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RESEARCH ARTICLE

Determination of out-core fuel burnup in TRIGA PUSPATI

Mohamad Annuar Assadat Husain [a, d,*], Suhairul Hashim [a, b], Sib Krishna Ghoshal [a], Mohamad Hairie Rabir [c], Norasalwa Zakaria [c], Muhammad Rawi Mohamed Zin [c], David Bradley [e, f]

[a] Department of Physics, Faculty of Science, Universiti Teknologi Malaysia, 81310 UTM Skudai, Johor, Malaysia
[b] Centre for Sustainable Nanomaterials (CSNano), Ibnu Sina Institute for Scientific and Industrial Research (ISI-SIR), Universiti Teknologi Malaysia, 81310 UTM Skudai, Johor, Malaysia
[c] Malaysia Nuclear Agency, Bangi 43000 Kajang, Selangor, Malaysia
[d] Atomic Energy Licensing Board, 43800 Dengkil, Selangor, Malaysia
[e] Department of Physics, Faculty of Engineering and Physical Sciences, University of Surrey, Guildford, United Kingdom
[f] Institute for Healthcare Development, Sunway University, 46150 Petaling Jaya, Malaysia

[*] Corresponding author: annuarassadat@gmail.com

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Abstract

The investigation was conducted on the out-core neutron flux and burn-up at irradiated fuel stored in TRIGA PUSPATI research reactor tank. This is required to examine whether the thermal and/or fast neutron flux can influence burn-up of the irradiated fuel stored in the same vicinity of the reactor core, the fuel rack being located 1 m above the core. MCNPX code was used to simulate fast and thermal neutron flux for the reactor operating at 750 kW. In this work, the computational model was created using MCNPX version 2.7 with the evaluated nuclear data file for thermal neutron scattering law data (ENDF7) cross-section data library and using a 10 cm x 10 cm x 10 cm mesh model. The results showed that the axial distribution for thermal neutrons occurred at energy lower than 1 x 10-6 MeV. Thermal neutron traveled at the maximum distance of 78 cm due to thermalization by moderator. Based on the maximum distance traveled by the thermal neutron, the thermal neutron did not reach the storage rack located 1 m from the core, hence there was no burn-up occurring at the irradiated fuel since burn-up can only occur in the thermal neutron region. For fast neutron, the axial distribution energy was higher than 1 x 10-6 MeV and traveled more than 158 cm. The reaction time for the fast neutron was too short to result in burn-up due to its fast travel.

Keywords: Neutron flux, thermal neutron, fast neutron, MCNPX, RTP

INTRODUCTION

The PUSPATI TRIGA Mark II reactor (typically referred as RTP, Reactor TRIGA PUSPATI) at the Malaysian Nuclear Agency, Malaysia, is the only research reactor in Malaysia. TRIGA stands for Training Research Isotopes, General Atomics while PUSPATI stands for Pusat Penyelidikan Tenaga Atom Tun Dr. Ismail (the Tun Dr. Ismail Center for Atomic Energy Research). The reactor achieved its first criticality on June 28, 1982, and since then, it has been operated for an average of 500 hours per year. The RTP is loaded with 3 types of standard TRIGA UZrH fuels of differing uranium composition: 8.5 wt %, 12 wt %, and 20 wt %. In terms of the design, the RTP is a custom model that allows storage of the irradiated fuel in the reactor tank, at the out-core. In other words, the fuel storage rack is located 1 m from the core, in the vicinity of the reactor tank.

The research reactor is built as a pool-type reactor. The core is immersed in an open water-pool 2 m in diameter, 6.5 m deep and lined with aluminum. The thickness of the reactor wall is 2.5 m, made from high-density concrete. The reactor uses light water as the moderator and for cooling. The cylindrical core is surrounded by a graphite reflector. Other in-core assemblies include the central thimble (CT), pneumatic transfer tube (PTS), dry tube (DT), delayed neutron activation-bare (DNA-bare), and Neutron activation-cadmium (DNA-CD) monitors. The RTP is also equipped with a rotary rack and thermal column. The out of core facilities are comprised of 4 beam ports, referred to as beam port 1, 2, 3, and 4. Fig. 1 shows a cross-sectional view and the internal components of the RTP.

