffin (thickness=15.24 cm) | 0.0208-0.0493 |
| Experiment U233-SOL-THERM-005 (2 cases) | U | 98.7 wt% $^{233}$U; 0.01 wt% $^{235}$U | Uranyl nitrate | None | Nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X= 405 and 514(reported) | 2 configurations: first has solution contained in a spherical Al vessel (radius = 13.2118 cm);<br>-second has solution contained in single Al cylindrical vessel (radius = 12.7287 cm) | Reflected by water (thickness=15.24 cm) | 0.0078-0.0094 |
| Experiment U233-SOL-THERM-008 (1 case) | U | 97.67 wt% $^{233}$U; 0.03 wt% $^{235}$U | Uranyl nitrate | None | Nitric acid; Thorium (trace) | Water in aqueous solution of uranyl nitrate<br>H/X= 1985 | Solution contained in single Al spherical vessel (radius = 61.011 cm) | Unreflected | 0.0030 |

Table 6-23 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Fort Saint Vrain HTGR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment U233-SOL-THERM-006 (12 cases) | U | 97.56 0r 97.54 wt% $^{233}U$; 0.025 or 0.026 wt% $^{235}U$ | Uranyl nitrate | None | SS cans, Al trays, SS rods, Thorium (trace) | Water in aqueous solution of uranyl nitrate<br>H/X=69-121 (calc) | Complex arrays of cans containing uranyl nitrate solution in rectangular geometry (2x2x2 or 3x3x3) | Unreflected | 0.0344-0.0599 |
| Experiment HEU-COMP-THERM-002 (25 cases) | U | 93.15 wt% $^{235}U$ | Uranium dicarbide | None | Al tubes; Plexiglas supports | Graphite and Water<br>C/X = 87-88.9 | Various arrays (triangular or square pitched lattices) of Al tubed or bare fuel elements (hexagonal graphite blocks containing uranium dicarbide beads) surrounded by water | Assembly reflected by water | 0.0094-0.0244 |
| Experiment HEU-COMP-MIXED-001 (26 cases) | U | 93.15 wt% $^{235}U$ | $UO_2$ | None | Carbon steel cans | Alcohol-water solution in some cases; Plexiglas<br>H/X=0-49 (calc) | Complex arrays of cans in rectangular geometry | Reflected by polyethylene (thickness 15.24 cm); Plexiglas also present | 0.1045-0.8015 |
| Experiment HEU-MET-INTER-006 (2 cases) | U | 93.2 wt% $^{235}U$ (average) | U metal discs | None | SS diaphragm; Al | C discs<br>C/$^{235}U$=52 | Cylindrical assembly of alternating U and C discs (53.34 cm diameter) | Reflected by copper | 0.3423-0.3864 |

Table 6-24 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Fort Saint Vrain HTGR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment U233-SOL-INTER -001 (33 cases) | U | 98.6 wt% $^{233}$U; 0.04 wt% $^{235}$U | Uranyl fluoride | None | Excess HF | Water in aqueous solution of uranyl fluoride H/X range = 24.66-40.34 (calc) | Solution contained in single SS vessel (inner radius varied for each experiment, range 7.8726 to 11.4152 cm) | Reflected by steel, Be and/or polyethylene (variable thickness) | 0.0685-0.1325 |
| Experiment U233-MET-FAST-001 (1 case) | U | 98.13 at% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.9838 cm) | Unreflected | 1.7739 |
| Experiment U233-MET-FAST-002 (2 cases) | U | 98.2 wt% $^{233}$U(sphere); 93.2 wt% $^{235}$U in reflector) | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by enriched uranium (93.2 wt% $^{235}$U ) (thickness 1.2217 and 1.9888 cm) | 1.7079-1.7370 |
| Experiment U233-MET-FAST-003 (2 cases) | U | 98.2 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by natural uranium (thickness 2.3012 and 5.3086 cm) | 1.7483-1.7623 |
| Experiment U233-MET-FAST-004 (2 cases) | U | 98.2 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by tungsten (thickness 2.4384 and 5.7912 cm) | 1.5178-1.6134 |
| Experiment U233-MET-FAST-005 (2 cases) | U | 98.2 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by beryllium (thickness 2.0447 and 4.1961 cm) | 1.5187-1.6195 |

Table 6-24 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Fort Saint Vrain HTGR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment U233-MET-FAST-006 (1 case) | U | 98.13 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 4.2058 cm cm) | Reflected by natural uranium (0.72 wt% $^{235}$U (thickness = 19.9136 cm) | 1.7740 |
| Experiment HEU-MET-FAST-007 (43 cases) | U | 93.15 wt% $^{235}$U | U metal slabs | None | SS stringers | Polyethylene, Plexiglas or teflon plates<br>H/X range = 0 to 5 (exp) | Multiple layers of rectangular plates of uranium and moderator (base dimension 25.4 cm x 25.4 cm or 12.7cm x 25.4 cm) | Unreflected;<br>some configurations are reflected by polyethylene | 0.3145-1.5855 |
| Experiment HEU-MET-FAST-011 (1 case) | U | 89.61wt% $^{235}$U<br>(average) | U metal | None | | None | The spherical assembly core (radius=7.55 cm) incorporates multiple (7) spherical layers of fissile material and has a central spherical cavity (radius =2 cm) | Reflected by polyethylene (thickness 10.45 cm) | 1.1607 |
| Experiment HEU-MET-FAST-019 (1 case) | U | 90 wt% $^{235}$U | U metal | None | | None | The spherical assembly core (radius=9.15 cm) incorporates multiple (7) spherical layers of fissile material and has a central cavity (radius = 4.029 cm) | Reflected by graphite (thickness=3.45 cm) | 1.4793 |
| Experiment HEU-MET-FAST-020 (1 case) | U | 90 wt% $^{235}$U | U metal | None | | None | The spherical assembly core (radius=8.35 cm) incorporates multiple spherical layers of fissile material | Reflected by polyethylene (thickness=1.45 cm) | 1.4341 |

Table 6-24 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Fort Saint Vrain HTGR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-024 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | | None | The spherical assembly core (radius=7.54 cm) incorporates multiple (10) spherical layers of fissile material and has a central cavity (radius = 0.6 cm) | Reflected by steel (thickness =0.8 cm) and polyethylene (thickness=9.65 cm) | 1.2553 |
| Experiment HEU-MET-FAST-031 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | Polyethylene | Polyethylene in the central cavity | The spherical assembly core (radius=7.55 cm) incorporates multiple spherical layers of fissile material. The central area contains a spherical block of polyethylene (radius= 4.02) with a central cavity (radius= 1.4 cm) | Reflected by polyethylene (thickness =17.45 cm) | 0.9660 |
| Experiment HEU-MET-FAST-033 (2 cases) | U | 96 wt% $^{235}U$ | U metal discs | None | SS discs | Polyethylene discs | Cylindrical assembly of alternating U, Polyethylene, and SS discs (19.99 cm diameter) | Unreflected | 0.7170-0.8500 |
| Experiment HEU-MET-FAST-034 (3 cases) | U | 96 wt% $^{235}U$ | U metal discs | None | Ti, Al or Steel discs | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and metal discs (19.99 cm diameter) | Unreflected | 0.9530-0.9580 |
| Experiment HEU-MET-FAST-036 (2 cases) | U | 96 wt% $^{235}U$ | U metal discs (both HEU and Depleted Uranium | None | SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and depleted uranium discs (19.99 cm diameter) | Unreflected | 0.8660-1.0970 |

Table 6-24 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Fort Saint Vrain HTGR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-041 (6 cases) | U | 94 wt% $^{235}U$ | U metal | None | | None | Single homogeneous unit (sphere) with a radius between 5.64 and 7.38 cm | Reflected by Be(2 cases) and graphite (4 cases) | 1.1280-1.4660 |
| HEU-MET-MIXED-001 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | Ti discs; SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and Ti discs (19.99 cm diameter) | Reflected by polyethylene (thickness = 10cm) | 0.7620 |
| HEU-MET-MIXED-004 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, depleted U, and polyethylene discs (19.99 cm diameter) | Reflected by depleted uranium (metal) | 0.9140 |
| HEU-MET-MIXED-005 (5 cases) | U | 90 wt% $^{235}U$ | U metal pellets | B as $B_4C$ present in some rods | SiO2, Al tubes | Polyethylene, sand ($SiO_2$) | Hexagonally pitched lattice of tubes containing fuel and/or moderator | $SiO_2$, depleted uranium | 0.2120-0.3770 |
| Experiment HEU-COMP-MIXED-001 (26 cases) | U | 93.15 wt% $^{235}U$ | $UO_2$ | None | Carbon steel cans | Alcohol-water solution in some cases; Plexiglas H/X=0-49 (calc) | Complex arrays of cans in rectangular geometry | Reflected by polyethylene (thickness 15.24 cm); Plexiglas | 0.1045-0.8015 |
| Experiment HEU-MET-MIXED-002 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | polyethylene | Spherical core of polyethylene | Assembly core incorporates 3 spherical layers of fissile material( max radius 9.15 cm) | Reflected by polyethylene (thickness 12.85 cm) | 0.7880 |
| Experiment HEU-MET-MIXED-003 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | polyethylene | Spherical core of polyethylene | Assembly core incorporates 3 spherical layers of fissile material( max radius 9.15 cm) | Reflected by polyethylene (thickness 15.85 cm) | 0.7840 |

Table 6-25 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport LWBR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-COMP-MIXED-001 (26 cases) | U | 93.15 wt% $^{235}U$ | $UO_2$ | None | Carbon steel cans | Alcohol-water solution in some cases; Plexiglas H/X=0-49 (calc) | Complex arrays of cans in rectangular geometry | Reflected by polyethylene (thickness 15.24 cm); Plexiglas | 0.1045-0.8015 |
| Experiment U233-SOL-THERM-006 (12 cases) | U | 97.56 or 97.54 wt% $^{233}U$; 0.025 or 0.026 wt% $^{235}U$ | Uranyl nitrate | None | SS cans, Al trays, SS rods, Th (trace) | Water in aqueous solution of uranyl nitrate H/X=69-121 (calc) | Complex arrays of cans containing uranyl nitrate solution in rectangular geometry (2x2x2 or 3x3x3) | Unreflected | 0.0344-0.0599 |
| Experiment HEU-COMP-THERM-003 (15 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding Al-lattice plates | Water H/X range(fuel region) = 51-349 | Cylindrical two-zones hexagonally pitched lattices (active fuel length =69.2 cm); varied pitches | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0139-0.0467 |
| Experiment HEU-COMP-THERM-005 (1 case) (detailed benchmark model evaluated) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding Al-lattice plates | Water H/X (fuel region) = 23 | Hexagonally pitched array of fuel rod clusters (each containing a hexagonally pitched lattice of fuel rods); hex pitch = 0.52 cm | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0764 |
| Experiment HEU-COMP-THERM-006 (3 cases) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | SS-cladding Al-lattice plates | Water H/X range (fuel region) = 30-716 | Cylindrical hexagonally pitched lattice (pitches of 0.56, 1, and 2.113 cm) | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0104-0.0720 |

Table 6-25 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport LWBR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-COMP-THERM-007 (3 cases) (detailed benchmark models evaluated) | U | 79.66 wt% $^{235}U$ | $UO_2$+Cu (cross shaped fuel rods) | None | ZrH rods, SS-cladding Al-lattice plates | Water +ZrH rods H/X range (fuel region) = 60 - 91 | Cylindrical hexagonally double lattice of fuel rods and ZrH rods. Fixed pitch for fuel rods (0.61 cm) and variable for ZrH rods | Assembly reflected by water (thickness >20 cm) and SS cylindrical tank | 0.0339-0.0475 |
| Experiment HEU-COMP-THERM-011 (3 cases) | U | 80 wt% $^{235}U$ | $UO_2$ +Al alloy (sintered) | None | SS cladding; Al lattice plate | Water; partially flooded cores H/X=170 | Four square clusters of cylindrical fuel rods (21x21, pitch =1.4 cm)) arranged in a square geometry (dist between clusters = 0 to 4.8 cm) | Assembly reflected by water (except top of assembly - partially flooded) | 0.0470-0.0533 |
| Experiment HEU-COMP-THERM-012 (2 cases) | U | 80 wt% $^{235}U$ | $UO_2$ +Al alloy (sintered) | None | SS cladding; Al lattice plate | Water; partially flooded cores H/X=35 | Four square clusters of cylindrical fuel rods (18x18, pitch =1.4 cm)) arranged in a square geometry (dist between clusters =1.4 and 2.8 cm) | Assembly reflected by water (except top of assembly - partially flooded) | 0.0509-0.0551 |
| Experiment HEU-COMP-THERM-013 (2 cases) | U | 80 wt% $^{235}U$ | $UO_2$ +Al alloy (sintered) | None | SS cladding; Al lattice plate | Water; partially flooded cores H/X=40 | Nine square clusters of cylindrical fuel rods (14x14, pitch =1.4 cm)) arranged in a square geometry (dist between clusters =1.4 and 2.8 cm) | Assembly reflected by water (except top of assembly - partially flooded) | 0.0432-0.0481 |

Table 6-26 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport LWBR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment U233-MET-FAST-001 (1 case) | U | 98.13 at% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.9838 cm) | Unreflected | 1.7739 |
| Experiment U233-MET-FAST-002 (2 cases) | U | 98.2 wt% $^{233}$U(sphere); 93.2 wt% $^{235}$U in reflector) | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by enriched uranium (93.2 wt% $^{235}$U ) (thickness 1.2217 and 1.9888 cm) | 1.7079-1.7370 |
| Experiment U233-MET-FAST-003 (2 cases) | U | 98.2 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by natural uranium (thickness 2.3012 and 5.3086 cm) | 1.7483-1.7623 |
| Experiment U233-MET-FAST-004 (2 cases) | U | 98.2 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by tungsten (thickness 2.4384 and 5.7912 cm) | 1.5178-1.6134 |
| Experiment U233-MET-FAST-005 (2 cases) | U | 98.2 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by beryllium (thickness 2.0447 and 4.1961 cm) | 1.5187-1.6195 |
| Experiment U233-MET-FAST-006 (1 case) | U | 98.13 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 4.2058 cm cm) | Reflected by natural uranium (0.72 wt% $^{235}$U (thickness = 19.9136 cm) | 1.7740 |

