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Maximum Fuel Utilization in Fast Reactors Without Chemical Reprocessing

Integrated University Programs

Dr. Ehud Greenspan
University of California, Berkeley

Steven Piet, Technical POC
Jack Wheeler, Federal POC

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Summary Report

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Principal Investigator: Ehud Greenspan, 510-643-9983, gehud@nuc.berkeley.edu

Collaborators: Jasmina Vujic, vujic@nuc.berkeley.edu

Contributing students

1. Florent Heidet, completed his Ph.D. in September 2010; now at ANL
2. Christian Di Sanzo, Ph.D.
3. Staffan Qvist, Completed Master project and working on Ph.D.
4. Remi Cognet, Completed Master project in 2010.
5. Soto Gonzalez, Completing Master project.

Additional contributing graduate students:

Anselmo Cisneros, Guanheng (George) Zhang, Alejandra Jolodosky, Madicken Munk, Lasshana Huddar, Zachery Beauvais, Guervan Adnet, Christopher Varela, Seungmin Woo

Contributing undergraduate students:

D.J. Anderson, J.C. Curtis, A. C. Kaplan, S.A. Dionisio, A.J. Dixon, J. Doojphibulpol, N.A. Fischer, A.M. Hernandez, J.A. Miller, C.F. Montegrande, S.S. Parker, A.P. Priest, F.V. Rubio, A.P. Shivprasad, K.C. Tirohn, T.M. Wilks

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Project Abstract

The overall objective of this project was to identify fast reactor core design and fuel cycle options that could enable to achieve burnups that are significantly higher than 100 GWd/tHM without use of fuel processing technologies that can extract plutonium or can separate the actinides from most of the fission products.

Following is a brief summary of the findings (see also Quad Chart below):

- It is possible to start implementing the B&B mode of operation without exceeding the already proven cladding dpa levels (200 dpa) by driving a sub-critical B&B blanket with neutrons that leak-out from a critical driver. The blanket (and driver) fuel discharge burnup could be gradually increased as a cladding material that is certified to operate to higher dpa levels becomes available. With a 200 dpa constraint a depleted uranium B&B blanket can generate at least 50% of the total core power. Due to the much smaller cost of unirradiated depleted uranium fuel relative to sodium-fast-reactor (SFR) recycled (after reprocessing) fuel, the fuel cycle cost of such a seed-blanket SFR is expected to be significantly lower than of a conventional SFR. When cladding materials are certified to operate up to 300 dpa, the blanket of the seed-and-blanket SFR could generate ~2/3 of the total power. Hopefully, eventually cladding/fuel materials/design that could safely operate up to ~550 dpa/20% FIMA could be certified so that self-sustaining B&B reactors could be deployed.
- It is neutronically feasible to establish a self-sustaining B&B mode of operation in large fast reactor cores provided the fuel/clad will be able to safely operate up to an average burnup of at least ~20% FIMA (Fissile per Initial Metal Atom). The uranium utilization of such B&B reactors is ~40 times that of LWR.
- Fuel discharged from a B&B core at an average burnup of 20% FIMA can be used, after reconditioning, for the “starter” fuel of a new B&B reactor core, thus enabling to expand the fleet of B&B reactors without need for supply of external fissile fuel beyond that required for starting the first generation of B&B reactors. The primary functions of the fuel reconditioning are to remove the gaseous fission products (FP) and encase the fuel rod in a new cladding; no separation of actinides from solid FP is needed.
- Alternatively, after reconditioning the fuel can be recycled into the B&B reactor and operate up to a maximum possible cumulative average burnup of ~55% FIMA. The corresponding uranium utilization is ~100 times that of LWRs.
- The depleted uranium stockpiles (“waste”) that will be accumulated in the US until ~2050 will be able to generate, when used in the B&B fast reactors, the total US 2010 demand of electricity for, at least 8 centuries and, if ~50% burnup is attainable, for 20 centuries. No fuel reprocessing and very little (if at all) uranium enrichment will be required.
- It is possible to design even large low-leakage B&B cores to be passively safe.
- It is possible to design a B&B core to fit within the reactor vessel of SUPER-PRISM.
- The amount of TRU and, particularly, fissile inventory in B&B UNF can be smaller, per unit of electricity generated, than that in LWR UNF.
- If LWR UNF could be economically reconditioned to convert it to the feed fuel of B&B reactors, it will be possible to generate at least twice the amount of electricity the same fuel

already generated in LWRs while operating on the once-through fuel cycle. That is, the B&B core could provide a very efficient “interim” solution for the LWR UNF.

It is concluded that successful development of B&B reactors and associated fuel re-conditioning technologies offer promising new options for the nuclear fuel cycle that could provide a great measure of energy security and energy cost stability. This prospect justifies addressing the challenging technological issues that need be solved before B&B reactors.

A number of computational methods were developed or improved as a by-product of this project. They greatly facilitate the design of fuel assemblies and the search for optimal fuel shuffling scheme for B&B and other types of fast reactors.

The work performed and the findings are described in detail in 8 journal papers, 9 conference proceedings papers, 2 white papers, 22 presentations and one invention disclosure. A couple of white papers on the subject of this project were prepared for the director of the DOE NE. 3 Master and one Ph.D. dissertations were completed and 2 additional students are currently pursuing their Ph.D. work. 9 additional graduate students and 16 undergraduate students did a research project on the subject matter of this NEUP project.

Project Summary

1. Background and Introduction

Present day commercial nuclear power reactors, mostly Light-Water-Reactors (LWRs), utilize less than one percent of the natural uranium feed: the uranium enrichment level presently preferred by the industry is approximately 4.5% ^{235}U . As natural uranium contains only 0.72% of ^{235}U , it takes 8 to 10 tons of natural uranium to make 1 ton of 4.5% enriched uranium. The remaining 7 to 9 tons of depleted uranium, typically containing 0.2% to 0.3% ^{235}U , is discarded as a waste. Of the enriched uranium that is loaded into the core, only about 5% is actually fissioned, making the overall uranium utilization only $\sim 1/9$ of 5% or, approximately, 0.6%.

The amount of natural uranium that has been mined so far for fueling the fleet of commercial LWRs that presently generates close to 20% of the U.S. electricity consumption is approximately 700 thousand tons. Out of these, more than 60,000 tons ended up as used nuclear fuel (UNF)—the enriched uranium fuel that was fed into the LWRs and discharged after few percent of the uranium has been fissioned. More than 600,000 tons ended up as depleted uranium “waste”. Additional depleted uranium has been accumulated from the military programs.

By using fast breeder reactors it is possible, in principle, to fission close to 100% of the depleted uranium “waste”. However, this high uranium utilization cannot be achieved in a single irradiation campaign because neutron-induced radiation damage effects constrain the burnup level the fuel can withstand to the order of 10% to 15% FIMA (Fissions per Initial heavy Metal Atom), depending on the core neutron spectrum. Consequently, attainment of high uranium utilization also necessitates multiple fuel recycling. Traditionally, fuel recycling includes removal of the fuel cladding, removal of most of the fission products, addition of some depleted uranium make up fuel, fabrication of new fuel elements and reloading them into the reactor core for another irradiation cycle. Although technically feasible, there is a significant objection in the U.S. and other countries towards fuel reprocessing due to economic viability and proliferation concerns.

Fast breeder reactors could, in principle, also operate without fuel recycling; that is, using a once-through fuel cycle as do all of the LWRs presently operating in the USA. Although a discharge burnup of 10% to 15% FIMA is 2 to 3 times higher than that of contemporary LWRs, the uranium utilization from a once-through fast reactor is not significantly different from that of a once-through LWR because the uranium enrichment required to fuel the fast reactor is more than twice that required to fuel the LWR.

Nevertheless, it may be possible to realize a significant increase in the uranium utilization by “reconditioning” the fuel that reached its radiation damage limit limited fuel. The functions of the fuel re-conditioning are to remove a fraction of the fission products, primarily the gaseous ones, and replace the fuel clad prior to fuel re-use in the reactor. The objective of the fuel reconditioning is to overcome material performance limits in a way that cannot be used to

extract plutonium and that is, hopefully, not as expensive as conventional fuel reprocessing. After reconditioning the re-fabricated fuel is to be loaded back into the core (either original or a new one) and keep accumulating additional burnup, thereby increasing the uranium utilization.

This NEUP project initially proposed to explore the feasibility of attaining an accumulated burnup that is significantly higher than 10% FIMA by reconditioning the fuel. The focus of this project was to be the core design and fuel cycle analysis assuming the technology for fuel reconditioning could be successfully developed to be commercial. A couple of fuel reconditioning were considered: a melt-refining process similar to that experimented with during the EBR-II project [B1], and an AIROX-like process that was originally developed in the USA [B2] but more recently experimented with in the Republic of Korea [B3]. We have found, in the early part of the project, that it is neutronically (and thermal-hydraulically) feasible to achieve very high cumulative burnup – possibly exceeding 60% FIMA, by reconditioning the fuel whenever its clad reaches its 200 dpa radiation damage limit corresponding to an average burnup of ~10% FIMA.

The next major question addressed in this project was what is the optimal mode-of-operation – the in core fuel management and reactivity control strategy, for fast reactors that are to discharge their fuel at very high burnups with the assist of fuel reconditioning. The most promising approach identified is use of a special class of fast reactors, referred to as “breed-and-burn” (B&B) or “travelling wave” reactors, such as the TWR under development by Terra-Power [B4–B6]. The unique feature of a B&B reactor is that it can breed plutonium in depleted uranium feed fuel and then fission a significant fraction of the bred plutonium, without having to reprocess the fuel. In order to initiate the chain reaction, the B&B core must first be fed with an adequate amount of fissile fuel such as enriched uranium. Plutonium or TRans-Uranium elements (TRU) extracted from UNF could also be used for this “starter”. Thereafter, the B&B core is capable of continued operation while being fed solely with depleted uranium. Eventually, the uranium utilization will approach the fraction of the loaded uranium that has been fissioned.

The principles and concepts of B&B reactors had been proposed in the past; [B7–B15] is a partial list of references. These references describe either one of two basic variants of B&B reactors—one is the Travelling-Wave-Reactor (TWR) like the highly published CANDLE reactor concept [B14,B15] and the TWR concept initially pursued by Terra-Power [B4,B12]. The other is the Stationary-Wave-Reactor (SWR) in which the location of the fission reaction in the core is maintained nearly stationary by adequate radial shuffling of fuel assemblies.

However, we judged the TWR concepts to be very difficult to implement and the state of knowhow concerning the acceptable design space for SWR very limited. Consequently, the major thrust of this project was turned into a thorough analysis of the physics of B&B reactors (particularly SWR), definition of the accessible design space, and identification of the most promising design options as well as of a path forward for a phased commercialization of B&B reactors.

As a result of the evolutionary process described above, the scope of work of this project has been greatly expanded – we have addressed many more issues than originally promised and discovered novel options we did not initially think about. The list of issues we addressed during the three years of the project is given in Section 2 while our findings pertaining to each of these issues are briefly summarized in Section 3. Detailed description of the studies performed in order to reach these findings can be found in the referenced publications a subset of which is included to this report as Appendices. Section 4 summarizes the more important conclusions derived from this project and gives recommendations for future work.

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2. List of Issues Addressed

Following is a list of the issues addressed in this project:

1. Establish maximum attainable burnup in large volume fast reactor cores accounting for criticality constraints and thermal-hydraulic constraints. Assume fuel is reconditioned when reaching radiation damage constraints. Ignore (initially) passive safety requirements. Cores are not of the Breed-and-Burn (B&B) type; they are fed with fissile fuel.
2. Establish the maximum attainable burnup in a large ($3000\text{MW}_{\text{th}}$) and medium ($1000\text{MW}_{\text{th}}$) size B&B sodium cooled reactor cores accounting for criticality constraints and thermal-hydraulic constraints. Assume fuel is reconditioned and recycled when reaching radiation damage constraints.
3. Determine minimum burnup required for establishing the Breed and Burn (B&B) mode of operation
4. Clarify the difference between Travelling Wave Reactors (TWR) and Stationary Wave Reactors (SWR) – two variants of B&B reactors
5. Determine the sensitivity of the minimum required burnup and of the maximum attainable burnup to the core composition and dimensions
6. Assess the feasibility of designing lead-cooled B&B cores (so as to eliminate need for intermediate coolant loop)
7. Establish the minimum volume for B&B cores; that is, the minimum core volume in which a B&B mode of operation can be sustained
8. Assess the feasibility of spawning a new B&B core using fuel discharged from a B&B core at the minimum required burnup and estimate the implications of such a spawning mode of operation
9. Assess the feasibility of reconditioning B&B fuel by encasing a fuel rod discharged when its cladding reaches its radiation damage limit in a new clad
10. Assess the feasibility of early introduction of the B&B reactor technology starting with 200 dpa peak cladding radiation damage using seed & blanket fast reactor cores
11. How to design large-volume low-leakage fast reactor cores to be inherently safe
12. Evaluate the waste characteristics of B&B reactors and assess the feasibility of reducing fissile inventory in fuel discharged from B&B reactors so as to improve their proliferation resistance
13. Quantify implications of successful development and deployment of B&B reactors on fuel utilization, energy sustainability and economic security
14. A strategy for phased introduction of B&B reactors
15. Explore possible synergism between LWR and B&B reactors fuel cycles

3. Summary of Findings

Issue # 1: What is the neutronically maximum burnup that can be achieved in sodium-cooled fast reactor (SFR) core if the fuel is to be recycled without separation of the majority of the solid fission products. Cores are not of the Breed-and-Burn (B&B) type; they are fed with fissile fuel. Ignore (initially) passive safety requirements. [Work is summarized in references 11, 12 (Appendix A), 22]

The upper bound on the burnup attainable in large fast reactors has been quantified for a fuel cycle that does not resort to conventional fuel processing. The fuel is recycled when reaching its radiation damage limit but undergoes limited reconditioning that involves removal of, at least, the gaseous fission products and recladding the fuel. The specific process assumed for most of the analysis is the so called Melt-Refining process that has been developed in the EBR-II project. This upper bound study assumed that initial criticality is achieved by uniformly loading TRans-Uranium (TRU) isotopes from Light Water Reactor (LWR) Used Nuclear Fuel (UNL). The core keeps operating without any fuel addition, but with fuel shuffling, until it runs out of reactivity.

The upper bound on the attainable burnup (corresponding to the use of infinite number of batches in an infinite core) was found to be ~72% FIMA (Fissions per Initial Metal Atom) for metallic uranium based fuel and ~67% FIMA for metallic thorium fuel. The upper bound on the attainable burnup is smaller in a finite core due to neutron loss via leakage. In a large 3000MW_{th} core the upper bound for metallic uranium based fuel is ~65% FIMA while for a medium-size 1000MW_{th} core it is ~52%. It is significantly lower for uranium oxide based fuel for thorium based fuel.

Issue # 2: Establish the maximum attainable burnup in a large (3000MW_{th}) and medium (1000MW_{th}) size Breed-and-Burn (B&B) sodium cooled reactor cores accounting for criticality constraints and thermal-hydraulic constraints. Assume fuel is reconditioned and recycled when reaching radiation damage constraints. [Work is summarized in references 4 (Appendix B), 14, 22].

A “Breed-and-Burn” (B&B) reactor is a breeder reactor that converts into fissile fuel a significant fraction of its fertile feed fuel and then fissions a significant fraction of the bred fissile fuel, without having to reprocess the fuel. In order to initiate the chain reaction, the B&B core has to be started with adequate amount of fissile fuel such as enriched uranium, plutonium or transuranium elements (TRU) extracted from LWR used nuclear fuel (UNF).

Unless stated explicitly otherwise, all B&B cores addressed in this project are of a stationary wave (rather than travelling wave) type – the fertile fuel is loaded at the outermost core radial zone and is gradually shuffled inward as it builds up its fissile fuel content. At equilibrium (and most of this work focuses on the equilibrium core), the fuel batch that reached the design burnup objective is discharged, the other fuel assemblies are shuffled according to a pre-defined optimal pattern, and fresh fertile fuel assemblies are loaded at the core periphery.

Whenever a fuel batch reaches the presently demonstrated clad radiation damage limit (200 dpa), it is extracted from the core, reconditioned, and reloaded back. Most of the studies performed in this project assumed that an ideal Melt-Refining process is used for the fuel reconditioning, although a limited number of studies assumed an AIROX-like process that can remove only gaseous and certain volatile fission products. The ideal melt-refining process is assumed to remove 100% of Br, Kr, Rb, Cd, I, Xe and Cs; 95% of Sr, Y, Te, Ba and lanthanides and 95% of Th and Am. An AIROX-like process is assumed to remove 100% of T, C, Kr, Xe and I, 90% of Cs and Ru and 75% of Te and Cd.

Using an ideal melt-refining process it was found that, using metallic uranium fuel, the maximum attainable burnup is 57% FIMA in a large (3000MW_{th}) core and 43% FIMA in an S-PRISM size 1000MW_{th} core. Using an AIROX-like process, the maximum burnup attainable in a large core is 47% FIMA. In general, the maximum attainable burnup is very sensitive to the core design variable; it tends to increase with hardening of the neutron spectrum and with the lowering of neutron loss via leakage and parasitic capture (See discussion under Issue # 5).

Issue # 3: Determine minimum burnup required for establishing the B&B mode of operation without use of fuel reconditioning. [Work is summarized in references 1, 2 ([Appendix C](#)), 6, 10, 13, 14, 22 and 23].

The minimum average burnup in a large (3000MW_{th}) B&B reactor is ~20% FIMA. The peak discharge burnup is ~30%; the corresponding peak radiation damage to the HT-9 clad is ~550 dpa. The minimum required burnup is highly sensitive to the neutron spectrum and to the fraction of neutrons that leak out from the core or parasitically absorbed in the core.

The contour plots in Figure 1 illustrate the kind of information generated in this project that pertains to Issues # 2 and # 3. A contour defines the burnup domain that is neutronicly accessible in a class of B&B cores that use sodium coolant, very tight pitch-to-diameter ratio of 1.088, HT-9 clad and metallic fuel made of an alloy of depleted uranium (0.2% ²³⁵U) and zirconium for a specific weight % of the alloying zirconium. The “% neutron loss” variable is the fraction of fission-born neutrons that is lost via leakage – the dominant mechanism, and via parasitic absorption in control elements that are used for compensate for the burnup reactivity swing over the cycle. Conventional IFR fuel used 10 wt. % Zr, but 6 wt. % may be acceptable. Typical neutron loss fraction from a large volume 3000 MW_{th} fast reactor core is 6% to 9%.

The data used for generating the contour plots of Figure 1 was obtained using a simple yet very useful neutron balance condition developed in this project to predict the minimum burnup required for sustaining a B&B mode of operation and the maximum burnup that can be reached in a B&B reactor provided radiation damage can be mitigated:

$$N_{HM} \int_0^{BU_m} \left[P_{NL} * P_{NRC} - \frac{1}{k_{\infty}(BU)} \right] \bar{\nu}(BU) * dBU = 0$$

In the above, N_{HM} is the Heavy Metal (HM) atom density, BU is expressed in FIMA, $\bar{\nu}(BU)$ is the average number of neutrons emitted per fission, P_{NL} (= 1 – P_L) is the non-leakage probability

and $P_{NRC} (= 1 - P_{RC})$ is the probability that a fission-born neutron will escape capture in the control elements used to compensate for the burnup reactivity swing over the equilibrium cycle. In the above we ignored the contribution of (n,2n) and (n,3n) reactions. The values of P_L and P_{RC} are deduced from 3-D analysis of a representative core; the other parameters that go into the above equation can be deduced from a batch-by-batch neutron balance analysis in the specific core being analyzed; they can also be well approximated from a much simpler unit cell analysis.

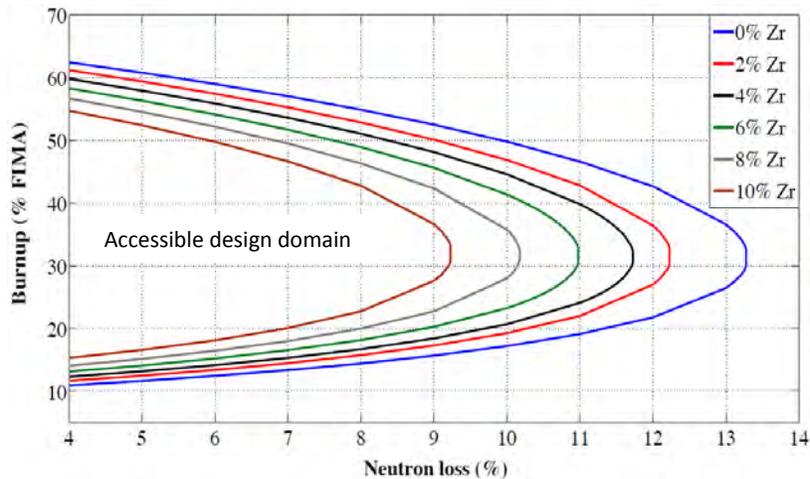


Figure 1 Illustration of burnup domain accessible with sodium-cooled tight-lattice metal-fueled B&B cores

In a conventional B&B core the fuel assemblies are shuffled in the radial direction but cannot be shuffled in the axial direction. As a result, there is a significant axial distribution of burnup levels in the discharged fuel – the axial peak-to-average discharge burnup is at least 1.5. A novel concept of pebble fuel B&B core which enable to shuffle the fuel both radially and axially was developed. The minimum average burnup discharge burnup in such a core was estimated to be ~15%; the corresponding radiation damage to the clad is ~300dpa. (23)

Issue # 4: Clarify the difference between Travelling Wave B&B Reactors (TWR) and Stationary Wave B&B Reactors (SWR) – two variants of B&B reactors [Work is summarized in reference 6 (Appendix D)].

The minimum burnup required for establishing a sustaining B&B mode in a TWR is approximately twice that required for a SWR (assuming no clad replacement). This is due to the fact that excess neutrons that are originating from the fission “wave” and leak in the direction of the “tail” of the wave do not contribute to the conversion of ^{238}U into ^{239}Pu in the blanket fuel. That is, the fraction of the neutrons that are in excess of the number required for sustaining the chain reaction that can contribute to the B&B process is significantly smaller in a TWR than in a SWR of a comparable power level. (Comment: Even though the TerraPower company refers to the B&B reactor they are developing as the TWR, its present version is, in fact, a stationary wave reactor; the burnup it requires for sustaining the B&B mode of operation is nearly half that required for a CANDLE type reactor).

Issue # 5: Determine the sensitivity of the minimum required burnup and of the maximum attainable burnup to the core composition and dimensions. [Work is summarized in references [2 \(Appendix C\)](#), 3, 4, 6, 10, 13, 14 and 22].

The minimum burnup required for sustaining the B&B mode of operation is very sensitive to the neutron spectrum and to the neutron loss probability. The harder the neutron spectrum is, the smaller can be the minimum required burnup. As metallic fuel offers the hardest possible spectrum, it features the lowest minimum required burnup. As illustrated in Figure 1, the higher the uranium weight % in the fuel, the lower is the minimum required burnup and the larger is the maximum attainable burnup. Oxide fuelled core spectrum is too soft to enable design of a practical B&B reactor. Nitride fuel is only slightly inferior to metallic uranium alloy (with Zr) fuel in terms of minimum required burnup. The B&B mode of operation cannot be sustained in cores the neutron leakage probability from which exceeds ~9%.

No sustainable B&B core can be designed using thorium as the only fertile fuel. This is so because, in the SFR spectrum, (a) $(^{233}\text{U}) < (^{239}\text{Pu})$ and (b) the fast fission probability of ^{232}Th is significantly smaller than that of ^{238}U .

Issue # 6: Assess the feasibility of designing lead and lead-bismuth cooled B&B cores so as to eliminate need for intermediate coolant loop. [Work is summarized in references [3 \(Appendix E\)](#), 8 and 22].

Of the alternate composition B&B cores examined, a special attention was given to assess the feasibility of designing large lead and lead-bismuth cooled B&B cores. A specially developed fast reactor fuel Assembly Design OPTimization code (ADOPT) was utilized for a consistent comparison of optimal lead, Pb-Bi and Na cooled core designs; ADOPT finds the optimal number of fuel rods and lattice pitch in a fuel assembly that will maximize the permissible power density while abiding by the thermal-hydraulic and structural design constraint – most if not all based on first principles.

It was found that, in order to match the minimum burnup/dpa level of a sustainable sodium-cooled B&B core, the lead or lead-bismuth cooled cores feature a lower power density although a higher specific power. Nevertheless, the benefit resulting from elimination of the intermediate coolant loop may exceed the drawback of a reduced core power density. If a Pb or Pb-Bi cooled core is to be designed to have the same power density as a sodium-cooled core, its fuel will have to be irradiated to a ~40% higher burnup/dpa, primarily due to thermal-hydraulic constraints that dictate a significantly larger pitch-to-diameter ratio; that is, smaller fuel volume fraction, in the active core region.

Issue # 7: Establish the minimum volume for B&B cores; that is, the minimum core volume in which a B&B mode of operation can be sustained [Work is summarized in reference [19 \(Appendix F\)](#)].

The minimum radius for an idealized spherical B&B reactor is 136 cm or 110 cm for, respectively, 40% or 28% coolant volume fraction. The minimum required burnup is about 25%. The minimum volume of a realistic cylindrical B&B core made of prismatic fuel assemblies of 209 cm in active fuel height is 99 cm for 28% coolant volume fraction; only ~15% larger than that of the idealized spherical core of 28% Na volume fraction. Such a core can fit within a S-PRISM like reactor vessel. The required average burnup for the minimum volume core is ~29%.

Issue # 8: Assess the feasibility of spawning a new B&B core using fuel discharged from a B&B core at the minimum required burnup and estimate the implications of such a spawning mode of operation [Work is summarized in references 1 (Appendix G), 2, 4, 6 and 22].

It is possible to use fuel discharged from a B&B core at the minimum sustainable burnup (~20%), after reconditioning, for the “starter” fuel of a new B&B reactor. The effective doubling time for spawning new B&B cores is estimated to be 13.5 years. The asymptotic B&B reactors capacity growth rate is 3.86% per year. This capacity growth rate is larger than even that of the most optimistic scenario for nuclear energy expansion rate forecasted by the IASA—3.6% per year. If a single 3000 MW_{th}/1.2 GWe B&B core is started in 2020 and will be operated in the spawning mode featuring the 3.86%/y capacity growth rate, the total installed B&B capacity will be 25.2 GWe by 2100 and 40.8 GWe by 2120. The spawning mode of operation is schematically illustrated in the figure below. Except for the several tons of enriched uranium or plutonium or TRU required for establishing initial criticality in the first generation (“Mother”) core, this expanding fleet of B&B reactors requires only depleted uranium for its fuel feed and no fuel reprocessing (though fuel reconditioning).

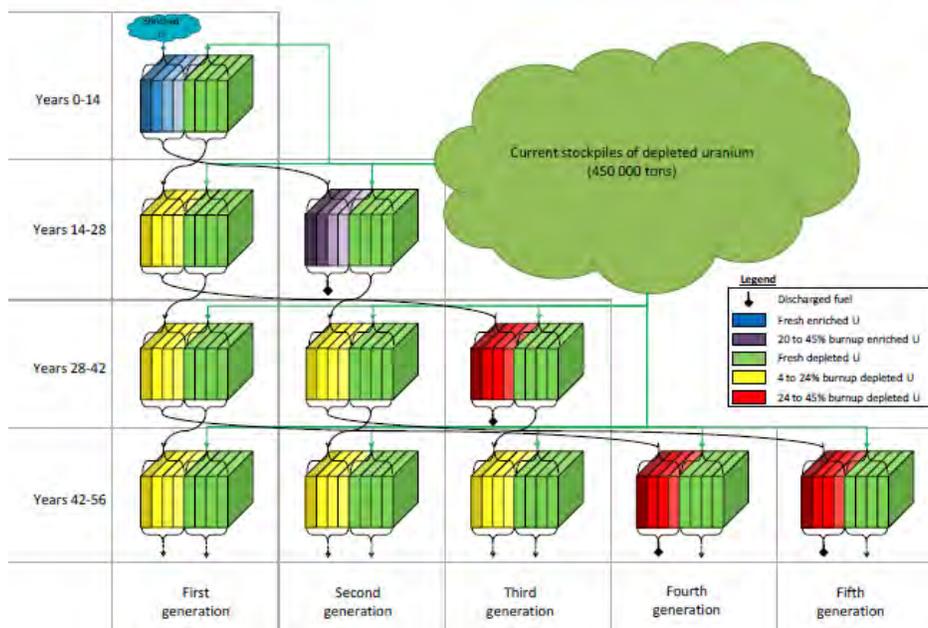


Figure 2 Schematics of the spawning mode of operation of B&B reactors

Figure 3 shows the maximum electrical capacity growth rate due to one large B&B reactor deployed in 2020 and operated in the spawning mode with the minimum possible doubling time of 13.5 years.

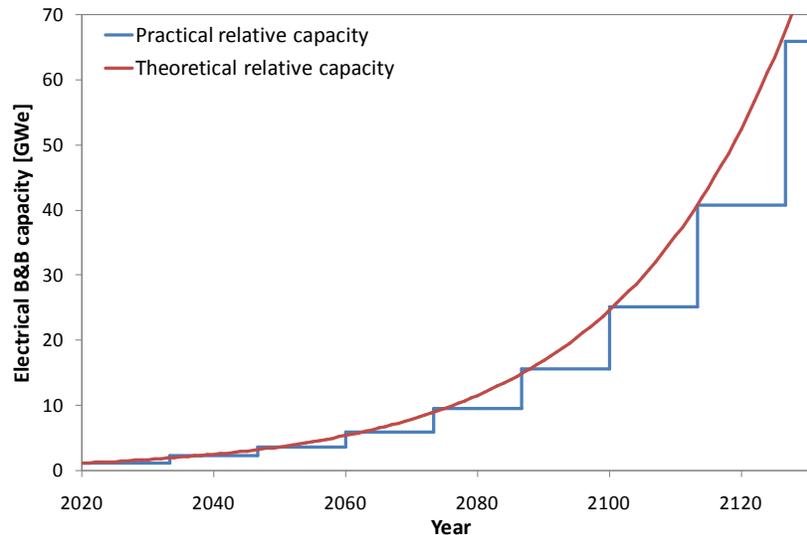


Figure 3 Electrical capacity evolution due to one large B&B reactor deployed in 2020 and operated in the spawning mode.

Issue # 9: Assess the feasibility of reconditioning B&B fuel by encasing a fuel rod discharged when its cladding reaches its radiation damage limit in a new clad [Work is summarized in references [15 \(Appendix H\)](#), [17](#)].

This study assessed the neutronics feasibility of reconditioning the fuel discharged from a B&B core at an average burnup of ~20% FIMA by enclosing the fuel along with its irradiated cladding, after venting the gaseous fission products, in a new cladding. The double clad fuel rods are to be reconstituted into new fuel assemblies featuring a somewhat larger pitch than the original fuel assemblies. It was found that the double-clad fuel can be irradiated in a stand-alone fast reactor up to an additional burnup of ~10% FIMA. Using an accelerator spallation neutron source to drive a subcritical core the attainable additional burnup is ~20% FIMA. The practical feasibility of double cladding the fuel is questionable but worth examining. Also worth examining is the feasibility of removing an outer layer of the irradiated cladding before encasing the fuel rod in a new cladding. The larger the fraction of the irradiated cladding that can be removed the larger the attainable extra burnup is expected to be.

Issue # 10: Assess the feasibility of early introduction of the B&B reactor technology starting with 200 dpa peak cladding radiation damage using seed & blanket fast reactor cores [Work is summarized in reference [6 \(Appendix D\)](#)].

A new approach to seed-and-blanket core design was conceived to enable to start benefiting from the B&B mode of operation in the near future by first using proven technology and gradually increasing the benefit as materials capable of withstanding higher and higher radiation

damage (burnup) levels become available. It is suggested to design the B&B core to be subcritical and to “drive” it by neutrons leaking from a critical TRU (or enriched U) sodium-cooled fast reactor (SFR). Conventional SFR cores are designed to be of a pancake shape and to have a large neutron leakage probability – typically ~20%. The large leakage probability is required for reducing the positive coolant temperature reactivity coefficient and coolant voiding reactivity effect – prerequisites for achieving passive safety. The leaking neutrons are not utilized. We are proposing to utilize the large number of neutrons that leak out from the SFR core without penalizing the core performance – instead of designing the SFR core to be of a pancake shape with the dominant neutron leakage being in the axial direction, it is proposed to design the SFR core to be of a cigar shape with the majority of the neutrons leakage in the radial direction and to make use of the leaking neutrons to “drive” a B&B blanket that radially surrounds the core. The primary design objective is to maximize the fraction of the total power generated by the B&B blanket. The larger this fraction is, the lower the SFR fuel cycle cost is likely to be, as the recycled TRU fuel is expected to be significantly more expensive than the cost of a once-through depleted uranium fuel. Whereas the “seed” (critical core) fuel can be recycled, as planned for SFRs, the blanket fuel is to operate on the once through fuel cycle – as planned for B&B cores. The blanket feed fuel is fertile – depleted uranium or thorium, and the blanket fuel management is to provide for the maximum tolerable discharge burnup. The initial design will limit the blanket and seed fuel discharge burnup to the licensable 200 dpa for HT-9 cladding. As cladding materials that are qualified to operate to higher dpa levels become available, the fuel management strategy will be modified to increase the discharge fuel burnup. This gradual increase in the acceptable dpa/burnup level will proceed until, hopefully, reaching the ~550 dpa range at which the B&B mode of operation could be sustained in a critical core.

Using depleted uranium feed for the blanket it was found possible to design the B&B blanket to generate as much as 50% of the total seed-and-blanket reactor power while discharging both the once-through blanket fuel and the seed fuel when their cladding accumulates 200dpa – that is, without exceeding acceptable radiation damage levels. When cladding materials are qualified to operate up to 300 dpa, the blanket will be able to generate ~60% of the total core power. At 400 dpa the blanket power fraction is likely to exceed ~70% while at 550 dpa it will reach 100%.

Earlier in this project we found that critical B&B cores cannot be designed using fissile-fuel-free thorium feed fuel. This is so because, in the SFR spectrum, (a) $\eta(^{233}\text{U}) < \eta(^{239}\text{Pu})$ and (b) the fast fission probability of ^{232}Th is significantly smaller than that of ^{238}U . However, thorium could be utilized without recycling in a B&B subcritical blanket driven by the neutrons that radially leak out from a cigar-shape SFR core. As in the case of depleted uranium blanket, the extra power to be generated by the thorium blanket will lower the SFR fuel cycle cost and, thus, improve the SFR economic viability. In addition, the use of thorium blanket will contribute to the sustainability of nuclear energy by providing an effective approach to fission a significant fraction of the mined thorium without having to recycle the thorium. It is estimated that at least 15% of the thorium loaded into the blanket could be fissioned – this is approximately 30 times the utilization of natural uranium in light-water-reactors. Simultaneously with extra power generation, the thorium blanket produces ^{233}U . In case the critical driver is designed to be a TRU transmuter, (i.e., function as an Advanced Burner Reactor – ABR), the proposed driver-blanket core concept will

effectively convert TRU into ^{233}U . This conversion could be valuable if there is an interest in deployment of a self-sustained thorium-based nuclear energy system for which significant inventories of ^{233}U will be required.

Issue # 11: How to design large volume fast reactor core to be inherently safe [Work is summarized in references 16, 18 (Appendix I), 21 and 26].

In order to achieve the good neutron economy required for establishing the B&B mode of operation at an average discharge burnup of $\leq 20\%$ FIMA the core need be large and feature a low neutron leakage probability and a hard neutron spectrum. These same features result in a large positive value of void reactivity worth and coolant temperature reactivity coefficient. These impair the ability of the core to safely respond passively to accidents such as unprotected loss of heat sink (ULOHS) and unprotected loss of flow (ULOF).

A number of approaches for enabling the low-leakage B&B cores to passively mitigate such accidents without impairing the neutron balance were investigated. The most promising approach identified is to counteract the positive reactivity induced by coolant temperature increase by passively driving a neutron absorbing material (^6Li) from the axial reflector region to the active core region. The lithium is contained in a special small diameter tube (on the order of a fuel rod diameter) that is integrated into the fuel assembly design. The ^6Li insertion mechanism is passively driven by the coolant temperature increase. Conceptual design of a number of embodiments of such novel passive safety systems, referred to as the Assembly Reactivity Control (ARC) system, was developed and evaluated. Results of preliminary analysis are very promising. A US Provisional Patent was granted for the liquid-liquid ARC system in August of 2012. Although developed with large volume B&B core in mind, the ARC devices can be installed in any liquid (and gas) cooled fast reactor core and improve the core passive safety. A detailed time-dependent transient and accident analysis need yet to be performed before the ARC system could be an acceptable design approach.

Issue # 12: Evaluate the waste characteristics of B&B reactors and assess the feasibility of reducing fissile inventory in fuel discharged from B&B reactors so as to improve their proliferation resistance [Work is summarized in references 4, 7, 20 and 25 (Appendix J)].

Table 1 compares selected fuel cycle characteristics of B&B reactors that discharge their fuel at (a) 20% FIMA (minimum) or (b) 55% FIMA (maximum) versus a once-through PWR and a conventional sodium fast reactor featuring breeding ratio of 1.0 that operates with unlimited fuel reprocessing/recycling (ARR). The comparison is done per unit of electricity generated, assuming that the energy conversion efficiency is 33% for the PWR and 40% for all the fast reactors. It was found that the TRU and plutonium inventory in fuel discharged from a B&B core at an average burnup of 20% is approximately twice that in LWR used nuclear fuel. The fissile plutonium fraction in the 20% B&B UNF is $\sim 81\%$ versus $\sim 64\%$ in PWR UNF. However, if the B&B fuel is discharged at the maximum feasible burnup of $\sim 55\%$ FIMA with the help of fuel reconditioning process, the specific TRU and Pu inventory is close to half that of LWR and the fissile Pu fraction is comparable at $\sim 67\%$.

Relative to conventional fuels-self-sustaining ARR fast reactor, the B&B reactors offer significantly smaller throughput of HM, TRU and Pu per unit of electricity generated – approximately 40% for the 20% B&B reactor and 15% for the 55% B&B reactor. This large difference is primarily due to the higher discharge burnup of the B&B fuel and ignores the throughput that the 55% B&B reactor fuel through the reconditioning plant. Nevertheless, the 2.5 folds reduction in the throughput of the 20% B&B fuel, that will need no reconditioning, is remarkable. It should be realized, though, that once cladding materials that could withstand a radiation damage exceeding 200 dpa are available, ARR type reactors could also be designed to discharge their fuel at a higher burnup and will, hence, feature a smaller HM specific throughput than given in Table 1.

Table 1 Comparison of selected fuel cycle characteristics of B&B reactors that discharge their fuel at (a) 20% FIMA (minimum) or (b) 55% FIMA (maximum) versus a once-through PWR and a conventional sodium fast reactor featuring breeding ratio of 1.0 that operates with unlimited fuel reprocessing/recycling (ARR)

Characteristic	PWR	ARR	20%B&B	55%B&B
Discharge burnup (MWD/MT)	50,000	73,000	189,455	540,930
Specific HM loading (Kg/GWeY)	2.21E+4	1.21E+4	4.82E+3	1.69E+3
Loaded fuel type	Enriched U	Recy U+TRU	Depleted U	Depl+Recon U
Natural uranium utilization (%)	0.6	→99	20	55
Total TRU discharged (Kg/GWeY)	251	1700	481	133
Total Pu discharged (Kg/GWeY)	225	1650	476	130
TRU/HM in discharge (%)	1.14	15.25	12.37	17.58
Fissile Pu/HM in discharge (%)	0.65	10.24	9.97	11.41
Specific fissile Pu discharge (Kg/GWeY)	143	1140	388	86
Fissile Pu/Pu in discharge (%)	63.7	69.3	81.4	66.6
Amount of Pu generated (Kg/GWeY)	224	~0	477	129

Figure 4 compare the specific radiotoxicity of the fuel and waste coming out from the B&B reactors with the specific radiotoxicity of the PWR and a couple of SFRs – the conversion ratio of one being 1.0 (ARR) and of the other 0.5 (ABR). It is seen that the radiotoxicity generated by the B&B reactors is smaller than that of the PWR with the exception of the B&B reactor that discharges its fuel at an average burnup of 20% FIMA in the period from ~2000 to ~ 50,000 years after discharge. A similar conclusion applies to the decay heat.

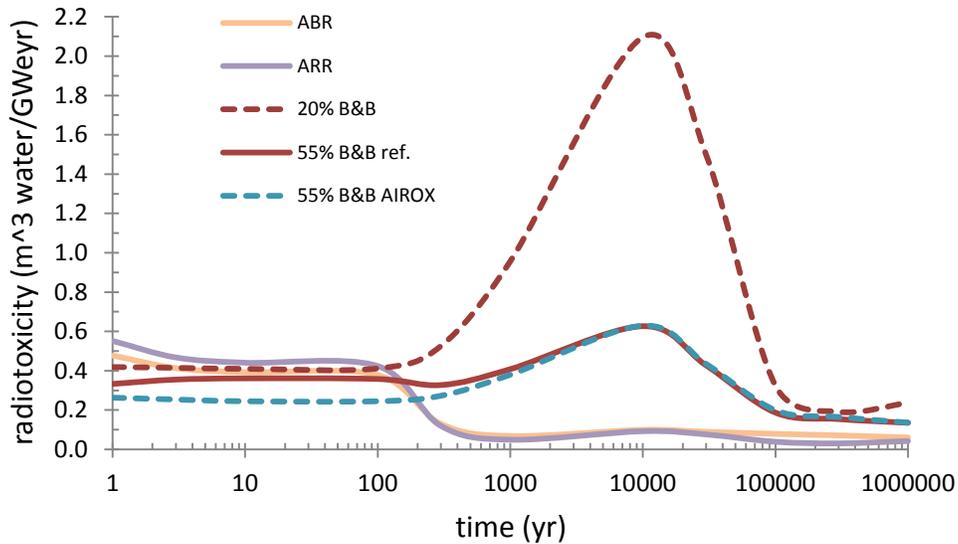


Figure 4 Radiotoxicity of the fuel and waste discharged from B&B reactors relative to that of a once-through PWR and of conventional sodium-cooled fast reactors [ARR (CR=1.0); and ABR (CR=0.5)]. Normalized per unit of electricity generated.

It was also found that by irradiating fuel discharged from B&B cores at 20% FIMA in a thermal spectrum an extra 5% FIMA, the fissile Pu fraction is reduced from more than 80% to less than 40%. The specific inventory of fissile plutonium is reduced to ~1/3 of its pre-thermal-irradiation value – from 388 to ~135 kg/GWeY); even slightly lower than the 143 of the PWR. Figure 5 shows the evolution of the plutonium isotopes under such a thermal irradiation. Three approaches were investigated for minimizing the fissile plutonium content in the fuel discharged from B&B cores at an average burnup of 20% FIMA so as to enhance this fuel proliferation resistance and to reduce waste disposal burden and radiotoxicity:

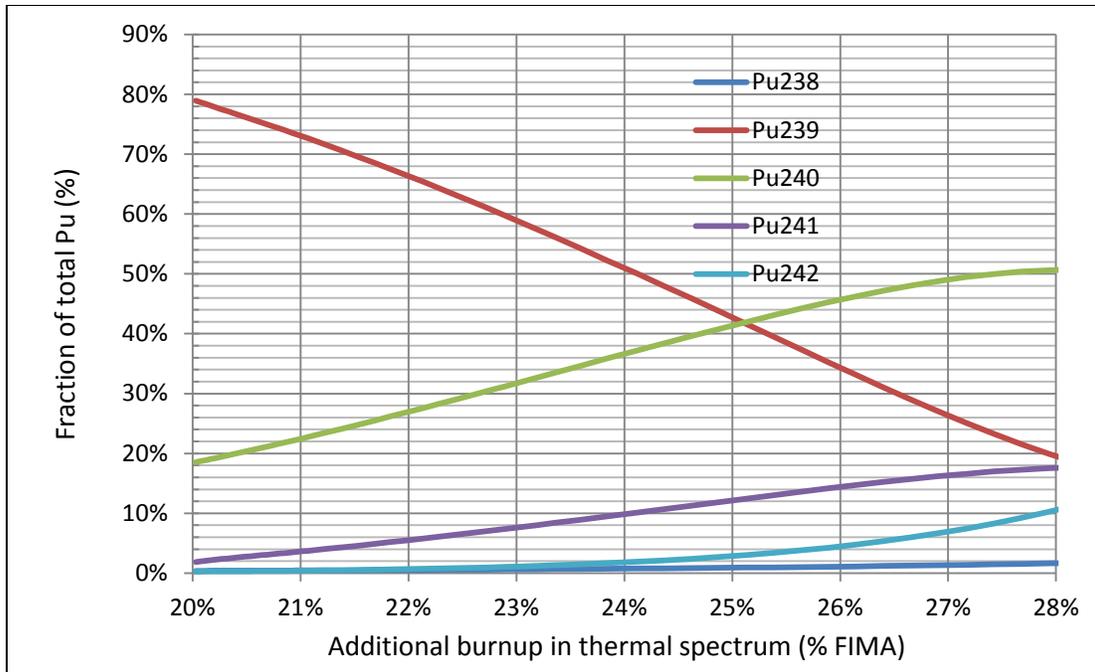


Figure 5 Evolution of Pu isotopic vector in a fuel discharged from a B&B core at 20% FIMA during irradiation in a PWR-like spectrum. Normalized to 100%.

(1) Irradiating the fuel discharged from the B&B core for additional few % FIMA in a stand-alone thermal spectrum core. It was found that the initial fuel k_{∞} , that was >1.1 in the B&B core spectrum, drops to between 0.4 to 0.6, depending on the softness of the spectrum. This drop was found due to high concentration of selected fission products (FP) such as ^{149}Sm (captures $\sim 50\%$ of neutrons captured by all fission products), ^{157}Gd (20%), ^{151}Sm (5%) and ^{155}Gd (5%). It was also found that irradiation of this B&B discharged fuel in a thermal neutron spectrum to an extra burnup of $\sim 0.5\%$ FIMA is sufficient to reduce the concentration of the bad FP players to bring the fuel k_{∞} above 1.0. Thereafter this fuel can be used in a stand-alone thermal reactor for an additional $\sim 5\%$ FIMA as a result of which its fissile plutonium content is reduced to nearly 25% of its value in the fuel discharged from the B&B core (at $\sim 20\%$ FIMA). This will significantly improve the proliferation resistance and the waste characteristics of the fuel discharged from the once-through B&B reactors. Alternatively, if the 20% FIMA fuel discharged from the B&B core undergoes a melt-refining process, this process will remove the most absorbing fission products and thereby make it possible to achieve criticality in a thermal spectrum core.

(2) Irradiating the fuel assemblies discharged from the B&B core at $\sim 20\%$ FIMA for an additional $\sim 0.5\%$ FIMA in a soft spectrum blanket (graphite was assumed for our analysis) that surrounds the B&B core and is driven by the neutrons that radially leak out from the core. This irradiation was found sufficient for “burning” enough of the bad FP players to bring the fuel k_{∞} above 1.0. This pre-treated fuel is then loaded, after reconditioning, into a stand-alone thermal reactor in which it is operated for an additional $\sim 5\%$ FIMA to achieve the fissile fuel concentration reduction stated above.

(3) Double cladding the fuel discharged (at ~20% FIMA) from the B&B core and loading it into a fast core that is surrounded by a soft spectrum blanket. The double clad B&B core discharged fuel is loaded in the fast spectrum core and, after a few % FIMA is shuffled into the soft spectrum blanket. The system is designed to have a relatively large neutron leakage probability from the core to the blanket. It is yet to be determine whether or not such a system can be designed to be critical and, if not, what spallation neutron source strength is required to drive it.

Issue # 13: Quantify implications of successful development and deployment of B&B reactors on fuel utilization, energy sustainability and economic security [Work is summarized in references 1 (Appendix G), 4, 6 (Appendix D), 14 and 22].

Table 2 compares the estimated uranium utilization that could be achieved with B&B reactors that are designed and/or operated in either one of the following five modes, all using depleted uranium for the blanket fuel feed:

- (a) A seed-driven subcritical B&B blanket the fuel of which is discharged at an average burnup of 10% FIMA. No fuel reconditioning is required. (See Issue # 10).
- (b) A critical stationary-wave B&B core (SWR) using a fuel that can maintain its integrity up to an average burnup of at least 20% FIMA. No fuel reconditioning is required unless the discharged fuel is to be used for spawning new B&B reactors.
- (c) Like “b” along with a successful development of the technology for a single fuel reconditioning at ~20% burnup. Spawning new SWR is possible.
- (d) A critical SWR or, possibly, TWR with 2 or more fuel reconditioning steps that will enable to achieve the maximum attainable burnup of ~50% FIMA.
- (e) Traditional fast breeder reactor approach in which fuel is reprocessed and recycled every 10% FIMA or so. It assumes extraction of all of the fission products and addition of depleted uranium makeup fuel at each recycle. There is no limit to the number of fuel recycles.

Also given in the table is the uranium utilization in the reference scenario of contemporary LWRs that operate with the once-through fuel cycle and discharge their fuel at 50 GWD/T.

The relative uranium utilization values given in the table are per unit of electrical energy generated. In converting thermal energy to electrical energy it is assumed that fast reactors convert thermal energy into electricity at 20% higher efficiency than LWRs.

The rightmost column in the table gives the number of years the B&B reactors could supply electricity at present day USA total annual consumption rate from all sources (assumed 4200 million MW_eh/year) if they are to be fueled only with the depleted uranium stockpiles (“waste”) that will be accumulated in the US from the fueling of LWRs (~1.3 × 10⁶ tons) and B&B reactors (~0.5 × 10⁶ tons) until the end of deployment of the first generation of B&B reactors—assumed in the second half of the 21st century.

Table 2 Implications of successful development of B&B reactors and of fuel reconditioning technology on the uranium utilization and energy security as measured by the total amount of electricity that could be generated from the existing stockpiles of depleted uranium (waste).

Mode of operation	U utilization	Relative U utilization	No. of years at present supply
Light Water Reactors (LWRs)—reference	0.6%	1	0
(a) subcritical B&B blanket; no reconditioning	10%	20	400
(b) SWR*; 20% average discharge BU	20%	40	800
(c) SWR, 1 reconditioning @ 20%; spawning is possible	40%	80	1600
(d) SWR or TWR*, with > 1 fuel reconditioning	50%	100	2000
(e) Fast reactor with continuous recycling	> 95%	> 190	3900

It is observed that using practically proven fuel technology in subcritical B&B blankets it is possible to achieve a uranium utilization that is 20-fold that offered by LWR. A successful development of B&B reactors that can achieve 20% average fuel burnup which, hopefully, could be achieved without fuel reconditioning, will offer 40-fold increase in the uranium ore utilization *versus* that presently achieved. A successful development of a fuel reconditioning technology could increase the attainable uranium utilization to close to 100-fold that achieved in contemporary LWRs. This corresponds to extraction of approximately 50% of the nuclear energy worth of depleted (and natural) uranium. All the above options do not require separation of most of the solid fission products from the actinides. For the utilization of the remainder 50% it will be necessary to develop economically viable and societal acceptable fuel reprocessing technology that will separate the fission products from the actinides. Such a reprocessing could be deferred, though, by several centuries, as the existing stockpiles of depleted uranium can provide all our electricity needs for between 400 to 2000 years (rightmost column of the table). Basically, a similar SFR technology could be used for implementing the different options.

Once Mode c could be implemented, it will be possible to rapidly increase the deployed capacity of B&B reactors without need for uranium enrichment or for another source of fissile fuel supply beyond that required for starting the first generation of B&B reactors. The capacity increase will be achieved by the first generation spawning the second generation of B&B reactors, the first and second generation spawning the third generation and so on and so forth. In an illustrative scenario worked out the amount of natural uranium required for starting a fleet of B&B reactors that will reach an electricity generation capacity of 1000 GWe by the end of this century is estimated to be the equivalent of 10 years of supply to the presently operating commercial fleet of LWRs in the US (86 GWe). No natural uranium and no enrichment capacity will be required to support this fleet beyond the later part of this century. The energy value of the depleted uranium stockpiles (“waste”) that will be accumulated in the US by that time is equivalent to, when used in the B&B reactors, up to 20 centuries of the total 2010 supply of electricity in the USA. It is therefore concluded that a successful development of B&B reactors and associated fuel reconditioning could provide a great measure of energy security, proliferation resistance and cost stability.

Issue # 14: A strategy for phased introduction of B&B reactors [Work is summarized in reference [6 \(Appendix D\)](#)].

A strategy is outlined for early introduction of B&B reactors followed by a gradual increase in the fuel utilization of such reactors. In the first phase the fast reactor core will consist of a subcritical B&B blanket driven by a relatively small critical seed which should probably be fueled with enriched uranium as the US does not have the reprocessing capability required for extracting Pu or TRU from LWR UNF. Both seed and blanket fuels will be discharged at the presently accepted dpa level of 200dpa for which fast reactor fuel had been qualified. As the required discharge radiation-damage to both driver and blanket fuel had already been proven, and as the depleted uranium fueled B&B blanket could generate close to 50% of the core power and will have very low fuel cycle cost, the deployment of such fast reactors could start in the near future.

The second phase will be improvement in the performance of the seed-and-B&B blanket reactors as soon as the cladding material is certified to operate up to higher dpa levels. At 300dpa the fraction of core power that could be generated by the depleted uranium B&B blanket will be ~60%, at 400 dpa the blanket power fraction is likely to exceed ~70% while at 550 dpa it will reach 100%. The larger is power fraction that is generated by the B&B blanket, the lower will be the reactor fuel cycle cost and the better will be its economic viability. The seed-and-blanket reactors could also make important contribution to the development of the technology required for the sustainable B&B reactors as they can provide a valuable test-bed and irradiation platform for testing under realistic conditions improved fuels and cladding materials.

The third phase consists of deploying self-sustaining stationary wave B&B reactors. It will require development of fuel technology that could withstand peak burnups of ~30% and peak radiation damage to the cladding of ~550 dpa.

The fourth phase requires development of a fuel reconditioning technology that will enable using the fuel up to an average burnup of ~50%—the upper bound permitted by neutron balance considerations when most of the fission products are not separated from the fuel. The increase in the uranium ore utilization relative to that provided by contemporary power reactors is quantified in Table 2. Likewise for the energy value of the depleted uranium stockpiles (“waste”) accumulated in the US.

Issue # 15: Explore the synergism between LWR and B&B reactors fuel cycles [Work is summarized in reference [6 \(Appendix D\)](#)].

It is suggested to explore the possibility of using reconditioned LWR used nuclear fuel for the blanket of the B&B core. The required functions of the LWR UNF reconditioning are removal of the gaseous fission products and zircaloy cladding, conversion into metal alloy form and fabrication of fuel rods and fuel assemblies of the dimensions and design that is suitable for the B&B core, using HT-9 or another acceptable type of cladding material. There is no need to remove from the LWR UNF any of the actinides or solid fission products. There is no must to

convert the fuel from an oxide to a metal alloy, although doing so will significantly improve its performance. Whereas oxide fuel cannot establish a sustainable B&B mode of operation in a critical core, it will be likely able to generate a significant amount of extra energy in the subcritical B&B blanket of a seed & blanket core —possibly more than twice the amount of energy it generated in the LWR. An AIROX or DUPIC-like process can be used for decladding the LWR UNF and removing the gaseous fission products. The leftover oxide fuel powder could be either sintered back into pellets or, probably simpler, be mixed with depleted uranium metal powder to make cermet-type fuel by vibropacking as recently proposed by Leon Walters and Dave Wade.

As indicated under Issue # 12, the fuel discharged from a B&B core at a burnup of 20% has high enough fissile plutonium concentration to operate in a LWR for another 5% FIMA or so. As the fuel discharged from the B&B core contains a very large concentration of rare-earth fission products that feature a large absorption cross section for thermal neutrons, these fission products need be either removed – which will happen in a melt refining process, or will have to be transmuted by thermal neutrons. Such a transmutation could, in principle, be performed in the LWR core thus, possibly, using the B&B UNF to function as a burnable poison in the LWR.

The options described in “Issue 15” were not studied quantitatively thoroughly enough in this project for us to be able to suggest whether or not they are of practical interest. Assessment of the feasibility of these concepts is recommended for future work.

Methods Development:

As a by-product of this project, the following computational methods were developed or improved so as to facilitate the design and analysis of B&B and other types of fast reactors:

1. An improved version of the MOCUP code system for coupling MCNP neutron transport code with the ORIGEN depletion code that enables handling larger number of isotopes and burnup zones.
2. A versatile driver code, referred to as Fast-BEAU, that enables an automated, efficient iterative search for the equilibrium composition of multi-fuel-batches B&B cores. It greatly facilitates the search for the optimal fuel shuffling strategy for B&B cores.
3. The ADOPT code [9] for fast reactor fuel Assembly Design OPTimization that finds the optimal number of fuel rods and lattice pitch in a fuel assembly that will maximize the permissible power density while abiding by thermal-hydraulic and structural design constraint.
4. A utility code that converts a multi-radial-zone cylindrical core model used for depletion analysis into a realistic assembly-by-assembly core layout that makes the optimal match to the approximate cylindrical model.

4. Conclusions and Recommendations

A successful development of metallic fuel and cladding that can maintain the fuel rod integrity up to a peak burnup of ~30% FIMA and peak radiation damage of ~550 dpa will enable the operation of stationary-wave fast reactors (SWR) in a sustainable Breed-and-Burn (B&B) mode using depleted uranium for the feed fuel. Such SWR reactors will offer 40-folds increase in the uranium ore utilization relative to contemporary LWR while operating in a once-through fuel cycle. A successful development of a fuel reconditioning technology could enable an increase in the attainable uranium utilization of B&B reactors to 100-folds its present value without separation of actinides from most of the fission products. It will also enable the use of reconditioned B&B fuel to provide the initial fissile fuel loading required to spawn new B&B reactors without the need for external supply of fissile fuel. The growth rate of the installed capacity of B&B reactors possible to achieve using such a spawning mode of operation is estimated to be nearly 3.9% per year. Only limited amount of enriched (and, therefore, natural) uranium is required for starting a fleet of first generation breed-and-burn reactors. The amount of natural uranium required for starting a fleet of B&B reactors that will reach an electricity generation capacity of 1000 GWe by the end of this century is estimated to be the equivalent of 10 years of supply to the presently operating commercial fleet of LWRs (86 GWe). No enriched uranium and no enrichment services will be required to support this fleet beyond the completion of the deployment of the 1st generation of B&B reactors – possibly second half of the 21st century. The energy value of the depleted uranium stockpiles (“waste”) that will be accumulated in the US from the fueling of LWRs and B&B reactors until end of deployment of first generation of B&B reactors is equivalent, when used in breeding fast reactors, to from 8 to 20 centuries of the total 2010 supply of electricity in the USA. It is therefore concluded that a successful development of the breed-an-burn reactors and/or the associated fuel re-reconditioning technologies could provide the US and many other countries a great measure of energy security and energy cost stability. This prospect justifies embarking upon an R&D effort aimed at developing the technology enablers for B&B reactors. Specifically it is recommended to undertake the following:

- (1) Development and/or qualification of cladding materials that are able to maintain their mechanical integrity up to peak burnups of at least 30%FIMA / 550dpa.
- (2) Development of improved fuel, fuel rods and fuel assembly designs. Relative to conventional fast reactors, the B&B reactor fuel will be typically twice as long, will have to operate for approximately twice as high a burnup and will have a significantly longer residence time in the core.
- (3) Development of economically viable technology for reconditioning the fuel discharged from B&B reactors and recycling it for additional use in B&B reactors. The primary functions of the reconditioning are (a) to remove the gaseous fission products accumulated in the fuel and (b) to replace the irradiated cladding by a new one. There is no need to separate the rest of the fission products from the actinides. Although not necessarily required (assuming a success in above listed items “1” and “2”) such a reconditioning process will open new

promising options for the development of B&B reactors that could greatly increase the benefits from B&B reactors. Three of the technological options suggested for evaluation for the fuel reconditioning are (i) an improved version of the EBR-II tested melt-refining process; (ii) a variant of the AIROX process that is applicable to metallic fuels; (iii) Recladding the fuel rod in a new cladding after relieving the accumulated gaseous fission products pressure and, possibly, after removing an outer layer of the irradiated cladding (See Issue # 9).

One of the fundamental questions that need to be addressed is the refabricability of fuel that contains a large amount of rare-earth and other fission products. A special liner may have to be introduced between such fuel and the clad in order to mitigate undesirable chemical interaction between fission products and the cladding material. Another important question is whether or not reconditioning of highly radioactive fuel containing plutonium, minor actinides and many fission products could be justified economically.

- (4) Verification of the safety and licensibility of B&B reactors. A unique challenge is to design the B&B reactor to be inherently safe. Most conventional fast reactor cores are designed to have a high neutron leakage probability (primarily in the axial direction) so as to reduce the typical positive reactivity effect of coolant density reduction—either by temperature increase or voiding. The B&B reactor cores must have significantly lower neutron leakage probability and, hence, tend to have more positive coolant temperature and void coefficients of reactivity. We are proposing to passively compensate for the relatively large positive coolant temperature reactivity coefficient by incorporating the Autonomous Reactivity Control (ARC) system in the B&B fuel assemblies (See details under Issue # 11); it is designed to insert into the active core region ${}^6\text{Li}$ neutron poison in a way that is passively actuated by coolant temperature increase. Detailed time-dependent simulation and experimental verification of the feasibility and license-ability of such passive ${}^6\text{Li}$ injection systems are yet to be performed.

As it may take significant time and R&D effort to develop the fuel technology that is required for operating a sustainable B&B reactor that is fed with depleted uranium, a phased commercialization of B&B reactors is recommended. It is proposed to start benefiting from the B&B mode of operation by deploying seed-and-blanket fast reactors in which a subcritical B&B blanket is driven by neutrons leaking from a critical seed, without exceeding $\sim 10\% \text{FIMA} / 200 \text{ dpa}$; that is, relying on proven fuel technology. Such seed-and-blanket reactors are expected to be more economically viable than conventional fast reactors. When using depleted uranium for its feed fuel, the subcritical B&B blanket could generate approximately 50% of the total core power without exceeding the radiation damage constraints. As fuel designs that can be certified to operate at higher than $\sim 10\% \text{FIMA} / 200 \text{ dpa}$ become available, the seed-and-blanket core could be designed to discharge the fuel at higher burnups and to offer higher uranium utilization. For example, when cladding materials are certified to operate up to 300 dpa, the blanket of the seed-and-blanket fast reactors could generate $\sim 2/3$ of the total power. The amount of fuel reprocessing and TRU fuel re-fabrication required for the seed fuel of such a seed-and-blanket core is only $\sim 1/3$ that required for a conventional fast reactor core, when measured on per unit of electricity generated by these cores. As a result, the fuel cycle cost of the seed-and-blanket reactor is expected to be significantly smaller than that of a conventional fast reactor.

The upper part of Figure 6 illustrates the phased commercialization approach recommended while the table at the bottom of Figure 6 gives the benefit expected from each phase in terms of improved uranium utilization relative to once-through LWR and in terms of the electricity value of the USA depleted uranium stockpiles.

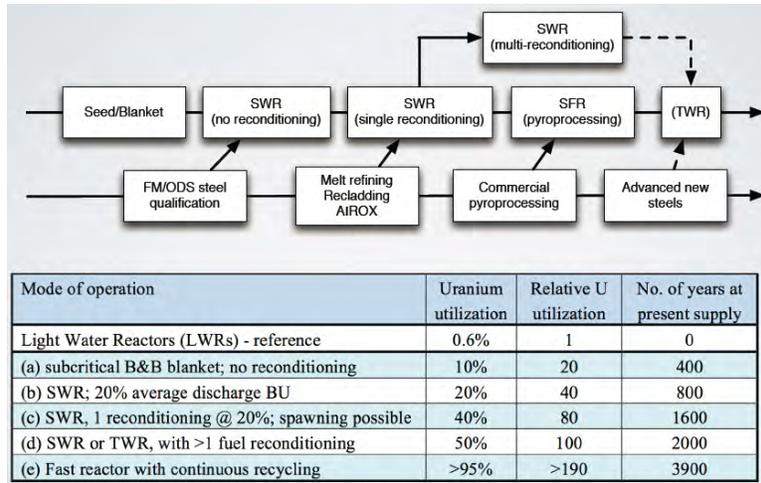


Figure 6 Approach for phased commercialization of B&B reactors

It is also recommended to study the feasibility of transforming the used nuclear fuel (UNF) from the LWR into the feed fuel for the B&B reactor blanket. The reconditioning of the LWR UNF should include removal of the gaseous fission products and zircaloy cladding and fabrication of fuel rods and fuel assemblies of the dimensions and design that is suitable for the B&B reactor blanket, using HT-9 or another acceptable type of cladding material. There is no need to remove from the LWR UNF any of the actinides or solid fission products and, although very desirable, there is no must to convert the fuel from an oxide to a metal alloy. Whereas oxide fuel cannot establish a sustainable B&B mode of operation in a critical core, it can generate a significant amount of extra energy in a subcritical B&B blanket—possibly more than twice the amount of energy it generated in the LWR.

Finally it is recommended to continue investigating design options that will enable to irradiate the B&B UNF in a soft spectrum in order to minimize the fissile fuel (primarily, plutonium) content in the discharged fuel to make it significantly more proliferation resistant, while extracting additional energy (Issue # 12).

ACKNOWLEDMENT

Financial support provided by Terra Power enabled more students to work on this project than would have been possible with only the NEUP funding

5. Project Publications

The following publications summarize most of the work performed in the framework of this project. They are included in the Appendices and should be considered as part of this projects' summary.

Journal papers:

1. E. Greenspan and F. Heidet, "Energy Sustainability and Economic Stability with Breed and Burn Reactors", [Progress in Nuclear Energy](#), **53** (September 2011), pp. 794-799 {Appendix G}
2. F. Heidet and E. Greenspan, "Neutron Balance Analysis for Sustainability of breed and Burn Reactors", Accepted for publication in [Nuclear Science and Engineering](#), 20 September 2011. {Appendix C}
3. F. Heidet and E. Greenspan, "Feasibility of Lead Cooled Breed and Burn Reactors", [Progress in Nuclear Energy](#), Vol. 54, pp. 75-80, 2012. {Appendix E}
4. F. Heidet and E. Greenspan, "Large Breed-and-Burn Core Performance", Accepted for publication in [Nuclear Technology](#). {Appendix B}
5. F. Heidet and E. Greenspan, "Super-PRISM Size Breed-and-Burn Sodium-Cooled Core Performance", Accepted for publication in [Nuclear Technology](#).
6. E. Greenspan, "A Phased Development of Breed-and-Burn Reactors for Enhanced Nuclear Energy Sustainability," [Sustainability](#) 2012, **4**, 2745-2764 {Appendix D}
7. R. Cognet and E. Greenspan, "Minimizing Fissile Content of Fuel Discharged from Once-Through Fast Reactors without Fuel Reprocessing," [Nuclear Technology](#), in preparation.
8. S. Qvist and E. Greenspan, "Design optimization of inherently safe liquid metal cooled breed & burn reactors," [Nuclear Technology](#), in preparation.
9. S. Qvist and E. Greenspan, "The ADOPT code for automated fast reactor core design," [Nuclear Engineering and Design](#), in preparation.
10. S. Qvist and E. Greenspan, "Neutronic limits of once-through breed & burn reactors," [Nuclear Engineering and Design](#), in preparation.

Conference papers:

11. F. Heidet and E. Greenspan, "Maximum Fuel Utilization in Fast Reactors without Chemical Reprocessing", [Proc. of Global'09](#), Paris, France, September 6-11 (2009)
12. E. Greenspan, F. Heidet, "Fast Reactor for Maximum Fuel Utilization without Chemical Reprocessing", [Proc. of ICENES2009](#), Ericeira, Portugal, June29 – July 3 (2009) {Appendix A}
13. F. Heidet, R. Petroski and E. Greenspan, "Minimum burnup required for sustainable operation of fast reactors without recycling", [Proceedings International Conference on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities FR09](#), Kyoto, Japan, 7-11 December 2009
14. F. Heidet and E. Greenspan, "Breed-and-burn depleted uranium in fast reactors without actinides separation", [PHYSOR 2010, Advances in Reactor Physics to Power the Nuclear Renaissance](#), Pittsburgh, Pennsylvania, USA, May 9-14, 2010
15. C. Di Sanzo, J. Cohen, T. Cisneros, B. Ludewigt and E. Greenspan, "An Accelerator Driven

[Energy Multiplier for Doubling Uranium Ore Utilization,” Accelerator Applications Topical Meeting, Knoxville, TN, April 3-7, 2011 {Appendix H}](#)

16. S. Qvist, F. Heidet and E. Greenspan, “Reactivity Feedbacks of the UCB Breed & Burn Reactor,” Transactions American Nuclear Society, June 2011.
17. F. Rubio, D. Anderson, A. Dixon, J. Doojhibulpol, C. Montegrando, K. Tirohn and E. Greenspan, “Sodium Cooled Reactor for Increasing Fuel Utilization without Reprocessing”, Transactions American Nuclear Society, November 2011.
18. S. Qvist and E. Greenspan, “Inherent Safety of Minimum Burnup Breed-and-Burn Reactors,” [proceedings of ICAPP-2012](#), Chicago, IL, June 24-28, 2012. {Appendix I}
19. C. Di Sanzo and E. Greenspan, “Search For Minimum Volume of Breed and Burn Cores,” [Proceedings of the ICAPP-2012](#), Chicago, IL. June 24-28, 2012. {Appendix F}

Dissertations

20. R. Cognet, “Neutronic Study on the Minimization of the Waste from GEN-IV Fast Reactors Operating on the Once-Through Fuel Cycle,” Final Project for INSTN (French School of Nuclear Engineering), December 2009. 68 pages
21. S. Qvist, “Reactivity Coefficients of the Berkeley Traveling-Wave Reactor”, Master Project Submitted to the Royal Institute of Technology, Stockholm, Sweden, June 2010. 95 pages
22. F. Heidet, “Maximum Fuel Utilization in Advanced Fast Reactors without Actinides Separation”, PhD (UC Berkeley), September 2010. 206 pages
23. S. Gonzalez, “Minimum Burnup Pebble-Bed Breed-and-Burn Reactor,” MSc Project (UC Berkeley), expected December 2012.
24. S. Qvist, “Design of Passively Safe Breed-and-Burn Fast Reactors,” PhD (UC Berkeley), expected May 2013.
25. C. DiSanzo, “Breed-and-Burn Reactors – New Design Options and Waste Minimization,” PhD (UC Berkeley), expected December 2013. [Part A: C. DiSanzo, F. Heidet and E. Greenspan “Assessment of waste characteristics of Breed and Burn reactors,” November 2012. {Appendix J}](#)

Invention Disclosure

26. S. Qvist, “Nuclear Fuel Assembly Reactivity Control System,” US Provisional Patent Application Serial number 61/676,881, Filed July 27, 2012.

Other papers and presentations:

27. E. Greenspan, “Uranium Utilization in Breed and Burn Reactors,” Technical Memo prepared for the Director of Nuclear Energy, DOE, March 31. 2010. 3 pages
28. E. Greenspan, “Physics of Breed-and-Burn Nuclear Reactors”, Presentation given at Reactor Physics Summer School, University of California, Berkeley, June 14, 2010.
29. E. Greenspan, “Recommendations for Enabling Technologies to Consider,” Technical Memo prepared for the Director of Nuclear Energy, DOE, July 25, 2010. 3 pages
30. D. Anderson, A. Dixon, J. Doojhibulpol, C. Montegrando, F. Rubio, K. Tirohn and E. Greenspan, “Sodium Cooled Reactor for Increasing Fuel Utilization without Reprocessing”, A Technical Report for ANS Design Competition, 2011.
31. E. Greenspan, “Promising Directions for Nuclear Energy,” Presentation given at Cornell

University, May 6, 2011.

32. E. Greenspan, "Material Enablers for Breed-and-Burn Reactors," Presentation given at the 5th Annual Asia-Pacific Nuclear Energy Forum on Materials for Nuclear Applications, Berkeley, CA, June 22, 2011.

APPENDICES

- Appendix A: E. Greenspan, F. Heidet, "Fast Reactors for Maximum Fuel Utilization without Chemical Reprocessing", Proc. of ICENES2009, Ericeira, Portugal, June29 - July 3 (2009)
- Appendix B: F. Heidet and E. Greenspan, "Large Breed-and-Burn Core Performance", Accepted for publication in Nuclear Technology.
- Appendix C: F. Heidet and E. Greenspan, "Neutron Balance Analysis for Sustainability of breed and Burn Reactors", Accepted for publication in Nuclear Science and Engineering, 20 September 2011.
- Appendix D: E. Greenspan, "A Phased Development of Breed-and-Burn Reactors for Enhanced Nuclear Energy Sustainability," Sustainability 2012, **4**, 2745-2764
- Appendix E: F. Heidet and E. Greenspan, "Feasibility of Lead Cooled Breed and Burn Reactors", Progress in Nuclear Energy, Vol. 54, pp. 75-80, 2012.
- Appendix F: C. Di Sanzo and E. Greenspan, "Search For Minimum Volume of Breed and Burn Cores," Proceedings of the ICAPP-2012, Chicago, IL. June 24-28, 2012.
- Appendix G: E. Greenspan and F. Heidet, "Energy Sustainability and Economic Stability with Breed and Burn Reactors", Progress in Nuclear Energy, **53** (September 2011), pp. 794-799
- Appendix H: C. Di Sanzo, J. Cohen, T. Cisneros, B. Ludewigt and E. Greenspan, "An Accelerator Driven Energy Multiplier for Doubling Uranium Ore Utilization," Accelerator Applications Topical Meeting, Knoxville, TN, April 3-7, 2011
- Appendix I: S. Qvist and E. Greenspan, "Inherent Safety of Minimum Burnup Breed-and-Burn Reactors," proceedings of ICAPP-2012, Chicago, IL, June 24-28, 2012.
- Appendix J: C. DiSanzo, F. Heidet and E. Greenspan, "Assessment of waste characteristics of Breed and Burn reactors," November 2012.

Fast reactors for maximum fuel utilization without chemical reprocessing

E. Greenspan and F. Heidet

Department of Nuclear Engineering, University of California, Berkeley, CA, 94720, USA

Abstract

This study assesses the feasibility of designing metallic fuel fast reactor cores to achieve burnups that are significantly higher than 100 GWd/tHM without use of chemistry-based process for fuel recycling. The functions of the recycling process include removal of gaseous and volatile fission products, mitigation of radiation damage effects in the fuel, addition of depleted uranium or thorium fuel makeup and clad replacement. This preliminary study indicates that it might be possible to achieve close to 600 GWd/tHM. Metallic uranium based fuel offers the highest possible burnup. This simplified recycling scheme is expected to (1) greatly increase the uranium ore utilization; (2) significantly reduce the inventory of plutonium and TRU that need be disposed of in a HLW repository per unit of electricity generated; (3) Likewise for the decay heat and spontaneous neutrons emission rate; (4) reduce the fuel cycle cost and, hence, improve the economics of fast reactors; (5) improve the proliferation resistance of fast reactors, and (6) enable sooner deployment of fast reactors. This recycling scheme is also highly proliferation resistant.

Keywords: Fast reactors; recycling; sustainability; proliferation-resistance

1. Introduction

The attainable burnup in fast reactors is limited by the mechanical integrity of the fuel rods that is constrained by, primarily, (a) radiation damage to the clad; (b) fuel swelling and hardening due to solid fission products accumulation; and (c) gaseous fission products pressure buildup. With presently available structural materials that are compatible with liquid metal coolants the radiation limit constraint is approximately 200 dpa. For relatively low fissile content metallic fuel cores this corresponds to an average fuel burnup (BU) in the vicinity of 100 GWd/tHM. This implies that for the introduction of fast reactors to make sense, the fast reactors discharged fuel needs to be recycled. For this reason the deployment of fast reactors is associated with the deployment of commercial reprocessing capability. However, near-term deployment of commercial reprocessing plants in the USA is presently disputable because of technology maturity, economic viability and proliferation resistance concerns.

The objective of this study is to perform a very preliminary assessment of the feasibility of designing fast reactor cores and fuel management to achieve burnups that are significantly higher than 100 GWd/tHM without use of chemistry-based process (including electro-chemical processes) for fuel recycling. The focus of this feasibility study is to estimate the burnup level that might be possible to achieve using multi-recycling without chemical processing and to perform a preliminary evaluation of the implications such a recycling approach may have on the fuel utilization and nuclear waste.

The recycling approach considered is described in Section 2 followed by a description, in Section 3, of the fast reactor cores examined and the study methodology. The attainable burnup results are summarized in Section 4 for both a single-batch and a multi-batch fuel

management scheme. Section 5 presents the discharge fuel composition and fuel mass balance and discusses fuel cycle and waste-related implications of multi-recycling in fast reactors without chemical processing. Other possible implications of multi-recycling without reprocessing are discussed in Section 6 that is followed with conclusions in Section 7.

2. Recycling approaches

The general approach assumed for this study is to use a physical rather than chemical process for recycling the fast reactor fuel. The purpose of the physical fuel process is to de-clad the discharged fuel, remove the gaseous and volatile fission products, relieve radiation and burnup induced defects in the fuel, add fuel makeup, refabricate fuel pellets or slugs, and load them into a new clad.

For oxide fuel the fuel de-cladding and gaseous plus volatile fission product removal can be achieved using the AIROX process [1] such as the DUPIC process [2] the feasibility of which was demonstrated by Korea for recycling fuel from PWR to HWR. The AIROX process is a dry process that involves oxidation of the irradiated UO_2 fuel by O_2 atmosphere at 400°C to U_3O_8 . The U_3O_8 is reduced back to UO_2 by exposure to H_2 at 600°C . When this process is repeated several times the fuel pellets decompose to a fine powder out from which it is possible to remove 100% of T, C, Kr, Xe and I, 90% of Cs and Ru and 75% of Te and Cd.

