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EFFECTS OF THE LOCATION OF A MISLOADED FUEL ASSEMBLY ON THE NEUTRON MULTIPLICATION FACTOR OF CASTOR X/28 F SPENT FUEL CASK.

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ABSTRACT

Nuclear fuel assembly misload incidents represent a serious nuclear safety concern from a nuclear criticality safety point of view, since they can potentially result in reactivity induced accidents (RIA) with severe consequences. Misloaded fuel assemblies are due to inadequate spent fuel management, such as poor record keeping of important data of the spent fuel such as the exact location of each fuel assembly in the spent fuel pool, the cooling period and burnup. These data are essential for deciding whether fuel assemblies will be acceptable for cask loading or not. The cask under study is CASTOR X/28 F which is designed for fuel assemblies with a 10-year cooling period and the average enrichment of 3.25 wt%. This paper examines the effect of enrichment, burnup, and location, of misplaced fuel assemblies on neutron multiplication factor.

Keywords

Misload, Single fuel assembly misload, Multiple fuel assembly misload, Misload of fresh fuel, End-effect, shielding, Castor X/28F,Burnup Credit

1.0 INTRODUCTION

It is important that spent fuel assemblies that are stored in casks, are loaded in casks that are designed for that particular burnup/enrichment range defined by the loading curve pertaining to that cask. Moreover, in the case of the spent fuel pool, it must be located in a specific location of a spent fuel pool region characterised by the burnup range which is dictated by the design and operational requirements of the spent fuel pool. Therefore if proper loading procedures and nuclear criticality safety requirements are not implemented and spent fuel assemblies are misloaded, it can result in reactivity induced accidents with undesirable consequences, the severity of which will depend on the enrichment level, the burnup history of the misloaded fuel assembly, the burnup history of the surrounding fuel assemblies, and the concentration of neutron absorber if used [1,2,3].The transfer of a spent fuel assembly from the spent fuel pool into the cask must follow a precise protocol in order to prevent: (a) selecting the incorrect fuel assembly from the spent fuel pool; (b) loading the fuel assembly into an incorrect cask (one not designed for that particular enrichment/burnup range); and (c) loading the fuel assembly into a correct cask, but in an incorrect location within the cask. Over many years of operation of nuclear power reactors, it has been observed that the majority of fuel assembly misload incidents are caused by [1,2]:

- Poor spent fuel management, which often manifests itself in poor record keeping of data, which is important to nuclear criticality safety. Record keeping information include data such as initial enrichment, burnup, date off-loaded from the core, duration of cooling in the spent fuel pool, and the exact location of each fuel assembly in the spent fuel pool.
- Inadequate fuel transfer procedures which make no provision for independent review to confirm whether the correct fuel assembly was selected and whether the fuel assembly was loaded in the correct cask and in the correct location within the cask.
- Human error – where; (a) the person responsible for reading the transfer procedure reads an incorrect section of the procedure, and (b) the crane operator picks up an incorrect fuel assembly.

Thus, in order to prevent these incidents, it is vital that adequate spent fuel management procedures are developed and adhered to at all time.

2.0 METHODOLOGY

In a nuclear power station that has been operating for more than 20 years, there could be a significant number of fuel assemblies at different burnup and enrichment levels in the spent fuel pool that need to be transferred into casks to create storage space for new fuel assemblies. In the absence of proper record keeping of the data of fuel assemblies in the spent fuel pool, there is a strong likelihood that misloading of a fuel assembly will occur. In a case where there are a number of fuel assemblies at various enrichment and burnup levels that are transferred into casks, a number of misloading scenarios may occur. These include [1,4]:

- Scenario 1: Misload of underburned fuel, where the degree of underburn can range from 1% to about 90%.
- Scenario 2: Misloading involving fresh fuel with ^{235}U enrichment levels varying from 2 to 5 wt%.
- Scenario 3: The effect of misload involving multiple fuel assemblies in one or both of the above two scenarios.

This paper will focus on exploring scenarios 2 and 3 to determine the effect of misloading *fresh fuel* with an initial enrichment of 2.4% in different locations of a cask that contains spent fuel with the same initial enrichment. The enrichment of the misloaded fuel assembly (i.e., fresh fuel) is then gradually increased up to 5%. In the first instance, a single misload case (i.e., one fuel assembly is misloaded) will be evaluated, while in the second instance multiple misloaded fuel assemblies will be evaluated. Furthermore, the effect of changing the location and enrichment of misloaded fuel assemblies on the neutron multiplication of the system will be analysed. The analysis was performed using KENO VI, a nuclear criticality analysis module of SCALE 6.1.3 [5].

