

## **Burn-Up Profiles for a New Beryllium Moderated Water Cooled Natural Uranium Reactor**

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### **Abstract**

A recently conceived nuclear reactor design is evaluated here for preliminary theoretical burn-up characteristics which is intended to meet Global Nuclear Energy Partnership (GNEP) goals. This reactor uses Natural Uranium (NU) metal as a single material fuel with beryllium moderation. The reactor also uses light water as a coolant. At constant power, the core reactivity initially decreases followed by a long term increase within 1 month. The reactivity increase eventually hits a maximum followed by a decline ending around a burn-up of over 30 GW-d when all the fuel is analyzed as a single material. By considering a strategic power history profile, this property can be exploited for advantage to allow power increases to eventually fall in the range of 100 to 500 MW for a once through cycle. Alternatively, by using a small fraction of the initial loading as either a slightly enriched mix of uranium or reprocessed spent fuel from this reactor, overall higher operating powers can be realized. By running the reactor for any moderate length of time sufficient to create appreciable amounts of plutonium, the hyper breeding nature of the reactor converts a large fraction of the isotope Pu-239 into Pu-240 supporting a proliferation resistant nature of the design. By reprocessing the spent fuel without ever separating the plutonium from the uranium, GNEP goals can be supported by mixing the spent fuel with natural or depleted uranium. In this way, the reactor might be used in normal operations for transmutation of the long lived actinides.

### **1. Introduction**

A large number of nuclear reactor designs have been engineered with many realizing actual power production (Lamarsh 1983). Typical of these latter designs, isotopic enrichment is used to create systems with a  $k_{eff} > 1$ . An example of this is that whenever Be has been used as a moderator for a uranium system, the fuel has always been enriched uranium. The two assemblies constructed in 1951 and 1954 (Zimmerman 1958) both utilized  $> 93\%$  enrichment as did the 1963 assemblies described by Mihalczko and Bentzinger (1997). Other examples include the BeO moderated reactor (AEC 1955) and the BR2 reactor (Posnard 2000) among others. The lowest enrichment (3.8%) found by the author from the literature was a critical experiment in 1963 by Stepnoi and Lomakin (1965). In fact, when defining safely subcritical systems of uranium moderated with beryllium for transportation purposes, only pure U-235 is considered with no limits specified for Natural Uranium (NU) as described by Parks et al. (1998).

It has previously been concluded by Wetzel (1996) that, "A mixture of U and Be has a larger minimum critical mass than a mixture of U and H<sub>2</sub>O. A mixture of U, Be, and H<sub>2</sub>O has a smaller minimum critical mass than either U and Be or U and H<sub>2</sub>O." A continuation of this work with a discussion of effects of system size and fissile concentration is given by Green (1997). Miller and Busch (1999) calculated that the minimum enrichments for homogenous systems was about 1.05% for water and 0.95% for Be. For heterogenous slab configurations the minimum calculated enrichments were about 0.7% for water and 0.47% for Be. The reactor design in this paper primarily evaluates commercial utility considerations for NU as fuel.

The previous work done on this reactor concept (Hayes 2006) began with noting that for infinite repeating slab geometries, Be and NU had a maximum reactivity ( $k_{inf}=1.08$ ) at about 16 cm Be and 0.6 cm NU. Similarly, light water layers with NU had regions of maximum reactivity ( $k_{inf}=0.97$ ) at 1 cm NU and 1.6 cm H<sub>2</sub>O. By first taking an infinite repeating Be-NU combination that was critical and then taking a layer thickness of light water that had maximum reactivity for

that same NU thickness, the repeating layers of Be-NU were broken up into infinite repeating layers of Be-NU-H<sub>2</sub>O-NU-Be such that the water gave sufficient neutronic isolation of the two NU layers to attain criticality (without the water, the NU layer became too thick and drove the system subcritical). So although the light water provides some moderation capability, its presence substantially reduced the system reactivity relative to the Be and NU layers alone. As such, the water is considered the coolant although it does offer some moderation to the core.

