ADVANCEMENT OF A DELAYED GAMMA DOSIMETER FOR SPENT NUCLEAR FUEL ASSAY

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ABSTRACT

Spent fuel assay is vitally important for spent nuclear fuel management. Spent fuels with high radiation dose levels also require an appropriate dosimeter. A wide spread detector used for nuclear radiation dosimetry, is Geiger-Muller detector. Pulses from this detector are of the same amplitude and no information about particle energy is provided. Dead time is a distorting effect as nonlinear response is observed at high counting rates which requires application of correction methods to avoid the nonlinearity. The especial measuring tool must be consistent with the structures and constrains of the spent nuclear fuels, the spent fuel storage, and its construction design in TRR. Spent nuclear fuel assemblies are kept in 10-meter water in depth in the open reactor pool for a specific cooling period of time before transportation to the spent fuel storage pool. The high dose rate measurement requires its specific measuring equipment. In this work, a spent fuel active dosimeter is developed for spent fuel assay in TRR. Due to high radiation exposure, parts of the equipment are specially designed to be resistive against gamma radiation. To eliminate pulse pile-up effect as well as noise cancellation, a band pass filter is employed. Time interval distribution (TID) method is also devised as an advanced technique by incorporating its related digital electronics circuitry. By utilizing a Co-60 standard gamma source in Karaj Secondary Standard Dosimetry Laboratory (SSDL), the tests and calibration of the detector are also accomplished. Validation of the system is performed with a commercial measuring tool.

KEYWORDS: Gamma Dosimeter, Geiger Muller Detector, Spent Fuel.

1. INTRODUCTION

Geiger-Muller detectors are commonly used for radiation particle detection. Observed pulses at the detector output are of the same amplitude, therefore, particle energy cannot be measured utilizing the detection systems which are on the basis of Geiger-Muller detectors. In Fig. 1, the total transient of a typical Geiger-Muller detector signal is illustrated. There are three regions for timing of the detector as follows [1]:

- Detector dead time: The time in which no secondary interactions are detected.
- Counting system dead time: The time in which secondary pulses have enough height to cross the discrimination level.
- Recovery time: The time in which the secondary pulses have the same amplitude as the first pulse.

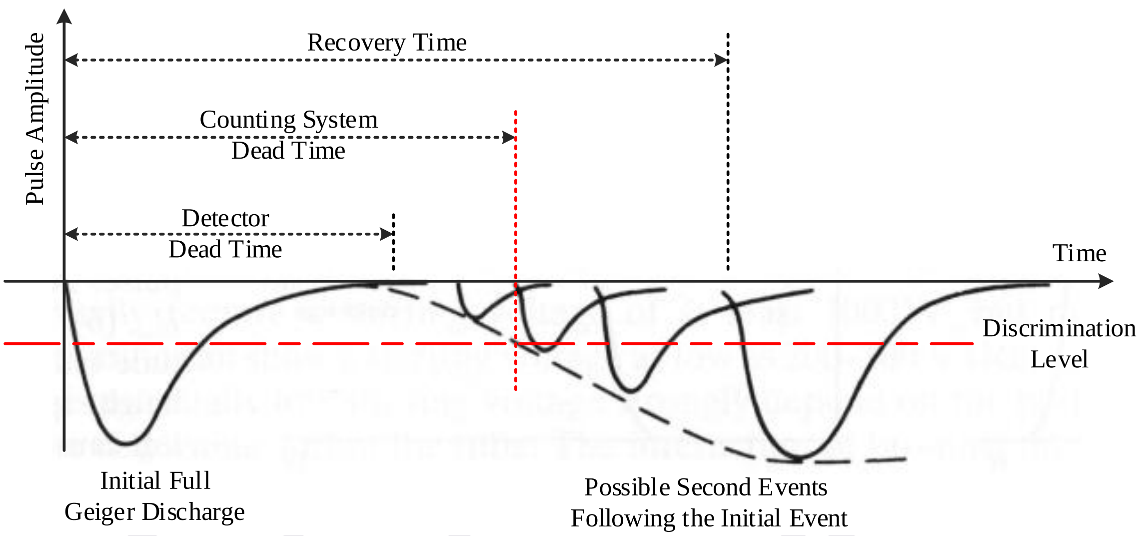

Fig. 1. Timing transient of Geiger-Muller detector.

Dead time makes the response of Geiger-Muller detectors as a nonlinear function of the observed count rate. Therefore, at high counting rates, dead time must be corrected to avoid the nonlinearity. There are various models for dead time correction. Among these models, three of them are more popular, Paralyzable, Non-Paralyzable, and Hybrid models [1-2]. When a particle is detected by the detector, within a duration of time, detector is involved, according to its physical structure. This duration is restarted in paralyzable model if a secondary radiation interacts with the sensitive material of the detector as shown in Figure 2, section A. this interaction is lost. Although three interactions are occurred, but only two distinct pulses are observed from the detector. In non-paralyzable model, section B of Figure 2, the duration is not restarted. Similar to paralyzable model, three interactions are occurred, but only two of them are observed as distinct pulses. In non-paralyzable model, the dead time duration is not extended as paralyzable model. In section C of Figure 2, hybrid model is exhibited. There are two durations of dead time after each interaction. First followed by non-paralyzable dead time and then followed by paralyzable dead time. As it can be seen in Figure 2, interactions during non-paralyzable dead time are ignored but interactions during paralyzable dead time causes restarting the dead time process. Five interactions are occurred but only three of them are observed as distanced pulses.

