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Investigation of Gamma Dose Changes of High-Degree Occupation Hall of Tehran Research Reactor up to a Few Days after the LOCA Accident

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Loss Of Coolant Accident (LOCA) investigation of Tehran Research Reactor (TRR) exactly at the time when the reactor has been operating at its maximum power (5 MW) and the consequent impact of it on the reactor operators and other people in the vicinity of the reactor containment is a very important subject which should be carefully evaluated. The effects of the events have been there since the first nuclear reactor built in 1954. It should be noted that according to the documents, high dose rate (1–2 Sv/h) has been reported inside the rooms of the damaged Chernobyl nuclear power plant after the accident. Considering the different nuclear accidents and the regularity body concerns, the present work aimed to calculate the gamma dose rates in high-degree occupation portions from the TRR hall after LOCA accident. The main purpose of upgrading the technical documentation of the reactor is to ensure the safe operation of the reactor and to ensure the safety of the whole site in such accidents. In the present work, the dose rate received by the staff working in the TRR entrance room after the LOCA accident was investigated when the situation resulted in the complete baring of the nuclear core. MCNPX code was used to model the TRR containment with the most details.

Introduction

Safety is very important concern for operational nuclear power plants (NPP). Yearly different documents are prepared to cover the different types of safety issues in NPPs. One of these documents is the personnel internal and external exposure during the normal and accident situation of an NPP. It should be mentioned, according to the ALARA (As Low As Reasonably Achievable) principal and standard regulations the limit value for the average effective dose for radiation workers is 20 mSv/year or 10 μSv/h but at accident condition if the gamma dose rate of the workers exceeds than 1000 μSv/h the area should be evacuated [1–3]. Hence, regularity body of any NPP should perform annually-inspections to control and update the required safety documents. For the mentioned purpose some activities are being done in different NPPs which some of them are reviewed in the following.

Hoq et al. (2017) reported the 3 MW TRIGA Mark-II Research Reactor of Bangladesh Atomic Energy Commission (BAEC) has been under operation for about thirty years since its commissioning at 1986. For determining dose rate at different strategic points of the reactor facility, they measured neutron, beta and gamma radiation with reactor power level of 2.4 MW to estimate the rising level of radiation due to its operational activities. They reported high radiation dose is observed at the measurement position of the piercing beam port, which is caused by neutron leakage. Their study also deals with the gamma dose rate measurements at a fixed position of the reactor pool top surface for different reactor power levels under both Natural Convection Cooling Mode (NCCM) and Forced Convection

Cooling Mode (FCCM) and results showed the radiation dose rate is higher for NCCM in compared comparison with FCCM [4].

Abrefah et al. (2018) used two computer codes of ORIGEN-S; for computing changes in the isotopic concentrations during neutron irradiation and radioactive decay as well as to determine the source term; and MCNP6; which uses the source term estimated by ORIGEN-S code to calculate the dose rate for GHARR-1 research reactor. The 30 kW pool-type Ghana Research Reactor-1 (GHARR-1) is a commercial Miniature Neutron Source Reactor (MNSR) similar to the Canadian SLOWPOKE in design. Their obtained results showed that above the reactor core-floor level at 595 cm the measured dose rate is 4.27E+04 ± 0.0006 mGy/h [5].

Hence, the present work aimed to investigate the gamma dose rate of the high-occupation areas in Tehran Research Reactor (TRR) in Loss Of Coolant Accident (LOCA). The carried out calculations were performed with the imagination of that the LOCA is happened exactly at the operational time of the reactor at 5 MW.

Material and methods

To investigate the gamma dose rates in the mentioned areas of TRR, the site containment was modeled using MCNPX code. A square-lattice 33-assembly pool-type TRR core has been modeled using MCNPX 2.6.0 code, which is a Monte Carlo-based code with ability of 34-particle transport and modeling different geometries powerfully [6]. Light water is used as coolant for of the fuel assemblies in TRR. Graphite is used as the core reflector. A 3D TRR model was set up using MCNPX 2.6.0. SDEF capability has been used to introduce gamma source. TRR is an open pool, MTR type light water-moderated with a thermal power of 5 MW.

