Burn-up credit in criticality safety of PWR spent fuel

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Abstract: The criticality safety calculations were performed for a proposed design of a wet spent fuel storage pool. This pool will be used for the storage of spent fuel discharged from a typical pressurized water reactor (PWR). The mathematical model based on the international validated Codes, WIMS-5 and MCNP-5 were used for calculating the effective multiplication factor, $k_{eff,}$ for the spent fuel stored in the pool. The data library for the multi-group neutron microscopic cross sections was used for the cell calculations. The k_{eff} were calculated for several changes in water density, water level, assembly pitch and burn-up with different initial fuel enrichment and new types and amounts of fixed absorbers. Also, k_{eff} was calculated for the conservative fresh fuel case. The results of the calculations confirmed that the effective multiplication factor for the spent fuel storage is sub-critical for all normal and abnormal states. The future strategy for the burn-up credit recommends increasing the fuel burn-up to a value greater than 60.0 GWD/MTU, which requires new fuel composition and new fuel cladding material with the assessment of the effects of negative reactivity build up.

[Rowayda F. Mahmoud, Mohamed K. Shaat, M. E. Nagy, S. A. Agamy and Adel A. Abdelrahman. **Burn-up credit in criticality safety of PWR spent fuel.** *J Am Sci* 2014;10(2):125-131]. (ISSN: 1545-1003). http://www.jofamericanscience.org. 19

Keyword: Burn-up,credit in,criticality,PWR, spent fuel.

1. Introduction

Un-irradiation of nuclear fuel has a well-specified nuclide composition that provides a straightforward and bounding approach to the criticality safety analysis of transport and storage casks. When the fuel irradiated in the reactor the nuclide composition changes, ignoring the presence of burnable poisons, this composition change will cause the reactivity of the fuel to decrease. Allowance in the criticality safety analysis for the decrease in fuel reactivity resulting from irradiation is typically termed burn-up credit.

The following recommendations made by Nuclear Regulatory Commission “NRC” based on Topical Report “TR” provide a cask-specific basis for granting burn-up credit, based on actinide composition. The NRC’s staff will issue additional guidance and/or recommendations as information is obtained from its research program on burn-up credit and as experience is gained through future licensing activities.

The licensing-basis analysis performed to demonstrate criticality safety should limit the amount of burn-up credit to that available from actinide compositions associated with PWR irradiation of UO_2 fuel to an assembly-average burn-up value of 40 GWD/MTU or less.

The initial enrichment of the fuel assumed for the licensing-basis analysis should be no more than 4.0 wt% ^{235}U unless a loading offset is applied. The loading offset is defined as the minimum amount by which the assigned burn-up loading value must exceed the burn-up value used in the licensing safety basis analysis. The loading offset should be less than 1.0 GWD/MTU for every 0.1 wt% increase in initial enrichment above 4.0 wt%. In any case, the initial enrichment shall not exceed 5.0 wt%. For example, if the applicant performs a safety analysis that demonstrates an appropriate subcritical margin for 4.5 wt% fuel burned to the limit of 40 GWD/MTU, then the loading curve should be developed to ensure that the assigned burn- up loading value is at least 45 GWD/MTU (i.e., a 5 GWD/MTU loading offset resulting from the 0.5 wt% excess enrichment over 4.0 wt%). Applicants requesting use of actinide compositions associated with fuel assemblies, burn-up values, or cooling times outside these specifications. Nuclear criticality safety is concerned with preventing the nuclear criticality accidents and mitigating its consequences [1].

In this work, we designed wet spent fuel storage for a typical pressurized water nuclear power plant (Sizewell-B). The criticality safety of the spent fuel storage was calculated using validated codes for designing criticality safety as WIMS-5 and MCNP-5 Codes. The behavior of the spent nuclear fuel at wet storage using burn-up credit methodology was applied and achieving safety criteria of sub-criticality during the life time of the spent fuel storage.

The variation of fuel pitch, water density, water level and percentage of the spent fuel burn-up on the degree of spent fuel sub-criticality were calculated and discussed.