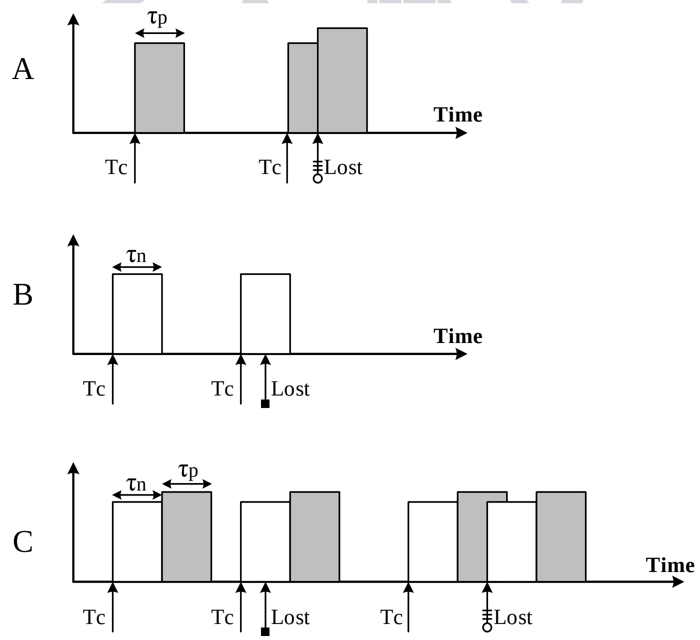

Fig. 2. Different dead time behaviours, A: paralyzable model, B: non-paralyzable model, C: hybrid model.

There are variety of different applications of Geiger-Muller detectors. Amongst them, dosimeters are more popular. In the present paper, a high range dosimeter for spent nuclear fuel assay is developed. After energy production of nuclear fresh fuels in the reactor, fission fragments are accumulated and trapped by the clad material. The source of gamma rays is the fission fragments. The nuclear fuel of Tehran Research Reactor (TRR) is shown in Figure 3. Each fuel plate is comprised of fuel meat and the clad material. Nineteen fuel plates are assembled as a standard fuel element or assembly. Spent fuels are activated and are sources of gamma rays. Their activity decay during the time. For the sake of spent fuel management, the activity of each fuel assembly must be measured. The irradiation time (the time in which the fuel assembly is resident within the reactor core), elapsed time after irradiation, and the reactor power are irradiation history of certain fuel assembly. Depending on the irradiation history, the activity of a spent fuel assembly is determined.

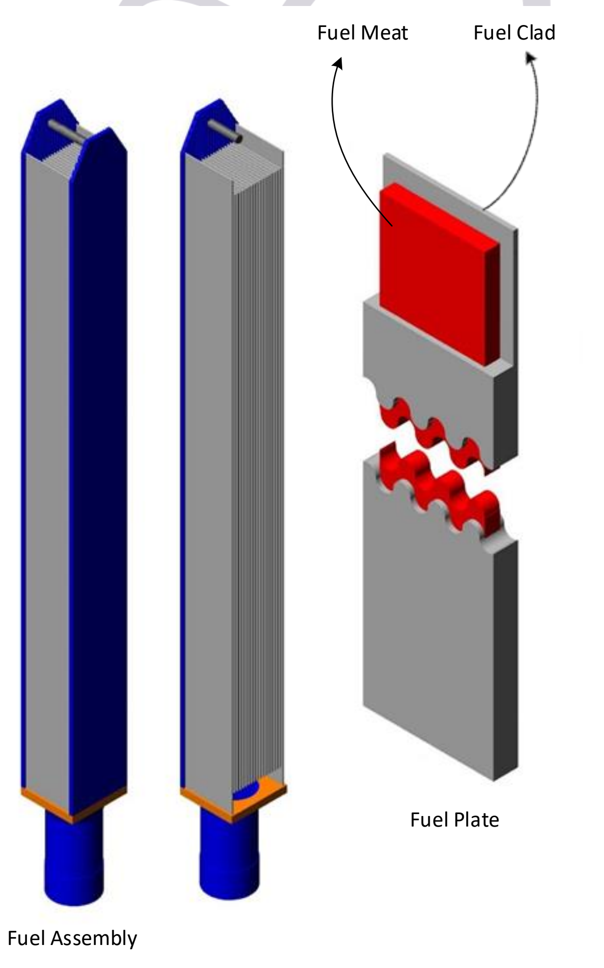

Fig. 3. A nuclear fuel assembly of TRR.

Spent fuels are stored in spent fuel storage or specially designed casks. The work performed by Kralik [3] is the dosimetry of an interim storage for spent nuclear fuels. In this research, the neutron and gamma doses of spent fuels (inside spent fuel casks) are measured using both neutron and gamma detectors. The neutron spectra around the casks are measured by means of a Bonner spheres spectrometer (BSS) which consisted of 13 polyethylene spheres. In their centres, cylindrical proportional counters are used. The corresponding response matrix was calculated with MCNP transport code [4]. During the first years of interim storage operation, the photon component was characterized by:

- An energy-compensated GM counter with suppressed sensitivity to neutrons, type ZP-1321/PTFE (Alrad Instruments, England), calibrated in kerma in air at ^{137}Cs and ^{60}Co photon beams.

- An FH 40 F2 radiameter (Eberline Instruments) with an incorporated GM counter indicating photon dose equivalent.

Passive detectors are also common for dosimetry purposes. The work performed by Viererbl and et. Al. [5] is about neutron and gamma radiations dosimetry of spent nuclear fuel assemblies based on passive detectors like alanine dosimeters for gamma radiation and track detectors for neutron radiation. Experiments are caried out on the IRT-2M spent nuclear fuel assemblies (used in the LVR-15 research reactor). The fuel assembly is located in a water storage pool at a depth of 6 m during irradiation of the detectors. Detectors are inserted into central hole of the assembly, and are irradiated for a certain time interval. After the irradiation, detectors are removed from the assembly, and then gamma and neutron doses are evaluated. Profiles of gamma dose rate and neutron fluence rate inside of the spent fuel assembly are measured.

Besides experimental methods of spent fuel dosimetry, theoretical methods are also considered to estimate the dose levels. Abrefah and his colleagues [6] calculated the dose rate of nuclear fuels of Ghanna Research Reactor-1 utilizing ORIGEN-S [7] (for computing changes in the isotopic concentrations during neutron irradiation and radioactive decay as well as to determine the source term) and MCNP (to calculate the dose rate based on the source term information provided by the ORIGEN-S code system) [4] calculational codes. The radionuclides are mainly produced after the core depletion contributed to the source term of 1.767 × 1013 ± 0.0008 photons/sec (observed after thirty days of the cooling period). The dose rates range between 3.51 × 10-25 ± 0.0003 mGy/h and 4.27 × 104 ± 0.0006 mGy/h at different positions above the reactor core, the control room (wall, door and window) and the rabbit room.