Table 6-26 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport LWBR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-COMP-MIXED-001 (26 cases) | U | 93.15 wt% $^{235}U$ | $UO_2$ | None | Carbon steel cans | Alcohol-water solution in some cases<br>H/X=0-49 (calc)<br>Plexiglas | Complex arrays of cans in rectangular geometry | Reflected by polyethylene (thickness 15.24 cm)<br>Plexiglas present | 0.1045-0.8015 |
| Experiment HEU-COMP-INTER-002 (5 cases) | U | 96 wt% $^{235}U$ | $UO_2$ | None | Al foil cladding | ZrH + water in various reactor cavities<br>H/X = 14 (calc) | Cilindrical array of fuel rods with Be axial reflector. | Assembly is radially reflected by sand and water; be end reflectors | 0.2377-0.2439 |
| Experiment HEU-MET-FAST-007 (43 cases) | U | 93.15 wt% $^{235}U$ | U metal slabs | None | SS stringers | Polyethylene, Plexiglas or teflon plates<br>H/X range = 0 to 5 (exp) | Multiple layers of rectangular plates of uranium and moderator (base dimension 25.4 cm x 25.4 cm or 12.7cm x 25.4 cm) | Unreflected;<br>some configurations are reflected by polyethylene | 0.3145 - 1.5855 |
| Experiment HEU-MET-FAST-033 (2 cases) | U | 96 wt% $^{235}U$ | U metal discs | None | SS discs | Polyethylene discs | Cylindrical assembly of alternating U, Polyethylene, and SS discs (19.99 cm diameter) | Unreflected | 0.7170-0.8500 |
| Experiment HEU-MET-FAST-034 (3 cases) | U | 96 wt% $^{235}U$ | U metal discs | None | Ti, Al or Steel discs | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and metal discs (19.99 cm diameter) | Unreflected | 0.9530-0.9580 |
| Experiment HEU-MET-FAST-036 (2 cases) | U | 96 wt% $^{235}U$ | U metal discs (both HEU and Depleted Uranium | None | SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and depleted uranium discs (19.99 cm diameter) | Unreflected | 0.8660-1.0970 |

Table 6-26 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact Shippingport LWBR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-MIXED-001 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | Ti discs; SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and Ti discs (19.99 cm diameter) | Reflected by polyethylene (thickness = 10 cm) | 0.7620 |
| HEU-MET-MIXED-004 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, depleted U, and polyethylene discs (19.99 cm diameter) | Reflected by depleted uranium (metal) | 0.9140 |
| HEU-MET-MIXED-005 (5 cases) | U | 90 wt% $^{235}U$ | U metal pellets | B as B4C in some rods | SiO2, Al tubes | Polyethylene, sand ($SiO_2$) | Hexagonally pitched lattice of tubes containing fuel and/or moderator | $SiO_2$, depleted uranium | 0.2120-0.3770 |

Table 6-27 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport LWBR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment U233-SOL-THERM-001 (5 cases) | U | 97.7 wt% $^{233}U$; 0.04 wt% $^{235}U$ | Uranyl nitrate | B (4 cases) | Nitric acid, thorium (trace) | Water in aqueous solution of uranyl nitrate<br>H/X range = 1324-1533 (reported) | Solution contained in an Al sphere (inner radius = 34.595 cm, thickness = 0.32 cm) | Unreflected | 0.0038-0.0043 |
| Experiment U233-SOL-THERM-002 (17 cases) | U | 98.7 wt% $^{233}U$; 0.01 wt% $^{235}U$ | Uranyl nitrate | None | Nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range = 57.1-752.6 (reported) | Solution contained in single Al cylindrical vessel (radius = 10.2645, 10.7641, 11.4351, 12.7287, or 15.2571 cm) | Reflected by paraffin (thickness=15.24 cm) | 0.0056-0.0490 |

Table 6-27 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport LWBR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment U233-SOL-THERM-003 (10 cases) | U | 98.7 wt% $^{233}U$; 0.04 wt% $^{235}U$ | Uranyl fluoride | None | HF | Water in aqueous solution of uranyl fluoride<br>H/X range = 39.4-775 (reported) | Solution contained in single Al cylindrical vessel (inner radius =6.3230, 6.8265, 7.5589, 8.3302, 9.5246, 10.2645, 10.7641, 11.4351, or 15.2571 cm) | Reflected by paraffin (thickness=15.24 cm) | 0.0056-0.0693 |
| Experiment U233-SOL-THERM-004 (8 cases) | U | 98.7 wt% $^{233}U$; 0.01 wt% $^{235}U$ | Uranyl nitrate | None | Nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range = 57.1-149.2 (reported) | Solution contained in single Al cylindrical vessel (radius = 7.5589 or 9.5246 cm) | Reflected (radial and bottom) by paraffin (thickness=15.24 cm) | 0.0208-0.0493 |
| Experiment U233-SOL-THERM-005 (2 cases) | U | 98.7 wt% $^{233}U$; 0.01 wt% $^{235}U$ | Uranyl nitrate | None | Nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X= 405 and 514(reported) | 2 configurations: first has solution contained in a spherical Al vessel (radius = 13.2118 cm);<br>-second has solution contained in single Al cylindrical vessel (radius = 12.7287 cm) | Reflected by water (thickness=15.24 cm) | 0.0078-0.0094 |
| Experiment U233-SOL-THERM-008 (1 case) | U | 97.67 wt% $^{233}U$; 0.03 wt% $^{235}U$ | Uranyl nitrate | None | Nitric acid; Thorium (trace) | Water in aqueous solution of uranyl nitrate<br>H/X= 1985 | Solution contained in single Al spherical vessel (radius = 61.011 cm) | Unreflected | 0.0030 |

Table 6-27 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport LWBR SNF (thermal spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment U233-SOL-THERM-006 (12 cases) | U | 97.56 or 97.54 wt% $^{233}U$; 0.025 or 0.026 wt% $^{235}U$ | Uranyl nitrate | None | SS cans, Al trays, SS rods, Thorium (trace) | Water in aqueous solution of uranyl nitrate<br>H/X=69-121 (calc) | Complex arrays of cans containing uranyl nitrate solution in rectangular geometry (2x2x2 or 3x3x3) | Unreflected | 0.0344-0.0599 |
| Experiment HEU-COMP-MIXED-001 (26 cases) | U | 93.15 wt% $^{235}U$ | $UO_2$ | None | Carbon steel cans | Alcohol-water solution in some cases; Plexiglas<br>H/X=0-49 (calc) | Complex arrays of cans in rectangular geometry | Reflected by polyethylene (thickness 15.24 cm); Plexiglas also present | 0.1045-0.8015 |

Table 6-28 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport LWBR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment U233-SOL-INTER -001 (33 cases) | U | 98.6 wt% $^{233}U$;<br>0.04 wt% $^{235}U$ | Uranyl fluoride | None | Excess HF | Water in aqueous solution of uranyl fluoride<br>H/X range = 24.66-40.34 (calc) | Solution contained in single SS vessel (inner radius varied for each experiment, range 7.8726 to 11.4152 cm) | Reflected by steel, Be and/or polyethylene (variable thickness) | 0.0685-0.1325 |
| Experiment U233-MET-FAST-001 (1 case) | U | 98.13 at% $^{233}U$ | U metal | None | Ni coating | None | Sphere of U metal (radius 5.9838 cm) | Unreflected | 1.7739 |

Table 6-28 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport LWBR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment U233-MET-FAST-002 (2 cases) | U | 98.2 wt% $^{233}$U(sphere); 93.2 wt% $^{235}$U in reflector) | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by enriched uranium (93.2 wt% $^{235}$U ) (thickness 1.2217 and 1.9888 cm) | 1.7079-1.7370 |
| Experiment U233-MET-FAST-003 (2 cases) | U | 98.2 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by natural uranium (thickness 2.3012 and 5.3086 cm) | 1.7483-1.7623 |
| Experiment U233-MET-FAST-004 (2 cases) | U | 98.2 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by tungsten (thickness 2.4384 and 5.7912 cm) | 1.5178-1.6134 |
| Experiment U233-MET-FAST-005 (2 cases) | U | 98.2 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 5.0444 cm and 4.5999 cm) | Reflected by beryllium (thickness 2.0447 and 4.1961 cm) | 1.5187-1.6195 |
| Experiment U233-MET-FAST-006 (1 case) | U | 98.13 wt% $^{233}$U | U metal | None | Ni coating | None | Sphere of U metal (radius 4.2058 cm cm) | Reflected by natural uranium (0.72 wt% $^{235}$U (thickness = 19.9136 cm) | 1.7740 |
| Experiment HEU-MET-FAST-007 (43 cases) | U | 93.15 wt% $^{235}$U | U metal slabs | None | SS stringers | Polyethylene, Plexiglas or teflon plates<br>H/X range = 0 to 5 (exp) | Multiple layers of rectangular plates of uranium and moderator (base dimension 25.4 cm x 25.4 cm or 12.7cm x 25.4 cm) | Unreflected; some configurations are reflected by polyethylene | 0.3145-1.5855 |

Table 6-28 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport LWBR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-011 (1 case) | U | 89.61wt% $^{235}U$ (average) | U metal | None | | None | The spherical assembly core (radius=7.55 cm) incorporates multiple (7) spherical layers of fissile material and has a central spherical cavity (radius =2 cm) | Reflected by polyethylene (thickness 10.45 cm) | 1.1607 |
| Experiment HEU-MET-FAST-019 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | | None | The spherical assembly core (radius=9.15 cm) incorporates multiple (7) spherical layers of fissile material and has a central cavity (radius = 4.029 cm) | Reflected by graphite (thickness=3.45 cm) | 1.4793 |
| Experiment HEU-MET-FAST-020 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | | None | The spherical assembly core (radius=8.35 cm) incorporates multiple spherical layers of fissile material | Reflected by polyethylene (thickness=1.45 cm) | 1.4341 |
| Experiment HEU-MET-FAST-024 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | | None | The spherical assembly core (radius=7.54 cm) incorporates multiple (10) spherical layers of fissile material and has a central cavity (radius = 0.6 cm) | Reflected by steel (thickness =0.8 cm) and polyethylene (thickness=9.65 cm) | 1.2553 |

Table 6-28 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport LWBR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment HEU-MET-FAST-031 (1 case) | U | ~90 wt% $^{235}U$ | U metal | None | Polyethylene | Polyethylene in the central cavity | The spherical assembly core (radius=7.55 cm) incorporates multiple spherical layers of fissile material. The central area contains a spherical block of polyethylene (radius= 4.02) with a central cavity (radius= 1.4 cm) | Reflected by polyethylene (thickness =17.45 cm) | 0.9660 |
| Experiment HEU-MET-FAST-033 (2 cases) | U | 96 wt% $^{235}U$ | U metal discs | None | SS discs | Polyethylene discs | Cylindrical assembly of alternating U, Polyethylene, and SS discs (19.99 cm diameter) | Unreflected | 0.7170-0.8500 |
| Experiment HEU-MET-FAST-034 (3 cases) | U | 96 wt% $^{235}U$ | U metal discs | None | Ti, Al or Steel discs | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and metal discs (19.99 cm diameter) | Unreflected | 0.9530-0.9580 |
| Experiment HEU-MET-FAST-036 (2 cases) | U | 96 wt% $^{235}U$ | U metal discs (both HEU and Depleted Uranium | None | SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and depleted uranium discs (19.99 cm diameter) | Unreflected | 0.8660-1.0970 |
| Experiment HEU-MET-FAST-041 (6 cases) | U | 94 wt% $^{235}U$ | U metal | None | | None | Single homogeneous unit (sphere) with a radius between 5.64 and 7.38 cm | Reflected by Be(2 cases) and graphite (4 cases) | 1.1280-1.4660 |

Table 6-28 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded Shippingport LWBR SNF (intermediate and fast spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| HEU-MET-MIXED-001 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | Ti discs; SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, polyethylene, and Ti discs (19.99 cm diameter) | Reflected by polyethylene (thickness = 10 cm) | 0.7620 |
| HEU-MET-MIXED-004 (1 case) | U | 96 wt% $^{235}U$ | U metal discs | None | SS diaphragm | Polyethylene discs | Cylindrical assembly of alternating U, depleted U, and polyethylene discs (19.99 cm diameter) | Reflected by depleted uranium (metal) | 0.9140 |
| HEU-MET-MIXED-005 (5 cases) | U | 90 wt% $^{235}U$ | U metal pellets | B as $B_4C$ present in some rods | SiO2, Al tubes | Polyethylene, sand ($SiO_2$) | Hexagonally pitched lattice of tubes containing fuel and/or moderator | $SiO_2$, depleted uranium | 0.2120-0.3770 |
| PU-MET-FAST-008 (1 case)(detailed model evaluated) | Pu | 93.59 wt% $^{239}Pu$ | Delta phase Pu metal sphere | None | Ga 1.01 wt% | None | Single homogeneous unit (sphere) with a radius = 5.310 cm | Reflected by metallic Thorium (thickness= 24.57 cm) | 1.7677 |
| Experiment HEU-COMP-MIXED-001 (26 cases) | U | 93.15 wt% $^{235}U$ | $UO_2$ | None | Carbon steel cans | Alcohol-water solution in some cases<br>H/X=0-49 (calc)<br>Plexiglas | Complex arrays of cans in rectangular geometry | Reflected by polyethylene (thickness 15.24 cm)<br>Plexiglas present | 0.1045-0.8015 |
| Experiment HEU-MET-MIXED-002 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | polyethylene | Spherical core of Polyethylene | Assembly core incorporates 3 spherical layers of fissile material( max radius 9.15 cm) | Reflected by polyethylene (thickness 12.85 cm) | 0.7880 |
| Experiment HEU-MET-MIXED-003 (1 case) | U | 90 wt% $^{235}U$ | U metal | None | polyethylene | Spherical core of Polyethylene | Assembly core incorporates 3 spherical layers of fissile material( max radius 9.15 cm) | Reflected by polyethylene (thickness 15.85 cm) | 0.7840 |

