DEVELOPMENT OF A FAST SPECTRUM IRRADIATION FACILITY FOR FUELS DEVELOPMENT IN THE HIGH FLUX ISOTOPE REACTOR

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Abstract

A concept for a fast spectrum irradiation facility has been developed for insertion in the High Flux Isotope Reactor at Oak Ridge National Laboratory. The design is based on the very large fast flux that is available in this reactor combined with the use of a strongly-absorbing thermal neutron shield. The preferred concept from the several considered consists of a three-pin design surrounded by a Eu_2O_3 thermal neutron shield located in the reactor flux trap. Preliminary analyses showed that this concept can provide a fast flux larger than 1×10^{15} n/cm^2·s and a fast-to-thermal flux ratio greater than 300 while having an acceptable impact on the HFIR operation. Additional analyses are necessary to confirm that this design is feasible and meets the requirements for fast fuel irradiation. If the design proves to be suitable, it can provide a relatively low-cost, near-term capability.

Introduction

The Global Nuclear Energy Partnership (GNEP) is proposing to develop a sodium-cooled fast-spectrum reactor (SFR) to transmute and consume actinides from spent nuclear fuel. The proposed fuels include metal and oxide mixed actinides (U-Np-Pu-Am-Cm) as well as target concepts with perhaps only Am-Cm. These concepts require the development of new and advanced fuels and targets that meet the performance objectives, including further development of fuel types and compositions. A significant part of the development process is the irradiation of the fuel and cladding system in a prototypic reactor environment to determine the performance of the fuels and cladding under irradiation.

A fast-spectrum irradiation facility does not exist in the United States. However, there is a strong desire to establish a near-term capability to irradiate materials in a fast spectrum in addition to efforts to gain access to international facilities through partnering arrangements. Currently, the DOE operates two high-power research reactors that potentially can be considered for materials and fuels irradiation: the High-Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) with the concept discussed in this paper and the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL) with a Boosted Fast Flux Loop as described in Ref. 1. An additional concept for a materials test station (MTS) at the Los Alamos National Laboratory (LANL) LANSCE facility has also been proposed and is discussed in Ref 2.

HFIR has a distinct advantage for consideration as an irradiation facility because of the extremely high neutron fluxes that this reactor provides over the full thermal to fast neutron energy range. As described below, an assessment of the capabilities for fast spectrum irradiations of the available irradiation positions in HFIR was performed, and it was found that the most appropriate location is the central flux trap target region, which is typically used for the irradiation of curium targets and other materials. The fast flux (> 0.1 MeV) in this region exceeds 1×10^{15} n/cm^2·s. The combination of this high fast flux and the use of a highly absorbing europia (Eu_2O_3) liner to eliminate thermal neutrons provides an irradiation environment that approximates that of a fast reactor. The primary benefits of this approach are that the irradiation facility is already in place and in use, the shielding technology has already been developed, and additional fast-flux boosting fuel plates or pins are not required. Therefore, this concept would provide a near-term capability that would be low-cost and based on target design and irradiation experience currently available at ORNL.

HFIR Description

The HFIR is a versatile, 85-MW isotope production and test reactor with the capability and facilities for performing a wide variety of irradiation experiments [3]. The HFIR provides one of the highest steady-state neutron fluxes available in any of the world's reactors. The original primary purpose of the HFIR was the production of transuranium isotopes; however, the current mission of HFIR has shifted to neutron sciences with the installation of a cold neutron source and the use of the available beam lines.

The HFIR is a beryllium-reflected, light-water-cooled and -moderated, flux-trap type reactor that uses highly enriched uranium-235 as the fuel. The reactor core consists of a series of concentric annular regions, each approximately 2 ft (0.61 m) high. A 5-in. (12.70-cm)-diameter hole, referred to as the "flux trap," forms the center of the core. The fuel region is composed of two concentric fuel elements. The

inner element contains 171 fuel plates, and the outer element contains 369 fuel plates. The fuel plates are curved, with an involute shape, thus providing a constant coolant channel width. The fuel (U_3O_8-Al cermet) is nonuniformly distributed along the arc of the involute to minimize the radial peak-to-average power density ratio. A burnable poison (boron) is included in the inner fuel element primarily to reduce the negative reactivity requirements of the control plates. A typical core loading is 9.4 kg of uranium-235 with 2.8 g of boron-10 in the inner fuel element.

