TRANSPORTATION CASK AND CONCRETE MODULE DESIGN FOR MANAGING NUCLEAR SPENT FUEL PRODUCED IN BUSHEHR NUCLEAR POWER PLANT

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Abstract

Bushehr Nuclear Power Plant, commercially operated in Sep 2013, is a Russian type VVER 1000/446; its reactor core consists of 163 fuel assemblies which have 3 up to 4 years' fuel life time. In this pressurized water reactor, removed spent fuel assemblies are transferred into a pool near the core. This pool has been designed to store spent fuels for 9 years and after this period, this pool will reach its maximum storage capacity and spent fuel assemblies will be transferred away from the reactor. The aim of this study was to design a safe and economic approach to manage spent fuel assemblies before the pool reaches to its maximum storage capacity. The study conducted in 3 stages. In the first stage, gamma and neutron flux calculated for spent fuel assemblies have 47000 burnup, 4.2 % enrichment, 4 years fuel life time and 3 and 6 years cooling time by using Origen 2.1. Dose rate and criticality by dint of Origen output were investigated for spent fuel assemblies by means of MCNPX 2.6. In the second stage, the thickness of transportation cask including stainless steel canister and the main body of the carbon steel cask was determined. Based on limitation in casting and fabrication in Iran, twelve fuel assemblies were considered for capacity of a transportation cask. Finally, the concrete module with 36 spent fuel assemblies capacity including 3 canisters and each canister contains 12 assemblies was designed. According to results, the thickness of stainless steel canister and the carbon steel cask were determined 2 cm and 35.5 cm respectively. Dose rates on the surface and 2 meter from the surface of the canister were calculated 5.23E+05 mSv/hr and 8.22E+04 mSv/hr respectively. Neutron multiplication factor was obtained 0.31463 for cask filled by air. Also, the dose rates obtained on the surface of the cask and in 2 meter from cask surface were about 10.21 µSv/h and 4.56 µSv/h respectively. It should be noted that these derived values met transportation regulations approved by Iran Nuclear Regulatory Authority. In addition, maximum dose rates on the surface and 2 meters from the surface of the concrete module were obtained 0.011 mSv/hr and 0.192 mSv/hr for front side (door side) and roof side respectively.

Key words: Bushehr Nuclear Power Plant, transportation cask, concrete module, spent nuclear fuel

1. INTRODUCTION

The spent fuel strategy is affected by the nuclear fuel cycle policies adopted by countries. Two definite nuclear fuel cycles are open or one-through fuel cycle and closed fuel cycle. In open fuel cycle, spent fuel is considered to be HLW. In other side, spent fuel is regarded as a valuable resource and seeks to utilize the resource through re-processing. Some countries have postponed the decision on which strategy to adopt and are taking the 'wait and see' or 'wait and do' approaches. Furthermore, several countries adopted return to the supplier approach based on bilateral agreements [1] [1].

Choosing 'wait and see' approach develops storage options, wet and dry storage, at reactor (AT) and away from reactor (AFR) for spent nuclear fuel management [2, 3, 4] [2, 3, 4]. Dry storage has been considered more than ever after Fukushima Daiichi nuclear disaster [5] [5]. Several studies indicated that dry storage is more cost effective than the wet storage [6, 7] [6, 7]. In parallel to development in storage options, several types of casks, canister based and non-canister based, were progressed and used for transportation of spent nuclear fuels [8, 9] [8, 9].

The Bushehr Nuclear power Plant is a Russian type VVER 1000/446 produced annually about 21 tons of heavy metals. Reactor core of BNPP consists of 163 fuel assemblies which have 3 up to 4 years' fuel life time [10] [10] [10]. In this pressurized water reactor, removed spent fuel assemblies are transferred into a pool adjacent the core. This pool has been designed to the store spent fuel for 9 years and the pool will reach its maximum storage capacity after this period and spent fuel assemblies will be transferred away from the pool.

