ESTIMATION OF DOSES RECEIVED BY OPERATORS IN THE 1958 RB REACTOR ACCIDENT USING THE MCNP5 COMPUTER CODE SIMULATION

by

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Scientific paper
DOI: 10.2298/NTRP1203199P

A numerical simulation of the radiological consequences of the RB reactor reactivity excursion accident, which occurred on October 15, 1958, and an estimation of the total doses received by the operators were run by the MCNP5 computer code.The simulation was carried out under the same assumptions as those used in the 1960 IAEA-organized experimental simulation of the accident: total fission energy of 80 MJ released in the accident and the frozen positions of the operators. The time interval of exposure to high doses received by the operators has been estimated. Data on the RB1/1958 reactor core relevant to the accident are given. A short summary of the accident scenario has been updated. A 3-D model of the reactor room and the RB reactor tank, with all the details of the core, created. For dose determination, 3-D simplified, homogenised, sexless and faceless phantoms, placed inside the reactor room, have been developed. The code was run for a number of neutron histories which have given a dose rate uncertainty of less than 2%. For the determination of radiation spectra escaping the reactor core and radiation interaction in the tissue of the phantoms, the MCNP5 code was run (in the KCODE option) and "mode n p e", with a 55-group neutron spectra, 35-group gamma ray spectra and a 10-group electron spectra. The doses were determined by using the conversion of flux density (obtained by the F4 tally) in the phantoms to doses using factors taken from ICRP-74 and from the deposited energy of neutrons and gamma rays (obtained by the F6 tally) in the phantoms' tissue. A rough estimation of the time moment when the odour of ozone was sensed by the operators is estimated for the first time and given in Appendix A.1. Calculated total absorbed and equivalent doses are compared to the previously reported ones and an attempt to understand and explain the reasons for the obtained differences has been made. A Root Cause Analysis of the accident was done and, for the first time, a Cause and Effect diagram has been created in Cause Mapping methodology and shown in Appendix A.2.

Key words: accident, absorbed dose, equivalent dose, RB reactor, MCNP5 code

INTRODUCTION

The RB reactor is a non-reflected, natural uranium, heavy water critical assembly designed by Yugoslav scientists, commissioned in former Yugoslavia at the "Boris Kidrič" (now Vinča) Institute of Nuclear Sciences, on April 29, 1958 [1, 2]. On October 15, 1958, a reactivity excursion accident in which six operators were heavily exposed to radiation occurred at the facility [3]. In the first paper published on the accident in 1959, an estimation of the equivalent doses received by the operators were calculated [3].The critical assembly, designed for operation at "zero power" (*i. e.*, within the mW range) reached the power maximum of 2.5 MW at the peak of the excursion. The first independent review of the accident was done by IAEA officials in an Internal Report [4]. As a consequence of the accident, in spite of the medical treatment received in France, a month later, one of the operators died. The medical treatment applied was the first ever human bone marrow transplantation in Europe [6]. The International Atomic Energy Agency (IAEA) offered help and, with the approval of the Yugoslav Government, in April 1960, prepared, organized and conducted the "Vinča Dosimetry Experiment" at the RB reactor with the aim of simulating accident conditions and estimating doses received by the operators [5]. Among other methods used, the absorbed doses received by the operators were estimated by measuring ^{24}Na activity in the water of the seven phantoms placed and irradiated around the RB core [5]. The uncertainty in the absorbed doses, within a 15%, was estimated based on applied methodologies, assumed approximations and

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the uncertain time interval of the accident and positions of the operators.

Since then, several papers on the accident, its description and estimates of its consequences, including the doses received by the operators, have been published [7-27, 50-56]. Most of these papers are merely references to the accident, lacking relevant physics or dosimetry evaluations. They also include the recollections [21] of the ORNL Division Head, Karl Z. Morgan, the founder of Health Physics in USA and head of the ORNL investigation team in Vinča Institute at the time [5, 12]. The ORNL team included such scientific dignitaries as G. S. Hurst, R. H. Ritchie, F. W. Sanders, J. A. Auxier, D. E. Callahan, P. W. Reinhardt, and G. H. Wigner. A recent novel on solidarity in bone marrow transplantation [26], a RTS TV documentary film [70] and an earlier TV drama ("Irradiated", directed by G. Poitou), realized in co-operation of the Belgrade TV studio (RTB) with the French State TV studio, ORTF, in 1976, deserve to be mentioned, too.

A comprehensive study of the accident, focusing on its technical and physics aspects, was done in 1992 at the Vinča Institute of Nuclear Sciences [7]. Aspects of biological and medical effects of the irradiation and the consequences of the radiation illness on the operators have not been studied in this work, because they have been elaborated in numerous previous papers of a medical nature [*e. g.*, 6, 9, 51, 59, 63, and 65]. Our work is the first attempt to evaluate doses received by the operators using a contemporary computation tool – the MCNP5 computer code [28]. The paper also compares computation results of doses received to a wide range of those previously published [3, 5, 6, 9, 12, 49, 51, 52, 59, 61, 63, and 64], attempting to explain the reasons for the discrepancies. Doses received by RB reactor operators were reported to be in the range of 207 rad (2.07 Gy) to 640 rad (6.4 Gy) and from 145 rem (1.45 Sv) to 1024 rem (10.24 Sv). Old, abandoned units of the absorbed doses, (rad), and the equivalent dose, (rem), have been retained in this paper with the aim of preserving authentical links to the published documents. Some estimated values of the doses received by the operators are published firstly and then re-evaluated afterwards in new papers [52, 59].

DESCRIPTION OF THE RB REACTOR AND SURROUNDINGS

A full description of the RB critical assembly with the first RB1/1958 core configuration used in 1958, when the accident occurred, is given in [1, 2]. Additional technical details are given in [5, 7]. A photograph of the RB reactor (1958) is given in fig. 1. In the photograph, the RB reactor control desk (console) can be seen on the right, at the edge of the pond, known as the "dry pool."

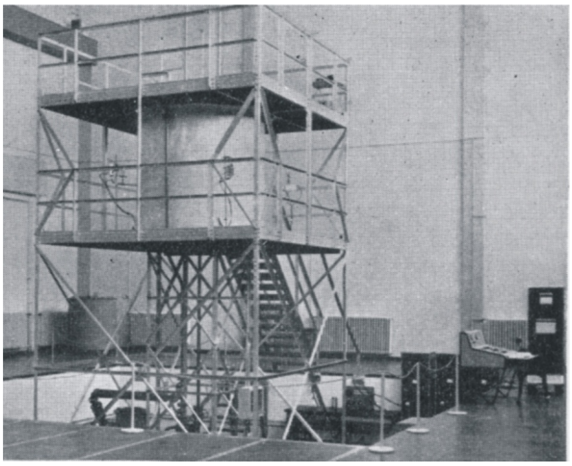


Figure 1. View of the RB reactor from 1958

Only basic data on the RB1/1958 core configuration necessary for the simulation and calculation of the escaping neutron and gamma ray spectra from the RB reactor tank and dose calculations by the MCNP5 computer code are given in this paper.

In 1958, the RB reactor's unreflected ("bare") RB1/1958 core was assembled from 208 fuel elements (rods) in a heavy water lattice with a 120 mm square pitch. Each fuel element was assembled of seven smaller natural uranium metal rods (25 mm in diameter and 300 mm in length), placed one above the other, inside an aluminium alloy cladding, 1 mm thick. The fuel manufacturer [29] specified the weight fraction of the ^{235}U nuclide at 0.714% and the average mass density of uranium metal at 18.92 0.01 g/cm^3. Three batches of fuel rods with various concentrations of impurities (B, C, N, Si, Mn, Fe, Ni, and Cu nuclides) were used. Unfortunately, no records of the assembling of uranium fuel elements exist today. For the purpose of this study, we have assumed that the fuel material was natural uranium metal, with given data on the impurities and a mass density of 18.64 g/cm^3 [30]. The said density of the uranium was evaluated from available data [30] and the assumption that there was no gap between the fuel meat and the aluminium alloy cladding. As for the aluminium alloy SAV-1 (uranium metal cladding), a Russian certificate for the composition of the material with a mass density of 2.729 g/cm^3, experimentally determined at the Vinča Institute, was applied [30]. Historical data on heavy water moderator, dating back to the time, indicate that the RB reactor had a stock of 6985.365 kg of heavy water, in total [7] (*i. e.*, 6.36 m^3 for a D_2O density of 1.1 g/cm^3). Heavy water purity was 99.76% (mol), while the remaining part to 100% was light water. With this volume of heavy water, a maximum level of 2.10 m of heavy water in the reactor tank could be achieved with 208 natural uranium metal rods placed in a lattice with a 12 cm square pitch of the RB1/1958 core. Horizontal and

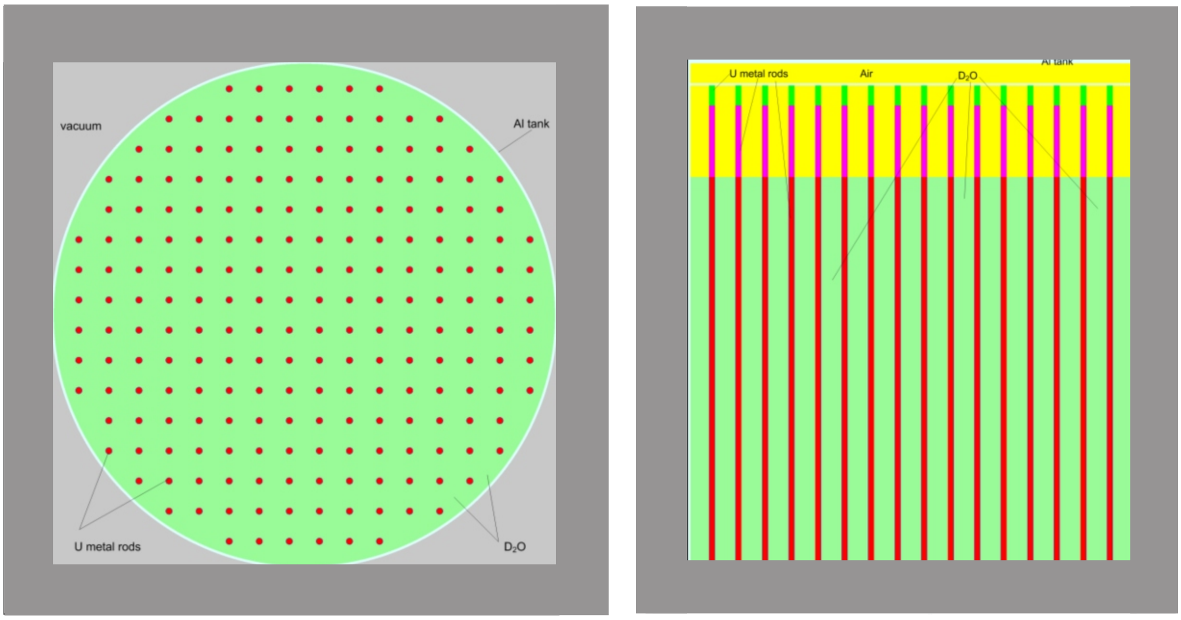


Figure 2. Horizontal and vertical cross sections of the RB1/1958 core 3-D model

vertical cross-sections of the 3-D model of the RB1/1958 reactor core are shown in fig. 2.

The experiments were performed in an aluminium cylindrical (RB) tank mounted on an aluminium platform (fig. 1). The platform was built in the centre of the dry pool (square cross section 8 m 8 m, depth 1.5 m) in the reactor room. The northern wall of the reactor room, facing the corridor in the reactor building, constructed in the lower part of the room, was fitted with large glass windows. The RB reactor tank inner diameter (ID) is 2000 mm wide and has a height of 2300 mm. The thickness of the aluminium bottom of the tank is 15 mm. The average thickness of the aluminium top cover of the tank is 25 mm. The bottom of the RB1/1958 reactor core is 2.5 m above the reactor room's floor. The centre of the RB1/1958 core (according to critical dimensions) is approximately 1 m (0.89 m) higher.

The surfaces of the RB reactor tank are distanced at least 3.75 m from any other surface (walls, floor or ceiling) of the reactor room. In this manner, the reflection of escaping neutrons from surrounding surfaces in the reactor room back to the RB reactor tank is determined to be less than 0.4% [1]. Beside the RB tank platform, an additional and separate aluminium platform for experimental equipment and personnel is mounted around the tank. The control room of the RB reactor, separated from the reactor room, had not been completed in the fall of 1958 and the facility was operated from a small reactor control console at the north side of the dry pool (fig. 1). Most of the experimental equipment was located in the northwestern corner of the dry pool (fig. 1). This mode of RB reactor operation was possible due to low radiation doses (due to neutrons and gamma rays escaping from the reactor core) at positions occupied by the operators and scientists, considered to be acceptable and in accordance with radiation limits at the time.