The present core configuration is represented by core No. 15, assembled in 2012. This configuration achieved power up to a maximum of 750 kW. This core consists of 114 fuel elements, arranged in six concentric rings, referred to as rings A to F. Fig. 2 shows a cross-sectional view of core configuration No. 15.

Nuclear fuel utilization in a reactor is referred to as burn-up. The burning of the nuclear fuel is a neutronic process, depending on neutron-induced fission. In the RTP, an americium-beryllium (Am-Be) source is used as the neutron source in order to initiate the fission process. The fission process emits more neutrons from the fuel in the form of fast neutron, epithermal/intermediate neutrons, and thermal neutrons (Rabir *et al.*, 2015). These neutrons are subjected to migration, diffusion, and slowing-down process. Neutrons are moderated and then diffused in thermal energy region and finally absorbed in surrounding media. It is assumed that there are leaked neutrons i.e neutrons that escape and travel relatively far from the vicinity of the fissionable material in the reactor core. Hence, there is a possibility of the fast neutron being thermalized and absorbed by irradiated fuels stored at the storage rack and induce fission.

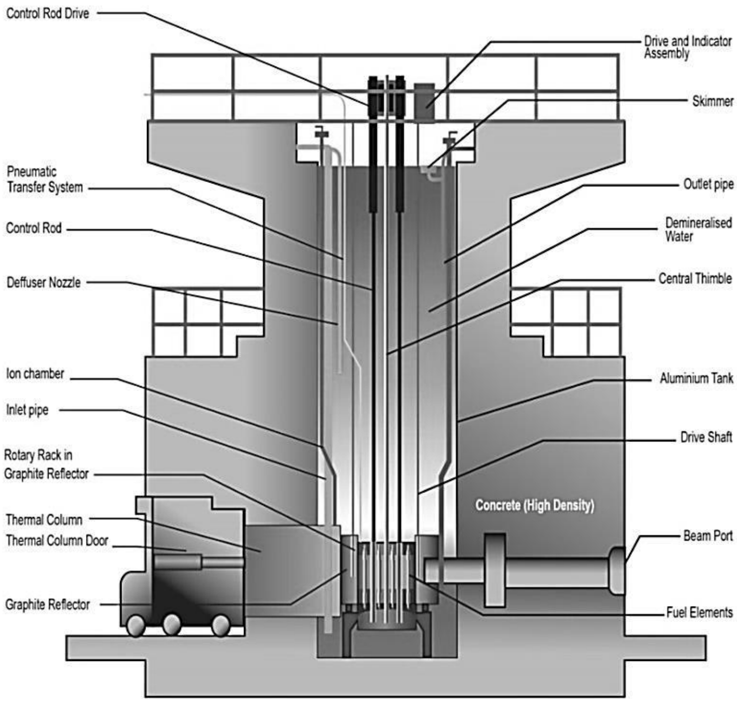

Fig.1 Cross-sectional view, showing internal components of the RTP.

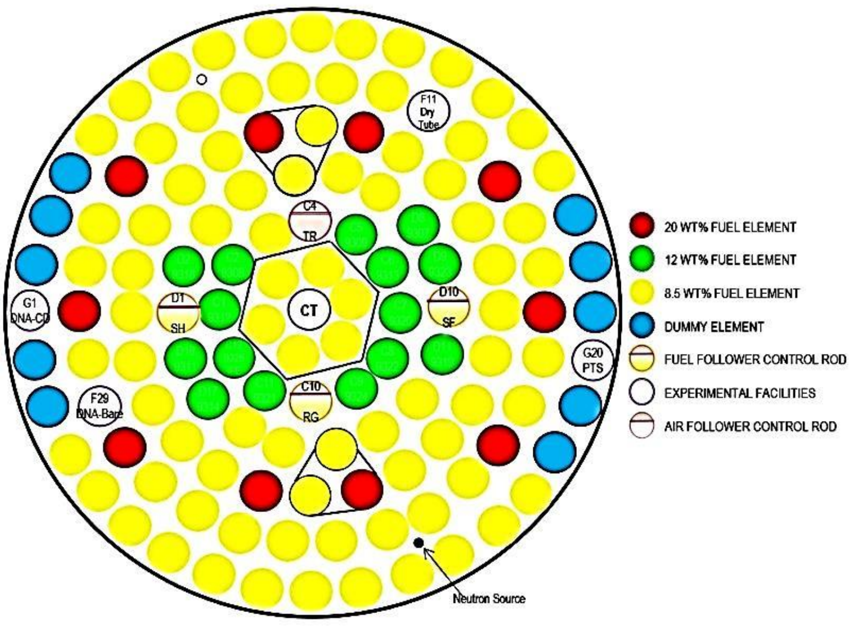

Fig. 2 Cross-sectional view of Core Configuration No. 15.

