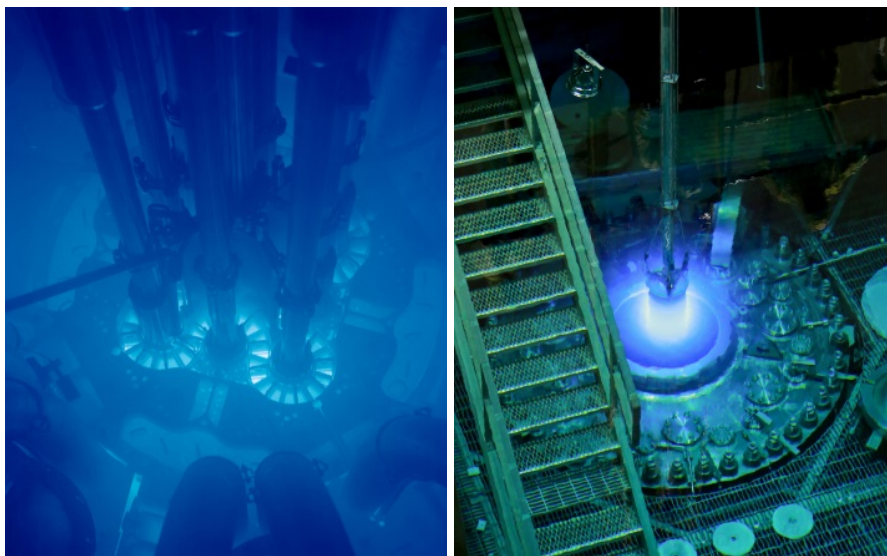


Preliminary Options Assessment of Versatile Irradiation Test Reactor: Neutron Fluxes in Test Reactors

Gilles J. Youinou

January 2017

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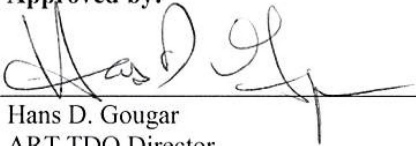


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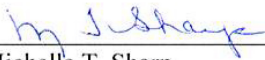
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ACRONYMS

AL	aluminum
ANL	Argonne National Laboratory
ART	Advanced Reactor Technologies
ATR	Advanced Test Reactor
BOR-60	fast research reactor in Russia
dpa	displacements per atom
EBR-II	Experimental Breeder Reactor II
FFTF	Fast Flux Test Facility
HFIR	High Flux Isotope Reactor
HT	Hydraulic Tube
INL	Idaho National Laboratory
JHR	Jules Horowitz Reactor
JOYO	Japan Experimental Fast Reactor
ORNL	Oak Ridge National Laboratory
PHYSOR	Physics of Reactors
PTP	Peripheral Target Positions
SST	stainless steel

PRELIMINARY OPTIONS ASSESSMENT OF VERSATILE IRRADIATION TEST REACTOR: NEUTRON FLUXES IN TEST REACTORS

1. INTRODUCTION

This report addresses the Applicability of water-cooled Material Test Reactors (MTRs) to the testing needs of current and future advanced reactor concepts. It forms part of the INL deliverables analyzing some of the options for designing a versatile test reactor (Sen et al. 2017). It is shown that high-power water-cooled MTRs such as the Advanced Test Reactor (ATR), the High Flux Isotope Reactor (HFIR), and the Jules Horowitz Reactor (JHR) cannot provide fast flux levels that are as high as fast flux levels produced in sodium-cooled fast-test reactors.

2. BASIC PHYSICS CONSIDERATIONS ABOUT NEUTRON FLUXES IN TEST REACTORS

In this section, some of the attributes and relevant reactor physics parameters of materials test reactors are discussed to provide the background against which different MTRs can be evaluated. Data from the ATR, HFIR, and JHR are then used to compare the performance of modern high-power water-cooled test reactors in terms of these physics parameters.

