RPE Radiation Physics & Engineering
An International Peer-Reviewed Journal
Published by K.N. Toosi University of Technology
http://rpe.kntu.ac.ir
Research Article

Radiation Physics and Engineering 2020; 1(4):65–76
https://doi.org/10.22034/RPE.2020.104843

Conceptual design of a high-performance hybrid object for applications of the fast neutron irradiation in MTRs

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HIGHLIGHTS

- Developing a high performance hybrid tool for advanced irradiating applications.
- Fundamental concepts, design criteria and safety issues are introduced coherently.
- The most numbers of core and safety parameters have been influenced.
- Introducing an innovative in-core tool for spectrum shift toward fast neutrons.
- High performance reduction of the unwanted thermal and epithermal neutrons.

ABSTRACT

Fast neutron irradiation is one of the most strategic radiation applications of research reactors. Usually, it is performed around the reactor core containing lower neutron flux. In this paper, a hybrid object has been introduced and analyzed to enhance irradiating applications of the fast neutrons in the core of a Material Testing Reactor (MTR). The tool includes an old-type low-consumed HEU control fuel element, a dry channel, and a Cd filter. It is supposed to be installed at the internal neutron trap (D4 positions) of TRR core configuration. Calculating results are very promising for using the proposed tool to increase neutron fluxes, reduce thermal and epi-thermal neutron fluxes, and shift the neutron spectrum toward the fast neutron region (hardening effect) at the chosen irradiating location. Primary safety parameters are also checked and passed successfully. Furthermore, there are also some other presented safety items which must be checked carefully and conservatively in order to refabricate and install such a irradiating tool in an in-core location of a MTR.

KEYWORDS
Research Reactor
TRR
Neutron Spectrum Shift
Fast Neutron
Irradiating Applications

HISTORY
Received: 25 December 2019
Revised: 1 February 2020
Accepted: 18 February 2020
Published: October 2020

1 Introduction

Presently, a significant number of operating research reactors have become multi-purpose facilities producing radioisotopes, performing neutron radiography, semiconductor doping, neutron activation analysis, material and fuel tests and experiments, Boron Neutron Capture Therapy (BNCT), and a wide range of industrial and medical applications (Hedayat, 2016b; IAEA, 1999, 2005a, 2007b,a; Choo et al., 2011; Raina et al., 2006). However, usually thermal neutrons have been more interesting and pointed out than the fast neutrons while the fast neutrons have own characteristics and benefits for nuclear irradiating applications.

Generally, in order to utilize a multi-purpose research reactor effectively, specific neutronic conditions must be provided while all of reactor Operating Limits and Conditions (OLCs) must be reserved assuming standard safety issues (IAEA, 2005b, 2006, 2008c,a,b). Developing model must include the scheduling and providing all irradiation requirements within safety margins (Hedayat et al., 2009b; Hedayat, 2014a,b; IAEA, 2008c,a,b). There are also some specific applications such as those applications that needs fast neutron requires some specific characteristics that cant be generally provided within the general design characteristics of a multi-purpose research reactor. Quality Assurance (QA) for products and services should also be provided for economic tasks (IAEA, 1999, 2007b,a; Choo et al., 2011). Then some specific tools may provide such a desired condition with a restricted area of the reactor.

Some specific applications of fast neutron flux (IAEA, 1999, 2007b,a; Choo et al., 2011; INRA and IAEA, 2015) in MTRs as follows:

- Study of ionizing and damaging effect of high energy

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neutron flux on the structure of fission power plant and fusion reactors especially for:

  - Enhancing material performances and choosing the most appropriate alloys (e.g. for core structures, core baffles, core barrels and pressure vessels).
  - Enhancing material resistance to develop the void formation and swelling effects.
  - Material tests due to analyses of the destructive effect of ionizing (fast neutron) fluxes on the reactor materials specially those structural materials that contribute to extended life-time of the new generation of the nuclear reactors in the future.
  - Some partial neutron spectrum shift to explore compatible and resistant materials and components against different type and values of neutron and fluxes specially for GEN IV reactors.

- Production of some specific radioisotopes, e.g. ^{89}Y(n, p)^{89}Sr.
- Aging identification of antiques and ancient samples (geochronology) using Argon gas.
- Gem-stone coloring;
- Medical applications as an special type of neutron therapy that can be used against tumors in depth or cancerous cells that are hardly distributed in the whole body.
- Studies of health physics including biological effects of the fast neutrons.
- Nuclear transmutation of high level radioactive waste.
- Specific Neutron Activation Analyses (NAA) for materials sensitive to fast neutrons.
- Taking High-resolution neutron radiography for thick samples and hydrogen based materials.
- Taking a specific energy window of neutron beam imposing suitable moderator and filters.

In 2004, Jang et al. (Jang et al., 2004) developed an out-core filtered fast neutron irradiating system in the Texas A&M University Nuclear Science Reactor using a lead-bismuth alloy. That system was used to study long-term biological effects of fast neutron on cultured cell and living animals. In 2008, Breitkreutz et al. (Breitkreutz et al., 2008) used thermal fission spectrum of U-235 for medical applications like the irradiation of human cancer for the radiography, tomography, and biological studies at FRM II reactor. In 2011, Marques et al. (Marques et al., 2011) characterized the fast neutron irradiating facility of the Portuguese Research Reactor. There are also some other reported fast neutron irradiating applications in RRs (IAEA, 1999, 2007b,a; Choo et al., 2011; INRA and IAEA, 2015). Briefly, this article propose an in-core hybrid high-performance tool for the particular fast neutron applications to be used in a conventional thermal-spectrum MTRs for the very first time.

