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EVALUATION OF NEUTRON SHIELDING PERFORMANCE OF CD-SS 316L AS A CANDIDATE ALLOY FOR DRY CASK OF RESEARCH REACTOR SPENT FUEL

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ABSTRACT

EVALUATION OF NEUTRON SHIELDING PERFORMANCE OF Cd-SS 316L AS A CANDIDATE ALLOY FOR DRY CASK OF RESEARCH REACTOR SPENT FUEL. Development of dry casks is necessary to support national strategy for the management of spent fuels. One of the requirements for the dry cask is shielding performance for neutrons emitted by the spent fuels. The objectives of this study were to determine the emitted neutrons by the spent fuels generated from GAS research reactor and to evaluate the neutron shielding performance of Cd-SS316L alloy as a candidate material to be used for dry cask for the spent fuels. The determination of the emitted neutrons was carried out using Origen 2.1 software while the evaluation of the neutron shielding performance was done using MCNP5. The result shows that the emitted neutrons by a spent fuel after 5 years discharged from GAS research reactor are 2.81×10^{3} and 3.32×10^{6} n/s for reactor core power of 15 and 30 MW, respectively. Addition of Cd improves the neutron shielding performance of SS 316L. It can be concluded that evaluation of neutron shielding performance of Cd-SS316L can be done using Origen 2.1 software for the neutron emission determination and MNCP 5 software for the neutron shielding performance by simulation. This study shows that the Cd-SS 316L alloy is worth further study to develop the dry cask design for the GAS research reactor.

Key words: Neutron shielding, cadmium, stainless steel, spent fuel.

ABSTRAK

EVALUASI KINERJA* SHIELDING NEUTRON *CD-SS 316L SEBAGAI KANDIDAT PADUAN UNTUK* DRY CASK *BAHAN BAKAR NUKLIR BEKAS REAKTOR RISET. *Pengembangan* dry cask *diperlukan untuk mendukung strategi nasional pengelolaan Bahan Bakar Nuklir Bekas (BBNB). Salah satu syarat untuk* dry cask *adalah unjuk kerja shielding bagi neutron yang dipancarkan oleh BBNB yang akan disimpan di* dry cask. *Tujuan dari penelitian ini adalah untuk menentukan emisi neutron BBNB yang dihasilkan dari reaktor riset GAS dan untuk mengevaluasi kinerja pelindung neutron paduan Cd-SS316L sebagai calon material yang akan digunakan dalam wadah sistem kering BBNB. Metode yang dilakukan adalah menggunakan perangkat lunak Origen 2.1, kemudian menggunakan MCNP5. Hasil penelitian menunjukkan bahwa emisi neutron BBNB setelah 5 tahun yang dikeluarkan dari reaktor riset GAS masing-masing sebesar 2,81×103 dan 3,32×106 n/s untuk daya teras reaktor 15 dan 30 MW. Penambahan Cd meningkatkan kinerja perisai neutron SS 316L. Evaluasi unjuk kerja penahan neutron SS 316L dengan penambahan Cd yang merupakan kandidat material dry cask bahan bakar bekas dari reaktor riset GAS dapat dievaluasi menggunakan software Origen 2.1 untuk emisi neutron, sedangkan unjuk kerja penahan neutron dievaluasi dengan simulasi menggunakan software MNCP 5. Studi ini menunjukkan paduan Cd-SS 316L dapat digunakan untuk studi lebih lanjut untuk mengembangkan* desain dry cask *untuk reaktor riset GAS.*

Kata kunci: *Neutron shielding, cadmium, stainless steel, BBNB.*

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INTRODUCTION

The operation of the G. A. Siwabessy (GAS) research reactor generated spent fuels that must be safely managed due to their high radioactivity. These spent fuels were stored in the reactor pool for at least 100 days to decrease their radioactivity and heat levels. Afterwards, the spent fuels were transferred to the storage pool in the Interim Storage Facility for Spent Fuels (ISSF) which is a wet storage type. Prolonged storage of the spent fuels in the storage pool should be avoided due to the risk of corrosion which may affect the integrity of the spent fuels. Further strategy that is planned for the management of this spent fuel is to transfer to dry storage facility prior to dispose of in a geological repository[1]. Due to the unavailability of the dry storage facility in Indonesia, the study to prepare this facility is necessary to support the national radioactive waste program.