2. Design description of a typical PWR station size well B

Size well B PWR is a four loop plant and is a development of the US Standardized Nuclear Unit Power Plant System (SNUPPS) design. The fuel is in the form of uranium dioxide pellets stacked in thin walled tubes of zirconium alloy, about 10 mm inside diameter and about 4 m long, referred to as fuel rods.

The fuel assembly comprises a square 17 x 17 array of such fuel rods, guide tubes and a central instrumentation guide sheath, all mounted vertically. Each assembly is held at its ends by top and bottom nozzles and supported at intervals along its length by grids. The reactor core is built up of 193 fuel assemblies. [2,3]

Spent fuel storage pool structures, systems, and components being designed to accomplish the following:

1- Should be sub-criticality in all fuel states.
2- Prevent loss of water from the fuel pool that would lead to water levels that are inadequate for cooling or shielding.
3- Protect the fuel from mechanical damage.
4- Provide the capability to limit potential offsite exposures in the event of a significant release of radioactivity from the fuel or significant leakage of pool coolant.
5- Provide adequate cooling to the spent fuel to remove residual heat.

Unless protective measures are taken, loss of water from a storage pool could cause the spent fuel to overheat, resulting in damage to fuel cladding integrity and, possibly, a release of radioactive materials to the environment. Natural events, such as earthquakes or high winds, could damage the fuel pool either directly or by generating airborne missiles, so mitigation actions must be taken.

The facility should be designed to withstand the obtained occurrences without significant loss of water to protect the spent fuel. Furthermore, adequate water levels above the top of the spent fuel assemblies should always be maintained because the water serves as a radiation shield for personnel. Provisions for maintaining adequate water levels are important not only for fuel cooling, but also for shielding.

Spent fuel storage pool facility should include a system for cooling, as the removal of decay heat from the spent fuel pool is an important safety consideration, providing a forced cooling and circulation system maintains the pool water at acceptable temperatures for spent fuel during all heat load conditions in order to maintain a bulk temperature below 60 °C (140 °F), including full-core offloads during refueling. Administrative controls may be used to ensure that this temperature limit is not exceeded. The pool water temperature during accident conditions should remain below 93 °C (200 °F).

The proposed design features of the spent fuel storage are based on the following assumptions:

- Pool is rectangular in cross section and approximately 40 feet deep
- Spent fuel dimensions can accommodate three reactor cores with additional space for defected assemblies
- Water pool type.
- Walls covered by stainless steel.
- Filled with treated light water.
- Spent fuel racks made of stainless steel plates sandwiched with B_4C/Ag-In-Cd of approximately 1/4-inch thickness to insure a leak tight system and assembly pitch in the range (4.0- 8.0) cm .
- The pool is equipped with necessary safety systems for forced cooling, measurements and control.

The design of the spent fuel racks should consider the ability of the coolant to naturally circulate through these racks. Improper design of the racks could prevent some fuel assemblies from receiving adequate coolant flow under certain conditions, resulting in overheating and possible cladding failures. Adequate coolant circulation ensures protection of the fuel from thermal damage, provided that the fuel remains covered by water.

Table 1.The core data of a Typical PWR "size-well B"

Plant data:	
Net electrical output, MW	1175
Fuel assembly	
Array	17X17
Number of fuel rods	264
Number of guide tubes:	
For absorber	24
For in-core instrumentation	1
Full length (without control spider), mm	4058
Width, mm	214
Rod pitch, mm	12.6
Mass of UO_2, Kg	460
Fuel rod:	
Outside diameter, mm	9.5
Initial internal pressure (He), bar	24.1
Fuel pellet:	
Material	UO_2
Density, g/cc	10.4
Temperature ,°C	811
Coolant temperature ,°C	570
Clad:	
Temperature ,°C	620
Material	Zircaloy-4
Thickness , mm	0.57

3 Mathematical simulation model

A wet storage of spent nuclear fuel has been designed for a typical nuclear power plant (Sizewell B) in order to study burn-up credit criticality safety under normal and abnormal conditions. These scenarios are high enrichment, high burn-up, different absorber type and decrease of water level change of water density and change of assembly pitch. We simulated the problem using WIMS-5 and MCNP-5 neutronic codes to perform criticality safety analysis.
The Basic conditions and assumptions which were applied are:

- Use of Winfrith Improved Multi-group Scheme (WIMS-5 Code) to get precise values of burned fuel concentrations at different burn-up, and using Mont-Carlo technique (MCNP-5) as a verified international neutronic code for modeling and calculation of multiplication factor for the system.
- Assuming the fuel is at its highest enrichment and maximum burn-up or fresh as a conservative case.