This present article reports a new hybrid method and tool introduced to increase fast neutron flux and shift the neutron spectrum toward the hardening effect just at the chosen irradiating location. It means by both regional and spectral shifts of neutrons toward fast neutrons at the chosen in-core irradiation location whereas we have still a thermal spectrum reactor core, entirely. Th results are promising to be used at conventional research reactors called MTRs. In this way, there are some important issues to be considered for such an application as follows:

- Hardening the neutron spectrum or neutron spectrum shift from thermal to fast region at the chosen irradiating location using HEU fuel plates.
- Omitting the thermal and epi-thermal neutron fluxes as much as possible but just at the irradiating location using neutron filters.
- Increasing the fast neutron flux and fluence at the chosen in-core irradiating location by means of direct positioning of fission spectrum (i.e. HEU fuel element) and eliminating of the localized moderator (i.e. positioning of a dry channel).
- Regarding safety issues for core management specially corresponding OLCs.
- Long-term corrosion and burn up effects for old consumed HEU fuel elements.
- Gas cooling of the dry channel.
- Suitable shielding procedure for further fabrication and installation.

The next section introduces the considered methodology to get a high performance object for the fast neutron applications. The case study and used material and method are also introduced. Moreover, the calculating tool and benchmarking process are introduced. Section 3 presents the results and discussion including neutron spectrums, neutron fluxes, and corresponding reactor OLCs via figures and tables. Finally, Section 4 presents the conclusion. Th results are highly promising. The introduced object can be located in the most effective irradiating location (the central neutron trap), shift the neutron spectrum toward the fast region (as hardening effect), reduce the thermal neutron fluxes, adapt with U-235 fission spectrum, and effectively filter the thermal and epithermal neutron fluxes, at the chosen in-core irradiating location.

2 Materials and Methods

2.1 Methodology: Enhancing the fast neutron flux and hardening neutron spectrum

In the present work, a hybrid object is proposed to be installed and used in the central neutron trap of the TRR (AEOI, 2009). Then the final hybrid object can be used for

increasing the fast neutron flux and hardening the neutron spectrum at the chosen irradiating location, effectively. The next methods and objects are chosen to handle this consideration as following:

- High performance installation at the most effective location (in-core central neutron trap).
- Using old low consumed HEU control fuel assemblies (AEOI, 1966; IAEA, 1980, 1992) to shift neutron spectrum toward the fast region known as hardening effect just at the chosen irradiating location.
- Using the dry channel to reduce the thermal neutron scattering and shift the neutron spectrum toward the fission spectrum similarly just at the irradiating location as much as possible.
- Using cadmium filters to omit the thermal and epithermal neutron fluxes at the chosen location as much as possible.

2.2 Case Study

Tehran Research Reactor (TRR) is chosen as the case study. Figure 1 presents a general overview of TRR as a pool type MTR. It is an open pool type 5 MW MTR. Fuel elements are made of LEU U_3O_8-Al alloy. Two different types of absorber rods including Shim Safety Rods (SSR) and Fine Regulating Rod (FFR) are used in a fork-shape. SSR and FRR are made of the Ag-In-Cd (80, 15, 5 Wt%) composition and the Stainless Steel Type AISI 316L, respectively. TRR uses water as the core cooling system and moderator. The major components or systems of the Pool-Type Research Reactor facility consist of the pool (including embedment and accessories), bridge and support structure, core, cooling systems, control and instrumentation, ventilation system, and experimental facilities (AEOI, 1966, 2009).

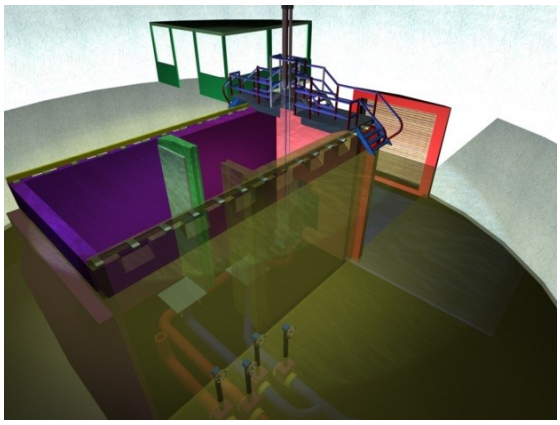

Figure 1: TRR (a large pool type MTR).

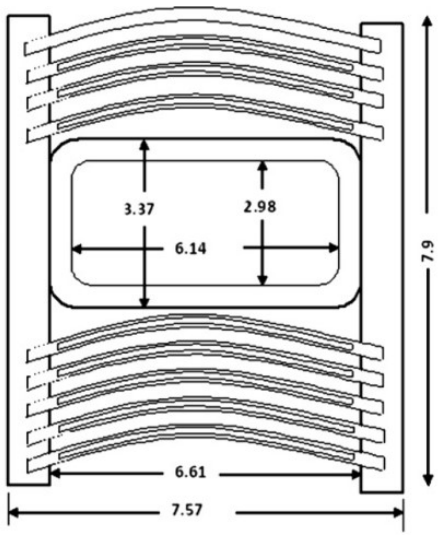

Figure 2: Old HEU control fuel element.

The reactor core (AEOI, 2009; Hedayat, 2014a,b) is composed of MTR-type core assemblies inserted in the grid plate. This is a conventional thermal-spectrum research reactor called MTR. In other words, MTRs are usually suitable for conventional thermal-neutron irradiating applications such as radioisotope productions. Furthermore, MTRs usually use Low Enriched (i.e. 20%) Uranium (LEU) instead of the obsolete High Enriched (i.e. 90%) Uranium (HEU) fuel elements. The assemblies may be arranged in a variety of lattice patterns depending on experimental requirements. They can be classified as: Standard Fuel Element (SFE), Control Fuel Element (CFE), Absorber Rods, Neutron Reflector Elements (GR-BOX for the Graphite Reflector Boxes), Irradiating Boxes (IR BOX) and or Empty Boxes. Table 1 shows the main specifications of the TRR core components.

In addition to general properties of TRR, Fig. 2 presents the geometrical properties of the suggested old type HEU control fuel element (AEOI, 1966; IAEA, 1980, 1992) and Fig. 3 shows the general view of the suggested sample holder. Table 2 presents a comparison between the new LEU and old HEU fuel elements. Figure 4 shows the chosen core configuration for the considered analyses, where the values present the percentage of the fuel burn ups.

