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Modeling and Validation of an Advanced Pressurized Water Reactor using Monte Carlo Technique

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Abstract:
The validation of reactor core design modeling codes is very essential, and can be achieved by comparing the code results with the available experiments or other computational models.
In this paper, the reactor core of the advanced pressurized water reactor, AP1000 was simulated, verified and modified using Monte Carlo Computer Code, MCNP. The physical and mathematical models of the MCNP-Code include, the Boltzmann neutron transport equation and the time dependent number densities of the depleted fuel inventory equations. The reactor core of AP1000 includes 157 fuel assemblies, and each assembly contains 264 UO_2 fuel rods arranged in 17x17 square array, with three different batches of fuel enrichment. The initial core contains two types of burnable absorbers, discrete burnable absorbers (PYREX), and Integrated Fuel Burnable Absorbers (IFBA) for the compensation of the initial excess reactivity and for increasing the reactor operation cycle length.
The simulated results were compared with the reference design parameters and validated with other calculations which were performed by other authors using SCALE and WIMS Codes.
The modeling results showed that the criticality (K_{eff}) for cold core was 1.204 for our model and 1.205 for the reference design, and for the clean assembly was 1.32728 for our model and 1.328 for the reference design which showed a good agreement. In addition to, some of thermal hydraulic safety parameters as; critical heat flux, actual heat flux, Departure from Nucleate Boiling Ratio (DNBR) and fuel power density were calculated using different correlations and other conservative equations. The results were compared with the reference design parameters, which showed a good acceptance, and confirm the safe design and efficient modeling.
Also, AP1000 core were modified by two approaches: the first, by using Uranium oxide fuel composition with three different enrichment regions (3.5, 4.5 and 4.95) % respectively and including burnable absorbers with the purpose of reaching high burnup and long cycle length. The modified core was modelled using MCNPX computer Code, and the results of the burn up calculations at cycle length of 21 months reached 25.3 GWd/MTU. So, in the three cycles it reaches to 75.9 GWD/MTU and preserving the fuel integrity by using high performance ZIRLO cladding material. The second core modification was by using Mixed Oxide fuel (MOX).This core consist of three different regions; first region contains MOX fuel (U+Pu) O_2 with 7% Pu- fissile content, second and third regions consist of two different enrichment of UO_2 (4.5 and 4.95) %. In addition to, the MOX core include the same types and number of BAs. as in the reference design core of the AP1000

Keywords: *AP1000, MCNPX, Burn-up, IFBA, PYREX, MOX*

1. Introduction

The development of Nuclear Power Plants is based on new design pattern of reactor core such as different fuel types, different enrichments and different Burnable Absorber rods (BAs). The main purpose of the development is to achieve high fuel burn-up, long cycle length, non- proliferation. Neutronic computer codes can assist in applying the necessary modeling and calculations in the field of; criticality control, depletion, source term calculation, power distribution and other safety related parameters [1, 8].

Several efforts and publications were dealt with reactor core modeling and core modifications with different approaches of different fuel types and different enrichments with different burnable absorbers and computing methodologies [2, 3]. The objective of the BAs is to reduce the excess reactivity in the initial core and to increase the operational cycle length with high fuel burn-up. The recent advanced BAs that used in advanced nuclear power reactors are the Discrete Absorbers (PYREX) and the Integral Fuel Burnable Absorbers (IFBA).
In this paper, we performed reactor core modeling and calculations for the Advanced Pressurized Water Reactor (APWR), AP1000, using MCNP Computer Code with the purpose of model verification for the initial core design to support our approach for subsequent core development. The results of the calculations include, reactor core burn-up, operational cycle length, criticality and concentration of the radionuclides in the depleted fuel for the initial core design and for the modified cores.
The excess reactivity for the clean core of the AP1000 was validated with the reference core design using MCNP Code, and the burn-up calculations of the first cycle with BAs were verified with the published results using other codes [2].
The reactor core neutronic design should be compatible with the thermal hydraulic design to preserve the safety of fuel integrity and to achieve high and reliable performance of the reactor. Also, the heat removal from the reactor core should be equal or greater than the heat generation rate, for avoiding overheating and susequently fuel failure.
Therefore, some thermal hydraulic safety related parametrs were calculated using the consevative correlations to find the Deparure from Nucleate Boiling Ratio (DNBR), Critical and average heat fluxes, Mass Flux, fuel power density and linear power density. Our modeling results using MCNP computer Code were compared and verified with the design reference date and other calculations using SCALE and WIMAS computational Codes.

2. Core Configuration of the AP1000 Advanced PWR

Westinghouse advanced pressurized water reactor, AP1000 is a Generation III+, two-loop pressurized water reactor (PWR). The reactor thermal power is 3400 MW_{th} with nominally electric power 1115 MW_e. The fuel type is$UO_2$with three enrichment regions and the reactor core is cooled/moderated by light water.
AP1000 core contains a matrix of fuel rods assembled into mechanically identical 157 fuel assemblies; each one contains 264 fuel rods, 24 guide tubes for control rod clusters, and one centrally located guide tube for in-core instrumentation, all of which are arranged in a 17 x 17 square lattice array. There are three different enrichment regions in the core to tune the flux and power profile over the core. The enrichment of the fuel in the first cycle of the core is (2.35, 3.4 and 4.45) % as shown in Figure 1. The core is designed for a fuel cycle length of 18 months with average burn-up of 60 GWd/MTU. The technical data of AP1000 is presented in Table 1[1].

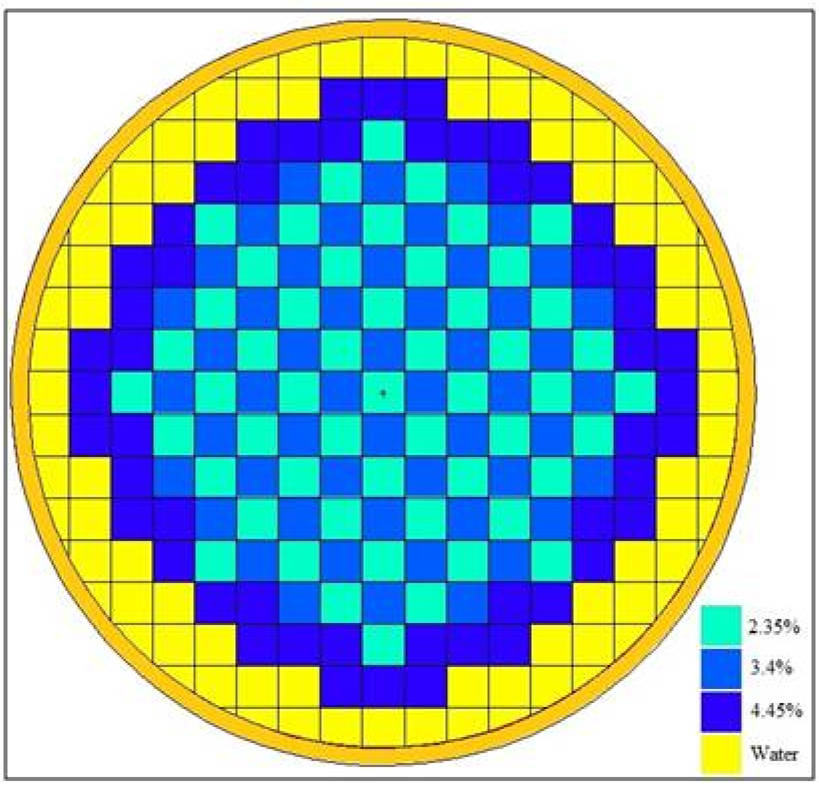

Figure 1: AP1000 First Cycle Core Configuration

Core Configuration	
Active core Equivalent diameter	304.038 cm
Active fuel height first core, cold	426.72 cm
Fuel Assembly	
Number	157
Rod array	17x17
Rods per assembly	264
Rod pitch	1.26 cm
Fuel Rod	
Outer diameter	0.94996 cm
Gab thickness	0.0165 cm
Clad thickness	0.0572 cm
Clad material	ZIRLO™
Clad density	6.5 g/cm^3
Material	UO_2 sintered
Density (% of theoretical)	95.5%
Fuel enrichments (wt.%)	
Region 1	2.35
Region 2	3.4
Region 3	4.45
Diameter	0.8192 cm
Length of Fuel pellet	0.98298 cm
Thermal Hydraulic Parameters	
System pressure, nominal	2250 psia
Minimum DNBR at nominal conditions Thimble(cold wall) flow channel	2.74
Coolant temperature[d,e]	
Nominal inlet	535.0 °F
Average rise in vessel	77.2 °F
Average rise in core	81.4°F
Average in core	578.1°F
Average in vessel	573.6°F
Effective flow area for heat transfer	$41.8 ft^2$
Average velocity along fuel rods [a]	15.8 ft/s
Average massve locity, [a]	$2.55*10^6$ lbm/hr-ft^2
Heat Transfer	
Active heat transfer surface earea [b]	56700 ft^2
Average heat flux	199300 BTU/hr-ft^2
Maximum heat flux for normal operation [c]	518200 BTU/hr-ft^2
Average linear power [d]	5.72 KW/ft
Peak linear power for normal operation [c]	14.9 KW/ft
Fuel Central Temperature	
Peak at peak linear power for prevention of center line melt	4700 °F

(a) Basedonthermaldesignflowand5.9percentbypassflow.
(b) Based on densified active fuel length. The value for AP1000isroundedto5.72KW/ft.
(c) Basedon2.60FQpeaking factor
(d) The value for AP1000isroundedto5.72KW/ft.

