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Investigation of under-containment gamma dose after total core uncovering accident in Tehran Research Reactor

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HIGHLIGHTS

- The happening of total core uncovering is investigated in a typical research reactor.
- Validation experiment and analytical calculations are conducted further using MCNPX2.6.0 and ORIGEN 2.1 codes.
- An emergency make-up tank is designed and located to prevent from fuel damage following core uncovering.

ABSTRACT

The occurrence of core uncovering following a loss of coolant accident is conceivable and should be taken into account for its significant possible consequences. Source terms are calculated using ORIGEN 2.1 code, and the gamma dose of the uncovered core is calculated for three different normal and anticipated accidents scenarios. Under containment gamma dose rates have been calculated analytically as well as using MCNPX 2.6.0 code. The uncovered core of the Tehran research reactor is supposed to operate in nominated power of 5 MW for 30 days. The results illustrated that the under-containment dose rate of gamma in some locations would be about 200 $Sv.h^{-1}$, far from the annual occupational exposure limit of 50 mSv. For preventing this occurrence, it would be possible to use an emergency make-up tank as an engineered safety feature, with functions of the avoidance of damaging fuel after the loss of coolant accident as well as controlling exposure from the core.

KEYWORDS
Research Reactor
Total core uncovering
Gamma dose
Emergency make-up tank

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1 Introduction

It should be ensured that for a research reactor to be built or undergo a significant modification, the highest reasonably achievable safety standards should be met thoroughly to protect people and the environment in the site vicinity. This assurance is provided by the governmental, legal, and regulatory framework. Safety principles intention is the protection of workers, the public, and the environment from the harmful effects of ionizing radiation. Two of these safety principles are the limitation of risks to individuals and prevention of accidents (IAEA, 2012). Providing sufficient cooling in a prolonged shutdown is one of the main operating characteristics. This provides the heat removal possibility and attenuation of the emitted gamma for preventing people from occupational exposure. The pool dewatering leads to direct exposure of the core to air and should be controlled as early as possible. The pool water could be drained in a damaged pool stemming from a severe accident such as a major earthquake, double-ended rupture of primary coolant pipe, or complete shearing off experimental beam tubes. Some accidents causing drainage of the pool water may leave the core partially or fully exposed to air. Such a situation would depend on the static head of water, mode of drainage, size, and form of the hole, as well as its location. In some cases, natural convection cooling by air is sufficient to prevent core damage if the power history before shutdown is not high or sufficient cooling time is provided after shutdown. Covering the core for preventing fuel damage and irradiation hazards is necessary. Furthermore, the pool dewatering and consequently direct exposure of the core to the air could lead to a considerable gamma dose rate in the reactor containment.

The radioactive hazards are highly dependent on the duration between the initiation of the accident and the core uncovering, operating history, and power of the reactor. If the core temperature exceeds the clad melting point, the radioactivity releases from the reactor containment and eventually to the surrounding population.

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Table 1: Some of the main characteristics of the TRR and the equilibrium core. SFE and CFE refer to Standard Fuel Element and Control Fuel Element, respectively. Also, FE is the Fuel Elements, GR is the Graphite box, and IR id Irradiation box.

Quantity	Value	Quantity	Value
Containment volume	15000 m^3	Pool dimensions	580 $\times$610 cm^2
Pool volume	500 m3	Internal diameter of containment	29.92 m
U-235 Enrichment	19.75%	Height form beam tube to upper containment	23.45 m
Plates No. in SFE	19	Containment wall height	18.55 m
Plates No. in CFE	14	Pool wall thickness	1.7 m
U mass in each plate	76 g	Average thermal flux	5.0×10^{13} $cm^{-2}.s^{-1}$
U mass	45.75 kg	FEs No. in core 76	33
1% burnup energy	73.57 MWd	GRs No. in core 76	13
Average burnup	28%	IRs No. in core 76	8
Fuel plate dimensions (cm^3)	$6.7\times0.15\times65.5$	Absorbing material (w/o)	Ag-In-Cd (80,15,5)
Meat dimensions (cm^3)	$6.0\times0.07\times61.5$	Fuel meat (w/o)	U-235, U-238, O, Al (12.45, 49.78, 11.18, 26.59)
Core dimensions (cm^3)	$47\times73\times100$	Grid plate array	6×9
Grid pitch (cm^2)	8.01×7.71	Clad (material, density)	Al-6061, 2.7 $g.cm^{-3}$

