Hindawi Publishing Corporation
Science and Technology of Nuclear Installations
Volume 2014, Article ID 395795, 23 pages
http://dx.doi.org/10.1155/2014/395795

Review Article

Compilation of Existing Neutron Screen Technology

N. Chrysanthopoulou,[1,2] P. Savva,[1] M. Varvayanni,[1] and N. Catsaros[1]

[1] *National Centre for Scientific Research "Demokritos", Institute of Nuclear & Radiological Sciences & Technology, Energy & Safety, 15310 Aghia Paraskevi, Greece*
[2] *Department of Physics, School of Applied Mathematical and Physical Sciences, National Technical University of Athens, 15780 Zografou, Greece*

Correspondence should be addressed to N. Chrysanthopoulou; nefeli@ipta.demokritos.gr

Received 17 October 2013; Revised 22 April 2014; Accepted 23 April 2014; Published 11 August 2014

Academic Editor: Alejandro Clausse

The presence of fast neutron spectra in new reactors is expected to induce a strong impact on the contained materials, including structural materials, nuclear fuels, neutron reflecting materials, and tritium breeding materials. Therefore, introduction of these reactors into operation will require extensive testing of their components, which must be performed under neutronic conditions representative of those expected to prevail inside the reactor cores when in operation. Due to limited availability of fast reactors, testing of future reactor materials will mostly take place in water cooled material test reactors (MTRs) by tailoring the neutron spectrum via neutron screens. The latter rely on the utilization of materials capable of absorbing neutrons at specific energy. A large but fragmented experience is available on that topic. In this work a comprehensive compilation of the existing neutron screen technology is attempted, focusing on neutron screens developed in order to locally enhance the fast over thermal neutron flux ratio in a reactor core.

1. Introduction

A neutron screen is a device, a model, a system, or a technology that provides the capability of locally tailoring the neutron spectrum. It contributes to the creation of desirable special irradiation conditions that cannot be achieved during normal reactor operation for technological or economic reasons. Spectrum tailoring is accomplished by using absorbing materials—acting as a filter to neutrons with specific energies—affecting thus the flux to which the specimen is exposed. In the literature the term "neutron screens" refers either to the absorber or to devices containing a neutron filter. In most of the cases the filter is combined with another material serving technological, physical, or chemical reasons.

The use of neutron screens technology is imperative within the framework of the materials qualification for the development of Generation IV (GEN-IV) reactors, since their performance can easily and inexpensively simulate the characteristics of a fast neutron flux facility and accomplish power transient experiments. Neutronic conditions similar to the ones prevailing in fast reactor cores can be achieved in MTRs by tailoring the neutron spectrum in order to properly tune the reaction rates. For the spectrum tailoring two options are offered, that is, either (a) to remove the thermal component using a neutron screen technique based for example, on Cadmium, Boron, or Hafnium shields, or (b) to increase the fast/thermal neutrons ratio inside an MTR by using fissile material. It should be noted that several complexities, mainly of technological nature, are involved in the use of thermal neutron absorption shields, since all candidate shielding materials have their specific problems induced by welding behaviour, swelling (e.g., Boron compounds), or melting (e.g., Cadmium). Therefore, research and exchange of information on neutron screens technology is of increasing interest.

The purpose of this report is to present the existing neutron screens technology; it is examining the neutron screens developed to address the lack of fast research reactors in sufficient number. A neutron screen achieves the required fast neutron environment by cutting off the thermal component of the neutron spectrum, using thermal neutron absorbing material. The absorber is almost transparent to fast—and much less to epithermal—neutrons. In order to

reproduce the irradiation conditions prevailing in a sodium-cooled fast reactor (SFR) environment or in a sodium-cooled fast breeder reactor (SFBR), coolant loops and booster fuels are combined in the irradiation facility. This work further investigates the power transition facilities that are also based on neutron screens utilization, where also thermal neutron absorbers are used. By varying the absorber's concentration, power transient on the irradiation sample is achieved. The utilization of variable screens is necessary in order to test the fuel behavior when exposed to sudden power variation conditions.

The report is divided into two sections, depending on the form of the absorbing material used that is, solid and fluid neutron screens. The first section is dedicated to neutron screens which use solid absorbing material and serve for simulation of fast neutron spectrum conditions. In the second section, neutron screens utilized for power transients are presented. Solid neutron screens can provide larger power transients than fluid screens, since they have higher density, but their utilization is complex. Fluid absorbing materials are often preferred for this purpose since their screening capability can easily change by varying their pressure and/or concentration. Several successful examples of neutron screens performance are reviewed while problems appearing in specific cases are pointed out. Computational results obtained in order to examine the various neutron screens behavior, using mainly MCNP code [1], are also reviewed.

2. Solid Neutron Screens

In this chapter, neutron screens which use a solid material are presented; four solid, thermal neutrons absorbing materials are reported, that is, cadmium, hafnium, boron, and europium. All of them are materials widely used in nuclear reactors in several applications and all of them have adequate thermal neutron absorption cross-sections. In each case, the selection of the appropriate material depends on the requirements of the experiment, the available space for screen loading, the safety issues related to screen loading, and the materials compatibility. Additionally, for the material selection the accumulated experience gained using the specific material for other reactor operations is also exploited. The factors that should be taken into account in a neutron screen design are the geometrical configuration of the screen, its depletion rate, the reactivity effect caused by its insertion in the core, the screen cooling medium, the acting field, and the required conditions. The impact that a neutron screen has on reactor operation (reactivity insertion) and on neighboring experiments, if any, should be thoroughly analyzed. In this report the last issue has not been raised.

The solid screens are classified by material and are divided into subclasses depending on whether they have already been used or they are under development or study. In cases for which neutron screens are under development or study, the parameters that should be considered for the safety and effective neutron screen design are still under investigation and have not accurately been predicted.

2.1. Cadmium Neutron Screens. Cadmium is a material widely used for thermal neutron filtering, because of its excellent thermal neutron capture capability. Its utilization in neutron screen technology is very common, since the reactors community is familiar with its mechanical properties and other technological issues. However, cadmium screen construction is complex from the engineering point of view, because of its low melting point and its large thermal expansion. Because of its extremely high thermal neutron cross-section, a slightly thin neutron screen is sufficient to tailor the neutron flux distribution in most cases. The more thermal neutron-absorbing cadmium isotope is ^{113}Cd, which constitutes only 12% of the natural cadmium. In high neutron fluxes, ^{113}Cd is quickly depleted, thus demanding frequent replacement of the screen (or additional material) in order to avoid unexpected increase of power deposited on the irradiated samples.

Cadmium screens are being used for many years now. In BR2 for instance, most of the irradiations from the 1960s until late 1980s were carried out in the framework of the fast reactor development program and most of the irradiation rigs contained a cadmium thermal neutron-absorbing screen [2].

2.1.1. Cadmium Neutron Screens Already Being Used

(1) Belgian Reactor 2, Center for Nuclear Energy Research (BR2, SCK.CEN). In BR2 core (Figure 1) at SCK.CEN a Cd screen was installed around a large Na loop (hosting a single fast reactor fuel pin) which was already surrounded by a gaseous ^{3}He screen at variable pressure. The experiment named VIC (variable irradiation conditions) was installed in a standard 84.2 mm channel of BR2 (Figure 1) and its utilization aimed at liquid-metal fast breeder reactor (LMFBR) fuel pins testing under transient operating conditions. The ^{3}He gas screen serves for fuel power transient, while the Cd screen provides a fast spectrum environment by cutting off the thermal neutrons. The optimization of the neutronic design of the loop was done by calculating an optimal thickness of the water annuli between the ^{3}He and Cd screens, in order to allow for a certain rethermalization (so that the ^{3}He screen would be able to induce a transient), keeping at the same time the thermal component of the neutron flux low enough, so as to be a representative of a SFR [3]. Utilization of ^{3}He screen for power transients in VIC is also examined in Section 3.

Figure 2 shows the neutron spectra in a BR2 experiment with (red line) and without (dashed line) Cd screen located axially in a fuel element channel. The installation of the screen cuts off the thermal component of the neutron spectrum, whereas it has practically no effect on the high fast flux, thus leading the radial fission density distributions across fuel pin bundles and inside the fuel pins themselves to become much flatter and therefore better simulating the conditions of fast reactor.

An additional experiment with cadmium screen, named MOL 7D was installed in the central H1 channel of BR2 (Figure 1) in order to study the safety parameters of a SFBR. The MOL 7D loop contained 19 fuel pins in a triangular pitch arrangement of 1 (central) + 6 (inner) + 12 (outer)

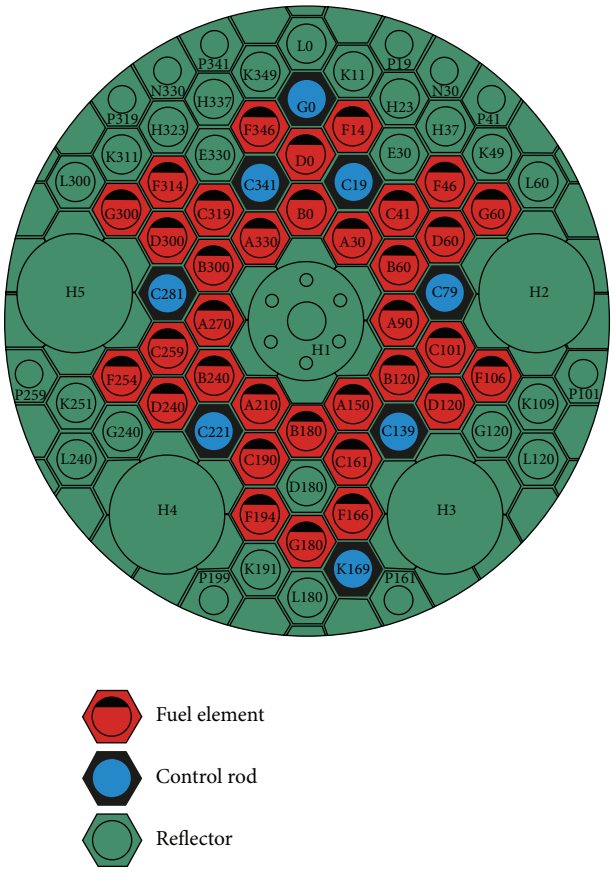


Figure 1: Horizontal cross-section of BR2 with a typical loading [2].

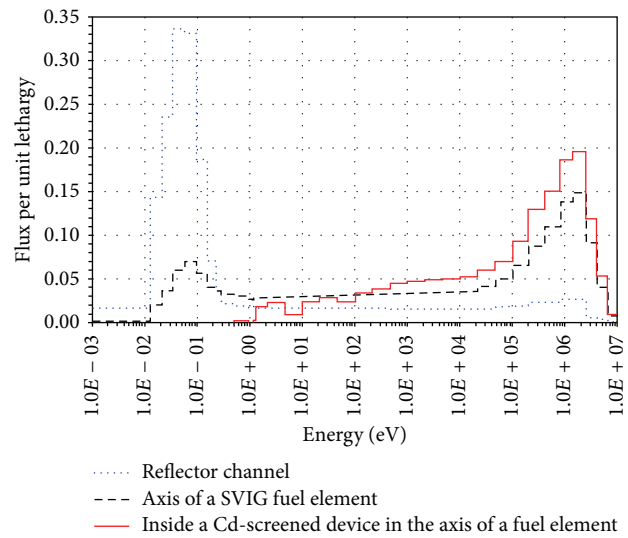


Figure 2: Neutron spectra in BR2 [2].

fuel pins and was loaded in the central hole of an Al plug in the central channel of BR2. A Cd screen surrounded the loop (Figure 3). Six fuel elements were loaded into the plug around the loop. Those elements acted as driver fuel elements, boosting the fast and epithermal neutron current entering the loop. Concurrently, they acted as neutralizers to the strong antireactivity of the neutron absorbing Cd screen [2].

(2) High Flux Reactor (HFR), Petten

(a) TRIO Modified for Irradiation of MOX Fuels (TRIOX) Capsule. TRIO is an irradiation device holding three circular cross-section thimbles, positioned radially per 120°. In HFR (Figure 4) a TRIO concept, that is, TRIOX, was utilized for MOX fuels irradiation, with a Cd screen adapted into the sample holder carrier for spectrum hardening at the location of testing fuel [3]. Starting from the center and moving towards periphery (i.e., from inside to outside), a TRIOX channel is configured in terms of its radially arranged materials as follows: sample (i.e. fuel pin), sodium (2.45 mm), molybdenum shroud (5 mm), sodium (1 mm), stainless steel (inner containment, 1 mm), gas gap (0.2 mm), stainless steel (outer containment, 1 mm), reactor coolant water (6.8 mm), aluminum, (6 mm), cadmium (3 mm, embedded into Al over part of the device height), and aluminum (the capsule material); the dimensions in parentheses correspond to the material thickness. The gas gap between primary and secondary containments, consisting of He/Ne or Ne/N, helps to control the fuel pin clad temperature. The molybdenum shroud is immersed in the helium pressurized stagnant sodium which fills the primary containment in order to prevent convection currents from appearing in Na when the gap between fuel pin and primary containment is too wide.