2.1 SINGLE MISLOADED FUEL ASSEMBLY

In modelling a single misloaded fuel assembly scenario, the study entails calculating and trending the k_{eff} of a system that results from a single misloaded fuel assembly at three different locations in the cask, where the X-Y co-ordinates in the cask are indicated in Table 1 and shown schematically in Fig. 1. At each location the enrichment of the fresh fuel was increased from 2.4 wt% to 5wt% and the resulting trend of the k_{eff} analysed.

Table 1: X-Y Co-ordinates of three cases of a single misload

Case #	X	Y	Comment
1	16.25	16.25	Inner Source
2	46.95	16.25	Inner Source
3	94.90	0.0	Outer Source

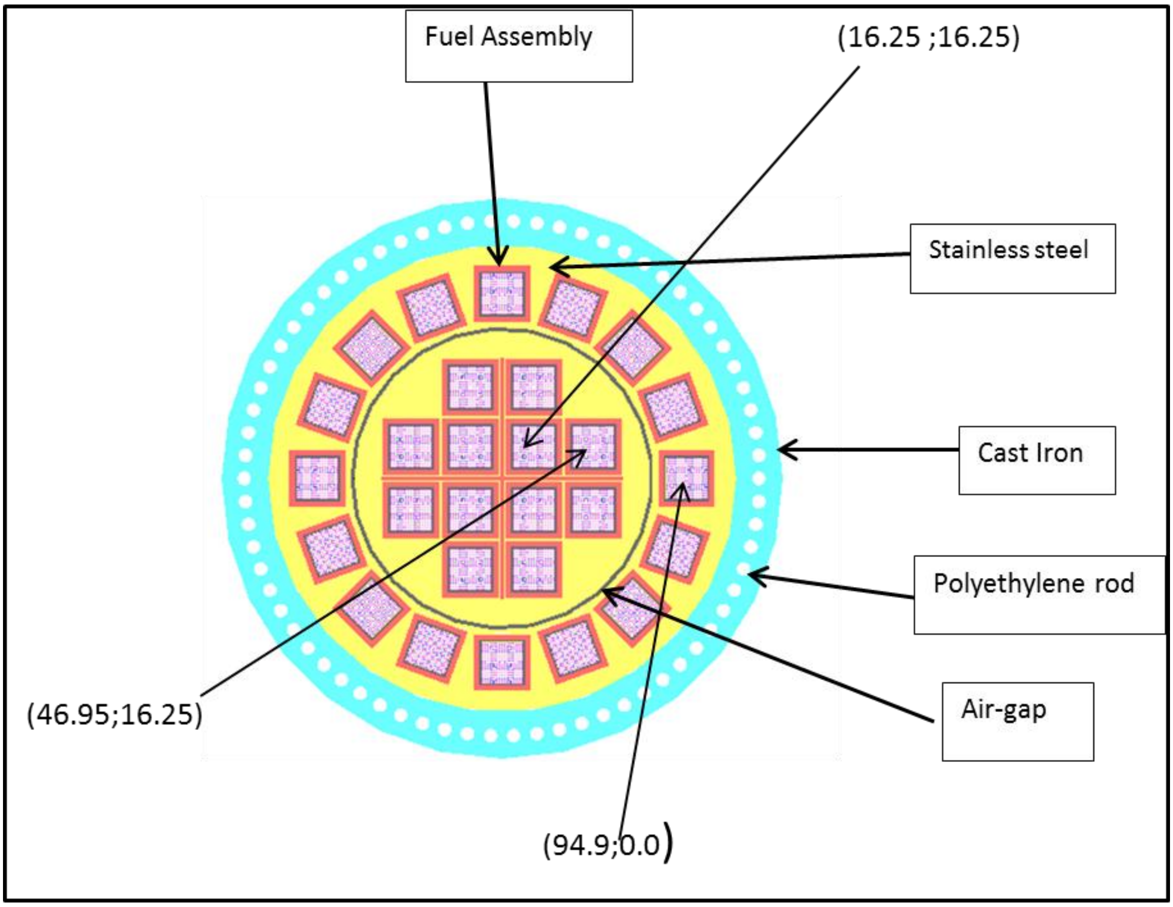


Figure 1: Cross-section of the cask: single-misloaded fuel assemblies**.**

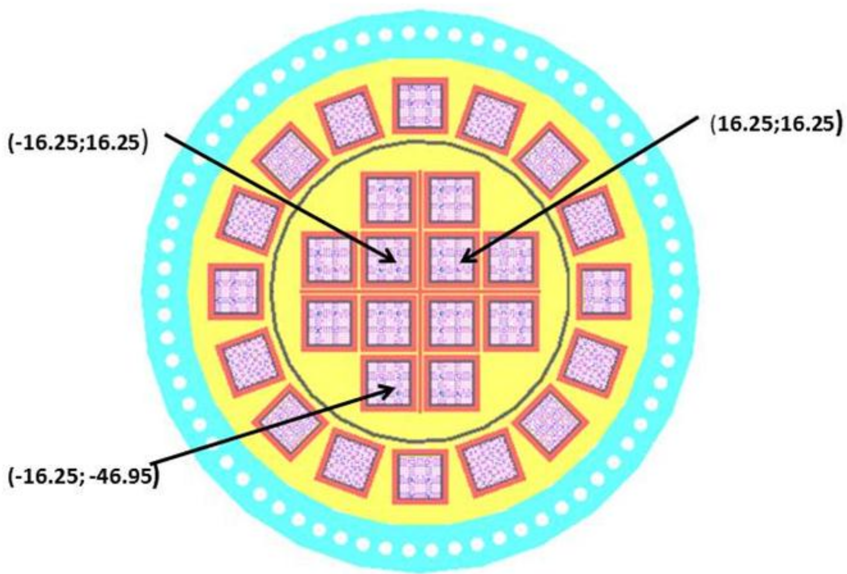
2.2 MULTIPLE MISLOADED FUEL ASSEMBLIES

In the case of modelling multiple misloaded fuel assemblies, the process started with a single misload located at (16.25; 16.25), then a second misload was added at (-16.25; 16.25) and finally a third misload at (-16.25; -46.95). The X-Y co-ordinates of the three misloaded fuel assemblies are shown in Table 2 and the diagrammatic representation of their locations within the cask are shown in Fig. 2. The misloaded fuel assembly cases were evaluated separately and in each case the enrichment of the fresh fuel was increased and the corresponding k_{eff} was recorded.

Table 2: X-Y Co-ordinates of three misloaded fuel assemblies

Location #	X	Y
1	16.25	16.25
2	-16.26	16.25
3	-16.25	-46.95

Figure 2: Cross-section of the cask: multiple misloaded fuel assemblies



3.0 RESULTS

In the case of a single misload fuel assembly, the results as shown in Fig. 3 indicate that if the misloaded fuel assembly is near the centre of the cask such as (16.25; 16.25) (refer to Fig. 1), it will result in the highest k_{eff} for enrichments higher than 2.5 wt%. For enrichments below that, the fuel assemblies farther away from the centre exhibits the highest k_{eff}. The increase in k_{eff} when the misloaded fuel assemblies are near the centre of the cask is as a result of a high material density of the fuel from adjacent fuel assemblies surrounding the misload, which increases the fission density.