Table 6-29 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact N-Reactor SNF (thermal and intermediate spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment with N-reactor Mark IA fuel elements (3 cases) | U | 0.947 and 1.25 wt% $^{235}U$ | U metal | None | Zircaloy-2 cladding | Water<br>H/X range (fuel region =994 - 1876 (calc) | Complex lattice of N-reactor mark IA fuel elements with pitches 7.112, 7.874, 8.636 cm (133.096 cm long) | Reflected by water | 0.3145-0.4085 |
| Experiment LEU-COMP-THERM-001 (8 cases) | U | 2.35 wt% $^{235}U$ | UO2 | None | Al cladding | Water<br>H/X range (fuel region)= 449 – 487 | Square-pitched clusters (pitch = 2.032 cm) in a rectangular geometry | Reflected by water | 0.1186-0.1239 |
| Experiment with 2.35 wt% $^{235}U$ and absorber plates (see also LEU-COMP-THERM-016) (4 cases) | U | 2.35 wt% $^{235}U$ | UO2 | Absorber plates of steel, borated steel and Al | Al cladding | Water | Square-pitched clusters (pitch = 2.032 cm) in a rectangular geometry | Reflected by water | 0.1201-0.1229 |

Table 6-29 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact N-Reactor SNF (thermal and intermediate spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment with 4.31 wt% $^{235}U$ subcritical clusters and reflecting walls (see also LEU-COMP-THERM-010) (3 cases) | U | 4.31 wt% $^{235}U$ | UO2 | None | Al cladding | Water | Square-pitched clusters (pitch = 1.892 cm) in a rectangular geometry | Reflected by lead, depleted uranium and carbon steel tank | 0.1778-0.2839 |
| Experiment LEU-COMP-THERM-042 (7 cases) | U | 2.35 wt% $^{235}U$ | UO2 | Absorber plates of stainless steel, borated stainless steel; boral; boroflex; Cd; Cu and Cd/Cu mix | Al cladding | Water | Square-pitched clusters (pitch = 1.684 cm) in a rectangular geometry | Reflected by SS walls and water | 0.1690-0.1750 |

Table 6-30 Area and Range of Applicability Benchmark Experiments for WP Configurations Containing Degraded N-Reactor SNF (thermal and intermediate spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiments with LEU UO3-H2O solutions (12 cases) | U | 1.0059 to 1.1586 wt% $^{235}U$ | UO3-H2O solutions | None | | Water in solution<br>H/X range (fuel region) =370 – 731 (calc) | Single homogeneous sphere | Unreflected | 0.1549-0.2541 |

Table 6-30 Area and Range of Applicability Benchmark Experiments for WP Configurations Containing Degraded N-Reactor SNF (thermal and intermediate spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment LEU-SOL-THERM-001 (1 case) | U | 5 wt% $^{235}U$ | Uranyl fluoride | None | HF | Water in aqueous solution of uranyl fluoride<br>H/X = 454 | Solution contained in SS304L cylindrical vessel (inner radius = 3.175 cm, outer radius = 24.4475cm) | Unreflected | 0.0519 |
| Experiment LEU-SOL-THERM-002 (3 cases) | U | 4.9 wt% $^{235}U$ | Uranium oxy-fluoride | None | | Water in aqueous solution of uranium oxy-fluoride<br>H/X range = 1001-1098 | Solution contained in single Al spherical vessel (radius =34.399, 34.4995, or 34.6710 cm) | Unreflected (1 case) or reflected by water | 0.0251-0.0283 |
| Experiment LEU-SOL-THERM-005 (3 cases) | U | 5.64 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | B4C rods | Exces nitric acid; SS cladding (B4C rods) | Water in aqueous solution of uranyl nitrate<br>H/X (solution region) =972 | The uranyl nitrate solution is contained in a cylindrical tank (inner diameter=55 cm) | Reflected by water (thickness > 20 cm) (radially and bottom | 0.0254-0.0260 |
| Experiment LEU-COMP-THERM-049 (18 cases) | U | 5 wt% $^{235}U$ | UO2 powder | None | Al alloy boxes, rubber seals | Water in moisture<br>H/X range (fuel region) 2 -3 | Array of boxes in a rectangular geometry | Reflected by polyethylene | 0.2270 0.3000 |

Table 6-31 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact TMI Waste Form (thermal and intermediate spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment LEU-COMP-THERM-001 (8 cases) | U | 2.35 wt% $^{235}U$ | UO2 | None | Al cladding | Water<br>H/X range (fuel region)= 449 - 487 | Square-pitched clusters (pitch = 2.032 cm) in a rectangular geometry | Reflected by water | 0.1186-0.1239 |

Table 6-31 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact TMI Waste Form (thermal and intermediate spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment LEU-COMP-THERM-027 (1 case) | U | 4.74 wt% $^{235}U$ | UO2 | None | Al cladding | Water | Square-pitched clusters (pitch = 1.6 cm) in a rectangular geometry | Reflected by lead walls (thickness 30 cm) | 0.1024 |
| Experiment LEU-COMP-THERM-042 (7 cases) | U | 2.35 wt% $^{235}U$ | UO2 | Absorber plates of stainless steel, borated stainless steel; boral; boroflex; Cd; Cu and Cd/Cu mix | Al cladding; | Water | Square-pitched clusters (pitch = 1.684 cm) in a rectangular geometry | Reflected by SS walls and water | 0.1690-0.1750 |
| Experiment LEU-COMP-THERM-041 (5 cases) | U | 3 wt% $^{235}U$ | UO2 | Absorber plates of stainless steel, borated steel, and steel+Cd | Zircaloy-4 cladding | Water | Square-pitched assembly (pitch = 1.26 cm) | Reflected by water | 0.1180-0.1850 |
| Experiment with 2.35 wt% $^{235}U$ and absorber plates (see also LEU-COMP-THERM-016) (4 cases) | U | 2.35 wt% $^{235}U$ | UO2 | Absorber plates of Stainless Steel, Borated Stainless steel and Al | Al cladding | Water | Square-pitched clusters (pitch = 2.032 cm) in a rectangular geometry | Reflected by water | 0.1201-0.1229 |

Table 6-31 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Intact TMI Waste Form (thermal and intermediate spectrum)

| Parameter/ Critical Experiments | Fissionable element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment with 4.31 wt% $^{235}U$ subcritical clusters and reflecting walls (see also LEU-COMP-THERM-010) (3 cases) | U | 4.31 wt% $^{235}U$ | UO2 | None | Al cladding | Water | Square-pitched clusters (pitch = 1.892 cm) in a rectangular geometry | Reflected by lead, depleted uranium and carbon steel tank | 0.1778-0.2839 |
| Experiment with 4.31 wt% $^{235}U$ rods in highly borated water lattices (4 cases) | U | 4.31 wt% $^{235}U$ | UO2 | B | Al cladding, Plexiglas, polypropylene lattice | Water | Square-pitched arrays in a rectangular geometry | Reflected by unborated water | 0.1781-0.2680 |
| Experiment with urania-gadolinia (19 cases) | U | 2.46 and 4.02 wt% $^{235}U$ 1.944 wt % enriched UO2-Gd2O3 rods | UO2 | Gd2O3, B4C and Ag-In-Cd rods | Zircaloy cladding | Water | Square-pitched clusters (15x15) in a rectangular geometry | Reflected by water | 0.1974-0.2101 |

Table 6-32 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded TMI Waste Form (thermal and intermediate spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiments with LEU UO3-H2O solutions (12 cases) | U | 1.0059 to 1.1586 wt% $^{235}U$ | UO3-H2O solutions | None | | Water in solution<br>H/X range (fuel region) =370 – 731 (calc) | Single homogeneous sphere | Unreflected | 0.1549-0.2541 |

Table 6-32 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded TMI Waste Form (thermal and intermediate spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment LEU-SOL-THERM-001 (1 case) | U | 5 wt% $^{235}U$ | Uranyl fluoride | None | HF | Water in aqueous solution of uranl fluoride<br>H/X = 454 | Solution contained in SS304L cylindrical vessel (inner radius = 3.175 cm, outer radius = 24.4475 cm) | Unreflected | 0.0519 |
| Experiment LEU-SOL-THERM-002 (3 cases) | U | 4.9 wt% $^{235}U$ | Uranium oxy-fluoride | None | | Water in aqueous solution of uranium oxy-fluoride<br>H/X range = 1001 – 1098 | Solution contained in single Al spherical vessel (radius = 34.399, 34.4995, , or 34.6710 cm) | Unreflected (1 case) or reflected by water | 0.0251-0.0283 |
| Experiment LEU-SOL-THERM-004 (7 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =719 - 1018 (calc) | The uranyl nitrate solution is contained in a cylindrical tank (diameter = 59 cm) | Reflected by water (thickness > 20 cm) | 0.0142-0.0188 |
| Experiment LEU-SOL-THERM-005 (3 cases) | U | 5.64 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | B4C rods | Exces nitric acid; SS cladding (B4C rods) | Water in aqueous solution of uranyl nitrate<br>H/X (solution region) = 972 | The uranyl nitrate solution is contained in a cylindrical tank (inner diameter = 55 cm) | Reflected by water radially and bottom (thickness > 20 cm) | 0.0254-0.0260 |
| Experiment LEU-SOL-THERM-007 (5 cases) | U | 9.97 wt% $^{235}U$ | Aqueous solution of uranyl nitrate | None | Exces nitric acid | Water in aqueous solution of uranyl nitrate<br>H/X range (fuel region) =709 - 942 (calc.) | The uranyl nitrate solution is contained in a cylindrical tank (diameter=59 cm) | Unreflected | 0.0159-0.0200 |

Table 6-32 Area and Range of Applicability of Benchmark Experiments for WP Configurations Containing Degraded TMI Waste Form (thermal and intermediate spectrum)

| Parameter/ Critical Experiments | Fissionable Element | Enrichment | Physical form | Neutron absorber | Non-fissionable materials in fuel region | Moderator | Geometry of the fissionable material | Reflector | AENCF range (MeV) |
|---|---|---|---|---|---|---|---|---|---|
| Experiment LEU-COMP-THERM-049 (18 cases) | U | 5 wt% $^{235}U$ | UO2 powder | None | Al alloy boxes, rubber seals | Water in moisture<br>H/X range (fuel region) = 2-3 | Array of boxes in a rectangular geometry | Reflected by polyethylene | 0.2270-0.3000 |

## 6.3 ESTIMATE AND PLOTS OF BIAS FOR DOE SNF TYPES

The calculation methods used to evaluate the critical experiments are discussed in Section 5. An appropriate description of known experimental uncertainties is given in the NEA criticality benchmark handbook (NEA [2001]). The MCNP4B2 code with ENDFB-V library was used to evaluate the benchmark experiments. The calculation results obtained for each experiment are presented in the tables in Section 5.

ANS-8.7/ANSI N16.5-1975(R1987) provides the guidelines for establishing subcriticality based on a numerical calculation of the multiplication factor ($k_{eff}$) for a fissile system. The calculated $k_{eff}$ for a fissile system is considered to be acceptable provided the calculated $k_{eff}$ plus 2 sigma is less than a specified upper subcritical limit (USL). The following relationship is used to establish the acceptance criteria for a calculated multiplication factor for a subcritical system, $k_s$.

$$k_s \leq k_c - \Delta k_c - \Delta k_s - \Delta k_m \qquad \text{Equation 6-1}$$

where

$k_c$ = mean value of $k_{eff}$ resulting from the calculation of benchmark critical experiments using a specific calculational method and data,
$\Delta k_c$ = uncertainty in the value of $k_c$,
$\Delta k_s$ = uncertainty in the value of $k_s$,
$\Delta k_m$ = administrative margin of subcriticality (if required).

In the Monte Carlo analysis, the uncertainty in the value for $k_s$, is typically two times the standard deviation (2 sigma) of the calculated $k_{eff}$ of the system (i.e., $\Delta k_s = 2\sigma_{MC}$). The acceptance criteria can then be written in the following form :

$$k_s + 2\sigma_{MC} \leq k_c - \Delta k_c - \Delta k_m \qquad \text{Equation 6-2}$$

The bias, $\beta$ in the calculational method is the difference between the mean value of the calculated $k_{eff}$ for the critical experiments, $k_c$, and 1.0 (i.e., $\beta = k_c - 1$). Based on the definition of the calculational bias, the uncertainty in the bias is identical to the uncertainty in $k_c$. Thus, $\Delta k_c = \Delta\beta$, and the acceptance criteria becomes

$$k_s + 2\sigma_{MC} \leq 1.00 + \beta - \Delta\beta - \Delta k_m \qquad \text{Equation 6-3}$$

A USL is an upper subcritical limit such that there is a specified level of confidence that a calculated $k_{eff}$ is considered to be subcritical. Using the acceptance criteria for a subcritical system, the USL can be defined as

$$USL = (1 - \Delta k_m) + \beta - \Delta\beta \qquad \text{Equation 6-4}$$

A fissile system is considered to be acceptably subcritical provided the following general condition is met for a model representing the fissile system:

$$k_{eff} + 2\sigma_{MC} \leq USL \qquad \text{Equation 6-5}$$

where $k_{eff}$ is calculated for a model representing the fissile system and $\sigma_{MC}$ is the Monte Carlo standard deviation of the calculated $k_{eff}$.

The calculational bias in the acceptance criteria can be positive if $k_c$ is greater than 1; however, a positive bias is not used in this evaluation. Therefore the bias for a given group is always ≤ 0.0.

The sources of uncertainty in the evaluation include the calculational method, the experimental data or technique and calculational models, as well as the particular analyst. The sources of the uncertainty are cumulatively observed in the variability of the calculated $k_{eff}$ results obtained for the modeled critical experiments. Furthermore this variability includes the Monte Carlo standard deviation in each calculated $k_{eff}$ for the critical experiment, as well as changes in the calculated value due to consideration of the experimental uncertainties. Consequently the noted uncertainties are included in the bias and uncertainty in the bias.