The initial work on this system (Hayes 2006) also showed that it should be capable of breeding at a rate high enough to increase its overall reactivity with time but that the initial start up power would probably be limited by fission product poisoning. The present work looks in more detail at some options for starting at a low power and increasing power over time so as to take advantage of the overall reactivity increase due to breeding. Alternatively, consideration is also given to replacing a couple of the initially loaded NU plates with Low Enriched Uranium (LEU), this would give the reactor enough initial reactivity to start off at a useful reactor power so as to enable the reactor to stay critical until the inherent breeding increased reactivity sufficient to start reloading/refueling the natural uranium plates with either depleted, natural uranium or reprocessed spent fuel plates for reactivity control and subsequent burn up. This work was done assuming all fuel plates were a single material throughout the burn-up process.

## 2. Experimental

The actual reactor modeled was that described by Hayes (2006) with some slight dimensional changes. The uranium layers were 1.2 cm, the beryllium layers were 16 cm and the water layers were 2.895 cm thick. The reactor size was just under 5.4 m on its outside edge and contained 48 plates of NU, 25 plates of Be with 24 light water channels. The uranium was modeled at a density of 19.05 g/cm<sup>3</sup> with the beryllium being modeled at 1.85 g/cm<sup>3</sup>. All plates were modeled as a single material at each successive time step in the burn up model with only the LEU and DU plates being modeled as separate materials. The entire reactor core was surrounded by a natural iron layer 5 cm thick at 7.8 g/cm<sup>3</sup> simulating the reactor vessel. The material properties utilized came from Weast (1987). These included the water layer which was modeled at a density of 0.6908 g/cm<sup>3</sup> (to model nominal reactor core water temperatures of around 300 °C). The thermal scattering law corrections in the S( $\alpha,\beta$ ) tables were used for both the water and beryllium. The evaluated nuclear data files (ENDF) 6 files were used for all isotopes (unless otherwise stated) except for the iron reactor vessel which used natural iron ENDF5 values.

The software utilized for the Monte Carlo calculations was MCNPX version 26b.c1 (Hendricks et al 2007) and utilized only the precompiled DOS version of the application. The computer used was a Dell Dimension 3000. When only constant power with NU or depleted uranium (DU) was run, the code options implemented were, 7770 source points per cycle, 20 pre-cycle full iterations to converge the source distribution and 100 final iterations to estimate the systems actual reactivity and burnup at each time interval. For all other runs, 10,000 source points per cycle, 40 pre-cycle convergence iterations and 200 final iterations were used. The initial starting source distribution was fixed from the command line to be that of the reactor prior to any burnup (initial criticality condition).

The method used by MCNPX in power burn up calculations is to follow the time dependent production and burn up of 3456 isotopes created as both fission and activation products. This is accomplished by coupling the CINDER90 code (Wilson et al. 1995) to MCNPX at the end of a reactivity calculation. In conducting a reactivity calculation, MCNPX also has calculated the neutron source distribution. By then specifying a reactor power, MCNPX is able to calculate both activation and fission product generation rates. Using these values along with the decay rates for generated isotopes, MCNPX is able to track the production and decay rates of all 3456 isotopes for each specified time step. This capability had previously only been available as an independent end processing code known as Monteburns (Trellue and Poston, 1999) but now is incorporated into the MCNPX code itself.

Although the MCNPX software presently assumes that changes in the neutron flux are linear between each specified time step, it uses a predictor corrector algorithm to correct for this discrepancy. This is done by calculating the neutron distribution midway between the two sequential time steps and then using that flux for the full time step (Fensin and Anghaie 2006). Also, all isotopes contributing an atom fraction less than  $1 \times 10^{-10}$  are ignored in subsequent transport calculations.

In general, MCNP has been shown to be accurate to 5 decimal places for over 70 simple systems Sood et al. (2003). The results in this paper are only presented out to 4 decimal places.