It should be mentioned that, according to an internal report made after the accident, the RB reactor began operation without any written License, Design documents, written Operation and Regulation rules or Safety Analysis Report. We may conclude that, in those pioneer days of nuclear technology in Yugoslavia, researchers at Vinča Institute regarded the RB reactor more as a new experimental tool than a facility involving a serious radiation risk.

The RB reactor is designed as unreflected and without any radiation shielding to provide "clean", simple "nuclear" geometry. The facility is operated when the two cadmium safety rods are out of the core. Criticality is achieved and maintained by a pump for adjusting the level of heavy water in the RB reactor tank, transferring heavy water from the storage tank to the RB reactor core. The heavy water pump is located in the underground room, adjacent to the dry pool, together with the heavy water storage tank. The level of the heavy water in the RB reactor tank is measured by a calibrated probe with a sensitive pin (known as the "levelmeter") that was set in the air above the current moderator level, at a new desired level. The pump is switched on to increase the heavy water to that of the new moderator level. When the heavy water touches the probe pin, it closes the electrical circuit, this is registered by an ammeter at the reactor control console, and at that point the operator is supposed to switch the pump off. The probe is then moved to the next desired (higher) level of heavy water and the process repeated. The heavy water is pumped into the reactor core by the pump with two possible speeds, the changing D_2O

moderator increasing the level at rates of 2.5 cm/min. and 0.8 cm/min. The level of D_2O is measured, with an uncertainty of 0.2 mm, by the described "levelmeter" observing the electrical contact between the water surface and the pin of the probe.

The experimental "start up" equipment consists of three pulse channels with BF_3 counters of different sensitivity, with scalers and 5-decade rate meters connected to a chart recorder. The BF_3 counters, with associated pre-amplifiers, were placed around the RB reactor tank, on the lower platform. They were used to measure neutron flux density (according to the ICRU, equal to the "fluence rate") and shown it at the experimental panel (racks) placed at the northwestern corner of the dry pool and at the pool edge, beside the reactor control console. During the start up procedure, a Ra-Be neutron source, of 17.5 GBq (500 mCi) Ra intensity, is inserted through the tank top cover into the RB reactor core, along a central axis tube, called the "source injector".

The safety system of the RB reactor (in 1958) consisted of a supervisor's control key, two safety rods, gamma ray dosimetry monitors coupled to audible alarms and an automatic shutdown circuit with a trip at the high level of thermal neutron flux density. No interlock system was in existence at the time [14] that would stop the RB reactor operation if the safety monitors or safety circuits were turn off or removed. No interlock system was designed at the time to prohibit the increase of the moderator level operating the pump with a higher speed nearing the criticality level too.

The first measured heavy water critical level in the RB1/1958 core was 177.6   0.1 cm, at a moderator temperature of 22 °C [1]. Basic reactor experiments regarding the determination of the reactor's parameters – critical mass, neutron flux density, temperature coefficient reactivity of the moderator, the reactivity of safety rods, buckling, migration length, *etc.* were carried out up to the end of September 1958 [31-33]. The gradient of reactivity near the critical level, *i. e.*, the change of reactivity with the level of heavy water, d /dH, was measured as (70.6   1.6) 10^{-5} *k*/*k* per cm [32]. Calculations done by the computer code KENO V.a [34], with a broad 44-group neutron cross section library based on ENDF/B-V data, gave a value of $d\rho/dH$ = (75   15) 10^{-5} $\Delta k/k$ per cm, while calculations by the computer code DENEB [35] gave us a value of $d\rho/dH$ = $67.6 \cdot 10^{-5}$ *k*/*k* per cm [7].

The measured value of the two safety rods was –2.12 β_{eff} = –0.017 *k*/*k*, for the calculated, using AVERY computer code [66], value of the total effective fraction of delayed neutrons and photo-neutrons in this heavy water system, β_{eff} = 8.0445 10^{-3}. The calculated value of the neutron generation time in the system, by the same AVERY computer code [66], is Λ = 0.50352 ms. The value of the neutron removal lifetime, calculated by the KENO V.a computer code [34], is 0.49019   0.00019 ms. The neutron removal lifetime is the average life-span of a neutron from the time it is born, until it is absorbed or leaks from the system [34, 67]. The neutron removal time, obtained from calculations by the MCNP5 computer code [28], is 0.49537   0.00025 ms, while the neutron generation time, *i. e.*, the average time between two fission production neutrons [67], amounts to 0.50072   0.00015 ms.

EVALUATION OF THE ACCIDENT SCENARIO

Only a few published references [3-5] of the accident course prior to 1990 exist. Based on these references, a new evaluation of the accident scenario was done in [7 and 8]. Additional data on the accident course can be found in testimonies of the Vinča Institute staff and officials from other institutes who took part in the evaluations [18]. It is unknown if any written testimonies of the accident by the participants themselves, except for a single one, given in [17], published almost 40 years after the event, exist. According to that testimony [17], the irradiated operators were not allowed to discuss the accident in public. The most comprehensive evaluation of the accident physics and course was presented in [7, 8]. Only the main findings were extracted from these evaluations, updated and given in this paper and Appendix A.2.

The accident occurred on October 15, 1958, during the third series of experiments that were carried out with the aim of determining the strength of the spontaneous fission source from natural uranium metal rods in the RB reactor core [4]. A fast increase in moderator level (2.5 cm/min.) was switched on by the pump, at a heavy water level of 175 cm (that was 3.5 cm below the expected critical level), to a new (expected) sub-critical level of 177 cm. The personnel operating the RB reactor were in the reactor room (fig. 1), near the reactor control console, on the north side of the reactor room and around the experimental panel at the north-west corner of the dry pool. After the pump was switched on, the operators were distressed by the entrance of a non-staff individual into the reactor room [4]. However, up to now, no confirmation of this particular event can be found in the testimonies [17, 55], nor a record of the dose received by that person. Moreover, there is another testimony [55] according to which the operators at the reactor control console were learning English from book(s) spread on the console. This testimony [55] was not refuted or commented in the written testimony [17]. Two other employees present in the RB reactor building have received elevated doses and have received medical treatment in a hospital in Belgrade [5, 51].

The D_2O moderator reached the 177 cm level in the reactor tank of the RB1/1958 core, but the ammeter reading of the 177 cm D_2O level was not observed by the operators. The heavy water in the RB reactor tank continued to increase, since the pump was not switched off. The instrumentation of the RB reactor, used regularly in dosimetry, alarm and safety systems, was either switched off or removed [14].

Without any supervision on the part of the staff at the RB reactor control console, reactivity and power continued to increase and, according to [4], the entire amount of the heavy water was transferred from the storage tank to the reactor core [4]. The maximum level (210 cm) of heavy water in the reactor tank of the RB1/1958 core would bring in a total reactivity of $1273 \cdot 10^{-5}$ k/k, as it was calculated by the KENO V.a computer code [34] and is in good agreement with data given in [5] and [7]. The information that all heavy water was transferred from the storage tank to the RB tank is in contradiction with the statement that the maximum level of heavy water in the RB reactor tank had reached 183 cm, corresponding to a positive reactivity of about $300 \cdot 10^{-5}$ $\Delta k/k$ [5]. In this study [7], the maximum heavy water level has been determined at 183 cm, based on the recorded readings of the ratemeter [4]. In addition, it is possible that the entire volume of the heavy water was not in the RB reactor storage tank during the experiment.

The power excursion continued until the operators in the RB reactor room sensed the odour of ozone [47] in the air. In the first instance, they checked the fuses of electrical installations and the valves of the pump [17] and only after that did the operator at the reactor control console switch off the heavy water pump and manually shut down the RB reactor with safety rods [17]. The operators then exited the reactor room and instructed the rest of the personnel to leave the RB building [17]. Various other technical details on the evaluation of the accident course can be found in [7, 8]. Only the Accident Scenario Summary in tab. 1, upgraded by new data on reactivity (ρ) obtained by using the MCNP5 [28] or KENO V.a [34] computer codes, is shown here. These data are shown in tab. 1, along with previous results for reactivity obtained by the computer code DENEB [35] in the study [7, 8]. Some new details in the column labelled "The Action of Operators" were also added to tab. 1, according to testimonies [17] and [6].

The initial analyses of the accident [3, 4] are done without including feedback effects arising from changes in the temperatures of the RB reactor fuel and moderator. A simple approximation of the power excursion, from the mW power range [7], by an exponential function with a 10 s power period, was assumed [4]. According to the measured activity of irradiated Au and Cu foils found in the RB building and metal objects that were carried by irradiated employees, it was estimated [3] that the total fission generated energy in the accident was 80 MJ. The duration of the accident was not recorded, but an automatic recorder for measuring airborne activity and the radioactive fallout at the Vinča Institute, 540 m away from the RB reactor building [18], registered the power rise by accompanying increased gamma ray background, lasting approximately 10 minutes. Based on that information, it was estimated that the time interval of the power excursion was between 4 and 10 minutes, while in [6], a time interval of 3 min. to 7 min. is mentioned.

The power and generated fission energy during the time interval of the accident are calculated in [7, 8] by two computer codes, SCM [36] and MACAN [37], developed at the Vinča Institute, and their results are shown in fig. 3. As can be seen in fig. 3, both computer codes have shown very good agreement, in spite of the differences in how the reactivity feedback is treated by them [7, 8]. Computer code SCM includes feedback using the generated energy coefficient of reactivity, while computer code MACAN includes the feedback via changes of temperatures of the fuel and moderator, using temperature reactivity coefficients.

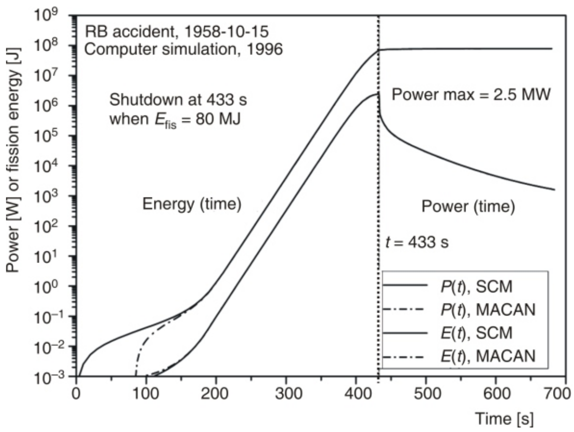


Figure 3. Power and fission generated energy *vs.* duration of the 1958 RB reactor accident

In numerical simulations of the accident course, *i. e.*, determinations of $P(t)$ and $E_f(t)$, it was found that the accident time was 433 s, measuring the time interval from the moment when the heavy water pump was switched on, at a D_2O level of 175 cm, until the RB reactor was shut down at a generated fission energy of 80 MJ. It was also concluded that the exclusion of fuel and moderator temperature feedback in the earlier analyses had not been the right thing to do. Depending on the shutdown time, the change in the fuel average temperature was between 80 °C and 100 °C, while the change in the average temperature of the moderator was about 2 °C, at the maximum of reactor power. These changes in temperatures of the RB reactor fuel and moderator were not high enough and capable of automatically shutting down the RB reactor with the negative, but nevertheless small temperature coefficients of the reactivity of the fuel ($-1.2 \cdot 10^{-5}$ $\Delta k/k$ per K) and moderator ($-24.1 \cdot 10^{-5}$ $\Delta k/k$ per K) [7]. The said temperature changes have influenced only the time

Table 1. Summary of the 1958 RB reactor accident scenario

Time [s]	RB reactor condition	Action of operators
0	D_2O level: 175 cm; ρ = –245.6 pcm*; P_0 = 0.25 mW MCNP5: ρ = – (210 25) pcm KENO V.a: ρ = – (210 31) pcm D_2O levelmeter position: 177 cm Increasing D_2O level to 177 cm	The heavy water pump is switched on at the reactor's control console to increase moderator level to the expected (determined) 177 cm level
48	D_2O level: 177 cm; ρ = –110.2 pcm MCNP5: ρ = – (94 25) pcm KENO V.a: ρ = – (74 31) pcm	Not observed by the operators on the ammeter at the reactor's control console
	Increase in the D_2O level in the core continues	The pump is not switched off
84	D_2O critical level: 178.5 cm; P = 0.59 mW	Not observed by the operators
	Increase in the D_2O level in the core continues	Not observed by the operators
192	D_2O level: 183 cm; all D_2O from storage tank is transferred into the core: ρ = +295.5 pcm MCNP5: ρ = + (280 25) pcm KENO V.a: ρ = + (307 31) pcm	Not observed by the operators
	ρ = +295.5 pcm; reactor power period T = 12.3 s	Odour of ozone is scented by the operators in the reactor room The pump is switched off
433	RB reactor is shutdown by safety rods P_{max} = 2.5 MW E_{tot} = 80 MJ	Safety rods are shut down manually by the operator at the reactor control console
	RB reactor is in shutdown state. Power decreases. Decreasing D_2O from the RB reactor tank is switched on	Operators leave reactor room and RB building

*Note: 1 pcm = 10^{-5} $\Delta k/k$

moment when the 80 MJ of fission energy was generated.