Table 1: AP1000 Core design data

Fuel	Enrichment (wt.%)	Nuclide	Atom density (atom/b.cm)
UO_2	2.35	U-235	5.56094375E-04
		U-238	2.28162251E-02
		O-16	4.67446389E-02
UO_2	3.4	U-235	8.04549293E-04
		U-238	2.23251989E-02
		O-16	4.67501601E-02
UO_2	4.45	U-235	1.0529963E-03
		U-238	2.2324844E-02
		O-16	4.6755681E-02
	Water	H-1	6.69111E-02
		O-16	3.35556E-02

Material	Nuclide ID	Mass Fraction
ZIRLO	Nb-93	-0.0120
	Fe-54	-0.00008
	Fe-56	-0.00119
	Fe-57	-0.00003
	Sn-112	-0.00008
	Sn-114	-0.00005
	Sn-115	-0.00003
	Sn-116	-0.00116
	Sn-117	-0.00061
	Sn-118	-0.00194
	Sn-119	-0.00069
	Sn-120	-0.00261
	Sn-122	-0.00037
	Sn-124	-0.00046
	O-16	-0.0016
	Zr-90	-0.50272
	Zr-91	-0.10963
	Zr-92	-0.16757
	Zr-94	-0.16982
	Zr-96	-0.02874
	Nb-91	-0.02736

Table 2: Material Composition

2.1. Discrete Burnable Absorber (PYREX)

Westinghouse has manufactured that type of Discrete Burnable Absorber (PYREX); which utilize borosilicate glass (B_2O_3-SiO_2 with 12.5 wt % B_2O_3) in the form of Pyrex tubing as a neutron absorber with a void central region filled by water located in the assembly guide tubes which are cladded in 304 stainless steel.

The discrete burnable absorber(PYREX)can be placed in any assembly which is not exist in a Reactor Core Control Assembly (RCCA) location, using several possible radial configurations as shown in Figure 2. The specification for Pyrex Rod is provided below in Table 3. [6, 7]

Boron-10 Loading	**6.24 mg/cm**
Concentrating	12.5 wt% B_2O_3
Density	2.24 g/cc
Inner Tube Inner Radius	0.214 cm
Inner Tube Outer Radius	0.231 cm
Pyrex Inner Radius	0.241 cm
Pyrex Outer Radius	0.427 cm
Cladding Inner Radius	0.437 cm
Cladding Outer Radius	0.484 cm
Absorber length	368.3 cm
Inner Tube Material	SS304
Clad Material	SS304
Number of rods	1558
PYREX composition	
Nuclide	**Weight fraction % of element**
B-10	0.699
B-11	3.207
O-16	53.902
Si	37.856

Table 3: PYREX rod specification

2.2. Integral Fuel Burnable Absorber (IFBA)

Use of Integral Fuel Burnable Absorber (IFBA), is a common modern technique for optimized fuel assembly reactivity control and for power distribution management. It is a very thin layer of ZrB_2 coated to UO_2 fuel pellets in an assembly. Because the boron is completely depleted quickly, and it does not displace fuel material, there is no residual reactivity penalty. The IFBA specification is provided in Table 4. [1,7]

Poison Material	**ZrB_2**
Boron-10 Loading	0.772 mg/cm
Boron-10 Enrichment	19.9%
Poison Height	304.8 cm
Poison Location	Centered axially

Table 4: IFBA fuel rod Specification

3. Core Modeling using Monte Carlo Simulation Technique

3.1. Assembly and Core Modeling

The Monte Carlo Computer Code, MCNP6, was used formodeling of AP1000 reactor core. The modeling was applied for the first cycle core configuration (initial core),which consist of three different enriched fuel assemblies, (2.35, 3.4, and 4.45)% with two types of burnable absorber; Discrete Burnable Absorber (PYREX) and Integral Fuel Burnable Absorber rods(IFBA). Pyrex rods are removed after the first cycle, however IFBA remains till the assembly discharged from the core[4-7]. Figure 2 shows schematic geomtry of fuel rod and PYREX rod. Figure 3 shows the PYREX and IFBA arrangment within the fuel assmblies in the initial core configuration.

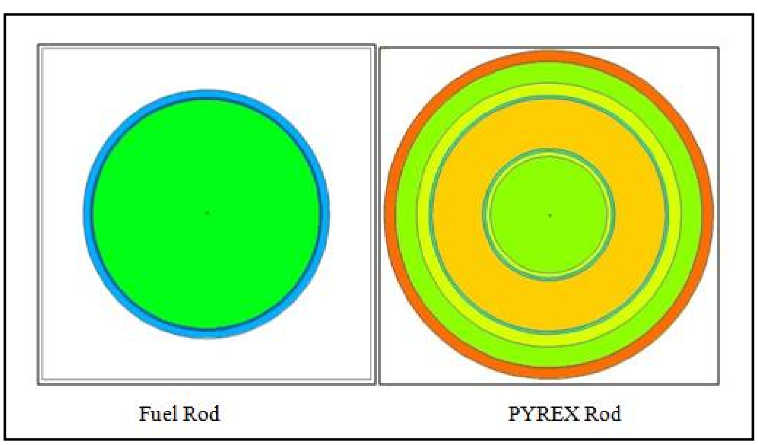

Figure 2:Schematic geomtry of Fuel Rod and PYREX Rod

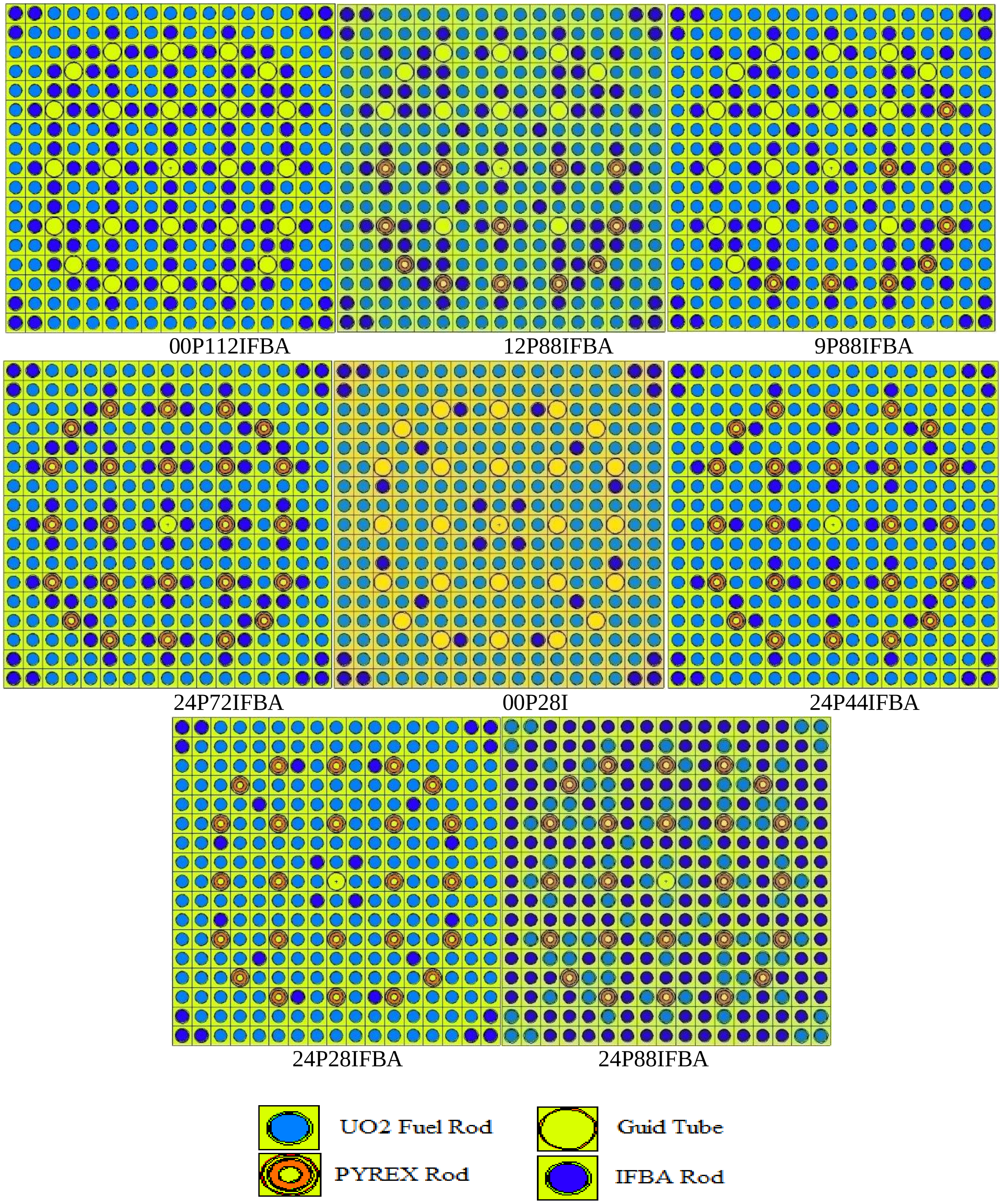

Figure 3: PYREX and IFBA arrangment within AP1000 Assmbly.
P: Pyrex, I: IFBA

4. Results and Analysis

4.1. Neutronic Model Validation

The simulation model considers, clean core without any absorbers, cold, zero power and zero soluble boron in the first cycle of the core. Table 5 presents the results of our calculations using MCNP6 computer Code against the reference design results and also against other published results. The results were found to be in a good consistent with the reference design values which were published in the Design Control Document of AP1000 issued by Westinghouse[1] ; and also in agreement with the other published results using SCALE and WIMS9 computer codes.