### 3. Results

#### 3.1 Constant Power With NU

The core reactivity was found to be both dependent on time and on the power level of the reactor itself. This reactivity dependence on time at multiple power levels is shown in Figure 1. Note that all values are shown in keff and have a calculated statistical uncertainty (one standard deviation) ranging between 0.0004 and 0.0006. Here it can be seen that at 1 MW and below, there is no substantial change in reactivity even after running the reactor for more than 1000 days (a peak is assumed to occur at longer times). At around 400 days of irradiation, the 1 MW system does begin increasing in reactivity. At and above 5 MW power, the system reactivity begins dropping within the first day of irradiation. At 5 MW, this initial drop in reactivity does not go below a keff of 1.003. After approximately 30 days of irradiation, the 5 MW system begins increasing in reactivity reaching a maximum beyond 1000 days of exposure. At 10 MW power, the system reactivity remains statistically indistinguishable at the 95% confidence level from unity from approximately 2 days out to 30 days of irradiation. After this, the 10 MW system increases to a maximum reactivity out beyond 1000 days of exposure.

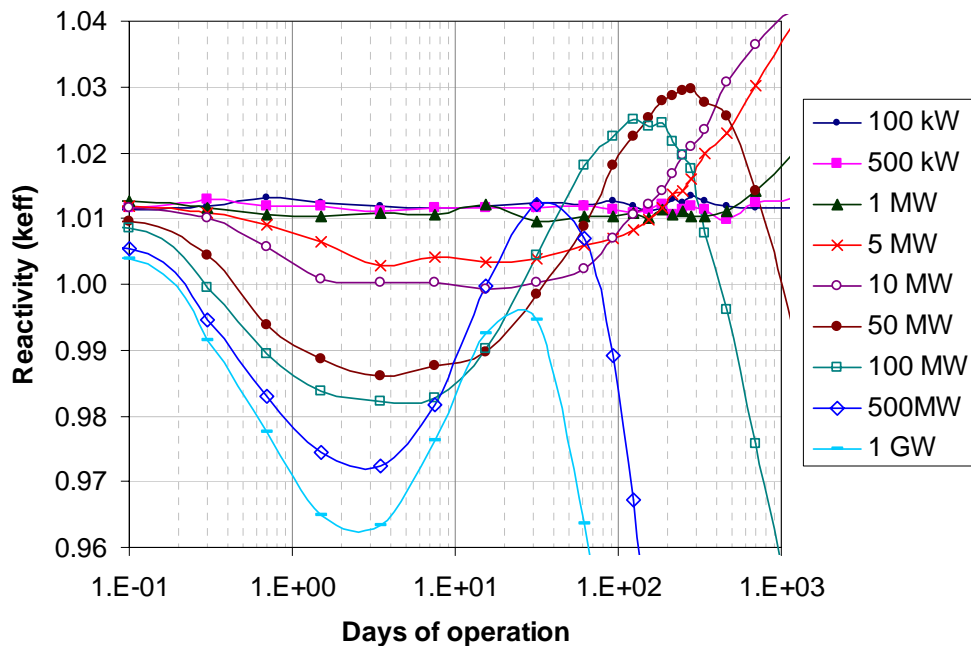


Figure 1. Reactivity profiles for the NU fuel as a function of time at the displayed constant power. All profiles were generated from the same reactor configuration with the only difference being the constant power level being utilized. No control rods were modeled and so are assumed negligible in generating these profiles. All plates were modeled as a single material so that region specific burnup was smeared over the entire reactor.

At those powers evaluated above 10 MW ( $\exists$ 50 MW), the system becomes subcritical within the first day of operation. If the system were to remain in operation at the same power level, one or more plates of the initial 48 plates of NU would need to be replaced by LEU at the initial loading or the reactor would shut itself down due to fission product poisoning. The minimum reactivity for the systems above 10 MW tends to occur between 3 and 7 days. At 1 GW, the system never becomes critical again ( $k_{eff} \exists$ 1). The subsequent maximum reactivity of the other systems appears to increase with decreasing reactor power. Likewise the time at which maximum reactivity takes place appears to also increase with decreasing reactor power. Similarly, the segments of time over which the systems stay critical after their initial low reactivity intervals appear to increase with decreasing reactor power. Qualitatively, the trends seem to be consistent with an overall functional dependence of reactivity to MW-d of burn up.