Metallic fuel offers a better neutron economy in fast reactors than oxide fuel and, hence, a higher discharge burnup. As is, the AIROX process is not directly applicable to metallic fuel. An even simpler process could possibly be applied to a metallic fuel, like the binary alloy U-Zr(10) or ternary alloy TRU-U-Zr(10) also referred to as the “IFR” type fuel. After the fuel discharge from the fast reactor and adequate cooling time, the fuel is uncladded and melted. The gaseous and volatile fission products (FP) will get out (and be captured) in the process. According to experimental data obtained in the EBR-II Fuel Cycle Facility [3], nearly 100% of the following gaseous and volatile fission products are expected to be removed during the melting phase – Br, Kr, Rb, Cd, I, Xe and Cs. In addition, nearly 95% of the following fission products are expected to be removed during, the so called, “melt refining” phase – Sr, Y, Te, Ba and Rare Earths. 95% of the americium and thorium are expected to be removed as well. After adding depleted (or natural) uranium or spent fuel from LWR to make up for the amount of Heavy Metal (HM) that has been fissioned, the resulting melt will be cast into new fuel slugs that will be inserted into a new clad and fabricated into new fuel assemblies that will provide the fuel loading for a subsequent recycling. No chemical separation is involved in this recycling operation.

Neither of the above recycling processes can partition plutonium or any other actinide from the fuel. For this reason, and also because the recycled fuel is seeded with relatively large amount of radioactive fission products in addition to all the actinides, this recycling approaches are expected to be highly proliferation resistant. They are also expected to be less expensive than any of the chemistry-based processes.

3. Study methodology

3.1. Fast reactor cores examined

Two fast reactor core models are examined in order to get an upper bound on the possible attainable burnup – an infinite core to provide an absolutely upper bound estimate, and a

large finite core to provide a more realistic upper bound estimate.

Three fuel types are examined: oxide and metallic fuels based on ^{238}U as the primary fertile isotope and metallic fuel based on thorium. Transuranium (TRU) isotopes from LWR discharged fuel are used for the initial fissile fuel loading. The oxide fuel is assumed to be $\text{UO}_2\text{-TRUO}_2$; its density is 10.8 g/cm^3 that is 95% of the theoretical density. The metallic uranium fuel is assumed to be a ternary metallic alloy U-TRU-Zr, with 10wt% Zr, also referred to as an IFR type fuel; its density is taken as 15.85 g/cm^3 . As the metallic thorium fuel crystalline structure is FCC, it is more stable than uranium and does not need to be alloyed with zirconium. The thorium density assumed is 11.65g/cm^3 ; somewhat smaller than that of the metallic uranium fuel. The clad material is the ferritic-martensitic steel HT-9 and the coolant is lead-bismuth eutectic [4]. Using sodium coolant will result in only a few percent smaller attainable burnup.

Table 1 gives the volume fraction of the fuel, clad and coolant assumed for the three infinite cores analyzed; the thorium and uranium metallic fuels have the same volume fractions.

Table 1.

Volume fraction of the core infinite core constituents

Component	Oxide fuel	Metallic fuels
Fuel	46.84%	39.04%
Gap	5.20%	13.01%
Clad	22.90%	22.90%
Coolant	25.05%	25.05%

The TRU composition used, is given in Table 2, it is typical to that recovered from LWR spent nuclear fuel that underwent 50 GWd/tHM and cooled for 10 years and taken from [4].

Table 2.

TRU composition obtained from LWR fuel discharged at 50 GWd/tHM and cooled for 10 years [4]

Isotope	Weight %
^{237}Np	6.663
^{238}Pu	2.758
^{239}Pu	48.813
^{240}Pu	23.056
^{241}Pu	6.949
^{242}Pu	5.050

^{241}Am	4.669
^{242}Am	0.019
^{243}Am	1.477
^{242}Cm	0
^{243}Cm	0.005
^{244}Cm	0.498
^{245}Cm	0.038
^{246}Cm	0.006

The finite core modelled is described in Figure 1 and Table 3. The core is designed to deliver 4000 MW_{th} with an average power density of 300 W/cm³. A large core was selected for this analysis as it has a relatively low neutron leakage probability and will therefore represent an upper bound on the burnup attainable from a finite core. There are no blanket elements in this core.

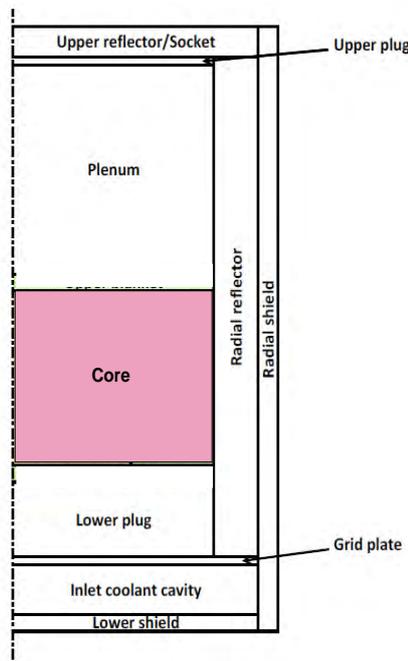


Figure 1. Vertical cut through the large core model used for the analysis

The fuel is U-TRU-Zr(10wt%) and the TRU composition is the same as in Table 2. The initial critical TRU concentration is found to be 10.3wt%. The coolant is lead-bismuth eutectic (LBE) and the dimensions and compositions of each of the regions modelled for the finite core analysis are specified in Table 3. The composition of each of the regions is modeled as homogenous.

Table 3.

Dimensions and composition of the regions modeled for the finite core analysis

Region	Height (cm)	Thickness (cm)	Material (Volume %)
Upper reflector	34.93	242.20	50% HT9- 50% Na
Upper end plug	2.54	201.36	22% HT9 - 78% Na
Plenum	250.00	201.36	22% HT9 - 28% Na
Core	209.36	142.38	37.5% Fuel - 22% HT9 - 28% Na
Lower end plug	90.42	201.36	22% HT9 - 78% Na
Grid plate	5.18	242.20	50% HT9 - 50% Na
Coolant inlet	60.00	242.20	22% HT9 - 78% Na
Bottom shield	20.00	242.20	43.1% B ₄ C - 29.7% HT9 - 27.2% Na
Radial reflector	592.96	40.84	50% HT9 - 50% Na
Radial shield	713.07	20.50	43.1% B ₄ C - 29.7% HT9 - 27.2% Na

At Beginning of Life (BOL), the axial and radial neutron leakage out of the core into the reflectors is, respectively, 2.85% and 4.3%.

The initial gap is 25% of the inner clad volume making an effective fuel smear-density of 75%. This provides for a fuel swelling of about 33%, at which point approximately 75% the volatile and gaseous fission product are released out of the fuel into the fission gas plenum. This phenomenon is modeled in the simulation: 75% of the total volatile and gaseous fission products are continuously removed from the core into the plenum.

3.2. Computational methodology

The core simulation is done using MCNP5 for the neutronics analysis and ORIGEN2.2 for the depletion analysis; the two codes are interfaced by the MOCUP module that has been upgraded at UC Berkeley. 34 actinides and 89 fission products are tracked with MCNP5; they account for more than 99.9% of the total absorption and fission probability of the fuel. An extended fuel composition is used for the depletion analysis with ORIGEN2.2; the total number of isotopes ORIGEN accounts for is ~1000.

The burnup analysis is performed assuming recycling takes place when the fast neutron fluence – the fluence of neutrons with energy exceeding 0.1 MeV, approaches 4×10^{23} n/cm². At that time, LWR spent nuclear fuel cooled for 10 years is added to the fuel to make up for the gaseous and volatile fission products that got out during the cycle.

4. Attainable burnups

4.1. Infinite cores

Figure 2 shows the k_{∞} evolution with burnup for the infinite cores fuelled with metallic or oxide fuel. The burnup results reported for the infinite cores correspond to the power density of the ENHS core [4] which is a low power density design. Somewhat higher burnups are expected for a higher power density cores. These results correspond to a “single-batch” burnup; that is no fuel shuffling is assumed and the excess reactivity is supposedly taken care of by the control system of the reactor. This is not the conventional way for operating fast reactors but used here only for comparison of the reactivity evolution for metallic versus oxide fuels. The maximum attainable burnup for a single-batch fuel management corresponds to the burnup for which the value of k_{∞} drops below unity.

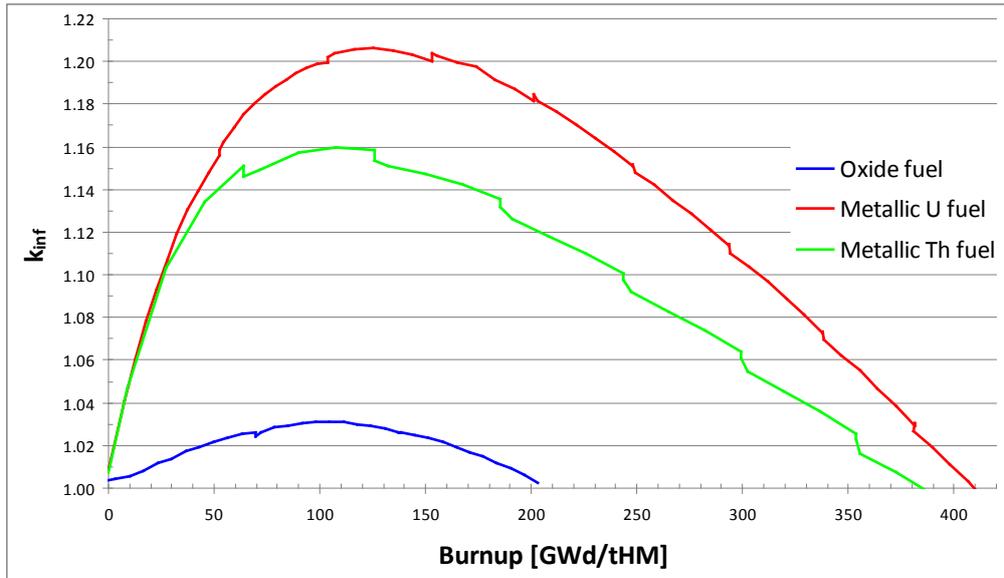


Fig. 1. k_{∞} evolution for infinite cores loaded with either uranium oxide, metallic uranium or metallic thorium fuel and operating in a single-batch fuel management with recycling without chemical processing. Gaseous and volatile fission products are partially removed and LWR spent nuclear fuel is used as makeup.

The maximum attainable burnup in a single-batch fuel management scheme is somewhat over 20% for oxide fuel versus 42% for metallic uranium fuel. This is due to the larger breeding ratio of the metallic fuelled core and is consistent with the initial TRU loading required for the two core types – 9.67 weight % for the metallic fuelled core versus 15.16 weight % for the oxide fuelled core. This large difference in the initial TRU loading is due to the differences in the HM loading and in the spectra – the spectrum of the oxide fuelled core being softer. The number of oxygen nuclei in the oxide core is significantly larger than the number of zirconium nuclei in the metallic fuelled core and the oxygen is more effective than zirconium in neutron slowing down.

The maximum burnup attainable from the single-batch metallic thorium core is 38% – this is 8% lower than for metallic uranium core. Its initial fuel composition is natural thorium with 13 weight % TRU. The maximum value k_{∞} gets to with burnup is 1.16 versus 1.21 for the metallic uranium fuel. This is because the η value of ^{233}U is smaller than that of, primarily, ^{239}Pu , in the hard spectrum of the systems examined.

Table 4 gives information on the neutron balance in the infinite metallic uranium fuelled core

and its variation from BOL to EOL. The fraction of neutrons absorbed in the fuel, clad and coolant stays approximately constant from BOL to EOL. However, as shown in Table 5, at EOL about 27% of the neutrons absorbed in the fuel are absorbed in the fission products.

Table 4.

Fraction of neutrons absorbed in the core constituents of the infinite metallic uranium fuelled core

Constituent	Neutron absorption fraction	
	BOL	EOL
Fuel	96.32%	96.90%
Gap (Na)	0.37%	0.00%
Cladding	2.78%	2.60%
Coolant	0.53%	0.50%

Table 5 compares the neutron balance in the thorium and uranium based metallic fuelled cores. A much larger fraction of neutrons are absorbed, at EOL, in the fission products of the metallic uranium fuel than of the thorium fuel, even for the same burnup of 38%. At this burnup the macroscopic absorption cross-section of the heavy metal of the thorium fuel is only 7% smaller than that for the metallic uranium fuel. On the other hand, the macroscopic absorption cross-section of the fission products is 43% smaller for the thorium fuel than for the metallic uranium fuel. This difference is primarily due to the different fission products yield; ^{233}U fissions yield slightly more volatile/gaseous fission products than the fission of ^{239}Pu .

Table 5.

Comparison of the fractional neutron absorption in the fuel isotopes for the infinite metallic uranium and metallic thorium cores; volatile & gaseous FP are continuously removed

Material	Metallic uranium			Metallic thorium	
	BOL	BU=38%	BU=42%	BOL	BU=38%
^{232}Th				53.8%	35.7%
^{233}U				0%	37.4%
^{238}U	59.2%	33.3%	31.3%	0%	0%
^{239}Pu	24.6%	32.8%	31.1%	28.7%	1.1%
^{240}Pu	4.2%	5.3%	5.3%	4.8%	2.1%
^{241}Pu	4.4%	1.2%	1.3%	5.1%	0.6%
FP	0%	23.2%	27.0%	0%	15.6%

The discharge burnup of any of the cores examined above can be significantly increased by

using multi- rather than single- batch fuel management – as commonly done in all commercial reactors. Figure 3 gives the upper bound on the attainable burnup using multi-recycling with limited processing in metallic uranium fueled core (with initial TRU loading). This upper bound is obtained by considering an infinite system and assuming a fuel management scheme having “infinite” number of batches; that is, by averaging the burnup-dependent k_{∞} plots (continuous and diamond lines) over the burnup, thus, obtaining the dotted line and “+” line of Figure 3. The upper bound on the attainable burnup is the burnup value at which the batch average k_{∞} equals 1.0; it is found to be 725 GWd/tHM – a 73% increase over the single-batch burnup. The corresponding attainable burnup for the infinite metallic thorium fuelled core, inferred from Figure 4, is 672 GWd/tHM – a 77% increase over the corresponding single-batch burnup. The amount of HM the quoted burnup values pertain to is the summation of the initial HM load and the HM from the makeup LWR spent fuel added each recycle.

The “bullet” and “diamond” lines in Figures 3 and 4 pertain to a mode of operation without any fuel processing. Although not realistic, the results are displayed to show that the recycling with limited processing mode-of-operation examined in this study provides some benefits also in terms of the amount of energy that can be generated per reactor, before the fuel has to be completely replaced.

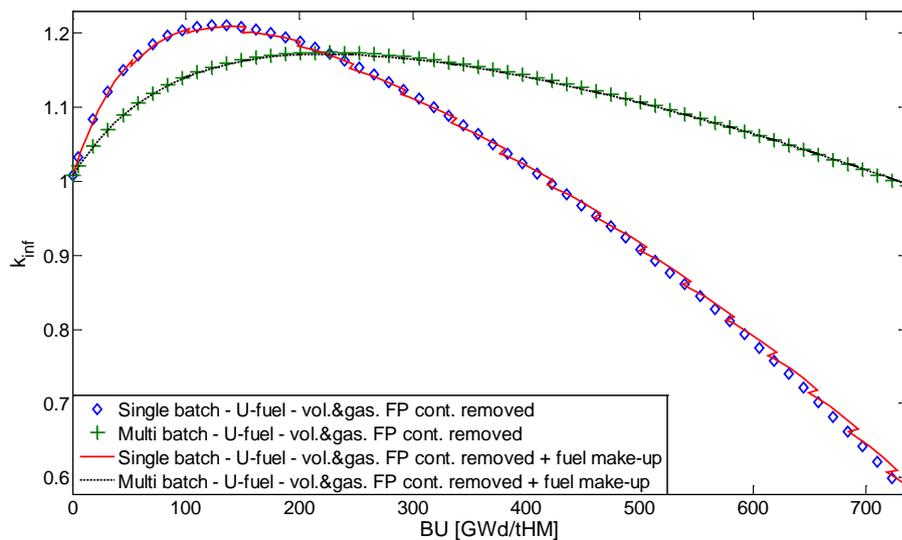


Fig. 3. k_{∞} evolution in infinite metallic uranium fuelled fast reactor cores operating in a single-batch mode (continuous and diamond lines) versus infinite-batch mode (dotted and “+” lines). “+” and “diamond” lines correspond to continuous core operation without recycling of any kind

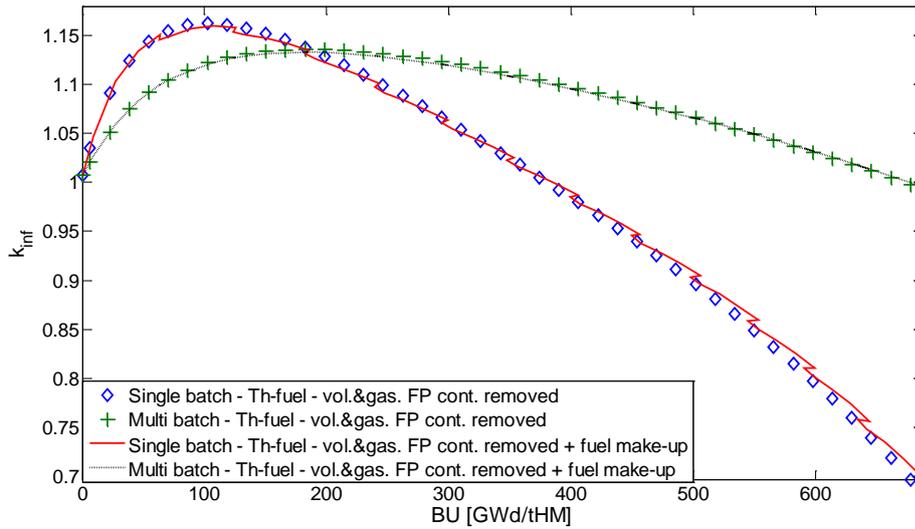


Fig. 4. k_{∞} evolution in infinite metallic thorium fuelled fast reactor cores operating in a single-batch mode (continuous and diamond lines) versus infinite-batch mode (dotted and “+” lines). “+” and “diamond” lines correspond to continuous core operation without recycling of any kind

4.2 Finite core

Figure 5 shows the burnup attainable from the 4000 MW_{th} finite core studied when the core operates with a single-batch burnup ($k_{\text{eff}} = 1.0$ limit for solid lines) and when assuming infinite-batch fuel management scheme (integral over BU of $k_{\text{eff}} = 1.0$ for dashed lines).

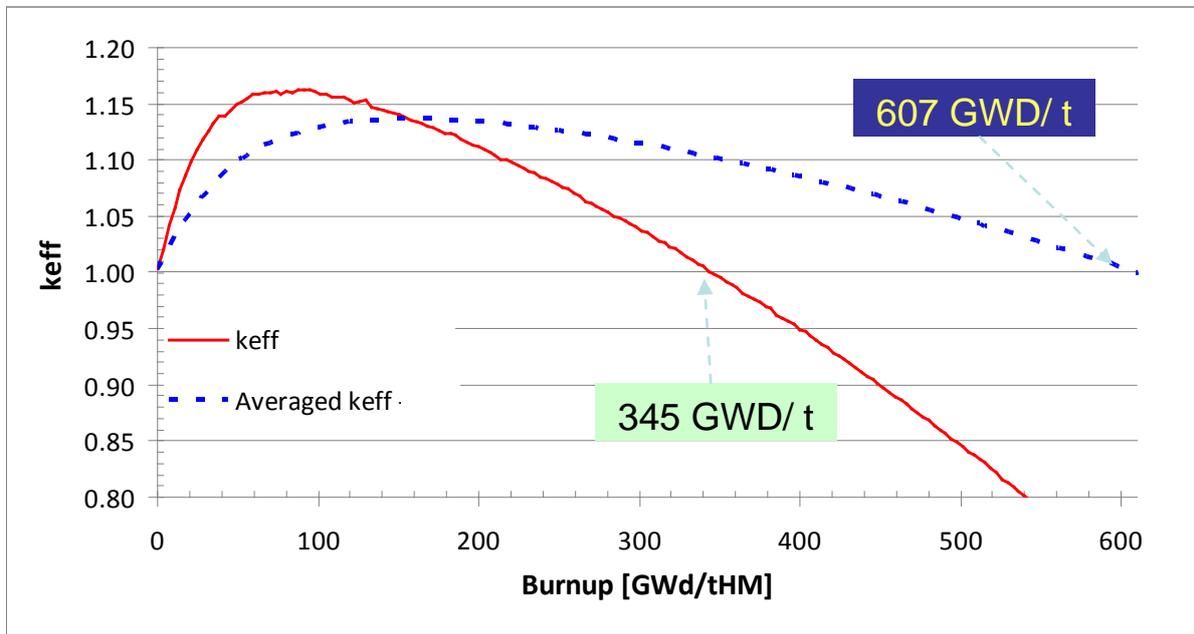


Fig. 5. k_{∞} evolution in the finite metallic uranium fuelled fast reactor core operating in a single-batch mode (solid lines crossing $k_{\text{eff}}=1.0$) versus infinite-batch mode (dotted lines crossing $k_{\text{eff}}=1.0$ line).

The attainable burnup is approximately 345 GWD/tHM for a single batch and 607 GWD/tHM for an infinite-batch fuel management scheme – only 16% smaller than of the

infinite core upper bound value. Had the infinite cores been analyzed at the power density assumed for the finite core, the difference in the burnup attainable from the finite and infinite cores would be slightly larger.

5. Discharged fuel composition

Table 6 compares our upper-bound estimates of fuel resource utilization and selected waste characteristics of the infinite metallic U and Th systems addressed in Figures 3 and 4 as well as of the finite 4000 MW_{th} fast reactor with that of a light water reactor that operates to 50GWd/tHM. It is found that multi-recycling in fast reactors with limited processing as hereby proposed offers a significant increase in the natural uranium ore utilization and a significant reduction in the amount of TRU that needs to be disposed of per unit of electricity generated, as compared to those attainable from LWR that operates on the once-through fuel cycle.

Consider, for example, the finite fast reactor. The amount of electricity it generates per given initial loading of TRU is [(328-100)/100=] 2.28 times larger than the amount of electricity generated in LWRs that would produce the required initial loading of TRU. Except for this amount of TRU, the fast reactor is fuelled with depleted uranium or spent fuel from LWR that can be considered as waste. Consequently, the introduction of the fast reactor with multiple recycling without chemical processing increases the natural uranium ore utilization relative to the once-through LWR by a factor of 3.28. The total amount of TRU discharged from this reactor is 44.3% that discharged from the reference LWR per unit of electricity generated. By further design optimization it is expected that the finite fast reactor could be designed to generate smaller amount of TRU per GW_eD. The lower limit on the amount of TRU discharged per unit of electricity generated in the uranium-fuelled fast reactors is 21% that discharged from the once-through LWR. The corresponding amount of TRU plus ²³³U discharged from the thorium-fuelled fast reactor is only 14.5% of that discharged from the LWR.

Table 6.

Fuel cycle characteristics of multi-recycling with no chemical separation versus once-through LWR

Characteristic	LWR	Reactor type		
		Infinite U	Infinite Th	4000MW _{th} U
Initial fissile fuel type	U	TRU	TRU	TRU
²³⁵ U or TRU wt. %	4.5	9.67	13	10.3
Discharge burnup (GWD/tHM)	50	725	672	607
<u>Natural uranium utilization</u>				
Relative amount of GW _e d generated per ton U _{nat} (%)	100	388	300	328
<u>left</u>				
Fraction of initial TRU (%)	-	56.6	5.1	67.8
Fraction of initial Pu (%)	-	61.5	4.3	75.3
Fissile/total Pu at BOL/EOL (%)	-/64.5	64.4/59.0	64.4/10.4	64.4/64.0
²³⁸ Pu/total Pu at BOL/EOL (%)	-/3.2	3.2/2.4	3.2/12.7	3.2/0.9
Fraction of initial ²³⁷ Np+ ²⁴¹ Pu+ ²⁴¹ Am+ ²⁴⁵ Cm (%)	-	22.3	5.2	21.2
<u>Discharged</u>				

TRU per GWD (g/GWD)	300	75.1	9.8	115.4
²³³ U per GWD (g/GWD)	-	-	42.8	-
Relative amount of TRU+²³³U per GW_eD (%)	100	20.9	14.5	44.3
²³⁷ Np+ ²⁴¹ Pu+ ²⁴¹ Am+ ²⁴⁵ Cm (g/GWD)	58.6	5.4	1.8	6.6
⁹⁰ Sr+ ⁹⁹ Tc+ ¹²⁹ I+ ¹³⁵ Cs+ ¹³⁷ Cs (g/GWD)	94.7	58.2	64.0	41.7
Decay heat 1 year after discharge (W/g TRU)	0.88	0.26	2.15	0.43
Neutron emission 1 year after discharge (n/s/g TRU)	1.0E+05	8.7E+04	6.5E+05	1.6E+05
<u>Radio-toxicity (m³ of water/MWD)</u>				
HM 1 year after discharge	4.52E+06	9.18E+05	5.48E+05	1.23E+06
HM 30 years after discharge	3.43E+06	6.72E+05	3.35E+05	7.65E+05
FP 1 year after discharge	1.57E+07	1.44E+06	2.36E+06	3.52E+06
FP 30 years after discharge	2.85E+06	3.14E+05	7.68E+05	9.12E+05
<u>Relative radio-toxicity per GW_eD (%)</u>				
HM 1 year after discharge	100	20.3	12.1	27.2
HM 30 years after discharge	100	19.6	9.8	22.3
FP 1 year after discharge	100	9.1	15.0	22.4
FP 30 years after discharge	100	11.0	27.0	32.0
<u>Relative decay heat per GW_eD (%)</u>				
1 year after discharge	100	6.1	6.7	15.8
30 years after discharge	100	13.2	17.9	20.6
<u>Relative spontaneous neutron yield per GW_eD(%)</u>				
1 year after discharge	100	18.2	17.6	28.1
30 years after discharge	100	21.4	22.0	31.5

Tables 7 and 8 compare the isotopic composition of, respectively, the TRU and U discharged from each of the reactors inter-compared in Table 6. The plutonium discharged from the thorium fuelled reactor has, as expected, the smallest fraction of fissile isotopes and the largest concentration of ²³⁸Pu and other even isotopes. The fraction of ²³⁹Pu plus ²⁴¹Pu in the Pu discharged from the finite fast reactor is comparable to that discharged from the LWR. The concentration of ²³⁷Np and its precursors – ²⁴¹Pu, ²⁴¹Am and ²⁴⁵Cm is the largest in the thorium-fuelled fast reactor, followed by the LWR; the uranium-based fast reactors have significantly smaller concentration.

Table 7.

Concentration, in weight percent, of the transuranium isotopes discharged from three of the fast reactors versus that of the fuel discharged from LWR ^a

	System			
	LWR	Infinite U	Infinite Th	Finite 4 GW _{th}
Burnup (GWd/tHM)	50	723	677	607
Wt. % of TRU	1.57	20.55	2.19	18.26
Np-237	5.03%	0.61%	7.60%	0.45%
Np-239	0.62%	0.03%	-	0.21%
Pu-238	2.25%	2.22%	9.24%	0.89%
Pu-239	52.21%	52.08%	3.34%	57.18%
Pu-240	18.86%	32.55%	36.22%	30.33%
Pu-241	13.98%	3.38%	4.21%	4.29%

Pu-242	4.51%	3.85%	19.82%	3.42%
Am-241	0.53%	3.11%	5.69%	0.78%
Am-242m	0.01%	0.22%	0.47%	0.06%
Am-243	0.91%	1.05%	6.77%	0.94%
Cm-242	0.17%	0.07%	0.14%	0.09%
Cm-244	0.83%	0.63%	4.72%	1.03%
Cm-245	0.08%	0.12%	1.01%	0.21%
Cm-246	0.01%	0.07%	0.68%	0.11%
Cm-247	-	0.01%	0.07%	0.01%
Cm-248	-	-	0.03%	-

^a The concentration of all TRU isotopes not given in this table is smaller than 0.01 weight %

Table 8.

Concentration, in weight percent, of the uranium isotopes discharged from the reactors inter-compared

	System			
	LWR	Infinite U	Infinite Th	Finite 4 GW _{th}
U-232	-	-	0.17%	-
U-233	-	-	61.86%	-
U-234	-	0.27%	26.60%	0.02%
U-235	1.37%	0.08%	6.21%	0.01%
U-236	0.69%	0.23%	5.17%	0.05%
U-238	97.95%	99.42%	-	99.92%

Figures 6 through 9 compare, respectively, the radio-toxicity of the actinides, the radio-toxicity of the fission products, the decay heat from the actinides and the spontaneous neutrons emission as a function of time after discharge from each one of the four reactors compared in Tables 6 through 8. The results are normalized per unit of thermal energy generated. A more realistic comparison would be per unit of electricity generated. Assuming that the fast reactors energy conversion efficiency is 40% versus 33% of the reference LWR, the fast reactor related curves shown in Figures 6 through 9 should be reduced by 33/40.

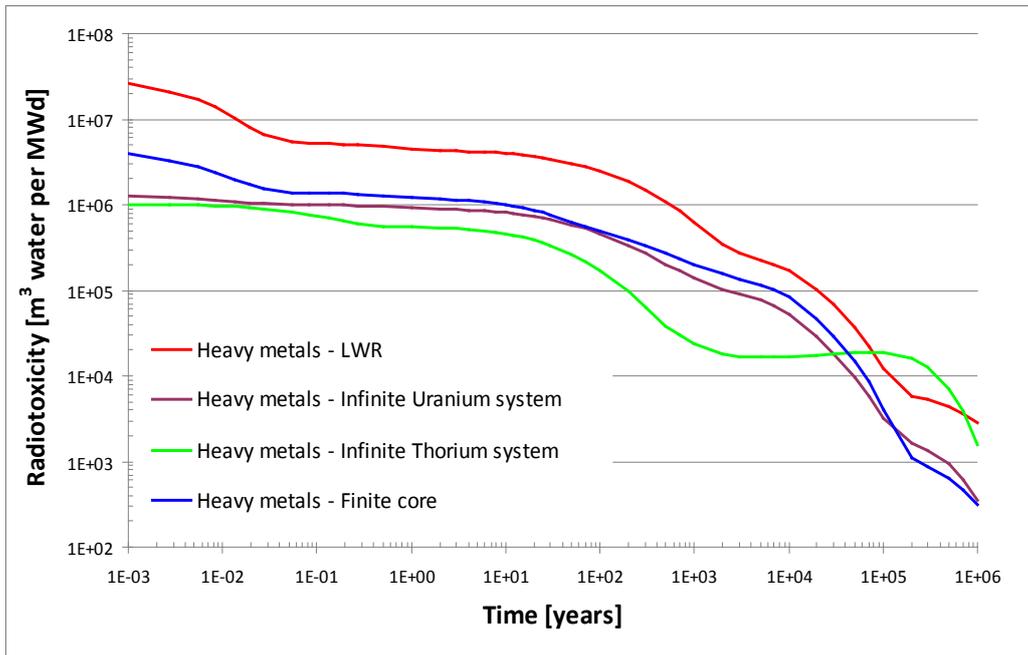


Fig. 6 Radio-toxicity of the actinides discharged from the fast reactors versus LWR, normalized per unit of thermal energy generated, as a function of time after discharge.

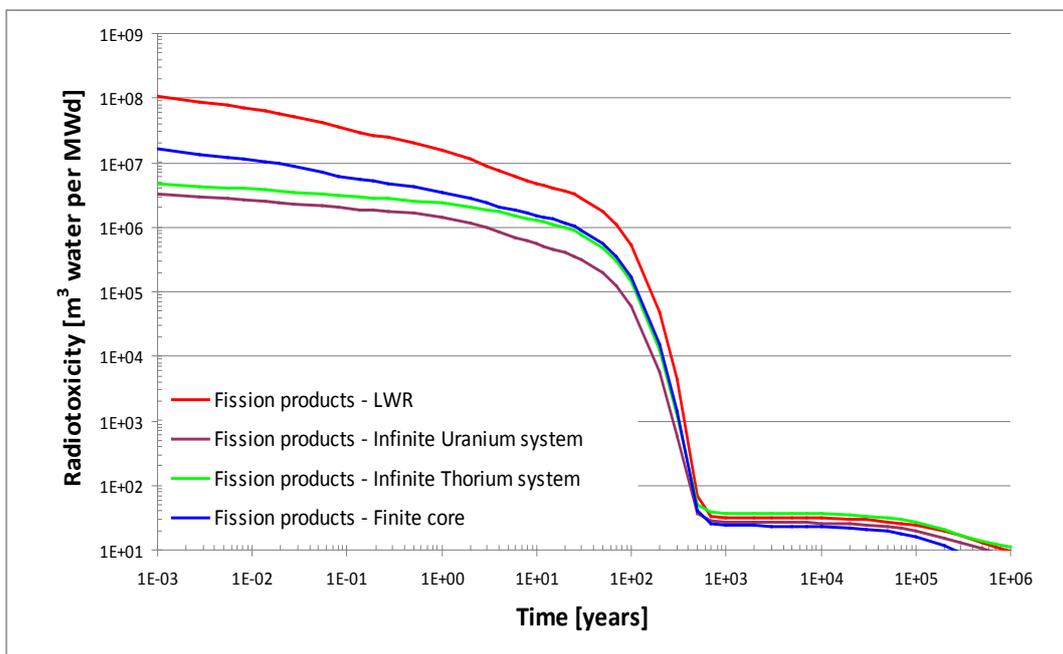


Fig. 7 Radio-toxicity of the fission products discharged from the fast reactors versus LWR, normalized per unit of thermal energy generated, as a function of time after discharge.

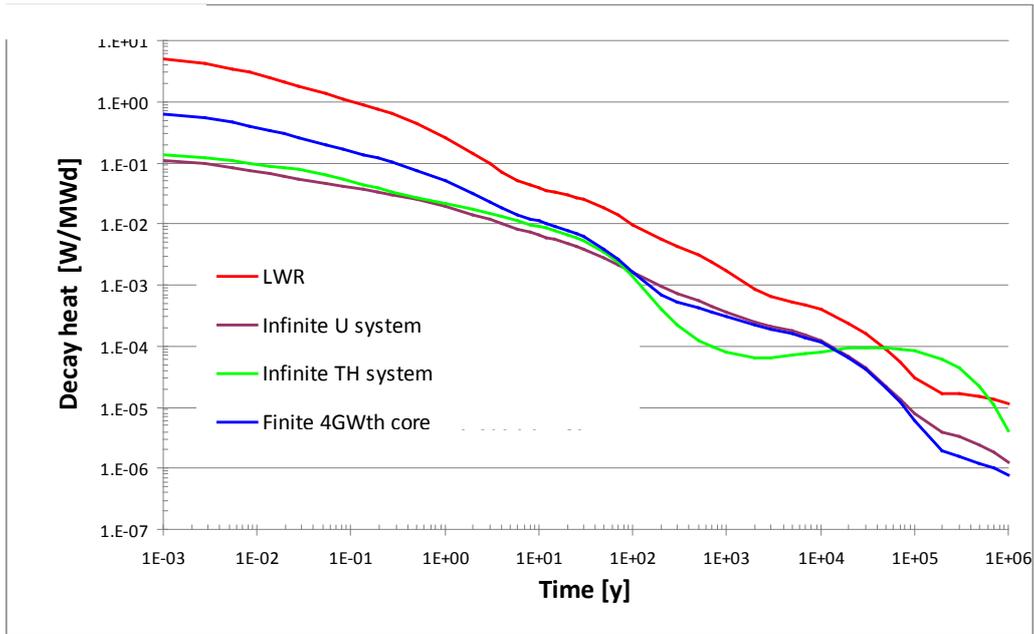


Fig. 8 Decay heat from the fuel discharged from the fast reactors versus LWR, normalized per unit of thermal energy generated, as a function of time after discharge.

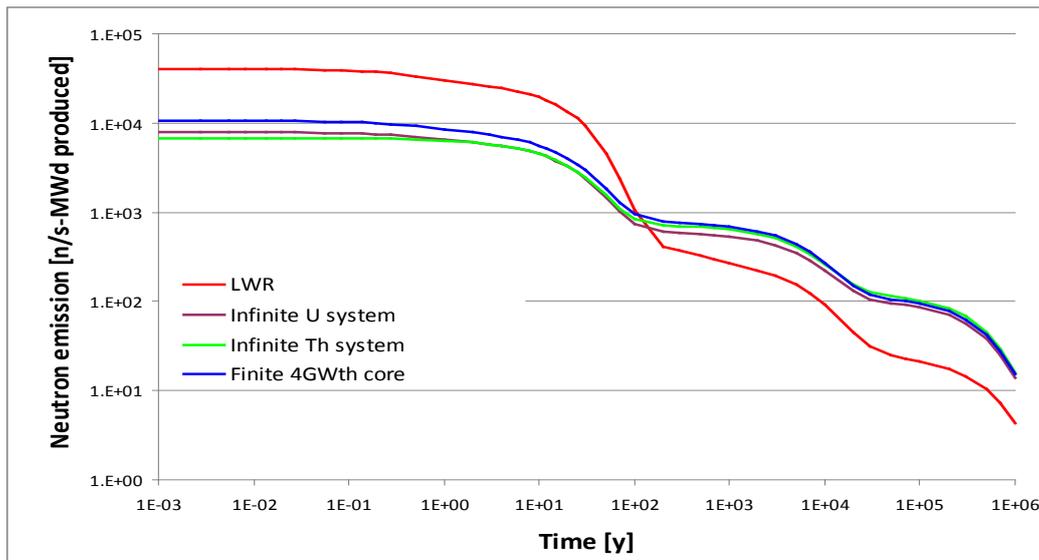


Fig. 9 Spontaneous neutrons emission from the fuel discharged from the fast reactors versus LWR, normalized per unit of thermal energy generated, as a function of time after discharge. Including primarily spontaneous fission neutrons and (α ,n) neutrons.

It is observed that the radio-toxicity of the FP from the fast reactors is initially one order of magnitude smaller than from the LWR until ~ 500 years following the fuel discharge, at which time most of the unstable FP decayed. After that, the differences in the radio-toxicity level become small. The radio-toxicity of the heavy metal discharged from the fast reactors is always $\sim 4-5$ times smaller than for the LWR discharged heavy metals except for the thorium fuelled fast reactor following $\sim 10^5$ years after discharge. The larger radio-toxicity at that time is mostly due to the build-up of ^{229}Th from the decay of ^{233}U the half-life of which is $1.59\text{E}+5$ years.

In the decay heat evolution (Figure 8), the deviation between the Th curve and the other curves is due to the same isotopes buildup mentioned above. Initially, there is more than one order of magnitude difference between the values obtained for the LWR fuel and for the other fuels, but this difference tends to decrease as time is increasing.

The neutron emission evolution from the discharged fuels displayed in Figure 9 exhibits a different behavior. The main contributors to spontaneous neutron emission are the curium and even plutonium isotopes. The three humps in the plots correspond each to the half-life of a major neutron emitter. The first hump is due to the decay of ^{244}Cm (18.1 y), the second to the decay of ^{240}Pu (6564 y) and ^{246}Cm (4730 y) and the last to the decay of ^{242}Pu ($3.733\text{E}+5$ y). Being rich in ^{244}Cm , the LWR has initially a higher spontaneous neutrons yield, but after ~100 years it drops to a level smaller than that of the other fuels because of the lower amount of the other primary spontaneous neutron emitters.

6. Discussion

The multi-recycling in fast reactors with limited processing is expected to reduce the fuel recycling cost and, hence, improve the economics of fast reactors and thus enable sooner deployment of fast reactors the technology of which is pretty much established¹ because (a) either no (in case of use of enriched U or weapons excess Pu) or a relatively simple process (maybe UREX +1A or COEX) is required for processing the LWR spent fuel to extract TRU without partitioning plutonium or any MA; and (b) no chemical reprocessing at all is required for recycling the fuel in the fast reactors. The resulting nuclear energy system will also be more proliferation resistant than an energy system that requires removal of all fission products and even partitioning of Pu or MA and chemistry-based processing.

If TRU from LWR is to be used to initially fuel the fast reactors, the fast reactor cores will effectively provide for a safe and proliferation resistant “interim storage” of the TRU for many years while using this TRU inventory as a “catalyst” for the fissioning of depleted uranium or thorium. As the TRU inventory at the end of the multi-recycling campaign will be significantly smaller than the initially loaded inventory (See Table 6) these fast reactors would also function as effective TRU burners.

Implementation of multi-recycling in fast reactors with limited processing will provide ample time (dozens of years) for the technologists to develop more economically viable and more proliferation resistant processes for extraction of all (or most) of the solid fission products and will provide the US policy makers and legislators ample time to decide upon the preferred approach for closing of the nuclear fuel cycle. The fuel discharged from our fast reactors after multi-recycling with limited processing could then be further recycled to extract the extra energy value of the discharged fuel and further minimize the leftover high level waste.

7. Conclusions

This preliminary analysis indicates that it is feasible to recycle fast reactor fuel several times without removal of solid fission products and without resort to chemical processing. Uranium-based metallic fuel offers a significantly higher cumulative discharge burnup than oxide fuel. The upper bound estimate of this attainable burnup is 725 GWd/tHM. The burnup

¹ Although more cost effective designs are desirable

that could be achieved in realistic fast reactor core designs using the simplified recycling with limited processing is at least 600 GWd/tHM. Based on the preliminary assessment it appears that the simplified recycling scheme proposed is expected to (1) reduce the fuel cycle cost and, hence, improve the economics of fast reactors; (2) improve the proliferation resistance of fast reactors and their fuel cycle, and (3) enable sooner deployment of fast reactors.

It thus appears justified and desirable to embark upon a more detailed core design and multi-batch fuel management optimization study having the objectives of reliably quantify the maximum practical cumulative discharge burnup from the multi-recycling campaign in fast reactors with no chemical processing and its implications on fuel utilization and waste management. Feasibility assessment of processes capable of efficiently recycling discharged fuel without chemical separation is also recommended.

Acknowledgments

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Large Breed-and-Burn Core Performance

Florent Heidet and Ehud Greenspan

University of California Department of Nuclear Engineering

Berkeley, CA 94720-1730, USA

fheidet@anl.gov; gehud@nuc.berkeley.edu

Abstract

A sodium cooled fast reactor Breed and Burn (B&B) core and fuel cycle concept are proposed for achieving uranium utilization in the vicinity of 50% without separation of most of the fission products from the actinides. This core is to be fuelled with depleted uranium with the exception of the initial core loading that uses fissile fuel to achieve initial criticality. When the cladding reaches its radiation damage limit, the fuel is reconditioned using the melt-refining process and reloaded into the core. This fuel reconditioning continues until the fuel reaches the neutronic maximum attainable burnup. When a fuel assembly is discharged at its maximum attainable burnup it is replaced with a fresh depleted uranium assembly.

The maximum burnup attainable in a large 3000MW_{th} B&B core is found to be 57% FIMA. The discharged fuel characteristics such as the inventory of actinides, radiotoxicity and decay heat are one order of magnitude smaller, per unit of energy generated, than those of a LWR operating with the once-through fuel cycle.

It is also found that the minimum burnup required for sustaining the breed and burn mode of operation is 19.4% FIMA. The fuel discharged at this burnup has sufficient excess reactivity for establishing initial criticality in a new large B&B core. The theoretical minimum doubling-time for new core spawning is estimated to be ~10 EFPY; there is no need for any external fissile material supply beyond that required for the initial “mother” reactor.

A successful development and deployment of the B&B core along with fuel reconditioning could possibly provide up to 3000 years worth of the current global nuclear

electricity generation by using the depleted uranium stockpiles already accumulated worldwide. However, a number of important feasibility issues are yet to be resolved.

I. Introduction

Present day Light-Water-Reactors (LWRs) utilize only approximately 0.6% of the energy value of the amount of natural uranium mined for making the LWRs fuel. About 90% of the unused uranium is left over as depleted uranium and the rest (over 9%) as used nuclear fuel (UNF). The depleted uranium can, in principle, be utilized in fast reactors. However, a high uranium utilization cannot be achieved in a single campaign. The approach commonly envisioned for attaining high uranium utilization in fast breeder reactors (FBRs) is to recycle the fuel many times; when the mechanical integrity of the nuclear fuel elements that are irradiated in the reactor core deteriorates beyond a certain level, the fuel elements are to be discharged, reprocessed and recycled back to the reactor core. The fuel reprocessing operation involves removal of the structural material that is cladding the fuel, removal of most of the products of the fission reaction, addition of some depleted uranium to make up for the uranium that has been fissioned, fabrication of new fuel elements and reloading them into the reactor core for another irradiation cycle. Typically such fuel recycling has to be done approximately every 10% FIMA (Fissions per Initial Metal Atoms) in a relatively low fissile content hard spectrum FBR and up to 15% FIMA in a soft spectrum FBR. Although technically feasible, there is a significant objection in the US to fuel reprocessing due to economic viability and proliferation concerns. The proliferation concern is that the chemical processes that have been developed for the extraction of the fission products can be used, possibly after some modification, for extraction of plutonium from the fuel and this plutonium could be used for nuclear weapons.

Fast breeder reactors could, in principle, also operate without fuel recycling; that is, using a once-through fuel cycle as do all of the LWRs presently operating in the USA.

Although a discharge burnup of 10% to 15% FIMA is 2 to 3 times higher than that of contemporary LWRs, the uranium utilization from a once-through FBR is not significantly different from that of a once-through LWR because the uranium enrichment required to fuel the FBR is more than twice that required to fuel the LWR.

Nevertheless, it may be possible to realize a significant increase in the uranium utilization without fuel reprocessing using a special class of fast reactors, referred to as “breed-and-burn” (B&B) or “travelling wave” reactors, such as the TWR under development by TerraPower [1]. The unique feature of a B&B reactor is that it can breed plutonium in depleted uranium feed fuel and then fission a significant fraction of the bred plutonium, without having to reprocess the fuel. In order to initiate the chain reaction, the B&B core has first to be fed with adequate amount of fissile fuel such as enriched uranium (EU). Plutonium or TRU extracted from used nuclear fuel could also be used for the “starter”. Thereafter, the B&B core is capable of continued operation while being fed with depleted uranium only. Eventually, the uranium utilization will approach the fraction of the loaded uranium that has been fissioned.

The principles and concepts of B&B reactors have been proposed in the past; [2-9] is a partial list of references. In order to sustain the chain reaction in the B&B mode of operation it is necessary to fission at least 20% of the depleted uranium [10]. The experimental and demonstration fast reactors that operated in the past have proven that, in a relatively low fissile content hard spectrum core such as required for a B&B reactor, the HT-9 fuel clad can maintain its mechanical integrity up to 200 displacements per atom (DPA), corresponding to an average burnup of ~10% FIMA. It is likely that the fuel could have withstood higher burnup without losing its mechanical integrity but there is no experimental evidence that this, indeed, is the case. Moreover, a combination of development of improved structural materials, improved fuel materials and improved core design is likely to increase the attainable average

burnup to at least 20% FIMA – corresponding to ~500 peak DPA in the clad, without having to reprocess or re-fabricate the fuel.

Alternatively, it might be possible to establish the B&B mode of operation with limited fuel “reconditioning” – an approach being presently studied at the University of California, Berkeley [11-15]. The functions of the fuel re-conditioning are to remove part of the fission products, primarily the gaseous ones, and replace the fuel clad prior to fuel re-use in the reactor. It is to overcome material performance limits in a way that cannot be used to extract plutonium and that is, hopefully, not as expensive as conventional fuel reprocessing. The re-fabricated fuel can either be re-introduced into the reactor core for additional use, or be used as the “starter” fuel for a new core. The latter option, to be referred to as the “spawning” mode of operation, offers a significant savings in the amount of enriched uranium and, therefore, natural uranium that is required to deploy a fleet of B&B reactors.

The objective of the present study is to perform a preliminary neutronics design of a B&B core, estimate the maximum burnup depleted uranium feed fuel can accumulate in such a core with limited fuel reconditioning, assess the feasibility of spawning new B&B cores using fuel discharged at ~20% FIMA from operating B&B cores, study the B&B core performance sensitivity to different fuel shuffling schemes and compare its discharged fuel characteristics with those of more conventional reactor technologies. A detailed description of the B&B core and of the mode of operation, including the reconditioning process assumed, is provided in Section II. The performance of the B&B core during the transition period to an equilibrium state is discussed in Section III for two different types of starters – enriched uranium and LWR TRU in depleted uranium. Sections IV and V describe the performance of the B&B core at equilibrium, when it is designed to operate at, respectively, the maximum attainable and minimum required discharge burnup. The performance of the equilibrium core is discussed for a simple and an improved shuffling scheme. Section VI discusses the

possibility of spawning additional B&B core without need for additional enriched fuel. The feasibility of reducing the required enriched uranium inventory is discussed in Section VII. A preliminary thermal-hydraulic analysis is performed in Section VIII to validate the cooling ability of the B&B core. The general conclusions of the study are provided in Section IX.

II. Methodology

II.1. B&B core description

The B&B core concept studied in this paper is a sodium cooled fast reactor that uses the ternary metallic fuel U-Pu-Zr with 10 wt% zirconium; the fuel nominal density is 15.85 g/cm^3 and a smear factor of 75% is assumed to accommodate the fuel swelling with burnup. The cladding material, the ferritic-martensitic alloy HT-9, and the coolant have a density of respectively 7.529 g/cm^3 and 0.830 g/cm^3 , both at 800 K. The pitch-to-diameter ratio assumed is 1.11 – near the lower limit used in liquid sodium cooled reactors [16]. This value corresponds to a coolant volume fraction of 26.4% in an infinite hexagonal lattice of fuel rods. The core average coolant volume fraction provided in Table 1 is larger as it accounts for the coolant that occupies the space between fuel assemblies and the space provided for several reactivity control assemblies. The exact number of required control assemblies has not been determined so the 28% core-average coolant volume fraction is an approximation. As the coolant volume in the control assemblies location is smeared over the entire core volume, the effect of the relatively large localized coolant volume on different phenomena in the vicinity of the control assembly locations is not accounted for. The cladding thickness was assumed to be 0.67 mm, corresponding to ~10% of the inner cladding diameter which is consistent with information in the IAEA database [16].

The core layout is presented in Figure 1. The inner radial half of the core is initially loaded with enriched uranium or fuel made of depleted uranium and TRU recovered from LWR UNF that has been cooled for 10 years. The starter fuel is radially surrounded by a

depleted uranium blanket that is surrounded by a thin radial reflector followed by a shield. The reactor dimensions, partially derived from the Argonne National Laboratory (ANL) Advanced Breeder Reactor (ABR) design [17], and material composition of the various components are given in Table 2. The depleted uranium radial blanket volume is equal to the enriched fuel volume.

The active core height is 209.36 cm and the core inner and outer diameters are 16.96 cm and 401.48 cm, respectively. The enriched fuel and radial blanket are divided each into four equal volume concentric burnup regions each of which is divided into three equal volume axial burnup regions. Each of the eight radial regions corresponds to a fuel batch. The assembly located along the core axis is empty in order to have an even number of fuel assemblies per batch arranged with azimuthal symmetry.

A large core was selected for this study since it has a relatively low neutron leakage probability and will therefore feature a favorable neutron economy, enabling to determine the upper bound on the burnup attainable from a finite core and lower bound on the minimum burnup required for sustaining the breed and burn mode of operation. The total core power is 3000 MW_{th}, corresponding to an average power density of 112.5 W/cm³. The peak fuel and cladding temperatures established in the core using this power level are estimated in Section VIII. A reduced power level may be required for certain fuel shuffling schemes so as not to exceed the maximum acceptable material temperatures.

II.2. Fast reactor modeling

As the neutrons mean free path in fast reactors is larger than the pitch of a fuel rod, it is common to represent the core by a homogenized composition; the fuel, bonding material, structural material and coolant are mixed together using their respective volume fraction. As the fuel burnup increases to 1-2% FIMA, the fuel swells and closes the gap, squeezing most of the bonding sodium out from the active core region into the fission gas plenum. Since the

targeted burnup of the B&B core studied is significantly larger than 2% FIMA, the bonding sodium is not modeled as a constituent of the active core regions.

Due to computation time limitations it was not possible in this study to perform burnup analysis for every fuel assembly. Instead, the core is modeled as consisting of concentric cylindrical regions, each representing a fuel batch. The error introduced by this assumption is illustrated in Figure 2 for the B&B core during the transition period: the multiplication factor (k_{eff}) evolution calculated for a cylindrical core model is compared against that calculated for an assembly-level core model in which every fuel, reflector and shield assembly is explicitly modeled. It is observed that the error between the two core models fluctuates from 0.4% to -0.4% $\Delta k/k$. This error is due to the radial power distribution differences between the two models: in the cylindrical model, all the assemblies of a given batch are depleted with the same power level and are assumed to have the same composition while in the assembly-level model, a different power level and composition is assigned to each assembly. Additional characteristics of the two core models are compared in Table 3. Apart from the radial power peaking factor, all the characteristics are in good agreement. It is concluded that it is sufficient to use the cylindrical model for getting a preliminary assessment of the B&B core performance with acceptable accuracy.

MCNP5 1.40 and ORIGEN2.2 coupled through MOCUP are used for this study. Two thousands neutron histories per cycle and 180 active cycles are used to obtain k_{eff} with a fractional statistical error of 50 ± 15 pcm. The (n,f), (n, γ), (n,2n), (n,3n) and (n, α) effective one group cross-sections are generated for the thirty-three actinide isotopes and ninety-nine fission product isotopes given in Table 4. All the cross-section data used by MCNP5 are based on the original ENDF/B-VI.8 libraries. These libraries were previously generated for different temperatures using NJOY [18], to account for the temperature dependence of Doppler broadening of resonances.

II.3. Mode of operation

The breed and burn mode of operation assumed for this study is similar to that which is presently pursued by TerraPower, LLC [1]. Except for the initial critical fissile fuel loading (i.e. the starter), the core is to be fuelled with depleted uranium only. When the fuel reaches its radiation damage limit, it is reconditioned and reloaded into the core. The fuel is being recycled this way as long as the core criticality can be maintained in the breed and burn mode of operation. A thorough evaluation of the neutron balance for the B&B core is described in [10]. At the end of a burnup cycle it is assumed that the fuel batch having the highest average burnup is discarded and the seven other fuel batches are discharged, reconditioned and reloaded into the core according to a pre-determined shuffling pattern. Practically, as the recycling process is likely to take more than one year, the fuel discharged at the end of cycle “n” will be reloaded at beginning of cycle “n+2”. It was found that cooling the discharged fuel for the duration of a cycle has a negligible effect on the B&B core performance.

The maximum radiation damage the cladding can sustain is not of much relevance for this study as it is assumed that, whenever the fuel reaches its fluence (or DPA) constraint it is removed from the core and reconditioned. The number of fuel batches (and therefore cycle length) and the shuffling scheme are determined based on the maximum fluence and DPA the cladding can sustain. In this study, twice the values demonstrated for HT-9 (~200 DPA and 4×10^{23} n/cm²) were assumed. Behavior of HT-9 cladding above these values is unknown because of the lack of experimental data. It is expected that new materials under development will be able to reach the assumed values. If not, fuel reconditioning will have to be performed more frequently and may increase the cost of the fuel cycle.

The reconditioning process aims at relieving the fuel radiation damage and fission gas pressure as well as replacing the cladding with a new one. For metallic fuel, this reconditioning can be performed by using the melt-refining process [19]. The discharged fuel

has axially varying composition – the heavy metal (HM) concentration tends to decrease and the fission products concentration tends to increase from the ends towards the axial center of the fuel rod. After the reconditioning process the fuel rods are assumed to have an axially uniform composition. The refabricated fuel batches are then reloaded into the core according to the shuffling scheme. A fresh depleted uranium batch is loaded at the outermost radial location to make up for the high burnup batch discarded. This shuffling process followed by burnup analysis is continued until an equilibrium core composition is reached.

II.4. Melt-refining

The melt-refining process had been developed for metallic fuel in the Experimental Breeder Reactor II project [19]. The melt-refining involves loading the decladded fuel into zirconia crucibles and melting the mixture at ~1300°C for several hours under argon atmosphere. The gaseous and volatile fission products are released and certain solid fission products are partially removed by oxidation with the zirconia of the crucible. Based on [19] it is assumed that this process can remove nearly 100% of Br, Kr, Rb, Cd, I, Xe and Cs, and at least 95% of Sr, Y, Te, Ba and the rare earths (lanthanides). Thorium and americium are also oxidized with zirconia, and 95% of these two elements will be removed from the fuel. A small fraction of other actinides may also be left in the crucible but the present analysis assumed that this fraction is negligible¹.

Even though the melt-refining process removes nearly a third of the fission products from the fuel, no heavy metal is added to make up for the mass deficiency; although the metallic fuel density is decreasing with burnup, the fuel volume is assumed, after [20], to

¹ In the “Melt-Refining” process experimented within the EBR-II program several percents of the plutonium and other actinides remained in the crud of the zirconia crucible. However, experts think that it is likely possible to develop a modified process that does not involve significant loss of actinides and cannot separate a specific or all of the actinides and, yet, can efficiently remove the gaseous and certain fraction of the volatile fission products. Although the results of this study are somewhat affected by the fraction and type of solid fission products that are removed in the fuel recycling process, the overall conclusions of this work are not sensitive to these uncertainties.

remain constant. This assumption needs to be confirmed with experimental data that will cover the range of operational conditions addressed in this study.

III. Transition core performance

This section assesses the feasibility of starting the B&B core using either enriched uranium (EU) or TRU from LWR UNF. The required enrichment level is smaller than the currently accepted limit of 20 wt% ^{235}U . The TRU vector composition assumed is that given in Table 5; it corresponds to a discharge burnup of 50 GWd/tHM and 10 years of cooling time. At the time of

TRU and EU starter cores become subcritical at an average core burnup of, respectively, 303 GWd/tHM and 306 GWd/tHM. The corresponding average starter burnups are 467 GWd/tHM and 468 GWd/tHM while the average blanket burnups are only 139 GWd/tHM and 145 GWd/tHM. The axially averaged burnup in the starter batch reaching the highest burnup is 500 GWd/tHM for the TRU starter and 501 GWd/tHM for the EU starter. These values are summarized in Table 6. By shuffling the eight fuel batches it is possible to decrease the excess reactivity during the transition period and increase the achievable starter burnup.

The power peaking is occurring in the innermost fuel batch when k_{eff} reaches its maximum value. When the reactor is started, the neutron flux amplitude is largest in the innermost batch that features the highest starter batch breeding ratio (due to low fissile concentration), and thus, the plutonium buildup rate is the largest in the central core region. As the plutonium concentration increases, the core multiplication factor and the power fraction of the core center are increasing. The power density distributions at BOL, when the power peak reaches its maximum and when $k_{\text{eff}}=1$ (~300 GWd/tHM) are shown in Figures 4 through 6 for the TRU starter core. The power distribution in the core using EU starter is almost the same. When the reactor becomes subcritical at ~300 GWd/tHM, the starter is producing only 40% of the total power and the radial power peaking factor is only 1.42 for

both the TRU and EU started cores. The maximum radial power peaking factor of 2.62 for the TRU starter and 2.56 for the EU starter is reached at 69 GWd/tHM.

The neutron spectrum in the TRU starter core and in the EU starter core is very similar as illustrated in Figure 7 at BOL and in Figure 8 at EOL (300 GWd/tHM). The batch-wise DPA accumulation in the cladding, shown in Figure 9, is sensitive to the spectrum and power fraction and is therefore identical for the two cores. The decreasing rate of DPA accumulation with the distance from the core center is due to a reduction in the neutron flux amplitude and, for the blanket batches, also to spectrum softening, as illustrated in Figure 10.

The infinite multiplication factor of the various fuel batches is estimated using the one-group cross-sections generated by MCNP5, assuming that the average number of neutrons emitted per fission is the same for all fuel batches. The batch k_{∞} value provides a useful guide for determining when the blanket batch can substitute a starter (or a driver) batch. The axially averaged k_{∞} value of the eight fuel batches are provided in Figure 11 for the TRU starter core and in Figure 12 for the EU starter core. Although batches 5 through 8 have an identical initial composition, their BOL k_{∞} values shown in Figures 11 and 12 differ because the neutron spectrum gets softer as the batch is further located from the core center. It is observed that, in both cores, the k_{∞} value of the innermost fuel batch is increasing until the average core burnup reaches ~ 70 GWd/tHM, confirming that the power fraction in the core center is increasing from BOL to 70 GWd/tHM and peaks around this burnup. At BOL, the k_{∞} values in the EU starter are larger than in the TRU starter, but when the first fuel reconditioning occurs the k_{∞} values of all the fuel batches of the two systems are almost equal. The first fuel reconditioning occurs when the peak fuel burnup reaches ~ 200 GWd/tHM, which happens after 8.4 EFPY and corresponds to an average core burnup of 65 GWd/tHM.

At ~ 180 GWd/tHM, the innermost blanket fuel batch reactivity is similar to that of the most reactive starter fuel batch (fourth). When the core becomes subcritical at ~ 300

GWd/tHM, the reactivity of the blanket fuel batches is sufficiently high to enable propagating the breed and burn mode to a fresh depleted uranium blanket, by progressively removing the starter batches, shuffling in the blanket batches and adding fresh depleted uranium batches. This scenario is discussed in Section VI.1.

The high uncontrolled k_{eff} values shown in Figure 3 are not desirable due to the large number of neutrons lost in the reactivity control systems. With an optimal starter design and an optimized shuffling scheme it is possible to decrease the excess reactivity of the B&B core and to improve its radial power distribution. It is also possible to discharge the TRU starter at a larger burnup than the EU starter. Acceptable shuffling schemes for the transition B&B core are discussed in Section VI.1.

IV. Equilibrium core –maximum burnup

In this section, the B&B core equilibrium composition, achieved after multiple fuel shufflings, is determined and the equilibrium core performance is quantified. The B&B core equilibrium performance is independent of the type of starter used; it depends only on the shuffling scheme, cycle length and core design characteristics (geometry, material...) discussed in Section II.2.

The k_{∞} evolution with burnup of the B&B core unit cell is shown in Figure 13; it is calculated [10] assuming depleted uranium for the initial fuel composition using the B&B core average power density. This k_{∞} evolution provides a useful guidance in the search for the optimal fuel shuffling pattern and for understanding the trends in the burnup reactivity swing. The actual k_{∞} evolution with burnup in the B&B core slightly differs from that of Figure 13 due to differences in the neutron spectrum and in the power density evolutions. The spectrum evolution obtained by eigenvalue calculations for the unit cell is not exactly the same as the spectrum evolution in the B&B core (illustrated in Figure 10). Nevertheless, Figure 13 exhibits the right trends in k_{∞} evolution – k_{∞} tends to increase up to a burnup of

approximately 13% FIMA and to drop with burnup thereafter. The initial rate of k_{∞} increase is very steep; it takes only 3% to 4% FIMA for k_{∞} to go from ~ 0.22 to 1.0. k_{∞} starts dropping with burnup beyond $\sim 15\%$ FIMA.

IV.1. Simple shuffling scheme

The simple shuffling scheme discussed in this section assumes an out-in shuffling; the fuel discharged from one batch is reloaded, after reconditioning, in a zone immediately inward to the zone it was discharged from. A fresh depleted uranium batch is loaded at the outermost radial zone to make-up for the high burnup batch discharged from the innermost zone. During operation, 75% of the gaseous fission products are continuously removed from the fuel to simulate the fission gas migration into the fission gas plenum above the core [23]. The above is repeated until equilibrium is achieved. At equilibrium, the maximum discharge burnup for the simple shuffling scheme is found to be 55% FIMA, corresponding to 8.80 years long cycles. The evolution of k_{eff} for several equilibrium cycles is shown in Figure 14.

A lower bound on the maximum achievable burnup is obtained by assuming that when the fuel cladding is replaced the fuel is not reprocessed. With this assumption all the fission products remains in the fuel, with the exception of the 75% of the gaseous fission products released during operation, and the fuel composition is not axially averaged. This approach might not be practical because the fuel may need to be reprocessed with a simple process such as the AIROX process [24] in order to relieve the radiation damage. The AIROX process would allow all the gaseous and a fraction of the volatile fission products to be released. With the assumption of no reprocessing, it is found that at equilibrium the maximum discharge burnup for the simple shuffling scheme is 42.7% FIMA. It is concluded that using the melt-refining process can enable a significant increase in the maximum discharge burnup; with the current layout the relative burnup increase is $\sim 29\%$ (to 55% FIMA).

The reactivity gain of ~4.5% with fuel reconditioning, shown in Figure 14, is mostly due to the axial mixing of the fuel composition: most of the fissions are occurring in the central axial part, where the End Of Equilibrium Cycle (EOEC) fission products content is the largest and the heavy-metal content is the lowest. The burnup reactivity swing can be decreased by increasing the number of radial fuel batches and decreasing the cycle length. This, however, will decrease the capacity factor of the reactor because of the more frequent shuffling.

Figures 15 and 16 show, respectively, the net neutron leakage probability out of each fuel batch and the radial power distribution at the Beginning Of Equilibrium Cycle (BOEC) and EOEC. The resulting equilibrium radial power distribution is quite favorable; the radial peak-to-average axially integrated power ratio is 1.66. The power density distribution shifts radially outward during the equilibrium cycle and the peak power density of 248 W/cm³ is reached at EOEC. The peak power density mentioned in this section and in the following ones accounts for the batch-wise radial peaking factor and for the axial peaking factor, but not for the radial peaking factor within a batch. The radial leakage probability increases from 1.9% at BOEC to 4.1 % at EOEC. The equilibrium core axial leakage probability is 4.1%, and remains approximately constant during the cycle. The peak DPA accumulated in the cladding per unit burnup is approximately 3.4 DPA/GWd/tHM, corresponding to approximately 400 DPA being accumulated during a cycle in the cladding of the innermost fuel batch.

The design parameters and performance of the equilibrium core are summarized in Table 7. The “conversion ratio” is the ratio between the number of transuranium atoms in the entire core at the end and beginning of an equilibrium cycle. The EOEC batch-wise concentration of the important actinides is given in Figure 17 as the atomic fraction of the given isotope relative to all the heavy metal and fission products present in the core region considered. The concentration of ²³⁹Pu peaks after the third shuffling – in the third batch from

the outer core boundary. Thereafter it is slightly decreasing as the ^{238}U concentration is depleted when approaching the central zone.

Tables 8 and 9 compare the heavy metal composition and selected characteristics of the fuel discharged from the B&B core at 55% FIMA and of the fuel discharged from a once-through LWR at 50 GWd/tHM. The uranium ore utilization is measured in a couple ways, both assuming that the initial starter fuel is made of 11% enriched uranium, and that the B&B core is being fed only with the depleted uranium left over from the uranium enrichment process: (1) Per unit weight of natural uranium ore.; (2) Per separative work unit (SWU) required. It is found that the B&B core and fuel cycle hereby proposed offer two orders of magnitude increase in the uranium ore utilization and nearly two orders of magnitude reduction in the SWU requirement without resorting to actinides separation. The restriction to only use the depleted uranium left-over from the natural uranium used for making the enriched uranium for the starter fuel is artificial; the same core could continue fissioning depleted or natural uranium – as well as used fuel discharged from LWRs. It is assumed that at the end of life of the reactor hardware, the equilibrium core will be transferred to a new reactor and no fissile fuel will have to be added to achieve criticality – as is required in the first reactor.

Also given in Table 8 is the total energy value of the 1.5 million tons of depleted uranium accumulated in the world so far [25], if all this depleted uranium was to be fed into the breed-and-burn reactors – 3000 times the world present annual generation of nuclear electricity. The current rate of depleted uranium being generated in the uranium enrichment plants in the world could support a breed-and-burn based power reactor capacity that is 100 times higher than the capacity of the presently operating nuclear power reactors, worldwide.

It is also observed in Tables 8 and 9 that, relative to LWR operating with the once-through fuel cycle, the fuel discharged from the B&B core under consideration features, per

unit of electricity generated: (a) ~40% the mass of TRU and plutonium; (b) ~12% the inventory of ^{237}Np and its precursors²; (c) ~12% of the decay heat one year following discharge; (d) ~28% of the radiotoxicity of heavy metal and fission products one year following discharge; and (e) ~7% the neutron emission rate one year following discharge. The fraction of the fissile isotopes in the discharged plutonium is comparable but the decay heat and neutron emission rate per unit mass of discharged plutonium are nearly half as large. Contributing to these differences, in addition to the amount of electricity generated per HM mass discharged, are the differences in the fission product and actinide (see Table 8) isotopic composition in the fuel discharged from the fast spectrum B&B core and the thermal spectrum LWR core.

IV.2. Improved shuffling scheme

The shuffling scheme studied in the previous section is a continuous inward relocation. This shuffling scheme does not necessarily provide the maximum feasible discharge burnup, the smallest possible burnup reactivity swing or optimal power distribution. The search for an improved shuffling scheme is guided by the following considerations:

- The two outermost fuel batches, being almost pure depleted uranium, should remain located at the outermost radial core location in order to minimize the fraction of neutrons that radially leak out from the core;
- A low burnup and, hence, a low fissile content fuel batch should be loaded near the core center in order to decrease k_{eff} at BOEC and reduce the radial power peaking;
- A medium burnup fuel batch should not be located at the innermost location. This is the fuel with the highest reactivity worth and, if located at the innermost location, will result in a very large power peaking.

² The concentration of ^{237}Np and its precursors (^{241}Pu , ^{241}Am and ^{245}Cm) is of concern because, of the actinides, ^{237}Np has the dominant contribution to the long-term radiological hazard of the repository.

By using the shuffling pattern schematically showed in Figure 18, it is found possible to extend the maximum discharge burnup by +2% FIMA while decreasing the burnup reactivity swing to about 2% but the radial power peaking factor is increased by +18% to 2.02; this corresponds to a peak power density of 293 W/cm³

Due to the larger discharge fuel burnup the cycle length is increased to 9.04 years. The radial power and burnup distributions at BOEC and EOEC are shown in Figures 19 and 20. The power distribution changes from BOEC to EOEC very differently than in the simple shuffling scheme (Figure 16). For the simple shuffling, the power distribution was shifting toward the outer batches, where the burnup was smaller. For the improved shuffling scheme, the radial batches having a burnup larger than 15% FIMA at BOEC see their power fraction decreasing, while for the other batches it is increasing. This phenomenon is particularly important for the second innermost batch. Its burnup at EOEC is about 15% FIMA, corresponding to the peak k_{∞} value – see Figure 13. The power peaking is occurring in this batch at EOEC due to its central location. It also makes the power density in the innermost batch slightly increasing at EOEC despite having a burnup above 50% FIMA. Perhaps the primary advantage of the improved shuffling scheme is that it offers a relatively small shift in the radial power distribution with burnup, with the exception of the second batch. This will simplify the design of the core cooling system.

V. Equilibrium cycle minimum required burnup

If the minimum burnup required for sustaining the breed and burn mode of operation is small enough, it may be possible to accommodate the radiation damage effects without fuel reconditioning. Moreover, it is theoretically possible [10] to use the discharged fuel, after reconditioning, for the starter fuel of a new B&B reactor. The practical minimum burnup required for sustaining the breed and burn mode of operation in the B&B core is discussed in this section.

In order to sustain the B&B mode of operation with a minimum discharge burnup the neutron leakage probability from the core needs to be minimized and the most reactive fuel needs to be loaded near the core center where the neutron importance is the highest. This tends to result in a relatively large radial power peaking factor. For the search for the minimum required burnup the core is divided into 12 radial fuel batches instead of the 8 batches used previously so as to increase the number of fuel shuffling scheme options in order to enable better reducing the peak power density while minimizing the required burnup. The study uses the assumptions defined in Section IV: (a) 75% of the gaseous fission products are continuously removed from the fuel; (b) at EOEC the highest burnup fuel batch is discarded; (c) at EOEC the fuel to be recycled is reconditioned with the melt-refining process; and (d) a fresh depleted uranium batch is added to the core at the outermost fuel zone.

V.1. Simple shuffling scheme

The reference shuffling scheme analyzed is the straight forward out-in fuel management: at EOEC the innermost fuel batch is discharged, the other eleven fuel batches are “instantaneously” shuffled inward, after reconditioning, and a fresh depleted uranium batch is added at the outermost location.

For a discharge burnup of 199.8 GWd/tHM – 20.4% FIMA – corresponding to a cycle length of 2.17 years, the multiplication factor, shown in Figure 21, is equal to unity at BOEC and increases up to 1.035 at EOEC. The multiplication factor is increasing during the early part of the cycle due to the increase of fissile material in all the fuel batches. When approaching the middle of the equilibrium cycle, the overall mass of fissile material in the core is increasing but the mass of fissile material in the innermost fuel batch is decreasing; this batch features the highest burnup and highest fission products concentration. This effect, along with increasing radial leakage probability due to the power distribution shifting outward, reduces the rate of increase of the multiplication factor and eventually causes it to

start decreasing towards EOEC (Figure 21). The power distribution at BOEC and EOEC is shown in Figure 22. The burnup reactivity swing is 3.5%, and approximately 2.5% of the total number of neutrons generated over a cycle need to be absorbed by the reactivity control systems to maintain k_{eff} equal to unity.

The power peaks in the innermost fuel batch and is the highest at EOEC. This batch produces more than 60% of the total core power in less than 8.5% of the core volume. The outer radial half of the core is producing practically no power and accumulates almost no burnup; most of the neutrons are generated near the center of the core and are absorbed before reaching the outermost fuel batches. The radial leakage probability is only 0.1%. The axial neutron leakage probability is constant from BOEC to EOEC and equals 3.8%. The axially averaged radial power peaking factor of this core at EOEC is 7.32. It is difficult to efficiently cool such a core and its average power density level is very low.

The k_{∞} value of the fuel discharged at 20.4% FIMA is only slightly below its maximum value (Figure 13). At BOEC, the innermost fuel batch has a burnup of 9.1% FIMA and its k_{∞} is 1.28, the highest of all the fuel batches. Over the cycle it accumulates +10.3% FIMA additional burnup. At EOEC its k_{∞} is 1.25. The k_{∞} and burnup distributions for the 12 fuel batches are shown in Figure 23.

V.2. Improved shuffling scheme

The primary objective of the improved shuffling scheme is to flatten the power distribution, so as to enable safe operation at the rated power level of 3000 MW_{th} without penalizing too much the minimum required burnup. The main idea pursued is to increase the volume fraction of the core producing power by building up fissile material in the fuel sooner. A potential disadvantage of power flattening is an increase in the radial neutron leakage probability that may result in a larger minimum required burnup.

The improved shuffling scheme arrived at is schematically represented in Figure 24; location #1 is the innermost (left on Figure 24) while location #12 is the outermost. The fresh depleted uranium batch is loaded at the outermost location and then shuffled toward the core center and back a couple of times. When it reaches location #5, it is discarded. The multiplication factor is linearly increasing from 1.00 at BOEC to 1.04 at EOEC. While the fissile material content of the highest burnup batches – locations #4 and #5 – is decreasing during the cycle, the fissile content of the inner batches is increasing, resulting in an overall increase of k_{eff} . The radial distributions of the infinite multiplication factor and burnup at BOEC and EOEC are shown in Figure 25. The minimum discharge burnup required to maintain the breed and burn mode of operation with the improved shuffling scheme is found to be 189.3 GWd/tHM (19.4% FIMA) and only 2.1% of the neutrons generated over the cycle need to be absorbed by the reactivity control systems to maintain k_{eff} at unity.

The infinite multiplication factor at BOEC is larger than unity for 5 out of the 12 fuel batches; all located in the inner half of the core. The radial power distribution of Figure 26 shows that practically no power is generated by the outermost fuel batch and the five outermost batches generate only ~ 5% of the fission neutrons. As a result, the radial neutron leakage probability is approximately constant during a cycle and equals to 0.7% – not significantly larger than for the simple shuffling scheme. The total neutron leakage probability is 4.5% at BOEC and decreases to 4.3% at EOEC.

The radial power distribution change between BOEC and EOEC, shown in Figure 26, is smaller than for the simple shuffling scheme (Section V.1). The power distribution varies only slightly in four out of five innermost fuel batches, and stays almost constant in the seven outer batches. The axially averaged radial power peaking factor is 2.58 and is occurring at BOEC. This peaking factor, while still large, is only a third of that of the simple shuffling scheme.

Despite of the somewhat smaller neutron loss in the simple shuffling scheme – a total of 6.3% (of which 2.5% are lost in the reactivity control systems and 3.8% by leakage) versus a total of 6.6% for the improved shuffling scheme (of which 2.2% are lost in the reactivity control systems and 4.4% by leakage) – the minimum required burnup was found slightly smaller for the improved shuffling scheme. This difference is due to the fuel reconditioning. The burnup being accumulated more progressively with the improved shuffling scheme, a larger amount of fission products is removed by the multiple fuel reconditionings than with the simple shuffling scheme. Therefore, the parasitic neutron capture in the fission products is lower with the improved shuffling scheme. Overall, about 5.2% of the neutrons are captured in the fission products with the simple shuffling scheme, while this fraction is only 4.5% with the improved shuffling scheme. This effect depends on the frequency of the fuel reconditioning. If the cladding could sustain the radiation damage corresponding to an average burnup of ~20% FIMA, no fuel reconditioning would be required in which case the fission products concentration and minimum required burnup would be slightly larger than the above reported values and the minimum required burnup for the improved shuffling scheme will be the larger than for the simple scheme.

In summary, using the improved shuffling scheme it is possible to maintain the minimum required burnup in the range of 20% FIMA while decreasing the radial power peaking factor from 7.32 to 2.58 and reducing the power distribution variation with burnup. It is possible to farther reduce the burnup reactivity swing by increasing the number of fuel batches and reducing the cycle length.

VI. Feasibility of spawning new cores

The neutron balance analysis performed in [10] indicates that it is theoretically possible to (a) start at least two B&B cores using a single starter, the design of which is described in

Section III, and (b) spawn new B&B cores using the fuel discharged at equilibrium with the minimum required burnup identified in the previous section.

The purpose of the study presented in this section is to show that it is practically feasible to establish the breed and burn mode in (a) two new B&B cores using a single EU starter; and (b) one new B&B core using the fuel discharged at 19.4% FIMA from the large equilibrium B&B core discussed in Section V.2. Although TRU is neutronically the preferred starter fissile material, EU is considered because of the existence of enrichment plants and the lack of commercial reprocessing plants in the USA for LWR used fuel.