3.1 WIMS-5 Input file

The input file for WIMS-5 Code is the lattice cell, shown in Fig 1, different geometries and material compositions to get flux, k-infinity and burn-up using 69 energy groups in which group condensation was performed and homogenization per material or region was carried out.
This code uses 69 neutron energy groups with neutron microscopic cross section values from the data library, ENDF IV, after executing WIMS-5 input we get its output, which includes data about integrated and averaged flux, diffusion coefficient, absorption, scattering matrix, infinite multiplication factor and burn-up calculations [4].

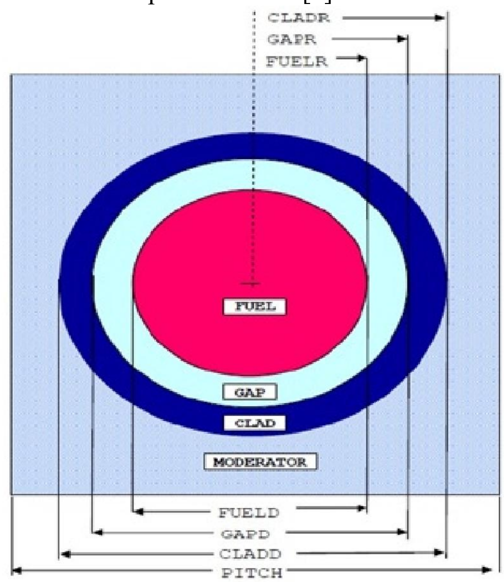

Fig 1. WIMS-5 lattice cell

3.2 MCNP-5 input file

The MCNP-5 input file contains the data of the spent fuel assemblies, racks, geometry of the pool, material compositions and tallies and the output of the file of WIMS-5 code as shown in Fig.2, after executing the input file we get the desired criticality results (i.e. multiplication factor "k_{eff}") [5].

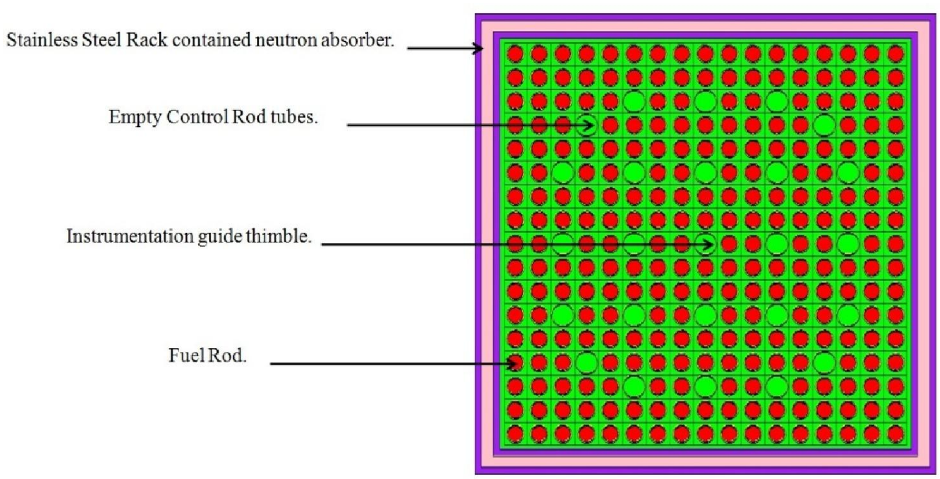

Fig 2. MCNP-5 Spent Fuel Assembly (17x17)

4. Results and discussion

In this section the effective multiplication factor was calculated for different values of assembly pitch, water level, water density and fuel burn-up.

4.1 Change of k_{eff} with change of Lattice Pitch

The following Tables2-5) and Figures(3-6)show that the effective multiplication factor decreases as the pitch increases as a result of the transverse leakage and most neutrons escape to the axial fuel direction.

Table 2. Fresh fuel without absorber, enrichment = 5% and water density= 0.7g/cc.