Figure 3: Suggested sample holder (Guide box region of an control HEU fuel assembly).

Table 1: TRR Core Elements.

Material or Characreistic	Value
Fuel elements:	
U-235 per Standard Fuel Element (SFE)	290 g
U-235 per Control Fuel Element (CFE)	214 g
U per fuel plate	214 g
Fuel Meat:	
Type	LEU U_3O_8 (20%)
U density	2.9617 g.cm^{-3}
Meat density	4.76 g.cm^{-3}
Void fraction	10.0%
Weight percentage	U-235 12.45%, U-238 49.78%, O 11.18%, Al 26.59%
Aluminium Meat	Purity 99.6% Density= 2.7 g.cm^{-3}
Frame and covers	Aluminium 6061 Density= 2.7 g.cm^{-3}
Shim and safety rods absorber	Ag-In-Cd Alloy (80, 15, 5% in weight respectively) Density= 10.17 g.cm^{-3}
Control rods' Cladding Material	AISI-316/L stainless steel Density= 7.95 g.cm^{-3}
Gap between absorber and clad	He (1 atm. pressure)
Regulating rod	AISI-316/L stainless steel Density= 7.95 g.cm^{-3}
Gird plate	Grid array X-Y Pitch: 7.71× 8.1 cm
Grid plate	AL-1100 (thickness= 12.7 cm) 54 holes (diameter: 6.17-6.19 cm); 40 holes (1.9-2.2 cm)
Reflectors	Water/Graphite

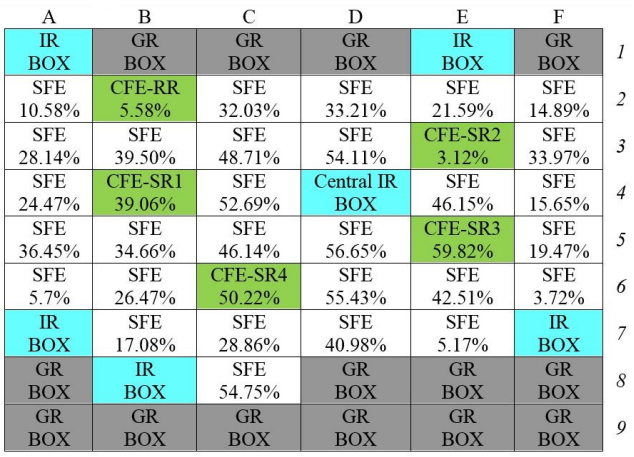

Figure 4: Core configuration and burn up distribution of the chosen operating core (for following studies).

2.3 Nuclear data and codes

The MTR-PC package is used for cell and core calculations based on macroscopic cross sections. It was developed by the INVAP S.E. to perform practical neutronic, thermal-hydraulic, and shielding calculations of MTR-type reactors in personal computers (INVAP, 2006c). The MTR-PC package is not a fully independent nuclear software package. It is composed of some updated well known nuclear codes such as WIMSD and CITVAP codes in addition to some supplementary codes such as POS-WIMS code or HXS utility. Furthermore, its nuclear data have been updated specially for research reactors by INVAP S.E. (INVAP, 2006a). It is worth mentioning that this package has been severally used for MTR calculations and design applications by the vendor (INVAP, 2006a; Villarino and Doval, 2011) successfully worldwide.

Macroscopic Cell Cross Sections (MCS) are calculated using the WIMSD5B code (INVAP, 2006c) which is an updated version of the well-known WIMSD4 code (ORNL, 1991). The default library (INVAP, 2006e), proposed for research reactors by the developer is used for WIMSD cell calculations. The library is updated and comprises 69 discrete energy-group structured microscopic cross-sections particularly for conventional MTRs by INVAP. The effective cross-sections for influencing isotopes in the library, such as Ag-109, Cd-113, and In-115 are in good agreement with resonance results (INVAP, 2006e).

The POS-WIMS code (INVAP, 2006d) is used to condense and homogenize the required macroscopic cross-sections for suggested core regions and the energy-group structures for the TRR (AEOI, 1989, 2009). Then the HXS utility (INVAP, 2006b) is applied to transform and save the resulting cross-sections in a suitable library for the CITVAP code (INVAP, 2006a). In order to calculate and influence OLCs, two different MCS libraries are created and used:

- Influencing fuel burn up and consumption at cold zero power state to simulate start up conditions. This method especially is suitable to reduce risks of refueling and start up procedures.
- Influencing fuel burn up and consumption at hot full power state to simulate operating conditions (AEOI, 1989, 2009).

Table 2: A comparison between new LEU and old HEU fuel assemblies.

Parameters	LEU	HEU
Meat material	U_3O_8-Al	(UAl alloy)-Al
Enrichment	20%	93.15%
Total fuel plate thickness	0.15 cm	0.127
Meat thickness	0.07 cm	0.0508
Cladding thickness	0.04 cm	0.0381
Water channel thickness	0.27 cm	0.310 cm
Meat width	6.0 cm	6.1595
Total plate width	6.700 cm	6.644 cm
Active Height	61.5 cm	59.69
Side wall thickness	0.45 cm	0.476
Length of side plates	8.010 cm	7.950 cm
Inner distance between side wall	6.7 cm	6.61 cm
CFE dimension	8.1×7.7×61.5 cm^3	8×7.61×87.31 cm^3
FP cladding and side walls material	Al-6061	Al-6061
Absorber type Fork Oval	Fork	Oval
Absorber material for shim safety rods	Ag In Cd	Ag In Cd
Absorber material for fine regulating rod	AISI-316L ss	AISI-316L ss
Material in the gap between cladding and absorber	He, at 1 atm.	None
Guide plates material	Al-6061	Al-6061
Cladding material for absorber plate	AISI-316L	AISI-316L
Uranium per fuel plate	15.26 g	12.2 g
Weight Of U-235 per Standard Fuel Element	196 g	290 g
Weight Of U-235 per Control Fuel Element	213.7 g	97.6 g
Density of total uranium in meat	3.0 $g.cc^{-1}$	0.69951 $g.cc^{-1}$
Total density of meat	4.8 $g.cc^{-1}$	3.16367 $g.cc^{-1}$
Density of U-235 in meat	0.591 $g.cc^{-1}$	0.6516 $g.cc^{-1}$

The CITVAP code is used for overall core calculations with macroscopic cell cross-sections (MCS), fuel consumption and burn up following, and simulation of refueling tasks (Hedayat, 2014a,b). It is a new version of the CITATION-II code (Fowler et al., 1971), developed by the Nuclear Engineering Department of INVAP (INVAP, 2006a). As noted previously, all of these mentioned codes and nuclear data are as different objects of MTR PC package that collected together and accessible within a unified software platform called MTR PC package.