The B&B core model used for this analysis is the cylindrical finite core made of 12 fuel batches described in Section V. Based on the neutron balance analysis of [10], in order to start two B&B cores from the same EU starter, the neutron losses by leakage and in the reactivity control systems must not exceed ~6%. The main constraints used for the current study are (1) a burnup reactivity swing no larger than 4%; (2) a cycle length no shorter than one year; and (3) a radial power peaking factor not exceeding the value of 2.62 obtained for the TRU igniter (Section III).

During the depletion, 75% of the fission gases are continuously removed from the active core and when the burnup limit of 20% FIMA is reached, only the fuel batches that are approaching this burnup are reconditioned using the melt-refining process. Melt-refining is assumed to occur instantly. The proposed shuffling schemes are not optimized and only aim at demonstrating the practical feasibility of the results predicted [10] using general neutron balance considerations rather than based on core design.

VI.1. Starting a couple of B&B cores using a single starter

It is assumed that whenever the excess reactivity of the first B&B core approaches ~4%, the EU and depleted uranium (DU) fuel batches are shuffled into a configuration the k_{eff} of which is between 1.0 and 1.003. Five shufflings were found needed in the initial B&B core;

their time of occurrence is given in Table 10 (left column) and the corresponding k_{eff} evolution is shown from 0 to 10.1 EFPY in Figure 27. At BOL, the axial and radial neutron leakage probabilities are 3.5% and 0.4%, respectively. The average burnup of the EU starter and DU batches after the 5th cycle (10.1 EFPY) is, respectively, 15.0% and 3.0% FIMA; hence, there was no need to recondition the fuel throughout this campaign. Of the six depleted uranium batches, the burnup of three exceeded 3.4% FIMA making their k_{∞} values larger than unity (Figure 13). The EU starter is then discharged, the relatively high k_{∞} blanket batches are shuffled inward, and fresh DU batches are loaded at the outer radial locations. After an additional 15 EFPY (marked in Figure 27 by the dashed vertical line), six shufflings and one reconditioning (specified in Table 10), the freshly loaded DU batches average burnup reaches 3% FIMA. At this point it is possible to shuffle the once burnt DU batches inward and to replace the twice burnt DU batches, having an average burnup of 23.8% FIMA, with fresh DU batches. The above fuel management scheme is, most likely, not the optimal. Two of the many possible options are keeping the twice burnt DU in the initial core longer to achieve a higher discharge burnup and/or discharging it progressively to achieve a smoother transition between cycles.

In this and the following sections, the number of neutron histories and cycles followed by MCNP were reduced in order to decrease the computation time and explore a larger number of shuffling schemes. As a result, the statistical uncertainty in the reported results is relatively large; typically ~150 pcm, and the curves shown are not smooth.

After discharged from the initial B&B core, the once burnt EU starter batches are reconditioned and loaded into a new B&B core along with the same volume of DU. After 15.1 EFPY, marked by the vertical dash line on Figure 28, two shufflings and one reconditioning (Table 10), the DU batches average burnup reaches 3.0% FIMA and they can replace the twice burnt EU starter to sustain the chain reaction. The resulting k_{eff} evolution is shown in

Figure 28. Alternately, since the starter burnup is only 28.9% FIMA, it is possible to keep it longer in the core to increase its burnup and/or to discharge it progressively.

The radial power peaking factor is 3.12 for the initial core that uses the fresh EU starter, but becomes as high as 5.0 for the initial core in which the DU starter is replaced with the once burnt DU batches, as well as for the second B&B core that uses the once burnt EU starter. In order to decrease this value, it is necessary to achieve an average burnup of approximately 4% FIMA in the fresh DU batches before removing the driver fuel (EU starter or once-burnt DU). This means discharging the EU starter after 12 EFPY instead of 10.1 EFPY. The “4% FIMA” criterion is reached after 17 EFPY in the initial core that uses the once-burnt DU batches to sustain criticality and after 18.8 EFPY in the second B&B core that uses the once-burnt EU starter.

It is concluded that using an EU starter made of 6.57 tons of ^{235}U , it is practically feasible to sequentially establish the breed and burn mode of operation in two similar B&B cores. As the excess neutrons that can be provided by TRU starter fuel is larger than that available from the same quantity of EU starter fuel [10], it is expected that a single TRU starter fuel will also have sufficient reactivity to sequentially start at least two B&B reactors.

VI.2. Spawning new B&B cores from a B&B core at equilibrium

The feasibility of spawning new cores from the equilibrium B&B core that is operating with the minimum required burnup, described in Section V.2, is assessed in this section. The equilibrium cycle lasts 2.05 years and at EOEC the discharged fuel average burnup is 19.4% FIMA. It is then reconditioned using the melt-refining process and stored until 50% of the B&B core fuel volume is accumulated; the core being made of 12 fuel batches, this is achieved after 6 cycles; i.e., 12.3 EFPY. When the required number of fuel assemblies have been discharged and reconditioned, they are loaded into a new B&B core along with the same volume of fresh depleted uranium fuel assemblies. This approach is schematically represented

in Figure 29. The two innermost and four outermost fuel batches of the new core are assumed loaded with depleted uranium. The six other batches, made of fuel having an average burnup of 19.4% FIMA that was discharged from the “mother” B&B core, are loaded between the depleted uranium fuel batches (right side of Figure 29).

At BOL, the new B&B core radial power peaking factor is 2.38; somewhat smaller than the 2.58 value obtained for the equilibrium core in Section V.2. The evolution of k_{eff} and of the radial power peaking factor in the new B&B core from BOL to 14.8 EFPY is shown in Figure 30. During the first several years, k_{eff} is slightly decreasing because the k_{∞} of the “19.4% burnup” fuel batches is decreasing (See Figure 13), and the increase in k_{∞} due to fissile material build-up in the depleted uranium batches is not sufficient to compensate for it. After ~8 EFPY, the k_{∞} of the two innermost depleted uranium fuel batches is larger than unity, and due to their favorable location near the center of the core, their k_{∞} increase causes the core k_{eff} to strongly increase. As the fission rate becomes very large in those two batches, the radial power peaking factor is also strongly increasing: it reaches 2.52 at 12.1 EFPY and 3.37 at 14.8 EFPY.

At 12.1 EFPY, the average burnup of the six “19.4% burnup” fuel batches is 34.2% FIMA and the average burnup of the depleted uranium batches is 4.0% FIMA. This burnup is high enough to sustain the breed and burn mode after removing the “19.4% burnup” fuel assemblies and replacing them with fresh depleted uranium assemblies. However, the resulting radial power peaking factor is very large (>4.0) since only three fuel batches have $k_{\infty} > 1.0$. A preferable approach is to recondition the “19.4% burnup” fuel at 12.1 EFPY (they almost accumulated +20% FIMA), shuffle the fuel to obtain an acceptable power distribution and operate the reactor until the “19.4% burnup” fuel batches reach 48% FIMA, at which point they cannot provide anymore excess neutrons (Figure 13). Discharging the starter

batches progressively enables to minimize the burnup reactivity swing and the radial power peaking factor with ease.

The neutron balance performed in [10] and the results obtained in the current study suggest that it is possible to start the new B&B core by loading it with only 40% of fuel discharged at 19.4% FIMA from the equilibrium B&B core – and 60% of depleted uranium. The doubling time of the B&B core at equilibrium is equal to the time required to discharge 5 fuel batches (corresponding to 41.7% of the core volume) – 10.3 EFPY. The doubling time is here defined as the time required for discharging from the equilibrium B&B core the minimum amount of fuel required to start a new B&B core of identical volume and power, without having to use fissile material from another source. The doubling time is provided in EFPY and therefore does not account for the capacity factor. The fuel reconditioning is assumed to occur during the operation of the reactor and therefore does not affect the doubling time at equilibrium. Furthermore, the heavy metal losses due to the melt-refining process are expected to be no larger than 2.5% [19], which is not a concern here: the five batches used to start the new core contain approximately 4% ($41.7\%/40\%-1$) excess fuel as compared to the minimum amount required.

VII. Feasibility of reducing the required enriched uranium inventory

It is possible to decrease the amount of ^{235}U required to establish the breed and burn mode of operation if the starter fuel is not required to spawn a new B&B core. This section discusses the feasibility of reducing the amount of ^{235}U in the EU starter by using higher enrichment fuel that is concentrated in the axial central part of the core. The idea is to maximize the fraction of excess neutrons generated by the starter fuel that are captured in the ^{238}U of the blanket fuel while minimizing the fraction of the starter generated excess neutrons that are either axially leaking out from the core or are captured in the ^{238}U of the starter fuel. The possible drawbacks of this approach are that the amount of separative work units for a

same ^{235}U mass is slightly larger when the enrichment is higher and that the peak fuel and cladding temperatures may increase because of the larger peak power density of a smaller volume starter.

The EU starter fuel batches height and location are determined so as to (a) maintain the reactor critical (b) minimize the mass of ^{235}U required; and (c) achieve a sufficiently flat BOL power distribution. The ^{235}U enrichment is taken to be 20% by weight, the highest presently acceptable because of proliferation concerns. The initial core layout yielding the best performance identified so far is presented on the left side of Figure 31. The dotted batches are made of depleted uranium, the hatched batches are made of 20% enriched uranium and the white regions are made of fuel rod sections filled with sodium. The active starter fuel lengths and ^{235}U masses are given in Table 11.

The mass of ^{235}U initially required is 5.68 tons – 13.5% smaller than for the EU starter discussed in the previous section, despite of using a larger enrichment level and a larger number of EU fuel batches. The starter heavy metal inventory is 99.7 tons instead of 142.6 tons of the reference starter. With a core power of 3000 MW_{th}, the average specific power at BOL increases from 21.2 W/gram of heavy metal (gHM) for the starter studied in Section VI.1 to 30.1 W/gHM for this starter. The BOL peak specific power is ~85.8 W/gHM instead of 52.6 W/gHM but the BOL radial power peaking factor is 1.93 versus 2.6 for the starter in Section III.

After 5.2 EFPY, corresponding to an average core burnup of 57.2 GWd/tHM, the starter average burnup is 16.08% FIMA and needs to be reconditioned. The reconditioned fuel is then reloaded into the core occupying only four batches, instead of six, as shown in the middle scheme of Figure 31. This is done by adjusting the active fuel rod length so as to flatten the core radial power distribution. Two fresh depleted uranium batches are added to the core, increasing the heavy metal inventory.

At 9.9 EFPY, the average burnup accumulated in the six depleted uranium batches having the highest burnups is 5.3% FIMA and is sufficient to maintain the core criticality when replacing all the starter fuel batches with fresh depleted uranium batches. The average starter burnup is 25.5% FIMA. The feasibility of establishing the breed and burn mode in a new B&B core using the once-burnt starter batches has not been studied; it appears that some additional fissile material may be required.

An alternative to discharging the starter batches and starting a new B&B core is to keep them in the current reactor and operate them to a higher discharge burnup. In this scenario the fuel is shuffled at 9.9 EFPY to flatten the radial power distribution and the starter fuel needs to be reconditioned for a second time at 12.3 EFPY. The reconditioned fuel is reloaded in three batches and a batch of fresh depleted uranium is added to the core, as shown on the right side of Figure 31. After ~23 EFPY the core becomes subcritical and the average starter and depleted uranium fuel burnups are 42.5% FIMA and 12.8% FIMA, respectively.

The evolution of k_{eff} , the radial power peaking factor and the leakage probabilities for this latter scenario are presented in Figures 32 and 33, from BOL to 23 EFPY. Compared to the core studied in Section VI.1, the axial leakage probability at BOL is smaller and the radial leakage probability is larger. This is due to the fact that the fissile material is concentrated near the core middle plane, but closer to the core radial periphery. Overall, the core studied in this section has a slightly larger leakage probability at BOL.

It is concluded that by decreasing the enriched fuel rod length and increasing the enrichment level, it is possible to establish the breed and burn mode of operation in the B&B core while decreasing the amount of ^{235}U required by 13.5% compared to the scenario studied in Section VI.1 and by 26% compared to the preliminary scenario discussed in Section III. Furthermore, when the shortened enriched fuel rods are reconditioned, it is possible to re-fabricate fewer of them having an increased length and load additional fresh depleted uranium

fuel batches into the core without discharging any fuel. This enables maintaining the radial power peaking factor approximately equal to 2.0 throughout the cycle. If it is found possible to start a second core using the same starter, the theoretical minimum doubling time of the first generation of B&B cores will be 9.9 EFPY – similar to the 10.25 EFPY found for the scenario presented in Section VI.2. For the transition period, the practical doubling time may be larger because of the fuel reprocessing which cannot occur in parallel with the core operation.

A concern associated with the use of the shortened starter fuel rods is whether they can deliver the nominal core power level without exceeding the thermal hydraulic constraints. This issue is examined in the following section.

VIII. Preliminary thermal-hydraulics analysis

Throughout the study it was assumed that the B&B cores are operating at 3000 MW_{th} without exceeding any of the thermal hydraulic constraints. The validity of this assumption is assessed in this section. The objective is to estimate the maximum achievable power for the B&B core, based on the power peaking factors obtained from the neutronics analysis in previous sections. The thermal hydraulic constraints accounted for are the following:

- maximum pressure drop: 1 MPa [16];
- maximum sodium velocity across the core: 12 m/s [16]
- maximum fuel-cladding interface temperature: 650°C [26]
- maximum fuel temperature: 1240°C

Since the breed-and-burn reactor fuel has to operate up to relatively high burnups, the practical constraint on the peak cladding temperature is likely to be smaller than assumed. The inlet coolant temperature is conservatively assumed to be 395°C and the maximum outlet coolant temperature is taken to be 580°C [16]. The maximum outlet temperature corresponds

to the outlet coolant temperature in the fuel assembly producing the highest power, corresponding to a mixed core coolant outlet temperature of approximately 510°C.

The coolant volume fraction used for this thermal hydraulic analysis is different from the fraction used for the neutronics analysis. For the neutronics analysis, the coolant volume fraction accounted for all the sodium in the core, including the sodium in the inter-assembly gaps and in a postulated number of control assembly locations. For the thermal hydraulic analysis, only the coolant inside a fuel assembly is accounted for. The fuel assembly pitch is taken to be 161.42 mm with an inter duct gap of 4.32 mm and a duct thickness of 3.94 mm. These are the values chosen for the S-PRISM core design by General Electric [27] and by ANL in their recent studies of ABR [17]. For those values and the overall volume fractions provided in Table 1, the in-assembly coolant volume fraction is 26.5%, corresponding to a P/D ratio of 1.11. The thermal hydraulic analysis is performed for the B&B equilibrium core operating with the simple shuffling scheme defined in Section IV.1. For this core, the radial power peaking factor is 1.67 and the axial power shape is approximated using the axial power distribution obtained from the neutronic analysis and a truncated cosine fit. The total fuel rod length, including fission gas plenum, is 418.72 cm. The design variable is the number of fuel rods per fuel assembly. The fuel rod diameter, cladding thickness and lattice pitch are adjusted with the number of fuel rods; the fuel, clad and coolant volume fractions are conserved. The cases examined are defined in Table 12.

When decreasing the number of fuel rods per fuel assembly, keeping the assembly dimension and the constituents' volume fractions constant, the fuel pin diameter is increased and the hydraulic diameter is increased. In order to sustain the same pressure build-up due to fission gas release inside the fuel rod, the cladding thickness is also increased. This makes the temperature gradient in the cladding larger. However, a larger hydraulic diameter decreases

the coolant friction along the fuel rods, thus making it possible to increase the coolant velocity and, as a result, the attainable core power.

For each case shown in Table 12, a pressure drop calculation is performed in order to determine the maximum coolant flow rate achievable without exceeding the maximum pressure drop and sodium velocity. The pressure drop calculation accounts for the coolant friction along the fuel rod, including the fission gas plenum and wire wrap effect, and for the coolant expansion and contraction as it flows through an assembly. The correlation proposed by Chang and Todreas [28] is used to calculate the coolant friction due to the wire wrap. The other correlations and values used for the material thermo-physical properties were obtained from [29, 30 and 31]. Using the calculated maximum coolant velocity, coolant inlet and outlet temperatures, radial power peaking factor, axial power distribution and material thermo-physical properties, the maximum core power and corresponding fuel and cladding peak temperatures are determined. All the results presented in this section are found to be constrained by the maximum pressure drop and coolant temperature increase but not by the maximum coolant velocity. The results obtained are summarized in Table 12.

When using 271 fuel rods per fuel assembly, it is possible to achieve 3000 MW_{th} in the B&B core at equilibrium without exceeding any of the assumed constraints. Decreasing the number of fuel rods to 91 per assembly, it may even be possible to reach 3900 MW_{th}. However, for this power level the peak linear heat rate is high, resulting in a peak cladding temperature and maximum coolant velocity close to the design limit and, hence, may not provide sufficient safety margins.

During the transition period the radial power peaking factor for the TRU and EU starters is as high as 2.62 and 2.56, respectively. Those peaking factors require the core power level to be decreased to 2510 MW_{th} for the TRU starter and to 2570 MW_{th} for the EU starter if using 91 fuel rods per assembly, and to 1930 MW_{th} and 1970 MW_{th} if using 271 fuel rods per

assembly. Those values are constrained by the core pressure drop and coolant temperature increase. Although these power levels are significantly lower than the 3000 MW_{th} assumed, they have little effect on the neutronics performance. Furthermore, it was observed that with a proper fuel management during the transition period, it is possible to maintain the radial power peaking factor below 2.2, and this may enable to reach 3000 MW_{th} with 91 fuel pins per assembly.

With the shorter starter fuel rods of the design presented in Section VI.3, the radial power peaking factor was found to be consistently lower than 2.0. The challenge with these fuel rods is the peak fuel and cladding temperatures because at BOL the maximum linear heat generation rate is higher than in the reference design. In this case it is favorable to use a larger number of thinner fuel pins and thus reduce the linear heat rate and, consequently, the peak fuel and cladding temperatures. For the fuel rod dimensions corresponding to 271 fuel pins per assembly (left column in Table 12), it is found possible to achieve 2550 MW_{th} without exceeding any thermal-hydraulic constraint. This power level is constrained by the core pressure drop and coolant temperature increase assumed in this study. For the shortest fuel rod presented in Table 11 – the length of which is 65.3 cm, the peak cladding temperature is 638°C and the peak fuel temperature is 860°C. Design margins may force this core power level to be further derated.

This preliminary thermal-hydraulic analysis of the B&B core suggests that, when at equilibrium, the nominal power can readily be achieved. However, during the transition period or when using fuel rods with a shorter enriched region it is not possible to operate the core at its nominal power of 3000 MW_{th}. It may be plausible to avoid the need for power de-rating by better optimization of the core neutronic design. .

IX. Conclusions and Discussion

The breed and burn mode of operation can be neutronically sustained in the B&B core for fuel discharge burnups ranging from 19.4% FIMA to 57% FIMA. This upper bound corresponds to a scenario in which the fuel is recycled every 20% FIMA using the melt-refining process. Except for the starter fuel required for initiating the chain reaction in the first core, only depleted uranium is used for the fuel feed. Either enriched uranium or TRU extracted from LWR used nuclear fuel can be used for the starter fuel. The fuel management scheme used to move from the transition period to the equilibrium mode of operation should be improved in order to optimize the radial power distribution and avoid excessive power peaking.

The B&B core enables to significantly improve the sustainability of the nuclear fuel cycle by increasing by two orders of magnitude the uranium utilization as compared to the once-through LWRs. The 1.5 million tons of depleted uranium that has been accumulated so far worldwide, when used as a feed for such B&B reactors, can provide the present global nuclear electricity generation capacity for more than 3000 years. Compared to once-through LWRs, the amount of plutonium and minor actinides discharged from the B&B reactor per unit of electricity generated is significantly reduced and the decay heat, radiotoxicity, and spontaneous neutron emission rate are about one order of magnitude lower. Furthermore, the lack of need of fissile fuel supply during equilibrium operation and the use of the melt-refining process to recondition the fuel are expected to offer favorable proliferation resistance.

When the B&B core is operated in the spawning mode and is at an equilibrium composition, its fuel discharged at the minimum possible burnup of 19.4% FIMA can be used to start a new B&B core without using any external fissile material every ~10.3 EFY. A similar theoretical minimum doubling time was determined for the transition period when the core is started with an EU or TRU. However, the time it takes for fuel cooling, transport and

reconditioning as well as the reactor capacity factor need to be accounted for to determine the practical doubling time.

Overall, the breed-and-burn mode of operation combined with a relatively simple fuel reconditioning is expected to improve the economics and the proliferation resistance of fast reactors and may justify sooner deployment of fast reactors. A deployment of the suggested fast reactor system (including the required fuel reconditioning technologies) will constitute a very significant step forward towards a sustainable nuclear energy system and energy security. However, a number of feasibility issues are yet to be addressed.

A first issue is the feasibility of an economically viable and environmentally acceptable fuel reconditioning process. The fraction of TRU that ended up in the waste streams of the melt-refining process experimented with in the EBR-II project was larger than desirable and the zirconia crucible used for the process was useable for a single melt; an improved fuel reconditioning technology need be developed

A second feasibility issue is the development of a cladding material and fuel design that could withstand the minimum burnup required to sustain the breed and burn mode of operation without the need for reconditioning. Although this issue could be bypassed by fuel reconditioning, reaching this objective would significantly simplify the fuel management and improve the economics of the breed and burn reactors.

A third feasibility issue is the performance of metallic fuel having a high concentration of solid fission products. As the fuel burnup increases more fission products may migrate, due to the temperature gradient, towards the fuel periphery and may lower the fuel-cladding eutectic temperature. Possible solutions include using a cladding material not forming low temperature eutectics with the problematic fission products; using a liner material between the fuel and cladding; or removing the problematic fission products during fuel reconditioning.

Another important feasibility issue is to design a B&B core to be sufficiently safe; the large low-leakage core features larger positive coolant temperature reactivity coefficient and coolant voiding reactivity feedback [32] than conventional “pancake” shape fast reactor cores. Novel negative reactivity insertion systems passively actuated by coolant temperature increase are being developed to compensate for the low leakage reactivity effects. In addition, an advanced core cooling strategy is required to accommodate the power distribution variations during a cycle.

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Table 1: Constituents Volume Fractions of the B&B Core

<u>Volume fractions</u>	
Fuel	37.5%
Gap	12.5%
Structural material	22.0%
Coolant	28.0%

Table 2: Dimensions and Composition of the Regions Modeled for the B&B Core

Region	Height (cm)	Thickness (cm)	Material (Volume %)	T [K]
Upper reflector	34.93	233.18	50% HT9- 50% Na	783
Upper end plug	2.54	200.74	22% HT9 - 78% Na	783
Plenum	250	200.74	22% HT9 - 28% Na	783
Inner duct	209.36	8.48	22% HT9 - 28% Na	783
Core	209.36	192.26	37.5% Fuel - 22% HT9 - 28% Na	800
Lower end plug	20.32	200.74	22% HT9 - 78% Na	628
Grid plate	5.18	233.18	50% HT9 - 50% Na	628
Coolant inlet	60	233.18	22% HT9 - 78% Na	628
Lower shield	20	233.18	43.1% B4C - 29.7% HT9 - 27.2% Na	628
Radial reflector	482.22	32.44	50% HT9 - 50% Na	628
Radial shield	602.33	15.22	43.1% B4C - 29.7% HT9 - 27.2% Na	628

Table 3: Characteristics Comparison Between Cylindrical Model (CY) and Assembly-level (AL) Model at BOL and After 39.4 Effective Full Power Years (EFPY) – ~300 GWd/tHM

Model	BOL		39.4 EFPY	
	CY	AL	CY	AL
Starter volume [m ³]	13.333	13.37	-	-
Blanket volume [m ³]	13.333	13.606	-	-
Average starter BU [% FIMA]	-	-	48.15%	47.81%
Total ²³⁸ U [tons]	134.2	135.8	84.9	86.5
Total ²³⁹ Pu+ ²⁴¹ Pu [tons]	4.556	4.560	9.662	9.690
Total TRU [tons]	8.184	8.190	13.172	13.22
Total FP [tons]	-	-	22.19	22.39
Radial power peaking factor	2.497	2.880	1.422	1.510

*CY=Cylindrical; AL=Assembly-level

Table 4: List of Isotopes Tracked in MCNP5

Element	Isotope number	Element	Isotope number
Thorium	232, 233	Palladium	104, 105, 106, 107, 108, 110
Protactinium	233	Silver	109
Uranium	234, 235, 236, 237, 238, 239	Cadmium	110, 111, 112, 113, 114
Neptunium	236, 237, 238, 239	Indium	115
Plutonium	236, 237, 238, 239, 240, 241, 242, 243, 244	Tin	117, 118
Americium	241, 241m, 242, 243, 244	Antimony	121, 123, 125
Curium	242, 243, 244, 245, 246, 247, 248	Tellurium	125, 128, 130
Berkelium	249	Iodine	127, 129
Californium	249	Xenon	130, 131, 132, 134, 135
Bromine	81	Caesium	133, 134, 135, 137
Krypton	83, 84	Barium	134, 137, 138
Rubidium	85, 87	Lanthanum	139
Strontium	90	Cerium	140, 142
Yttrium	89	Praseodymium	141
Zirconium	90, 91, 92, 93, 94, 96	Neodymium	143, 144, 145, 146, 148, 150
Molybdenum	95, 96, 97, 98, 100	Promethium	147
Technetium	99	Samarium	147, 148, 149, 150, 151, 152, 154
Ruthenium	100, 101, 102, 103, 104, 105	Europium	151, 152, 153, 154, 155
Rhodium	103	Gadolinium	154, 155, 156, 157, 158
		Terbium	159

Table 5: TRU Composition Obtained from LWR Fuel Discharged at 50 GWd/tHM and Cooled for 10 Years (Hong et al) [22]

LWR spent fuel composition	
Isotope	weight %
^{237}Np	6.641
^{238}Pu	2.749
^{239}Pu	48.652
^{240}Pu	22.98
^{241}Pu	6.926
^{242}Pu	5.033
^{241}Am	4.654
^{242}Am	0.019
^{243}Am	1.472
^{242}Cm	0
^{243}Cm	0.005
^{244}Cm	0.496
^{245}Cm	0.038
^{246}Cm	0.006

Table 6: Average and Peak Burnups for the TRU and EU Starter Cores at $k_{\text{eff}}=1$.

Burnups	TRU	EU
Overall average [GWd/tHM / % FIMA]	303/31.3	306/31.3
Starter average [GWd/tHM / % FIMA]	467/48.2	468/47.8
Blanket average [GWd/tHM / % FIMA]	138/14.3	145/14.9
Peak value [GWd/tHM / % FIMA]	500/51.6	501/51.2

Table 7: B&B Core Design and Performance Parameters at Equilibrium

Parameters	Value
Fuel/Gap/Clad/Coolant volume fractions [%]	37.5/12.5/22/28
Average power density [W/cm ³]	112.5
Peak power density [W/cm ³]	248
Average specific power density [W/g HM]	21.0
Cycle length [y]	8.8
Average core burnup at BOEC/EOEC [% FIMA]	19.9/26.9
HM mass at BOEC/EOEC [tons]	114.4/104.5
TRU mass at BOEC/EOEC [tons]	10.2/11.5
Conversion ratio	1.15
Discharge burnup [GWd/tHM]	540.8
Burnup reactivity loss [% Δk]	4.4
Core average flux [10 ¹⁵ n/cm ² -s]	1.8
Fast flux fraction > 0.1 MeV [%]	65.0
Axial neutron leakage at BOEC/EOEC [%]	4.1/4.1
Radial neutron leakage at BOEC/EOEC [%]	1.9/4.1

Table 8: Discharged Fuel Heavy Metal Composition of Breed and Burn Core and of Once-through LWR

	LWR		Breed & Burn	
	gram/GWed	wt% of TRU	gram/GWed	wt% of TRU
U-234	0.3	-	0.3	-
U-235	765.4	-	0.1	-
U-236	384.4	-	0.9	-
U-238	54887.5	-	1689.1	-
Np-237	45.1	5.0%	1.8	0.5%
Np-239	5.5	0.6%	0.2	0.1%
Pu-238	20.2	2.2%	2.7	0.7%
Pu-239	468.0	52.2%	224.4	62.1%
Pu-240	169.1	18.9%	109.4	30.3%
Pu-241	125.3	14.0%	11.9	3.3%
Pu-242	40.5	4.5%	6.0	1.7%
Am-241	4.7	0.5%	3.7	1.0%
Am-242M	0.1	0.0%	0.1	0.0%
Am-243	8.1	0.9%	0.7	0.2%
Cm-242	1.5	0.2%	0.1	0.0%
Cm-244	7.5	0.8%	0.3	0.1%

Table 9: Selected Fuel Cycle Characteristics of the Breed and Burn Core Versus Once-through LWR

Characteristic	LWR	Breed & Burn
Feed fuel type	Enriched U	Depleted U
^{235}U wt. %	4.5	0.2
Discharge burnup ($\text{GW}_{\text{thD}}/\text{tHM}$)	50	541
Natural uranium utilization		
1. GW_{eD} generated per ton U_{nat} ; FR uses same amount of U_{nat}	1	109
2. GW_{eD} generated per SWU; FR uses enriched U starter fuel + leftover U_{dep}	1	>90
GW_{eY} generated by all U_{dep} in world / GW_{eY} presently generated in world	1	3000
GW_{eY} generated by U_{dep} made per year / GW_{eY} presently generated in world	1	100
Discharged fuel		
TRU per GW_{thD} (g/ GW_{thD})	299	144
Relative amount of TRU per GW_{eD} (%)	100	40.3
Relative amount of Pu per GW_{eD} (%)	100	43.1
Fissile/total Pu at EOEC (%)	64.5	66.7
^{238}Pu /total Pu at EOEC (%)	3.2	0.8
$^{237}\text{Np} + ^{241}\text{Pu} + ^{241}\text{Am} + ^{245}\text{Cm}$ (g/ GW_{thD})	58.6	6.9
$^{99}\text{Tc}/^{129}\text{I}/^{135}\text{Cs}$ (g/ GW_{thD})	23.1/5.2/15.3	14.1/7.6/40.1
Decay heat 1 year after discharge (W/kg Pu)	16.8	7.9
Neutron emission 1 year after discharge (n/s/kg Pu)	4.58E+05	2.09E+05
Radio-toxicity (m^3 of water/ GW_{eD})		
HM 1 year after discharge	1.36E+10	2.42E+09
HM 30 years after discharge	1.03E+10	2.14E+09
FP 1 year after discharge	4.71E+10	1.43E+10
FP 30 years after discharge	8.54E+09	6.52E+09
Relative radio-toxicity per GW_{eD} (%)		
HM 1 year after discharge	100	17.8
HM 30 years after discharge	100	20.8
FP 1 year after discharge	100	30.4
FP 30 years after discharge	100	76.3
Relative decay heat per GW_{eD} (%)		
1 year after discharge	100	12.2
30 years after discharge	100	49.1
Relative spontaneous neutron yield per GW_{eD} (%)		
1 year after discharge	100	7.2
30 years after discharge	100	8.1

Table 10: Time of shuffling/reconditioning for the initial and new B&B cores

<i>Core</i>	Initial core		Initial core		New core	
<i>Driver</i>	Fresh EU		Once-burnt DU		Once-burnt EU	
<i>Operations</i>	EFPY	Event	EFPY	Event	EFPY	Event
	3.8	shuffling	2.2	shuffling	4.9	shuffling
	6	shuffling	3.3	shuffling+ reconditioning	9.6	shuffling+ reconditioning
	7.1	shuffling	4.9	shuffling	15.1	B&B mode established
	8.2	shuffling	6.3	shuffling		
	9.3	shuffling	7.9	shuffling		
	10.1	new core spawning	13.1	shuffling		
			15.0	B&B mode established		

Table 11: Active Fuel Length and ^{235}U Mass at BOL

Batch	1	2	3	4	5	6	7	8	9	10	11	12
Fuel length [cm]	209	100	209	66	65	209	73	82	115	209	209	209
^{235}U wt%	0%	20%	0%	20%	20%	0%	20%	20%	20%	0%	0%	0%
^{235}U mass [tons]	0	1.13	0	0.75	1	0	0.82	1	1.3	0	0	0

Table 12: Thermal Hydraulic Performance of the B&B Core for a Radial Power Peaking
Factor of 1.67

Parameters	Case 1	Case 2	Case 3	Case 4
p/d	1.11	1.11	1.11	1.11
Fuel rods per assembly	271	169	127	91
Fuel rod diameter [mm]	8.16	10.33	11.91	14.07
Fuel rod pitch [mm]	9.06	11.48	13.24	15.64
Cladding thickness [mm]	0.667	0.845	0.974	1.151
Coolant volume fraction	26.6%	26.6%	26.6%	26.6%
Reynolds number	8.39E+04	1.18E+05	1.44E+05	1.80E+05
Pressure drop [MPa]	1.00	1.00	1.00	1.00
Coolant velocity [m/s]	8.75	9.91	10.61	11.40
Maximum core power [MW _{th}]	3020.7	3421.2	3662.9	3935.6
Peak linear heat rate [kW/m]	15.8	28.7	40.9	61.3
Max. inner cladding temp. [°C]	596.7	610.9	624.1	645.7
Peak. fuel temperature [°C]	644.6	714.0	792.5	923.5

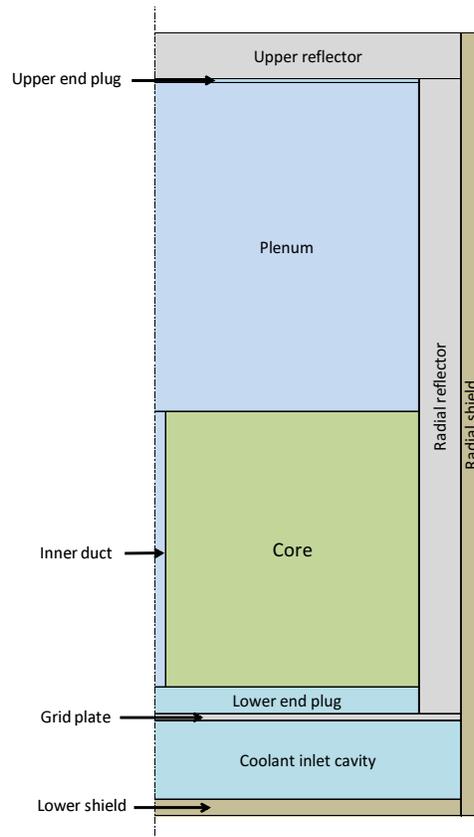


Figure 1: Layout of the B&B core and surrounding regions

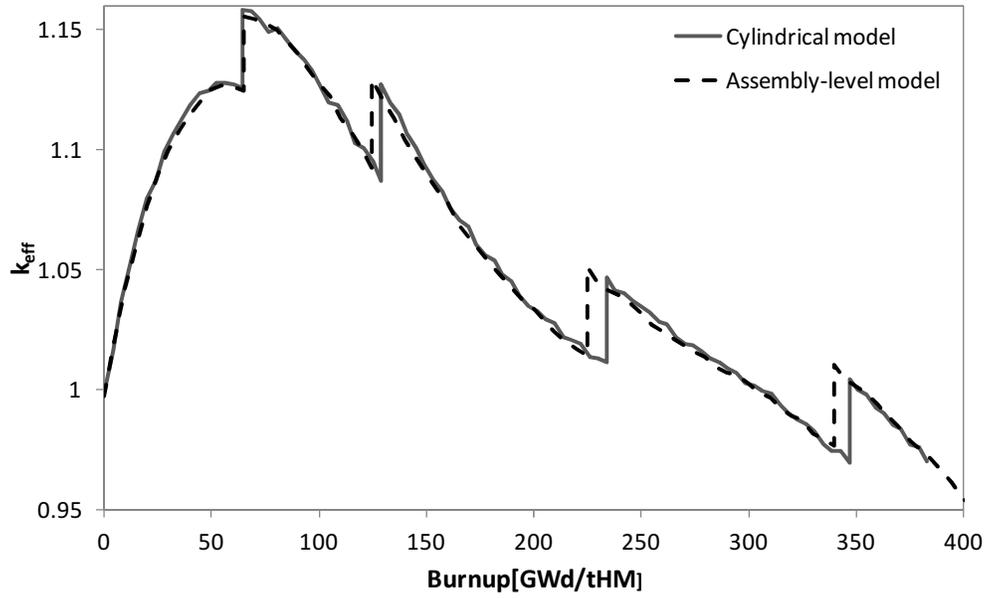


Figure 2: k_{eff} evolution for a couple of representations of the B&B core: a cylindrical core model and an assembly-level core model

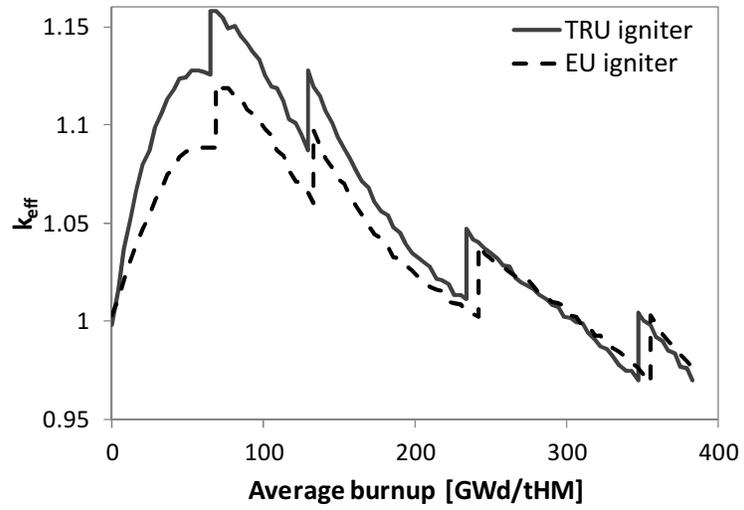


Figure 3: k_{eff} evolution during the B&B core transition period using a TRU or EU starter and melt-refining fuel reconditioning

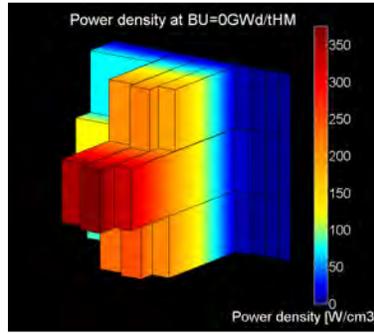


Figure 4: Power density distribution in the B&B core with TRU starter at BOL

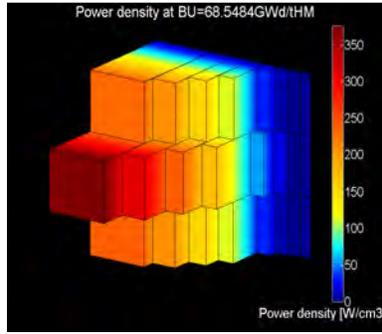


Figure 5: Power density distribution in the B&B core with TRU starter at 69 GWd/tHM (peak power)



Figure 6: Power density distribution in the B&B core with TRU starter at EOL (~300 GWd/tHM)

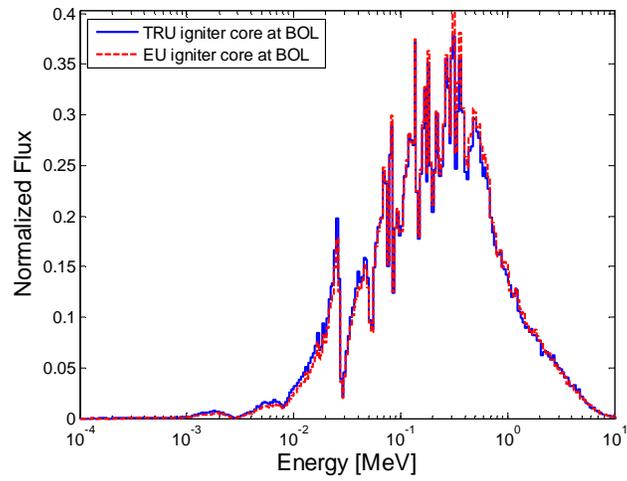


Figure 7: Spectra comparison at BOL for the fourth fuel batch of the B&B transition core with TRU starter and EU starter (igniter=starter)

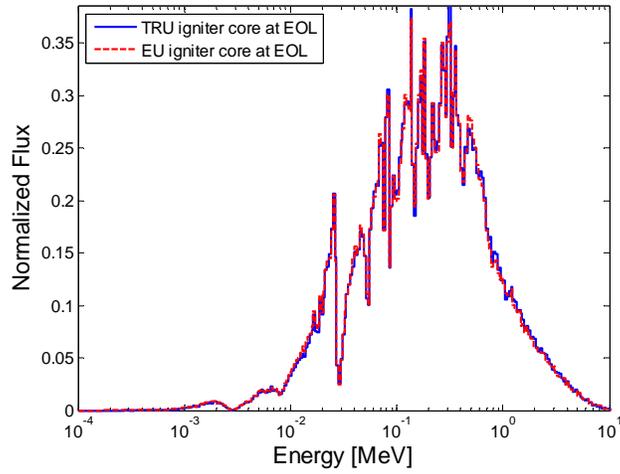


Figure 8: Spectra comparison at EOL for the fourth fuel batch of the B&B transition core with TRU starter and EU starter (igniter=starter)

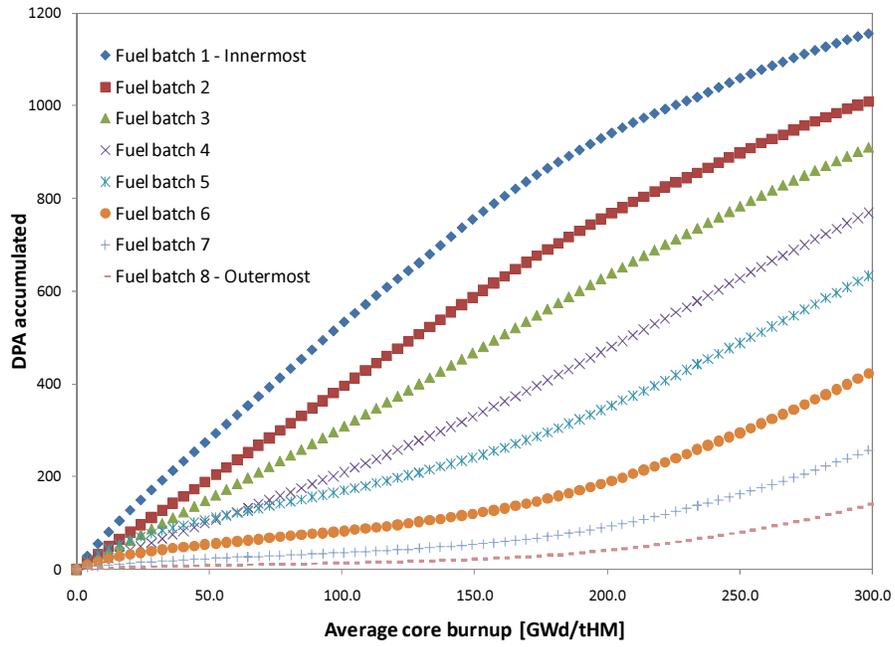


Figure 9: Peak DPA accumulation in the different fuel batches during the transition period of the B&B core

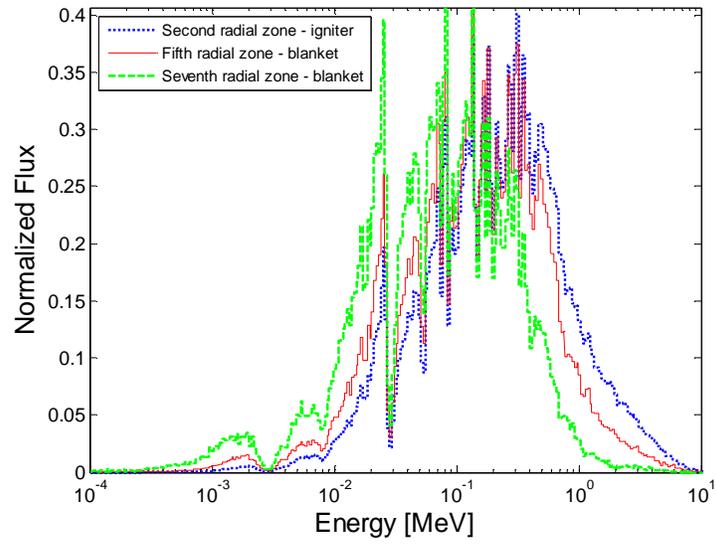


Figure 10: Neutron spectra at BOL for the second innermost EU starter fuel batch, and two of the blanket fuel batches of the transition core (igniter=starter)

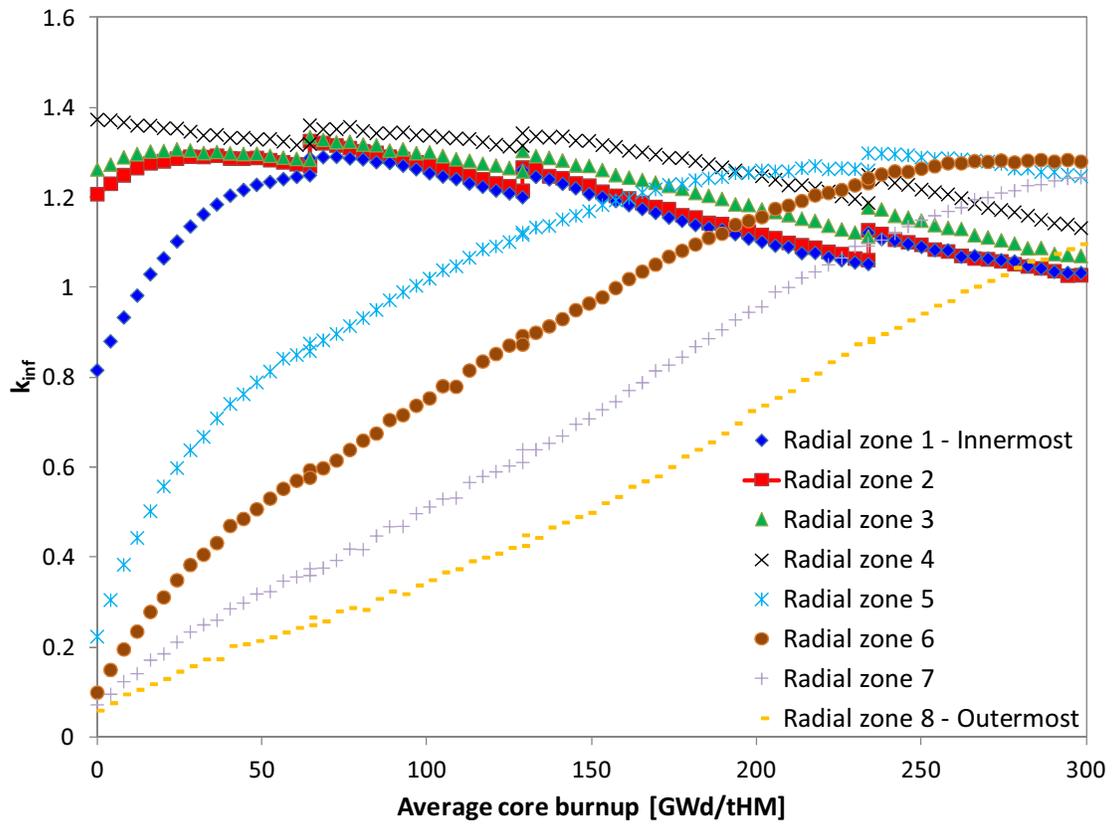


Figure 11: k_{∞} evolution of for the eight fuel batches of the B&B transition core with the TRU starter

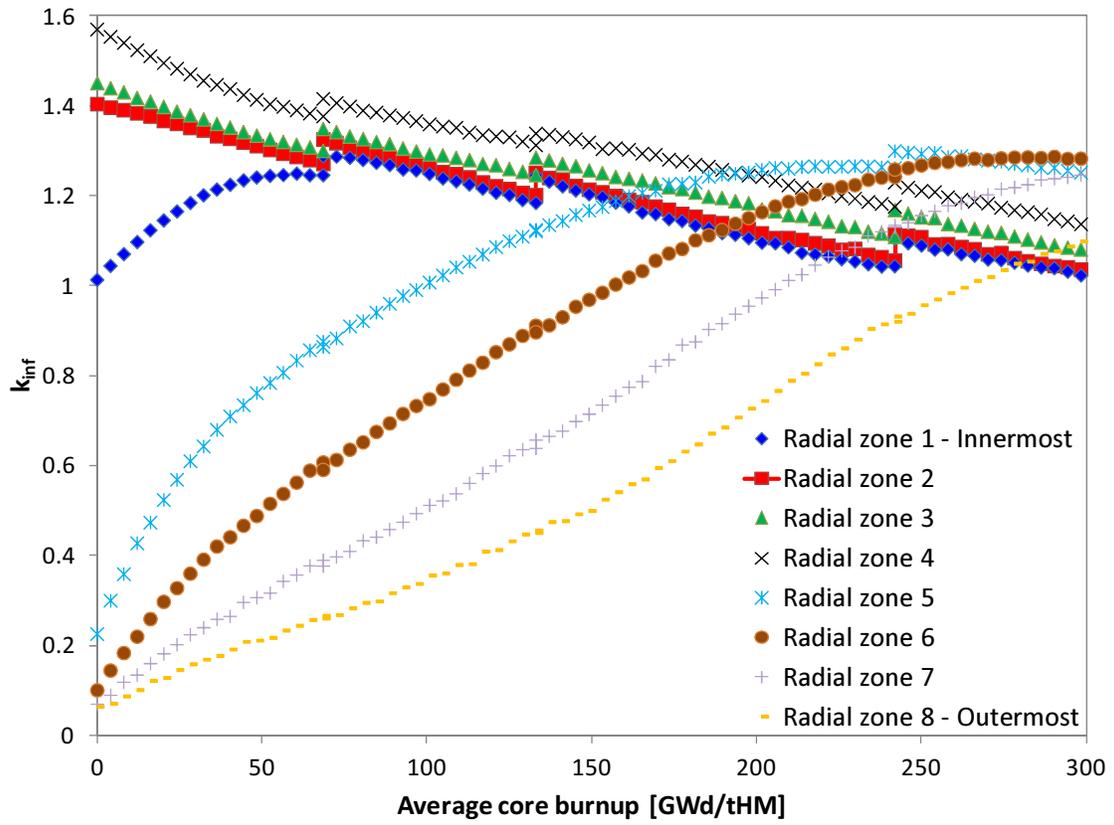


Figure 12: k_{∞} evolution for the eight fuel batches of the B&B transition core with the EU starter

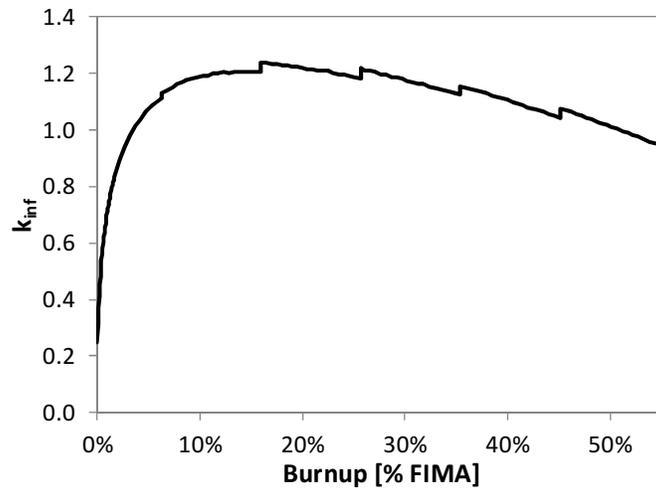


Figure 13: k_{∞} evolution with burnup of a depleted uranium unit cell (U-Zr 10%)

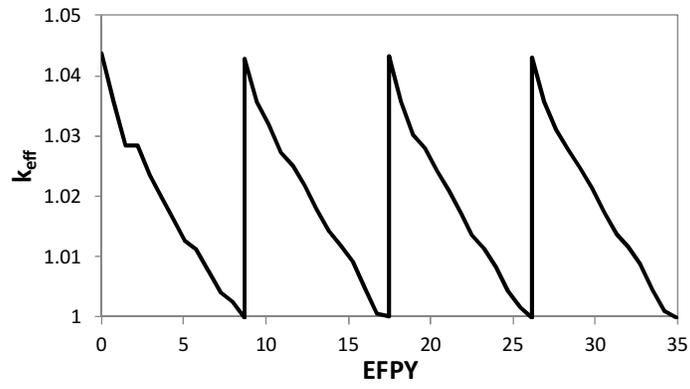
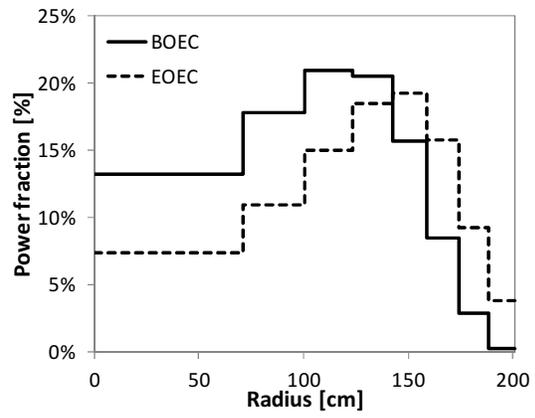
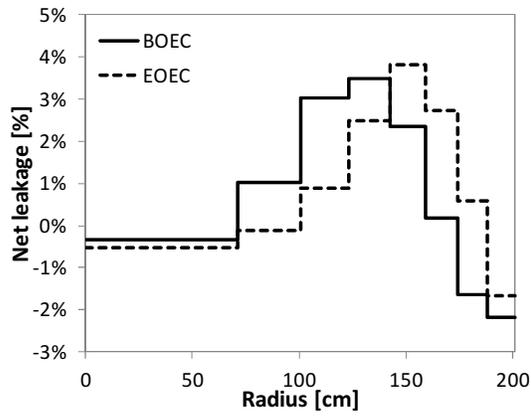


Figure 14: k_{eff} evolution with burnup for four equilibrium cycles of the B&B core with a fuel discharge burnup of 55% FIMA



Figures 15 and 16: Bath-wise net total neutron leakage and power fraction distribution at BOEC and EOEC for the B&B core having a discharge burnup of 55% FIMA.

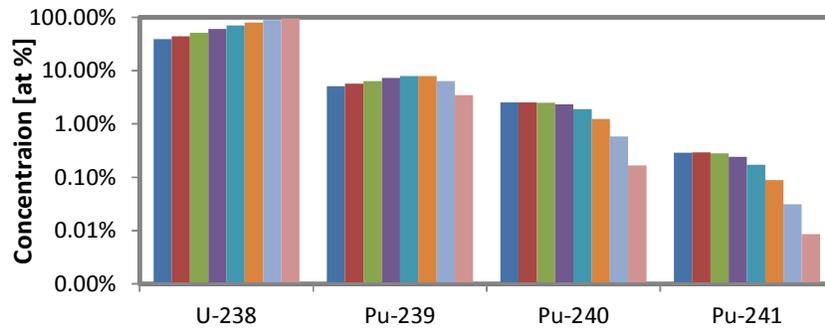


Figure 17: Fuel composition at EOEC, across the core, from the center (left bar of a given isotope) to the outer (right bar of a given isotope)

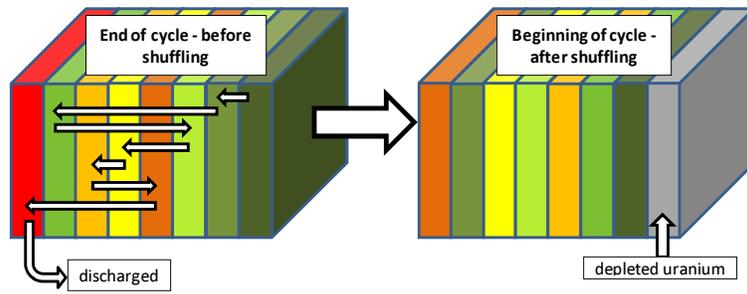
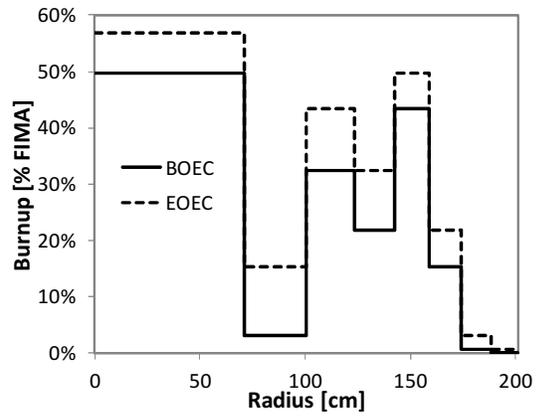
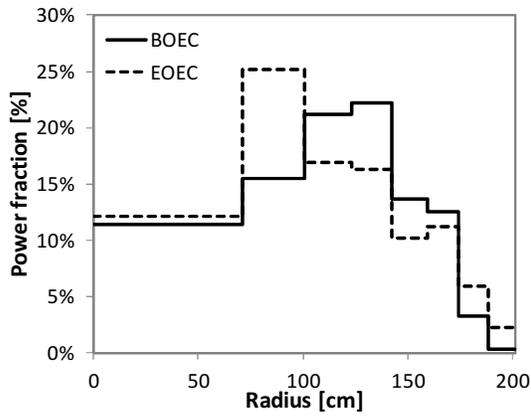


Figure 18: Schematic of the improved shuffling scheme used for the B&B core (core innermost fuel batch is the leftmost slab of the schemes)



Figures 19 and 20: Radial power and burnup distributions at BOEC and EOEC for the improved shuffling scheme for the B&B core

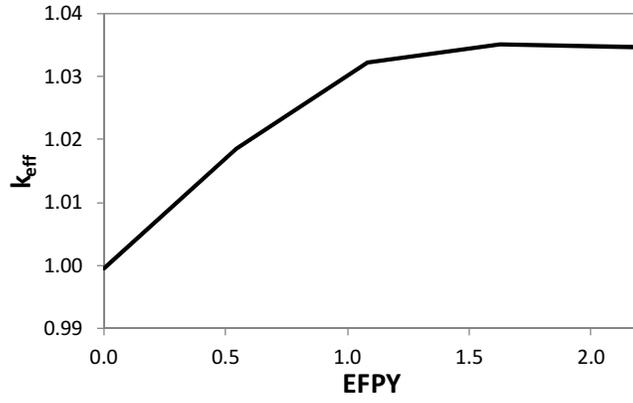


Figure 21: k_{eff} evolution for the B&B core operating with the minimum discharge burnup for the simple out-in shuffling scheme

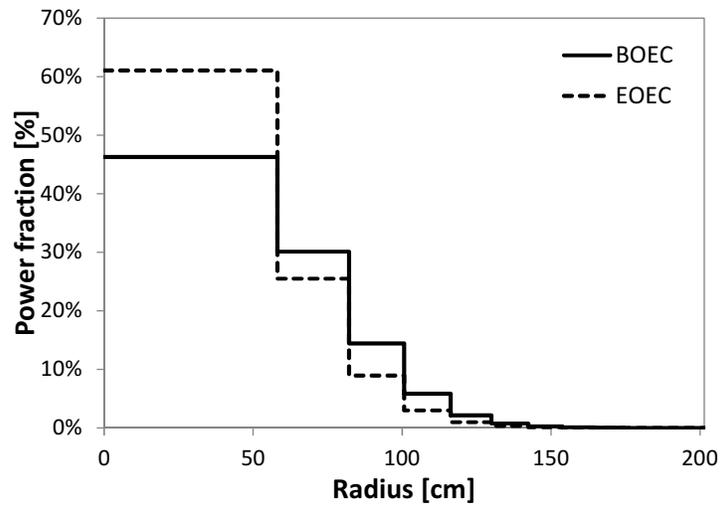


Figure 22: Power distribution at BOEC and EOEC for the B&B core operating with the minimum discharge burnup for the simple out-in shuffling scheme

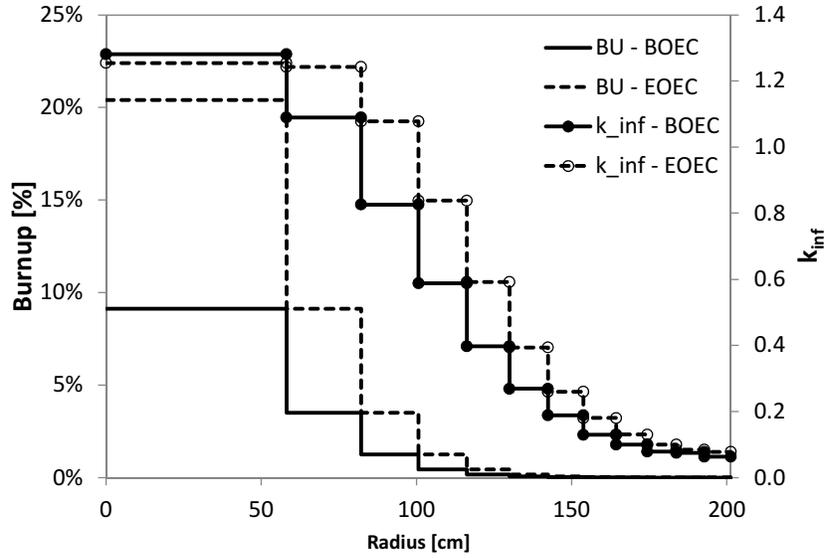


Figure 23: Burnup and k_{inf} distribution at BOEC and EOEC for the B&B core operating with the minimum discharge burnup for the simple shuffling scheme; 12 batches.

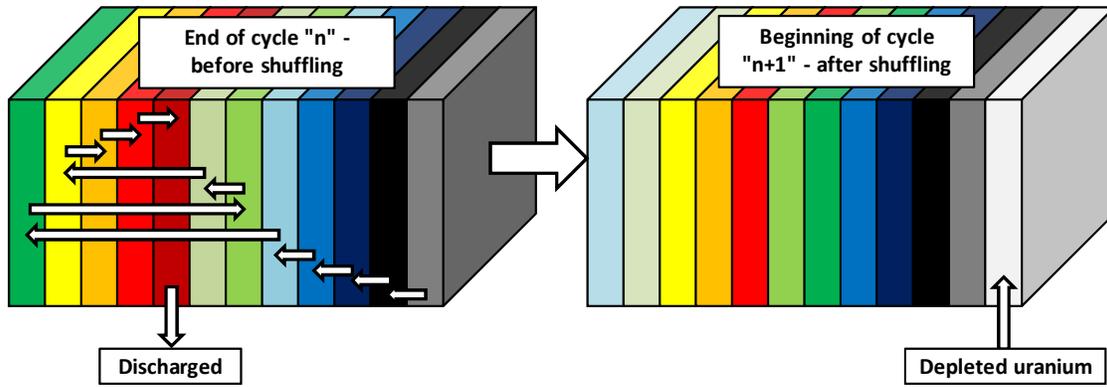


Figure 24: Schematic of the improved shuffling scheme pattern for minimum discharge burnup B&B core

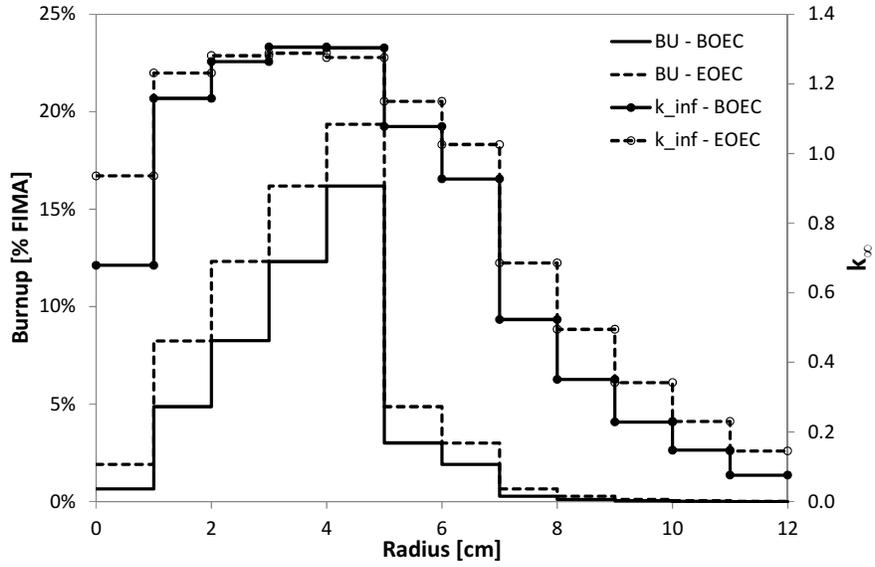


Figure 25: Burnup and k_{∞} radial distribution at BOEC and EOEC for the B&B core with the improved shuffling scheme for minimum discharge burnup

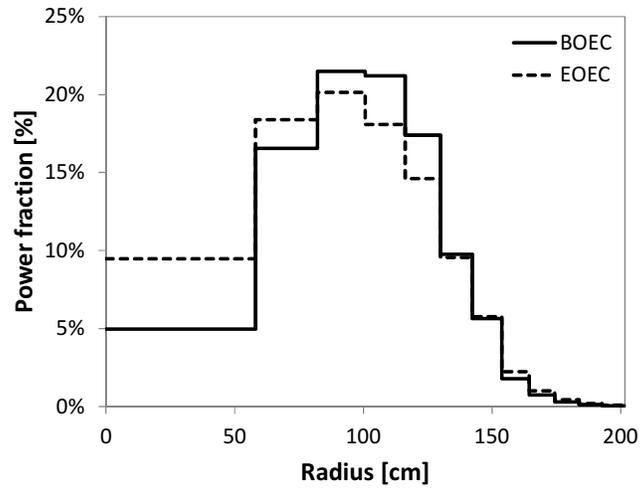


Figure 26: Radial power distribution at BOEC and EOEC for the B&B core with the improved shuffling scheme for the minimum discharge burnup

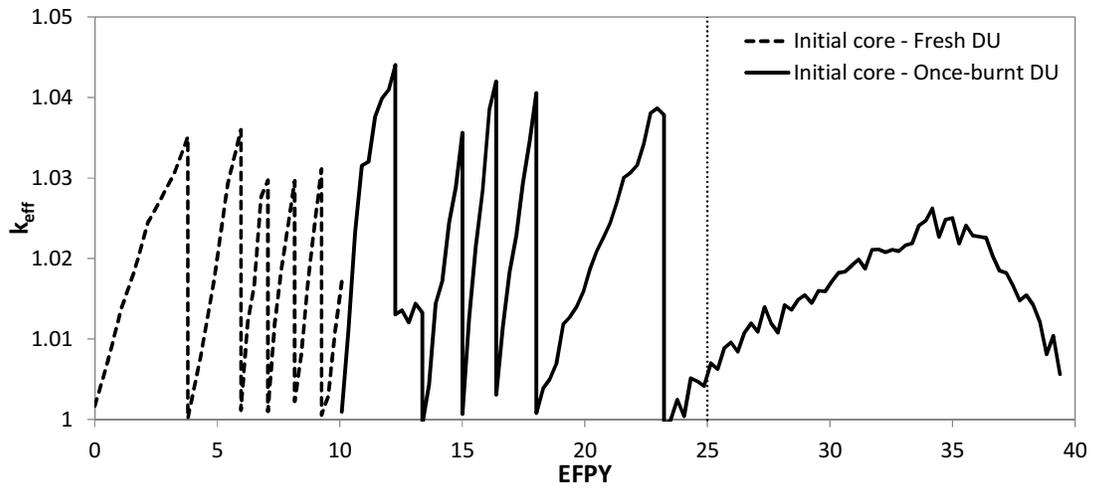


Figure 27: Evolution of k_{eff} for the initial B&B core driven by the EU starter (dotted curve) and then by the once-burnt DU fuel (plain curve)

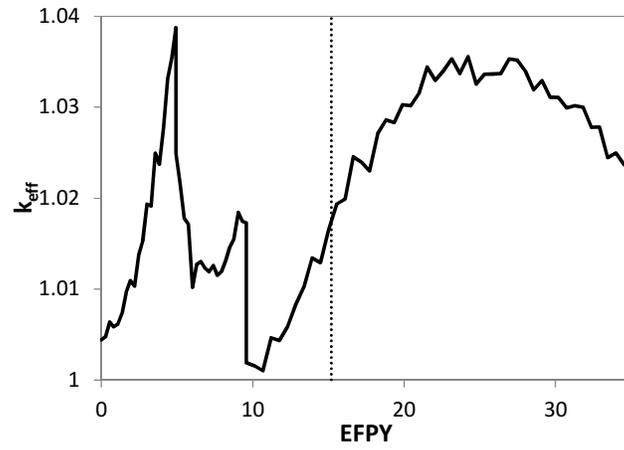


Figure 28: Evolution of k_{eff} for the new B&B core driven by the once-burnt EU starter

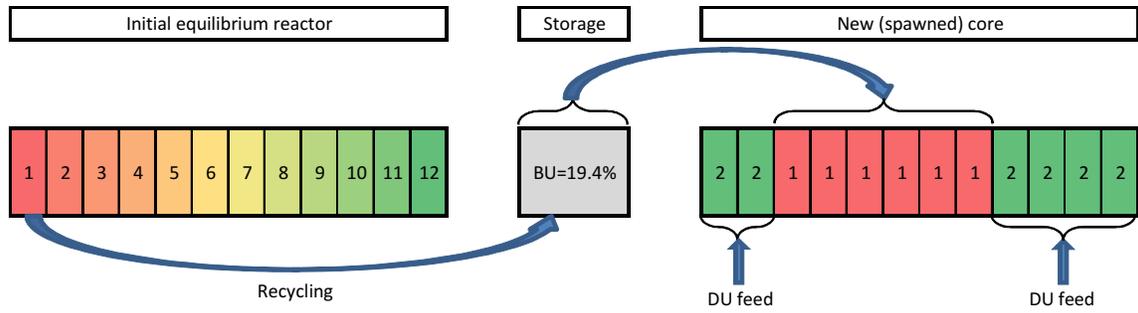


Figure 29: Schematic of core spawning from an equilibrium B&B core from which the fuel is discharged at 19.4% FIMA

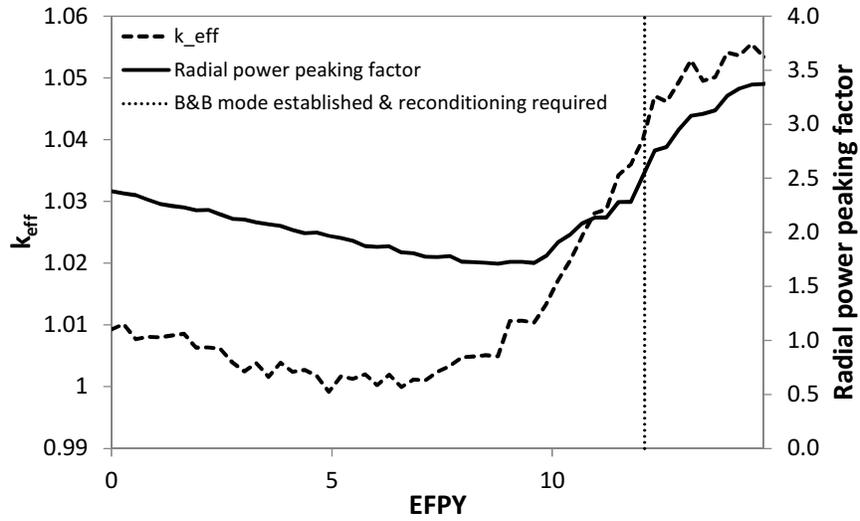


Figure 30: Evolution of k_{eff} and the radial power peaking factor for the new B&B core driven by the fuel discharged from the equilibrium B&B core

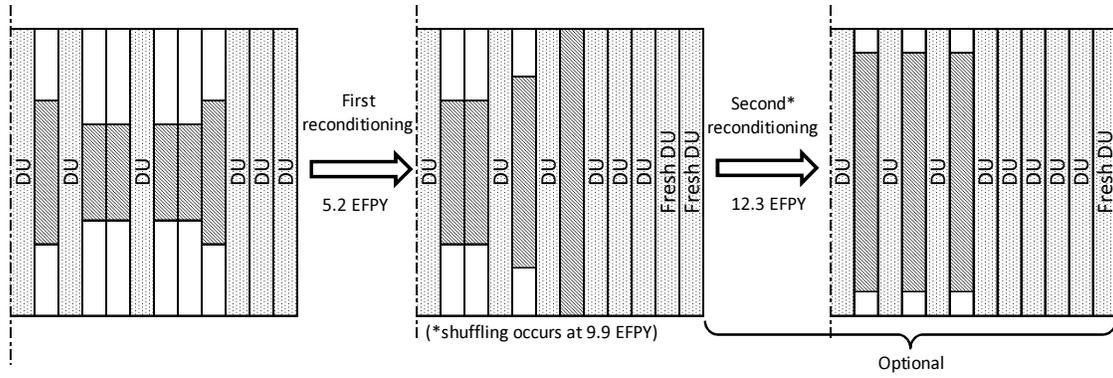


Figure 31: Optimized core layout with reduced length EU fuel assemblies at BOL – left, after the first reconditioning – middle, and after the second reconditioning – right

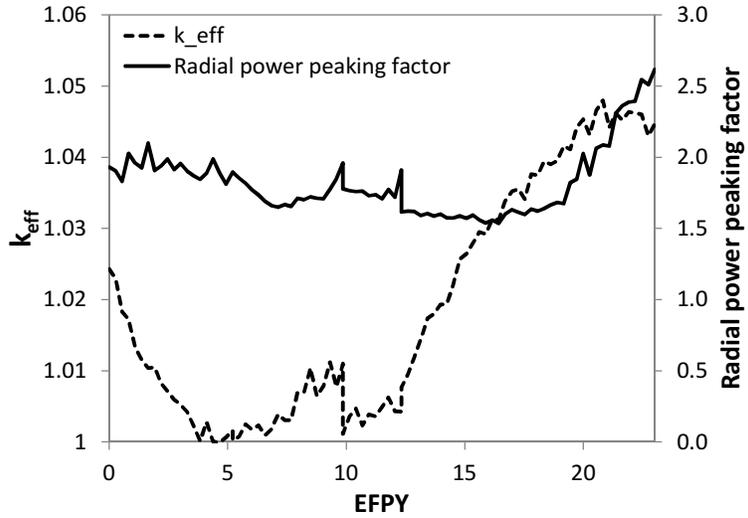


Figure 32: Evolution of k_{eff} and radial power peaking factor for the B&B core with a reduced EU inventory – shuffling occurs at 5.2, 9.9 and 12.3 EFY; reconditioning occurs at 5.2 and 12.3 EFY

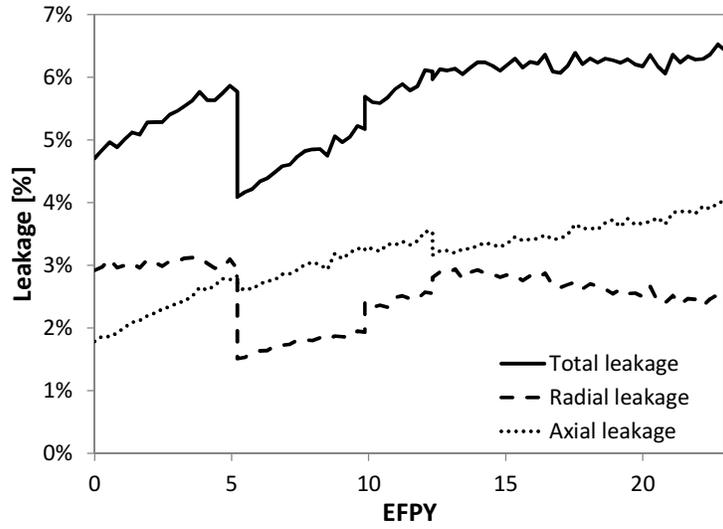


Figure 33: Evolution of the leakage probability for the B&B core with a reduced EU inventory

Neutron Balance Analysis for Sustainability of Breed-and-Burn Reactors

Florent Heidet* and Ehud Greenspan

University of California, Department of Nuclear Engineering
Berkeley, California 94720-1730

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Abstract—One objective of the present work is to determine the minimum burnup (BU) required to sustain a breed-and-burn (B&B) mode of operation in a large 3000-MW(thermal) sodium-cooled fast reactor core fed with depleted uranium-based metallic fuel. Another objective is to assess the feasibility of using the fuel discharged at the minimum required BU for fabricating the starter of an additional B&B core without separation of actinides and most of the solid fission products. A melt-refining process is used to remove gaseous and volatile fission products and to replace the cladding when it reaches its 200 displacements per atom radiation damage limit. Additional objectives are to assess the validity of a simplified zero-dimensional (0-D) neutron balance analysis for determination of the minimum BU required and the maximum BU attainable in a B&B mode of operation and to apply this 0-D methodology to assess the feasibility of establishing a B&B mode of operation in fast reactor cores made of different combinations of fuels, coolants, and structural materials.

It is found that the minimum BU required to sustain the B&B mode in the referenced depleted uranium-fueled B&B reactor is 19.4% FIMA. The number of excess neutrons that can be generated by the fuel discharged at 19.4% FIMA is found sufficient to establish the B&B mode in another B&B core. The net doubling time for starting new B&B reactors with fuel discharged from operating B&B reactors is 12.3 yr.

The minimum BU required to sustain the B&B mode of operation in alternative core designs was found to be 29% FIMA when using Pb-Bi coolant with metallic uranium fuel and 40% FIMA when using nitride fuel with sodium coolant. The B&B mode of operation cannot be established using thorium fuel and liquid-metal coolant.

The results derived from the neutron balance analysis strongly depend on the value of the estimated neutron leakage probability and the fraction of neutrons lost in the reactivity control systems. A neutron balance performed using a simplified 0-D core model, although not accurate due to, primarily, inaccurate spectra predictions, provides reasonable estimates of the minimum required and the maximum attainable BUs despite the fact that its k_{∞} evolution prediction is inaccurate. The 0-D approach can save much computational effort and time and is found to be useful for scoping analysis.

I. INTRODUCTION

The University of California, Berkeley, is studying breed-and-burn (B&B) reactor cores that are conceptually similar to the Tokyo Institute of Technology CANDLE reactor¹ and, in particular, to the TerraPower traveling wave reactors.² Except for the initial critical fissile fuel loading, the B&B reactor type is to be fueled

with fertile material. The basic principle of the B&B reactor concept is to have a sufficiently high breeding gain to build up the fissile concentration in blanket elements efficiently enough to enable these fuel elements to generate, before reaching their radiation damage limit, at least as many excess neutrons as they captured in the process of converting them to net neutron producers. If this condition can be achieved, it will be possible to operate fast reactors on fertile fuel feed without the need for fuel reprocessing.