Concrete modules as a storage system and transportation casks were widely used for spent nuclear fuel management [11, 12] [11, 12]. Thermal analysis in normal conditions and accidental situations were considered for both storage system and transportation cask [13, 14] [13, 14]. Criticality calculation was performed for spent nuclear fuel, transportation casks and storage system in several studies [15, 16] [15, 16]. Several works, also, have been done previously for behaviour of spent fuel in dry storage and concrete storage system [17, 18] [17, 18]. Radiological aspect and shielding calculation were investigated by several authors for spent fuel casks [19, 20] [19, 20].

In this paper, design of integrated system, concrete modular storage system as a primary option for storage of spent nuclear fuels produced in BNPP and canister based transportation cask, was investigated in consideration of 4 years' fuel life time and 3 and 6 years cooling time by using Origen 2.1 and MCNP X 2.6. With regard to limitation in casting and fabrication, capacity of transportation cask was calculated. Based on 44 kW as thermal power, final capacity of each concrete module was designed.

2. MATERIALS AND METHODS

Reactor core of BNPP consists of 163 fuel assemblies which have 3 up to 4 years' fuel life time. Spent fuels are transferred into a pool and stored for 9 years. After this period, this pool will reach its maximum storage capacity and spent fuel assemblies will be transferred away from reactor. The schematic of the fuel assembly and its fundamental features are presented in Fig. 1 and Table 1 [21, 22] [21, 22].

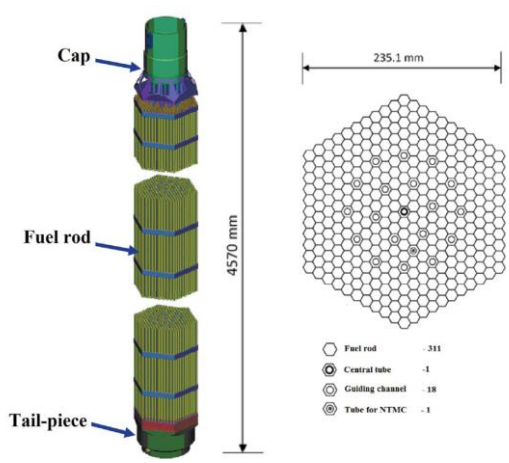

FIG 1. Schematic of Fuel Assembly of BNPP Used in This Work.

TABLE 1. FUNDAMENTAL FEATURES OF FUEL ASSEMBLY USED IN BNPP

Parameter	Value
Fuel Assembly	
Reactor type	VVER 1000/446
Geometry	Hexahedral prism
Maximum dimension of wrench, cm	23.51
Height, m	4.570
Fuel (UO_2) mass, kg	489.8
Fuel density, g/cm^3	10.4 to 10.7
Pith between the fuel rods, mm	12.75
Fuel enrichment	
Assembly type 40	
Average, ^{235}U wt%	4.02
Type 1, ^{235}U wt% (No. of rods)	4.1 (245)
Type 2, ^{235}U wt% (No. of rods)	3.7 (66)
Fuel rods	
Number of fuel rods	311
Clad inside diameter, mm	7.73
Clad outside diameter, mm	9.1
Clad material	Alloy Zr + 1%Nb
Active fuel length, m	3.53

For determining source term, spent fuels consist of 47000 MWD/MTU burnup, 4.2 % enrichment, 4 years' fuel life time and 3 and 6 years cooling time were considered in Origen 2.1. Gamma sources including primary gamma rays from actinides and fission products and secondary gamma rays from (α, n) reaction and neutrons produced from spontaneous fission and (α, n) reaction were obtained from Origen 2.1. Based on Origen outputs, criticality and dose rate were calculated for spent fuel assemblies by means of MCNPX 2.6. Regarding limitations in casting and fabrication in Iran, twelve fuel assemblies were considered for capacity of a transportation cask.