The reactor core is composed of two types of fuel assembly that are standard fuel elements and control fuel elements. The core consists of 28 standard fuel element (SFE) containing 19-fuel plates and 5 control fuel elements (CFE) containing 14 fuel plates according to the core specifications. The modeled reactor is a 5 MW reactor with 20% enriched MTR fuels and 500 m^3/h flow rate. The reactor core is composed of two types of fuel assembly that are standard fuel elements and control fuel elements. Two types of control rods are used in the TRR; which one made out of Ag-In-Cd alloy, and the other of stainless steel. Both have a set of two control plates as a fork type shape. The reactor fuel is U_3O_8-Al containing 20%-enriched uranium. Neutronic and thermal hydraulic characteristics of the fuel assemblies and control rods are summarized in Table 1 [7].

To define a gamma source as SDEF card in MCNPX code, the gamma source should be calculated using ORIGEN code, which is a widely used computer code for calculating the buildup, decay, and processing of radioactive materials. The code involves reactor models, cross sections, fission product yields, decay data and decay photon data [8].ORIGEN 2.1 is a versatile point-depletion and radioactive-decay computer code for use in simulating nuclear fuel cycles; it was developed at the Oak Ridge National Laboratory (ORNL) and distributed worldwide beginning in the early 1970s. It is used for existing reactors, including pressurized water reactors, boiling water reactors, liquid-metal fast breeder reactors, and Canada deuterium uranium reactors [9].

To calculate gamma dose rates two computational procedures were applied; one is simulation of gamma detectors to calculate a selected point dose rate, the second is the definition of mesh tally card in the MCNPX code input to calculate a dose rate distribution. Some Geiger Muller detectors (GM; LB 123 made by BETHOLD Company; the gamma detector sensitive volume dimension is 45 mm×170 mm (R×H).) were placed around the TRR check in room to determine the gamma dose rates at different positions of the place. In addition, the dose rate distributions were calculated using the mesh tally capability of the MCNPX code. F4 tally with DE/DF card were used to apply flux to dose conversion factors for the gamma dose rate calculation. Figure 1 shows the modeled containment of TRR using MCNPX code.