2.3.1 Code validation and general benchmarking calculations

The WIMSD and the CITVAP (CITATION) codes have been severally benchmarked against 10 MW IAEA benchmarking problem (AEOI, 2009; IAEA, 1980, 1992; Hedayat et al., 2009a,b) and comprehensively validated during several experiments at the RA-2, RA-6, RA-8, RP-10, NUR, TRR and ETRR-2 research reactors (AEOI, 1989, 2009; Hedayat, 2014a, 2016a; INVAP, 2006c; Khalafi and Gharib, 1999; Villarino and Doval, 2011).

In addition to previous benchmarking problems, several benchmarking calculations have been performed by the author for the TRR against available experimental results, reactor safety amendment (AEOI, 1989), and FSAR for TRR (AEOI, 2009). Results show a very good agreement between the experimental values and TRR core calculations (Hedayat, 2014a,b, 2016a,c,b).

In addition to the previous mentioned code validation, verifications and benchmarking process, the chosen operating core configuration of TRR is also benchmarked against the relevant experimental critical state. The next simulation and analyses of the reactor core are performed based on one of the operating core of the TRR.

3 Results and discussion

3.1 General overview of the chosen core configuration

Figure 4 shows the burn up distribution of the chosen core configuration. Table 3 presents the main core parameters of this core configuration assuming the corresponding reactor OLCs. Furthermore, Table 4 presents the results of the first critical core state of the chosen core configuration via the simulation of the control rod positions. The results completely pass the safety margins (AEOI, 1989, 2009) and compatible with the experimental situation.

Table 3: Main core parameters of the chosen operating core (for following studies).

Neutronic Parameters	Calculated Parameters	Safety Margin
Effective Multiplication Factor	1.03074	-
Core Excess Reactivity (pcm)	2982.5	-
Shutdown Margin (pcm)	8373.8	>3000
Total Power Peaking Factor	2.45	<3.0
Integral Worth of SSRs	11356.3	-
Safety Reactivity Factor (SRF)	3.8	>1.5
Fine Reg. Rod Worth	280.4	$< \beta_{eff}$

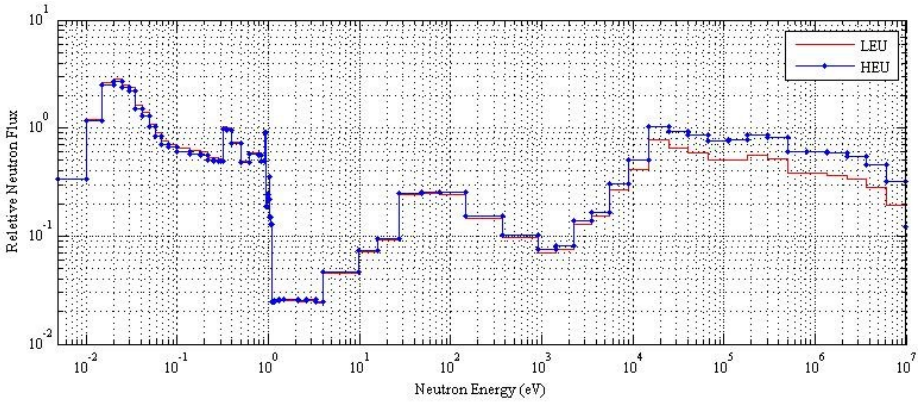

Figure 5: Neutron spectrum over LEU and HEU fuels.

Table 4: Critical state simulation results of the chosen operating core (for following studies).

Parameters		K_{eff}	Reactivity (pcm)
Control Rod	Extracting Percent		
SSR 1	61.5%		
SSR 2	61.5%		
SSR 3	61.5%	0.99835	-165.0
SSR 4	61.5%		
FRR	50%		

3.2 The location effect

Table 5 presents the radial three-group neutron flux distribution over the irradiating boxes. The results confirm that the central in-core (D4) location has the most suitable neutron fluxes for irradiating tasks. Table 6 presents the axial neutron flux distribution over the central in-core (D4) position. It depicts the most effective location and suitable capsule length that shall be required to get homogeneous radioactivity of irradiating samples.

3.3 The HEU effect

Figure 5 shows the neutron spectrum of the LEU and HEU fuels regarding geometrical buckling effect of core configuration using the WIMSD5B code. It is worth mentioning that the WIMSD5B code can estimate the neutron transport equation. The energy ranges are chosen according to the 69 energy-group distribution of the WIMSD code (ORNL, 1991). The results show the shift of the neutron spectrum toward the fast region of the HEU respect to the LEU fuels known as the hardening effect particularly at the chosen irradiating location. It is worth mentioning that we can not change the reactor type or increase the total net value of the flux because we have a fixed reactor core power distributed over definite core objects; However, we can relatively shift and enhance the fast neutron toward the chosen irradiating location regarding both regional and spectral shift.

Table 5 and Fig. 6 indicate that the central neutron trap is the best practical location that can be used for the installation of the sample holder. Figure 6 and 7 present the fast neutron flux distribution over the core configuration before and after inserting HEU sample holder respectively.

It is worth mentioning that, although results show that the C5 has the maximum value of the fast neutron flux, the D4 is more practical to be used for the installation. Then the D4 (central neutron trap) is chosen for the following studies because of its practicality to be used for irradiating sample holder. Furthermore, the neutron fluxes have movement toward the inserting location (D4) after the insertion of the HEU sample holder.