The cadmium incorporated into the TRIOX capsule acts as a neutron screen for the reduction of the thermal flux influence. It is a 0.5×1.9 mm^2 wire which is embedded into the Al structure in a spiral groove made in the Aluminum tube, giving a partial cadmium-cover in the TRIOX holder. Effective cadmium cover can be changed depending on the design requirements. In order to avoid design complications, cadmium screen is not directly incorporated to the sample holder. On the other hand, the Cd placement in the coolant water channel allows for the sufficient cooling required for cadmium, but has the drawback that some fast neutrons turn thermal once they have passed the neutron screen, which constitutes a compromise in the design. Moreover, all three TRIOX channels can contain cadmium but this causes a significant reactivity effect. From the HFR operation safety point of view, the TRIOX capsule should be placed in the lower flux positions, since in this way it limits the impact of cadmium wire on the neutronics of the reactor.

(b) High Neutron Fluence Irradiation of Pebble Stacks for Fusion (HICU) Project. The HICU irradiation project has been implemented in HFR (Figure 4) for Li-ceramics irradiation under conditions similar to those of a pebble bed reactor (PBR). Li-ceramics are candidate materials for the breeder blanket of fusion reactor. HICU combines two neutron screens: hafnium rings have been placed along with cadmium rings creating a neutron screen layer within the sample holder tube. The combination of the two absorbers allows for a longer irradiation under fast spectrum conditions.

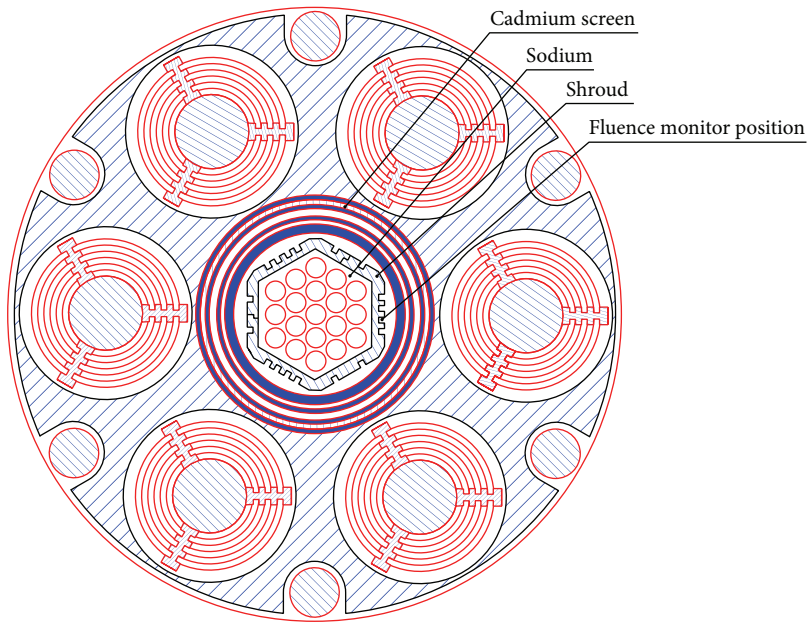


Figure 3: Horizontal cross-section of the MOL 7D loop with 19 fuel pins in the central 200 mm diameter BR2 channel [2].

Figure 5 shows the power density versus time generated by (n, α) reactions on Li_4SiO_4 (Li ceramics material) in the dominant spectrum inside the HICU at core position C7 for five cases. The power generated by both thermal and fast neutrons was calculated using the MCNP code [3]. The abrupt upturn in power generation (curves B, C, D, and E) is due to the fact that at that moment Cd is completely burnt. Having a lower cross-section, Hf is expected to cause a more gradual power upturn. In fact, it can be seen that curve E (a mixture of Cd with some Hf) significantly delays the upturn. Therefore shield in curve E allows the longest irradiation (over 500 days) in the fast spectrum. The behavior of the unshielded (not accounting for burnup) facility (curve A) is linear [3].

In order to study the screen's behavior, the neutron flux inside the HICU was computed with the MCNP code [5] (Figure 6). The computations were performed considering the HICU loaded at the third highest flux position of HFR core (C3 or C7, Figure 4) while two types of fusion breeder materials, containing lithium metatitanate, that is, Li_2TiO_3 with natural Lithium (7.5% 6Li) and Lithium-enriched material (30% 6Li), were assumed. The screen is based on a 2 mm Cd shield together with an Hf wire ($0.6 \times 0.6\,mm^2$) [5]. The installation of HICU in C7 irradiation position (Figure 4) causes a negative reactivity effect of about 1200 pcm (per cent mille) which is considered acceptable by the HFR limiting conditions for operation [3]. Concerning the utilization of breeding materials, it is noted that fusion reactions in ITER are fueled by deuterium and tritium, the resources of the latter being extremely limited (currently estimated at 20 kg). Lithium can be used as a solid breeder material of tritium in the blanket of fusion reactions in ITER while tritium is produced by the neutrons leaving the plasma and interacting with lithium in the blanket. Li_2TiO_3 is used as a tritium breeding material because it allows high tritium release and presents low activation whilst being chemically stable.

2.1.2. Cadmium Neutron Screens under Investigation or Development

(1) Massachusetts Institute of Technology (MIT). In the MIT reactor, a design for achieving significantly higher neutron flux has been proposed. The proposed facility would be hosted in the central fast flux trap of the MIT reactor core. The design includes a fast flux trap loop surrounded by fissionable material (^{233}U, ^{235}U, ^{239}Pu, and ^{242m}Am), reflected and cooled by liquid eutectic Pb-Bi coolant. The fissile material could be enriched in either ^{235}U or ^{233}U. The area containing the fissile pins, called amplifier ring, consists of 164 oxide fuel pins arranged in four rings of 32, 38, 44, and 50 pins, respectively, in a hexagonal arrangement. The Pb coolant acts as a "driver" by reflecting fast neutrons and sending them back to the central irradiation facility. Between the amplifier ring and the experimental irradiation facility area, a Cd filter was loaded [6]. The optimum thickness of Cd was investigated and the impact of various Cd thicknesses on the neutron flux is summarized in Table 1. An extremely thin layer of Cd (0.1 mm) was found able to reduce the thermal flux well over 50% while the fast flux decrease was limited to 2% [6]. Figure 7 illustrates the impact of the 0.1 mm Cd thickness on the neutron flux spectrum.

(2) Advanced Test Reactor (ATR), Idaho National Laboratory (INL). In the east flux trap (EFT) of the ATR (Figure 8), at INL, irradiation tests of high-actinides-content fuels, AFC, have been performed with the aim to examine the transmutation of long-lived isotopes in spent nuclear fuel into

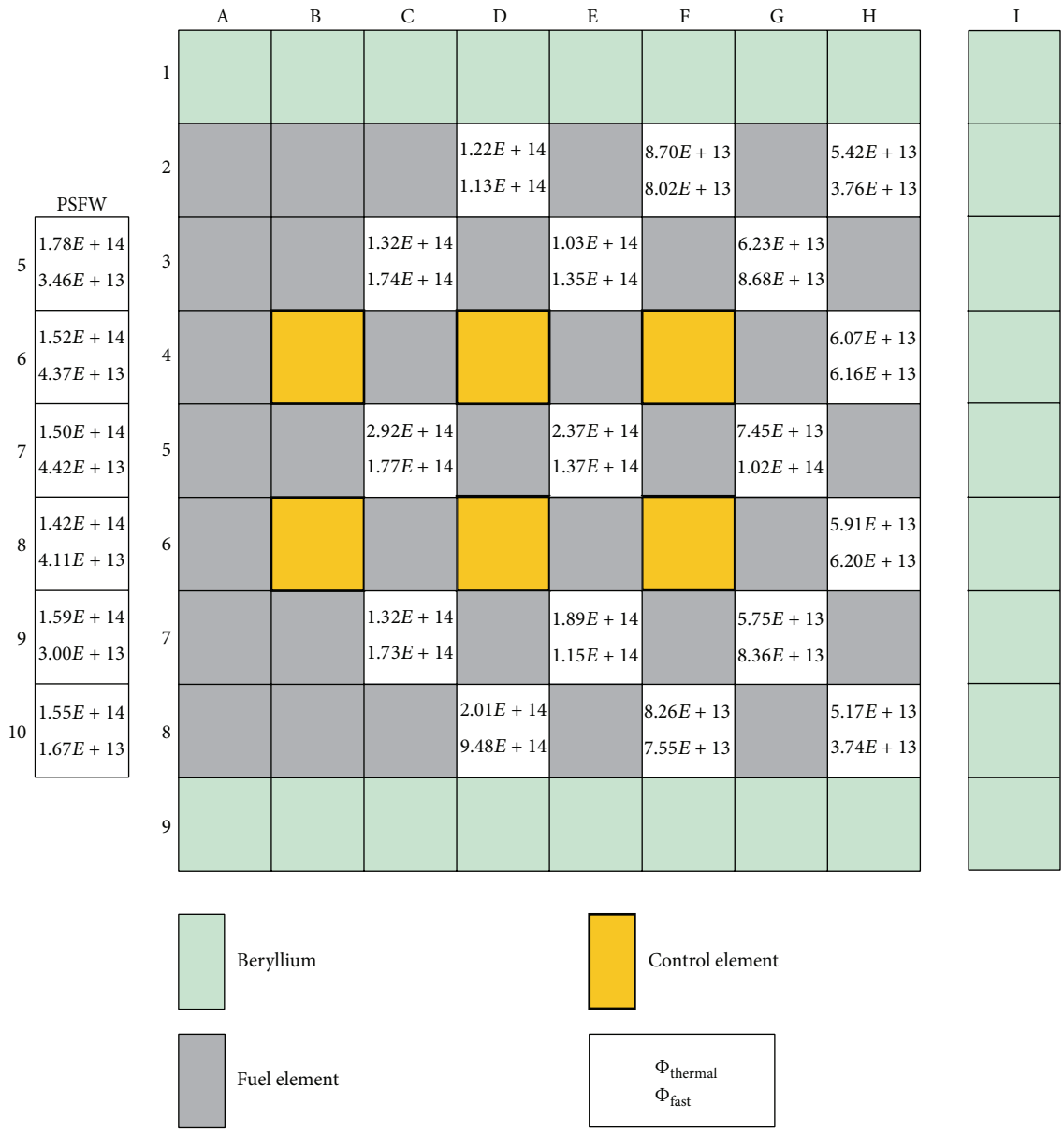


Figure 4: HFR cross-section [4].

shorter lived fission products in an irradiation environment similar to that of a fast reactor. The fuels were loaded in an experiment Al-sheathed basket with a 0.114 cm thick Cd absorber filter. The device was cooled by light water, which is the primary coolant of the reactor. From a study performed with the combination of MCNP and ORIGEN-2 codes [7] it was found that at the beginning of irradiation the peak of the linear heat generation rate by the metal fuel, with and without absorber filter, was 237 and 2174 W cm^{-1}, respectively [8]. The maximum basket lifetime was estimated to be about 48 effective full power days (EFPDs).

Figure 9 compares the neutron flux spectrum in two cases, that is, the neutron spectrum produced with and without Cd filter for the neutron flux spectrum type of a liquid metal fast breeder reactor (LMFBR) [9]. The results show that omission of a Cd filter (Al-basket) results in a softer neutron spectrum [8] and that Cd-filter can adequately harden the neutron spectrum in EFT position.

2.2. Hafnium Neutron Screens. Hafnium is also a widely used material in nuclear reactors. It has excellent mechanical properties; it is extremely corrosion-resistant and therefore can be used without doping. Another advantage is its high melting point. Hafnium thermal neutron cross-section is not as high as that of the other absorbers presented in this chapter. However, the formation of Hf isotopes under irradiation, which are good thermal neutron absorbers too, makes hafnium a good candidate material for neutron screen technology. Since it decays to good thermal absorbers, its depletion occurs slowly, delaying thus its replacement requirement

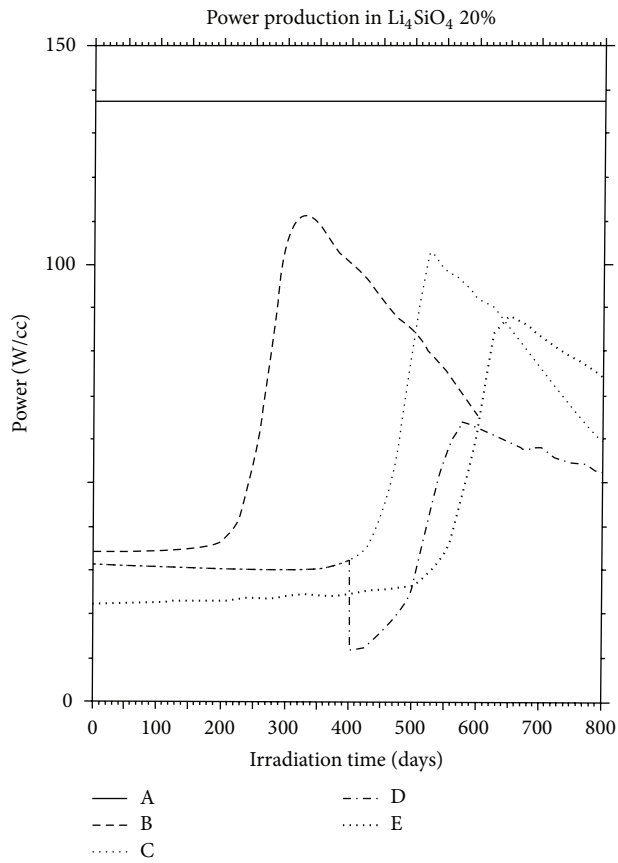


Figure 5: n-α power production versus irradiation time calculated for Li_4SiO_4 20% enriched in ^{6}Li. Notation for curves is as following: unshielded (A); 1 mm Cd (B) and 2 mm Cd (C); 2 mm Cd, 400 days at C7 and then at H8 (D); 2 mm Cd and 0.5 mm Hf (E) [3].

