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Center #: 10/24-6-R7057-0A0  
Contract#: SA 128069  
Prime #:  
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Title: AN ASSESSMENT OF MODULAR HIGH TEMPERATURE GAS COOLED REACTOR FOR ACTINIDE...  

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NOTICE OF PROJECT CLOSEOUT

Closeout Notice Date 12/10/92

Project No. E-25-677

Center No. 10/24-6-R7057-0A0_

Project Director LANCASTER D

School/Lab MECH ENGR_____

Sponsor GENERAL ATOMICS/SAN DIEGO, CA

Contract/Grant No. SA 128069

Contract Entity GTRC

Prime Contract No. 

Title AN ASSESSMENT OF MODULAR HIGH TEMPERATURE GAS COOLED REACTOR FOR ACTINIDE

Effective Completion Date 911031 (Performance) 911031 (Reports)

Closeout Actions Required:                Date Submitted

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                     FRED CAIN-OOD       Y
Final Report to the
General Atomics Corporation

on

AN ASSESSMENT OF THE MODULAR HIGH TEMPERATURE GAS COOLED REACTOR FOR ACTINIDE BURNING

by

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October 15, 1992
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I. Executive Summary

Actinide burning in MHTGRs was evaluated relative to a PWR, and LMR. The first analysis performed showed that the MHTGR was the best performing reactor for minimizing its own actinides. This result was mainly due to the MHTGR using thorium as the fertile material. The second analysis looked at using an MHTGR to burn actinides from LWRs. In this case the MHTGR showed inferior performance to the PWR, so further analysis was halted. The poor performance was due to the lack of self shielding of the large low energy resonances of Np-237, Pu-240, and Am-241. This lack of self shielding made it impossible to achieve criticality on the mix of PWR transuranic actinides. When Pu was separated from the other transuranic actinides it was possible to be critical with 91% Pu and 9% minor actinides. Unfortunately when this mixture was burned it was not possible to reach criticality for the next cycle without removing some of the minor actinides. Although burning actinides in a MHTGR is possible, it is clear that the performance is inferior to that of a PWR.
II. Introduction

Actinide burning has been proposed as a way of reducing the long term effects of high level radioactive waste. The fast reactor community has seized upon this concept in order to promote the construction of LMRs. Indeed it was hoped that this mission could save the FFTF for a few years. It is clear that in a fast spectrum the actinides have a higher fission cross section and pose a lower reactivity penalty; however, if LMRs are to breed then the total quantity of transuranic actinides actually increases. If LMRs are not to breed then the low reactivity penalty of the actinides is of little benefit. This study was commissioned to investigate burning actinides in MHTGRs and compare that to LMRs and PWRs.

The most difficult question in most actinide burning studies is to define exactly what the objective should be. For some, the assumption is that Pu is good and it is desired is to get rid of non-Uranium/non-Plutonium (or non-Thorium) actinides. This assumption is favored by those who see LMRs in our future and therefore the need for Pu. This assumption is also favored by those who believe that we will be burning Pu in LWRs, as is already proceeding in other parts of the world. This assumption assumes reprocessing and the objective is to decrease the toxicity of the
reprocessed waste actinides (ie: non-U/Pu/Th). Another point of view comes from those that believe that all comparisons should be made to the current once through cycle. This is where the LMRs get into trouble. From this point of view Pu is a waste material. This work both considers both points of view.

Since actinide burning requires more than ten cycles to approach equilibrium, spatial studies are prohibitively expensive. Although spatial studies have been attempted for the LMR, no such studies have been performed for thermal reactors. This implies any actinide burning study must make significant assumptions. Two key assumptions found in previous work have been: 1) Constant one group cross sections, and 2) Constant beginning of life (BOL) $k$. The constant one group cross section assumption is normally made with the ORIGEN code. Since the isotopics are changing greatly from the starting condition, the spectrum is changing as is the one group cross section. For this study, burnup is performed with one group cross sections; but, they are updated each cycle (4 times the fuel lifetime in the core). Since the reactivity change as a function of burnup decreases with increasing actinides, the BOL $k$ used should change. This is done for this study by iterating on the BOL $k$ in order to match the desired cycle length. Hand calculations were used for this iteration. Although there
is considerable uncertainty remaining, it is believed that the approach taken in this study is the best done to date.

III. Self Generated Actinide Burning

In order to fairly compare the actinide burning capabilities of differing reactor types one could argue that each reactor type should burn its self generated actinides. Mike Stone produced his masters thesis with this point of view. In his thesis he does a full literature review, method development, and presentation of the results. This report will not repeat this material but rather refer the reader to the thesis and discuss the work that has been performed following Stone’s work. Stone’s thesis was presented at the Charleston Topical Meeting on Advances in Reactor Physics in March 1992 and published in its proceedings (Appendix C).

If one is interested in reduction of self generated transuranic isotopes it was found that the MHTGR produced the best results. This is due to starting with a thorium cycle which implies more transmutations are required in order to arrive at a transuranic isotope. If one looks further into the topic, it is not clear actinide recycling of thorium cycle products is a good idea. The thorium cycle heavy metal waste products have a much lower radiotoxicity. Recycling the fuel in the thorium cycle just creates more problem isotopes.
IV. Method Improvements

In order to analyze the ability of reactor types to burn already generated transuranic actinides improvements were required to the code developed by Stone. These improvements actually reduced the size of the original code by a factor of three and reduced the number of input files required. The objective of the code reduction was to clarify the method so the modifications could be more easily implemented and understood. There still is considerable room for improvement but time was not allotted for this further effort. The new code is found in Appendix A. The key modification required was to allow refueling with any mixture of actinides. This was performed by identifying groups of isotopes as fuel or diluent. An index flag for each actinide indicates the appropriate group (0 = not used in refueling, 1 = fuel, and 2 = diluent). Another modification required was to allow double heterogeneity for the MHTGR.

Along with code modifications, it was desired to rearrange isotopes that receive updated cross sections with each cycle. The PC version of COMBINE only allows 20 isotopes in each run. Stone's previous work was reviewed and it was determined that it would be possible to include Cm-244, Cm-245 and Cm-246 in the COMBINE analysis. This
improvement greatly reduced the concerns over the cross sections that were not updated, since their concentrations are small even with high burnup.

In Stone's thesis a batch program that took twelve pages to present has been reduced to a system of three batch programs that can now fit on less than three pages. It is included in Appendix A.

Small improvements have been made in the input decks for the actinide studies. Sample input decks are found in Appendix B.

V. Method Check with PWR Analysis

For Stone's thesis he compared his results to that of a study by Gorrell.¹ This comparison gave a general confidence in the method but the level of agreement expected was fairly low since different methods were used. In order to look for finer details, a simpler problem was selected. It was decided to burn 3.2 w/o PWR fuel for 33,000 MWD/MTU, then let it decay 3 years, and compare the isotopics to two references. The first reference comes out of ANL. The paper was presented at the topical meeting in Marseilles in 1990.² In 1992 Downar³ is still using the same isotopics. The second paper is out of JAERI.⁴ This paper was presented at a topical meeting in Kyoto in 1991. It is a disappointment that for such a seemingly simple problem that
is so basic to actinide burning (and normal once through waste storage) such a large discrepancies exists. This study agrees best with the JAERI work. Table 1 shows the comparisons that can easily be made.

**TABLE 1**

Isotopics After 33,000 MWD/MTU Burning in a PWR

<table>
<thead>
<tr>
<th>Isotope</th>
<th>This Study</th>
<th>JAERI(^4)</th>
<th>ANL(^2)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Np-237</td>
<td>.569</td>
<td>.562</td>
<td>.491</td>
</tr>
<tr>
<td>Am-241</td>
<td>.263</td>
<td>.264</td>
<td>.227</td>
</tr>
<tr>
<td>Am-243</td>
<td>.128</td>
<td>.120</td>
<td>.225</td>
</tr>
<tr>
<td>Cm-243</td>
<td>.0004</td>
<td>.0003</td>
<td>.0007</td>
</tr>
<tr>
<td>Cm-244</td>
<td>.0361</td>
<td>.0511</td>
<td>.050</td>
</tr>
<tr>
<td>Cm-245</td>
<td>.0021</td>
<td>.0028</td>
<td>.0046</td>
</tr>
</tbody>
</table>

Fraction of Each Isotope Among the Pu Isotopes

<table>
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<tr>
<th>Isotope</th>
<th>This Study</th>
<th>JAERI(^4)</th>
<th>ANL(^2)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu-238</td>
<td>.0169</td>
<td>-</td>
<td>.0114</td>
</tr>
<tr>
<td>Pu-239</td>
<td>.5615</td>
<td>-</td>
<td>.571</td>
</tr>
<tr>
<td>Pu-240</td>
<td>.256</td>
<td>-</td>
<td>.224</td>
</tr>
<tr>
<td>Pu-241</td>
<td>.111</td>
<td>-</td>
<td>.151</td>
</tr>
<tr>
<td>Pu-242</td>
<td>.0546</td>
<td>-</td>
<td>.0436</td>
</tr>
</tbody>
</table>

Ratio of Minor Actinides to Pu Isotopes

<table>
<thead>
<tr>
<th></th>
<th></th>
<th>JAERI(^4)</th>
<th>ANL(^2)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>.0899</td>
<td>-</td>
<td>.124</td>
</tr>
</tbody>
</table>
Since fairly good agreement seems to exist between this work and the JAERI work it was believed that the method was sufficiently accurate to continue the study. It was shocking however that the two major laboratories used as a reference were so different on this test. A 14% disagreement in the main minor actinide, Np-237, and close to a factor of two for Am-243 were not expected. These isotopes are very important in any of the waste disposal work so it was believed that better agreement would exist.

VI. PWR Actinide Burning Study

For PWR actinide burning a mixture of all transuranic actinides as fuel and depleted uranium as a diluent was selected. The transuranic actinide actinide mixture came from PWR discharged fuel (33,000 MWD/MTU) after ten years in storage. The isotopic content of this fuel is given in Table 2. Notice that this fuel has about five percent Np-237 and due to the significant decay of Pu-241 it has about five percent Am-241. The Pu-241 content is down to seven percent. This mixture contains 88.7% Pu and 11.3% minor actinides.
### Table 2: Isotopic Fractions of Transuranics in PWR Discharged Fuel (33,000 MWD/MTU)

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Fraction</th>
</tr>
</thead>
<tbody>
<tr>
<td>Np-237</td>
<td>0.04737</td>
</tr>
<tr>
<td>Pu-238</td>
<td>0.01468</td>
</tr>
<tr>
<td>Pu-239</td>
<td>0.51511</td>
</tr>
<tr>
<td>Pu-240</td>
<td>0.23587</td>
</tr>
<tr>
<td>Pu-241</td>
<td>0.07097</td>
</tr>
<tr>
<td>Pu-242</td>
<td>0.05020</td>
</tr>
<tr>
<td>Am-241</td>
<td>0.05277</td>
</tr>
<tr>
<td>Am-243</td>
<td>0.01060</td>
</tr>
<tr>
<td>Cm-244</td>
<td>0.00226</td>
</tr>
<tr>
<td>Cm-245</td>
<td>0.00017</td>
</tr>
</tbody>
</table>

The fuel cycle scheme selected for the actinide burning assumed a four batch core with an average discharge burnup of 64,000 MWD/MTU. In an AP600 type core this would mean each cycle was about two years. The cross sections were updated each cycle. It was assumed that the effect of each batch on the keff was the same. At the end of any cycle there would be assemblies with 1, 2, 3 and 4 cycles of burnup. This means the average burnup in cycles is 2.5. The beginning of life (BOL) k needed was calculated as 2.5/4 times the delta k for 64,000 MWD/MTU. Since the change in k with burnup was dependent on the isotopics, the BOL k was
determined through iteration. Table 3 shows the beginning and end of life $k$ for each fuel loading.

Table 3: Beginning and End of Life $K_{\text{eff}}$ For Each Fuel Loading For PWR Actinide Burning

<table>
<thead>
<tr>
<th>Fuel Loading</th>
<th>$k_{\text{eff}}$ BOL</th>
<th>$k_{\text{eff}}$ EOL</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1.0719</td>
<td>0.9613</td>
</tr>
<tr>
<td>2</td>
<td>1.034</td>
<td>0.9809</td>
</tr>
<tr>
<td>3</td>
<td>1.029</td>
<td>0.9835</td>
</tr>
<tr>
<td>4</td>
<td>1.025</td>
<td>0.9857</td>
</tr>
<tr>
<td>5</td>
<td>1.022</td>
<td>0.9867</td>
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<tr>
<td>6</td>
<td>1.021</td>
<td>0.9885</td>
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<tr>
<td>7</td>
<td>1.019</td>
<td>0.9889</td>
</tr>
<tr>
<td>8</td>
<td>1.018</td>
<td>0.9894</td>
</tr>
<tr>
<td>9</td>
<td>1.017</td>
<td>0.9897</td>
</tr>
<tr>
<td>10</td>
<td>1.017</td>
<td>0.9905</td>
</tr>
<tr>
<td>11</td>
<td>1.017</td>
<td>0.9912</td>
</tr>
</tbody>
</table>

Notice that the delta $k$ with burnup is extremely low. This implies that the actinides work as excellent burnable absorbers. There could be significant advantages due to this effect. For example, the moderator temperature coefficient could be more negative since less soluble boron would be required. Power distribution control could be easier for these long cycles since the $k$ of all the fuel would be similar. The down side of this is that any
uncertainty in the cross sections could have a major effect on the results. Since there is a large amount of rarely used isotopes in these cores this uncertainty concern is serious.