The objectives of this work were two-fold: to determine thermal neutron flux around irradiated fuel in the core and to determine the burn-up of the stored fuel. In other words, this work investigated whether the thermal and/or fast neutron flux could influence the burn-up of the irradiated fuel stored at the storage rack in the same vicinity of the reactor core. The neutron flux (ϕ) is a measure of the intensity of neutron radiation, determined by the rate of flow of neutrons. The neutron flux value is calculated as the neutron density (n) multiplied by neutron velocity (v), where n is the number of neutrons per cubic centimeter (expressed as neutrons/cm3), and v is the distance the neutrons travel in 1 second (expressed in cm/s). The unit for neutron flux is $n.cm^2.s^{-1}$. The flux shape is the term applied to the density or relative strength of the flux as it moves around the reactor core.

A method to solve the neutron transport equation is by using Monte Carlo method which is a stochastic statistical simulation method. According to Bakkari *et al.* (2013), Monte Carlo N-Particle (MCNP) code is capable of providing the correct representation of detailed geometry, transport effects, and continuous energy cross sections. It is a favored choice because of its general modeling capability, the correct representation of detailed geometry, transport effects, and continuous energy cross-sections. There are numerous codes that use the Monte Carlo method. The widest spread being used is Monte Carlo N-Particle Extended (MCNPX) utilized to calculate the interrogating neutron flux and predict acquired spectra accounting for the interrogation geometry and detector effect (Pelowitz *et al.*, 2008). Table 1, adapted from work by Alnour *et al.* (2013), summarizes the neutron flux distribution at the RTP calculated using MCNPX.

Table 1 Recorded average thermal neutron flux and epithermal neutron flux.

Study	Year	Thermal neutron flux (x 10+12 n cm–2 s–1)	Epithermal neutron flux (x 10+11 n cm–2 s–1)
Liew	2010	2.26 ± 0.10	1.11 ± 0.11
Alnour	2012	2.33 ± 0.08	1.23 ± 0.02

Burn up refers to the utilization of nuclear fuel in the reactor. According to Ueda *et al.* (1993), the burn-up parameters can be obtained from measurement of the neutron flux produced by the burn- up dependent neutron source in a spent fuel bundle. Several works on burning up the determination of TRIGA fuel in the core were conducted by many researchers (Rabir *et al.*, 2013; Snoj *et al.*, 2011; Dalle *et al.*, 2002), however, there has been no discussion on burning up for out-core fuel.

METHODOLOGY

In this work, the computational model was created using MCNPX version 2.7 with the evaluated nuclear data file for thermal neutron scattering law data (ENDF7) cross-section data library. Core 15 configuration is used in the calculation. Tallies F1, F4 and F8 are the most important output data, which refers to counts of particles passing a given surface (F1), mapping of thermal neutron travel through the moderator in the reactor pool (F4) and radiation pulses (F8). The MCNP model developed in this study is a full 3D model as shown in Fig. 3. The mesh size adopted for this model is 10 cm x 10 cm x 10 cm.

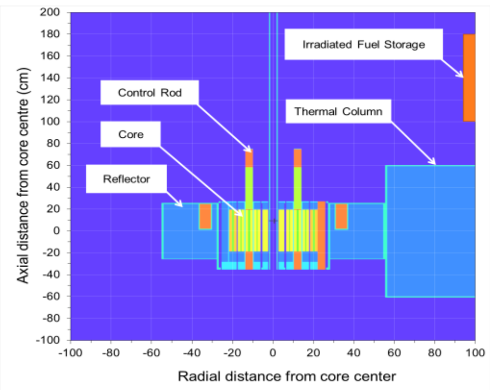

Fig. 3 Axial view of the MCNPX model for RTP.