*E-mail: fheidet@anl.gov

The objective of the present work is to perform a neutron balance analysis of a large B&B reactor core to quantify the minimum burnup (BU) the feed fuel needs to accumulate in order to sustain the B&B mode of operation. Another objective is to assess the feasibility of using the fuel discharged at the minimum required BU to fabricate the initial critical mass (the “starter”) of a new B&B core. If feasible, the corresponding doubling time of B&B reactors is to be quantified as well.

The neutron balance analysis uses a more generalized version of the methodology recently applied to simplified zero-dimensional (0-D) B&B cores.^{3,4} The resulting methodology enables quick assessment of the feasibility of the B&B mode of operation of various core design concepts using a simplified core geometry. However, to obtain exact quantitative estimates, it is necessary to use a full core model with a realistic fuel shuffling scheme and to account for realistic neutron losses by leakage and to reactivity control elements.

The methodology used is presented in Sec. II. The results obtained with this methodology for the full core model⁵ and the 0-D core models are discussed in Sec. III for the equilibrium period, when the core is fed only with depleted uranium. The results obtained for the full core model during the transition period, when the core is driven by a starter made of either a transuranic (TRU) containing fuel or an enriched uranium (EU) fuel, are discussed in Sec. IV. A preliminary assessment of the feasibility of establishing the B&B mode of operation for different types of fuel and/or coolant is provided in Sec. V.

II. METHODOLOGY

II.A. Fast Reactor Cores Studied

The large B&B core is studied using two core models: a full core model and a 0-D core model. The full core model provides accurate results as well as the neutron leakage probability and BU reactivity swing to be controlled by the reactivity control systems. The 0-D core model provides less accurate results and does not provide the neutron leakage or the BU reactivity swing but requires significantly less computation time and is useful for parametric studies.

II.A.1. Large B&B Core

The large sodium-cooled fast reactor core examined is a B&B core using ternary metallic fuel U-Pu-Zr with 10 wt% zirconium, a fuel density of 15.85 g/cm³, and a smear density of 75% to accommodate the fuel swelling with BU. This corresponds to an initial heavy metal atom density of 3.6×10^{22} atoms/cm³ of fuel. To obtain a lower bound on the minimum required BU, the fuel and gap volume fraction is taken to be 50%, corresponding

TABLE I
Constituents Volume Fractions of the B&B Core

Volume Fractions	(%)
Fuel	37.5
Gap	12.5
Cladding	22.0
Coolant	28.0

to a pitch-to-diameter (P/D) ratio of 1.122—near the lower limit used in liquid sodium-cooled reactors that feature a hexagonal lattice.⁶ To simplify the core design and get upper bound estimates, no control assemblies, gas expansion modules, assembly ducts, or interduct gaps are accounted for in calculating the volume fractions presented in Table I. The cladding thickness was assumed to be 20% of the inner cladding diameter.

The radial central part of the B&B core is initially loaded with “starter” fuel made either of EU or of depleted uranium mixed with TRU recovered from light water reactor used nuclear fuel that was discharged at 50 GWd/tonne HM and has been cooled for 10 yr. This starter fuel zone is radially surrounded by a depleted uranium blanket that is surrounded by a thin radial reflector followed by a shield, as shown in Fig. 1. The dimensions, partially derived from the Argonne National Laboratory advanced burner reactor design,⁷ and material composition of the various components are given in Table II. The depleted uranium radial blanket volume is equal to the starter fuel volume. The active core height is 2.09 m and the active core diameter is 4.03 m in order to meet the power level mentioned below. The starter region and the radial blanket are each divided into four equal-volume concentric BU zones, each of which is divided into three equal-volume axial BU zones. The total mass of heavy metal initially loaded in the core is 142 tons.

The total core power is 3000 MW(thermal), corresponding to an average power density in the starter region of 225 W/cm³ (112.5 W/cm³ overall), yielding a peak power density within the imposed constraint of 450 W/cm³. This constraint is based on the existing fast reactor design database of the International Atomic Energy Agency.⁶ To minimize the beginning-of-life (BOL) radial power peaking factor while maintaining criticality, the starter volume was divided into four equal-volume radial zones and loaded with fissile fuel of increasing concentration from the innermost to the outermost radial zone of the starter region (inner half of the core). The optimal concentrations found are 6.6%/11.7%/12.2%/15.3% TRU by weight for the TRU-based fuel and 6.1%/10.8%/11.3%/14.8% ²³⁵U by weight when using EU.

At the end of a BU cycle, the fuel assemblies in the highest BU radial zone are discarded, and all the rest of

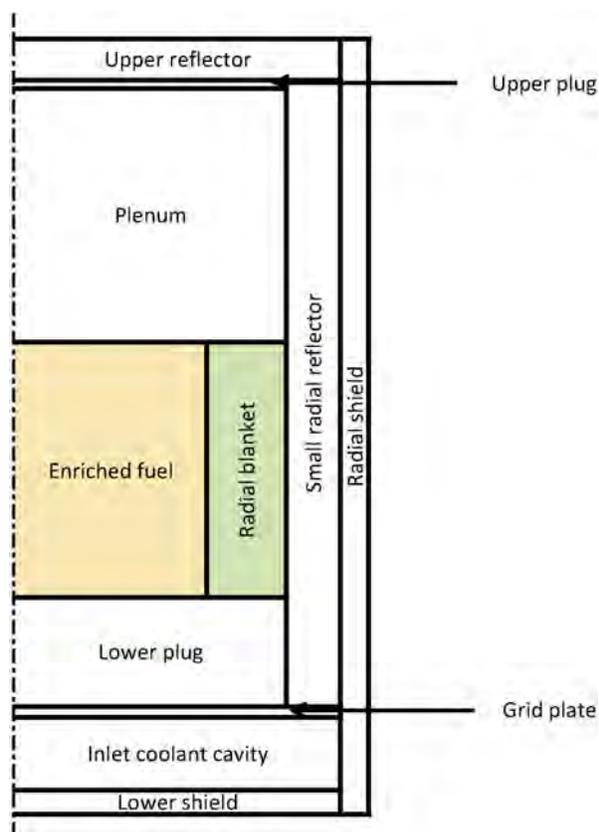


Fig. 1. Layout of core and surrounding regions for the B&B core studied.

the fuel assemblies are assumed to be discharged, undergo the melt-refining process (defined below), reloaded, and reloaded. The discharged fuel has axially varying composition—the heavy metal concentration tends to

decrease and the fission products concentration tends to increase from the ends toward the axial center of the fuel rod. After the melt-refining process, the fuel rods have axially uniform composition. The recycled fuel assemblies are then reloaded in the core according to the shuffling scheme assumed. Fresh depleted uranium assemblies are loaded at the core outermost zone so as to minimize neutron loss via leakage. This shuffling process followed by BU analysis is continued until an equilibrium core composition is reached. For this preliminary study, the entire process of fuel recycling described above is assumed to be instantaneous. This assumption causes an overestimation of the beginning of cycle concentration of ^{241}Pu and other short-lived actinides but has insignificant effect on the conclusions of this study. Future studies will simulate the fuel recycling process more realistically by recycling a fuel batch only when it reaches its displacement per atom limit and by accounting for the time it takes to cool the discharge fuel, to take it through the melt-refining process, and to refabricate new fuel assemblies. The fuel density change with BU will also be estimated to determine whether or not it is possible to add depleted uranium to the fuel during the recycling.

II.A.2. Zero-Dimensional Core

The 0-D core model is a unit cell having reflective boundary conditions and made of fuel, cladding, and coolant with the same volume fraction as for the B&B core given in Table I. The metallic fuel characteristics are also the same as for the B&B core. Rather than simulating the depletion of a fuel assembly as it is shuffled across the full core while being exposed to variable flux amplitudes and spectra both depending on the whole core composition, the 0-D unit cell is initially made of depleted uranium and is undergoing BU while exposed to a

TABLE II
Dimensions and Composition of the Regions Modeled for the B&B Core Studied

Region	Height (cm)	Thickness (cm)	Material (vol %)	T (K)
Upper reflector	34.9	242.2	50% HT9–50% Na	783
Upper end plug	2.5	201.4	22% HT9–78% Na	783
Plenum	209.4	201.4	22% HT9–28% Na	783
Enriched fuel	209.4	142.4	37.5% Fuel–22% HT9–28% Na	800
Blanket	209.4	59.0	37.5% Fuel–22% HT9–28% Na	800
Lower end plug	90.4	201.4	22% HT9–78% Na	628
Grid plate	5.2	242.2	50% HT9–50% Na	628
Coolant inlet	60.0	242.2	22% HT9–78% Na	628
Lower shield	20.0	242.2	43.1% B ₄ C–29.7% HT9–27.2% Na	628
Radial reflector	511.6	40.8	50% HT9–50% Na	628
Radial shield	631.8	20.5	43.1% B ₄ C–29.7% HT9–27.2% Na	628

constant power density and to a neutron spectrum that depends solely on the unit cell composition. The fuel of the unit cell is depleted assuming the same power density as that of the B&B core average power density—112.5 W/cm³. The problem with imposing a fixed power density for BU analysis is that for the initially very low fissile material content, the neutron flux amplitude is highly overestimated. In reality, the neutron flux seen by the low BU fuel is the flux leaking out of the higher BU zone and is smaller than the average core value. The spectrum of the 0-D core was found (Sec. II.C) to be softer at low BUs and somewhat harder at high BUs than the spectrum of the fuel loaded in the B&B core.

II.B. Melt Refining

The melt-refining process has been developed for metallic fuel in the Experimental Breeder Reactor II project.⁸ The melt refining involves loading the declad fuel into a zirconia crucible and melting the mixture at ~1300°C for several hours under argon atmosphere. The gaseous and volatile fission products are released, and certain solid fission products are partially removed by oxidation with the zirconia of the crucible. Based on Ref. 8, it is assumed that this process can remove nearly 100% of Br, Kr, Rb, Cd, I, Xe, and Cs and at least 95% of Sr, Y, Te, Ba, and the rare earths (lanthanides). Thorium and americium are also oxidized with zirconia, and 95% of these two elements will be removed from the fuel.

II.C. Neutron Balance Analysis

The first phase in a B&B mode of operation consists of building up the fissile content in fertile feed element until its $k_{\infty} \times P_{NL} \times P_{RC}$ reaches unity, where P_{NL} is the neutron nonleakage probability and P_{RC} is the fraction of neutrons not captured in the reactivity control systems. The concept of neutron balance has been introduced in Refs. 3 and 4. As $k_{\infty} \times P_{NL} \times P_{RC}$ increases beyond unity, it produces excess neutrons and becomes a driver element. The theoretical minimum BU required for establishing the B&B mode of operation is that BU for which the total excess neutron production from the BU corresponding to $k_{\infty} \times P_{NL} \times P_{RC} = 1$ until reaching the minimum required BU equals the total neutron consumption (absorption in the core + leakage + absorption in the reactivity control systems) from loading the fuel until $k_{\infty} \times P_{NL} \times P_{RC}$ becomes 1.0. This condition is expressed by Eq. (1):

$$\int_0^{BU_{th}} P_{NL} \times P_{RC} \times \text{neutron production rate (BU)} \\ = \int_0^{BU_{th}} \text{neutron absorption rate (BU)} . \quad (1)$$

The neutron production and absorption rates are expressed by Eqs. (2) and (3),

$$\begin{aligned} & \text{neutron production rate (BU)} \\ &= N_{HM} \times \left(\bar{\nu}(\text{BU}) + \frac{2 \times \bar{\Sigma}_{n2n} + 3 \times \bar{\Sigma}_{n3n}}{\bar{\Sigma}_f} \right) \\ & \quad \times d\text{BU} \end{aligned} \quad (2)$$

and

$$\begin{aligned} & \text{neutron absorption rate (BU)} \\ &= \text{neutron production rate (BU)} / k_{\infty}(\text{BU}) , \end{aligned} \quad (3)$$

where N_{HM} is the heavy metal atomic density at BOL, $\bar{\nu}$ is the average number of neutrons emitted per fission, and the BU is measured in FIMA. $\bar{\Sigma}_{n2n}$, $\bar{\Sigma}_{n3n}$, and $\bar{\Sigma}_f$ are the average effective one-group $(n, 2n)$, $(n, 3n)$, and (n, f) macroscopic cross sections, and $d(\text{BU})$ is given in FIMA. The magnitude of the theoretical minimum required BU is that value of BU(thermal) for which Eq. (4) is satisfied:

$$\begin{aligned} & N_{HM} \int_0^{BU_{th}} \left[\frac{1}{k_{\infty}(\text{BU})} - P_{NL} \times P_{RC} \right] \\ & \quad \times \left(\bar{\nu}(\text{BU}) + \frac{2 * \bar{\Sigma}_{n2n} + 3 * \bar{\Sigma}_{n3n}}{\bar{\Sigma}_f} \right) \times d\text{BU} = 0 . \end{aligned} \quad (4)$$

As the $(n, 2n)$ and $(n, 3n)$ reaction rates are significantly smaller than the fission reaction rate, the equation used to determine the minimum required BU is

$$\begin{aligned} & N_{HM} \int_0^{BU_{th}} \left[1 - \frac{1}{k_{\infty}(\text{BU}) \times P_{NL} \times P_{RC}} \right] \bar{\nu}(\text{BU}) \times d\text{BU} \\ &= 0 . \end{aligned} \quad (5)$$

A similar neutron balance equation has already been discussed by Yu et al.⁹ in 2002 in studying B&B reactors. The evolution of the neutron balance can be graphically presented by plotting the left side of Eq. (5) as a function of BU_{th} . Since most of the depletion codes provide only discrete data, the graphical representation of the neutron balance is a convenient way to determine the minimum required BU. The plot can also be used to determine the maximum achievable BU while still enabling maintenance of the B&B mode of operation.

A difficulty in using the 0-D model for determining the BU range (minimum and maximum), which can support a B&B mode of operation, is the determination of the neutron leakage probability and the fraction of neutrons absorbed in control elements. Those values strongly depend on the core dimensions and fuel management

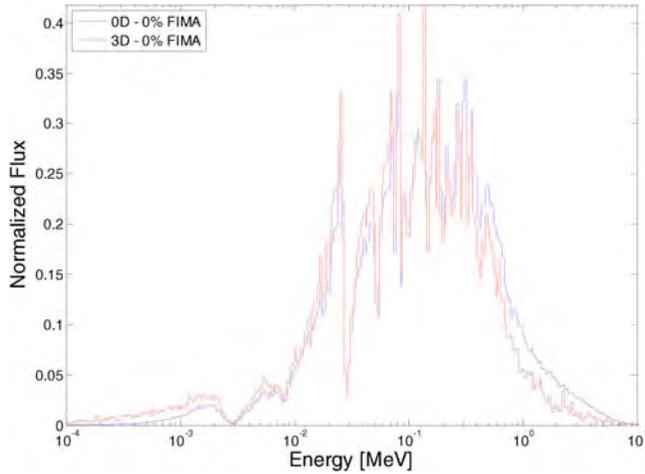


Fig. 2. Neutron spectrum comparison at 0% FIMA between the 0-D and full core models.

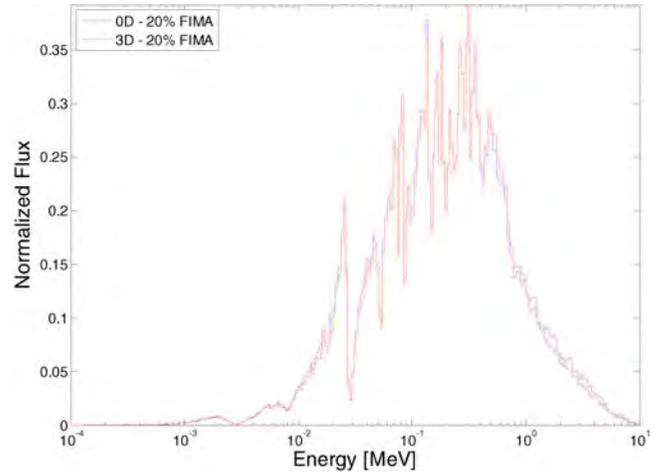


Fig. 3. Neutron spectrum comparison at 20% FIMA between the 0-D and full core models.

scheme. For example, changing the targeted fuel discharge BU or the fuel shuffling scheme affects the neutron leakage probability via, primarily, their effect on the radial power distribution. Nevertheless, the simple 0-D model is useful for performing quick scoping analysis. The simple 0-D model can also be used for estimating the effect of a postulated neutron loss fraction on the minimum BU required for establishing the B&B mode of operation. The initial spectrum the feed fuel is exposed to in the full core is significantly softer than in the 0-D model, as shown in Fig. 2, and the initial power density in the 0-D model is significantly higher. The feed fuel is loaded at the core periphery, where its spectrum is mostly due to the neutrons leaking out of the other fuel zones. As the fuel BU increases, the feed fuel is shuffled toward the core center, and the spectra of the two core models become consistent with each other, as shown in Fig. 3 at 20% FIMA.

III. NEUTRON BALANCE AT EQUILIBRIUM

III.A. Reference B&B Reactor

The performance analysis of the reference B&B core was performed in Ref. 5 when recycling the fuel with the melt-refining process. During operation, 75% of the fission gases are removed from the fuel, and at the end of a cycle the fuel is instantaneously recycled with the melt-refining process neglecting recycling loss. In this preliminary study, it is assumed that all the fuel assemblies are recycled at each end of equilibrium cycle (EOEC); discrete fuel recycling is economically preferable but does not change the results of this study. The same assumptions are used for the studies reported below. It was found possible to sustain the B&B mode with a discharge BU

ranging from 19.4% FIMA to 57% FIMA with appropriate shuffling schemes. Although the approach used in Ref. 5 is exact (subjected to the simplifying assumptions about the instantaneous recycling), it is computationally expensive and time consuming. It was found that the B&B core operating at 19.4% FIMA has an average cycle leakage probability of 4.4%, and 2.2% of the neutrons are absorbed, on the average, in the reactivity control systems required to compensate for the BU reactivity swing. The k_{eff} evolution of the B&B core at equilibrium is provided in Fig. 4 for two cycles. With a simple “out-in” shuffling scheme where the fuel discharged from one batch is reloaded, after recycling, in a zone immediately inward to the zone it was discharged from, the maximum discharge BU is 55% FIMA. For this discharge BU, the average neutron leakage probability of the B&B core is 5.95%, and 2.1% of the neutrons are absorbed in the reactivity control systems.

The neutron balance analysis approach described in Sec. II.C is applied to the full core and to the 0-D core models to confirm the minimum required BU and

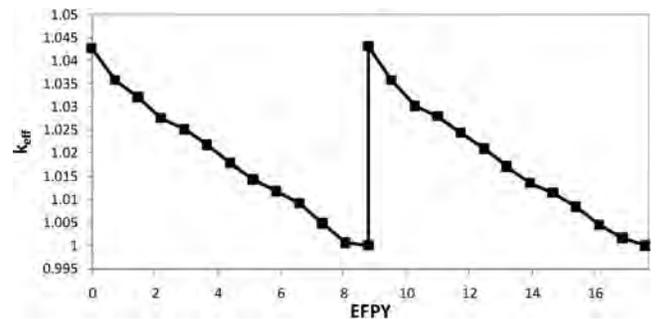


Fig. 4. The k_{eff} evolution of the B&B core at equilibrium.

maximum achievable BU for establishing and sustaining the B&B mode in the B&B core. It also provides the number of neutrons that need to be absorbed in the fuel to sustain the B&B mode and the number of excess neutrons that can still be produced by the fuel that is discharged from the B&B core. The latter information indicates whether or not this discharged fuel can be used as a starter for a new B&B core without having to purchase additional fissile fuel.

III.B. Neutron Balance of the Reference B&B Core

The BU range for a sustainable B&B mode is first determined for the reference B&B core described in Sec. II.A.1 when the fuel is shuffled and recycled with the melt-refining process. The neutron balance analysis for the full core model is performed using the BU-dependent infinite multiplication factor for each radial fuel zone that is obtained using Eq. (6):

$$k_{\infty} = \frac{\nu \times \sum_{i=1}^{i=3} [\phi_i \times \Sigma_{F,i}^{HM}]}{\sum_{i=1}^{i=3} [\phi_i \times (\Sigma_{A,i}^{HM} + \Sigma_{A,i}^{FP} + \Sigma_{A,i}^{Zr} + \Sigma_{A,i}^{Na} + \Sigma_{A,i}^{HT-9})]} \quad (6)$$

The reaction rates are summed over the three axial zones (i varies from 1 to 3). The average number of neutrons emitted per fission is assumed to be constant in all

the zones and equal to 2.92, and the neutron production from $(n, 2n)$ and $(n, 3n)$ reactions is assumed to be negligible. The zone flux ϕ_i and the effective one-group macroscopic cross sections $\Sigma_{x,i}$ are the values at a given time of the cycle and are inferred from the MCNP calculations of the full core model. For instance, the k_{∞} of the 0% FIMA fuel is obtained by using the cross sections and fluxes of the three axial zones of the outermost radial zone at beginning of equilibrium cycle, where the depleted uranium has been loaded.

The neutron balance for the reference B&B core with fuel recycling is performed by calculating the left side of Eq. (5) as a function of BU, assuming a neutron loss of 8.0% corresponding to the core being operated at its maximum BU. The neutron loss is predicted by the full core MCNP calculation. The k_{∞} evolution and the neutron balance obtained for the reference B&B core are shown in Fig. 5.

A negative sign in the neutron balance of Fig. 5 implies that there is a net cumulative neutron loss whereas a positive sign pertains to BUs over which a net excess of neutrons has been generated by a unit fuel volume. The discontinuities observed on the k_{∞} curve in Fig. 5 correspond to the fuel recycling and shuffling.

The maximum achievable BU is the discharge BU for which the lowest k_{eff} value of the B&B at equilibrium is equal to unity. In Fig. 5 it corresponds to the BU for which the neutron balance plot crosses the zero line for the second time: 55% FIMA. Approximately 1.9×10^{21} n/cm³ of fuel should initially be provided from the outside to convert a depleted uranium assembly into an

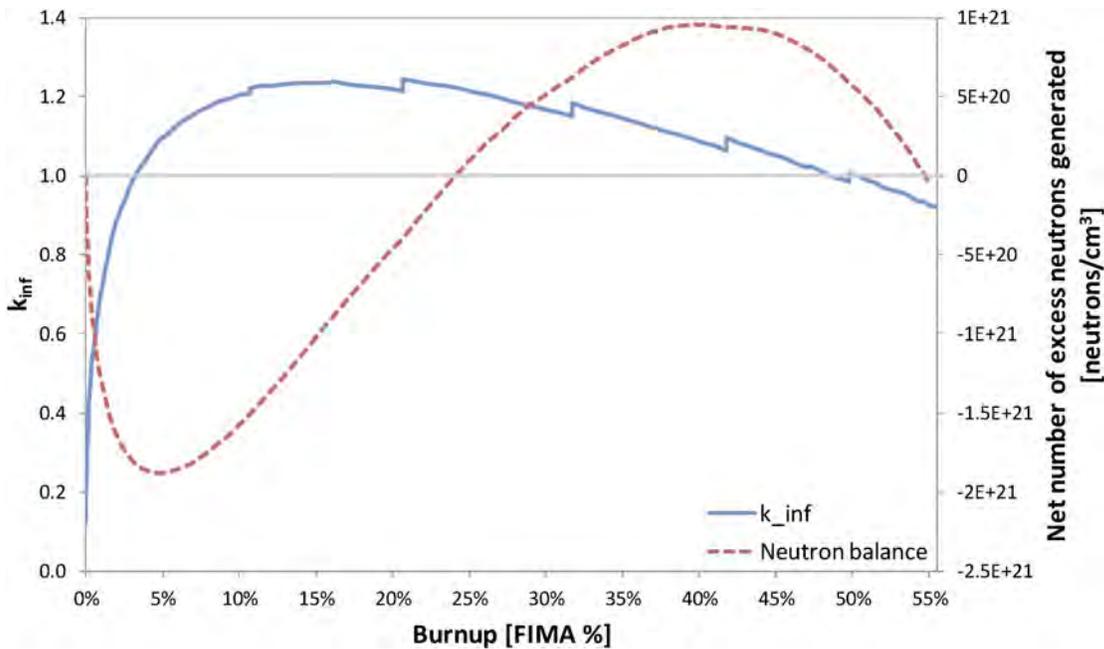


Fig. 5. The k_{∞} evolution and neutron balance for the reference B&B core with fuel recycling.

excess neutrons—producing assembly; the latter occurs at 5% FIMA. Thus, from 0% to 5% FIMA the assembly is a net consumer of neutrons, and from 5% to 24% FIMA it provides excess neutrons sufficient to make up for the net number of neutrons it had to absorb in order to get to 5% FIMA. This fuel assembly can maintain its $P_{RC} \times P_{NL} \times k_{\infty}$ above unity ($\Leftrightarrow k_{\infty} > 1/P_{RC} \times P_{NL} = 1/0.92$) up to 42% FIMA at which point it has an additional neutron excess of approximately 9.6×10^{20} n/cm³ of fuel, sufficient to extend the fuel BU up to $\sim 55\%$ FIMA.

The same maximum achievable BU as predicted by the neutron balance performed above, 55% FIMA, has been found in the full core analysis performed in Ref. 5 for the B&B core.

III.C. Impact of the Melt Refining

The impact of fuel recycling with the melt refining on the neutron economy is assessed. With no recycling, all the fission products remain in the fuel, and it conserves an axially varying composition. The k_{∞} evolution and neutron balance obtained for the B&B cores with and without fuel recycling (both with shuffling) are compared in Figs. 6 and 7, respectively. For the B&B core without fuel recycling, the neutron loss is taken to be 6.5% as predicted by the full core MCNP calculation. In case of the melt refining, the discontinuities amplitude observed in Fig. 6 is larger because of the reactivity change resulting from both the fission products partial removal and the fuel shuffling.

The maximum achievable BU is 43% FIMA for the core without recycling. The melt refining enables the increase of the maximum achievable BU by 28% ($= 55/43 - 1$) because almost one-third of the fission products are removed by this process, thus significantly decreasing the parasitic neutron capture. As a result, the k_{∞} values without recycling are up to 10% smaller at high BU. Despite the lower average neutron leakage probability and lower BU reactivity swing of the B&B core without recycling—both due to the lower discharge BU and, hence, smaller BU per cycle and more centrally peaked radial power distribution—the neutron economy is significantly impaired by the accumulated fission products.

For the B&B core without fuel recycling, because of the lower neutron loss fraction, approximately 1.7×10^{21} n/cm³ of fuel should be provided from the outside so as to make this fuel $P_{RC} \times P_{NL} \times k_{\infty} = 1.0$. When the fuel reached 22.5% FIMA, it generated sufficient numbers of excess neutrons to pay back for the number of neutrons it initially consumed. From 22.5% FIMA to 33% FIMA, it can provide an additional neutron excess of approximately 4.8×10^{21} n/cm³ of fuel, sufficient to extend the fuel BU up to $\sim 43\%$ FIMA.

III.D. Results from 0-D Analysis

The neutron balance results obtained with the 0-D core model are compared to those obtained in Sec. III.C

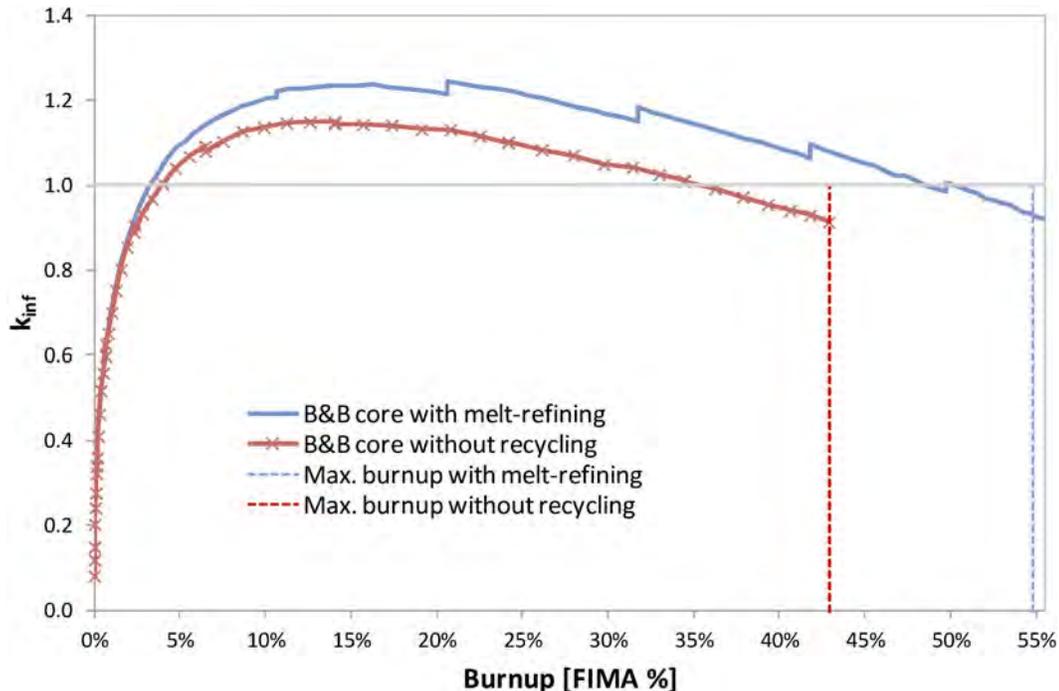


Fig. 6. The k_{∞} evolution with BU of the B&B cores with and without fuel recycling.

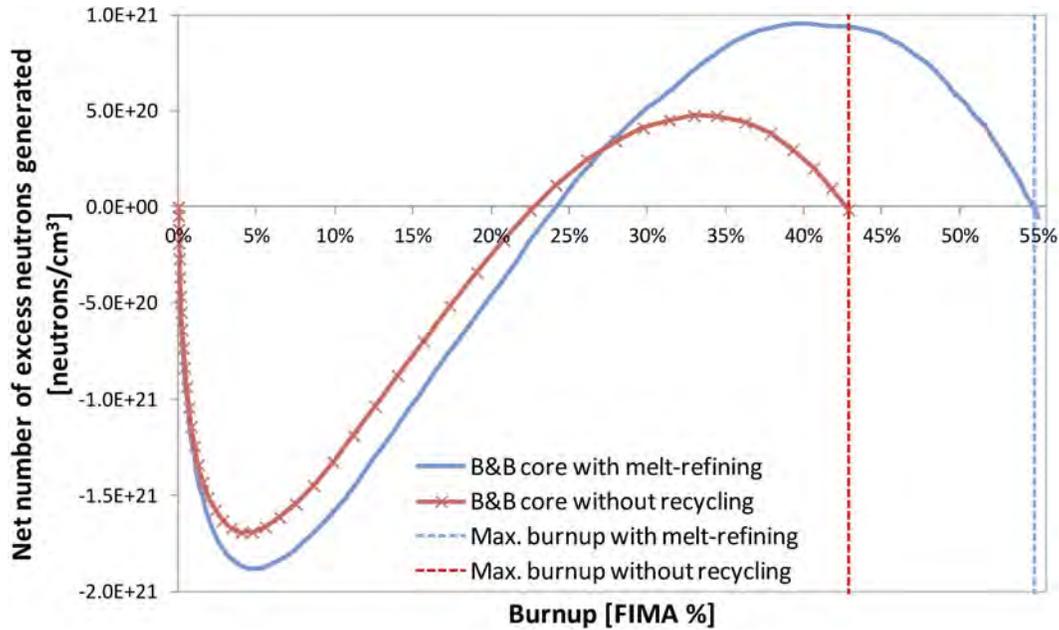


Fig. 7. Neutron balance for the B&B cores with and without fuel recycling.

from the full core model analysis for the following systems:

Scenario 1. *The B&B core at equilibrium operating at 55% FIMA discharge BU:* The average neutron leakage probability is assumed to be 5.95%, and the average fraction of neutrons absorbed in the reactivity control systems is 2.1%. All the fuel is recycled at every EOEC with the melt refining. These are the values reported in Sec. III.A.

Scenario 2. *The B&B core at equilibrium operating at the minimum discharge BU of 19.4% FIMA:* The average neutron leakage probability is 4.4%, and the fraction of neutrons absorbed in the reactivity control systems is 2.2%. All the fuel is recycled at every EOEC with the melt refining.

Scenario 3. *The 0-D core initially made of depleted uranium, assuming the same neutron leakage probability and fraction of neutrons absorbed in the reactivity control systems as for Scenario 1:* All the fuel is recycled with the melt refining at the same BU at which it is recycled in scenario 1.

Scenario 4. *The 0-D core initially made of depleted uranium, assuming the same neutron leakage probability and fraction of neutrons absorbed in the reactivity control systems as for Scenario 2:* The k_{∞} evolution used is the same as obtained for Scenario 3. Therefore, all the fuel is recycled with the melt refining at the same BU at which it is recycled in scenarios 1 and 3.

The evolution of k_{∞} with BU for the four scenarios is given in Fig. 8. The evolution of k_{∞} is identical for the

two unit cell scenarios 3 and 4, and only one curve is shown for both of them: The differences between these two scenarios are the different neutron leakage probabilities and fractions of neutrons absorbed in the reactivity control systems assumed when performing the neutron balance. After a few percent BU, the k_{∞} for scenario 2 is larger than for scenario 1, which is itself larger than for scenarios 3 and 4. For scenario 2, seven fuel recyclings occur over a BU of 19.4% FIMA, while for scenario 1 the seven fuel recyclings occur over a BU of 55% FIMA. As a result of the higher discharge BU, the scenario 1 core has a higher TRU and higher fission products inventory, making its spectrum somewhat harder than for scenario 2. Scenario 1 power density also peaks closer to the core radial boundary making its radial neutron leakage probability higher. For scenarios 3 and 4, seven fuel recyclings occurred by the time the fuel reached 55% FIMA, but only three recyclings were made by the time the fuel of scenario 4 reached 19.4% FIMA.

Despite those differences, the three k_{∞} evolutions show good agreement for BU above $\sim 1\%$ FIMA. Below $\sim 1\%$ FIMA BU, the unit cell displays a significantly higher k_{∞} than the two B&B core scenarios. This behavior is due to the harder spectrum (Fig. 2) in the unit cell at very low BU. With a harder spectrum, the fission cross section of ^{238}U is larger, yielding a larger k_{∞} for the unit cell model.

The BU-dependent neutron balance evolution is calculated [the integral of Eq. (5)] using the k_{∞} evolution and the estimated neutron losses. The balance obtained for the high BU, high neutron loss scenarios 1 and 3 and the balance obtained for the low BU, low neutron loss scenarios 2 and 4 are presented in Figs. 9 and 10, respectively.

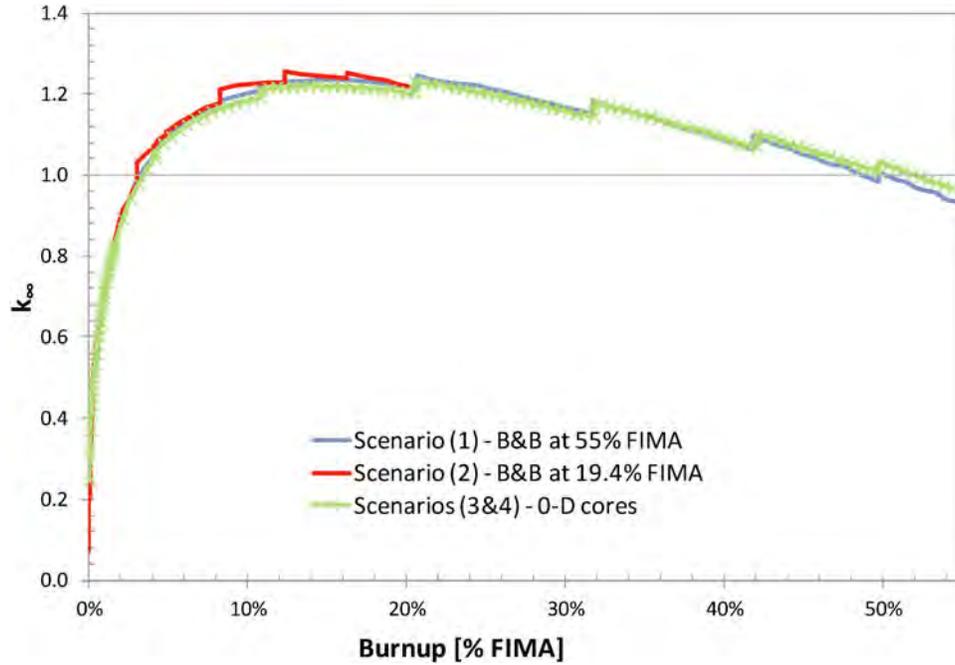


Fig. 8. The k_{∞} evolution with BU for the four scenarios studied.

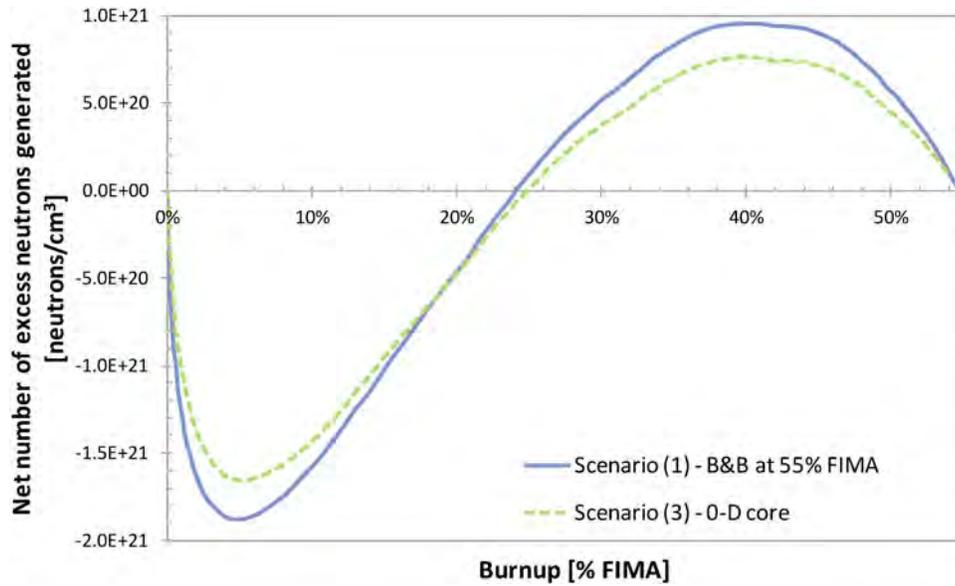


Fig. 9. Neutron balance for scenarios (1) and (3): high BU and high neutron losses.

The analysis of scenario 3 predicts that approximately 1.7×10^{21} neutrons should be provided from the outside per cubic centimeter of fuel to convert depleted uranium fuel into driver fuel that will pay back these neutrons as it reaches $\sim 25\%$ FIMA. This is 12% smaller than the amount of neutrons required for scenario 1, discussed in Sec. III.A. The scenario 3 analysis also pre-

dicts an additional neutron excess of approximately 7.6×10^{20} n/cm³ of fuel, 20% smaller than for scenario 1, sufficient to extend the fuel BU up to $\sim 54\%$ FIMA. The neutron balances obtained with the 0-D core model and the full core model feature a similar attainable BU, but the number of required neutrons and number of excess neutrons, per unit of fuel volume, are smaller for the

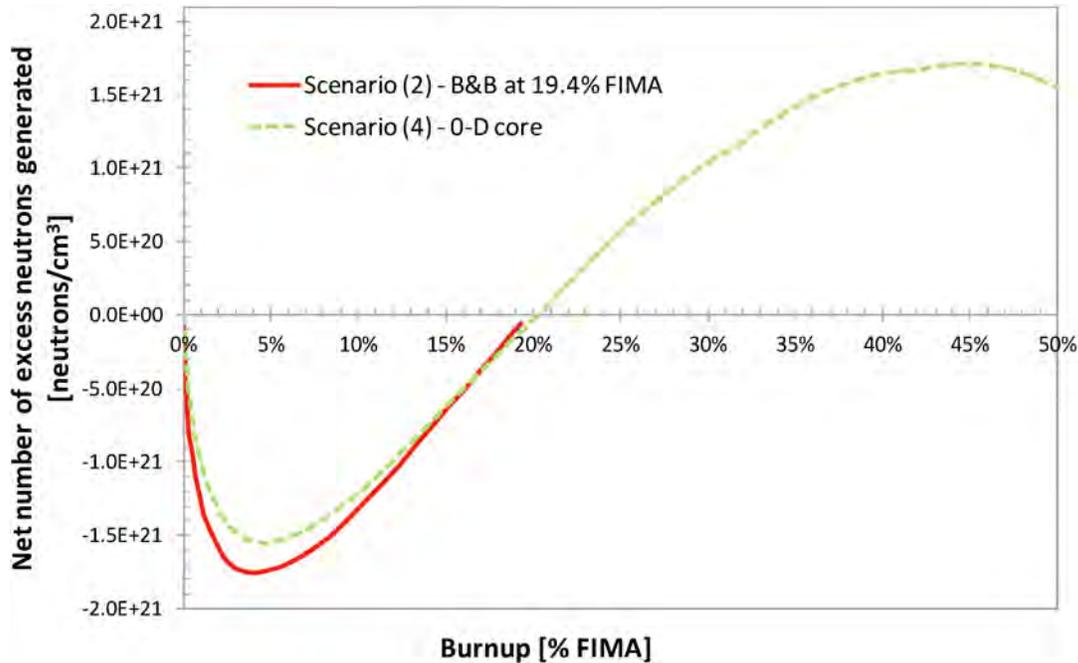


Fig. 10. Neutron balance for scenarios (2) and (4): low BU and low neutron losses.

0-D model than for the full core model of scenario 1. These compensating effects are due to the spectral differences that vary with BU (Figs. 2 and 3) and affect both the k_{∞} values and the $\bar{\nu}$ values used in Eq. (5). The difference of excess neutrons generated observed between the 0-D models and the full core models increases from 0% to $\sim 1\%$ FIMA, is approximately constant up to 4% to 5% FIMA and then decreases and eventually reverts at higher BUs.

For scenario 2, approximately 1.75×10^{21} neutrons should be provided from the outside per cubic centimeter of fuel to convert depleted uranium fuel into driver fuel. For scenario 4, this number is 1.55×10^{21} . As the neutron leakage probability is relatively low for this scenario, it takes 3.5% FIMA to supply this number of external neutrons. By the time the fuel reaches $\sim 20\%$ FIMA (190 GWd/tonne HM), it paid back as many excess neutrons as it consumed from 0% to 3.5% FIMA. Hence, the minimum BU required for sustaining the B&B mode of operation is $\sim 20\%$ FIMA. In Fig. 10, the neutron balance for scenario 4 shows a minimum BU of 21% FIMA and features an $\sim 10\%$ smaller number of neutrons to be supplied per unit of fuel volume than for the full core model (scenario 2). Based on the neutron balance shown in Fig. 10 for scenario 4, from 3.5% to 45% FIMA the fuel generates a total of approximately 3.6×10^{21} excess neutrons per cubic centimeter of fuel; after discharging the fuel at $\sim 20\%$ FIMA, approximately 1.7×10^{21} excess neutrons per cubic centimeter of fuel can still be generated before $k_{\infty} = 1/P_{RC} \times P_{NL} (= 1/0.935)$, which occurs at 45% FIMA. This is sufficient to start a new B&B core using the fuel discharged at $\sim 20\%$ FIMA

from the original core. For scenario 4, the number of neutrons that need to be provided initially and the number of excess neutrons accumulated later are, respectively, 11% and 9% smaller than for scenario 2.

The doubling time of the B&B reactor is defined as the time it takes to accumulate 50% of the core volume worth of discharged fuel—the amount assumed (conservatively) necessary to make a “starter” for a new core. The net doubling time is the doubling time for which the capacity factor and time required for fuel shuffling and recycling are not accounted for. The net doubling time of the B&B core discharging fuel at 20% FIMA would be $(190 \text{ GWd/tonne HM} \times 142 \text{ tonne HM} \times 50\% / 3 \text{ GW(thermal)}) = 12.3 \text{ yr}$.

It is concluded that performing the neutron balance analysis using the 0-D core model yields a minimum required BU value and a maximum achievable BU value that are similar to those obtained by the neutron balance performed for the full core model. However, because of the spectral differences between the 0-D and full core models, it was found that the 0-D model underestimates the number of neutrons that need to be absorbed in the fuel and the number of excess neutrons that can be produced by the fuel by $\sim 10\%$.

IV. NEUTRON BALANCE FOR THE TRANSITION PERIOD: TRU- AND EU-BASED STARTERS

A neutron balance analysis similar to that performed for the B&B core at equilibrium is performed for the B&B core, when started with either a TRU starter or an

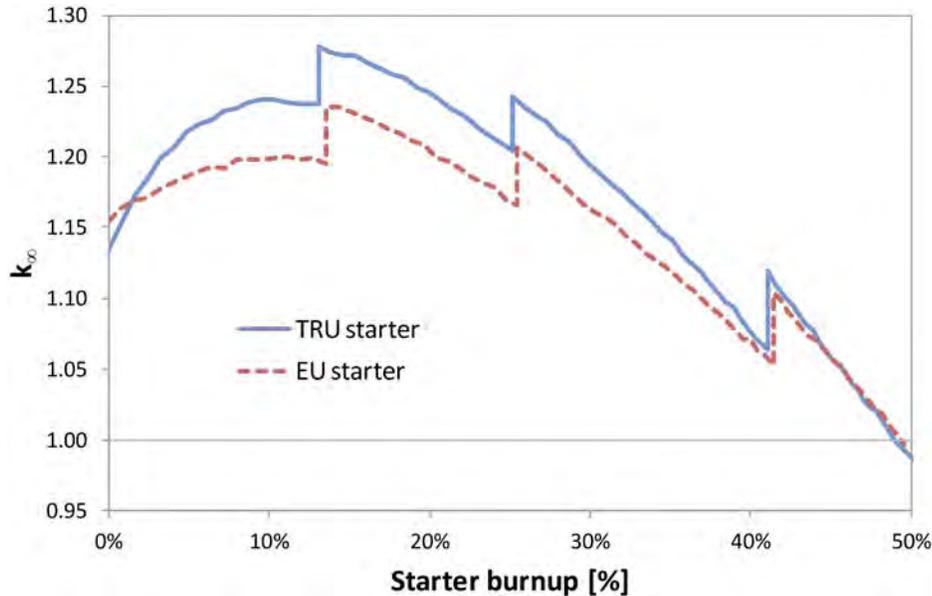


Fig. 11. The k_{∞} evolution for the TRU and EU starters of the B&B core.

EU starter. The core geometry and starters compositions are described in Sec. II.A.1. During the transition period, there is no fuel shuffling, and therefore, the neutron balance is performed for the innermost half-core volume corresponding to the whole starter volume. The depleted uranium assemblies are expected to have a similar behavior to that of the depleted uranium assemblies studied in Sec. III. The k_{∞} values of the starter are obtained using Eq. (6), summing the reaction rates over the three axial and four radial fuel zones composing the starter. It is assumed that 3% of the neutrons are absorbed by the reactivity control systems, corresponding to a BU reactivity swing of $\sim 6\%$. The average neutron leakage probability is deduced from a previous study¹⁰; its value is 3.85% for the TRU starter and 4.05% for the EU starter.

The k_{∞} evolution thus obtained for the two starters is shown in Fig. 11. Although the enrichments of the TRU and EU starters is chosen such as to get the reactor k_{eff} equal to unity at BOL, the neutron leakage probability for the EU starter is larger than for the TRU starter. At BOL the EU starter initially has a larger amount of fissile material, and its k_{∞} is larger than for the TRU starter. However, as $\eta(^{235}\text{U})$ is smaller than $\eta(^{239}\text{Pu})$, more neutrons are available for breeding in the TRU starter. Hence, the initial conversion ratio of the TRU starter is larger than for the EU starter, making the TRU starter k_{∞} increase more rapidly with BU. As the ^{235}U of the EU starter is being consumed and replaced by the bred plutonium isotopes, the EU starter k_{∞} converges toward the TRU starter k_{∞} .

The neutron balance, presented in Fig. 12, shows that the TRU starter generates 4.27×10^{21} excess neutrons per cubic centimeter of fuel while the EU starter generates 3.17×10^{21} excess neutrons per cubic centi-

meter of fuel. It was previously found (scenario 2, Sec. III.D) that 1.75×10^{21} neutrons need be absorbed per cubic centimeter of depleted uranium feed fuel in order to convert it into driver fuel. The neutron balance of Fig. 12 indicates that it takes a BU of only 14% FIMA for the TRU starter and of 19.5% FIMA for the EU starter to provide this number of excess neutrons. Using a single TRU starter, it is possible to start two B&B cores, while when using a single EU starter, it is possible to start only a single B&B core. Theoretically, the number of excess neutrons that can be generated by a TRU starter is sufficient to establish the B&B mode of operation in 2.4 ($= 4.27/1.75$) B&B cores. The corresponding value for an EU starter is 1.8 ($= 3.17/1.75$).

Additional studies¹⁰ showed that it is possible to design an improved fuel shuffling scheme for the transition period that yields a BU reactivity swing no larger than 3% versus the 6% assumed for the above-reported analyses. Using this improved shuffling scheme in which only 1.5% rather than 3% of the neutrons are lost, on the average, for excess reactivity control, a single EU starter is theoretically able to establish the B&B mode of operation in two cores, while a single TRU starter would be able to establish it in 2.5 cores. The corresponding net doubling time of the B&B core is 8.4 yr with the TRU starter and 10.6 yr with the EU starter.

V. BREED AND BURN FEASIBILITY FOR ALTERNATIVE CORE DESIGNS

The neutron economy of the B&B mode of operation worked out in Secs. III and IV pertains to cores

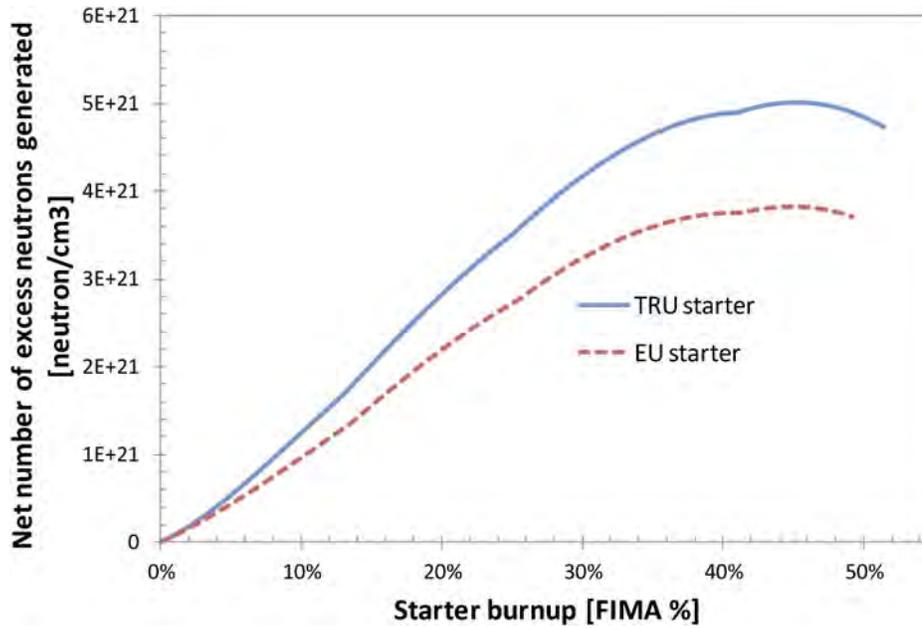


Fig. 12. Excess number of neutrons in the B&B for the TRU and EU starters with 3% of neutrons absorbed in the reactivity control systems.

cooled with sodium and that use uranium-based metallic fuel. The feasibility of establishing the B&B mode of operation for alternative core design concepts is assessed in this section for the following design variants:

1. uranium-based metallic fuel with lead-based coolants and HT9 cladding
2. thorium-based metallic fuel with helium coolant and SiC cladding
3. uranium-based nitride fuel with sodium coolant and HT9 cladding.

The analysis is performed using the 0-D core model described in Sec. II.A.2. It was found that the neutron balance performed with this core model yields results consistent with the full core model in terms of maximum achievable and minimum required BU, although with a low accuracy regarding the total number of neutrons available from and required by the fuel. Embarking upon a thorough space-dependent study of all these designs would require a significant effort, while using the neutron balance method with a 0-D core model provides an acceptable preliminary feasibility assessment within a limited amount of time.

V.A. Lead-Bismuth-Cooled B&B Core

In Ref. 11 it has been found that when using 271 fuel rods per assembly with $P/D = 1.3$, it is possible to operate a lead-bismuth eutectic (LBE)-cooled B&B core at 3000 MW(thermal) with an average neutron leakage probability of 4.66%, approximately equal to that of the

sodium-cooled B&B core— $P/D = 1.122$ —operating at 19.4% FIMA. The larger P/D required for the LBE-cooled core is due, primarily, to the lower LBE velocity versus sodium, which is due to the higher erosion rate of the cladding.

The minimum BU required to sustain the B&B mode is determined for the LBE-cooled core by performing the depletion analysis and neutron balance in a 0-D unit cell and using the above-mentioned neutron leakage probability, 4.66%. The BU reactivity swing is assumed to be 4.4% so that an average of 2.2% of the neutrons is lost in the reactivity control systems. The average core (including depleted uranium blanket) power density used for the depletion analysis is 112.5 W/cm^3 , and the volume fractions used for the core constituents are given in Table III. Interassembly coolant gaps and duct walls are not accounted for. The fuel characteristics are the same as those presented for the B&B core in Sec. II.A.1. During operation, 75% of the fission gases are removed from

TABLE III
0-D Homogenized Unit Cell Composition

P/D	1.30
Fuel (U-Zr10)	28.0%
Gap (empty)	9.3%
Structural material (HT9)	16.4%
Coolant (LBE)	46.3%

the fuel, and approximately every 10 yr the fuel is instantaneously recycled with the melt-refining process neglecting recycling loss. The evolution of k_∞ with BU is shown in Fig. 13, and the corresponding neutron balance is shown in Fig. 14 in comparison with that of the reference sodium-cooled core.

The minimum BU required to sustain the B&B mode of operation in the LBE-cooled core is 29% FIMA. At this BU, there are 299.0 kg of ^{239}Pu per cubic meter in the LBE-cooled core and the ^{239}Pu -to-HM weight fraction is 10.8%.

Compared to the sodium-cooled core with a P/D ratio of 1.122, the minimum required BU for the LBE

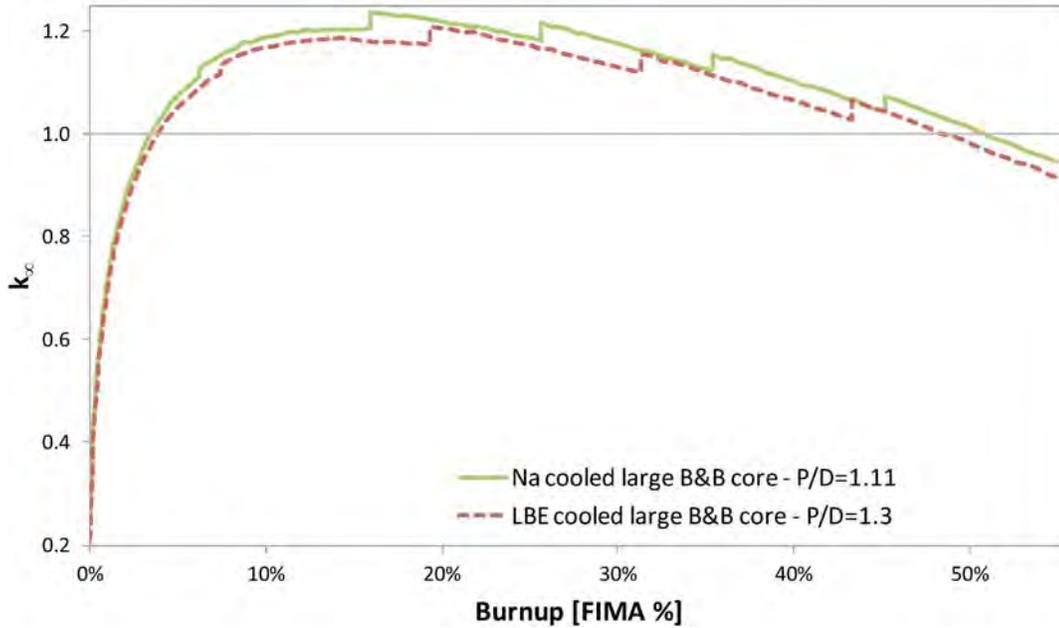


Fig. 13. The k_∞ evolution for the LBE-cooled core with P/D = 1.30 and the sodium-cooled core with P/D = 1.11.

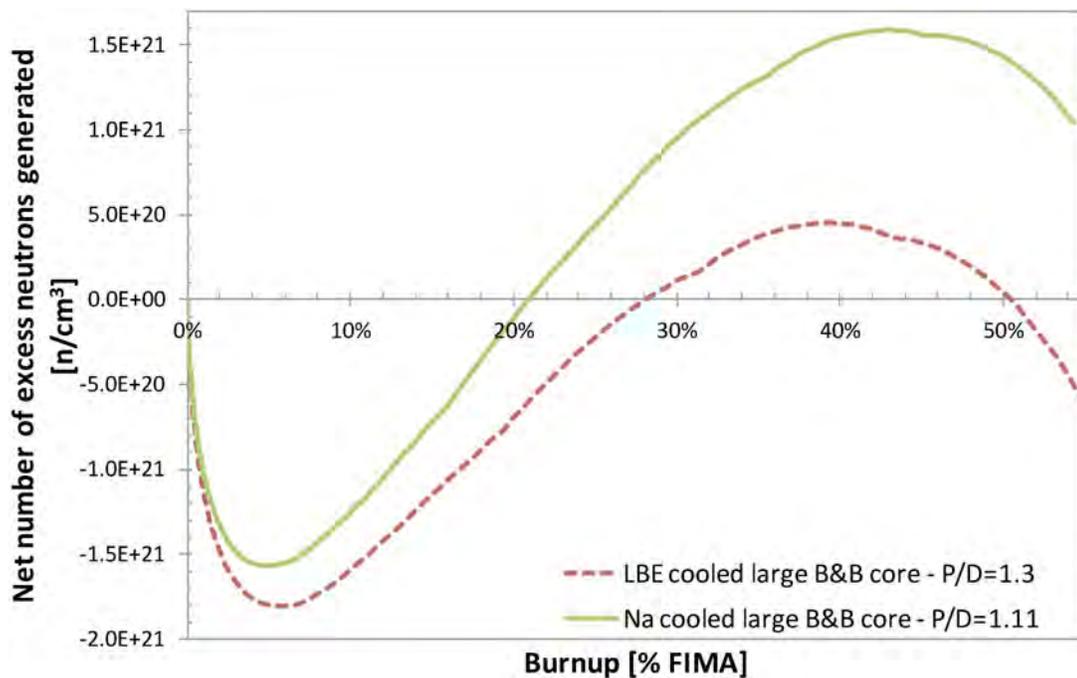


Fig. 14. Neutron balance for the LBE-cooled core with P/D = 1.30 and the sodium-cooled core with P/D = 1.11.

coolant is increased by +9% FIMA, and the maximum achievable BU is significantly impaired because of spectral effects and the higher parasitic neutron absorption in the coolant that is due to the significantly larger LBE-coolant volume fraction.

Equation (5) can also be used to search for the maximum $P_{NL}P_{CR}$ value for which there is a BU(thermal) satisfying the equation. This value corresponds to the maximum neutron loss by leakage and absorption in the reactivity control system for which the B&B mode of operation is attainable. For the LBE-cooled core, the maximum tolerable neutron loss is 8%, and the corresponding discharge BU is 38% FIMA.

V.B. Thorium-Based Metallic Fuel

The feasibility of establishing a B&B mode of operation in a thorium-fueled fast reactor core is assessed using the 0-D neutron balance methodology. Metallic thorium fuel is made of 100% ^{232}Th , has a density of 11.65 g/cm^3 , and does not need to be alloyed with Zr because its crystal structure is stabler than that of pure metallic uranium (U-Zr). This corresponds to an initial heavy metal atomic density of 3.0×10^{22} thorium atoms per cubic centimeter of fuel. The volume fractions and smear density used for the metallic thorium-based 0-D core are those used for the uranium-based metallic fuel given in Table I. The initial heavy metal density is de-

creased from 5.35 g/cm^3 of core for the uranium-based unit cell to 4.37 g/cm^3 for the thorium-based unit cell, both using the same P/D ratio of 1.122. Also, the simulation assumptions are the same: continuous removal of 75% of the fission gases and instantaneous fuel recycling every ~ 10 yr using the melt refining. Although the melt-refining process is not compatible with thorium, if this study shows that thorium features attractive performance, it will be justified to embark upon developing a recycling process for thorium fuel without actinides separation. The neutron leakage probability and the fraction of neutrons lost in the reactivity control systems are assumed to be 2% and 1%, respectively. It was found that despite using these optimistically low neutron loss probabilities, no B&B mode of operation could be established when using HT9 cladding and sodium coolant.

To get an upper possible bound on the neutron economy of a thorium-based core, the sodium coolant and HT9 structural material are replaced with helium and silicon carbide, respectively; several gas-cooled fast reactor core designs using SiC cladding with helium coolant have recently been proposed.^{12,13} A helium pressure of 70 bars and average temperature of 700 K are assumed; the corresponding helium density is calculated to be 4.88 kg/m^3 .

The k_∞ evolution for this thorium-based core is compared in Fig. 15 with that of the reference sodium-cooled uranium-based core. The maximum k_∞ value reached is

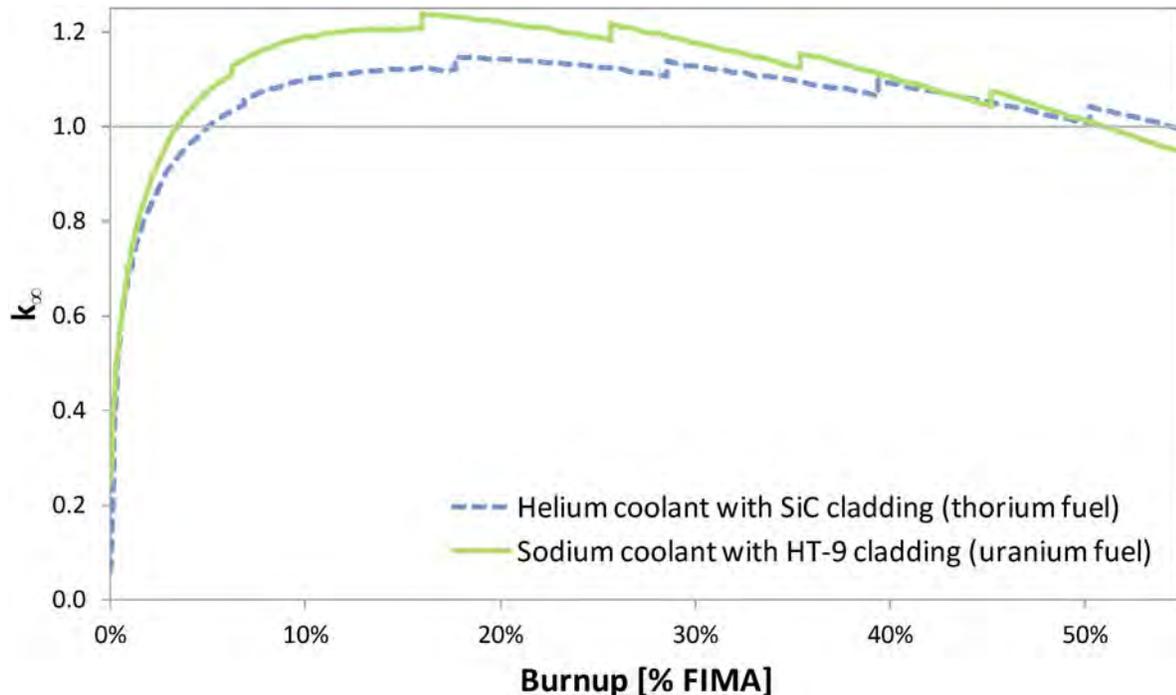


Fig. 15. The k_∞ evolution for the Th/SiC/He and UZr/HT9/Na 0-D core models.

1.1 for the Th/SiC/He core, while it is 1.2 for the UZr/HT9/Na core. This is because the spectrum-averaged value of the reproduction factor (η) of ^{233}U is $\sim 15\%$ smaller than that of ^{239}Pu . Figure 16 compares the spectra of the two systems; the two main spectrum depressions in the Th/SiC/He spectrum occurring around 60 and 200 keV are due to the ^{28}Si scattering cross-section resonances.

The initial k_{∞} value for the thorium system is smaller than for the uranium system—due to the smaller fast

fission cross section of thorium. Above $\sim 10\%$ FIMA k_{∞} evolution with BU decreases slower in the thorium system due to the smaller fission product poisoning effect and the smaller number of neutrons captured in the coolant and cladding.

Figure 17 compares the neutron balance evolution in the two cores. It is theoretically possible to sustain the B&B mode of operation with pure ^{232}Th feed; the minimum required BU is $\sim 39\%$ FIMA. However, the 3%

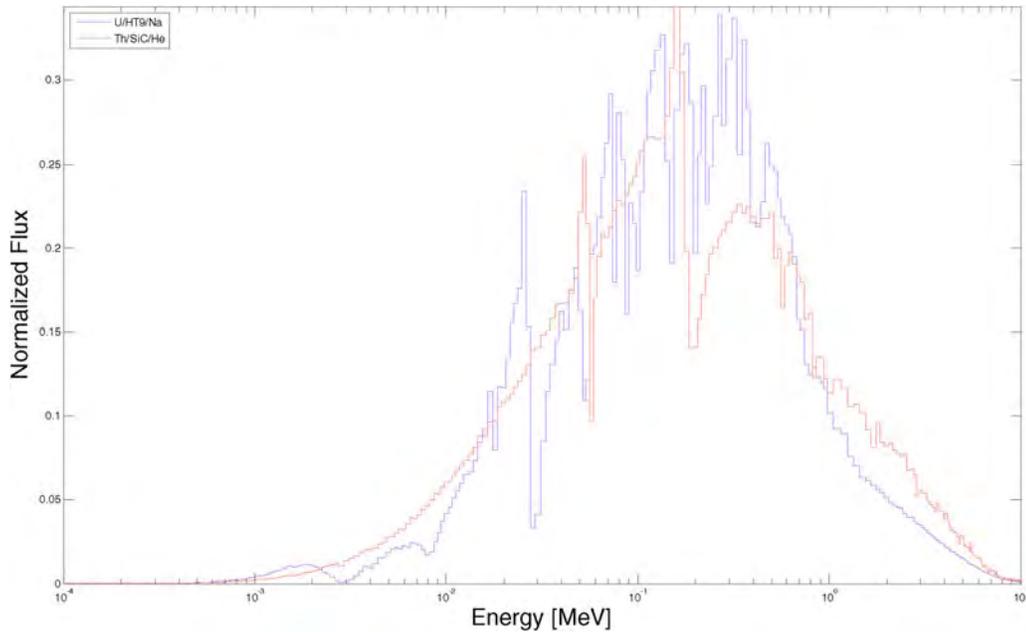


Fig. 16. Spectra comparison in the UZr/HT9/Na 0-D core (blue) and in the Th/SiC/He 0-D core (red) (color online).

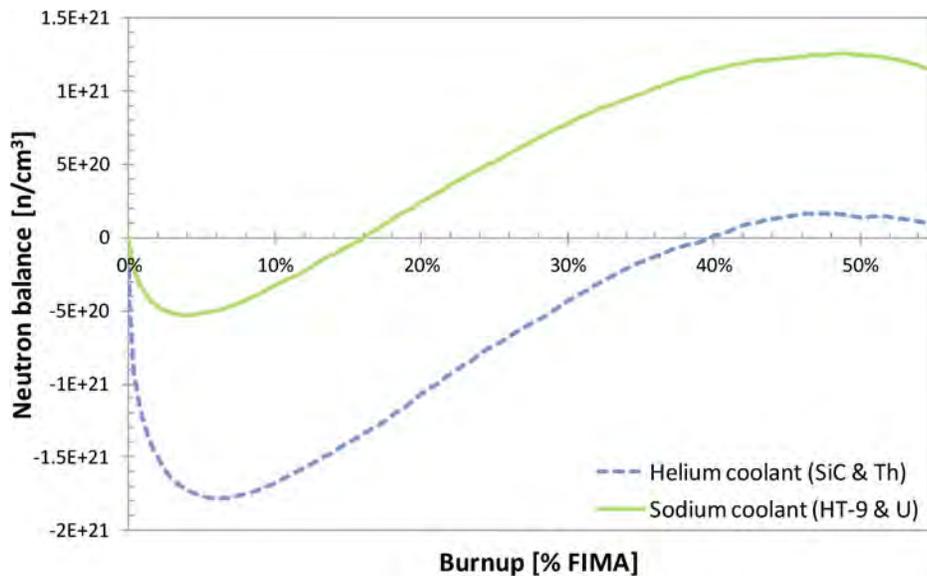


Fig. 17. Neutron balance for the helium-cooled and sodium-cooled cores with SiC cladding and thorium fuel. A small neutron loss of 3% is assumed.

neutron loss probability assumed for this analysis is not very realistic; the minimum neutron leakage probability possible to achieve¹⁰ in the sodium-cooled, uranium-fueled B&B core was 4.4%, and helium is a worse neutron reflector than sodium. The maximum neutron loss by leakage and absorption in the reactivity control systems for which the B&B mode can be established in the Th/SiC/He core is estimated to be 3.5%; it corresponds to a minimum required discharge BU of 46% FIMA.

It is concluded that it is not practically feasible to establish a B&B mode of operation when feeding the core with pure thorium.

V.C. Nitride Fuel

According to Refs. 14, 15, and 16, it is possible to fabricate uranium nitride (UN) pellets having up to 90% of the theoretical density, 14.32 g/cm³. This corresponds to an initial heavy metal atomic density of 3.1×10^{22} atoms/cm³ of fuel. In addition, UN swells less with BU than metallic fuel so that the smear density taken is 80% (Ref. 15), somewhat larger than the 75% assumed for metallic fuel. Since the neutron capture cross section of ¹⁴N is larger than that of ¹⁵N and neutron capture in ¹⁴N results in generation of the environmentally hazardous ¹⁴C, the nitrogen is assumed enriched to 99 at. % ¹⁵N, as is often done.^{17,18} The cladding and coolant volume fractions are the same as for U-Zr fuel, but the fuel and gap volume fractions are changed because of the different smear density. The volume fractions used are provided in Table IV.

The k_{∞} evolution for the UN-fueled unit cell is compared in Fig. 18 with that of the U-Zr reference. The UN-fueled core k_{∞} values are ~4% smaller because of the softer spectrum due to the neutron slowing down by nitrogen nuclei. The spectra of the two cores at a BU of 20% FIMA are compared in Fig. 19. Because of the softer spectrum, the UN-fueled core fissile-to-HM atom ratio is 10.3% larger than in the U-Zr-fueled core. It is also found that the parasitic neutron absorption is lower by ~0.7% when using UN instead of U-Zr fuel because of the significantly smaller fast spectrum average macroscopic capture cross section of ¹⁵N compared to zirconium.

TABLE IV
Volume Fractions of Nitride Fuel
for the Unit Cell Study

	U-Zr (%)	UN (%)
Fuel	37.5	40.0
Gap (empty)	12.5	10.0
Structural material (HT9)	22.0	22.0
Coolant (Na)	28.0	28.0

The neutron balance for the UN-fueled 0-D core is performed assuming the same neutron leakage probability, 4.4%, and the same fraction of neutrons absorbed in the reactivity control systems, 1%, as obtained for the U-Zr-fueled B&B core in Ref. 5. The results are compared in Fig. 20. The minimum BU required to establish the B&B mode of operation for the UN-fueled core is found to be ~33% FIMA versus ~19% FIMA for the reference U-Zr-fueled core. It is possible to sustain the B&B mode in a B&B core fueled with UN with a discharge burnup of ~40% FIMA, as long as the fraction of neutrons lost by leakage or absorbed in the reactivity control systems is smaller than 5.7%. It is however impractical to spawn additional cores from a UN-fueled B&B core because of the very small amount of excess neutrons that can be spared beyond sustainment of the B&B mode in the first core.

VI. CONCLUSIONS

During the transition period of the B&B core, the number of excess neutrons that can be generated by a TRU-based starter fuel used for initiating the chain reaction is sufficient for establishing the B&B mode in at least 1.4 additional B&B cores; the net doubling time is 8.4 yr. For EU starter, the corresponding values are at least an additional 0.8 B&B cores and 10.6 yr doubling time.

The maximum BU attainable, at equilibrium, in the B&B core when recycling the fuel using an ideal melt-refining process and an "out-in" fuel shuffling scheme is 55% FIMA. This is 28% higher as compared to the BU attainable in the same core without fuel recycling. The minimum BU required to sustain the B&B mode of operation in a B&B core fueled with metallic fuel having 10 wt% Zr and 75% smear density, clad with HT9, and cooled with sodium is 19.4% FIMA. The fraction of neutrons lost in this core by leakage or absorption in the reactivity control systems is 6.6%. Approximately 1.75×10^{21} n/cm³ of fuel should initially be provided to convert depleted uranium feed fuel into driver fuel. From 0% to 3.5% FIMA, the feed fuel is a net neutron consumer, and from 3.5% up to 45% FIMA it provides approximately 3.6×10^{21} excess neutrons per cubic centimeter of fuel. This amount of excess neutrons is sufficient to establish the B&B mode in the original core and to start another B&B reactor without need for additional enriched fuel. For the equilibrium B&B reactor fed with depleted uranium and operated at the minimum required BU, the net doubling time is 12.3 yr.

When the coolant is LBE, the minimum BU required for a 3000 MW(thermal) core to sustain the B&B mode of operation at equilibrium is ~29% FIMA. It was also found possible to establish the B&B mode in a large sodium-cooled B&B core fueled with nitride fuel in which

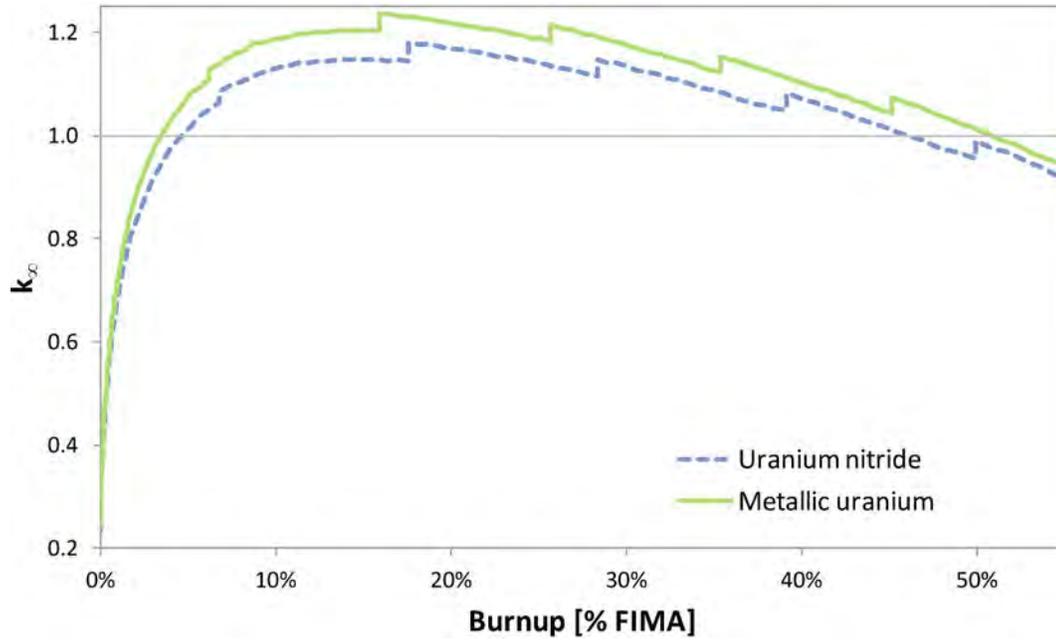


Fig. 18. The k_{∞} evolution for UN and U-Zr-fueled 0-D unit cells cooled with sodium and clad with HT9.

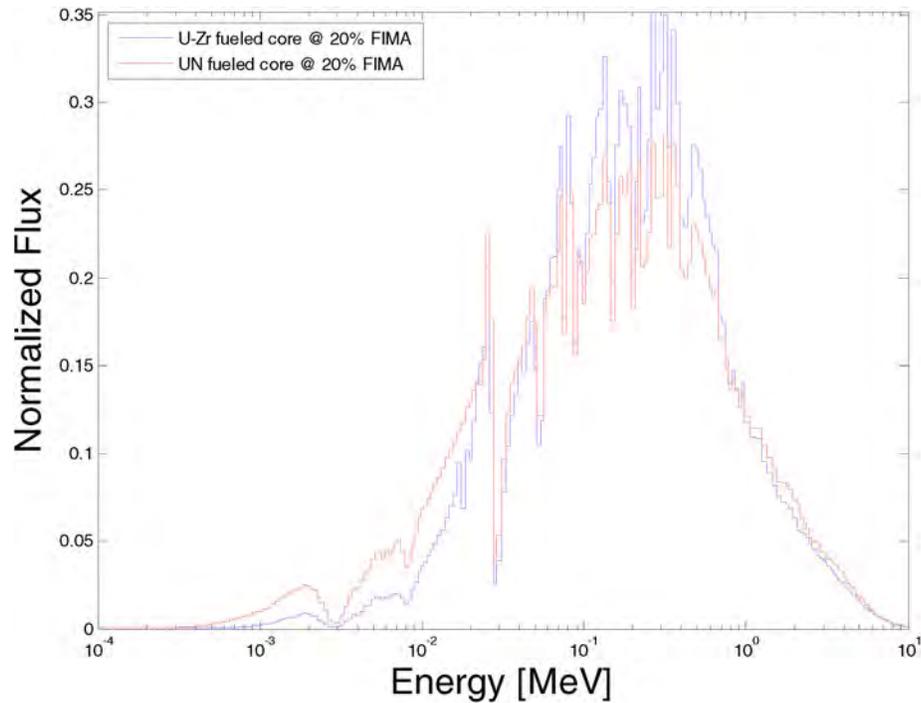


Fig. 19. Spectra comparison between the U-Zr fuel and the UN fuel in the 0-D core model at 20% FIMA.

the nitrogen is enriched to 99% ^{15}N . However, the required discharge BU at equilibrium is in the vicinity of 40% FIMA for a maximum neutron loss of 5.7%. It is not realistically possible to establish the B&B mode when

the core is fueled with thorium, even when using helium coolant and silicon carbide cladding.