### 3.2 Variable Power with NU

Table 1 shows one possible set of results from incrementally increasing reactor power after the initial reactivity drop is passed. Specifically, by starting at 8 MW power for a period of about 3 weeks, the reactor power is increased according to the pattern given in Table 1 with the associated overall reactivities being presented concomitantly. The results in Table 1 indicate that the reactor could be run for 80 MW for approximately 150 days followed by 150 days at 200 MW.

Table 1. Demonstration of incrementally increasing power with burn-up.

Total days	MW	$k_{eff}$
0.1	8	1.0123
0.2	8	1.0115
0.4	8	1.0098
0.7	8	1.0063
1	8	1.0048
2	8	1.0019
4	8	1.0011
7	8	1.0011
10	8	1.0008
15	8	1.0014
20	8	1.0011
30	8.8	0.9998
40	9.6	1.0010
50	10	1.0013
65	15	0.9991
80	40	0.9975
100	80	1.0048
150	80	1.0178
250	200	1.0187
500	256	0.9669

In this way, not only is the reactor power increased to values large enough for commercial interest (in the range of 100's of MW) but the reactor would not be able to attain prompt critical providing the steady increases of power are continually instituted. In essence, reactivity control could be accomplished through fission product generation by running the reactor at useful power levels. This feature has inherent beauty from a safety perspective in that it would be physically incapable of a prompt critical approach due solely to operating it at commercially viable power levels.

When the reactivity in Table 1 is seen to be less than unity, this means the corresponding power evaluated was too great. Likewise, if the reactivity was greater than unity, a higher power could

have been used during that time interval or fuel replacement could begin (where the new fuels initial reactivity drop compensate for the excess reactivity in the core due to breeding). The dynamic dependence of changes in power in earlier times is not evaluated for later times outside of what is shown in Table 1. The results in Table 1 are only one example of power histories and were not optimized for the starting 8 MW power regime for a core without refueling. Future evaluations might focus on a more detailed evaluation of the actual high power time interval to more accurately determine the length of time the reactor could run at greater than 100 MW and or evaluate refueling capabilities during this high reactivity interval (between 100 and 300 days post start-up).

### 3.3 Constant Power with 2 LEU Replacement Plates

By initially replacing 2 of the 48 NU plates (4.2% of the initial fuel volume) with 2 w/o LEU plates, the initial drop in reactivity below critical can be compensated for as shown in Figure 2 for a reactor operating at 100 MW. Although no U-234 was modeled in the LEU for this particular run, Figure 2 shows that less LEU could have been used as the reactivity never went below 1.03 keff during the initial reactivity drop. Alternatively, this indicates an initial higher power could have been run for this loading configuration as the added reactivity would have compensated for the larger initial reactivity drops associated with higher power as shown in Figure 1. In addition, a variable power profile with time could be implemented as shown in the previous section (Table 1). If the reactor were allowed to run at a varying maximum power level and so prevent having excess reactivity capable of a prompt critical, a lower enrichment (<2 w/o in the 2 plates) would likely be required for the configuration than that evaluated here if LEU were to be utilized.