Also, in our calculations we used the Data ENDF/B-VII.0 library with fuel temperatures at 900K and with moderator temperatures at 600K. Depletion calculations were performed using 550 cycles with 10,000 neutron histories per cycle.

AP1000 Neutronic data	Reference values[1]	MCNP6	SCALE[2]	WIMS9[3]
Max fuel assembly k_∞ (cold, clean, unborated water)	1.328	1.32728±0.00019	-	-
Max core reactivity K_{eff} (cold, zero power, BOC, Zero soluble boron)	1.205	1.20403±0.00021	1.2026	1.2038

Table 5: AP1000 Reactor Core Criticality validation by MCNP6 compared with Reference values, SCALE and WIMS9Codes

4.2. Burn-up Calculations and Analysis

In this section, we verified our depleted core with published results from Sandia Lab of the first core cycle of AP1000[2]. The results are shown in Tables 6,7 and 8 and presented in Figures 4,5 and 6.

AP1000 Neutronic data	MCNP6
Max fuel assembly k_∞ (cold, clean, unborated water)	1.32728±0.00019
Max core reactivity K_{eff} (cold, zero power, BOC, Zero soluble boron)	1.20403±0.00021
Max core reactivity $K_{eff}$with PYREX and IFBA	1.18785±0.00045

Table 6: AP1000 Reactor Core Criticality

Time (days)	K_{eff}
0.00E+00	1.18785
1.00E+00	1.16015
8.00E+00	1.14929
4.00E+01	1.14168
9.70E+01	1.13264
1.69E+02	1.12457
2.41E+02	1.11777
3.13E+02	1.10995
3.88E+02	1.09753
4.63E+02	1.07839
5.50E+02	1.04982

Table 7: Cycle Length and Criticality for AP1000 Reactor

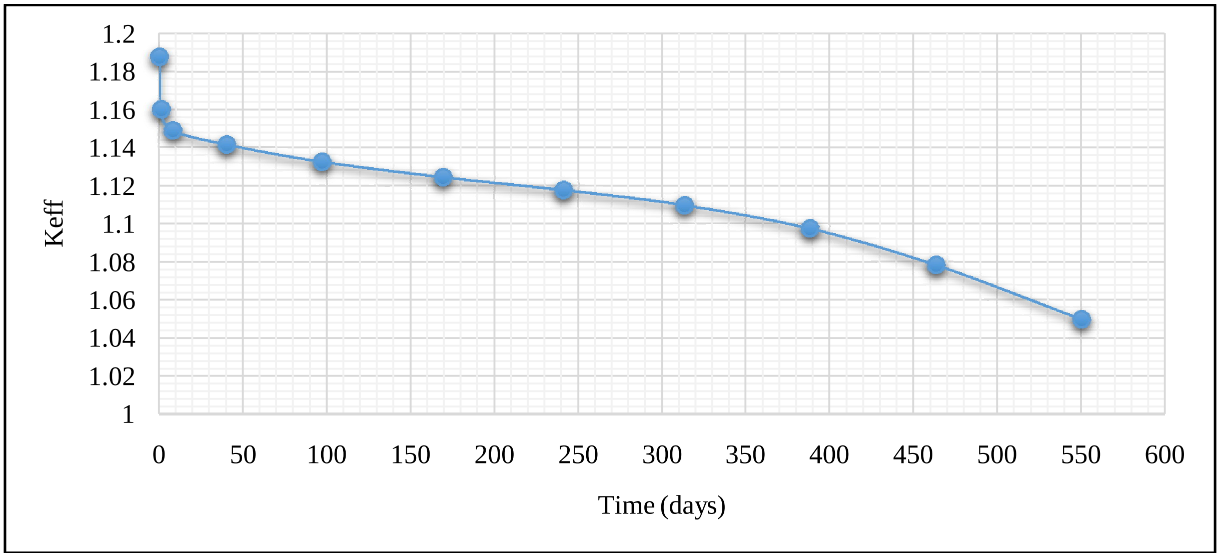

Figure 4: K_{eff} Vs Time

Time (days)	Burn-up (GWd/MTU)
0.00E+00	0.00E+00
1.00E+00	3.96E-02
8.00E+00	3.16E-01
4.00E+01	1.58E+00
9.70E+01	3.84E+00
1.69E+02	6.68E+00
2.41E+02	9.53E+00
3.13E+02	1.24E+01
3.88E+02	1.53E+01
4.63E+02	1.83E+01
5.50E+02	2.18E+01