The primary cause of the accident was the rise of D_2O over the critical level of 4.5 cm, because the pump was not switched off at the moderator level of 177 cm, since the operators at the reactor's control console did not register the ammeter reading. The other contributing causes were:

– there was no interlock system designed to stop the operation of the RB reactor when the alarm and safety systems were switched off, or removed,
– there was no interlock system designed which would prevent the operation of the D_2O pump (at a higher speed) for an indefinite time near the moderator's critical level,
– a D_2O over-level safety trip was not included in the safety system of the RB reactor, and
– the disturbance caused by the entrance of a non-staff individual – according to the testimony [17], now under question.

It is to be understood that the action of the operator to switch on (open the valve) in order to decrease the moderator in the RB reactor tank immediately after the RB reactor was shut down after the accident [17], was done with the best intention of the operator to assure (additional) subcriticality of the RB reactor's core. Unfortunately, his action had some drawbacks, too. Firstly, removing the heavy water from the RB reactor tank increased gamma ray radiation from the reactor core, since the moderator also had a shielding effect on the delayed gamma rays originating from fuel elements. Secondly, removing the heavy water around the fuel elements also reduced their cooling efficiency, since the heat transfer from the fuel was directed to the surrounding air in the RB reactor tank, instead to the heavy water. Third, it deprived us of a possibility to, later on, determine easily and precisely the heavy water level in the RB core achieved at the time of the accident.

In the various descriptions of the accident, there was no mention of irradiation damages to RB reactor components and systems, except for (in an internal report) those stating that, in some cases, fuel rods had small swellings on the surface of their claddings. This initial fuel cladding (assumed to be made of SAV-1) was replaced, May 1960-1962, with a new aluminium alloy fuel cladding made in Yugoslavia (known as Yu_Al). Unfortunately, this aluminium alloy contained highly absorbing neutron impurities (B, Cd) which excluded the possibility of the RB1/1958 core of ever again reaching the critical level. Calculations done by the KENO V.a code [34] give the value of k_{eff} = 0.99980   0.00021 for the maximum level (210 cm) of heavy water (0.18% mol H_2O and T = 22

°C) in the RB reactor tank and 208 fuel elements of natural uranium metal rods with a cladding made of Yu_Al in the lattice with a square pitch of 12.0 cm.

This was one of the main reasons that the RB reactor was upgraded, 1961-1962. The upgrade allowed RB reactor operation with a fuel of 2% enriched uranium metal slugs, known as the Soviet (Russian) TVR-S fuel type [30]. These fuel elements were also used at the Vinča Institute 6.5 MW heavy water RA research reactor designed in USSR.

Finally, the consequences of the RB accident were fatal for one RB reactor operator (coded VZ). He died due to radiation overexposure, four weeks [6] after he was checked in for medical treatment at the Curie Foundation hospital in Paris, France. All evaluations of the absorbed or equivalent doses, regardless of the methodology used, agree that this operator received the highest dose. According to an unconfirmed account, he was the operator who climbed (?!) to the top of the RB reactor to shutdown the safety rods manually and the one who, by so doing, put a stop to the accident.

EXPERIMENTAL EVALUATION OF THE ABSORBED DOSES

Immediately after the event, the International Atomic Energy Agency (IAEA) offered help in the evaluation of the accident. After the Government of Yugoslavia accepted the offer, in spring of 1960, IAEA prepared, organized and conducted the "Vinča Dosimetry Experiment" at the RB reactor with the aim of simulating accident conditions and estimating the doses received by the operators [5, 12]. The main participants of the dosimetry experiments carried out at the RB reactor in April 1960, besides the Yugoslav experts (from the Federal Nuclear Commission and RB reactor experts of the "Boris Kidrič" Institute of Nuclear Sciences), came from USA (ORNL), France (CEA, CEN de Saclay), UK, and the IAEA. The ORNL team has already been mentioned above, the French team included, among others, Dr. H. Jammet from CEA Saclay, the IAEA team experts such as G. W. C. Taft and R. Baker, while the UK sent J. W. Smith from AERE Harwell. The heavy water for the experiments was obtained from UK (AERE, Harwell), since the original heavy water was transferred to the 6.5 MW RA heavy water research reactor at the Vinča Institute which had begun operation on December 29, 1959. Similarities and experiences gained in the evaluation of the June 1958 accident at the USA ORNL Y-12 plant [15, 16, and 57] were used, too. Individual doses received during the accident at the Y-12 Plant have been evaluated again in 1984 [59] and, more recently, in 2006 [60].

The original control and safety systems of the RB reactor were found to be inappropriate for the operation of a RB reactor in said accident simulation experiments, in fact, unsuitable and unsafe for any operation of the RB reactor at powers of an order of a watt or more [5]. CEA, France, designed, manufactured and delivered new control and safety systems for the RB reactor in a couple of months [5]. A new, additional safety rod, which operated as a continuous heavy water level follower, was designed, manufactured and delivered, too. The construction of the RB reactor control room was finished by March 1960. The new equipment was installed and tested. In these accident simulation experiments at the RB reactor, during operations at high power levels, the operators in the RB control room were additionally protected from radiation coming from the RB reactor core with provisional walls made of concrete and lead blocks [5].

On the assumption that the RB reactor went to an accidental (exponential) power excursion at a initial power of 0.3 mW and that the total generated fission energy amounted to 80 MJ, it has been concluded [5] that the period, *i. e.*, RB reactor time constant, was about 10 s during the event, and that the duration of the excursion was about 400 s.

The absorbed neutron doses received by the operators were also estimated by a method [24] based on the measurement of activated ^{23}Na in a (n, γ) reaction taken from samples of the exposed operators' blood and tissue. This method of neutron dose evaluation for determining the ^{24}Na/^{23}Na ratio from human blood and tissue samples taken after irradiation has the advantage of being independent of the position of the personnel involved in the accident. In said accident simulation experiments, neutron doses have been evaluated using ^{24}Na activity in samples taken from irradiated ^{23}NaCl dissolved in water-filled, plastic models of humans – phantoms. The phantoms were irradiated by neutrons in two "high power" runs (of 1 kW and 5 kW, each lasting about 30 minutes) of the refurbished RB reactor. This method of determining the neutron absorbed doses by the ^{24}Na/^{23}Na ratio has also been re-evaluated later on [59-61].

Seven phantoms (of a Bomab, Calvin, and Tyrone type) placed around the RB core [5] were used. The probable positions of the operators at the time of the accident, most likely situated around the RB reactor core, are shown in the sketch given in fig. 4 (left), as their positions were determined by [5, 12]. These positions were considered "probable", because according to chapter 1 of the same document [5], four operators (not three, as was assumed in consequent accident simulation experiments and according to available written testimonies [17]), were actualy under the RB reactor tank, in the dry pool. The sketch of the positions of the four operators in the dry pool near the experimental equipment is shown in fig. 1(b) of the paper [6], drawn by the French physicians, and in fig. 4 (right) of this paper.

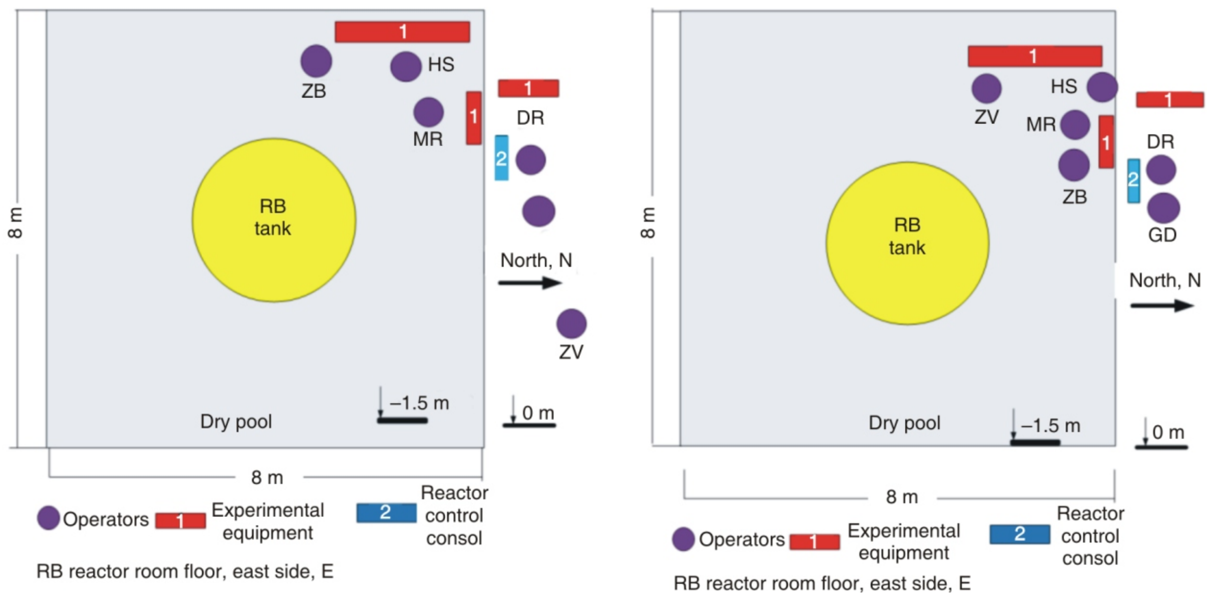


Figure 4. Sketch of the probable positions of the operators in the vicinity of the 1958 RB reactor core at the time of the accident, according to [5] (left) and [6] (right)

In this paper, the operators are coded by the initial letters of their family and first names. According to [6], operators coded HS, VZ, MR and BZ were in the dry pool, while operator DR (a female) was sitting behind the reactor control console, with GD standing beside her, to her left. Their precise positions and their distances from the RB reactor tank were not given in [6], but it was stated that two of the operators (behind the reactor control console) "were at a distance of around 4 m", while "the other four operators were grouped around the experimental equipment", in a corner of the dry pool, "at a similar distance" from the reactor core.The operators were between 24 and 28 years old, except for BZ, aged 34 [59]. Four technicians (DR, GD, BZ, and HS), as well as two senior undergraduates (apsolvents) of the Belgrade Faculty of Natural Sciences (ZV and MR), were present, too. No senior member of the RB reactor staff or a radiation expert happened to be in the reactor room at the time of the experiment.

In addition, an experimentally determined ratio of the absorbed gamma ray dose to the fast neutron absorbed dose ($D_\gamma/D_{\text{fast n}}$) for the various positions (LPS-1 ... LPS-10) of the operators in the facility was used, as well. This ratio is measured by using irradiated neutron threshold foils (made from Au, S, U, Np, and Pu) and the readings from the γ-ray sensitive ionisation carbon wall – CO_2 gas chamber [58]. The D /$D_{\text{fast n}}$ ratio was determined as an almost constant factor. Its value for positions of GD, DR, and BZ was 3.6, while in the case of positions of HS, VZ, and MR, the $D_\gamma/D_{\text{fast n}}$ factor was 4.1. The ratio was measured in horizontal axis as being at a distances of 4 to 7 m (in 1 m steps) from the RB reactor core. It is claimed [5] that this ratio of the gamma ray absorbed dose to the fast neutron absorbed dose is unchangeable at low and high power runs of any reactor. The said ratio is used to estimate the gamma ray absorbed dose. The ratio of the gamma absorbed dose to the neutron absorbed one has also been determined [5] by the two types of gamma dosimeters (based on the GM counter [62], a proportional ionisation chamber [58]) and the Radsan fast neutron dosimeter [38]. A sensitivity of the used gamma ray carbon-CO_2 ionization chamber to thermal neutrons not to be neglected was reported and has consequently been corrected [5], upon the conclusion of the experiments. The ionisation chamber's response (threshold) to high-energy gamma rays has not been reported, but for the gamma dosimeter based on the GM counter, according to [62], the gamma high-energy threshold amounted to a mere 1.5 MeV for gas at atmospheric pressure.

