Transactions, SMiRT-25
Charlotte, NC, USA, August 4-9, 2019
Division X

A FUTURE GENERATION THERMAL BREEDER USING NATURAL URANIUM FUEL AND BE MODERATION

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ABSTRACT

This research explored a novel reactor design capable of hyper-breeding, potentially greater than any design previously evaluated (fast or thermal). Reactor breeding estimates at power were predicted to exceed 1.55 (Hayes 2006) using only natural uranium making it of substantial interest for obtaining a high burnup fuel cycle. It has also been shown capable of starting with only natural uranium fuel making it highly attractive from a nonproliferation perspective. The breeding capability also allows for both partial refueling with depleted uranium with a new core burnup capability > 3 GW yr (Hayes 2007, 2008). Previous scoping calculations demonstrated great promise having only evaluated metallic, unclad slab fuel geometries without considerations for material performance after irradiation nor the thermal hydraulics of such a system. This work reviews the technical basis for such a neutron economy where reactor physics considerations demonstrate how such a heterogenous configuration enables apparently drastic changes in reactor performance. In addition, novel non-proliferation technology is also considered which could further mitigate closed fuel cycle models for this reactor concept. Construction, design, capital and mechanics of resultant system configurations still need to be evaluated.

INTRODUCTION

This work details the theoretical viability of a thermal breeder capable of activating into a prompt critical state using natural uranium (NU) as a fuel, light water as a coolant and Be as a moderator. Historical uranium breeders have required either HEU (Clayton 1993) or ^{233}U (Atherton 1987) central seed cores to achieve breeding. The proposed design for natural uranium (NU) breeding utilizes Be moderation such that the ^{9}Be(n,2n)2α reaction provides an additional boost of neutrons to the reactor per thermal fission (effectively the thermal fission factor). This is then coupled with an optimization of the resonance escape probability and fast fission factor for this design allows such a conservative neutron economy. Note the material properties of Be and NU during irradiation are not considered here but clearly warrant future attention.

The term "hyper-breeder" being coined here refers to a design which can breed at such a rate as to eventually become prompt critical without intervention. It is believed that the present design concept is the only currently known reactor which has this property. The hyper-breeder was initially analysed without consideration for depletion (Hayes 2006) but did show that under the proper configuration of both the Be moderator and natural uranium (NU) fuel (in slab geometry) could be placed into a prompt critical state. Addition of water for a coolant then allowed for a critical design to theoretically function as a power reactor. The results of this paper now detail how thermal breeding becomes possible for natural uranium alone based on rather unique design specifications along with associated non-proliferation coupled issues.

Issues commonly associated with power reactors are the issues of waste and nuclear proliferation. With nuclear waste, there are no technical issues unsolved but with proliferation, the controls can be more diverse and may require custom designs for each reactor concept.

Technical Basis

It was important for this design to first recognize that although homogenous mixes of natural uranium and other compounds cannot become critical, at optimum repeating slab thickness of both NU and light water, much higher reactivity values can be attained (Hayes 2006). In this way, a unique heterogeneous configuration becomes substantially more reactive than any homogenous mix of water and NU. Performing a similar analysis with Be and NU showed that there were a range of repeating Be and NU slab thickness' which could become prompt critical despite all homogenous mixes of the same elements being overtly subcritical (Hayes 2006). Burnup calculations then allowed additional reactor dynamics to be considered (Hayes 2007, 2008).

The additional optimization of the fast fission factor ε, and resonance escape probability p used in this design help to increase reactivity values sufficient for the hyper-breeding to occur. Although the Be fission component from n(Be,2α)2n could be looked at as an additional factor in the $\varepsilon p\eta f$ formula, it does effectively increase the number of neutrons produced per neutron absorbed in the fuel which is the terminology ascribed to the thermal fission factor η. This contribution could be a separate factor to combine with the $\varepsilon p\eta f$ formula but for simplicity is folded into η although this is clearly not required. So one can assume η is unchanged and have a separate Be contribution or just fold these into an increased value of η.