In a recent research work performed by Bagheri and et. Al. [8] gamma dose rates of TRR irradiated fuel assemblies are determined as an information for the context of the design and construction of transport and storage casks for the spent nuclear fuels. The work assesses gamma radiation dose rate radiated from irradiated nuclear fuels in TRR based on various theoretical and experimental methods. For this purpose, four different irradiated fuel assemblies with different operating histories are considered. Amber dosimeters at different distances from the fuel assembly were located to measure the dose rate of the gamma radiation. In another experiment, axial profile of gamma radiation dose-rate has been measured by placing 10 Amber dosimeters arranged in the axial active length of the fuel assembly. ORIGEN2.1 [9] and MCNPX2.7 [10] computer codes are also utilized for verification and comparison of the results.

There are various available scientific and research works published in the literature about spent nuclear fuel dosimetry and its challenges [11-16]. In this paper, an instrument for spent nuclear fuel dosimetry is developed and its calibration as well as its application are explained. Followings are detail description of the methods and results.

2. MATERIALS AND METHODS

2.1. SYSTEM ARCHITECTURE

In Figure 4, architecture of the detection system is shown. As all spent fuel assemblies are active, they are kept in water depth. Therefore, housing of the detector plus detector itself and its amplifier are also located under the water where spent fuel measurements are performed. The amplified signal is sent to the band pass filter out of the water. The rising edge of the pulses is passed through the filter. The signal noise is also cancelled as the central frequency of the design is set at 1 MHz. At high counting rates, pulse pile up is a problem which is also eliminated by the aid of the band pass filter. In the next stage, the signal is amplified and inverted, making it appropriate for comparison and discrimination. The background is discriminated by the comparator block. The resulting signal is passed through a Schmitt trigger gate to shape it as standard 3.3V LVTTL signal. Then, pulses are counted every ten second and based on calibration data, dose rate is exhibited in display unit.

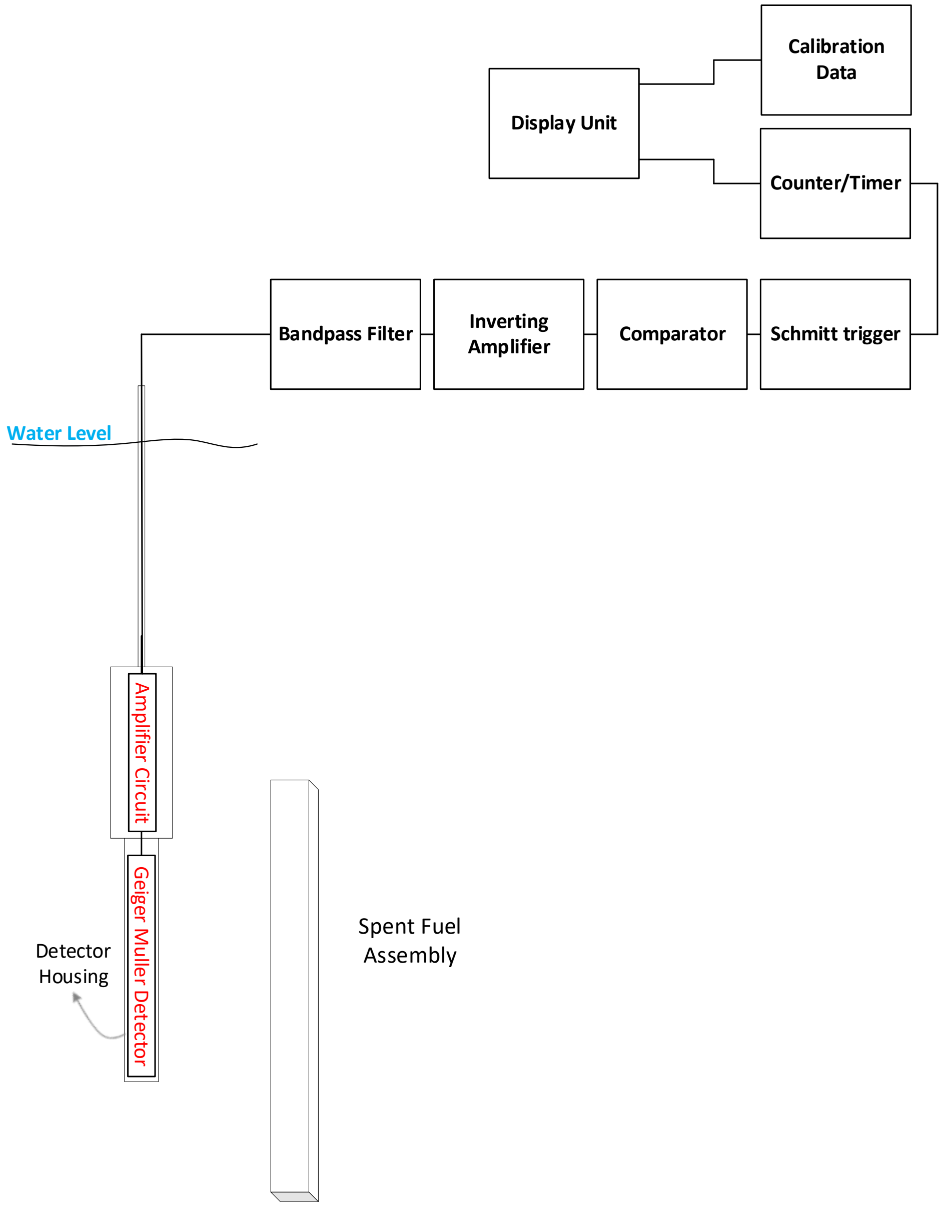

Fig. 4. Architecture of the detection system.