Table 8: Cycle Length and Burn-up for AP1000 Reactor

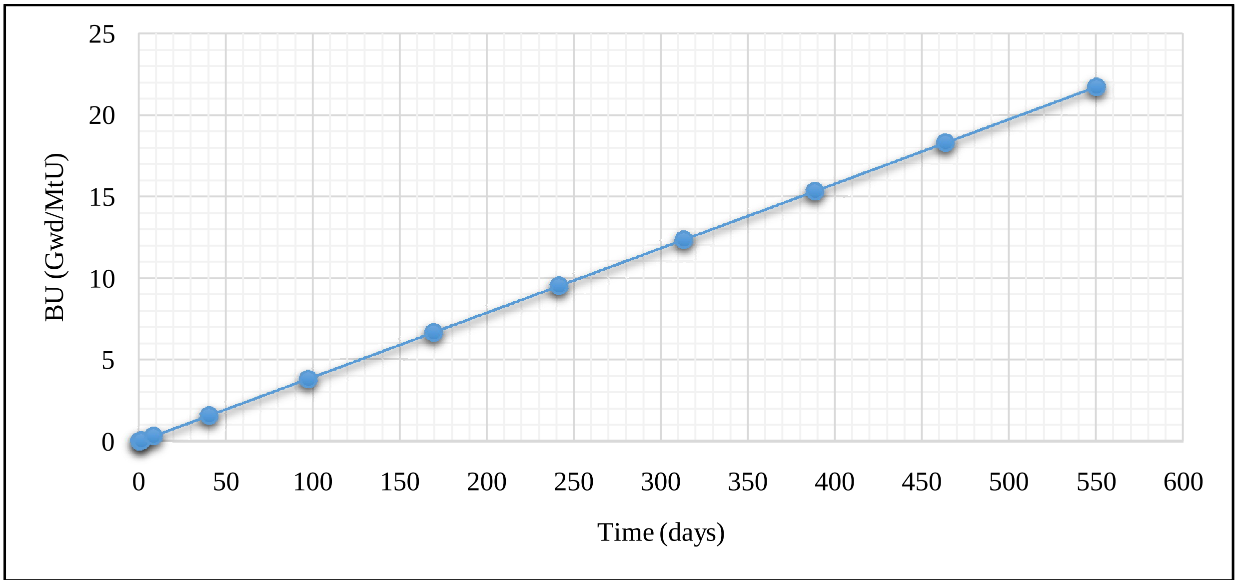

Figure 5: Burn Up Vs Time

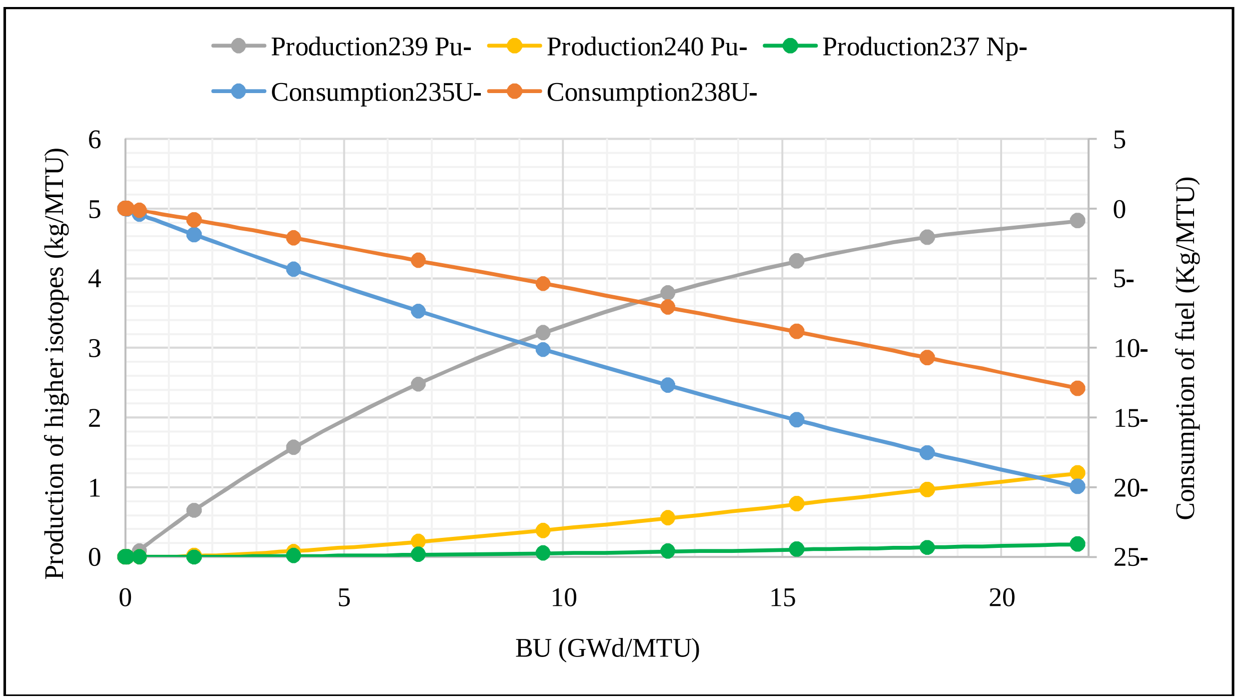

Figure 6: Inventory of Major Actinides

From Figure 4, AP1000 reference core k_{eff} has an initial value of 1.18785, after the participation ofBAs, K_{eff} decreased rapidly then stabilized in decreasing over the operation time due to the consumption of PYREX and IFBA in the core. The reactor core reaches at the end of cycle to 21.8 GWd/MTU which is equivalent to 550 days (i.e. 18 month)

As the core was loaded by 86 tons of UO_2, Uranium was depleted due to the fission process which lead to the buildup of fission products which are mainly actinides. The major consumed elements were, U-235 and U-238 and the produced actinides were, Pu-239 and Pu-240, which were increased with increase of fuel burn-up, as shown in Figure 6 which illustrates the consumption of UO_2 and the production of actinides.

4.1. Verfication of Some Thermal Hydraulic Safety Related Parameters

4.1.1. Departure from Nucleate Boiling Ratio (DNBR)

As DNBR is the *ratio* of the heat flux needed to cause *departure from nucleate boiling* to the actual local heat flux of a fuel rod, then we'll calculate the Critical Heat Flux using the following Correlation [9], Based on the thermal hydraulic data for AP 1000 at Table 1, We get;

$$q_c'' = C * 10^{-6} (G * 10^{-6})^m \Delta T_{sub}^{0.22} \tag{1}$$

Where:

C and m are constants with the following values 0.445 and 0.5 respectively.

G = Average mass velocity= $\frac{\text{coolant mass flow rate per fuel rod}}{\text{area of coolant channel (lb/hr.}ft^2)}$

Where;

Coolant mass flow rate per fuel rod = **0.297 kg/sec.**

Coolant channel area for square lattice = **0.87 cm^2**,

Then, average mass velocity is = **2.52 x 10^6 lb$_m$/hr.ft^2**

Mean coolant temperature at normal operating conditions = **302.05 °C** (575.6 °**F**),

At coolant pressure = 15.51 MP_a = **2250 psia**,

Saturation Temp.= 652 °**F**,

ΔT_{sub}= **76.4 °F**

Substitute in Eq. (1),

we get, the critical heat flux=**578W/cm^2** (2)

And the critical fuel power density = **3233.6 W/ cm^3** (3)

Figure 7, shows the axial distribution along the fuel rod of DNBR. The minimum DNBR (MDNBR) reaches the value 2.50 at the fuel center line which is considered greater than 1.3 which is the safety limit.

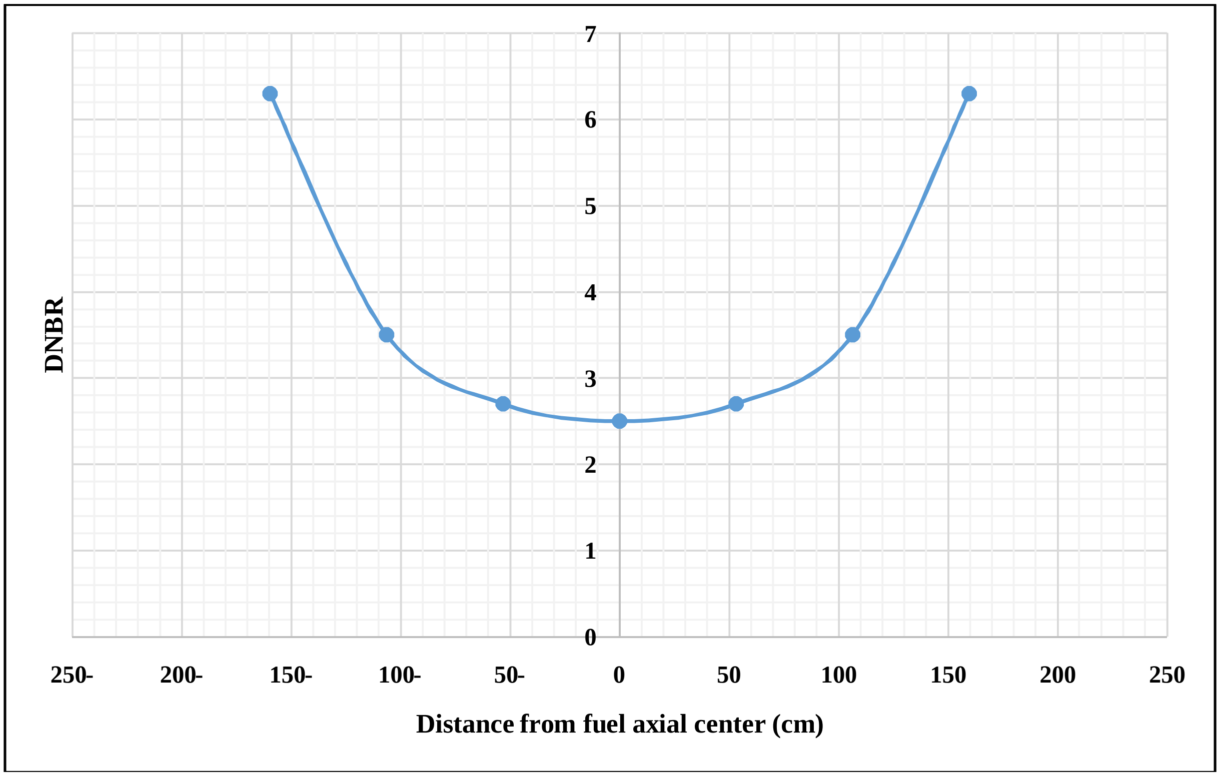

Figure 7: DNBR at axial fuel distance

4.1.2. Calculation of Actual Maximum Heat Flux and Power Density

Using the formula obtained from Ref. [9],

Actual maximum power density, $q'''_{(actual)} = \frac{P}{0.275} * N * A_c * H_e$ (4)

Where,
P: Reactor power,
N: Number of fuel rods in the core,
A_c: Fuel cross section area,
H_e: Extrapolation fuel length.
Then, we get,
Actual Maximum Power Density = **1306.95 W/cm^3** (5)
Actual Maximum Heat Flux =**2.34 x 10^6 W/m^2** (6)

Dividing Eq. (2) by Eq. (6), we get the DNBR value is 2.47, at the hottest coolant channel, which is greater than 1.3 as recommended by the designer, and the reference value which was calculated by the designer using (WRB-2M) correlation [1], was = 2.74,
Also, we can calculate from previous equations, the peak and average linear power and the hot channel factor as:
Peak linear power = **14.93 KW/ ft** (7)
Average linear power= **5.74 KW/ft** (8)
Hot channel factor = peak linear power/ average linear power = **2.60** (9)

Figure 8, shows the axial distribution of actual heat flux, fuel power density compared to critical heat flux, and average heat flux.