The dry storage facility uses dry casks in which the spent fuels are enclosed. These dry casksshall comply many rigorous criteria and standard requirements, among which is shielding performance for neutron emitted by the spent fuels[2]. Stainless steel is widely used for canister of a dry cask for spent fuels because of its good corrosion resistance [3]-[5]. The canister is enclosed in overpack made of concrete or carbon steel for gamma radiation shielding. To improve neutron shielding performance, borated stainless steel has been used as a material for canister[6]. Stainless steel with addition of Gd has been studied as an alternative material because Gd has higher neutron cross section[7]. Another candidate additive for stainless steel is Cd because it also has higher neutron cross section but relatively cheaper compared to Gd.

The objectives of this study are to determine the emitted neutrons by the spent fuel generated from GAS research reactor and to evaluate the neutron shielding performance of Cd-SS316L alloy as a candidate material to be used in dry cask for the spent fuels. The emitted neutrons were calculated using Origen 2.1 software for spent fuels from the research reactor with core power 15 and 30 MW. The neutron shielding performance evaluation was carried out by comparing the neutron transmittance of SS316L and Cd-SS316L which were calculated using MCNP5. This software has been widely used for gamma and neutron shielding evaluation [8]-[12].

METHODOLOGY

a. Calculation of neutron inventory in spent fuel.

Neutron inventory in the spent fuel generated from the GA. Siwabessy research reactor was calculated using Origen 2.1 software[13][14]. Two values of reactor core power were used in the calculation i.e., 15 and 30 MW which represented the existing operation and design values, respectively. The input parameters for the calculation of the spent fuels assembly from the research reactor with those core power values were shown in Table 1. The emitted neutrons from the spent fuels were then calculated as a function of time after being discharged from the reactor using the same software.

Table 1. Input parameter for calculation of the emitted neutrons from the research reactor spent fuels[15].

Parameter	Core power	
	15 MW	30 MW
Power (MW/FE)	3.75×10^{-1}	7.50×10^{-1}
Burn-up (MWD)	7.6×10^{1}	1.54×10^{2}
Flux (N/cm^2.s)	7.18×10^{13}	1.31×10^{15}
Enrichment of ^{235}U (%)	19.75	19.75
Weight of ^{235}U per FE (g)	250	250

b. Neutron shielding performance of Cd-SS 316L.

The sample used for neutron shielding performance evaluation was based on SS 316L, the composition of which is shown in Table 2. This alloy was then modified with the addition of 3, 5, 7.5, and 10 wt.% Cd. The density of those alloys is shown in Table *3*.

Table 2. Composition of SS 316L[16].

\	Composition (%w/w)
C	0.03
Si	0.3
P	0.027
Si	0.30
Cr	17
Mn	1.94
Fe	47.854
Ni	10.30
Mo	22.21
N	0.039
Total	100

Table 3. The material density used for shielding performance simulation in MCNP 5.

Sample	Density (g/cm^3)
SS 316L	8.0000
3 wt.% Cd-SS 316L	8.0195
5 wt.% Cd-SS 316L	8.0325
7.5 wt.% Cd-SS 316L	8.0488
10 wt.% Cd-SS 316L	8.0650

The neutron shielding performance was evaluated by the simulation of neutron transmittance of the sample using MCNP5 program with ENDF/B-VI.8 database. For thermal neutron shielding performance evaluation, a neutron point source with energy of 0.0253 eV was used in the simulation, of which the geometry model is shown in Figure 1. The distance from the neutron source to the incident plane of the sample was 10 cm. Two points detectors, one each was placed at the point of incident and at another end of the sample. The former was to obtain the incident intensity (I_0), while the latter was for the neutron intensity passing through the sample (I). The neutron transmission was then calculated from the ration of I and I_0. The simulation was carried out for each variation of Cd-SS 316L sample with thickness of 0.1, 0.2, 0.3, 0.4, 0.5, and 1.0 cm. The Cd-SS316L is a solid composite material that will be used as a neutron absorber. Additionally, the simulation was also carried out for SS 316L with various Gd contents as reported by other researchers to compare the simulation result[17].