Table 4 shows the actinide contents at the end of each fuel lifetime. As can be seen from this table the reactor is actually an actinide storage device. Table 5 shows the quantities of the feed actinides in each cycle. This feed material is used to replace the fission products. All other isotopes are mixed with this feed to make the fuel for the next cycle. Since PWRs produce about 1 gm actinides/MWD or about 25,000 kg/yr with our current operating PWRs, it is clear that it would require a new actinide burner for each year of operation of the current reactors. You would also need on the order of ten of these reactors to reduce the current backlog. Table 6 normalizes this data by the energy produced. From this table it can be observed that for about every two new reactors ordered an actinide burner would be required. The actinides represent an attractive fuel and since the minor actinides act as burnable absorbers the advantage of removing the minor actinides from the Pu is not significant.
Table 4: Inventory (Kg) of Actinides in a 600 MWe PWR Actinide Burner

<table>
<thead>
<tr>
<th>End Load</th>
<th>U234</th>
<th>U235</th>
<th>U236</th>
<th>U238</th>
<th>Np237</th>
<th>Pu238</th>
<th>Pu239</th>
<th>Pu240</th>
<th>Pu241</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>24.0</td>
<td>46.9</td>
<td>17.1</td>
<td>5.69E+4</td>
<td>108.</td>
<td>411.</td>
<td>1660.</td>
<td>1500.</td>
<td>496.</td>
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<tr>
<td>2</td>
<td>56.9</td>
<td>26.6</td>
<td>19.4</td>
<td>5.42E+4</td>
<td>131.</td>
<td>715.</td>
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<th>End Load</th>
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<th>Am241</th>
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<th>Am243</th>
<th>Cm242</th>
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<td>2.99</td>
<td>530.</td>
<td>141.</td>
<td>87.3</td>
</tr>
<tr>
<td>10</td>
<td>2180.</td>
<td>694.</td>
<td>9.03</td>
<td>480.</td>
<td>.0773</td>
<td>3.03</td>
<td>548.</td>
<td>146.</td>
<td>95.5</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>End Load</th>
<th>Cm247</th>
<th>Cm248</th>
<th>Bk249</th>
<th>Cf249</th>
<th>Cf250</th>
<th>Cf251</th>
<th>Cf252</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>.122</td>
<td>.019</td>
<td>1.77E-5</td>
<td>8.06E-4</td>
<td>4.47E-5</td>
<td>3.33E-5</td>
<td>5.49E-6</td>
</tr>
<tr>
<td>2</td>
<td>.608</td>
<td>.185</td>
<td>2.13E-4</td>
<td>1.17E-2</td>
<td>1.26E-3</td>
<td>1.04E-3</td>
<td>2.47E-4</td>
</tr>
<tr>
<td>3</td>
<td>1.40</td>
<td>.616</td>
<td>7.58E-4</td>
<td>4.56E-2</td>
<td>6.25E-3</td>
<td>5.34E-3</td>
<td>1.52E-3</td>
</tr>
<tr>
<td>4</td>
<td>2.35</td>
<td>1.33</td>
<td>1.69E-3</td>
<td>1.07E-1</td>
<td>1.69E-2</td>
<td>1.46E-2</td>
<td>4.53E-3</td>
</tr>
<tr>
<td>5</td>
<td>3.33</td>
<td>2.26</td>
<td>2.94E-3</td>
<td>1.93E-1</td>
<td>3.32E-2</td>
<td>2.90E-2</td>
<td>9.40E-3</td>
</tr>
<tr>
<td>6</td>
<td>4.29</td>
<td>3.34</td>
<td>4.40E-3</td>
<td>2.96E-1</td>
<td>5.44E-2</td>
<td>4.76E-2</td>
<td>1.58E-2</td>
</tr>
<tr>
<td>7</td>
<td>5.18</td>
<td>4.50</td>
<td>5.98E-3</td>
<td>4.10E-1</td>
<td>7.86E-2</td>
<td>6.90E-2</td>
<td>2.32E-2</td>
</tr>
<tr>
<td>8</td>
<td>6.01</td>
<td>5.68</td>
<td>7.59E-3</td>
<td>5.28E-1</td>
<td>1.05E-1</td>
<td>4.20E-1</td>
<td>3.12E-2</td>
</tr>
<tr>
<td>9</td>
<td>6.77</td>
<td>6.84</td>
<td>9.19E-3</td>
<td>6.45E-1</td>
<td>1.32E-1</td>
<td>1.16E-1</td>
<td>3.94E-2</td>
</tr>
<tr>
<td>10</td>
<td>7.45</td>
<td>7.96</td>
<td>1.07E-2</td>
<td>7.60E-1</td>
<td>1.59E-1</td>
<td>1.39E-1</td>
<td>4.75E-2</td>
</tr>
</tbody>
</table>
### Table 5: Kilograms Inserted Each Loading in a 600 MWe PWR Actinide Burner

<table>
<thead>
<tr>
<th>Loading</th>
<th>Depleted Uranium</th>
<th>Plutonium Mixture</th>
<th>Minor Actinides</th>
<th>Total Non U Actinides</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>59949</td>
<td>5907</td>
<td>754</td>
<td>6661</td>
</tr>
<tr>
<td>2</td>
<td>24</td>
<td>4013</td>
<td>512</td>
<td>4525</td>
</tr>
<tr>
<td>3</td>
<td>1088</td>
<td>3081</td>
<td>393</td>
<td>3475</td>
</tr>
<tr>
<td>4</td>
<td>1355</td>
<td>2863</td>
<td>365</td>
<td>3228</td>
</tr>
<tr>
<td>5</td>
<td>1563</td>
<td>2694</td>
<td>344</td>
<td>3037</td>
</tr>
<tr>
<td>6</td>
<td>1621</td>
<td>2655</td>
<td>339</td>
<td>2994</td>
</tr>
<tr>
<td>7</td>
<td>1770</td>
<td>2532</td>
<td>323</td>
<td>2855</td>
</tr>
<tr>
<td>8</td>
<td>1817</td>
<td>2499</td>
<td>319</td>
<td>2818</td>
</tr>
<tr>
<td>9</td>
<td>1876</td>
<td>2454</td>
<td>313</td>
<td>2767</td>
</tr>
<tr>
<td>10</td>
<td>1870</td>
<td>2465</td>
<td>315</td>
<td>2779</td>
</tr>
<tr>
<td>11</td>
<td>1887</td>
<td>2454</td>
<td>313</td>
<td>2768</td>
</tr>
</tbody>
</table>

### Table 6: Grams Inserted Per MWD Electricity Produced in a PWR Actinide Burner

<table>
<thead>
<tr>
<th>Loading</th>
<th>Depleted Uranium</th>
<th>Plutonium Mixture</th>
<th>Minor Actinides</th>
<th>Total Non U Actinides</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>43.95</td>
<td>4.33</td>
<td>0.55</td>
<td>4.88</td>
</tr>
<tr>
<td>2</td>
<td>0.02</td>
<td>2.94</td>
<td>0.38</td>
<td>3.32</td>
</tr>
<tr>
<td>3</td>
<td>0.80</td>
<td>2.26</td>
<td>0.29</td>
<td>2.55</td>
</tr>
<tr>
<td>4</td>
<td>0.99</td>
<td>2.10</td>
<td>0.27</td>
<td>2.37</td>
</tr>
<tr>
<td>5</td>
<td>1.15</td>
<td>1.97</td>
<td>0.25</td>
<td>2.23</td>
</tr>
<tr>
<td>6</td>
<td>1.19</td>
<td>1.95</td>
<td>0.25</td>
<td>2.19</td>
</tr>
<tr>
<td>7</td>
<td>1.30</td>
<td>1.86</td>
<td>0.24</td>
<td>2.09</td>
</tr>
<tr>
<td>8</td>
<td>1.33</td>
<td>1.83</td>
<td>0.23</td>
<td>2.07</td>
</tr>
<tr>
<td>9</td>
<td>1.38</td>
<td>1.80</td>
<td>0.23</td>
<td>2.03</td>
</tr>
<tr>
<td>10</td>
<td>1.37</td>
<td>1.81</td>
<td>0.23</td>
<td>2.04</td>
</tr>
<tr>
<td>11</td>
<td>1.38</td>
<td>1.80</td>
<td>0.23</td>
<td>2.03</td>
</tr>
</tbody>
</table>
VII. MHTGR Heterogeneity Analysis

After performing the PWR actinide analysis, it was expected that the MHTGR analysis would be a straightforward extension of the work done by Stone with the updated code capabilities. Unfortunately, it was found that the worth of the actinide mixture was very low to negative. This forced a re-evaluation of the methods previously used by Stone. Stone had not used the double heterogeneity option in COMBINE. The implementation of the double heterogeneity option did not seem clear since it appeared desirable to have the average number density of the fuel in the grain as well as in the pin but only one input was allowed. This forced a review of the literature which COMBINE referenced to see how this could be done. As it turns out, in the reference’s by Stamatelatos, he comes up with a correlation for the heterogeneous background cross section, e, that uses the pin average number density. Unfortunately, one still needs the background cross sections for the non-absorber isotopes, 0, which should be based on the grain atom densities. At first, it may appear that this makes it impossible to force COMBINE to give correct results. INEL personnel have been putting in pin average atom densities into COMBINE. It is possible to yield the correct results by putting in the pin averaged absorber atom density and
then using non-physical atom densities for the non-absorbers so that the ratio of non-absorber to absorber is the same as in the grain.

The effect of the double heterogeneity is fairly large. In order to quantify, this a standard U/Th fueled MHTGR was analyzed. The core is fully described by Stone in his thesis as the starting core for his actinide burning study. For this case, $k_{\text{eff}}$ using a single heterogeneity model was 1.212. Using the INEL input method $k_{\text{eff}}$ was 1.224. Using the corrected double heterogeneity method $k_{\text{eff}}$ was 1.276. Double heterogeneity was worth over 6% delta $k$. It is important to point out that in order for double heterogeneity to have an important effect there must be a significant amount of a resonance absorber. In a production reactor that is fully enriched or in a reactor designed to burn weapons grade Pu the effect of double heterogeneity would be much less or negligible.

VIII. MHTGR Actinide Burning Results

With the double heterogeneity now correct it was attempted to design an actinide burning MHTGR. With 90% actinide mix from Table 2 and 10% depleted uranium, $k_{\text{eff}}$ was .9592 ($k_{\text{inf}} = 1.0068$). This result suggested that the minor actinide to Pu ratio that exists in discharged PWR fuel is
too high to be used in an MHTGR. The next step was to fuel the MHTGR with PWR Pu and PWR minor actinides (MA) in a ratio that would be acceptable to the MHTGR. It was found that with 91% Pu and 9% MA mix the $k_{\text{eff}}$ was 1.0449. This $k$ was high enough that a long burnup was possible (about 300,000 MWD/MTU). This seemed very attractive but unfortunately when the fission products were replaced by a pure Pu mix it was not possible to be critical ($k_{\text{eff}} = .9907$). This would imply that some of the minor actinides would have to be removed to start the next cycle. Hence instead of an actinide burner the reactor would be simply a Pu burner that would produce some actinide waste with higher atomic weight than the original actinide waste. This eliminates the MHTGR from the front runners in actinide burning. In order to test this result further, Pu after three years decay (ANL concentrations) was used. $k$ was still too low. Using a fuel burnup of 97,435 MWD/MTU and fresh Pu (ANL concentrations) allowed two fuel lifetimes; but on the third refueling, it was not possible to become critical.

Some of the actinides have very large absorption resonances at low energies. Np-237, Am-241, and Pu-240 have very large negative worths in a thermal reactor. In a PWR there is sufficient heterogeneity to produce flux depressions to lower the impact of these isotopes. In an MHTGR there is insufficient heterogeneity. Larger grain
sizes (600 microns) were investigated at one enrichment and although it helped the effect on k was 0.1% which is not nearly enough. Of some concern was whether the Pu worth was too low or the MA worth too high. It was found that using the cross sections for this study a Pu/U core would require a fissile Pu enrichment of about 30%. In a PWR this value is about 3%. Given this result it is clear that the Pu worth is low as well as the minor actinide worth is very high.

Selected sample input files are shown in Appendix B for independent review.

IX. Conclusions

This project has looked into actinide burning with an MHTGR. The method of the review was to perform a comparative analysis to a PWR and LMR. The first step was to evaluate the reactors on their ability to burn their own actinides. This study was a Masters thesis for Mike Stone. He concluded that in this context the MHTGR faired best of the three but mainly due to its thorium cycle start.

The next step was to confirm in more detail methods for actinide burning by an analysis of a standard PWR burnup. Surprisingly, it was found that there is poor
international agreement on the content of trans-plutonium actinides even for a standard 33,000 MWD/MTU burn in a PWR. This study agreed with those reported by JAERI.