This report calculates an estimate of the USL by means of two methods. A statistical average that assumes normal distribution of the data (Method 1) and a non-statistical method that uses the absolute minimum of the difference in the calculated $k_{eff}$ and the experimental result and the maximum standard deviation to calculate a conservative estimate of the USL (Method 2).

The techniques/tools currently used in this analysis are non-parametric. As such they are not suitable as-is for extending the range and area of applicability beyond the range and area of applicability of the benchmark experiments.

**Method 1**

The calculational bias (β) can be calculated by using the general definition:

$$K_c = k_{eff} + \beta \qquad \text{Equation 6-6}$$

Where $k_c$ represents the calculated estimate of $k_{eff}$. In this analysis a lower tolerance limit is established such that there is a 95% confidence that at least 95% of the population is above the limit. The prescription for evaluating this lower limit is given in the topical report as below:

$$\beta_L = \beta_{ave} - K_b s_b \qquad \text{Equation 6-7}$$

where $\beta_{ave}$ is the mean value of the difference between the value of the calculated $k_{eff}$ for the critical experiments, $k_c$, and 1.0 (i.e., average of $\beta = k_c - 1$) and $s_b$ is the corresponding pooled sample variance. The multiplier $K_b$ is normally found from statistical tables of the t-distribution and depends on the number of degrees of freedom (DOF) for the supporting measurement. The pooled sample variance is calculated by first combining the reported experimental uncertainty and the Monte Carlo variance as a statistical sum in the form:

$$\sigma_U = \sqrt{\varepsilon_b^{\,2} + \sigma_{MC}^{\,2}} \qquad \text{Equation 6-8}$$

where $\varepsilon_b$ is the reported benchmark model (experiment) uncertainty and $\sigma_{MC}$ is the Monte Carlo variance. Note that the Monte Carlo variance an normally be reduced by increasing the number of Monte Carlo histories whereas the experimental uncertainty is a constant. The standard deviation of the bias $\sigma_\beta$ is combined with $\sigma_U$ to obtain $s_b$ as.

$$s_b = \sqrt{\sigma_U^{\,2} + \sigma_\beta^{\,2}} \qquad \text{Equation 6-9}$$

Thus $\beta_L$ with all its terms can be written as:

$$\beta_L = \beta_{ave} + K_b \cdot \sqrt{\sigma_\beta^{\,2} + \sqrt{\varepsilon_b^{\,2} + \sigma_{MC}^{\,2}}} \qquad \text{Equation 6-10}$$

For the case that $\beta_{ave}$ is greater than 0 a positive bias is implied and $\beta_{ave}$ is set to 0.0 to account for the no positive bias rule. Note that equation 6-10 is analogous to equation 6-7. The plus sign in equation 6-10 is used in order to faciliate summation of all terms.

A value for the multiplier $K_b$ can only be determined for a known number of degrees-of-freedom. Nevertheless for a 95% confidence limit the standard t distribution yields a value for $K_b$ from 1.7 to 1.65 as the DOF ranges from 20-120 (see table in Beyer 1987, page 571). The same reference shows that $K_b$ for a 99% confidence limit ranges from 2.52 to 2.35 while DOF ranges from 20-120. For this evaluation $K_b$ = 3.0 is chosen. The rationale for this value is that it provides more conservatism for $\beta_L$ than the values listed above while at the same time allows to state with a 95% confidence that at least 95% of the sample population will fall above the resulting estimate for the USL over that particular group.

**Method 2**

In this method the Monte Carlo standard deviation is combined with the experimental model uncertainty via a standard statistical sum of the form:

$$\sigma = \sqrt{\sigma_e^2 + \sigma_c^2} \qquad \text{Equation 6-11}$$

where $\sigma_e$ is the experimental uncertainty and $\sigma_c$ is the MCNP standard deviation. The tables in Section 5 provide the value of σ.

A Statistically significant use of bias and σ would include evaluation of the variability in the bias and establishment of a confidence margin. For this method a conservative estimate of bias is made using the following formula:

$$bias_{est} = |bias_{\min}| + 3 \cdot \sigma_{\max} \qquad \text{Equation 6-12}$$

where $bias_{est}$ is the estimated conservative bias for the experiment grouping applicable to a DOE SNF type as given in Section 6.2 , $bias_{min}$ is the minimum of the bias values given in Section 5, and $\sigma_{max}$ is the maximum of the combined uncertainty, σ . The minus sign in front of $bias_{min}$ facilitates absolute summation of the two numbers and the multiplier in front of $\sigma_{max}$ is choosen to provide more conservatism in the estimate. The $bias_{est}$ value has no statistical significance except as a measure of the lower bound of the bias in the respective grouping for the present study.

The following Sections provide for each DOE SNF type a table with the estimated bias calculated as specified in Method 1 and Method 2 and according to the grouping provided in Section 6.2. The numbers have been obtained by inspecting of the results provided in the MCNP criticality calculation result listed in tables in Section 5 and performing the arithmetic manipulation shown in equations for Method 1 and Method 2 respectively. Plots of bias for each case of the experiments is also provided to facilitate a visual representation of the information presented in Section 5. The purpose of reporting these values is to provide means to compare between the different DOE SNF groups established in Section 6.2 only.

Table 6-33 Estimate of bias for DOE SNF groupings using method 1 and method 2

| Fuel Type/Grouping | Method 1 | | Method 2 | |
|---|---|---|---|---|
| | $\beta_L$ | $1-\beta_L$ | $bias_{est}$ | $1-bias_{est}$ |
| FFTF (MOX)/ Intact Thermal | 0.0252 | 0.9748 | 0.0449 | 0.9551 |
| FFTF (MOX)/ Intact Fast&Int. | 0.0369 | 0.9631 | 0.0367 | 0.9633 |
| FFTF (MOX)/ Degr. Thermal | 0.0257 | 0.9743 | 0.0361 | 0.9639 |
| FFTF (MOX)/ Degr. Fast&Int. | 0.0182 | 0.9818 | 0.0278 | 0.9722 |
| Melt&Dilute/Intact Thermal | 0.0176 | 0.9824 | 0.0243 | 0.9757 |
| Melt&Dilute/Intact Fast&Int. | 0.0140 | 0.9860 | 0.0201 | 0.9799 |
| Melt&Dilute/Degr. Thermal | 0.0216 | 0.9784 | 0.0450 | 0.9550 |
| Melt&Dilute/Degr. Fast&Int. | 0.0140 | 0.9860 | 0.0201 | 0.9799 |
| SPWR(HEU)/Intact Thermal | 0.0289 | 0.9711 | 0.0449 | 0.9551 |
| SPWR(HEU)/Intact Fast&Int. | 0.0277 | 0.9773 | 0.0238 | 0.9762 |
| SPWR(HEU)/Degr. Thermal | 0.0277 | 0.9723 | 0.0521 | 0.9479 |
| SPWR(HEU)/Degr. Fast&Int. | 0.0231 | 0.9769 | 0.0352 | 0.9648 |

Table 6-33 Estimate of bias for DOE SNF groupings using method 1 and method 2

| Fuel Type/Grouping | Method 1 | | Method 2 | |
|---|---|---|---|---|
| | $\beta_L$ | $1-\beta_L$ | $bias_{est}$ | $1-bias_{est}$ |
| TRIGA(UZrH)/Intact Thermal | 0.0338 | 0.9662 | 0.0449 | 0.9551 |
| TRIGA(UZrH)/Intact Fast&Int. | 0.0359 | 0.9641 | 0.0527 | 0.9473 |
| TRIGA(UZrH)/Degr. Thermal | 0.0319 | 0.9681 | 0.0541 | 0.9459 |
| TRIGA(UZrH)/Degr. Fast&Int. | 0.0222 | 0.9778 | 0.0352 | 0.9648 |
| FSVrain(Graphite/Carbide)/Intact Thermal | 0.0339 | 0.9661 | 0.0395 | 0.9605 |
| FSVrain(Graphite/Carbide)/Intact Fast&Int. | 0.0224 | 0.9776 | 0.0334 | 0.9666 |
| FSVrain(Graphite/Carbide)/Degr. Thermal | 0.0346 | 0.9654 | 0.0619 | 0.9381 |
| FSVrain(Graphite/Carbide)/Degr. Fast&Int. | 0.0309 | 0.9691 | 0.0522 | 0.9478 |
| Fermi(U Alloy)/Intact Thermal | 0.0237 | 0.9763 | 0.0439 | 0.9561 |
| Fermi(U Alloy)/Intact Fast&Int. | 0.0140 | 0.9860 | 0.0201 | 0.9799 |
| Fermi(U Alloy)/Degr. Thermal | 0.0215 | 0.9785 | 0.0450 | 0.9550 |
| Fermi(U Alloy)/Degr. Fast&Int. | 0.0140 | 0.9860 | 0.0201 | 0.9799 |
| SLWBR( ThU )/Intact Thermal | 0.0300 | 0.9700 | 0.0439 | 0.9561 |
| SLWBR( ThU )/Intact Fast&Int. | 0.0220 | 0.9780 | 0.0334 | 0.9661 |
| SLWBR( ThU )/Degr. Thermal | 0.0325 | 0.9675 | 0.0619 | 0.9381 |
| SLWBR( ThU )/Degr. Fast&Int. | 0.0309 | 0.9691 | 0.0522 | 0.9478 |
| N Reactor ( U Metal)/ Intact Thermal & Int. | 0.0314 | 0.9686 | 0.0247 | 0.9753 |
| N Reactor ( U Metal)/ Degraded Thermal & Int. | 0.0237 | 0.9763 | 0.0333 | 0.9667 |
| TMI ( LEU)/ Intact Thermal & Int. | 0.0173 | 0.9827 | 0.0451 | 0.9549 |
| TMI ( LEU)/ Degraded Thermal & Int. | 0.0206 | 0.9794 | 0.0333 | 0.9667 |

Based on the statistical averaging method (Method 1) the lowest USL for the groups as given in section 6.2 is found to be 0.9631, corresponding to the (FFTF (MOX)/ Intact Fast&Int.) Fuel type/Grouping. The average for all groups per Method 1 as given in table above is 0.9749. The lowest USL based on Method 2 is 0.9381 corresponding to the FSVrain and SLWBR Degraded Thermal Modes. The average for all groups per Method 2 as given in the table above is 0.9611.

Note that Method 1 is not sensitive to removing or adding experiments to a given group since it uses statistical averaging while Method 2 is extremely sensitive to removing or adding experiments/cases. For example removing case ust03-41 (U233-Sol-Therm-003 case 41) from the group would change 0.9381 to 0.9456 while the result for Method 1 would change from 0.9604 to 0.9612 if the same case were omitted from the grouping.

The results given in Table 6-33 are useful for comparing the groups against each other and estimating a conservative USL for the experiments/cases within a given group. But they are not meant to be as a absolute validation for the calculational models used for the different DOE SNF

fuel types. The effect of the range of physical parameters important to criticality (such as enrichment, H/fissile isotope) as used in the models, to the computed $k_{eff}$ needs to be established in order to provide a meaningful USL for the models developed.

### 6.3.1 Plots of bias for the DOE SNF Groupings in Section 6.2

The following plots provide a graphical representation of $\Delta k_{eff}$ (MCNP Model Calculation $k_{eff}$ – Benchmark Model $k_{eff}$) presented in the tables in Section 5. The information provided in this Section is the same as that provided in Section 5, except that each plot provides the $\Delta k_{eff}$ for the cases contained in the groupings in Section 6.2.

#### 6.3.1.1 Plot of Bias for Benchmark Experiments Applicable to HEU Oxide (Shippingport-PWR) DOE-Owned Fuel.

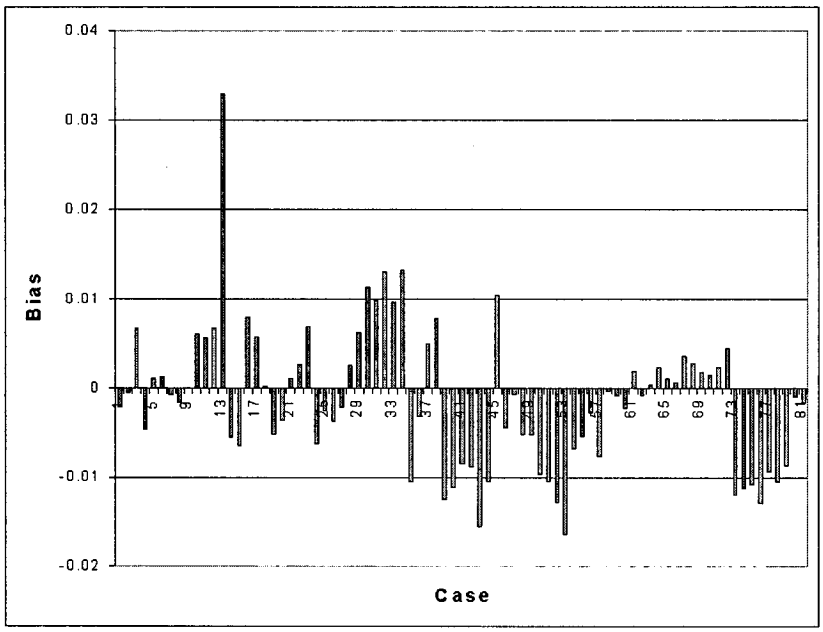


Figure 6-1 Bias for Benchmarks Applicable to HEU Oxide DOE-Owned Intact Fuel (Thermal Spectrum)