The energy border for the fast neutrons in the determined neutron spectrum escaping from the RB reactor core is not mentioned in [5] and is assumed to be, conventionally, 0.1 MeV. The neutron spectrum escaping from the RB reactor tank is also calculated using a multi-group, multi-regional, single-dimensional GNU-II computer code in diffusion approximation [39]. The obtained escaping neutron spectrum [5] is shown in the graph as $EN(E)$, *i. e.*, per unit of lethargy, with E_{max} = 10 MeV, in the function of neutron energy E (fig. 5).

The gamma ray spectrum escaping from the RB reactor core was analytically estimated upon the review of the attenuation of fission-prompt gamma rays through natural uranium metal rods and heavy water. In addition, a determination of the absorbed doses [5, 12], under the assumption that the neutron-generated gamma dose absorption rate from the H(n, γ) reaction

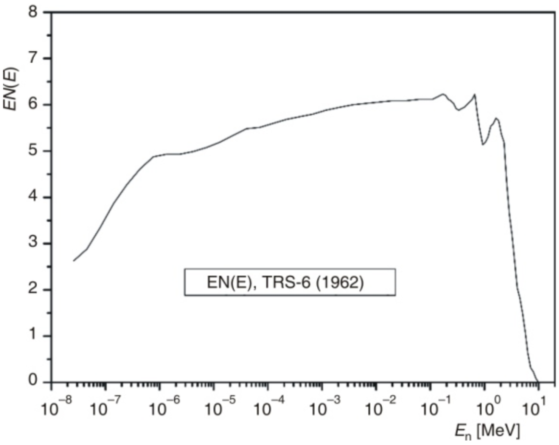


Figure 5. Spectrum of neutrons escaping the RB1/1958 core obtained by the GNU-II code

in the human tissue was exactly 1.5 times higher than the neutron absorbed dose, was made. The IAEA-assembled international team has estimated that the overall uncertainty in the total absorbed doses was within a 15% margin, based on the applied methodology, accepted approximations, unconfirmed duration of the accident and unclear positions of the operators.

The total generated fission energy in these simulations of the RB reactor accident, in the two runs of the RB reactor, was 3 kWh: approximately 7.5 times less than the total fission energy (80 MJ) released during the accident. It was assumed that such a simple scale of generated fission energy is valid. After the evaluation of doses received by the phantoms and corrections due to the issues mentioned above, as well as the respective masses of the operators [5], the absorbed doses were associated to each individual, as shown in tab. 2. The doses received by the operators, estimated by Savić [3] and the French physicians' group [6, 9] are shown in tab. 3. The abbreviations used are: "n" for neutrons and E for neutron energy.

An additional evaluation of the doses absorbed by the operators is shown in [49]. It is based on the activity of ^{24}Na taken from the blood samples of the irradiated personnel and on the assumption that the neutron spectrum escaping the RB reactor was such that the thermal and epithermal components were equal. Moreover, the authors have assumed that the total escaping neutron spectrum was equal to the experimentally determined epithermal (>5 keV) spectrum in the Y-12 plant [15, 16] and that the thermal flux density equalled the total flux density above 5 keV. Based on the measured data for ^{24}Na activity and the assumption that the bodies of each operator contained 105 g of ^{23}Na, the neutron absorbed doses were determined. The gamma ray dose was estimated from the relation between the known (measured) intensity of the thermal neutron flux density (155 /cm^{2}s) and the exposition dose of gamma rays (1 mR/h, 1 R = 2.58 10^{-4} C/kg) at the RB reactor [10], judged to be equal to that of the neutron absorbed dose. The results of this evaluation [49] are given in tab. 4, along with ambient equivalent doses published initially by Pendić [51]. It can be seen that the doses given by Pendić (shown in "rem") are equal to the ones given in [49] (shown in "rad") with RBE = 1, which is obviously incorrect.

The re-evaluation of the published doses in [5, 12] was done in 1984 [59]. Based on the contents of ^{23}Na in the tissue of the operators and ^{24}Na activity measured in blood samples taken from the exposed individuals, it was concluded that the published doses

Table 2. Doses attributed to the operators after the accident simulation experiments of April, 1960 [5, 12]

Operator (code)	Phantom (type)	Operator mass [kg]	Neutron dose [rad]	^{1}H(n, γ)^{2}H dose [rad]	Gamma ray exposure dose [rad]	Total (15%) absorbed dose [rad]
HS	Tyrone	65	66	99	158	323
VZ	Calvin	80	89	133	214	436
GD	Remab	70	90	135	189	414
MR	Calvin	72	87	130	209	426
DR	Tyrone	52	91	136	192	419
BZ	Remab	90	45	67	95	207

Note: the values are reproduced in old units for the absorbed dose (100 rad = 1 Gy), as they were reported in [5]

Table 3. Doses of the operators estimated by Savić [3] and the French physicians' group [6, 9]

Operator (code)	Savić, dose [rem]					French physicians' group, dose [rem]		
	Fast n, E > 1 MeV	1/E n, thermal –1 MeV	Thermal neutrons	Gamma rays	Total 15%	Neutrons	Gamma rays (extreme range)	Total (range)
VZ	116	223	49	295	683	210	630 (450-1000)	840 (1000-1200)
MR						214	642 (450-1000)	856 (700-1000)
GD						230	690 (450-1000)	920 (700-1000)
DR						256	768 (450-1000)	1024 (700-1000)
HS						174	522 (450-700)	696 (600-800)
BZ						102	306 (250-600)	408 (300-500)

Note: the values are reproduced in old units for the ambient dose equivalent(100 rem = 1 Sv), as they were reported in [6, 9]

Table 4. Doses attributed to the operators after the evaluation done by ORNL in May 1961 [49] and ones initially published by Pendić [51]

Operator (code)	Neutron dose [rad]	Gamma ray dose [rad]	ORNL [49] total absorbed dose [rad]	Pendić [51] total dose [rem]
HS	210	210	420	420
VZ	320	320	640	640
GD	300	300	600	600
MR	290	290	580	580
DR	250	250	500	500
BZ	175	175	350	350

Note: the values are reproduced in old units for the absorbed dose (100 rad = 1 Gy), and ambient dose equivalent (100 rem = 1 Sv), as they were reported in [49] and [51], respectively

required corrections. Dose estimates in [5, 12] are based on the assumed average (for all individuals) concentration of ^{23}Na per 1.5 g/kg of body mass ([48] for the Reference Man weighing 70 kg. However, this concentration was found in the range of 1.00 g/kg (woman) to 1.04 g/kg (men), depending on the individual [59]. Another issue requiring attention was the statement that the dose component attributed to the H(n, γ) reaction "was probably too large, by a factor of 2" [12]. The corrections in the published total doses require (according to [59]), an increase of 40% for male individuals and that of 50% for the female operator, *i. e*., a factor of 1.29-1.70, depending on the operator in question. On average, the values of the total doses given in [5, 12] for the four largest doses, should be increased by a factor of 1.31 to 1.34, according to [59]. The new values of the corrected absorbed doses are not given in [59], but can be easily calculated for each individual because of the evaluation done in [5, 12] which assumes the linear dependence of the gamma ray absorbed dose on the neutron absorbed dose of the individual. The published values of absorbed doses from [5, 12], shown in tab. 2, are corrected in this paper, according to the reevaluation [59] for the ^{24}Na/^{23}Na contents and given in tab. 5. In another reference [61], based on previous re-evaluations, it was also concluded that "... the doses at Vinca were much higher than those assumed earlier".

Table 5. Doses attributed to the operators after [5, 12] and corrections by Mole in [59]

Operator (code)	Correction to ^{24}Na/^{23}Na	Neutron dose [rad]	^{1}H(n, γ)^{2}H dose [rad]	Gamma ray exposure dose [rad]	Total (±15%) absorbed dose [rad]
HS	1.35	89	134	214	437
VZ	1.41	125	188	298	611
GD	1.29	116	174	245	535
MR	1.35	117	176	281	574
DR	1.29	117	176	247	540
BZ	1.70	76	115	160	351

Note: the values are reproduced in old units for the absorbed dose (100 rad = 1 Gy), as they were reported in [5]

During the accident, the RB reactor was operated, as mentioned above, by the six operators present in the reactor room, at the RB reactor control console and racks of the experimental equipment, as shown on the right (north, N), in fig. 1. According to [5], three operators (VZ, GD, and DR) were at the northern (N) side of the RB reactor room, at floor level (level 0.0 m), close to the reactor control console at the edge of the dry pool. The three remaining operators (coded HS, MR, and BZ), given in northwestern corner of the dry pool (–1.5 m level), close to the experimental equipment. Two of these six operators (DR and MR), were sitting, the first one in front of the reactor control console and the other one in front of the experimental panel.

As can be seen in tabs. 2-5, the absorbed doses are given in "rad" units in the case of the IAEA international team [5, 12], ORNL team [49] and corrections done by Mole [59]. It should be noted that the absorbed doses from different radiations are not supposed to simply sum in respect to the biological effects in human tissue. These absorbed doses are shown here in "rad" units with the aim of preserving compatibility with the results of the previously published data. The said absorbed dose were different of their ambient dose equivalents (shown in "rem" units) were given by the Savić [3], Pendić [51], and French task group [6]. Some of the values were estimated after the operators had received medical treatment (under the auspices of Dr. G. Mathe) in the Paris "Maria Currie" Foundation hospital and after the accident simulation experiments at the "Boris Kidrič" (now Vinča) Institute of Nuclear Sciences [5].

Anyway, the said accident simulation experiments were defined by H. Jammet as representing "a great contribution, another brick built into the edifice of medical research" [53]. IAEA Director General and the then Yugoslav Undersecretary of State, in a joint statement from 1960, stated "... We are convinced that this will be of great value to all mankind." [54].

THE MCNP5 COMPUTER CODE 3-D MODEL FOR NUMERICAL SIMULATION OF DOSES RECEIVED IN THE ACCIDENT

As for the numerical experiments concerning the determination of doses received by the operators in the RBI/1958 core accident of October 15, selected information from [5, 12] regarding the positions of the operators and information extracted from data on the dependence of the generated fission energy time (fig. 3) were selected from [7, 8]. The Monte Carlo computer code MCNP5 (version 1.60), with neutron cross sections based on the evaluation of the ENDF/B-VII.0 library was applied. Libraries of cross sections on the interactions of gamma rays and electrons, distributed by the MCNP5 computer codes MCPLIB04 and

EL03, respectively, were used as well. The energy range of radiation used in the MNCP5 calculations along with the cross sections of these libraries were selected out of 0.01 meV to 20 MeV ranges for neutrons and those of 10 keV to 20 MeV for gamma rays and electrons.

An updated thermal neutron scattering library (TSL) ENDFB70SAB for ENDF/B-VII, based on $S(\alpha, \beta)$ laws for neutron scattering at hydrogen atoms bounded in H_2O molecules and deuterium atoms bounded in D_2O molecules, is used. Cross-section data in TSL are evaluated at 293.6 K and applied in the entire neutron thermal energy range. Heavy water is used at a temperature of 22 °C with 0.24% (mol) light water, while all other materials are used at a temperature of 20 °C, except for the human tissue material that is used at 37 °C. Neutron and gamma ray transport and interactions are done by the MCNP5 computer code in a 3-D model in all cells with materials. Electron transport and interactions, including the bremsstrahlung, are done by the MCNP5 computer code only in cells of the phantoms containing human tissue material. Some impurities in the materials do not have gamma ray production cross-sections in the ENDF/B-VII.0 library (*e. g.* ^{17}O, ^{40}Ar, so that natural Ar in air, isotopes of Cd and natural Zn are used instead). This version of the MCNP5 computer code is not capable of generating delayed gamma rays from fissions.

The calculations by the MCNP5 computer code were done at a four Intel i7-processor Toshiba Satellite laptop A660, with a 64-bit Windows 7 Home Premium Operating System. The MCNP5.1.60 computer code was run in the "mode n p e", with KCODE and TOTNU options. The initial neutron source (KSRC option), originating from each fuel element in the RB core, was used.

All materials pertaining to the RB reactor were used with known impurities [30], while the material for human tissue was used according to the ICRP (1959) recommendation for components in the soft tissue of the Standard Man [40], with 11 main elements, slightly modified [41]. The density of this tissue-equivalent material was 1.063 g/cm^3, and the temperature, as already mentioned, 37 °C.

