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Criticality Safety Analysis for Cask Design with High Discharged Fuel Burnup Using MCNPX Code

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ABSTRACT

MCNPX computer code is used to model the general PWR cask which contain 32 typical PWR spent fuel assemblies. For Safe storage and transportation of the cask, factors that affect the criticality were studied using the concept of burn up credit. Several parameters such as initial fuel enrichment , fuel burnup history, cooling time, and axial burnup profile were analysed. The analysis was performed in two different steps , first burn the fuel assembly at different burnup and storage (cooling) conditions , secondly, incorporate the details of the assemblies into the cask (canister) model and perform a criticality calculations for the cask. Several cases of unnormal storage conditions are considered in the case of UO_2- PWR only. In this research high discharged fuel assemblies burn up include Standard UO_2 - PWR and next generation MOX fuel. The present results are compared with similar GBC-32 benchmark and satisfactory agreements were found.

1. INTRODUCTION

Safety of the storage and transportation of the high level radioactive waste is one of the most important tasks in the field of criticality safety. The spent fuel waste is usually stored in spent fuel casks which should remain subcritical during storage and transportation under normal and accident conditions. There are many factors affecting the criticality of the cask such as cask design , fuel type , and burn up credit of the spent fuel. Burn up credit of the fuel describes the reduction in reactivity due to the production of radioactive isotopes during fuel burn up. Burn up credit plays an important role in nuclear fuel cycle criticality safety studies. The decrease of the irradiated fuel cycle reactivity is taken into account in order to optimize safety margins and increase the storage capacity of the caske [1,2,3]. In this research high discharged fuel assemblies burn up include Standard UO_2 - PWR and next generation MOX fuel.

In Burnup credit the criticalty of the cask is calculated based on the concentration of all isotopes (fission product and actinides) at the discharge burnup taking into account the axial discharge burnup. The reactivity margin associated with fixed boron carbide or soluble boron is inherently credited in cask or spent fuel design with burnup credit analyses to compensate uncertainties associated with the utilization of burnup credit [4].

In this research , MCNPX code package, based on Monte Carlo method , is used to model The cask or canister which is loaded with thirty two standard PWR or next generation MOX spent fuel assemblies [5, 6]. The concept of burnup credit is applied and the axial burnup of the assemblies , actinides and fission product are calculated and incorporated in MCNPX model of the cask. The multiplication factor for the cask is determined. Factors that affect multiplication factors such as Fuel enrichment , fuel burn up , cooling time are studied under the concept of burnup credit. Accident conditions in which fuel assembly is placed in an incorrect positions or low burnup assembly are considered in the case of PWR-UO_2 fuel [7,8,9,10,11].

In the following, section 2 describes the computational MCNPX model, section 3 contains the results, discussions, and the conclusion as given in section 4, and the references are given at the end of the paper.

2. COMPUTATIONAL BENCHMARK PROBLEM

The generic cask design, GBC-32 [5,6], or TN-32 [7] could accommodate 32 typical spent PWR fuel assemblies of type 17x17 fuel rods. Dimensions for the GBC-32 cask and fuel assemblies are listed in Table 1. The assembly is equipped with a layer of thickness 0.2565 cm of boron carbide along the active height of

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the assembly to reduce the criticality during storage. Boral panel composed of three different layers. The outer layers is aluminum cladding which form a sandwich with the central layer composed of boron carbide[6].

PWR –UO_2 Fuel Assembly

The fuel assembly is typical PWR of size 17 x 17 fuel rod positions. The assembly contains 20 burnable poisons, 5 water channels and 264 fuel rods as illustrated in Figure 1. The full dimensions and compositions of fuel, clad and burnable poison rods can be found in reference [6]. MCNPX computer code are used to model the burnup of the assembly in a typical operating condition of PWR reactor, the power assigned to the assembly is 17.1 Mw.

The axial fuel rod height are divided into 20 zones to take into consideration the axial fuel burn up in each zone. The concentration of actinides and fission product are calculated at every time step. Fig. 1 shows a typical MCNPX model of the a PWR assembly. In the assembly model The tallies are calculated with 1200000 neutron histories which are divided into 60 cycle with 20,000 neutron each and 10 cycles are skipped. The burnup step are adjusted to be calculated every 10,000 Mwd/T. The depletion calculations were performed using reasonably conservative cycle average operational parameters for fuel temperature (900 K), clad temperature (600 K), moderator temperature (570 K), soluble boron concentration (0 ppm) , and specific power (36 .22 MW/MTU).

