Analysis of a fast spectrum irradiation facility in the high flux isotope reactor

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Abstract

The Global Nuclear Energy Partnership (GNEP) is proposing to develop a sodium-cooled fast-spectrum reactor (SFR) to transmute and consume actinides from discharged nuclear fuel. To meet performance objectives, new and advanced fuels and targets need to be developed. The fuels to be irradiated include metal and oxide mixed actinides (U-Np-Pu-Am-Cm); for the target concept, Am-Cm has been considered. A significant part of the development process is the irradiation of the fuel and cladding in a prototypic fast reactor environment to determine the performance under irradiation. Analysis results are presented in this paper for a fast-neutron irradiation facility design based on the large fast neutron flux available in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL) combined with the use of a strongly-absorbing thermal neutron shield. Several designs were assessed; the preferred concept consists of a three-pin design with an europium oxide thermal neutron shield, to be situated in the HFIR flux trap. Analyses show that this design could provide a fast to thermal neutron flux ratio greater than 400, a fast neutron flux larger than 1×10^{15} n/cm^2·s, and with an acceptable impact on HFIR operation. This design feasibly will be a relatively low-cost near-term facility that meets the requirements for fast fuel irradiation.

1. 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. To meet the performance objectives, new and advanced fuels and targets need to be developed. The fuels to be irradiated include metal and oxide mixed actinides (U-Np-Pu-Am-Cm); for the target concept, Am-Cm has been considered. 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 neutron spectrum in addition to efforts to gain access to international facilities through partnering arrangements. Currently, the U.S. Department of Energy (DOE) operates two high-power research reactors that potentially can be considered for materials and fuels irradiation: the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL), and the High Flux Isotope Reactor (HFIR) at Oak Ridge National

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Laboratory (ORNL)) (Cheverton and Sims, 1971). An additional concept for a materials test station (MTS) at the Los Alamos National Laboratory (LANL) Los Alamos Neutron Science Center (LANSCE) facility has been proposed (Pitcher, 2007).

HFIR has a distinct advantage for consideration as an irradiation facility because of the extremely high neutron fluxes provided over the full thermal to fast neutron energy range. An assessment of the use of the available irradiation locations in HFIR was performed and will be further described in this paper. The HFIR region considered the most appropriate for fast-neutron irradiation applications is the central flux trap target region that 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 europium-oxide (Eu_2O_3) liner to eliminate thermal neutrons will ensure an irradiation environment that closely approximates that of a fast reactor. Primary benefits of this approach are that the irradiation facility is already in place and in use, the thermal-neutron shielding technology is already developed and additional fast-flux boosting fuel plates or pins are not required.

2. HFIR description

The HFIR (Cheverton and Sims, 1971) is a versatile, 85-MW isotope production and test reactor with the capability and facilities for performing a wide variety of irradiation experiments. The HFIR (Fig. 1) is a beryllium-reflected, light-water-cooled and -moderated, flux-trap type reactor that uses highly enriched ^{235}U as the fuel.

The HFIR is unique in the sense that it provides one of the highest steady-state neutron fluxes available in any of the world's reactors, and the neutron currents from the four horizontal beam tubes are among the highest available. 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.

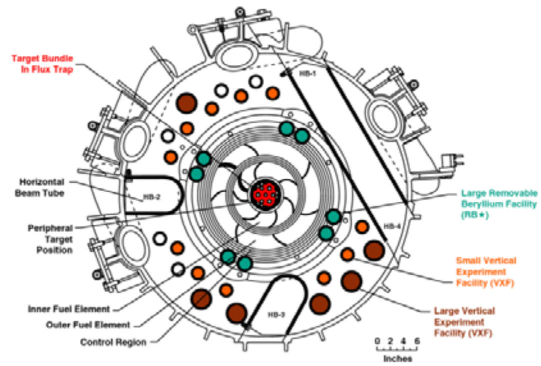

Fig. 1. View of the HFIR reactor system at midplane.

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)-diam hole, referred to as the flux trap forms the center of the core. The fuel region is composed of two concentric fuel elements made up of many involute-shaped fuel plates: the inner element contains 171 fuel plates, and the outer element contains 369 fuel plates. The fuel plates are curved in the shape of an involute, which provides constant coolant channel width between plates. 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 (Ellis et al, 2006). A burnable poison (B_4C) is included in the inner fuel element primarily to reduce the negative reactivity requirements of the reactor control plates. A typical highly enriched uranium (HEU) core loading in HFIR is 9.4 kg of ^{235}U and 2.8 g of ^{10}B.