Actinide burning in a PWR was found not only to be possible but would have significant benefits. The minor actinides were found to act as excellent burnable absorbers for the PWR core. The actinide burner would start with 90% depleted uranium and 10% transuranic actinides. This enrichment would allow a 64,000 MWD/MTU burnup. All of the following cycles could be done by removing only the fission products and adding the transuranic mixture and some depleted uranium. This concept never requires separation of the plutonium from the other transuranic actinides. In equilibrium, a 600 MWe PWR actinide burner would burn about 400 kilograms of transuranic actinides a year. In equilibrium about one third of the reactors would have to be actinide burners.

The MHTGR, unfortunately, did not fare as well for actinide burning. Due to the large low energy resonances of Np-237, Pu-240, and Am-241, self shielding is very important. The MHTGR does not have nearly as much self shielding as a PWR and this makes the positive worth of plutonium less and the negative worth of the minor actinides greater. Double heterogeneity is important for these cores and it was found that there is not clear agreement on how
this should be performed (using non GA codes). It was
determined that for the MHTGR the plutonium had to be
separated from the minor actinides in order to reach
criticality in the MHTGR. Even when this is done it was
found that it was not possible to achieve recriticality
using the Pu mix if no actinides were to be removed from the
actinide burner. These findings do not mean that actinide
burning is impossible in a MHTGR but rather that the MHTGR
has a significant disadvantage when compared to a PWR for
this purpose.

X. References

1. T. C. Gorrell, "Transmutation of Waste Actinides in

2. R. N. Hill, D. C. Wade, E. K. Fujita, and H. S. Khalil,
"Physics Studies of Higher Actinide Consumption in an LMR,"
Proceedings of the International Conference on the Physics
of Reactors, Operation, Design, and Computation, Marseilles,

3. H. B. Choi and T. J. Downar, "The Neutronics Design and
Analysis of a LMR for Burning Minor Actinides," Proceedings
of the 1992 Topical Meeting on Advances in Reactor Physics,


APPENDIX A: Improved Code System

Al: The new code.

The following is a code listing of the improved burnup code first reported in Stone's thesis:
C MAIN PROGRAM: USE INPUT COMBINE CROSS-SECTION DATA TO OBTAIN ONE
C GROUP CROSS-SECTIONS AND ONE GROUP FLUX; DETERMINE ISOTOPIC DEPLETION
C OR BUILDUP FOR A GIVEN TIME STEP; CONTINUE BACK AND FORTH BETWEEN
C THIS PROGRAM AND COMBINE UNTIL THE DESIRED BURNUP IS ACHIEVED;
C REENRICH AND RUN FOR ADDITIONAL CYCLES
C

IMPLICIT REAL*8(A-H,O-Z)
COMMON B2, N, MN, SIGTR1(40), SIGTR2(40), SIGCA1(40), SIGCA2(40)
COMMON SIGFS1(40), SIGFS2(40), XNU1(40), XNU2(40), SIG12(40)
COMMON DIFF1, DIFF2, SGTAB1, SGTAB2, SGTFS1, SGTFS2, SGTNFS1
COMMON SGTNFS2, SIGT12, TFR, XMASS(40)

OPEN(5,FILE='crx')
C READ VARIABLE TO DETERMINE WHICH REACTOR TYPE IS TO BE RUN (1:PWR,
C 2:HTGR, 3:LMR) and whether or not an enrichment step (0/1 yes/no)
C
READ(5,*) NRX, icho
C
CALL SUBROUTINE TO READ IN CROSS-SECTIONS
CALL COMISO(nrx)
C
CALL SUBROUTINE TO READ IN NUMBER DENSITIES AND CALCULATE TOTAL
C MACROSCOPIC CROSS-SECTIONS FOR ALL ISOTOPES AS WELL AS DETERMINE
C THERMAL TO FAST RATIO and collapse cross sections
C
CALL NDMAC
C
CALL SUBROUTINE TO DO DEPLETION OVER GIVEN TIMESTEP
CALL MARCH
C
CALL SUBROUTINE TO CONVERT ATOMS/BARN-CM TO G/MTM AS WELL AS TO
C CALCULATE K EFFECTIVE AT THE END OF A TIMESTEP (DATA WRITTEN TO
C KCON.OUT)
C
ien=0 for non-enrich call  ien=1 if called from enriching routines
ien=0
open(2, file='kcon.out')
call KC(ien)
close(2)
C
IF/THEN CLAUSE TO CHOOSE BETWEEN NORMAL RUN FOR A TIMESTEP DURING
C A CYCLE (1) OR RE-ENRICHMENT RUN AT END OF CYCLE (0)
C
IF(icho .EQ. 1) goto 38
C
C RE-ENRICHMENT SEQUENCE OF SUBROUTINES
C
C WRITE THE FINAL NUMBER DENSITIES OF A CYCLE TO A
C STORAGE FILE 'CYCLE' WHICH WILL BE RENAMED IN THE BATCH FILE TO A
C UNIQUE FILE NAME FOR THE PARTICULAR CYCLE
C
OPEN(1,FILE='pcycle')
DO 345 I=2, MN
345 WRITE(1,235) XMASS(I), I
235 FORMAT(1X,1E13.7,2X,I2)
WRITE(1,*)
C CALL SUBROUTINE TO DELETE FISSION PRODUCT ATOMS, decay for
C reprocessing time, and replace, with fissile mix.
C Then adds fertile mix in order to match the desired keff
C
CALL newnd
C
38 Continue
C CALL SUBROUTINE TO WRITE A NEW COMBINE INPUT FILE USING NEW NUMBER
C DENSITIES
C
CALL fwrite(nrx)
stop
END
C
**************************************************************************
C SUBROUTINE COMISO(nrx)
C
C SUBROUTINE TO READ IN CROSS-SECTION DATA
C
IMPLICIT REAL*8(A-H,O-Z)
COMMON B2, N, MN, SIGTR1(40), SIGTR2(40), SIGCA1(40), SIGCA2(40)
COMMON SIGFS1(40), SIGFS2(40), XNU1(40), XNU2(40), SIG12(40)
COMMON DIFF1, DIFF2, SGTAB1, SGTAB2, SGTF1, SGTF2, SGTF3
COMMON SGTF3, SGT12, TFR, XMASS(40)
COMMON PHIB, FXS(40), AXS(40), VOL, POWER
CHARACTER*1 B
C
C READ IN SYSTEM MACROSCOPIC CROSS-SECTIONS AND ISOTOPIC MICROSCOPIC
C CROSS-SECTIONS FROM COMBINE ISOTXS FILE (2 GRP DATA)
C
C MAKE FORMAT STATEMENTS FOR READING ISOTXS FILE
C
798 FORMAT(1X,I1,A1,1X,8I6)
800 FORMAT(1X,I1,A1,2X,A6,1X,A6,1X,A6)
801 FORMAT(6E12.5)
802 FORMAT(12I6)
803 FORMAT(1X,I1,A1,1X,5E12.5)
C
C SET UP INPUT FILE FOR DATA (ALSO OUTPUT FOR ECHO)
C
OPEN(5, FILE='acrx.iso')
OPEN(6, FILE='echo.out')
C
C READ N FOR NUMBER OF MATERIALS IN FILE!!!!!!!!!!!!!!!!!!!!!!!DON'T FORGET!!
READ(5,798)
READ(5,798) J, B, J, N
WRITE(6,*) 'N:'
WRITE(6,802) N
WRITE(6,*)

C C READ VARIABLE NUMBER OF LINES BEFORE MACRO ENTRY
C
DO 339 I=1, 5
339 READ(5,798)
IF(N .gt. 9) read(5,798)
IF(N .gt. 12) read(5,798)
IF(N .gt. 19) read(5,798)

C C READ IN CROSS-SECTION DATA
C
DO 376 MATN=1, N
READ(5,800) J
READ(5,801) X
READ(5,802) J
READ(5,802) J
READ(5,803) J, B, SIGTR1(MATN), SIGTR2(MATN), X, X, SIGCA1(MATN)
READ(5,801) SIGCA2(MATN), SIGFS1(MATN), SIGFS2(MATN),
6 XNU1(MATN), XNU2(MATN)
if (nrx.ne.3) then
  READ(5,803) J, B, X, X, X, SIG12(MATN)
else
  READ(5,803) J, B, X, X, SIG12(MATN)
endif
READ(5,801) X

WRITE(6,*) 'MATN, SIGTR1, SIGTR2, SIGCA1, SIGCA2:'
WRITE(6,901) MATN, SIGTR1(MATN), SIGTR2(MATN), SIGCA1(MATN),
6 SIGCA2(MATN)
WRITE(6,*)
WRITE(6,*) 'MATN, SIGFS1, SIGFS2, XNU1, XNU2, SIG12:'
WRITE(6,901) MATN, SIGFS1(MATN), SIGFS2(MATN), XNU1(MATN),
6 XNU2(MATN), SIG12(MATN)
WRITE(6,*)
901 FORMAT(1X,I2,5E12.5)
C
376 CONTINUE
CLOSE(5)

C C READ IN NON-COMBINE ISOTOPES MICROSCOPIC CROSS-
C SECTIONS (2 GRP FROM SOMEWHERE???????????) AND OTHER DATA
C
OPEN(5,FILE='noncom.iso')

C C READ IN DATA AND ECHO TO OUTPUT FILE [# MATLS, BUCKLING, POWER, VOL, C MODL-3 FLAGS (INTEGERS REPRESENT MODERATOR ISOTOPES WHOSE NUMBER C DENSITIES ARE NOT TO BE REWRITTEN), MCLAD1-3 FLAGS (CLAD ISOTOPES C WITH CONSTANT NUMBER DENSITIES), MODF1-2 (FUEL MODERATOR, E.G. OXYGEN, C WITH A CONSTANT NUMBER DENSITY) AND THEN MICROS]
C
READ(5,*)
READ(5,*) M, B2, POWER, VOL

C
WRITE(6,*) '# MATLS, BUCKLING, POWER, VOL:'
WRITE(6,908) M, B2, POWER, VOL
908 FORMAT(1X,I2,3E12.5,I2,1X,I2,1X,I2)
WRITE(6,*)

C
MN = M + N
NP1 = N + 1

C
DO 691 I=NP1, MN
   READ(5,*) SIGTR1(I), SIGTR2(I), SIGCA1(I), SIGCA2(I)
   READ(5,*) SIGFS1(I), SIGFS2(I), XNU1(I), XNU2(I), SIG12(I)
   WRITE(6,*) 'I, SIGTR1, SIGTR2, SIGCA1, SIGCA2:'
   WRITE(6,904) I, SIGTR1(I), SIGTR2(I), SIGCA1(I), SIGCA2(I)
   WRITE(6,*)
   WRITE(6,*) 'I, SIGFS1, SIGFS2, XNU1, XNU2, SIG12:'
   WRITE(6,904) I, SIGFS1(I), SIGFS2(I), XNU1(I), XNU2(I), SIG12(I)
   WRITE(6,*)
691 FORMAT(1X,I2,5E12.5)
CLOSE(5)
RETURN
END

C
****************************************************************
C
SUBROUTINE NDMAC
C
READ IN NUMBER DENSITIES AND CALCULATE TOTAL MACROSCOPIC CROSS-
SECTIONS FOR ALL ISOTOPES; DETERMINE THERMAL TO FAST RATIO
C
IMPLICIT REAL*8(A-H,O-Z)
COMMON B2, N, MN, SIGTR1(40), SIGTR2(40), SIGCA1(40), SIGCA2(40)
COMMON SIGFS1(40), SIGFS2(40), XNU1(40), XNU2(40), SIG12(40)
COMMON DIFF1, DIFF2, SGTAB1, SGTAB2, SGTSF1, SGTSF2, SGTNFS1
COMMON SGTNFS2, SIGT12, TFR, XMASS(40)
COMMON PHIB, FXS(40), AXS(40), VOL, POWER
COMMON XMASSP(40), ALAM(40), LD(40,2), LCN(40,2), CFRAC(40,2)
COMMON YIELD, DELT, LFP, DFRAC(40,2), XMOLWT(40), TDM
DIMENSION CXS(40)

C
OPEN(5,FILE='numden.in')

C
READ(5,*)
READ(5,*) TDM
WRITE(6,*) 'TOTAL DENSITY/MASS (G/CM3):'
WRITE(6,786) TDM
786 FORMAT(1X,1E13.7)
READ(5,*)
DO 323 I=2, MN
   READ(5,*) XMASS(I), L, xmolwt(i)
WRITE(6,*) 'I, XMASS, XMOLWT:'
WRITE(6,987) I, XMASS(I), XMOLWT(I)
323 WRITE(6,*)

C call kcal(xkeff)

C DETERMINE THE THERMAL TO FAST RATIO

WRITE(6,*) 'keff before burnup'
WRITE(6,906) xkeff
TFR = (SIGT12)/(DIFF2*B2 + SGTAB2)
WRITE(6,*) 'THERMAL TO FAST RATIO:'
WRITE(6,906) TFR
906 FORMAT(1X,1E12.5)
WRITE(6,*)

C COLLAPSE GROUPS

DO 501 I=2, MN
CXS(I) = (SIGCA1(I) + SIGCA2(I)*TFR)/(1.0 + TFR)
FXS(I) = (SIGFS1(I) + SIGFS2(I)*TFR)/(1.0 + TFR)
AXS(I) = CXS(I) + FXS(I)
WRITE(6,*) 'I, CXS, FXS, AXS:'
WRITE(6,987) I, CXS(I), FXS(I), AXS(I)
987 FORMAT(1X,I2,3E12.5)
501 WRITE(6,*)