MOX Fuel Assembly

A PWR MOX fuel assembly is the same geometrical configuration as a 17×17 type PWR fuel design. The assembly containg 3 types of plutonium isotopes , Low Pu content fuel rods , middle Pu content fuel isotops and high Pu content fuel rods (Figure 2). MOX is the fuel of the next generation reactors. Plutonium total concentrations (wt %) are 7.5 , 14.4 , and 19.1 % resprctively for low, medium and high Pu content and MOX density is 10.4 gm /cm^3. (For more details in MOX fuel compositions reference No. 12)

Cask (canister) Model

Figure 3 shows MCNPX model for GBC-32 canister (cask) . The canister (or cask) contains 32 similar fuel assemblies (It is assumed that all fuel assemblies are similar in composition). Table 1 contains the date for the canister dimensions. The cask is divided axially into the same axial assembly divisions (20 zones) and the axial concentrations of the fuel are incorporated into the corresponding cask zones.

In determining which additional nuclides to include for the estimation of the additional reactivity margin, MCNPX computer codes calculates in this model a total of 100 isotopes (15 actinides and 85 fission product) .

Procedure of the Calculation

Figure 4 shows flowchart and calculation steps of the model. MCNPX [13] code is used to model as a first step an assembly of standard PWR reactor of type 17 x17 fuel rods and or MOX fuel assembly. The burnup card is used to burn the assembly for different burnup and cooling times. The concentrations of Actinides , Fission Product and structure materials are calculated and stored at every burnup and cooling time. The nubmer of isotopes considered are 15 actinides and 85 fission product. These isotopes are incorporated in the canister (or cask) model which contains 32 fuel assemblies. It is assumed that all assemblies (32 assemblies) have the same material compositions. The canister (or cask) model are used to determine the criticality condition and K_{eff} for the cask.

Table (1): Dimension of GBC Cask [6]

Parameter	Dimension (cm)
Cask Inside Diameter	175.0
Cask outside diameter	215.0
Cell inside diameter	22
Cell outside diameter	23.5
Cell wall thickness	0.75
Boral panel thickness	0.2565
Boral central thickness	0.2057
Boral AL. plate thickness	0.0254
Cell Pitch	23.7565
Boral panel width	19.05
Cell height	365.76
Top assembly hardware thickness	30.0
Bottom assembly hardware thickness	15.0
Cask Radial thickness	20.0
Base plate thickness	30.0
Cask Lid thickness	30.0
Active fuel height and boral panel height	365.76
Cask inside height	410.76