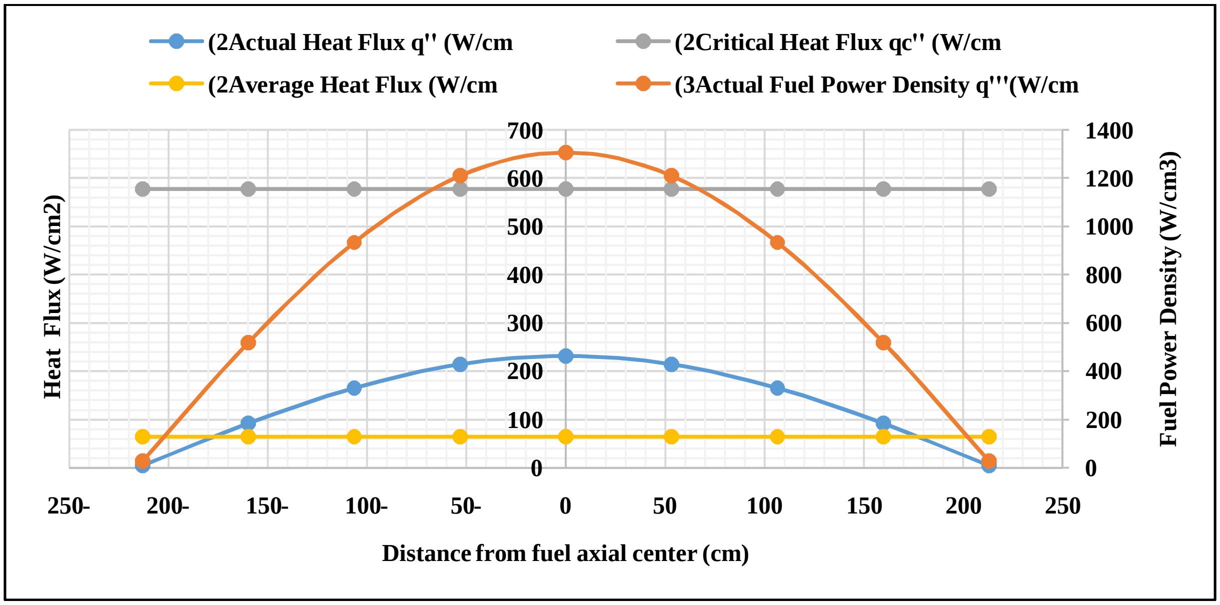

Figure 8 Actual, average, critical heat flux and actual fuel power density

4.1.3. Calculation of linear power, heat flux and DNBR in case of overpower (118%)

In this part, we calculated the linear power, actual heat flux and DNBR for the hot channel of AP1000 in case of reactor power increase by 118%. The results were compared with the safety margin of the reference design safety parameters.

If it was assumed the reactor power was increased by 118 % of the nominal design value, then, the new reactor thermal power will be 4012 MW_t,

After calculations we get, the maximum fuel power density is 1542.26 W/cm^3, the maximum actual heat flux is 272.98 w/cm^2, peak linear power = 17.97 kw/ft which is still less than the design limiting value (< 22.45 kw/ft) and DNBR is 2.12, which still safe and greater than the safety limit 1.3.

So, the reactor design safety parameters were validated with the conservative models and correlations, which support our simulated neutronic modeling in case of core development with other fuel compositions and other core structure materials.

4.1.4. Comparison with Design Calculations

Table 9 presents the calculation of some thermal hydraulics safety parameters using conservative approach which are compatible with the neutron modeling, compared with the calculated design values using other methods [1].

Item	Our Calculation	Reference Values
DNBR at Hot Channel	**2.50**	**2.74**
DNBR at overpower 118%	**2.12**	**-**
Average mass Velocity ($lb_m/hr\ ft^2$)	**2.52×10^6**	**2.55×10^6**
Coolant Pressure (psia)	**2250**	**(1459 – 2425)**
Max Actual Heat Flux ($Btu/hr.ft^2$)	**524,012**	**518, 200**
Average Actual Heat Flux ($Btu/hr.ft^2$)	**199244**	**199300**
Peak Linear Power (Kw/ft)	**14.93**	**14.90**
Average Linear Power (Kw/ft)	**5.74**	**5.72**
Hot channel factors	**2.62**	**2.60**
Average fuel power density (KW/KgU)	**39.6**	**40.2**
Average power density (MW/m^3)	**109.8**	**109.7**

Table 9: Claculated Thermal Hydraulics Safety Parameters compared to Reference data

From Table 9, we can see that our calculations are in good agreement with the reference value, which verify and support our core neutronic modeling at the maximum reactor power of 3400 MW and burn-up of 60GWD/MTU. Also, this can support our approach for core modification safely. Any uncertainty is due to the utilization of different correlations in the calculation of DNBR and volumetric thermal source strength.

5. Modified Core Patterns

5.1. UO_2 Core

The purpose of the modified core patterns is to increase the fuel burn-up and to extend the cycle length. The first core modification was proposed by using UO_2 fuel with higher enrichment for reaching to 21 months cycle length, taking into consideration the maximum acceptable value of the enrichment limit. In addition to that, we used 1558 PYREX rods and 8832 fresh IFBA rods which serve as a Burnable Absorbers (BAs). The addition of the BAs in the core will maintain the core operation for 21 months by reducing the multiplication factor, K_{eff} at the BOC then its effect decreases at the EOC. We considered the same distribution of BAs as in the reference core design of the AP1000 with the proposed three different batches (3.5, 4.5 and 4.95) % of UO_2.

Criticality, burn-up and concentration of radionuclides calculations were performed to the UO_2 modified core pattern using MCNPX for the input model for the three cases of study;

1. Cold, clean and un-borated water assembly;
2. Cold, zero power, zero soluble boron for the beginning of first cycle of the core;
3. Introducing Burnable Absorbers (PYREX and IFBA) in the core to compensate the initial excess reactivity

The results of the operation burn-up, criticality control and depleted core inventory are shown in Tables [10-12] and Figures [9-11].

Neutronic data	MCNP6
Max fuel assembly k_∞ (cold, clean, unborated water)	**1.39532±0.00019**
Max core reactivity K_{eff} (cold, zero power, BOC, Zero soluble boron)	**1.28395±0.00023**
Max core reactivity K_{eff} with PYREX and IFBA	**1.26948±0.00038**

Table 10: Criticality of Modified UO_2 Core

EFPD	K_{eff}
0.00E+00	**1.26948**
1.00E+00	**1.23691**
8.00E+00	**1.22525**
4.00E+01	**1.21538**
9.70E+01	**1.20193**
1.69E+02	**1.19027**
2.22E+02	**1.18231**
2.75E+02	**1.17609**
3.66E+02	**1.16484**
4.58E+02	**1.14824**
5.49E+02	**1.12429**
6.41E+02	**1.09500**

Table 11: Cycle Length and criticality for the modified UO_2 Core

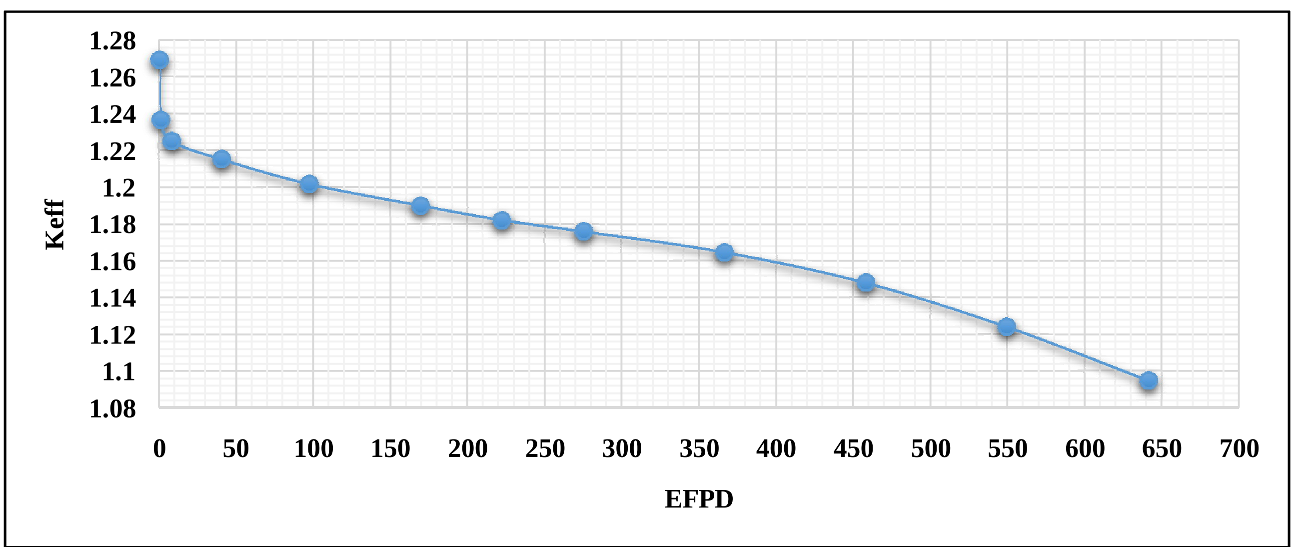

Figure 9: K_{eff} vs Effective Full Power Days (EFPD)

EFPD	BU (GWd/MTU)
0.00E+00	**0.00E+00**
1.00E+00	**3.95E-02**
8.00E+00	**3.16E-01**
4.00E+01	**1.58E+00**
9.70E+01	**3.84E+00**
1.69E+02	**6.68E+00**
2.22E+02	**8.78E+00**
2.75E+02	**1.09E+01**
3.66E+02	**1.45E+01**
4.58E+02	**1.81E+01**
5.49E+02	**2.17E+01**
6.41E+02	**2.53E+01**