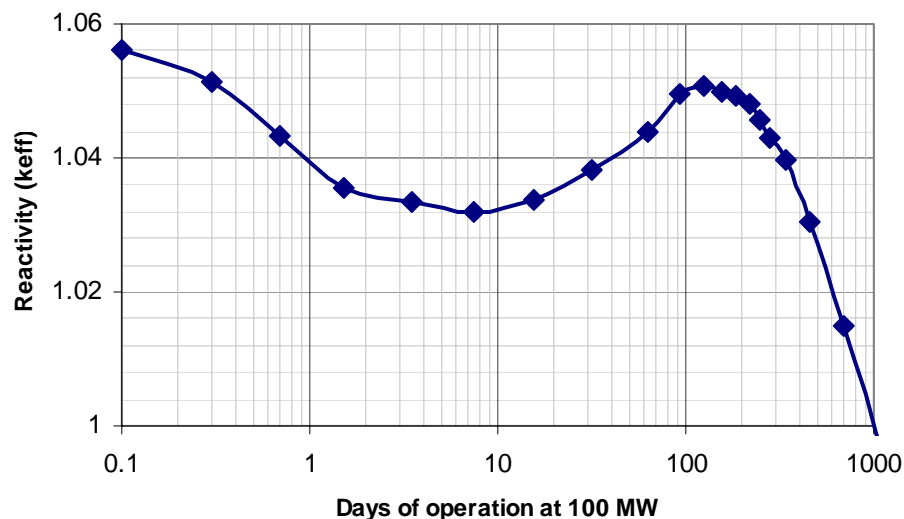


Figure 2. Reactivity profile from the NU reactor at 100 MW thermal power with 2 of the 48 plates being replaced by 2 w/o LEU. This demonstrates that a very small amount of enrichment on less than 5% of the fuel elements is sufficient to maintain constant power for many years. Again, all plates were modeled as a single material.

### 3.4 Refueling with DU

In order to understand the effects of placing unirradiated DU plates into the core, a core fully loaded with DU was modeled at 10, 50, 100 and 500 MW. The resulting profiles as a function of time are shown in Figure 3. Note that the initial minimum reactivity region would have to be compensated for by positive reactivity from other sources. This could either be done during the reactivity peaks shown in Figures 1 and 2, Table 1 or any other location in the positive reactivity lobes provided the number and location of the DU plates would not drive the core subcritical during the DU's burn-up process.

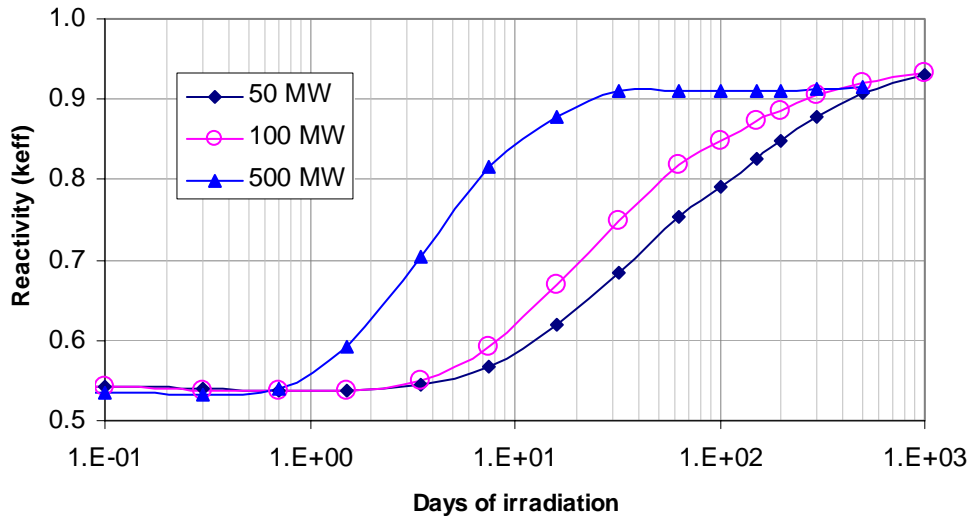


Figure 3. Reactivity profile of the reactor fully loaded with DU run at multiple power levels (with all plates modeled as a single material). Note that an external neutron source would have been required to maintain the reactor at the evaluated power levels. This figure demonstrates that if DU were used during an initial loading that it would increase in reactivity with burnup and so could be used to replace or augment burnable poisons.

Alternatively, a first approximation on the initial reactivity worth is provided in Figure 4 which shows initial core reactivity as a function of the number of initial NU fuel plates being replaced with DU. Here, only initial core reactivity is shown, it can be assumed to a first approximation that a mixture of DU and NU would be a weighted superposition of the effects seen in Figure 3 with those of Figure 1 or Table 1 etc. The placement of the DU plates in Figure 4 tried to approximate a centrally based distribution throughout the core.