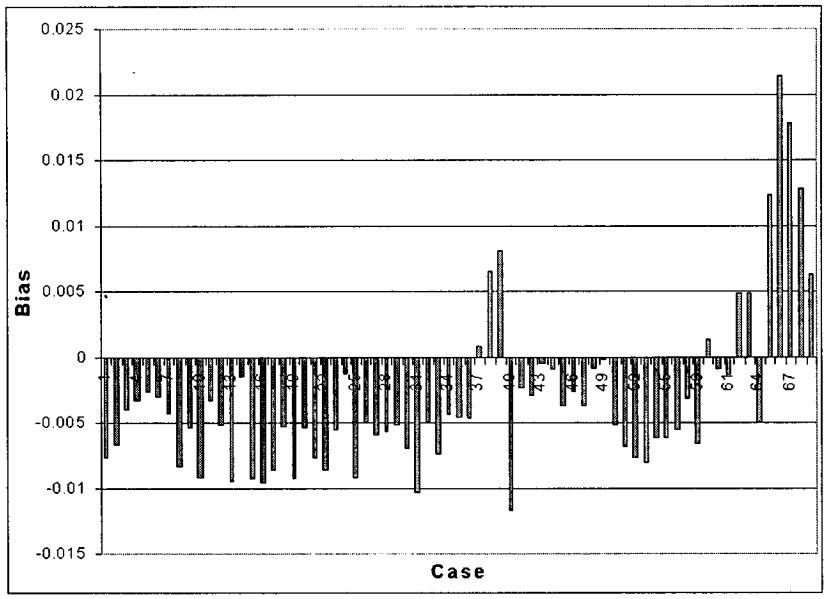


Figure 6-2 Bias for Benchmarks Applicable to HEU Oxide DOE SNF ( Intermediate and Fast Spectrum)

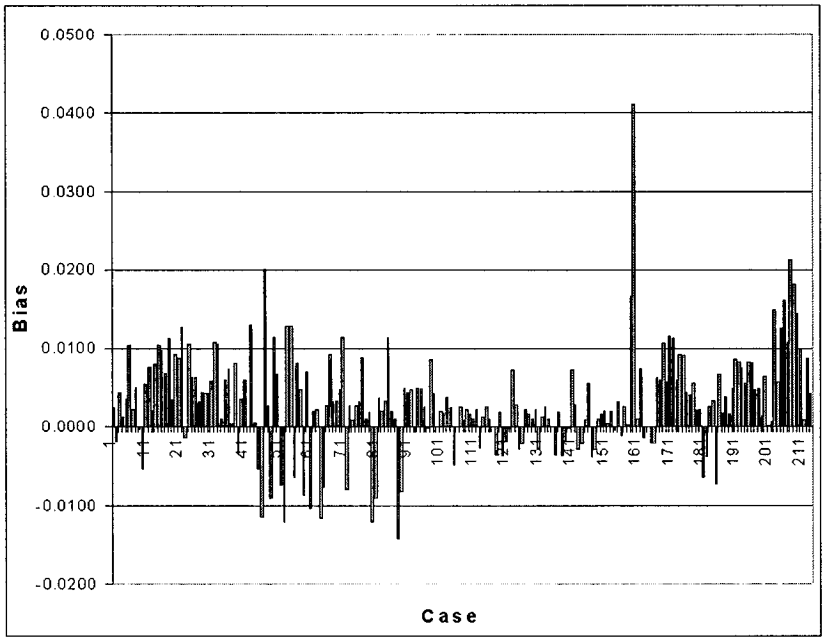


Figure 6-3 Bias for Benchmarks Applicable to HEU Oxide Degraded DOE-SNF (Thermal Spectrum)

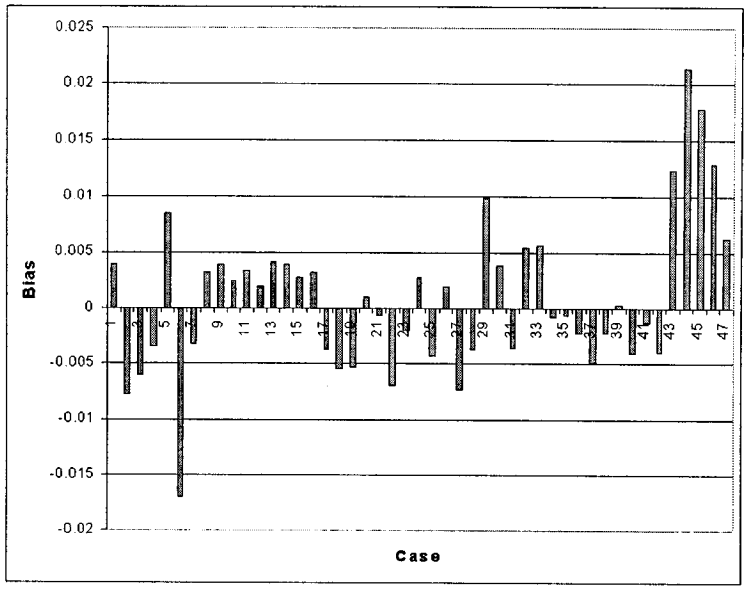


Figure 6-4 Bias for Benchmarks Applicable to HEU Oxide Degraded DOE-SNF (Intermediate and Fast Spectrum)

### 6.3.1.2 Plot of Bias for Benchmark Experiments Applicable to UZrH (TRIGA) DOE-Owned Fuel.

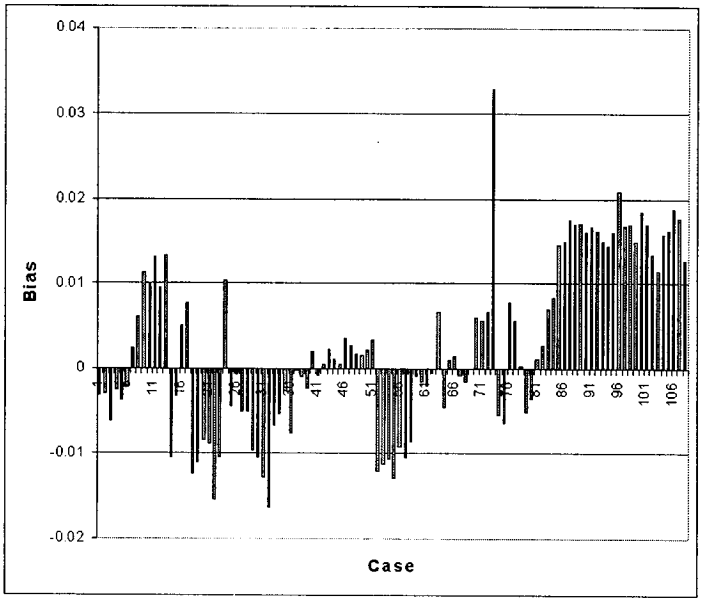


Figure 6-5 Bias for Benchmarks Applicable to UZrH DOE-Owned Intact Fuel in Thermal Spectrum

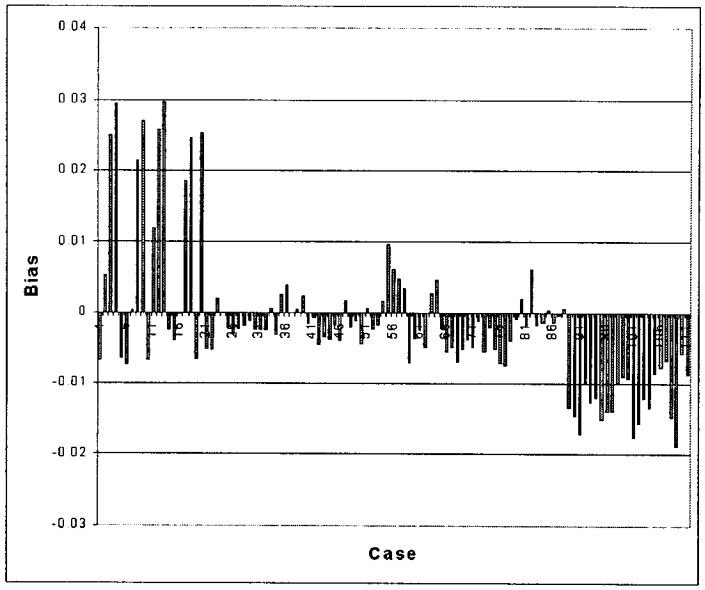


Figure 6-6 Bias for Benchmarks Applicable to UZrH DOE-Owned Intact Fuel in Intermediate and Fast Spectrum

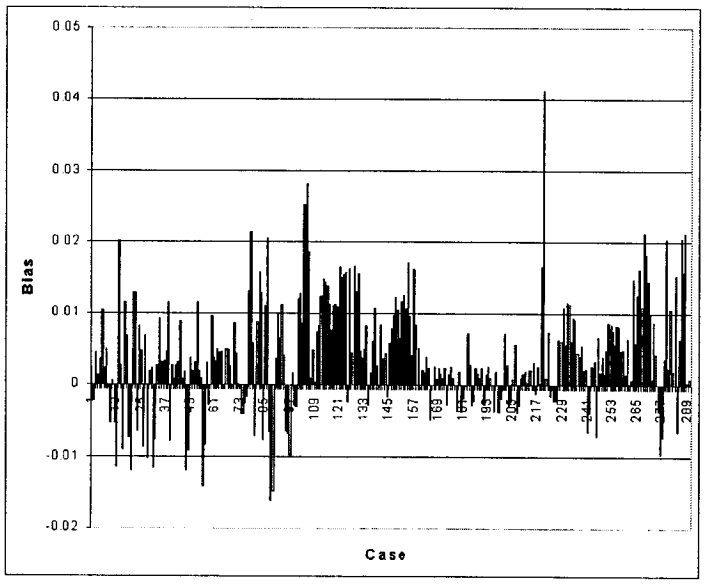


Figure 6-7 Bias for Benchmarks Applicable to UZrH DOE-Owned Degraded Fuel in Thermal Spectrum

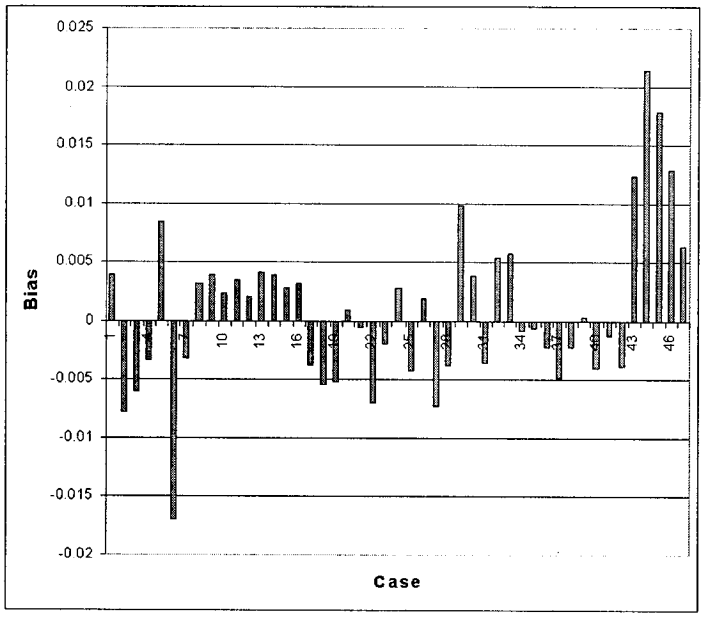


Figure 6-8 Bias for Benchmarks Applicable to UZrH DOE-Owned Degraded Fuel in Intermediate and Fast Spectrum

### 6.3.1.3 Plot of Bias for Benchmark Experiments Applicable to Melt and Dilute DOE-Owned Fuel.

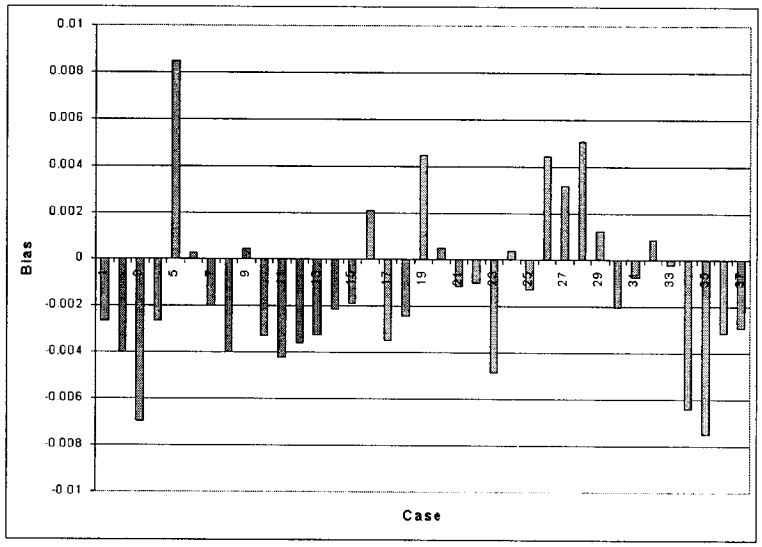


Figure 6-9 Bias for Benchmarks Applicable to Melt and Dilute DOE-Owned Intact Fuel in Thermal Spectrum

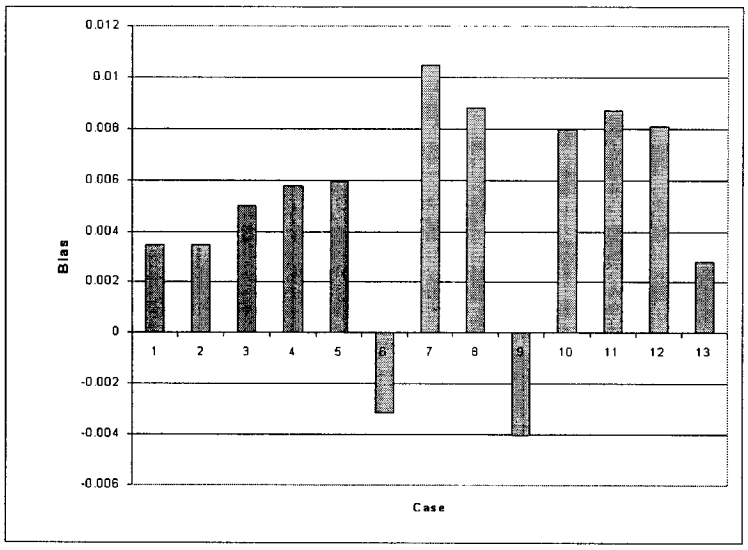


Figure 6-10 Bias for Benchmarks Applicable to Melt and Dilute DOE-Owned Intact Fuel in Intermediate and Fast Spectrum