Safety requirements provided in IAEA transport regulations and Iran Nuclear Regulatory Authority (INRA) were used for criticality and dose assessments in designing the transportation cask[23, 24] [23, 24]. Origen 2.1 and MCNP X 2.6 were used as well-known codes for source term determination and criticality and dose assessments respectively [25, 26] [25, 26]. ENDF/B-IIV cross section library was used for shielding calculations. The geometric scheme of transportation cask and canister used in MCNP calculations, are shown in Fig. 2. The main features of transportation cask and canister used in this work are presented in Table 2.

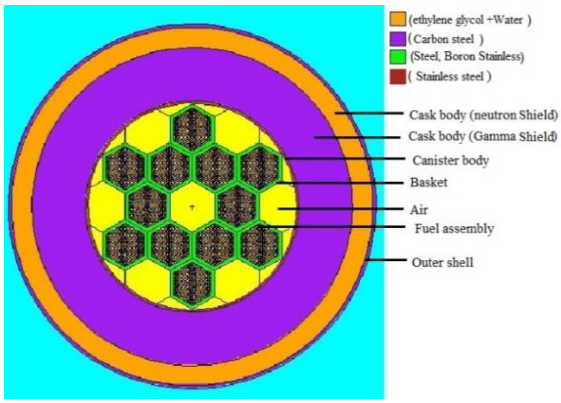

FIG 2. Cross section of designed transportation cask.

TABLE 2. PRINCIPAL SPECIFICATIONS OF CASK AND CANISTER

Component	Value	Component	Value
Basket			
Thickness (cm)	2		
Material	Borated Stainless Steel		
Canister		Cask (inner body)	
Thickness (cm)	1.5	Thickness (cm)	33.5
Inner diameter (cm)	130	Inner diameter (cm)	133
Outer diameter (cm)	133	Outer diameter (cm)	200
Thickness of Top Lid (cm)	25.4	Thickness of Outer Top Lid (cm)	26.6
Thickness of Bottom Lid (cm)	20	Material	Carbon Steel
Material	Stainless Steel		
Neutron absorber		Cask (Outer Shell)	
Thickness (cm)	12.7	Thickness (cm)	2
Inner diameter (cm)	200	Inner diameter (cm)	225.4
Outer diameter (cm)	225.4	Outer diameter (cm)	229.4
Material	Ethylene Glycol (67%)	Material	Carbon Steel

After loading canisters into casks, casks will be transported into the interim storage inside the reactor site. Casks will store in the interim storage on reactor site. After this time, casks will transfer to the generic storage outside the power plant site. Concrete modules were considered in this generic storage. For designing concrete module, international guidelines and regulations were considered [27, 28] [27, 28]. The geometric scheme of concrete module used in MCNP calculations, are shown in Fig. 3. The main features of a concrete module used in this work, are presented in Table 3.

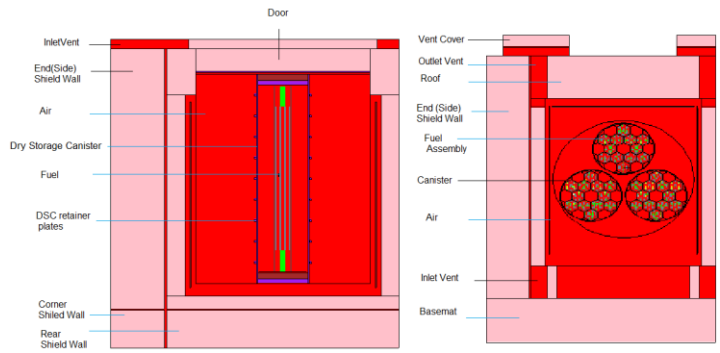

FIG. 3. Cross Sections of storage concrete module.