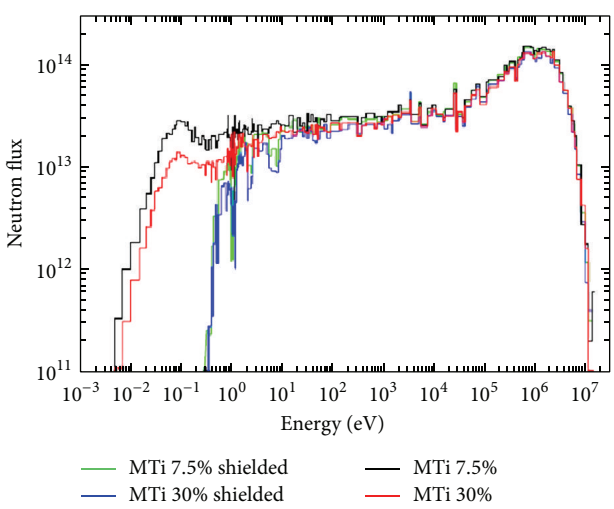


Figure 6: Neutron flux inside the HICU experiment computed for four conditions: with and without neutron screen, with natural Li (7.5% 6Li), and with Lithium-enriched material (30% 6Li) [5].

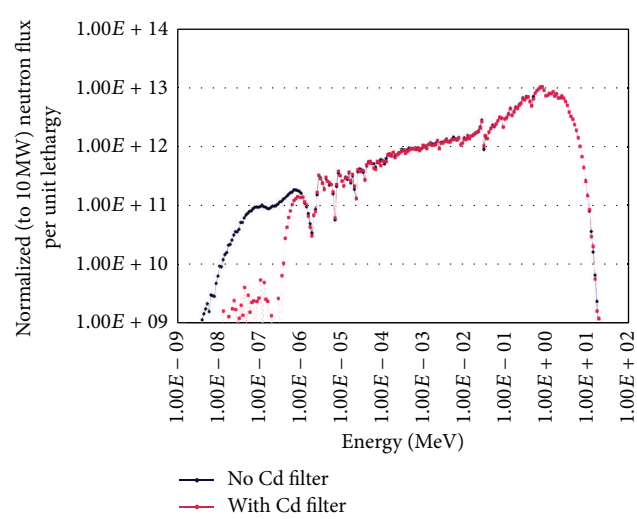


Figure 7: Effect of a 0.1 mm-thick Cd filter on neutron spectrum [6].

in a neutron screen. Hafnium is usually combined with aluminum, which is a good heat conductor and is transparent to fast neutrons.

2.2.1. Hafnium Neutron Screens Already in Operation

(1) HFR, Petten. A TRIO-facility sample-holder called CONFIRM has been used for fast reactor fuel irradiation in HFR (Figure 4). In CONFIRM the fuel pin (typical diameter

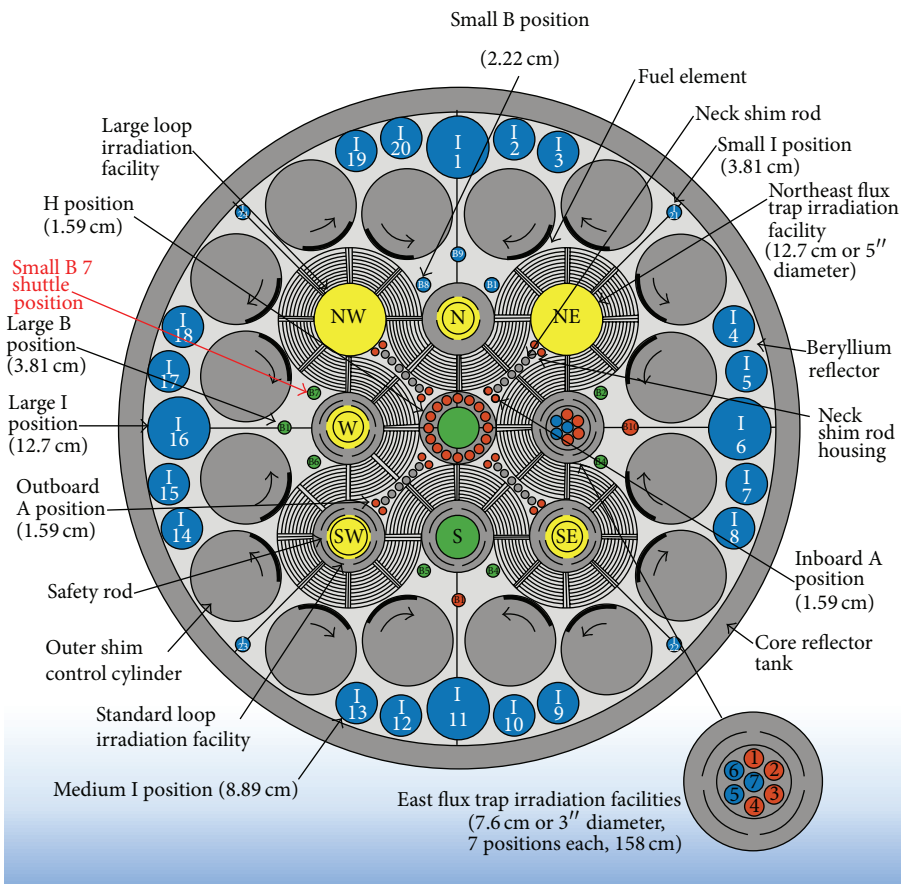


Figure 8: ATR core cross-section—4 flux traps, 5 in-pile tubes, and 68 in reflector—[10].

Table 1: Effect of Cd filter thickness on the neutron flux [6].

Cd filter thickness (cm)	Neutron flux modification (%) for four neutron energy groups			
	0–0.4 eV	0.4 eV–2 keV	3 keV–1 Mev	1–10 MeV
0.00	0	0	0	0
0.01	−55	0	−2	−2
0.02	−65	−1	−2	−3
0.05	−69	−2	−2	−4
0.07	−71	−2	−1	−4
0.08	−72	−5	−1	−4
0.13	−73	−5	−2	−5
0.21	−73	−6	−2	−8

6.55 mm) is surrounded by a sodium layer (1.325 mm thick) enclosed in a Mo-shroud (2.4 mm thick) for heat conductivity purposes. The material is placed in a stainless steel containment (1.4 mm thick) while a sodium zone (2.5 mm thick) is interposed between the containment and the Mo-shroud. A second stainless steel containment (1 mm thick) with a 0.5 mm Hf shield endues the first containment; a 0.1 mm gap exists between the two containments. The whole structure is placed axially in the TRIO wet channel of 31.5 mm diameter confined by a 1 mm thick stainless steel tube [3].

The irradiation holder was fabricated with Hf and was loaded at the lower flux positions of the core. The CONFIRM irradiation was carried out in a wet channel of a dry-wet-dry (DWD) TRIO with a standard rig head. The outer surface of Hf was cooled by the water flowing of the wet channel. The height of the shield was larger than that of the fuel, providing an effective shielding. In order to study the impact of Hf on the power density, computations using MCNP [3] were performed for various Hf thicknesses (Figure 10) and two fuel pins of plutonium with 87% Pu-239. As can be seen, changing of the shield thickness has a large impact on the power density. The spectrum in the molybdenum shroud which surrounded the fuel sample was also computed. Figure 11 shows the impact of a 4 mm Hf shield on the spectrum, while Figure 12 shows the thermal energy range in detail for five Hf thicknesses. Figure 13 shows the variation of the normalized power as well as of the normalized flux per unit lethargy as a function of the Hf thickness; the variations are very similar.

(2) High Flux Isotope Reactor (HFIR), Oak Ridge National Laboratory (ORNL). In HFIR, ORNL (Figure 14), an irradiation facility that allows testing of advanced nuclear fuels under prototype LWR (Light Water Reactors) operating conditions in approximately half the time it takes in other research

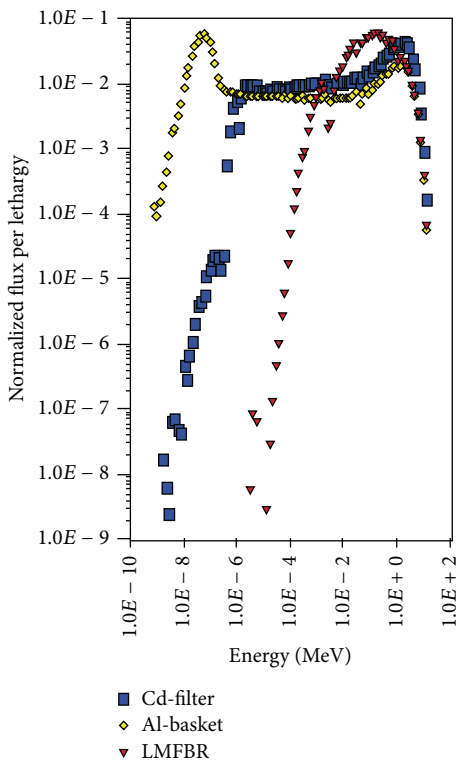


Figure 9: Comparison of fast-fission spectrum (LMFBR) with the neutron spectrum produced with and without Cd [9].

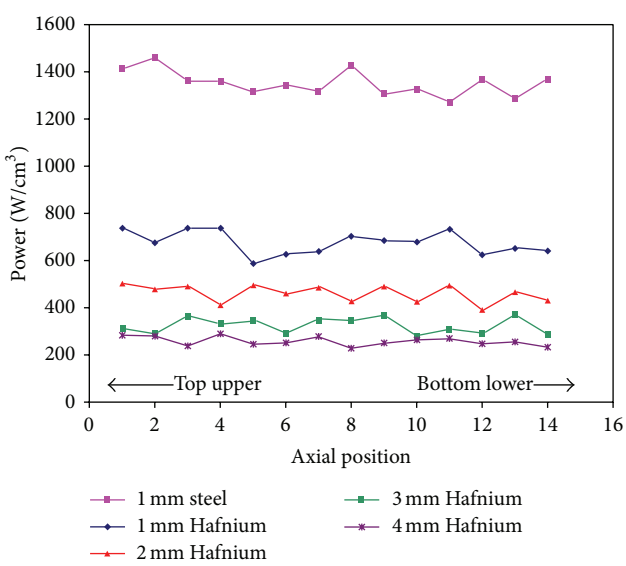


Figure 10: The influence of the thickness of the Hf shield on the power density in the two fuel pins [3].

reactors, has been developed. The cylindrical irradiation device holds three axial thimbles of circular cross-section, with their centers located radially per 120°, forming an inscribed concentric circle. The intermediate space around irradiation holes is filled with aluminum and each capsule is surrounded by a coolant channel. Three flux monitor tubes,

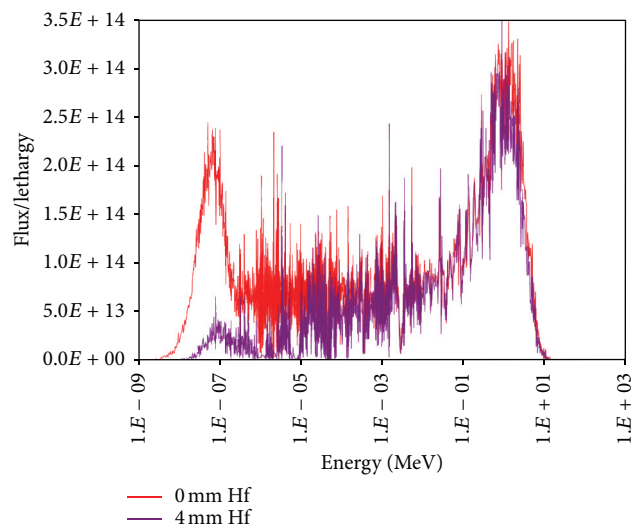


Figure 11: The spectrum in the molybdenum shroud surrounding the fuel [3].

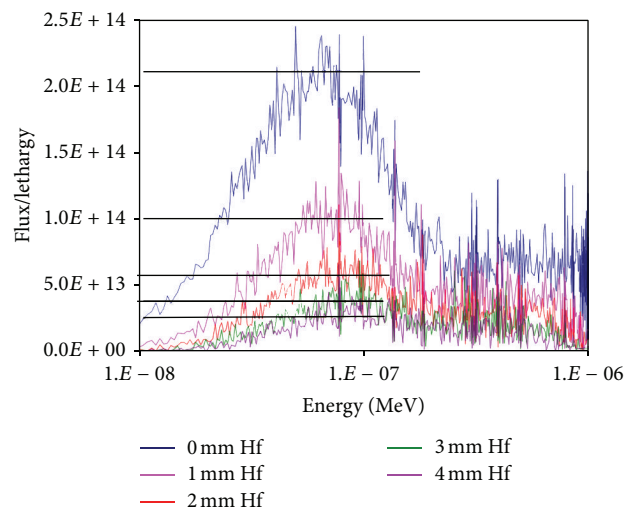


Figure 12: The spectrum in the molybdenum shroud surrounding the fuel. Detail of the thermal energy range for five Hf thicknesses. The horizontal lines mark the height of the thermal peak [3].

also surrounded by coolant, are placed radially, each one between two successive irradiation holes. The goal of this design is to maintain a relatively constant linear heat rate. An Hf screen surrounds the facility basket which is located in the reflector region of HFIR.