Table 7 shows the numerical effect of the HEU sample holder on the neutron flux distribution. The results show that the fast neutron fluxes relatively enhanced 31% and 43% (i.e. at the chosen irradiating location) without and with Al guide boxes respectively. Figure 8 shows the axial neutron distribution along the D4 with and without inserting the HEU sample holder. The results are highly promising and indicate following impact as:

- The major effect of the internal HEU as a powerful fission convector;
- The minor effect of the Al guide box without any sensible interaction of Al box with fast neutrons.

Table 8 presents the safety parameters against the safety margins of the FSAR for TRR (AEOI, 2009). The results show that the safety margins completely passed even for a fresh HEU (the most reactive one). There are two important safety items which must be checked before any further experiment in the reactor:

- Corrosion factors and any leakage condition from an old consumed HEU fuel assembly shall be checked even for a low consumed fuel.
- Old irradiated HEU fuels are radioactive even a low consumed fuel. This means that a suitable shielding shall be required during the fabrication process.
- Other safety margins of the reactor OLCs (i.e. particularly those parameters that relate to the kinetic parameters) shall be checked out specially to keep safe against possible Reactivity Induced Accidents (RIA).

Table 5: Radial three-energy group neutron flux distribution over the irradiating locations.

Irradiating Boxes		Neutron Flux ($n.s^{-1}.cm^{-2}$)			
		Fast (0.821 MeV-10.0 MeV)	Epithermal (0.625 eV-0.821 MeV)	Thermal (0-0.625 eV)	Total
IR 1	A1	4.38E+12	9.69E+12	2.44E+13	3.84E+13
IR 2	E1	6.33E+12	1.38E+13	3.06E+13	5.07E+13
IR 3	D4	2.04E+13	4.18E+13	7.67E+13	1.39E+14
IR 4	A7	9.37E+12	1.88E+13	4.11E+13	6.93E+13
IR 5	F7	8.62E+12	1.79E+13	3.80E+13	6.45E+13
IR 6	B6	8.28E+12	1.73E+13	3.63E+13	6.19E+13

Table 6: Axial three-energy group neutron flux distribution over the central irradiating location.

Distance from down (cm)	Neutron Flux ($n.s^{-1}.cm^{-2}$)			
	Fast (0.821 MeV-10.0 MeV)	Epithermal (0.625 eV-0.821 MeV)	Thermal (0-0.625 eV)	Total
3.075	1.25E+13	2.58E+13	5.16E+13	8.98E+13
9.225	1.85E+13	3.81E+13	6.95E+13	1.26E+14
15.375	2.22E+13	4.57E+13	8.30E+13	1.50E+14
21.525	2.35E+13	4.83E+13	8.78E+13	1.60E+14
27.675	2.24E+13	4.60E+13	8.35E+13	1.52E+14
33.825	1.96E+13	4.00E+13	7.23E+13	1.32E+14
39.975	1.63E+13	3.31E+13	5.97E+13	1.09E+14
46.125	1.29E+13	2.63E+13	4.72E+13	8.64E+13
52.275	9.53E+12	1.93E+13	3.50E+13	6.38E+13
58.425	5.99E+12	1.22E+13	2.41E+13	4.23E+13

Table 7: Effect of the HEU on the D4 neutron flues.

Irradiating Boxes	Neutron Flux ($n.s^{-1}.cm^{-2}$)			
	Fast (0.821 MeV-10.0 MeV)	Epithermal (0.625 eV-0.821 MeV)	Thermal (0-0.625 eV)	Total
Empty Box	1.79E+13	3.69E+13	9.35E+13	1.48E+14
HEU CFE without Guide Box	2.34E+13	4.84E+13	8.17E+13	1.54E+14
HEU CFE with Al Guide Box	2.59E+13	5.24E+13	7.41E+13	1.52E+14

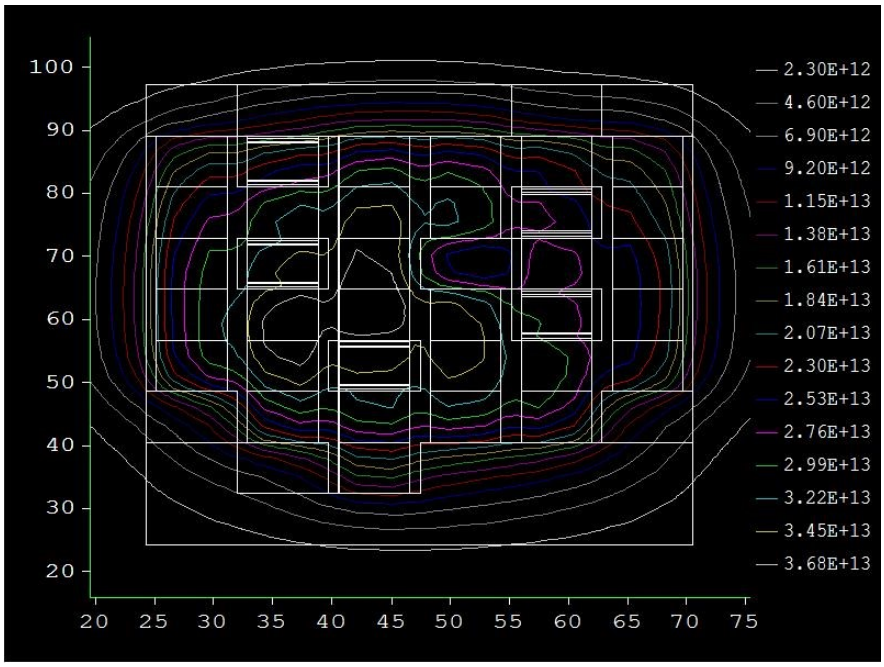

Figure 6: Fast neutron flux distribution over the chosen core configuration (before inserting HEU).

