A Hybrid Subcritical Testbed for Fast Neutron Irradiation of Novel Fuels and Claddings in Fast Reactors

P. Deng,[1] W. S. Yang,[1] V. P. Chellapandi,[1] F. Y. Odeh,[2] M. Mamtimin,[2] T. L. Grimm[2]

[1]*Department of Nuclear Engineering & Radiological Sciences, University of Michigan, Ann Arbor, MI 48109*
[2]*Niowave, Inc., 1012 N Walnut St, Lansing, MI 48906*
wonyang@umich.edu

INTRODUCTION

Research and development (R&D) of Generation IV fast reactors relies on novel materials that are resistant to both radiation damage and corrosive environments. The lack of data for most of these novel and advanced materials calls for renewed testing efforts performed in an environment of intense fast neutron flux. Fast neutron fluxes greater than 10^{14} n/cm^{2}s are required for in-situ radiation damage studies, and fast fluxes greater than 10^{15} n/cm^{2}s are required for radiation damage studies attempting to reach significant displacements per atom (DPA). These fluxes are conventionally achieved with fast reactors, which are not currently available in the US since the shutdown of the Fast Flux Test Facility (FFTF) and will not become available without major investments from the private and public sector [1]. Instead, it was proposed to develop a hybrid subcritical testbed that can provide a fast reactor like environment to support experimentation and demonstration of novel fuels and materials, without the logistical and regulatory challenges of fast reactors [2].

The objective of this proposed work is to develop a conceptual design of a hybrid fast/thermal spectrum subcritical testbed that is coupled to a superconducting electron linear accelerator (linac) through a lead-bismuth eutectic (LBE) neutron source converter. The targeted flux level is a peak fast neutron flux (neutron energy > 0.1 MeV) over 10^{15} n/cm^{2}s and a peak annual fast neutron fluence over 2×10^{22} n/cm^2 within a test chamber that contains at least 100 cm^3 of space. The design goals are to minimize the required fuel loading and the electron beam power to drive the subcritical system while achieving a reasonable cycle length.

METHODS

Material Selection and Design Constraints

Most design constraints come from the selected materials. To reach a high fast neutron flux level with a typical energy spectrum in fast reactor, a fast core region is essential in the proposed hybrid system. The fuel volume fraction of the fast core region should be high enough to enhance the neutron economy and thus to reduce the fuel loading and core size. To remove the resulting high power density with small coolant volume fraction, liquid metals are typically chosen as the coolant materials due to their good thermal conductivity. Since LBE was used as the target material in the neutron source converter, it was also selected as coolant in both the fast and thermal core regions to avoid complicated multiple coolant systems. For fuel, U-10Zr metallic fuel with ^{235}U enrichment of 19.75 wt.% was selected because of its high heavy metal density and high thermal conductivity [3,4]. In the fast region, HT-9 steel was chosen as the cladding material for its excellent irradiation resistance [5]. In the thermal region, a zirconium alloy was chosen as the cladding material to reduce the thermal neutron absorption. Zirconium hydride ($ZrH_{1.6}$) was selected as the moderator material.

Several design constraints were imposed to ensure fuel integrity. The fuel centerline temperature was limited by the fuel melting point, and the cladding inner wall temperature was limited by the eutectic temperature of the fuel and cladding. The fuel melting and eutectic temperatures were based on the current material database [6]. The LBE coolant velocity was also limited to 2 m/s because of its erosive and corrosive nature [7]. The peak linear power and core-averaged power density were determined to satisfy these three constraints. In order to ensure a subcritical state for all anticipated abnormal conditions, no active reactivity control mechanism was used and a tentative 2% reactivity margin to criticality was imposed at the cold shutdown state. A reactivity control scheme with burnable poison (BP) was developed to minimize the reactivity swing over the cycle since the required electron beam power increases with increasing sub-criticality level.

Design Approaches

The design work was initiated by developing a reference fast core design which yields the targeted flux level with minimal fission power and fuel loading. Then, in order to reduce the required fuel loading while maintaining a typical fast reactor spectrum in the irradiation test chamber, the periphery part was progressively replaced by a thermal region. An optimum ratio of fast to thermal core dimension was determined to minimize the fuel loading for a targeted multiplication factor without deteriorating the fast flux level and the flux spectrum in the test chamber.