1. A high fast flux in the driver fuel is not a necessary requirement to obtain a high thermal flux in a test zone.

To illustrate this point, let's assume an idealized case where the fission neutrons born in the driver fuel zone, D , are either absorbed in the same driver zone or reach the test zone, T , where they are thermalized and eventually absorbed. In this case, the fast flux in the driver zone, ϕ_D^{fast} , and the thermal flux in the test zone, ϕ_T^{th} , are related by the following neutron balance equation:

$$[\nu\Sigma_{fis} - \Sigma_{abs}]_D^{fast} \phi_D^{fast} + [\nu\Sigma_{fis} - \Sigma_{abs}]_D^{th} \phi_D^{th} = [\Sigma_{abs}]_T^{th} \phi_T^{th} \quad (1)$$

$$\phi_T^{th} = \frac{[\nu\Sigma_{fis} - \Sigma_{abs}]_D^{fast} \phi_D^{fast} + [\nu\Sigma_{fis} - \Sigma_{abs}]_D^{th} \phi_D^{th}}{[\Sigma_{abs}]_T^{th}}$$

Hence, independent of the characteristics of the driver zone providing the neutrons, the thermal flux in the test zone can be made large by designing it so that the neutron absorption, $[\Sigma_{abs}]_T^{th}$, is small. This can be accomplished in practice by using materials such as heavy water, graphite, or beryllium, which have very small thermal neutron absorption cross sections. Having a high flux in the driver zone is, of course, a plus, but, as explained above, it is not the only way to obtain a high thermal flux in a test zone.

2. A high fast flux in the driver fuel is necessary to obtain a high fast flux in a test zone.

As obvious as it may seem, it is important to emphasize this point. The only source of fast neutrons is the driver fuel, and the only way to obtain a high fast flux in a test zone is to ensure that these fast neutrons originating from the driver zone are not slowed down before they reach the test zone. A proper core design can ensure that the thermal flux is higher in a test zone than in the driver zone, but the fast flux in a test zone cannot be higher than that in the driver zone.

3. Obtaining a high fast flux in the driver fuel requires a high power density and a low thermal neutron flux.

The power density, P , in the driver fuel (i.e., the number of watts per unit mass of fissile material), the neutron flux, and the macroscopic fission cross sections are related by the following expression:

$$P \propto [\Sigma_{fis}]_D^{fast} \phi_D^{fast} + [\Sigma_{fis}]_D^{th} \phi_D^{th} \quad (2)$$

From this simple expression, it is clear that, for a given a core configuration, the higher the power density P , the higher the fast and thermal-neutron fluxes in the driver zone. This expression also highlights the fact that, for a given power density, the fast flux will be higher if the core is designed such that the thermal flux is minimized. Minimizing the thermal flux is all the more important when we consider that thermal fission cross sections are typically 2 to 3 orders of magnitude higher than fast fission cross sections. Figure 1 illustrates this point; whereas uranium-235 and plutonium-239 fission cross sections are between about 1 and 2 barns for fast neutrons above 10 keV, they are between a few hundred to a few thousand barns for thermal neutrons below 0.1 eV.