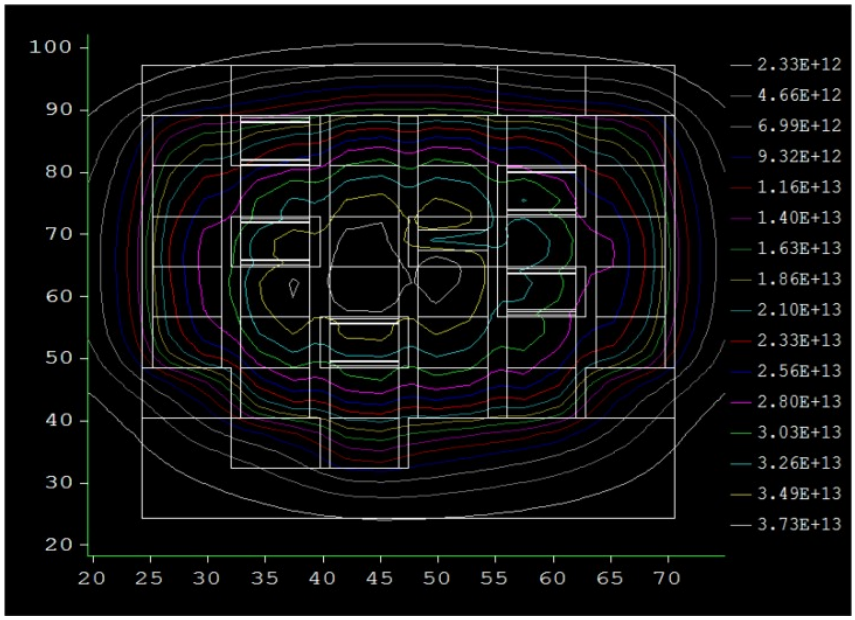

Figure 7: Fast neutron flux distribution over the chosen core configuration (after inserting HEU).

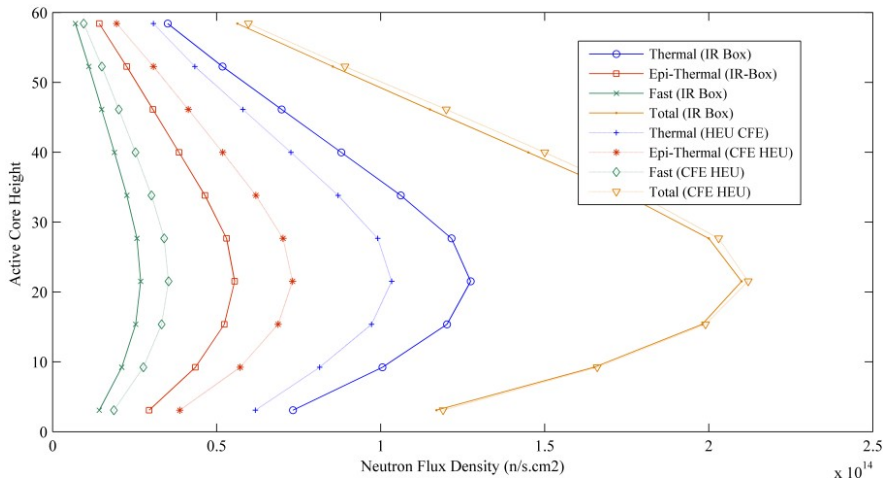

Figure 8: Axial distribution of three-energy group neutron flux distribution along the D4 with and without the HEU sample holder.

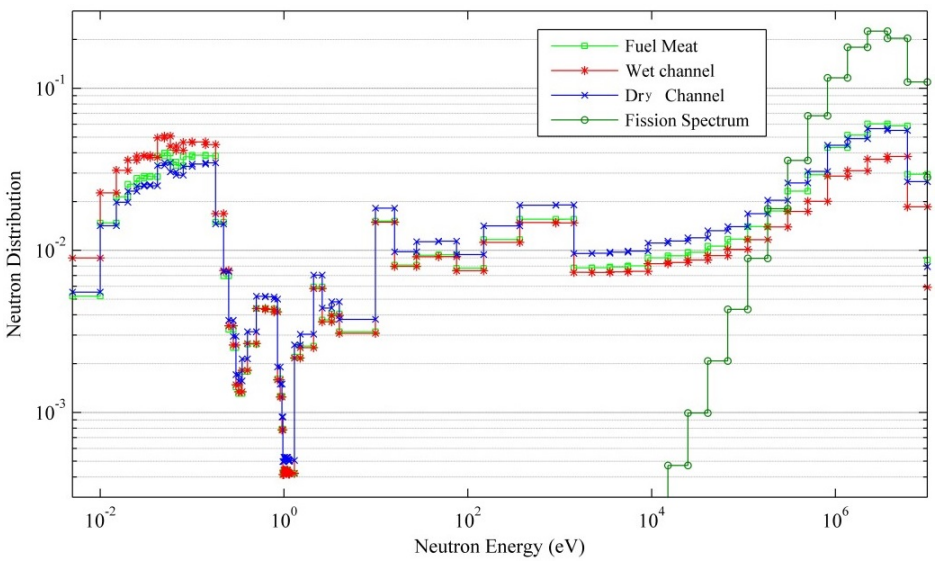

Figure 9: Neutron spectrum of the fission, fuel meat, wet and dry channel based on cell calculations.

- To check out fine regional and geometrical effect of strong filters via accurate MonteCarlo codes such as MCNP code at the irradiating location.

Table 8: Prediction of safety parameters after inserting a virtual fresh HEU sample holder.