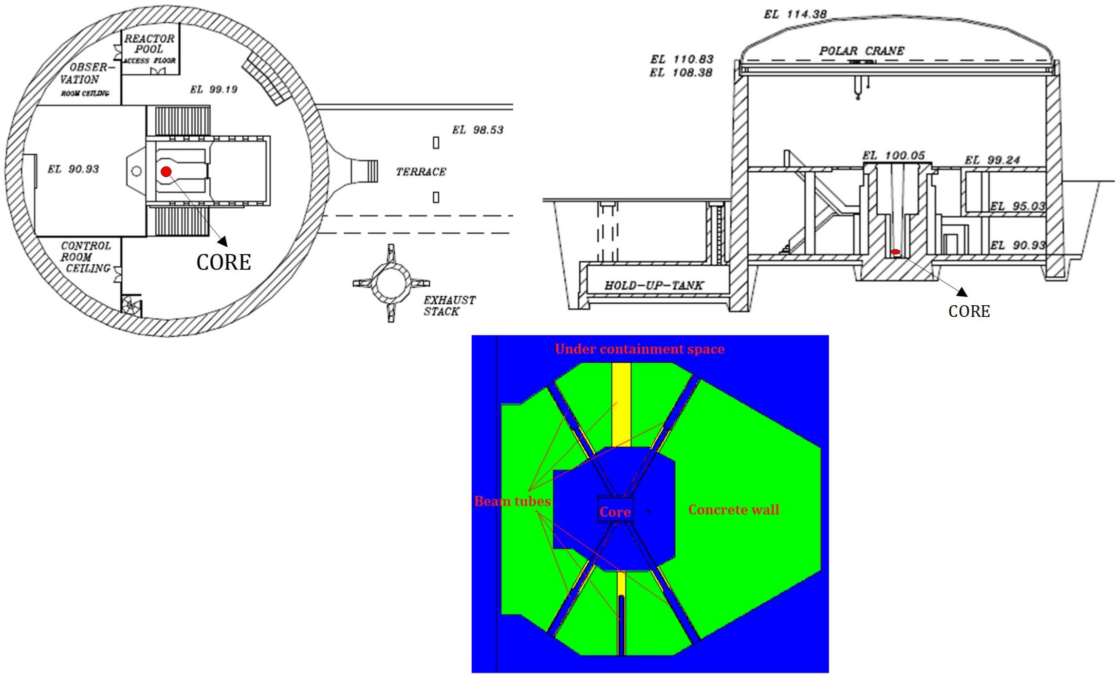

Figure 1: Lateral and top view of the TRR (up) and the core beam tubes (down).

Therefore, it is necessary for demonstration of the ambient air capability for keeping the core temperature below the melting point (Khan et al., 1993).

There are some researches in investigating the core damage phenomenon in some reactors in low power ones as Miniature neutron source up to 10 MW pool-type, but there is not any undamaged core case in those studies. Estimation of core inventory, source term, and dose results under a hypothetical accident in different meteorological conditions is conducted for 1 MW and open-pool reactor of NUR in Algeria (Foudil et al., 2017). TRICO II is a 1MW TRIGA reactor in which source term derivation and radiological safety analysis are done using the Karlsruhe KORIGEN and the HotSpot Health Physics codes for outside the reactor (Muswema et al., 2015). Source term evaluation and atmospheric dispersion modeling for an accidental release from the Pakistan Research Reactor-1 (PARR-1) are investigated for different meteorological and core conditions using ORIGIN and Hotspot codes (Raza and Iqbal, 2005; Ullah et al., 2010). Assessment of the total effective dose equivalent for accidental release from the Tehran Research Reactor (TRR) is conducted focused on the investigation of the contamination of outside the reactor containment for different initial conditions (Anvari and Safarzadeh, 2012; Sadeghi et al., 2013). The radiological dose assessment is done for the hypothetical severe accident in the TRR and corresponding emergency responses in another research mainly investigating dose assessment and related responses in the TRR outside regions (Ahangari et al., 2017). As could be inferred from given researches, the outside contamination was studied as there is no sound investigation about under containment regions. It would illustrate the unavoidable importance of the emergency make-up tank in the research reactor safety considerations. The following procedure is foreseen in this research: Section 2 describes the TRR as one typical Material Testing Reactor (MTR) and our methodology for calculating source terms. Section 3 is devoted to the validation of source and dose rate calculations and our simulation results for proposed scenarios. Finally, the conclusions are given in Section 4.

2 Materials and methods

2.1 MTR type research reactor

One MTR is selected as a case study. It is a swimming pool-type research reactor using plate-type fuel and demineralized light water as coolant and moderator. The generated heat in fuels is removed by natural convection in power levels less than 100 kW and through a forced circulation at higher power levels. The core is reflected by graphite on two sides and light water on the remaining sides. The reactor core is immersed in one section of the pool that contains beam tubes and other experimental facilities, which is called the stall pool. The other section of the pool is an open area for bulk irradiation and is called the open pool. The reactor can be operated at full power in either section which is separated by a concrete wall having an opening that can be opened or closed by a watertight aluminum gate. Some of the main characteristics of the TRR with the equilibrium core No. 76, which is used in this research are given in Table 1 (AEOI, 2018, 2015).

Perspectives of the TRR core, containment, outside, and beam tubes are shown in Fig. 1. As could be inferred from this figure, large regions of the under containment are exposed to the core radiation. The value of dose rate is highly dependent on source strength and the other accident-established circumstances. Furthermore, being filled or dewatered the indicated pool and beam tubes are one of the main parameters determining the accident severity. The operating time of 30 days is given as the reactor history before the total core uncovering. As large as operating in nominal power before the total core uncovering, there would be a higher dose rate of gamma. There must be a delay time between the reactor shutdown and the core uncovering to the removal of residual heat and assuring from the core intactness. According to a late investigation on the Loss of Coolant Accident (LOCA) in this reactor, one day was assumed as a delay time for conservation in this study (Boustani and Khakshournia, 2020).

2.2 The calculation method

This investigation methodology is simultaneous using of analytical method, simulation, and experimental measurements for reliably determining the under-containment gamma dose. Three scenarios are studied for normal and after the complete LOCA as the accident scenarios involve dewatered and full of water beam tubes. An introduction of the TRR is done at first. In the next step, the validation and calculation of source terms are done through comparison with the Safety Analysis Report (SAR) and using ORIGEN 2.1 code (Bell, 1973). The validation of dose calculations is done using the experimental measurement and the comparison with the calculation and the analytical results. The under-containment dose of determined points is calculated using MCNPX 2.6.0 code in the last step (Pelowitz et al., 2005).