Fig. 2 shows the inner and outer fuel elements and the enclosed flux trap region, with its numerous target irradiation locations. The thermal neutron flux in the flux trap region can exceed 2.5×10^{15} n/cm^2·s while the fast flux in this region exceeds 1×10^{15} n/cm^2·s; this feature makes a fast irradiation facility in the HFIR flux trap region viable.

The inner and outer fuel elements are in turn surrounded by a concentric ring of beryllium reflector approximately 1 ft (0.30 m) thick. The beryllium reflector consists of three regions: the removable reflector, the semi-permanent reflector, and the permanent reflector. It is surrounded by a water reflector of effectively infinite thickness. In the axial direction, the reactor is reflected by water above and below the reactor.

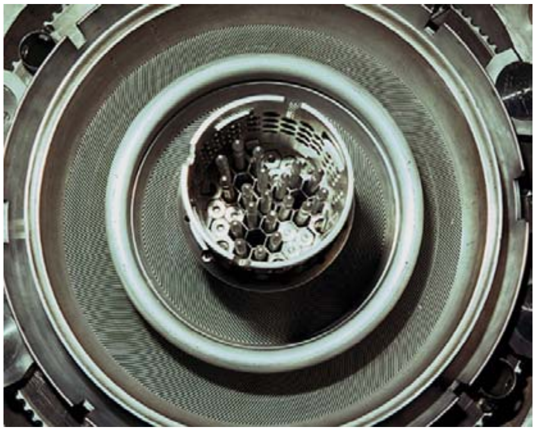

Fig. 2. The HFIR fuel elements and flux trap region

A fuel cycle for the HFIR normally consists of full-power operation at 85 MW (thermal) for a period of 22 to 26 days (depending on the experiment and radioisotope load in the reactor), followed by an end-of-cycle outage of two to three weeks for refueling, control plate changeout, calibrations, maintenance, and inspections. 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 for activation analysis while the reactor is operating; (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. Additionally, spent fuel assemblies are used for gamma irradiation in the gamma irradiation facility in the HFIR pool.

3. Irradiation assembly design concepts

Several design options were considered for fast-spectrum irradiation including the following: (1) a shielded single pin that fits into a flux trap target location; (2) a somewhat larger single pin that provides for additional coolant flow, (3) a single-pin design with flux-boosting fuel plates for situating in an irradiation site in the removable beryllium reflector, and (4) a tri-pin design (illustrated in Fig. 3) that occupies seven target locations in the HFIR flux trap region (a "septa-foil" design) and provides an increased irradiation volume. The analyses performed for the boosted irradiation facility design (#3) for the removable beryllium reflector indicated that the fast neutron flux levels were too low so this design was not deemed viable for the current purposes. The performance parameters for the designs intended for the flux trap region (#1,#2,#4) were similar, but since the tri-pin design provided a larger irradiation volume, it was selected as the preferred fast-neutron irradiation facility design. The tri-pin assembly design provides flexibility, a fast-reactor irradiation environment, and supports multiple irradiation targets. Fig. 3 is an illustration of the tri-pin irradiation facility indicating the main components including the three fuel pins and the europia (Eu_2O_3) neutron shield.

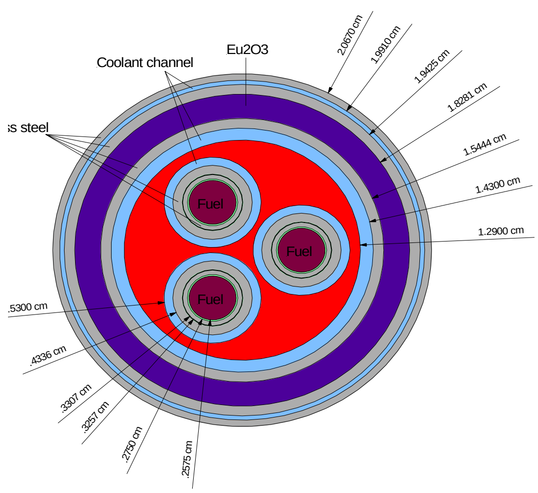

Fig. 3. Tri-pin fast-flux shielded irradiation target design

The key design features of this irradiation facility target design are (1) it is sized to fit within the irradiation volume available when seven existing target sites are vacated and (2) the thickness of the europium-oxide liner was chosen to match the desired performance characteristics, such as linear heat generation rate (LHGR), fast/thermal ratio, as closely as possible. Within the europia shield liner the amount of coolant was limited to keep the low thermal neutron flux level that can be obtained with the strongly absorbing liner, yet provide adequate coolant flow to maintain appropriate experimental temperatures.