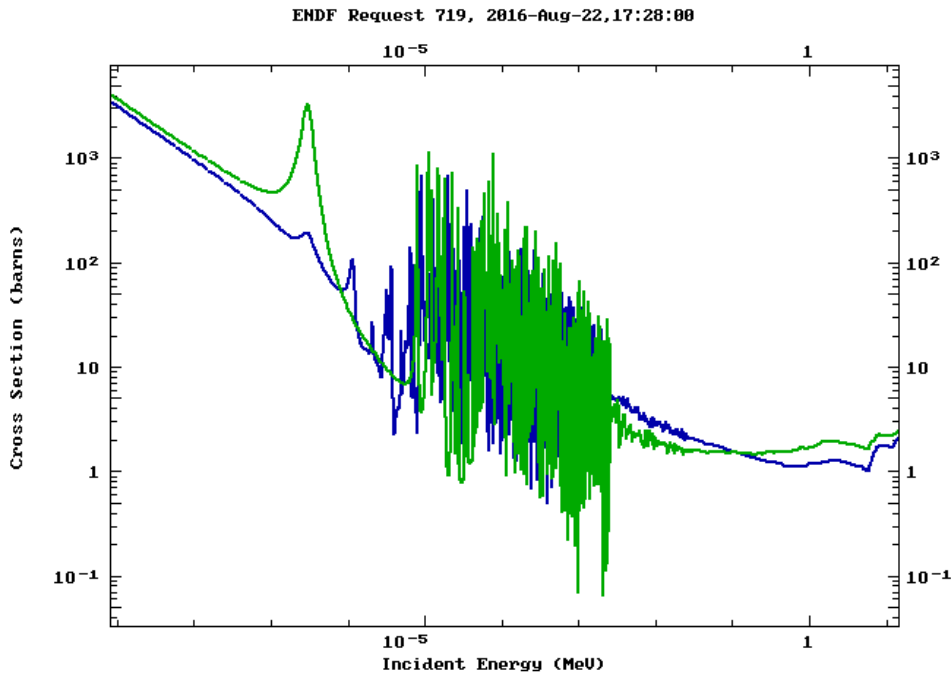


Figure 1. Fission cross sections of uranium-235 (blue) and plutonium-239 (green).

The most efficient way to suppress the thermal-neutron flux in the driver zone is to minimize the presence of certain light elements—in particular hydrogen^a—that would otherwise be responsible for significant neutron moderation. Reactors using coolants such as sodium,^b sodium-potassium, lead, or lead-bismuth fall into this category of design characterized by the presence of only a very small amount of thermal neutrons.

a. Hydrogen is a particularly efficient neutron moderator. Indeed, in the slowing-down region, between about 100 keV and 0.1 eV, its neutron elastic scattering cross section is about 20 barns, whereas that of deuterium, beryllium, and carbon—the other common neutron moderators—is only about 4 to 6 barns. Furthermore, neutron slowing down theory shows that, on average, it takes a neutron only about 15 collisions with hydrogen to slow down from 2 MeV to 1 eV, whereas it takes about 20, 70, and 90 collisions with deuterium, beryllium, and carbon, respectively.

b. Sodium is also a light material and is responsible for some neutron moderation. However, being about twice as heavy as graphite, sodium's capacity to slow down neutrons is only about half that of graphite, and, for all practical purposes, its neutron-moderation effect is negligible when used as a coolant. It takes a neutron about 170 collisions with sodium to slow down from 2 MeV to 1 eV. A large volume of sodium—too large to be relevant to the design of a reactor—would be necessary to thermalize a significant amount of fast neutrons.

On the other hand, in standard water-cooled test reactors such as ATR, HFIR, and JHR, the presence of water to cool the driver fuel elements is sufficient to generate a significant thermal neutron flux. Figure 2 illustrates the impact of the presence of water; the red curve is typical of the neutron spectrum in ATR, HFIR, and JHR water-cooled driver fuel elements, whereas the green curve is typical of the neutron spectrum in a fast test reactor such as the Fast Flux Test Facility (FFTF) and Experimental Breeder Reactor II in the United States or the Japan Experimental Fast Reactor (JOYO) in Japan.

Hence, because of the presence of thermal neutrons, the fast flux level attainable in a water-cooled test reactor is intrinsically more limited than that in a fast test reactor where the presence of thermal neutrons are several orders of magnitude lower.