The ORIGEN 2.1 code is a versatile point burnup and a decay tool that solves decay and growth equations for numerous isotopes for arbitrary coupling. This code uses the matrix exponential method and first-order, linear, and coupled ordinary differential equations with the constant coefficient for solving large systems. The general nature of the exponential method allows for the investigation of complex decay and transmutation schemes. An extensive library of nuclear information has been collected, including half-lives, decay schemes, neutron absorption cross-sections, fission yields, disintegration energies, and multigroup photon release data. This code is used to calculate the compositions and radioactivity of fission products, cladding, and fuel materials in some types of reactors, such as light water reactors (Bell, 1973).

The output of the ORIGEN code is used as the MCNPX 2.6.0 code input to compute the under-containment dose rate. The MCNPX is a Monte Carlo transport code for tracking numerous particles in extensive energy and complicated geometries. The ENDF/B-IV and MCPLIB libraries are used for neutron and photon computations, respectively. The DFn card is used for the conversion coefficient from flux to dose rate (Pelowitz et al., 2005). It is worth mentioning that the MCNPX 2.6.0 code relative error is controlled by the number of source histories on the NPS card. This number is increased to $1.90E+9$ that giving rise to less than 1.0% relative error in calculated dose rates.

3 Results and discussion

The source terms determination includes the calculation of Fission Products (FPs), Actinides and daughters (ADs), and Activation Products (APs) terms. The source terms are written for the average values of emitted photons in 18 groups from $1.0E-2$ to 9.5 MeV. At first, one validation is conducted using the SAR data of the TRR. Then, the source terms were computed considering the burnup of each fuel element operated in 5 MW power for 30 days following 1-day cooling.

3.1 Source term validation

The source terms and produced radionuclides of one equilibrium core involving 28 SFEs with 295 days operation at 5 MW which is equivalent to 40000 MWD per ton uranium are computed and given in SAR of the TRR reactor (AEOI, 2018). As the reactor has been operated for a long time, the produced radionuclides have a high activity and are in category I of sources classification according to the IAEA document (IAEA, 2003). These terms are recalculated for the validation presented in Table 2 for a better comparison. Due to good agreement of calculation and

Table 2: Source terms and some radionuclides' activities for equilibrium core.

	Source term /Radionuclide	Activity (Ci) Calculation	(AEOI, 2018)	Relative difference
Source terms	Activation Products (APs)	3.03e+05	3.00e+05	1.00%
	Actinides and Daughters (ADs)	8.11e+05	8.00e+05	1.31%
	Fission Products (FPs)	2.36e+07	2.37e+07	-0.59%
Radionuclides	Kr-87	1.02e+05	1.03e+05	-1.07%
	Sr-91	2.32e+05	2.34e+05	-0.77%
	Zr-95	2.54e+05	2.53e+07	0.47%
	Mo-99	2.44e+05	2.47e+05	-1.21%
	Tc-99m	2.14e+05	2.16e+05	-1.11%
	Xe-135	7.81e+04	7.63e+04	0.02%
	I-131	1.20e+05	1.20e+05	0.01%
	I-136	1.24e+05	1.25e+05	-0.01%

the refereed data in Table 2, it is concluded that the obtained data is reliable and this method can be appropriate for conducting this research.

3.2 Gamma source

Our concentration in this research is on gamma source, whereas the neutron numbers 1 second after the core uncovering decrease abruptly to zero from an initial value of 10^{13} as seen from the given Eq. (1) (Lamarsh et al., 2001):

$$T \equiv \frac{l_p}{k_\infty - 1} \quad , \quad n = n_0 e^{\left(\frac{t}{T}\right)}$$
$$\rightarrow n = 10^{13} \times e^{-\left(\frac{1}{56 \times 10^{-6}}\right)} = \frac{10^{13}}{e^{17.88\times 10^3}} \approx 0 \tag{1}$$

The prompt neutron lifetime, l_p, in this reactor is about 45 μs. The infinite multiplication factor, is calculated by MCNPX 2.6.0 code, and the reactor period is according to Eq. (1).

The gamma sources are radiations from the produced source terms in the reactor core. The source terms are the result of the core elements activation, mainly SFEs and CFEs, through neutron irradiation during operation. This is straightforward to have a more intensive gamma source the higher the reactor operation and power are.

The calculation of the core inventory using the ORIGEN code is performed knowing the reactor history, operating powers, and also the composition of the fuel elements and coolant. It is worth mentioning that this code is independent of geometry, unlike MCNP code. Also, the possibility of reporting radionuclides mass, radionuclides activity and special activity, power, and neutron flux exist in the output file of this code.

The gamma spectra of fuels with different amounts of burnup are calculated. As the APs have a short lifetime, the activity of this group is smaller than to other two groups and then can be ignored from the survey. The Actinides and Daughters spectra as photon release rate per energy for different burnup are calculated using ORIGEN and shown in Fig. 2. As can be seen, the number of photon sources distinctly decreased for energies upper than 0.5 MeV.