C SOLVE FOR ONE GROUP FLUX (IN UNITS OF PER BARN - SEC)

C CALCULATE SYSTEM ONE GROUP MACROSCOPIC FISSION CROSS-SECTION

FXSMS = 0.0

DO 697 I=2, MN
FXSMS = FXSMS + XMASS(I)*FXS(I)
WRITE(6,*) 'MACRO SYSTEM FXS:'
WRITE(6,906) FXSMS
WRITE(6,*)
697

PHIB = ((POWER*3.121*(10.0**16))/(FXSMS*VOL))*10.0**(-24)
WRITE(6,*) 'PHIB:'
WRITE(6,906) PHIB
write(6,*)
RETURN
END

C ****************************************************************
C
SUBROUTINE MARCH

C PERFORM DEPLETION OVER GIVEN TIMESTEP (THIS SUBROUTINE COMPUTES THE
C TIME DEPENDENT ISOTOPIC CONCENTRATION)
IMPLICIT REAL*8(A-H,O-Z)
COMMON B2, N, MN, SIGTR1(40), SIGTR2(40), SIGCA1(40), SIGCA2(40)
COMMON SIGFS1(40), SIGFS2(40), XNU1(40), XNU2(40), SIG12(40)
COMMON DIFF1, DIFF2, SGTAB1, SGTAB2, SGTS1, SGTS2, SGTFSl
COMMON SGTFS2, SIGT12, TFR, XMASS(40)
COMMON PHIB, FKS(40), AXS(40), VOL, POWER
COMMON XMASSP(40), ALAM(40), LD(40,2), LCN(40,2), CFRAC(40,2)
COMMON YIELD, DELT, LFP, dfrac(40,2), XMOLWT(40), TDM

C READ IN DECAY CONSTANTS, AND DECAY, CAPTURE, AND FISSION MATRICES;
C and TIMESTEP

OPEN(5, FILE='mdata.in')
C READ IN TIMESTEP, DELT (IN DAYS), FROM MDATA.IN;
C ALSO INPUT MAT ID FOR THE FP; Finally read in refueling time in days

READ(5,*) DELT, LFP, rftime
WRITE(6,*) 'DELT, LFP, RFTIME:'
WRITE(6,930) DELT, LFP, RFTIME
930 FORMAT(1X, E12.5, 1X, I2, 1E15.6)
WRITE(6,*)

C READ IN DECAY CONSTANTS; DECAY AND CAPTURE LOGIC; AND DECAY AND
C CAPTURE FRACTIONS FOR SOURCE ISOTOPES

DO 266 I=2, MN
READ(5,*)
READ(5,*) ALAM(I), LD(I,1), LD(I,2), LCN(I,1), LCN(I,2),
      DFRAC(I,1), DFRAC(I,2), CFRAC(I,1), CFRAC(I,2)
WRITE(6,*) 'I, ALAM, LD1, LD2, LCN1, LCN2, DF1, DF2, CF1, CF2:'
WRITE(6,931) I, ALAM(I), LD(I,1), LD(I,2), LCN(I,1), LCN(I,2),
      DFRAC(I,1), DFRAC(I,2), CFRAC(I,1), CFRAC(I,2)
266 WRITE(6,*)
931 FORMAT(1X, I2, 1E12.5, 1X, I2, 1E15.6)

C READ IN YIELD VALUE FOR CALCULATION OF LFP (THE LUMPED FP)(XN AND
C SM WILL BE KEPT AT EQUILIBRIUM LEVEL THROUGHOUT THE CYCLE), YIELD
C IS AMOUNT OF FP WHICH IS NON-XE135/SM149

READ(5,*)
READ(5,*) YIELD
WRITE(6,*) 'YIELD:'
WRITE(6,871) YIELD
871 FORMAT(1X, 1E13.7)
WRITE(6,*)

C CONVERT DELT (IN DAYS) INTO ONE-TENTH OF DELT (OR TEMP), IN SECONDS;
C ALSO INITIALIZE TEMP1

TEMP = DELT*24.0*3600.0/10.0
TEMP1 = 0.0
LOOP THROUGH ALL ISOTOPES (MN) TO OBTAIN THE TOTAL FISSION RATE FOR THE INITIAL MASS OF ISOTOPES; FXS: FISSION XS (BARNs), PHIB: FLUX (PER BARN-SEC), MASS: ISOTOPE NUMBER DENSITY (ATOMS PER BARN - CENTIMETER), VOL: VOLUME (CUBIC CENTIMETERS), AND A CONVERSION CONSTANT TO OBTAIN THE RATE

DO 5 KN = 2, MN
   5 TEMP1 = TEMP1 + FXS(KN)*PHIB*XMASS(KN)*VOL*(10.0**24)

TIME LOOP TO DO TEN TIME STEPS OF SIZE TEMP; INITIALIZE TEMP3 CHANGE FOR HAND CALC>>>>>>>>>>>

DO 200 KT = 1, 10
   TEMP3 = 0.0

RENAME INITIAL NUMBER DENSITIES TO VARIABLE MASSP, SO AS TO USE VARIABLE MASS FOR COMPUTATION OF NEW NUMBER DENSITIES

DO 20 KN = 2, MN
   20 XMASSP(KN) = XMASS(KN)

ITERATION LOOP FOR IMPLICIT METHOD USED TO DETERMINE NEW ISOTOPIC COMPOSITION -- CHANGED FROM 5 TO 1 FOR HAND CALC>>>>>>>>>

DO 50 KKK = 1, 5

CALCULATE ISOTOPIC CHANGES

DO 50 KN = 2, MN

CHECK FOR LIGHT ISOTOPES; THESE ARE TO BYPASS THE DEPLETION LOOP

IF (KN .EQ. LFP) goto 313
IF (xmolwt(KN) .LE. 230.0) goto 50

CALCULATE THE LOSSES DUE TO DECAY AND ABSORPTION

   TEMP2=-(XMASS(KN)+XMASSP(KN))*(ALAM(KN)+AXS(KN)*PHIB)

CALCULATE SOURCE FROM DECAY (TWO DECAY SOURCES ALLOWED), DFRAC IS USED WHEN THERE ARE TWO TYPES OF DECAY FROM A SOURCE ISOTOPE

DO 30 K=1, 2
   KK = LD(KN,K)
   DD = DFRAC(KN,K)
   IF (KK) 30, 30, 28
   28 TEMP2 = TEMP2 + (DD*ALAM(KK))*(XMASS(KK) + XMASSP(KK))
   30 CONTINUE

CALCULATE SOURCE FROM CAPTURE (TWO CAPTURE SOURCES ALLOWED), CFRAC IS USED WHEN THERE ARE TWO DIFFERENT CAPTURES FROM A SOURCE ISOTOPE (I.E. AM241 TO AM242M AND AM242)
DO 32 K = 1, 2
KK = LCN(KN,K)
CC = CFRAC(KN,K)
IF (KK) 32, 32, 31
31 TEMP2 =TEMP2 + (AXS(KK)-FXS(KK))*CC*PHIB*(XMASS(KK)+XMASSP(KK))
32 CONTINUE

GOTO 123

C CALCULATE SOURCE FOR LFP FROM FISSION OF ACTINIDES
C 313 TEMP2 = 0.0
C DO 36 KK = 2, MN
36 TEMP2 = TEMP2 + YIELD*FXS(KK)*PHIB*(XMASS(KK)+XMASSP(KK))
C CALCULATE NEW NUMBER DENSITY FOR EACH ISOTOPE
C 123 XMASS(KN) = XMASSP(KN) + 0.5*TEMP*TEMP2
C 50 CONTINUE
C CALCULATE NEW FISSION RATE FOR ALL ISOTOPES
C DO 100 KN = 2, MN
100 TEMP3 = TEMP3 + FXS(KN)*PHIB*XMASS(KN)*VOL*(10.0**24)
C ADJUST FLUX FOR NEXT ITERATION TO KEEP CONSTANT POWER LEVEL
C IF (TEMP3) 200, 200, 110
110 PHIB = PHIB*TEMP1/TEMP3
200 CONTINUE
C WRITE(6,*), 'TEMP3, FISSION ADJUSTED PHIB:'
WRITE(6,306) TEMP3, PHIB
306 FORMAT(1X,2E12.5)
WRITE(6,*)
C LOOP TO DECAY ACTINIDES AT OTHER THAN EOC POWER OFF PERIOD
C RFTIME IS THE LENGTH OF THE REFUELING TIME
C IF(rftime.lt.1.0) goto 177
delt=rftime
call marchd
177 continue
C WRITE THE NUMBER OF ISOTOPES AND THEN THEIR NUMBER DENSITIES TO A
C OUTPUT FILE
OPEN(2, FILE='numden.out')
WRITE(2,*) 'TOTAL DENSITY/MASS (G/CM3):'
WRITE(2,786) TDM
786 FORMAT(1X, 1E13.7)
WRITE(2,961) MN
961 FORMAT(I2)
DO 59 KN = 2, MN
59 WRITE(2,956) XMASS(KN), KN, XMOLWT(KN)
956 FORMAT(I2, 1E13.7)
WRITE(2,*)
RETURN
END

C ****************************************************************
C SUBROUTINE FWRITE(nx)
C SUBROUTINE TO WRITE A NEW COMBINE INPUT FILE USING NEW NUMBER
C DENSITIES FROM THE DEPLETION SUBROUTINE
C
IMPLICIT REAL*8(A-H,O-Z)
COMMON B2, N, MN, SIGTR1(40), SIGTR2(40), SIGCA1(40), SIGCA2(40)
COMMON SIGFS1(40), SIGFS2(40), XNU1(40), XNU2(40), SIG12(40)
COMMON DIFF1, DIFF2, SGTAB1, SGTAB2, SGTFS1, SGTFS2, SGTNF1
COMMON SGTNF2, SIGT12, TFR, XMASS(40)
DIMENSION INVALID(40), THERM(40), HMASS(40), DANF(40)
DIMENSION ABH(20), NABH(20)
CHARACTER*72 AA

C SET UP INPUT, OUTPUT FILES
C OPEN(5, FILE='fwdata.in')
C READ IN DATA FOR INPUT FILE WRITING LOOP(S):
C RTEMP IS REACTOR TEMP; FVF is THE FUEL MODERATOR VOLUME FRACTION;
C FRAD IS THE FUEL REGION OUTER RADI (CM);
C MOS1-3 ARE THE ID NUMBERS NEEDED FOR THE PRIMARY MODERATING MATERIALS
C IN THE FUEL; NMF IF ZERO INDICATES NO ABH, IF NON ZERO IT IS
C THE NUMBER OF ISOTOPES IN THE FUEL
C
READ(5,*)
READ(5,*) RTEMP, FVF, FRAD
READ(5,*) MOS1, MOS2, MOS3, NMF
WRITE(6,*) 'RTEMP, FVF, FRAD:'
WRITE(6,940) RTEMP, FVF, FRAD
940 FORMAT(1X, 1F8.1, 2F12.6)
WRITE(6,*)
WRITE(6,*) 'MOS1, MOS2, MOS3, NMF'
WRITE(6,937) MOS1, MOS2, MOS3, NMF
937 FORMAT(4I5)
WRITE(6,*)

C
C FAST(I) AND THERM(I) ARE THE COMBINE FAST AND THERMAL ISOTOPE ID
C NUMBERS; DANF(I) ARE THE ISOTOPE DANCOFF FACTORS; ABH ARE THE COMBINE
C THERMAL ID NUMBERS INCREMENTED IN ABH RECORDS SECTION ORDER; NABH ARE
C THE ID NUMBERS CORRESPONDING TO THE HETEROGENEOUS NUMBER DENSITIES
C (HMASS) OF THE ISOTOPES LISTED IN ORDER IN THE ABH RECORDS SECTION
C
READ(5,*)
DO 457 I=2, N
  if (nrx.eq.3) then
    READ(5,*) FAST(I), THERM(I), DANF(I)
    WRITE(6,*), 'I, FAST(I), THERM(I), DANF(I):'
    WRITE(6,939) I, FAST(I), THERM(I), DANF(I)
  else
    READ(5,*) FAST(I), THERM(I), DANF(I), ABH(I-1), NABH(I-1)
    WRITE(6,*), 'I, FAST(I), THERM(I), DANF(I), ABH(I-1), NABH(I-1):'
    WRITE(6,939) I, FAST(I), THERM(I), DANF(I), ABH(I-1), NABH(I-1)
  endif
939 FORMAT(1X,1I2,1F6.1,1F10.5,1F5.1,1F10.5,2X,I2)
457 WRITE(6,*)

C
C CALC HETEROGENEOUS ATOM DENSITIES (HMASS) FROM HOMOGENEOUS (XMASS)
C
DO 889 I=2, N
  HMASS(I) = XMASS(I)/FVF
C
C SET UP OUTPUT FILE **ACTUAL OUTFILE WILL BE NAME OF COMBINE INPUT FILE
C
close(2)
OPEN(2, FILE='tacrx.in')