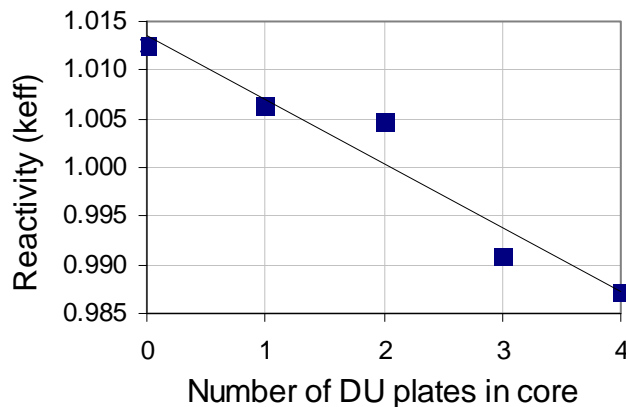


Figure 4. Initial reactivity of the NU reactor as a function of the number of initial DU plates in the core. The displayed fit is represented by  $y = -0.0066x + 1.0136$ .

### 3.5 Nonproliferation

In order to assess how robust these reactor configurations would be in the efforts to enhance nonproliferation goals of nuclear professionals and politicians, the resultant actinide inventory should be considered. Shown in Table 2 are those values which are obtained at the maximum reactivity values of 50 MW, 100 MW and 500 MW. These correspond to time since startup values of 280 days, 125 days and 31 days respectively. These values are chosen as they are reasonable dates for reloading to occur so that the month or so dwell time required for new fuel to start becoming more reactive would easily be compensated for with the rest of the core being at an otherwise elevated reactivity configuration during the refueling. Note that for the initial NU

loading, there were 16.48 kg U-234,  $2.167 \times 10^3$  kg U-235 and  $302.6 \times 10^3$  kg U-238. As a unit of measure for nonproliferation, the ratio of Pu-240 to Pu-239 is consistently 14% or greater. This would mean that if the irradiated fuel were removed from the core at this time for reloading, the plutonium would not be weapons grade (the latter being a ratio of 10% or less).

Table 2. Actinide inventory at the maximum reactivity values in the burn up timeline of NU. All values are in kg. Some values not reported (NR) due to low atom fraction limit.

Actinide	50 MW	100 MW	500 MW
Th-230	3.17E-05	NR	NR
U-234	1.47E+01	1.49E+01	1.45E+01
U-235	1.61E+03	1.66E+03	1.53E+03
U-236	9.01E+01	8.28E+01	1.03E+02
U-237	1.70E-01	3.27E-01	1.55E+00
U-238	3.02E+05	3.02E+05	3.02E+05
U-239	5.98E-02	1.22E-01	6.21E-01
Np-237	3.58E+00	3.00E+00	2.89E+00
Np-238	5.51E-03	9.12E-03	3.70E-02
Np-239	8.64E+00	1.75E+01	8.88E+01
Pu-238	2.02E-01	1.43E-01	1.28E-01
Pu-239	4.86E+02	4.49E+02	4.94E+02
Pu-240	7.26E+01	6.09E+01	8.35E+01
Pu-241	1.35E+01	1.07E+01	1.64E+01
Pu-242	8.72E-01	6.10E-01	1.14E+00

The last reasonable date in which the fuel would be replaced (and the one having the highest assumed fuel economy value) would be just before the core would otherwise go subcritical. The actinide contents in the core for 50, 100 and 500 MW are given in Table 3 for times 700, 340 and 63 days respectively. Besides noting that the burnup in each of these cases is similar (near 33 GW-days) the isotopic mixes are likewise similar with a fairly constant Pu-240 to Pu-239 ratio of 30%. At this point, for each of the powers listed in Table 2, the heavy metal content was only decreased by less than 0.3%.

If the fuel were reprocessed having only the fission products removed, this would serve as a very attractive source of fuel for this same reactor design. By again modeling only the heavy metal fraction given in Table 2 for the 100 MW and 500 MW columns as the starting isotopic mix, the initial starting reactivity of the system would be  $k_{eff}=1.0813 \pm 0.0006$  and  $k_{eff}=1.0909 \pm 0.0005$  respectively (higher than any other reactivities evaluated). This assumed the same reactor dimensions stated above. As done previously, only the Np-239 and U-239 were modeled as having decayed at all and then only to Pu-239 with the Np-238 not being modeled due to lack of cross section data. Given the high reactivity seen for utilizing reprocessed fuel from this reactor back into the same reactor, this is an attractive option. This would even allow blending the reprocessed fuel with depleted uranium either at the initial reprocessing stage or a later stage if the cycle were to be continued so as to limit the maximum reactivity present in the core. In this way, not only could NU and DU be used as fuel but transmutation of the actinide activation products could also be accomplished as well in furtherance of GNEP goals.