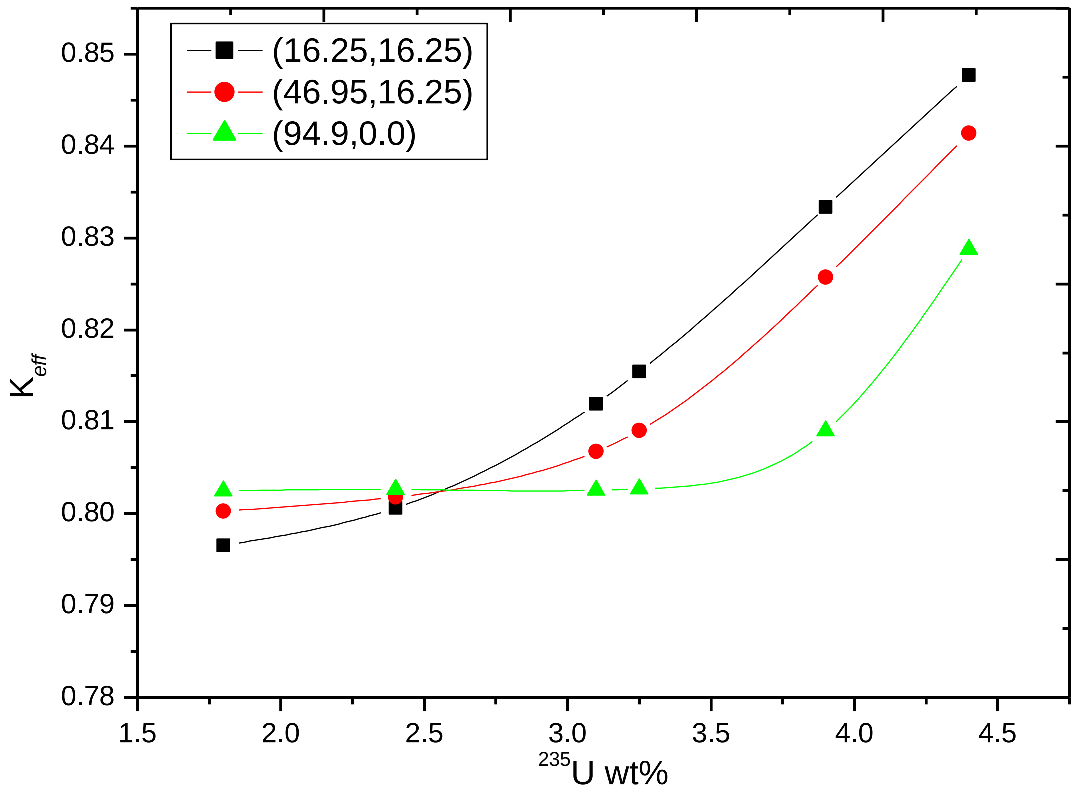
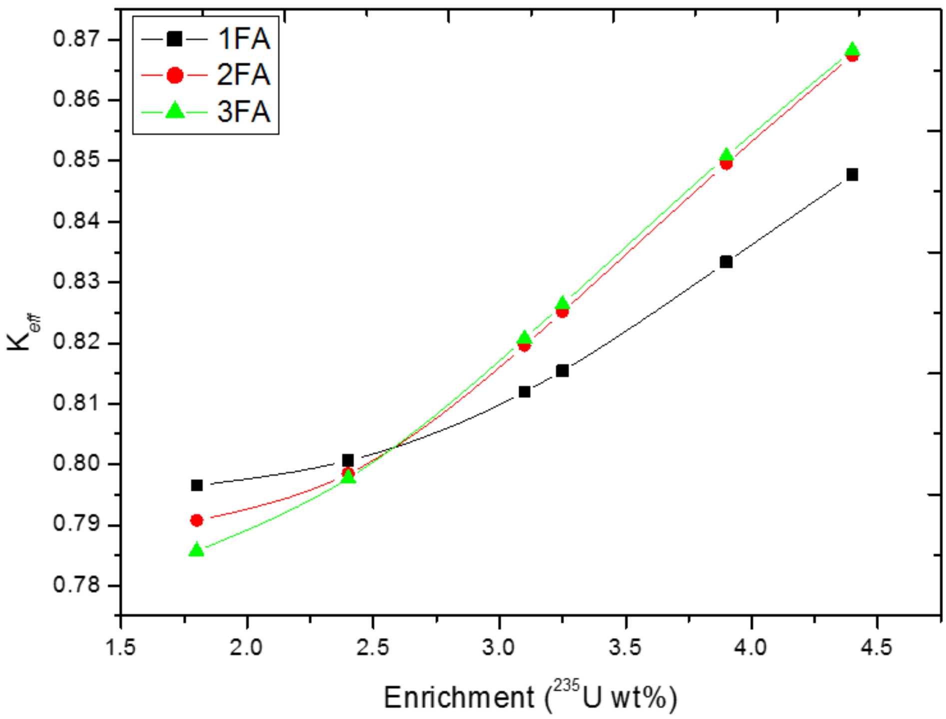


Figure 3: Effect of the location of a single misloaded fuel assembly on the k_{eff} of the system