Initial parametric studies were performed to determine the important fast and thermal core design parameters using cylindrical-z core models and homogenized compositions. The reference fast core design was developed by diffusion calculations using the multi-group cross sections generated for each homogenized composition with the MC^2-3 code [8] and the DIF3D code [9]. Hybrid core designs were developed by Monte Carlo simulations using the MCNP6 code [10]. Heterogeneous configurations of the thermal core were explicitly modeled to account for the local heterogeneity effects. Preliminary fuel cycle analyses were performed to quantify the reactivity swing over a 45-day fuel depletion cycle using the depletion capability of MCNP6. Various reactivity control options were investigated to compensate for the reactivity changes due to fuel depletion and fission product buildup. The power and temperature defects were also evaluated via separate MCNP6 simulations for the normal operating condition, the hot zero power condition, and the cold shutdown condition and they were incorporated in the design of reactivity control schemes.

At the current design stage, the properties of the linac driven LBE neutron source converter were investigated by independent Monte Carlo simulations. The neutron yield per electron and the neutron source spectrum depending on the electron beam energy were evaluated.

RESULTS

Reference Fast Core Design

Using a metallic U-10Zr fuel of 19.75 wt.% enrichment, a reference fast core design was developed. An optimum fuel volume fraction of 46% was determined to minimize the critical fuel loading while satisfying the aforementioned three material related design constraints. The core height to diameter ratio was fixed to 1.0 around which the critical volume is minimized. A typical U-10Zr fuel pin design was adopted. Table 1 summarizes the key design parameters and operating conditions. The power level was normalized to yield the average fast flux of 10^{15} n/cm^{2}s in the test chamber.

Table 1. Design Parameters of Reference Fast Core

Fuel material (^{235}U enrichment)	U-10Zr (19.75 wt. %)
Fuel volume fraction	0.46
Fuel pin diameter, cm	0.635
Cladding thickness, cm	0.045
Cladding material	HT-9
Coolant material	LBE
Coolant inlet/outlet temp., K	600/750
Coolant flow velocity, m/s	2.0
Active core height, cm	59.4
Core height to diameter ratio	1.0
Test chamber radius, cm	2.0
Test chamber height, cm	8.0
Reflector (LBE) thickness, cm	180.0
Uranium loading, kg	1050
Average power density, W/cm^3	133
Total fission power, MW	21.71

Parametric Studies for Hybrid Core Design

Based on the reference fast core design, the hybrid core design was developed by replacing the outer part of the fast core with moderated thermal region. The same U-10Zr fuel and LBE coolant were adopted in the thermal region, and zirconium hydride ($ZrH_{1.6}$) was selected as the moderator. The HT-9 steel cladding was replaced by zircaloy to reduce the thermal neutron absorption. The active core height and the central test chamber radius were fixed to the values of the reference fast core. Outside the thermal core, a beryllium reflector was introduced to enhance the neutron economy.

Parametric studies were performed by varying the key design parameters, including the fuel volume fractions in fast and thermal regions, the moderator to fuel ratio in the thermal region, and the ratio of the radial dimensions of fast and thermal regions. These parametric studies were initially performed with region homogenized models without considering the local heterogeneity effects in the thermal region. Then, several promising candidate designs were reanalyzed with heterogeneous configurations of thermal region for further optimization.

For a fixed fuel volume fraction, the fast flux level in the thermal region required to yield the targeted fast flux level in the test chamber is roughly constant. On the other hand, the thermal flux level (and thus power level) increases as the moderator to fuel ratio increases. As a result, the total fission power required to yield the targeted fast flux in the test chamber increases with increasing moderator to fuel ratio. Therefore, replacing part of the fast core with a thermal region increases the required total power while reducing the required fuel loading.

A high fuel volume fraction was used in the reference fast core to produce a hard spectrum and to reduce neutron leakage. However, with the introduction of a thermal booster region, smaller fuel volume fractions were adopted in the fast and thermal regions to reduce the fuel loading further and to reduce the required fission power level to yield the targeted fast flux level. The moderator volume fraction was adjusted to maintain the desired reactivity level.