The platforms and the supporting construction of the RB reactor tank are neglected in the 3-D model used in MCNP5 computer code calculations. The RB1/1958 reactor core is modelled in 3-D, with the level of heavy water at 183.0 cm. Even the part of the natural uranium metal rods in the air, above the moderator level in the reactor tank and the tank top cover, are modelled in a 3-D model. All instruments and reactor equipment placed on the top cover and the platforms of the reactor are neglected. It is also assumed that the walls, floor and ceiling of the RB reactor room were constructed from ordinary concrete, with a material composition (NBS ordinary concrete) taken from [42] and a density of 2.35 g/cm^3. The large glass windows, initially designed and constructed for the corridor in the reactor building at the lower part of the north wall of the reactor room (at the right side in the photograph, fig. 1), were replaced by a concrete wall after the accident, in early 1960, and modelled in a 3-D model as a wall made from ordinary concrete.

The 3-D model used in MCNP5 calculations also includes data indicating that the RB reactor was operated by six operators present in the reactor room. In the said 3-D model, the seven phantoms are designed in the positions they probably occupied at the time of the accident, in the vicinity of the RB reactor, as shown in the sketch given in fig. 4 (left). The analyses of positions in question show that the phantoms were placed at distances of 5.4 m to 6.9 m from the RB reactor tank bottom, which differ from values given (estimated) in [14] and in [49]. In [14], the distances of the operators from the reactor tank are estimated as 4 m for the operators at the northwestern corner of the dry pool in the vicinity of the experimental equipment and as 6 m for operators in the vicinity of the reactor's control console, at the north edge of pool. Their values were estimated to be 5 m and 10 m, respectively, [49].

Numerous phantom models are known and used nowadays [46]. Because they include various human organs, they were considered too complex for this study. The idea was to make a simple 3-D model of the operators, similar to the phantom (Bomab) used in [5], so as to reduce the overall calculation time of the computer code. Thus, the operators modelled in 3-D, were presented as simplified, homogeneous, sexless and faceless phantoms, based on average human proportions given in Chapter 9 of [43]. As already mentioned, these 3-D models were made of a tissue-equivalent homogeneous liquid composition based on the 11 main elements of the human body [41, 48]. No human organ was modelled in phantoms considered in this study.

The model of the "standing phantom" has a height of 168 cm, while the "sitting phantom" was modelled as being 144.5 cm tall. Both phantoms have a mass of 70 kg, a volume of 65904 cm^3, and a total surface of 20685 cm^2. Each of the 3-D models has a separately designed head, neck, trunk, arms (made in one piece), and legs (made of two separate pieces). Each body element of the 3-D model phantom is designed as a cylinder, except for the torso, modelled as a parallelepiped. The seventh, reference phantom (RF), as in the IAEA experiments done in April 1960 [5], was placed in the northeastern (NE) corner of the dry pool, as a reference. Graphical images of 3-D models of the "standing phantom" and the "sitting phantom" are shown in fig. 6.

The positions of the phantoms in the RB reactor room relative to the RB tank in the 3-D model developed for the MCNP5 numerical simulations are shown in fig. 7. The RB reactor control console and the experimental panels in the RB reactor room were not modelled. In dose evaluations by the MCNP5 computer

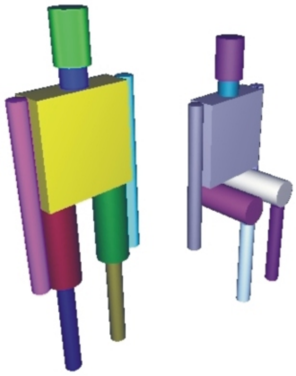


Figure 6. Models of simplified, homogeneous, sexless and faceless standing and sitting phantoms

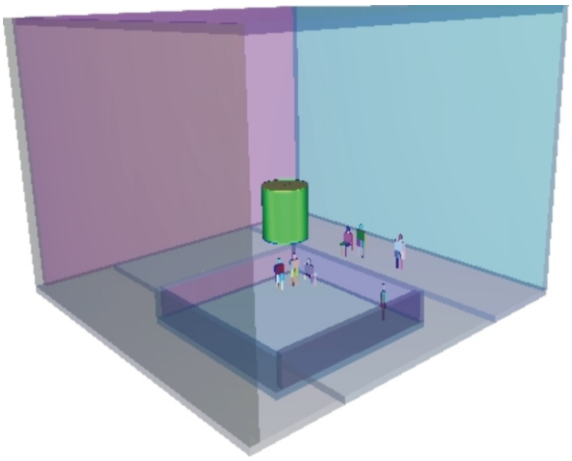


Figure 7. MCNP5 model of positions of the operators around the RB tank in the reactor room

code, apart from mass corrections, no corrections for the real height of the operators in relation to the height of the phantoms were done.

As in other methodologies, the MCNP5 computer code estimation of the doses received by the operators has to do with the knowledge of the time they had spent in the mixed neutron – gamma ray radiation field of the RB reactor at the the time of the accident. As already mentioned, in the short period of time the accident lasted, the power of the RB reactor shifted from the mW range to that of 2.5 MW. Such time dependence of radiation flux density (neutron and gamma rays escaping the RB tank), related to the power of the RB reactor, is impossible to simulate in the MCNP5 computer code. Because of this, for the purpose of estimating the doses received, the MCNP5 computer code was run for an equivalent of the RB reactor's stationary power, for a time interval chosen to allow for the exposition of the operators to a high range of doses, up to the total generated fission energy of 80 MJ. This methodology of dose determination is also valid under the assumption that all of the operators held stationary positions ("were frozen") during the said exposition time t_{irr}.

Obviously, since the absorbed doses (determined by other methods) were in the range of a few hundred 'rad' and the fact that the power-time relation underwent an exponential change, it is clear that, in the last stage of the power excursion, such high doses must have been received by the operators. In this study it is assumed that the exposition to high doses was initiated when the generated fission energy achieved 0.1% of the total generated fission energy (*i. e.*, 0.001   80 MJ). From data used to plot $E_f(t)$, shown in fig. 3, this time moment was found to be t_{d1}(80 kJ) = 337 s. The accident was interrupted in t_{acc}(80 MJ) = 433 s, when the power rise was discontinued by the shutdown of the reactor with safety rods. Thus, the effective irradiation time to high doses was estimated as t_{irr} = 433 s – 337 s = 96 s. The equivalent stationary power, P_{eq}, of the RB reactor is then 80 MJ divided by 96 s = 833.333 kW. The MCNP5 computer code neutron flux density normalized constant, F_{eq}, can be determined from the equation

$$F_{eq} \quad \frac{P_{eq}\,\nu}{E_f C_f k_{eff}}$$
$$7.041953 \quad 10^{16} \text{ per cm}^2\text{s}$$

where ν = 2.456 (MCNP total number of neutrons generated per fission in the RB1/1958 core), E_f = 180.88 MeV (MCNP energy generated per ^{235}U fission), C_f = $1.602 \cdot 10^{-13}$ J/MeV (conversion factor from units MeV to units J), and k_{eff} = 1.003 (effective multiplication factor for the RB1/1958 core at a heavy water level of 183 cm – assumed D_2O level).

This constant is used in the MCNP5 computer code (FM option) for the normalization of neutron, gamma ray and electron flux densities. According to MCNP5 code calculations (with the exception of reactor room walls), in all energy ranges, the escaping gamma ray current rate from all the surfaces of the RB tank is almost the same as the escaping neutron current rate from the RB tank itself. In other words, the ratio of neutrons to gamma rays, as mentioned above, is 1.1:1. The calculated ratio of the flux density of escaping neutrons to the flux density of escaping gamma rays is 1.4:1, along the entire energy range of neutrons and gamma rays.

The doses were determined in the MCNP5 computer code using F4 and F6 tallies [28]. For the F6 tally, the deposited energy of neutrons and gamma rays (including photon-generated electrons) in human tissue material (phantoms), the value of unit's conversion factor (FM option) was calculated as $11.281209 \cdot 10^8$. Using this factor, the F6 tally result is directly obtained in units of rad/s (absorbed dose rate), if the tissue mass of the operator is inserted into the code (as a SD option), in grams. The total absorbed dose in the tissue of the phantom is obtained by multiplying the F6 results for radiation dose rates (given in rad/s) with the estimated time of exposition t_{irr} (96 s) and the corrected individual masses. The correction for an individual mass is obtained by simply multiplying the calculated dose with the value of the ratio of the individual mass to the mass of the reference phantom (70 kg).

For the F4 tally, the neutron flux density (fluence rate) was first calculated in all cells, including the phantom, in several neutron energy groups. Neutron energy 55-group structure was selected to match the neutron group structure used in the VEGA [44] computer code, with a maximum neutron energy of 20 MeV. As for the VITAMIN-E library [45], the gamma ray energy 35-group structure, with a gamma ray energy maximum of 15 MeV, was the one selected. Since the interactions of gamma rays in human tissue may create electrons, an electron energy 10-group structure of up to 10 MeV was selected, with the first energy bin of up to 1 MeV.

In order to convert the radiation flux density (given by F4 tallies) into the absorbed dose rate, ambient dose equivalent conversion factors per unit of neutron fluence in the function of radiation energy were used, taken from tables A.21, A.42 and A.44, given in ICRP-74 [46], for gamma rays, neutrons, and electrons, respectively. Since none of the human organs were modelled in the phantoms, in order to obtain the neutron absorbed dose rate, neutron factors from table A.42 were converted by dividing them with the value of the neutron tissue weighting factor, $w_T(E_n)$, given in the function of neutron energy E_n (MeV). Dimensionless, $w_T(E_n)$ factors, are to be found in the chapter entitled "Quantities" of ICPR-7 [46], given for neutrons with approximate relations

$$w_T(E_n) \quad 5 \quad 17\exp \quad \frac{[\ln(2E_n)]^2}{6}$$

The radiation weighting factor, w_T, for gamma rays and electrons of all energies, is 1.0 [46]. Then, the total absorbed dose in the tissue of the phantoms is obtained by simply multiplying the F4 sum[1] results for radiation dose rates (given in rad/h) with the estimated time of exposition t_{irr} (96 s), corrected for each individual mass. Again, the correction for an individual mass is obtained by simply multiplying the calculated dose with the value of the ratio of an individual mass to the mass of the reference phantom (70 kg).

The effective doses (given in rem) received by the operators were also calculated in the MCNP5 computer code using the F4 tally and neutron fluence – effective dose conversion factors taken from Table A.41 of the ICRP-74 [46] for neutrons. They equalled 1.0 for gamma rays and electrons, *i. e.*, proved to be independent of the energy of particular radiation types. Factors affecting the neutrons were also applied in the computation of the AP (anterior – posterior) geometry [46], regarding phantoms 3, 4, 5, and 6, supposed to have been facing the RB reactor (fig. 7). In addition, the same factors were applied in the PA (posterior – anterior) computation geometry [46] for phantoms 1, 2, and 7, supposed to be exposed with their backs turned to the radiation source – the RB reactor (fig. 7). The said phantom positions should, as close as possible, correspond to the actual geometry (positions) of the operators in the RB reactor room at the time of the accident. The calculated doses are a sum of the values of radiation doses obtained after the F4 tally dose rates were multiplied by the time of the exposition and the applied corrections for individual masses compared to those of the reference phantoms.

RESULTS AND DISCUSSION

The MCNP5 computer code was run for a total of 1000 million neutron histories, a 1 000 000 histories in each of the 1000 active cycles, after the initial 100 cycles. The said number of histories has provided us with the certainty that the dose rates obtained from the F4 tallies have a statistical relative uncertainty of 1 (standard error), less than 1.5%, for neutrons, gamma rays and electrons, respectively. At the same time, the dose rates obtained by the F6 tallies exhibited a statistical relative uncertainty of a 1 (standard error), less than 0.4%. The run of the MCNP5 code with the ENDF70 library took, in a parallel mode, several days on four i7 processors (1.73 GHz each) laptop for the mentioned 1 G neutron histories.

Neutron and gamma ray (photon) current spectra (J) escaping the RB reactor tank, normalized per unit of lethargy, $u = \ln(E_{g+1}/E_g)$, calculated by the MCNP5 computer code for 1000 million neutron histories, are given in the function of energy (E) in figs. 8 and 9, respectively.

The average energies of neutron or gamma ray spectra escaping the RB core, shown in figs. 8 and 9, were calculated as $\langle E \rangle = \ _g(E_g\Psi_g)/ \ _g(\Psi_g)$. For neutrons, they were found to amount to 0.15 MeV, for gamma rays, 1.12 MeV. In the relation cited above, these ($_g$) are over energy group g, quantity Ψ_g is the radiation (neutron or gamma ray) group current or flux density in group g with a group energy E_g. The label "total" in the legend accompanying figs. 8 and 9 is the MNCP5 calculated group neutron or gamma ray current of flux density averaged over all surfaces of the RB reactor tank. The total number of neutrons (#NG) and photon groups (#PG) and corresponding energy ranges are given in figs. 8 and 9. These figures also include the relative uncertainty (in percentage points) of the calculated radiation quantity and the total number of neutron (n) histories per computer code run.