Table 12: Burn-up and Cycle Length for the modified UO_2 Core

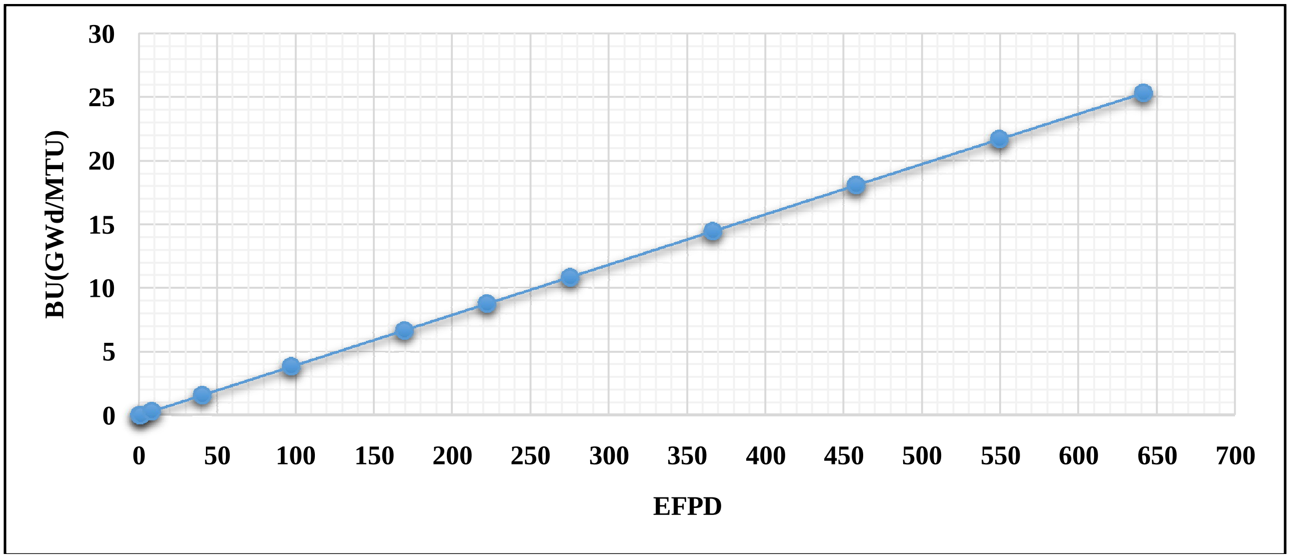

Figure 10: Burnup vs Effective Full Power Days (EFPD)

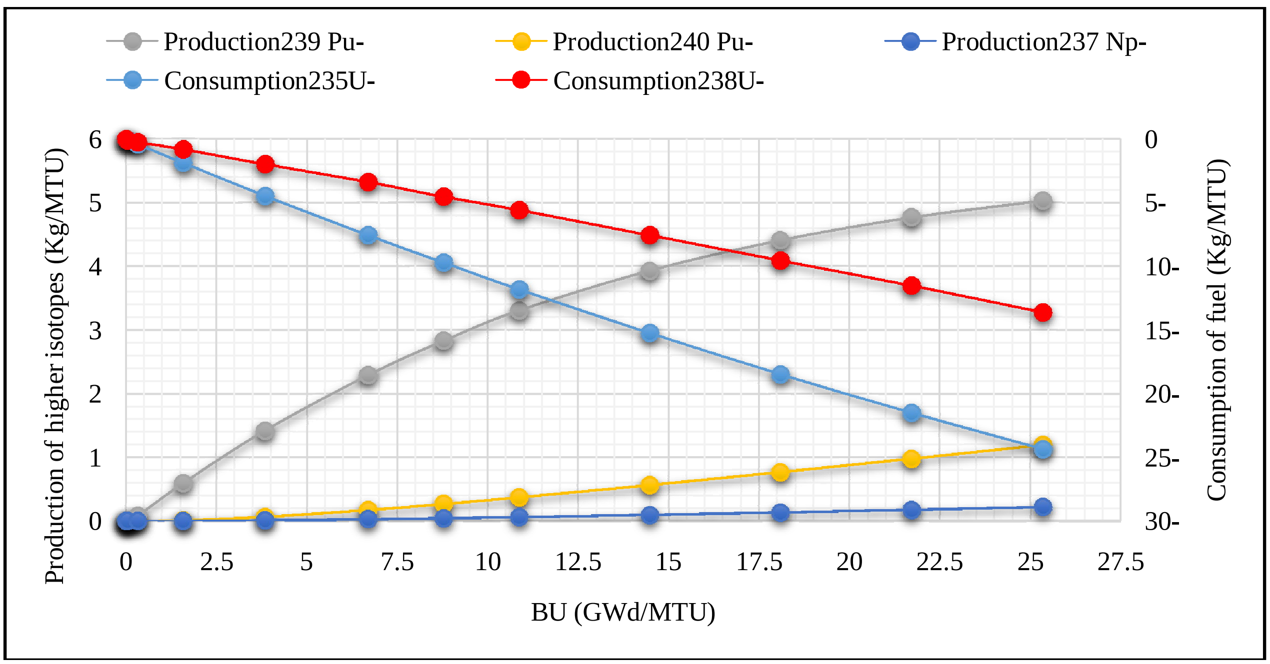

Figure 11: Inventory of major actinides for the modified UO_2Core

Similar to the refernce core, cricticalty control parameter K_{eff} behavair was the same as at the BOC it has an initial value of 1.26948, which is decreased in the beginning due to the neutron absorption in the BAs. in the core as shown in Figure 9.

Due to increasing the operational days to 641 days (about 21 months), the burnup value reaches to 25.3 GWd/MTU.As the core consist of 3 batches which take three cycles of operation to be changed, so the reactor core burn-up value may reach approximately to 75.9 GWd/MTU.

Also, as the core was loaded by 86 ton of heavy metal fuel, Uranium was depleted due to fission process which lead to the buildup of fission products, which are mainly actinides; the major actinides are the depleted U-235 and U-238 and the production of Pu-239 and Pu-240, as shown in Figure 11 which illustrate the consumption of UO_2 and the production of actinides.

5.2. Mixed Oxide Core (MOX)

This modified core consist of three different regions; first region contains MOX fuel (U+Pu) O_2 with 7% Pu- fissile content, second and third regions consist of two different enrichment of UO_2 (4.5 and 4.95) %. In addition to, the core includes the same types and numbers of the PYREX and IFBA burnable absorbers as the reference design core of the AP1000.

The core Criticality, burn-up, cycle length, and radionuclides inventory were calculated for the modified MOX core using MCNPX. The results of the operation burn-up, criticality control and core inventory are shown in Tables [13-15] and Figures [12-16].

EFPD	K_{eff}
0.00E+00	**1.18824**
1.00E+00	**1.16578**
8.00E+00	**1.15715**
4.00E+01	**1.1473**
9.70E+01	**1.14069**
1.69E+02	**1.1346**
2.22E+02	**1.1323**
2.75E+02	**1.13026**
3.67E+02	**1.12416**
4.59E+02	**1.11168**
5.59E+02	**1.08861**
6.59E+02	**1.06163**

Table 13: Cycle Length and criticality for the modified MOX core

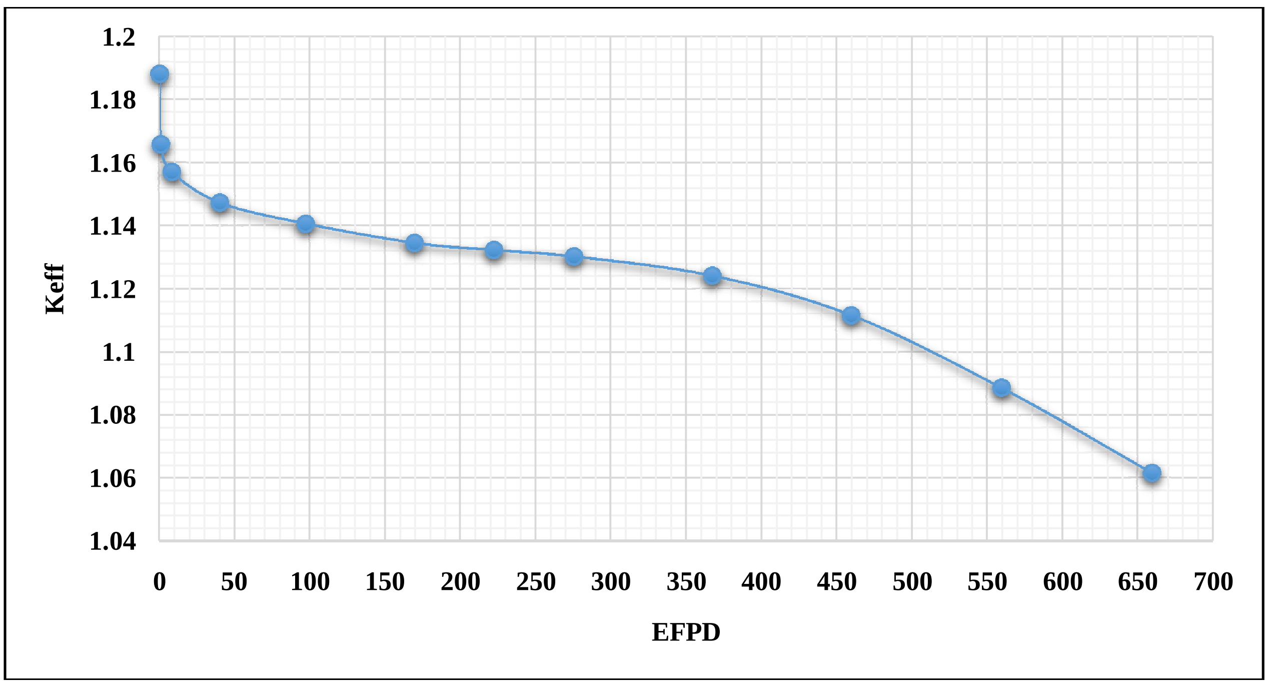

Figure 12: K_{eff} vs Effective Full Power Days (EFPD)