Two LWR experiments have taken place using the above irradiation facility; the first contained UN (uranium nitride) and the second contained UO_2 fuel pellets inside SiC cladding. The facilities contain nine fuel pins—each comprising 10 fuel pellets—arranged as three fuel rods (Figure 15) [11]. Design calculations indicate that a 1.61 mm thick Hf shield degrades at a rate similar to the burnup of 3.8%-enriched UO_2 fuel, so that the linear heat generation rate of the fuel remains relatively constant, at least over the first few cycles. With this configuration and shield thickness, linear power

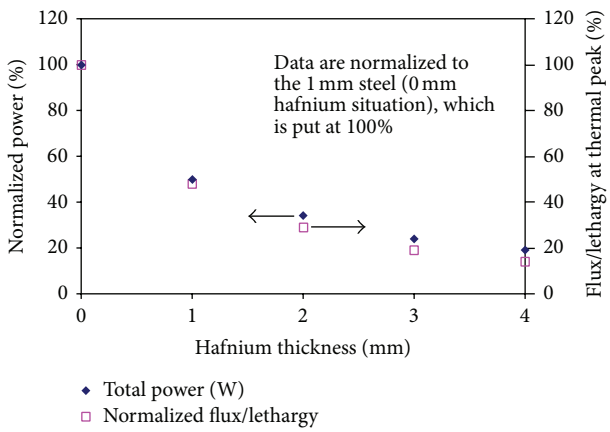


Figure 13: The normalized power density and the normalized flux per unit lethargy at the thermal peak plotted as a function of the Hf thickness [3].

ratings average 22 kW/m for the upper and lower capsules and 32 kW/m in the medium capsule. Eventually, the screen was fabricated with a thickness of 0.272 cm [18].

2.2.2. Hafnium Neutron Screens under Development or Investigation

(1) ATR, INL. Regarding the development of irradiation capabilities under fast neutron spectrum conditions at ATR (Figure 9), a boosted fast flux loop (BFFL) concept (Figure 16) has been proposed. The desired fast to thermal neutron flux ratio is over 15. The BFFL was designed to be hosted in a gas test loop (GTL) in one of ATR corner lobes, that is, NW or NE (Figure 9), where large space is available. In the framework of the GTL Project Conceptual Design, several configurations of an experiment facility that could replicate a fast flux test environment have been studied. The BFFL combines boosters for neutron flux enhancing and thermal neutron absorber for neutron filtering and for reinforcing the increase of fast to thermal flux ratio. The absorbing material is a Hf-Al composite. This material retains the high thermal conductivity of Al and has the thermal neutron absorption properties of Hf. With this approach, the produced heat can be removed by conduction and can be transferred from the experiment to pressurized water cooling channels [13, 19]. The fuel meat in the booster fuel is Si meat enriched to 93% (U_3Si_2) in ^{235}U. In U-Si plates of the required fuel loading, meat thickness and curvature are prototypes [13]. The design shown in Figure 16 provided a fast flux of approximately 1.05×10^{15} n/cm^{2}s and a fast (>0.1 MeV) to thermal (here <0.625 eV) neutron flux ratio of about 23, averaged over 16 cm in the three tubes (test spaces) [12]. The coolants of the configuration include ATR primary light water coolant, He, and Na. Helium being inert, single phase, and without reactivity effects, it was chosen as the gas coolant medium. In order to study the impact of Hf concentration on the fast/thermal neutron flux ratio, MCNP calculations were performed (Figures 17 and 18) [14]. The greater the presence of Hf in Al, the greater the removed fraction of thermal neutrons (Figure 17) would be. The heating rates in the Hf-Al as a function of the Hf concentration in the absorber are shown in Figure 18. The heating rate appears to saturate at about 6% to 7% Hf concentration, which suggests that this percentage may correspond to an optimum Hf loading [14]. It was indicated that with a 6.5% or greater concentration in Hf, a fast to thermal neutron flux ratio greater than 40 can be produced [18].

(2) OSIRIS, Saclay. An Hf neutron screen has been installed between OSIRIS reactor core (Figure 19) and ISABELLE test rig. ISABELLE 4 loop has been designed for the irradiation of fuel elements under conditions representative of those of a pressurized water reactor (PWR) and power transient irradiation. The loop can house 1–4 fuel elements or absorbent elements and can be inserted and removed while the reactor is operating. The power that can be evacuated by the loop is 90 kW and the maximum linear power on the fuel elements is 500 W·cm^{-1}. The aim of the screen was a higher than 0.1 MW fast neutron flux (2.10^{13} n/cm^2·s) to be reached in the test rig through the thermal component cutoff by Hf. The 2 mm thick screen is surrounded by a block of 10 mm thick Al and is cooled by forced convection from two water gaps. The Al block consists of 3 removable plates with 3 mm water gaps in-between that provide cooling by natural convection. Hf depletion is low, making its replacement during the irradiation unnecessary.

2.3. Boron Neutron Screens. Boron and its compounds find extensive application in nuclear reactors. Natural boron consists of 20% ^{10}B and 80% ^{11}B. The extremely high thermal cross-section of ^{10}B together with its low abundance cause quick depletion of the boron screen. The depletion can be delayed by ^{10}B enrichment. In contrast with other absorbers, that is, cadmium, Boron has a significant neutron absorption in the epithermal energy range. Boron is usually combined as a carbide or oxide. However, an Al-B alloy has been successfully used as neutron screen and another is under development. A factor that should be taken into account in a boron neutron screen design is the swelling that can be caused due to helium generation after boron irradiation with fast neutrons.

2.3.1. Boron Neutron Screens Already in Operation

(1) ATR, INL. In order to achieve a high fast (>0.1 MeV) to thermal (here <0.1 eV) neutron flux ratio [16], a 0.25 cm thick neutron screen of Al-B alloy filter has been inserted in the irradiation test vehicle (ITV, Figure 20) in ATR (Figure 8). The ITV is loaded in the central flux trap where the highest neutron flux occurs. The inert inner region of the ITV is filled with Al (Figure 20), in order to avoid water that would increase neutrons thermalization. He or N_2 gases have been selected for the specimens cooling. The screen should be replaced after a certain irradiation time. In Table 2 the neutron flux values for a case with the Al-B alloy filter (4.3% wt of ^{10}B in Al) and a case without the Al-B alloy filter are presented. As can be seen, the presence of the Al-B screen increases the fast over thermal flux ratio about twice.

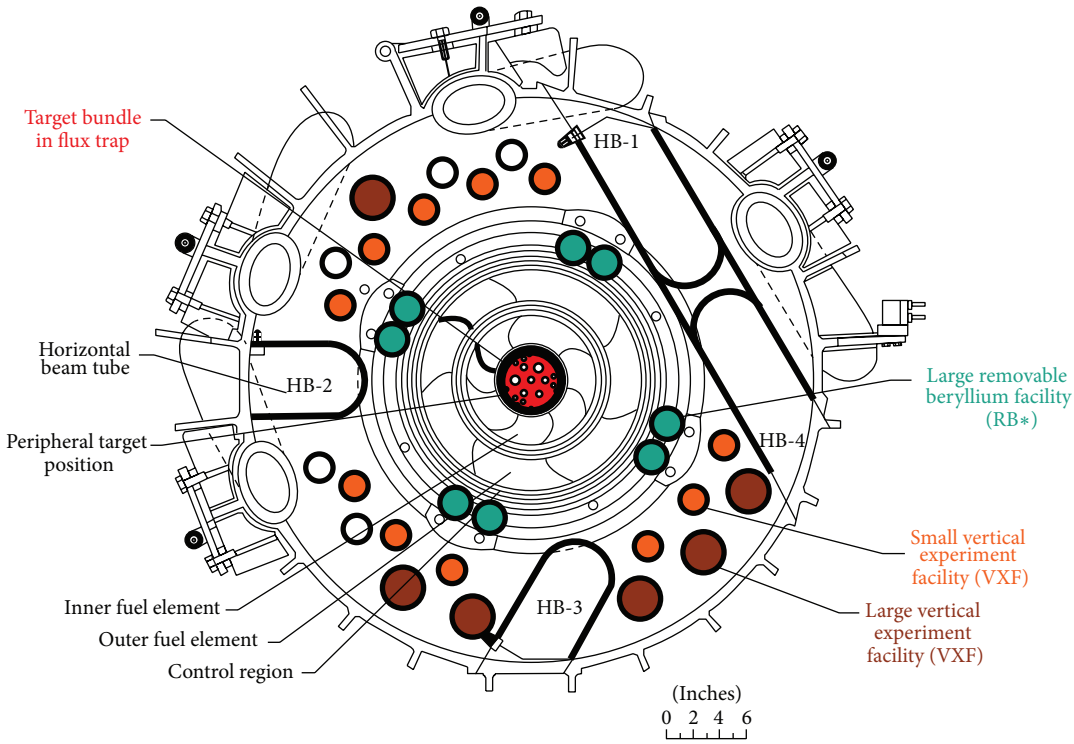


Figure 14: HFIR horizontal cross-section [11].

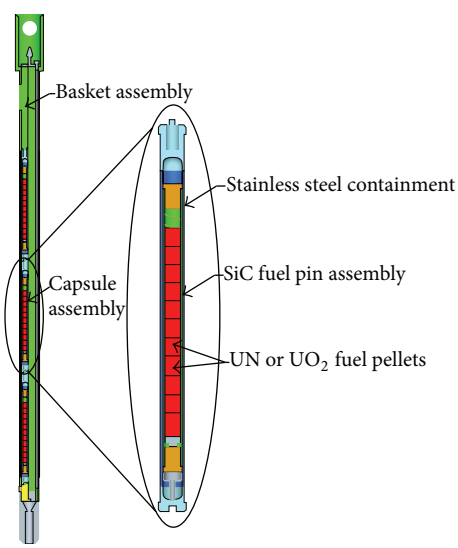


Figure 15: Vertical view of the thermal neutron irradiation facility with a close-up of one of the nine capsule assemblies [11].

Table 2: Neutron fluxes for ATR ITV midplane specimens [16].

	Neutron flux at 26 MW Center lobe power ($n/cm^2 \cdot s$)	
	Filtered	Unfiltered
Thermal (here <0.1 eV)	$1.13 \cdot 10^{14}$	$1.76 \cdot 10^{14}$
Fast (>0.1 MeV)	$4.55 \cdot 10^{14}$	$4.54 \cdot 10^{14}$
F/T	4.03	2.58

2.3.2. Boron Neutron Screens under Development or Study

(1) ATR, INL. An Al-B screen has been proposed to be loaded in a test facility in the ATR (Figure 8), in order to harden the spectrum, providing an acceptable environment for V alloy testing (The two basic design goals of the test design assembly were the achievement of 10 displacements per atom (dpa) per year in vanadium while limiting the ^{51}V transmutation to less than 0.5% for a 30 dpa experiment.). The ITV is installed in the central flux trap, where the highest value of the neutron flux occurs. The fast (>0.1 MeV) to thermal (here <0.415 eV) [20] neutron flux ratios for a 2.54 cm Al-B screen (with 95% ^{10}B enrichment) have been calculated with MCNP and ORIGEN-2 codes [21]. The results are presented in Table 3. As can be seen, the fast over thermal flux ratio (F/T) decreases linearly over operation time, because of the ^{10}B depletion in the Al-B alloy.

Figure 21 shows the impact of an Al-B filter installation on the neutron. As the enrichment increases, the hardening of the neutron spectrum is more intense. Three different cases are depicted; (a) without filter, (b) with Al-B (^{10}B 60% enrichment) filter, and (c) with Al-B (^{10}B 95% enrichment) filter. The F/T ratios for the unfiltered case have been estimated to reach14.6, whereas with the highly enriched Al-B filter (^{10}B 95%), the F/T ratio in BOL (beginning of life) reaches 142.5 (Table 3). In order to hold the averaged ratio over 80, the filter needs to be replaced after 160 EFPDs (4

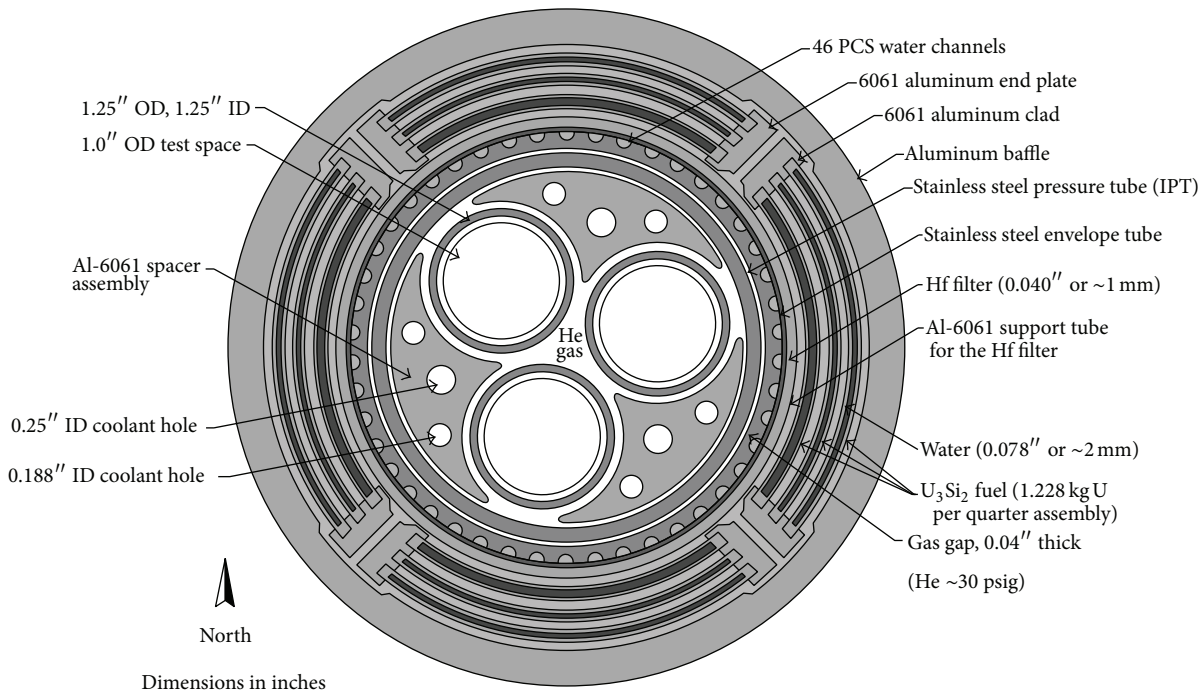


Figure 16: A graphical representation of the Gas Test Loop conceptual design [12].