[1] Please note that the absorbed doses from different radiations cannot be simply added to ensuing biological effects

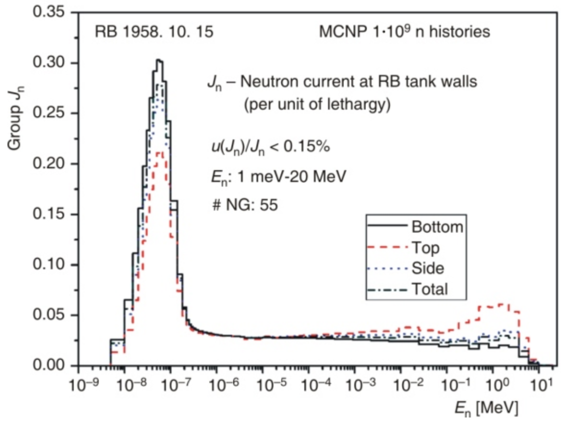


Figure 8. Group neutron current spectra escaping the RB reactor tank walls

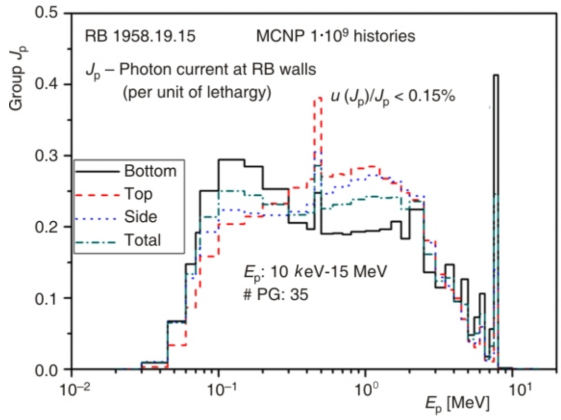


Figure 9. Group photon current spectra escaping the RB reactor tank walls

Neutron and gamma ray (photon) flux density (F4 and F2 tallies) and current (F1 tally) normalized spectra escaping the RB reactor tank, calculated by the MCNP5 code for 1000 million neutron histories, as given in the function of energy in figs. 10 and 11, respectively. The analyses of this 55-group neutron flux density spectra (F4 tally) escaping the RB tank has shown that the thermal (<5 keV) component is 4.6-4.8 times higher than the epithermal component (>5 keV) and that the assumptions made at the ORNL evaluation [49] is not valid.

Neutron and gamma ray (photon) flux density spectra in the tissue of the phantoms, normalized per unit of lethargy, after MCNP5 calculations done for 100 million histories, are given in figs. 12 and 13. The figures show that there is, practically, no difference between (neutron or gamma ray) spectra impacting phantoms placed at different distances from the reactor core.

Analyses of figs. 12 and 13 show that the neutron spectrum reaching the tissue of the phantoms is pre-dominantely thermal, while the gamma ray spectrum in the tissue of the phantoms exhibits three "peaks". One wide peak around 0.1 MeV, a second narrow peak around 2.5 MeV and a third narrow peak around 8 MeV. An additional peak of 0.5 MeV, in the spectrum of the photon current escaping the RB reactor core, can be seen in fig. 9, as well. Such a spectrum is a consequence of the multiplicity of the source of the gamma ray emission from the RB reactor core. The flux density spectra are similar in shape to the spectra of the current of respective radiations, given in figs. 8 and 9, since their transport from the RB reactor tank walls to

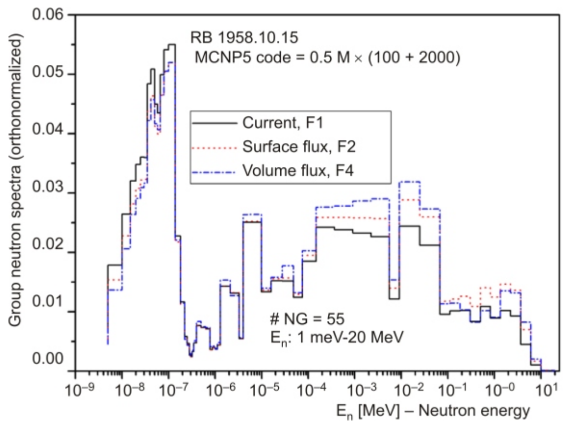


Figure 10. Spectra of the group neutron flux density and current escaping the RB reactor tank walls

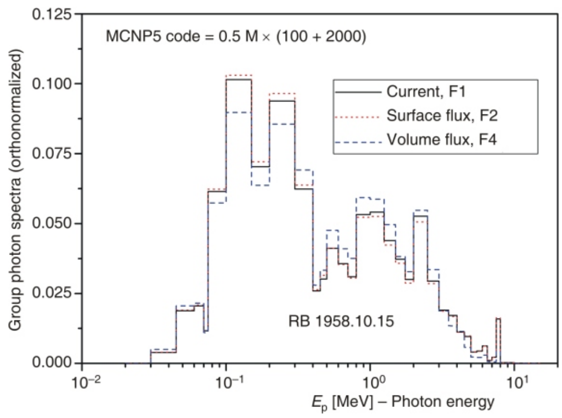


Figure 11. Spectra of group photon flux density and the current escaping the RB reactor tank walls

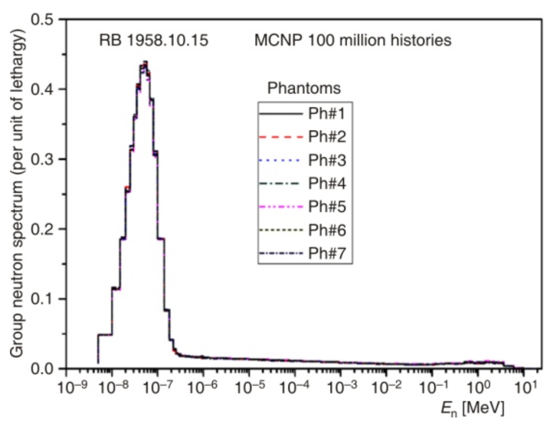


Figure 12. Group neutron flux density spectra in the tissue of the phantoms

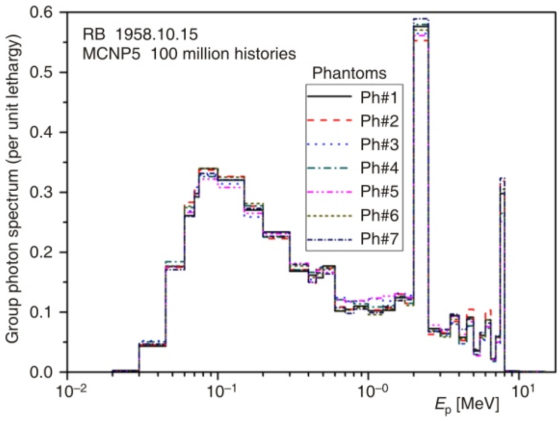


Figure 13. Group photon flux density spectra in the tissue of the phantoms

the phantoms is done through the air in the reactor room.

The absorbed dose rates of the phantoms, obtained by the F6 tally for deposited neutron and gamma ray energies in the tissue of the phantoms, are multiplied by exposition time, t_{irr} = 96 s, and corrected for the individual operator's mass, so as to obtain the doses attributed to the operators. The calculated results are shown as the total calculated absorbed dose of individuals D_{F6} (given in rad) in tab. 6. The uncertainty, $u(D_{F6})$, expressed as a statistical 1σ standard error, and the ratio of the calculated total doses to the total doses given in [5, 12], and in [49], as well as those calculated after corrections found by Mole [59], are given too. The calculated absorbed doses, obtained by the F6 tally, are found to be around 50% (except for BZ) of the published absorbed doses. Explanations for these discrepancies are given in the text following tab. 8 and summarized in Conclusions. Note that data for absorbed dose are not given for RF in reports [5, 12, and 49].

The absorbed dose rates of individual phantoms obtained by the sum[2] of the F4 tally for contributions of neutron, gamma ray and electron radiation in the tissue of the phantoms are multiplied by irradiation time, t_{irr} = 96 s, and then corrected for the individual's mass, so as to obtain the absorbed doses attributed to individuals. The said doses are shown as the total calculated absorbed dose of individuals D_{F4} (given in rad) in tab. 7. The value of the ratio of the calculated total dose to the total dose reported in [5, 12], [49] and after corrections found by Mole in [59], is given too.

As can be seen, the calculated total absorbed dose of the operators is, in most cases, within 25% of the total absorbed dose determined in [5, 12] and within 50% of the ones reported in [49]. This can be attributed to the assumptions, approximations and uncertainties applied in the input data for MCNP5 computer code calculations. Outside of this range of uncertainty are the calculated total absorbed doses for phantom #1 (BZ), phantom #2 (HS) and phantom #3 (DR). If the reevaluation, done in [59] and briefly presented in chapter 4 of this article is recalled, calculation results for the absorbed doses of individuals (tab. 7) by the MCNP5 computer code are still quite different from previously reported dose values.

For phantom #1, the calculation result is unexpected, since this phantom is attributed to the operator (BZ) with the highest mass (90 kg), compared to phan-

Table 6. MCNP5 calculated absorbed doses according to the deposited energy (F6) in the tissue of the phantoms

Phantom number (#)	Operator code	Total D_{F6} [rad] MCNP5	$u(D_{F6})$ [%]	$D_{F6}/D_{[5, 12]}$	$D_{F6}/D_{[49]}$	D_{F6}/D_{Mole}
1	BZ	479.27	0.35	2.32	1.37	1.37
2	HS	311.65	0.26	0.96	0.74	0.71
3	DR	265.81	0.20	0.63	0.53	0.49
4	GD	350.66	0.26	0.85	0.58	0.66
5	VZ	362.99	0.32	0.83	0.57	0.59
6	RF	339.69	0.27	–	–	–
7	MR	352.16	0.29	0.83	0.61	0.61

Table 7. MCNP calculated (F4) absorbed dose according to the radiation type in phantom tissue

Ph. #	Operator code	$D_{n,F4}$ [rad] $\pm u\ D_{n,F4}$	$D_{\gamma,F4}$ [rad] $u\ D_{\gamma,F4}$	$D_{e,F4}$ [rad] $\pm u\ D_{e,F4}$	$D_{t,F4}$ [rad] $u\ D_{t,F4}$	$D_{t, F4}/D_{[5, 12]}$	$D_{t, F4}/D_{[49]}$	$D_{t, F6}/D_{Mole}$
1	BZ	243.04 0.61%	420.26 1.22%	67.96 0.75%	731.26 1.56%	3.53	2.09	2.08
2	HS	155.89 0.47%	276.31 0.83%	45.19 0.52%	477.38 1.09%	1.48	1.14	1.09
3	DR	134.23	232.11 0.67%	33.44 0.39%	399.78 0.85%	0.95	0.80	0.74
4	GD	183.99 0.48%	302.86 0.85%	44.17 0.53%	531.02 1.11%	1.28	0.89	0.99
5	VZ	186.09 0.52%	313.75 0.94%	46.41 0.51%	546.61 1.19%	1.25	0.85	0.89
6	RF	178.22 0.46%	297.63 0.89%	48.37 ± 0.56%	524.22 1.15%	–	–	–
7	MR	174.27 0.45%	314.80 0.98%	52.19 0.58%	541.26 1.22%	1.27	0.93	0.94

[2] Please note that the absorbed doses from different radiations cannot be simply added to respective biological effect

tom #6 with a reference mass of (70 kg). In the evaluation done in [5], the lowest absorbed dose is attributed to this operator, which is unexpected, again. The explanation for this discrepancy between the calculated and the attributed total absorbed dose for this operator is found in testimonies [17] and [6] – the operator was not exposed to radiation for the same time interval as the other operators present in the RB reactor room. In his testimony [17], author (HS) wrote "... I suspected that the scaler, perhaps, did not show acceptable values and asked BZ, who was in charge of the electronic (equipment), to go to the next room and fetch a spare (scaler)"[3]. If this, as written in [6] was the case, BZ could not have been exposed to the radiation at the same time as the other operators or had not occupied the same position as they did.