C INPUT FILE WRITING LOOPS
C
C FIRST TWO SECTIONS OF COMBINE INPUT FILE
C
NN = N - 1
  if (nrx.eq.1) WRITE(2,450) 'ACTINIDE LWR '
  if (nrx.eq.2) WRITE(2,450) 'ACTINIDE HTGR'
  if (nrx.eq.3) WRITE(2,450) 'ACTINIDE LMR'
450 FORMAT(A14)
  if (nrx.eq.3) then
    WRITE(2,941) '1010101 0 2 1 1 ', NN, ' 3 3 2 0 0 0 0 0 61 1 0'
    WRITE(2,876) '1010201 166, 113, 110'
876 FORMAT(A21)
  else
    WRITE(2,941) '1010101 2 0 1 1 ', NN, ' 2 3 1 0 0 0 0 0 61 1 0'
    WRITE(2,451) '1010201 166, 61'
  endif
941 FORMAT(A16,I2,A23)
451 FORMAT(A15)
WRITE(2,*), '1030101 1 3 32 0 1 2'
452 FORMAT(A20)
C
J1 = 1030201
C
DO 675 I=1, NN
WRITE(2,942) J1,' 2 ', I, ' 0 1 1'
675  J1 = J1 + 1
942  FORMAT(I7,A3,I2,A6)
C
WRITE(2,*)
WRITE(2,943) '1041001', B2, B2, RTEMP, ' 0. 0. 0. 0.'
943  FORMAT(A7,2F10.8,1F7.1,A12)
WRITE(2,*)
C
LOOP TO WRITE ALL THE INDIVIDUAL ISOTOPE ENTRIES
C
J2 = 1042001
J3 = 1043011
J4 = 1043012
J5 = 1043013
C
DUE TO DOUBLE HETEROGENEITY THE MHTGR MUST BE DONE SEPARATELY
IF (NRX.EQ.2) GOTO 298
C
DO 361 I=2, N
C
WRITE(2,944) '*ISOTOPE ', I
944  FORMAT(A9,I2)
WRITE(2,945) J2, FAST(I), THERM(I), XMASS(I), ' 0.0'
945  FORMAT(I7,1F6.1,1F9.5,1E13.7,A4,2F10.5)
C
IF(DANF(I) .GT. 0.0) THEN
C
WRITE(2,946) J3, RTEMP, ' 2.0', HMASS(I), FRAD, DANF(I), ' 5.0'
946  FORMAT(I7,1F7.1,A4,1E13.7,1F8.6,1F4.1,A4)
C
IF(I .EQ. MOS1) THEN
WRITE(2,947) J4, HMASS(MOS2), HMASS(MOS3), ' 0.0',
947  FORMAT(I7,2E13.7,A4,2F6.1,A4)
C
ELSE IF(I .EQ. MOS2) THEN
WRITE(2,947) J4, HMASS(MOS1), HMASS(MOS3), ' 0.0',
947  FORMAT(I7,2E13.7,A4,2F6.1,A4)
C
ELSE IF(I .EQ. MOS3) THEN
WRITE(2,947) J4, HMASS(MOS1), HMASS(MOS2), ' 0.0',
947  FORMAT(I7,2E13.7,A4,2F6.1,A4)
C
ELSE
WRITE(2,948) J4, HMASS(MOS1), HMASS(MOS2), HMASS(MOS3),
948  FORMAT(I7,3E13.7,3F6.1)
ENDIF

WRITE(2,949) J5, ' 1.0 1.0 1.0'
949 FORMAT(I7,A12)

ENDIF

WRITE(2,*)
J2 = J2 + 1
J3 = J3 + 10
J4 = J4 + 10
J5 = J5 + 10

361 CONTINUE
GOTO 224

C MHTGR SECTION

C ELZERO AND CZERO ARE DESCRIBED IN COMBINE MANUAL
C THE HMASS VALUES MUST BE FOR THE OX AND C CORRECTED FOR GRAIN EFFECT

298 READ(5,*) ELZERO, CZERO, HMASS(MOS1), HMASS(MOS2)
DO 261 I=2, N
   WRITE(2,944) '*ISOTOPE ', I
   IF(DANF(I) .GT. 0.0) THEN
      WRITE(2,945) J2, FAST(I), THERM(I), XMASS(I), ' 0.0', ELZERO, CZERO
      WRITE(2,946) J3, RTEMP, ' 2.0', HMASS(I), FRAD, DANF(I), ' 5.0'
      IF(I .EQ. MOS3) THEN
         WRITE(2,947) J4, HMASS(MOS1), HMASS(MOS2), ' 0.0 ',
         6       FAST(MOS1), FAST(MOS2), FAST(MOS3)
      ENDIF
   ELSE
      WRITE(2,948) J4, HMASS(MOS1), HMASS(MOS2), HMASS(MOS3),
       6       FAST(MOS1), FAST(MOS2), FAST(MOS3)
   ENDIF
   WRITE(2,949) J5, ' 1.0 1.0 1.0'
ELSE
   WRITE(2,945) J2, FAST(I), THERM(I), XMASS(I), ' 0.0'
ENDIF
WRITE(2,*)
J2 = J2 + 1
J3 = J3 + 10
J4 = J4 + 10
J5 = J5 + 10

261 CONTINUE
224 CONTINUE

C READ IN AND THEN WRITE DANCOFF PARAMETERS SECTION; LL IS NUMBER
C OF LINES TO BE READ

READ(5,*)
READ(5,*) LL
DO 496 I=1, LL
   READ(5,342) AA
496 WRITE(2,342) AA
SUBROUTINE NEWND
C
SUBROUTINE TO DELETE FISSION PRODUCT ATOMS, and replace with fissile material, decay the actinides for some period outside of the reactor, then mix in fertile material to match the desired keff (removing fissile material to maintain fuel mass)
C
IMPLICIT REAL*8(A-H,O-Z)
COMMON B2, N, MN, SIGTR1(40), SIGTR2(40), SIGCA1(40), SIGCA2(40)
COMMON SIGFS1(40), SIGFS2(40), XNU1(40), XNU2(40), SIG12(40)
COMMON DIFF1, DIFF2, SGTAB1, SGTAB2, SGTFS1, SGTFS2, SGTNFS1
COMMON SGTNFS2, SIGT12, TFR, XMASS(40)
COMMON PHIB, FXS(40), AXS(40), VOL, POWER
COMMON XMASSP(40), ALAM(40), LD(40,2), LCN(40,2), CFRAC(40,2)
COMMON YIELD, DELT, LFP, DFRAC(40,2), XMOLWT(40), TDM
Dimension ifuel(40), ffrac(40)
C
SET UP INPUT AND OUTPUT FILES
C
OPEN(5,FILE=’check.in’)
C
TOTFUEL IS THE TOTAL NUMBER OF HEAVY METAL FUEL ATOMS ALLOWED,
C DTIME IS THE LENGTH OF DECAY TIME OUTSIDE THE REACTOR (IN SECONDS)
C NORM IS ONE FOR NORMAL RUN AND ZERO FOR BENCHMARK (-U234/U236)
C
C READ IN DESIRED K EFFECTIVE FOR BOC AND TOLERANCE
C
READ(5,*)
READ(5,*) XKBOC, TOLER, FRAC, DENOM
WRITE(1, *) 'K BOC, TOLERANCE, CHANGE FRACTION, C.FRAC. CHANGE:
WRITE(1,280) XKBOC, TOLER, FRAC, DENOM
280 FORMAT(1X,1F8.5,1F12.9,2F7.3,1X,I2)

WRITE(1,*)
READ(5,*)
READ(5,*) TOTFUEL, DTIME, norm
WRITE(1, *) 'TOTFUEL, DTIME, NORM:'
WRITE(1,40) TOTFUEL, DTIME, norm
read(5,*)
read(5,*) (ifuel(i),i=1,mn)
read(5,*)
read(5,*) (ffrac(i),i=1,mn)
write(1,41) (ifuel(i),i=1,mn)
41 format(' IFUEL(i),i=1,mn): 0/1/2 no change/enricher/fertile’,
6’ dilutant’/,,(40I2))
write(1,42) (ffrac(i),i=1,mn)
42 format(' ffrac(i),i=1,mn): fraction of either enricher or’,
6’ dilutant as specified by ifuel’/,,(8e10.4))
40 FORMAT(2E13.7,1X,I2)

WRITE(1,*)

C REMOVE FISSION PRODUCT ATOMS (LUMPED FP)
C
C XMASS(LFP) = 0.0
C
C DECAY THE ACTINIDES FOR TIME PERIOD, DTIME (in days)
C
delt=dtame
CALL marchd
C
CALL SUBROUTINE KC TO FIND G/MTM’S AND K EFFECTIVE AT EOC AND
C AFTER COOLING PERIOD (DTIME); WRITE DATA TO KCEOCY.OUT
C
close(2)
OPEN(2, FILE = 'kceocy.out')
ien=1
Call kc(ien)
close(2)

OPEN(2,FILE = 'cycle')
C
C CHECK FOR NORMAL OR BENCHMARK RUN
C
IF(NORM .EQ. 1) goto 632
C FOR BENCHMARKING WITH DP-1518: SUBTRACT 10% OF THE MASS FROM U234 AND U236 BEFORE CALCULATING RE-ENRICHMENT VALUES

C

XMASS(8) = XMASS(8) - XMASS(8)*.10
XMASS(9) = XMASS(9) - XMASS(9)*.10
WRITE(2,*) 'NEW U234, U236; XMASS 8,9:'
WRITE(2,727) XMASS(8), XMASS(9)
727 FORMAT(1X,2E13.7)
WRITE(2,*)

C CALCULATE TOTAL NUMBER OF WASTE ACTINIDES

C

TAXMASS = 0.0

DO 459 I=2, MN
459 IF(FXS(I) .GT. 0.0) TAXMASS = TAXMASS + XMASS(I)

C

WRITE(2,*) 'TOTAL ACTINIDE XMASS:'
WRITE(2,81) TAXMASS
81 FORMAT(1X,8E13.7)
WRITE(2,*)

C CALCULATE NEW ATOMS OF FISSILE TO BE ADDED

C

XNFUEL = TOTFUEL - TAXMASS
WRITE(2,*) 'TOTFUEL, XNFUEL:'
WRITE(2,81) TOTFUEL, XNFUEL
WRITE(2,*)

C MAKE XNFUEL INTO fissile ISOTOPES

C

do 983 i=1,mn
if (ifuel(i).ne.1) goto 983
XMASS(i) = XMASS(i) + ffrac(i)*XNFUEL
WRITE(2,*) ' i, XMASS(i) AFTER XNFUEL ADDED:'
WRITE(2,82) i, XMASS(i)
983 continue
82 format(i6,e13.7)
WRITE(2,*)

C CHECK K EFFECTIVE

C

MFLAG = 3
ICHECK = 1
91 call kcal(xkeff)
C

WRITE(2,*) 'KEFF AND ICHECK:'
WRITE(2,543) XKEFF, ICHECK
543 FORMAT(1X,1F11.8,1X,I3)
WRITE(2,*)
C COMPARE CALCULATED K EFFECTIVE TO DESIRED K BOC. IF THE TOLERANCE IS C MET THEN GO TO NEXT SECTION, IF NOT CHECK TO SEE IF KEFF IS GREATER C THAN OR LESS THAN THE DESIRED KBOC. C EXCHANGE fertile for fissile.
C SET LIMIT ON ITERATIONS TO AVOID AN INFINITE LOOP, PRINT MESSAGE IF C THIS LIMIT IS REACHED
C
CTOLER = ABS((XKEFF - XKBOC)/XKBOC)
IF(CTOLER .LT. TOLER) goto 877
   IF(XKEFF .GT. XKBOC) THEN
      IF(MFLAG .EQ. 0) FRAC = FRAC/DENOM
      eee=0.0
      do 734 ii=1,mn
         if (ifuel(ii).EQ.1) EEE = FRAC*XMASS(ii)*ffrac(ii)+eee
      continue
      do 735 ii=1,mn
         if (ifuel(ii).EQ.1) XMASS(ii) = XMASS(ii) - EEE*ffrac(ii)
         if (ifuel(ii).EQ.2) XMASS(ii) = XMASS(ii) + EEE*ffrac(ii)
      continue
      MFLAG = 1
   ELSE
      IF(MFLAG .EQ. 1) FRAC = FRAC/DENOM
      eee=0.0
      do 834 ii=1,mn
         if (ifuel(ii).EQ.1) EEE = FRAC*XMASS(ii)*ffrac(ii)+eee
      continue
      do 835 ii=1,mn
         if (ifuel(ii).EQ.2) XMASS(ii) = XMASS(ii) - EEE*ffrac(ii)
         if (ifuel(ii).EQ.1) XMASS(ii) = XMASS(ii) + EEE*ffrac(ii)
      continue
      MFLAG = 0
   ENDIF
C ICHECK = ICHECK + 1
IF(ICHECK .GT. 101) GOTO 937
C
GOTO 91
C
937 WRITE(2,*) 'COUNTER LIMIT WAS REACHED'
WRITE(2,*)
C
877 sgtca1=sgtabl-sgtfs1
sgtca2=sgtab2-sgtfs2
sgtttr1=1/(3*diff1)
sgtttr2=1/(3*diff2)
WRITE(2,*) 'SGTTR1, SGTTR2, SGTCA1, SGTCA2, SGTFS1:'
WRITE(2,620) SGTTR1, SGTTR2, SGTCA1, SGTCA2, SGTFS1
WRITE(2,*)
WRITE(2,*) 'SGTFS2, SGTNFS1, SGTNFS2, SIGT12:'
WRITE(2,620) SGTFS2, SGTNFS1, SGTNFS2, SIGT12
WRITE(2,*)
620 FORMAT(1X,5E13.7)
CLOSE(2)