Looking only at the irradiated DU, the plutonium ratio is also demonstrated to be aligned with GNEP goals of a fuel cycle not requiring plutonium separation as shown in Table 4. Here, the isotopic ratios of plutonium are given for the various powers shown in Figure 3 at 100, 500 and 1000 days of constant power irradiation. Except for low power operation for short amounts of time, the generated Pu is not weapons grade material. Furthermore, if the irradiated DU were reprocessed in the same manner as the NU (such that the uranium was not removed from the Pu) leaving all the actinides still in the fuel, the reactivity of the resultant fuel is also given in table

4. These latter values demonstrate the utility of reuse of spent fuel allowing a closed fuel cycle to be considered rather than a once through cycle.

Table 3. Actinide inventory after approximately 33 GW-d burn-up at multiple constant power levels in NU. All values are in kg. Some values not reported (NR) due to low atom fraction limit.

Actinide	50 MW	100 MW	500 MW
Th-230	6.87E-05	3.36E-05	NR
U-233	4.78E-05	4.70E-05	4.50E-05
U-234	1.26E+01	1.27E+01	1.29E+01
U-235	1.10E+03	1.12E+03	1.15E+03
U-236	1.72E+02	1.68E+02	1.63E+02
U-237	2.30E-01	4.56E-01	2.07E+00
U-238	3.01E+05	3.01E+05	3.01E+05
U-239	5.83E-02	1.16E-01	5.94E-01
Np-237	1.05E+01	1.04E+01	7.88E+00
Np-238	1.56E-02	3.01E-02	1.04E-01
Np-239	8.41E+00	1.67E+01	8.50E+01
Pu-238	1.56E+00	1.29E+00	7.20E-01
Pu-239	7.72E+02	7.63E+02	7.32E+02
Pu-240	2.42E+02	2.36E+02	2.16E+02
Pu-241	6.33E+01	6.11E+01	5.58E+01
Pu-242	1.03E+01	9.57E+00	7.72E+00

Table 4. The isotopic ratios of Pu-240 to Pu-239 for the various powers and times shown in Figure 3 from spent DU fuel along with keff values from the same DU if reprocessed without any subsequent mixing. Values do not include any decay.

Days post irradiation	50 MW		100 MW		500 MW	
	Pu-240 / Pu-239	reprocessed fuel keff	Pu-240 / Pu-239	reprocessed fuel keff	Pu-240 / Pu-239	reprocessed fuel keff
100	0.13	0.8252	0.20	0.8948	0.59	1.0026
500	0.37	0.9530	0.56	0.9893	1.01	1.0342
1000	0.56	0.9863	0.79	1.0069	1.07	1.0371

#### 4. Discussion

Various options of using incremental increases in power with time or initial loading of a small number of plates with LEU were calculated to have potential viability for commercial power production. The proposed approach would largely rely upon fission product poisoning for reactivity control rather than moderator poisons or control rods. Still, there remain details prerequisite for commercial viability which have not been addressed. These include the thermal expansion in the Be and NU plates which would tend to squeeze out some of the water in the channels which would change the reactivity of the system as a function of both power and burn up and so would have a dynamic component. Likewise, the actual amount of mixing spent fuel actinides with either DU or NU should be looked at over multiple cycles through a reactor to insure thorough transmutation and burnup will readily occur to eliminate the need to utilize long term geological disposal of wastes. In addition, use of improved cross sections should be used for future work (such as endf66) to the extent that these are available. Perhaps even more importantly, the more computer resource intensive analysis of modeling each plate as a separate

material throughout the burn up (or even multiple regional materials in each) plate should also be undertaken when that becomes possible\*.