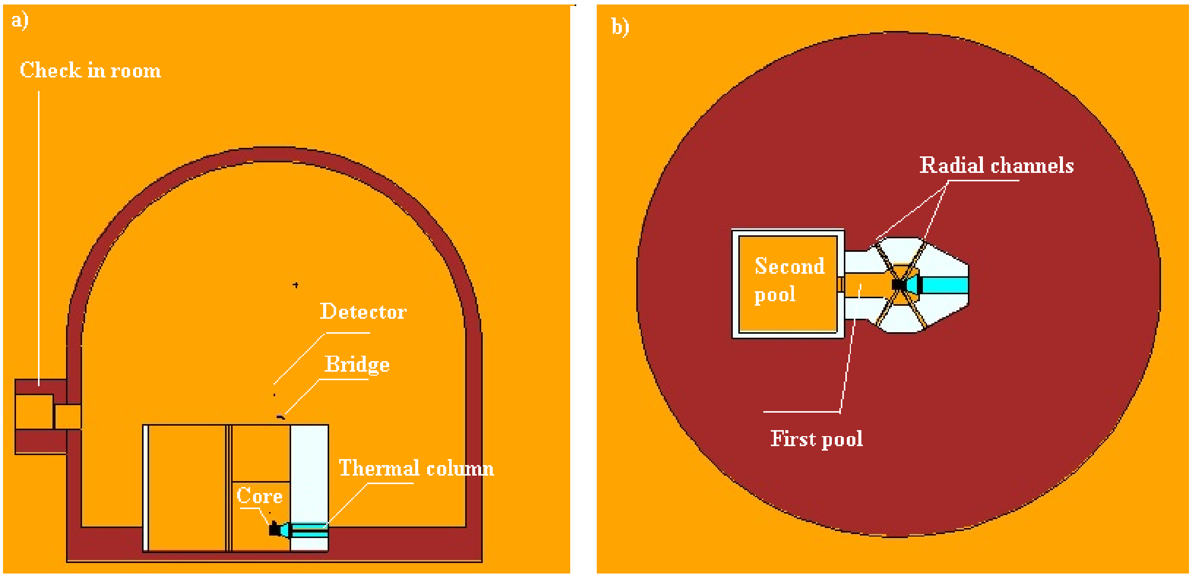

Figure 1: The modeled containment of TRR using MCNPX code, a) Axial view, b) Radial view.

Results and discussion

Figure 2 shows the average gamma dose rates at the height that is near the second floor just immediately and 7-days after LOCA, which was happened so that both first and second pools are bare (empty).

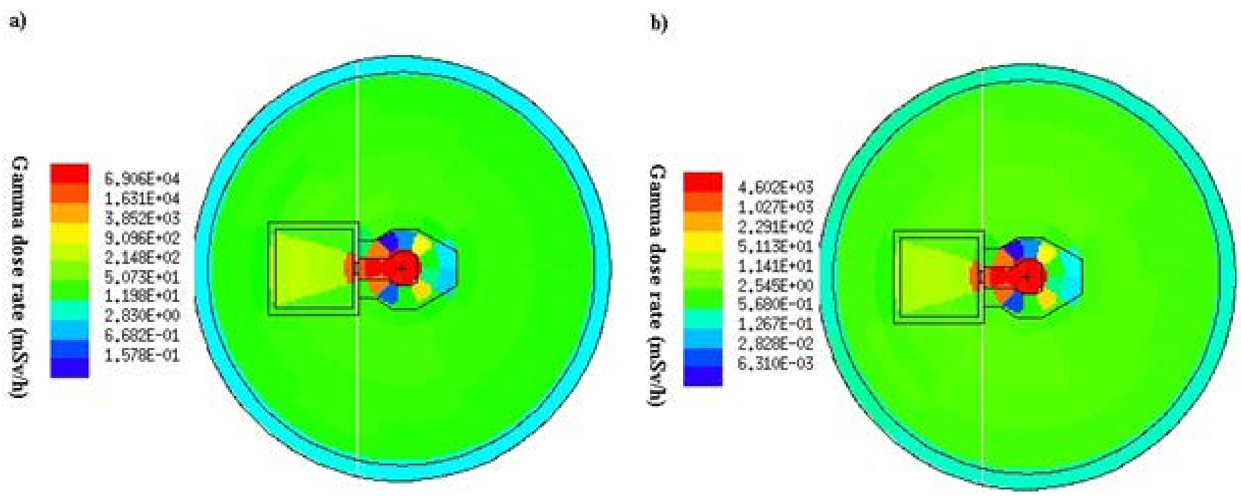

Figure 2: Gamma dose rate in the modeled containment of TRR using MCNPX code, a) immediately after LOCA, b) 7-days after LOCA; both of the pools are empty.

Fig.3 shows the average gamma dose rates at the height that is near the second floor exactly immediately and 7-days after LOCA, which was happened so that only first pool is bare (empty).