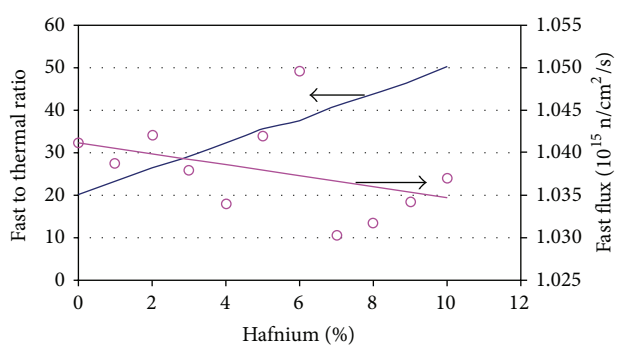


Figure 17: Sensitivity of fast-to-thermal ratio (blue line) and fast flux intensity (magenta line) to the Hf content in the Al central filler piece (absorber). The cycles on the graph correspond to the fast flux measurements [13].

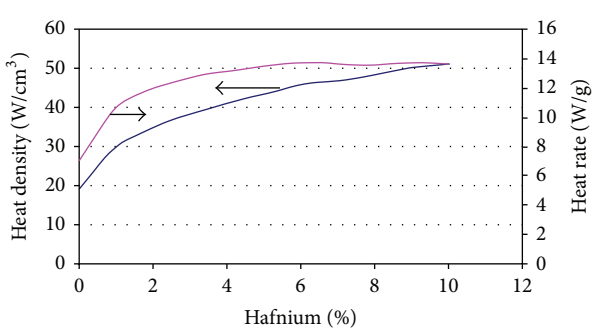


Figure 18: The heating rate (magenta line) and heat density (blue line) of the Hf-Al averaged over the central 40 cm (16 inches) of the core height [14].

Table 3: Fast (>0.1 MeV) to thermal (here <0.415 eV) neutron flux ratios versus irradiation EFPDs at ITV [21].

Irradiation days	Neutron flux F/T ratio
BoL	142.51
EFDPs 20	130.48
EFDPs 40	127.21
EFDPs 60	120.25
EFDPs 80	113.84
EFDPs 100	108.54
EFDPs 120	99.24
EFDPs 140	91.78
EFDPs 160	87.99

typical operation cycles). The Al-B (^{10}B 95%) screen can meet the desirable requirements [21].

(2) Budapest Research Reactor (BRR). In the frame of the MTR+I3 project a parametric neutron screen study was made for the largest irradiation channel of the BRR (Figure 22). The B_4C filter was inserted between the walls of an Al irradiation tube. The area outside the irradiation tube was surrounded by Al displacers. The dimensions of the irradiation tube are shown in Table 4. Two changing parameters were investigated, that is, the thickness of the filter and the enrichment of ^{10}B. In total, eight cases were studied and compared with the reference case, in which the B_4C filter was replaced by Al. The energy spectrum was divided in five groups with their upper boundaries given in Table 5. The influence of the thickness and enrichment variations on the energy spectrum and on

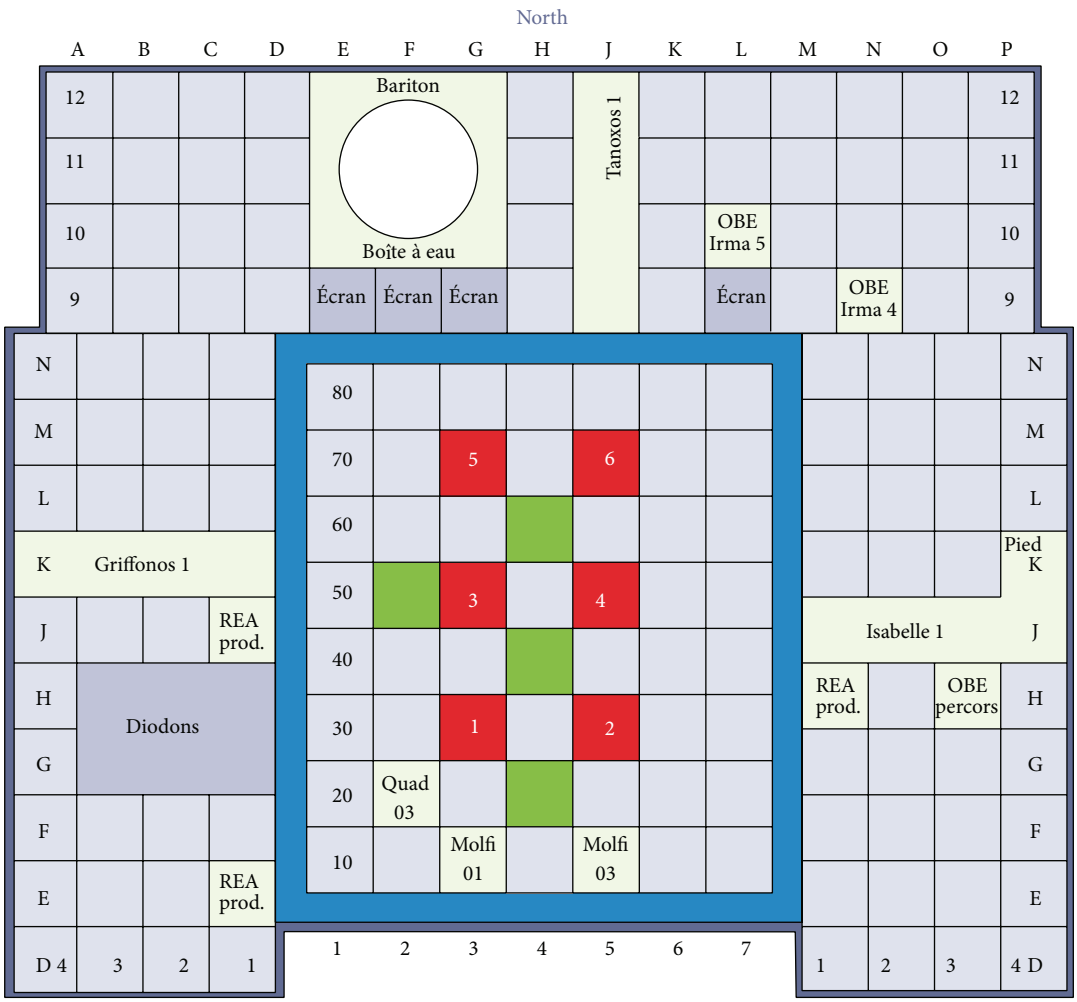


Figure 19: Horizontal cross-section of OSIRIS core [15].

Table 4: Dimensions of the irradiation tube [17].

Component	(mm)
Irradiation tube inner radius	28
Irradiation tube inner wall thickness	2
Neutron filter thickness, maximum	7
Irradiation tube outer wall thickness 3 mm	3
Irradiation tube outer radius 40 mm	40

Table 5: BRR energy spectrum boundaries [17].

Group number	Upper boundary (MeV)
1	0.0001
2	0.1
3	0.5
4	1.0
5	20.0

neutron flux is shown in Figures 23 and 24, respectively. The following results were derived by this study.

(i) The maximum ^{10}B density causes the maximum tailoring in the spectrum.

(ii) All filtering configurations in the thermal energy group have as a result an effective reduction of the thermal part of the spectrum.

(iii) For groups 3 and 4 the filtering effect is very similar.

(iv) The same filtering of the spectrum can be obtained with different filtering configurations (e.g., different enrichment in ^{10}B or different filter thickness).

(v) In filtering cases, the thermal neutron fluxes have the smallest value in contrast with the reference case where the opposite occurs.

(vi) The presence of B_4C causes an increase of 1100–1400 pcm on the reactivity which is not acceptable in reference to the limiting conditions for operation of the reactor.

Although the results of the study have shown the great thermal neutron absorption capability of B_4C, its utilization was not considered in practice because of its reactivity effect and

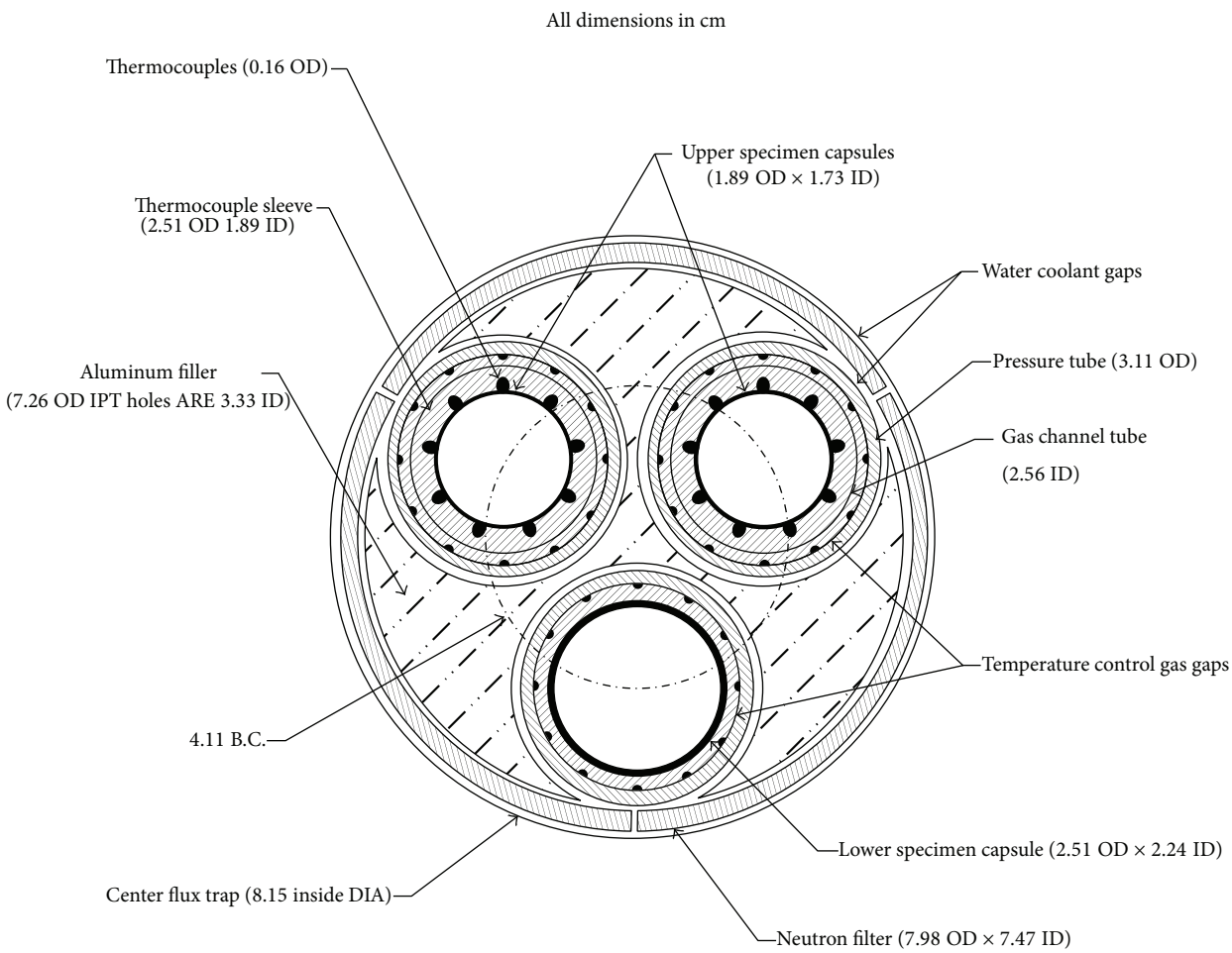


Figure 20: Core region ITV cross-section without specimens [16].

its low heat conductivity [17]. The Budapest Advanced Gas-cooled Irradiation Rig Assembly (BAGIRA) rig is recently redesigned with B_4C shield [22].

2.4. Europium Neutron Screens. Europium is a material with extremely high thermal neutron cross-section. The use of europium is limited since it is rare and hence expensive. In nuclear reactors, europium is utilized in control rods. In this section, only one neutron screen with europium is presented, that is currently under development in HFIR. The F/T flux ratio that is predicted to be achieved with europium screen is remarkable (>400).