TABLE 3. MAJOR FEATURES OF CONCRETE MODULE

Item	Value (cm)	Item	Value (cm)
Base/roof width	417	Rear/end (side)/corner shield wall height	564
Base/roof length	652	End (side) shield wall length	297
Base upper side wall thickness	30.5	Corner shield wall width (square)	114.44
Base upper rear wall thickness	30.5	Door inner steel plate thickness	2.5
Roof thickness	135	Door concrete thickness (centerline)	50,67
Rear/end (side) shield wall thickness	114.44		

Concrete module was designed to store 3 canisters including 36 spent fuels containing 6 years cooling time. Dose rates were calculated in 5 directions for concrete module as presented in Fig. 4. The design of the concrete module was completed in accordance with the thermal power consideration and the radiation safety criteria.

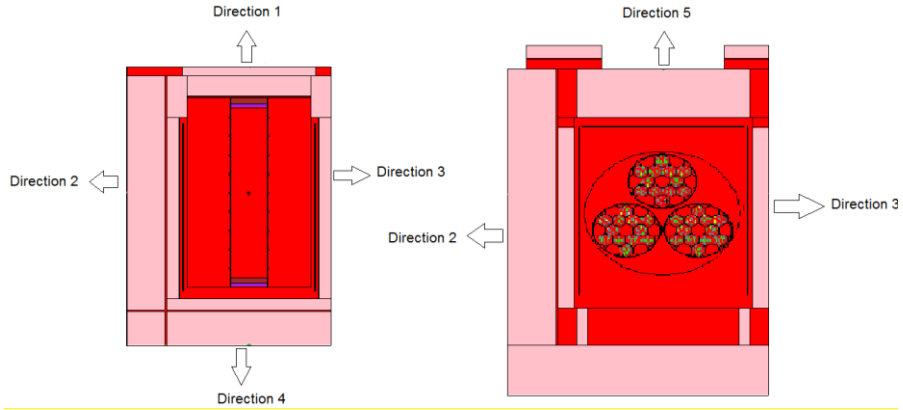

FIG. 4. Directions assumed for dose rate calculations

According to operational procedures of BNPP, spent nuclear fuels will be stored in the reactor pool for 9 years. Considering 3 and 6 years for cooling time is a conservative assumption in this study. The second conservative assumption, in addition, was 47000 MWD/MTU burnup for source term definition.

3. RESULTS

In order to calculating dose rate, F2, F4, and F5 Tallies together with DEn (dose energy) and DFn (dose function) cards were used in MCNP 2.6. ANSI/ANS 6.1.1 was used for flux to dose conversion factors [29] [29]. Relative errors were less than 5 %, 10 % and 10 % for F5, F4, and F2 Tallies respectively. Geometry splitting was considered for variance reduction in MCNP simulation. Mesh Tallies errors are lower than 1% in all calculations.

3.1. Source terms

Source terms were determined by using Origen 2.1. Gamma source terms were derived from two sources: 1. gamma rays came from fission products and actinides and 2. Secondary gamma rays produced in (n, γ) reactions. Gamma rays resulted from Co-60 decay which produced in activation in fuel assemblies' parts were excluded. Majority of energy gamma in source term are less than 4 MeV after 3 years cooling time. Fission products account for more than 80% of photon spectrum. There is a direct relative between gamma and neutron source strength and burnup. Gamma source strength increases if burnup increases. Neutron source terms consist of neutrons from (α, n) reaction and spontaneous fission. The neutron source to the power of four increases with increasing in burnup. Cm-244 has been the most influential radioisotope in neutrons flux. Photon flux spectrum and neutron source due to (α, n) reaction and spontaneous fission after 3 and 6 years cooling time are shown in Table 4, Table 5. Based on Origen outputs, total gamma flux were determined 1.44E+16 photons/sec and 1.04E+16 photons/sec. for 3 and 6 years cooling time respectively.