As for the operator coded HS, (phantom #2), the discrepancy is probably a result of the imprecise position he occupied, given in plots [5] and [12], shown in fig. 4. Another discrepancy is the one regarding operator MR (phantom #7) – according to one testimony [17], he was standing, not sitting, as in [5] and in the MNCP computer code 3-D model. The discrepancy concerning the third operator (DR) in the calculated total absorbed dose and the one found in [5], can be explained by the fact that between this (sitting) operator (with the lowest body mass) and the RB reactor tank, stood the RB reactor control console which acted as some sort of shielding. Moreover, according to the testimony of HS [17], the operator (DR) at the RB reactor's control console, (after the odour of ozone appeared) "... ran to check the reactor (valves of the)[3] pump..." The "pump valves" were under the reactor tank's bottom, in the dry pool. Therefore, DR changed his position from that of being at a greater distance from the reactor tank, to that of bringing herself closer to the source of irradiation. This operator "went back to the reactor control console and manually shut down the RB reactor by safety rods" [17].

This testimony is in conflict with the account that the accident was ended by operator VZ, VZ being attributed the one who had actually shut down the safety rods. According to an oral account, he did it from the top of the reactor (?), or more realistically, as operator MR recalled in his oral testimony [70], at the reactor control console. If this account holds true, operator VZ had to cross the path from his position in the pool (fig. 4, left) or from the floor of the reactor room (fig. 4, right) heading to the top of the reactor or the reactor control console. In addition, according to a testimony [5], these actions were carried out after the appearance of the odour of ozone. Therefore, it was appropriate to make a rough estimation of the time moment when the odour of ozone was sensed by the operators in the reactor room. This estimation is shown in Appendix A.1. In [5] and the 3-D model for the MCNP5 computer code, operators BZ and DR are standing still, since they are "frozen" in their positions, like the rest of the operators.

In [6] it is stated: "In addition, certain individuals moved around during the course of the exposure. MR, GD, DR, and HS kept more or less to one place, while VZ approached the reactor after the shutdown, exposing himself to additional exposure; during the accident, BZ left the room for three minutes and, in so doing, reduced his rate of exposure by about a half". This description [6] supports the explanation for the obtained discrepancies between the calculated dose results for BZ, DR and VZ and the doses attributed to these individuals in published evaluations. Obviously, for any calculation of doses, the precise positions of the operators are crucial, apart from the time of their exposure to radiation. It is quite impossible to simulate, in the MCNP5 computer code, a situation in which the operators move around a RB reactor core. The ratio of the total absorbed dose calculated by the MCNP5 computer code to the absorbed dose for operators estimated after applied corrections [59] found by Mole are shown in the last column of tab. 7. These $D_{t,F4}/D_{Mole}$ ratios are within the expected 10% (except for BZ and DR).

Ambient dose equivalents (shown in rem) received by the operators were also calculated by the MCNP5 computer code using the F4 tally and neutron fluence – equivalent dose conversion factors taken from table A.42, ICRP-74 [46] for neutrons. These con-

Table 8. MCNP5 calculated (F4) ambient dose equivalent in tissue of phantoms according to radiation type

Ph. #	Operator code	$D_{n,F4}$ [rad]	$u\ D_{n,F4}$	$D_{\gamma,F4}$ [rad] $\pm u\ D_{\gamma,F4}$		$D_{e,F4}$ [rad] $u\ D_{e,F4}$		$D_{t,F4}$ [rad] $\pm u\ D_{t,F4}$		$D_{t,F4}/D_{[3]}$	$D_{t,F4}/D_{[6]}$
1	BZ	1886.42	0.41%	420.26	1.22%	67.96	0.75%	2299.33	1.49%	3.37	5.64
2	HS	1173.08	0.43%	276.31	0.83%	45.19	0.52%	1524.85	1.07%	2.23	2.19
3	DR	1056.96	0.42%	232.11	0.67%	33.44	0.39%	1380.70	0.88%	2.02	1.35
4	GD	1459.22	0.41%	302.86	0.85%	44.17	0.53%	1806.25	1.08%	2.64	1.96
5	VZ	1489.13	0.44%	313.75	0.94%	46.41	0.51%	1818.62	1.16%	2.66	2.17
6	RF	1369.62	0.42%	297.63	0.89%	48.37	0.56%	1715.63	1.13%	–	–
7	MR	1307.26	0.43%	314.80	0.98%	52.19	0.58%	1667.42	1.22%	2.44	1.95

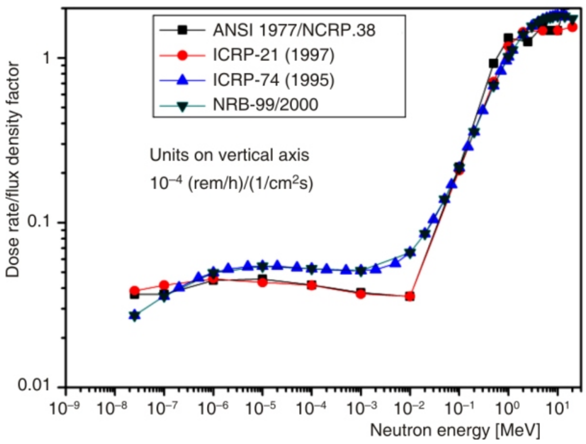
[3] Note that in the quotes, the author of this paper has inserted words in brackets in order to clarify the quotes

version factors are taken from tables A.21 and A.44 of the ICRP-74 publication for gamma rays and electrons. The ambient dose equivalent rate of individual phantoms is obtained as the sum of the F4 tally results for the contributions of neutron, gamma ray and electron radiation in the tissue of the phantoms, multiplied by exposition time, t_{irr} = 96 s, and corrected for the individual's mass. These doses are shown as the total calculated ambient dose equivalent of individuals D_{F4} [rem] in tab. 8. The ratio of the calculated total doses to the total doses given in [3] and [6] are shown, too.

Table 8 shows that the calculations of equivalent doses by the MCNP5 computer code give roughly twice higher doses in individuals (with the exception of BZ) than the ones reported in [3] and [6]. It is believed that such high ratios are a consequence, besides the reasons mentioned above, of the fact that applied new fluence – equivalent dose factors include a corrected, higher contribution of neutrons from the intermediate and fast (>1 MeV) energy range to the dose than was with earlier versions of these factors, given *e. g.*, in [48]. Moreover, at the beginning of the sixties of the last century, these factors were not yet established clearly and underwent change over time, as can be seen in fig. 14.

The effective dose rates of individual phantoms, obtained by the sum of F4 tallies for contributions of neutron, gamma ray and electron radiation in the tissue of the phantoms, are multiplied for irradiation time, t_{irr} = 96 s, corrected for the individual's mass, and shown as the total calculated effective doses of individuals D_{F4} (given in rem) in tab. 9. The ratio of the calculated total doses to the total doses given in [3] and [6] is shown, as well.

As can be seen in tab. 9, the effective doses calculated by the MCNP5 computer code remain high. Roughly, between one and a half to two times higher doses are obtained in individuals than the ones reported in [3] and [6]. Such results from MCNP5 code calculations are understandable, since the orientation of the operators to the source of radiation – the RB reactor core – is not taken in account with the simplified, homogeneous, faceless model of operators without internal organs. Moreover, the distribution (along the depth and height of the torso) of the absorbed dose within the human body also has an influence on the total dose received by the operators [71] and was included in MCNP5 calculation results. Some of the reasons for the obtained discrepancies were already explained in previous paragraphs. The fact that the operators were moving around the RB reactor core and were not all exposed



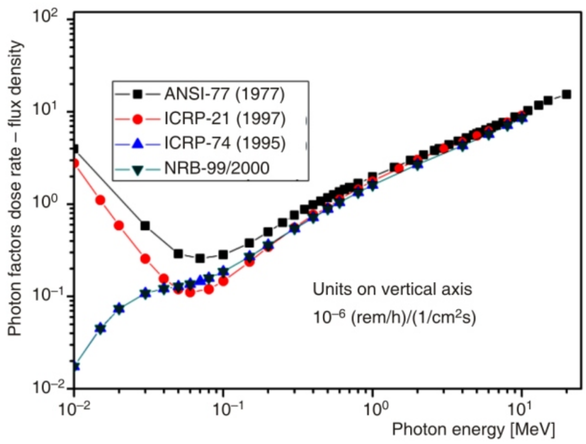


Figure 14. Neutron and photon flux density to dose rate conversion factors

Table 9. MCNP calculated (F4) effective doses in the tissue of phantoms according to radiation type

Ph. #	Operator code/geometry	$D_{n,F4}$ [rem]	$u\ D_{n,F4}$	$D_{\gamma,F4}$ [rem]	$u\ D_{\gamma,F4}$	$D_{e,F4}$ [rem]	$u\ D_{e,F4}$	$D_{t,F4}$ [rem]	$u\ D_{t,F4}$	$D_{t,F4}/D_{[3]}$	$D_{t,F4}/D_{[6]}$
1	BZ/PA	1019.45	0.44%	420.26	1.22%	67.96	0.75%	1507.67	1.50%	2.21	3.70
2	HS/PA	641.17	0.47%	276.31	0.83%	45.19	0.52%	962.67	1.09%	1.41	1.38
3	DR/AP	861.52	0.44%	232.11	0.67%	33.44	0.39%	1127.07	1.37%	1.65	1.10
4	GD/AP	1190.87	0.43%	302.86	0.85%	44.17	0.53%	1537.90	1.09%	2.25	1.67
5	VZ/AP	1215.29	0.46%	313.75	0.94%	46.41	0.51%	1575.45	1.16%	2.31	1.88
6	RF/AP	1109.20	0.44%	297.63	0.89%	48.37	0.56%	1455.20	1.14%	–	–
7	MR/PA	703.58	0.46%	314.80	0.98%	52.19	0.58%	1070.56	1.23%	1.57	1.25

to radiation for a same period of time remain the principal causes for these discrepancies.

CONCLUSIONS

An attempt to estimate the doses received by the operators during the accident at the "Boris Kidrič" (now the Vinča) Institute of Nuclear Sciences, former Yugoslavia, of October 15, 1958, was carried out with a modern tool: the MNCP5 computer code with cross-section data, based on the ENDF/B-VII.0 data library. A 3-D model of the RB reactor, RB reactor room and the operators at their probable positions was created (according to [5]) and explained. In our work, basic data on the RB reactor and accident scenario are given. A rough estimation of the time moment when the odour of ozone was detected by the operators is estimated for the first time and given in Appendix A.1. A new Root Cause Analysis of the accident and a Cause Mapping diagram, based on the cause-and-effect methodology, are shown in Appendix A.2. An attempt to evaluate the time of exposition to high doses in the RB reactor room, based on the estimated equivalent stationary power of the reactor during the accident, has been made. An evaluation of doses received by the operators and methodologies applied and published in previous papers, is given here as well.

Assumptions, approximations, uncertainties and unknown details in the applied methodologies, including data from the input of the MCNP5 computer code, are cited in this article. The main uncertainties in MCNP5 results are a consequence of:

- frozen (still unclear) positions of the (sitting and standing) operators,
- the use of simple, geometry-based homogeneous phantom models,
- the assumed equivalent stationary power of the reactor in the transient, due to the reactivity excursion during the accident,
- estimated (time) duration of the power excursion,
- the fact that this version of the MCNP5 computer code cannot generate delayed gamma rays from the fissions,
- the fact that the north wall of the reactor hall was made of concrete, while the lower part (to the corridor in the RB building) was fitted with glass windows in 1958,
- neutron, gamma ray and electron flux density – fission power normalization factors, and
- neglecting in the 3-D model possible shielding, reflecting or absorbing radiation effects of the control console and experimental equipment panels, chairs and tables on the doses received by the operators.

The uncertainty in the fluence-to-dose factors which have influenced the dose results obtained should also be considered. The said fluence to dose equivalent coefficients is not supposed to have uncertainties. But, this is a mere convention, since the calculated values of these factors, for neutrons only, show estimated uncertainties of 5% to 10% below 20 MeV [68], and of the order of 10-15% above, at a 1 confidence level [69]. As for neutron dose determination, one should mention the statement in [69]: "At high doses involved in criticality accidents, deterministic effects are the most important, and protection quantities, such as the dose equivalent, should not be used because they only take into account the stochastic biological effects at low-dose. The measured quantity should be the absorbed dose".

A wide variety of reported dose values received by the operators show how difficult a task it was to asses them in those days when the operators did not have personal accident dosimeters at their disposal. This study also proves how difficult it is to estimate previously determined doses by applying a contemporary computer code, with all the uncertainties, simplifications and approximations taken into account. An attempt to understand and explain the causes for the discrepancies between the calculated doses in this study and the doses reported in previous papers have been given in the data provided by our evaluations and in the written testimony of one of the participants in the accident. Certain details of the accident scenario and actions of the operators, their positions and time of exposure, still remain unclear. In spite of this, the doses estimated by the MCNP5 code are within expected values (the best results, $D_{t,F4}/D_{Mole}$ ratios, are within the expected 10%, apart from those for BZ and DR), taking into account the uncertainties in previously published results obtained by different methodologies and corrections applied, as cited in [59].