The core height to diameter ratio was fixed to 1.0 for the reference fast core to reduce neutron leakage. With a hybrid core configuration, this constraint was released. Several parametric studies were performed by varying the core outer radius and the fast region size while the active core height was fixed to 59.4 cm. A minimum radius of 7 cm was determined for the fast region in such a way that the flux spectrum in the test chamber is not deteriorated. Since the reduced core size increases the power density in the thermal region, a minimum outer radius of 20 cm was then determined for the thermal region in a way to meet the power density limit.

Based on the results of these parametric studies, various candidate designs for the hybrid fast/thermal core were developed within the power density constraint imposed by the heat removal capacity of the LBE coolant with 2 m/s flow velocity. The fuel loading and power level of the candidate hybrid designs are compared in Fig. 1 with those of the reference fast core design. For comparison, although it is a thermal reactor, the LEU core design of the High Flux Isotope Reactor (HFIR) [11] is included here since it was designed to produce a similar fast flux level at the irradiation site. The hybrid core designs need less fuel loading than the fast core design and lower fission power than the thermal HFIR design. Relative to the HFIR design, the most promising hybrid design could reduce the fission power by a factor of 4.5 with a comparable uranium loading while providing typical fast spectrum neutron fluxes in the test chamber.

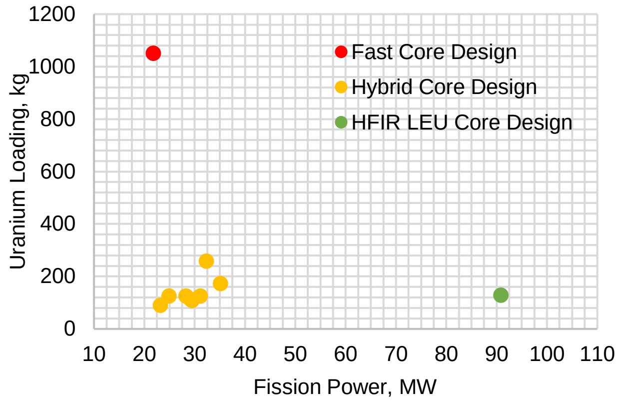

Fig. 1. Comparison of Required Fuel Loading and Fission Power to Yield 10^{15} n/cm^{2}s Fast Flux

Candidate Hybrid Core Design

Further improvements were made on the most promising candidate hybrid design. A heterogeneous configuration was introduced into the thermal region using annular fuel and $ZrH_{1.6}$ solid moderator plates. The thick LBE reflectors were replaced with realistic steel and beryllium reflectors (cooled by LBE). The resulting core configuration is shown in Fig. 2, where the dimensions are in the unit of centimeter. The configuration of the thermal region is schematically shown in Fig.3, where a BP plate was introduced for reactivity control. In addition, Table 2 summarizes the design parameters that were varied from the reference fast core design. The power level was determined to yield the targeted 10^{15} n/cm^{2}s fast flux in the test chamber at the beginning of cycle (BOC).