TABLE 4. PHOTON FLUX SPECTRUM AFTER 3 AND 6 YEARS COOLING TIME FOR 12 FUEL ASSEMBLIES

Energy (MeV)	Gamma Flux (photons/sec)		Energy (MeV)	Gamma Flux (photons/sec)	
	After 3 years	After 6 years		After 3 years	After 6 years
0.01~0.05	4.85E+15	2.1E+15	1.33~1.66	3.03E+14	1.06E+15
0.05~0.10	1.10E+15	4.82E+14	1.66~2.00	1.49E+14	2.22E+14
0.10~0.20	1.14E+15	5.72E+14	2.00~2.50	6.70E+12	6.45E+12
0.20~0.30	9.54E+14	4.12E+14	2.50~3.00	1.94E+13	1.51E+12
0.30~0.40	6.52E+14	2.72E+14	3.00~4.00	1.75E+11	6.93E+10
0.40~0.60	6.83E+14	3.00E+14	4.00~5.00	2.07E+10	8.98E+09
0.60~0.80	5.42E+14	2.21E+14	5.00~6.50	6.18E+04	48270000
0.80~1.00	3.04E+14	1.25E+14	6.50~8.00	7.00E+03	5567000
1.00~1.33	3.70E+15	4.24E+15	8.00~10.0	7.98E+02	639600

TABLE 5. NEUTRON SOURCE DUE TO (α, n) REACTION AND SPONTANEOUS FISSION AFTER 3 AND 6 YEARS COOLING TIME FOR 12 FUEL ASSEMBLIES

	(α, n) reaction			Spontaneous fission	
Radionuclides	per fuel element (neutrons/sec)		Radionuclides	per fuel element (neutrons/sec)	
	After 3 years	After 6 years		After 3 years	After 6 years
PU-238	3.50E+06	3.45E+06	PU-240	1.22E+06	1.22E+06
PU-239	1.16E+05	1.16E+05	PU-242	9.97E+05	9.97E+05
PU-240	2.31E+05	2.32E+05	CM-242	2.54E+06	1.34E+05
AM-241	4.60E+05	6.89E+05	CM-244	1.21E+09	1.12E+09
AM-243	3.30E+04	3.30E+04	CM-246	1.57E+07	1.57E+07

CM-242	5.24E+05	2.77E+04	CF-252	7.23E+06	4.27E+06
CM-243	4.56E+04	4.34E+04			
CM-244	1.01E+07	9.31E+06			

3.2. Thermal power for fuel assembly

Another output from Origen 2.1 was thermal power. According to outputs, thermal power was obtained 2.13 kW and 1.2 kW after 3 and 6 years cooling time respectively. The first value, 2.13 kW, was considered for cask design and second value, 1.2 kW, taken into account for module design. Thermal power for spent fuel assembly after removing from the core is illustrated in Fig. 5.

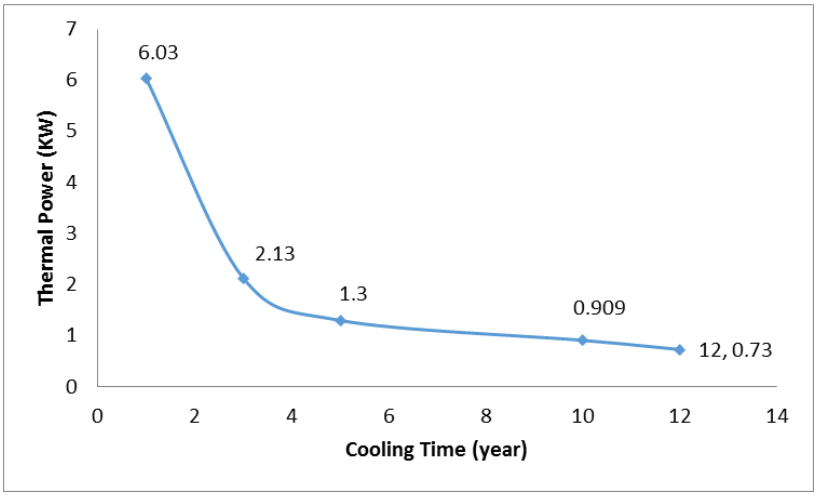

FIG. 5. Thermal power for spent fuel assembly after removing from the core

3.3. Dose rate for canister

Dose rates on the surface and 2 meters from surface of canister containing 12 spent fuel assemblies were quantified 3.95E+08 mSv/hr and 6.24E+07 mSv/hr respectively. Due to radiation protection purposes and satisfying ALARA principle, placing of spent fuels in canister and loading canister into the cask must perform in storage pool of the core. Also, radiation protection considerations should be taken into consideration in the loading canister into the concrete module. Calculated dose rates in different distances and dose distributions for canister are presented in Table 6.