C OPEN ANOTHER OUTPUT FILE
C CHANGE NUMDEN.TP TO NUMDEN.IN FOR next cycle in batch program
C
OPEN(2, FILE='numden.tp')

C WRITE THE NUMBER OF ISOTOPES AND THEN THEIR NUMBER DENSITIES AND ID
C NUMBERS TO NUMDEN.TP

WRITE(2,*) 'TOTAL DENSITY/MASS (G/CM3):'
WRITE(2,786) TDM
786 FORMAT(1X, 1E13.7)
WRITE(2,963) MN
963 FORMAT(I2)
DO 62 KN = 2, MN
62 WRITE(2,959) XMASS(KN), KN, XMOLWT(KN)
959 FORMAT(1E13.7, 2X, I2, E13.7)

WRITE(2,*)

C CALL SUBROUTINE KC TO WRITE BOC G/MTM'S AND K EFFECTIVE (DATA
C WRITTEN TO KCBOC.OUT)

close(2)
Open(2, file='kcboc.out')
ie=1
Call kc(ien)
close(2)
RETURN
END

C ****************************************************************
C Subroutine kc(ien)
C
C SUBROUTINE TO CONVERT ATOMS/BARN-CM TO G/MTM AS WELL AS TO CALCULATE
C K EFFECTIVE AT EOC AND COOLING PERIOD
C
IMPLICIT REAL*8(A-H, O-Z)
COMMON B2, N, MN, SIGTR1(40), SIGTR2(40), SIGCA1(40), SIGCA2(40)
COMMON SIGFS1(40), SIGFS2(40), XNU1(40), XNU2(40), SIG12(40)
COMMON DIFF1, DIFF2, SGTAB1, SGTAB2, SGTFS1, SGTFS2, SGTNFS1
COMMON SGTNFS2, SIGT12, TFR, XMASS(40)
COMMON PHIB, FXS(40), AXS(40), VOL, POWER
COMMON XMASSF(40), ALAM(40), LD(40, 2), LCN(40, 2), CFRAC(40, 2)
COMMON YIELD, DELT, LFP, DFRAC(40, 2), XMOLWT(40), TDM

C WRITE ATOM/B-CM VALUES BEFORE THE CALCULATED G/MTM'S
C
WRITE(2,*)
WRITE(2,*) 'XMASS(I) AND I:'
WRITE(2,*)
DO 345 I=2, MN
C CHANGE UNITS FROM ATOMS/BARN-CM TO G/CM3 WHILE ALSO FINDING
C THE FRACTION OF EACH ISOTOPE WITH RESPECT TO THE INITIAL TOTAL
C G/CM3 (TDM) AND MULTIPLYING BY 1E6 TO OBTAIN G/MTM (GMTM(I))
C ALSO CALCULATE TOTAL G/MTM (GMTMT)

C if (ien.eq.0) close(2)
if (ien.eq.0) open(2, file='kcon.out')
GMTMT = 0.0

C WRITE(2,*)
C WRITE(2,*) 'G/MTM AND I:'
C WRITE(2,*)

C DO 228 I=2, MN
   IF(FXS(I) .GT. 0.0) THEN
      GMTM = (((XMASS(I)*XMOLWT(I))/6022)/TDM)*(10.0**6)
      GMTMT = GMTMT + GMTM
      WRITE(2,101) GMTM, I
   101 FORMAT(1X,1E13.7,1X,I2)
   ENDIF
228 CONTINUE

C WRITE OUT TOTAL G/MTM VALUE

C WRITE(2,*)
C WRITE(2,*) 'TOTAL G/MTM:'
C WRITE(2,888) GMTMT
888 FORMAT(1X,1E13.7)
C WRITE(2,*)

C CALCULATE K EFFECTIVE OF THE CELL

C INITIALIZE TOTAL MACRO XS COUNTERS

C Call kcal(xkeff)

C WRITE(2,*)
C WRITE(2,*) 'KEFF:'
C WRITE(2,543) XKEFF
543 FORMAT(1X,1F11.8)
C WRITE(2,*)
C return
C end

C ****************************************************************
C Subroutine kcal(xkeff)
C
C SUBROUTINE TO CONVERT ATOMS/BARN-CM TO G/MTM AS WELL AS TO CALCULATE
C K EFFECTIVE AT EOC AND COOLING PERIOD
IMPLICIT REAL*8(A-H,O-Z)
COMMON B2, N, MN, SIGTR1(40), SIGTR2(40), SIGCA1(40), SIGCA2(40)
COMMON SIGFS1(40), SIGFS2(40), XNU1(40), XNU2(40), SIG12(40)
COMMON DIFF1, DIFF2, SGTAB1, SGTAB2, SGTFS1, SGTFS2, SGTNFS1
COMMON SGTNFS2, SIGT12, TFR, XMASS(40)

SGTTR1 = 0.0
SGTTR2 = 0.0
SGTCA1 = 0.0
SGTCA2 = 0.0
SGTFS1 = 0.0
SGTFS2 = 0.0
SGTNFS1 = 0.0
SGTNFS2 = 0.0
SIGT12 = 0.0

NEED NON-COMBINE TWO GRP DATA!!!

DO 323 I=2, MN
   SGTTR1 = SGTTR1 + XMASS(I)*SIGTR1(I)
   SGTTR2 = SGTTR2 + XMASS(I)*SIGTR2(I)
   SGTCAl = SGTCAl + XMASS(I)*SIGCA1(I)
   SGTCA2 = SGTCA2 + XMASS(I)*SIGCA2(I)
   SGTFS1 = SGTFS1 + XMASS(I)*SIGFS1(I)
   SGTFS2 = SGTFS2 + XMASS(I)*SIGFS2(I)
   SGTNFS1 = SGTNFS1 + XMASS(I)*XNU1(I)*SIGFS1(I)
   SGTNFS2 = SGTNFS2 + XMASS(I)*XNU2(I)*SIGFS2(I)
323

SIGT12 = SIGT12 + XMASS(I)*SIG12(I)

CONVERT CROSS-SECTION DATA: TRANSPORT TO DIFFUSION COEFF; CAPTURE AND
FISSION TO OBTAIN ABSORPTION

DIFF1 = 1/(3*SGTTR1)
DIFF2 = 1/(3*SGTTR2)
SGTAB1 = SGTCAl + SGTFS1
SGTAB2 = SGTCA2 + SGTFS2

CALCULATE K EFFECTIVE

write(6,*) 'sal,sa2,s1-2,snf1,snf2,d1,d2,b2'
write(6,137) sgtabl,sgtab2,sigt12,sgtnfs1,sgtnfs2,diff1,diff2,b2
137 format(6e12.5)
VV = SGTNFS1 + ((SGTNFS2*SIGT12)/(DIFF2*B2 + SGTAB2))
WW = DIFF1*B2 + SGTAB1 + SIGT12

XKEFF = VV/WW
RETURN
END

****************************************************************
SUBROUTINE MARCHD

PERFORMS Decay chain depletion at zero power

IMPLICIT REAL*8(A-H,O-Z)
COMMON B2, N, MN, SIGTR1(40), SIGTR2(40), SIGCA1(40), SIGCA2(40)
COMMON SIGFS1(40), SIGFS2(40), XNU1(40), XNU2(40), SIG12(40)
COMMON DIFF1, DIFF2, SGTAB1, SGTAB2, SGTS1, SGTS2, SGTNFS1
COMMON SGTNFS2, SIGT12, TFR, XMASS(40)
COMMON PHIB, FXS(40), AXS(40), VOL, POWER
COMMON XMASSP(40), ALAM(40), LD(40,2), LCN(40,2), CFRAC(40,2)
COMMON YIELD, DELT, LFP, dfrac(40,2)

C CONVERT DELT (IN DAYS) INTO ONE-TENTH OF DELT (OR TEMP), IN SECONDS;
C Then start TIME LOOP TO DO TEN TIME STEPS OF SIZE TEMP
C
TEMP = DELT*24.0*3600.0/10.0
DO 200 KT = 1, 10

C RENAME INITIAL NUMBER DENSITIES TO VARIABLE MASSP, SO AS TO USE
C VARIABLE MASS FOR COMPUTATION OF NEW NUMBER DENSITIES
C
DO 20 KN = 2, MN
20 XMASSP(KN) = XMASS(KN)

ITERATION LOOP FOR IMPLICIT METHOD USED TO
DETERMINE NEW ISOTOPIC COMPOSITION
C
DO 50 KKK = 1, 5
DO 50 KN = 2, MN

CALCULATE THE LOSSES DUE TO DECAY
C
TEMP2=-(XMASS(KN)+XMASSP(KN))*ALAM(KN)
C
CALCULATE SOURCE FROM DECAY (TWO DECAY SOURCES ALLOWED), DFRAC IS
C USED WHEN THERE ARE TWO TYPES OF DECAY FROM A SOURCE ISOTOPE
C
DO 30 K=1, 2
   KK = LD(KN,K)
   DD = DFRAC(KN,K)
   IF (KK) 30, 30, 28
30 TEMP2 = TEMP2 + (DD*ALAM(KK))*(XMASS(KK) + XMASSP(KK))

CALCULATE NEW NUMBER DENSITY FOR EACH ISOTOPE
C
50 XMASS(KN) = XMASSP(KN) + 0.5*TEMP*TEMP2
200 CONTINUE
RETURN
END
A2. New Batch Files

The following set of batch files were used to run the actinide burning cases. SDBUST.BAT calls BUSTB.BAT and BUSTE.BAT.

SDBUST.BAT:

LDBURN
ren ACRX.ISO acrx1.iso
ren NUMDEN.IN numden1.in
DEL ECHO.OUT
COPY NUMDEN.OUT CLTS1
COPY CLTS1+KCON.OUT
DEL KCON.OUT
REN NUMDEN.OUT NUMDEN.IN
REN TACRX.IN ACRX.IN

call bustb.bat 1 2
call bustb.bat 1 3
call buste.bat 1 2

ren mdata.in mdata1.in
ren mdata2.in mdata.in

call bustb.bat 2 1
call bustb.bat 2 2
call bustb.bat 2 3
call buste.bat 2 3

call bustb.bat 3 1
call bustb.bat 3 2
call bustb.bat 3 3
call buste.bat 3 4

call bustb.bat 4 1
call bustb.bat 4 2
call bustb.bat 4 3
call buste.bat 4 5

call bustb.bat 5 1
call bustb.bat 5 2
call bustb.bat 5 3
call buste.bat 5 6
call bustb.bat 6 1
call bustb.bat 6 2
call bustb.bat 6 3
call buste.bat 6 7

call bustb.bat 7 1
call bustb.bat 7 2
call bustb.bat 7 3
call buste.bat 7 8

call bustb.bat 8 1
call bustb.bat 8 2
call bustb.bat 8 3
call buste.bat 8 9

call bustb.bat 9 1
call bustb.bat 9 2
call bustb.bat 9 3
call buste.bat 9 10

call bustb.bat 10 1
call bustb.bat 10 2
call bustb.bat 10 3
call buste.bat 10 11

call bustb.bat 11 1
call bustb.bat 11 2
call bustb.bat 11 3
call buste.bat 11 12

BUSTB.BAT:

COPY ACRX.IN d:
D:
COMBINE <ACRX.IN >ACRX.OUT
REN COMBINE.SIG ACRX.ISO
COPY ACRX.ISO C:
DEL ACRX.IN
DEL ACRX.ISO
DEL ACRX.OUT
C:
DEL ACRX.IN
LDBURN
DEL ACRX.ISO
DEL NUMDEN.IN
DEL ECHO.OUT
COPY NUMDEN.OUT C%1TS%2
COPY C%1TS%2+KCON.OUT
DEL KCON.OUT
REN NUMDEN.OUT NUMDEN.IN
REN TACRX.IN ACRX.IN
BUSTE.BAT:

COPY ACRX.IN D:
D:
COMBINE <ACRX.IN >ACRX.OUT
REN COMBINE.SIG ACRX.ISO
DEL ACRX.IN
COPY ACRX.ISO C:
DEL ACRX.ISO
DEL ACRX.OUT
C:
DEL ACRX.IN
REN Crx NCHOICE
REN ENRICH Crx
REN CHECK.IN CHECK.OUT
REN CHECK%2.IN CHECK.IN
LDBURN
REN CHECK.IN CHECK%2.IN
REN CHECK.OUT CHECK.IN
DEL ACRX.ISO
DEL NUMDEN.IN
DEL ECHO.OUT
REN Crx ENRICH
REN NCHOICE Crx
REN NUMDEN.OUT EOC%1
COPY EOC%1+KCON.OUT
DEL KCON.OUT
REN PCYCLE PCYCLE%1
REN CYCLE CYCLE%1
REN KCEOCY.OUT KCEOCY%1
REN KCBOC.OUT KCBOC%2
COPY NUMDEN.TP BOC%2
REN NUMDEN.TP NUMDEN.IN
REN TACRX.IN ACRX.IN
APPENDIX B: Sample Input

Bl: Input files for the Burnup Program.