RESULTS AND DISCUSSION

The path of neutron distribution, representing the distance a neutron travels between interactions, depends on the type of material and the energy of the neutron. After each collision, the energy is decreased and the neutron path length is affected accordingly. The axial thermal neutron flux at 750kW core power level has been simulated using MCNPX and is presented in Fig. 4. The behavior of the thermal neutron flux distributions displays the typical "bell-shaped" for the positions between -100 cm to +100 cm, while a rapid decline of the flux occurs from the position +100 cm up to +200 cm from the core. According to Matsumoto (1998), the reactivity is larger at the center of the core.

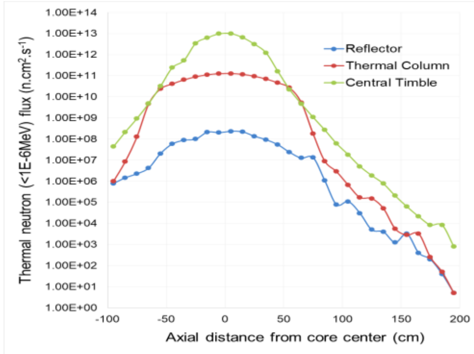

Fig.4 Axial thermal neutron (<1E-6MeV) flux (n.cm^2.s^{-1}) profile at full core power of 750 kW.

As indicated in Fig. 4, the thermal neutron flux is the greatest at the position of the central thimble at 1 x 10+13 n.cm2.s-1. The maximum neutron fluxes at the thermal column and the reflector are greatly reduced to 1 x 10+11 n.cm2.s-1 and 4 x 10+8 n.cm2.s-1 respectively as a result of elastic and inelastic neutron-nucleus scattering collisions. The results are consistent with previous work by (Rabir *et al.*, 2015), in which the thermal and fast neutron flux was reported to be concentrated at the central thimble while there was a significant reduction of the flux particularly at the position of the DNA-Cd monitor due to the thermal neutrons being absorbed by the cadmium.

The fast neutron (neutrons of energy higher than 1 x 10-6 MeV) travel profile was also analyzed in this work. A similar pattern to that for the thermal neutrons is obtained for the fast neutrons shown in Fig. 5, with a "bell-shaped" flux distribution occurring between +100 cm and -100 cm from the core. The fast neutron flux then shows a declining trend from +100 cm up to +200 cm.

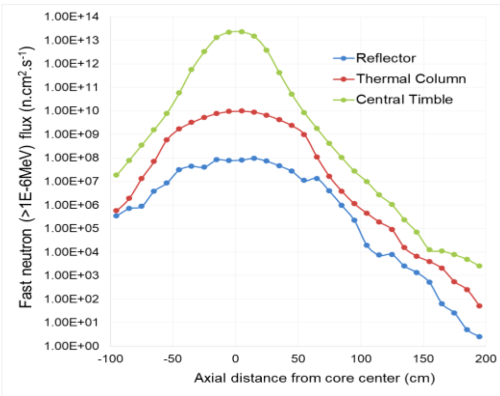

Fig.5 Axial fast neutron (>1E-6MeV) flux (n.cm^2.s^{-1}) profile at full core power of 750 kW.

The "bell-shape" becomes less apparent for the fast neutron flux at the reflector and the thermal column as a result of attenuation due to scattering and collisions. The maximum fast neutron flux is observed to occur at the central thimble, at 5 x 10+13 n.cm^2.s^{-1} due to its central position in the core, as depicted in Fig. 2 and Fig. 3. When the neutron flux density is plotted using MATLAB on the 20 cm x 20 cm grid, one can easily determine the distance traveled by the thermal neutrons. Fig. 6 illustrates the contour of the thermal neutron flux axial distribution and the corresponding radial 3D view.