Reactor Configuration and Calculations

The optimum range of NU thickness for both the infinite slab configurations with H_2O and Be had some overlap, so by separating each of these NU slabs with the optimal H_2O and then Be in a repeating pattern resulted in a LWR type design as shown in Figure 1. When moderating with water, as shown in Figure 1, the loss of water due to boiling from an off-nominal operating condition results in a higher than optimal fuel/moderator ratio with the Be, indicating the coolant is required at startup for criticality and would provide a negative void coefficient during operation. Following the initial realization of the capability for a reactor, as surmised in Figure 1, depletion calculations were performed to determine reactivity as a function of burnup using homogenous material distributions at each burnup step (Hayes 2007).

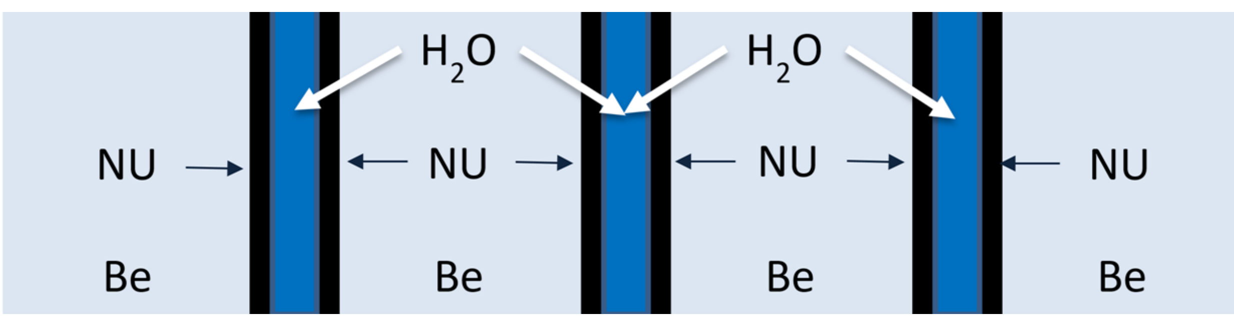

Figure 1. Beryllium moderated, light water cooled, hyper breeding natural uranium reactor concept schematic. Layer thickness of slabs shown are not proportional but are considered representative.

Although prior research only included simple preliminary scoping calculations for this design, it was found that not only did the core reactivity increase with time during the first 100 GWd but that this could be controlled through replacing NU fuel with depleted uranium (Hayes 2008) in an iterative fashion (although not for every fuel element). Furthermore, it was found for a reactor having a volume of 1000 m^3, the total burnup potential from an initial NU core loading prior to any refueling was only 3.4 GWd/TU (from running at a full GWth for almost 40 months). As suggested by Figure 1, the fractional content of the reactor by volume of NU is less than ~7%, demonstrating another very unique aspect of this reactor (high thermal output with a low fuel loading). Where a typical LWR core contains almost 3E6 kg of enriched uranium, this hyper-breeder design only requires closer to 0.3E6 kg NU despite having a volume of 1 dm^3. The ability to generate the same GW levels of heat is therefore also a novel facet of interest with a hyper-breedder design. Whether this will allow multiple niche applications and/or full power generation with load following abilities is of interest but not explored further in this work.

All calculations were done with MCNPX. Example input stack portions are provided in the appendix for reference. Burnup, effective η, isotopics and eigenvalue results were all generated by the MCNP® code.

Previous Burnup Calculations

The initial burnup calculations for this design (Hayes 2007) did not consider the heterogeneity in the fuel plates due to the energy and intensity dependence of the flux on spatial distribution (which approximates a cosine through the symmetry axis). As such, each plate was initially homogenized after each burn step. In order to account for the spatial dependence on fission and activation, the plates were later divided into only 2 regions to allow the central portions to be evaluated separate from the edges which resulted in substantially more detailed isotopic analysis as a function of burnup (Hayes 2008).

RESULTS

As a simplification approach, the components of the effective multiplication eigenvalue will be evaluated via the four factor formula.