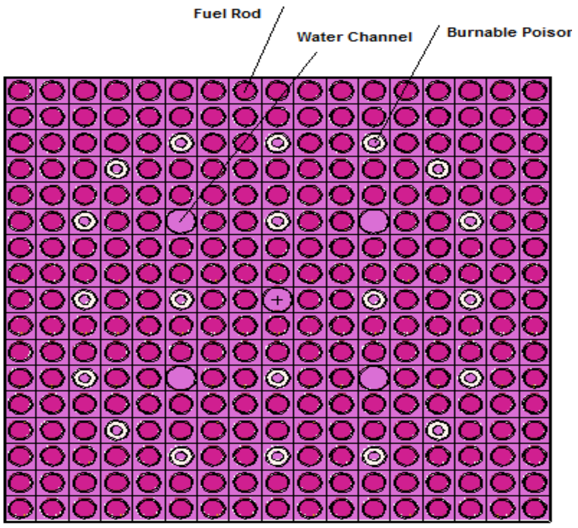

Fig. (1): MCNPX model for PWR- UO2 Assembly Model

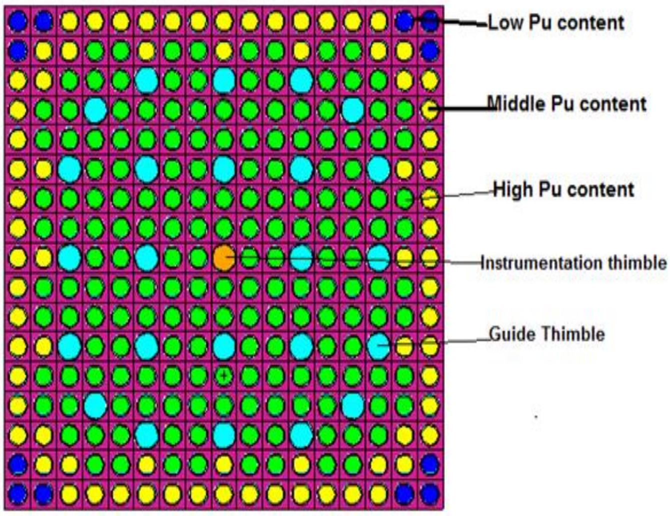

Fig. (2):MCNPX Horizontal Cross section of MOX assembly Model

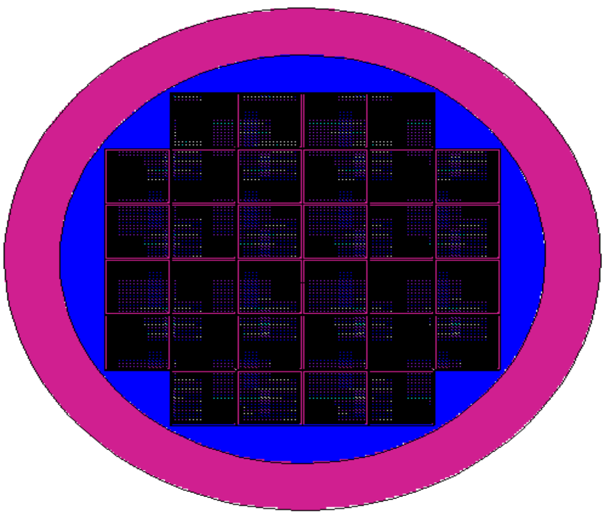

Fig. (3): MCNPX cross sectional view of GBC-32 cask model

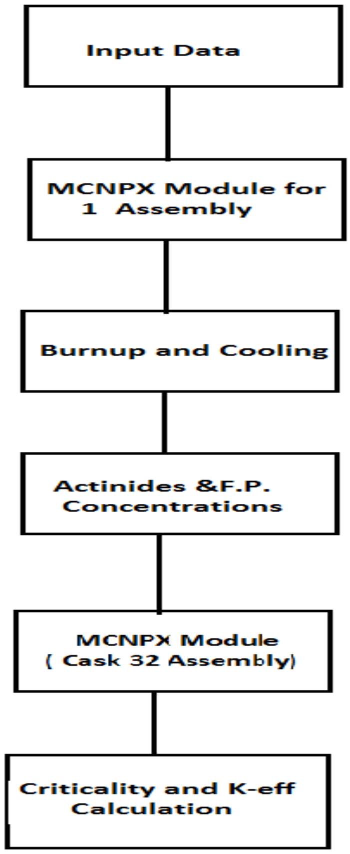

Fig. (4): Computational Procedure of The Model

3. RESULTS AND DISCUSSIONS

3.1 Fuel Assembly Burnup

Figure 5 shows the axial fuel burnup (Gwd/T) versus axial fuel distance for different fuel enrichments 2 , 3 , 4 , and 5 %. The axial burn up illustrates cosine shape and peak at the fuel center, the minimum burnup at fuel bottom is 36 Gwd/T while the maximum value approach 84 Gwd/T with maximum/ minimum ratio 2.33

Figure 6 shows the concentration of U-235 and Np-237 (atom/barn.cm) versus axial fuel distance (cm) for fuel of initial enrichment 4 % at burnup 60 Gwd/T and 5 years cooling. The results indicate that U-235 are consumed higher in the middle zones. While Np-237 remains constant along the axial distance.

Figure 7 shows Pu- isotopes concentration (atom/barn.cm) versus axial fuel distance (cm) for fuel of initial enrichment 4 % at burnup 60 Gwd/T and 5 years cooling. The concentration of Pu-239 after 60 Gwd/T is 2.0E-4 (atom/barn.cm) and higher than other isotopes.