The spectra of FPs for the different amounts of burnup are given for the released photon rate per energy in Fig. 3. As can be seen from Fig. 2 and Fig. 3, the gamma spectrum of the FPs has more significant amounts of the released photon in more extent of energy than the ADs. More considerable amounts in energies less than 0.575 MeV and 1.75 MeV for ADs and FPs are explicit, respectively. The total activities of FPs and ADs are given in Table 3 for a more accurate survey.

As Table 3 presents, the FPs activities are several times higher than the ADs leading to more quantities of gamma dose rate in comparison with the ADs.

As a typical study, the long-term variation of FPs and ADs for 442 days operation in 5 MW following 158 days after shut down for a 30% SFE is calculated and given in Fig. 4.

According to Fig. 5, the amount of FPs activity in the operating time (from 0 to 442 days) as well as after the reactor shutdown abruptly decreases and is several times higher than ADs. The difference between the activities of FPs and ADs increased as well as the increment of burnup. Decrease of the ADs activity to the final value happens suddenly, although the FPs activity fell abruptly after the reactor shutdown and with a lower slope for additional times.

Table 3: Total activity of FPs and ADs for all fuel elements.

Burnup (%)	Activity (Ci) ADs	FPs
0	5.56e+14	3.57e+15
5	1.20e+15	8.33e+15
10	9.64e+15	4.52e+15
15	1.99e+15	1.33e+16
20	5.54e+15	1.12e+16
25	2.30e+15	1.34e+16
30	2.89e+15	1.57e+16
35	4.00e+15	2.01e+16
40	3.36e+15	1.56e+16
45	3.11e+15	1.33e+16
50	3.38e+15	1.32e+16
55	3.67e+15	1.31e+16
Sum	4.16e16	1.45e17

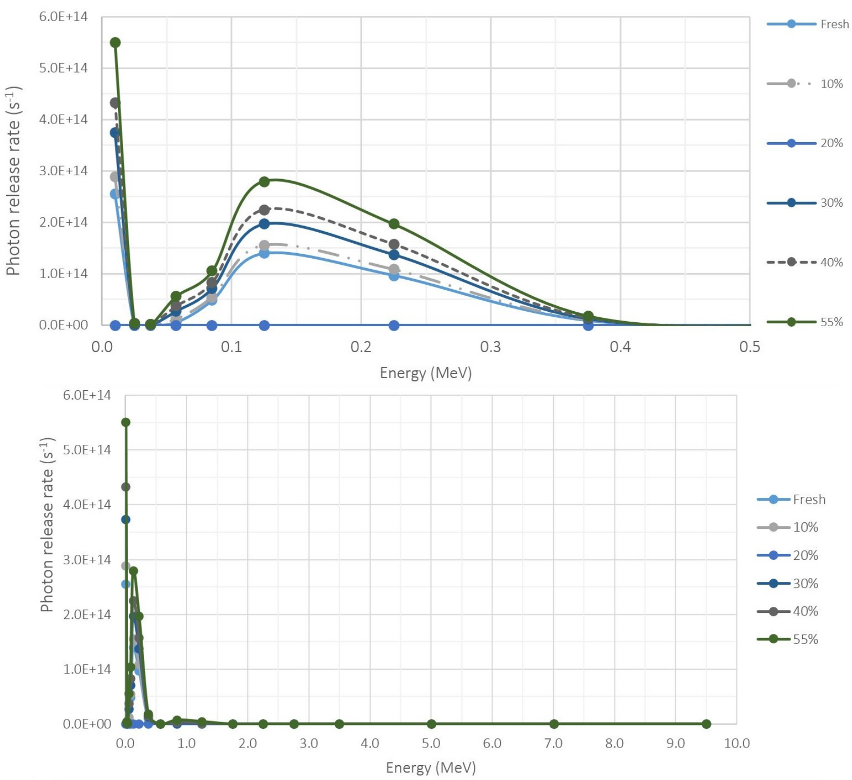

Figure 2: Actinides and Daughters spectrum for lowe (up) and all (down) energies.

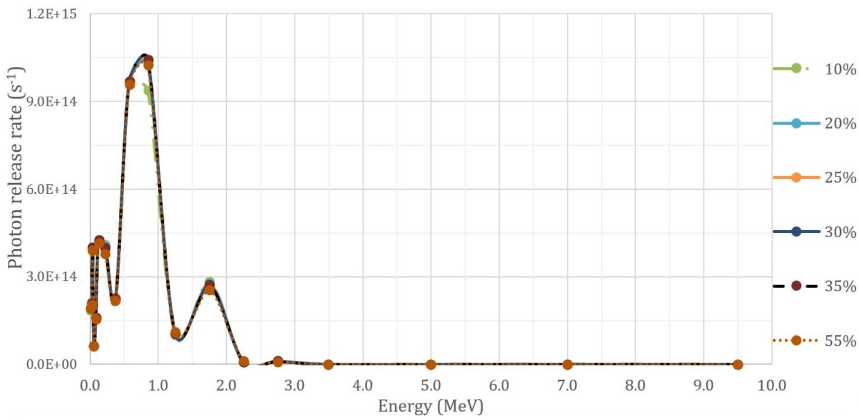

Figure 3: Fission products spectrum for different fuels.