The results derived from the neutron balance analysis strongly depend on the value of the estimated neutron

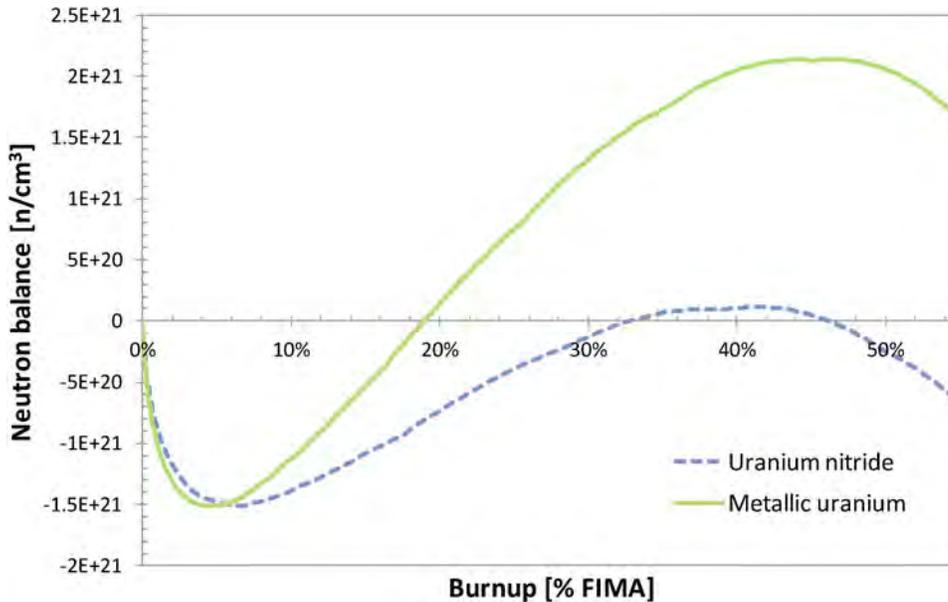


Fig. 20. Neutron balance for the UN-fueled and reference U-Zr-fueled 0-D cores using HT9 cladding and sodium coolant.

leakage probability and the fraction of neutrons lost in the reactivity control systems. A neutron balance performed using a simplified 0-D core model, although not accurate because of, primarily, inaccurate spectra predictions, provides reasonable estimates of the minimum required and maximum attainable BUs despite the fact that its k_{∞} evolution prediction is inaccurate. The 0-D approach can save much computational effort and time and is therefore useful for scoping analysis.

ACKNOWLEDGMENTS

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Article

A Phased Development of Breed-and-Burn Reactors for Enhanced Nuclear Energy Sustainability

Ehud Greenspan

University of California Nuclear Engineering Department, Berkeley, CA 94720, USA;
E-Mail: gehud@nuc.berkeley.edu; Tel.: +1-510-643-9983; Fax: +1-510-643-9685

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Abstract: Several options for designing fast reactors to operate in the Breed-and-Burn (B&B) mode are compared and a strategy is outlined for early introduction of B&B reactors followed by a gradual increase in the fuel utilization of such reactors. In the first phase the fast reactor core will consist of a subcritical B&B blanket driven by a relatively small critical seed. As the required discharge burnup/radiation-damage to both driver and blanket fuel had already been proven, and as the depleted uranium fueled B&B blanket could generate close to 2/3 of the core power and will have very low fuel cycle cost, the deployment of such fast reactors could start in the near future. The second phase consists of deploying self-sustaining stationary wave B&B reactors. It will require development of fuel technology that could withstand peak burnups of ~30% and peak radiation damage to the cladding of ~550 dpa. The third phase requires development of a fuel reconditioning technology that will enable using the fuel up to an average burnup of ~50%—the upper bound permitted by neutron balance considerations when most of the fission products are not separated from the fuel. The increase in the uranium ore utilization relative to that provided by contemporary power reactors is estimated to be 20, 40 and 100 folds for, respectively, phase 1, 2 and 3. The energy value of the depleted uranium stockpiles (“waste”) accumulated in the US is equivalent to, when used in the B&B reactors, up to 20 centuries of the total 2010 USA supply of electricity. Therefore, a successful development of B&B reactors could provide a great measure of energy sustainability and cost stability.

Keywords: fast reactors; breed-and-burn; stationary wave; travelling wave; seed-and-blanket; sustainability; depleted uranium; thorium

1. Introduction

Present day commercial nuclear power reactors, mostly Light-Water-Reactors (LWRs), utilize less than one percent of the natural uranium feed: the uranium enrichment level presently preferred by the industry is approximately 4.5% ^{235}U . As natural uranium contains only 0.72% of ^{235}U , it takes 8 to 10 tons of natural uranium to make 1 ton of 4.5% enriched uranium. The remaining 7 to 9 tons of depleted uranium, typically containing 0.2% to 0.3% ^{235}U , is discarded as a waste. Of the enriched uranium that is loaded into the core, only about 5% is actually fissioned, making the overall uranium utilization only $\sim 1/9$ of 5% or, approximately, 0.6%.

The amount of natural uranium that has been mined so far for fueling the fleet of commercial LWRs that presently generates close to 20% of the U.S. electricity consumption is approximately 700 thousand tons. Out of these, more than 60,000 tons ended up as used nuclear fuel (UNF)—the enriched uranium fuel that was fed into the LWRs and discharged after few percent of the uranium has been fissioned. More than 600,000 tons ended up as depleted uranium “waste”. Additional depleted uranium has been accumulated from the military programs.

By using fast breeder reactors it is possible, in principle, to fission close to 100% of the depleted uranium “waste”. However, this high uranium utilization cannot be achieved in a single irradiation campaign because neutron-induced radiation damage effects constrain the burnup level the fuel can withstand to the order of 10% to 15% FIMA (Fissions per Initial heavy Metal Atom), depending on the core neutron spectrum. Consequently, attainment of high uranium utilization also necessitates multiple fuel recycling. Traditionally, fuel recycling includes removal of the fuel cladding, removal of most of the fission products, addition of some depleted uranium make up fuel, fabrication of new fuel elements and reloading them into the reactor core for another irradiation cycle. Although technically feasible, there is a significant objection in the U.S. and other countries towards fuel reprocessing due to economic viability and proliferation concerns.

Fast breeder reactors (FBR) could, in principle, also operate without fuel recycling; that is, using a once-through fuel cycle as do all of the LWRs presently operating in the USA. Although a discharge burnup of 10% to 15% FIMA is 2 to 3 times higher than that of contemporary LWRs, the uranium utilization from a once-through FBR is not significantly different from that of a once-through LWR because the uranium enrichment required to fuel the FBR is more than twice that required to fuel the LWR.

Nevertheless, it may be possible to realize a significant increase in the uranium utilization without fuel reprocessing using a special class of fast reactors, referred to as “breed-and-burn” (B&B) or “travelling wave” reactors, such as the TWR under development by Terra-Power [1–3]. The unique feature of a B&B reactor is that it can breed plutonium in depleted uranium feed fuel and then fission a significant fraction of the bred plutonium, without having to reprocess the fuel. In order to initiate the chain reaction, the B&B core must first be fed with an adequate amount of fissile fuel such as enriched uranium. Plutonium or TRU extracted from UNF could also be used for this “starter”. Thereafter, the B&B core is capable of continued operation while being fed solely with depleted uranium. Eventually, the uranium utilization will approach the fraction of the loaded uranium that has been fissioned.

The principles and concepts of B&B reactors had been proposed in the past; [3–11] is a partial list of references. These references describe either one of two basic variants of B&B reactors—one is the

Travelling-Wave-Reactor (TWR) like the highly published CANDLE reactor concept [11,12] and the TWR concept initially pursued by Terra-Power [1,8]. The other is the Stationary-Wave-Reactor (SWR) like the concepts proposed in [2,3,5,7,13] that is also presently pursued by Terra-Power [2,3]. However, in order to sustain the chain reaction in the B&B mode of operation it is necessary to fission, on the average, approximately 20% of the depleted uranium fed (see section 2). This corresponds to a peak discharge burnup of close to 30% FIMA. This peak burnup corresponds to peak radiation damage to the fuel rod cladding material of about 550 displacements per atom (dpa). The experimental and demonstration fast reactors that operated in the past have proven that the HT-9 fuel clad can maintain its mechanical integrity up to 200 dpa, corresponding to a burnup of ~10% FIMA in a hard-spectrum core such as required for a B&B reactor. It is likely that the fuel could have withstood higher burnup without losing its mechanical integrity but there is no experimental evidence that this, indeed, is the case. Moreover, a combination of the development of improved structural materials, improved fuel materials, and improved core design is likely to increase the acceptable burnup.

The minimum of 20% average burnup pertains to large volume SWR cores. The situation is aggravated in TWR cores because a smaller fraction of the excess neutrons can be used for building up the fissile content in the depleted uranium feed—as will be elaborated upon in Section 3.

Alternatively, it might be possible to establish the B&B mode of operation with limited fuel “reconditioning” [13–19]. The functions of the fuel re-conditioning are to remove a portion of the fission products, primarily the gaseous ones, and replace the fuel clad prior to fuel re-use in the reactor. This procedure overcomes material performance limits in a way that cannot be used to extract plutonium and that is, hopefully, not as expensive as conventional fuel reprocessing. The re-fabricated fuel can either be re-introduced into the reactor core for additional burnup, or be used as the “starter” fuel for a new core. The latter option, to be referred to as the “spawning” mode of operation, offers a significant savings in the amount of enriched uranium and, therefore, natural uranium that is required to deploy a fleet of B&B reactors.

However, significant R&D is required before an acceptable fuel reconditioning process will be developed, and it is not certain today whether or not such a process will be acceptable. Likewise, the accumulation of experimental evidence that HT-9 or another structural material can maintain its integrity up to 550 dpa is a long campaign that will take significant time to complete and, although likely to succeed, success is not certain.

The objectives of the present paper are to describe and compare several options for designing fast reactors to operate in the B&B mode and to suggest a strategy for phased commercialization of B&B reactors that could provide a significant measure of energy sustainability significantly sooner than otherwise possible.

Section 2 gives an estimation of the minimum burnup required for establishing the B&B mode of operation as well as the maximum burnup that is attainable in such B&B reactors when fuel reconditioning can be used for recycling the fuel in the B&B reactor as long as the fuel has sufficient reactivity to maintain criticality. The feasibility of spawning new B&B reactors using fuel discharged from previous generation B&B reactors is also discussed in this section. Section 3 explains the difference between a TWR and a SWR in terms of the minimum burnup required for sustaining the B&B mode of operation. Section 4 introduces the concept of a subcritical B&B blanket driven by a critical seed and suggests an approach for phased development of the technology required for B&B

reactors while Section 5 gives a brief summary of the impact B&B reactors and fuel reconditioning could have on energy security and economic stability.

2. Minimum Required and Maximum Attainable Burnup

2.1. Neutron Balance Analysis

The minimum burnup required for sustaining the B&B mode of operation, as well as the maximum burnup that can be achieved if fuel reconditioning could be implemented, were established in previous studies [15,17–21]. It is insightful to estimate these values using a simple neutron balance analysis that counts the number of neutrons that are absorbed and that are generated by fissions in a unit volume of fuel as a function of burnup in the core, starting from the fresh feed fuel. The minimum required burnup is the lowest burnup (BU_m), other than zero, for which Equation (1) is satisfied where the maximum attainable burnup is the largest BU_m for which Equation (1) is satisfied [15,17–21].

$$N_{HM} \int_0^{BU_m} \left[P_{NL} * P_{NRC} - \frac{1}{k_{\infty}(BU)} \right] \bar{\nu}(BU) * dBU = 0 \quad (1)$$

In the above, N_{HM} is the Heavy Metal (HM) atom density, BU is expressed in FIMA, $\bar{\nu}(BU)$ is the average number of neutrons emitted per fission, $P_{NL} (= 1 - P_L)$ is the non-leakage probability and $P_{NRC} (= 1 - P_{RC})$ is the probability that a fission-born neutron will escape capture in the control elements used to compensate for the burnup reactivity swing over the equilibrium cycle. In the above we ignored the contribution of (n,2n) and (n,3n) reactions. The values of P_L and P_{RC} are deduced from 3-D analysis of a representative core; the other parameters that go into Equation (1) can be deduced from a batch-by-batch neutron balance analysis in the specific core being analyzed; they can also be well approximated from a much simpler unit cell analysis [15,17].

A quantitative analysis performed at the University of California, Berkeley (UCB) for a large sodium-cooled fast reactor B&B core [17] is briefly summarized. This core uses a ternary metallic fuel U-Pu-Zr with 10 wt% zirconium, a fuel density of 15.85 g/cm³ and a smear factor of 75%—to accommodate the fuel swelling with burnup. The assumed volume fraction of fuel, initial gap, HT-9 clad and Na coolant is, respectively, 37.5%, 12.5%, 22% and 28%. These correspond to a hexagonal lattice pitch-to-diameter ratio of 1.122—near the lower limit used in liquid sodium cooled reactors. The active core height is 209 cm and its diameter is 402 cm. The core is divided into 8 radial batches. At the end of an equilibrium cycle the highest burnup batch is discharged, the other batches are shuffled in a predetermined optimal pattern and a fresh depleted uranium fuel batch is loaded at the outermost core zone.

When a fuel batch reaches its radiation damage limit, it undergoes a melt-refining process like that developed for metallic fuel in the Experimental Breeder Reactor II project [13,25]. The melt-refining involves loading the declad fuel into a zirconia crucible and melting it at ~1300 °C for several hours under argon atmosphere. The gaseous and volatile fission products are released and certain solid fission products are partially removed by oxidation with the zirconia of the crucible. Based on [25] it is assumed that this process can remove 100% of Br, Kr, Rb, Cd, I, Xe and Cs, and 95% of Sr, Y, Te, Ba and the rare earths (lanthanides). Thorium and americium are also oxidized with zirconia, and 95% of

these two elements are assumed removed from the fuel. In fact, in the melt-refining” process experimented with in the EBR-II program, several percent of the plutonium and other actinides remained in the crud of the zirconia crucible. However, experts think that it is likely possible to develop a modified process that does not involve significant loss of actinides and, yet, can efficiently remove the gaseous and certain fraction of the volatile fission products. Although the results of this study are somewhat affected by the fraction and type of actinides and solid fission products that are removed in the fuel recycling process, the overall conclusions of this work are not expected to vary by assuming an ideal process with no loss of actinides other than americium and thorium.

The minimum required average burnup deduced from a detailed search for the equilibrium cycle was found [17,18] to be 19.4% FIMA. This core features an average neutron leakage probability of $P_L = 4.4\%$ and a fraction of neutrons absorbed in the control systems of $P_{RC} = 2.2\%$ [17,18]. The maximum possible average burnup, also deduced from a detailed search for the equilibrium cycle, was found to be 55% FIMA [17,18]. The P_L and P_{RC} values pertaining to this core are, respectively, 6.95% and 2.1%. The leakage probability of the maximum discharge burnup core is higher than that of the minimum required burnup core since the radial power distribution in the high burnup core peaks closer to the outer core periphery than in the minimum burnup core.

Figure 1a shows the burnup-dependent neutron balance evolution in a core that features the P_L and P_{RC} values of the minimum required burnup core. Shown in the figure is a plot of the left hand side integral of Equation 1, using the above mentioned values of P_L and P_{RC} and the burnup-dependent k_∞ values derived from the full core analysis [17,18]. The minimum required average burnup inferred from Figure 1a is close to 20%—slightly larger than the value obtained from the batch-wise full core analysis [18]. Likewise, Figure 1b shows the burnup-dependent neutron balance evolution in a core that features the P_L and P_{RC} values of the maximum attainable burnup core. The maximum attainable average discharge burnup inferred from Figure 1b is 54% FIMA—very close to the 55% calculated from the detailed fuel shuffling and burnup analysis [18].

Figure 1. Neutron balance *versus* burnup in large hard spectrum Stationary-Wave-Reactor (SWR) core designed to sustain the Breed-and-Burn (B&B) mode of operation at (a) the minimum required average burnup ($P_L = 4.4\%$; $P_{RC} = 2.2\%$) and (b) at the maximum possible average burnup ($P_L = 6.95\%$; $P_{RC} = 2.1\%$).

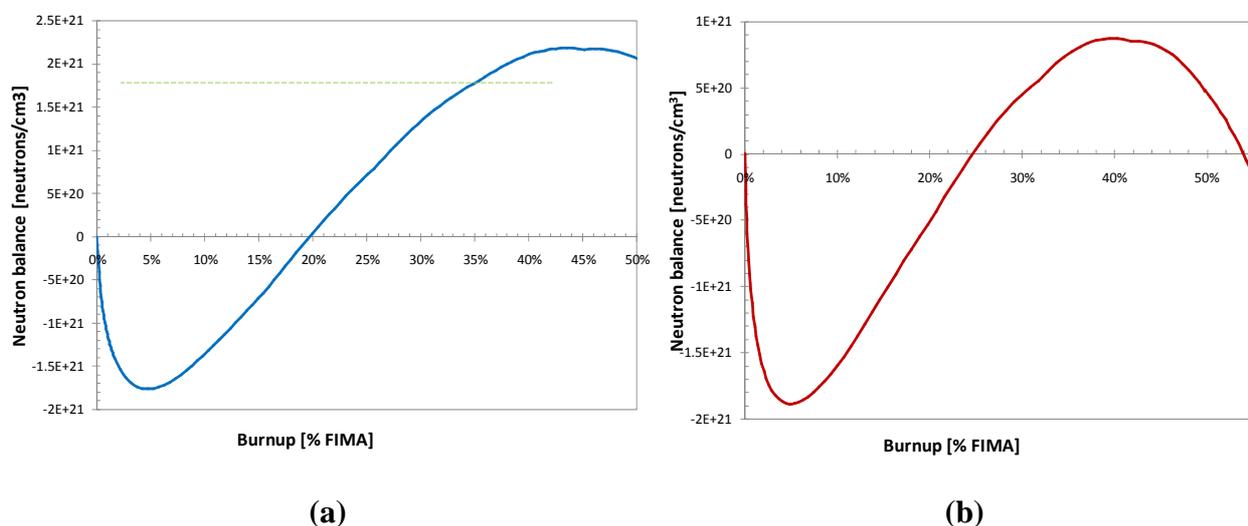
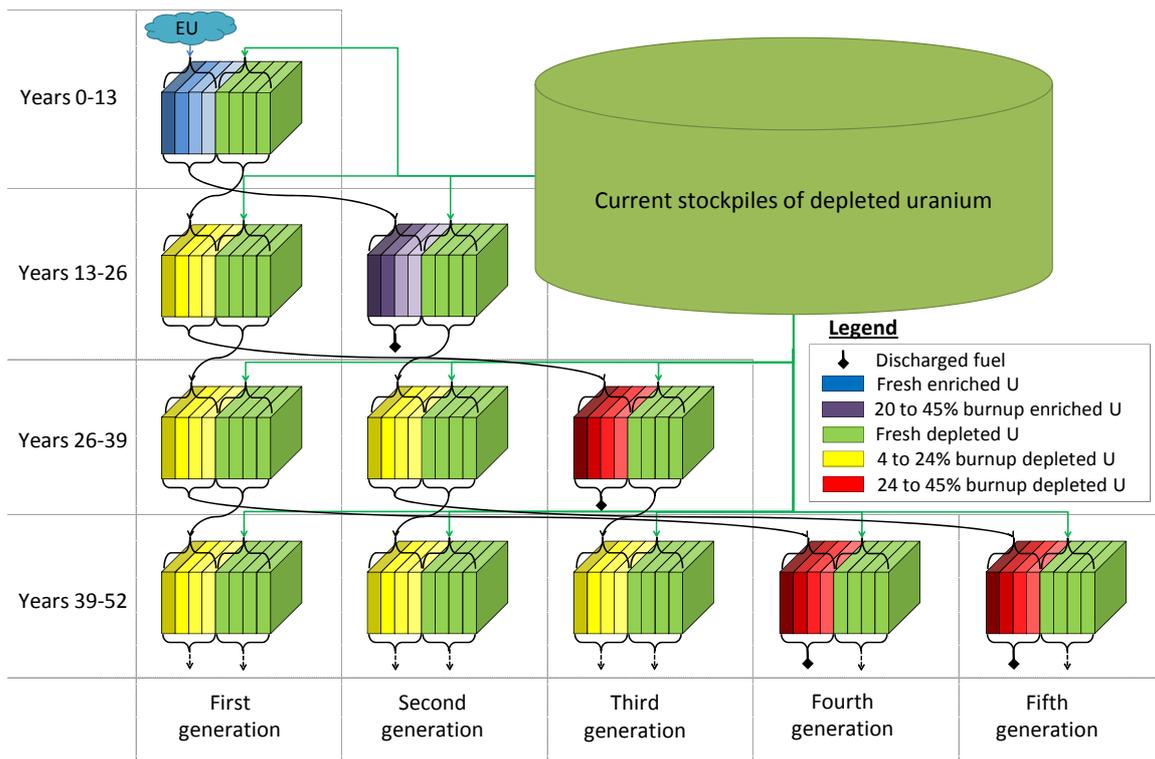


Figure 1a shows that the fuel discharged at an average burnup of 19.4% FIMA has sufficient excess reactivity to provide a total of additional $2.2E + 21$ excess neutrons per cm^3 of fuel—reached at a cumulative average discharged burnup of 42.5% FIMA. This is more than the $\sim 1.8E + 21$ neutrons that need to be provided per cm^3 of depleted uranium feed in order to turn it into a net neutron producer—corresponding to the minimum of Figure 1a curve. That is, the fuel discharged at 19.4% FIMA can serve, after reconditioning (aimed at relieving the radiation damage constraints) as the starter fuel for a new B&B core as described in the following sub-section.

2.2. Spawning Feasibility

The spawning mode of operation of B&B reactors is illustrated schematically in Figure 2. The number of B&B cores at generation “i” equals the number of B&B cores at generation “i-1” plus the number at generation “i-2”. Fissile fuel needs to be purchased only for the first core; thereafter, depleted uranium is the only fuel supply required for the growing fleet of B&B reactors.

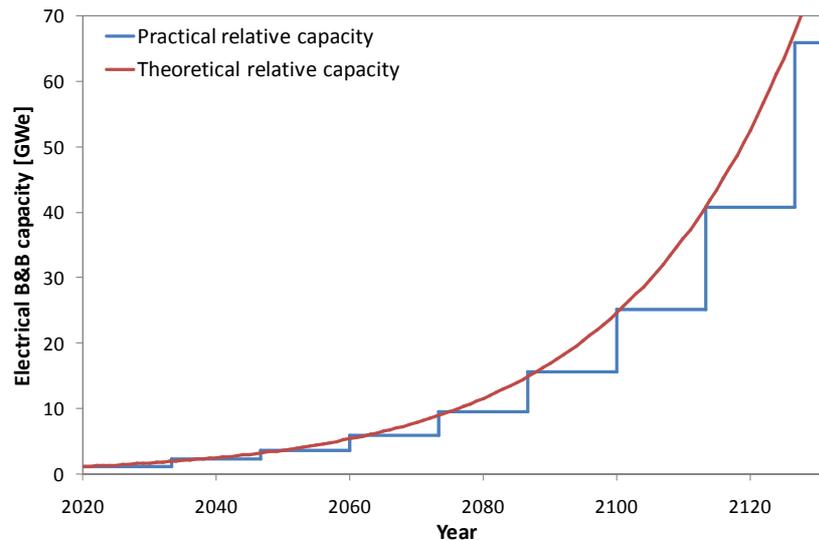
Figure 2. Schematic illustration of the spawning mode of B&B reactors.



The doubling time of such a spawning fleet of B&B reactors is defined as the time it takes to accumulate 50% of the core volume worth of discharged fuel—an amount found sufficient [17,18] to make a “starter” for a new core. As the equilibrium cycle lasts 2.05 years and there are 12 fuel batches in the B&B core analyzed, the doubling time is 12.3 effective full-power years (EFPY). Assuming a capacity factor of 90%, the doubling time is approximately 13.5 years. Figure 3 shows the resulting installed capacity evolution; the asymptotic B&B reactors capacity growth rate is 3.86% per year. This capacity growth rate is larger than even that of the most optimistic scenario for nuclear energy expansion rate forecasted by the IIASA—3.6% per year. If a single $3000 \text{ MW}_{\text{th}}/1.2 \text{ GW}_e$ B&B core is

started in 2020 and will be operated in the spawning mode featuring the 3.86%/y capacity growth rate, the total installed B&B capacity will be 25.2 GW_e by 2100 and 40.8 GW_e by 2120. Except for the several tons of enriched uranium or plutonium or TRU required for establishing initial criticality in the first (“Mother”) core, this expanding fleet of B&B reactors requires only depleted uranium for its fuel feed.

Figure 3. Electrical capacity evolution due to one large B&B reactor deployed in 2020 and operated in the spawning mode.



3. Travelling-Wave versus Stationary-Wave

3.1. Neutron Balance Considerations

The minimum burnup required for sustaining the B&B mode of operation is highly sensitive to the number of excess neutrons available for converting fertile into fissile fuel. For a given fuel type the number of excess neutrons tends to increase with the hardening of the neutron spectrum—due to increase in the average value of the fuel, and tends to decrease with enhanced parasitic neutron capture and enhanced neutron leakage probability.

The neutron balance in a TWR core is significantly different from that of a SWR core. Figure 4 presents a schematic illustration of the “fission wave” propagation along the core of a TWR like the CANDLER reactor developed by Professor Hiroshi Sekimoto *et al.* [11,12,22–24]. The “fresh fuel” (sometimes referred to as the “blanket”) is typically depleted uranium. Neutrons that leak from the “burning region” (sometimes also referred to as the “fission wave”) in the direction of the wave propagation have high probability of being captured in the blanket fuel and increase its fissile fuel concentration by, primarily, converting ^{238}U into ^{239}Pu . This process is illustrated in Figure 5 that shows, among other things, the ^{239}Pu concentration distribution along the core axis. When the ^{239}Pu concentration in a blanket zone gets high enough to make this blanket zone k_{∞} larger than 1.0, this blanket zone becomes a net neutron producer. While k_{∞} at a given location in front of the wave keeps increasing with time, the value of k_{∞} at a given location at the tail of the fission wave goes down with time due, primarily, to accumulation of fission products but also due to depletion of uranium and,

correspondingly, plutonium. A typical axial distribution of k_{∞} along a TWR core is illustrated in Figure 6. This k_{∞} evolution is responsible for the propagation of the fission wave in the direction of the blanket.

Due to enhanced probability for parasitic neutron capture in fission products, neutrons that leak from the high fission density zone in the backward direction have a smaller probability of contribution to the B&B process than neutrons that leak from the same high fission density zone in the direction of the wave propagation. Moreover, neutrons that leak-out from the core in the radial direction and are not scattered back do not contribute to the B&B process.

Figure 4. Schematic illustration of the Travelling-Wave-Reactors (TWR) core evolution with burnup. Courtesy of Hiroshi Sekimoto [11,12,22–24].

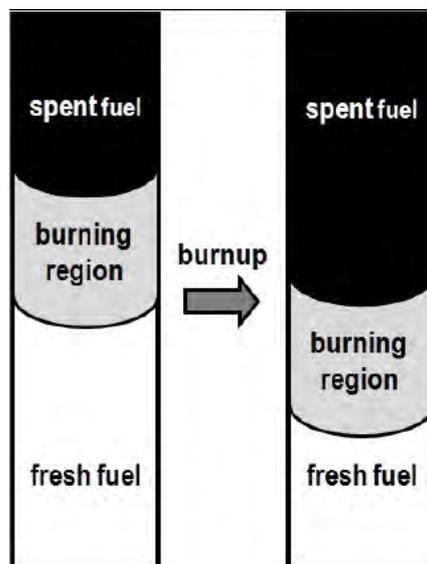


Figure 5. A snap shot of the concentration of selected fuel isotopes and of the neutron flux along the axis a TWR core. The fission wave propagates to the left. Courtesy of Hiroshi Sekimoto [11,12,22–24].

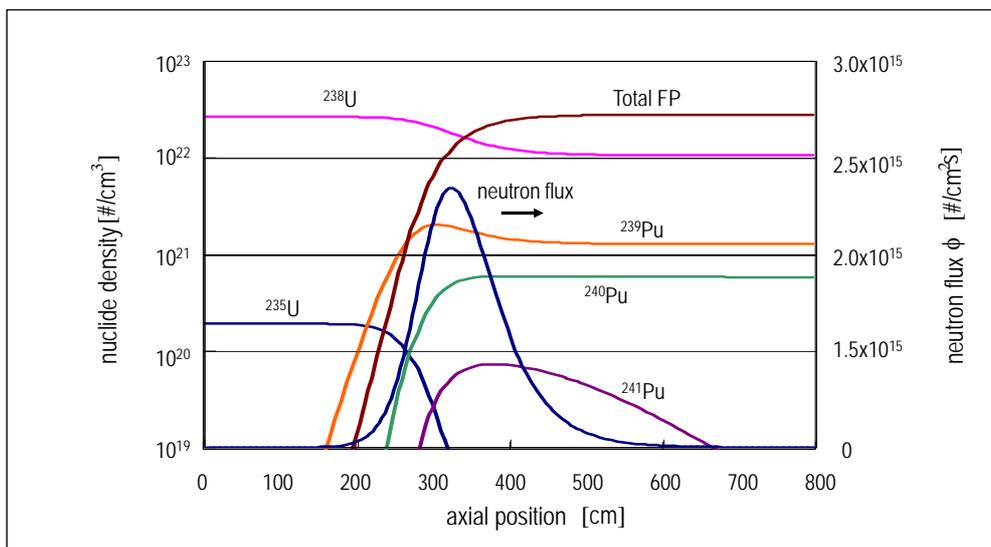
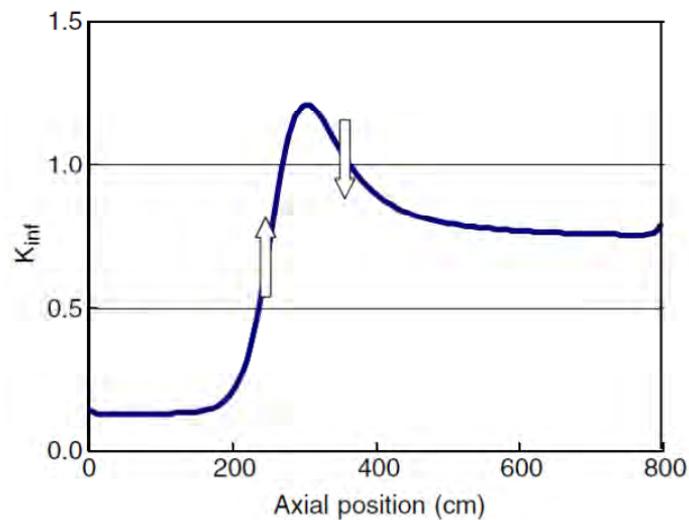


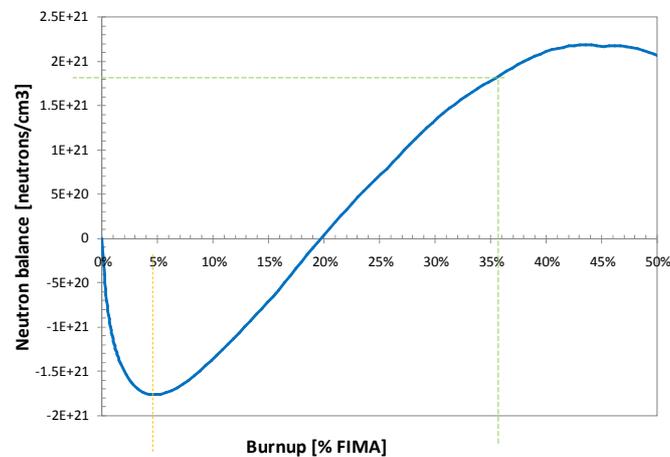
Figure 6. A snap shot of the k_{∞} distribution along the axis a TWR core. The fission wave propagates to the left. Courtesy of Hiroshi Sekimoto [11,12,22–24].



The situation is different in a SWR; the blanket, typically made initially of depleted uranium, radially surrounds the fission zone and is shuffled inward when accumulating adequate amount of fissile isotopes. Consequently, neutrons that leak from the fission zone in the radial direction have a high probability to contribute to the B&B process while neutrons that leak in the axial direction do not. In the minimum burnup B&B core designed at UCB [18] and briefly described in Section 2, the axial neutron leakage probability is 0.7% while the probability that neutrons will leak out from the radial blanket and, therefore, will not contribute to the B&B process is $\sim 3.7\%$. As a typical TWR core diameter is similar to that of a SWR core diameter, the TWR net radial leakage probability is expected to be larger since a TWR core does not have an effective radial blanket as SWR cores typically have. Based on the above considerations it is concluded that the fraction of the neutrons that are in excess of the number required for sustaining the chain reaction that can contribute to the B&B process is significantly smaller in a TWR than in a SWR of a comparable power level.

It is difficult to accurately estimate the fraction of the excess neutrons that do contribute to the B&B process in TWR and SWR cores without performing detailed 3-D core burnup analyses. It is possible, nevertheless, to get a rough estimate of the burnup implications of the difference in the wave propagation mode by assuming that the number of excess neutrons per unit burnup that are available for building up the fissile content in the blanket fuel in a TWR is only half that in a SWR core. This assumption is based on the simplified supposition that neutrons that leak in the backward 2 directions do not contribute to the buildup of fissile fuel in the TWR blanket. Factoring this assumption into the neutron balance analysis performed in Section 2, the total number of excess neutrons that need to be generated per cm^3 of TWR blanket fuel in order to turn it into a net neutron producer is twice the $1.8 \times 10^{21} \text{ n/cm}^3$ value inferred from Figure 1a for the SWR fuel. Figure 7 shows that the minimum burnup required for providing this number of excess neutrons is $\sim 36\%$ FIMA. In fact, the burnup of many of the CANDLE cores designed by Sekimoto *et al.* [11,12,22–24] feature an average burnup level that is in the vicinity of 40% FIMA—nearly twice that required for sustaining a B&B mode of operation in a SWR.

Figure 7. Plot of the left hand side of Equation 1 *versus* burnup for a large SWR core designed [17] to operate at the minimum burnup required for sustaining a B&B mode of operation.

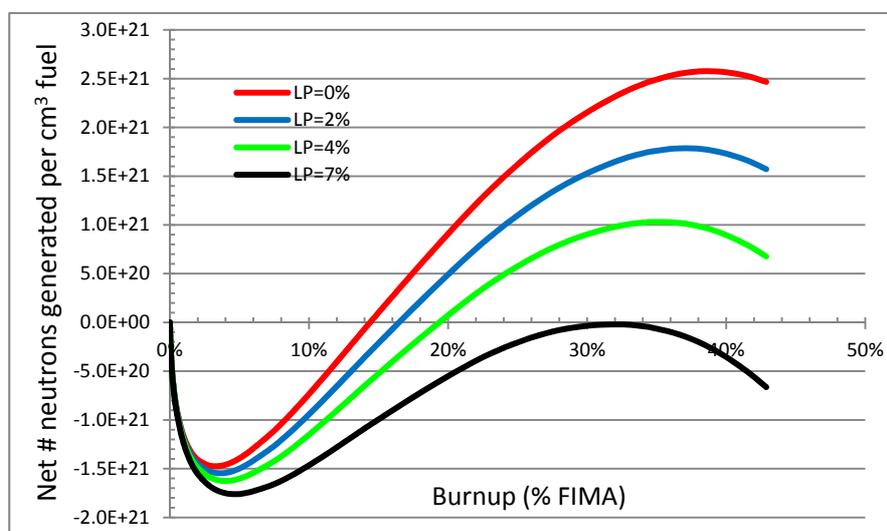


3.2. Practical Considerations

Perhaps the most challenging practical design feasibility issue for the SWR and, even more so, TWR is the ability of the fuel clad to maintain its mechanical integrity over the extended burnup required for sustaining the B&B mode of operation. The peak fuel burnup corresponding to a minimum required SWR batch average burnup of 20% FIMA is close to 30% FIMA. The corresponding radiation damage to an HT-9 clad is in the vicinity of 550 dpa. This is more than double the maximum value of 200 dpa HT-9 structure was subjected to so far. It is possible that future irradiation experiments along with innovative design of the fuel rods will prove that clad made of HT-9 or, more likely, of an improved structural material will be able to safely accommodate ~550 dpa. It is most unlikely, though, that a TWR core could reach its minimum required burnup without fuel reconditioning. This is one of the reasons that make SWR more practical than TWR for a near-term implementation of the B&B mode of operation. As suggested in Section 4, it is possible to start introducing the stationary wave type B&B mode of operation using already proven technology.

Another unique challenge is to design the SWR to be inherently safe. Most conventional fast reactor cores are designed to be oblate—having a relatively small height-to-diameter ratio so as to enhance the neutron leakage probability in the axial direction. The primary objective of this design approach is to reduce the typical positive reactivity effect of coolant density reduction—either by temperature increase or voiding. The neutron leakage probability in typical fast reactor core designs is close to 20%—significantly larger than the 4.4% of the B&B SWR core designed at UCB for which the results of Figure 1a and associated discussion in Section 2 pertain. In fact, neutron balance analysis performed using the simplified approach presented in Section 2 suggests that, as illustrated in Figure 8, it is not feasible to establish a sustainable B&B mode of operation when the neutron leakage probability exceeds ~7%. Detailed 3-D B&B core design performed at UCB [26] extends the maximum feasible leakage probability to up to 9% or, possibly, 10%.

Figure 8. Plot of the left hand side of Equation 1 *versus* burnup for several values of the neutron leakage probability. $P_{CR} = 1\%$.



It has been proposed [8,27,28] to passively compensate for the relatively large positive coolant temperature reactivity coefficient by insertion into the active core region of ${}^6\text{Li}$ neutron poison in a way that is passively actuated by coolant temperature increase. Recent analysis [28] shows that such ${}^6\text{Li}$ injection systems can be designed to provide a strong negative reactivity feedback without significantly impairing the neutron economy, reactor operation and cost, while giving safety margins that equal or exceed those of smaller and leakier fast reactor core designs. Nevertheless, detailed time-dependent simulation and experimental verification of the feasibility and license-ability of such passive ${}^6\text{Li}$ injection systems are yet to be performed.

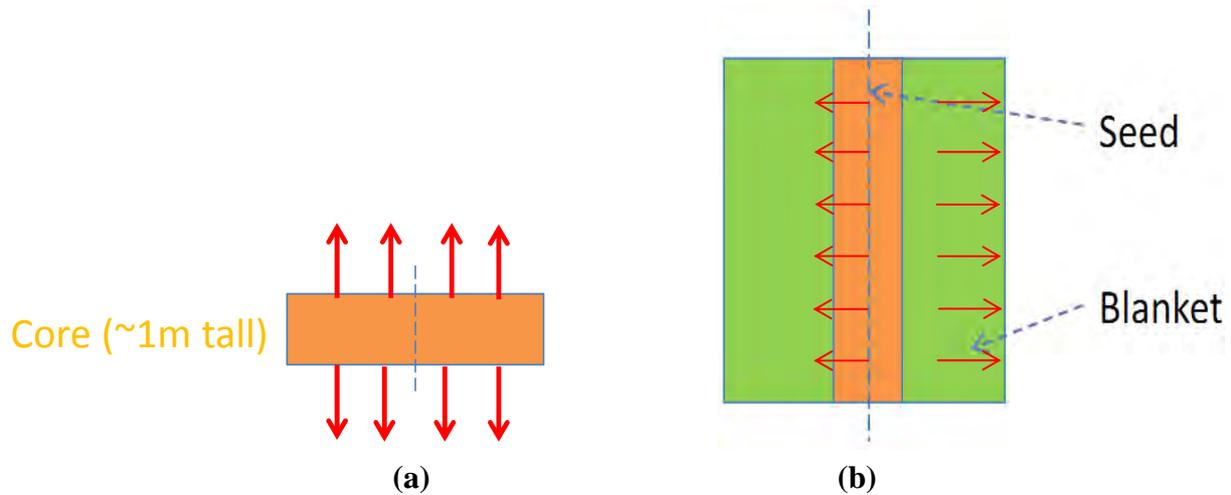
4. Phased Commercialization of Breed-and-Burn Reactors

4.1. Concept Introduction

Fast sodium-cooled critical reactors (SFR), such as the Advanced Recycling Reactor (ARR) and the Advanced Burner Reactor (ABR), are designed to have an oblate (“pancake” shape) core the dominant neutron leakage from which is in the axial direction, as schematically illustrated in Figure 9a [29–32]. A typical neutron leakage probability from such an SFR core is on the order of 20%; the majority of the neutrons leak in the axial direction and do not have any constructive usage. The relatively high neutron leakage probability helps designing the core to have a smaller positive (and seldom, negative) coolant temperature and coolant voiding reactivity effect. ABR cores are sometimes designed to have an even larger neutron leakage probability in order to reduce their conversion ratio (typically $CR = 0.5$ to 0.75).

Instead of designing the SFR core to be oblate with the dominant neutron leakage being in the axial direction, it is proposed [33,34] to design the SFR core to be of a prolate (“cigar” like) shape for which the majority of the neutron leakage is in the radial direction, and to make use of the leaking neutrons to “drive” a subcritical B&B blanket that radially surrounds the core, as illustrated in Figure 9b.

Figure 9. Schematic layout of (a) a conventional fast reactor core and (b) the proposed core having a “cigar shape” seed surrounded by a blanket. Not to scale.



The composition of the driver fuel will be similar to that of a conventional SFR core; it can use TRU from LWR UNF and can be designed to have a low conversion ratio as in an ABR—in case there is interest in LWR TRU transmutation, or to be TRU self-sustaining, as in ARR cores. It is envisioned that the driver fuel will be multi-recycled, as is the case for conventional SFR—typically when the fuel accumulates ~ 100 GWD/tHM corresponding to 200 dpa in an HT-9 fuel clad, or to a fast neutron fluence constraint of 4×10^{23} n/cm² [29–32]. The blanket can be fueled with depleted uranium, thorium or another type of low fissile content fuel such as reprocessed LWR UNF. The blanket is to operate on the once-through fuel cycle; when a blanket fuel assembly reaches its irradiation induced design constraint—initially assumed to be 200 dpa in the fuel clad, it will be discharged and a fresh fuel assembly will be loaded into the blanket. In order to maximize the neutron economy, it is desirable to load the fresh blanket fuel in the outer part of the blanket and gradually shuffle it inward as it builds up more fissile fuel.

The proposed seed (driver)-and-blanket core and fuel cycle concept could facilitate the development and early introduction of depleted uranium-fed B&B reactor technology by designing the B&B blanket to be subcritical and “driving” it by the excess neutrons leaking out from the TRU driver—initially up to the licensable 200 dpa for HT-9 cladding (average burnup $\sim 10\%$ FIMA). The blanket fuel discharge burnup will be progressively increased as fuel/cladding materials that are licensable to higher FIMA/dpa level become available; up until, hopefully, reaching a level of $\sim 30\%$ FIMA/550 dpa that enables sustainment of the B&B mode of operation in a critical stationary-wave core.

The proposed seed-and-blanket core is likely to significantly reduce the SFR fuel cycle cost and, thus, to improve the SFR economic viability. This is because the cost of a depleted uranium fuel assembly required for the blanket is significantly smaller than the cost of a TRU-containing fuel assembly required for the driver, while the amount of energy to be generated per unit weight of blanket fuel and driver fuel is comparable; it is limited by similar radiation damage constraints. The larger the fraction of the core power to be generated by the blanket is, the smaller need be the installed capacity of UNF reprocessing and recycling plants per unit of electricity generated in the seed-and-blanket core and, hence, the smaller will be the fuel cycle cost.

4.2. Depleted Uranium B&B Blankets

A study recently initiated at UCB [35] indicates that it is possible to generate nearly 2/3 of the total core power from a subcritical B&B blanket fueled with depleted uranium without exceeding 300 dpa. Both seed and blanket use an IFR type metallic fuel made of depleted uranium alloyed with Zr and, in case of the seed fuel, also TRU. The HT-9 ferritic-martensitic steel is used for the cladding and sodium for the coolant and bonding material (filling the initial fuel-clad gap).

The preliminary design analysis assumed that the metallic fuel contains 10 weight % Zr; that its smear density is 68% and that the hexagonal lattice pitch to fuel outer diameter ratio is 1.24. The corresponding fuel, structural material and sodium volume fractions are, respectively, 43.0%, 16.9% and 40.2%. The core height is 250 cm, the seed outer diameter is 54.7 cm and the blanket outer diameter is 179.4 cm. The TRU loading in the seed fuel is adjusted so as to be TRU self-sustaining as in an ARR core and the seed diameter is the minimum required for generating a total of 800MW_{th} in the core without exceeding thermal-hydraulic design constraints.

Table 1 presents selected characteristics of a very preliminary seed-and-B&B blanket equilibrium core conceptual design recently arrived at [35]. The seed has a couple of batches; each cycle, that lasts 1150 EFPD (3.15 EFPY), the inner seed batch is discharged, the outer seed batch is shuffled inward, and a recycled seed fuel is loaded at the outer seed batch location. The blanket is made of 16 batches. Each cycle the innermost blanket batch is discharged, the other blanket fuel batches are shuffled inward, and a fresh blanket fuel is loaded at the outermost blanket batch location. The discharged blanket fuel is not recycled. The small burnup reactivity swing will facilitate the design of the control and safety systems of such a core.

Table 1. Selected performance characteristics of an illustrative seed-and-blanket core *.

Characteristic	Value
Number of seed batches	2
Number of blanket batches	16
Seed/Blanket batch number	2/16
Fuel cycle length (EFPD)	1150
Seed/Blanket total resident time (EFPD)	2300/18400
BOEC/EOEC k_{eff}	1.00242/1.00580
Discharged seed fuel peak/average burnup (%)	16.37/12.30
Discharged blanket fuel peak/average burnup (%)	14.36/9.56
Peak DPA in seed/blanket fuel clad at discharge	280/293 ^a
Power fraction from blanket (%)	60.3

* Design is far from optimal.

The above fuel management scheme is far from optimal and is not the most practical; it is desirable to reduce the number of blanket batches and the need for fuel shuffling. It is likely possible to halve the number of blanket batches and to double the blanket batch cycle time without significantly degrading the overall core performance. It is also possible to eliminate seed fuel shuffling with little penalty on the core performance. Optimal seed-and-blanket core designs will be thoroughly explored in the framework of the NEUP project [34] that is soon to be initiated.

The time required for the blanket to reach an equilibrium composition exceeds (16 batches \times 3.15 EFPY \Rightarrow 50.4 EFPY). It is possible, nevertheless, to initially load the inner part of the blanket with enriched fuel so as to get the blanket to generate from day one a similar fraction of the total core power as the blanket generates in the equilibrium core. The total amount of TRU in the preliminary equilibrium core described above is 1.22 tons in the seed and 3.33 tons in the blanket. Instead of TRU, it is possible to use for the initial blanket enriched uranium. After the first fuel loading, only depleted uranium (or other fertile fuel) will be fed into the blanket. Optimal strategies for the approach to the equilibrium core performance will be explored as part of the NEUP project [34].

4.3. Thorium B&B Blankets

A recent study found [17] that critical B&B cores cannot be designed using pure thorium feed fuel. This is so because, in the SFR spectrum, (a) $(^{233}\text{U}) < (^{239}\text{Pu})$ and (b) the fast fission probability of ^{232}Th is significantly smaller than that of ^{238}U .

Nevertheless, thorium can be used as the feed-fuel for a subcritical B&B blanket that is driven by neutrons that leak out from a critical seed. In fact, there is a unique synergism between an ABR-type seed and a thorium blanket; rather than just incinerating TRU in ABR cores, the proposed core concept will convert part of the fissioned TRU into ^{233}U . This ^{233}U may be valuable for starting in the future a self-sustaining ^{233}U -Th energy system (a number of which are being proposed). If and when ^{233}U is a commercial commodity, its value will further improve the economic viability of the TRU driver—Th blanket SFR concept. No ^{233}U /Th fuel recycling capability is required for utilization of thorium for nuclear energy generation in the seed-and-blanket reactor.

A preliminary analysis indicates [33,36] that it is possible to generate in thorium-fueled B&B blankets at least 1/3 of the total power of the seed-and-blanket core while fissioning up to 15% of the fed thorium without exceeding the cladding 200 dpa constraint. The amount of energy thus obtained per kg of mined Th is more than 30 times the amount of energy extracted in the once-through LWRs per kg of natural uranium mined. This is because about 90% of the mined U turns into depleted U and only ~5% of the enriched U is fissioned; also, the efficiency of converting thermal energy to electricity in SFR is expected to be ~40% *versus* ~32% in LWR.

4.4. Other B&B Blanket Options

It may be possible to use for the blanket reconditioned LWR used nuclear fuel. The required functions of the LWR UNF reconditioning are removal of the gaseous fission products and zircaloy cladding and fabrication of fuel rods and fuel assemblies of the dimensions and design that is suitable for the SFR blanket, using HT-9 or another acceptable type of cladding material. There is no need to remove from the LWR UNF any of the actinides or solid fission products and there is no need to convert the fuel from an oxide to a metal alloy. Whereas oxide fuel cannot establish a sustainable B&B mode of operation in a critical core [17], it can generate a significant amount of extra energy in a subcritical B&B blanket—possibly more than twice the amount of energy it generated in the LWR. An AIROX or DUPIC-like process can be used for decladding the LWR UNF and removing the gaseous fission products [37–41].

Alternatively, a cermet-type fuel similar to that recently proposed by Walters and Wade [42] might be used. Walter and Wade are proposing to replace the depleted uranium, that is the commonly used makeup for recycled SFR fuel, by an equivalent amount of LWR used nuclear fuel in the form of crushed oxide particles. For their application, Walter and Wade are suggesting that the “crushed U/Pu/MA/fission product oxide particles, that can be generated in an AIROX-like process described above, would be well blended with the uranium/transuranic metal alloy particles recovered by the pyro-recycle process of the SFR fuel and then the mixed powder would be vibropacted into the fuel cladding. The processes would all be done remotely. After return to the reactor, and upon ~1 atom % burnup, the mixed particle bed will swell under fission gas production and restructure into a solid cermet fuel form comprised of oxide particles embedded in a metallic fuel alloy matrix—containing interconnected porosity and filling the interior radius of the cladding at a smear density of 70–75%” [42]. For the once-through B&B blanket, we envision mixing as much crushed U/Pu/MA/fission-product oxide particles from reconditioned LWR UNF with metallic particles made of depleted uranium alloyed with Zr or with metallic thorium particles. The implications of such options will be studied in a later stage of the NEUP project [34].

5. Impact on Energy Sustainability and Economic Stability

Table 2 compares the estimated uranium utilization that could be achieved with B&B reactors that are designed and/or operated in either one of the following five modes, all using depleted uranium for the blanket fuel feed:

- (a) A seed-driven subcritical B&B blanket the fuel of which is discharged at an average burnup of 10% FIMA. No fuel reconditioning is required.
- (b) A critical stationary-wave B&B core using a fuel that can maintain its integrity up to an average burnup of at least 20% FIMA. No fuel reconditioning is required unless the discharged fuel is to be used for spawning new B&B reactors.
- (c) Like “b” along with a successful development of the technology for a single fuel reconditioning at ~20% burnup. Spawning new SWR is possible.
- (d) A critical SWR or, possibly, TWR with 2 or more fuel reconditioning steps that will enable to achieve the maximum attainable burnup of ~50% FIMA (*versus* 55% obtained in the UCB large B&B core analysis [18]) without separation of most of the solid fission products.
- (e) Traditional fast breeder reactor approach in which fuel is reprocessed many times (every 10% FIMA or so). It assumes extraction of all of the fission products and addition of depleted uranium makeup fuel at each recycle. There is no limit to the number of fuel recycles.

Also given in Table 2 is the uranium utilization in the reference scenario of contemporary LWRs that operate with the once-through fuel cycle and discharge their fuel at 50 GWD/T.

The relative uranium utilization values given in Table 2 are per unit of electrical energy generated. In converting thermal energy to electrical energy it is assumed that fast reactors convert thermal energy into electricity at 20% higher efficiency than LWRs.

The rightmost column in Table 2 gives the number of years the B&B reactors could supply electricity at present day USA total annual consumption rate from all sources (assumed 4200 million

MW_e/year) if they are to be fueled only with the depleted uranium stockpiles (“waste”) that will be accumulated in the US from the fueling of LWRs ($\sim 1.3 \times 10^6$ tons) and B&B reactors ($\sim 0.5 \times 10^6$ tons) until the end of deployment of the first generation of B&B reactors—assumed in the second half of the 21st century.

Table 2. Estimated uranium utilization limits and energy value of depleted uranium when used in B&B reactors and in Light Water Reactors (LWR).

Mode of operation	Uranium utilization	Relative U utilization	No. of years at present supply
Light Water Reactors (LWRs)—reference	0.6%	1	0
(a) subcritical B&B blanket; no reconditioning	10%	20	400
(b) SWR; 20% average discharge BU	20%	40	800
(c) SWR, 1 reconditioning @ 20%; spawning possible	40%	80	1600
(d) SWR or TWR, with >1 fuel reconditioning	50%	100	2000
(e) Fast reactor with continuous recycling	>95%	>190	3900

It is observed that using practically proven fuel technology (except for the length of the fuel rods which is envisioned to be 2 to 3 meters versus ~ 1 meter for conventional SFR and ~ 4 m for LWR) in subcritical B&B blankets it is possible to achieve a uranium utilization that is 20-fold that offered by LWR. A successful development of B&B reactors that can achieve 20% average fuel burnup which, hopefully, could be achieved without fuel reconditioning, will offer 40-fold increase in the uranium ore utilization *versus* that presently achieved. A successful development of a fuel reconditioning technology could increase the attainable uranium utilization to close to 100-fold that achieved in contemporary LWRs. This corresponds to extraction of approximately 50% of the nuclear energy worth of depleted (and natural) uranium. All the above options do not require separation of most of the solid fission products from the actinides. For the utilization of the remainder 50% it will be necessary to develop economically viable and societal acceptable fuel reprocessing technology that will separate the fission products from the actinides. Such a reprocessing could be deferred, though, by several centuries, as the existing stockpiles of depleted uranium can provide all our electricity needs for between 400 to 2000 years (rightmost column of Table 2). Basically, the same SFR technology can be used for implementing the different options.

6. Conclusions

A successful development of metallic fuel and cladding that can maintain the fuel rod integrity up to a peak burnup of $\sim 30\%$ FIMA and peak radiation damage of ~ 550 dpa will enable the operation of stationary-wave fast reactors in a sustainable Breed-and-Burn (B&B) mode using depleted uranium for the feed fuel. Such SWR reactors will offer 40-folds increase in the uranium ore utilization relative to contemporary LWR while operating in a once-through fuel cycle. A successful development of a fuel reconditioning technology could enable an increase in the attainable uranium utilization of SWR to 100-folds its present value without separation of actinides from most of the fission products. It will also enable the use of reconditioned B&B fuel to provide the initial fissile fuel loading required to spawn new SWR without the need for external supply of fissile fuel. The growth rate of the installed

capacity of SWR possible to achieve using such a spawning mode of operation is estimated to be nearly 4% per year. A successful development of a fuel reconditioning technology could also enable deployment of traveling-wave fast reactors.

As it may take significant time and R&D effort to develop the fuel technology that is required for operating a sustainable SWR that is fed with depleted uranium, it is proposed to start benefiting from the B&B mode of operation by deploying seed-and-blanket fast reactors in which a subcritical B&B blanket is driven by neutrons leaking from a critical seed, without exceeding $\sim 100\%$ FIMA/200 dpa; that is, relying on proven fuel technology. Such seed-and-blanket reactors are expected to be more economically viable than conventional fast reactors and can facilitate the phased commercialization of critical B&B reactors. When using depleted uranium for its feed fuel, the subcritical B&B blanket could generate close to $2/3$ of the total core power without exceeding the radiation damage constraints. The amount of fuel reprocessing and TRU fuel re-fabrication required for the seed fuel of such a seed-and-blanket core is only $\sim 1/3$ that required for a conventional fast reactor core, when measured on per unit of electricity generated by these cores. As a result, the fuel cycle cost of the seed-and-blanket reactor is expected to be significantly smaller than that of a conventional fast reactor. As fuel designs that can be certified to operate at higher than $\sim 100\%$ FIMA/200 dpa become available, the seed-and-blanket core could be designed to discharge the fuel at higher burnups and to offer higher uranium utilization.

The B&B blankets of seed-and-blanket cores can also be fed with thorium. The amount of energy that can be generated per kg of mined thorium is estimated to be more than 30 times the amount of energy extracted in the once-through LWRs per kg of natural uranium mined.

An additional interesting option for the feed fuel of the B&B blanket is reconditioned LWR used nuclear fuel. This may enable the generation of additional energy from the UNF without reprocessing; possibly close to twice the amount of energy it generated in the LWR.

The energy value of the depleted uranium stockpiles (“waste”) to be accumulated in the USA until the middle of the present century is equivalent, when used in the B&B reactors, to the total 2010 USA supply of electricity up to 8 centuries without fuel reconditioning and up to 20 centuries with fuel reconditioning. Therefore, a successful development of B&B reactors could provide a great measure of energy security and cost stability.

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Conflict of Interest

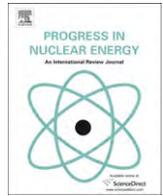
The authors declare no conflict of interest.

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Feasibility of lead cooled breed and burn reactors

Florent Heidet*, Ehud Greenspan

Department of Nuclear Engineering, University of California, Berkeley, CA 94720-1730, USA

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ABSTRACT

The objective of the present work is to assess the feasibility and achievable performance of a large Breed-and-Burn (B&B) reactor core that is cooled with heavy-liquid-metal (HLM), such as lead or lead-bismuth eutectic (LBE), relative to a sodium-cooled B&B core. With the exception for the initial critical fissile fuel loading the B&B reactor systems are to be fuelled only with fertile material. When the fuel reaches its radiation damage limit it is reconditioned and recycled without separating the actinides from most of the fission products.

It was found that in order to get a total power of 3000 MWth as of the reference sodium-cooled B&B core, core pitch to diameter ratio needs to be increased from the 1.112 value of the reference sodium-cooled core to 1.3 for Pb and LBE coolants and to 1.24 for PbLi coolant. As a consequence, the minimum burnup required to sustain the B&B mode of operation is approximately 29% FIMA for the LBE and PbLi cooled cores – larger than the 21% FIMA required for the reference sodium cooled B&B reactor. Had the Pb been enriched to nearly 100% ^{208}Pb , the minimum required burnup would be 22.5%; less than 10% higher than for Na-cooled core. The maximum possible accumulated burnup in the LBE-cooled B&B core is $\sim 45\%$ FIMA – smaller than the 55% FIMA possible to achieve in the reference sodium cooled core. For the same core volume HLM-cooled cores require 26% smaller mass of fissile material to establish initial criticality.

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1. Introduction

The Breed and Burn (B&B) reactors being studied at the University of California, Berkeley (UCB), are similar in concept to the TIT CANDLE (Sekimoto and Ryu, 2000) and the Terra Power Traveling Wave (Gilleland et al., 2008) reactors. Except for the initial critical fissile fuel loading this reactor type is to be fuelled with fertile material. The B&B reactor systems under study at UCB feature reconditioning of the fuel whenever its clad reaches its radiation damage limit. The reconditioning operation includes removal of the gaseous and volatile fission products and recladding of the fuel. The performance attainable from sodium-cooled B&B cores has recently been quantified (Heidet and Greenspan, 2010).

The primary objective of the present work is to assess the feasibility of establishing the B&B mode of operation in a large B&B core that is cooled with heavy-liquid-metal (HLM), such as lead, lead-bismuth eutectic (LBE) or lead-lithium (PbLi). The study also assesses the feasibility of achieving a similar average power density as in the reference sodium cooled B&B core. The interest in PbLi, discussed in (Cognet and Greenspan, in press), is due to its lower

melting temperature, lower radiotoxicity of its activation products and the high abundance and low cost of lithium. However, it may not be practical because of the high cost associated with the need to enrich the Li with ^7Li as well as because of its chemical reactivity and difficulty in corrosion control. A thorough evaluation of these effects needs to be performed. For academic interest, a limited analysis is also done for Pb coolant that is highly enriched in ^{208}Pb – the Pb isotope that features the lowest neutron capture cross section and highest threshold for inelastic scattering.

The primary potential advantage of using lead-based coolant instead of sodium coolant is the elimination of the need for a secondary coolant loop – since lead-based alloys do not chemically react with air and water. This could significantly decrease the cost of the reactor cooling system. As the boiling temperature of lead and its alloys is close to 800 °C, higher than that of sodium (Table 1), the probability of coolant boiling and coolant voiding accidents is smaller in the HLM reactors. However, lead is more corrosive and erosive than sodium and may cause early failure of fuel rods, if the lead chemistry is not adequately controlled.

2. Fast reactor core studied

The HLM cooled fast reactor core examined is similar to the large sodium-cooled B&B core recently studied by Heidet & Greenspan in

* Corresponding author. Tel.: +1 510 643 9983.

E-mail addresses: fheidet@anl.gov (F. Heidet), gehud@nuc.berkeley.edu (E. Greenspan).

Table 1
Comparison of Selected Thermo-physical Properties of Na, LBE and PbLi at 800 K.

Coolant	Na	Pb	LBE	PbLi
Melting temperature [°C]	97.7	327	123.5	235
Boiling temperature [°C]	883	1749	1670	N.A.
Density [g/cm ³]	0.828	10.411	10.013	9.569
Specific heat [J/kg.K]	1260	144.1	146.5	187.7
Volumetric specific heat [J/cm ³ .K]	1.043	1.500	1.467	1.796
Dynamic viscosity [N.s/m ²]	2.27E-04	1.73E-03	1.30E-03	1.10E-03
Thermal conductivity [W/m.K]	62.9	18.0	14.9	17.6
Thermal expansion coefficient [K ⁻¹]	2.82E-04	1.15E-04	1.38E-04	1.24E-04

(Heidet and Greenspan, 2010). It uses ternary metallic fuel U–Pu–Zr with 10 wt% zirconium, a fuel nominal density of 15.85 g/cm³ and a smear density of 75%. The structural and cladding materials are the ferritic-martensitic alloy, HT-9. The overall fuel, gap, coolant and structural material volume fractions are 37.5%, 12.5%, 28% and 22%, respectively. Assuming that the fuel assemblies have a pitch, inter-duct gap and duct thickness of, respectively, 161.42 mm, 4.32 mm and 3.94 mm, the above mentioned volume fractions correspond to a pitch-to-diameter ratio (P/D) of 1.112 – near the lower limit used in liquid sodium cooled reactors (IAEA, 2006) – and a coolant volume fraction of 26.5%, inside an assembly. The core height is 209.36 cm and the equivalent core diameter, made of 564 fuel assemblies, is 4.03 m. The general core layout is provided in Fig. 1.

For the reference core designed with sodium coolant, it was found (Heidet and Greenspan, in press) possible to generate 3000 MWth when the fuel assemblies are made of 271 fuel rods,

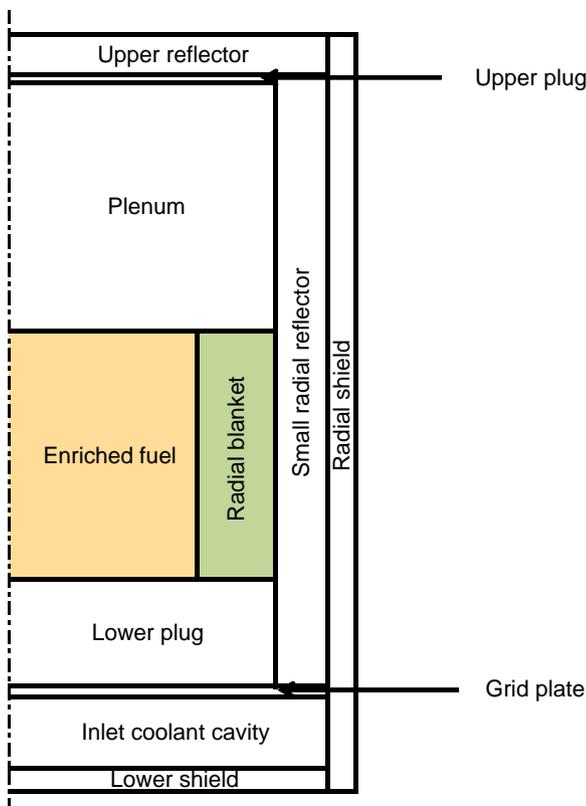


Fig. 1. Large B&B core layout.

with P/D = 1.112. The corresponding peak cladding and fuel temperatures are 597 °C and 650 °C, respectively. Those results were obtained with an assumed coolant inlet temperature of 395 °C, a peak coolant outlet temperature of 580 °C and a maximum pressure drop of 1 MPa.

3. Thermal hydraulic/neutronics trade-off study

3.1. Thermal hydraulic considerations

The maximum velocity of lead based alloys in the core is limited by the erosion rate of the protective oxide layer formed on the cladding. Oxidation of iron and/or other additives to the cladding, such as Si or Al, forms a protective layer on the cladding, the stability of which can be controlled by oxygen pressure. Erosion of this layer will reduce the cladding thickness and may, eventually, cause it to fail. The commonly accepted velocity limit for lead-based coolants is 2–3 m/s (Zhang and Li, 2008), but improved alloys, such as are currently under development by Prof. Ballinger et al. at MIT may enable increasing the velocity up to 6 m/s (Short et al., 2010). In this study, 6 m/s is assumed for the upper permissible velocity for determining the maximum achievable power. As the composition of Prof. Ballinger's material was not known, the HT-9 composition is used instead. The impact of a different cladding composition on the neutronics performance is expected to be small and not to change the conclusions. The maximum permissible coolant pressure drop across the core is taken to be 1 MPa (IAEA, 2006) and the coolant inlet temperature is assumed to be 395 °C. The same values were used in (Heidet and Greenspan, in press) for the sodium cooled B&B core. Additional constraints assumed are the maximum coolant outlet temperature and maximum acceptable cladding and fuel temperatures of, respectively, 580 °C, 650 °C (Hofman et al., 1997; Kim et al., 2008) and 1050 °C. The maximum outlet temperature corresponds to the outlet coolant temperature in the fuel assembly producing the highest power. The thermo-physical properties of lead, LBE and PbLi are taken from (Sobolev, 2007; Suspuglas and Greenspan, 2005) and (Masdelevalls et al., 2008). Selected properties of lead, LBE, PbLi and sodium at 800 K are compared in Table 1.

The thermal hydraulic analysis is performed with Pb, LBE and PbLi coolants at different P/D ratio and different number of fuel rods per assembly in the large B&B core. The radial power peaking factor is assumed to be 1.7 – the same as in the sodium cooled reference core. Without using flow orificing, the mixed coolant outlet temperature corresponding to the maximum coolant outlet temperature of 580 °C is approximately 510 °C. This is somewhat below the temperature usually observed in fast reactors (IAEA, 2006). However, flow orificing could, in principle, be used to reduce the flow rate in low power fuel assemblies thus increasing the average core outlet temperature. Due to the absence of a secondary coolant loop, lead-based coolants could theoretically be operated with a lower outlet temperature than sodium without penalizing the energy conversion efficiency. For convenience of comparison, the same inlet and outlet temperatures were assumed for all the cores studied.

3.2. Thermal hydraulic results

The thermal-hydraulic analysis (Suspuglas and Greenspan, 2005) is performed for the hottest assembly and the results are provided in Table 2 for Pb, Table 3 for LBE and Table 4 for PbLi, along with the volume fractions inside an assembly. For every configuration analyzed the core power is limited by the maximum coolant pressure drop. The coolant velocity ranges from 2.3 m/s to almost 4.5 m/s, depending on the number of fuel rods per assembly.

Table 2

Thermal Hydraulic Performance of the Pb Cooled Large B&B Core with a Radial Power Peaking Factor of 1.7.

Parameters	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6
p/d ratio	1.1	1.2	1.3	1.3	1.3	1.3
Fuel rod pitch [mm]	9.06	9.06	9.06	11.48	13.24	15.64
Fuel rod diameter [mm]	8.24	7.55	6.97	8.83	10.19	12.03
Fuel rods per assembly	271	271	271	169	127	91
Clad thickness [mm]	0.667	0.611	0.564	0.715	0.824	0.974
Coolant volume fraction	25.0%	37.0%	46.3%	46.3%	46.3%	46.3%
Cladding volume fraction	22.3%	18.7%	16.0%	16.0%	16.0%	16.0%
Fuel volume fraction	39.5%	33.2%	28.3%	28.3%	28.3%	28.3%
Pb velocity [m/s]	2.22	2.91	3.59	3.89	4.06	4.25
Maximum power [MW]	1016	1969	3040	3295	3439	3599
Max. inner cladding temp. [°C]	589	596	604	619	632	653
Max. fuel temperature [°C]	611	638	670	769	859	1004

It is found that due to the core pressure drop constraint, the attainable power for the same P/D ratio is 2.6 times smaller with Pb and LBE and 2.1 times smaller with PbLi than with sodium. In order to achieve the same core power level with lead based coolant as with sodium coolant, it is necessary to increase the P/D ratio of those systems. When the P/D ratio is changed, the core dimensions – height and radius – are not changed, but the fuel rod dimensions are changed. By increasing the P/D ratio to 1.30 for the Pb and LBE coolant and to 1.24 for the PbLi coolant and decreasing the number of fuel rods per assembly to 91, it is theoretically possible to increase the maximum core power up to 3600 MWth for any of the lead-based cooled system. For the Pb and PbLi cooled cores operating at this power level, the cladding maximum temperature constraint is exceeded by 3 °C. However, no hot channel factor has been factored into this thermal-hydraulic analysis. When accounting for all the design uncertainties (flow distribution, physics modeling, material distribution...), it is expected that the maximum attainable power will be smaller than 3600 MWth. This also applies to the sodium cooled large B&B core but does not change the main result: by increasing the P/D ratios it is possible to achieve similar power levels with lead-based coolant than with sodium. With the increased P/D ratios, the HM loading is decreased by 45% for the Pb and LBE cooled cores and by 39.5% for the PbLi cooled core, compared to the sodium cooled core. For a same P/D ratio, it is observed that PbLi enables removing 28% more heat from the core than Pb and 26% more than LBE, enabling increasing the core power by the same fraction. This is due to the higher specific heat and lower dynamic viscosity of PbLi.

3.3. Neutronics considerations

Due to their high macroscopic scattering cross sections and low slowing-down power due to elastic scattering, lead and its alloys function as very good neutron reflectors. Lead also features a small capture cross section; for the same fuel and coolant volume fraction. HLM cooled reactors feature a harder neutron spectrum,

a lower neutron capture probability and a higher k_{eff} than sodium-cooled reactors. However, due to their inferior thermal-hydraulic characteristics that dictate higher P/D ratio designs, the neutron economy of HLM cooled cores is often comparable to that of sodium-cooled cores.

In addition to lead of natural isotopic composition, this study evaluates the performance of pure ^{208}Pb . The capture cross-section of ^{208}Pb is 2–3 orders of magnitude smaller than of the other Pb isotopes and bismuth. In addition, its inelastic scattering cross-section threshold occurs at a higher energy. The net result is that, relative to Pb cooled core, the neutron spectrum to be established in a ^{208}Pb -cooled core is expected to be harder and the neutron economy better. Although Pb highly enriched with ^{208}Pb is expected to be very expensive, it has been recently proposed for use in an accelerator-driven subcritical system, conventional fast reactors (Khorasanov et al., 2009) and in the TIT CANDLE reactor (Okawa and Sekimoto, in press). It is considered in this study for its academic interest.

3.4. Neutronics results

Table 5 compares the k_{eff} values and cycle average neutron leakage probabilities of a couple of LBE-cooled cores and of a natural lead-cooled and ^{208}Pb -cooled cores relative to those of the reference large sodium-cooled core. The LBE, Pb and ^{208}Pb cores use the fuel composition of the equilibrium sodium cooled reference core. It is found that, for the same P/D ratio, the LBE core leakage probability is almost 25% smaller (–1.1%) and the k_{eff} value is 2.8% larger than of the reference sodium-cooled core. This is because the LBE core spectrum is harder and yields a larger reproduction factor. In reality the amount of fissile material in the equilibrium LBE-cooled core will be smaller than in the sodium cooled core so that both cores BOEC k_{eff} value will be the same.

By increasing the LBE, Pb and ^{208}Pb cores P/D to 1.3, the value needed for attaining the reference B&B core power level, the total

Table 3

Thermal Hydraulic Performance of the LBE Cooled Large B&B Core with a Radial Power Peaking Factor of 1.7.

Parameters	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6
p/d ratio	1.1	1.2	1.3	1.3	1.3	1.3
Fuel rod pitch [mm]	9.06	9.06	9.06	11.48	13.24	15.64
Fuel rod diameter [mm]	8.24	7.55	6.97	8.83	10.19	12.03
Fuel rods per assembly	271	271	271	169	127	91
Clad thickness [mm]	0.667	0.611	0.564	0.715	0.824	0.974
Coolant volume fraction	25.0%	37.0%	46.3%	46.3%	46.3%	46.3%
Cladding volume fraction	22.3%	18.7%	16.0%	16.0%	16.0%	16.0%
Fuel volume fraction	39.5%	33.2%	28.3%	28.3%	28.3%	28.3%
LBE velocity [m/s]	2.33	3.05	3.76	4.07	4.25	4.44
Maximum power [MW _{th}]	1029	1991	3072	3325	3472	3627
Max. inner cladding temp. [°C]	584	589	594	605	615	631
Max. fuel temperature [°C]	608	634	664	758	847	991

Table 4
Thermal Hydraulic Performance of the PbLi Cooled Large B&B Core with a Radial Power Peaking Factor of 1.7.

Parameters	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6
p/d ratio	1.1	1.24	1.3	1.24	1.24	1.24
Fuel rod pitch [mm]	9.06	9.06	9.06	11.48	13.24	15.64
Fuel rod diameter [mm]	8.24	7.31	6.97	9.26	10.68	12.61
Fuel rods per assembly	271	271	271	169	127	91
Clad thickness [mm]	0.667	0.592	0.564	0.749	0.864	1.021
Coolant volume fraction	25.0%	41.0%	46.3%	41.0%	41.0%	41.0%
Cladding volume fraction	22.3%	17.6%	16.0%	17.6%	17.6%	17.6%
Fuel volume fraction	39.5%	31.1%	28.3%	31.1%	31.1%	31.1%
PbLi velocity [m/s]	2.4	3.43	3.88	3.77	3.96	4.18
Maximum power [MW _{th}]	1298	3038	3882	3339	3507	3702
Max. inner cladding temp. [°C]	591	603	609	618	632	653
Max. fuel temperature [°C]	614	657	683	739	820	954

neutron leakage probability of any of those cores increases to ~4.65%; slightly larger than the leakage probability of the reference sodium cooled core. The resulting k_{eff} values of those cores are significantly smaller than of the reference sodium-cooled core. The impaired neutron economy of the P/D = 1.3 cores is due to spectrum softening, enhanced leakage probability and enhanced parasitic neutron capture in the coolant associated with an increase in the coolant volume fraction (going from P/D of 1.112–1.3). Enriching Pb with 100% ^{208}Pb enables to achieve k_{eff} values ~2.5% larger than with natural Pb and 2.0% larger than with LBE.

4. Minimum required burnup

The minimum burnup required for sustaining the breed-and-burn mode of operation in the Pb, ^{208}Pb , LBE and PbLi cooled cores designed to give the nominal power is determined by performing a neutron balance analysis as described in (Heidet and Greenspan, in press). The k_{∞} evolution is deduced from a 0-D burnup analysis using the MOCUP (MCNP + ORIGEN) code and ENDF/B-VI cross sections. The k_{eff} value is estimated by multiplying the calculated k_{∞} by the neutron non-leakage probability (1.0–0.0466) and by the probability that the neutrons will not be captured in the excess reactivity control elements. The burnup reactivity swing is estimated to be 4.4% so that an average of 2.2% of the neutrons are assumed lost in the reactivity control systems. The average power density used for the depletion analysis is 112.5 W/cm³ and the volume fractions used are given in Table 6. The loaded fuel characteristics are the same as of the sodium-cooled reference B&B core. It is also assumed that 75% of the fission gases are continuously removed from the fuel and approximately every 10 years the fuel is instantaneously reconditioned and recycled with the melt-refining process (Hesson et al., 1963) without recycling loss. The evolution of k_{∞} with burnup for the Pb, ^{208}Pb , LBE and PbLi cooled large B&B cores is shown in Fig. 2 and the corresponding neutron balance is shown in Fig. 3. The lithium of the PbLi coolant is assumed, for all the neutronic analysis, to be depleted to 1% ^6Li . The neutron balance is expressed in terms of the net number of excess neutrons generated per unit fuel volume, defined as the total number of fission (also (n,2n)) neutrons generated per unit fuel volume minus the total number of neutrons absorbed in the same unit fuel volume,

integrated over the accumulated burnup, when starting with the fresh depleted uranium feed. It is mathematically computed from the expression $\int d(BU) \cdot \nu(BU)[1 - 1/k(BU)]$ in which the burnup (BU) is expressed in FIMA (Heidet and Greenspan, in press).