EFPD	Burnup (GWd/MTU)
0.00E+00	**0.00E+00**
1.00E+00	**3.97E-02**
8.00E+00	**3.18E-01**
4.00E+01	**1.59E+00**
9.70E+01	**3.85E+00**
1.69E+02	**6.72E+00**
2.22E+02	**8.82E+00**
2.75E+02	**1.09E+01**
3.67E+02	**1.46E+01**
4.59E+02	**1.82E+01**
5.59E+02	**2.22E+01**
6.59E+02	**2.62E+01**

Table 14: Burnup and operational Cycle Length for the modifiedMOX core

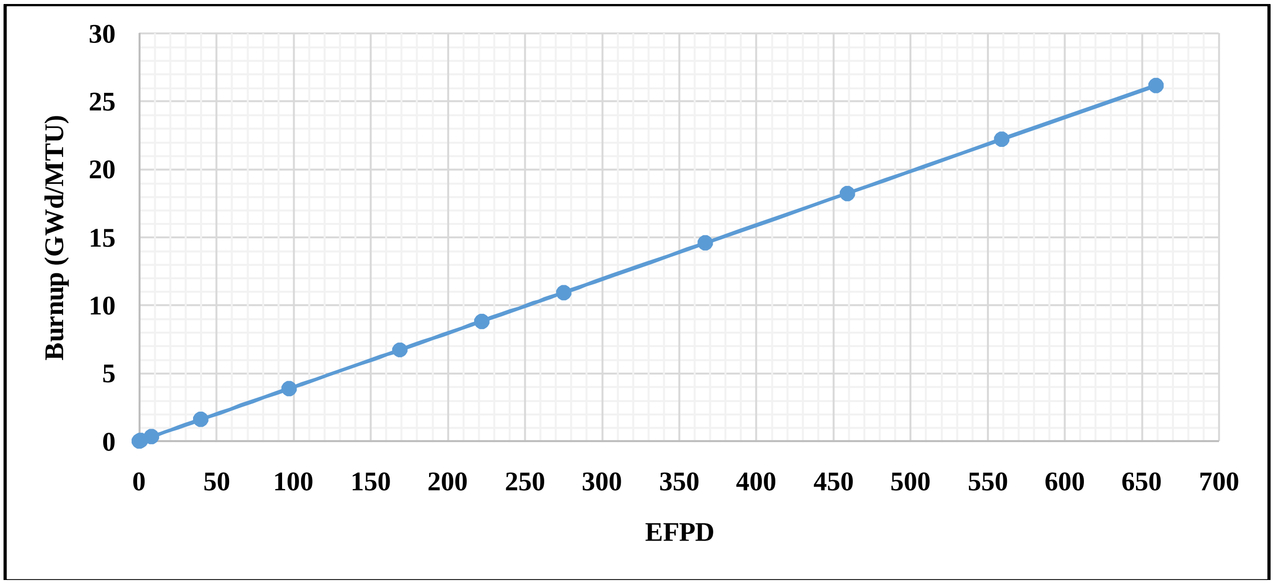

Figure 13:BU vs Effective Full Power Days (EFPD)

Burnup (GWd/MTU)	K_{eff}
0.00E+00	**1.18824**
3.97E-02	**1.16578**
3.18E-01	**1.15715**
1.59E+00	**1.1473**
3.85E+00	**1.14069**
6.72E+00	**1.1346**
8.82E+00	**1.1323**
1.09E+01	**1.13026**
1.46E+01	**1.12416**
1.82E+01	**1.11168**
2.22E+01	**1.08861**
2.62E+01	**1.06163**