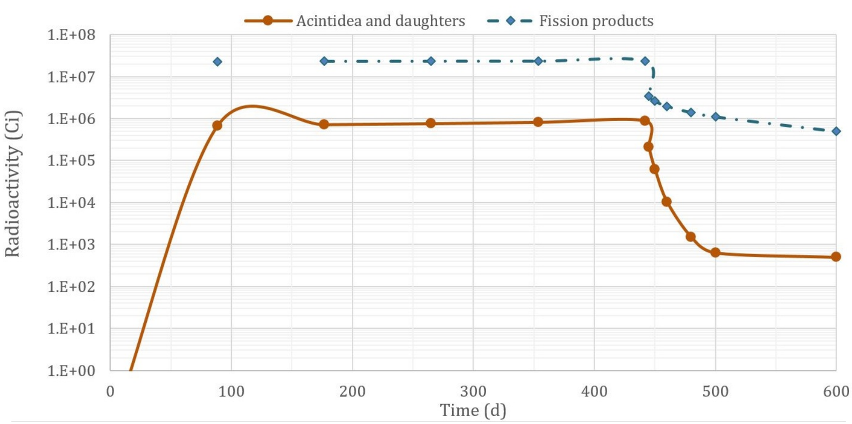

Figure 4: Activity of FPs and ADs for a 30% burnup fuel.

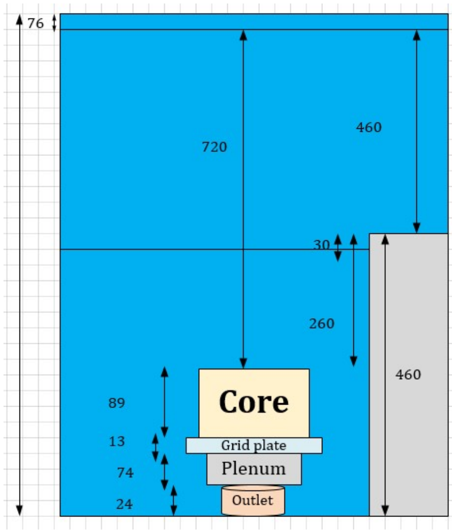

Figure 5: TRR core configuration during validation experiment (The given numbers are in cm).

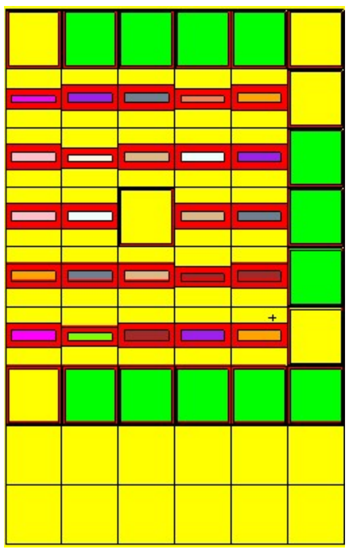

Figure 6: Simulated compact core No. 1 with the MCNPX code.

Table 4: Total activity of all fuel elements in the compact core No. 1.

SFE		CFE	
Burnup (%)	(photon/s)	Burnup (%)	(photon/s)
0	6.11e+13	1	9.97e+13
5	5.00e+14	9	1.44e+14
10	2.78e+14	18	1.39e+14
15	5.93e+14	27	1.31e+14
20	4.03e+14	35	2.04e+14
25	6.41e+14		
30	5.10e+14		
35	8.47e+14		

Table 5: Calculated and measured gamma dose rate in the compact core No. 1.

Water height (cm)	Measurement height (cm)	Dose rate (μSv.h^{-1})	
		Measurement	Calculation
230	400	20 $\pm$ 1	3.46 $\pm$ 0.51

3.3 Validation of dose rate calculation

The TRR core is being on a grid plate that the height of its upper water in a full pool is 7.2 m. The pool water is drained to 230 cm above the core for the validation experiment, as shown in Fig. 5.

The compact core No. 1 comprises 19 SFEs, 4 CFEs, 1 RR (Regulating Rod), 12 GRs, and 6 IRs. The RR is a regulating rod containing two plates made of stainless steel and used for fine adjustments of power in the core. According to Log book No. 50 of the TRR, this core is operated for 550 MWh with 4 MW power following 50 days shutdown (AEOI, 2020). The gamma spectrum of each fuel element is calculated in the averaging 18 energy groups and given in Table 4.

For simulation of the core, the amount of meat and clad of all fuel elements are calculated. Then, each fuel element is simulated in the form of a meat slab and one peripheral aluminum clad as shown in Fig. 6. The green and yellow boxes are GR and IR boxes, respectively. The other boxes, including two tangled rectangles, are the SFEs and CFEs.

The exact determination of upper core water level is not possible, unlike simplicity in simulation. Therefore, two simulations with a few differences in the height of water are conducted for indication of the water height effect in the calculated dose value and also the possibility of a better comparison with the measured dose. The measurements are performed by health physics experts of the TRR using LB 123 UMo dosimeters of the Berthlod Company. The LB123 consists of a battery-driven data logger unit, where one of seven available radiation detectors could be connected. High voltage supplies, preamplifier, and discriminators are integrated into the probes (Operating-Manual, 1993).

The calculated and measured dose rate at 370 cm above the upper level of the core for two different water heights are given in Table 5.