Assembly Pitch (cm)	k_{eff}
0	1.19111±0.00014
2	1.05981±0.00014
4	0.96535±0.00015
6	0.89782±0.00013
8	0.84680±0.00014

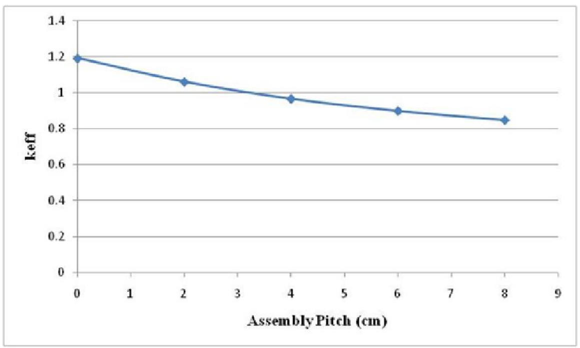

Fig 3, k_{eff} versus Assembly pitch (cm) at no absorber

From Table 2, it can be seen that if the pitch less than 4.0 cm, leads to a super critical condition for the pool (i.e. k_{eff} >1). At 4.0 cm pitch, even though the fuel is fresh and without absorber, the pool is subcritical.

We will see the effects of Boron Carbide (B_4C) or Ag-In-Cd absorbers in the fuel pool on the values of k_{eff} in case of fresh and burned fuel.

Table 3. Fresh fuel with B_4C absorber total volume, enrichment = 5% and water density= 0.7g/cc.

Assembly Pitch(cm)	k_{eff}
0	0.70932±0.00016
2	0.61234±0.00016
4	0.56564±0.00016
6	0.55568±0.00015
8	0.54942±0.00015

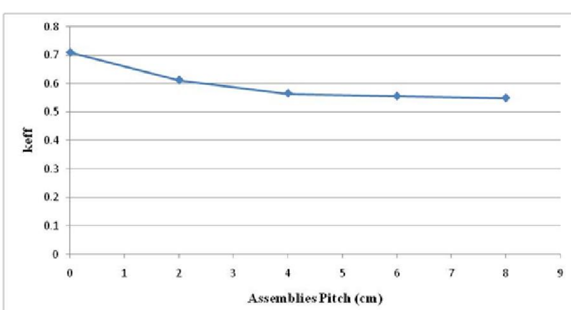

Fig 4. k_{eff} versus Assembly Pitch (cm) with B_4C absorber

Table 4. Fresh fuel with Ag-In-Cd absorber, enrichment = 5% and water density= 0.7g/cc.

Assembly Pitch (cm)	k_{eff}
0	0.91358±0.00041
2	0.79302±0.00046
4	0.71677±0.00046
6	0.66757±0.00046
8	0.65564±0.00057

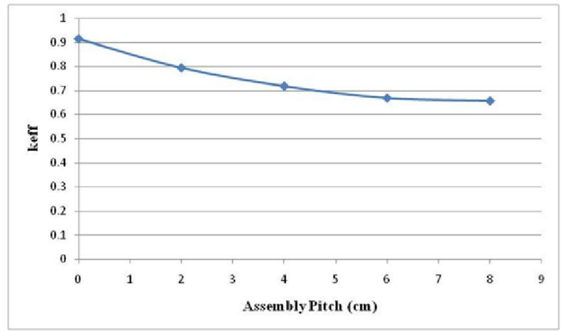

Fig 5. k_{eff} versus Assembly pitch (cm) with Ag-In-Cd absorber.

In Table 4 and Fig.5 using Ag-In-Cd as an absorber instead of boron carbide we get higher values of k_{eff} than of B_4C as a result of the high value of absorption cross section for B_4C.