2.2. ELECTRONIC DESIGN CIRCUITRY

The specifications of the Geiger-Muller tube used for detection of the gamma radiations are listed in Table 1. The main items are operating or recommended high voltage, recommended anode resistor, and the nominal dead time which are 550 V, 1.8 MΩ, and 20 µs respectively. The amplifier design is shown in Figure 5. In order to reduce the dispersion capacitance of the resistors (at high impedance part of the circuit), three resistors (R1, R2, and R3) are made in series. The high voltage is connected to the detector through the RLC filter (L1, C1, and R4) and 1.8 MΩ series resistances. The diode D1 is protection component against high amplitude input pulses. The signal generated by the Geiger-Muller tube is amplified by the Operational Amplifier (OpAmp). The resulting pulse is shown in Figure 6. The rise time is about 500 ns, and the fall time is about 3 µs. The amplitude of the signal is about 2.3 V. Rise of the signal is rapid, while its falling is a slow exponential function shape (in comparison with the rise of the signal). This is the same for all signals shown in Figures 8 to 10.

The output signal from the amplifier stage is fed to the band pass filter circuit (refer to Figure 7). The first stage of this circuit (labelled as U1 amplifier) is the band pass filter designed for eliminating low frequencies (slow portion of the pulses and pulse pile-up effect) and noises of the signal. The second stage is an inverting amplifier, and the final stage functions as a comparator and a Schmitt trigger converting the analogue pulses to the digital ones with LVTTL standard. The resulting output is shown in Fig. 8. The bottom signal is the differentiation and inversion of the detector signal. If this signal is again inverted, the signal shown in Fig. 9 is resulted (the bottom signal). The last stage is the comparator (the components U3 and U4a in Fig. 7) which is designed for conditioning of the signal appropriate for the next stage, counter/timer circuit (refer to Fig. 4). In order to tolerate stochastic fluctuations in low counting rate, moving average filter is employed over the ten previous counts of the detection system. This is implemented in the software of the system.

Table 1. Specifications of the Geiger-Muller tube.

Part Number	VACUTEC 70 018 E
Physical Data	
Sensitivity (662keV, ^{137}Cs)	16 counts/s mGy/h
Dose Rate Range (662keV, ^{137}Cs)	(0.05 … 20.10^{3}) mGy/h
Photon Energy Range	(0.05 … 2) MeV
Background (Shielded by 5 cm Pb with a 2 mm Al surface)	≤2 counts/min
Length of Active Volume	5 mm
Cathode Diameter	3 mm
Anode Diameter	1.0 mm
Mass	47 g
Filling Gas	Helium/Halogen
Life Expectancy	>6.10^{10} Counts
Electrical Data	
Starting Voltage	< 410 V
Plateau Voltage Range	(520 … 620) V
Plateau Length	>100 V
Plateau Slope	<0.3 %/V
Recommended Supply Voltage	550 V
Recommended Anode Resistor, R_{Anode}	1.8 MΩ
Dead Time (R_{Anode}=1.8 MΩ)	≤20 µs
Anode to Cathode Capacitance	<2.5 pF

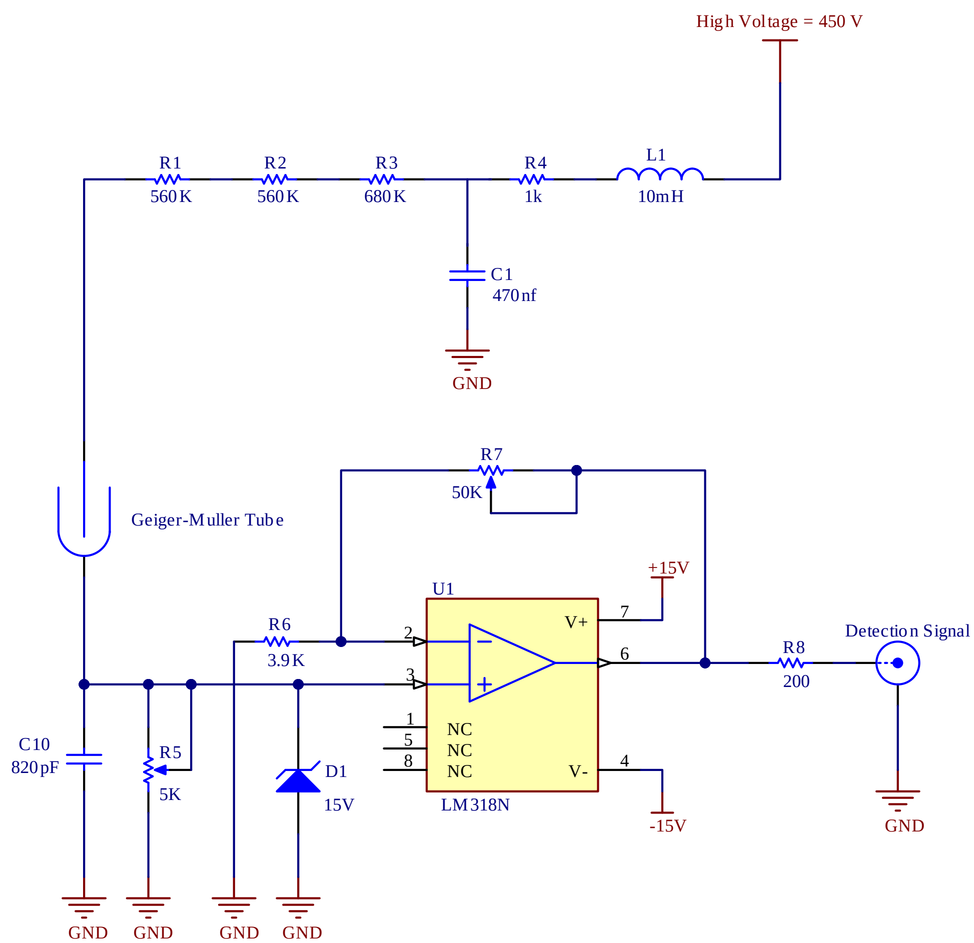

Fig. 5. Schematic diagram of the amplifier circuit.