Table 15: Burnup against $K_{eff}$for themodified MOX core

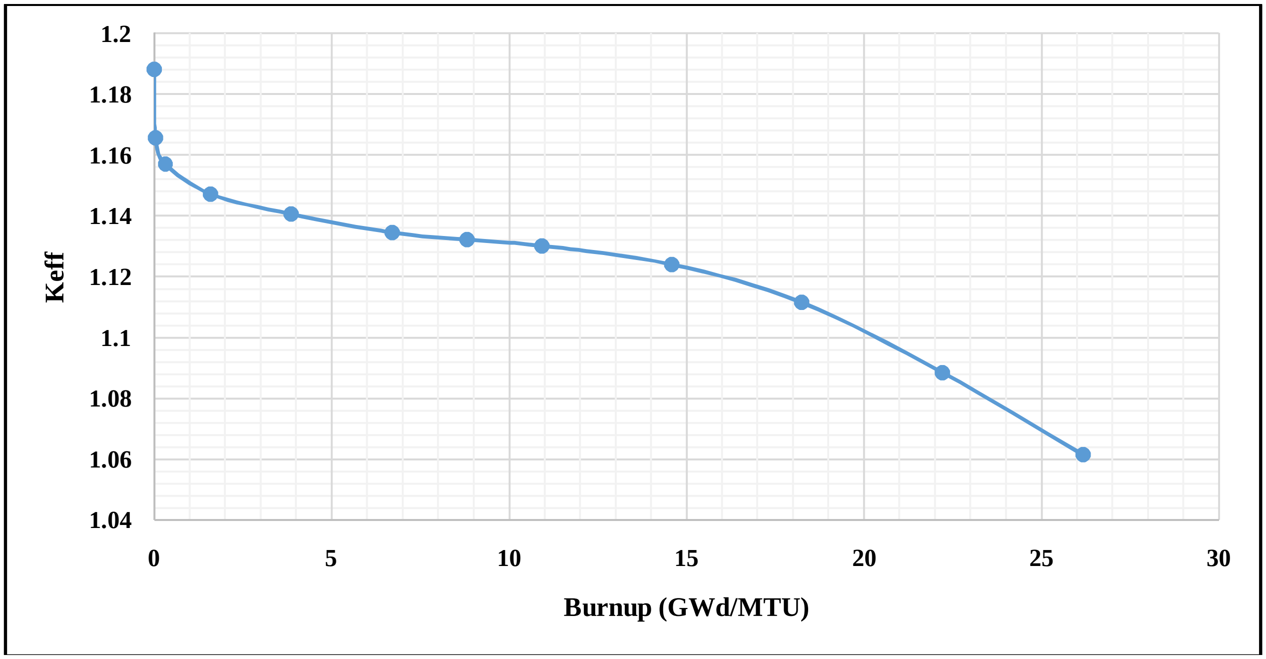

Figure 14: K_{eff} vs BU

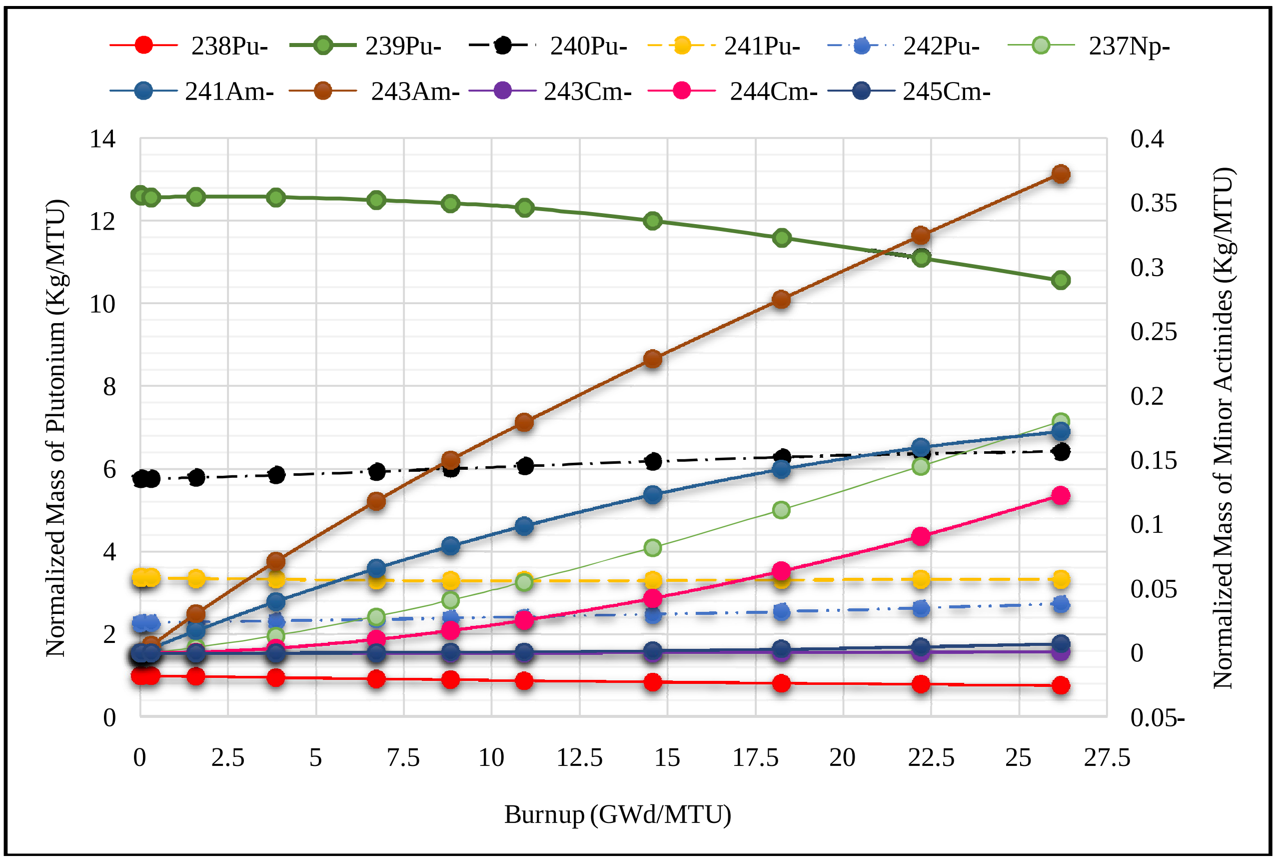

Figure 16: Inventory of major actinides for the modified MOX core

5.3. Comparison between modified UO_2, MOX cores and UO_2 Reference core

The fuel burnup and multiplication factor and the cycle length for the modified cores (UO_2 and MOX) were compared to the reference core. We can noted that the cycle length reached to 21 months for UO_2 core and 22 months for MOX core, where it was 18 months for the references core as shown in Figures [17 and 18].

As the MOX core were more econmic than UO_2 core due to the the cycle length extension and U-content mass loaded in the core at the BOC, and that support the nuclear non-profilration strategy.

As a result of the cycle length extension, BU reached 78.6 GWd/MTU for the MOX discharge core and for UO_2 it reached 75.9 GWD/MTU compared to Westinghouse BU value 60 GWD/MTU for the advanced AP1000 refernce design core.

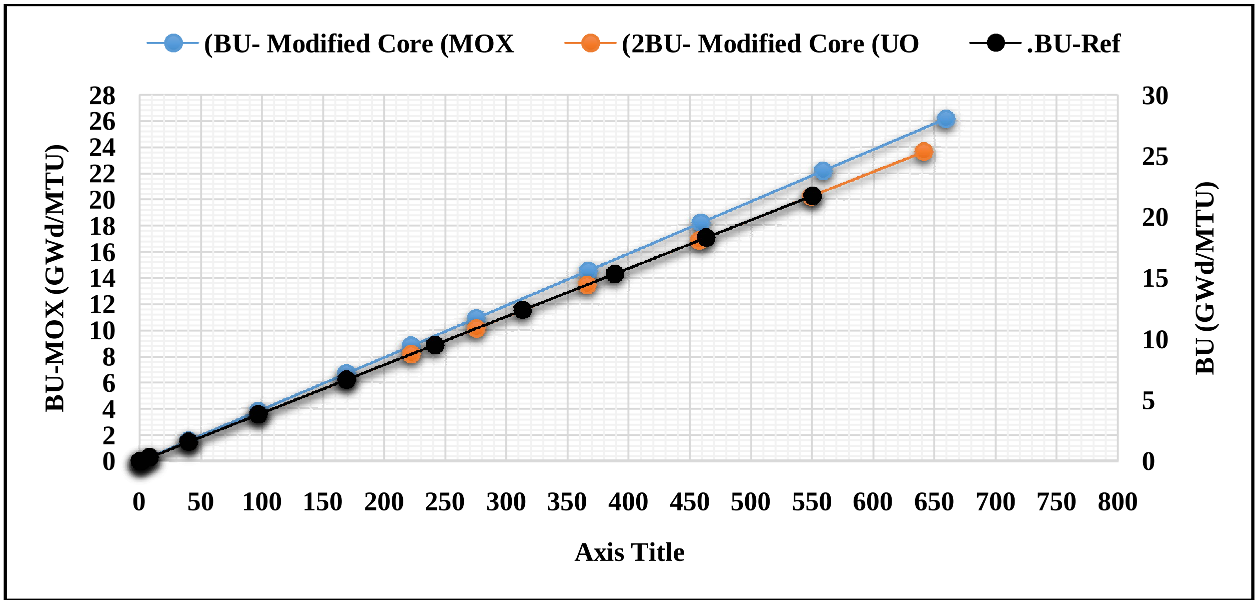

Figure 17: Burnup vs Effective Full Power Days (EFPD)

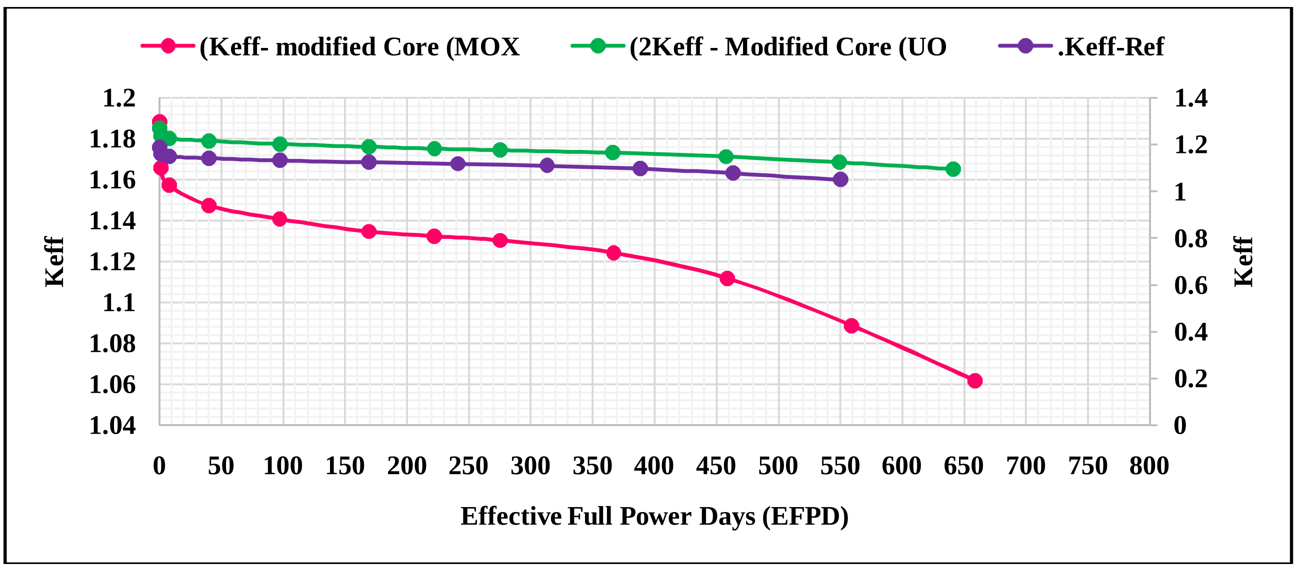

Figure 18: K_{eff}vs Effective Full Power Days (EFPD)

6. Conclusions

Based on the results of this paper we can conclude that;

- AP1000 reactor core was modeled using MCNP 6.1 computer Code. The Code calculated the effective multiplication factor, burn-up and operational cycle length for core configuration loading with and without BAs.
- The MCNP6 criticality results for cold core and clean assembly were validated with the reference data of the reactor and with other published results using SCALE and WIMS computer Codes.The comparison of criticality (K_{eff}) for cold core was 1.204 for our model and 1.205 for the reference design, and for the clean assembly was 1.32728 for our model and 1.328 for the reference design which showed a good agreement.
- The utilization of Burnable Absorbers, as IFBA and PYREXaffect significantly on the critiality and burnup of the initial reference core.
- Theburnup calculations by MCNPX at 18 months was 21.8 GWd/MTU, which reaches the discharge burnup value of 65.4 GWd/MTUthat agreed with the refernce value.
- The validated thermal hydraulic safety parameters as, DNBR , maximum heat flux, mass velocity and linear power density, were in good agreement with the corresponding reference values. These results verify our core neutronic design model for AP1000 at normal operating conditions with maximum reactor power of 3400 MW_{th} and burnup of 60GWD/MTU.

- The validation results using MCNP6.1Computer Code can support our approach for reactor core development and modification safely.
- For the modified core pattern using UO_2 fuel with three different enrichment regions (3.5,4.5,4.95) %, it can reach criticality of 1.06375 at EOC for 21 months.
- The criticality results for cold core and clean assembly of the UO_2 modified core modelled by MCNP6, Keff for cold core was 1.28395 and for the clean assembly was 1.39532.
- For the modified UO_2 core, the burnup calculations by MCNPX-code at 21 months reached 25.3 GWd/MTU per cycle, so in the three cycles it reaches to 75.9 GWD/MTU and preserving the fuel integrity by using ZIRLO cladding material.
- For the modified MOX core, , the burnup calculations by MCNPX-code at 22 months reached 25.6 GWd/MTU per cycle, so in the three cycles it reaches to 76.8 GWD/MTU.

7. References

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