Figure 8 shows the concentration of four fission products Mo-95, Nd-145, Nd-143 and Tc-99 (atom/barn.cm) versus axial distance for assembly of initial enrichment 4 % at burnup 60 Gwd/T and 5 years cooling. The results of both Nd-143 and Tc-99 coincide with each other.

Figure 9 shows the concentration of 3 fission products Sm-149, Gd-155 and Ag-109 ((atom/barn.cm) versus axial distance for assembly of initial enrichment 4 % and 60 Gwd/T and 5 years cooling. The results of both Sm-149 and Gd-155 coincide with each other.

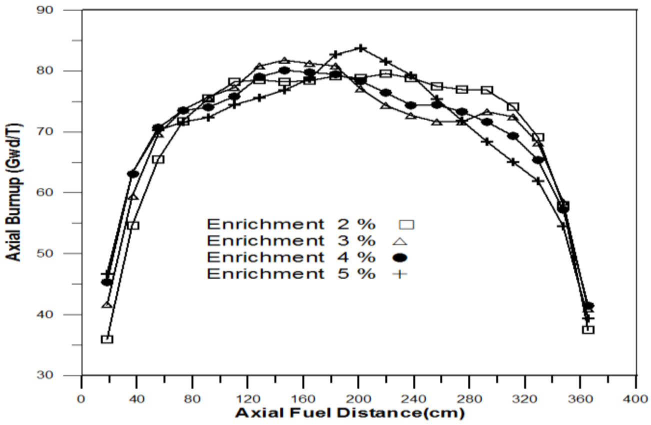

Fig. (5): Fuel Burnup (Gwd/T) versus axial fuel distance (cm)

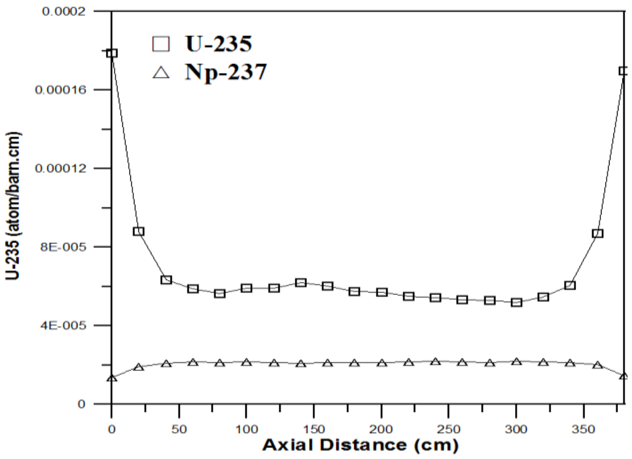

Fig. (6): U-235 and Np-237 versus Axial core distance (cm) for 4 % Enrichment at discharge burnup 60 Gwd/T and 5 year cooling.

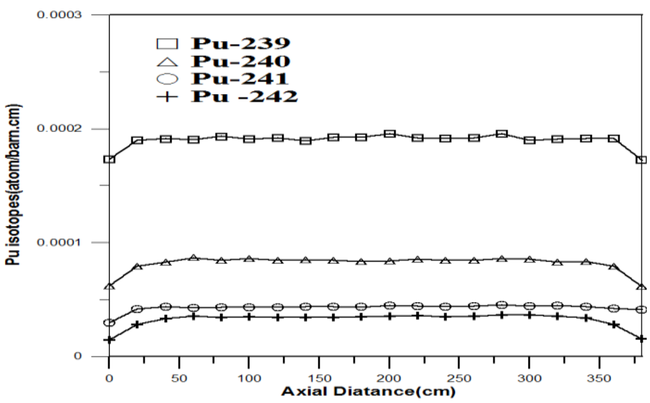

Fig. (7): Pu isotipes versus Axial core distance (cm) for 4 % Enrichment at discharge burnup 60 Gwd/T and 5 year cooling.