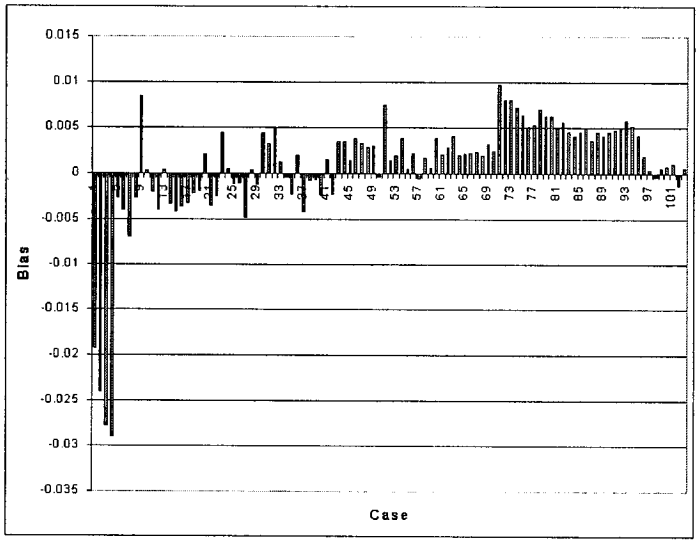


Figure 6-11 Bias for Benchmarks Applicable to Melt and Dilute DOE-Owned Degraded Fuel in Thermal Spectrum

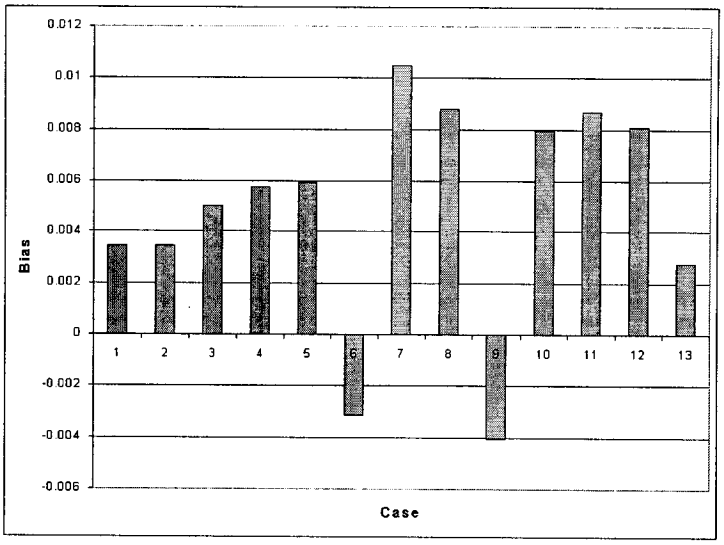


Figure 6-12 Bias for Benchmarks Applicable to Melt and Dilute DOE-Owned Degraded Fuel in Intermediate and Fast Spectrum

### 6.3.1.4 Plot of Bias for Benchmark Experiments Applicable to U Alloy (Enrico Fermi) DOE-Owned Fuel.

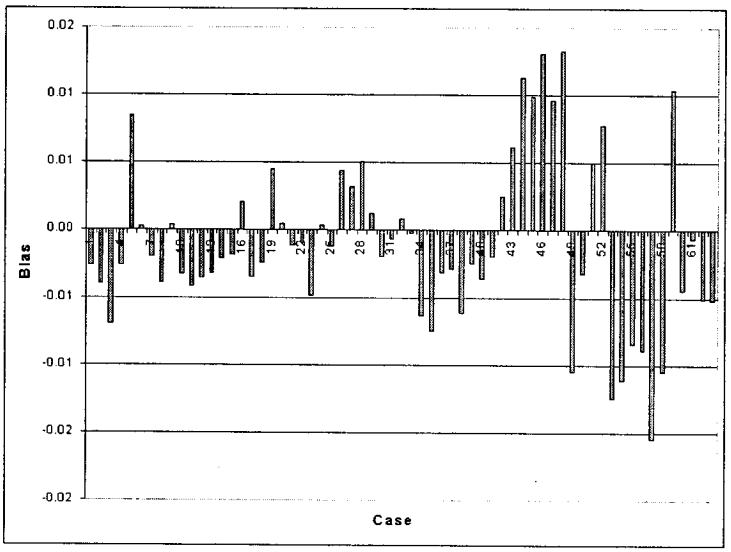


Figure 6-13 Bias for Benchmarks Applicable to U Alloy DOE-Owned intact Fuel in Thermal Spectrum

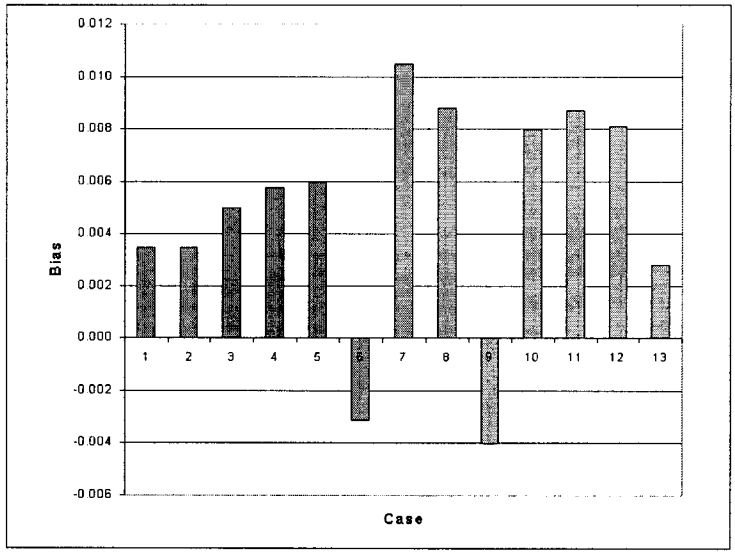


Figure 6-14 Bias for Benchmarks Applicable to U Alloy DOE-Owned intact Fuel in Intermediate and Fast Spectrum

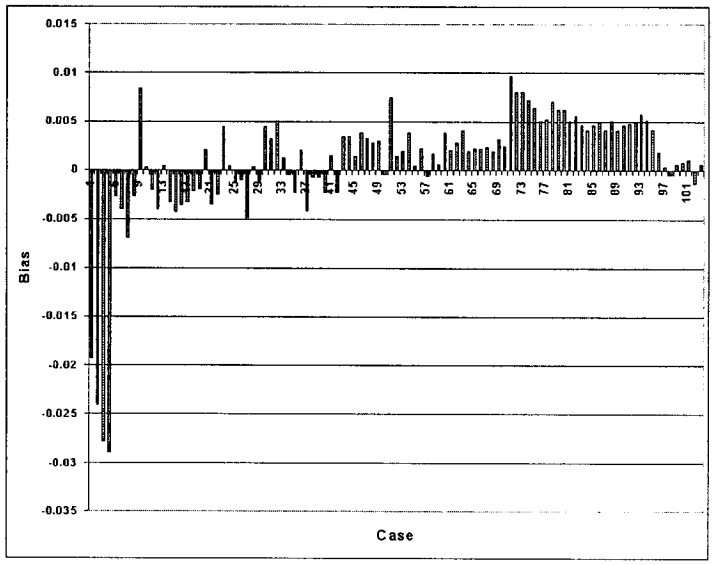


Figure 6-15 Bias for Benchmarks Applicable to U Alloy DOE-Owned Degraded Fuel in Thermal Spectrum.

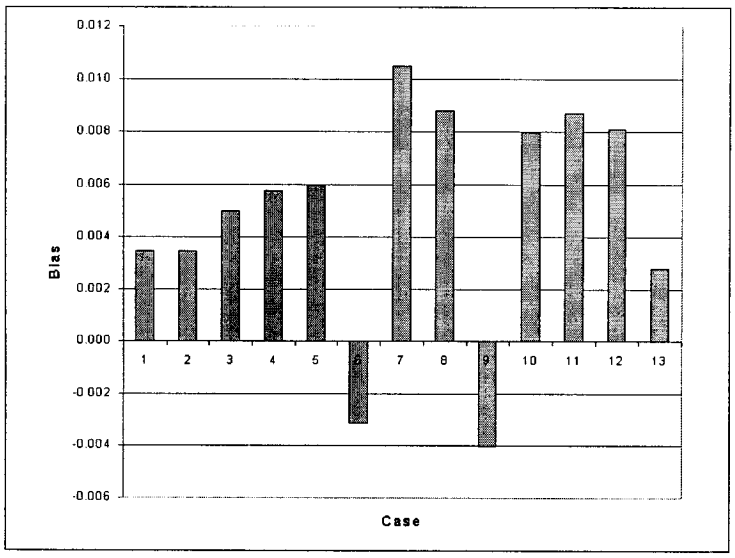


Figure 6-16 Bias for Benchmarks Applicable to U Alloy DOE-Owned Degraded Fuel in Intermediate and Fast Spectrum

### 6.3.1.5 Plot of Bias for Benchmark Experiments Applicable to MOX (FFTF) DOE-Owned Fuel.

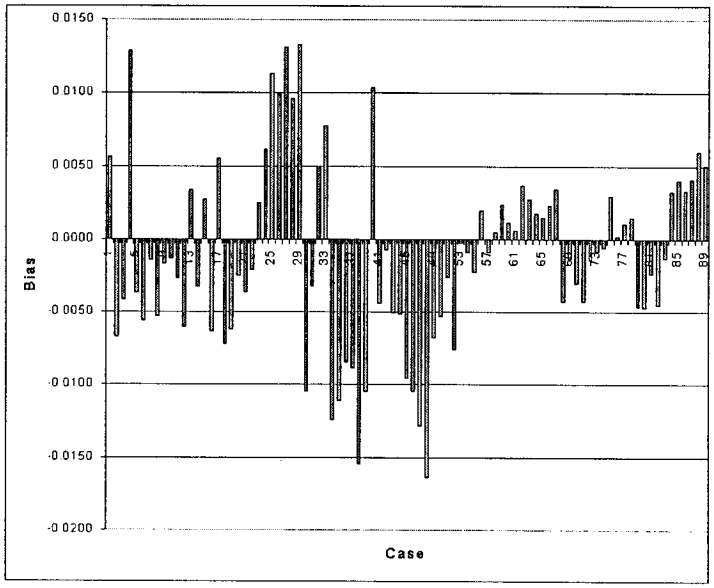


Figure 6-17 Bias for Benchmarks Applicable to MOX DOE-Owned intact Fuel in Thermal Spectrum.

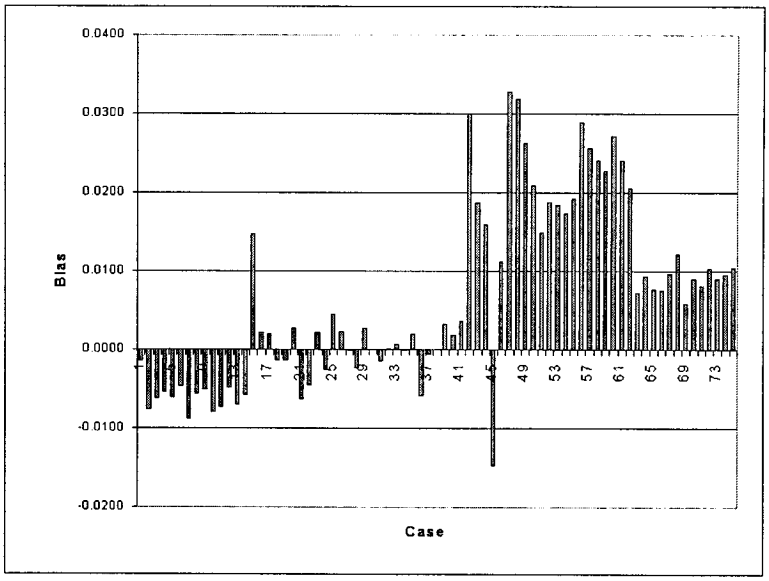


Figure 6-18 Bias for Benchmarks Applicable to MOX DOE-Owned intact Fuel in Intermediate and Fast Spectrum

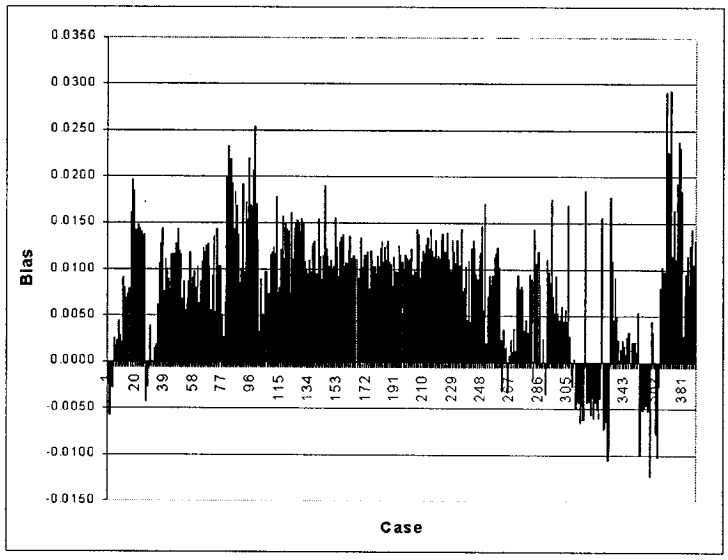


Figure 6-19 Bias for Benchmarks Applicable to MOX DOE-Owned Degraded Fuel in Thermal Spectrum