This pin design, which provides a sizable irradiation volume, is the currently preferred design. Performance parameters were computed for all design configurations considered; those corresponding to the tri-pin, Eu_2O_3 shielded flux trap target are included in this paper.

Fig. 4 shows the tri-pin irradiation facility located in the central flux trap region of the HFIR. As mentioned, this irradiation facility fills seven target pin locations in this region.

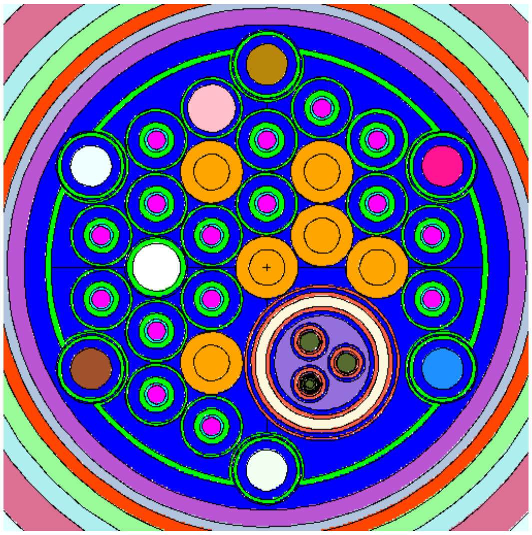

Fig. 4 Flux trap region of HFIR with the tri-pin assembly.

4. Analysis results

The usual neutron flux spectrum in the HFIR flux trap region, with the very large thermal neutron component clearly seen, is illustrated in Fig. 5. The neutron flux spectrum within the europia shield of the irradiation facility is seen to be predominantly made up of fast neutrons, as the thermal neutrons have been effectively removed.

Performance characteristics of this target were estimated using the MCNP5 Monte Carlo Code (X-5 Monte Carlo Team, 2005) calculations based on ORNL reference HFIR models (Peplow, 2004; Xoubi and Primm III, 2005). The calculations were performed to determine the key performance parameters that were requested including linear heat generation rate, neutron flux level magnitude, fast-to-thermal neutron flux ratio, displacements per atom (DPA), helium (He) production, etc. The specific results along with the proposed requirements are provided in Table 1 on the following page.

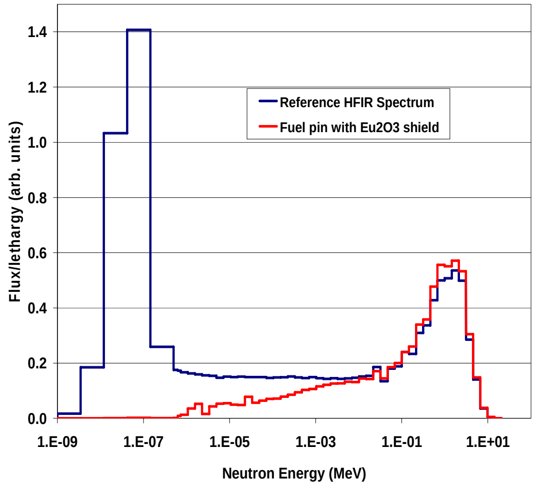

Fig. 5. Neutron flux spectra within the flux trap region of HFIR and the tri-pin fast irradiation facility

The performance characteristics of the shielded target compare favorably with those needed to perform fast flux irradiations with a fast neutron flux (>0.1 MeV) $>1\times10^{15}$ n/cm^2·s and a fast to thermal (< 0.625 eV) flux ratio greater than 300. The burnup and DPA rates are within the range of given performance measures. The radial fission density profile for a fresh fuel pin indicates that there is a slight “rim-effect” that is not typical of fast reactor irradiations; however, a value of less then 10% is acceptable. This is difficult to avoid in HFIR because of the presence of epi-thermal neutrons and may require additional evaluation to investigate the impact of this distribution and the period of time until the surface power peaking burns down and subsides. 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. Detailed analysis of the effect of the burnout of the europia liner needs to be performed to ensure that the neutron flux variations during the HFIR fuel cycles and the irradiation period are acceptable.

Table 1
HFIR irradiation parameters for the proposed design.