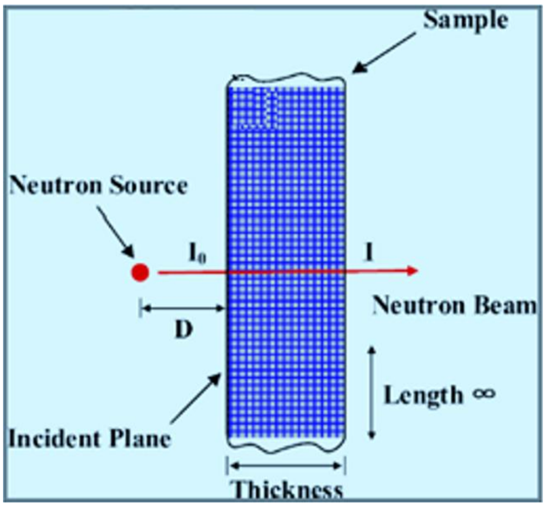

Figure 1. Schematic geometry for simulation of thermal neutron shielding.

For fast neutron shielding evaluation, simulation was carried out with a geometry model as shown in Figure 2. The neutron source used in the simulation was based on^{241}Am^{9}Be with density of 1.3 g/cm^3 and average energy of 4.5 MeV[18]. The source was placed in a paraffin which have inner cylinder with one open end. The open end was closed with a 5 cm thickness of paraffin. Neutron detection was simulated using a rectangular neutron detector with dimensions of 50.8× 19.8 × 140.9 mm. The sample was placed between the paraffin and the detector. The Cd contents in SS 316L samples were the same as the previous simulation with a 0.0253 eV neutron source while the sample thicknesses were 0.3, 0.5, and 1.0 cm.

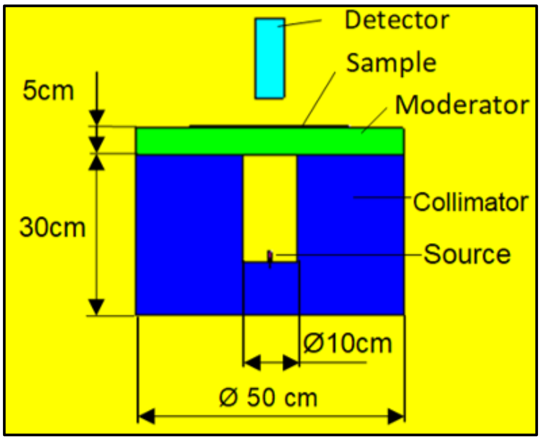

Figure 2. Schematic geometry for simulation of fast neutron shielding.

RESULTS AND DISCUSSION

a. Neutron emission from a spent fuel

Figure 3 shows the total neutron emitted by a spent fuel assembly generated from GAS research reactor with core power of 15 and 30 MW. The emitted neutron mainly originated from spontaneous fission of actinides and (α,n) reaction. For core power of 15 MW, the total neutron emitted by a spent fuel freshly discharged from the research reactor was 8.86 × 10^3 n/s. This value sharply decreased to 2.81× 10^3 n/s after 5 years, then the value became relatively constant as the time increased. On the other hand, the total neutron emitted by a spent fuel freshly discharged from the research reactor with core power of 30 MW was 3.32 × 10^6 n/s which approximately three order magnitude greater than that of the 15 MW. This value decreased to 2.56× 10^6 n/s after 5 years. These results show that shielding evaluation for the calculated emitted neutron is required in designing dry cask for the spent fuel from GAS research reactor.