The fuel region is surrounded on the radial direction by a concentric ring of beryllium reflector approximately 1 ft (0.30 m) thick. This reflector consists of three regions: the removable reflector, the semi-permanent reflector, and the permanent reflector, as shown in Figure 1. The beryllium reflector is surrounded by a water reflector. In the axial direction, the reactor is reflected by water. A graph showing an overview of the available neutron fluxes in the HFIR is given in Figure 2. Note that the shown fluxes are unperturbed fluxes at 100 MW and should be reduced to 85% to account for the current power level of 85 MW. As this figure shows, the fast flux in the flux trap region exceeds 10^{15} n/cm^2·s, which is a key feature of HFIR that is exploited in the concept proposed in this paper.

A fuel cycle for the HFIR normally consists of full-power operation at 85 MW for a period of time from 22 to 26 days (depending on the experiment and radioisotope load in the reactor), followed by an end-of-cycle outage for refueling. Experiment insertion and removal may be accomplished during any end-of-cycle outage. Deviations from the schedule are infrequent and are usually caused by periodic changeout of major reactor components, reactor and experiment component malfunctions, etc. On average, seven cycles are performed per year.

Several materials irradiation facilities were provided in the HFIR original design and others have been added over time. Irradiation facilities available include (1) the hydraulic tube facility, located in the very high flux region of the flux trap, which allows for insertion and removal of irradiation samples while the reactor is operating; (2) thirty target positions in the flux trap, which normally contain transuranium production rods or materials irradiation experiments; (3) six peripheral target positions located at the outer edge of the flux trap; (4) numerous vertical irradiation facilities of various sizes located throughout the beryllium reflector; (5) two pneumatic tube facilities in the beryllium reflector, which allow for insertion and removal of irradiation samples while the reactor is operating for activation analysis; (6) four horizontal beam tubes, which originate in the beryllium reflector; and (7) four slant access facilities, called "engineering facilities," located adjacent to the outer edge of the beryllium reflector. In addition, spent fuel assemblies are used for gamma irradiation in the gamma irradiation facility in the reactor pool.

Proposed Irradiation Concept

Three options for irradiation facilities were considered: (1) a shielded single pin design that fits into a target location of the central flux trap region; (2) a single pin design with flux boosting fuel plates located in the removable beryllium reflector region; and (3) a tri-pin design that occupies seven target locations (a “septa-foil” design) in the central flux trap region and provides an increased irradiation volume. As the analysis performed for the boosted removable beryllium design indicated that the fast flux levels were too