As already observed in Section 3b, it is found (Fig. 2) that the multiplication factor values of the ^{208}Pb cooled core are on the average 2.1% larger than for the LBE cooled core. In addition, the LBE, Pb and PbLi cooled cores k_{∞} evolution is very similar apart for the burnup at which the fuel reconditionings occur. Consequently, the neutron balance (Fig. 3) of those three cores is almost identical. The ^{208}Pb cooled core features the best neutron economy of all the cores cooled with lead and its alloys. The minimum burnup required to sustain the breed and burn mode of operation is 28.5% FIMA in the LBE cooled core, 29% FIMA in the Pb and PbLi cooled cores and only 22.5% FIMA for the ^{208}Pb cooled core; it is the burnup at which the net number of excess neutrons is zero (Fig. 3). The mass of ^{239}Pu per cubic meter and ^{239}Pu -to-HM weight fraction are provided in Table 7 when those cores reach the minimum burnup required to sustain the breed and burn mode.

Although the ^{239}Pu -to-HM weight fraction is 10.8% for both the PbLi and LBE cooled cores, the total mass of ^{239}Pu is different, primarily due to the difference in the P/D ratio. The ^{239}Pu -to-HM weight fraction of the Pb and ^{208}Pb cooled cores are slightly different from the LBE values because of the different minimum required burnups. At the same burnup, the ^{239}Pu -to-HM ratio value obtained for the ^{208}Pb -cooled core is the same as for the LBE cooled core. The minimum required burnup of the ^{208}Pb cooled core is smaller than for the other cores due to the harder neutron spectrum and smaller parasitic neutron capture in the coolant. Similarly, the spectrum of the Pb cooled core is slightly softer than that of the LBE cooled core because of the higher Pb density, resulting in a slightly higher minimum required burnup.

5. Comparison with sodium-cooled B&B core

The results obtained for the LBE cooled core having P/D = 1.3 are compared in Fig. 4 through 6 with the results obtained for the sodium cooled reference core for which P/D = 1.112 (Heidet and Greenspan, in press). For the sodium cooled system, the neutron

Table 5
 k_{eff} and Average Leakage Probabilities at Equilibrium for the Large Sodium Cooled B&B Core with P/D = 1.112 and for the Large LBE Cooled B&B Core with P/D = 1.112 and P/D = 1.3.

Coolant	Na	LBE	LBE	Natural	^{208}Pb
P/D	1.112	1.112	1.3	Pb 1.3	1.3
k_{eff} [BOEC/EOEC]	0.99934/1.04217	1.0288/1.07218	0.95496/1.0011	0.94935/0.9956	0.97585/1.0198
Axial leakage	3.73%	2.86%	3.80%	3.80%	3.82%
Radial leakage	0.67%	0.48%	0.86%	0.81%	0.86%
Total leakage	4.40%	3.33%	4.66%	4.61%	4.69%

Table 6
Unit Cell Fuel Volume Fraction – Inter-Duct Gap and Duct Wall are not Accounted for.

P/D	1.24	1.30
Fuel (U-Zr10)	31.1%	28.0%
Gap (empty)	10.4%	9.3%
Structural material (HT9)	17.6%	16.4%
Coolant (HLM)	41.0%	46.3%

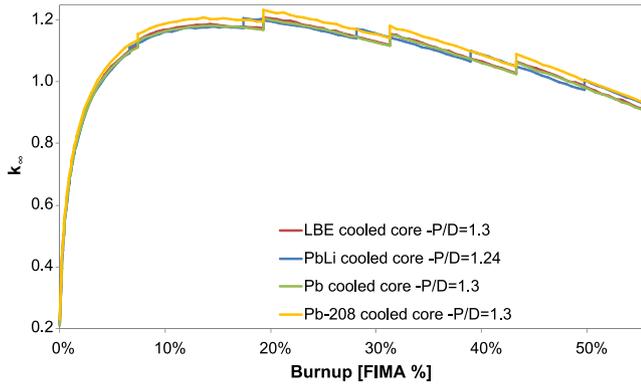


Fig. 2. k_{∞} evolution for the Pb, ^{208}Pb and LBE cooled cores with P/D = 1.30 and the PbLi cooled core with P/D = 1.24.

leakage probably was found to be 4.40% and the fraction of neutrons absorbed in the control elements was estimated to be 2.2%. The minimum required burnup of the reference sodium cooled core was found $\sim 21\%$ FIMA (Fig. 4) – significantly lower than of the LBE cooled core (29%). This larger burnup is due, primarily, to the lower k_{∞} , by approximately 2%, of the LBE core relative to that of the sodium cooled core (Fig. 4). The slightly larger leakage probability for the LBE cooled core (4.66% vs. 4.40%) also contributes to this trend. Fig. 5 shows that the number of neutrons that need to be absorbed in the depleted uranium to make the unit cell k_{∞} equals unity – corresponding to the minimum in the neutron balance plot, is larger for the LBE than for the sodium cooled core. The burnup at which $k_{\infty} = 1$ occurs is 3.83% FIMA in the LBE and 3.46% FIMA in the sodium cooled cores. Fig. 5 also shows that the number of excess neutrons that can be generated by the fuel of the LBE cooled core after reaching the minimum required burnup – $\sim 4\text{E} + 20 \text{ n/cm}^3$ – corresponding to the maximum in the neutron balance plot of Fig. 5, is not sufficient to

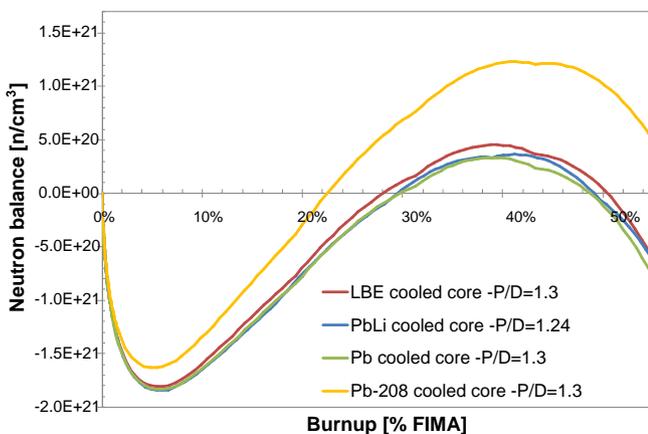


Fig. 3. Neutron balance for the Pb, ^{208}Pb and LBE cooled cores with P/D = 1.30 and the PbLi cooled core with P/D = 1.24.

Table 7
Mass and Weight Fraction of ^{239}Pu in the B&B Cores at the Minimum Required Burnup.

Coolant	Burnup [% FIMA]	Mass ^{239}Pu [kg]	^{239}Pu to HM wt%
LBE	28.5%	299	10.80%
PbLi	29%	335.5	10.80%
Pb	29%	305.6	10.93%
^{208}Pb	22.5%	318.2	10.30%

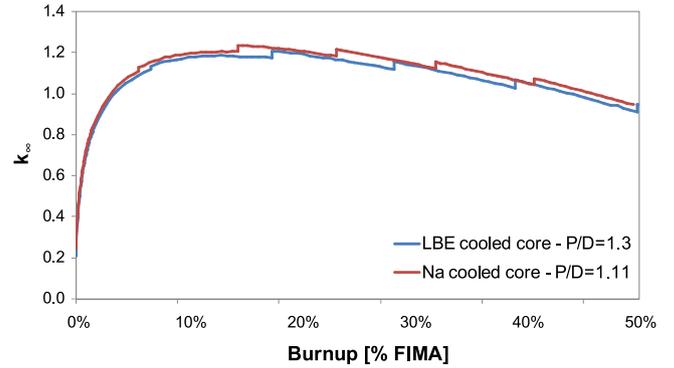


Fig. 4. k_{∞} evolution for the LBE cooled core with P/D = 1.3 and sodium cooled core with P/D = 1.112.

establish the breed-and-burn mode in a new core that requires $\sim 1.8\text{E} + 21 \text{ n/cm}^3$. If the fraction of neutrons lost by leakage and in the reactivity control systems can be decreased below 4%, it will be possible to use the fuel discharged from the LBE core at the minimum required burnup for starting a new LBE cooled B&B core. When operating the core at its maximum achievable burnup, the neutron leakage probability is slightly increased and, based on Fig. 4, the maximum burnup that the fuel can accumulate is estimated to $\sim 45\%$ FIMA in the large LBE cooled core and to $\sim 55\%$ FIMA in the large sodium cooled core (Heidet and Greenspan, in press); at this burnup the net number of excess neutrons gets back to zero implying that it will not be possible to maintain criticality at higher burnups. The neutron leakage probability is increased to 5.4% and 6.0% for respectively the LBE cooled and sodium cooled cores.

The evolution of the plutonium mass for the two cores studied is compared in Fig. 6. When the unit cell becomes critical ($k_{\infty} = 1.0$; corresponding to the minima in Fig. 5) the mass of fissile

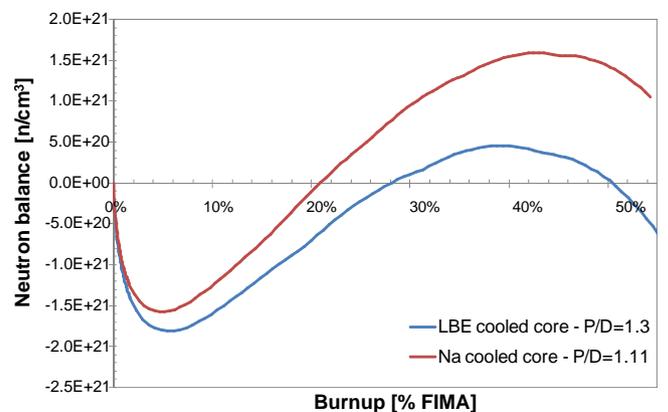


Fig. 5. Neutron balance for the LBE cooled core with P/D = 1.30 (4.66% leakage) and for the sodium cooled core with P/D = 1.112 (4.4% leakage).

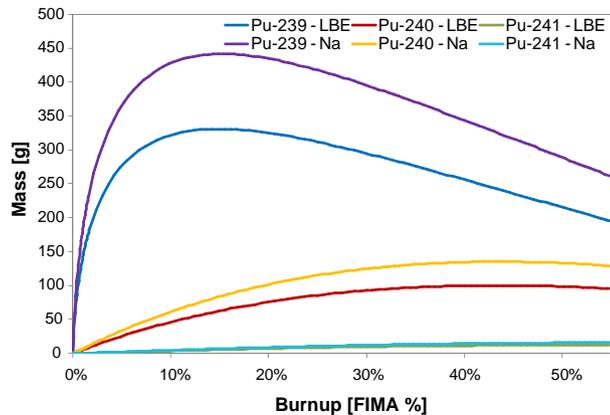


Fig. 6. Plutonium isotopes mass evolution, per m^3 , for the LBE and sodium cooled cores for a unit cell.

plutonium per cubic meter of core is 323 kg for the reference sodium cooled core and only 238 kg for the LBE cooled core. This is due to the lower HM inventory in the LBE core that features a larger P/D ratio; the fissile plutonium weight fraction is nearly the same – 6.25%, for both cores. Due to its smaller HM loading, the initial mass of fissile material required for establishing the initial criticality is $\sim 26\%$ smaller for the LBE cooled core than for the sodium cooled core.

6. Discussion and conclusions

In order to get from the lead-alloy-cooled cores the power level of the reference sodium cooled core, the P/D ratio needs to be increased from the reference 1.112 value to 1.24 for PbLi in which the Li is depleted to 1% ^6Li and 1.30 for lead and LBE. The increase in the coolant volume fraction impairs the neutron economy. As a consequence, the minimum burnup required for sustaining the B&B mode of operation in the LBE and PbLi cooled B&B cores is larger than in the reference sodium-cooled core – $\sim 29\%$ FIMA versus 21% FIMA. If the Pb was to be enriched to nearly pure ^{208}Pb , the minimum required burnup would be 22.5% FIMA; very close to that required for a Na-cooled core. The maximum possible accumulated burnup in the LBE-cooled B&B core is $\sim 45\%$ FIMA – smaller than the 55% FIMA possible to achieve in the reference sodium cooled core (Heidet and Greenspan, 2010).

Potential advantages of the HLM-cooled reactors is that they do not need a secondary coolant loop and require 26% smaller mass of fissile material to establish initial criticality. Also, as the boiling temperature of lead and its alloys is close to 800°C higher than that of sodium, the probability of coolant boiling and coolant voiding accidents is much smaller in the HLM reactors.

The major challenge faced by the HLM-cooled B&B cores is the successful development of an erosion- and corrosion-resistant cladding material that could withstand coolant velocities higher than 2 m/s at temperatures above 550°C , typically assumed as constraints for HT-9 made claddings. Another challenge is a significant reduction of the fuel-cladding eutectic temperature in fuel with high content of fission products. So far the ferritic-martensitic steel under development at MIT (Short et al., 2010) shows good compatibility with LBE up to a temperature of 650°C , for 50 years. However, it also needs to demonstrate good irradiation performance, acceptable chemical interactions with high burnup fuel and compatibility with high coolant velocities.

Acknowledgement

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A search for minimum volume of Breed and Burn cores

Christian Di Sanzo¹ and Ehud Greenspan¹

¹*Department of Nuclear Engineering, University of California, Berkeley
Etcheverry Hall, MC 1730, Berkeley, CA, 94720
Tel. +1-510-643-9983, Fax. +510-643-9685, e-mail: cdisanzo@berkeley.edu*

Abstract – *The objective of the present study is to quantify the minimum volume a Breed and Burn (B&B) core can be designed to have and the corresponding burnup required for sustaining the breed-and-burn mode of operation based on neutronics; radiation damage constraints are ignored. The minimum radius for an idealized spherical B&B reactor is 136 cm or 110 cm for, respectively, 40% or 28% coolant volume fraction. The peak required burnup is about 25%. The minimum volume of a more realistic cylindrical B&B core is estimated to be only ~15% larger than that of the idealized spherical core but is only 43% of the volume of the medium-size B&B core previously designed to fit within the S-PRISM reactor vessel. Thus it appears that SMR's can, in principle, be designed to have a B&B core. It was also found that the minimum volume B&B core does not necessarily coincide with the maximum permissible leakage from a core that can sustain the B&B mode of operation.*

I. INTRODUCTION

The Breed and Burn (B&B) reactors addressed in this study are sodium-cooled fast reactors that, once initial criticality is established, can sustain criticality “indefinitely” when fueled with depleted uranium only while operating in a once-through fuel cycle. In a B&B mode of operation depleted uranium is first converted into plutonium part of which is fissioned in situ. Background information about B&B reactors, also referred to as Travelling Wave Reactors (TWR), can be found in References 1 through 3 and references thereof¹⁻³.

Previous studies^{4,5} quantified the value of the minimum burnup that is required for sustaining the breed-and-burn mode of operation and the sensitivity of the minimum burnup to a number of design variables such as fuel type, fuel volume fraction and neutron leakage probability. The practical minimum required average discharge burnup using metallic fuel and very-low leakage core was found^{4,5} to be in the vicinity of 20% FIMA (Fissions per Initial Metal Atom). In order to achieve a low neutron leakage probability the B&B core is typically designed to be of a significantly larger height and volume than of a conventional sodium-cooled fast reactor core. These large volume-low leakage cores feature relatively large positive coolant temperature and void reactivity coefficients that make it challenging to design to be passively safe. Despite of the large core volume, the

minimum required burnup is more than double the burnup IFR-type metallic fuel rods have been irradiated to.

The objective of the present study is to quantify the minimum volume a B&B core can be designed to have and the corresponding burnup required for sustaining the breed-and-burn mode of operation. The smaller and leakier the core is, the easier it will be to design it passively safe. The identification of the minimum volume core will also clarify whether or not a B&B reactor can be designed to fit within the category of Small Modular Reactor (SMR) the interest in which is recently increasing.

The present study ignores radiation damage constraints; it is set to quantify the minimum core volume that can support a B&B mode of operation based on, primarily, neutron balance considerations. It assumes that either improved cladding materials and fuel designs that are able to maintain the fuel integrity up to the minimum burnup required for maintaining the B&B mode of operation will be developed, or that fuel reconditioning that does not involve separation of actinides could be applied^{3,5}.

Three approaches are pursued in this preliminary study: (1) A unit-cell neutron balance analysis to determine related numerical values of leakage and burn-up (Section II); (2) An idealized one-dimensional spherical core made of a bed of HT-9 clad metallic fuel pebbles that are circulated from the outer periphery inwardly (Section III); and (3) A more realistic multi-batch cylindrical core made

of hexagonal fuel assemblies having a similar composition to that of the idealized spherical core pebbles (Section IV).

II. UNIT CELL BALANCE

A neutron balance analysis that is based on the k_{∞} evolution with burnup of a representative unit cell initially loaded with depleted uranium can provide an indication on the burn-up that is needed for sustainability of B&B mode of operation as a function of the assumed neutron leakage probability. The present analysis uses the necessary condition for sustaining a breed-and-burn mode of operation that was derived in references 3 to 5:

$$\int v(BU) \left\{ 1 - \frac{1}{k(BU)} \right\} d(BU) = 0 \quad (1)$$

with:

$$k(BU) = k_{\infty}(BU) * (1 - P_L) * (1 - P_{RC}) \quad (2)$$

where P_L and P_{RC} are the fraction of the fission-born neutrons that is lost in, respectively, leakage out from the core and capture in the reactivity control systems; v is the number of neutrons generated per fission; and BU is the burn-up expressed in FIMA. The integration is to be performed from zero up to the burnup value for which the value of the integral is zero. The corresponding upper limit of integration is the minimum burnup for which a breed-and-burn mode of operation can be established. If the k_{∞} evolution with burn-up is known, expression (1) can be used to derive the maximum value of $(1 - P_L) * (1 - P_{RC})$. If the value of P_{RC} is known or can be estimated, expression (1) can yield an estimate of the maximum value of P_L a B&B core can have.

The k_{∞} evolution with burn-up is calculated for a unit cell that is representative of the B&B reactor core. Although the neutron spectrum the fuel experiences in a B&B core somewhat differs from the unit cell spectrum – it tends to be softer at low burnups and harder at high burnups. This difference was found not to significantly affect the conclusions derived from the neutron balance analysis⁴⁻⁶.

Two unit cell compositions of different coolant volume fraction are examined; they are reported in Table I. Following the common practice in fast reactor core simulations, the unit-cell constituents are homogenized. The coolant volume fraction of 40% assumed for one unit-cell corresponds to a pebble-bed core while the coolant volume fraction of 28% corresponds to a tight pitch hexagonal array of fuel rods⁷. The geometrical parameters from which the volume fractions are derived are also reported in Table I. The initial fuel composition is an alloy of depleted uranium (0.2% ²³⁵U) and zirconium (6% by weight) with a nominal density of 17.12 g/cc and a smear

density of 75% to accommodate fuel swelling. The cladding is made of HT-9.

The unit cell calculations were performed using the MCNP5 1.51 neutron transport code and the ORIGEN 2.2 burn-up codes, coupled with MOCUP 2.0. For neutron cross sections, ENDF/B-VII library is used, assuming a core temperature of 900 K. The burnup calculations assume operation at a constant power density of 112.5 W/cm³ which is the average power density of our reference B&B core design⁷.

TABLE I.
Composition of the Two Unit Cells Examined and the B&B System Geometry they Represent.

Volume fractions			Pebble bed		Hexagonal assembly		
sodium	cladding	fuel & gap	pebble diam. (mm)	clad thickness (mm)	P/D	rod diam. (mm)	clad thickness (mm)
40%	18.8%	41.2%	34 (40% coolant VF)	2	1.23	10	0.855
28%	22%	50%	34 and smaller (28% coolant VF)	2	1.12	10	0.83

The k_{∞} evolution of the two unit cells is plotted in Fig.1 and the corresponding neutron balance analysis – a plot of the integral of the left hand side of Eq. (1) as a function of burnup, is displayed in Fig. 2 and 3.

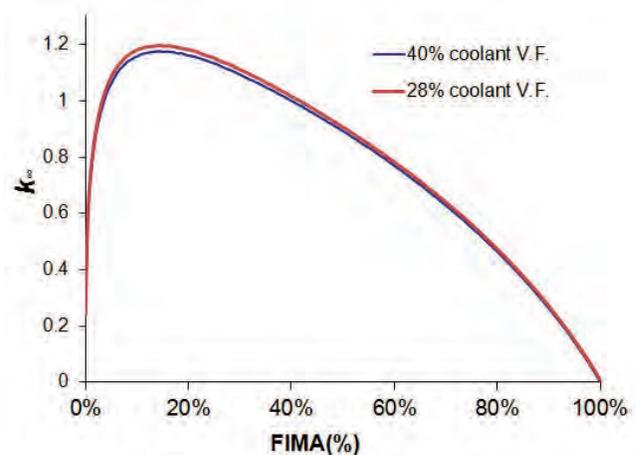


Fig. 1. k_{∞} evolution of unit cells for 40% sodium volume fraction and 28% sodium volume fraction. Initial fuel is depleted uranium alloyed with 6% Zr.

The neutron balance expresses the net number of neutrons absorbed, per unit burnup, per cubic centimeter of fuel volume⁵. When the slope is negative, the system is a net neutron consumer; i.e., its k_{∞} is smaller than unity. When

k_{∞} is >1 , the system is a net neutron producer and the slope of the neutron balance evolution curve is positive. The minimum burnup required for establishing a B&B mode of operation is that burnup for which the cumulative neutron balance is zero; that is, the burnup at which the neutron balance evolution curve crosses the zero line for the first time. The smaller is the neutron leakage probability, the smaller becomes the minimum required burnup. The scenario in which the neutron balance evolution curve is tangent to, but does not cross the zero line represents the condition of maximum neutron leakage probability that can be tolerated; a B&B mode of operation could not be established at a higher leakage probability. The specific values of the neutron leakage probability with the corresponding minimum required burnup for the two unit cell systems examined are reported in Table II.

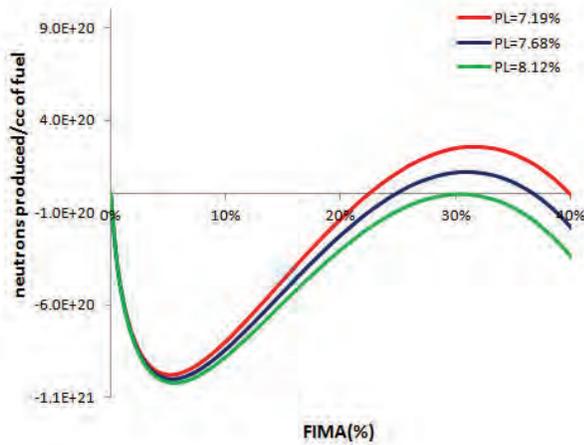


Fig. 2. Neutron balance curves for different leakage probabilities for 40% coolant volume fraction.

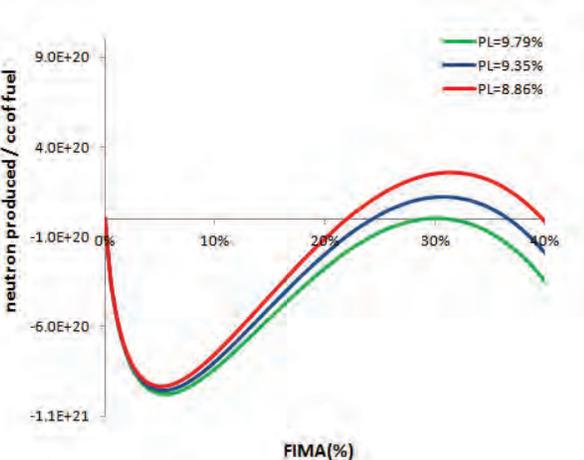


Fig. 3. Neutron balance curves for different leakage probabilities for 28% coolant volume fraction.

TABLE II
Sets of Probability and Burnup for 40% and 28% Coolant Volume Fractions, Derived from Figure 2 and 3 that are based on the Unit Cell Analysis.

Coolant V. F.	P_{CR}	P_L	Burnup (FIMA)
40%	0%	7.19%	22.5%
40%	0%	7.68%	25%
40%	0%	8.12%	30%
28%	0%	8.86%	22.5%
28%	0%	9.35%	25%
28%	0%	9.79%	30%

The maximum tolerable neutron leakage probability is higher for the unit cell featuring a smaller coolant volume fraction. This is consistent with prior observations⁴⁻⁶ that the harder is the neutron spectrum, the better becomes the neutron economy.

III. SPHERICAL CORES

Since a sphere is the geometrical shape that offers the lowest possible neutron leakage probability per unit core volume, an idealized spherical core is first analyzed to establish a lower bound on the minimum possible B&B core volume. The core is assumed to consist of a bed of fuel pebbles that are charged at the outermost core region and are gradually moved radially inward toward a central discharge duct (not simulated). Pebbles are discharged from the core center after accumulating the minimum required burnup. Although the spherical pebble core model considered cannot be realized in practice as simulated, it is analyzed to provide a bounding estimate. A schematic of such a spherical core is provided in Fig. 4.

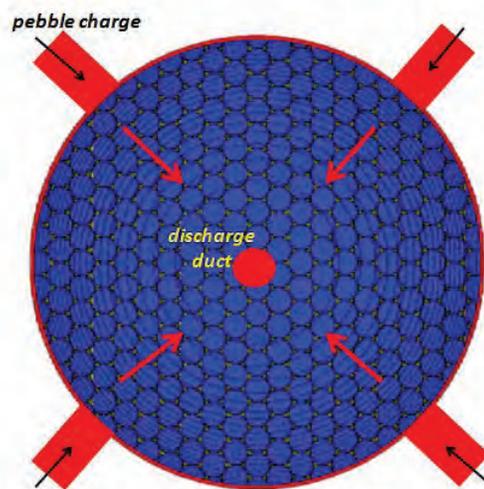


Fig. 4. Schematic of a Spherical Pebble Reactor. Pebbles are not in scale. The Reactor Reflector is not shown.

The reactor is divided for depletion calculation into 20 equal volume batches (shown as circles in Fig. 4). The power is constrained not to exceed a peak power density of 450 W/cm³; it occurs at the core center. A 1 meter thick reflector made of 50% sodium-50% HT-9 (by volume) surrounds the core. The feed fuel is depleted uranium alloyed with 6% (by weight) Zr and having a smear density of 75%.

The search for the minimum core volume was iterative; starting with an initial volume guess, burnup analysis was performed for over 20 cycles until an equilibrium core composition was established. The cycle length is initially chosen so that the equilibrium discharged fuel burnup will be ~30% FIMA, corresponding to the maximum tolerable leakage deduced from the unit cell analysis of Section II. If the minimum k_{eff} value of the equilibrium cycle exceeds 1.0, the core radius is reduced by 5 cm and the above described computational process is repeated. Further simulations to refine the core design are performed when the radius interval of interest is identified.

Figure 5 shows the reactor k_{eff} evolution during two consecutive equilibrium cycles for three cores having 40% coolant volume fraction. Statistical scattering is present with $\sigma \sim 0.00040$ for all simulations, due to the stochastic method used. Reactor parameters are summarized in Table III. The burnup reactivity swing is less than 0.5%. Had the pebbles been continuously circulated, the burnup reactivity swing would have been close to zero.

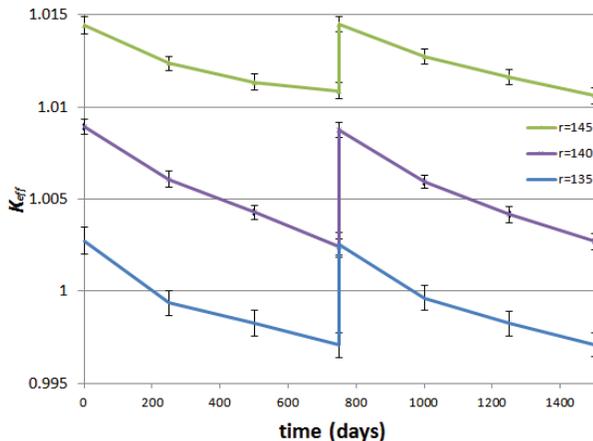


Fig. 5. Equilibrium cycle k_{eff} evolution for a spherical core of different dimensions; 40% coolant volume fraction and 20 fuel batches.

TABLE III

Selected Parameters for Three Spherical Cores with 40% Coolant Volume Fraction.

Core radius (cm)	Power (MW)	Cycle length (days)	BOEC k_{eff}	EOEC k_{eff}	EOEC P_L	Discharge burnup (FIMA)
145	1200	750	1.01443	1.01088	6.37%	28.8%
140	1090	750	1.00891	1.00240	7.00%	29.0%
135	995	750	1.00273	0.99707	7.73%	29.0%

The minimum critical volume deduced from Fig. 5 is between 135 cm and 140 cm. The corresponding discharge burnup is ~30% FIMA. To refine the minimum volume a further search is performed varying the burnup value, with the goal of identifying a cycle with EOEC (End of Equilibrium Cycle) $k_{eff} = 1$. Fig. 6 shows equilibrium cycle k_{eff} evolution for a 138 cm core for different values of discharge burnup. The EOEC k_{eff} values are reported in Fig. 7 with the corresponding leakage probabilities. It is observed that the trend of k_{eff} evolution depends on the discharge burnup. For low burnup (22%) the k_{eff} increases during a cycle, while it decreases at higher burnup (30%). For intermediate values that k_{eff} evolution has a parabolic type profile. This change in trend is due to the k_{∞} evolution with burnup (Fig. 1).

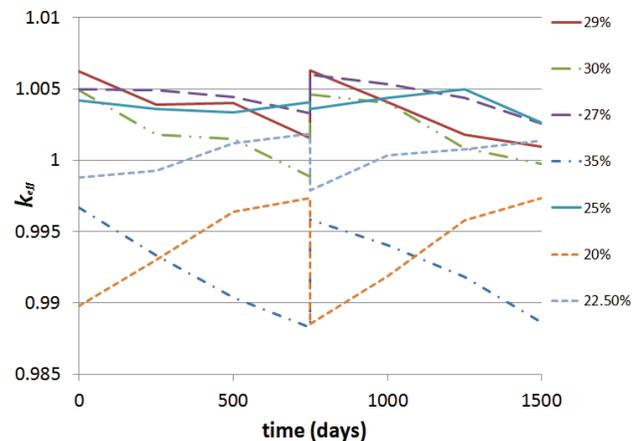


Fig. 6. Equilibrium cycle k_{eff} evolution for a spherical core that is 138 cm in radius for discharge burnup from 22.5% to 29%; 40% coolant volume fraction and 20 fuel batches.

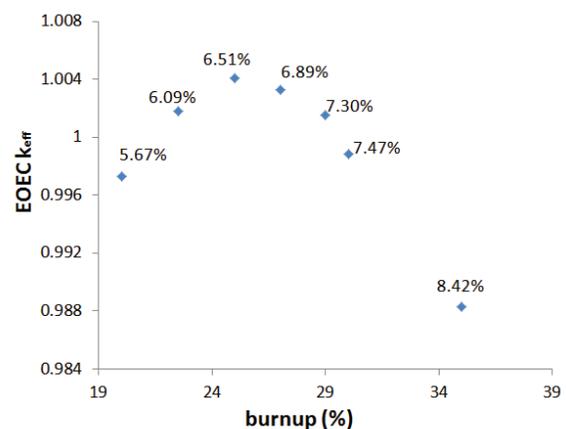


Fig. 7. EOEC k_{eff} values as a function of discharge burnup in the 138 cm core. Neutron leakage probability values are reported next to each point.

Fig. 8 shows how the discharge burnup affects the power density profile in the spherical reactor. The power

shape flattens and, correspondingly, the radial neutron leakage probability increases, as the discharge burnup is increased. The power flattening occurs since the central batch produces less power at higher burnup because its k_{∞} goes down with burnup.

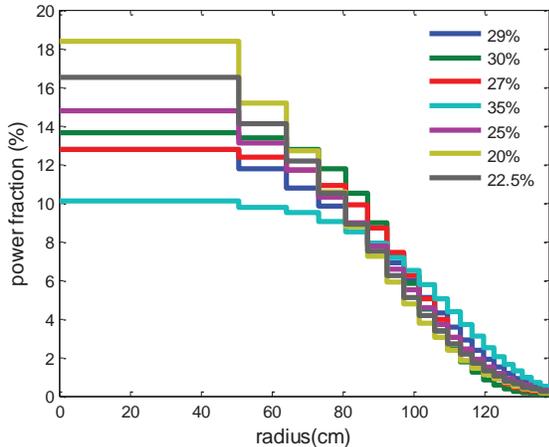


Fig. 8. EOE Power distribution along the 138 cm reactor for different discharge burnup from 22.5% to 29%.

The results of Figs. 5 and 6 imply that the maximum tolerable leakage is 7.30% corresponding to 29% burnup. This leakage value is smaller than the value inferred from the unit cell balance (Sec. II) for 29% burnup. Moreover, Fig. 7 shows that the k_{eff} of the core designed to discharge its fuel at 25% FIMA is highest implying that at this discharge burnup it is possible to design a B&B core of the smallest possible volume.

The minimum volume core was found to have a radius of 136 cm and a discharge burnup of 25%, as shown in Fig. 9 and Table IV. The corresponding leakage probability is 6.82% that is smaller than the 7.68% predicted from the unit cell analysis and smaller than the 7.30% for the 138 cm core. It is concluded that our initial assertion that the minimum volume core corresponds to the maximum tolerable neutron leakage probability is not correct. This is due to the fact that both the core k_{eff} and the neutron leakage probability are a function of the discharge burnup and in-core fuel management; these effects are not accounted for in the zero-dimensional model of Sec. II.

Although larger than the ~20% FIMA minimum required average burnup for a large cylindrical B&B core⁷, the resulting burnup of 25% is smaller than the peak discharge burnup of the large cylindrical core; the peak-to-average discharge burnup is approximately proportional to the peak-to-average axial power density which is close to 1.5 for the cylindrical core made of fuel assemblies. Featuring on-line refueling and continuous circulation, every pebble can be discharged at very closely the same burnup.

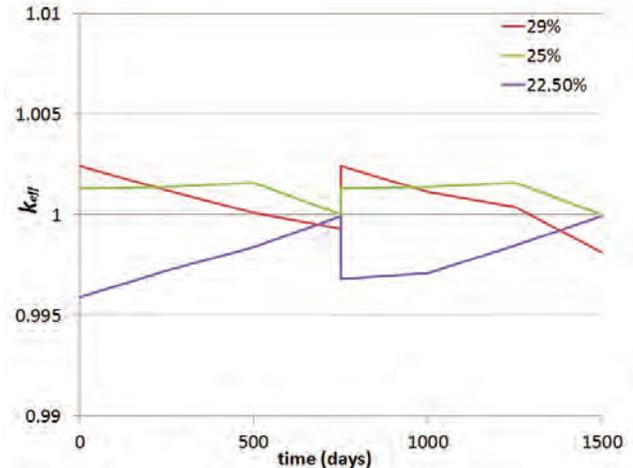


Fig. 9. Equilibrium cycle k_{eff} evolution for a spherical core that is 136 cm in radius for discharge burnup from 22.5% to 29%; 40% coolant volume fraction and 20 fuel batches.

TABLE IV
Selected Parameters for Three 136 cm Spherical Cores with 40% Coolant Volume Fraction.

discharge burnup	BOEC k_{eff}	EOEC k_{eff}	EOEC P_L
29%	1.00244	0.99932	7.56%
25%	1.00131	1.00002	6.82%
22.5%	0.99591	0.99995	6.39%

The analysis was repeated for a core in which the coolant volume fraction is 28%. Although no pebble bed core made of same diameter pebbles can have a coolant volume fraction that is smaller than 40%, a core made of pebbles of a number of different diameters can. The minimum volume reactor having a 28% coolant volume fraction is found to be 110 cm with 25% discharge burnup as shown in Fig. 10. This core volume is approximately half of the minimum volume core having a 40% coolant volume fraction.

Selected reactor characteristics are reported in Table V; resulting neutron leakage probability is close to 8.74% as compared with the 9.35% predicted by the infinite medium analysis (Table II).

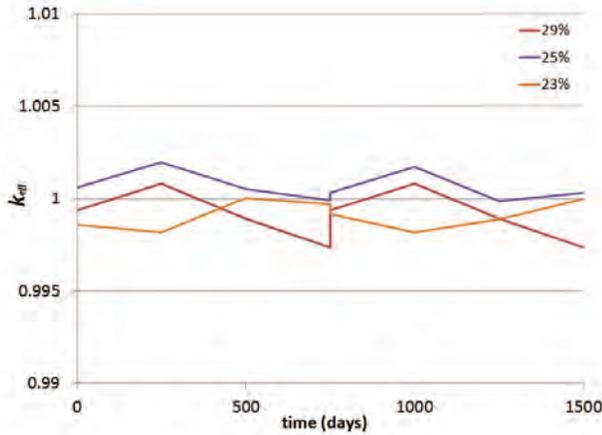


Figure 10. Equilibrium cycle k_{eff} evolution for a spherical core that is 110 cm in radius for discharge burnup from 23% to 29%; 28% coolant volume fraction and 20 fuel batches.

TABLE V.

Selected Parameters for Three 110 cm Spherical Cores with 28% Coolant Volume Fraction

discharge burnup	BOEC k_{eff}	EOEC k_{eff}	EOEC P_L
29%	0.99935	0.99736	9.44%
25%	1.00059	1.00001	8.74%
23%	0.99857	0.99971	8.41%

IV. CYLINDRICAL CORES

The cylindrical core design considered uses hexagonal fuel assemblies made of HT-9 clad metallic fuel rods. Except for the core radius and number of fuel batches, the core is similar to our reference B&B core design⁷. The active core height is fixed at 209 cm whereas the core radius is a design variable. A 2 m high plenum extends on top of the fuel and a 1-m thick reflector made of 50% sodium and 50% HT-9 (by volume) surrounds the core. Depleted uranium feed fuel is loaded at the outermost radial batch of the core and is gradually shuffled towards the central radial batch.

The core is assumed to be made of 20 equal volume fuel batches, in order to be consistent with the spherical core analysis (Section III). The batches are represented as cylindrical shells so as to simplify the analysis; the cylindrical approximation of the core for the purpose of burnup analysis was proven to be reasonable⁷. A schematic layout of the core is given in Figure 11.

Figure 12 shows that, at a first approximation, the minimum radius for a cylindrical core that has 40% coolant volume fraction is below 139 cm; interpolation leads to an approximate value of 137 cm. This is only 17% larger than the corresponding minimum volume spherical core identified in Section III that is 136 cm in radius. Table VI

gives selected characteristics of cylindrical B&B cores with 40% coolant volume fraction.

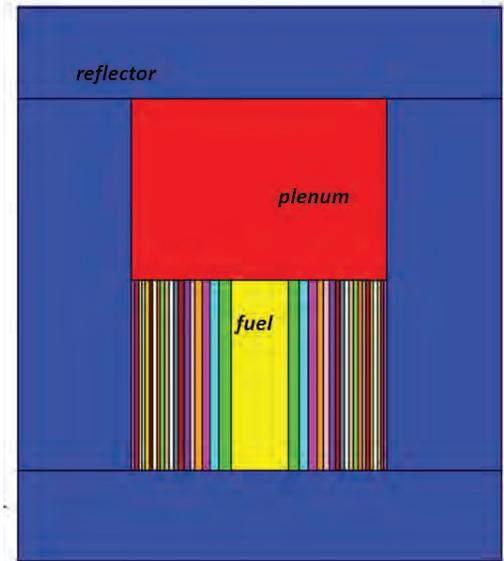


Figure 11. Schematic layout of the cylindrical reactor. Reflector is shown in blue. Fuel batches are shown in different colors.

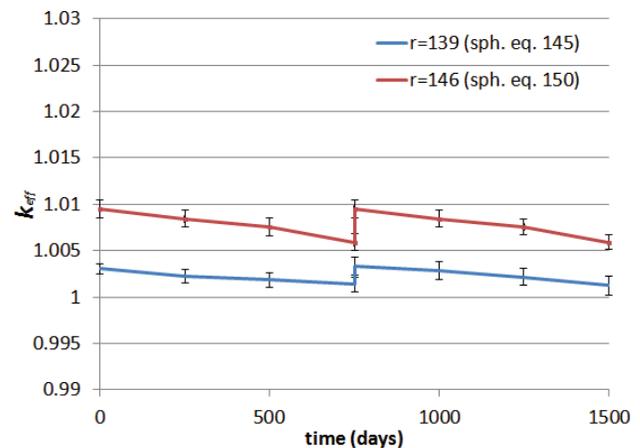


Figure 12. Equilibrium cycle k_{eff} evolution for cylindrical cores of different outer radius; 40% coolant volume fraction and 20 batches.

Figure 13 and Table VII give the corresponding results for cylindrical cores the coolant volume fraction of which is 28%. The minimum cylindrical core radius is found to be approximately 99 cm. The volume of this cylindrical core is only about 15% larger than the minimum volume of the corresponding idealized spherical core (110 cm in radius). It is close to half the volume of the 40% coolant volume fraction core and is only 43% of the volume of the medium-size B&B core previously designed to fit within the S-PRISM reactor vessel⁸. Thus, it appears that SMR's can, in principle, be designed to have a B&B core.

TABLE VI.
Selected Parameters of Cylindrical Cores with 40% Coolant Volume Fraction.

H (cm)	R (cm)	R sphere equivalent (cm)	Power (MW)	Cycle length (days)	BOEC k_{eff}	EOEC k_{eff}	P_L (RADIAL)	P_L (AXIAL)	P_L (TOTAL)	Discharge burnup (FIMA)
209	146	150	1330	750	1.00950	1.00591	2.22%	4.15%	6.37%	29.0%
209	139	145	1200	750	1.00306	1.00138	2.79%	4.16%	6.95%	29.0%

TABLE VII.
Selected Parameters for Cylindrical Cores with 28% Coolant Volume Fraction.

H (cm)	R (cm)	R sphere equivalent (cm)	Power (MW)	Cycle length (days)	BOEC k_{eff}	EOEC k_{eff}	P_L (RADIAL)	P_L (AXIAL)	P_L (TOTAL)	Discharge burnup (FIMA)
209	111	125	780	900	1.0172	1.0148	3.53%	3.38%	6.91%	29.0%
209	105	120	690	900	1.00889	1.0080	4.91%	3.41%	8.32%	29.0%
209	98	115	630	900	0.9993	0.9988	5.84%	3.45%	9.30%	29.0%

A more detailed analysis needs to be performed for the cylindrical cores in order to come up with a more refined minimum radius value.

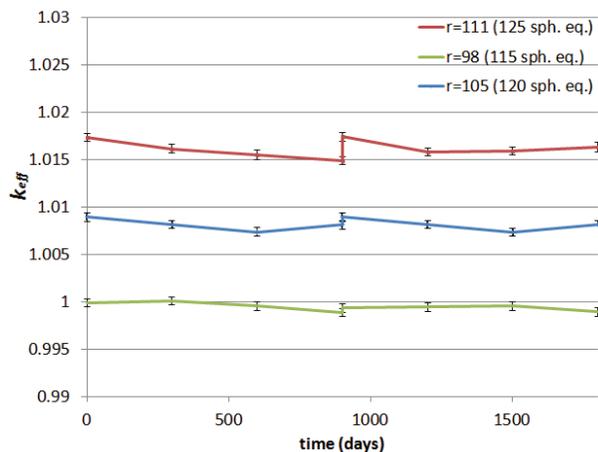


Fig. 13. Equilibrium cycle k_{eff} evolution for cylindrical cores of different outer radius; 28% coolant volume fraction and 20 batches.

IV. CONCLUSIONS

The minimum radius for an idealized spherical B&B reactor is 136 cm or 110 cm for, respectively, 40% or 28% coolant volume fraction. The peak required burnup is about 25%. The minimum volume of a more realistic cylindrical B&B core is estimated to be only ~15% larger than that of the idealized spherical core but is only 43% of the volume of the medium-size B&B core previously designed to fit within the S-PRISM reactor vessel⁸. Thus it appears that SMR's can, in principle, be designed to have a B&B core. It

was also found that the minimum volume B&B core does not necessarily coincide with the maximum permissible leakage from a core that can sustain the B&B mode of operation.

ACKNOWLEDGMENTS

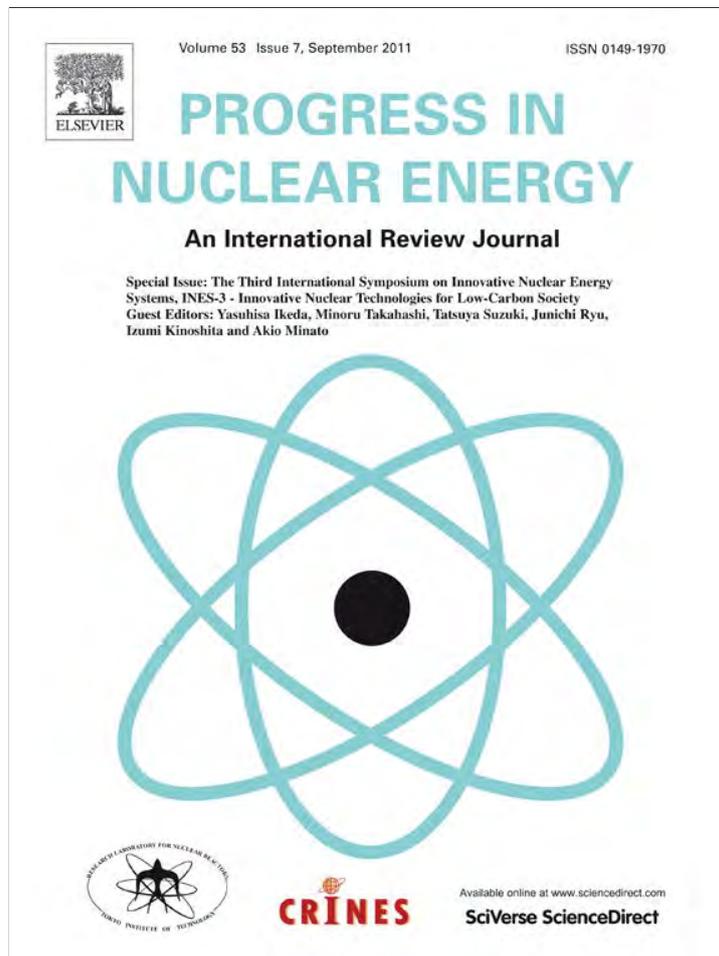
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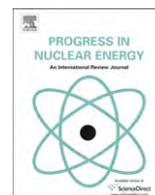
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Energy sustainability and economic stability with Breed and Burn reactors

Ehud Greenspan*, Florent Heidet

Department of Nuclear Engineering, University of California, Berkeley, CA 94720-1730, USA

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ABSTRACT

The purpose of the present study is to evaluate the impact a successful development of Breed and Burn (B&B) fast reactors and their fuel reconditioning technologies could have on the uranium ore utilization, uranium enrichment capacity, nuclear waste and energy security. It is found that a successful development of B&B reactors will offer 40-folds increase in the uranium ore utilization versus that presently achieved. A successful development of a fuel reconditioning technology could increase the attainable uranium utilization to 100-folds its present value. The growth rate of the installed capacity of B&B reactors possible to achieve using the “spawning” mode of operation is estimated to be nearly 4% per year. The amount of natural uranium required for starting a fleet of B&B reactors that will reach an electricity generation capacity of 1000 GWe by the end of this century is estimated to be the equivalent of 10 years of supply to the presently operating commercial fleet of LWRs in the US (86 GWe). No natural uranium and no enrichment capacity will be required to support this fleet beyond the later part of this century. The energy value of the depleted uranium stockpiles (“waste”) that will be accumulated in the US by that time is equivalent to, when used in the B&B reactors, up to 20 centuries of the total 2010 supply of electricity in the USA. It is therefore concluded that a successful development of B&B reactors and associated fuel reconditioning could provide a great measure of energy security, proliferation resistance and cost stability.

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1. Introduction

Present day commercial nuclear power reactors, mostly Light Water Reactors (LWRs), utilize less than one percent of the natural uranium feed: the uranium enrichment level presently preferred by the industry is approximately 4.5% ^{235}U . As the natural uranium mined contains only 0.72% of ^{235}U , it takes 8–10 tons of natural uranium to make 1 ton of 4.5% enriched uranium. The remaining 7–9 tons of depleted uranium, typically containing 0.2%–0.3% ^{235}U , is discarded as a waste. Of the enriched uranium that is loaded into the core, only about 5% is actually fissioned, making the overall uranium utilization only $\sim 1/9$ of 5% or, approximately, 0.6%.

The amount of natural uranium that has been mined so far for fueling the fleet of commercial LWRs that presently generates close to 20% of the US electricity consumption is approximately 600,000 tons. Out of these, more than 60,000 tons ended up as used nuclear fuel – the enriched uranium fuel that was fed into the LWRs and discharged after few percents of the uranium has been fissioned.

More than 500,000 tons ended up as depleted uranium “waste”. Additional depleted uranium has been accumulated from the military programs.

By using fast breeder reactors it is possible, in principle, to fission close to 100% of the depleted uranium “waste”. However, this high uranium utilization cannot be achieved in a single irradiation campaign because radiation damage effects constrain the burnup level the fuel can withstand to the order of 10%–15% FIMA (Fissions per Initial heavy Metal Atom), depending on the core neutron spectrum. Consequently, attainment of high uranium utilization also necessitates multiple fuel recyclings. Traditionally, fuel recycling is to include removal of the fuel cladding, removal of most of the fission products, addition of some depleted uranium makeup fuel, fabrication of new fuel elements and reloading them into the reactor core for another irradiation cycle. Although technically feasible, there is a significant objection in the US to fuel reprocessing due to economic viability and proliferation concerns.

Fast breeder reactors could, in principle, also operate without fuel recycling; that is, using a once-through fuel cycle as do all of the LWRs presently operating in the USA. Although a discharge burnup of 10%–15% FIMA is 2–3 times higher than that of contemporary LWRs, the uranium utilization from a once-through

* Corresponding author. Tel.: +1 510 643 9983; fax: +1 510 643 9685.
E-mail address: gehud@nuc.berkeley.edu (E. Greenspan).

FBR is not significantly different from that of a once-through LWR because the uranium enrichment required to fuel the FBR is more than twice that required to fuel the LWR.

Nevertheless, it may be possible to realize a significant increase in the uranium utilization without fuel reprocessing using a special class of fast reactors, referred to as “breed and Burn” (B&B) or “travelling wave” reactors, such as the TWR under development by TerraPower (Gilleland et al., 2008). The unique feature of a B&B reactor is that it can breed plutonium in depleted uranium feed fuel and then fission a significant fraction of the bred plutonium, without having to reprocess the fuel. In order to initiate the chain reaction, the B&B core has first to be fed with adequate amount of fissile fuel such as enriched uranium. Plutonium or TRU extracted from used nuclear fuel could also be used for the “starter”. Thereafter, the B&B core is capable of continued operation while being fed with depleted uranium only. Eventually, the uranium utilization will approach the fraction of the loaded uranium that has been fissioned.

The principles and concepts of B&B reactors had been proposed in the past; (Feyberg and Kunegin, 1958; Driscoll et al., 1979; Fischer et al., 1979; Slesarev et al., 1984; Toshinsky, 1997; Teller et al., 1996; Sekimoto and Ryu, 2000; Sekimoto et al., 2001) is a partial list of references. However, in order to sustain the chain reaction in the B&B mode of operation it is necessary to fission at least 20% of the depleted uranium (See Section 2). The experimental and demonstration fast reactors that operated in the past have proven that, in a hard-spectrum core such as required for a B&B reactor, the HT-9 fuel clad can maintain its mechanical integrity up to 200 dpa, corresponding to a burnup of ~10% FIMA. It is likely that the fuel could have withstood higher burnup without losing its mechanical integrity but there is no experimental evidence that this, indeed, is the case. Moreover, a combination of development of improved structural materials, improved fuel materials and improved core design is likely to increase the attainable burnup. The objective of TerraPower is to use these approaches to enable achieving approximately 20% FIMA average burnup without having to reprocess or re-fabricate the fuel.

Alternatively, it might be possible to establish the B&B mode of operation with limited fuel “reconditioning” – an approach being presently studied at the University of California, Berkeley (Heidet and Greenspan, 2009; Greenspan and Heidet, 2009; Heidet et al., December 2009; Heidet and Greenspan, May 9–14, 2010). The functions of the fuel reconditioning are to remove part of the fission products, primarily the gaseous ones, and replace the fuel clad prior to fuel re-use in the reactor. It is to overcome material performance limits in a way that cannot be used to extract plutonium and that is, hopefully, not as expensive as conventional fuel reprocessing. The re-fabricated fuel can either be re-introduced into the reactor core for additional use, or be used as the “starter” fuel for a new core. The latter option, to be referred to as the “spawning” mode of operation, offers a significant savings in the amount of enriched uranium and, therefore, natural uranium that is required to deploy a fleet of B&B reactors.

The objective of the present study is to evaluate the impact a successful development of B&B reactors and their fuel reconditioning technologies could have on the uranium ore utilization, uranium enrichment capacity, nuclear waste and energy security. The numerical analysis is performed for the USA energy economy but the general conclusions apply to the global energy economy.

Section 2 gives an estimation of the minimum burnup required for establishing the B&B mode of operation as well as the maximum burnup that is attainable in such B&B reactors in case that fuel reconditioning can be used for recycling the fuel in the B&B reactor as long as the fuel has sufficient reactivity to maintain criticality. Section 3 evaluates the impact B&B reactors and fuel reconditioning could have on energy sustainability and economic stability while Section 4 discusses additional options and R&D requirements.

2. Minimum required and maximum attainable burnup

The estimation presented in this paper for the minimum burnup required for sustaining the B&B mode of operation while using depleted uranium feed fuel, as well as for the maximum attainable burnup are based on neutron balance (Heidet et al., December 2009; Heidet and Greenspan, May 9–14, 2010; Petroski et al., May 9–14, 2010) considerations. The analysis follows a unit volume of fuel as a function of burnup in the core, starting from the fresh feed fuel. At any given burnup level,

- Number of fission neutrons generated per unit fuel volume per unit burnup (in FIMA) is

$$\sum_i v^i \Sigma_f^i \Phi / \sum_i \Sigma_f^i \Phi = \sum_i v^i \Sigma_f^i / \sum_i \Sigma_f^i \quad (1)$$

where Σ_f^i is the effective one group macroscopic fission cross section of fuel isotope i , v^i is the average number of neutrons emitted per fission of isotope i and Φ is the total neutron flux over the fuel volume. The summation is overall fuel isotopes and is, in fact, the average number of neutrons generated per fission at a given burnup (see Eq. (4)).

- Number of neutrons absorbed per unit fuel volume per unit burnup is

$$\sum_i \Sigma_a^i / \sum_i \Sigma_f^i \quad (2)$$

where Σ_a^i is the effective one group macroscopic absorption cross section of fuel isotope i . The ratio of Eq. (2) is, in fact, number of neutrons absorbed per number of fission events at a given unit volume at a given burnup.

- Net number of neutrons generated per unit fuel volume up to a given burnup level (BU) is

$$\int d(\text{BU}) \left\{ \sum_i v^i \Sigma_f^i / \sum_i \Sigma_f^i - \sum_i \Sigma_a^i / \sum_i \Sigma_f^i \right\} = \int d(\text{BU}) \underline{v}(\text{BU}) \{1 - 1/k_\infty(\text{BU})\} \quad (3)$$

where BU is expressed in number of initial heavy metal atoms fissioned per unit volume,

$$\underline{v}(\text{BU}) = \sum_i v^i \Sigma_f^i / \sum_i \Sigma_f^i \text{ and } k_\infty = \sum_i v^i \Sigma_f^i / \sum_i \Sigma_a^i \quad (4)$$

- The minimum required burnup is that BU for which

$$\int d(\text{BU}) \underline{v}(\text{BU}) \{1 - 1/k_\infty(\text{BU})\} = 0 \quad (5)$$

In the above we ignored the contribution of (n,2n) and (n,3n) reactions. For more realistic representation of practical core designs one should account in the neutron balance for neutrons that are lost via leakage and capture in the reactivity control elements. These neutron losses can be accounted for by replacing, in Eq. (5), $k_\infty(\text{BU})$ by $k(\text{BU})$ defined as

$$k(\text{BU}) = k_\infty(\text{BU}) * (1 - P_L) * (1 - P_{RC}) \quad (6)$$

in which P_L is the leakage probability and P_{RC} is the probability that the neutron will be captured in the reactivity control elements. The values of P_L and P_{RC} are deduced from 3-D analysis of a representative core.

The quantitative analysis is performed for a large sodium cooled fast reactor core like the B&B core studied in (Greenspan and Heidet, May 9–14, 2010). It uses a ternary metallic fuel U–Pu–Zr with 10 wt% zirconium, a fuel density of 15.85 g/cm³ and a smear density of 75%, to accommodate the fuel swelling with burnup. The volume fraction of fuel, initial gap, HT-9 clad and Na coolant is, respectively, 37.5%, 12.5%, 22% and 28%. These correspond to a hexagonal lattice pitch-to-diameter ratio of 1.122 – near the lower limit used in liquid sodium cooled reactors. The active core height is 209 cm and its diameter is 402 cm. The core is divided into 8 radial batches. At the end of an equilibrium cycle the highest burnup batch is discharged, the other batches are shuffled in a predetermined optimal pattern and a fresh depleted uranium fuel batch is loaded at the outermost core zone. When a fuel batch is close enough to its radiation damage limit it undergoes the melt-refining process developed for metallic fuel in the Experimental Breeder Reactor II project (Hesson et al., 1963). The melt-refining involves loading the declad fuel into a zirconia crucible and melting the mixture at ~ 1300 °C for several hours under argon atmosphere. The gaseous and volatile fission products are released and certain solid fission products are partially removed by oxidation with the zirconia of the crucible. Based on (Hesson et al., 1963) it is assumed that this process can remove nearly 100% of Br, Kr, Rb, Cd, I, Xe and Cs, and at least 95% of Sr, Y, Te, Ba and the rare earths (lanthanides). Thorium and americium are also oxidized with zirconia, and 95% of those two elements will be removed from the fuel. A small fraction of other actinides may also be left in the crucible but the present analysis assumed that this fraction is negligible.¹

The minimum required burnup was found (Greenspan and Heidet, May 9–14, 2010) to be 19.4% FIMA. The corresponding average neutron leakage probability from the core analyzed is $P_L = 4.4\%$ and the fraction of neutrons absorbed in the control systems is $P_{RC} = 2.2\%$. Fig. 1a shows the burnup-dependent neutron balance evolution of this core, which is a plot of the left hand side integral of Eq. (5), using the pertinent values of the P_L and P_{RC} . Likewise, the maximum possible burnup was found (Greenspan and Heidet, May 9–14, 2010) to be 55% FIMA. The P_L and P_{RC} values pertaining to this core are, respectively, 6.95% and 2.1%. The resulting neutron balance evolution is shown in Fig. 1b. In both scenarios the neutron balance analysis predicts burnup values very similar to those obtained from the detailed multi-batch fuel shuffling analysis performed in order to define the equilibrium core composition that will maintain criticality throughout the cycle when fed with only depleted uranium.

Fig. 1a shows that the fuel discharged at 19.4% FIMA has sufficient excess reactivity to provide a total of additional $2.2E + 21$ excess neutrons per cm³ of fuel – reached at a cumulative discharged burnup of 42.5% FIMA. This is more than the $1.7E + 21$ neutrons that need to be provided per cm³ of deplete uranium feed in order to turn it into a net neutron producer – corresponding to the minimum of Fig. 1a curve. That is, the fuel discharged at 19.4% FIMA can serve, after reconditioning (aimed at relieving the radiation damage constraints) as the starter fuel for

a new B&B core. As a consequence, one B&B core can spawn many new B&B cores, as illustrated schematically in Fig. 2. The number of B&B cores at generation “i” equals the number of B&B cores at generation “i-1” plus the number at generation “i-2”. Fissile fuel needs to be purchased only for the first core; thereafter, depleted uranium is the only fuel supply required for the growing fleet of B&B reactors.

The doubling time of such a spawning fleet of B&B reactors is defined as the time it takes to accumulate 50% of the core volume worth of discharged fuel – the amount assumed (conservatively) necessary to make a “starter” for a new core. As the equilibrium cycle lasts 2.05 years and there are 12 fuel batches in the reference B&B core analyzed,² the doubling time is 12.3 effective full-power years (EFPY). Assuming a capacity factor of 90% the doubling time is approximately 13.5 years. Fig. 3 shows the resulting installed capacity evolution; the asymptotic B&B reactors capacity growth rate is 3.86% per year. This capacity growth rate is larger than even the most optimistic scenario forecasted for nuclear energy by the IASA (Fischer et al., 1979) – 3.6% per year. If a single 3000 MW_{th}/1.2 GW_e B&B core is started in 2020 and operated in the spawning mode with the 3.86%/y capacity growth rate, by 2100 the electrical B&B capacity will be 25.2 GW_e and by 2120 it will be 40.8 GW_e. Except for the several tons of enriched uranium or plutonium or TRU required for establishing initial criticality in the first (“Mother”) core, this expanding fleet of B&B reactors requires only depleted uranium for its fuel feed.

3. Impact on energy sustainability and economic stability

Table 1 compares the estimated uranium utilization that could be achieved in fast breeder reactors that are designed to operate in either one of the following five scenarios:

- B&B mode without spawning. Average discharge burnup of 20% is assumed attainable without fuel reconditioning.
- A default for Scenario “a” in case advanced fuel design will not enable getting to 20% average burnup: use of a single fuel reconditioning will enable establishing the B&B mode of operation with a cumulated discharge burnup of, at least, 20% FIMA. No spawning possible.
- B&B mode assuming a successful development of advanced fuel capable of withstanding 20% average burnup and a successful development of the technology for a single fuel reconditioning at ~20% burnup. Spawning is possible.
- B&B mode with 2 or more fuel reconditioning steps that will enable to achieve the maximum attainable burnup of ~50% FIMA (versus 55% obtained in our large B&B core analysis) without separation of most of the solid fission products. Spawning is possible.
- Traditional fast breeder reactor approach in which fuel is reprocessed many times (every 10% FIMA or so). It assumes extraction of all of the fission products and addition of depleted uranium makeup fuel at each recycle. There is no limit to the number of fuel recycles.

Also given in Table 1 is the uranium utilization in the reference scenario of contemporary LWRs that operate with the once-through fuel cycle and discharge their fuel at 50 GWD/T.

¹ In the “Melt-Refining” process experimented within the EBR-II program several percents of the plutonium and other actinides remained in the crud of the zirconia crucible. However, experts think that it is likely possible to develop a modified process that does not involve significant loss of actinides and cannot separate a specific or all of the actinides and, yet, can efficiently remove the gaseous and certain fraction of the volatile fission products. Although the results of this study are somewhat affected by the fraction and type of solid fission products that are removed in the fuel recycling process, the overall conclusions of this work are not expected to vary.

² The radial mesh was refined for these calculations by increasing the number of batches from 8 (used before) to 12. As the fuel discharged from a B&B core undergoes the melt-refining process before being loaded into a new B&B core – to make the new core “starter”, its initial composition is assumed to be axially uniform.

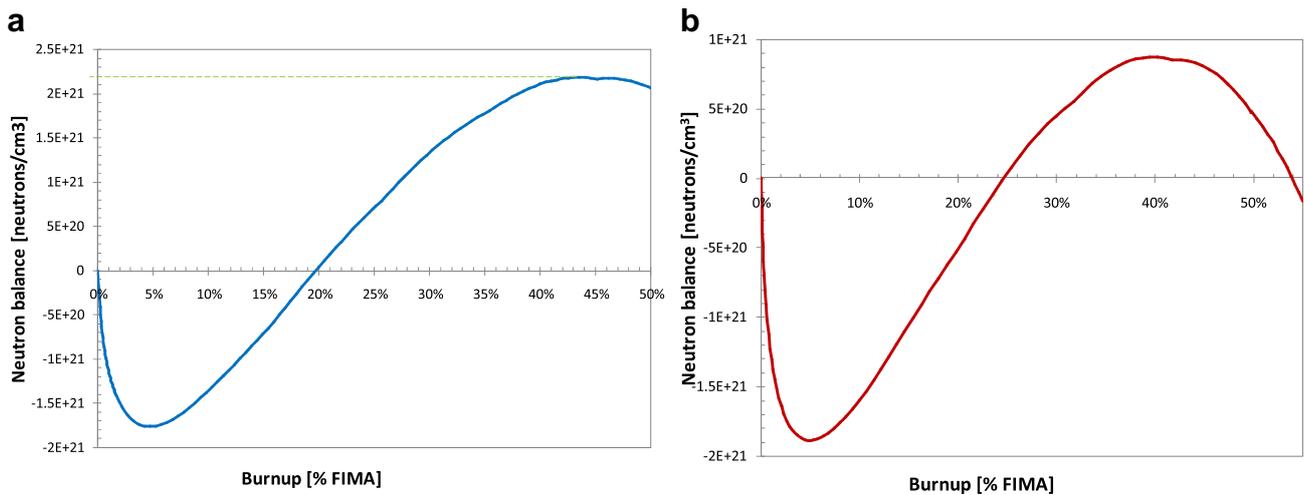


Fig. 1. Neutron balance versus burnup in large fast spectrum core designed to sustain the B&B mode of operation at the minimum required burnup (a) and at the maximum possible burnup (b).

The relative uranium utilization values given in Table 1 are per unit of electrical energy generated. In converting thermal energy to electrical energy it is assumed that fast reactors convert thermal energy into electricity at 20% higher efficiency (relative) than LWRs.

The rightmost column in Table 1 gives the number of years the B&B reactors could supply electricity at present day US total annual consumption rate (from all sources; 4200 million MW_eh/year) if they are to be fueled only with the depleted uranium stockpiles (“waste”) that will be accumulated in the US from the fueling of LWRs ($\sim 1.3 \times 10^6$ tons) and B&B reactors ($\sim 0.5 \times 10^6$ tons) until the end of deployment of the first generation of B&B reactors – assumed in the second half of the 21st century.

It is observed that a successful development of B&B reactors that can achieve 20% average fuel burnup without fuel reconditioning will offer 40-folds increase in the uranium ore utilization versus

that presently achieved. A successful development of a fuel reconditioning technology could increase the attainable uranium utilization to close to 100-fold that achieved in contemporary LWRs. This corresponds to extraction of approximately 50% of the nuclear energy worth of depleted (and natural) uranium. For the utilization of the remainder 50% it will be necessary to develop economically viable and societal acceptable fuel reprocessing technology that will separate the fission products from the actinides. Such a reprocessing could be deferred, though, by several centuries, as the existing stockpiles of depleted uranium can provide all our electricity needs for between 800 and 2000 years (rightmost column of Table 1).

The three options addressed in Table 1 can be supplementary. It is possible to start deployment of B&B reactors as soon as either fuel technology/design that can withstand 20% burnup without

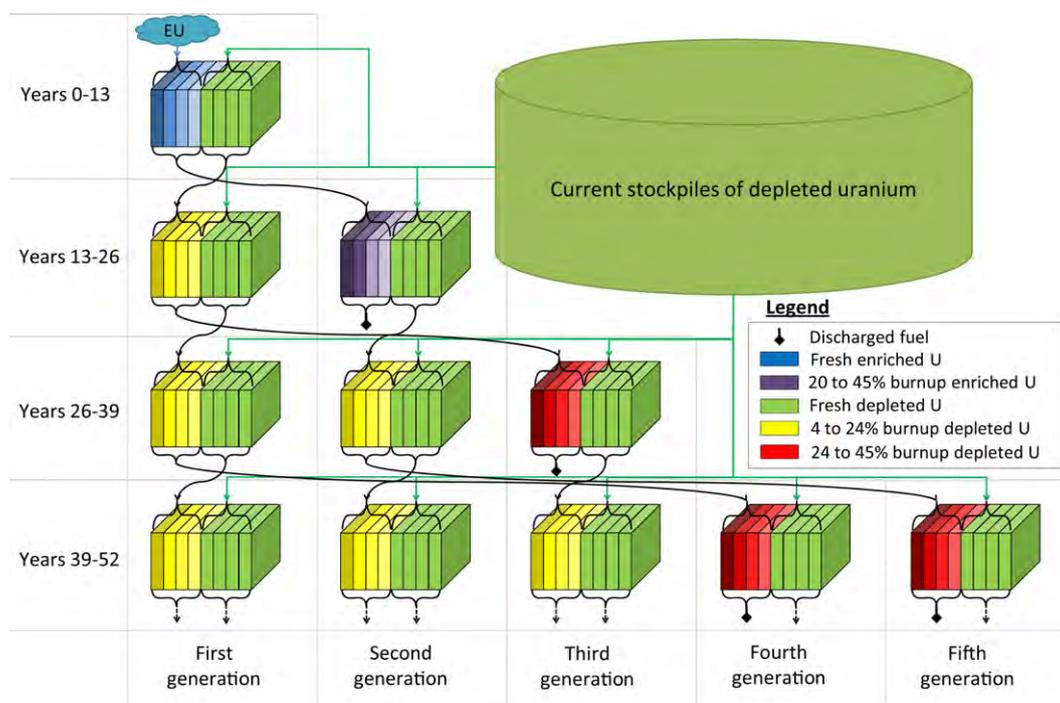


Fig. 2. Spawning schematic of large B&B cores.

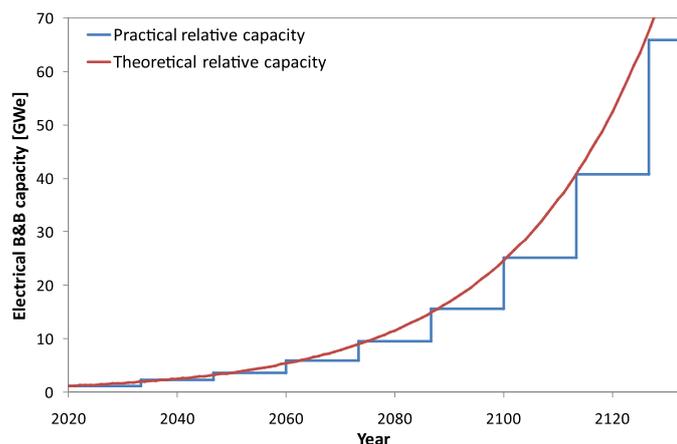


Fig. 3. Large B&B reactor electrical capacity evolution for one large B&B reactor deployed in 2020 and operated in the spawning mode.

reconditioning or recycling will be available, or the technology for fuel reconditioning is developed. When fuel reconditioning becomes feasible, it will be possible to recycle the fuel discharged at relatively low burnup for additional operation in breed and burn reactors until utilizing 40 or 50% of the energy value of the initially fed uranium. If and when fuel reprocessing technology that can remove most of the fission products will be developed to be economically viable and acceptable proliferation resistance wise, it will be possible to use the fuel discharged at ~50% burnup and increase the uranium utilization to >95%. Basically, the same reactor technology is required for these three modes of operation.

4. Discussion

In case the US will decide to proceed with reprocessing of the used nuclear fuel (UNF) that was discharged from the LWRs over the last 50 years or so, it will be possible to use the transuranium isotopes (plutonium and heavier elements) accumulated in this UNF to start the first generation of B&B reactors. In this case there will be need for no natural uranium at all for fueling a large and expanding fleet of B&B reactors for several centuries. The starting-up of such a fleet will make a very effective use of the UNF the disposal of which has, currently, no long term solution. Alternatively, it may be possible to develop new technologies that will recondition the UNF by removing from this fuel a fraction of the fission products along with a fraction (close to 90%) of the uranium and, desirably, converting the oxide to metallic fuel. Such a reconditioned UNF could be used to start B&B reactors, although not as efficiently as the previously mentioned options.

Another option worth considering in the use of excess weapons grade plutonium or enriched uranium to provide the fissile fuel required for the Starter of the first generation of B&B reactors. Thereafter, only depleted uranium (or used nuclear fuel discharged

Table 1
Estimated uranium utilization limits and energy value of depleted uranium.

Mode of operation	Uranium utilization	Relative U utilization	No. of years at present supply of depleted U
Light water reactors (LWRs) - reference	0.6%	1	0
(a) B&B reactor, no fuel reconditioning	20%	40	800
(b) B&B reactor, 1 fuel reconditioning @ 10%	20%	40	800
(c) B&B reactor, 1 fuel reconditioning @ 20%	40%	80	1600
(d) B&B reactor, with >1 fuel reconditioning	50%	100	2000
(e) Fast reactor with continuous recycling	>95%	>190	3900

from LWRs) need be fed. This option offers, perhaps, the most effective utilization of the excess material from the military programs.

The realization of the tremendous amount of clean energy that might be offered by B&B reactors depends on a successful development of the technology for these reactors, the fuel they need and of the reconditioning of this fuel. Among the major technological challenges are the following: (1) Development of improved structural materials for the cladding of the nuclear fuel that will maintain their mechanical integrity up to peak burnups of at least 30% (corresponding to an average burnup of 20%); (2) Development of improved fuel element designs that are capable to safely achieve high burnup; (3) Verification of the safety and economic viability of the B&B reactors; and (4) Development of economically viable technology for reconditioning the fuel discharged from a B&B reactor for additional use in B&B reactors.

Performance of clad and fuel irradiation experiments to higher than so far tested fast neutron fluencies and burnups, combined with development of advanced structural and fuel materials along with other technology and design innovations may enable commercialization of B&B reactors in the foreseeable future. In view of the large potential benefit expected it appears justified to invest in the aforementioned R&D activities.

5. Conclusions

It is concluded that a successful development of the breed and burn reactors and/or the associated fuel re-reconditioning technologies could provide the US and many other countries a great measure of energy security and energy cost stability. Only limited amount of enriched (and, therefore, natural) uranium is required for starting a fleet of first generation breed and burn reactors. The fuel discharged from these 1st generation reactors can be used, after reconditioning, to start the 2nd generation of breed and burn reactors and so on. The growth rate of the installed capacity of B&B reactors possible to achieve using this “spawning” mode of operation is estimated to be close to 4% per year. The amount of natural uranium required for starting a fleet of B&B reactors that will reach an electricity generation capacity of 1000 GWe by the end of this century is estimated to be the equivalent of 10 years of supply to the presently operating commercial fleet of LWRs (86 GWe). No enriched uranium and no enrichment services will be required to support this fleet beyond the completion of the deployment of the 1st generation of B&B reactors – possibly second half of the 21st century. The energy value of the depleted uranium stockpiles (“waste”) that will be accumulated in the US from the fueling of LWRs and B&B reactors until end of deployment of first generation of B&B reactors is equivalent, when used in breeding fast reactors, to from 8 to 20 centuries of the total 2010 supply of electricity in the USA.

Acknowledgment

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AN ACCELERATOR DRIVEN ENERGY MULTIPLIER FOR DOUBLING URANIUM ORE UTILIZATION

Christian Di Sanzo^a, Jeremy Cohen^a, Anselmo T. Cisneros^a, Bernhard Ludewigt^b and Ehud Greenspan^a

^a*Department of Nuclear Engineering, University of California, Berkeley
4155 Etcheverry Hall, MC 1730, Berkeley, CA 94720-1730, USA*

^b*Lawrence Berkeley National Laboratory,
1 Cyclotron Rd, Berkeley, CA 94720, USA*

e-mail: cdisanzo@berkeley.edu

A preliminary assessment is performed of the feasibility of doubling the fuel utilization of fast reactor fuel, without fuel reprocessing, using an Accelerator Driven Energy Multiplier (ADEM). Instead of reprocessing, the fuel discharged from a Breed-and-Burn critical core at an average burnup of 20% it is proposed to re-clad the fuel over the pre-existing cladding after venting the gaseous fission products, avoiding costly reprocessing and consequent proliferation risks. The double-clad fuel cannot establish criticality for significant burnup, but can generate fission energy in an ADEM. The goal set for this ADEM is doubling of the fuel utilization – that is, irradiating the double clad fuel in the ADEM for an additional 20% burn-up. Using a 10 MW_e accelerator and assuming 20 neutrons production per 1 GeV proton, the electric power gain at equilibrium is found to be about 8 – too low for commercial viability.

Moreover, when using a 10 MWe of cyclotron – near present-day technology, the attainable average specific power and average power density of the blanket are one order of magnitude smaller than of the critical fast reactor core that provided the fuel for the ADEM.

I. INTRODUCTION

There is growing interest in Breed and Burn (B&B) reactors, also referred to as Travelling Wave Reactors (TWR) – sodium-cooled fast reactors that, once initial criticality is established, can sustain criticality “indefinitely” when fueled with depleted uranium only while operating in a once-through fuel cycle¹⁻⁴. In a B&B mode of operation depleted uranium is first converted into plutonium, part of which is fissioned in situ. In order to do so, a minimum burn-up is required to sustain criticality, so that enough plutonium is bred in the system. It was found³ that the practical minimum required average

discharge burn-up using metallic fuel and very-low leakage core was to be in the vicinity of 20% FIMA (Fissions per Initial Metal Atom). With 20% burn-up, B&B can increase the fuel utilization by one order of magnitude compared to commercial Light Water Reactor (LWR). The main limitation to reaching higher than 20% burn-up is the accumulation of neutron-induced damage in the cladding⁵ and not the neutronics performance of the fuel. Consequently, the fuel discharged from B&B reactors could still be used, without reprocessing, for further energy production, provided the fuel will be “reconditioned” so as to relieve it from the radiation damage.²⁻⁴ One approach examined²⁻⁴ for fuel reconditioning involves application of the melt-refining process²⁻⁴. Another approach is examined in this paper – re-cladding the fuel discharged after 20% FIMA and using it to fuel an Accelerator Driven Energy Multiplier (ADEM).

Accelerators had been proposed as drivers of subcritical fission systems for a variety of applications including breeding fissile material and transmutation of transuranic elements⁶⁻⁷. In this study, an accelerator driven system (ADS) is designed to increase the fuel utilization of B&B reactor discharged fuel so as to avoid the need for fuel reprocessing.

II. DESCRIPTION OF FUEL CYCLE AND REACTOR SYSTEM

II.A Fuel used in ADEM

An insight on the neutron balance in the ADEM blanket that is loaded with B&B discharged fuel can be obtained from the k_{∞} evolution with burn-up of a single unit cell. The unit cell consists of a homogenized composition of fuel, coolant and cladding, as is commonly the practice for fast spectrum neutronic

analysis. The analysis is performed using MCNP5 1.51⁸ and the ORIGEN 2.2⁹ codes that are coupled with the MOCUP 2.0¹⁰ driver. The initial heavy metal fuel composition is depleted uranium and the clad is assumed to be HT-9 - 12% Cr ferritic-martensitic steel. The reactor coolant is sodium. The power density assumed is 300 W/cc, which is well within the heat-removal capability of liquid-metal-cooled cores.