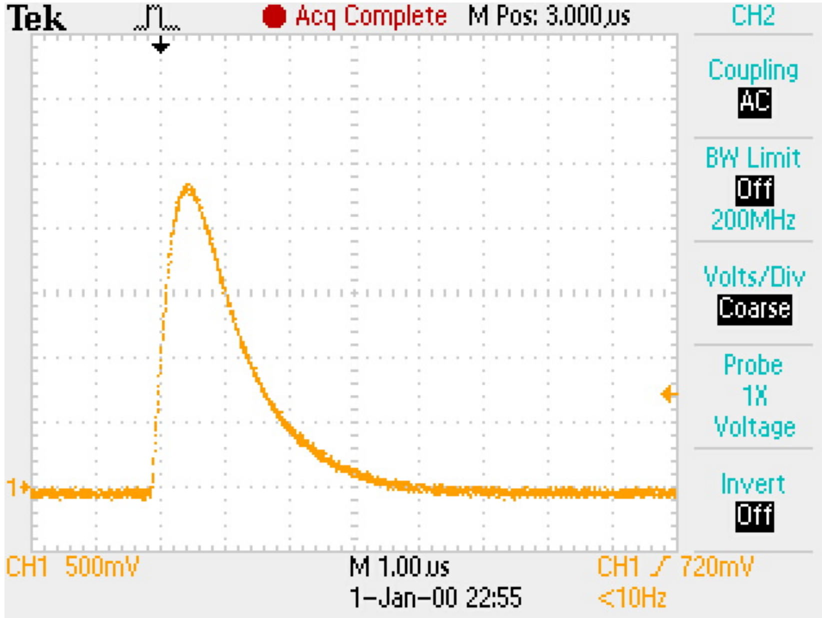

Fig. 6. A detection pulse at the output of the amplifier circuit.

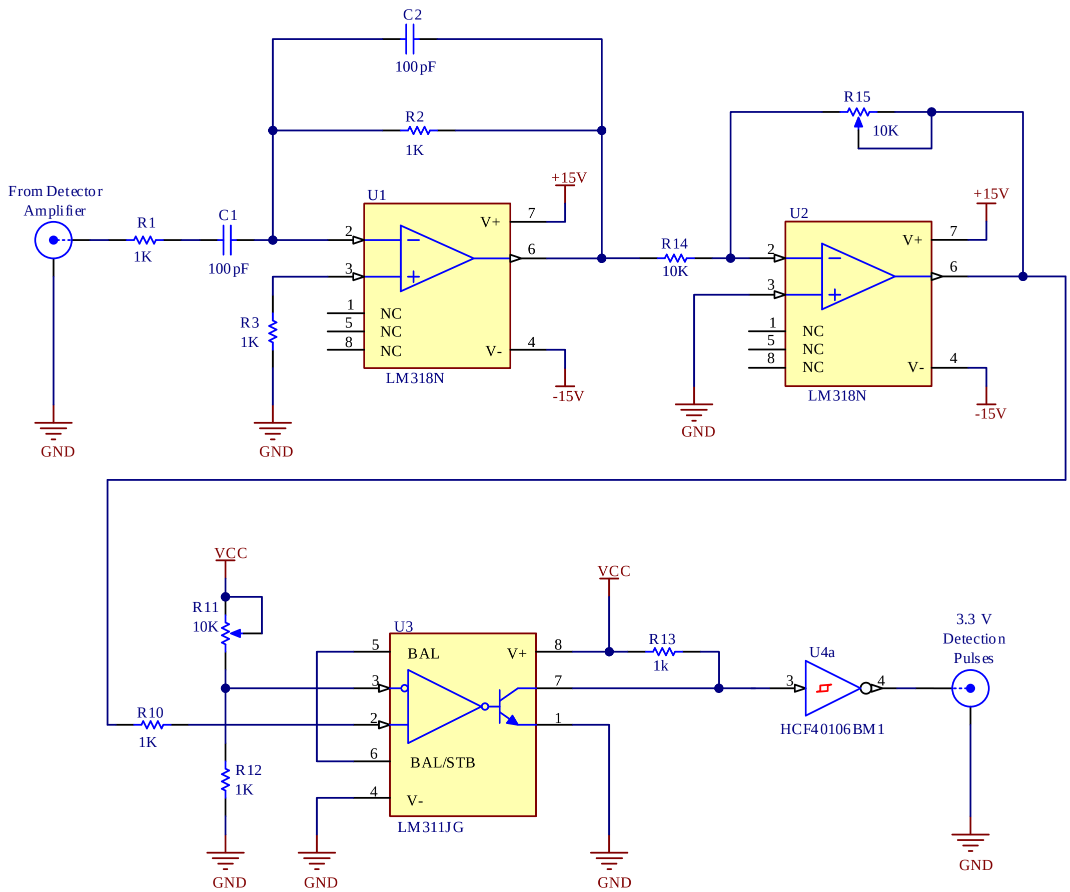

Fig. 7. Band pass filter, inverting amplifier, and comparator circuit.