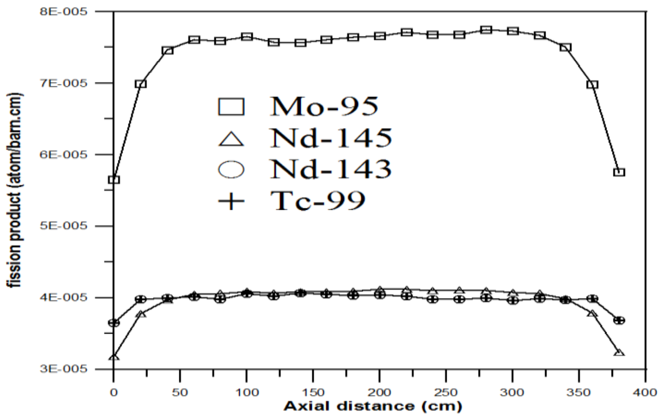

Fig. (8): Fission product versus Axial core distance (cm) for 4 % Enrichment at discharge burnup 60 Gwd/T and 5 year cooling.

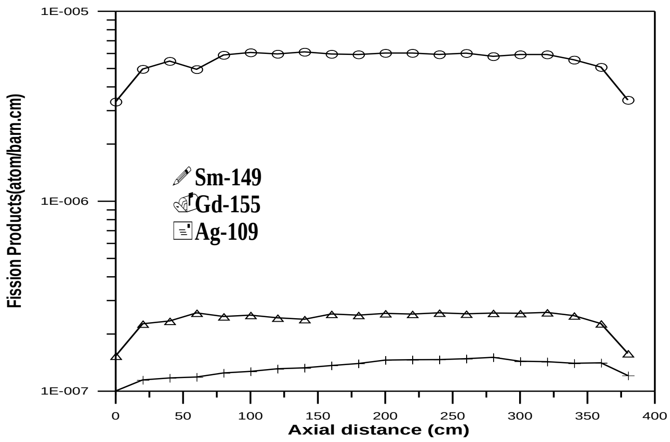

Fig. (9): Fission product versus Axial core distance (cm) for 4 % Enrichment at discharge burnup 60 Gwd/T and 5 year cooling.

3.2 Cask Loaded with UO_2 – PWR fuel Assemblies

Based on the nuclide sets identified in the previous section, calculated k_{eff} values for the cask Loaded with 32 fuel assemblies of type UO_2 (Figure 1) are provided as a function of initial enrichment, fuel bumup history , and cooling time within the ranges relevant to storage and transportation.

Table 2 shows K_{eff} for cask GBC-32 loaded with assemblies with different burn up fresh , 5000, 30000 , 40000 MWd/T for fuel with initial enrichment 5 % without cooling (0.0 cooling time). The results indicate that ($K_{eff} < 1$) the cask is subcritical only for storage of 40,000 Mwd/T. K_{eff} for storage cask > 1.0 because we assumed zero cooling time which is not a real transport case.

Table 3 shows K_{eff} for the cask of different initial enrichment 2 , 3 , 4 , and 5 % with all having discharge burnup 60,000 Mwd/T and cooling time 5 years , the results indicate that the cask are subcritical in all conditions. The cask is subcritical for all types of initial fuel enrichments even at 5 % which coresponds to advanced Generation III PWR. The difference between the present results and reference [6] is due to the difference in the amount of actinides and fission products between the two methods which affect the cask multiplication factor.

Table 4 shows cask multiplication factor (K_{eff}) with initial enrichment 5 % with discharge burnup 60000 Mwd/T and different cooling times 0.0 , 5 , 20,30, 50, 100 years. The results indicate that K_{eff} for the cask decrease as cooling time increase because actinides and fission products decay with time.

Abnormal conditions: Three cases of unnormal storage are considered :

- Loss of water cooling inside the cask
- The cask is totally immersed in water
- 4 fresh casks are misloaded into the cask which is filled with high burned fuel.

Table 5 shows K_{eff} for cask GBC-32 loaded with fuel with initial enrichment 5 % and burnup 40000 Mwd/T and in the second case water cooling is lost from the cask. The results show that K_{eff} decrease from 0.96186 to 0.38497 in the case of loss of water cooling inside the cask.

Table 6 shows K_{eff} for cask GBC-32 loaded with fuel with initial enrichment 5 % and burnup 40000 Mwd/T , the second case if the cask is totally immersed in water, the results show that K_{eff} increase slightly from 0.96186 to 0.96215 in the case of cask drops in water.