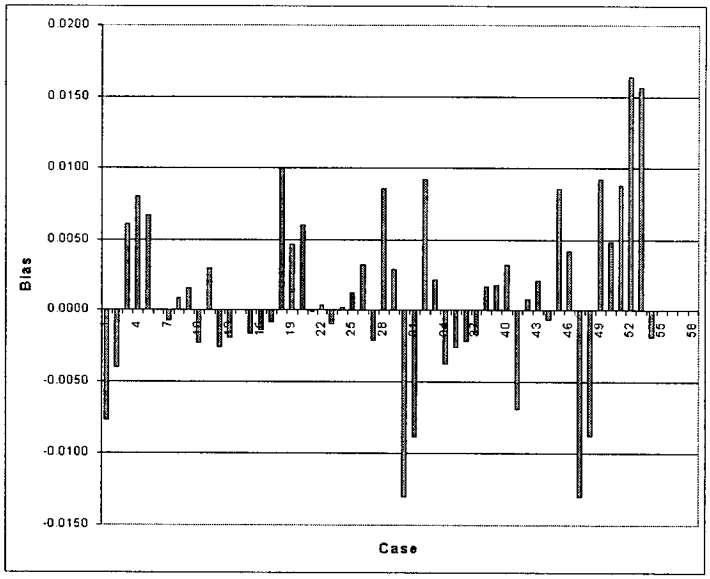


Figure 6-20 Bias for Benchmarks Applicable to MOX DOE-Owned Degraded Fuel in Intermediate and Fast Spectrum

### 6.3.1.6 Plot of Bias for Benchmark Experiments Applicable to Graphite/Carbide (Ft. St. Vrain) DOE-Owned Fuel.

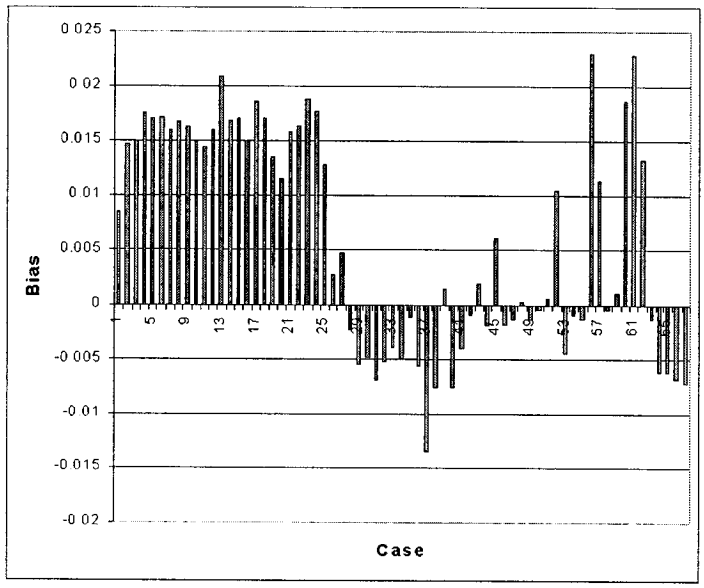


Figure 6-21 Bias for Benchmarks Applicable to Graphite/Carbide DOE-Owned intact Fuel in Thermal Spectrum.

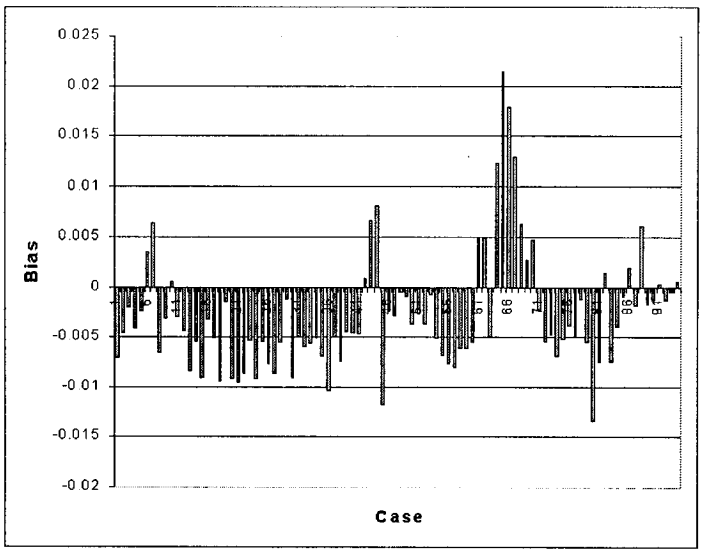


Figure 6-22 Bias for Benchmarks Applicable to Graphite/Carbide DOE-Owned intact Fuel in Intermediate and Fast Spectrum.

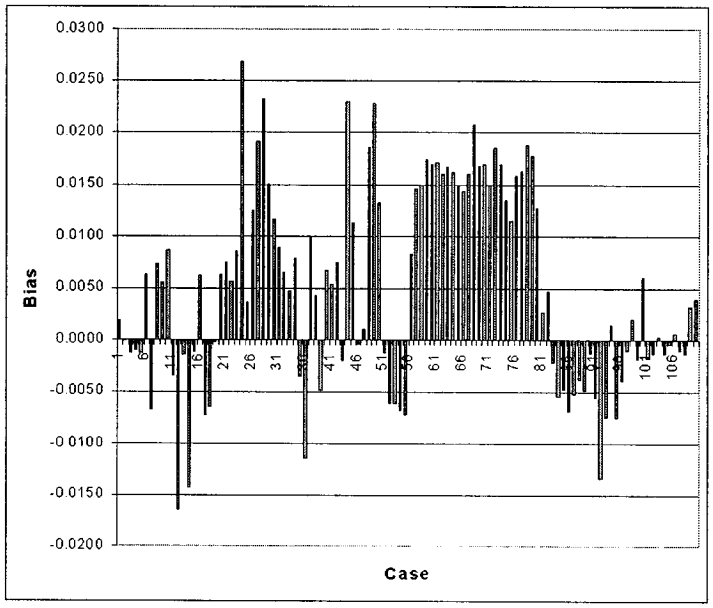


Figure 6-23 Bias for Benchmarks Applicable to Graphite/Carbide DOE-Owned Degraded Fuel in Thermal Spectrum.

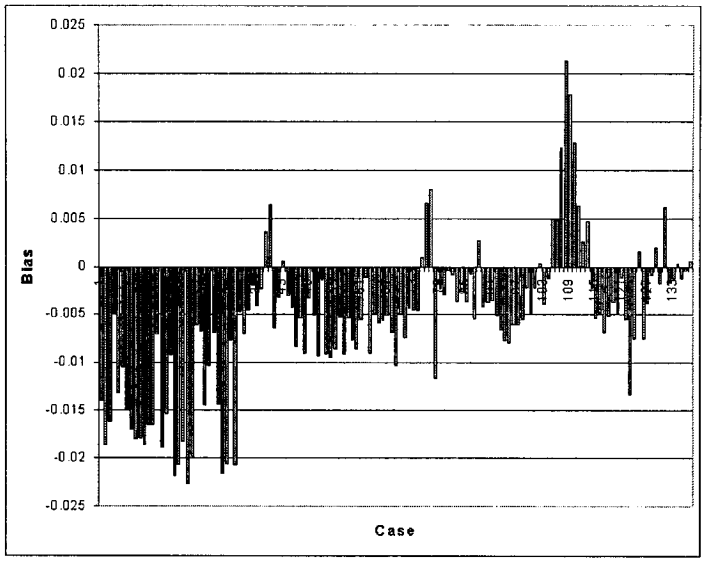


Figure 6-24 Bias for Benchmarks Applicable to Graphite/Carbide DOE-Owned Degraded Fuel in Intermediate and Fast Spectrum.

### 6.3.1.7 Plot of Bias for Benchmark Experiments Applicable to Th/U (Shippingport LWBR) DOE-Owned Fuel.

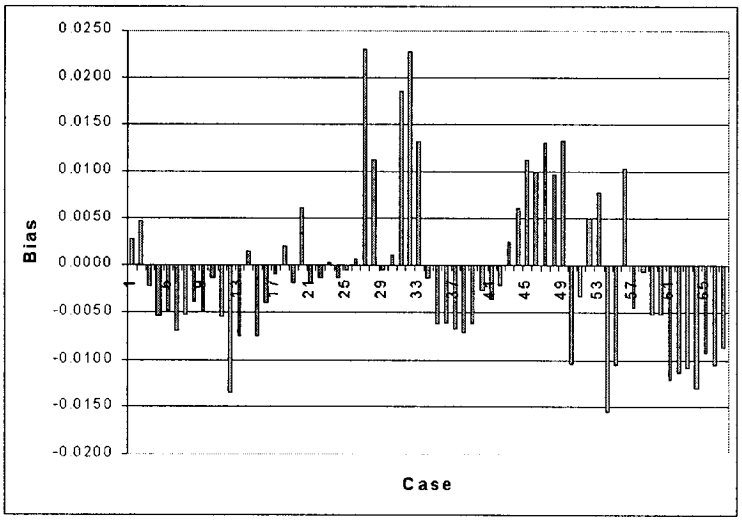


Figure 6-25 Bias for Benchmarks Applicable to Th/U DOE-Owned intact Fuel in Thermal Spectrum.

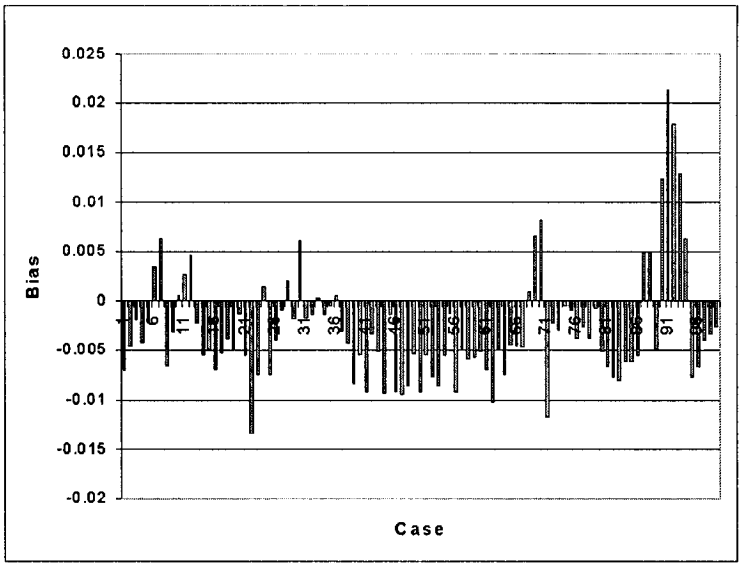


Figure 6-26 Bias for Benchmarks Applicable to Th/U DOE-Owned intact Fuel in Intermediate and Fast Spectrum.

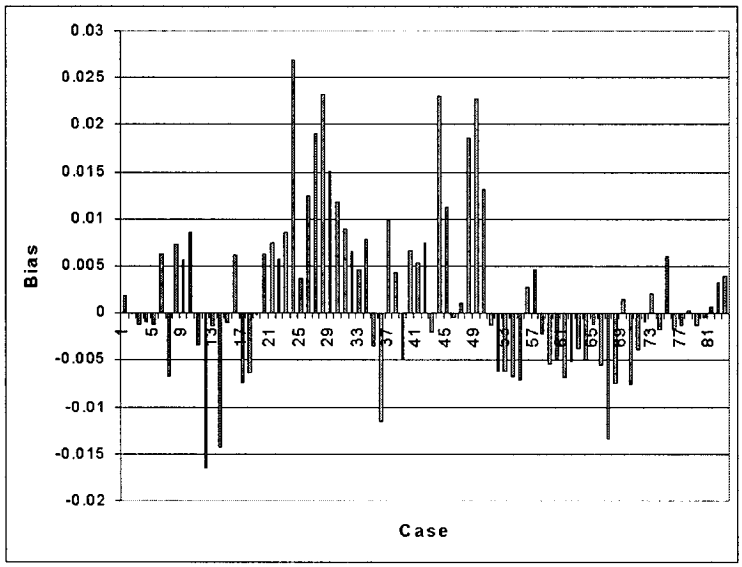


Figure 6-27 Bias for Benchmarks Applicable to Th/U DOE-Owned Degraded Fuel in Thermal Spectrum.

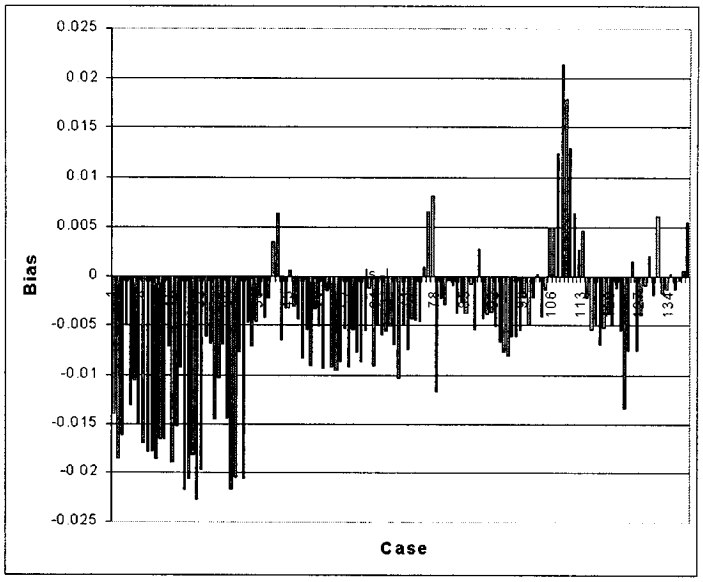


Figure 6-28 Bias for Benchmarks Applicable to Th/U DOE-Owned Degraded Fuel in Intermediate and Fast Spectrum.

### 6.3.1.8 Plot of Bias for Benchmark Experiments Applicable to U-metal (N Reactor) DOE-Owned Fuel.

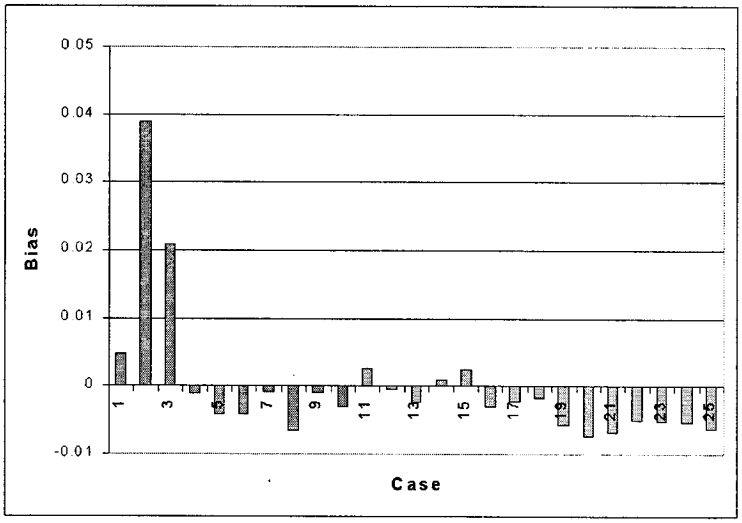


Figure 6-29 Bias for Benchmarks Applicable to U-metal DOE-Owned intact Fuel in Thermal and Intermediate Spectrum.