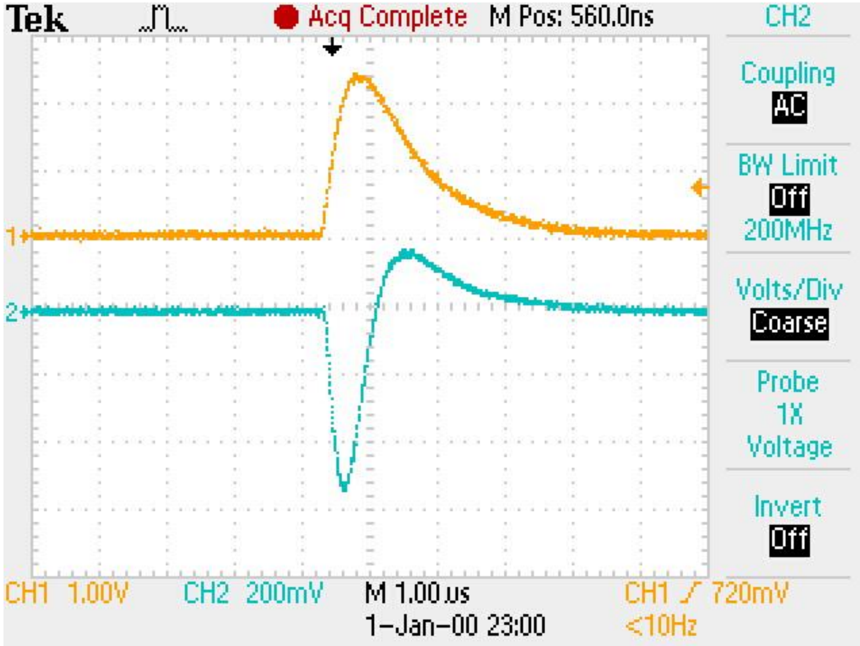

Fig. 8. Detector signal (the top signal which is the same as the signal shown in Fig. 6) and band pass filter output (the bottom signal).

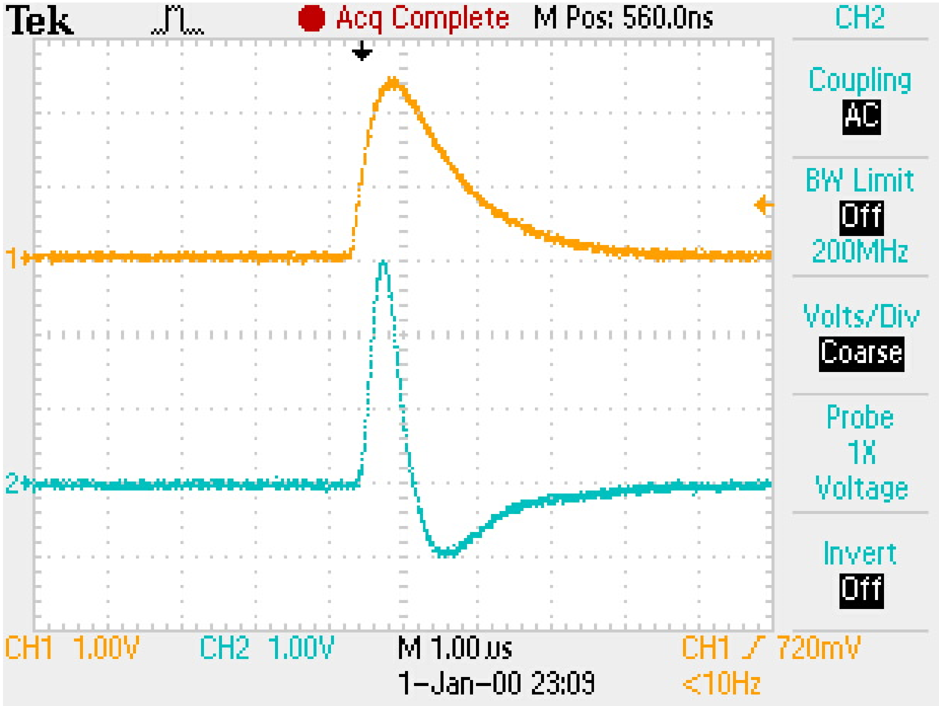

Fig. 9. Detector signal (the top signal) and inverting amplifier output (the bottom signal).

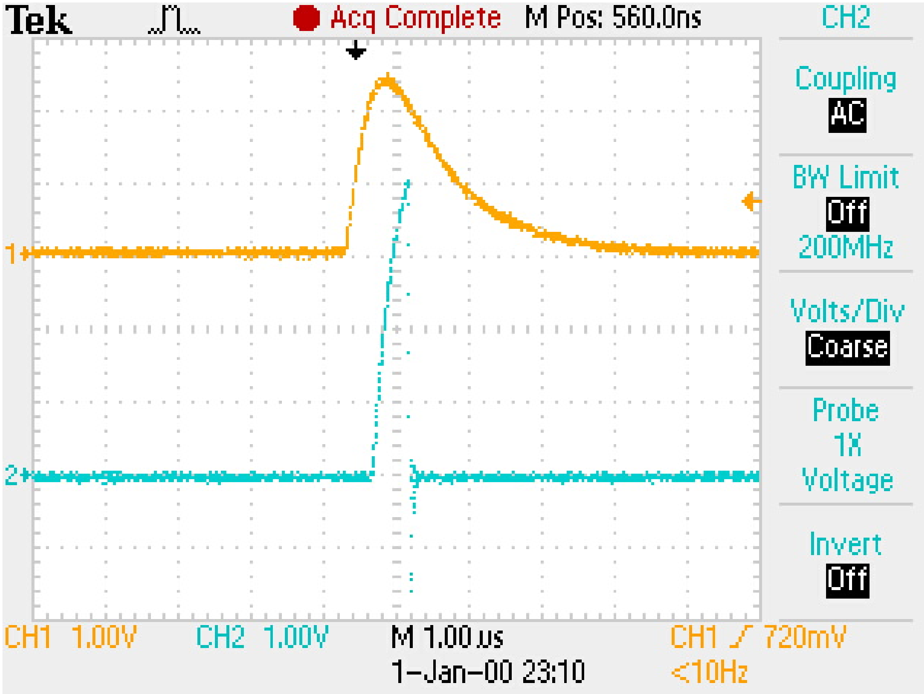

Fig. 10. Detector signal (the top signal) and comparator signal (the bottom signal).