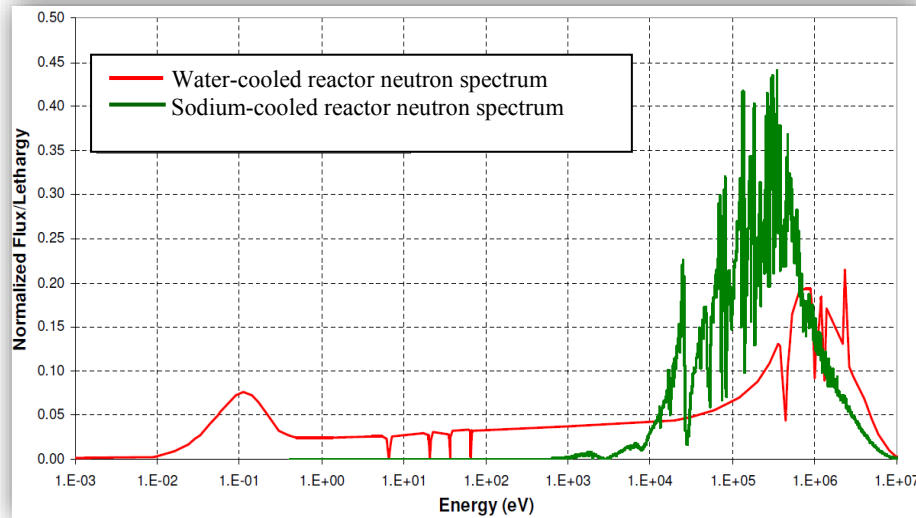
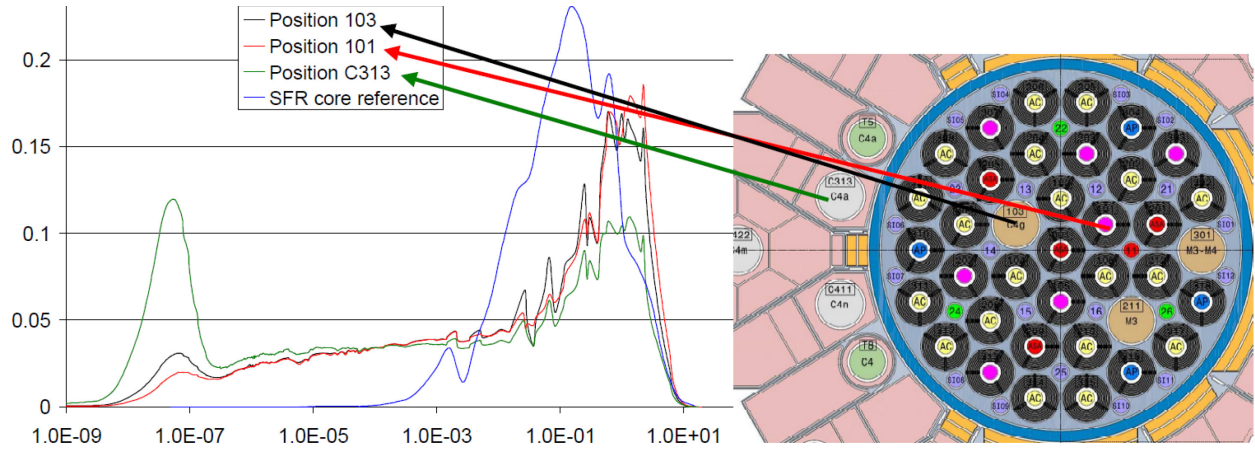