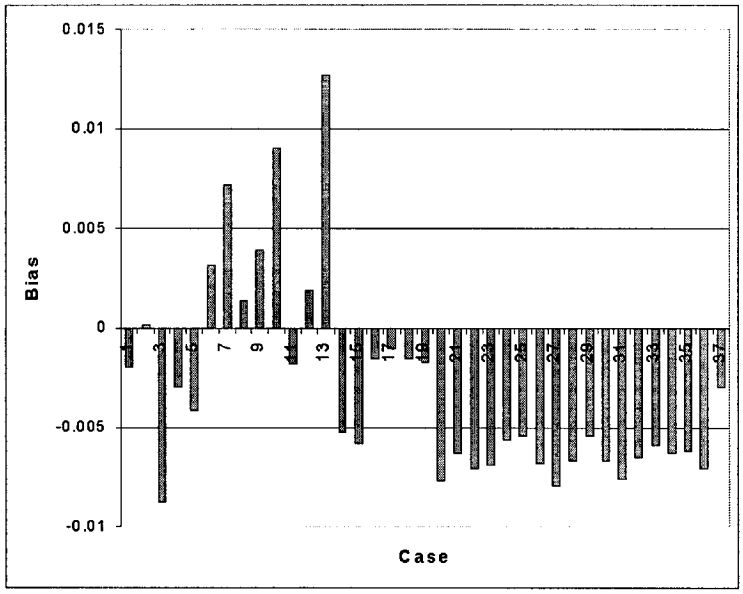


Figure 6-30 Bias for Benchmarks Applicable to U-metal DOE-Owned Degraded Fuel in Thermal and Intermediate Spectrum.

### 6.3.1.9 Plot of Bias for Benchmark Experiments Applicable to LEU Oxide (N Reactor) DOE-Owned Fuel.

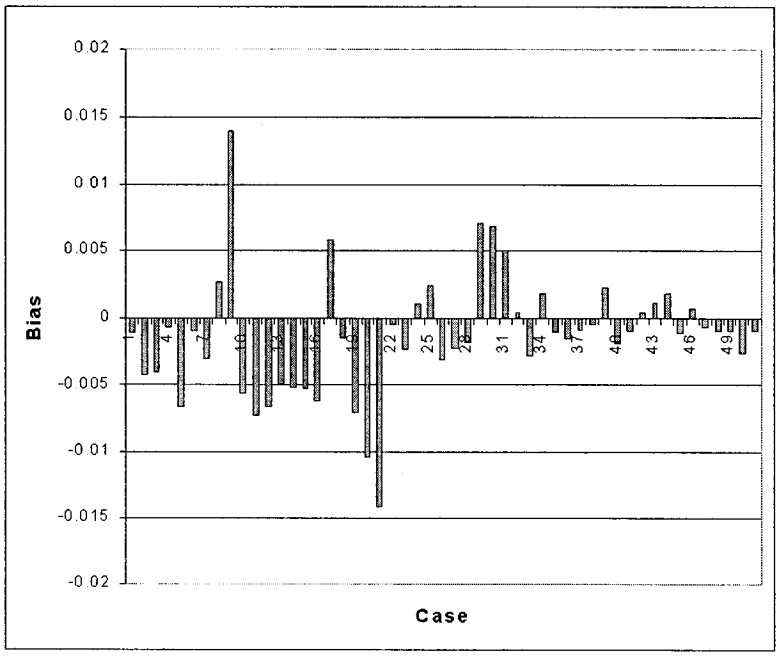


Figure 6-31 Bias for Benchmarks Applicable to LEU Oxide DOE-Owned intact Fuel in Thermal and Intermediate Spectrum.

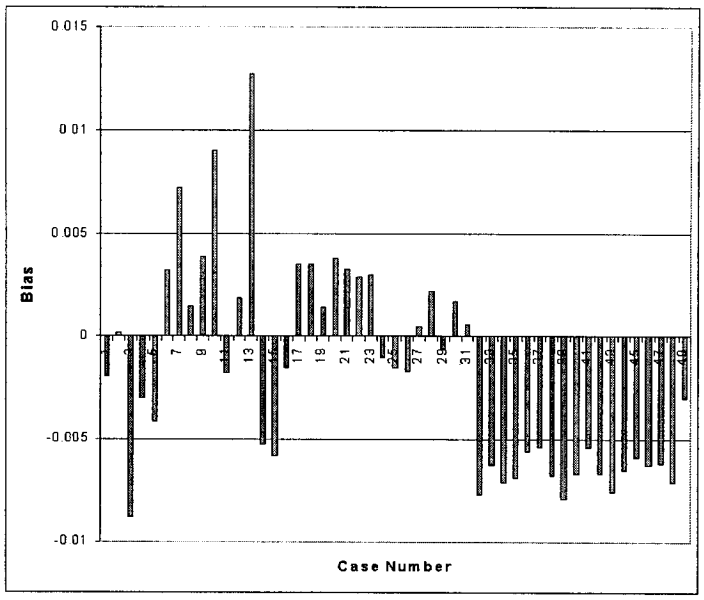


Figure 6-32 Bias for Benchmarks Applicable to LEU Oxide DOE-Owned intact Fuel in Thermal and Intermediate Spectrum.

# 7. REFERENCES

## 7.1 DOCUMENTS CITED

Beyer, W.H., ed. 1987. *CRC Standard Mathematical Tables.* 28th Edition. 3rd Printing 1988. Boca Raton, Florida: CRC Press. TIC: 240507.

Briesmeister, J.F., ed. 1997. *MCNP–A General Monte Carlo N-Particle Transport Code.* LA-12625-M, Version 4B. Los Alamos, New Mexico: Los Alamos National Laboratory. ACC: MOL.19980624.0328.

BSC (Bechtel SAIC Company) 2002. *Technical Work Plan for: Department of Energy Spent Nuclear Fuel Work Packages.* TWP-MGR-MD-000010 REV 01. Las Vegas, Nevada: Bechtel SAIC Company. URN-1028.

CRWMS M&O 1997. *MCNP Evaluation of Laboratory Critical Experiments: Homogeneous Mixture Criticals.* BBA000000-01717-0200-00045 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19971230.0134.

CRWMS M&O 1998a. *Software Qualification Report for MCNP Version 4B2, A General Monte Carlo N-Particle Transport Code.* CSCI: 30033 V4B2LV. DI: 30033-2003, Rev. 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980622.0637.

CRWMS M&O 1998b. *Software Code: MCNP.* 4B2LV. HP. 30033 V4B2LV.

CRWMS M&O 1999a. *LCE for Research Reactor Benchmark Calculations.* B00000000-01717-0210-00034 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990329.0394.

CRWMS M&O 1999b. *Laboratory Critical Experiment Reactivity Calculations.* B00000000-01717-0210-00018 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990526.0294.

CRWMS M&O 1999c. *Summary Report of Laboratory Critical Experiment Analyses Performed for the Disposal Criticality Analysis Methodology.* B00000000-01717-5705-00076 REV 02. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990920.0167.

CRWMS M&O 1999d. *Evaluation of Codisposal Viability for MOX (FFTF) DOE-Owned Fuel.* BBA000000-01717-5705-00023 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991014.0235.

CRWMS M&O 2000a. *Evaluation of Codisposal Viability for UZrH (TRIGA) DOE-Owned Fuel.* TDR-EDC-NU-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000207.0689.

CRWMS M&O 2000b. *Evaluation of Codisposal Viability for U-Zr/U-Mo Alloy (Enrico Fermi) DOE-Owned Fuel.* TDR-EDC-NU-000002 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000815.0317.

CRWMS M&O 2000c. *Evaluation of Codisposal Viability for HEU Oxide (Shippingport PWR) DOE-Owned Fuel.* TDR-EDC-NU-000003 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000227.0240.

CRWMS M&O 2000d. *Evaluation of Codisposal Viability for Th/U Oxide (Shippingport LWBR) DOE-Owned Fuel.* TDR-EDC-NU-000005 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20001023.0055.

CRWMS M&O 2001. *Evaluation of Codisposal Viability for U-Metal (N-Reactor) DOE-Owned Fuel.* TDR-EDC-NU-000004 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20010314.0004.

DOE (U.S. Department of Energy) 2000. Quality Assurance Requirements and Description. DOE/RW-0333P, Rev. 10. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20000427.0422.

NEA (Nuclear Energy Agency) 2001. *International Handbook of Evaluated Criticality Safety Benchmark Experiments.* 2001 Edition. NEA/NSC/DOC(95)03. Paris, France: Nuclear Energy Agency, Organization for Economic Co-operation and Development. TIC: 251945.

Radulescu, H.R. 2001a. *Evaluation of Codisposal Viability for Melt and Dilute DOE-Owned Fuel.* TDR-EDC-NU-000006 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010809.0070.

Radulescu, H.R. 2001b. *Evaluation of Codisposal Viability for Th/U Carbide (Fort Saint Vrain HTGR) DOE-Owned Fuel.* TDR-EDC-NU-000007 REV 00. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20011017.0092.

YMP (Yucca Mountain Site Characterization Project) 2000. *Disposal Criticality Analysis Methodology Topical Report.* YMP/TR-004Q, Rev. 01. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: MOL.20001214.0001.

## 7.2 CODES, STANDARDS, REGULATIONS, AND PROCEDURES

ANS-8.7/ANSI N16.5-1975(R1987). 1975. *American National Standard for Nuclear Criticality Safety in the Storage of Fissile Materials.* La Grange Park, Illinois: American Nuclear Society. TIC: 205016.

AP-SI.1Q Rev. 3, ICN 3. *Software Management.* Washington, D.C.: U.S. Department of Energy. ACC: MOL.20020102.0200.

## 8. ATTACHMENTS

Attachment I: One Compact Disk (CD) containing the MCNP input and output files for the critical benchmarks denoted in Section 5.

Attachment II: Description of files contained in Attachment I (2 pages).

## ATTACHMENT II

This attachment contains description of the zip file contained on the attachment CD of this calculation (Attachment I). The CD was written using a Sony Model CRX160E Internal CD-rewritable drive for personal computers.

The Zip archive was created using ZipCentral version 2.06. The zip file attributes on the CD are as follows:

| Archive Filename | File Size (bytes) | File Date | File Time |
|---|---|---|---|
| doebench.zip | 113,490,085 | 03/29/02 | 7:43a |

There are 2138 files contained in a unique directory structure. Upon file extraction the top level directories will correspond as follows:

/inputs/ - contains the MCNP input files
/outputs/ - contains the MCNP output files

The files contained under the directories */inputs/* and */outputs/* are for the MCNP input and output files respectively, for the benchmark model calculations performed in this analysis for the experiments denoted as "unevaluated" in Section 5.

The output files are denoted the same as the input files but have an "*.out*" at the end of the name. The files are arranged in a directory structure that separates files with the same filenames. The directory structure is as follows:

| | |
|---|---|
| /Heu/: | MCNP benchmark model pertaining to Highly Enriched Uranium Critical Experiments. |
| /IEU&LEu/: | MCNP benchmark model pertaining to Intermediately and Low Enriched Uranium Critical Experiments. |
| /Pu&Mix/: | MCNP benchmark model pertaining to Plutonium and MOX Critical Experiments. |
| /U233/: | MCNP benchmark model pertaining to U-233 Critical Experiments |

For each Experiment Set the Folder names are abbreviated versions of the Experiment Classification from NEA 2001 in the form XXXYY.ZZZ where XXX is either HEU, IEU, LEU, U-233, Pu or MIX and YY is from the following table

| YY | Classification |
|---|---|
| CI | COMP-INTER |
| CM | COMP-MIXED |
| CT | COMP-THERM |
| MI | MET-INTER |
| MF | MET-FAST |
| MM | MET-MIXED |
| ST | SOL-THERM |
| SI | SOL-INTER |

ZZZ is the critical experiment number for the XXXYY critical experiment class as given in the Section 5 of the calculation. For example using this abbreviation the input cases for critical benchmark experiment *HEU-COMP-INTER-001* are under the directory *HEUCI.001*

The file names under each XXXYY.ZZZ critical experiment correspond to the case names provided in the MCNP results tables in the relevant section of the document.

# OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT
## SPECIAL INSTRUCTION SHEET

1. QA: ~~N/A~~ QA 4-16-02

Page: 1 of: 1

filelist 4-16-02 nfc

***Complete Only Applicable Items***

**This is a placeholder page for records that cannot be scanned.**

| | |
|---|---|
| 2. Record Date<br>03/29/2002 | 3. Accession Number<br>ATT TO: MOL.20020416.0053 |
| 4. Author Name(s)<br>MEHMET SAGLAM | 5. Author Organization<br>N/A |

6. Title/Description
BECHMARK AND CRITICAL LIMIT CALCULATION FOR DOE SNF

(See Addendum 1)

| | |
|---|---|
| 7. Document Number(s)<br>CAL-EDC-NU-000008 | 8. Version Designator<br>REV. 0 |
| 9. Document Type<br>DATA | 10. Medium<br>CD-ROM DISK |

11. Access Control Code
PUBLIC

12. Traceability Designator
DC# 31197

13. Comments
THIS DOCUMENT IS ONE OF A KIND DUE TO (ONE) CD-DISK

THIS DATA SUBMITTAL TO THE
RECORDS PROCESSING CENTER IS
FOR ARCHIVE PURPOSES ONLY, AND
IS NOT AVAILABLE FOR VIEWING OR
REPRODUCTION