Parameter	HFIR Value
Fast (>0.1 MeV) neutron flux in irradiation volume	1.2×10^{15} n/cm^{2}s
Annual fast (>0.1 MeV) neutron fluence at peak irradiation position	1.7×10^{22} n.cm^{-2} per year 5.9% burnup/year 19.5 DPA/year (based on seven reactor cycles per year)
Fuel burnup–to–clad DPA ratio at peak irradiation position	0.3 atom % per DPA
Fast (>0.1 MeV) to thermal (<0.625 eV) neutron flux ratio	400
He–to–DPA ratio in Fe at peak irradiation position	0.2 appm He per DPA
Linear heat rate	280 W/cm (380 W/cm with the inclusion of gamma heating) (variable depending upon design of Eu_2O_3 shield)

Fig. 6 shows the expected fast neutron flux distribution in the axial direction in the tri-pin irradiation facility at beginning of cycle (BOC) and end of cycle (EOC). The fast neutron flux levels are seen to be greater than 1×10^{15} n/cm^{2}s for a substantial length of the irradiation facility.

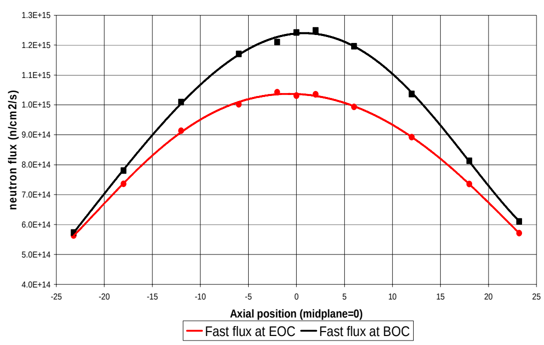

Fig. 6. Axial distribution of fast neutron flux levels in the irradiation facility

Limited analysis has been performed to date to assess the impact on the reactor operation. One concern is the potential reduction in the HFIR fuel cycle length owing to reactivity loads. Based on the MCNP calculations, the reactivity worth of the tri-pin design is comparable to the use of two europium-oxide shielded experiments in the large removable beryllium facilities (RB*) in the reflector (see Fig. 1), for which there is HFIR experimental experience. Therefore, the impact on this key parameter is not outside of current experience.

The insertion of a tri-pin irradiation facility in the HFIR flux trap region was estimated to have a reactivity worth of approximately -\$2.9 which is estimated to have an impact on HFIR cycle length of about -3.6 days.

Three options for offsetting this negative reactivity include: (1) the 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 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 minimum HFIR fuel cycle length requirement. The net result of these measures is that the reactivity worth of the tri-pin assembly would be about -\$1.2 with a cycle length reduction of 1.6 days.

An important and necessary HFIR safety and operational requirement is to ensure that the HFIR fuel element power distribution computed for both BOC and EOC operational conditions are not perturbed or tilted by more than the maximum 9% limit for experimental power density changes at the HFIR fuel element hot spots. Calculations of the distributions were performed with MCNP and it was seen that the maximum increase for the inner fuel element hot spot (at the lower outer edge) is 1.8±0.8% at BOC and 1.4±0.8% at EOC. The maximum increase at the outer fuel element hot spot (at the lower inner edge of the outer fuel element) is 3.3±0.9% at BOC and 4.3±1.0% at EOC. These results show that the insertion of the tri-pin irradiation assembly has an acceptable power tilt impact on the power distribution, well within the 9% maximum requirement.

Additional analysis, per the standard irradiation experiment procedure, would need to be performed before the actual insertion of the experiment in the reactor. HFIR safety procedures require a full safety assessment report to include necessary calculations performed with QA qualified codes, descriptions of methods, material details, and model differences to base cases.

5. Summary and conclusions

The HFIR reactor has numerous irradiation sites available in which facilities can be inserted to allow irradiation in a particular neutron spectrum for a chosen neutron energy range and neutron flux levels. A concept has been developed to provide fast-flux irradiations in the HFIR to support GNEP. The proposed design concept provides performance parameters that are consistent with the target values specified in the GNEP program objective. Additional studies on the impact of the experiment insertion to the HFIR irradiation performance need to be carried out for a complete evaluation. The major advantage of the proposed design is that it is compatible with the existing HFIR design and operational experience and it can provide a very near-term capability with no reactor modifications required. As stated, this concept should provide a near-term capability that would be low-cost and based on target design and irradiation experience currently available at ORNL.

In addition to fast-neutron irradiation facilities, plans for experimental irradiation of pressurized water reactor (PWR) fuel pins have been designed for the HFIR. This irradiation facility is to be located in the beryllium reflector region of HFIR. The assembly will utilize a thin Hf shield approximately 4 cm in diameter surrounding the facility basket to reduce the thermal neutron flux sufficiently such that the linear power rating in the irradiated fuel pins will be similar to PWR operating conditions.

Acknowledgment

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