Fuel assemblies with lower fuel material density around them such as those at (46.95; 16.25) and (94.9; 0.0) show much lower k_{eff} values than those with higher fuel material density, indicating that there is a correlation between the number of fuel assemblies around the misloaded fuel assembly and the k_{eff}. In addition to being affected by the reduction in fuel material density, the k_{eff} of the fuel assemblies in the periphery, such as that at (94.9; 0.0), are also affected by the distance from their nearest fissile neighbours and also their orientation with respect to one another. The effect of material density, distance from the source and other applicable factors on the k_{eff} of the system will be discussed in detail in section 4. Because fuel assemblies at locations (16.25; 16.25) and (46.95; 16.25) are closer to each other, compared to (46.95; 16.25) and (94.9; 0.0), it implies that neutrons originating from (46.95; 16.25) have a much greater chance of reaching (16.25; 16.25) and causing fission than they do with (94.9; 0.0). The neutron multiplication factor is further reduced by the fact that there is a much thicker stainless steel shielding material between (46.95; 16.25) and (94.9; 0.0) than between (46.95; 16.25) and (16.25; 16.25). Thus, by virtue of the material density of the stainless steel shield between (46.95; 16.25) and (94.9; 0.0) and the separation gap between the two, neutrons travelling from (46.95; 16.25) to (94.9; 0.0) have a much greater chance of being captured than would be the case with neutrons travelling from the same source to (16.25; 16.25); hence a low k_{eff}. Lastly, the angular orientation of adjacent fuel assemblies with respect to the misloaded fuel assembly plays a significant role in determining the fission rate from the misloaded fuel assembly. Given that the angular orientation of the fuel assemblies in the periphery varies from -45° to 45° relative to the centre of the cask, it is expected that they would form a reflector with a focal point towards the centre of the cask (refer to Fig. 1) [3]. If however, the angle between the adjacent assembly and the misload is such that neutrons are scattered away from either assembly, they will have very little or no effect on the fission rate and hence lower k_{eff} values will result. This is the case with all fuel assemblies on the periphery, e.g. the two fuel assemblies on either side of (94.9; 0.0). As a result of the combination of these factors, the misloaded fuel assembly at (94.9; 0.0) will result in the lowest k_{eff} of all the misloaded fuel assemblies in the cask, as illustrated in Fig. 3.

In the case of misloading multiple fuel assemblies, Fig. 4 demonstrated that, the k_{eff} of the misloaded cask will increase with an increase in the number of misloaded fuel assemblies. This is because the misloaded assembly is a fresh fuel assembly with uniform enrichment throughout the length of the fuel assembly and fission products with parasitic absorption are not taken into account in the calculation. The other factor is that the misloaded fuel assembly has a minimum enrichment of 2.4 wt% and the enrichment level is increased gradually up to 5 wt%. This will therefore increase the amount of ^{235}U atom density in the cask and result in an increase in the fission rate and consequently in the increase in k_{eff}. In addition to the increase in the *number* of misloaded fuel assemblies, the *location* of the misloaded fuel assembly in the cask plays a significant role in the value of the k_{eff} as shown in Fig. 3 and Fig.



4.

Figure 4: Multiple misload where two misloaded fuel assemblies are in the centre

In the case of two misloaded fuel assemblies, the following is noted:

1. When the two misloaded fuel assemblies are both near the centre of the cask, such as (16.25,16.25) and (-16.25,16.25), they will yield a higher k_{eff} than the same fuel assemblies located in the periphery such as (94.90,0.00) or any of its immediate neighbours.

2. When all misloaded fuel assemblies are located in the periphery, they will have the lowest k_{eff} of all combinations.

3. When one of the two misloaded fuel assemblies is located near centre, for example (16.25; 16.25) and the other one is farther away from the centre at say (-16.25;-46.95), the k_{eff} of such an arrangement will be lower than that of the system where both fuel assemblies are in the centre, but higher than a combination where both are near the periphery.

4.0 DISCUSSION

Stainless steel is composed of the following iron isotopes; ^{54}Fe, ^{56}Fe, ^{57}Fe and ^{58}Fe. To elucidate an understanding of the relationship between enrichment, location and k_{eff}, as presented in the figures above, an evaluation of the respective neutron absorption and scattering cross-sections of the isotopes (as listed in Table 3 [5, 8]) is required.