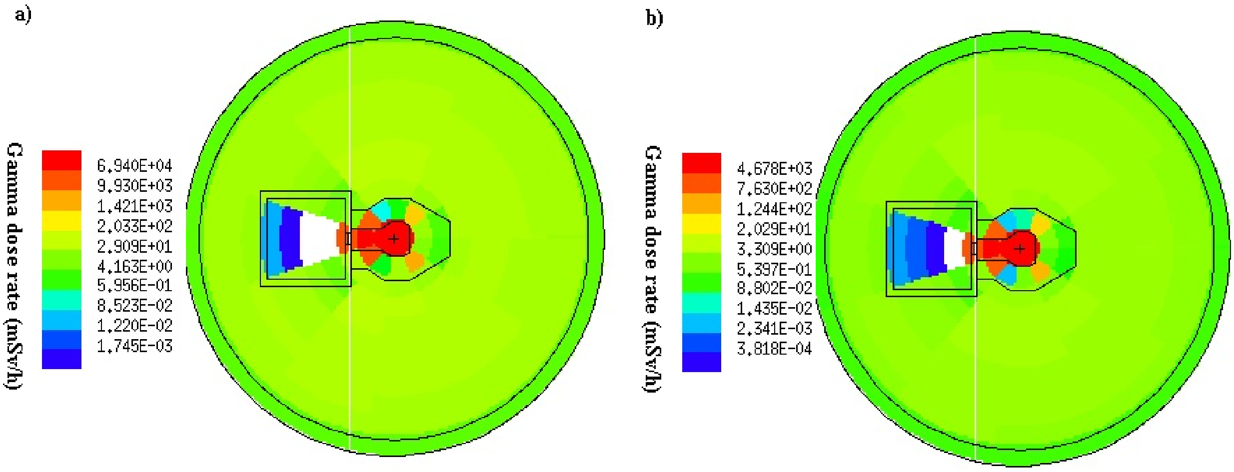

Figure 3: Gamma dose rate in the modeled containment of TRR using MCNPX code, a) immediately after LOCA, b) 7-days after LOCA; first pool is empty of water.

Clearly, the evacuation of the first pool of water has the highest impacts on the reactor hall dose rate because of the bared core high gamma emission rate. One of the important sections is the entrance door to the reactor hall as well as the check in room of the TRR reactor. Table 1 shows gamma dose rates at different position, which are very important regard to keeping the personnel exposure limitations. Obviously, the worse situation belongs to water evacuation accident of both pools. The second critical situation can be considered when the first pool and then the nuclear core are bare. In both situations the hall exit location experiences high gamma dose rates 55.09 and 38.21 mSv/h respectively. When both of the pools miss their water, the gamma dose rate between the lead doors of the TRR containment is about 41 µSv/h in the middle of the corridor between the two lead doors. The calculations shows after the second lead door at the worse accident the gamma dose rate would be in order of 4 µSv/h.

Table 1: Dose rate at the positon of the TRR containment entrance door immediately after LOCA

Position	**Gamma dose rate (mSv/h)**
Both of the pools are empty	
Before lead doors	55.09
Between lead doors	0.041
After second lead doors	0.004
First pool is empty	
Before lead doors	40.43
Between lead doors	0.00074
After second lead doors	< nSv/h

When the first pool is bare, after 7-days of LOCA the gamma dose rate exactly before the lead door would be 3.43 ± 0.17 mSv/h. The value is 3.2 ± 0.09 mSv/h when both of the pools are empty. Average gamma energy of the radioactive fuel assemblies is about 950 keV after LOCA and 763 keV at 7-days after LOCA, which the less average energy causes less gamma dose rates far from the bared core.

Conclusion

Investigation of gamma dose rate distribution inside the TRR containment after immediately LOCA and during the next days is very important in point of the operator's exposure views. The carried out calculations showed immediately after LOCA in condition that both the pools are empty a 50 mSv/h average gamma dose rate would be available in whole of the second floor of the containment. The value drops 10 times after 7-days of cooling. In addition, when the first pool only misses its water during the LOCA, the gamma dose rate of the second floor is about 4 mSv/h on average. Similarly after 7-days of cooling the average gamma dose rate decreases about 10 times. Clearly when the second pool keeps its water, some portion of gammas shield by the available water of the second pool. The calculations showed that even in the worse condition (immediately after LOCA and water evacuation of both pools) the check in room would not experience the gamma dose rates of more than 10 µSv/h.

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