TABLE 6. CALCULATED DOSE RATES IN DIFFERENT DISTANCES FOR CANISTER

Distance	Dose rate (mSv/hr)	error
surface	5.23E+05	0.0008
1 m from the surface	1.74E+05	0.00080
2 m from the surface	8.22E+04	0.00330

3.4. Dose rate and criticality calculation for cask

Several thicknesses were investigated for the cask thickness from 25 cm to 45 cm. The optimum value for thickness of the cask assuming the transportation regulations and limitations in casting and the fabrication, was determined about 33.5 cm for inner body and 2 cm for outer shell of the cask. The calculated dose rates in different distances for the cask are shown in Table 7. Also, variation of the dose rate along the length of the cask, are illustrated in Fig. 6. These results can be used for estimating the working time for workers during the transportation process. As seen in Fig. 6, primary gamma rays have significant contributions in dose. In addition, dose rate increases in middle of the cask because of the centre of the fuel active length located in this section. The neutron multiplication factor and its relative error were obtained about 0.31463 and 0. 019% for cask filled by the air.

TABLE 7. CALCULATED DOSE RATES IN DIFFERENT DISTANCE FOR CASK

dose measurement Location	Gamma dose rate (μSv/h)	Neutron dose rate (μSv/h)	Total dose rate (μSv/h)
Outer surface	10.10	1.08E-01	10.21
2 m from surface	4.54	2.02E-02	4.56

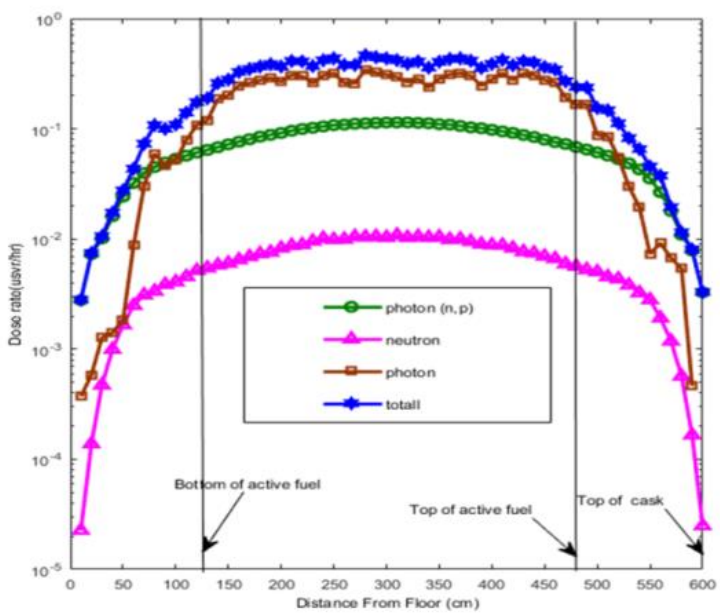

FIG. 6. Dose rate variations along the length of the cask

3.5. Dose rate for concrete module

Concrete modules were designated for storing spent nuclear fuel. Operational considerations such as loading canisters into these modules, monitoring activity and maintenance of module were taken into account in designing processes. Concrete module plays the same shielding role that the cask has got in transportation process. Dose rates based on directions showed in Fig. 4 are presented in Table 8.