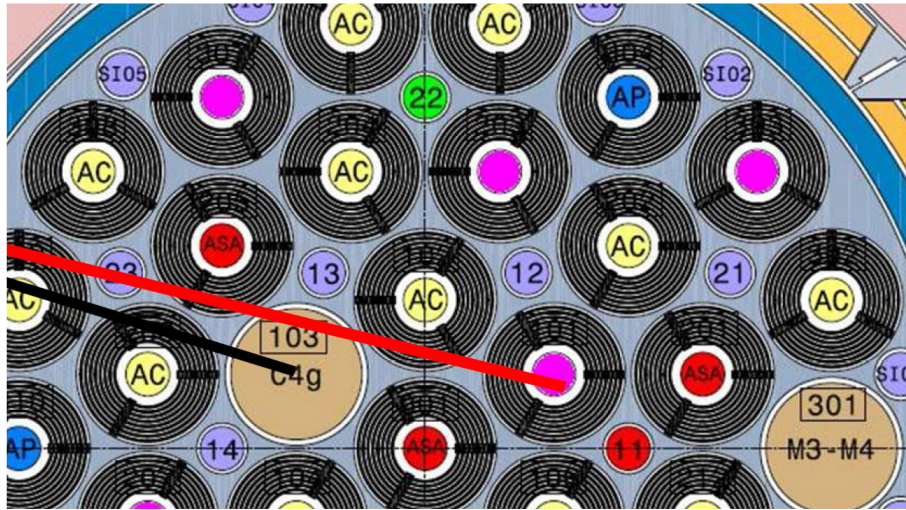
Figure 2. Typical neutron spectra for water-cooled reactors (red) and sodium-cooled reactors (green).

3. EXAMPLE OF THE JHR AND PRELIMINARY COMPARISON WITH ATR AND HFIR

The JHR core has 34 fuel assemblies—each containing about 0.9 kg of uranium-235—positioned in an aluminum casing and surrounded by a beryllium reflector. It can accommodate up to 10 experimental devices (Colin et al. 2014) distributed on three rings: seven in small locations of 37-mm diameter (pink locations on Figure 3a and Figure 3b) and three in larger locations of 91-mm diameter (brown locations on Figure 3a and Figure 3b). The reflector can also receive devices with 20 fixed positions, mainly of 100-mm diameter, but there will be one location of 200-mm diameter. Ten cycles of about 27 days each are expected per year corresponding to 270 full power days per year (2009).



3a



3b

Figure 3. Neutron spectra for different locations in the JHR core.

For the startup configuration (70 MW, U_3Si_2 fuel), the highest fast flux above 0.1 MeV in the core is expected to reach about $7.5 \times 10^{14} \text{ n.cm}^{-2}.\text{s}^{-1}$. In this relatively small test position ($\varnothing=3.2 \text{ cm}$) at the center of a fuel element located close to the center of the core (Position 101, red curve, Figure 3a), damage rates reach about 9 displacements per atom (dpa)/year. When JHR operates at its nominal power of 100 MW with UMo fuel, the fast flux above 0.1 MeV is expected to reach about $10^{15} \text{ n.cm}^{-2}.\text{s}^{-1}$; in this configuration, the damage accumulation on test materials is expected to reach about 16 dpa/year, assuming 270 days of full-power operation per year.

For comparison, the maximum fast flux above 0.1 MeV attainable in ATR, when operated at 110 MW, is about $5.5 \times 10^{14} \text{ n.cm}^{-2}.\text{s}^{-1}$ in the small ($\varnothing=1.7 \text{ cm}$) central A positions (Figure 4). In this position, about 6 dpa/year can be reached, assuming five 50-day cycles at full power per year. The higher fast-flux level attained in JHR compared to ATR may be explained, at least partially, by a higher power density and a lower thermal-neutron flux in the driver fuel (JHR driver fuel elements have a lower water-to-fuel volume ratio than ATR's and, furthermore, are farther away from the beryllium reflector

than ATR's). If ATR were to operate at its maximum rated power of 250 MW, and assuming eight 22.5-day cycles^c at full power per year, then it could reach about 10 dpa/year.

The fast flux in HFIR's six small ($\varnothing=1.8$ cm) peripheral target positions located in the central flux trap (ORNL 2015) (Figure 5), when operated at 85 MW, can reach about 1.3×10^{15} n.cm⁻².s⁻¹, i.e., similar to JHR (Daly and McDuffee 2012; Ilas and Primm 2012). HFIR's high flux is due in large part to its very high power density (85 MW generated from about 9 kg of uranium-235 versus 110 MW for 43 kg in ATR and 100 MW for about 30 kg in JHR). In these high fast-flux positions, and considering a typical year with seven 23-day cycles at full power, damage accumulation reaches about 14 dpa/year.