Neutron Economy

With the fuel being large plates as shown in Figure 1, the fast fission neutrons which travel perpendicular to their slab of origin (normal to the face) will next traverse H_2O or Be and so contribute to the thermal population. Those neutrons which travel parallel to the plate will traverse a potentially long path of pure NU and so have a much higher probability of fast fission in the ^{235}U than that seen in a standard LWR's. For fast fission in an LWR, the fission neutrons basically have to traverse a pin axis to attain a high probability of fast fission resulting in a much larger solid subtended angle in this reactor for fast fission than that of an LWR. Basically concentrating more fuel together will increase fast fission as a general principle and so maximizing the fast fission factor ε as done here.

The value of η is energy dependent so the temperature of the reactor and thermal average energy of fission fix its value for a Maxwellian distribution. As such, the resultant distribution from the mixed H_2O and Be reflectors straddling each NU plate do provide a slight increase in the calculated η found from this reactor design but the biggest increase would come in forcing it to have a definition of the number of neutrons produced per neutron absorbed in the fuel. With each neutron absorbed in the fuel, a certain fraction of those produced from fission can multiply in the moderator via the n(Be,2α)2n reaction and so cause an effective increase in the thermal fission factor for this reactor design.

The very large Be slab results in a large volume of pure moderator in which the neutrons can reside and thermalize so as to increase the likelihood that they will not be in the ^{238}U resonances when they diffuse back into the fuel. In this way, the resonance escape probability is again increased over and against that in a typical LWR as previously shown by Hayes (2006). In this way, the first 3 of the 4 terms in the $\varepsilon p\eta f$ expression are qualitatively increased over and above that attainable by modern LWR's.

The thermal utilization described by Equation 1 could also be evaluated for this design. Rather than evaluate this along with the fast and thermal escape probabilities, it is more straightforward to simply calculate k_{eff} using high quality reactor codes such as MCNPX.

$$f = \frac{\Sigma_a^F}{\Sigma_a^F + \Sigma_a^{NF}(V^{NF}/V^F)(\phi^{NF}/\phi^F)} = \frac{\sigma_a^F}{\sigma_a^F + \sigma_a^{NF}(N^{NF}/N^F)(V^{NF}/V^F)(\phi^{NF}/\phi^F)} \quad (1)$$

MCNP Reactor Predictions

The starting value and growth of the effective ν (from $\eta = \nu \frac{\Sigma_f^F}{\Sigma_a^F}$) is shown in Figure 2 for this reactor design. Here the monotonic growth starting from an initial value of 2.468 is demonstrated with an expectation to continue increasing towards 2.8 with only the first 100 days shown. This shows how this reactor has both a larger starting value than that expected in an LWR (2.42) but characterizes the kind of growth expected with burnup as a property of the design.

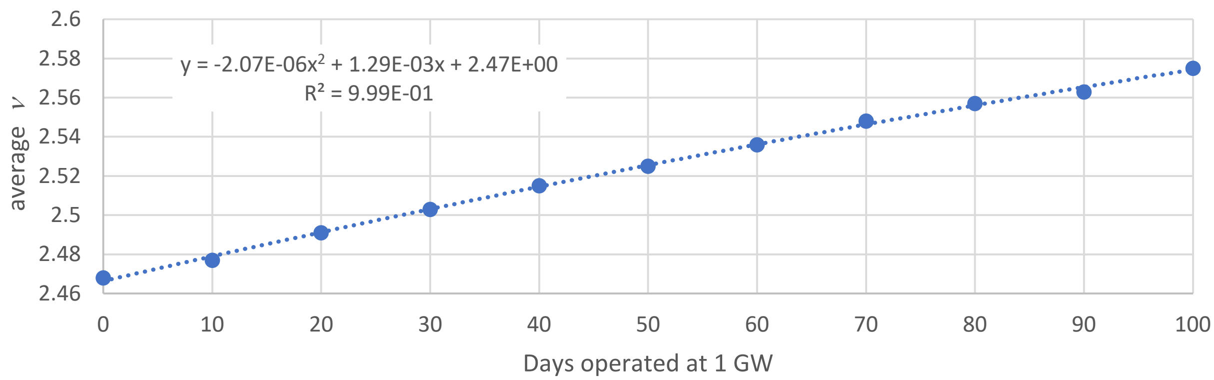

Figure 2. Average MCNPX calculated values of ν based on continual operation at 1 GWth.