As could be seen from Table 5, there is some difference between the calculated and experimental results. Simulation of the sub-merged core is conducted with approximate level of the pool water (1 to 5 cm error) as there was not accessible tool for precise determination of the water level. Furthermore, radiations from core structures, walls, pool water, depleted fuels in another pool, and in-containment air particulates were not considered in simulation due to complexity and inability of the MCNPX code. Also, it is worth mentioning that the background radiation of gamma in the upper region of the reactor pool could be up to 20 μSv.h^{-1} even in shutdown condition. Therefore, having a more accurate result needing more deep investigation and time which could be down as another research work and is beyond the scope of this study. Finally, considering the above-mentioned arguments, the calculated dose rate of 3.46 μSv.h^{-1} for 20 μSv.h^{-1} measurement seems rationale. According to the conducted simulation and comparison with experimental measurement, it appears that the simulations are reliable and could be used for the calculation of under-containment dose.

Table 6: Absorbed dose rate for one fresh SFE.

Average energy (MeV)	Photons (s^{-1})	Photons density at 500 cm ($cm^{-2}.s^{-1}$)	$(\frac{\mu_a}{\rho})^{air}$	$\dot{X}$ ($mR.h^{-1}$)	$\dot{X}_R$ ($Gy.h^{-1}$)
0.010	4.08E+14	1.30E+08	0.0233	2.00E+03	0.02
0.025	1.90E+14	6.03E+07	0.0233	2.32E+03	0.02
0.037	3.46E+14	1.10E+08	0.0233	6.34E+03	0.06
0.057	6.08E+13	1.93E+07	0.0233	1.71E+03	0.01
0.085	1.98E+14	6.29E+07	0.0251	8.84E+03	0.08
0.125	4.70E+14	1.49E+08	0.0251	3.09E+04	0.27
0.225	5.03E+14	1.60E+08	0.0268	6.37E+04	0.56
0.375	2.09E+14	6.64E+07	0.0288	4.72E+04	0.41
0.575	8.34E+14	2.66E+08	0.0296	2.98E+05	2.61
0.850	5.70E+14	1.82E+08	0.0289	2.94E+05	2.58
1.250	9.64E+13	3.07E+07	0.0268	6.77E+04	0.59
1.750	2.22E+14	7.06E+07	0.0238	1.94E+05	1.70
2.250	6.98E+12	2.22E+06	0.0238	7.84E+03	0.07
2.750	8.01E+12	2.55E+06	0.0211	9.75E+03	0.09
3.500	6.94E+10	2.21E+04	0.0211	1.07E+02	0.00
5.000	8.78E+08	2.80E+02	0.0181	1.67E+00	0.00
7.000	1.31E-01	4.16E-08	0.0172	3.30E-10	0.00
9.500	1.49E-02	4.74E-09	0.0160	4.75E-11	0.00
		Total dose rate ($Gy.h^{-1}$)			9.07

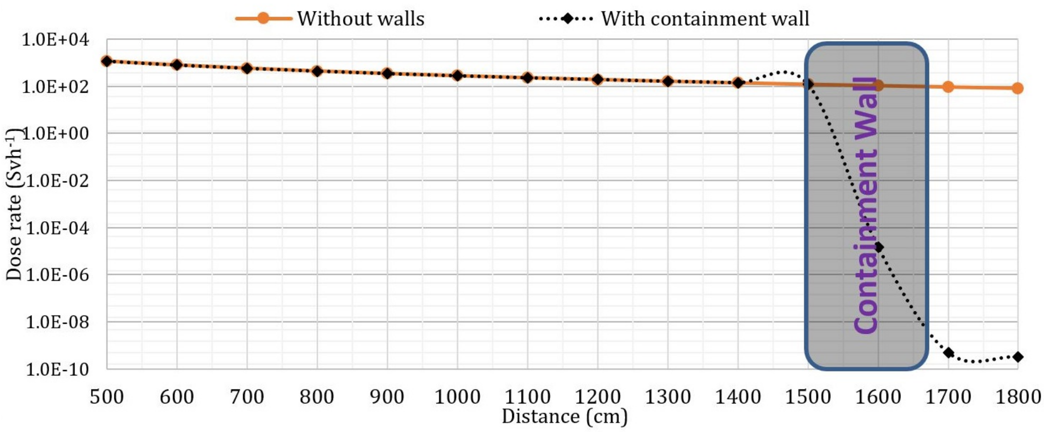

Figure 7: Dose rate inside and outside of the containment.

3.4 Dose rate calculations

The dose rate calculation is conducted for two general areas depending on the studied areas. One area is in the direct radiation of the core, such as the upper core bridge, entrance door, or front regions of the beam tubes. Another area is without direct radiation, such as the Beam Hole Floor (BHF), in which the concrete pool wall prevents direct gamma radiation to this area. Furthermore, the calculation of the outside reactor dose rate is necessary due to the presence of the population.

The reactor containment is a right cylinder domed with reinforced concrete exterior walls, and a steel-domed roof that paly the main role in prevention from passing gamma to the out of containment. It is windowless and has 1.80 m thick from elevation 90.43 to 96.48 m from where it tapers to 1.20 m at elevation 108.98 m at which point it reduces to 0.85 m for connection to the steel dome (AEOI, 2018).

3.4.1 Dose rate calculation with considering no wall for pool

a. Simulation

Two conditions are considered for the calculation of the direct dose rate. In the first, the gamma source is considered in origin for the calculation of the dose rate at different distances. In the second, one concrete wall with 185 cm thickness is supposed at a distance of 1500 cm to include the reactor wall effect in calculating the outside containment dose rate. The dose rate of inside and outside containment is calculated and given in Fig. 7.