Appendix A.1

Estimation of ozone generation time

A rough estimation of the time moment when the odour of ozone was sensed by the operators in the RB reactor room is given in this Appendix for the fist time. This estimation is based on several assumptions. First, it is assumed that the RB power excursion was an exponential function ($e^{t/T}$) in time t (192 s after the accident scenario was initiated, see section Evaluation of the accident scenario) with a constant period T = 13.2 s. At that moment (192 s selected as the zero time moment), the heavy water in the reactor tank achieved the highest level (183 cm), as it was determined in the accident scenario. The generation of ozone before that moment is supposed to be neglected. It has also been assumed that the flux density of neutrons and gamma rays (escaping from the reactor tank) in the RB reactor room follow the same (like power) exponential time dependence with proportional initial constants (at t = 0, *i. e.* 192 s after the accident). The value of the proportionality constant of

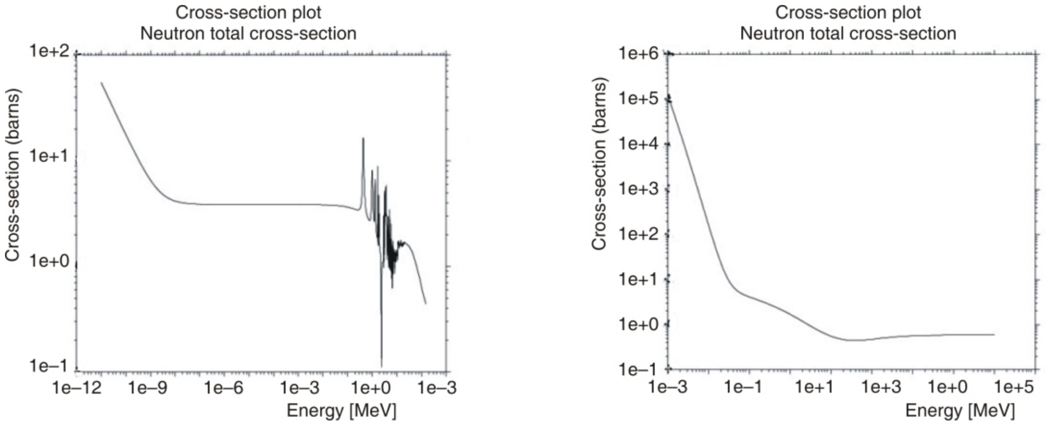


Figure A.1.1 Total cross-section for neutron and photoatomic (gamma) interactions with oxigen

1.4 is taken from the ratio of neutron flux density to the flux density of gamma rays, determined in section Conclusions. Further, it is assumed that ozone molecules, once generated in the interaction of neutrons and gamma rays with the oxygen in the reactor room, did not decay or dilute within the time frame of the accident. It is also assumed that all escaping neutrons had an average energy of 0.15 MeV and all escaping gamma rays had an average energy of 1.12 MeV, as determined in section Results and Discussion. Since the ionisation potential of oxygen and nitrogen is between 13 eV and 15 eV, it is further assumed that each interaction of ionising radiation, involving average energies mentioned above, can produce only two ozone molecules out of three oxygen molecules in the air. Total cross sections, for neutron and gamma ray interactions with oxygen, were taken from the MCNP5 ACE type libraries (fig. A.1.1) as $\sigma_n = 4.06$ b (1 b = 10^{-24} cm^2), for neutron average energy and σ = = 1.60 b, for the gamma ray average energy.

The neutron total cross-section is, at energies bellow the MeV energy range, almost entirely composed of elastic neutron scatterings at oxigen. The photo-atomic (gamma) total cross-section is composed of coherent and incoherent gamma ray scattering (a dominant component when this nuclide is concerned) in photoelectric, fluorescence and pair-production processes.

According to [47], a human is able to detect the odour of ozone at a threshold of ozone concentration in the air as low as 0.01 mol per mol, *i. e.*, 1.66 10^{-8}, if the molar mass of ozone amounts to 48 g per mol and that of air to 28.97 g per mol. The estimation in this Appendix is based on the assumption that the operators in the reactor room were able to detect such a low concentration of ozone. The RB reactor room (length: 26.8 m, width: 15.7 m, height: 11.8 m) has a volume of $4.965 \cdot 10^9$ cm^3. For an air density equalling 1.21 mg/cm^3 of the given molar mass and the Avogadro number amounting to $6.022 \cdot 10^{23}$ atoms per mole, the air concentration is $2.505 \cdot 10^{20}$ per cm^3. The number of air molecules in the reactor room is obtained as $1.244 \cdot 10^{30}$, while the number of oxygen atoms, N(O), is obtained for a 0.2095 volume fraction of oxygen in the air. The minimal number of ozone atoms in the air $N(\mathrm{O_3})$ is determined as $2.0606 \cdot 10^{22}$, taking into account the sensitivity of humans to ozone, as mentioned above.

At the (supposed) initial time moment of ozone generation (192 s), the power level of the RB reactor was determined at 0.05 W, according to data presented in fig. 3, given in section Evaluation of the accident scenario of the paper. Using the equation given in the section The MCNP5 computer code, the neutron flux density initial constant A_0 can be calculated as $4.225 \cdot 10^9$ cm^{-2} s^{-1}, while the gamma ray flux density initial constant should be 1.4 lower, under the above mentioned assumption. Therefore, the task is to determine time t_1, after the initial time set to 0 (after 192 s), in which $N(\mathrm{O_3})$ ozone molecules were produced in the interactions of neutron and gamma rays with oxygen molecules, from the equation

$$2N(\mathrm{O_3}) \quad 3N(\mathrm{O_2}) \quad \int_0^{t \; t_1} \sigma_n A_0 \mathrm{e}^{t/T} \quad \sigma_\gamma \frac{A_0}{1.4} \mathrm{e}^{t/T} \, \mathrm{d}t$$

By solving this simple integral and by replacing the numerical values for variables in the equation obtained, one can determine the value of t_1 as being 114 s. This is the time interval (192 s upon the beginning of the accident) in which the odour of ozone was sensed by the operators. Therefore, the operators in the RB reactor room were able to detect the odour of ozone 306 s after the accident scenario was initiated. It gave them enough time, about 2 minutes (433 s – 306 s = 127 s) to take actions described in [17], *i. e.* to evaluate the situation and shutdown the RB reactor.

Appendix A.2

RCA CM diagram of the RB accident

This simple Root Cause Analysis (RCA) diagram of the cause-and-effect methodology is based on

[4] Think Reliability, POB 301252, Houston, Tex., USA

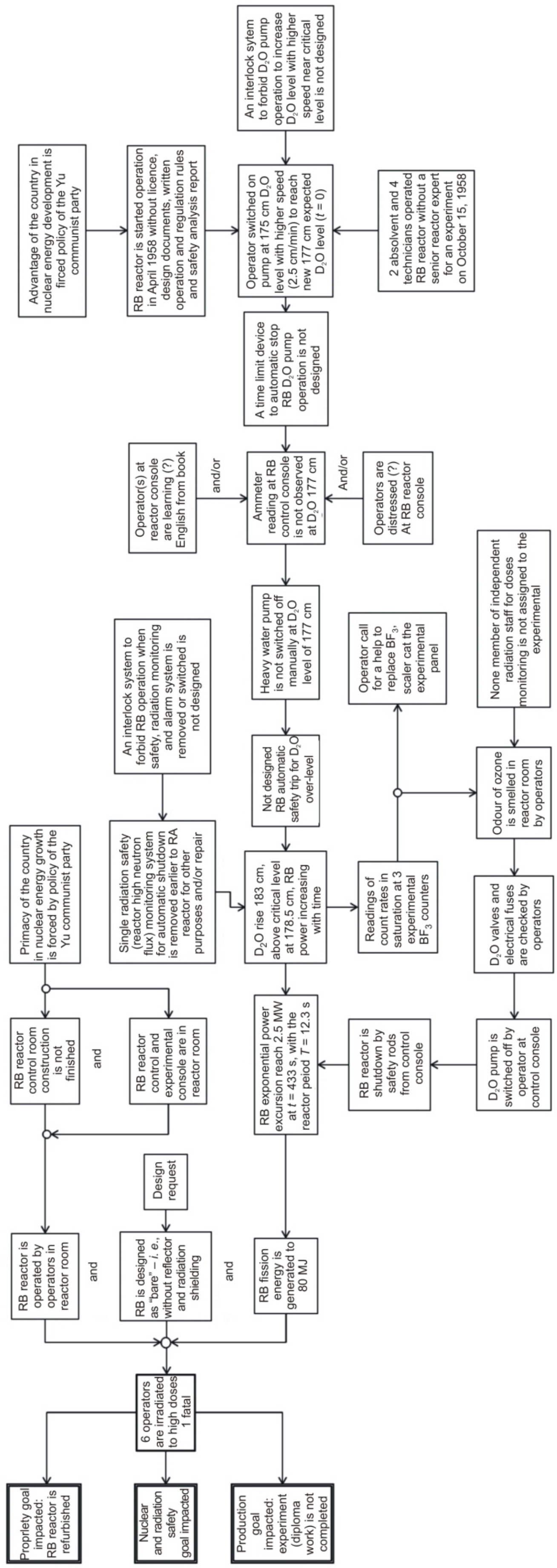


Figure A.2.1. A RCA CM diagram of the RB 1958 accident

data from the RB accident and the Cause Mapping (CM) approach developed by the Think Reliability[4] Company. The diagram below (fig. A.2.1) shows the causes and effects of the RB reactor accident considered at a medium complexity level. The diagram should be read in the direction opposite of the arrows, (for the most part) from the left ("effect") to the right, the question "why?" being answered by the "cause". A more detailed diagram is not shown, due to its size and complexity. As can be seen, the causes of the accident are complex. Possible (unconfirmed) causes are marked with a question mark in the text.

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Received on March 21, 2012
Accepted on June 15, 2012

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ПРОЦЕНА ДОЗА КОЈЕ СУ ПРИМИЛИ УЧЕСНИЦИ У АКЦИДЕНТУ НА РЕАКТОРУ РБ 1958 ГОДИНЕ ПОМОЋУ СИМУЛАЦИЈЕ MCNP5 РАЧУНАРСКИМ ПРОГРАМОМ

Нумеричко симулирање акцидента, који се догодио на реактору РБ 15. октобра 1958. године, и процена доза које су примили учесници урађени су рачунарским програмом MCNP5 под истим условима који одговарају експерименталној симулацији акцидента коју је организовала МААЕ 1960. године: тотална ослобођена фисиона енергија у акциденту је 80 MJ и учесници који се налазе у реакторској хали на стационарним положајима. Процењен је интервал излагања учесника високим дозама. Наведени су и подаци о језгру реактора РБ1/1958 важни за акцидент, а кратак преглед тока акцидента је допуњен новим сазнањима. Тродимензионални (3-Д) модел реактора са свим детаљима језгра и реакторске хале направљени су за ову сврху. За потребе оцене доза које су примили учесници, развијени су такође 3-Д модели – хомогени, бесполни, једноставни фантоми учесника и распоређени у хали реактора. Програм је радио са довољним бројем историја неутрона, у “mode n p e” који је обезбедио да статистичка несигурност одређивања доза учесника буде мања од 2%. Одређени су спектри зрачења на излазу из реактора и у ткиву фантома: спектри неутрона у 55 енергетских група до 20 MeV, спектри гама зрака у 35 енергетских група до 15 MeV и спектри електрона само у ткиву фантома учесника, у 10 енергетских група до 10 MeV. Груба процена тренутка времена у коме су оператори у реакторској хали први пут осетили мирис озона је извршена по први пут и дата у Додатку А.1. Дозе су одређене коришћењем ICRP-74 конверзионих фактора којима се резултати F4 талија за густину флукса зрачења претварају у апсорбовану или еквивалентну дозу и према резултатима F6 талија који одређују депоновану енергију неутрона и гама зрака (апсорбовану дозу) у ткиву фантома. Израчунате апсорбоване и еквивалентне дозе упоређене су са претходно објављеним вредностима доза и учињен је напор да се разлози за добијене разлике схвате и објасне. Урађена је анализа узорка акцидента (RSA) и, по први пут, дијаграм узрок-последица је креиран према методологији мапирања узрока (CM) и приказан у Додатку А.2.

Кључне речи: акцидент, апсорбована доза, реактор РБ, програм MCNP5