Neutronic Parameters	Calculated Parameters	Safety Margin
Effective Multiplication Factor	1.05100	-
Core Excess Reactivity (pcm)	4852.7	-
Shutdown Margin (pcm)	6409.9	> 3000
Total Power Peaking Factor	2.49	< 3.0
Integral Worth of SSRs	11262.6	-
Safety Reactivity Factor (SRF)	2.32	> 1.5
Fine Reg. Rod Worth	273.2	$< \beta_{eff}$

3.4 Dry channel effect

Figure 9 shows the neutron spectrum of the fission spectrum, fuel meat region, wet and dry channel regarding geometrical buckling effect of core configuration using the WIMSD5B code. It is worth mentioning that the WIMSED code is an estimator solver of the neutron transport equation (Duderstadt and Hamilton, 1976; ORNL, 1991; Stacey, 2007). The results show the effective hardening effect. It means the shift of the neutron spectrum toward the fast region by means of the dry channel. It is well-matched with the fuel meat region but it is completely different with the fission spectrum because of existing water moderator in the reactor core. In other words, the fission spectrum in the fuel region can be simulated via the introduced HEU fission converter and dry channel. The dry channel is proposed to be located thinly around the last fuel plate before sample holder without any effective neutron moderator.

Table 9 presents the numerical investigation of the neutron spectrum shift from the thermal to the fast region captured by using the dry channel in addition of the HEU fission converter. Based on the core calculations, results show that fast and epi-thermal neutron fluxes relatively enhanced 9.6 and 9.5% (i.e. just at the irradiation location) respectively while the thermal neutron fluxes relatively decreased 49%. Then, this can be a very useful method to reduce thermal neutrons just at the fast-neutron-irradiating location.

Although some MTRs like NUR use out-core dry channels (IAEA, 1980, 1992, 2007b,a) for irradiating applications, there are some major safety items which must be checked out accurately and conservatively to insert a dry-channel in an internal region of the MTR core configuration as following:

- A suitable cooling system for irradiating samples and inner region of the last fuel plates surrounding the dry channels shall be required.

- High positive reactivity insertion coefficient during water leakage to the inner region of sample holder shall be considered for further safety analyses perhaps as the most impressive problem during DBA.

3.5 Cadmium filters

Usually, Cd capsules use to reduce thermal neutron flux (i.e. similarly just at the irradiating location) because of large neutron absorbent cross section while it is a non-burnable absorbent known as gray-type absorber (Hedayat, 2016a,b, 2017; Matzie, 2008). Figure 10 shows the Cd neutron absorption cross section including $(n, total)$ and (n, γ). It is captured by using JANIS software and depicts the type of this neutron absorbent.

In this research, the Al guide box of the inserted control HEU fuel is virtually replaced with a mixed alloy with the Cd. Figure 11 shows the effect of imposing Cd mixture within Al layer on the neutron fluxes. They are calculated based on the transport neutron theory using WIMSD5B code.

Because of large impact of the Cd on the core excess reactivity, usually a thin layer of Cd is used. Then the 20% (weight percentage) of Cd is preferred. This is approximately equivalent with the 0.18 mm of a Cd capsule.

Table 10 presents main characteristics of Cd mixture performances to omit the thermal neutron flux based on the core calculations using CITVAP code. This is an approximation based on the homogenized materials at the chosen irradiating location.

The results show that:

- According to the both solvers of neutron transport and neutron diffusion codes 20% (weight percent) of using Cd in the Al guide boxes would be recommend. It is approximately equivalent with a 0.18 mm of Cd capsule thickness. This can be so effective to reduce 98% of the thermal neutron fluxes.

- Core calculations show that a 10% (weight percent) of using Cd within the Al Guide boxes can also be effective to reduce 56% of the thermal neutron fluxes at the chosen irradiating location. It is equivalent to about 0.09 mm of Cd capsule thickness. Further investigations could be performed for the fine regional and geometrical effects of strong filters by using the MCNP code.

4 Conclusion

In this article, applications and the importance of the fast neutron irradiation in conventional thermal-spectrum MTRs are briefly reviewed and discussed via samples. A new hybrid tool is proposed for irradiating applications of fast neutron in MTR. Then conceptual analyses are performed via corresponding neutronic and safety parameters by using MTR-PC package (i.e. including WIMSD5B, POS-WIMS, HXS, and CITVAP codes).

An operating core configuration of the TRR (a 5 MW MTR) is chosen to develop studies. Calculating codes have been severally verified and benchmarked successfully against available experimental data, the 10MW IAEA benchmarking problem, and safety reports. Then the first critically state of the chosen core configuration is also simulated within a 3D core modeling. Results are completely satisfactory. Then the corresponding safety parameters

Table 9: Effective hardening effect of the dry channel based on the core calculations.

Irradiating Boxes	Neutron Flux ($n.s^{-1}.cm^{-2}$)			
	Fast (0.821 MeV-10.0 MeV)	Epithermal (0.625 eV-0.821 MeV)	Thermal (0-0.625 eV)	Total
Wet Channel	2.40E+13	4.98E+13	7.65E+13	1.50E+14
Dry Channel	2.63E+13	5.45E+13	3.88E+13	1.20E+14

Table 10: Effects of the Cd mixture in the Al guide box for omitting the thermal neutron fluxes using homogenized core calculations.

Irradiating Boxes	Cadmium Weight Percent (%)	ρ (pcm)	Neutron Flux ($n.s^{-1}.cm^{-2}$)			
			Fast (0.821 MeV-10.0 MeV)	Epithermal (0.625 eV-0.821 MeV)	Thermal (0-0.625 eV)	Total
Wet Channel	0.0	4828.8	2.40E+13	4.98E+13	7.65E+13	1.50E+14
	0.1	4637.7	2.37E+13	4.94E+13	7.37E+13	1.47E+14
	1	3540.8	2.21E+13	4.66E+13	5.78E+13	1.26E+14
	5	2289.9	2.01E+13	4.34E+13	3.89E+13	1.02E+14
	10	1969.5	1.95E+13	4.25E+13	3.39E+13	9.59E+13
	20	1886.0	1.94E+13	4.15E+13	3.25E+13	9.34E+13
	30	1744.1	1.92E+13	4.18E+13	3.06E+13	9.16E+13
Dry Channel	0.0	4051.5	2.63E+13	5.45E+13	3.88E+13	1.20E+14
	0.1	3909.0	2.60E+13	5.41E+13	3.66E+13	1.17E+14
	1	3125.2	2.47E+13	5.20E+13	2.39E+13	1.01E+14
	5	2261.5	2.31E+13	4.96E+13	9.41E+12	8.21E+13
	10	2034.4	2.26E+13	4.89E+13	5.38E+12	7.69E+13
	20	2051.0	2.27E+13	4.81E+13	8.76E+11	7.17E+13
	30	1898.0	2.23E+13	4.84E+13	1.96E+12	1.50E+14