TABLE 8. DOSE RATES IN 5 DIRECTIONS IN CONCRETE MODULE

Location	Distance (m)	Dose rate (mSv/hr)	Relative	Dose rate (mSv/hr)	Relative
		Before adding other module		After adding other (new) module	
	On the surface	0.011	0.027		
Direction No. 1	1 m from the surface	0.027	0.013		
	2 m from the surface	0.030	0.010		
	On the surface	0.0007	0.0535		
Direction No. 2	1 m from the surface	0.0005	0.0509		
	2 m from the surface	0.0004	0.0486		
	On the surface	4.3399	0.0016	0.0321	0.025
Direction No. 3	1 m from the surface	2.1914	0.0016	0.0206	0.0237
	2 m from the surface	1.2152	0.0016	0.0154	0.0224
	On the surface	0.00010	0.0996		
Direction No. 4	1 m from the surface	0.00194	0.0307		
	2 m from the surface	0.00421	0.0186		
	On the surface	0.0043	0.0399		
Direction No. 5	1 m from the surface	0.1861	0.0055		
	2 m from the surface	0.1923	0.0044		

According to results, the dose rate in direction No. 3 is more than dose rates in other directions. This problem solved with designing and connecting another concrete module in proximity of the previous module. The second empty module presented in Fig. 7. In addition, after connecting the new concrete module to primary module, dose rate was calculated in the new arrangement. Results were shown in Table 8 and presented that the dose rate reduced in acceptable level.

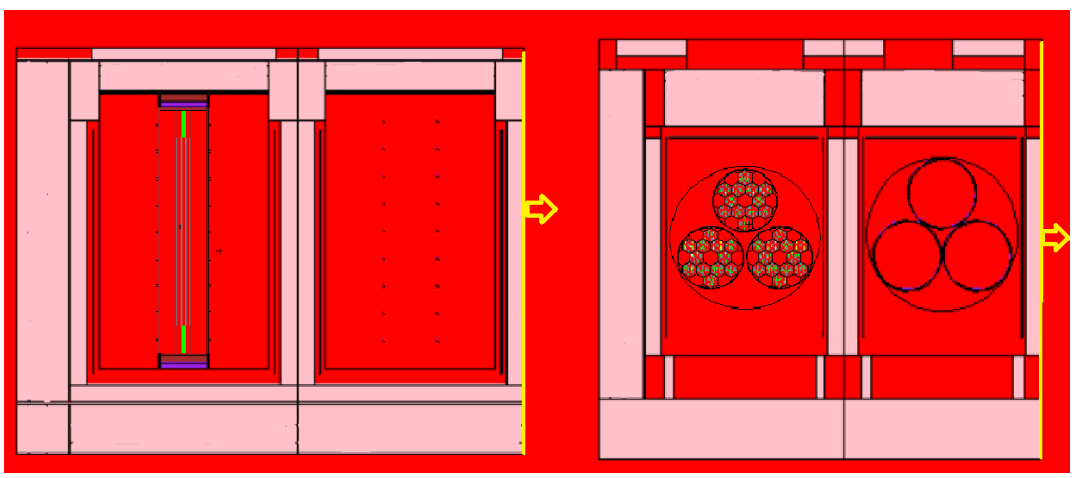

FIG. 7. Connecting additional module to reduce dose rate in direction No. 3 (left picture: YZ plane and right picture: XY plane).

Based on the results shown in Table 8, in directions No. 1, 4, and 5, dose rate increased with increasing distance because of emission of gamma rays from other surfaces and invent and outlet vents. These issues are illustrated in Fig. 8.