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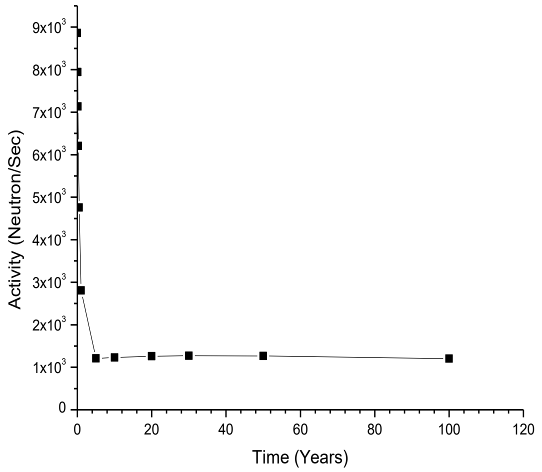
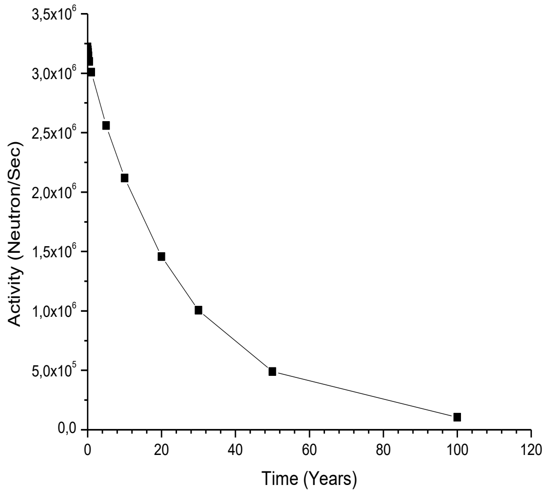


(a)



(b)

Figure 3. Neutron emission from the RSG-GAS spent fuel with core power of (a) 15 and (b) 30 MW.

b. Neutron shielding performance of Cd-SS316L

Figure 4 shows the neutron transmittance of the SS 316L-based alloys for thermal neutron with single energy of 0.0253 eV. For a test case, the resultsof simulation with 7.78wt.% Gd-SS 316L were compared with those have been previously reported by other researchers that carried out the simulation with different software but the same geometry with this study. As can be seen in the figure 4, the result for 7.78wt.% Gd-SS316L was closely fit with another study which prove the validity of the simulation result of this study. The figure 4 also shows that the SS 316L with addition of Cd have significantly better performance to attenuate neutrons from a 0.0253 eV neutron source, albeit to a lesser extent compared to that of with Gd addition. The neutron transmittance of a1 cm thick of SS316L was 3.31×10^{-1}. The neutron transmittance value significantly decreased to 8.30×10^{-4} for the same sample with addition of 7.78wt.% Cd. It shows that the SS 316L with the addition of 7.78 wt.% Cd has better performance in absorbing neutrons than SS 316L. This value corresponds to more than 99.9% of neutron from the source was attenuated by the sample. This result suggests that the Cd-SS 316L alloy can efficiently attenuate thermal neutrons and therefore can be used as an alternative for thermal neutron shielding material used in the dry cask.

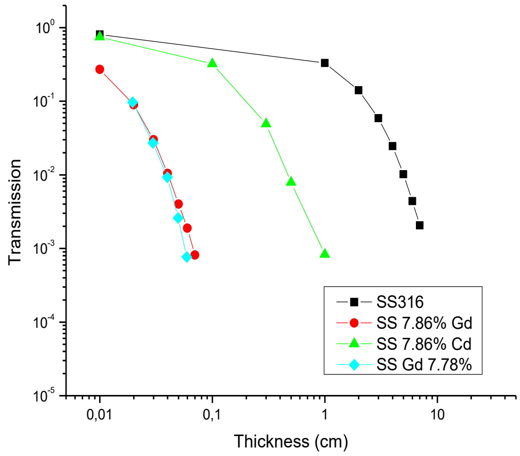


Figure 4. Neutron transmittance of SS316L, Cd-SS316L, and Gd-SS316L for thermal neutron calculated using MCNP 5.

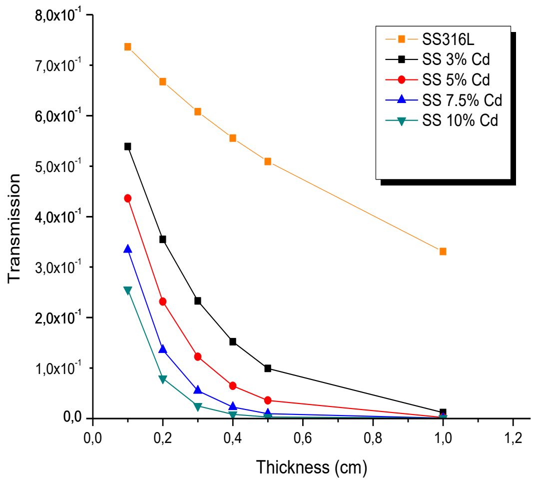


Figure 5. Neutron transmittance of SS316L with addition of various wt.% Cd as at different thicknesses from a 0.0253 eV neutron source.