2.4.1. Europium Neutron Screens Already in Operation. Cases of europium neutron screens already used in reactor facilities have not been found in the open literature, possibly due to the reasons described above.

2.4.2. Europium Neutron Screens under Development or Investigation

(1) HFIR, ORNL. An analysis of a fast spectrum irradiation facility design has been performed in HFIR (Figure 14) at ORNL. The screen has been planned to be installed at the reactor flux trap (FT) (southeastern core part) where the fast neutron flux exceeds 1.10^{15} n/cm^2·s and the thermal neutron flux may exceed $2.5.10^{15}$ n/cm^2·s [23]. A tri-pin assembly design (Figure 25), occupying seven target locations, was selected for the application.

Calculations of performance characteristics such as linear heat generation rate, neutron flux magnitude, fast-to-thermal flux ratio, and displacements per atom (dpa) were performed in HFIR using the MCNP code [24]. The results are provided in Table 6. From the obtained results, it appears that the proposed requirements of fast neutron flux greater than 1×10^{15} n/cm^2·s and fast-to-thermal flux ratio greater than 300 are achieved. It was concluded that this design could provide a fast (>0.1 MeV) to thermal (here <0.625 eV) neutron flux ratio over 400. Figure 26 presents a comparison between the existing neutron spectrum in the FT and the spectrum inside a 3 mm thick europium shield. The insertion of the screen has an acceptable impact on the power distribution (less than 9% as required by the HFIR safety analysis [25]).

2.5. Summarized Results on Solid Neutron Screens. Neutron screen performances can easily and inexpensively meet

Table 6: HFIR irradiation parameters for proposed tripin concept [23].

Parameter	Value
Fast (>0.1 MeV) neutron flux in irradiation volume	$1.2 \cdot 10^{15}$ n/cm^2·sec
(a) Annual[3] fast (>0.1 MeV) neutron fluence at peak irradiation position	$1.6 \cdot 10^{22}$ n/cm^2
(b) Burnup/year	5.9%
(c) dpa	19.5
Fuel burnup-to-clad dpa ratio at peak irradiation position	0.3 atoms % per dpa
Fast (>0.1 MeV) to thermal (here <0.625 eV) flux ratio	400
He-to-dpa ratio in iron at peak irradiation position	0.2 appm He per dpa
Linear heat rate (variable, depending upon design of Eu_2O_3 shield)	300 W/cm

[3] Annual estimates are conservatively based on seven cycles per year.

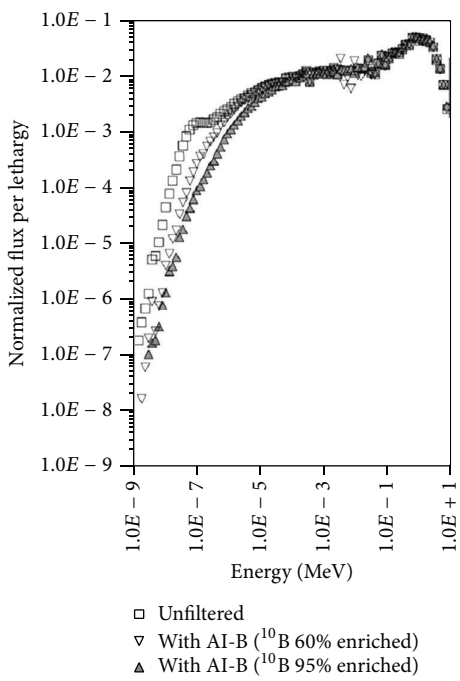


Figure 21: Neutron spectrum comparison for three cases: unfiltered, with Al-B (^{10}B 60% enrichment) filter, and with Al-B (^{10}B 95% enrichment) filter [8].

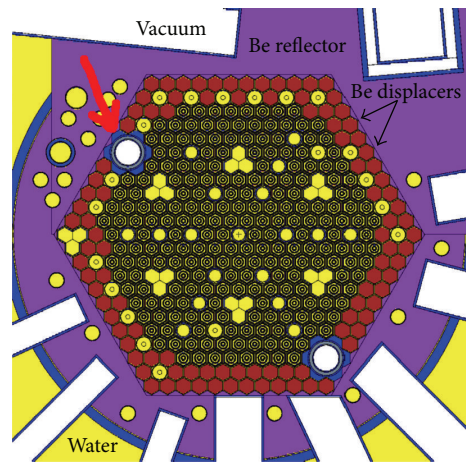


Figure 22: The geometry of the B shield (marked by the red arrow) inside BRR [5].

demands for a fast neutron flux facility, which arise due to the absence of adequate number of fast experimental reactors. In Section 2, neutron screens which use a solid shielding material were presented. The screens that were reported are either already successfully implemented and used or are still under development or study. The purpose of their use is the creation of an environment with a neutron energy spectrum free as much as possible from thermal and epithermal neutrons.

Four solid materials were presented, that is, boron, cadmium, hafnium and europium. In principle the material selection depends on the reactor which will host the screen and on the required conditions that should be achieved. Europium has not met great industrial development due to its excessively high cost compared to boron, cadmium, and hafnium which are widespread in nuclear reactors. From the safety point of view, europium oxide reacts readily with water and boron gets highly corrosive at high temperatures. On the contrary, hafnium and cadmium are both corrosion-resistant metals. Regarding their mechanical properties, cadmium has the disadvantages of low melting and low boiling points. Irradiating boron with fast neutrons induces the generation of helium, causing swelling. Hafnium exhibits the best mechanical properties.

The size of the screen depends on the materials depletion and on the experiment requirements. The fact that europium and hafnium activation products have high-cross sections, delays their depletion with time, extending thus the neutron screen operational time. On the contrary, boron and cadmium have only one isotope with high cross-section, and in low abundance, so that both deplete fast with time. Therefore, boron and cadmium neutron screens demand frequent replacement in order to prevent a sudden upturn on the reactor's power deposited on the irradiated sample. Furthermore, in order to compensate the fast depletion, increment of the absorber's content in the screen is sought. This is achieved

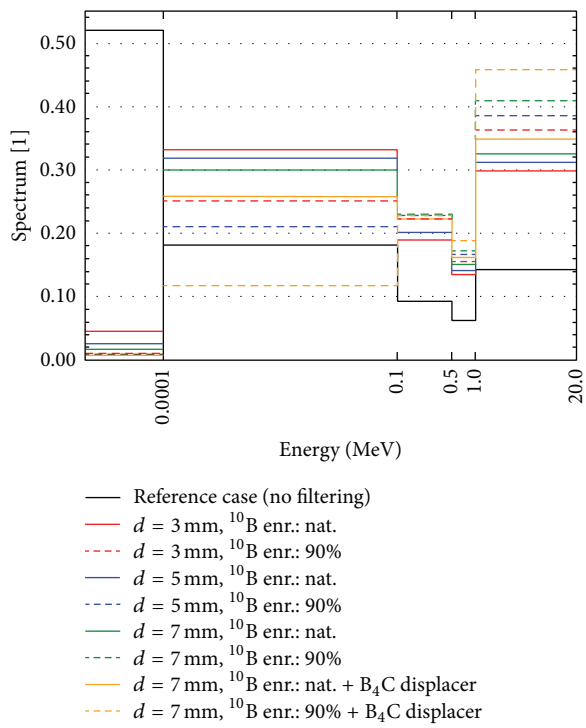


Figure 23: The spectrum in the whole volume of the irradiation tube for each case [17].

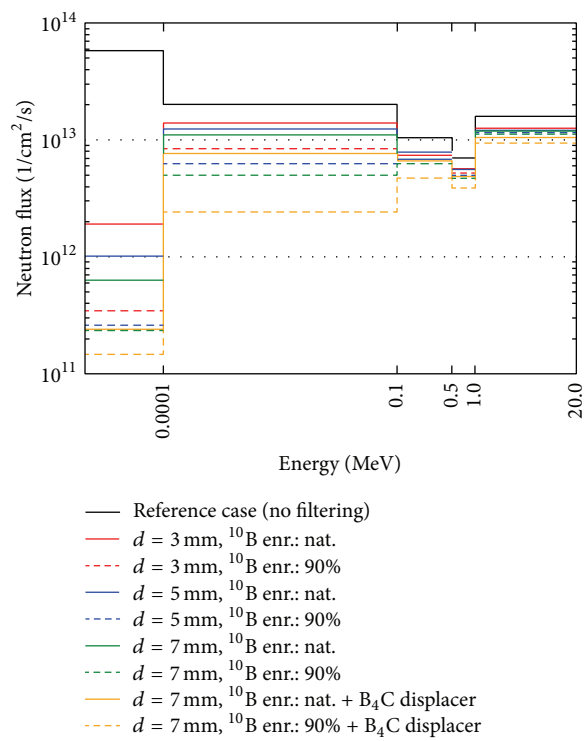


Figure 24: The multigroup flux—corresponding to the 10 MW thermal power of the BRR—in the whole volume of the irradiation tube for each case [5].

either by increasing the volume of the material or through enrichment of the absorber (i.e., ^{10}B enrichment), the cost of the screen being increased accordingly. On the other hand, hafnium has the lowest cross-section of all four materials. Therefore, in order to fulfill the same requirements, larger volume of hafnium must be used, compared to the necessary volumes of other screening materials. In general, few mm of absorber thickness are sufficient.

From all of the four materials, boron has the highest cross-section in the epithermal energy region and its behavior is regular. Europium and its isotopes have also very important cross-section in that region (order of 10^4 barns), but in contrast with boron, they present many resonances. Hafnium and its isotopes also present high neutron capture in the epithermal region with many resonances (order of 10^3–10^5 barns). Finally, cadmium has the lowest epithermal cross-sections (order of 10^3-10^4 barns). The epithermal neutron capture capability of the absorbers results to a neutron spectrum which better represents that of a fast reactor. A combination of absorbers can be utilized in order to create a more efficient neutron screen.

3. Fluid Neutron Screens

In this chapter, neutron screens which use a gaseous or a liquid thermal neutrons absorbing material are presented. The purpose of fluid screens is to generate power transients, that is, power increase or decrease in a few seconds on the volume of a fuel sample. Fuels fission power alteration is achieved through the fluctuation of the absorbent content in the screen (thickness of the screen or absorber's enrichment). The fluctuation is accomplished by pressure variation (i.e., ^{3}He case) or variation of the concentration of the absorber (i.e., ^{10}B in boron compounds) in the screen surrounding the fuel. The performance of variable screens is necessary in order to examine the fuels behavior in case of exposure to sudden power variation conditions.

Three different types of variable neutron screens are reported in this chapter, that is, gaseous ^{3}He, liquid H_3BO_3, and gaseous BF_3. Gaseous BF_3 and ^{3}He screens performance has been abandoned in most of the reactors, since both gases present characteristics that can jeopardize reactor's safety. ^{3}He screens have been replaced by liquid H_3BO_3 ones.

Parameters that can enhance the transient amplitude comprise the core location where the screen is adapted, the distance between the screen and the sample, the absorber type that is used in the screen, the absorber content in the screen, and the operational phase of the reactor. In addition, large amplitude transients can be achieved by combining the screen modification and the reactor power changes, for example, scram.

The fluid screens are classified by material and are divided into subclasses depending on whether they have already been used or they are under development or study.

3.1. Helium-3 Neutron Screens. ^{3}He is a gaseous synthetic isotope of natural helium and is often used in nuclear

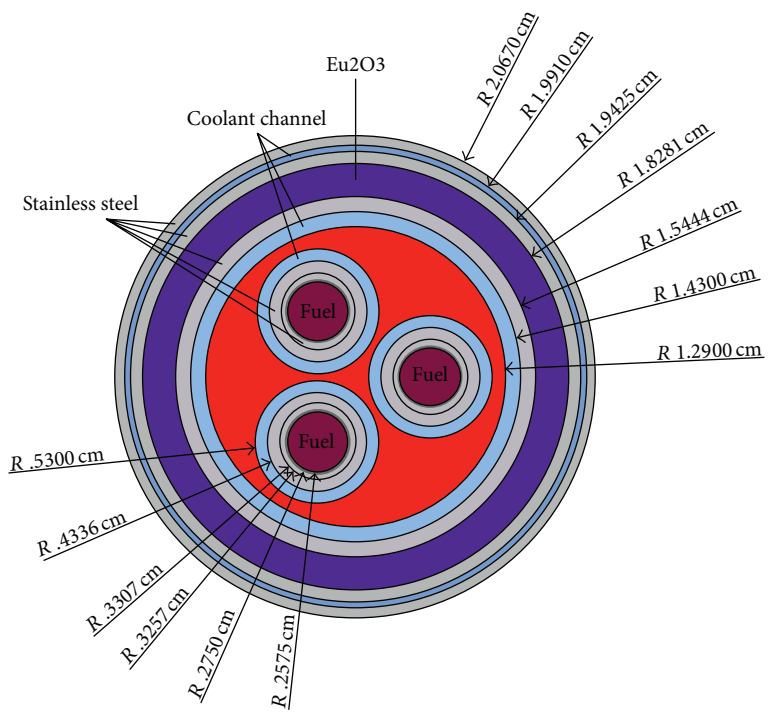


Figure 25: Tri-pin fast flux shielded irradiation design at HFIR (ORNL) [25].