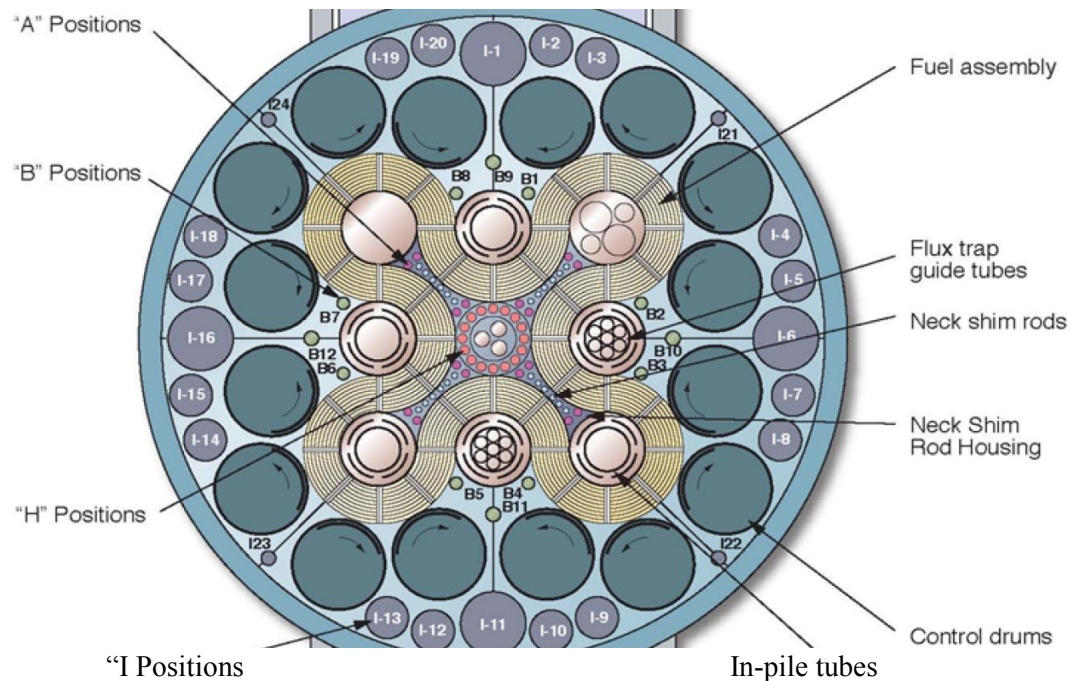


Figure 4. ATR core cross section showing fuel elements and test positions.

^c Increasing the power from 110 to 250 MW would decrease the cycle length by the same ratio, because the fuel would burn faster. Assuming that the outage time is independent of the cycle length, increasing the power leads to fewer days of operation at full power.