Table 5. Burnup at 56.5874 GWD/MTU with B_4C absorber total volume, enrichment=5% and water density= 0.7g/cc

Assembly Pitch (cm)	k_{eff}
0	0.70860±0.00018
2	0.63376±0.00015
4	0.56564±0.00016
6	0.55991±0.00017
8	0.54876±0.00016

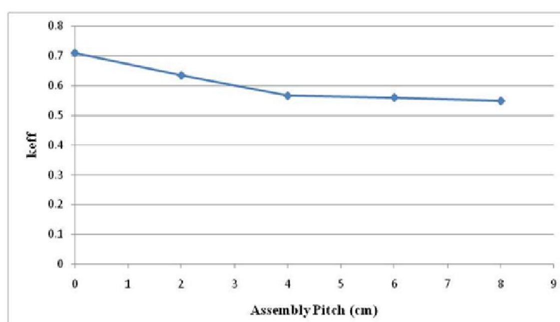

Fig 6. k_{eff} versus Assembly pitch (cm) at BU= 56.5874 GWD/MTU

In Table 5 and Fig.6 with high burn-up value with total volume absorber of Boron Carbide, we notice that the behavior of k_{eff} decreased. However the values are less than that for fresh fuel, due to the buildup of fission products and minor actinides and the depletion of fissionable material.

4.2 Change of k_{eff} versus change of Water level

The effective multiplication factor k_{eff} was calculated for different reduction in water pool level and assuming different fuel burn-up. From Tables (6-8) and Figs. (7-9), the values of the effective multiplication factor k_{eff} decrease as water level decreases due to the reduction of neutron moderation and critical volume decreases.

Table 12 shows that the behavior of k_{eff} is sharply decreased because of the relative depletion of fissile material and build up of fission products and minor actinides. The behavior of criticality changes at high burn-up due to higher decay heat which need specific value of water level to maintain cooling of the spent fuel.

Table 6. Burnup at 21.17215GWD/MTU, with B_4C half volume absorber, enrichment = 5%, water density= 0.7g/cc and assembly pitch= 0.0 cm.

Water Level (cm)	k_{eff}
0	0.15544±0.00026
92.55	0.46555±0.00069
112.55	0.46941±0.00064
142.55	0.47844±0.00074
182.55	0.48099±0.00072
192.55	0.48039±0.00071
202.55	0.47982±0.00069
242.55	0.48254±0.00070
272.55	0.48280±0.00066
292.55	0.48037±0.00069
385.1	0.48766±0.00070

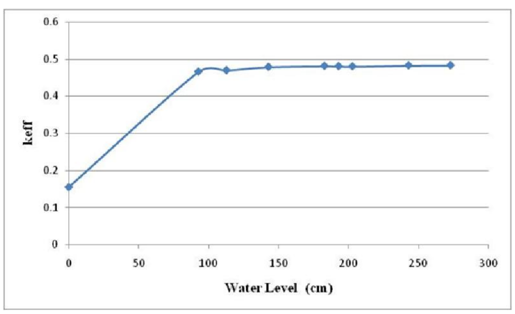

Fig.7 k_{eff} versus of Water level (cm) at BU= 21.17215GWD/MTU, pitch=0.0cm

Table 7. Burn-up at 35.030 GWD/MTU, with B_4C half volume absorber, enrichment = 5%, water density= 0.7g/cc and assembly pitch= 0.0 cm

Water Level (cm)	k_{eff}
0	0.14488±0.00024
92.55	0.41235±0.00064
112.55	0.41776±0.00066
142.55	0.42083±0.00064
182.55	0.42331±0.00060
192.55	0.42453±0.00062
202.55	0.42660±0.00064
242.55	0.42578±0.00062
272.55	0.42761±0.00064
292.55	0.42680±0.00062
385.1	0.42857±0.00057

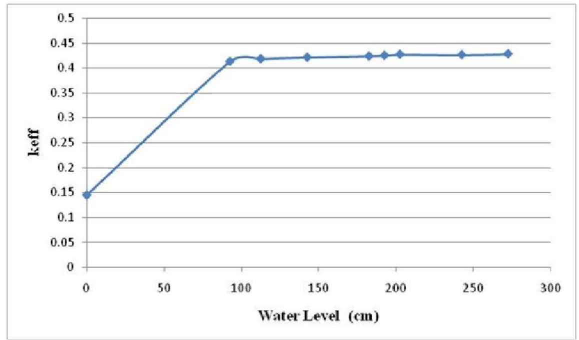

Fig.8. k_{eff} versus Water level (cm) at BU= 35.030 GWD/MTU, pitch=0.0cm

Table 8 Burn- up at 64GWD/MTU with B_4C half volume absorber, enrichment = 5%, water density= 0.7g/cc and assembly pitch= 0.0cm