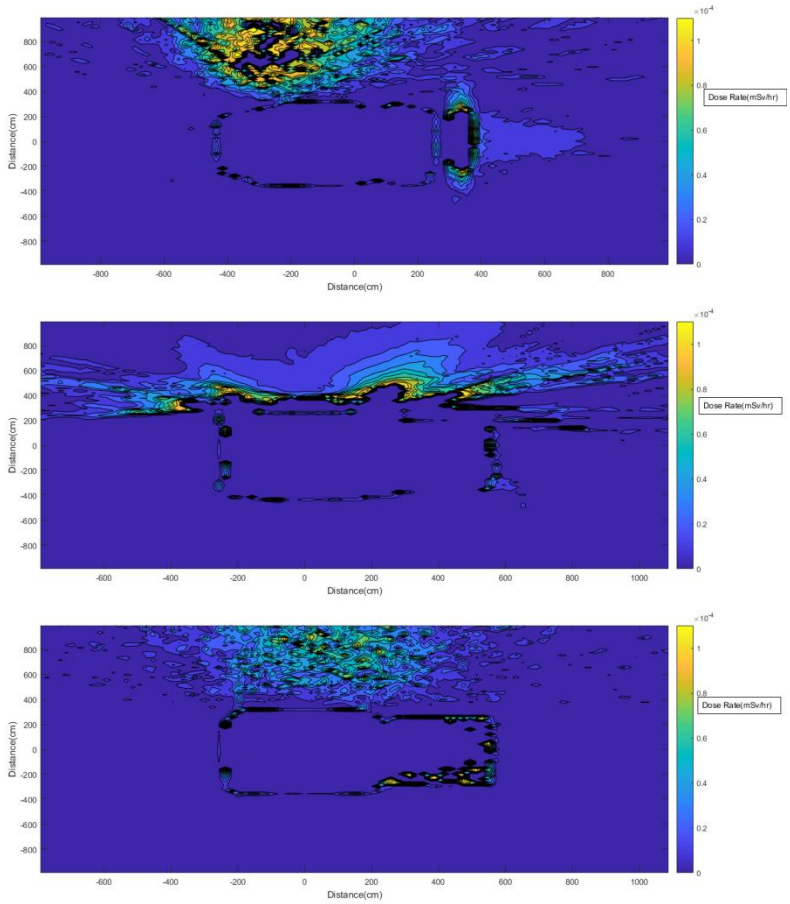

FIG. 8. Dose distribution around concrete module (above: YZ plane, middle: XY plane, bottom: XZ plan).

4. CONCLUSION

In this study, determination of the source term for spent nuclear fuel, designing the canister and cask for transportation, designing the concrete module for storing spent fuel were carried out using Origen 2.1 and MCNPX 2.6. According to these investigations, the thickness of the stainless steel canister was obtained about 2 cm. Dose rate on the canister surface, on 1 meter and 2 meters from the surface and at the top of the canister were gained 3.95E+08 mSv/hr, 1.32E+08 mSv/hr, and 6.24E+07 mSv/hr, and 7.07E+03 mSv/hr respectively. For radiation protection purpose and meet ALARA principles, the placing of spent fuels in canister and loading canister into the cask should be performed in reactor pool.

The thickness of carbon steel cask was obtained about 35.5 cm. The neutron multiplication factor was obtained 0.31463 for the cask filled by the air. The relative errors were less than 10% for all calculations. Also, the total dose rates were determined 10.21 µSv/h, and 4.56 µSv/h in the cask surface and 2 meters from the surface of the cask, respectively. These obtained values are below the criteria approved by INRA for the road route transportation.

Thermal power was obtained about 2.13 kW and 1.2 kW for each spent fuel assembly after 3 and 6 years cooling time respectively. The results of the concrete module simulation have shown that inlet and outlet vents have a crucial of role in the dose rate around the module. In addition, maximum dose rates on the surface and 2 meters from the surface were obtained 0.011 mSv/hr and 0.192 mSv/hr for front side (door side) and for roof side respectively

In continuation of the project, recommended scenarios such as Water spray, Drop test, Stacking test, Penetration test, and finally Thermal test (30 min and 800 C°) will conduct. Other concept of transportation and storage of spent nuclear fuel management, Dual Purpose Cask, will finalize and present in other article. At the end of the project, two independent concepts, Dual Purpose Cask and Transportation Cask and Concrete Module, will compare in technical, economical, and safety aspects.

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