Another detail which should be looked at further is the lack of any cladding being modeled between the NU and light water. Although both elemental Be and NU are listed as being insoluble in water (Weast 1987), fuel cladding may be desired if the added load on fission products in the primary coolant loop cannot readily be removed as suggested previously (Hayes 2006). With less than 0.3% of the heavy metal being lost to fission products, the amount of transfer to the primary coolant is open to question. As the fuel has presently only been modeled being quite intimate with the coolant, the increased heat transfer would keep the fuel at temperatures comparable to the coolant and so would provide a relatively high heat flow from the fuel which would tend to keep the fuel from undergoing any deformation or other high temperature effects.

If Be were used as a clad, this would be expected to have negligible if not preferential effects on the neutronics, although most other clad materials would be expected to reduce system reactivity. Still, no systematic attempt to optimize the combination H<sub>2</sub>O, Nu, Be layers has been done and so further increases in reactivity might be realized with additional efforts.

The results in Figure 1 show that operation at low powers tends to give rise to higher long term overall reactivity peaks. Similarly, higher powers tend to have lower initial reactivity minima. When the power levels become high enough, strategic staggered partial refueling with either DU, NU or reprocessed fuel could be undertaken for reactivity control under an approved fuel management plan. The trends in Figure 1 might also be expected to demonstrate that maximum reactivity for each power would be at the same burn-up levels. The isotopic ratios from Table 3 however suggest that lower thermal powers produce lower Pu-240/Pu-239 ratios although long irradiation times would have to be evaluated to validate this conjecture.

Although only 2 plates of LEU (at 2 w/o) were evaluated for burn-up evaluation at only 100 MW. Other plate distributions at lower enrichment or higher powers could have been used including the use of reprocessed fuel. Other characteristics for optimization consideration are reactor shape differences and overall size. This could include modeling the plate fueled reactor in a shape closer to that of a right circular cylinder or possibly even variations in plate thickness of the fuel, moderator or coolant over one or more channels or other variants.

The process by which refueling would occur has not been evaluated (whether a full core reload or multiple partial core reloads). Another option would be to consider individual pressure lines for each coolant channel. In this way, real-time on-line refueling could be accomplished as presently done for the CANDU reactor designs (Lamarsh 1983). This option further opens the possibilities of using some LEU in the core with DU such that the increase in reactivity from DU will occur when the decrease in reactivity from the LEU or the rest of the core is taking place to approach continuous steady state power operation.

DU itself could not be used as the sole fuel but could be folded into a comprehensive fuel management plan as its reactivity will increase with time and so compensate for reactivity decreases from high burn-up in NU. Similarly, if heavy water were used as the coolant, it is possible that further analysis would demonstrate that DU could be optimized as a primary fuel material and warrants additional analysis.

Use of spent fuel for the primary source of fuel is a very promising property of this reactor provided the plutonium is kept with the uranium in the final uranium plates made from the reprocessed fuel. Use of reprocessed DU could also be used as a primary fuel if it had been

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\* At present this has been done and the work submitted for publication. The results of this analysis show improved performance of the reactor in terms of longer burn up and higher available thermal power (>1000 GW-days) operation.

irradiated at sufficiently high powers and times, otherwise it would still serve reactivity control purposes being that it would start out with negative reactivity and increase its reactivity with burn up.

## 5. Conclusion

Based on the calculations presented, the beryllium moderated reactor has alluring properties regarding possible burn up profiles that might be used in support of GNEP goals. The spent fuel at 30 GW-d burn-up would not be attractive for illicit weapons programs but if reprocessed could be reused in the reactor after the fission products had been removed. Present calculations indicate that burn ups in excess of 30 GW-d are possible for NU, and that DU could be used in place of certain fuel elements for reactivity control. Higher burn-up would be possible with the use of LEU or reprocessed fuel. Use of reprocessed fuel where the plutonium had not been separated from the uranium is seen to be a viable long term fuel source. Reactivity control appears to be possible due primarily to creation of fission products in the fuel limiting maximum reactivity and power. Unresolved questions include reactivity optimization, material characteristics with increasing irradiation, fuel cladding and the thermal hydraulics. Further analysis should include analyzing individual plates as separate materials for more detailed burn-up predictions.

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