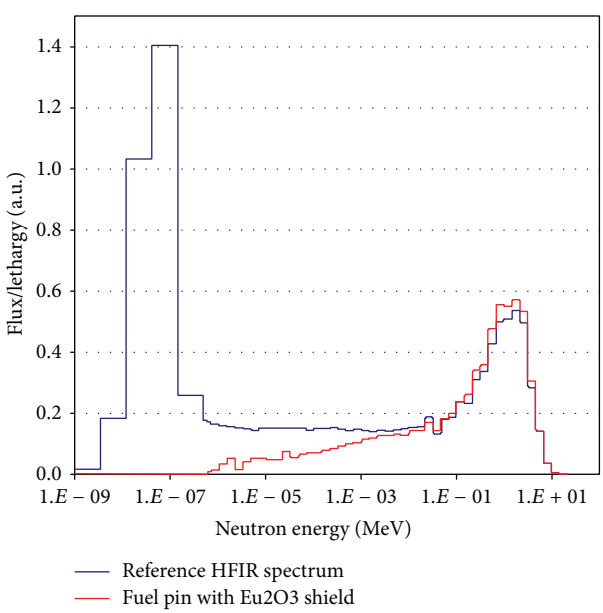


Figure 26: Neutron flux spectra within the FT region of HFIR and the tri-pin fast irradiation facility [23].

reactors. ^{3}He thermal neutron cross-section is extremely high, making this material an excellent absorber. Most of power transients with a ^{3}He neutron screen have nowadays been abandoned for safety reasons aroused due to production of Tritium with ^{3}He activation. The first experiment with a variable ^{3}He neutron screen was made in 1977.

3.1.1. Helium-3 Neutron Screens Already Operational

(1) BR2, SCK.CEN

(a) Pressurized Water Capsule/Cycling and Calibration Device (PWC/CCD). In BR2, at SCK.CEN, a pressurized water capsule/cycling and calibration device (PWC/CCD) (Figure 27) device has been used for fast power transients on fuel pins, under conditions similar to those of a boiling water reactor (BWR) or a PWR. This is accomplished through ^{3}He pressurization/depressurization (1–38 bars). With a 2 mm wide gas gap, a transient factor of 1.75 has been achieved. The PWC/CCD device has been applied in several fuel ramping tests during the last 30 years.

(b) VIC Experiment. A variable pressure ^{3}He screen has been utilized in BR2 for fuel power transient in VIC experiment (Figure 2). The screen surrounded a Na gas loop, in which the sample was inserted (Figure 28). The use of ^{3}He screen provided transient amplitude of 80%. As it has been stated in Section 2 the VIC experiment also consisted of a Cd screen, whose presence reduces the transient amplitude to 25% [3].

(2) Halden, Norway. Although the He gaseous screen performance in most reactors has been abandoned, ramp test facilities with ^{3}He screen are still in operation in the Halden reactor. More precisely, in-pile loops with gaseous ^{3}He are utilized for studying the fuel rods performance under power transient conditions. The experiment is surrounded by Zircaloy, in order to be isolated from the reactor environment. The pressure variance of the gaseous ^{3}He can provide a ramping factor up to 4 and a ramp rate between 100 and

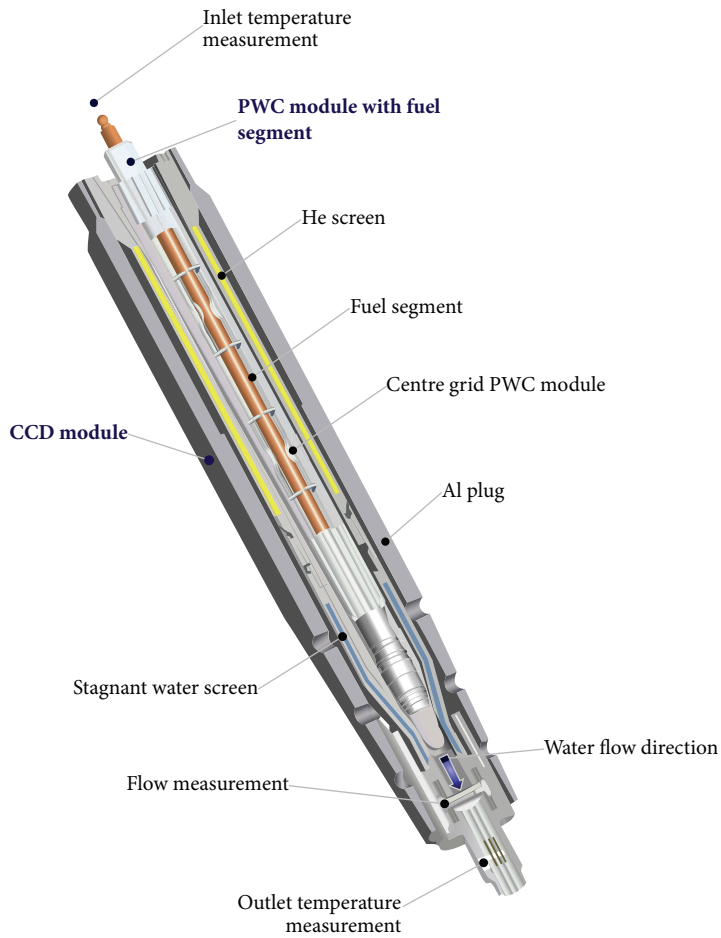


Figure 27: Exploded 3D view of the PWC/CCD assembly [26].

200 W/cm.min. Several power test series have been executed for the investigation of fuels' behavior during normal/off normal operational transients [20]. In Figure 29 four typical power ramp tests are shown.

3.2. Boron Neutron Screens. Boron isotope ^{10}B has an extremely high thermal cross-section. For power transient experiments boron compounds are used. ^{10}B enrichment can be utilized in order to increase the power transient. As stated in Section 2.3 boron utilization should seriously be considered due to its high reactivity effect. Two types of boron fluid screens are described: gaseous BF_3 and liquid H_3BO_3. BF_3 utilization has been abandoned due to its corrosive and poisonous nature and its relatively low absorption cross-section, which requires screen utilization at high pressure.

3.2.1. Boron Neutron Screens Already Operational

(1) HFR, Petten

(a) MOKA-POTRA-BISAR Experiments. A BF_3 gas screen has been used in the MOKA, POTRA, and BISAR boiling water fuel capsules in HFR (Figure 4) for power control in fuel irradiation experiments. The capsule carrier is placed in an aluminum filled space and its horizontal cross-section is described by nine concentric rings surrounding the 10.4 mm radius fuel rod. The rings' compositions and widths in mm, from center to periphery, are respectively Zirconium, 1.34; water coolant, 14.6; aluminum, 9; water coolant, 2; aluminum, 11; stainless steel, 6; BF_3, 13.5; stainless steel, 6; water coolant, 1.5. The high thermal neutron absorption of ^{10}B can cause a power reduction, although the BF_3 gas was not sufficient for large power ramps in the in-pile experiments performed. Stainless steel was used as a construction material and BF_3 gas was inserted into a special annular space surrounding the fuel. The typical pressure of the gas screen was 50 bar, causing 20% power reduction. Due to the low cost of enriched B, the possibility of using enriched BF_3—instead of natural—was considered. With the same amount of enriched BF_3 less space would be necessary and more neutron absorption could be achieved, without pressure changes. Moreover, safety would also be increased. BF_3 is no longer in use at the HFR, because of its corrosiveness, poisonous nature, and its relatively low absorption cross section which requires screen utilization at high pressure.

(b) TOP Experiment. BF_3 gas screen has been used in a TOP-scenario experiment, in order to study the influence of

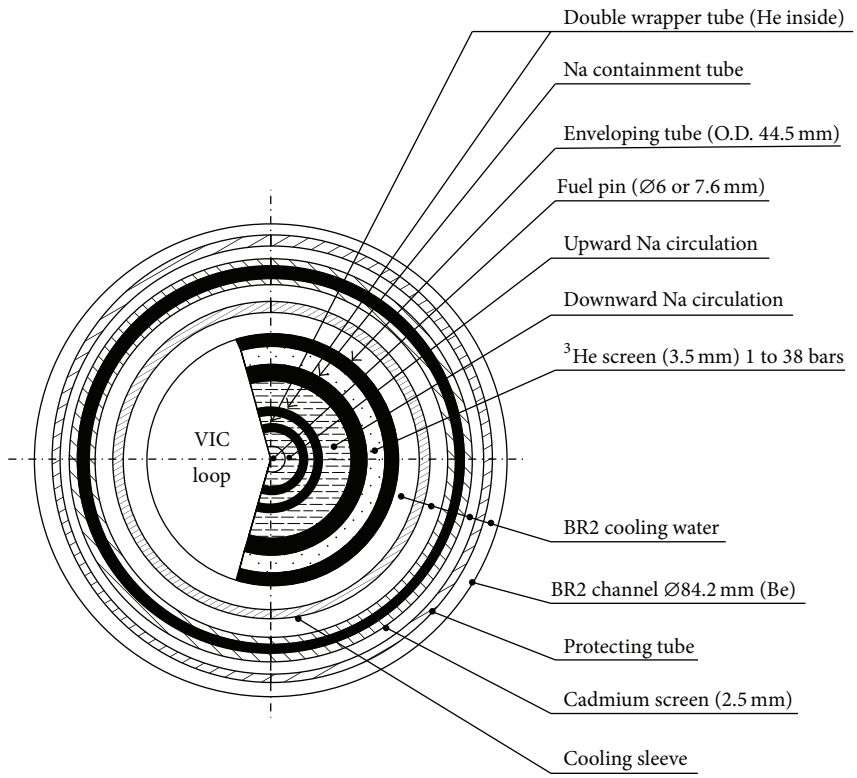


Figure 28: VIC loop cross-section [3].

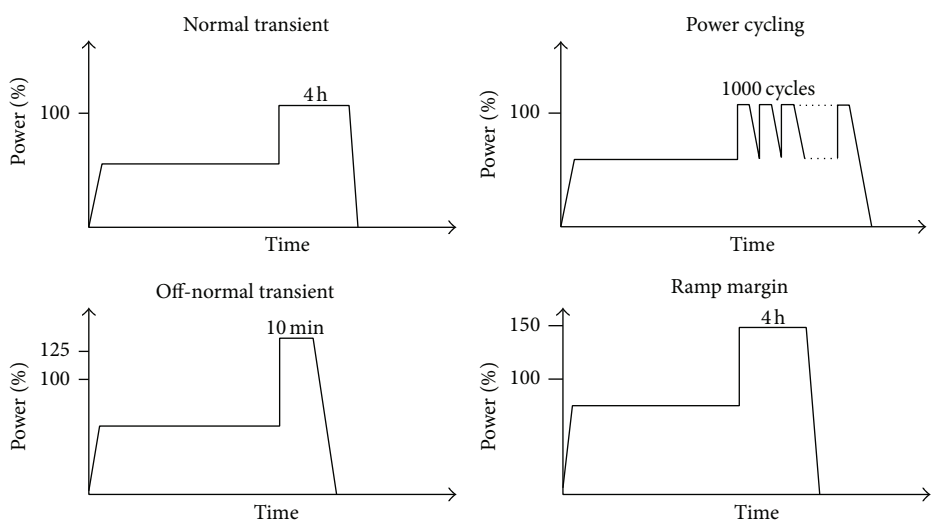
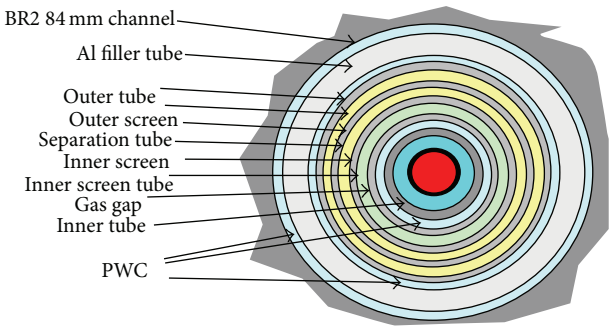


Figure 29: Four typical power ramp tests at Halden [20].

fast reactor fuel's structure under overpower conditions. The facility was placed outside the core in the pool side facility (PSF). The sodium-cooled fast reactor fuel was irradiated with 1800 W/cm maximum. The region around the sodium containment was filled with BF_3 gas, acting as neutron shield. The variation of BF_3 concentration, the displacement of the facility (to and from the core), or the combination of both allowed a power transient factor of 2 to 4. The maximum pressure of BF_3 into the irradiation device was 45 bar. B pressure increase varied slightly the power reduction and neutron absorption.

3.2.2. Boron Neutron Screens under Development or Study

(1) BR2, SCK.GEN. A water screen with a variable concentration of H_3BO_3 was developed at SCK.CEN for fuel power transient experiments (Figure 30). The screen, called VANESSA, surrounds a "classic" pressurized water capsule



PWC diameter	30	mm			
BR2 water channel thickness	2	mm			
Pressure tube thickness	2	mm	Aluminium tube	34–38	mm
Isolation gap thickness	2	mm	He isolation gap	38–42	mm
Screen inner tube thickness	2	mm	Aluminium tube	42–46	mm
Inner screen thickness	3	mm	Inner screen	46–52	mm
Separation tube thickness	2	mm	Aluminium tube	52–56	mm
Outer screen thickness	3	mm	Outer screen	56–62	mm
Screen outer tube thickness	2	mm	Aluminium tube	62–66	mm

Figure 30: Horizontal cross-section of VANESSA with dimensions [28].