Table 7 shows K_{eff} for cask GBC-32 loaded with fuel initial enrichment 5 % and burnup 40000 Mwd/T in the case of misloading with 4 fresh fuel assemblies. the results show that K_{eff} increase from 0.96186 to 0.99432 in the case of 4 casks with fresh fuel are misloaded. K_{eff} increases because the mass of fissile isotopes increased in the second case.

3.3 Cask Loaded with MOX – PWR fuel Assemblies

Table 8 shows K_{eff} for Cask Loaded with 32 MOX fuel assemblies for fuel burned to 50000 and 75000 Mwd/ T with zero cooling time The results indicate that K_{eff} decreases with increasing fuel burn up because Pu isotopes were consumed as fuel burned inside the reactor.

Table 9 shows K_{eff} for Discharge of burnup 75,000 MWd/T and different cooling time varying from 5 to 10 years , the results indicate the K_{eff} decreases with cooling time , because cooling the fuel permits for cask isotopes (actinides and fission product) to decay.

CONCLUSION

- MCNPX computer code is used to model a generic PWR cask which contains 32 spent fuel assemblies. The cask is loaded with two types of fuel assemblies, UO_2 - PWR and MOX.
- Several parameters that affect the safety of the storage such as fuel enrichment, burn up history, and cooling time are analyzed.
- Increasing the initial fuel enrichment will increase the criticality of the cask, while increasing the discharge burnup will decrease the cask criticality.
- Larger cooling time of the fuel, reduces the cask criticality.
- Losses of cooling water inside the cask reduces cask criticality.
- If the spent fuel cask dropped in water, K_{eff} increases.
- if 4 fresh assemblies are misloaded into the storage cask that contains spent fuel, K_{eff} increases.
- Calculations should be performed for each fuel package separately to analyze and avoid the boundaries of the criticality and accident conditions.

Table (2): K_{eff} for cask GBC-32 with different burnup and zero Cooling time of initial fuel 5 % Enrichment

Burnup Mwd/T	0.0	5000	30,000	40,000
MCNP Model	1.17607±0.00098	1.13908±0.00097	1.04691±0.00068	0.96186±0.00079
Reference [6]	1.19142±0.00056	-----------	---------------	-----------

Table (3): K_{eff} for cask GBC-32 with different fuel enrichment at burnup 60,000 Mwd/T and 5 years cooling

Enrichment	2 %	3 %	4 %	5%
MCNPX model	0.73828±0.00064	0.77057±0.00067	0.81418±0.00067	0.87433±0.00052
Reference[6]	0.64783±0.00047	0.71563±0.0.00051	0.79043±0.00056	0.85549 ±0.00052

Table (4): K_{eff} for cask GBC-32 with burnup 60,000 Mwd/T and different Cooling time (year)

Cooling Time (years)	0.0	5	20	30	50	100
MCNP Model	0.88563 ±0.00059	0.87433 ±0.00052	0.85642 ±0.00062	0.85549 ±0.00048	0.84133 ±0.0005	0.83738 ±0.00042
Reference [5]	0.88140 ±0.00048	0.85549 ±0.00048	0.82850 ±0.0005	------	-------	-------

Table (5): K_{eff} for GBC-32 Cask with fuel initial 5% enrichment and 40,000 Mwd/T ,

	Normal	Loss of cooling
MCNP Results	0.96186±0.00079	0.38497±0.00015

Table (6): K_{eff} for GBC-32 Cask with fuel initial 5% enrichment and 40,000 Mwd/T

	Normal	Over moderated
MCNPX results	0.96186±0.00079	0.96215±0.00079

Table (7): K_{eff} for GBC-32 Cask with fuel initial 5% enrichment and 40,000 Mwd/T

	Normal	Misloading 4 fresh assemblies
MCNPX results	0.96186±0.00079	0.99432±0.00121

Table (8): K_{eff} for Different Discharge burnup and zero Cooling time

Fuel Burnup	50,000 MWd/T	75,000 MWd/T
K_{eff}	0.95924±0.0009	0.91897±0.00072

Table (9): K_{eff} for Discharge burnup 75,000 MWd/T and different cooling time

Cooling time (years)	5 years	10 years
K_{eff}	0.88301±0.00072	0.84564±0.00074

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