CRX:

2 1

ENRICH:

2 2

ACRX.IN:

0V ISOTXS *COMBINE VERS* 321 1
1D 2 20 1 1 1 1 1 1 1
2D * ACTINIDE HTGR *
* IID 32 IID 1 IID 2 IID 3 IID 4 IID 5 IID 6 IID 7 IID 8
IID 9 IID 10 IID 11 IID 12 IID 13 IID 14 IID 15 IID 16 IID 17 IID 18
IID 19
1.00000E+00 0.00000E+00
1.89610E+02 8.58207E+00 1.69046E+07 6.83000E-01 1.00000E-03
0 3 6 9 12 15 18 21 24 27 30 33

36 39 42 45 48 51 54 57

4D SEQ 1 ENDF/B IID 32
0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01
0 0 1 0 0 0 0 0 0 1 0 0

2 2 2 2 2 2 1

5D 2.52949E-01 4.13277E-01 3.03790E-01 4.17681E-01 2.14441E-03
4.19086E-02 1.23407E-03 3.96897E-02 2.91030E+00 2.89559E+00
7D 1.35360E-02 2.99747E-01 3.22547E-01 6.65009E-04 2.41319E-04
1.97619E-02 1.71743E-02 1.35100E-04

4D SEQ 2 ENDF/B IID 1
0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01
0 0 1 0 0 0 0 0 0 1 1 0 0

2 2 2 2 2 2 1

5D 2.52495E-01 8.324125E+00 3.24125E+01 8.18699E+00 1.15466E+01
6.22451E-01 4.43924E-02 1.13643E-06 2.74865E+00 2.41358E+00
7D 0.00000E+00 2.08206E+01 7.56454E+00 9.06897E-04 0.00000E+00
1.16906E+00 2.19029E-02 2.08115E-04

4D SEQ 3 ENDF/B IID 2
0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01
0 0 1 0 0 0 0 0 0 1 1 0 0

2 2 2 2 2 2 1

5D 2.09806E+01 1.11858E+02 2.68932E+01 1.16199E+02 5.80530E+00
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<td>2.21845E+02</td>
<td>3.38206E+01</td>
<td>2.08778E+02</td>
<td>1.74422E+01</td>
<td>1.60346E+01</td>
<td>1.37548E-02</td>
<td>2.83263E-04</td>
<td>0.00000E+00</td>
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<tr>
<td>7D</td>
<td>1.16924E+00</td>
<td>1.16384E-04</td>
<td>6.58315E-05</td>
<td>0.00000E+00</td>
<td>1.60346E+01</td>
<td>3.78548E-02</td>
<td>2.83263E-04</td>
<td>0.00000E+00</td>
<td>1.07616E+00</td>
</tr>
<tr>
<td>4D</td>
<td>3.80612E+01</td>
<td>4.29148E+01</td>
<td>5.34487E+01</td>
<td>4.19237E+01</td>
<td>4.16937E+01</td>
<td>3.69517E+01</td>
<td>2.20522E+01</td>
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<td>1.97201E+00</td>
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<td>1.12419E-04</td>
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<td>1.07616E+00</td>
<td>1.39710E-02</td>
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<td>3.80612E+01</td>
</tr>
<tr>
<td>7D</td>
<td>3.69517E+01</td>
</tr>
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0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01
 0 0 1 0 0 0 0 0 1 1 0 0
 2 2 2 2 1
5D 1.89643E+01 5.34374E+01 2.42618E+01 5.5133E+01 6.34222E+00
4.05278E+01 1.77912E+00 1.23714E+00 2.95449E+00 0.00000E+00
7D 0.00000E+00 1.61393E+01 1.37484E+01 1.12320E-03 0.00000E+00
1.32130E+00 3.95372E-02-2.76080E-04
4D SEQ 18 ENDF/B IID 17
0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01
 0 0 1 0 0 0 0 0 1 1 0 0
 2 2 2 2 1
5D 1.33718E+01 1.72702E+01 1.67150E+01 8.61188E+00 3.98609E+00
3.34291E-01 3.97451E-01 1.65296E-02 3.73045E+00 3.47999E+00
7D 0.00000E+00 1.23305E+01 8.26106E+00 9.55621E-04 0.00000E+00
1.26110E+00-8.44583E+00-2.10056E-04
4D SEQ 19 ENDF/B IID 18
0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01
 0 0 1 0 0 0 0 0 1 1 0 0
 2 2 2 2 1
5D 2.75186E+01 8.56310E+02 3.74020E+01 8.54797E+02 2.58431E+01
8.40508E+02 4.07269E-01 4.04275E+00 3.30273E+00 3.09000E+00
7D 0.00000E+00 1.11504E+01 1.02463E+01 1.21260E-03 0.00000E+00
1.66720E+00 2.98528E-02-2.43415E-04
4D SEQ 20 ENDF/B IID 19
0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01 0.10000E+01
 0 0 1 0 0 0 0 0 1 1 0 0
 2 2 2 2 1
5D 3.00476E+01 2.71232E+02 3.87144E+01 2.66645E+02 3.55050E+00
2.46120E+01 2.54779E+01 2.35035E+02 3.61115E+00 3.60520E+00
7D 0.00000E+00 9.68537E+00 6.99801E+00 5.87588E-04 0.00000E+00
9.87809E-01-7.13223E+00-1.17443E-04
MDATA.IN:

DELT, LFP, RFTIME
300.0 33 60.
U238 (DECAY CONST; DECAY, CAPT LOGIC; DECAY FRACTIONS)
4.870E-18 14 0 0 0 1.0 1.0 1.0 1.0
U235
3.090E-17 9 0 8 0 1.00 1.0 1.0 1.0
O MOD
0.0 0 0 0 0 1.0 1.0 1.0 1.0
O FUEL
0.0 0 0 0 0 1.0 1.0 1.0 1.0
H2/H2O
0.0 0 0 0 0 1.0 1.0 1.0 1.0
U234
8.890E-14 17 0 0 0 1.00 1.0 1.0 1.0
U236
9.190E-16 10 0 3 0 1.00 1.0 1.0 1.0
PU239
9.010E-13 16 23 2 17 1.0 0.997 1.0 1.0
PU240
3.340E-12 15 0 9 0 1.00 1.0 1.0 1.0
PU241
1.660E-09 20 0 10 0 1.0 1.0 1.0 1.0
ZR
0.0 0 0 0 0 1.0 1.0 1.0 1.0
NP237
1.026E-14 19 11 8 0 1.0 0.000023 1.0 1.0
PU242
5.800E-14 18 21 11 19 1.00 0.1791 1.0 0.162972
CM244
1.250E-09 0 0 23 16 1.0 1.0 1.0 1.0
AM243
2.760E-12 24 23 21 14 1.0 0.003 1.0 1.0
PU238
2.540E-10 22 21 13 0 1.00 0.0048 1.0 1.0
CM246
3.990E-12 28 0 20 0 1.00 1.0 1.0 1.0
AM241
4.800E-11 11 0 0 0 0.999977 1.0 1.0 1.0
CM245
2.360E-12 27 26 15 0 1.00 0.000022 1.0 1.0
AM242M
1.450E-10 0 0 19 0 1.0 1.0 0.0946 1.0
CM242
4.920E-08 21 0 19 0 0.8161 1.0 0.742428 1.0
CM243
6.860E-10 0 0 22 0 1.0 1.0 1.0 1.0
CM247
1.370E-15 29 0 18 0 1.0 1.0 1.0 1.0
CM248
4.670E-14 30 0 24 0 0.97 1.0 1.0 1.0
BK249
2.580E-08  0  0  25  0  1.0  1.0  1.0  1.0
CF249
6.100E-11  26  0  0  0  .999978  1.0  1.0  1.0
CF250
1.660E-09  0  0  27  0  1.0  1.0  1.0  1.0
CF251
2.750E-11  0  0  28  0  1.0  1.0  1.0  1.0
CF252
8.350E-09  0  0  29  0  1.0  1.0  1.0  1.0
XE135
0.0   0  0  0  0  1.0  1.0  1.0  1.0
SM149
0.0   0  0  0  0  1.0  1.0  1.0  1.0
LFP
0.0   0  0  0  0  1.0  1.0  1.0  1.0
YIELD (PG. 4 OF BAT FILE SECTION, .96151 (leopard))
   .96151

FWDATA.IN:

RTEMP, FVF, FRAD
1000. .2389 .6223
MOS1,M-2,M-3,NMF
3  5  9  17
FAST, THERM, DANF, ABH, NABH
501.0  92.23805  2.0  6.01002   5
502.0  92.23505  3.0  8.01050   3
505.0  8.01050   0.0  92.23405   7
506.0  6.00802   0.0  92.23505   3
506.0  6.01002   0.0  92.23605   8
551.0  92.23405  3.0  92.23805   2
552.0  92.23605  0.0  93.23705  13
553.0  94.23905  3.0  94.23805  17
554.0  94.24005  3.0  94.23905   9
555.0  94.24105  0.0  94.24005  10
584.0  40.00005  0.0  94.24105  11
587.0  93.23705  3.0  94.24205  14
592.0  94.24205  3.0  95.24105  19
612.0  96.24452  3.0  95.24305  16
613.0  95.24305  3.0  96.24452  15
644.0  94.23805  3.0  96.24505  20
675.0  96.24605  3.0  96.24605  18
676.0  95.24105  3.0  40.00005  12
679.0  96.24505  3.0  1.00600  6
0.0267  .81  .001276  .000188
DANCOFF SECTION (FIRST NUMBER IS NUMBER OF LINES FOLLOWING)
   3
*Dancoff calculation parameters
1044011  1.0  2.0  1.2731  0.0  0.0  0.0  0.0
1044012  .3  0.0  0.0  0.0  0.0  0.0  0.0
2010401 3  .6223  .77230  1.2731
2010521 2 6.00802 6.771E-02 40.00005 .1E-19
2010531 1 6.00802 6.771E-02

CHECK.IN:

K BOC, TOLERANCE, FRAC, DENOM
1.02  .000001  .50   2.0
TOTFUEL, DTIME (DAYS) NORM(1),BENCH(0)
1.8703E-04  365.25  1
Ifuel 4 5 6 7 8 910 1 2 3 4 5 6 7 8 920 1 2 3 4 5 6 7 8 930 1 2 3 4
0 0 0 0 0 0 0 0 1 1 1 0 2 1 2 2 1 0 2 2 0 0 0 0 0 0 0 0 0 0 0 0 0 0
ffrac 2 3 4 5 6 7 8 9 10 11 12 13 14
0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.5710 .2230 .1510 0.0 0.4185 .0436
.0200 0.0937 .0114 0.0  .4663 0.0015 0.0
0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0

0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0 0.0
B2. Sample COMINE input for MHTGR

=ACTINIDE HTGR
1010101 2 0 1 1 19 2 3 1 0 0 0 0 0 61 1 0
1010201 166, 61

1030101 1 3 32 0 1 2
1030201 2 1 0 1 1
1030202 2 2 0 1 1
1030203 2 3 0 1 1
1030204 2 4 0 1 1
1030205 2 5 0 1 1
1030206 2 6 0 1 1
1030207 2 7 0 1 1
1030208 2 8 0 1 1
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1030210 2 10 0 1 1
1030211 2 11 0 1 1
1030212 2 12 0 1 1
1030213 2 13 0 1 1
1030214 2 14 0 1 1
1030215 2 15 0 1 1
1030216 2 16 0 1 1
1030217 2 17 0 1 1
1030218 2 18 0 1 1
1030219 2 19 0 1 1

1041001 .00014 .00014 1000. 0. 0. 0. 0.

*ISOTOPE 2
1042001 501.0 92.23805 .1E-19 0.0 0.0267 .81
1043011 1000. 2.0 .1E-19 .6223 2.0 5.0
1043012 .001276 .000188 0.00041380 505.0 506.0 553.0
1043013 1.0 1.0 1.0

*ISOTOPE 3
1042002 502.0 92.23505 .1E-17 0.0 0.0267 .81
1043021 1000. 2.0 .1E-17 .6223 3.0 5.0
1043022 .001276 .000188 0.0004138 505.0 506.0 553.0
1043023 1.0 1.0 1.0

*ISOTOPE 4 Oxygen in fuel
1042003 505.0 8.01050 .0003491 0.0

*ISOTOPE 5 Carbon in moderator
1042004 506.0 6.00802 .051538 0.0

*ISOTOPE 6 Carbon in fuel
1042005 506.0 6.01002 .019923 0.0
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<th>Other Properties</th>
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<td>13</td>
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<td>17</td>
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1043162 .001276 .000188 0.000413800 505.0 506.0 553.0
1043163 1.0 1.0 1.0

*ISOTOPE 18
1042017 675.0 96.24605 .1E-19 0.0 0.0267 .81
1043171 1000. 2.0 .1E-19 .6223 3.0 5.0
1043172 .001276 .000188 0.000413800 505.0 506.0 553.0
1043173 1.0 1.0 1.0

*ISOTOPE 19
1042018 676.0 95.24105 7.849E-6 0.0 0.0267 .81
1043181 1000. 2.0 3.286E-5 .6223 3.0 5.0
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*ISOTOPE 20
1042019 679.0 96.24505 2.525E-08 0.0 0.0267 .81
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1043193 1.0 1.0 1.0

*Dancoff calculation parameters
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1044012 .3 0.0 0.0 0.0 0.0

*ABH records
2010401 3 .6223 .77230 1.2731
2010511 17 6.01002 .8339E-01 8.01050 1.461116E-3 92.23405 .1E-19
2010512 92.23505 1.E-20 92.23605 .1E-19
2010513 92.23805 .1E-19 93.23705 2.949E-5
2010514 94.23805 1.183E-5 94.23905 .00041380
2010515 94.24005 .00018950 94.24105 5.700E-5
2010516 94.24205 4.032E-5 95.24105 3.286E-5
2010517 95.24305 6.602E-6 96.24521 1.409E-06
2010518 96.24505 1.057E-7 96.24605 .1E-19
2010521 2 6.00802 6.771E-02 40.00005 .1E-19
2010531 1 6.00802 6.771E-02
APPENDIX C: Charleston Topical Publication
UNIFORM ANALYSIS OF RECYCLING SELF GENERATED ACTINIDES IN A PWR, LMR, AND MHTGR

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Nuclear Engineering Program
Georgia Institute of Technology
Atlanta, GA 30332
*BMO/MGSM, Norton Air Force Base, CA 92409

ABSTRACT

The recycling of all self generated actinides in a PWR, LMR and MHTGR has been consistently analyzed. Since the actual quantities of the non-plutonium actinides is small, homogeneous recycling with the fuel is assumed. Thus a cell representation of each core is sufficient. A burnup routine was written to link with INEL's cross section code, COMBINE, so accurate cross sections could be attained throughout the multiply loadings. The burnup routine also searched for the appropriate feed enrichment so that the target fuel burnup could be achieved. The analysis was performed for eleven reloads of the fuel. Tables of actinide masses are presented for each reactor type. All three reactor types were effective in burning the plutonium. The non uranium/plutonium actinides never reached equilibrium in the 11 fuel reloads (33 years for the PWR). There was, however, a reduction of the actinides in each case when compared to non-recycling.