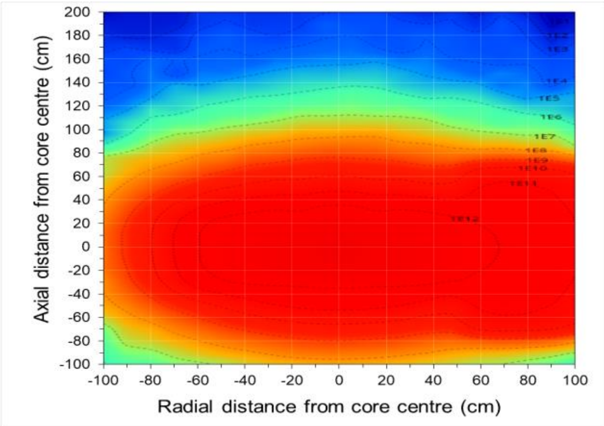

Fig.6 Axial distribution of thermal neutron contour flux at full core power of 750 kW

Fig. 6 also indicates that the thermal neutron flux of 1 x 10+9 n.cm^2.s^{-1} travels a length up to 78 cm. When Fig. 6 is interpreted with Fig. 7, the results suggest the epithermal region of the neutron flux starts at distance 78 cm up to 158 cm from the core. In brief, the higher of thermal neutron flux is 1 x 10+13 n.cm^2.s^{-1} while the greatest fast neutron flux is 5 x 10+13 n.cm^2.s^{-1} shown in Fig. 7

The axial flux distribution contour for fast neutrons is illustrated in Fig. 8 together with the corresponding radial 3D view of the flux in Fig. 9.

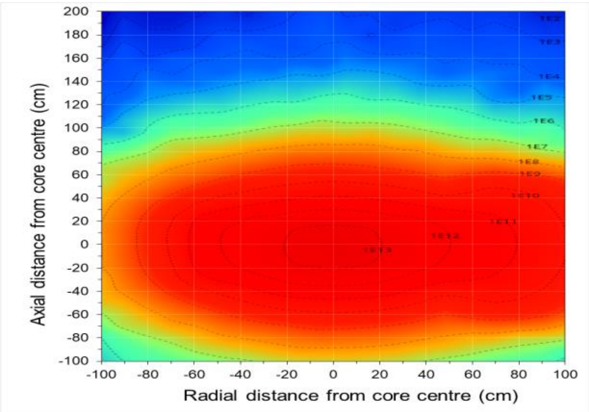

Fig.7 Axial distribution of fast neutron flux contour at full core power of 750 kW.

The fast neutron started to thermalize at a distance of 158 cm from the core. The thermalization is mainly due to water moderation but to a certain extent, it is also due to the graphite at the thermal column. Mitsui and Sugiyama (1973) indicated the decay constant of neutron flux in graphite to be 2341/s and graphite thermalization to occur in 250 µsec to 750 µsec depending on the degree of graphite buckling.

The trend line in Fig. 8 indicates that the thermalization process in this region occurs rapidly, water being an effective moderator, with hydrogen being of similar size to the neutron. The observation shows that the thermal neutron is thermalized at 78 cm. The green region plots the thermal neutron with fluxes between 1 x 10+5 and 1 x 10+9 n.cm^2.s^{-1}. The red region indicates a contour profile for thermal neutrons with fluxes between 1 x 10+9 n.cm^2.s^{-1} and 1 x 10+12 n.cm^2.s^{-1}

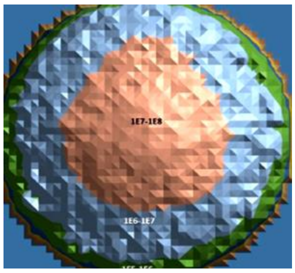

Fig. 8 Radial Distribution of Thermal Neutron Flux at Full Core Power of 750 KW at 1 m from Irradiated Fuel Storage.

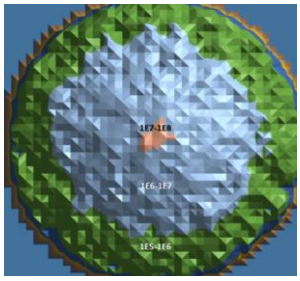

Fig. 9 Radial Distribution of Fast Neutron Flux at Full Core Power of 750 KW at 1 m from Irradiated Fuel Storage.

The red region in Fig. 9 indicates the contour profile of fast neutron's distribution for fluxes between 1 x 10+7 n.cm^2.s^{-1} and 1 x 10+13 n.cm^2.s^{-1}. The gray region in Fig. 9 represents the energy of neutron distribution. The green region maps fluxes between 1 x 10+2 and 1 x 10+9 n.cm+2.s–1.