3. RESULTS AND DISCUSSION

3.1. DEAD TIME MEASUREMENT AND CALIBRATION

The dead time of the system is measured utilizing Time Interval Distribution (TID) method [17]. The results are exhibited in Figure 11. As it is obvious, the data is distorted in less than 100 µs. The figure is semi-logarithmic and data should have linear distribution, but due to dead time effect, at small time intervals it is nonlinear. The dead time affects at short time intervals as shown in Figure 2. The red line is the fitted linear curve to the data for time intervals of greater than 100 µs where the dead time distortion is negligible. The dead time parameters of the detection system are measured and the results are shown in Table 2. The observed counting rate is about 1001.5 CPS. It is 1447.0 CPS for true counting rate. If the behaviour of the detection system is assumed to be as parayzable model, the dead time is about 25.4 µs. on the other hand, if it is assumed to be as non-paralyzable model, the dead time is about 30.7 µs.

Table 2. Dead time parameters of the detection system measured by TID method.

Observed Counting Rate	True Counting Rate	Paralyzable Dead time, μs	Non-Paralyzable Dead Time, μs
1001.5	1447.0	25.4	30.7

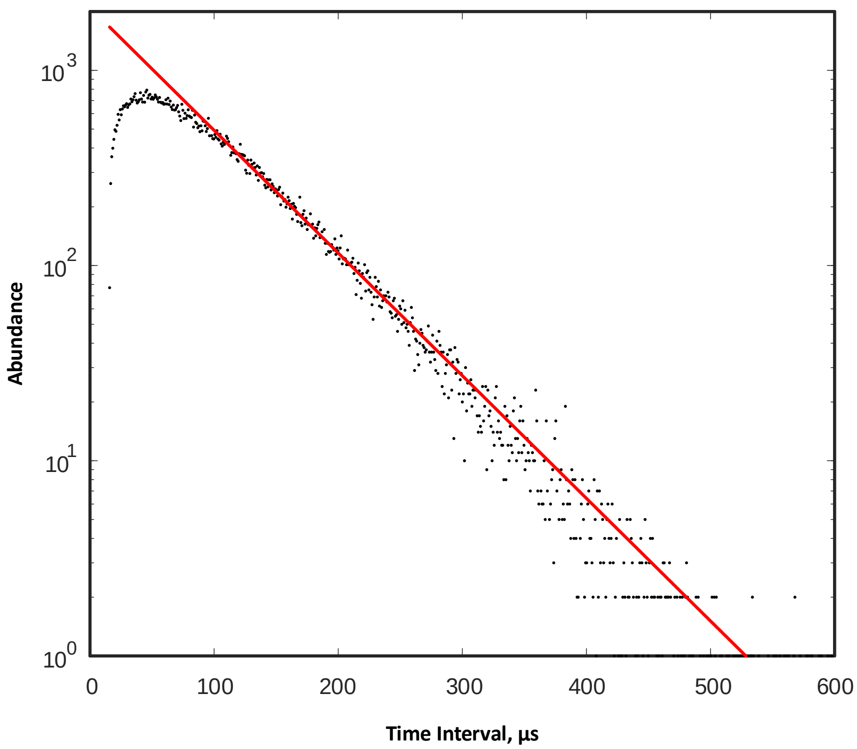

Fig. 11. Time interval distribution of the observed pulses and its fitted curve.

Calibration of the system is performed by a standard Co-60 gamma source in Karaj Secondary Standard Dosimetry Laboratory (SSDL). In Table 3 the experimental data is observed. Below is the calibration equation of the detection system. As dead time interferes at high counting rates, the calibration equation is derived at low counting rates where the dead time effect is negligible. As the relation between dose rate and the counts rate is linear, the calibration function is also linear. The second term (0.0037), is the background dose rate and slope of the linear fitted curve is 0.007191. Note that the counts per ten second should be inserted into the relation.

$$DoseRate\left(\frac{mSv}{h}\right) = (0.007191 \times Counts\ per\ Ten\ Second) - 0.0037 \tag{1}$$

Table 3. Experimental calibration data of the detection system.

Index	Gamma Dose Rate, μSv/h	Counts Per 10 Seconds*
1	110.5	16.0 ± 1.5
2	381.3	53.4 ± 5
3	711.5	98.9 ± 9
4	1360.0	190.6 ± 18
5	2645.8	368.1 ± 35
6	5157.8	703.5 ± 66
7	10266.9	1375.2 ± 120
8	20303.9	2670.4 ± 260
9	42572.5	5435.0 ± 500
10	86547.8	10746.4 ± 900
11	167505.9	19729.5 ± 1400
12	333944.4	35971.9 ± 3000
13	640823.8	60640.3 ± 5100
14	977723.8	83759.0 ± 7500
15	1878607.1	131660.0 ± 12300
16	3474800.1	188119.8 ± 18000
17	6396471.3	247578.6 ± 23270
18	7256053.0	258410.4 ± 24200

* Averaged over ten consecutive observed counts of the detector over every 10 seconds.

3.2. TEST AND VALIDATION

The International Atomic Energy Agency (IAEA) in an irregular way, check the dose rate of the spent nuclear fuel in TRR. They measure the dose rate in water using the IF104 dose rate meter designed and manufactured by SAPHYMO. This equipment, with various types of probes, which allows to measure high gamma and X dose rate by means of a is Geiger-Muller detector housed in a stainless-steel casing including all the electronics. The probe can be connected to the monitor through a 25-meter cable connection [18]. This dose rate meter plus its probe are shown in Fig. 12.

Fig. 12. IF104 dose rate meter and its probe used for dosimetry of TRR spent nuclear fuels.

For the sake of comparison between the results measured by IF104 dose rate meter and the dosimeter advanced in this work, a spent nuclear fuel is chosen and measurements are performed on it. The fuel assembly is located in a specific gamma rack, 20 cm in distance with the detectors. Before placement of the fuel element, the background dose rate is measured to be differentiated from the dose rate in the case the fuel assembly is resident in the gamma rack. In Table 4, the counts of the detector within ten seconds are listed. The average count is 18592.2 Counts per ten seconds. Based on the equation (1), the dose rate can easily be calculated by using the average counts. In Table 5, the dose rates measured by IF104 equipment and the dosimeter developed in this work are reported. The error is mainly due to the geometrical locations of the detectors as measurements are performed in 10 m of water in depth.