Although the neutron economy has previously been shown sufficient to allow thermal breeding (Hayes 2008), an additional metric to assess this ability is the generation of the various plutonium isotopes. These values are shown in Figure 3 over the first 100 days. The initial value of (^{239}Pu+^{241}Pu)/(total Pu) was calculated to be 0.992 with the final value at 100 days being 0.985. Herein lies a potentially major proliferation concern in that spent fuel from such a reactor could pose such a concern comparable to those from molten salt reactors or any thorium fuel cycle. Any fuel containing weapons grade fissile nuclides would need to be subject to commensurate controls accordingly.

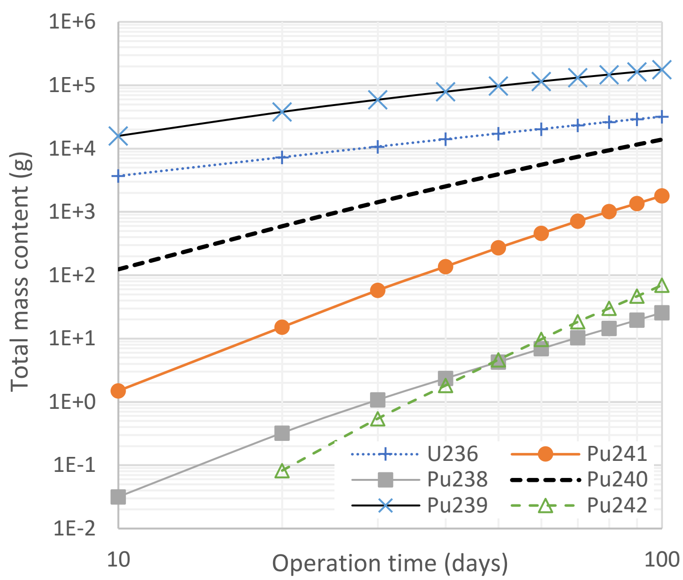

Figure 3. Activation isotopics of concern for closed fuel cycles demonstrating breeding potential.

If the PUREX process is used to recycle the fuel and mitigate proliferation concerns, the ^{236}U will fold into the ^{235}U and ^{238}U content creating elevating radiation fields from the fresh fuel bundles. This is readily addressed with engineered shielding for transportation and operation in addition to modern ALARA programs which are generally very robust. The total reactor content of the ^{236}U is shown in Figure 3 to highlight this additional consideration for utilizing such a reactor design in a closed fuel cycle.

DISCUSSION

The potential for using only natural uranium as a feedstock with partial depleted uranium in fractional portions of refuelling is very attractive for a nuclear fuel cycle. The use of a hyper–breeder to accomplish this then warrants controls on any subsequent attempts to close the fuel cycle so that plutonium extraction would not occur separate from the uranium. Similar controls are required when thorium is used in a fuel cycle as a target given that the resulting ^{233}U would then constitute a pure fissile mix with no ^{238}U or ^{240}Pu to degrade the materials attractiveness for weapons production.

Perhaps the single most important uncertainty remaining on the proposed reactor design are the material properties expected from burnup. The swelling and cracking expected from irradiated Be would occur from Helium build-up generated from the n(Be,2α)2n reaction. Initial estimates of the source terms possible are shown in Figure 4. Here, the material isotopics with irradiation are shown for the first 100 days at 1 GWth and then for a subsequent 500 days at 10 GWth. The shaded area shows the transition period where the elevated power is being used although the functional response between the lower and higher power range are not detailed and so shaded out for this reason.