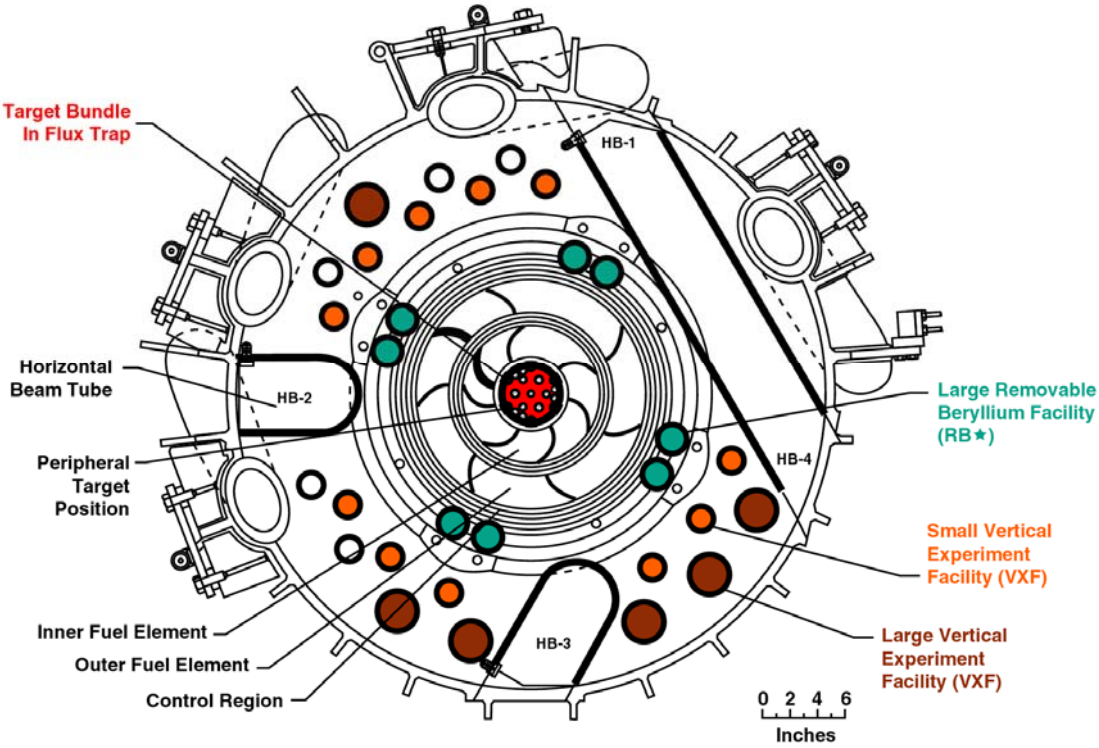

Figure 1. Plan view of HFIR showing experiment irradiation locations

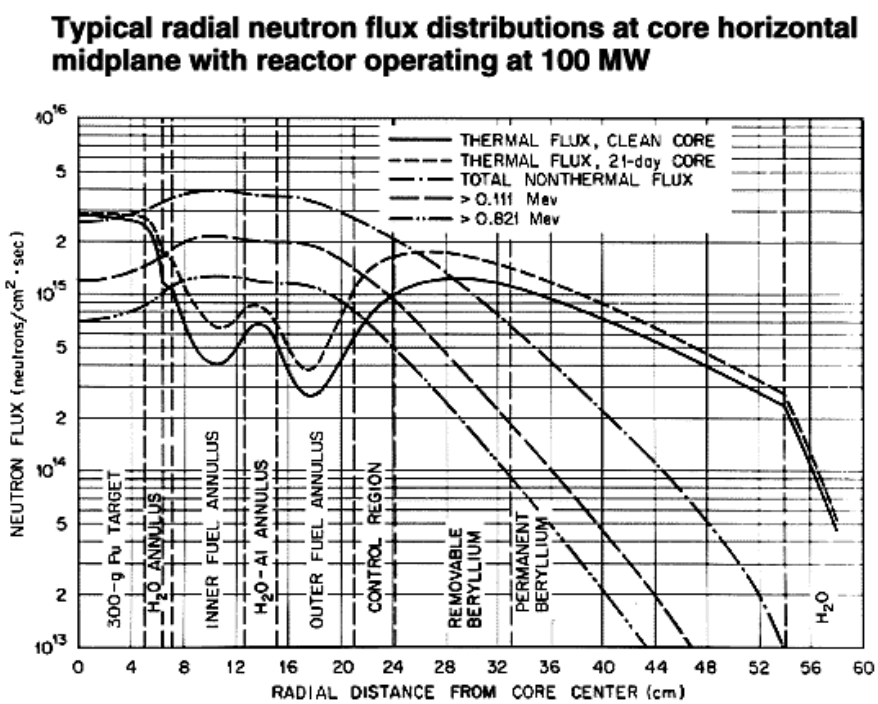

Figure 2. HFIR neutron flux vs radial location at midplane [3].

low, this design was removed from further consideration. The performance parameters for the two flux trap designs were found to be similar; however, since the tri-pin design provides a larger irradiation volume, it was selected as the preferred concept. Figures 3 and 4 show the details of the shielded tri-pin design and its location in the central flux trap.