Table 3: Cross-section of Iron isotopes [5, 8]

Isotope	Capture σ (b)			Capture RI (b)			Elastic σ (b)		
	σ^{th} (b)	$\Delta\sigma^{th}$ (b)	$\Delta\sigma^{th}$ (%)	RI^c (b)	ΔRI^c (b)	ΔRI^c (%)	σ^{el} (b)	$\Delta\sigma^{el}$ (b)	$\Delta\sigma^{el}$ (%)
^{54}Fe	2.25	0.18	8	1.27	0.1	7.874	2.17	0.1	4.6083
^{56}Fe	2.59	0.14	5.4054	1.36	0.15	11.0294	12.62	0.49	3.8827
^{57}Fe	2.48	0.3	12.0968	1.51	0.15	9.9338	0.68	0.06	8.8235
^{58}Fe	1.32	0.03	2.2727	1.5	0.07	4.6667			

These have to be looked in context as their magnitude becomes even more meaningful if they are part of the sensitivity coefficient of that nuclide than if used in isolation [6,7]. Table 4 lists selected nuclear reactions that are likely to take place when the neutron interacts with the nuclide.

Table 4: Sensitivity of Iron Isotopes for Fresh fuel at enrichment of 3.9 wt%

	S							
Isotope	Total	Std Dev	scatter	Std Dev	Elastic scattering	Std Dev	capture	Std Dev
^{54}Fe	-2.11E-03	1.16E-06	-1.45E-03	1.11E-06	-1.40E-03	1.11E-06	-6.61E-04	1.66E-07
^{56}Fe	-1.25E-02	6.78E-06	-8.46E-03	6.55E-06	-6.76E-03	6.40E-06	-4.03E-03	8.23E-07
^{57}Fe	-1.64E-03	9.17E-07	-1.47E-03	8.99E-07	-3.76E-04	4.70E-07	-1.75E-04	4.99E-08
^{58}Fe	-8.01E-05	5.31E-08	-5.72E-05	5.14E-08	-5.19E-05	5.12E-08	-2.29E-05	9.60E-09

It must be noted that the magnitude of the nuclide cross-section is already included in the calculation of the sensitivity of a cross-section. Therefore, the effect of the cross-section on the nuclide-reaction pairs can only be used on its own if the sensitivity of that nuclide-reaction pair is unknown. If the sensitivity coefficient is known, it must take preference over the cross-section. To that effect, the sensitivity coefficient of various nuclide-reaction pairs of iron isotopes that make up the stainless steel are as listed in Table 4 to aid in the analysis of the graphs.

It is important to note that when ranking or prioritising nuclear reactions to determine which one will be a dominant one in a system where there is more than one reaction occurring, that the determining factor is the absolute value |S| of the total (*integrated*) sensitivity coefficient of the nuclide-reaction pair [8]. This is an important consideration in the analysis of the trends of graphs in Fig. 3 and Fig. 4, and is also applicable in predicting the degree to which the k_{eff} will change when the input parameter is perturbed by a small fraction.

When interpreting the sensitivity coefficient, it is important that they are interpreted correctly as incorrect interpretation might lead to incorrect application in the analysis of the graphs. With that in mind it, the sign of the sensitivity coefficient indicates the direction of the effect on the k_{eff}; a negative sensitivity coefficient means that increasing the cross-section will result in the decrease in k_{eff}, while the positive sensitivity coefficient means that increasing the cross-section will result in the increase in the k_{eff} [8]. Although there are many nuclide-reaction pairs that take place when a neutron interacts with matter, when their respective sensitivity coefficients are applied individually they may not make a significant contribution to the neutron multiplication factor of the system [7]. To make sense of the trends of the graphs it is the total sensitivity coefficient of the nuclide-reaction pair that needs to be applied. Some of the most common reactions observed in a fissile system include: *inelastic scattering*, *elastic scattering*, (n,n'), (n,2n), capture, (n,γ), (n,p), (n,d), (n,t) and (n,α).

Applying the above, an analysis of Table 4, shows that ^{56}Fe has the highest total sensitivity coefficient, (|S|= $1.25\text{X}10^{-2}$), which implies that ^{56}Fe will make the largest contribution in the neutron multiplication factor of the system. Furthermore, given that its sensitivity coefficient is negative, it implies that an increase in the cross-section of ^{56}Fe will result in the decrease in the k_{eff} of the system.

Also, due to the angular orientation of the outer source on the periphery and the thickness of the shield between the inner and outer source, whenever neutrons escape from inner sources to the outer source, they will be scattered back into the inner source by ^{56}Fe.