Fig.1 shows the calculated k_{∞} as a function of burn-up for a couple of unit cells – one made of a single-clad fuel and the other of a double clad fuel. It is found that at 20% FIMA the single-clad fuel unit cell has a k_{∞} of about 1.18 and is therefore still a net neutron producer. By double cladding the fuel at 20% FIMA k_{∞} drops to approximately 1.05. Accounting for leakage from the finite core the double-clad fuel will not be able to sustain criticality for additional burn-up of interest. Consequently, the use of a source-driven subcritical blanket is examined for achieving a significant additional burn-up from the double-clad fuel.

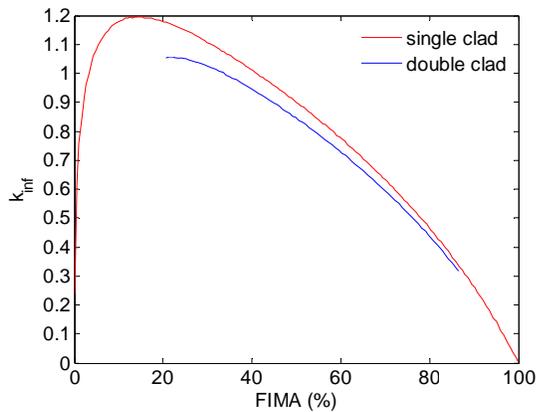


Fig. 1. Single clad and double-clad unit cell k_{∞} as a function of burn-up.

II.B Fuel reconditioning Process

The reconditioning process considered in this study consists of relieving the gaseous fission products pressure and encasing the fuel with its pre-existing cladding in a new, larger diameter clad. New fuel assemblies are made of the double clad fuel and are loaded into the ADEM blanket. Fig. 2 shows a graphical illustration of fuel unit cell before and after the reconditioning process.

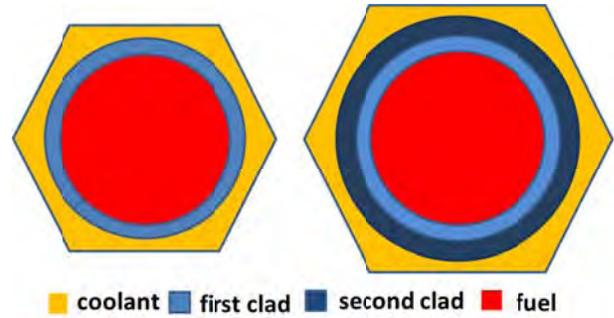


Fig. 2. Unit cell of B&B reactor fuel (left) and unit cell of the double-clad fuel (right).

Table I reports the dimensions of the unit cell for single clad case, with a pitch-to-diameter ratio of 1.122. Two reconstituted fuel assemblies are examined. One has the same pitch-to-diameter ratio of the original unit cell so that the total coolant volume in the reactor is the same. In the second case the unit-cell coolant cross-section area is preserved. The resulting volume fractions of the unit cell elements are given in Table II, based on the geometrical parameters of Table I.

TABLE I. Unit cells geometrical parameters.

dimension (cm)	single clad	double clad - same P/D	double clad- same coolant area
rod (D)	1	1.166	1.166
clad (t)	0.083	0.166	0.166
P/D	1.122	1.122	1.079

TABLE II. Unit cell volume fractions.

volume fraction	single clad	double clad - same P/D	double clad- same coolant area
coolant	28%	28%	22%
fuel	50%	37%	40%
clad	22%	35%	38%

II.C ADEM Design

The ADEM design consists of a central neutron source surrounded by a subcritical radial blanket. The spallation neutron source is provided by a 10 MWe cyclotron impinging on a central cylindrical lead target. The lead target is a 10 cm radius cylinder with a height of 20 cm. A schematic of ADEM is shown in Fig. 3. The accelerator beam tube is inserted from the top into a central cavity in the subcritical blanket. The beam tube is cooled from the outside by liquid-lead which flows from bottom to top in the central channel and functions as both coolant and spallation target. A 3 mm HT-9 wall separates the central channel from the sodium cooled subcritical blanket. The subcritical blanket is made of 10 radial batches, each divided into 3 axial zones for burn-up calculations. Sodium was chosen as the coolant of the subcritical blanket for its better than lead thermal-hydraulic performance in B&B cores¹².

The external diameter of the subcritical blanket is 3 m and it is surrounded by a 1 m thick reflector made of HT-9 and sodium at equal volume. The fuel elements are 209 cm long and have a 2 m plenum on top to accommodate gaseous fission products. The blanket is divided into 10 equal volume fuel batches.

II.D Neutron Source Description

The spallation neutron source is modeled in MCNP5 1.51 as 20 cm diameter 20 cm long cylinder of a uniform intensity having an energy spectrum that is typical to that of spallation neutrons produced by 1 GeV protons impinging on a lead target¹³. The neutron source spectrum input to MCNP5 is shown in Fig. 4; it is a discrete approximation of the neutron spectrum derived from continuous energy distribution data, presented in Ref. 13.

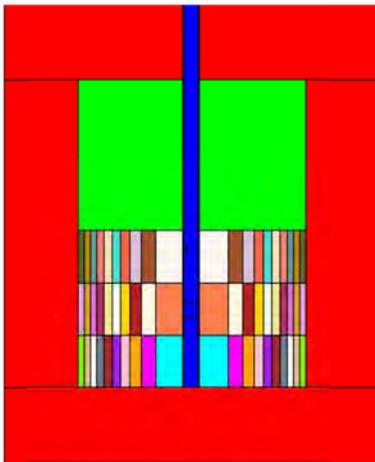


Fig. 3. ADEM design in MCNP geometry

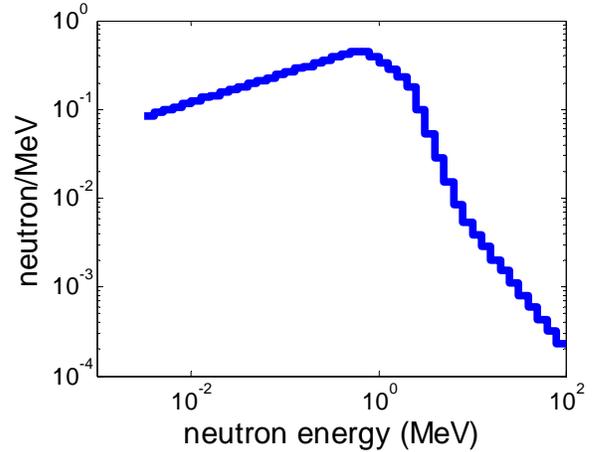


Fig. 4. Energy spectral distribution of the spallation neutron source simulated in MCNP5 1.51. Derived from the continuous energy distribution of Ref. 13

To estimate the intensity of the neutron source it was assumed that the accelerator efficiency is 40% and that 1 GeV proton impinging on a lead target produces 20 neutrons. With these assumptions a 10 MWe accelerator is estimated to produce a spallation neutron source of 5×10^{17} neutrons/second; this estimate was used for burn-up calculations.

III. DESCRIPTION OF COMPUTATIONAL METHODS

III.A Calculation of k_{src} for subcritical systems

A metric that is useful for understanding ADEM performance is the source multiplication factor of the subcritical blanket, k_{src} . The value of k_{src} differs from the k_{eff} multiplication factor that is usually obtained from the solution of the source-free transport equation. k_{src} accounts for the actual neutron flux distribution that is established in the subcritical blanket by the driving neutron source distribution. This neutron flux distribution tends to have its highest amplitude close to the driving source and to decline with distance from the source – it can be significantly different from the flux distribution calculated for the same blanket by solving the k_{eff} eigenvalue transport equation.

To infer the k_{src} value from MCNP5 calculated results it is assumed that each source neutron produces, on the average, k fission neutrons and that each of these 1st-generation fission neutrons produces k 2nd-generation fission neutrons and so on. The total number of fission neutrons generated in such a sub-critical chain reaction induced by one source neutron, to be denoted by F , is:

$$k+k^2+\dots=\frac{k}{1-k}=F$$

Solving for k one obtains:

$$k=\frac{F}{F+1}$$

which is k_{src} for a source driven system.

If neutron generation via n, xn reactions is to be accounted for, F need be replaced by $F+N$ where N stands for the number of n, xn neutrons generated per source neutron in the subcritical chain reaction so that:

$$k_{src}=\frac{F+N}{F+N+1}$$

For ADEM the number of neutrons from n, xn reactions is about two order of magnitudes lower than the number of fission neutrons.

MCNP5 outputs N and F so the k_{src} value can be easily inferred using the above expression. The k_{src} value calculated for a typical ADEM system was found to be about 1% larger as compared to the k_{eff} value calculated for the same system. This is due to the fact that the center-peaked neutron flux distribution in the ADEM features a reduced radial neutron leakage probability and that the k_{∞} value of the innermost fuel batch is the highest.

IV. RESULTS OF BURN-UP SIMULATIONS

Simulations for ADEM are performed using MCNP5 1.51 and ORIGEN 2.2, coupled with MOCUP 2.0. ENDF-VII/B cross-sections for a temperature of 900 K are used for all simulations.

The subcritical blanket is initially loaded with double clad fuel assemblies after they went through the reconditioning process described in Sec. II.B. The Beginning Of Life (BOL) fuel composition is radially uniform. It corresponds to the composition of the fuel discharged from B&B reactors at 20% average burnup³. The fuel composition is not uniform in the axial direction due to axially varying burnup. Three axial burnup zones were considered sufficient for capturing the axial fuel composition variation for this study.

The BOL fuel composition assumed for the three axial zones is given in Table III for the most important nuclides.

As in previous studies³ of B&B reactors, it is assumed that 75% of the produced gaseous FPs instantaneously migrate out of the fuel and are accumulated in the fission gas plenum above the fuel.

TABLE III. BOL weight percent of selected fuel nuclides with respect to the total mass of heavy metal and fission products.

nuclide	bottom zone	middle zone	top zone
²³⁸ U	80.50%	79.61%	80.54%
²³⁷ Np	0.07%	0.07%	0.07%
²³⁸ Pu	0.04%	0.04%	0.04%
²³⁹ Pu	9.03%	9.03%	9.02%
²⁴⁰ Pu	2.01%	2.11%	2.01%
²⁴¹ Pu	0.19%	0.20%	0.18%
²⁴² Pu	0.03%	0.03%	0.03%

IV.A. RESULTS FOR P/D=1.122 (original lattice pitch)

The shuffling scheme for ADEM cycle is assumed to be in-to-out – the reconditioned double-clad fuel is charged in the innermost radial batch and after each cycle it is shuffled outwardly. The assumed cycle length is 5500 days. The calculated BOL k_{src} is 0.985. However, as shown in Fig. 5, k_{src} decreases with time until an equilibrium fuel composition is reached.

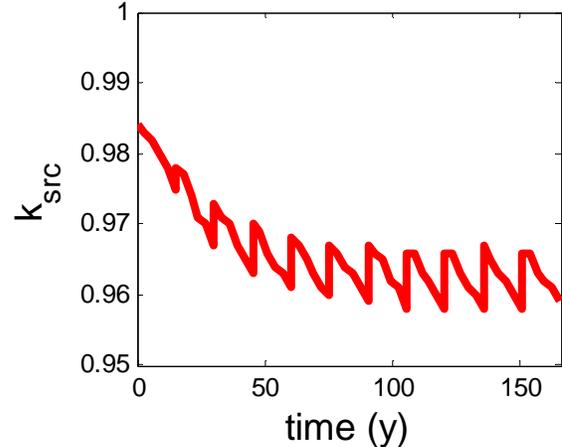


Fig. 5. k_{src} evolution in the ADEM; P/D=1.122.

The equilibrium core k_{src} varies from 0.966 at Beginning Of Equilibrium Cycle (BOEC) to 0.959 at End Of Equilibrium Cycle (EOEC). The corresponding fission power produced by ADEM that is driven by a 10 MW_e proton cyclotron is shown in Fig. 6.

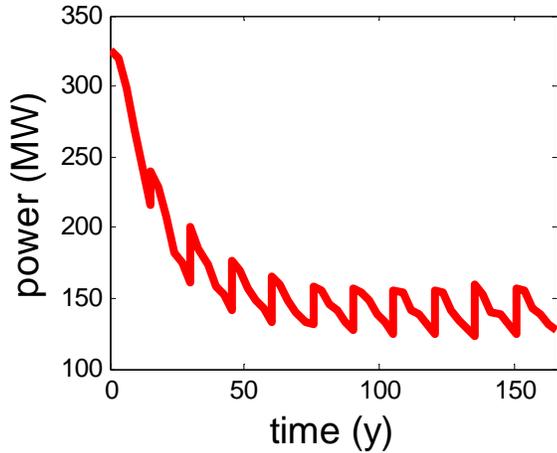


Fig. 6. Fission power evolution in ADEM that is driven with a 10 MW_e proton accelerator; P/D=1.122.

It is observed that the power generation rapidly drops, following the k_{src} behavior. Assuming a 40% energy conversion efficiency, the electricity gain offered by the ADEM drops from a factor of $(320 \cdot 0.4 / 10 =) 12.8$ at BOL to a factor of between $(160 \cdot 0.4 / 10 =) 6.4$ at BOEC and $(125 \cdot 0.4 / 10 =) 5.0$ at EOEC. This implies that, at EOEC, approximately 20% of electrical power generated needs to be recirculated for driving the cyclotron. Additional several percents of the generated electricity is needed for driving the cooling pumps and other plant auxiliary systems. The combined recirculating power fraction is too high for an economically viable power plant. Moreover, the average specific power and average power density of the blanket are less than one order of magnitude smaller than of the B&B core that provided the fuel for the ADEM.

Fig. 7 shows the batch-wise burn-up at BOEC and EOEC. It is observed that only the inner 6 fuel batches undergo a significant burning in a cycle. The discharge burn-up is about 35% - short of the design goal of 40%. It can be increased by increasing the fuel residence time, but this will result in a smaller equilibrium cycle k_{src} and, hence, smaller blanket energy multiplication.

The batch fractional power generation is shown in Fig. 8. As expected, it is highly peaked in the central radial zone, while the outer zones do not undergo significant fissions and do not contribute much to power generation. The power fraction is almost constant from the BOEC to EOEC, resulting in a stable profile for power generation.

The neutron leakage probability is given in Table IV. The total leakage is less than 10%. Use of liquid lead as a coolant and reflector could reduce the leakage probability. However, lead inferior thermal-hydraulic performance could penalize the attainable power level.

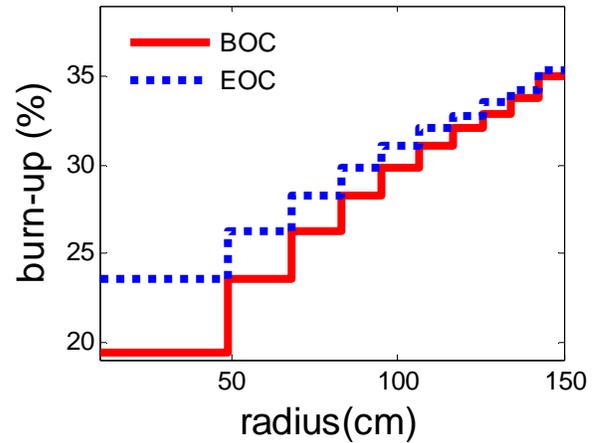


Fig. 7. Average batch burn-up at BOEC and EOEC; P/D=1.122

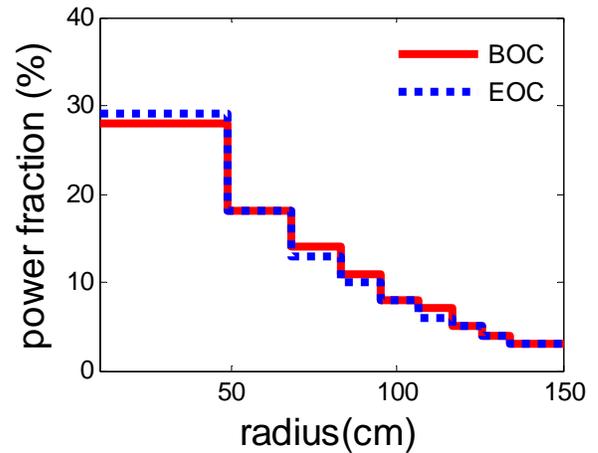


Fig. 8. Power fraction at BOEC and EOEC; P/D=1.122

TABLE IV. Neutron leakage probability from ADEM (P/D=1.122)

leakage	BOEC	EOEC
bottom	2.50%	2.48%
top	2.44%	2.48%
radial	4.65%	4.57%

IV.B. RESULTS FOR P/D=1.078 (preserved coolant area)

The same type of analysis was carried out for a blanket having a pitch-to-diameter ratio of 1.078. The equilibrium composition was searched starting from the already known equilibrium composition for P/D=1.122 blanket, thus enabling to accelerate the convergence process, as displayed in Fig. 9.

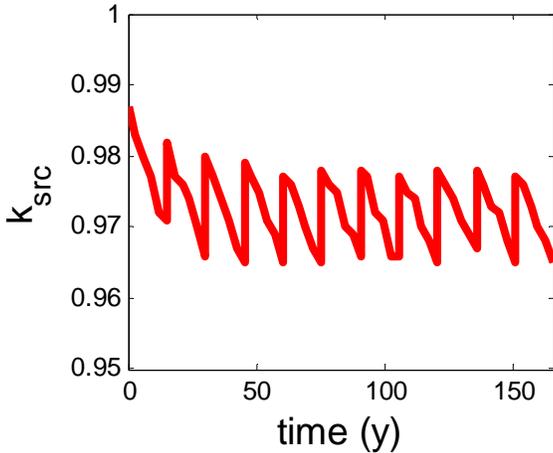


Fig. 9. k_{src} evolution in the ADEM; P/D=1.078

It is observed (Fig. 9) that at equilibrium the k_{src} is between 0.98 and 0.965, for a corresponding power production varying from 250 MW_{th} to 150 MW_{th} (Fig. 10). This corresponds to an average equilibrium electricity gain of 8. Although higher than attainable with the large P/D (Section IV.A) the recirculating power fraction is still too high for an economically viable power plant. Likewise for the average specific power and average power density of the blanket.

Fig. 11 shows the batch burn-up distribution in ADEM. It is found that the discharge burn-up is 39.5%, basically reaching the design goal of 40%. Fig.12 shows the power fraction in each batch; as in the previous case, the power shape is essentially constant from BOEC to EOEC.

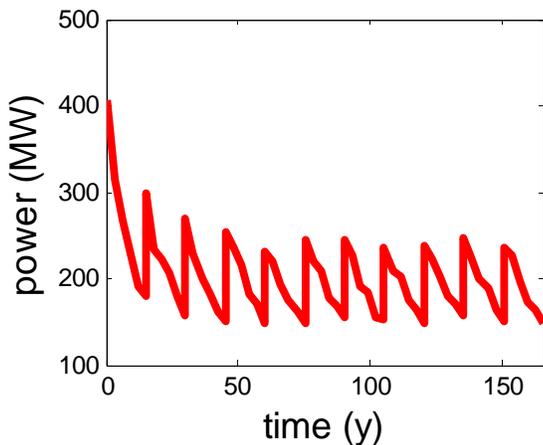


Fig. 10. Fission power evolution in ADEM that is driven with a 10 MW_e proton accelerator; P/D=1.078

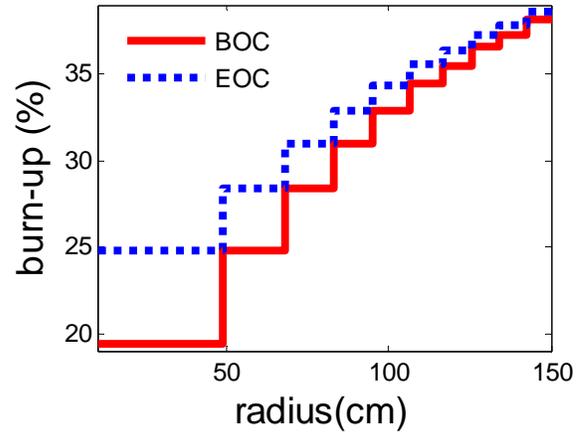


Fig. 11. Average batch burn-up at BOEC and EOEC; P/D=1.079

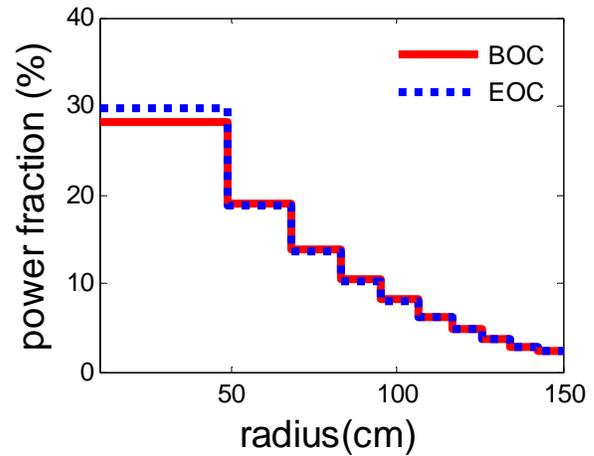


Fig. 12. Power fraction at BOEC and EOEC; P/D=1.079

V. CONCLUSIONS AND DISCUSSION

The maximum energy multiplication attainable from an ADEM consisting of a spallation neutron source featuring 20 neutrons per 1 GeV proton beam energy and a fission blanket that is fed with 20% FIMA fuel that was discharged from a critical breed-and-burn reactor and was reconditioned – by venting the gaseous fission products and double cladding, is found to be approximately 8, when the fuel is discharged after accumulating additional 20% FIMA. This energy multiplication is too low for commercial viability.

Moreover, when using a 10 MWe of cyclotron – near present-day technology, the attainable average power density and average specific power of the blanket are one order of magnitude smaller than of the B&B core that provided the fuel for the ADEM.

Additional blanket design optimization studies are needed to find out if the ADEM approach is promising

enough for increasing the uranium fuel utilization without resorting to fuel reprocessing.

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Inherent safety of minimum-burnup breed & burn reactors

Staffan Qvist and Ehud Greenspan

*Department of Nuclear Engineering,
University of California Berkeley, CA 94720-1730, USA
Tel: 510-508-4490, Fax: 510-643-9685, Email: staffan@berkeley.edu*

Abstract – Reactors that aim to sustain the breed & burn (B&B) mode of operation at minimum discharge burnup require excellent neutron economy. Minimum-burnup B&B cores are generally large and feature low neutron leakage probability and a hard neutron spectrum. While highly promising fuel cycles can be achieved with such designs, the very same features are pushing the limits of the core's ability to passively respond safely to unprotected accidents. Low leakage minimum-burnup sodium-cooled B&B cores have a large positive coolant void-worth and coolant temperature reactivity coefficient. In this study, the applicability of major approaches for fast reactor void-worth reduction is evaluated specifically for B&B cores. The design, shuffling scheme and performance of a new metallic-fueled, sodium-cooled minimum burnup B&B core, used as basis for the void-worth reduction analysis, is presented. The analysis shows that reactivity control systems based on passive ${}^6\text{Li}$ injection during temperature excursions are the only option able to provide negative void-worth without significantly increasing the minimum burnup required for sustaining the B&B mode of operation. A new type of lithium expansion module (LEM) system was developed specifically for B&B cores and its effect on core performance is presented.

I. INTRODUCTION

Breed and Burn (B&B) reactors are designed to produce (breed) and burn (fission) plutonium & minor actinides without chemical separation in the fuel cycle. B&B cores require fissile fuel, typically enriched uranium or transuranics, to establish initial criticality. Thereafter, the B&B reactor can be fed solely by fertile fuel such as depleted uranium (DU). The achievable fuel utilization for B&B reactor could be at least 40 times higher than that of light water reactors and the entire US electricity demand could be met using B&B reactors fed by readily available DU for at least 800 years.¹ In addition, discharged fuel from one B&B reactor can serve as starter fuel for additional B&B reactors. Feinberg first proposed the general B&B concept in 1958, and since then two types of B&B concepts have been developed:²

Travelling wave – in which a nuclear breed & burn wave slowly moves axially through the length of the fuel, which stays stationary in the core. ${}^{239}\text{Pu}$ is bred from ${}^{238}\text{U}$ in the wave front and then fissions and supplies the neutrons needed to perpetuate the movement of the wave. The concept was developed by Teller et al., used in the CANDLE concept by Sekimoto et al. and further mathematically analyzed by Van Dam and others.³⁻⁵

Standing wave – Rather than a burnup wave moving through the fuel, the standing (or soliton) wave concept relies on radially shuffling fuel assemblies (moving fuel) while maintaining a relatively constant spatial power distribution. The standing wave core fuel cycle can include

fuel charge/discharge and thus eventually reach an equilibrium cycle. The FMSR design from BNL, the SSFR design from ANL and the design proposed by Toshinsky are examples of multi-batch equilibrium cycle B&B core concepts.⁶⁻⁷ B&B cores that use fuel reconditioning by melt-refining have been the main focus of research at UC Berkeley.⁸ Standing wave B&B cores with no in-cycle fuel charge/discharge are currently being pursued commercially with the TWR reactor from TerraPower LLC and the EM² from General Atomics.⁹⁻¹⁰

In order to sustain the B&B mode of operation, the burnup the DU feed fuel needs to accumulate has to exceed a certain threshold. The value of this threshold strongly depends on the core neutron balance. In a high fuel volume fraction, hard spectrum low-leakage sodium-cooled metallic-fueled B&B core, the minimum required burnup is about 20%.¹¹ This corresponds to a peak burnup exceeding 30% and peak radiation damage of the fuel cladding in the vicinity of 500 dpa (displacements per atom).

Experimental fast reactors such as the Fast Flux Test Facility (FFTF) have proven that, in a hard spectrum core such as required for a B&B reactor, the HT-9 fuel clad can maintain its mechanical integrity up to 200 dpa corresponding to a burnup of ~10% FIMA (Fission per Initial heavy Metal Atom).¹² Further irradiation of HT-9 to higher dpa levels and the development of new materials may push these limits above 20% FIMA (400 dpa), making it technically feasible to sustain the B&B mode of

operation without the need for fuel re-cladding and reconditioning. Because of this, much of the effort in B&B research is focused on reducing the value of average and maximum discharge burnup toward its theoretical minimum. Minimizing B&B discharge burnup requires minimizing neutron leakage from the core. Therefore, B&B cores are very large and their reactivity feedback response is quite different from that of conventional sodium-cooled fast reactors (SFRs).

In conventional SFRs, an increase in temperature will cause a significant increase in the neutron leakage probability due to the reduced density of the core and reflector constituents. In B&B cores, in which the leakage probability is very small, a given relative increase in leakage has a minor effect on core reactivity. Specifically, the leakage-based negative component of the coolant temperature reactivity coefficient is greatly reduced in the large B&B core versus conventional SFR core, leading to a more positive value of this coefficient. Thus, the reactivity feedback response and coolant void worth of B&B reactors are unfavorable compared to conventional SFRs.

The objective of this work is to identify ways of designing B&B cores that lowers the value of the coolant density reactivity coefficient and void worth. The ambition is to design B&B cores with equal or greater safety margins than conventional SFRs.

The design, shuffling scheme and performance parameters of a new minimum burnup B&B core, used as a basis for this study, are given in Section II. Section III evaluates different design approaches for void worth reduction in B&B cores. In Section IV, new systems for inherent reactivity control using passively injected Lithium-6 are presented and discussed. Section V summarizes conclusions and the outlook for future research.

II. MINIMUM BURNUP B&B CORE DESIGN

II.A. General design and performance parameters

A minimum burnup B&B reactor core concept *ORB*² (*Optimized Radially shuffled Breed & Burn*) was developed and employed as the reference reactor for this study. A side and top view of the *ORB*² core geometry are shown in Figure 1 and 2 and the core geometry and performance characteristics are summarized in Table 1. Figure 3 shows its 8-batch shuffling scheme. The only fuel input in the *ORB*² equilibrium cycle is DU. The concept does not employ any in-cycle fuel reconditioning, assuming that advanced fuel and cladding materials will be able to maintain the fuel integrity up to the minimum required burnup.

TABLE I

Main parameters of the *ORB*² core

Power	3000 MW thermal
Coolant	Sodium
Feed fuel	DU-6Zr (w%)
Cladding, Duct & Wire-wrap	HT-9
Discharge burnup	21.5% FIMA (average)
Fast fluence (E > 0.1 MeV)	12 x 10 ²³ n/cm ²
Burnup reactivity swing	2.2%
$\beta_{\text{eff}} (=1\$)$	0.00342
Inlet/outlet coolant temperature	350°C / 500°C
Radial max/average power	2.3
Core radius	250 cm
Active core height	175 cm
Gas plenum height	200 cm
Specific power density	16 MW/MT
Average power density	90 W/cm ³
Maximum power density	265 W/cm ³
Radial/Axial neutron leakage	1.6% / 4.3%
Void worth (\$)	+12\$

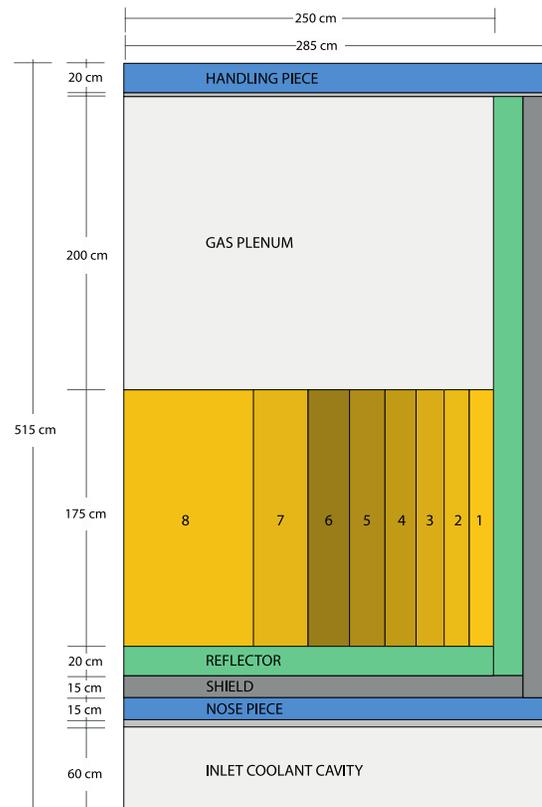


Figure 1, Radial side-view of the *ORB*² core design (the center of the core is at left edge of the figure), fuel batches are labeled with numbers from out-to-in with a darker hue at higher burnup.

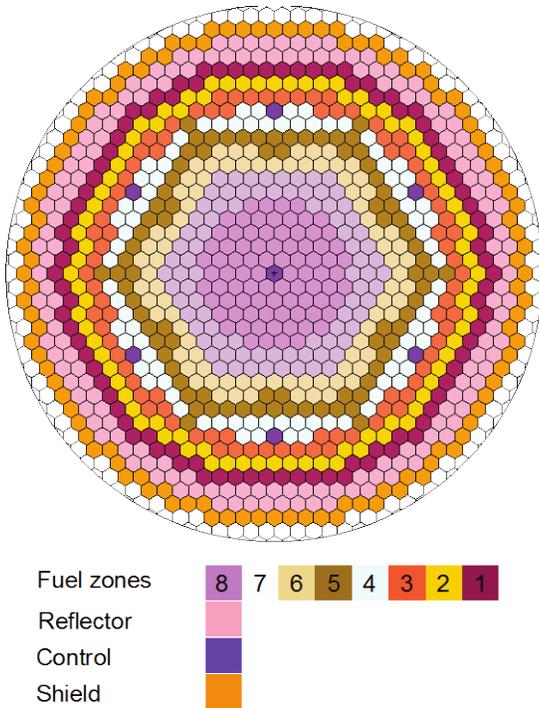


Figure 2, Top view of the ORB2 fuel assembly layout

II.B. ORB² shuffle scheme optimization

The reactor core design and fuel assembly layout is the result of an optimization toward both neutron efficiency and simplicity of operation and construction. A small number of fuel batches means fewer shuffling operations (higher capacity factor) and a more simple design, but is generally neutronically inefficient. A larger number of batches enable a more efficient neutronic design (lower discharge burnup) with a lower burnup reactivity swing and a smaller power peaking factor. The optimum 8-batch shuffling scheme was found in a search that was based on the following criteria:

Requirements

- k_{eff} at or above 1.0 throughout the burnup cycle
- Cycle reactivity swing < 3%

Optimization, Minimize

- Discharge burnup
- Radial power peaking
- Orificing mismatch
- Burnup reactivity swing
- Void reactivity worth

The resulting optimal shuffling scheme is given in Figure 3.

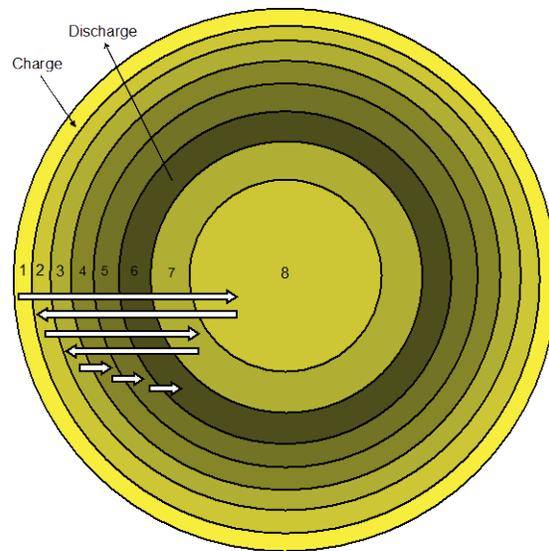


Figure 3, The ORB² 8-batch optimal shuffling scheme

A search for the optimum 8-batch shuffling scheme does not lend itself to direct “brute force” numerical evaluation, as a thorough evaluation of the 8! possible schemes would be prohibitively computationally expensive. For the ORB² core, a comparative point system for the minimization requirements was implemented to find the optimal solution from an initial set of scheme-types (in-to-out, out-to-in etc.). Schemes were awarded points by their relative success in minimizing the 5 previously mentioned parameters. In the end, the scheme with the highest combined score was chosen for implementation. Computations were made for 8 radial x 3 axial burnup zones of homogenized composition using a coupled MCNP/ORIGEN code system^{a, 13-15}.

Some general features for successful B&B equilibrium cycle shuffling schemes were identified during this work:

Fuel zone loading for maximum neutron economy

In order to reduce radial neutron leakage to very low values, the outermost fuel batch needs to be loaded with fuel assemblies with the lowest infinite multiplication factor (k_{∞}) available in the core. That is, with fresh fertile fuel (DU). In minimum-burnup B&B cores, the fuel k_{inf} increases monotonically with burnup. The lowest available BU fuel assemblies should be loaded in the batches next to the outermost DU batch to occupy an annulus of low k_{∞} fuel that is at least two neutron mean free paths in thickness.

^a The equilibrium-cycle analysis code FAST-BEAU developed at UC Berkeley by A. T. Cisneros (not published) was used for coupling of MCNP and ORIGEN. In MCNP, the ENDF/B-VII.0 neutron reaction cross-section package was used.¹⁵

In minimum burnup B&B cores, k_{∞} peaks when the fuel burnup gets to ~12.5% FIMA after which it gradually declines due to fission products accumulation and uranium depletion. The rate of k_{∞} decline is very moderate as compared to the rate of k_{∞} increase at low burnups. It is therefore preferential to place high burnup fuel in zones of high neutronic worth. However, in this analysis it was found that doing so leads to unacceptably high levels of radial power peaking – most of the power is produced in the central zone. The general conclusion is that high k_{∞} fuel should be loaded in intermediate batches but not at the radial center of the core.

Void worth reduction zoning

Very large low leakage B&B cores tend to have very large positive values of coolant density reactivity and void worth. To ensure a safe and benign response to anticipated transients, these values need to be minimized or ideally turned negative. One ambition of this work was to identify ways to shuffle the fuel that would minimize the void worth. When the coolant density decreases (or boils/voids), the average neutron energy and mean free path increases. Neutrons will on average travel further between birth and absorption. If the fuel can be arranged in such a way that neutron escaping the fuel zones where they are born are preferentially absorbed in low k_{inf} fuel, or leak out of the core entirely, the void worth may be reduced.

Unfortunately, shuffling schemes that are optimal for void worth reduction are the exact opposite of schemes aimed at maximized neutron economy. A compromise was found where both inner and outer radial zones are made up of low k_{inf} fuel, ensuring that regardless of radial direction of neutron leakage, neutrons will encounter low k_{inf} fuel. This approach managed to lower core void worth (using an out-to-in shuffle scheme reference) by about -2\$ -- from 14\$ to 12\$, with very little effect on neutron economy (same discharge burnup).

II.C. ORB² fuel assembly design

Neutronic analysis of fast reactor systems can be carried out with a high degree of accuracy without explicit modeling of small-scale heterogeneities, such as individual fuel pins. This is due to the long mean free path of neutrons in fast systems. While explicit heterogeneous modeling can be avoided during calculations, fuel assembly design is still a necessity since the volume fractions of different materials in the core are needed for the homogenized zones.

The fuel assembly design process for a B&B core is conceptually simple – maximize the fuel volume fraction without violating thermal hydraulic constraints. An iterative single (hot) channel thermal-hydraulics coupled with a fuel assembly geometry solver was developed to find the optimum fuel assembly design. The optimal ORB²

assembly design was found using the constraints given in Table 2.

TABLE 2

ORB² fuel assembly design constraints

Peak cladding temperature	600°C
Maximum coolant velocity	12 m/s
Coolant temperature increase across core	150°C
Maximum channel friction pressure drop	1 MPa
Peak fuel temperature	900°C

Details of fuel pin and assembly geometry are given in Table 3 and 4, and the resulting volume fractions for the neutronic analysis is given in Table 5.

TABLE 3

Optimal ORB² fuel pin parameters

Pin pitch-to-diameter (P/D) ratio	1.10
Cladding thickness	0.56 mm
Fuel diameter	0.89 cm
Cladding inner diameter	1.026 cm
Cladding outer diameter	1.138 cm
Allowable fuel swelling	25%

TABLE 4

Optimal ORB² fuel assembly parameters

Maximum outer diameter	20.0 cm
Duct thickness	0.4 cm
Inter-assembly gap thickness	0.35 cm
# of pins per assembly	169
# of fuel assemblies/core	672

TABLE 5

Optimal ORB² core volume fractions %
(as manufactured, excluding control assemblies)

Inter-assembly coolant	3.5
Duct	7.7
Fuel	40.3
Cladding	12.4
Bond	13.4
Coolant (inside duct)	22.7

III. FAST REACTOR VOID WORTH REDUCTION APPROACHES AND THEIR APPLICABILITY FOR B&B CORES

In fast reactors in which the relative importance of neutron leakage on core reactivity is small, core reactivity

is generally increased by a decrease in coolant density. The extreme case of such an event is when coolant temperatures increase to the point where a phase change occurs; the effect on reactivity of such an event is referred to as void worth. When mentioned in this report, void worth is the change in k_{eff} (measured in %) by complete coolant voiding of the entire core, including the plenum and inlet cavity. While neutron leakage increases upon coolant voiding, the spectral hardening component dominates the reactivity feedback (changes in coolant absorption are irrelevant).

A large research effort aimed at fast reactor void worth reduction has been ongoing since the inception of the fast reactor technology in many countries. In this section the potential for B&B core implementation of the following void worth reduction approaches are evaluated:

- Leakage-based
 1. Large coolant plenum above core
 2. Reduction of H/D ratio
 3. Strongly heterogeneous designs
 4. Gas Expansion modules
- Moderation-based
 - Fuel replaced by BeO or ZrH
- Absorption-based
 1. Ca_3Na_2 permanent solid absorber
 2. ^6Li injection absorber

III.A. Leakage-based void worth reduction designs

The main focus of void worth reduction work worldwide has been devoted to leakage-based methods. This is because the main negative component of the reactivity effect of coolant voiding is the increase in neutron leakage from the core (as stated earlier, changes in coolant absorption rates can effectively be ignored). When the relative importance of neutron leakage in the core is increased, the leakage component of the void reactivity will increase and eventually dominate the positive spectral component, resulting a negative total void worth. Ideally, a leakage-based void worth reducing design should feature low leakage during standard operation, while strongly increasing leakage in the voided state. However, such ideal (for neutron economy) designs have not been identified.

Four main leakage-based void worth reduction approaches are presented and analyzed for implementation in a B&B-type core:

1. Large coolant plenum above active core

Most modern large fast reactor designs feature a large plenum of coolant above the active core region. This is an inherent feature of metallic-fueled cores because of the need for a fission gas plenum volume that is comparable to the fuel volume. Coolant voiding is likely to initiate in the plenum where the local boiling point is lowest (low pressure) and coolant temperature the highest. Voiding of

coolant outside of the active core region reduces neutron reflection back into the core and provides a way of reducing void worth without penalizing noticeably the neutron economy of the core in standard operation. In the ORB^2 B&B design, the effect of the plenum region on total void worth is, however, minor – contributing about -1% (from +13% to +12%).

2. Reduction of core height-to-diameter (H/D) ratio

The main approach in the void worth reduction research has been focused on designing the core to have a large axial neutron leakage probability through reducing the height of the active core region. This is a very effective approach and was utilized successfully in, for example, versions of the Integral Fast Reactor (IFR) designs that aimed at negative void worth.¹⁶ For minimum-burnup B&B cores this is not a viable approach due the detrimental effects it has on neutron economy. A core with negative or even small positive void worth due to reduced H/D alone cannot sustain the B&B mode of operation.

3. Strongly heterogeneous core designs

It has been shown that carefully designed strongly heterogeneous cores with different fuel heights and fuel diameters in different parts of the core can reduce the void worth without significantly damaging neutron economy. Concepts such as the “diabolo” design with an axially shorter central core region show great promise.¹⁷⁻¹⁸ Since voiding will initiate in the plenum above this central zone, both axial leakage from the central core zone as well as radial leakage from peripheral zones provide a strongly negative reactivity feedback. Unfortunately, the total nominal leakage probability from such cores is prohibitively high for B&B cores. Moreover, fuel shuffling in the B&B cores requires uniform fuel assemblies throughout the core.

4. Gas expansion module (GEM)

The first system specifically designed to reduce reactivity through leakage in accident scenarios is the GEM system developed at FFTF in the 1980s.¹⁹ GEMs are essentially empty canisters located between the outer fuel assembly row and the reflectors. They are sealed at the top but open at the bottom and filled in their upper part with a pressurized gas. During standard operation, coolant flow provides upward pressure inside the GEM that compresses the gas so that it stays above the active core region. When flow decreases in a loss of flow (LOF) accident, the gas expands into the core region and thereby reduces the neutron reflection and hence, the core reactivity. While GEMs are conceptually both simple and brilliant and worked well in the FFTF, they are effective only if the neutron importance in the outer core region is sufficiently high. This is not the case in B&B cores in which depleted uranium fuel is loaded at the core periphery. In addition, GEMs only respond to one accident scenario (LOF), and

concerns have been raised about the potential response of the core to gas leakage from the GEMs. For large B&B cores, the radial neutron leakage is minimal (1-3%), so GEMs do not have a significant effect on reactivity.

III.B. Moderation-based void worth reduction designs

The positive reactivity component of coolant voiding is due, primarily, to the fact that the fission/capture probability in the fuel actinides and the number of neutrons produced per fission (ν) rise sharply with increased neutron energy. At above 1 MeV, this rise is nearly exponential. In addition, non-actinide absorption cross-sections decrease and the fertile actinide isotopes that have energy thresholds for fission see sharply increased fission rates upon hardening of an already hard spectrum. If the neutron spectrum can be kept "soft" enough upon coolant voiding, the void worth can be greatly reduced. Because of this, core designers have proposed incorporating solid moderating material such as BeO or ZrH in fast reactor cores.²⁰ However, it is vital for the neutron economy of B&B cores to both maximize fuel volume fraction and maintain the hardest possible neutron spectrum during standard operation. Therefore, the addition of solid moderator material is not a viable option for B&B cores.

III.C. Absorption-based void worth reduction designs

The void worth can be reduced by an increase in neutron absorption either by solid-state stationary neutron absorbers or thermally actuated liquid absorbers.

1. High-energy neutron absorbers

The use of permanent solid neutron absorbers that preferentially absorb high-energy neutrons (such as Ca_3Na_2) can introduce negative reactivity during core voiding. The problem, as with moderation-based approaches, is that these systems are present and active during standard operation as well. They will impair the neutron economy and, therefore, is not a viable option for minimum-burnup B&B cores.

2. Lithium-based passive injection systems

The use of ${}^6\text{Li}$ for reactivity control was introduced along with the original B&B travelling wave reactor design by Teller et. al in 1996.³ Lithium has several unique properties that make it suitable for reactivity control. Natural lithium consists of 92.5% ${}^7\text{Li}$ and 7.5% ${}^6\text{Li}$. ${}^7\text{Li}$ is relatively neutronically benign while ${}^6\text{Li}$ is a very potent neutron poison with a high absorption (n,T) cross-section also at high neutron energies. Isotopic separation of the lithium isotopes is relatively inexpensive, and lithium stays liquid throughout sodium-cooled reactor operating temperature regimes. Because of this, lithium has been the focus of at least two innovative passive reactivity control systems.

The system devised by Teller et. al for the traveling wave reactor consists of two connected metallic compartments, one filled with ${}^6\text{Li}$ and the other with ${}^7\text{Li}$, fed by capillary tubes. The ${}^7\text{Li}$, which is permanently located within a compartment in the fuel region, expands upon a temperature increase, which in turn actuates a piston that injects ${}^6\text{Li}$ into a separate compartment located inside a coolant channel. When temperatures decrease, the ${}^6\text{Li}$ retracts down a tube and leaves the in-core compartment. In this way, a passive thermostating reactivity control system with negligible impact on core neutronics during standard operation was devised.

In 1998, Kambe et. al developed the Lithium Expansion Module (LEM) system for reactivity control for the RAPID cores.²¹ LEMs consist of one or more large reservoirs of ${}^6\text{Li}$ located above the core, with close ended tubes leading down through the active core region. During standard operation, the ${}^6\text{Li}$ in the tubes is suspended above the active core by argon gas. When temperatures increase, the ${}^6\text{Li}$ inside the reservoir expands. This pushes ${}^6\text{Li}$ down the tube and into the core region while compressing the argon gas (see Figure 4).

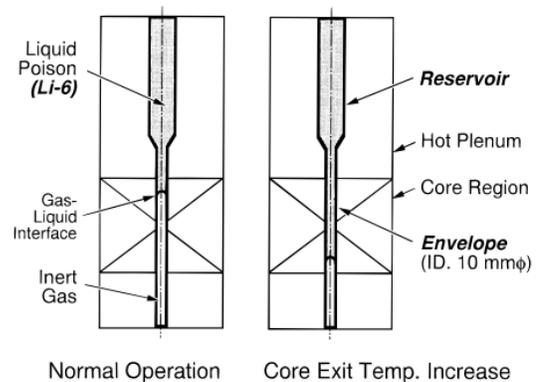


Figure 4, Operational principle of the LEM system (from ref. 21)

Lithium-expansion systems can provide large amounts of negative reactivity, operate passively and do not affect core neutronics noticeably during standard operation. Both conceptual and detailed analyses of lithium-based control systems were carried out within the scope of this work. Several different new systems based on the passive injection of ${}^6\text{Li}$ that integrate seamlessly into conventional fast reactor fuel assemblies were developed in this work specifically for use in shuffled B&B cores. This work is partially presented in Section IV, while specifics of some of the new systems developed are left out due to ongoing intellectual property rights processing.

III.D. Conclusions

Leakage-based void worth reduction approaches should not be implemented (apart from the inherently implemented large plenum above the core) in minimum

burnup B&B cores due to their impairment of the neutron economy. Moderation-based and permanent solid absorber approaches are ill suited for B&B cores for the same reason. Of all approaches analyzed, systems based on the passive injection of liquid ${}^6\text{Li}$ appear to be the only option capable of providing large amounts of negative reactivity without severely impairing the sensitive B&B neutron economy during standard operation.

IV. INHERENT REACTIVITY CONTROL BY LITHIUM-6 INJECTION

IV.A. Lithium control system design and operation

The general idea is to drive liquid ${}^6\text{Li}$, initially at a location of low reactivity worth outside the active core, into the core when temperatures increase. This is to be done passively by thermal expansion.

For the ORB² core, several new systems were designed to passively provide a strong negative reactivity response to increasing temperature that is of a larger absolute magnitude than the positive reactivity introduced by coolant voiding. While highly innovative and effective, the Teller et. al type systems were discarded for use in the ORB² core due to their reliance on the proper function of moving mechanical parts. In addition, having several thousand pipes going in and out of the core region, along with thousands of pistons seem overly complex compared to the relative simplicity of the Kambe et. al LEM-system. The LEM-system is advantageous since it does not have moving mechanical parts and relies solely on reliable physical phenomena (thermal expansion). The LEM-system does, however, rely on the stability of a liquid/gas interface, where the heavier liquid is suspended above the gas. Calculations, balancing the buoyancy force with surface tension, show that a stable interface can be upheld within a tube with a diameter such as the inner cladding diameter of a standard fuel pin. These calculations have also been validated experimentally.²¹

For the ORB² core, new systems were designed so as not to interfere with standard shuffling operations of the assemblies. This is achieved by integrating the control systems within the fuel assemblies themselves -- one miniature system in each fuel assembly.

Apart from the method of ${}^6\text{Li}$ delivery into the core and the effect on reactor shuffling and refueling operations, a number of design variables have to be defined and/or identified:

Total reactivity worth of the system – There are two responses for safety that the system can be designed for – either the ambitious aim of a negative reactivity state at coolant void, or a less extensive system that counteracts the single-phase (liquid) positive coolant density coefficient. In addition, systems can be specifically designed for passive thermostating load following. In lithium-based control

systems, regardless of the details of engineering and implementation, an estimate of the ${}^6\text{Li}$ -fraction in the active core that provides the appropriate reactivity response is needed. The calculational flow-chart for this procedure for negative void-state system is given in Figure 5. If the voided reactivity state is positive, additional volume of ${}^6\text{Li}$ corresponding to one fuel pin is added until a negative state is reached. The resulting new volume fractions are then plugged in to re-calculate the equilibrium fuel cycle. The general reactivity response of such a lithium control system designed in this work is given in Figure 6.

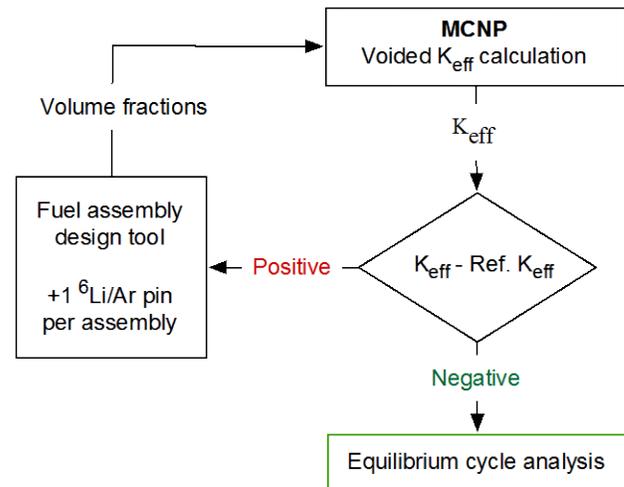


Figure 5, Generalized calculation procedure to determine B&B core parameters for negative void worth with a passive lithium-based control system.

Rate of negative reactivity injection – During certain accidents, such as the very severe transient following an abrupt loss of flow due to a shaft break in the primary pumping system, time constants for different effects are of great importance. In such an event, a temporary power/flow mismatch may raise temperatures above boiling on a timescale of less than a minute. A slowly actuated control system may not be able to provide enough negative reactivity to ensure sub-criticality at coolant boiling for this transient. The rate of negative reactivity injection of the system depends on the efficiency of the heat transfer and properties and size of the reservoir. Detailed thermal-hydraulic design of the reservoir heat-pipes may increase the speed of actuation. A thorough time-dependent transient analysis for these systems is high priority for our future work in B&B inherent safety research.

Temperature range of actuation – The change in coolant outlet temperature that is needed to fully actuate the control system is a design parameter that determines the volume of the reservoir driving the lithium insertion into the core. For a system designed specifically to

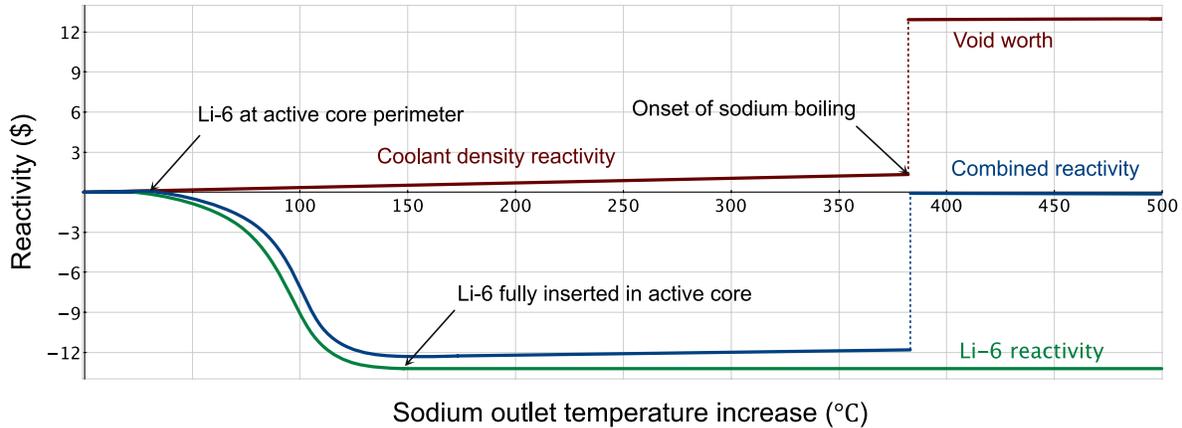


Figure 6, Reactivity response plot of a lithium-based control system for a sodium-cooled core. This system is designed to counteract +12.5\$ of reference void worth to 0\$ void worth with a full-stroke actuation temperature span of 125°C.

counteract coolant void reactivity, the maximum actuation temperature should be well below the coolant boiling point. Finding the optimal minimum temperature at which the system starts to have a major impact on reactivity (when the ${}^6\text{Li}$ reaches the active core region) is not trivial. Ideally, the ${}^6\text{Li}$ should stay in a low flux region during temperature transients that can be expected during standard operation ($\pm 15^\circ\text{C}$). Such a design may however cause a time delay of actuation that will interfere with the requirements for initial negative reactivity insertion rates.

Consequences of a system failure – While system failures such as major leakages are highly unlikely events, the effects of such failures must be benign in nature for any well-designed safety system. For instance, a break in the gas/liquid interface of a fully actuated LEM system will lead to gas bubbles traveling up through the core adding positive reactivity. New ${}^6\text{Li}$ delivery methods have the potential of eliminating such concerns.

In addition, cost must be considered as a design variable if systems require the use of rare and expensive materials or are expected to increase difficulty in core operation (capacity factor decrease). A sample design of a new type of LEM-based ${}^6\text{Li}$ control system and its impact on the performance on the ORB^2 core are presented in the following subsection.

IV.B. Fuel assembly integrated LEM-system

In this new type of LEM system, a reservoir of ${}^6\text{Li}$ is located above the fission gas plenum in each fuel assembly. Pipes are connected at the top to the top of the reservoir and at the bottom to a set number of rods that are emptied of fuel and filled with Argon gas. Implemented in this way, only very minor changes are needed in the fuel assembly design – the addition of a ${}^6\text{Li}$ reservoir and a small number

of pipe connections. Fuel shuffling operations are not affected at all, while the core does suffer a modest increase in core pressure drop and a small decrease in fuel volume fraction. In the reference design, the rods containing the Argon/ ${}^6\text{Li}$ system are simply standard fuel rods that have their fuel removed.

For the ORB^2 design, which can be considered an upper bound case due to its very high void worth, the volume fraction of ${}^6\text{Li}$ needed to offset the reactivity of full core voiding is about 0.8%. In the 169-pin reference fuel assembly design, this means replacing the fuel in 3 standard pins with Argon/ ${}^6\text{Li}$. A system designed to merely counteract the positive coolant density coefficient only requires the replacement of a single pin per assembly. The general design concept of this system is shown in Figure 7.

Neglecting the expansion force needed to compress the small volume of argon gas in the pipe, the volume of the ${}^6\text{Li}$ reservoir is simply given by Equation 1,

$$V_{Reservoir} = \frac{V_{Displacement} - V_{Plenum} \cdot \alpha_{Li} \cdot \Delta T}{\alpha_{Li} \cdot \Delta T} \quad (1)$$

where V denotes the volume of ${}^6\text{Li}$ of different components, α is the volumetric expansion coefficient of liquid lithium and ΔT is the temperature range of actuation for the system. For the ORB^2 fuel assembly implementation and a 150°C actuating range, the reservoir height needed on top of the gas plenum is about 60 cm. Careful reservoir design work is needed to ensure that coolant flow out of the assembly is not obstructed and increases the pressure drop are kept to a minimum. The coolant volume fraction in the expansion reservoir (physically separated from the ${}^6\text{Li}$) should remain about 20%, same as in the rest of the assembly. The exact design of piping leading the coolant through the reservoir will be the focus of future work. The fuel volume fraction in the

active core is decreased from 40.3% in the reference case to 39.6% with the system implemented.

The average discharge burnup required for B&B operation, which is the neutronic benchmark for this type of core, is hardly affected (it increases from 21.5% to 22%). To put this in perspective, it is not possible to achieve negative void worth with any other strategy outlined in Section III while maintaining a B&B mode of operation in a sodium-cooled, metallic DU-fueled core without significantly exceeding the minimum ~20% burnup.

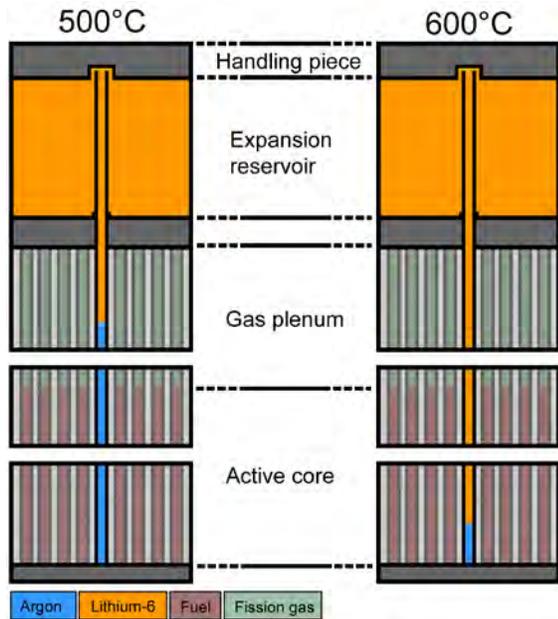


Figure 7, Fuel assembly integrated LEM-system seen from the side, not to scale. Standard (left, 500°C coolant outlet temperature) and transient (right, 600°C) states shown.

IV. CONCLUSIONS

Conventional approaches for reducing the void worth of fast reactors are not suitable for minimum burnup B&B cores due to the significant neutron economy impairment they cause. Careful optimization of B&B shuffling schemes and the use of a large upper sodium plenum can provide significant void worth reduction (-2\$ and -1\$ respectively) but are insufficient to assure passive safety.

A control system based on the passive injection of ${}^6\text{Li}$ neutron poison during temperature excursions provides a strong negative reactivity feedback without significantly hurting the neutron economy during standard operation. A minimum burnup B&B reactor with metallic fuel and sodium coolant was developed that features negative void worth using a new lithium-based passive control system. The analysis shows that relatively simple, inherently passive systems based on ${}^6\text{Li}$ -injection can be implemented in B&B cores with minimal impact on core neutronics, operation and cost, while giving safety margins that equal

or exceed those of smaller and leakier fast reactor core designs.

Future work will focus on new passive methods of ${}^6\text{Li}$ injection delivery and detailed transient analysis of accident scenarios with such systems installed. The core response with such systems to transients, such as loss of flow, loss of heat sink and transient overpower needs to be analyzed for reliable assessment of the potential for inherent safety with these new systems.

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- DRAFT -

Assessment of waste characteristics of Breed and Burn reactors

C. DiSanzo, F. Heidet and E. Greenspan,
University of California, Berkeley

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1. Introduction

Two of the findings of the Breed and Burn (B&B) reactors study at UC Berkeley are that the minimum average burnup required for establishing the B&B mode-of-operation in a large core is 20% FIMA (Fissions per Initial Metal Atom) while the maximum attainable average burnup is 55% FIMA [1]. In arriving at these values it was assumed that whenever the fuel reaches its radiation damage limit, it will be taken out of the core, be “reconditioned”. A couple of reconditioning processes have been examined [2-3]: the melt-refining process [3] and the AIROX process [2]. After reconditioning the fuel is recycled back to the core. The purpose of the present study is to compare waste characteristics of B&B reactors which discharge their fuel at the bounding burnup levels:

- a. 20% average burn-up. A B&B reactor with 12 radial batches, each divided in 3 axial zones for burn-up simulations [1]. The cycle length is 2.06 years, divided into 4 time-steps.
- b. 55% average burn-up. The core consists of 8 radial batches, each divided into 3 axial zones for burn-up simulations [1]. The cycle length is 8.8 years (divided in 12 time-steps for simulations).

Both scenarios pertain to the same core; its dimensions and composition are defined in Section 2. In the following, we will refer to these scenarios as the 20% B&B reactor and the 55% B&B reactor. Section 3 briefly describes the fuel reconditioning processes examined while Section 4 compares the fuel and waste mass flows and composition for the five energy system being examined. Sections 5 through 8 compare selected characteristics of the discharged fuel and waste streams.

All the results of B&B reactors, reported in this summary, were obtained using the coupled MCNP 1.51 transport and ORIGEN 2.2 depletion calculations.

2. Large B&B reactors Design

A layout of the B&B reactors under study is given in Fig. 1. Dimensions and compositions of the regions of B&B reactors are given in Table I and Table II. This data is common to both of the studied reactors; the different scenarios are obtained by using different fuel management scheme and cycle length.

Fig. 2 shows the fuel management scheme of the 55% reactor. This reactor uses 8 fuel batches with an outward to inward shuffling scheme. These studies [1] assumed that material R&D will result in cladding that could withstand an average burn-up of 20% so that the fuel will be reconditioned when accumulating an average batch burnup of ~20% and 40%. Specifically, it had been assumed that a fuel batch is reconditioned at the end of fourth cycle (i.e. batch #4 in Fig. 3) and at the end of the sixth cycle (i.e. batch #6 in Fig. 3) in the reactor, when it accumulated an average burnup of, respectively, 20% and 41.4% FIMA.

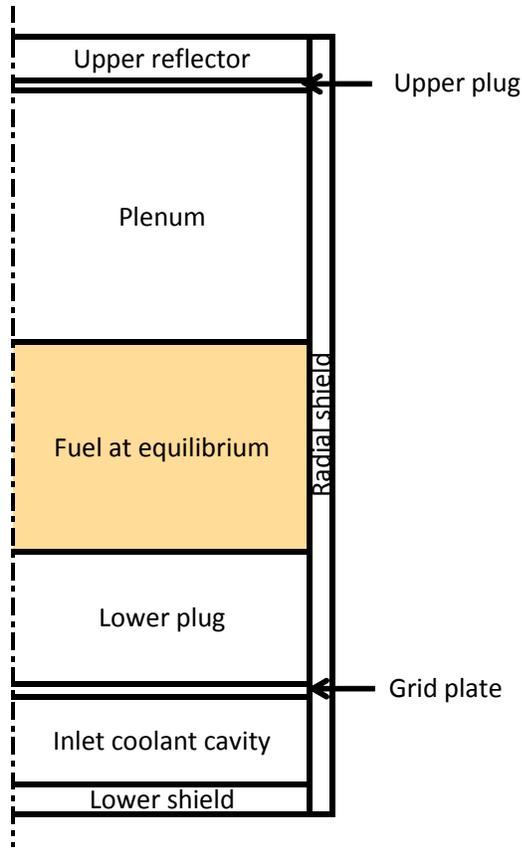


Fig.1 Simplified vertical cut through the B&B reactors studied

Table I. Dimensions and compositions of the regions of the B&B reactors studied

Region	Height (cm)	Thickness (cm)	Material (Volume %)	T [K]
Upper reflector	34.93	233.18	50% HT9- 50% Na	783
Upper end plug	2.54	200.74	22% HT9 - 78% Na	783
Plenum	250	200.74	22% HT9 - 28% Na	783
Inner duct	209.36	8.48	22% HT9 - 28% Na	783
Core	209.36	192.26	37.5% Fuel - 22% HT9 - 28% Na	800
Lower end plug	20.32	200.74	22% HT9 - 78% Na	628
Grid plate	5.18	233.18	50% HT9 - 50% Na	628
Coolant inlet	60	233.18	22% HT9 - 78% Na	628
Lower shield	20	233.18	43.1% B4C - 29.7% HT9 - 27.2% Na	628
Radial reflector	482.22	32.44	50% HT9 - 50% Na	628
Radial shield	602.33	15.22	43.1% B4C - 29.7% HT9 - 27.2% Na	628

Table II. Volume fractions of fuel

Fuel	37.5%
Gap	12.5%
Clad	22.0%
Coolant	28.0%

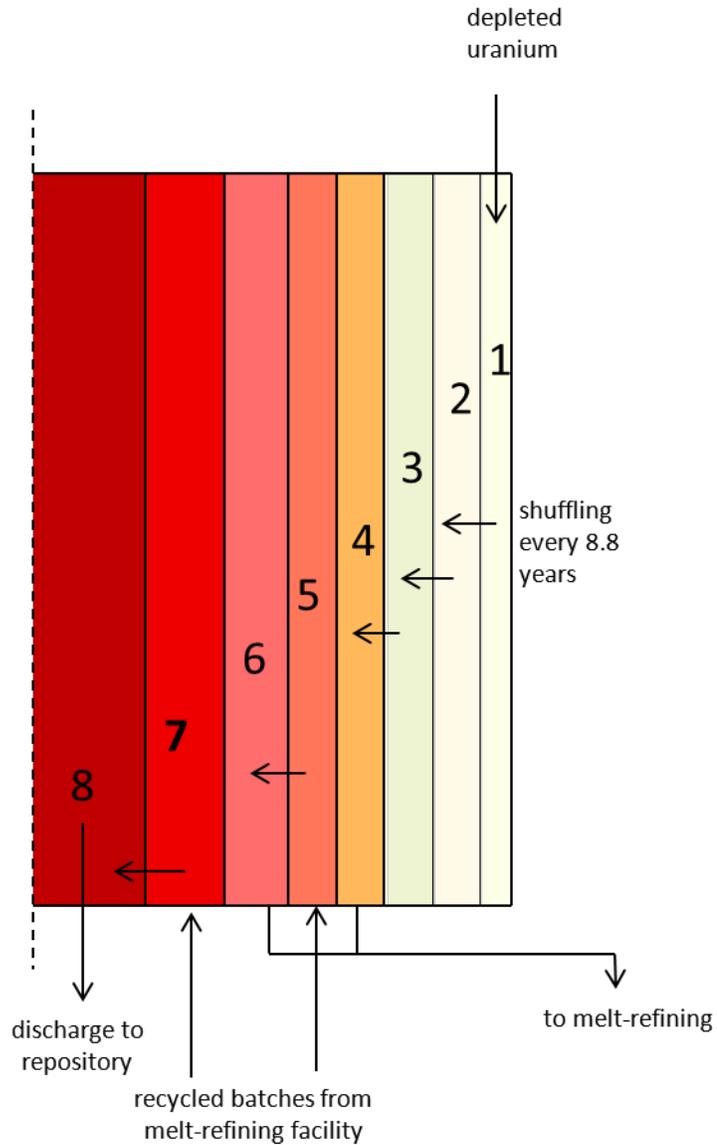


Fig.2 Shuffling scheme of the 55% B&B reactor; depleted uranium is charged in batch 1 and shuffled inward to the inner batch, where it is discharged after reaching 55% FIMA. Batch #4 and #6 are sent to the melt-refining facility; recycled batches are moved to position #5 and #7 respectively.

Fig. 3 shows the fuel management scheme of the 20% B&B reactor. This shuffling scheme is optimized for the reactor to operate at 20% FIMA. The 20% average burn-up is the minimum discharge burn-up required for establishing and maintaining a B&B mode of operation for the assumed core composition. It is assumed that future cladding materials will be able to withstand 20% burn-up thus eliminating the need for reconditioning the fuel for this reactor.

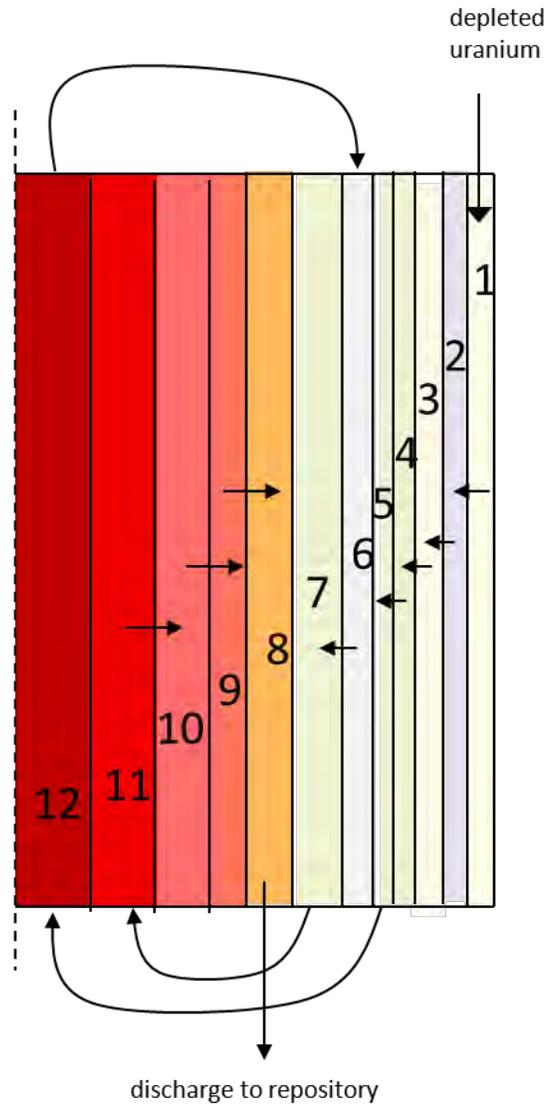


Fig.3 Shuffling scheme of the 20% B&B reactor; depleted uranium is charged in batch 1 and shuffled as shown by the arrows.

3. Fuel reconditioning and fuel cycle scenarios

The following fuel reconditioning scenarios are examined for the 55% reactor:

- 1) Melt refining whenever 20% BU is reached. The recovery efficiency is that reported in the EBR-II project report [3]
- 2) Melt refining whenever 20% BU is reached; the recovery efficiency of all actinides is 99%.
- 3) Melt refining whenever 20% BU is reached; the recovery efficiency of all actinides is 90%.
- 4) AIROX reprocessing whenever 20% BU is reached, using the reference AIROX recovery efficiency [2]

The 20% reactor is assumed to discharge its fuel without reconditioning.

3.1 Melt Refining During the melt refining (MR) process the gaseous fission products (FPs) accumulated in the fuel and in the fission gas plenum of the fuel rod are released. It is assumed that the gaseous FPs are released when the cladding is breached and are collected in special filters. After collection, they are recovered from the filters and solidified in a waste form that is suitable for disposal. Some of the solid FPs are partially removed from the melt by oxidation with the zirconia of the crucible. It is assumed, after the results of the EBR-II project, that this process can remove [3] nearly 100% of Br, Kr, Rb, Cd, I, and Cs, and 95% of Sr, Y, Te, Ba and rare earths (lanthanides). Moreover, 95% of Th and Am are also oxidized with zirconia and removed from the fuel. The materials collected in the crucible will have to be periodically removed from the crucible and disposed in a suitable waste form. Therefore the B&B reactor waste is made of 3 components: 1) B&B fuel discharged at the end of cycle; 2) gaseous fission products embedded in a proper waste form (possibly, borosilicate glass); 3) waste from the melt-refining process (possibly, borosilicate glass). Waste streams 2 and 3 could be combined during waste form fabrication. The recovery efficiencies reported above are used in scenario 1 analysis. As it is likely that further R&D will result in significantly more efficient melt refining process, we are evaluating a couple of bounding scenarios: 1% (scenario 2) and 10% (scenario 3) of all actinides are not recovered. We will refer to scenario 2 and 3 as, respectively, the 1% melt-refining and the 10% melt-refining scenario.

3.2 AIROX The AIROX process was originally developed for an oxide fuel [2]. It removes, in gaseous form, 100% of T, C, Kr, Xe and I, 90% of Cs and Ru and 75% of Te and Cd. It is assumed that a similar process could be developed for metallic fuel, thus providing a most optimistic scenario for fuel reconditioning – no loss of actinides.

3.3 Reference systems The characteristics of the B&B waste are compared to those of a three reference systems: (1) a PWR operating on the once-through cycle; (2) a sodium-cooled Advanced Burner Reactor (ABR) having a conversion ratio of $CR=0.5$ [4] and, (3) an Advanced Recycling Reactor (ARR) having a conversion ratio $CR=1$ [4]. The reference PWR uses 4.5% enriched fuel up to 50,000 MWd_{th}/MT , while the ABR and ARR wastes are derived from ANL data for reactor at equilibrium assuming 99% recovery of actinides in the recycling process (possibly a pyro-metallurgical fuel reprocessing). Fuel from ABR and ARR is discharged with a burnup of, respectively, 73,000 MWd_{th} /MT and 131,900 MWd_{th} /MT [4].

Table III summarizes the burn-up characteristics of the reactor systems inter-compared while Table IV summarizes the fraction of the actinides and FPs assumed recycled for the different assumed processes. It is also assumed that 75% of the gaseous FPs generated in the B&B reactor fuel migrate out from the

fuel and accumulates in the fission-gas plenum. The fission products that are removed this way are: H, He, N, O, F, Ne, Cl, Ar, Kr, Xe and Rn.

Table III. Burn-up and thermal conversion efficiency of studied reactors

	PWR	55% B&B	20% B&B	ARR	ABR
final burn-up (MWd _{th} /MT)	50,000	540,930	189,455	73,000	131,900
Fuel cycle	Once-thru	Once-thru via reconditioning	Once-thru	Recycled	Recycled
thermal conversion efficiency	33%	40%	40%	40%	40%

Table IV. Actinides and FPs removal fractions for AIROX and melt-refining process

Actinides and FPs removal fractions				
	AIROX	melt-refining		
		reference	1%	10%
U	-	-	1%	10%
Th	-	95%	95%	95%
Am	-	95%	95%	95%
other TRU	-	-	1%	10%
FPs	100%	100% Br,Kr,Rb,Cd,I,Cs		
	T,C,Kr,Xe, I			
	90% Cs,Ru	95% Sr,Y,Te,Ba, La-Lu		
	75% Te,Cd			
gas FPs	100% H,He,N,O,F,Ne,Cl,Ar,Kr,Xe,Rn			

Reactor simulations were performed using MCNP 1.51 coupled with ORIGEN 2.2. As depleted uranium is charged into the reactor and recycling takes place at the pre-determined steps the reactor reaches an equilibrium composition.

It is assumed that the MCNP calculated one group cross-sections at equilibrium, using reference melt-refining, do not vary when using other recycling processes, so that it is possible to use one-group cross-sections previously calculated in all the scenarios. Since the one-group cross sections for each single nuclide varies weakly with the reactor composition, while composition varies only of about 10% between a scenario and another, the constant one-group assumption is legit. When the system is simulated a mass flow of materials is established at equilibrium.

4. Mass Balance

Tables V, VI and VII summarize the fuel specific mass flow, in Kg/GWeY, for each of the reactor systems examined. Included in the tables are the important actinides as well as Sr-90 and Cs which are responsible for most of the short-term radio-toxicity and decay heat of the waste. The data shown in these tables is for the equilibrium composition cores; it does not include the initial fissile fuel feed that is required for starting a new B&B reactor. The mass flow is attained normalizing the feed and the materials discharged from the reactor at the beginning and end of each cycle to the energy produced during the cycle.

Table V. Mass flow in kg/GWeYr of PWR, 20% B&B and 55% B&B (reference recovery). The streams of B&B reactor are: discharge, i.e. the discharged fuel, MR waste, i.e. waste from melt-refining process, and gas FP, i.e. the gaseous fission products accumulated in the plenum.

	PWR		B&B-20%			B&B-55% - reference			
	charge	discharge	charge	discharge	gas FP	charge	discharge	MR waste	gas FP
mass	2.21E+04	2.21E+04	4.82E+03	4.72E+03	9.78E+01	1.69E+03	1.23E+03	3.56E+02	9.64E+01
U	2.21E+04	2.07E+04	4.82E+03	3.41E+03	-	1.69E+03	6.24E+02	-	-
Pu		2.25E+02	-	4.76E+02	-	-	1.30E+02	-	-
Np		1.86E+01	-	3.49E+00	-	-	7.23E-01	-	-
Am		5.38E+00	-	1.43E+00	-	-	2.24E+00	2.29E+00	-
Cm		2.03E+00	-	1.30E-01	-	-	3.95E-01	-	-
U-235	9.95E+02	1.56E+02	9.64E+00	7.17E-01	-	3.37E+00	4.92E-02	-	-
Pu-238		6.97E+00	-	1.70E+00	-	-	1.27E+00	-	-
Pu-239		1.10E+02	-	3.80E+02	-	-	8.18E+01	-	-
Pu-240		5.89E+01	-	8.55E+01	-	-	3.98E+01	-	-
Pu-241		3.31E+01	-	8.05E+00	-	-	4.59E+00	-	-
Pu-242		1.59E+01	-	1.31E+00	-	-	2.35E+00	-	-
Am-241		1.49E+00	-	1.24E+00	-	-	1.79E+00	1.90E+00	-
Np-237		1.73E+01	-	2.94E+00	-	-	6.61E-01	-	-
Cm-245		6.03E-02	-	5.25E-03	-	-	4.40E-02	-	-
Sr-90		1.75E+01		7.05E+00	-	-	1.59E+00	4.62E+00	-
Cs		9.44E+01		9.62E+01	-	-	2.31E+01	7.01E+01	-
U-235/U	4.50%	0.75%	0.20%	0.02%	-	0.20%	0.01%	-	-
HM	2.21E+04	2.10E+04	4.82E+03	3.89E+03	-	1.69E+03	7.57E+02	2.29E+00	-
TRU		2.52E+02	-	4.81E+02	-	-	1.33E+02	-	-
TRU/HM		1.20%	-	12.37%	-	-	17.58%	-	-
Fiss. Pu		1.43E+02	-	3.88E+02	-	-	8.64E+01	-	-
Fiss. Pu/HM		0.68%		9.97%	-	-	11.41%	-	-
Fiss. Pu/Pu		63.69%		81.43%	-	-	66.55%	-	-

Table VI. Mass flow, in kg/GWeYr, of three variants of 55% B&B reactor: 1) MR with 1% actinides lost in the process; 2) MR with 10 % loss; and 3) using AIROX reprocessing.