The key design features of this target design is that it is sized to fit in seven flux-trap irradiation positions, and the thickness of the europium-oxide liner was chosen to match the desired performance characteristics (linear heat generation rate, fast/thermal neutron flux ratio, etc.) as closely as possible. Within the liner, the amount of coolant was minimized to ensure the low thermal neutron flux level that can be obtained with a strongly absorbing liner, yet provide adequate coolant flow to maintain experiment temperatures.

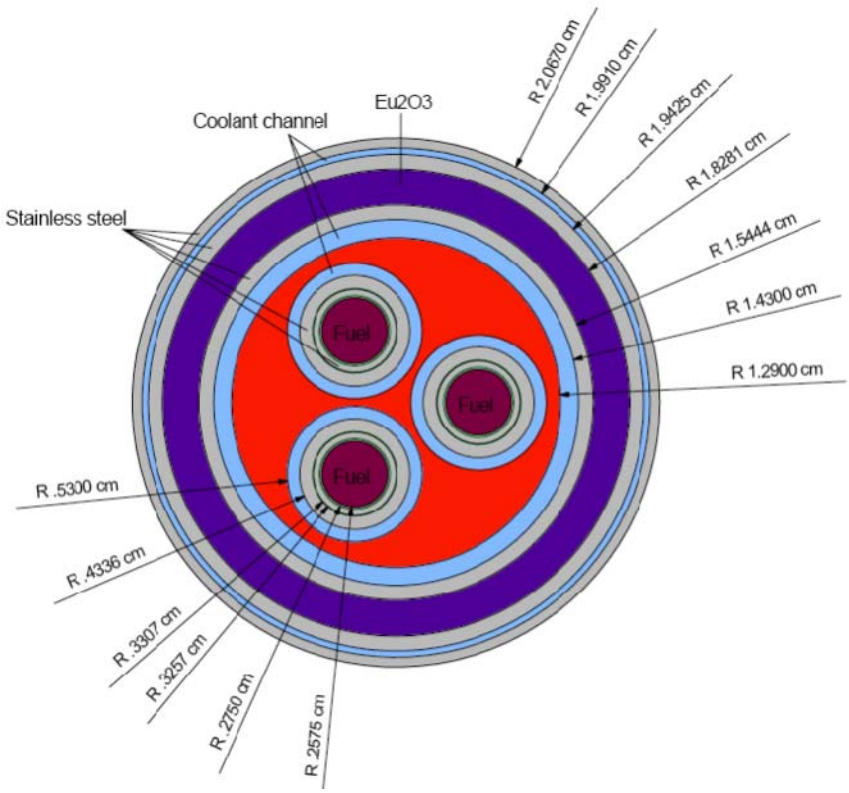

Figure 3. Tri-pin HFIR flux trap shielded target design.

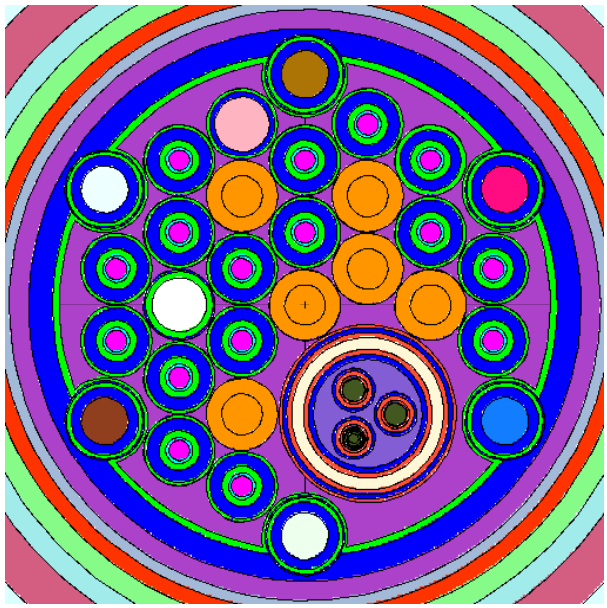

Figure 4. Tri-pin target design shown in HFIR flux trap location

Performance characteristics of this target were calculated using the MCNP Monte Carlo Code [4] based on the ORNL reference HFIR model [5]. The calculations were performed to determine the key performance parameters that were requested including linear heat generation rate (LHGR), neutron flux magnitude, fast -to-thermal flux ratio, dislocations per atom (DPA), etc. A summary of the analysis is provided in Table 1. Analysis of the effect of the burnup of the liner has yet to be performed to ensure that flux variations over the fuel cycle are acceptable.