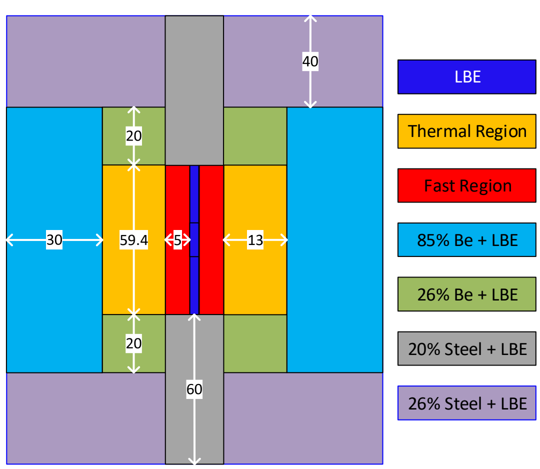

Fig. 2. Configuration of Candidate Hybrid Core Design

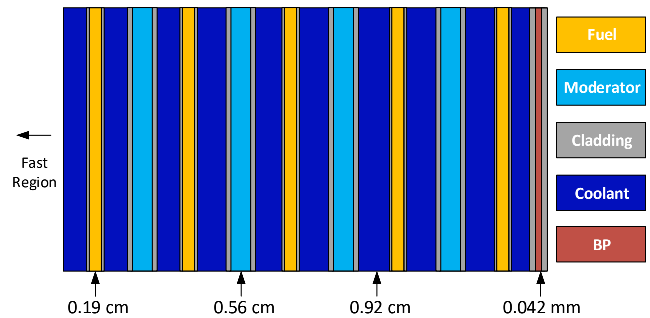

Fig. 3. Configuration of Thermal Region with BP Plate

Table 2. Additional Parameters of Candidate Hybrid Core

Fuel volume fraction in fast region	0.29
Fuel volume fraction in thermal region	0.057
Moderator volume fraction in thermal region	0.17
Cladding in thermal region	Zircaloy
Cladding thickness, cm	0.03
Total uranium loading, kg	86
Ave. power density in fast region, W/cm^3	299
Ave. power density in thermal region, W/cm^3	274
Total fission power, MW	20.49

A 45-day single-batch fuel depletion cycle was simulated using MCNP6 for this candidate hybrid core design. The fuel depletion performances at a constant fission power level of 21.71 MW are shown in Fig. 4, where the green and orange curves represent the reactivity history of the core with and without the BP plate, respectively. The blue line shows the core-averaged burnup vs. irradiation time. The core-averaged burnup at the end of cycle is about 11 MWd/kgU.

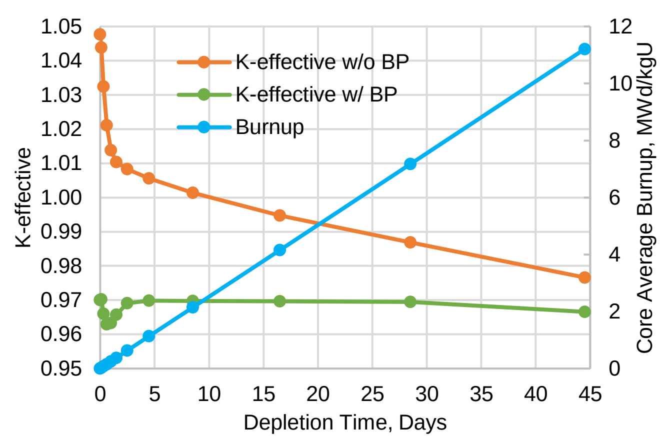

Fig. 4. Depletion Characteristics in 45-day Fuel Cycle with Constant Fission Power of 21.71 MW

Without the BP plate, a total of ~7% reactivity loss is observed and is composed of two major components: ~4% quick reactivity loss in the first 2.5 days mainly due to xenon buildup, and ~3% reactivity loss due to fuel depletion after reaching the equilibrium xenon concentration. To minimize the reactivity change during fuel cycle, boron and gadolinium based burnable poisons were introduced in the thermal region by adding a 0.042 mm thick BP plate that contains ZrB_2 of natural boron (97.75 vol.%) and Gd-157 enriched Gd_2O_3 (2.25 vol.%). Addition of BP helps maintain the reactivity level around 0.97 over the 45-day cycle length. Since the sum of power and temperature defects of reactivity is ~1%, this leaves 2% margin to criticality when the core is at the cold shutdown state.

The average fast flux in the test chamber at a fixed fission power level of 21.71 MW is shown in Fig. 5. It is seen that the required 10^{15} n/cm^{2}s flux can be obtained through the whole cycle. The higher fast flux level at BOC is caused by the poisoning effects of xenon and BPs that increased the fast to thermal flux ratio in the thermal region.