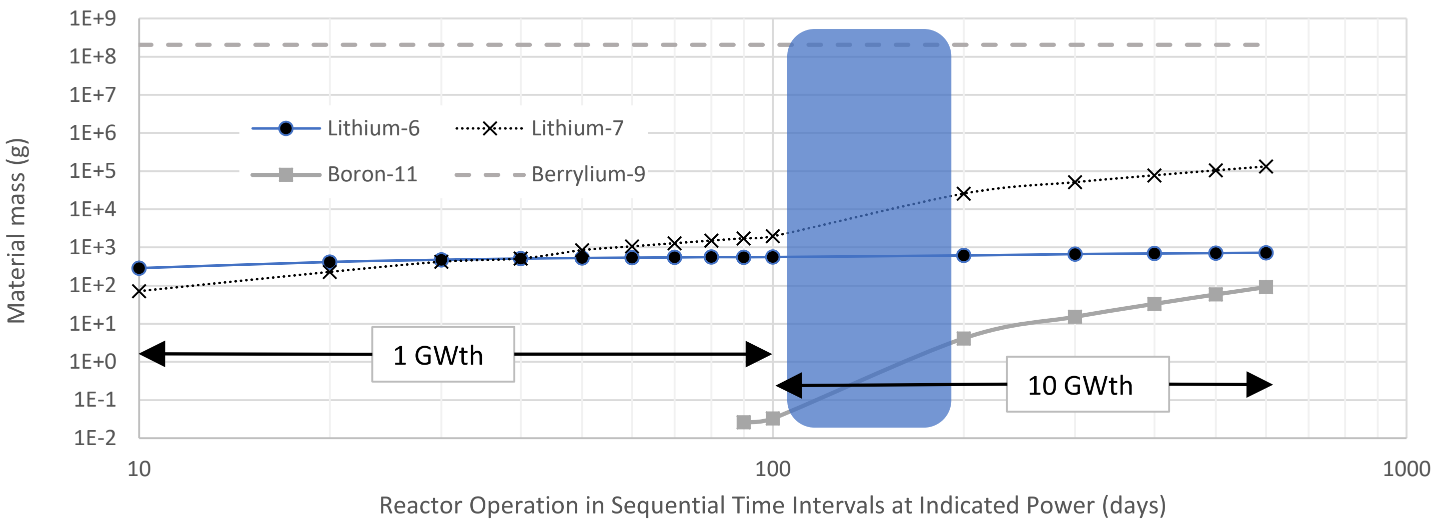

Figure 4. Moderator material predictions with burnup from mcnp simulations. The reactor was simulated at 1 GW for the first 100 days and then at 10 GW for the subsequent 500 days.

Swelling and elongation in the fuel has not been analysed nor the effects from ignoring any clad materials. The uranium fuel plates cannot contact the coolant water so the neutronic, thermal and mechanical properties of any clad have also not been evaluated.

Novel Non-proliferation Technologies

Recent research in nuclear safegaurds, forensics and non-proliferation technology has shown how thermoluminescence dosimetry, optically stimulated luminescence dosimetry and electron paramagnetic resonance can be used in novel ways (Hayes 2016). Specific examples of how this has been demonstrated to date include showing how a common red brick can function as a gamma ray spectrometer with 10% energy resolution for ^{241}Am (O'Mara and Hayes 2017). Similarly, surface mount electronics were shown to be able to measure background radiation levels sufficient to age components to within 10 years (Hayes and O'Mara 2018). Using cored building materials is then theoretically capable of demonstrating retrospective assay of UF_6 enrichment (Hayes 2019). The whole point being made here is that in addition to all the traditional detection and material balance controls presently available for regulatory compliant fissile material handling, newer cutting edge technologies can further prevent proliferant activity. As such, this technology can further enable a closed fuel cycle to be considered for this reactor concept.

CONCLUSIONS

This work has shown that a water cooled Beryllium moderated natural uranium reactor is fully capable of hyper-breeding. As such, a fuel cycle using this technology would not require enrichment and so addresses proliferation concerns on this point if a once through cycle is used. If a closed fuel cycle is desired, this reactor's spent fuel could pose proliferation concerns comparable to those inherent to thorium fuel cycles and some molten salt reactor designs. Novel approaches to nuclear forensics were considered towards addressing proliferation concerns using retrospective dosimetry methodologies.

ACKNOWLEDGEMENTS

This work was funded in part by federal grant NRC-HQ-84-14-G-0059 from the Nuclear Regulatory Commission and through a joint faculty appointment between North Carolina State University and Oak Ridge National Laboratory in coordination with the Office of Defense Nuclear Nonproliferation R&D of the National Nuclear Security Administration sponsored Consortium for Nonproliferation Enabling Capabilities under Award Number DE-NA0002576.

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