The performance characteristics of the shielded target compare favorably with those needed to perform fast flux irradiations with a fast neutron flux greater than 1×10^{15} n/cm^2·s and a fast-to-thermal flux ratio greater than 300 (the fast neutron range corresponds to energies > 0.1 MeV; the thermal neutron range corresponds to energies < 0.625 eV). The burnup and DPA rates are within the range of desired performance measures. The radial fission density profile for the fresh fuel pin in the shielded target indicates that there is a slight "rim-effect" that is not typical of fast reactor irradiations, but a value of less then 10% is acceptable. This effect difficult to avoid in HFIR because of the presence of epi-thermal neutrons and may require additional evaluation to investigate the impact of this radial fission density distribution and the period of time until the surface peaking burns down. With the appropriate thickness of the europium oxide shield and flexibility in the fuel pellet diameter, the linear heat generation rate is well within the desired value.

Temperature control in the flux trap is commonly accomplished without instrumentation through appropriate design. However, the flux trap region has two target positions capable of irradiating fully instrumented, temperature-controlled experiments. Experience has shown that temperatures can be very closely controlled with a fully instrumented experiment throughout a cycle. Historical experience on temperature-controlled irradiation experiments performed in the Removable Beryllium (RB) irradiation

positions in HFIR indicate that the temperature at any given thermocouple reading can be readily controlled to within 2 °C over the duration of the full-power irradiation cycle. Depending on the size of the fuel pellet and the thickness of the shield, multiple adjacent target positions may be required to provide adequate space for inert gas flows and controllable coolant channels.

Actinide-based targets are routinely irradiated in the HFIR flux trap. Given that the proposed concept would use a double-encapsulated design, there is no limitation on the fuels or materials that can be irradiated.

Table 1. Performance parameters for HFIR tri-pin design

Parameter	HFIR Value	Parameter	HFIR Value
Fast (> 0.1 MeV) neutron flux in irradiation volume	1.2×10^{15} n/cm^2·s (midplane)	Linear heat rate	280 W/cm (380 W/cm including gamma heating) (variable depending upon design of Eu_2O_3 shield)
Annual fast (> 0.1 MeV) neutron fluence at peak irradiation position	1.7×10^{22} n/cm^2 per year 5.9% burnup/year 19 DPA/year (miplane values based on 7 reactor cycles per year)	Fuel clad irradiation temperature	350ºC-550ºC is within the achievable range.
Minimum irradiation volume	Sufficient volume for 30 fuel pellets and 30 rodlets with minimum 10 cm high fuel height	Materials sample irradiation temperature range	Materials irradiation experiments in HFIR are routinely designed for temperatures between 250ºC - 1300ºC .
Radial dependence of fission rate (gradient) in fresh fuel pellet at peak irradiation position	~20% ratio of fission rate from pellet edge to center	In-situ measurement capability	Experiments can be instrumented with multiple thermocouples

Impact on Reactor Performance

As previously stated, HFIR's primary mission is to support applications related to neutron science and therefore it is important that the insertion of this experiment not have an adverse impact on these activities. In addition, there are safety requirements that must be met to ensure that the reactor does not

operate outside of its approved design margins. Criteria have been established for two key parameters that are likely to be impacted by the insertion of a highly-absorber experiment: (1) cycle length must be maintained at a value greater than 22 days; and (2) power distribution in the fuel elements must not be impacted by more than 9%. Additional analysis, per the standard irradiation experiment procedure, would need to be performed before the actual insertion of the experiment in the reactor.