It has been found that the following parameters have a profound effect on the reaction rate and consequently on the k_{eff} of the system [9]; density and thickness of the shielding material, cross-section of the nuclide with which neutrons will react, and the amount of fissile material available for interaction with neutrons. The relationship of these parameters can be described by the following mathematical equation [9]:

$$\boldsymbol{I_{\mathbf{x}} = I_0 e^{-\mathbf{N}\sigma\mathbf{x}}}, \qquad \text{Eqn 1}$$

where,

I_0 = the initial intensity of the neutron beam with no shielding,

I_x = the intensity of the beam after traversing x cm of the shielding material,

N = the number of nuclei of the material (nuclei/cm^3), and

σ = the microscopic cross-section (cm^2/nucleus).

Given that,

$$\sum = N\sigma \qquad \text{Eqn 2}$$

where Σ (cm^{-1}) is a macroscopic cross-section,

by substituting Eqn 2 into Eqn 1, Eqn 1 can be rewritten as [9]:

$$I_x = I_0 e^{-\Sigma x}. \qquad \text{Eqn 3}$$

Furthermore, because

$$\mathrm{N} = \rho \mathrm{N_0}/\mathrm{A} \qquad \text{Eqn 4}$$

where,

ρ = density (g/cm^3) of the shielding material,

N_0 = Avogadro's number (6.023×10^{23} mol^{-1}), and

A =atomic mass (g/mol) of the material,

By substituting Eqn 4 into Eqn 2, Eqn 2 can be written as,

$$\Sigma = \rho \frac{\mathrm{N_0}}{\mathrm{A}} \sigma \qquad \text{Eqn 5}$$

I we substitute Eqn 5 into Eqn 3, and re-write Eqn 1, it can be re-written as:

$$I_x = I_0 e^{-\rho \frac{\mathrm{N_0}}{\mathrm{A}} \sigma \mathrm{x}}. \qquad \text{Eqn 6}$$

Because neutrons have to travel through thicker and denser shielding material to cause fission on the fuel assemblies on the periphery (which are also at an angular position), it results in a slower fission rate (ranging between 2 wt% and 3.5 wt%) which translates to a decreased k_{eff} as shown Fig. 3.

Factors that cause the k_{eff} below 2.5 wt% swopping around is the end-effect in the fuel assemblies [2, 3, 4]. Since the initial enrichment of spent fuel is 2.4 wt%, it is expected that the top and bottom of the fuel assemblies will stay at approximately the same levels of enrichment because of low neutron flux in that region, while the centre will be depleted because of the build-up of fission products and actinides with parasitic absorption. In the fresh fuel assembly, on the other hand, the enrichment is higher in the centre than at the top and bottom ends because of neutron leakage. This is consistent with the findings of Wagner that "*the highest k_{eff} occurs at the location where neutron production is maximized and neutron loss in minimized*" [1,2,4]. In the fresh fuel this occurs near the centre while in spent fuel this occurs near the top and bottom of the fuel assembly.

Therefore, when a fresh fuel assembly is misloaded into the cask, it will have a higher neutron density and ^{235}U, which are largely concentrated in the centre of the fuel assembly. Thus, the neutron-rich centre region of the fresh fuel will coincide with the depleted centre region of the spent fuel. Similarly, the underburned top and bottom region of the spent fuel will coincide with the reactive region of the fresh fuel. As the enrichment of fresh fuel is increased, the combined (fresh and spent fuel) fuel material density at the top and bottom ends of the fuel assembly will experience a much larger increase due to the end-effect. Hence, the largest fission rate will occur at the top and bottom end of both fuel assemblies due to end-effect. If the number of fuel assemblies is increased, the concentration of ^{235}U per unit volume of the cask will increase, which will result in an increase in reaction rate, especially near the ends of the fuel assemblies.

5.0 CONCLUSION

It has been shown that that multiple misloaded fuel assemblies result in a higher k_{eff} than in the case of a single misload and that misloaded fuel assemblies located in the centre of the cask result in higher k_{eff} than fuel assemblies misloaded further away from the centre i.e., in the periphery. It can therefore be concluded that the worst case misload scenario resulting in the highest k_{eff} would be the multiple misload involving four fresh fuel assemblies (i.e., 0 burnup) located near the centre of the cask, i.e., at (16.25; 16.25), (-16.25; 16.25), (-16.25; -16.25), (16.25; -16.25). The scenario with the smallest consequence (i.e., with the lowest k_{eff}) would be the misload of a single spent fuel assembly that has undergone 95% burnup and is located at the periphery of the cask, e.g. at (94.9; 0.0), if credit for burnup is assumed for major actinides *and* principal fission products [3,4] .

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