Water Level (cm)	k_{eff}
0	0.12831±0.00018
92.55	0.12793±0.00018
112.55	0.12736±0.00017
142.55	0.12736±0.00017
182.55	0.12736±0.00019
192.55	0.12695±0.00019
202.55	0.12658±0.00018
242.55	0.12587±0.00017
272.55	0.12587±0.00017
292.55	0.12286±0.00019
385.1	0.10833±0.00013

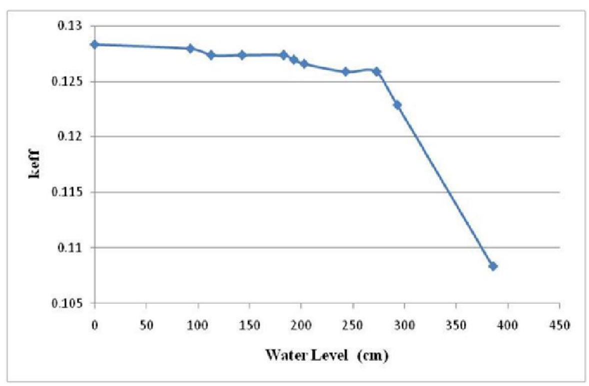

Fig.9. k_{eff} versus Water level (cm) at BU= 64 GWD/MTU

4.3Change of k_{eff} versus change of Build Up Value

It can be noticed from Tables (9-10) and Figs. (10-11) of all different burn up values, at low value of burn up the value of k_{eff} is the highest, and at high value of burn up the value of k_{eff} is the lowest. This decreasing behavior is due to the accumulation of fission products and minor actinides, and also as burn- up increases pool temperature increases and the negative reactivity increases.

Table 9 Burnup with B_4C absorber total volume, enrichment = 5%, water density= 0.7g/cc and Assembly pitch = 0.0 cm

Burnup Level (GWD/MTU)	k_{eff}
21.172	0.60911±0.00016
35	0.53510±0.00014
50.967	0.10951±0.000030

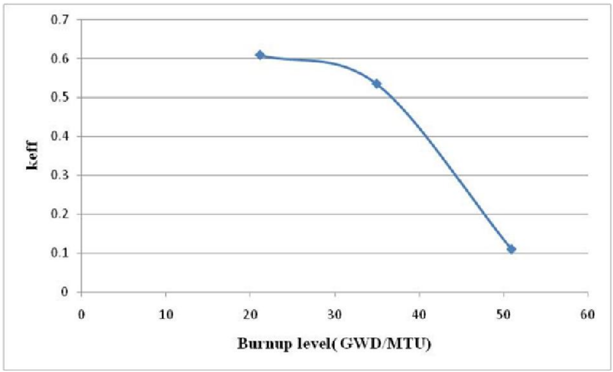

Fig.10 k_{eff} versus Burn-up level when assembly pitch = 0.0 cm

Table 10 Burnup with B_4C absorber, enrichment = 5%, water density= 0.7g/cc and Assembly pitch = 8.0cm

Density (g/cc)	k_{eff}
1	0.09417±0.00002
0.7295	0.11145±0.00003
0.2	0.14336±0.00004
0.01	0.16464±0.00004

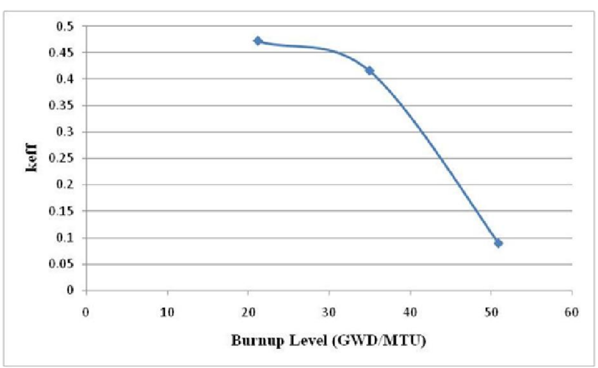

Fig.11. k_{eff} versus Burn-up level when Assembly pitch = 8.0 cm

4.4 Change of k_{eff} with change of water Density

The following Tables (11-12) and Figures (12-13) present the results of water density change with the effective multiplication factor k_{eff}, in which k_{eff} decreases with the decrease of water density due to the decrease of moderation and the increase of negative reactivity.