(PWC) in BR2 reactor (Figure 1) and was developed in order to replace the existing screen, based on ^{3}He pressurization/depressurization [27]. The modification of boron concentration allows the tuning of the basic linear power. The H_3BO_3 concentration can vary between zero (i.e., pure water) and slightly below the saturation value of about 60 g H_3BO_3 per liter at room temperature. VANESSA screen was loaded in a BR2 high flux channel for transient testing on very high burnup fuel. VANESSA not only provides the variable thermal neutron absorption but, via the thermal balance method, also serves for fuel rod power determination. Through the volume of H_3BO_3 solution generated during the operation, limited number of transients during a cycle takes place.

Various content/enrichments of ^{10}B have been studied. The results showed that with natural boron a fuel power transient factor of about 2.0 could be achieved, while with 100% enriched ^{10}B the power transient factor was increased to 4.5 [27]. The ramp factor has been calculated for two different screen thicknesses; that is, (a) 2 mm: ramp factor 2 (almost linear dependence on B concentration) and (b) 6.5 mm: ramp factor 4 (strongly nonlinear) [28]. A study of the thermal behavior of the VANESSA part showed that H_3BO_3 should be continuously refreshed (at least every 3 s) in order to avoid its stagnancy, that could lead to unacceptably high temperatures. The cooling of the screen is performed by the primary BR2 cooling system water. VANESSA can also be used in combination with RODEO (rotatable device for the execution of operational transients) device (Figure 31). The RODEO concept is a rotating plug inserted in the peripheral 200 mm diameter channel of BR2. The PWC is inserted commonly in an eccentric 84 mm channel within RODEO and can be rotated by 180°. Three solutions for the plug material were studied; RODEO filled with water gives the highest transient ratio [27]. The results are shown in the Table 7.

The power increase in the H3 and H4 channels (Figure 1) was calculated. With zero H_3BO_3 concentration in VANESSA, the power in these channels increases by 0 to 4% by RODEO rotation (from the remote position to the near position). With saturated natural boron H_3BO_3, the rotation has a result of 1 to 5% decrease. For intermediate B concentrations, the influence of the RODEO rotation on the neighboring fuel elements will therefore be extremely small. RODEO rotation can be achieved on a time scale of the order of a few seconds. This leads to the conclusion that very fast power transients with RODEO are possible [27]. Figure 32 shows the combined utilization of VANESSA and RODEO.

3.3. Summarized Results on Fluid Neutron Screens. In Section 3, neutron screens which use a gaseous or a liquid thermal absorbing material were presented. The purpose of fluid screens is to generate power transients in order to examine fuel's behavior in case of exposure to sudden power variation conditions.

Two different materials are typically utilized, helium and boron, that is, gaseous ^{3}He, gaseous BF_3, and liquid H_3BO_3. BF_3 and H_3BO_3 screens were developed with the purpose to replace the widely used ^{3}He screen for safety reasons. However, BF_3 performance also causes safety concerns: through neutron capture, ^{3}He generates tritium and BF_3 can potentially form F_2 which is a poisonous and corrosive gas. Compared to the previous materials, H_3BO_3 can be considered a safe candidate for performing power transients, since it is not corrosive and does not produce active by-products.

^{3}He has an extremely high neutron absorption cross-section. For this reason its replacement with boron compounds having much lower cross-sections cannot provide the same power transients. This can be compensated either by pressure increment or by boron enrichment. Moreover, the screen can be combined with a displacement system (RODEO).

Fluid neutron screens can provide successful power transients. In cases requiring higher power deviations, the screens can be combined with reactor's power alterations or with displacement systems.

4. Overall Conclusions

Neutron screen technology has been developed in order to fulfill several application requirements. The key idea of the neutron screens is the utilization of some materials capability to absorb neutrons at specific energy range. In this report only screens which utilize thermal (and epithermal) absorbers were presented.

First, the capability to simulate fast nuclear reactor conditions in a specific area by adapting a neutron screen is reviewed through the presentation of relevant studies and applications. As arises from the available literature, several neutron spectra have been simulated in different reactors. Four solid materials were presented, that is, boron, cadmium, hafnium, and europium. Their performance characteristics differ in terms of their mechanical properties, compatibility,

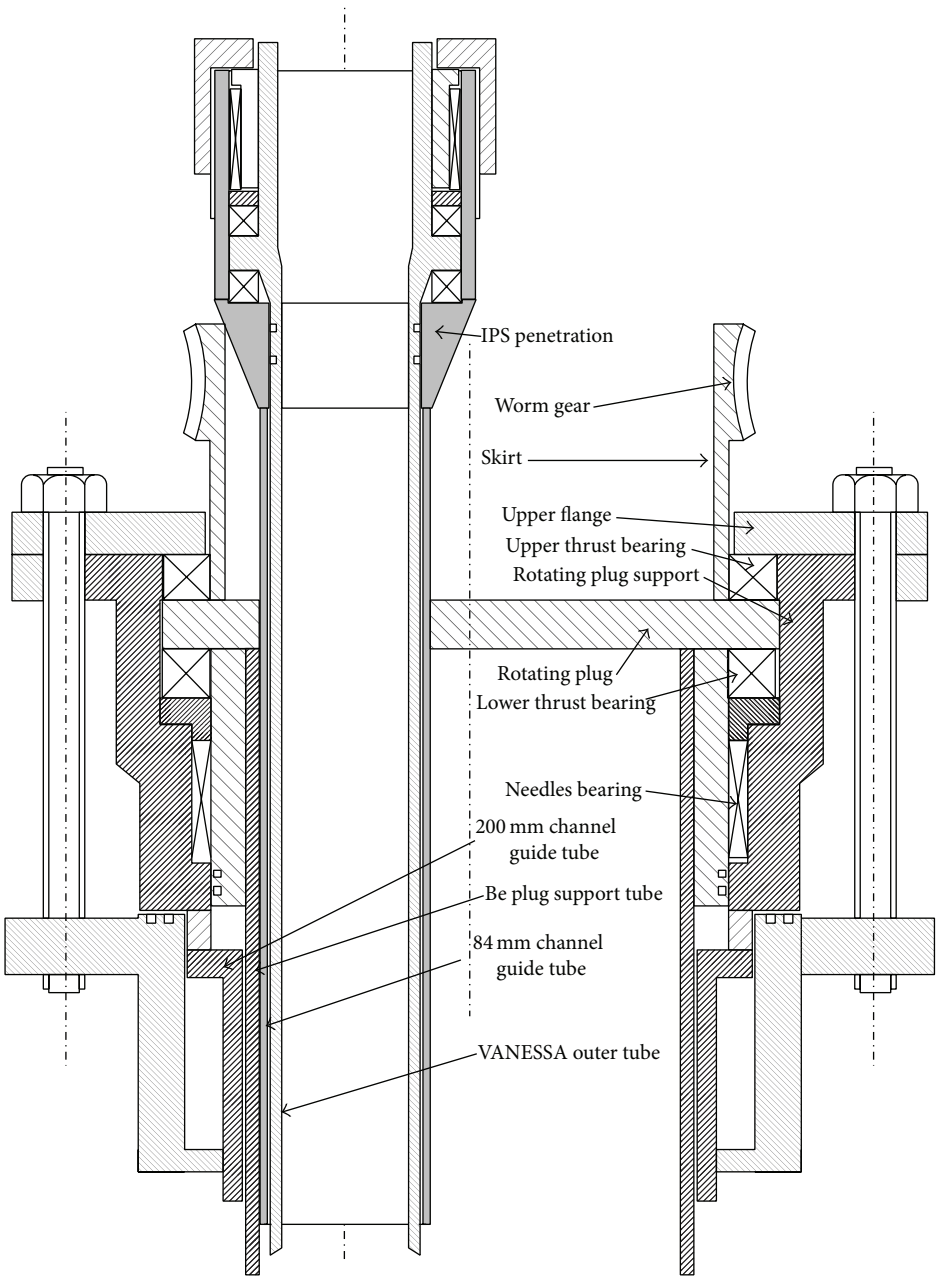


Figure 31: RODEO rotating plug at BR2 [27].

depletion, and so forth. However, in most cases presented in this chapter, the utilized material could be replaced by one of the rest. The material selection is determined by factors such as the reactor conditions desired to be simulated, the reactor type in which the screen will be inserted (and its operational conditions), the reactor coolant (in order to prevent safety issues arising from possible interaction with coolant), and the available space for the screen.

Second, the capability of the existing reactors to irradiate samples under power transients by exploiting the performance of neutron screens is examined based on reported experience. More specifically, the utilization of neutron screens with fluid materials is reviewed. The method is based on the gas pressure or the liquid concentration variation so that power transients can be initiated. Two different materials have been reported, that is, helium and boron, namely, gaseous ^{3}He, gaseous BF_3, and H_3BO_3. The last two neutron screens have been developed in order to replace the unsafe utilization of ^{3}He, because of the tritium generation problem. BF_3 utilization has been limited and eventually abandoned because of its corrosive and poisonous nature. Moreover, its performance could not reach power transient ranges equivalent to those obtained using ^{3}He. Likewise, by using H_3BO_3 the achieved power transients appear much lower, so that the screen is combined with a displacement system (RODEO), thus providing very fast power transients.

Table 7: Three solutions for the plug material have been evaluated: beryllium, light water, and aluminum [27].

	Min linear power (kW/cm)	Max linear power (kW/cm)	RODEO transient ratio
Be			
Without H_3BO_3	1.23	2.45	2.0
63 g/L H_3natBO_3	0.61	1.21	2.0
63 g/L H_310BO_3	0.20	0.44	2.2
H_2O			
Without H_3BO_3	0.39	1.87	4.8
63 g/L H_3natBO_3	0.165	0.67	4.1
63 g/L H_310BO_3	0.055	0.24	4.4
Al			
Without H_3BO_3	1.20	1.52	1.3
63 g/L H_3natBO_3	0.69	0.92	1.3
63 g/L H_310BO_3	0.31	0.42	1.3

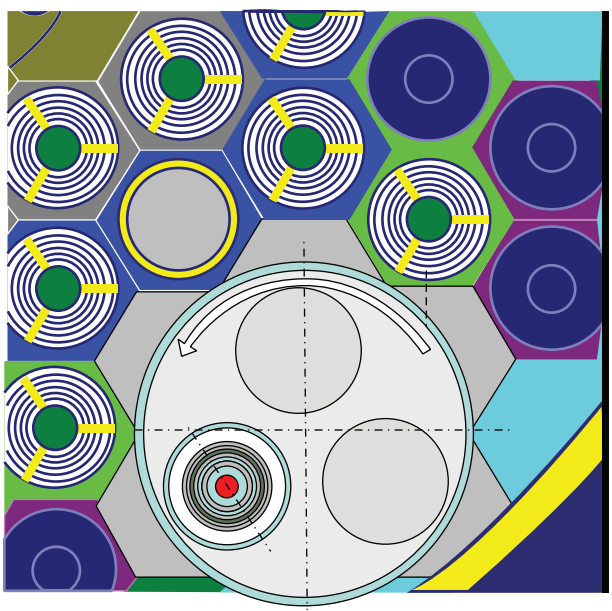


Figure 32: RODEO and VANESSA cross-section [29].

The main conclusion is that neutron screens are worth studying and developing since they can successfully contribute to the creation of desirable special irradiation conditions, which cannot be achieved during normal reactor operation due to technological or economic reasons.

Abbreviations

AFC: Advanced Fuel Cycle
appm: Atomic parts per million
ATR: Advanced Test Reactor
BAGIRA: Budapest Advanced Gas-cooled Irradiation Rig Assembly
BFFL: Boosted Fast Flux Loop
BOI: Beginning Of Irradiation
BOL: Beginning Of Life
BR2: Belgian Reactor 2
BRR: Budapest Research Reactor
BWR: Boiling Water Reactor
CCD: Cycling and Calibration Device
CONFIRM: Collaboration On Nitride Fuel Irradiation and Modeling
dpa: Displacements per atom
DWD: Dry-wet-dry
EFDPs: Effective full power days
EFT: East flux trap
ESF: European Social Fund
F/T: Fast to thermal
FR: Fast reactor
FT: Flux trap
GEN-IV: Generation IV (GEN-IV)
GTL: Gas test loop
GTL: Gas test loop
HFIR: High flux isotope reactor
HFR: High flux reactor
HICU: High neutron fluence irradiation of pebble stacks for fusion
INL: Idaho National Laboratory
ITV: Irradiation test vehicle
LHGR: Linear heat generation rate
LMFBR: Liquid metal fast breeder reactor
LWR: Light water reactor
LWR: Light water reactors
MCNP: Monte Carlo neutron particle
MCWO: MCNP with origen 2
MIT: Massachusetts Institute of Technology
MOX: Mixed oxide
NSRF: National Strategic Reference Framework
ORNL: Oak Ridge National Laboratory
PBR: Pebble bed reactor
pcm: Per cent mille
PSF: Pool side facility
PWC: Pressurized water capsule
PWR: Pressurized water reactor
RODEO: Rotatable device for the execution of operational transients
SCK/CEN: Center for nuclear energy research
SFBR: Sodium-cooled fast breeder reactor

SFR: Sodium-cooled fast reactor
TRIO: Irradiation device with three thimbles
TRIOX: TRIO modified for irradiation of MOX fuels
VIC: Variable irradiation conditions
WP: Work package.

Conflict of Interests

The authors declare that there is no conflict of interests regarding the publication of this paper.

Acknowledgments

This research paper has been cofinanced by the European Union (European Social Fund—ESF) and by the Greek national funds through the Operational Program "Education and Lifelong Learning" of the National Strategic Reference Framework (NSRF)—Research Funding Program: Thales. Investing in knowledge society through the European Social Fund.

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