In this study, the neutron flux distribution has been calculated and the result is very close with agreement that has been discuss in the previous literature data to the thermal neutron flux and epithermal neutron flux for RTP as summarized in Table 2. The works from Liew (2010) and Alnour (2012) reported the fluxes for thermal neutron are 2.26 × 10±0.10 n.cm–2s–1 and 2.33 × 10±0.08 n.cm–2s–1, for the epithermal neutron is 1.11 × 10±11 n.cm–2s–1 and 1.23 × 10±0.02 n.cm–2s–1.

Table 2 Recorded average thermal neutron flux and epithermal neutron flux.

Study	Year	Thermal neutron flux (×10+12n.cm–2s–1)	Epithermal neutron flux (×10+11 n.cm–2s–1)
Liew	2010	2.26 ± 0.10	1.11 ± 0.11
Alnour	2013	2.33 ± 0.08	1.23 ± 0.02
Present Study	2018	2.30 ± 0.09	1.16 ± 0.07

Table 3 Irradiation of a single fuel in the core and in the fuel rack above the core.

Fuel Location	**At 750 kW Core Power Level**			**At 1kW Core Power Level**		
	Fission Rate (#/s)	Energy Release (Watt)	U-235 burned (in g) from 38g initial mass after 1 day	Fission-Rate (#/s)	Energy Release (Watt)	U-235 burned (in g) from 38g initial mass after 1 day
Fuel in-core	3.18 x 10+14	1.02 x 10+4	1.26 x 10–2	4.24 x 10+11	1.36 x 10+0	1.68 x 10–5
Irradiated fuel in storage rack (out-core)	1.51 x 10+7	4.84 x 10–4	6.01 x 10–10	2.02 x 10+4	6.46 x 10–7	8.01 x 10–13

As indicated in Table 4, the simulation shows the fission-rate of the fuel in the core to be 3.18 x 10+14 particle/s compared to 1.51 x 10+7 particle/s in the irradiated fuel in the storage rack. This indicates the fission-rate in the storage rack to be negligible compared to the fission-rate at the core, implying that to any practical extent the neutron does not induce burn up of the irradiated fuel at the storage rack.

This is consistent with the flux distribution results shown in Fig. 4 to Fig. 5. Assuming the irradiated fuel is to be stored in the storage rack for the next 30 years, with continuous 24/7 exposure to a power level of 750 kW, the simulation confirms that the level of the associated burn-up would be negligible, as depicted in Table 3. Only some 7 µg of U-235 is projected to be consumed throughout this span of storage time, representing an insignificant quantity. Based on these results, it is concluded that the thermal neutron flux density at the storage rack is very small and the level of burn-up of these fuels is negligible.

Table 4 Burn-up rate of one fuel element located at storage rack.

Years of RTP operation at full power 750 kW	U-235 burned (g) from 38 g mass	Burn-up [%]
1	2.19 x 10–7	1 x 10–6
2	4.39 x 10–7	1 x 10–6
3	6.58 x 10–7	2 x 10–6
4	8.77 x 10–7	2 x 10–6
5	1.10 x 10–6	3 x 10–6
6	1.32 x 10–6	3 x 10–6
7	1.53 x 10–6	4 x 10–6
8	1.75 x 10–6	5 x 10–6
9	1.97 x 10–6	5 x 10–6
10	2.19 x 10–6	6 x 10–6
20	4.39 x 10–6	1.2 x 10–5
30	6.58 x 10–6	1.7 x 10–5

CONCLUSIONS

MCNPX code was used to simulate fast and thermal neutron flux for the reactor operating at 750 kW, using a 10 cm x 10 cm x 10 cm mesh model. For fast neutron, the axial distribution energy was higher than 1 x 10-6 MeV and traveled more than 158 cm indicating that the fast neutron can travel far distance reaching the area of the fuel rack. The reaction time for the fast neutron was too short to result in burn-up due to short interaction time. Thermal neutron in-core traveled the maximum distance of 78 cm due to thermalization by moderator, thus it can be confirmed that the thermal neutron from the core did not reach the storage rack located 1 m from the core. The axial distribution of thermal neutrons occurred at energy lower than 1 x 10-6 MeV. Above this energy, the neutrons are regarded as epithermal and/or fast neutrons. The fission reaction of the irradiated fuel in the fuel rack is 7 orders of magnitude lower than the fission rate at the core, indicating very small to negligible fission reaction of these fuels and the burn up of stored irradiated fuel can be neglected.

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