Table 11 Fresh fuel, B_4C absorber total volume when Assembly pitch = 0.0 cm and enrichment = 5%.

Density (g/cc)	k_{eff}
1	0.90558±0.00017
0.7295	0.75758±0.00017
0.2	0.35522±0.00021
0.01	0.18790±0.00047

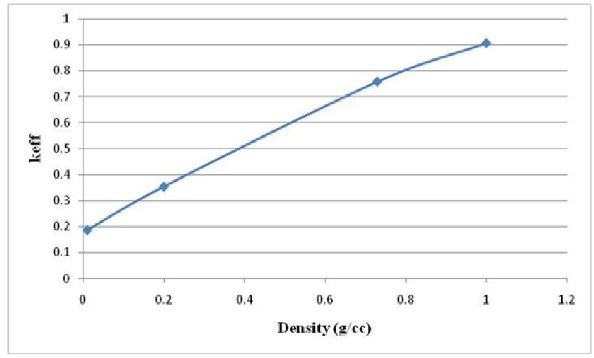

Fig.12. k_{eff} versus Water density, Fresh fuel, B_4C absorber total volume when assembly pitch = 0.0 cm and enrichment = 5%.

Table 12 Burnup at 51.967GWD/MTU at B_4C absorber total volume, enrichment = 5% and assembly pitch = 0.0 cm

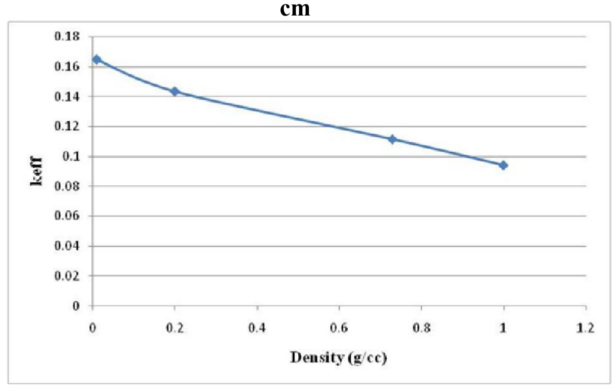

Fig.13. k_{eff} versus Water density, BU=51.967GWD/MTU at B_4C absorber total volume, enrichment = 5% and assembly pitch =0.0 cm

5 Conclusions

- Multiplication factor dependence on lattice pitch is crucial as for fresh fuel was higher than for burned fuel which can be close to each other in range of space by 4.0 cm which is the optimum space.
- Studying the behavior of storing fresh fuel, we found that when racks don't involve any absorber, multiplication factor was high against safety limits and condition, so we need to put an absorber in the racks like B_4C.
- With B_4C absorber, fresh fuel could be stored with assembly pitch 4.0cm.

- For burned fuel with the existence of B_4C its behavior was found to be safer when racks were at pitch in the range (4.0- 8.0) cm. The multiplication factor was low enough to satisfy safety condition and the pool was subcritical.
- Studying water level, for different burn up values, the behavior of k_{eff} increases as water level increases due to the increase in moderation. However for high burn up values, the multiplication factor tends to decrease with the increase of water level and the reason of that is the reduction of fissile material with buildup of fission product and minor actinides.
- Studying the effect of the change of burn- up values on the multiplication factor under certain pitches, we found that the higher value of burn up is the lower the value of k_{eff}. This decreasing behavior is due to the accumulation of fission products and minor actinides as burn up increases.
- Studying water density, for fresh fuel the k_{eff} increases as water density increases, and for burned fuel k_{eff} decreases as water density increases due to the buildup of fission products and high temperature which causes reduction in reactivity as moderation decreases(Doppler Broadening).
- The results of the calculations confirmed that the effective multiplication factor for the spent fuel storage is sub-critical for all normal and abnormal states. The future strategy for the burn-up credit recommends increasing the fuel burn-up to a value greater than 60.0GWD/MTU, which requires new fuel composition, new fuel cladding material and new core structure materials.

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