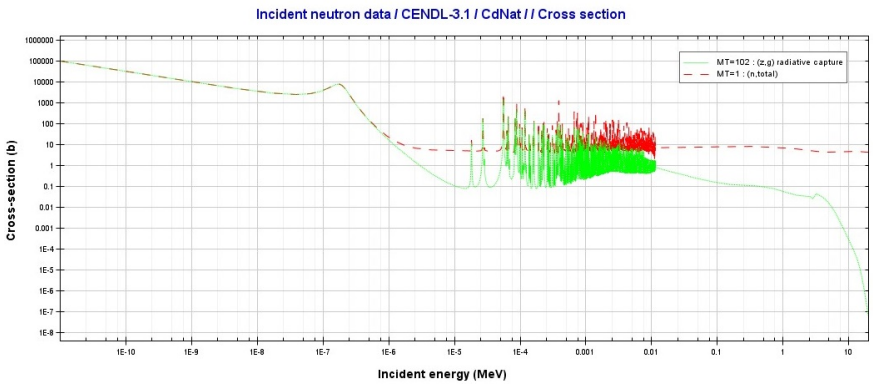

Figure 10: Microscopic neutron absorption cross section of Cd.

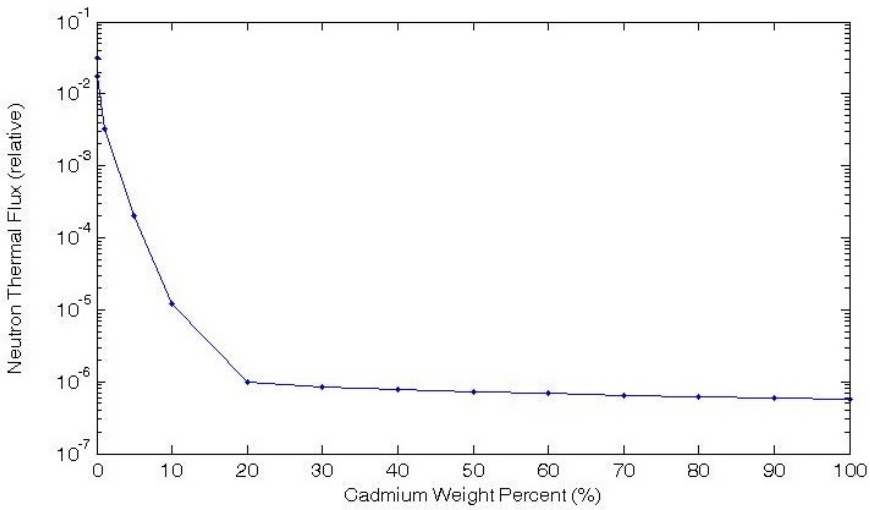

Figure 11: The effect of Cd mixture within the Al layer on the neutron flux using WIMSD5B.

are also calculated and checked against the FSAR for TRR, successfully.

After that, three different materials and objects are introduced to design a virtual hybrid in-core sample holder which can harden the neutron spectrum toward the fast neutron region and relatively reduce the thermal and epithermal neutrons. The hybrid irradiating tool includes a low consumed old HEU control fuel element, a dry channel, and Cd filter. The Cd filter is composed of a mixture of Cd within the Al guide box region or even a Cd capsule.

Calculations are performed and analyzed to explore geometrical and material studies via choosing sensitive and impressive parameters. Calculating results are very promising to be used effectively at central (in-core) location of MTRs for the installation. Then further results shows suitable hardening effect as shifting neutron spectrum toward the fast energy region, increasing the fast neutron flux, and omitting the thermal neutron fluxes but just in the hybrid tool instead of the whole reactor core. They can be mainly mentioned in detail as follows:

- HEU fuels can be so effective for the hardening shift of neutron spectrum and relatively enhancing the fast neutron fluxes up to 40% at the chosen irradiating location;
- Dry channel can reduce thermal neutron fluxes up to 50% at the irradiating location;
- And using Cd filters particularly if it is mixed within a wide structure such as the structural parts; then it can approximately omit the thermal neutron flux even up to 98% but just within the designed hybrid tool.

Then the introduced hybrid tool can be promisingly a high performance fast-neutron-irradiating utility, but on the other hand, there are some nuclear safety items for old type HEU fuels which must be checked out accurately and passed conservatively before any physical fabrication or installation as follows:

- Corrosion factors and any leakage condition from an old consumed HEU fuel assembly should be checked carefully even if it would be a low consumed fuel.
- Old irradiated HEU fuels are radioactive even an old low consumed fuel. This requires a suitable shielding procedure during the fabrication process.

There are also some items to be considered to install an in-core dry channel as follows:

- A suitable cooling system for irradiating samples and inner region of last fuel plates surrounding the dry channel shall be required.
- The possible positive reactivity coefficient shall be considered for further safety analyses against a possible RIA during a water leakage to the inner region of sample holder.

It is also worth mentioning that, such safety parameters and items must be completely checked and passed before any physical deployment, fabrication, and installation. But if such safety items passes successfully, this is a very high-performance hybrid tool for such economic irradiating applications of fast neutrons.

Abbreviations

CFE	Control Fuel Element
DBA	Design Basis Accident
GR	Graphite Boxes
IR	Irradiating Boxes
FRR	Fine Regulating Rod
HEU	High Enriched Uranium
LEU	Low Enriched Uranium
MTR	Material Testing Reactor
OLC	Operating Limits and Conditions
RIA	Reactivity Induced Accident
SFE	Standard Fuel Element
SSR	Shim Safety Rods
TRR	Tehran Research Reactor

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