Table 4. Counts per ten seconds measured by the detector developed in the present work.

Index	Counts per Ten Seconds
1	18412
2	18310
3	18527
4	18350
5	18205
6	18411
7	18653
8	18890
9	18839
10	18623
11	18886
12	18748
13	18641
14	18800
15	18615
16	18565
Average	**18592.2**

Table 5. Comparison of the results measured by IF104 equipment and the dosimeter developed in the present work.

$DoseRate\left(\frac{mSv}{h}\right)$		Error (%)
IF104 Equipment	The Dosimeter Developed in the Present Work	
110	133.7	21.5

4. CONCLUSION

Dosimetry of irradiated fuel assemblies are of prime importance for providing information for the context of the design and construction of transport and storage casks for the spent nuclear fuels. A very status instrument for experimental studies is an especial measuring tool consistent with the features and limitations of the spent nuclear fuel, the spent fuel storage, and its construction design. In TRR, spent nuclear fuel assemblies are first stored in the reactor pool itself for a certain cooling period of time before being transported to the spent fuel storage pool utilizing transportation casks especially designed for this purpose. Therefore, spent fuels are stored in 10-meter water in depth in the open reactor pool for a while. The high dose rate fuel assemblies located in the reactor pool, needs its specific measuring tool for spent fuel management in TRR. In this research, a spent fuel dosimeter is developed for spent fuel assay in TRR pool based on Geiger-Muller detector as an active measuring instrument. As some parts of the equipment are exposed at high gamma radiation levels, special care is taken at the design stage to implement electronic devices which are more radiation resistive. Intelligent design is also performed to eliminate pulse pile up effect and noise cancellation by implementation of a band pass filter. Correspondingly, moving average filter is implemented in the software of the system to moderate statistical fluctuations at low counting rates. Dead time of the system is also measured utilizing TID method which is an advanced and new technique at the expense of complicated digital electronics equipment. Tests and calibration of the measuring tool are also performed utilizing a Co-60 standard gamma source in Karaj SSDL laboratory. To validate the detection system, the results are compared with a commercial measuring tool, and consistent results are observed.

REFERENCES

1. G.F. Knoll, Radiation Detection and Measurement, John Wiley & Sons, Inc, 2010.
2. S.H. Lee, R.P. Gardner, A new G–M counter dead time model, Applied Radiation and Isotopes, Volume 53, Issues 4–5,2000, Pages 731-737.
3. M. Kralık, V. Kulich, J. Studeny, P. Pokorny, Dosimetry at an interim storage for spent nuclear fuel, Radiation Protection Dosimetry (2007), Vol. 126, No. 1-4, pp. 549-554.

4. Briesmeister, J. F., MCNP: A general Monte Carlo N-particle transport code, LA-12625-M, March 1997.
5. L. Viererbl, Z. Lahodová, A. Voljanskij, V. Klupák, M. Koleška, M. Cabalka, K. Turek, Measurement of gamma and neutron radiations inside spent fuel assemblies with passive detectors, Nuclear Instruments and Methods in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment, Volume 652, Issue 1, 2011, Pages 90-93.
6. R.G. Abrefah, P.A.A. Essel, H.C. Odoi, Estimation of the dose rate of nuclear fuel of Ghana Research Reactor-1 (GHARR-1) using ORIGEN-S and MCNP 6, Progress in Nuclear Energy, Volume 105, 2018, Pages 309-317.
7. Hermann, O. W., and R. M. Westfall. "ORIGEN-S: SCALE system module to calculate fuel depletion, actinide transmutation, fission product buildup and decay, and associated radiation source terms." Vol. II, Sect. F7 of SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200, Rev 6 (1995).
8. S. Bagheri, H. Khalafi, F. Faghihi, A. Ezzati, M. Keyvani, H. Ghods, Gamma dose rate determination of TRR irradiated fuel assemblies, Progress in Nuclear Energy, Volume 142, 2021, 103950.
9. Croff, A.G., 1980. User’s Manual for the ORIGEN2 Computer Code. Oak Ridge National Lab., TN (USA).
10. Waters, L.S., 2002. MCNPX User’s Manual. Los Alamos. Accesed in Apr 15, 2012 at. http://mcnpx.lanl.gov/opendocs/versions/v230/MCNPX.2.3.0.Manual.pdf.
11. Lee, Yi-Kang, Sharma, Kabir, 2013. TRIPOLI-4 gamma-ray dose calculation for spent PWR fuels. In: 2013 21st International Conference on Nuclear Engineering. American Society of Mechanical Engineers Digital Collection.
12. Lloyd, W.R., Sheaffer, M.K., Sutcliffe, W.G., 1994. Dose Rate Estimates from Irradiated Light-Water-Reactor Fuel Assemblies in Air. No. UCRL-ID–115199. Lawrence Livermore National Lab.
13. Ohno, Akio, Matsuura, Shojiro, 1980. Measurement of the gamma dose rate distribution in a spent fuel assembly with a thermoluminescent detector. Nucl. Technol. 47 (3), 485-493.
14. Vogt, Jan, et al., 1999. Measurements of Decay Heat and Gamma-Ray Intensity of Spent LWR Fuel assemblies. No. IAEA-TECDOC-1089.
15. Willingham, C.E., 1981. Radiation dose Rates from Commercial PWR and BWR Spent fuel Elements. No. PNL-3954. Pacific Northwest Lab., Richland, WA (USA).

16. Winston, Philip Lon, James William, Sterbentz, 2002. Gross gamma dose rate measurements for TRIGA spent nuclear fuel burnup validation. International Conference on Nuclear Engineering 35987.
17. M. Arkani, G. Raisali, Measurement of dead time by time interval distribution method, Nuclear Instruments and Methods in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment, Volume 774, 2015, Pages 151-158.
18. IF104 dose rate meter datasheet, SAPHYMO corporation, http://www.saphymo.de/ftp/ecatalogue/161/29294868EN_C_-_Data_Sheet_IF104.pdf