	B&B-55% - 1% rec.				B&B-55% - 10%			B&B-55% - AIROX		
	charge	discharge	MR waste	gas FP	discharge	MR waste	gas FP	discharge	AIROX waste	gas FP
mass	1.69E+03	1.21E+03	3.79E+02	9.64E+01	1.02E+03	6.72E+02	9.65E+01	1.36E+03	2.28E+02	9.61E+01
U	1.69E+03	6.04E+02	2.00E+01	-	4.40E+02	1.89E+02	-	6.25E+02	-	-
Pu	-	1.26E+02	3.16E+00	-	9.52E+01	3.00E+01	-	1.31E+02	-	-
Np	-	7.02E-01	2.20E-02	-	5.20E-01	2.09E-01	-	7.33E-01	-	-
Am	-	2.17E+00	2.27E+00	-	1.61E+00	2.09E+00	-	2.49E+00	-	-
Cm	-	3.99E-01	3.12E-03	-	4.40E-01	3.25E-02	-	5.02E-01	-	-
U-235	3.37E+00	4.73E-02	2.81E-03	-	3.12E-02	2.71E-02	-	5.36E-02	-	-
Pu-238	-	1.24E+00	1.85E-02	-	9.49E-01	1.73E-01	-	1.50E+00	-	-
Pu-239	-	7.93E+01	2.34E+00	-	5.81E+01	2.21E+01	-	8.20E+01	-	-
Pu-240	-	3.88E+01	7.10E-01	-	3.03E+01	6.73E+00	-	4.01E+01	-	-
Pu-241	-	4.51E+00	7.37E-02	-	3.79E+00	7.09E-01	-	4.62E+00	-	-
Pu-242	-	2.33E+00	2.13E-02	-	2.09E+00	2.09E-01	-	2.47E+00	-	-
Am-241	-	1.73E+00	1.88E+00	-	1.21E+00	1.72E+00	-	1.97E+00	-	-
Np-237	-	6.39E-01	1.92E-02	-	4.58E-01	1.81E-01	-	6.70E-01	-	-
Cm-245	-	4.47E-02	2.40E-04	-	5.24E-02	2.63E-03	-	6.07E-02	-	-
Sr-90	-	1.59E+00	4.62E+00	-	1.59E+00	4.62E+00	-	3.33E+00	-	-
Cs	-	2.30E+01	7.01E+01	-	2.27E+01	6.99E+01	-	2.30E+01	6.55E+01	-
U-235/U	0.20%	0.01%	0.01%	-	0.01%	0.01%	-	0.01%	-	-
HM	1.69E+03	7.34E+02	2.54E+01	-	5.38E+02	2.22E+02	-	7.59E+02	-	-
TRU	-	1.30E+02	5.46E+00	-	9.78E+01	3.24E+01	-	1.34E+02	-	-
TRU/HM	-	17.66%	21.47%	-	18.18%	14.60%	-	17.71%	-	-
Fiss. Pu	-	8.38E+01	2.41E+00	-	6.19E+01	2.28E+01	-	8.66E+01	-	-
Fiss. Pu/HM	-	11.42%	9.49%	-	11.51%	10.31%	-	11.41%	-	-
Fiss. Pu/Pu	-	66.40%	76.29%	-	65.01%	76.25%	-	66.30%	-	-

Figure 4 shows the flow streams for the 55% B&B reactor. Every 8.8 years batch # 8 is discharged at 55% FIMA and melt-refining or AIROX waste streams are generated from the reprocessing of batch #4 and #6. The latter include the waste stream of gaseous FPs; their mass is 6% of the charged fuel mass. The total produced waste is normalized to the electrical energy production during the cycle (in this case 540,930 MWdth/MT or 216,372 MWde/MT). For the ARR and ABR the waste stream consists of only the FPs and unrecovered actinides (1% loss) each time the fuel is recycled. The waste stream for the PWR and 20% B&B reactor is just the discharged fuel.

Table VII. Mass flow, in kg/GWeYr, of the ABR and ARR [4].

	ARR-ANL		ABR	
	Charge	Discharge	Charge	Discharge
mass	1.25E+04	1.25E+04	6.71E+03	6.71E+03
U	1.04E+04	9.46E+03	4.47E+03	3.94E+03
Pu	1.62E+03	1.65E+03	1.97E+03	1.60E+03
Np	1.10E+01	1.20E+01	4.77E+01	2.65E+01
Am	3.50E+01	3.30E+01	1.42E+02	1.16E+02
Cm	9.00E+00	1.00E+01	6.48E+01	6.82E+01
U-235	4.00E+00	2.00E+00	3.40E+00	2.40E+00
Pu-238	1.80E+01	1.80E+01	7.42E+01	6.41E+01
Pu-239	1.07E+03	1.09E+03	8.61E+02	6.54E+02
Pu-240	4.43E+02	4.50E+02	6.86E+02	5.94E+02
Pu-241	5.40E+01	5.70E+01	1.44E+02	1.03E+02
Pu-242	3.70E+01	3.70E+01	2.09E+02	1.82E+02
Am-241	2.30E+01	2.10E+01	7.05E+01	5.09E+01
Np-237	1.10E+01	1.06E+01	4.77E+01	2.56E+01
Cm-245	1.55E+00	1.57E+00	1.17E+01	1.19E+04
Sr-90	-	7.89E+00	-	6.75E+00
Cs	-	1.08E+02	-	9.60E+01
U-235/U	0.04%	0.03%	0.08%	0.06%
HM	1.21E+04	1.12E+04	6.70E+03	5.75E+03
TRU	1.68E+03	1.70E+03	2.23E+03	1.81E+03
TRU/HM	13.86%	15.25%	33.26%	31.44%
Fiss. Pu	1.13E+03	1.14E+03	1.98E+03	1.60E+03
Fiss. Pu/HM	9.30%	10.24%	15.00%	13.16%
Fiss. Pu/Pu	69.33%	69.35%	50.92%	47.39%

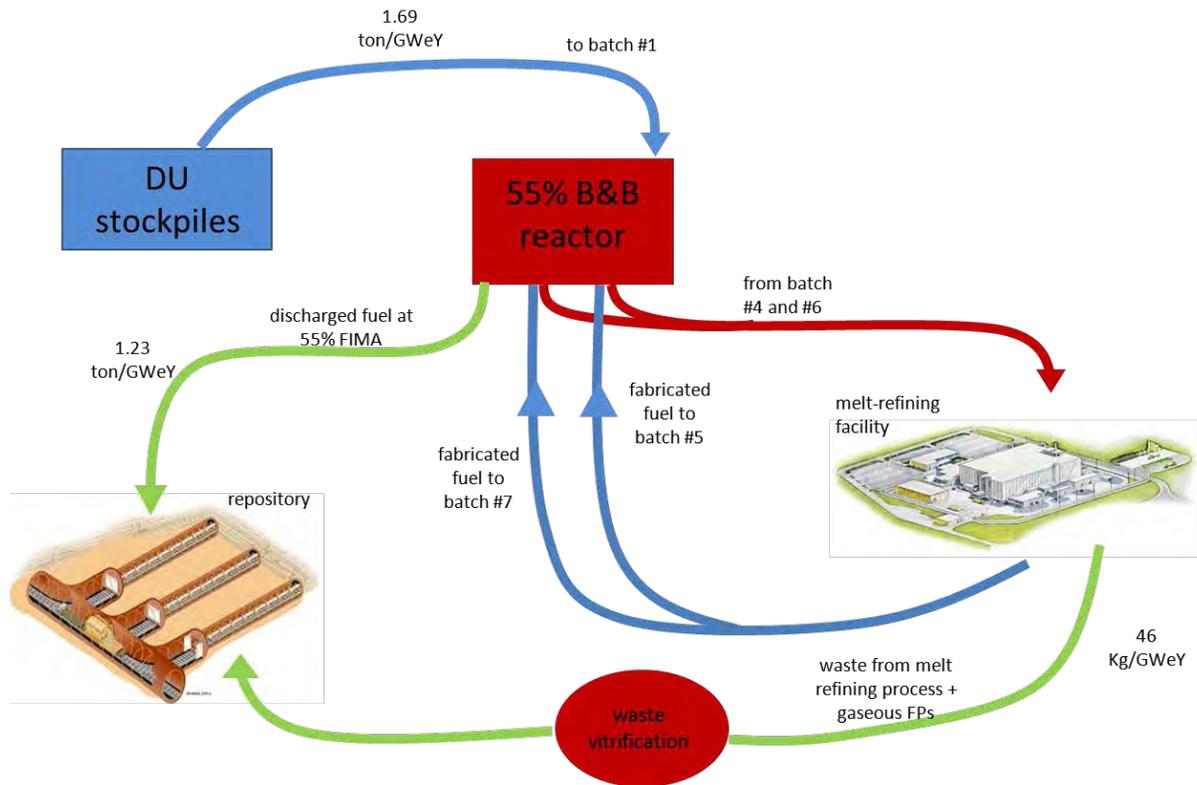


Figure 4. Fuel cycle of the 55% B&B reactor using melt-refining

Table VIII compares selected fuel cycle characteristics of the reactor systems excluding that of the ABR the primary mission of which – LWR TRU transmutation, is different. It is observed that the B&B cores feature the lowest specific HM loading – i.e., the heavy metal that needs to be loaded into the core per unit of electricity the core generates. Although the table says that the specific HM loading into the 55%B&B core is nearly 1/3 that into the 20%B&B core, the fuel reconditioning and recycling required after every 20% FIMA will actually make the 55%B&B specific HM loading very similar to that of the 20%B&B core.

The natural (depleted, in case of the fast reactors) uranium utilization of the B&B reactors far exceeds that of the PWR but do not match that attainable in fast breeder reactors that recycle their fuel, after reprocessing, unlimited number of times.

On the other hand, the specific inventory of fissile plutonium discharged from the B&B cores is significantly smaller than that discharged from the ARR. The fissile Pu specific inventory of the 55%B&B is even smaller than that of the PWR. The fissile Pu isotopes (239+241) fraction in the plutonium in the fuel discharged from the 55%B&B core is only slightly higher than that from PWR, but slightly lower than that of the ARR. However, the 20% B&B discharged fuel has a significantly higher fissile Pu fraction.

Relative to a PWR, the net amount of plutonium accumulated per GWe-Year of electricity generated is close to one-half for the 55%B&B reactor, but approximately double in case of the 20%B&B reactor.

There is practically no Pu accumulation in the ARR, while there is a net plutonium destruction in the ABR (not shown in Table VIII).

Table VIII Comparison of selected fuel cycle characteristics of the 20%B&B, 55%B&B, ARR and PWR

Characterstic	PWR	ARR	20%B&B	55%B&B
Specific HM loading (Kg/GWeY)	2.21E+4	1.25E+4	4.72E+3	1.69E+3
Loaded fuel type	Enriched U	Recy U+TRU	Depleted U	Depl+Recon U
Natural uranium utilization (%)	0.6	→99	20	55
Specific fissile Pu discharge (Kg/GWeY)	143	1140	388	86
Fissile Pu/Pu (%)	63.7	69.3	81.4	66.6
Amount of Pu generated (Kg/GWeY)	224	~0	477	129

The following sections compare additional waste characteristics of the reactor systems examined.

5. Ingestion Radiotoxicity

The ingestion radiotoxicity is the main measure to assess the hazard of the waste from a nuclear fuel cycle. It is a measure of how much the waste needs to be diluted in water so that it is not dangerous for the population. This section compares the specific ingestion radiotoxicity of the 5 reactor systems; it is measured in m³ of water by which the waste generated per GWe-yr of generated electricity needs to be diluted.

Fig. 5 compares the total radiotoxicity of the fuel and waste streams discharged from the different reactor systems examined. Time zero is taken to be the moment of fuel discharged from the core and, in case of the 55%B&B reactor, the associated waste streams from fuel reconditioning. No cooling time and no reprocessing time are assumed. In this way it is possible to fairly assess the different reactor systems and fuel cycles, independently from the particular reprocessing technology used. In fact, different cooling times might be needed for the different technologies inter-compared. Since the discharged fuel and waste streams need to be managed from the moment they are created, an analysis without a cooling time will show up the intrinsic differences between the characteristics of the waste from the different reactor systems.

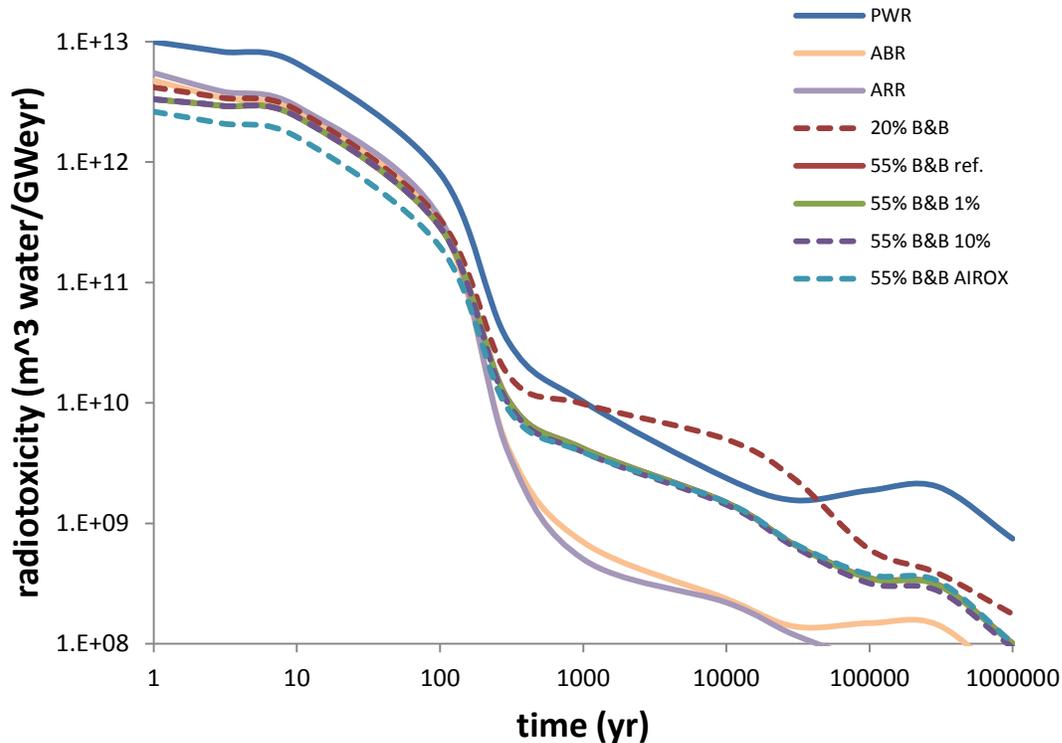


Fig.5 Total specific radiotoxicity for the studied reactor fuel cycles

Fig. 5 shows that the total specific radiotoxicity of all the fast reactor systems examined is smaller than that of PWR with one exception – the radiotoxicity of the 20% B&B reactor around the 10,000 year time range. This is due, in particular, to the contribution of Pu-239 -- the specific Pu-239 content in the 20%B&B reactor discharged fuel is ~3.5 times that in PWR (Table V). The 55%B&B reactor has a lower long-term specific radiotoxicity because their specific Pu-239 and other TRU inventories are smaller. Notice that the specific radiotoxicity of all four 55%B&B scenarios is very similar. This is due to the fact that the sum of the actinides in the discharged fuel and in the reconditioning process waste stream is practically the same, regardless of the fraction that remains in the reconditioning waste streams. The specific radiotoxicity of both the ARR and ABR fast reactors is significantly lower, after several hundred years, than the specific radiotoxicity of all the other systems. This low long-term radiotoxicity level is due to the fact that these reactors keep recycling all their actinides, with the exception of 1% that gets to the reprocessing waste stream. Up to several hundred years from discharge, the specific radiotoxicity of all the B&B reactors is very similar to that of the ARR and ABR with the exception of the 55%B&B scenario that reconditions its fuel using an AIROX-like process; the latter is smaller.

The lower short-term specific radiotoxicity of the 55%B&B reactor that reconditions its fuel using an AIROX process is due to the fact that the AIROX process does not separate Sr-90 from the fuel while the melt-refining process does. As shown in Fig. 6, Sr-90 is the dominant contributor to the short term radiotoxicity of both the PWR and the reference 55% B&B reactor (which uses the melt-refining

process). In the 55% AIROX scenario the Sr is recycled with the fuel back to the core. As long as the Sr-90 resides in the core, it does not contribute to the radiotoxicity of the waste streams and discharged fuel. While in the core the Sr-90 keeps decaying and, as illustrated in Fig. 7, its concentration levels off towards an equilibrium value. The sum of the three Sr-90 peaks in MR fuel cycle shown in Fig. 7 makes the Sr-90 inventory in MR waste. This sum is significantly larger than the amount of Sr-90 inventory discharged with the fuel that is reconditioned with the AIROX process. The inventory differences are also reported in Table V -- the specific Sr-90 discharge with the MR process is nearly twice that in the AIROX process.

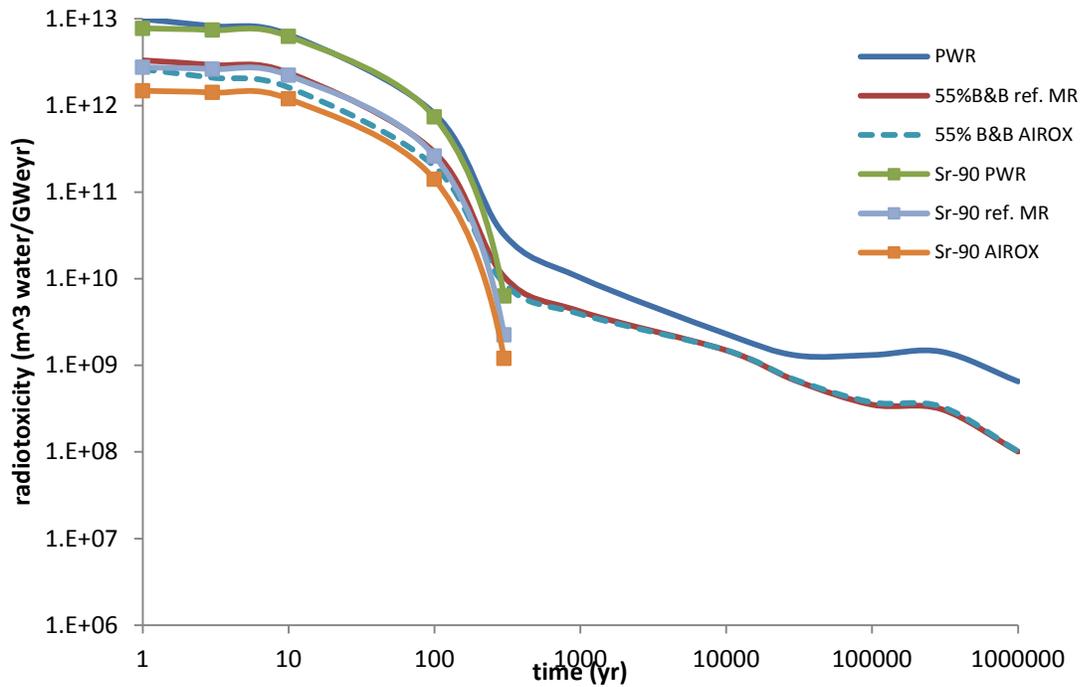


Fig.6 Total radiotoxicity and Sr-90-only radiotoxicity for PWR once-through, 55%B&B AIROX, and 55% B&B using reference MR. It is shown that, for PWR and MR, Sr-90 radiotoxicity is the major component of total short-term radiotoxicity.

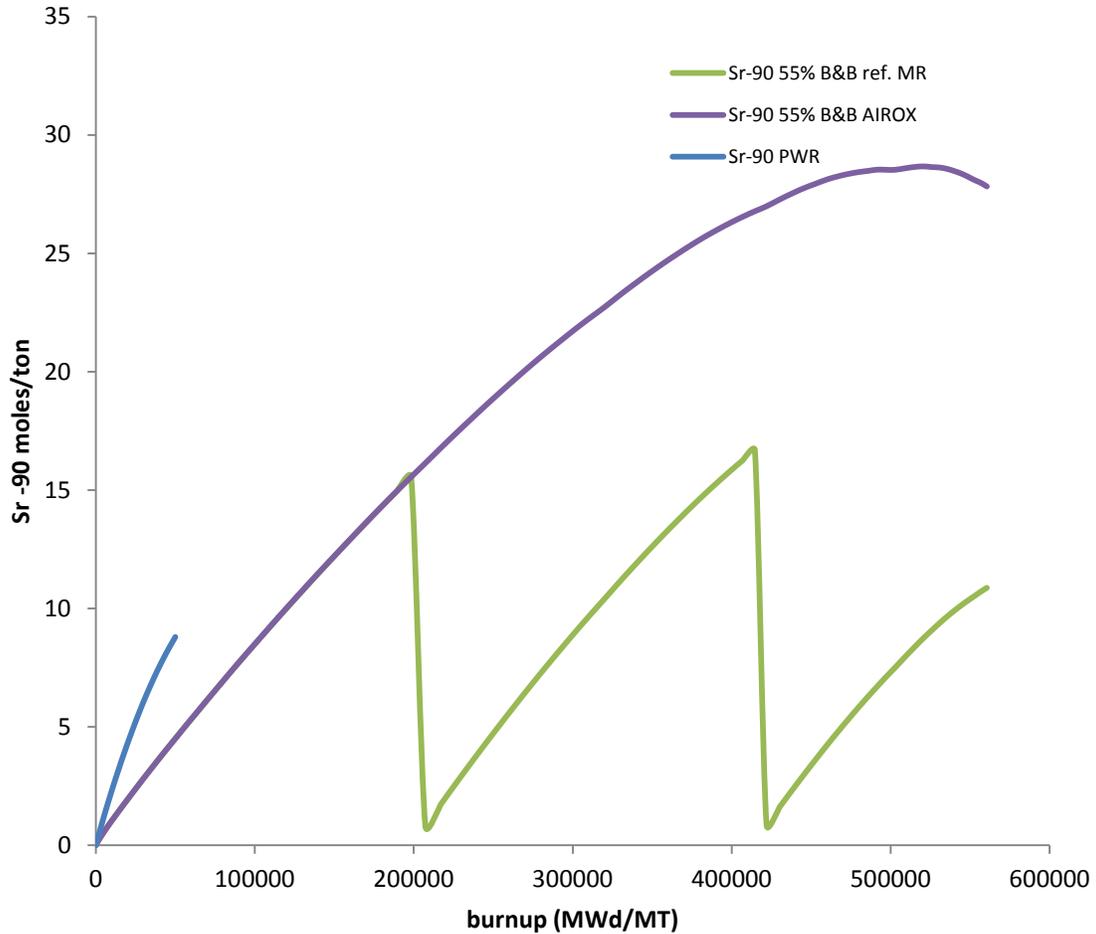


Fig. 7 Sr-90 concentration evolution with burn-up in PWR, 55%B&B ref. MR and AIROX cycle.

Sr-90 is also responsible for the higher short-term radiotoxicity of the PWR discharged fuel (Fig. 5). As shown in Fig. 7, the Sr-90 build-up rate in PWR fuel is higher than in fast reactors. Contributing to this difference are the relative short fuel residence time (relative small fraction of the Sr-90 decays before fuel discharge, different fission yield, as well as smaller thermal conversion efficiency).

Fig. 8 shows the contribution of the discharged fuel and of the waste streams for the two 55% B&B reactor systems – one using the reference melt-refining and the other using AIROX for fuel reconditioning. In the long-term, the discharged fuel for AIROX and MR has essentially the same radiotoxicity. However, in the short term, most of the radiotoxicity of the reference MR system comes from the MR waste stream (dashed – green line) rather than discharged fuel (yellow line); this is due mostly to the Sr-90 that ends up in this stream. With AIROX reconditioning, the trend is the opposite -- the discharged fuel (purple solid line) has a higher radiotoxicity than the waste stream (dashed red). The radiotoxicity contribution of the gaseous FPs, also shown in Fig. 8, is negligible compared to the other contributions; it will be neglected in all the subsequent analysis.

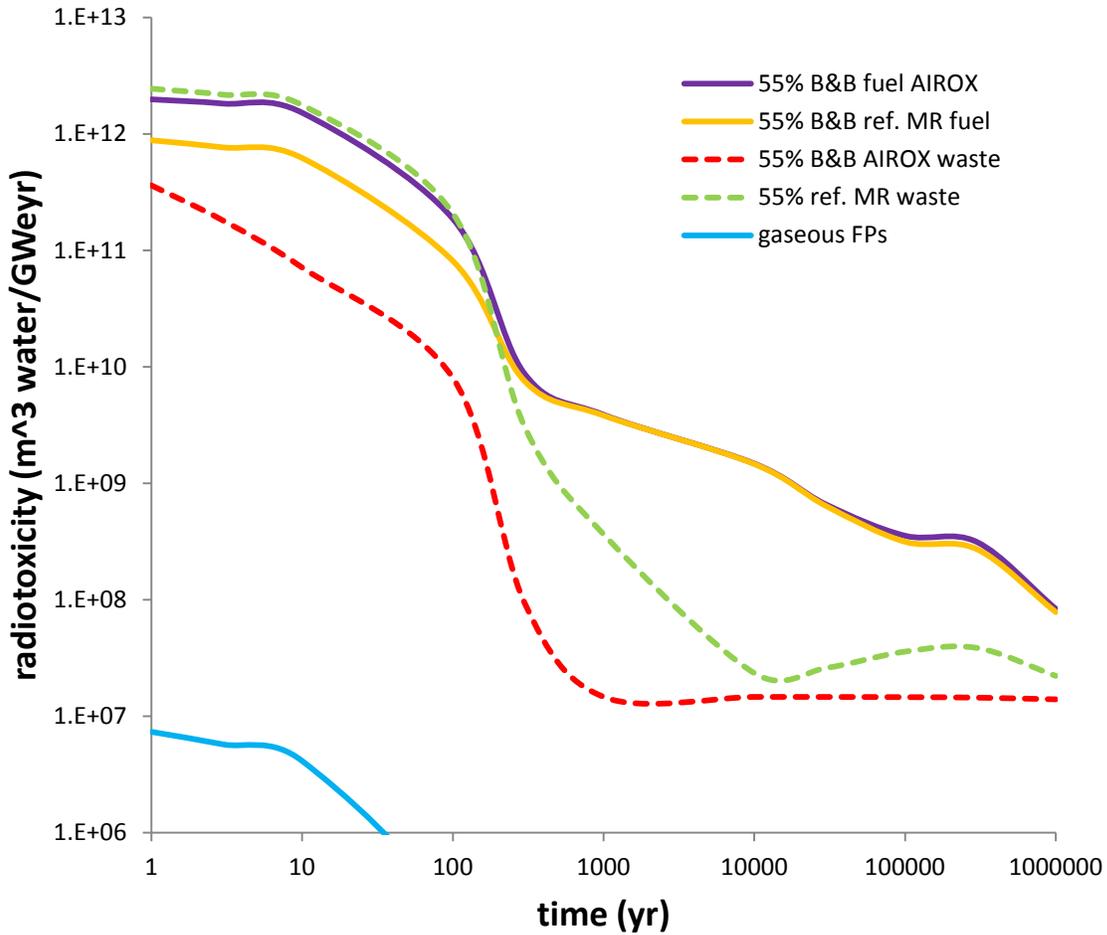


Fig. 8 Radiotoxicity components of the 55% B&B reactor system using either MR or AIROX reconditioning.

The effect of different recovery efficiencies in the melt refining on the radiotoxicity contribution of the Melt-refining waste (MRW) and of the discharged fuel is shown in Fig. 9. It is observed that the discharge fuel radiotoxicity is the same for all the 3 cases in the short term but somewhat differ in the long-term – it is lower for the 10% loss fraction. This is expected since nearly 30% of the actinides end up in the waste stream (Table VI).

As a summary of this section, Fig. 10 plots the ratio of the total radiotoxicity of each of the fast reactor systems examined to that of the PWR radiotoxicity. The 55% reference MR B&B reactor represents the other 55%B&B reactors, since the total radiotoxicity of all these systems is very close. It is concluded that with fuel reconditioning, the specific radiotoxicity of B&B reactors can be lower than that of a once-through PWR while the uranium utilization the B&B reactors offer is higher by up to 2 orders of magnitude. Without fuel reconditioning the radiotoxicity of the B&B reactors is comparable to that of the PWR – it is approximately half the PWR radiotoxicity up to ~200 years and up to twice as much

10,000 years from discharge. The 20% FIMA B&B reactor offers, though, nearly 40 times the PWR uranium utilization and requires a very small fraction of the PWR uranium enrichment capacity.

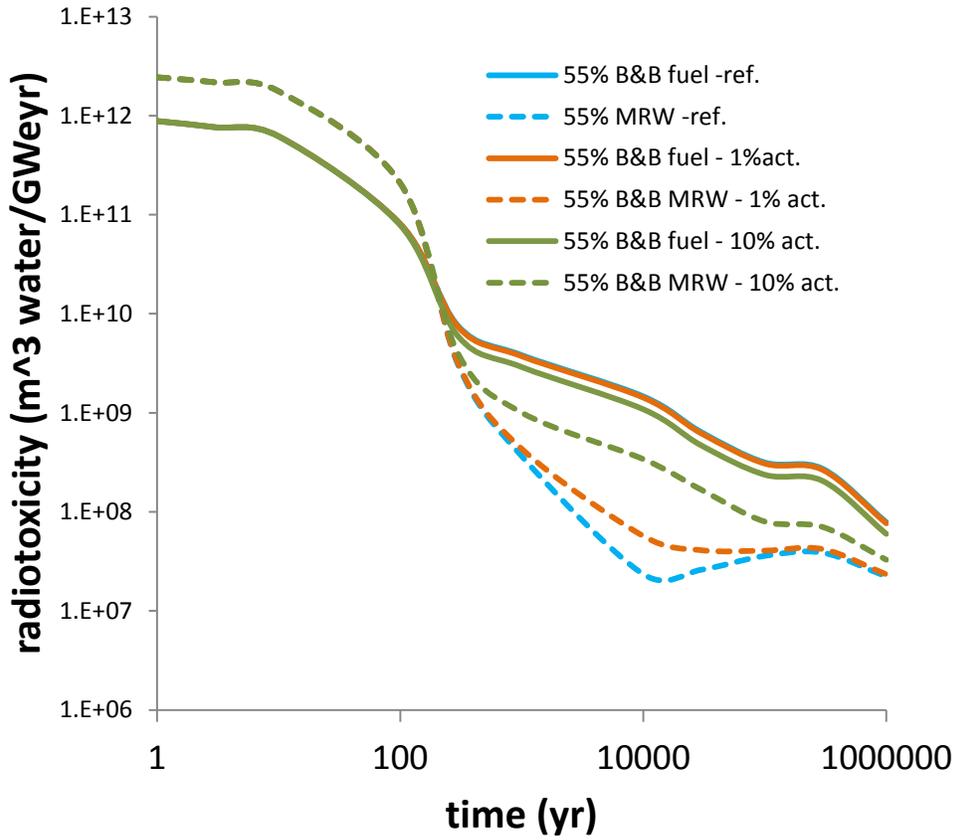


Fig. 9 Radiotoxicity MR 55% B&B cycles, using different recovery efficiency. Melt-refining waste (MRW) is plot together with discharged fuel for reference recoveries and 1% and 10% actinides lost in waste.

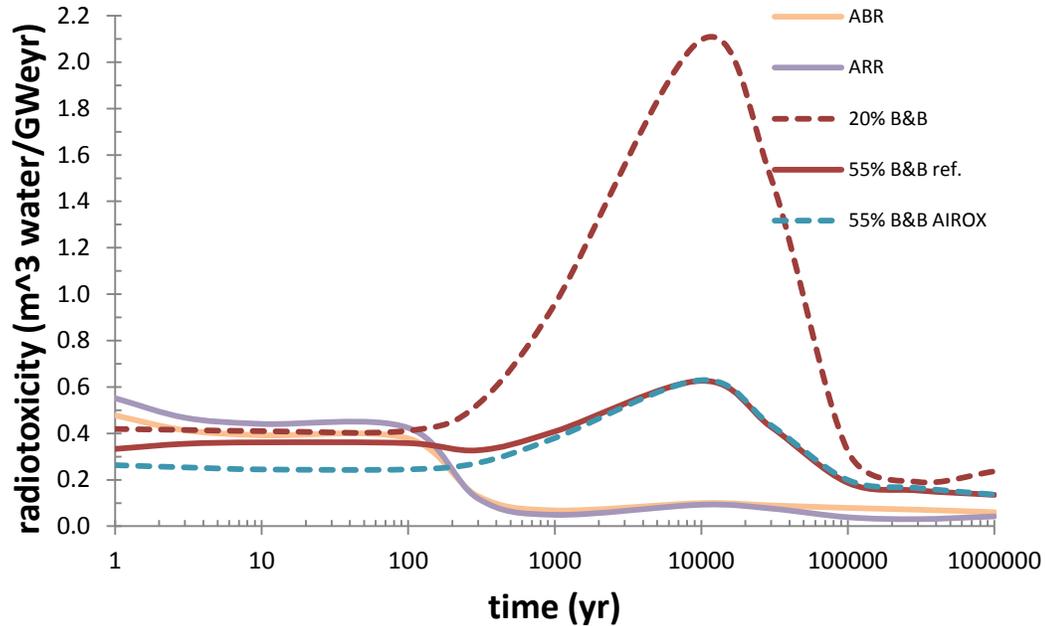


Fig. 10 Radiotoxicity of the various fuel cycles as a ratio to PWR once-through radiotoxicity.

6. Decay Heat

The total decay heat from the discharged fuel and waste streams from the analyzed reactor systems is compared in Fig. 11. The overall trend is similar to that of the radiotoxicity (Fig. 5); there is no significant difference for the total decay heat of the 55% reactors, using MR with different recovery efficiencies. The specific decay heat of all fast reactor systems is smaller than that of the PWR with the exception of the 20% B&B reactor system that has a higher decay heat around the 10,000 years. In the short term, the B&B reactors specific decay heat is even somewhat smaller than that of the ARR and ABR but after ~100 years the ARR and ABR specific decay heat becomes significantly smaller. This is again attributed to the low amount of actinides that get into the ARR and ABR waste stream combined with lack of discharged fuel for disposal.

Fig. 12 shows the difference in the decay heat of the 55%B&B reactor systems when using the MR versus the AIROX process for fuel reconditioning. The discharged fuel decay heat is essentially the same but there is a significant difference in the decay heat emitted by the waste streams – the AIROX waste stream specific decay heat is higher in the short-term but gets significantly lower after ~100 years. This difference is due to the actinide contribution that is only present in melt-refining waste.

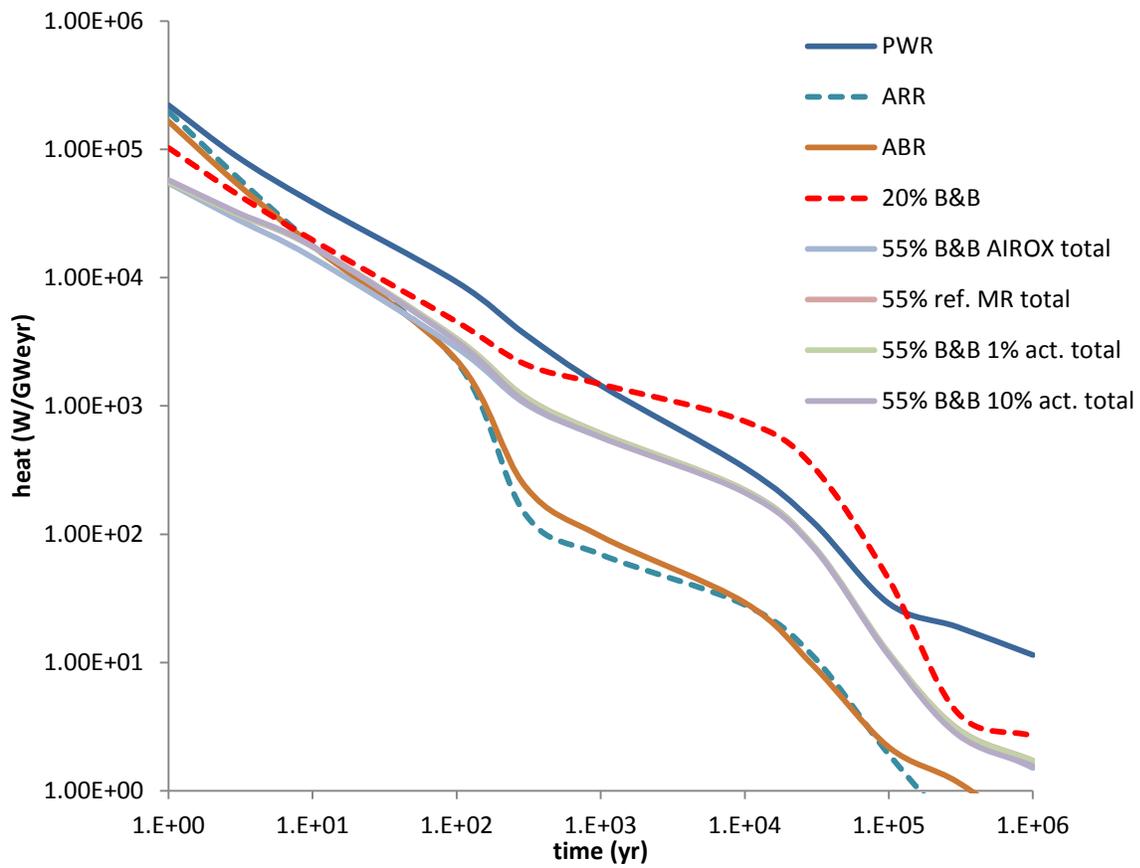


Fig. 11 Total specific decay heat of analyzed fuel cycles

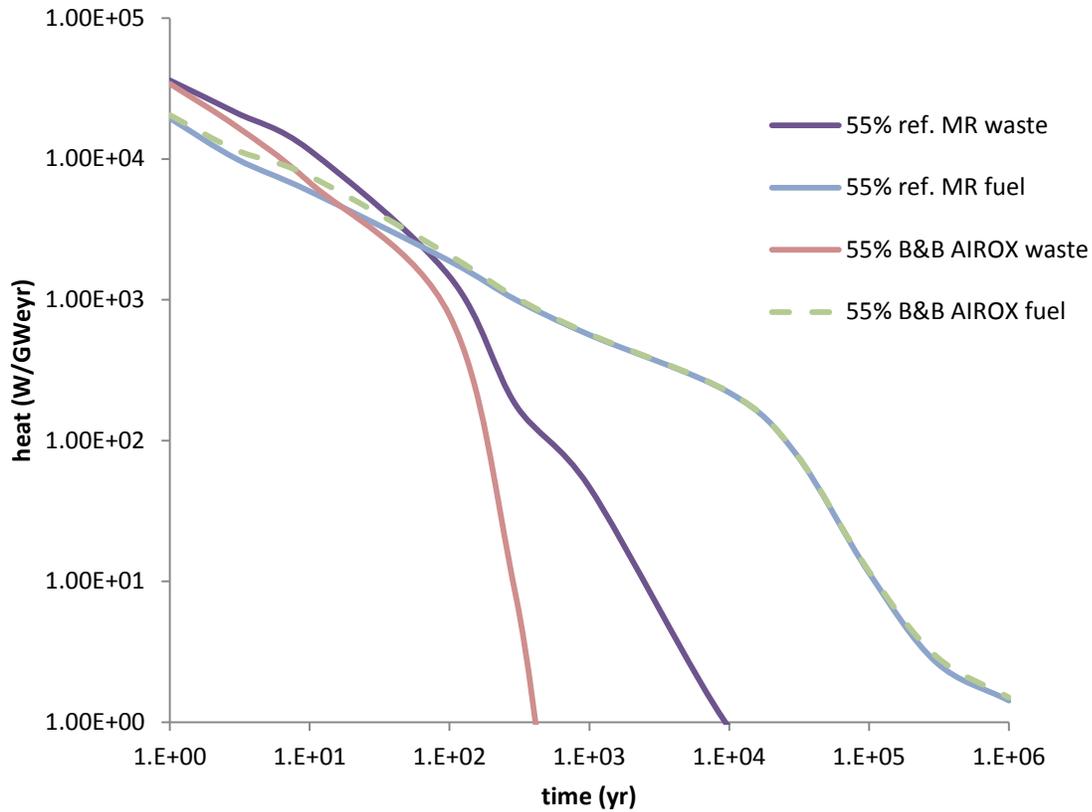


Fig. 12 Specific decay heat of the reference 55%B&B reactor system using the melt-refining and AIROX reconditioning

Fig. 13 shows the effect of the MR recovery fraction on the 55% B&B system decay heat. Very small difference is observed in the short term decay heat from the waste streams but the difference becomes significant in the long time range when the majority of the heat comes from actinides decay.

Fig. 14 shows the decay heat as a ratio of decay heat of PWR. It is observed that the 55% B&B reactors have significant less decay heat than PWR. On the other side, the 20%B&B reactor has a significant higher decay heat than PWR, still due to the high content of Pu-239

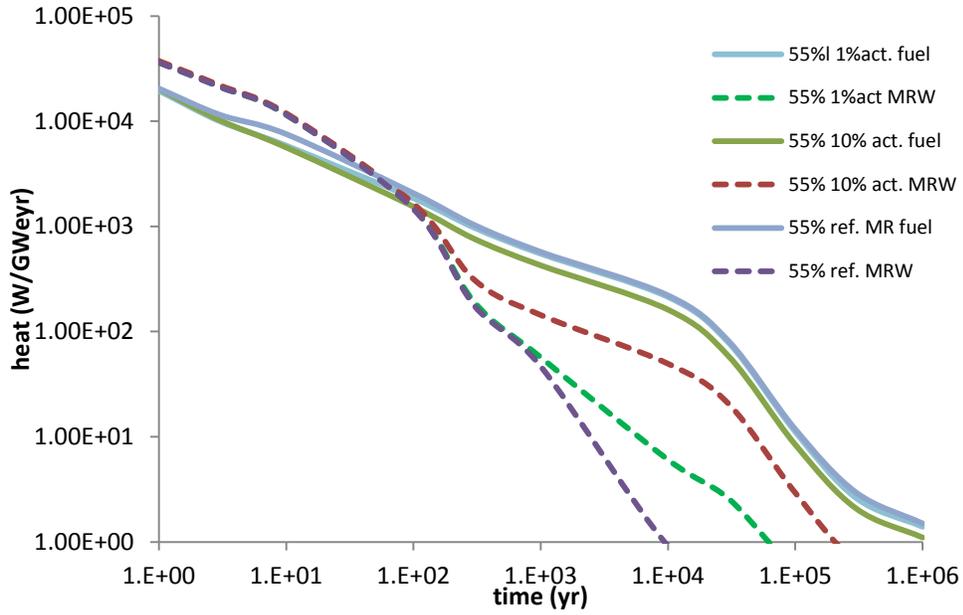


Fig. 13 Effect of recovery efficiency on the decay heat of the discharged fuel and waste streams from the 55%B&B reactor systems

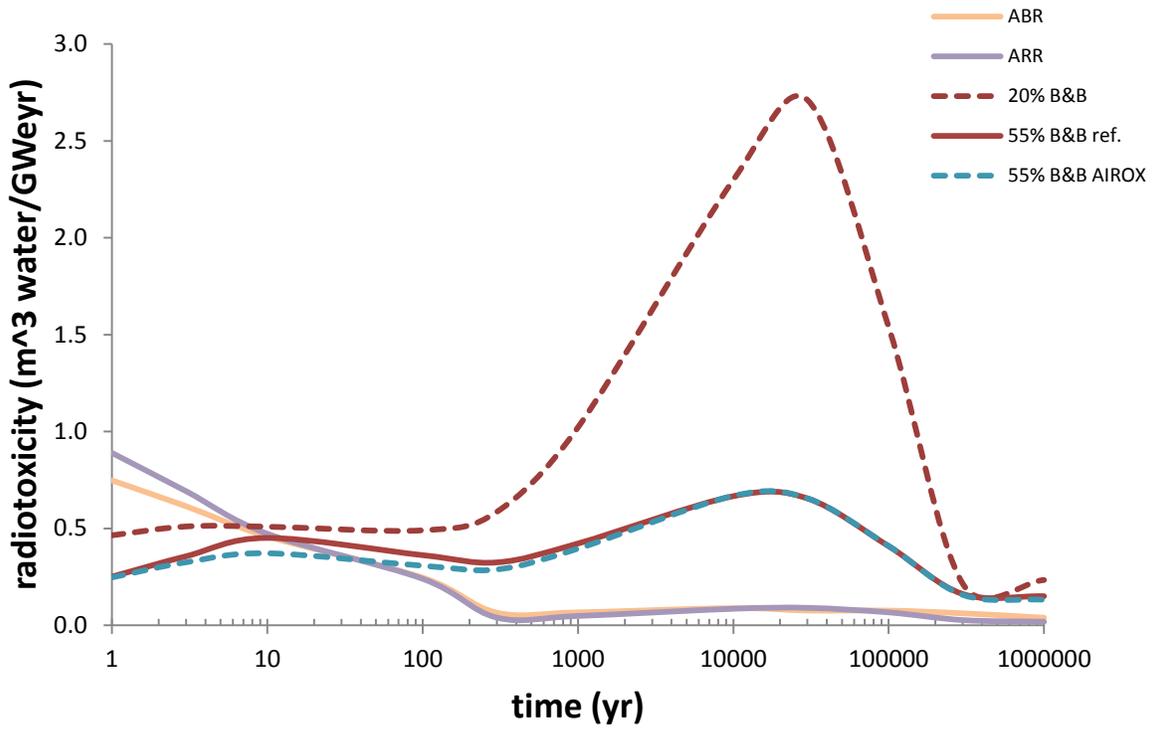


Fig. 14 Decay heat of the various fuel cycles as a ratio to PWR once-through radiotoxicity.

7. Proliferation resistance

The proliferation resistance of the fuel and of the waste discharged from the B&B reactor systems examined is compared against that of the fuel and waste discharged from the reference PWR, ARR and ABR systems using a recently proposed Figure of Merit [5]:

$$FOM_1 = 1 - \log_{10} \left(\frac{M}{800} + \frac{Mh}{4500} + \frac{M}{50} \left[\frac{D}{500} \right]^{\frac{1}{\log_{10} 2}} \right)$$

Here, M is the critical mass, in kg, of a bare sphere made of the metal in question; h is the specific heat of the metal in W/kg; and D is the dose rate, in rad/hr, at a distance of one meter away from a quantity of $0.2M$. A fuel is considered to be proliferation resistant if its FOM value is negative; the more negative the more proliferation resistant the fuel is; and vice versa – the larger positive, of more proliferation concern the material is.

The values of critical mass are calculated as a weighted average of the mass reported in Table IX [5].

Table IX. Nuclide data for FOM calculations

Isotope	M (kg)	Mh (W)	Dose (rad/h @ 1 m for 0.2M)	FOM ₁
²²⁷ Ac	>5(10) ¹¹			
²²⁸ Th	>6(10) ¹¹			
²²⁹ Th	2780.3	1.69(10) ⁴	6.18(10) ⁰	0.1
²³⁰ Th	>6(10) ¹¹			
²³² Th	>6(10) ¹¹			
²³¹ Pa	>1(10) ¹²			
²³² Pa	105.3	2.96(10) ⁸	3.64(10) ⁸	-18.8
²³³ Pa	>6(10) ¹⁰			
²³² U	6.7	4.76(10) ³	2.82(10) ⁻¹	1.0
²³³ U	15.3	4.30(10) ⁰	1.46(10) ⁻⁴	2.7
²³⁴ U	126.1	2.26(10) ¹	3.59(10) ⁻⁴	1.8
²³⁵ U	46.5	2.79(10) ⁻³	1.04(10) ⁻⁵	2.2
²³⁶ U	>1(10) ¹²			
²³⁸ U	>1(10) ¹¹			
²³⁶ Np	7.0	1.88(10) ⁻¹	1.10(10) ⁻²	3.1
²³⁷ Np	62.8	1.26(10) ⁰	4.69(10) ⁻⁴	2.1
²³⁶ Pu	7.2	1.31(10) ⁵	6.98(10) ⁰	-0.5
²³⁸ Pu	9.7	5.51(10) ³	2.11(10) ⁻¹	0.9
²³⁹ Pu	10.0	1.92(10) ¹	3.95(10) ⁻⁴	2.8
²⁴⁰ Pu	37.3	2.64(10) ²	7.17(10) ⁻³	2.0
²⁴¹ Pu	13.0	4.27(10) ¹	1.45(10) ⁻³	2.6

²⁴² Pu	89.1	1.04(10) ¹	5.45(10) ⁻³	1.9
²⁴⁴ Pu	256.2	1.29(10) ⁻¹	1.50(10) ⁻²	1.5
²⁴¹ Am	57.3	6.55(10) ³	1.22(10) ⁰	0.8
²⁴² Am	10.9	1.01(10) ⁷	1.29(10) ⁵	-6.4
^{242m} Am	11.7	4.94(10) ¹	2.74(10) ⁻¹	2.6
²⁴³ Am	144.8	9.30(10) ²	2.82(10) ⁻¹	1.4
²⁴² Cm	368.2	4.45(10) ⁷	8.55(10) ²	-3.0
²⁴³ Cm	11.9	2.25(10) ⁴	1.37(10) ²	0.3
²⁴⁴ Cm	27.1	7.66(10) ⁴	1.40(10) ¹	-0.2
²⁴⁵ Cm	9.5	5.43(10) ¹	1.47(10) ⁻¹	2.6
²⁴⁶ Cm	49.4	4.93(10) ²	1.67(10) ¹	1.8
²⁴⁷ Cm	8.4	2.42(10) ⁻²	1.62(10) ⁻³	3.0
²⁴⁸ Cm	42.5	5.06(10) ⁰	7.25(10) ¹	2.2
²⁵⁰ Cm	24.8	3.66(10) ³	5.04(10) ³	-2.0
²⁴⁹ Bk	193.7	6.20(10) ⁴	7.49(10) ⁻¹	-0.1
²⁴⁹ Cf	7.2	1.10(10) ³	5.36(10) ¹	1.6
²⁵⁰ Cf	6.6	2.64(10) ⁴	2.95(10) ³	-0.7
²⁵¹ Cf	5.6	3.16(10) ²	1.59(10) ⁰	2.1
²⁵² Cf	5.8	1.12(10) ⁵	5.77(10) ⁵	-8.2

Table X gives the relative composition of the actinides in the discharged fuel and in the waste streams of each of the reactors systems examined while Table XI compares the FOM of the TRU present in each of these systems. It is found that none of the B&B reactor systems meets the FOM<0 criterion, although the 55%B&B discharged fuel comes very close to it. The FOM strongly depends on the mass percent of fissile Pu. As this is highest in the 20%B&B reactor (Table VIII), this reactor fuel features the highest FOM. The reference 55% B&B MR waste stream has the smallest FOM of -3.17. This is due to the fact that this waste stream has only a very little amount of TRU, consisting only of the 95% Th and Am that remains stuck to the crucible [1].

In a follow-up study we found that it is possible to greatly reduce the fissile plutonium content in the fuel discharged from B&B reactors at an average burnup of 20% FIMA by irradiating this fuel in a soft neutron spectrum for few % FIMA. After such a soft-spectrum irradiation the FOM of the fuel discharged from the 20% B&B core will be negative.

Table X. Fraction of TRU nuclides to the total TRU in each waste stream

Fraction	PWR	AIROX 55% B&B fuel	1% act. 55% B&B fuel	1% act. 55% B&B MRW	10% act. 55% B&B fuel	10% act. 55% B&B MRW	ref. 55% B&B fuel	ref. 55% B&B MRW	20% B&B	ABR	ARR
NP236	1.45E-09	6.92E-10	3.36E-10	3.58E-10	4.29E-10	6.19E-10	3.28E-10	0.00E+00	7.44E-10	0.00E+00	0.00E+00
NP237	6.93E-02	6.28E-03	4.93E-03	3.41E-03	4.69E-03	5.57E-03	4.96E-03	0.00E+00	6.12E-03	1.41E-02	6.03E-03
PU236	2.08E-07	1.76E-07	1.20E-07	9.15E-08	1.33E-07	1.53E-07	1.19E-07	0.00E+00	1.74E-07	3.20E-07	9.89E-08
PU238	2.79E-02	6.22E-03	9.55E-03	3.29E-03	9.71E-03	5.35E-03	9.52E-03	0.00E+00	3.55E-03	3.54E-02	1.05E-02
PU239	4.41E-01	7.18E-01	6.12E-01	4.17E-01	5.94E-01	6.83E-01	6.14E-01	0.00E+00	7.90E-01	3.61E-01	6.38E-01
PU240	2.36E-01	2.29E-01	3.00E-01	1.27E-01	3.09E-01	2.08E-01	2.99E-01	0.00E+00	1.78E-01	3.28E-01	2.64E-01
PU241	1.33E-01	2.40E-02	3.48E-02	1.31E-02	3.87E-02	2.19E-02	3.44E-02	0.00E+00	1.67E-02	5.70E-02	3.33E-02
PU242	6.35E-02	6.46E-03	1.79E-02	3.79E-03	2.14E-02	6.45E-03	1.76E-02	0.00E+00	2.72E-03	1.01E-01	2.21E-02
PU244	4.08E-06	1.76E-07	9.75E-07	1.22E-07	1.47E-06	2.27E-07	9.36E-07	0.00E+00	3.87E-08	6.78E-07	1.03E-07
AM241	5.95E-03	6.46E-03	1.34E-02	3.36E-01	1.24E-02	5.29E-02	1.34E-02	7.77E-01	2.57E-03	2.81E-02	1.25E-02
AM242m	1.36E-04	5.76E-04	1.07E-03	2.86E-02	1.09E-03	4.59E-03	1.06E-03	6.61E-02	1.71E-04	2.03E-03	8.18E-04
AM242	1.36E-04	5.76E-04	1.07E-03	2.86E-02	1.09E-03	4.59E-03	1.06E-03	6.61E-02	1.71E-04	2.03E-03	8.18E-04
AM243	1.55E-02	6.93E-04	2.33E-03	3.97E-02	2.94E-03	6.86E-03	2.28E-03	9.10E-02	2.19E-04	3.38E-02	5.99E-03
CM242	2.05E-03	2.94E-04	4.42E-04	1.57E-04	5.43E-04	2.67E-04	4.32E-04	0.00E+00	1.28E-04	1.78E-03	6.47E-04
CM243	7.40E-05	2.68E-05	4.74E-05	1.43E-05	6.21E-05	2.48E-05	4.61E-05	0.00E+00	1.00E-05	1.67E-04	4.47E-05
CM244	5.73E-03	5.15E-04	2.16E-03	3.35E-04	3.22E-03	6.16E-04	2.07E-03	0.00E+00	1.20E-04	2.51E-02	3.64E-03
CM245	2.41E-04	6.08E-05	3.45E-04	4.27E-05	5.36E-04	8.12E-05	3.30E-04	0.00E+00	1.09E-05	6.55E-03	8.89E-04
CM246	4.07E-05	6.88E-06	7.89E-05	5.99E-06	1.33E-04	1.21E-05	7.50E-05	0.00E+00	7.94E-07	3.58E-03	4.72E-04
CM247	4.89E-07	3.37E-07	6.17E-06	3.46E-07	1.09E-05	7.16E-07	5.84E-06	0.00E+00	2.89E-08	3.05E-04	3.57E-05
CM248	3.32E-08	2.55E-08	9.61E-07	3.59E-08	1.92E-06	7.81E-08	8.97E-07	0.00E+00	1.41E-09	1.44E-04	1.69E-05
CM250	1.48E-16	2.29E-17	1.59E-15	5.08E-17	4.83E-15	1.34E-16	1.43E-15	0.00E+00	1.12E-18	6.70E-11	5.62E-12
BK249	3.38E-10	3.06E-10	1.01E-08	4.80E-10	2.66E-08	1.17E-09	9.19E-09	0.00E+00	1.78E-11	3.23E-06	3.17E-07
CF249	7.86E-11	5.08E-10	3.55E-08	9.26E-10	7.26E-08	2.01E-09	3.31E-08	0.00E+00	1.61E-11	9.89E-06	1.12E-06
CF250	8.08E-11	1.04E-10	8.01E-09	2.24E-10	2.03E-08	5.40E-10	7.30E-09	0.00E+00	3.44E-12	2.27E-06	1.95E-07
CF251	3.42E-11	6.88E-12	7.74E-10	1.84E-11	2.08E-09	4.55E-11	7.02E-10	0.00E+00	1.70E-13	2.92E-07	2.30E-08
CF252	2.55E-11	2.73E-13	3.25E-11	8.85E-13	1.09E-10	2.40E-12	2.89E-11	0.00E+00	5.73E-15	2.00E-08	1.38E-09

Table XI. TRU FOM for the different discharged fuels and waste streams

	PWR	AIROX 55% B&B fuel	1% act. 55% B&B fuel	1% act. 55% B&B MRW	10% act. 55% B&B fuel	10% act. 55% B&B MRW	ref. 55% B&B fuel	ref. 55% B&B MRW	20% B&B	ABR	ARR
FOM	-0.32	0.37	0.16	-1.78	0.10	-0.14	0.17	-3.17	0.77	-0.36	0.08

8. Dose at 1m from discharged fuel

Since the fuel to be reconditioned in a melt-refining or an AIROX process will contain many fission products, the reconditioning operation will have to be done in a well shielded facility. As a measure of the hot fuel handling difficulty the dose-rate at 1 m from B&B fuel, at the moment of discharge, is calculated and compared against the dose from the fuel discharged from the ARR, ABR as well as PWR fuel.

The dose is calculated using the method of Unger & Trebuy [6]. The UNGER code, provided by RSICC, calculates the mSv/h per MBq of each nuclide. An ad-hoc python program CALCDOSE was written to fold this data with the nuclide inventory as provided by the ORIGEN 2.2 calculations.

Table XII reports the dose at 1 m for the discharged batch of various reactors. The dose is normalized to GWeY and to ton of initial Heavy Metal. For the 55% reactor only the ref. MR scenario is analyzed as there is no difference in batch #4 for all fuel cycles and batch #6 of the reference scenario will be more radioactive than the 1% and 10% loss scenario.

Table XII. Dose at 1m from discharged fuel. Dose is normalized to the GWeY produced and ton of Initial Heavy Metal.

dose rate at 1 m	mSv/h/ GWeY at discharge	mSv/h/tonIHM at discharge	mSv/h/ GWeY at 1 month	mSv/h/tonIHM at 1 month
PWR	7.58E+09	3.42E+08	4.24E+08	1.92E+07
ARR	6.24E+09	9.31E+08	3.51E+08	5.24E+07
ABR	9.22E+09	7.61E+08	4.43E+08	3.66E+07
20% B&B	2.80E+09	5.85E+08	1.77E+08	3.69E+07
55% B&B batch #4	3.01E+08	4.91E+08	1.89E+07	3.08E+07
55% B&B batch #6	5.07E+08	4.04E+08	3.73E+07	2.97E+07
55% B&B batch #8	3.53E+08	2.10E+08	2.78E+07	1.65E+07

The dose from B&B fuel is higher than the dose from PWR, but lower than the dose of other advanced reactors. This is true for all the options: the 3 batches of 55% B&B and for 20% B&B. This means that recycling of B&B fuel would not pose any additional requirement for fuel handling technology compared to other advanced reactor concepts

9. Fuel Cycle Scenarios

In this final section, scenarios for the deployment of B&B reactors will be analyzed and discussed.

B&B reactors at equilibrium will be fed only by depleted uranium, however to start operations B&B reactors will need a starter fissile fuel. As fissile materials for the starter, enriched uranium and TRU from Used Nuclear Fuel (UNF) are being considered.

In the scenarios considered, an annual growth of nuclear power of 2% is assumed. Deployment of B&B reactors is assumed to start in 20 years, i.e. in 2032. As B&B reactors are deployed, existing LWRs are decommissioned at the end of life. The existing LWRs population is simplified in 3 main categories: reactors 10-19 years old, reactors 20-29 years old, reactors 30-39 years old. The current U.S. nuclear power production (104 GWe) [8] is then divided in these categories according to the number of reactors, leading to an approximated population of 9712 GWe (10-19 years), 40788 GWe (20-29 years) and 50500 GWe(30-39 years). Each population is assumed to end the life at 60 years of age; the rate of decommissioning of GWe per year is assumed to be constant within each population. Fig. 15 shows the growth of nuclear power at the rate of 2%, with the replacement of LWRs with B&B, as LWRs are dismissed. It is assumed that LWR will be introduced until the beginning of 2031, and as a consequence the last LWR will shut down at the end of 2090. Fig. 15 shows how the LWRs are dismissed according to the population they belong to, this gives the 4 different rates of decommissioning in the figure, the last rate being LWRs introduced from 2012 to 2031. Fig. 15 also shows how the B&B population will need to evolve to replace decommissioning of LWRs and the 2% annual growth. It is also assumed that there are about 65,000 MT of used nuclear fuel in the US [9]. ORIGEN calculations show that there are about 1% TRU for each ton of UNF and therefore 650 tons of TRU are available. It was shown in [1] that each B&B reactor would use 8.15 tons of TRU as starter. Fig. 15 shows therefore that if all B&B reactors were started with TRU, and considering TRU production from new LWRs, the TRU reserves would run out in 2044.

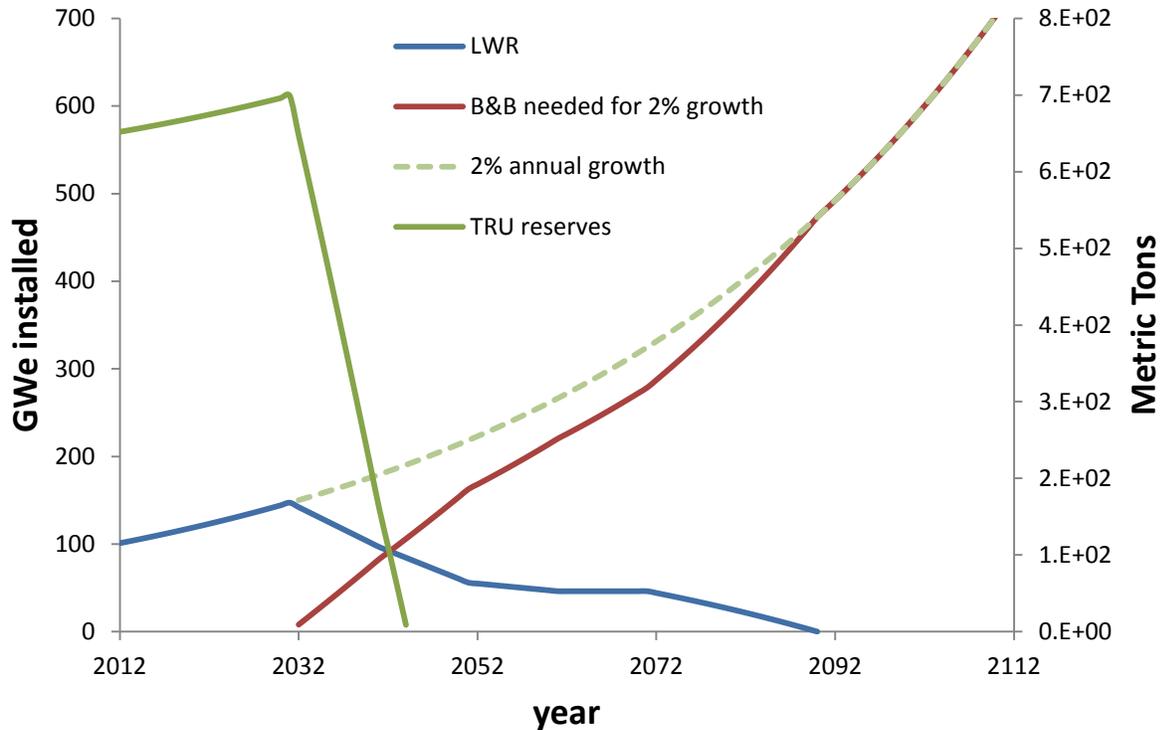


Fig. 15 GWe installed and TRU reserve as B&B reactors are deployed.

Another way is to use enriched uranium to start B&B reactors. It was found [1] that to start a B&B reactor, a starter would need 4 batches with enrichment of 6.1%, 10.8%, 11.3% and 14.8% for a total of 7.66 tons of U-235. The Separative Work Units (SWU) needed to enrich such an amount of uranium can be calculated using [10]:

$$S = W(2x_w - 1) \ln \frac{x_w}{1 - x_w} + P(2x_p - 1) \ln \frac{x_p}{1 - x_p} - F(2x_f - 1) \ln \frac{x_f}{1 - x_f}$$

where P is the amount of product (enriched uranium), F is the amount of natural uranium feed, W is the amount of tailings, x_f is the fraction of U-235 in natural uranium ($x_f = 0.00711$), x_w is the amount of U-235 in the tailings (assumed = 0.002), x_p is the fraction of U-235 in the product.

W and F can be calculated as:

$$F = P \frac{x_p - x_w}{x_f - x_w}$$

$$W = P \frac{x_p - x_f}{x_f - x_w}$$

Assuming $P=1$ Kg, it is possible to calculate SWU per 1 Kg of enriched U. Since each batch contains 17.8 tons of HM, SWU for each batch depending of U-235 enrichment are given in Table XIII for a total of 1620.6 thousands SWU for each starter.

Table XIII. Enrichment of U-235 in each batch.

enrichment	Kg SWU (thousands)
6.1%	203.6
10.8%	406.3
11.3%	428.2
14.8%	582.5
total	1,620.6

Fig.16 shows the thousands of SWU required each year for the starters of B&B reactors. The curve reflects the changes in rate of B&B reactors construction. These changes depend on the current population of LWRs, as these are slowly decommissioned. The first three steps correspond to dismissal of existing reactors; the fourth to a period with no LWR decommissioning (as no reactor has been built in the last 10 years), followed by a ramp up period to replace LWRs and then a slow growth as B&B are fully utilized for the entire energy production. By the year 2100 an enrichment capacity of about 15,000 SWU (thousands) would be needed. This capacity is less than current enrichment capacity of Russia (23,000 SWU) and if construction is undertaken would not be a problem for US.

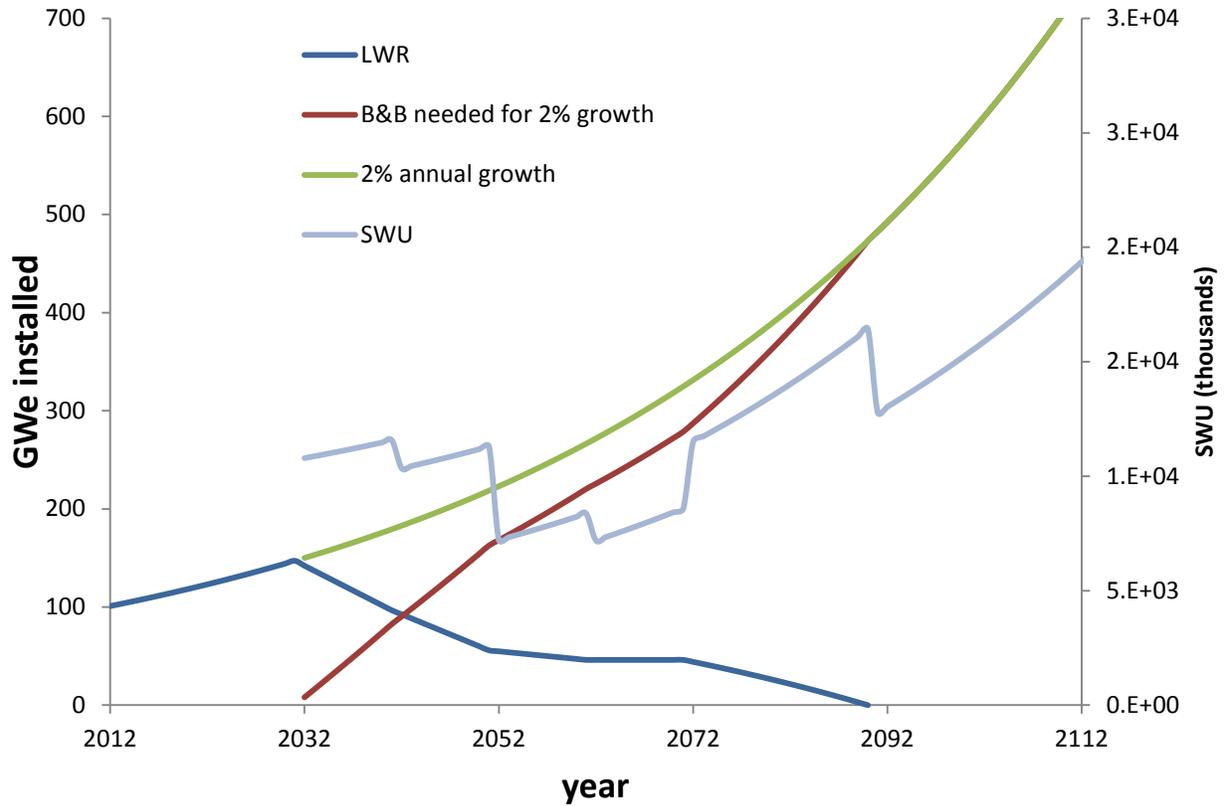


Fig. 16 GWe installed and TRU reserve as B&B reactors are deployed.

However, B&B reactors are also able to establish an almost self-sustaining fuel cycle with little requirement of enrichment capacity or TRU reserves. It was found [1] that the 20% B&B reactor used fuel can be used to start another B&B reactor core, if the used fuel is taken out before it reaches the end of life burn-up. The spawning mode of B&B reactors is shown in Fig. 17. It was found [1] that the doubling time of B&B reactors is about 14 years.

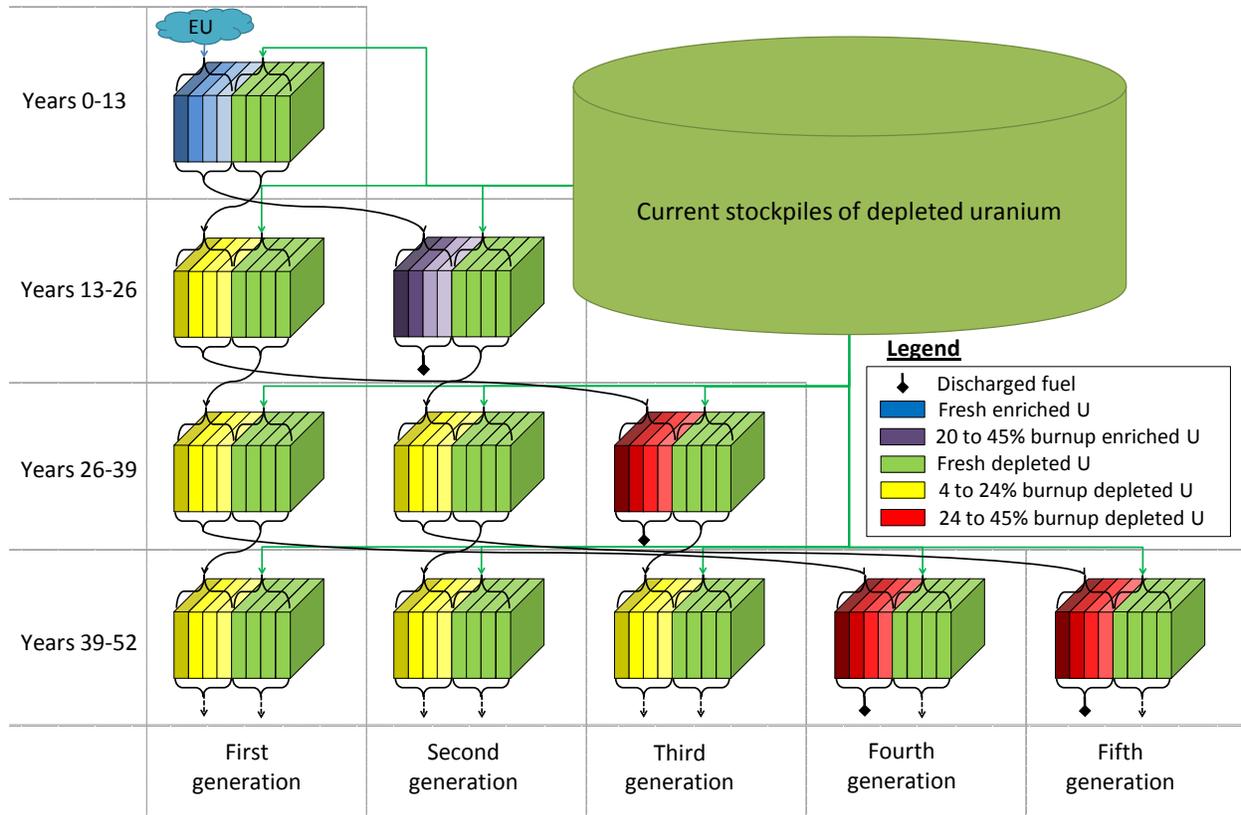


Fig. 17 Spawning mode of B&B reactors from "mother" reactor

Using this spawning mode is possible to spawn a large population of B&B only starting building a few reactors each year. Fig.18 shows the evolution if spawning happens from 7 reactors built each year for 14 years and then let them spawn or from 5 reactors built each year for 14 years and then let spawn. After 14 years, new reactors are built each year, but these new reactors use fuel that has been used in B&B reactors and do not use TRU or enriched fuel. It is found that spawning from 7 reactors/year for 14 year would be required to satisfy the 2% growth curve. The use of spawning from 7 reactors would require the consumption of 54 TRU tons/year for 14 years, or 11,340 thousands SWU each year for 14 years. After the first 14 years no more TRU or SWU would be required for new B&B reactors to be deployed. An assessment of reconditioning capacity that would be needed for melt-refining is also done. Using spawning mode from 7 reactors built each year for 14 years, 320 tons reconditioning capacity by 2100 would be needed. This is only 1/3 of the reprocessing capacity of La Hague plant.

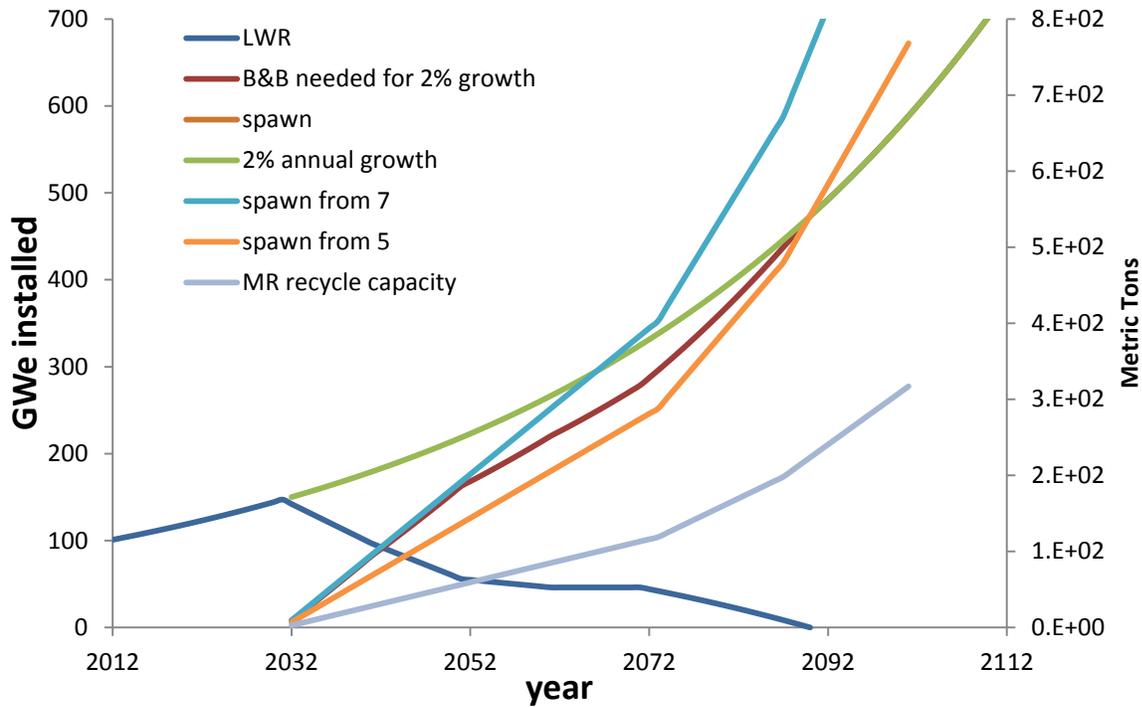


Fig. 18 Spawning mode of B&B reactors from 7 reactors and from 5 reactors. The necessary recycling capacity is also plot.

In conclusion, B&B reactors can be deployed to gradually replace LWRs. The requirements on enrichment capacity of uranium are not particularly stringent. On the other hand, starting B&B only on TRU would not be possible due to TRU reserve consumption after only 12 years of deployment.

Using spawning of new reactors, enrichment or TRU would be needed only for the first 14 years. However, a 2% growth can be satisfied only from spawning after building 7 reactors each year for 14 years. This might be a hard objective to achieve. On the other hand, a 1% year growth in nuclear power could be satisfied using only 5 reactors built every year for 14 years. Given the current economic crisis and the trend in higher efficiency of energy use, the 1% growth is still a plausible scenario.

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