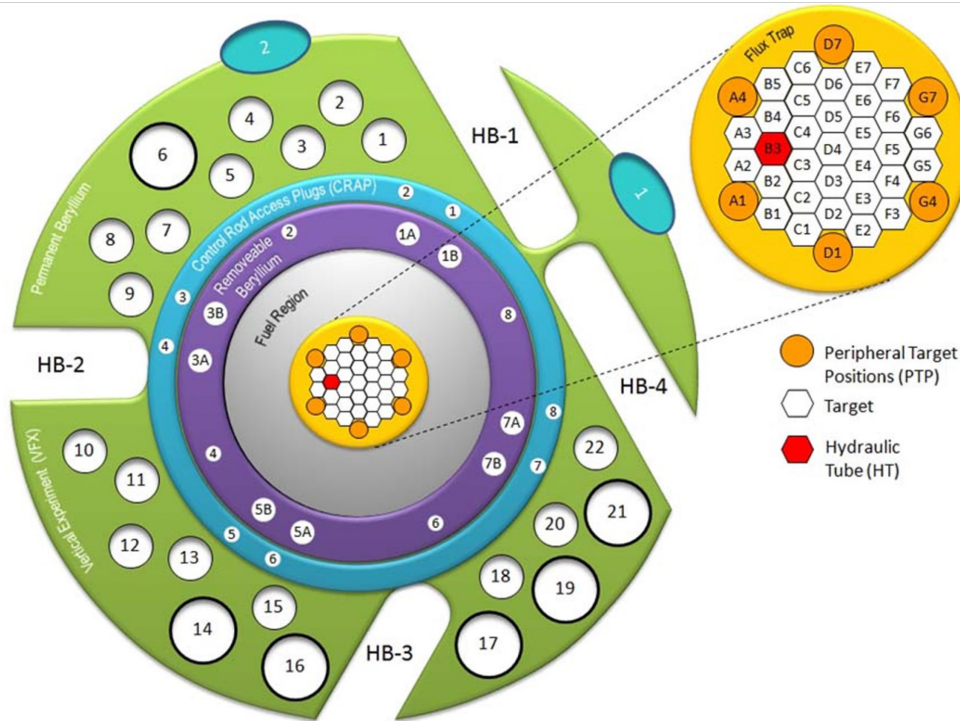


Figure 5. HFIR core cross section showing fuel region and test positions.

These fast fluxes, though very high for light-water reactor testing purposes, still pale in comparison to the $4\text{--}5 \times 10^{15} \text{ n.cm}^{-2}.\text{s}^{-1}$ (above 0.1 MeV) that can be achieved in typical fast-test reactors such as the JOYO or Russia's BOR-60, for example. With such fast-flux levels, up to 50 dpa/year should be attainable (assuming about 300 days of full-power operation per year); hence, whereas it would take only about 2 to 2.5 years to accumulate 100 dpa on a test sample in a fast-test reactor, it would take at least 6 years in a high-power water-cooled test reactor to reach the same damage. In order for JHR or HFIR to reach the same fast flux levels as these fast test reactors, their power densities would need to be increased by a factor of about 4 to 5, which is technically impossible because the fuel could not be adequately cooled.

Furthermore, the ATR, HFIR, and JHR test locations with the highest fast flux have a significant level of intermediate and thermal neutron flux (Figure 3a). One of the consequences of testing fast reactor fuel in this environment is that the dpa over burnup ratio will not be representative of what would be experienced in an actual fast reactor. Another important metric, the ratio of helium production in the clad over the dpa accumulated, will also not be prototypical of actual fast reactors.

It is, of course, possible to filter part of the epithermal and thermal neutrons using appropriate neutron filters (see Figure 6 and Figure 7), but this will not increase the fast-flux level; it will just get rid of some of the lower-energy neutrons. Another possible approach to minimize the thermal flux—and consequently to maximize the fast flux—in a high-power water-cooled test reactor may be to use heavy water (D_2O) instead of the standard light water (H_2O) as the coolant.^d This would lower the thermal-neutron flux in the driver fuel and, consequently, increase the fast-flux level (assuming the same reactor power for the D_2O -cooled and H_2O -cooled cores) to an extent that would need to be quantified.

^d Heavy water is more transparent to neutrons than light water. In any given volume of water, a neutron is about 5 times less likely to collide with the nucleus of a deuterium atom in D_2O than with that of a hydrogen atom in H_2O .