As can be found from Fig. 7, the direct radiation dose rate is in the range of more than 100 $Sv.h^{-1}$. This caused to exceed from allowable absorbed dose value in less than some seconds leading to considerable radiation damages. Accordingly, preventing exposure of personnel from the direct radiation of the core in these parts is very important.

Table 7: Absorbed dose rate for the compact core No. 1.

Burnup (%)	$\dot{X}_R$ (Gy.h^{-1})	SFEs NO.	CFEs No.	$\dot{X}_R$ (Gy.h^{-1})
0	9.1	1	0	9.1
5	22.3	2	0	44.6
10	32.4	2	1	97.2
15	36.2	3	0	108.6
20	33.0	1	1	66
25	36.9	3	0	110.7
30	36.8	2	1	110.4
35	46.3	3	1	185.2
40	34.0	2	1	102
45	36.9	3	0	110.7
50	36.9	3	0	110.7
55	33.9	3	0	101.7
Total dose rate (Gy.h^{-1})				1156.9

b. Analytical method

For assurance, the dose rate calculation is conducted analytically for a point 500 cm away from the bare core as an origin for the equilibrium core No. 76. The exposure rate for a gamma source with energy E (MeV), the intensity I (cm^{-2}.s^{-1}), and mass attenuation coefficient μ/ρ (g.cm^{-2}) is given by Eq. (2) (Lamarsh et al., 2001).

$$\dot{X} = 0.0659\, I\, E\, \Big(\frac{\mu_a}{\rho}\Big)^{\text{air}} \quad (\text{mR.h}^{-1}) \tag{2}$$

Then, the exposure rate at distance R from the source is given by Eq. (3):

$$\dot{X}_R = 0.0659\, \Big(\frac{I}{4\pi R^2}\Big)\, E\, \Big(\frac{\mu_a}{\rho}\Big)^{\text{air}} \quad (\text{mR.h}^{-1}) \tag{3}$$

The mass attenuation coefficients exist in different references, such as (Lamarsh et al., 2001). The approximate equivalence of Gray and Sievert are considered and also the relation between Roentgen, Rem, and Gray are used. After this, the absorbed dose rate for one fresh SFE is calculated as an example in averaging the 18 energy groups and given in Table 6.

As could be seen from the calculations of Table 6, the dose rate in 500 cm from a fresh SFE is approximately 9.1 Sv.h^{-1}. The calculation of the dose rates for fuels of the equilibrium core No. 76 is conducted and given in Table 7. As could be seen from this Table, the analytically calculated dose rate is 1157 Gy.h^{-1} which has good accordance with the given simulation result of 1160 Sv.h^{-1} in Fig. 7. The difference is less than 0.3% infers the reliability of our simulation for dose rate calculations.

3.4.2 Dose rate calculation with considering wall for pool

The under-containment dose rate is calculated for three different scenarios of some selected points, as an example shown in Fig. 8. This is due to the extent of the studied area and the impossibility of calculation for all spaces.

As the considered height for all calculations is 60 cm, this parameter is deleted from the location in all given tables. The dose limits for protective actions from references are given in Table 8 (Manual, 2013; IAEA, 2011).

Three different scenarios for under-containment dose rate are addressed in the following depending on the different conditions of the pool and the beam tubes.

Scenario 1. The pool and beam tubes are full of water

There is no gamma leakage from the core to the under-containment region due to the full water of the beam tubes and the completely submerged core. The calculation results are given in Table 9.

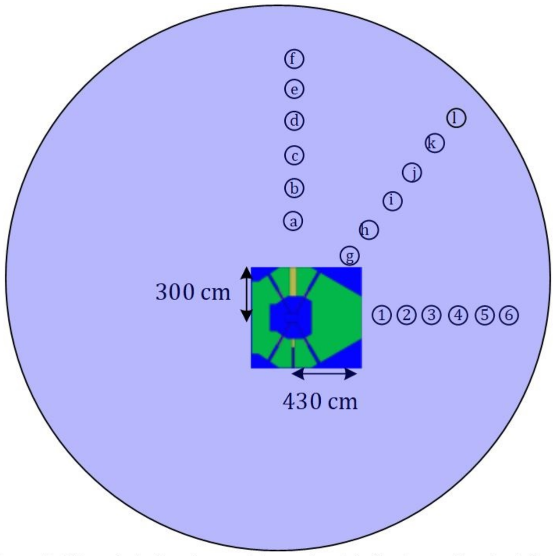

Figure 8: The selected under containment points for dose rate calculation.

Table 8: Dose limits for protective actions.

Dose limits (mSv)	Protective actions
10 to 50	Exit or sheltering
$\geq$ 50	Using Iodine tablet

Table 9: Calculated gamma dose rate in the compact core No. 1 for scenario 1.