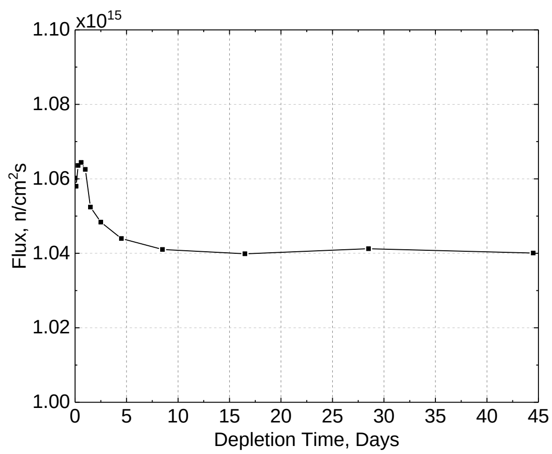

Fig. 5. Average Fast Flux Level in Test Chamber

Properties of Neutron Source Converter

Through the simulation of a LBE target pot bombarded by monoenergetic electron beams using MCNP6, the neutron yield per electron was estimated as ~1% for 40 MeV electrons and ~2.3% for 80 MeV electrons. The neutron sources have similar spectra like the U-235 fission sources, as shown in Fig. 6, which indicates that the external neutrons would behave like fission source neutrons produced in the core.

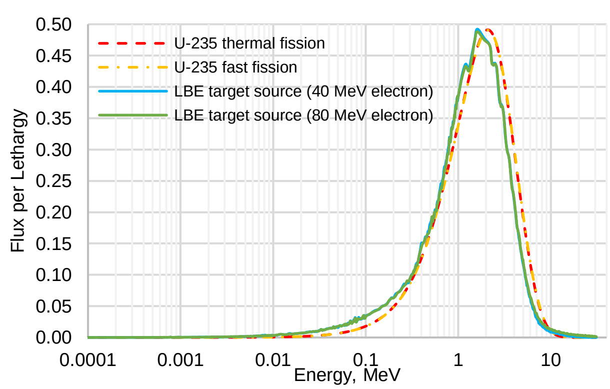

Fig. 6. LBE Target Neutron Source Spectra

CONCLUSION

To support experimentation and demonstration of novel fuels and materials for advanced fast reactors, a linac driven, subcritical testbed concept was developed to provide >10^{15} n/cm^{2}s fast flux in a 100 cm^3 irradiation chamber. A hybrid fast/thermal core design concept was developed with ~86 kg of 19.75% enriched uranium. At a fission power level of 21.71 MW, this hybrid core could yield the targeted fast flux level with a reactivity swing less than 1% over a 45-day fuel depletion cycle.

ACKNOWLEDGMENTS

This work is supported by DOE grant number DE-SC0017733.

REFERENCES

1. "US Nuclear Power Policy," World Nuclear (2017). Webpage: http://www.world-nuclear.org/information-library/country-profiles/countries-t-z/usa-nuclear-power-policy.aspx
2. TERRY L. GRIMM, et al., "A Subcritical Testbed for Fast Neutron Irradiation of Novel Fuels and Cladding in Fast Reactors," DOE-NE SBIR Phase I Proposal, Niowave Inc., February (2017).
3. J. H. KITTEL, et al., "History of Fast Reactor Fuel Development," *Journal of Nuclear Materials*, **204**, 1 (1993).
4. Y. I. CHANG, "Technical Rationale for Metal Fuel in Fast Reactors," *Nuclear Engineering and Technology*, **39**, 161 (2007).
5. YIREN CHEN, "Irradiation Effects of HT-9 Martensitic Steel." *Nuclear Engineering and Technology*, **45**, 311 (2013).
6. A. D. PELTON AND L. LEIBOWITZ, "Calculation of the Liquidus and Solidus in the Plutonium-Uranium-Zirconium System," ANL-IFR-93, August 1988.
7. INPRO Group, "Challenges Related to the Use of Liquid Metal and Molten Salt Coolants in Advanced Reactors," IAEA-TECDOC-1696, May (2013).
8. C. H. LEE and W. S. YANG, "MC^2-3: Multigroup Cross Section Generation Code for Fast System Analysis," ANL/NE-11-41, Argonne National Laboratory (2011)
9. K. L. DERSTINE, "DIF3D: A Code to Solve One-, Two-, and Three-Dimensional Finite-Difference Diffusion Theory Problems," ANL-82-64, Argonne National Laboratory (1984).
10. D. B. PELOWITZ et al., "MCNP – A General Monte Carlo N-Particle Transport Code," LA-CP-13-00634, Los Alamos National Laboratory, May (2013).
11. DAVID CHANDLER et al., "Conceptual Design Parameters for HFIR LEU U-Mo Fuel Conversion Experimental Irradiations," ORNL/LTR-2013/132, Oak Ridge National Laboratory (2013).