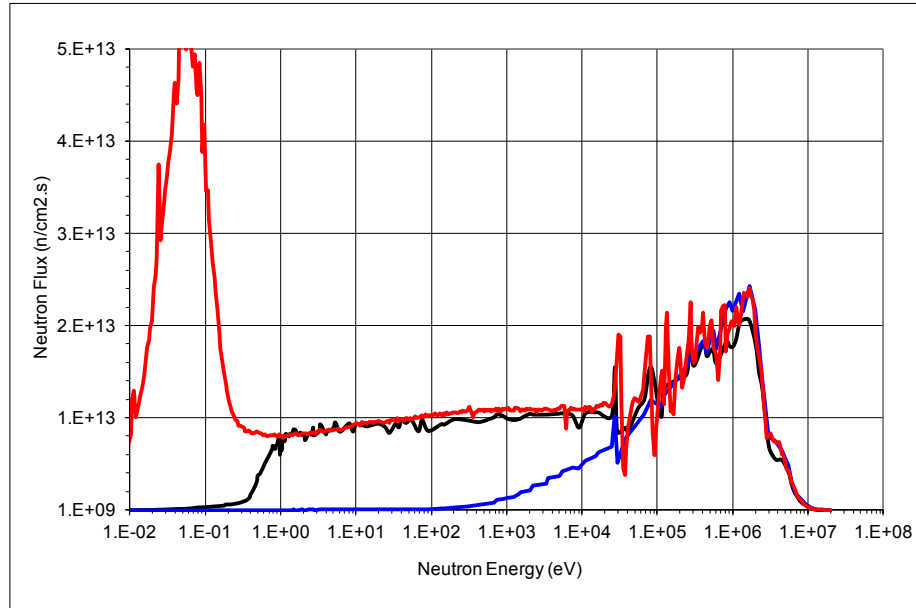


Figure 6. Effect of neutron filters on the neutron spectrum in the ATR B10 test position (see Figure 4) (INL 2015). Red curve = no filter; blue curve = 1-mm cadmium filter; and black curve = 10-mm boron filter (70% enriched).

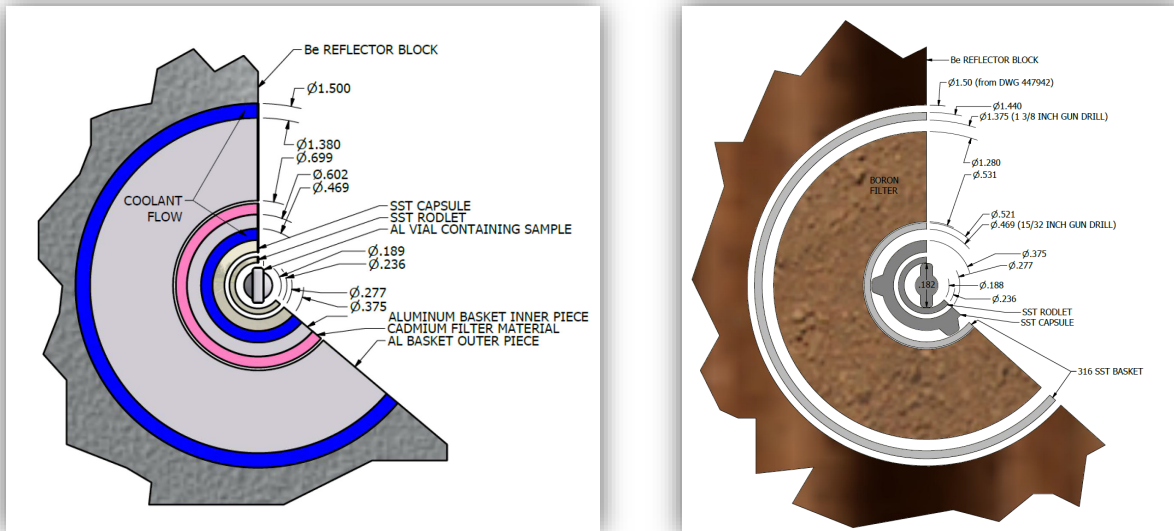


Figure 7. Schematic of a cadmium filter (left) and boron filter (right) located in ATR B10 test position.

Therefore, irradiations conducted in high-power (light)-water-cooled test reactors—such as ATR, HFIR, or JHR—with proper thermal-neutron filtering can provide some important information for understanding certain fast-reactor fuel behaviors under irradiation, but they are not able to provide information to assess many important phenomena under fast-spectrum irradiations, nor is the information adequate to assess the integral behavior of a fuel pin that is needed to demonstrate fuel reliability or burnup potential.

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