Point	Location (x,y) (cm, cm)	Dose rate (μSv.h^{-1})
1	(500, 0)	5.1
2	(650, 0)	5.3
3	(800, 0)	4.0
4	(950,0)	2.9
5	(1100, 0)	2.3
6	(1250, 0)	6.1
a	(0, 500)	40.7
b	(0, 650)	31.5
c	(0, 800)	23.3
d	(0, 950)	17.4
e	(0, 1100)	15.8
f	(0, 1250)	11.3
g	(360, 360)	14.5
h	(500, 500)	10.6
i	(650, 650)	6.9
j	(800, 800)	4.9
k	(950, 950)	3.4
l	(1100, 1100)	2.8

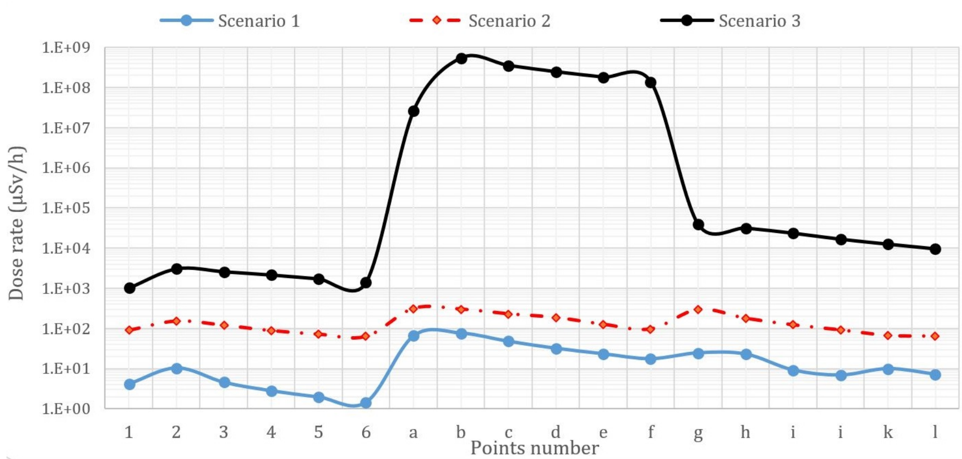

Figure 9: comparison of dose rate for the three studied scenarios.

Table 10: Calculated gamma dose rate in the compact core No. 1 for scenario 2.

Point	Location (x,y) (cm, cm)	Dose rate (μSv.h^{-1})
1	(500, 0)	19.2
2	(650, 0)	46.1
3	(800, 0)	89.2
4	(950,0)	50.3
5	(1100, 0)	34.9
6	(1250, 0)	68.9
a	(0, 500)	66.9
b	(0, 650)	70.7
c	(0, 800)	64.2
d	(0, 950)	41.5
e	(0, 1100)	31.6
f	(0, 1250)	24.8
g	(360, 360)	27.5
h	(500, 500)	53.0
i	(650, 650)	185.5
j	(800, 800)	61.0
k	(950, 950)	33.1
l	(1100, 1100)	27.7

Table 11: Calculated gamma dose rate in the compact core No. 1 for scenario 3.

Point	Location (x,y) (cm, cm)	Dose rate (μSv.h^{-1})
1	(500, 0)	0.23
2	(650, 0)	0.65
3	(800, 0)	0.58
4	(950,0)	0.51
5	(1100, 0)	0.44
6	(1250, 0)	0.38
a	(0, 500)	7.92e+04
b	(0, 650)	5.45e+04
c	(0, 800)	3.71e+04
d	(0, 950)	2.62e+04
e	(0, 1100)	1.92e+04
f	(0, 1250)	1.47e+04
g	(360, 360)	27.63
h	(500, 500)	17.26
i	(650, 650)	13.54
j	(800, 800)	10.32
k	(950, 950)	7.72
l	(1100, 1100)	5.88

As could be seen from Table 9, the under-containment dose rates are low in this condition which are well-matched with experimental data of the health physics monitoring system.

Scenario 2. The bare core and beam tubes full of water

The emitted gamma would cause a high dose rates in the upper core regions despite the BHF area. The calculated dose rate values are given in Table 10 for a better understanding.

As could be seen from Table 10, the core dewatering could cause high dose rates in the under-containment areas that necessitate considering the appropriate protective actions.

Scenario 3. The bare core and emptied beam tubes

The condition in this scenario is the worst for the under-containment dose rate. The calculated dose rate values are given in Table 11.

As illustrated in Table 11, unlike the two last scenarios, the gamma dose rate in some points would cause considerable quantities in a short time. These values significantly exceeding the allowable limit of gamma dose enforces considering the mentioned actions of Table 8. A comparison of dose rate for investigated scenarios is given in Fig. 9 for a better understanding of accident consequences. As shown in this figure, dewatering of the pool and the beam tubes could cause considerable dose rates in the under-containment.

4 Conclusion

The main goal of this research is the illustration of the uncovered core gamma radiation hazards and the necessity of providing solutions for the prevention of this accident as early as possible. The core dewatering occurrence one day after the core sub-criticality would not cause core damage. The gamma dose rate of the direct radiation is in the range of 200 Sv.h^{-1}. This leads to too much radiation dose exceeding the allowable limit of 50 mSv per year. Thus, due consideration should be given to the modification and improving reactor safety. This study intends to attend research reactors planners to safety for protecting the reactor personnel against detrimental health effects during a reactor core uncovering accident through improvement and modification of safety systems.

The probability of happening this scenario in the TRR is very low due to the following reasons:

1. The results of this conservative study are based upon the continuous full-power operation of the reactor for 30 days which seldom occurs.

2. Happening a severe accident caused by the core uncovering is rarely probable due to various diverse and redundant systems in this reactor.

3. This reactor is already equipped with one emergency make-up water system preventing core uncovering as soon as happening LOCA.

Locating an emergency make-up water system has a major safety significance in enhancing nuclear safety for preventing severe accidents such as core damage, and undue radiation risks following the total core uncovering.

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