Introduction

Recently, there has been a lot of debate on the value of burning actinides in order to decrease the long term toxicity of the high level nuclear waste.1,2,3 Actinides can be burned in all reactor types but with differing effects. Although PWRs and LMRs have been investigated for actinide burning,4,5 this study contains the first analysis of an MHTGR for this purpose. When reviewing the literature on actinide burning, it is difficult to compare different reactor systems due to differing methods and assumptions. This study analyzes all three reactor types with a consistent method.
Method

In order to analyze the actinide burning capability of any reactor accurately, a technique for the calculation of appropriate cross sections must be developed. For this study the COMBINE code\(^6\) from INEL was used. This cross section processing code was developed for analysis of a broad range of reactors and has been benchmarked with good results for a PWR, LMR, and MHTGR.

COMBINE, at the time of this study did not have burnup capabilities so a computer code, ABURN, was written that would burn and decay the isotopes of interest for actinide burning. The burnup technique used was modeled after the MARCH subroutine of 2DB. In addition to burning the actinide isotopes, ABURN was used to remove fission products and add the correct amounts of fissile and fertile materials so each new fuel loading would reach the targeted discharge burnup. ABURN was linked to COMBINE so new cross sections could be generated several times per fuel load.

The above procedure assumed the reactors analyzed could be represented by a cell. The selection of the cell for a PWR was straightforward but for the LMR and the MHTGR significant homogenization had to be performed. It is important to point out that the actinide fuel was irradiated in the average spectrum of the reactor under study. If small volumes of actinides were to be burned, the use of a modified loading plan allowing for a harder spectrum would be desirable. This study ignored any such spatial effects.

For this study it will be assumed that no reactor was built for burning actinides. Rather, each reactor type is required to burn its own actinides.

Figure 1 shows a flow chart of the procedure used.

Each of the three reactor types analyzed had a different base fuel cycle. The PWR uses a straight uranium fuel cycle and hence is fed with enriched uranium. The LMR is based on a U/Pu fuel cycle and is enriched with LWR discharge Pu. The MHTGR is based on the thorium fuel cycle and is enriched by highly enriched U-235.

The code developed for this project used a target \(k\) for the enrichment search. This \(k\) value was an input value that was changed for each cycle. The target \(k\) was selected so that the typical discharge burnup of the reactor type would be maintained. For example, for the PWR it was assumed that a three batch loading scheme was typical. Further it was assumed that a beginning of life \(k\) of 1.2 would yield a typical discharge burnup. If the discharge burnup were 45,000 MWD/MTU, it was desirable that all loadings would
have the same $k$ at 30,000 MWD/MTU. This $k$ would be close to 1.0 but the actual $k$ at 2/3 of the discharge burnup was kept constant. Since the buildup of actinides changed the rate of the loss of reactivity with burnup, the target beginning of life $k$ (BOLK) had to be gained by iteration. A constant BOLK was used for the first run through the system and the slope of reactivity versus burnup was determined. The system was then run a second and third time using the derived slopes to determine BOLK values. Table 1 shows the BOLK values used in the final analysis.

Results

The results of an actinide burning study are often hard to follow because of the lack of a uniform concept of what is desired to be burned. Uranium, plutonium, and the higher isotopes are all actinides, but in different studies uranium and/or plutonium may be separated out. Tables 2, 3 and 4 show the amount of all actinides that would exist from running a 600 MWe plant for 11 loadings. A loading here does not mean the time from startup to shutdown of the reactor but rather the time from when fuel was loaded into the core until that fuel was removed. For example, this would be three cycles in the PWR. The tables show the actinide "recycle" case and "non-recycling" below for comparison. The "non-recycling" cases are just the discharge from a single loading multiplied by the number of loadings and corrected for decay. All values are given for the end of a loading.

Figures 2 and 3 illustrate some of the problems in interpreting the worth of actinide burning. If uranium and plutonium are viewed as fuels for the LMR, then recycling in a PWR produces more "bad" actinides for the first century rather than reducing them. On the other hand, if any non-uranium is a "bad" actinide then almost immediate benefits take place. If one looks at this benefit, it is clear that it comes from the burning of plutonium which could have been achieved with a standard mixed oxide PWR. If one is interested in removal of all the world's actinides (including uranium), then it does not matter whether you recycle or not, since the energy of a fission is about the same for all actinides and the only way to remove an actinide is to fission it.

Figures 4-8 allow for a consistent comparison of the reactor types with regard to their actinide production and burning. First, the non-uranium/thorium actinide inventory for each reactor as a function of the energy produced is shown as Figure 4. It should be noted that the LMR uses a plutonium feed and hence starts with a large inventory.

For those that do not think plutonium is a "bad" actinide, Figure 5 shows the true picture of actinide
burning. Here the best performance is from the MHTGR. This is due to the use of the thorium cycle. One might object to this being artificial but it should be pointed out that the toxicity of the thorium cycle products tends to be less due to their half lives. Next you would notice that the LMR is superior to the PWR due to its harder spectrum.

Figures 6, 7 and 8 demonstrate the effect of actinide burning on the higher actinides.

Conclusions

This paper has studied the effect of recycling each reactor’s fuel into itself. It has used a tool developed for this study which should give superior results when compared to a constant cross section ORIGEN² type model due to its frequent update of the cross sections. Reactor spatial distributions are not considered. For the first time three different reactor types are compared with a consistent method. From these results it would appear that the MHTGR has an advantage over the other two reactor types. This comes from the thorium cycle used. If one were interested in burning previously generated actinides from other reactors, the LMR has a clear advantage. For burning previously generated actinides in a thermal reactor (PWR or MHTGR), the analysis is not yet conclusive and work is continuing.

Acknowledgments

The work presented here was done under support from a grant from General Atomics. The authors would like to express their appreciation for the support and the specifically the project manager, Richard Turner.

References


### Table 1
K Effective Values at the Beginning and End of the Fuel Life

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<th>MHTGR</th>
<th>LMR</th>
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<td>BOLK</td>
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**TABLE 2**

ISOTOPIC INVENTORY FOR A PWR WHEN DISCARDED FUEL

(BASED ON 400 T coherence IN PLANT ASSUMING ALL FUEL IN THE CORE HAS THE DISCARDED RUMP)

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**ACCUMULATED WASTE ASSUMING NO RECYCLING**

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**Figure 3**

ACCU...
### Table 3

**Isotopic Inventory for an Water When Discharging Fuel**

*(Basis: Grams for God Man Plant Assuming All Fuel in the Core Has the Discharge Runup)*

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#### Reactor Inventory

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#### Accumulated Waste Assuming No Recycling

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#### Additional Notes

- The data is presented in a tabular format, listing various isotopes and their inventories, masses, compositions, and clarieties.
- The reference column indicates the source of the data.
- The table provides a summary of the isotopic inventory for various core elements, including uranium, plutonium, americium, and other actinides.
- The data is relevant for understanding the isotopic composition and potential waste management strategies for nuclear reactors.
<table>
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<th>RECYCLING ALL ACTIVITIES</th>
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**ACCELERATED WASTE ASSUMING 0% RECYCLING**

| 1.34E+05 | 3.61E+04 | 1.33E+04 | 1.69E+04 | 1.69E+04 | 1.69E+04 | 1.69E+04 | 1.69E+04 | 1.69E+04 | 1.69E+04 |
| 2.08E+04 | 2.08E+04 | 2.08E+04 | 2.08E+04 | 2.08E+04 | 2.08E+04 | 2.08E+04 | 2.08E+04 | 2.08E+04 | 2.08E+04 |
| 2.81E+04 | 2.81E+04 | 2.81E+04 | 2.81E+04 | 2.81E+04 | 2.81E+04 | 2.81E+04 | 2.81E+04 | 2.81E+04 | 2.81E+04 |
| 3.54E+04 | 3.54E+04 | 3.54E+04 | 3.54E+04 | 3.54E+04 | 3.54E+04 | 3.54E+04 | 3.54E+04 | 3.54E+04 | 3.54E+04 |
| 4.27E+04 | 4.27E+04 | 4.27E+04 | 4.27E+04 | 4.27E+04 | 4.27E+04 | 4.27E+04 | 4.27E+04 | 4.27E+04 | 4.27E+04 |
| 5.00E+04 | 5.00E+04 | 5.00E+04 | 5.00E+04 | 5.00E+04 | 5.00E+04 | 5.00E+04 | 5.00E+04 | 5.00E+04 | 5.00E+04 |
| 5.73E+04 | 5.73E+04 | 5.73E+04 | 5.73E+04 | 5.73E+04 | 5.73E+04 | 5.73E+04 | 5.73E+04 | 5.73E+04 | 5.73E+04 |

**DISCHARGE 1:**

- **Grain:** 400,000 lb
- **Plant:** 3,000,000 lb
- **Fuel:** 2,000,000 lb
- **Core:** 1,000,000 lb

**DISCHARGE 2:**

- **Grain:** 400,000 lb
- **Plant:** 3,000,000 lb
- **Fuel:** 2,000,000 lb
- **Core:** 1,000,000 lb
READ IN CROSS-SECTIONS FROM COMBINE ISOTOPES FILE

READ IN REACTOR DATA (BUCKLING, POWER, VOLUME). FLAGS FOR IDENTIFYING ISOTOPES NOT TO BE DEPLETED, AND CROSS-SECTIONS FOR NON-COMBINE ISOTOPES

READ IN NUMBER DENSITIES OF ALL ISOTOPES. CALCULATE TOTAL MACROSCOPIC CROSS-SECTIONS FOR THE SYSTEM, AND DETERMINE THE THERMAL TO FAST FLUX RATIO

COLLAPSE THE TWO GROUP CROSS-SECTIONS TO ONE GROUP AND FIND THE ONE GROUP FLUX

READ IN TIMESTEP SIZE FOR DEPLETION. FLAGS FOR IDENTIFYING THE FISSION PRODUCTS, ISOTOPES DECAY CONSTANTS, AND LOGIC FOR THE DEPLETION ROUTINE FOR DETERMINING ISOTOPIC CHANGES BY DECAY, CAPTURE, AND FISSION PROCESSES

PRODUCE NEW NUMBER DENSITIES FOR THE TIMESTEP BY PERFORMING THE ISOTOPIC BURNUP OR DEPLETION ANALYSIS FOR THE SYSTEM

END OF FUEL LIFE

NO

WRITE A NEW COMBINE INPUT FILE WITH THE NEW NUMBER DENSITIES AND RUN COMBINE

YES

WRITE END OF LIFE NUMBER DENSITIES TO A FILE FOR THE GIVEN FUEL LIFE

REMOVE FISSION PRODUCT ISOTOPES AND MAKE GUESS FOR NEW ENRICHMENT FOR THE NEXT FUEL LIFE

ITERATE ON ENRICHMENT TO FIND DESIRED K EFFECTIVE FOR THE NEXT FUEL LIFE AND SAVE NEW NUMBER DENSITIES OF THE ISOTOPES

WRITE A NEW COMBINE INPUT FILE. RUN COMBINE. AND THEN RETURN TO THE FIRST BOX OF THE FLOW CHART FOR THE FIRST TIMESTEP OF THE NEW FUEL LIFE
Figure 4
Non-U Actinide Isotopes in PWR, MHTGR, and LMR (Recycle)

Figure 5
Non-U,Pu Actinide Isotopes in PWR, MHTGR, and LMR (Recycle)
Figure 6
Np-237 in PWR, MHTGR, and LMR (Recycle)

Figure 7
Americium isotopes in PWR, MHTGR, and LMR (Recycle)

Figure 8
Curium isotopes in PWR, MHTGR, and LMR (Recycle)