The impact of the insertion of the tri-pin design on the cycle length is presented in Table 2 along with three approaches for offsetting the reactivity loss. The tri-pin target design alone is worth nearly $3 in reactivity, which may reduce the cycle length below the required 22-day, depending upon other experiments that are loaded in the reactor. Three options for offsetting this negative reactivity include: (1) use of beryllium inserts in four of the removable beryllium sites (adding 0.9 days); (2) adding beryllium targets to the flux trap location (adding an additional 0.7 days); and (3) adding additional aluminum to displace water in the flux trap. These options provide sufficient reactivity addition to meet the 22 day requirement.

The impact on the power distribution was computed for both BOC and EOC conditions to ensure that the insertion of the strong absorber do not increase the power density by more than 9% at the hot spot locations, as required by the HFIR safety analysis. Calculations of the power distributions were performed with MCNP and are presented in Table 3. As these results show, the impact on the power distribution is well within the required 9%.

Table 2. Impact on HFIR cycle length for tri-pin design

Case identification	Reactivity Worth ($)	Estimated Impact on Cycle Length (Days)
(a) Tri-pin in HFIR Flux Trap	-2.9	-3.6
(b) Same as (a) but with removable beryllium positions filled with beryllium rather than aluminum	-2.0	-2.5
c) Same as (b) but with 12 beryllium target rods in flux trap	-1.5	-1.8
d) Same as (c) but with water in flux trap displaced by aluminum	-1.2	-1.5

Table 3. Potential impact on fuel element power distribution as indicated by the maximum increase in power density at the inner and out element hot spot locations for Tri-pin concept in flux trap.

Case	Inner Element	Outer Element
BOC	1.8 ± 0.8 %	3.3 ± 0.9 %
EOC	1.4 ± 0.8 %	4.3 ± 1.0 %

ORNL Experiment Design and Irradiation Experience

The technology and experience necessary to design and build this type of experiment has already been developed. HFIR has hosted many experiments designed to qualify nuclear fuel. For example, the New Production Reactor (NPR) program sponsored two experiments in the outer reflector for the purpose of qualifying tritium-producing nuclear fuel. The Modular High-Temperature Gas-cooled Reactor (MHTGR) program sponsored approximately twenty fuel qualification experiments in the HFIR RB positions. Although fuel qualification experiments have not been conducted in the flux trap region of the HFIR to date, fissile material (curium) is routinely placed in the flux trap for the production of californium. Likewise, thermal neutron shields have been used and continually improved for many years in HFIR. The lifetime of the existing liners in the removable beryllium positions is seven cycles. The proposed concept is similar to the standard irradiation target pin that has been used dozens of times over many years by many different programs, and it can be either instrumented or uninstrumented.

In summary, there is nothing about the proposed fast-reactor fuel irradiation that is fundamentally different than previous experiments. The thermal-neutron shield that would be required for this experiment is a well-developed technology that has been used many times before in HFIR, and HFIR management has established guidelines for its use. There have been numerous fueled experiments in the reflector region of HFIR, and fissile material is routinely irradiated in the flux trap region.

Summary and Conclusions

A concept has been developed to provide a capability to fast-flux irradiations at HFIR. This proposed concept:

1. Requires no large capital investment to develop a new facility. The cost is essentially that of the design, manufacture, and irradiation of the experiment.

2. Provides performance parameters that are near those desired for fast fuel irradiation. There are a few exceptions that may require additional evaluation on impact to the irradiation performance.

3. Is well-within the ORNL design and HFIR irradiation experience.

4. Can provide a very near term capability with no reactor modifications required.

Additional design optimization and performance improvement can be performed, but short of the addition of neutron flux boosting pins or plates, the parameters provided represent reasonable expected values. Operation of spectrally-shielded, temperature-controlled experiments with sweep-gas monitoring for fission product release is within the current operating envelope for the reactor. The major cost expenditure is associated with the design, fabrication and irradiation of monitored and temperature-controlled irradiation capsules. Additional analysis is planned to assess the impact of the burnup of the europia liner and the ability to maintain temperature control over the irradiation period.

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