Figure 5 shows the neutron transmittance of SS 316L with various Cd contents and thicknesses. At the same thickness, the neutron transmission values continuously decreased as the content of Cd in the SS 316L sample increased from 0 to 10 wt.%. For example, the transmission of SS316L with addition of 0, 3, 5, and 10 wt.% Cd at 0.5 cm thickness were5.09×10^{-1}, 9.93×10^{-2}, 3.58×10^{-2}, 9.5×10^{-3}, and 3.07×10^{-3}, respectively. This result further corroborates the fact that addition of Cd significantly improves the neutron shielding performance of SS 316L as mentioned previously.

Figure 6 shows neutron transmission from a 4.5 MeV neutron source moderated with 5 cm thickness paraffin passing through the SS 316L sample with various contents of Cd and thicknesses..

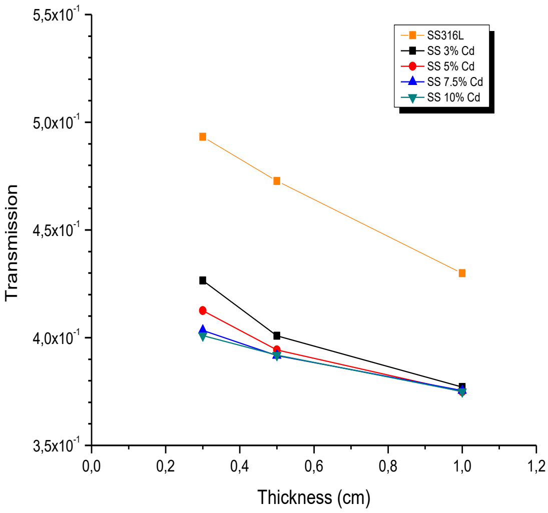


Figure 6. Neutron transmittance of SS316 with addition of various wt.% Cd at different thicknesses from a 4.5 MeV neutron source moderated with 5 cm thickness paraffin.

As can be seen from the content figure 6, addition of Cd only slightly improved the neutron shielding performance of SS 316L. At 0.5 cm thickness, for example, the neutron transmission for SS 316L was 4.73×10^{-1}. The value decreased to 3.79×10^{-1} with the addition of 3 wt.% of Cd. As the Cd contents increased to 10%, the values were approximately constant which were 3.71×10^{-1},3.68×10^{-1}, and $3.68\times10^{-1}$for 5, 7.5, and 10wt.% Cd-SS 316L, respectively. These values correspond to about 63% of the neutrons were attenuated by the sample which were smaller than those for a 0.0253 eV neutron source previously discussed. These results suggest that the Cd-SS 316L has less shielding performance for high neutron energy compared to that of thermal energy. This result is consistent to the properties of Cd which has higher neutrons cross section for thermal neutrons than that of fast neutrons. Further study such as additional material for slowing down the high energy neutrons will be carried out to improve the performance of Cd-SS 316L to attenuate this type of neutrons

CONCLUSIONS

Evaluation of neutron shielding performance of SS 316L with addition of Cd which is the candidate material for dry cask of the spent fuels from the GAS research reactor can be evaluated using origen 2.1 software for neutron emission, while the neutron shielding performance was evaluated by the simulation using MNCP 5 software. The result shows that the emitted neutrons by spent fuel after 5 years discharged from GAS research reactor were $2.81\times10^{3}$and 3.32×10^{6}n/s for reactor core power of 15 and 30 MW, respectively. The neutron transmission of a 0.0253 eV neutron source passing through a 0.5 cm thickness SS 316L was 55% and decreased to 3.58% with the addition of 5 wt.% Cd. On the other hand, the neutron transmission of a 4.5 MeV neutron source with 5 cm thickness paraffin as moderator passing through SS 316L and 5 wt.% Cd-SS 316L were 47.3 and 37.1%. Addition of Cd to SS 316L improves the neutron shielding performance and the alloy can be used for further study to develop the dry cask design for the GAS research reactor.

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