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# Design of a neutron flux measurement channel using the ionization chamber KNK-3 at the Dalat Nuclear Research Reactor

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**Abstract:** This paper presents a design of the neutron flux measurement channel that consists of a Boron-contained gamma-compensated ionization chamber (CIC) named KNK-3 and operates in current mode, a current to frequency (I to F) converter, and a neutron flux measurement and control module (FPGA-WR). The designed measuring channel allows to measure and control the neutron flux density from $1.0 x 10^6$ to $1.2 x 10^{10}$ n/cm$^2$.s corresponding to the range from 0.1 to 120% of the nominal power of 500 kW of the Dalat nuclear research reactor (DNRR). The measurement and control module uses FPGA Artix-7 and digital signal processing algorithms to measure and calculate the reactor power and period values, and generate warning and emergency signals by the reactor power and period. The measurement channel was tested by using simulated signals and examining in the reactor to compare with the neutron flux measurement channel using the BPM-107R neutron flux controller of the existing complex ASUZ-14R for the DNRR control and protection system (CPS). The comparison results show that the measurement channel fully meets the requirements on the accuracy of the reactor power and period parameters as well as the ability to respond at once to the warning and emergency signals of the reactor power and period. Therefore, the measurement channel can be used for testing, research and training. The FPGA-WR measurement and control mudule can replace the BPM-107R controller for the working range of the CPS.

**Keywords:** *Control and protection system, reactor period, reactor power, moving average filter, FPGA, KNK-3, ASUZ-14R.*

## I. INTRODUCTION

In a nuclear reactor instrumentation and control (I&C) system, the neutron flux monitoring equipment (NFME) plays an important role in determining the reactor power and period values, their emergency preset thresholds, etc. in order to control and protect the reactor. The neutron flux density is monitored through the reactor power (P) and the time rate of power level change is indicated by the reactor period (T). On July 2007, the analog control system (AKNP-5A) of the DNRR was replaced by the digital control system based on ASUZ-14R complex, but the principles and basic functions are in compliance with the old system [1]. The reactor power and period are monitored by three independent NFME channels, which were designed for formation and generation of actuation signals based on the “2 out of 3” selection principle. Every NFME channel can monitor the neutron flux density from $1.0\times10^0$ to $1.2\times10^{10}$ n/cm$^2$.s including two ranges: start range and working range [1, 2]. The purpose of this study is to design an independent neutron flux measurement channel for testing of reactor parameters, as well as for research and training. The designed channel is composed of the ionization chamber KNK-3, the I to F converter, the FPGA-based control module

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using moving average (MA) filters, and digital signal processing (DSP) technique. Comparison with the current module BPM-107R which is based on 8-bit microprocessor of the CPS shows that the obtained parameters by simulated signals and real neutron signals from the reactor are quite similar in the reactor power, the reactor period and the response time in the working range. Therefore, the neutron flux measurement module FPGA-WR based on DSP algorithm can replace BPM-107R module to monitor the reactor in the working range with the current design configuration and can extend to monitor for the start-up range of the CPS of DNRR. Then, when FPGA-WR module is connected to the logic control module BFM-29R of the ASUZ-14R complex [1, 3], the emergency signals of reactor power and period will send to the relay unit “2 out of 3” to keep the reactor at a subcritical state by dropping neutron absorbers into the reactor core.

## II. METHOD AND DESIGN

### A. Method of determining the reactor power and period

The Russian KNK-3 Boron-lined CIC type is operated in the current mode and used for the working range of the analog control system AKNP-5A of DNRR. It was installed in dry channel and outside the reactor core. The neutron flux density at this location is smaller than about 3 decades at the core center. The output current of the KNK-3 is proportional to the neutron flux density. That current is converted to frequency (Fwr) which is applied to the FPGA-WR control module. The power value at the working range of the reactor can be calculated as:

$$Pwr = Kwr \times Fwr \times 10^{-3} \qquad (1)$$

where Pwr is the reactor power, Kwr is the coefficient, Fwr is the pulse frequency from the I to F converter output.

The reactor power of a nuclear reactor follows the exponential function:

$$P(t) = P_0 \times e^{t/T} \qquad (2)$$

where P(t) is transient reactor power at time *t*, $P_0$ is initial reactor power at time $t_0$. The time interval that takes for the reactor power changed by a factor of e (e = 2.718) is called the reactor period T.

The output frequency from the I to F converter is proportional to the reactor power, therefore from function (2) we can determine the reactor period as below [4, 5].

$$\frac{1}{T} \approx \frac{1}{F} \times \frac{dF}{dt} \ hay \ \ T = \frac{F_k}{F_k - F_{k-1}} \times \Delta t \qquad (3)$$

where $F_{k-1}$, $F_k$ are the (k-1)-th and k-th pulse frequency samples from the output of the I to F converter, $\triangle t$ is sample time in seconds.

From Eq. (3) we can see that the reactor period can reflect the state of nuclear reactor. When $F_k$-$F_{k-1}$ = 0, the reactor is in a stable state (T～999 s); when $F_k$-$F_{k-1}$ > 0, the reactor power increases; and when $F_k$-$F_{k-1}$ < 0, the reactor power decreases.

In practice, the experimental measurements indicated that the physical processes in the reactor are reflected by the output pulse frequency from the converter, especially large fluctuations often occur at low counts, so the MA filters are used for determining the mean value.

The MA filter is a type of finite impulse response filter (FIR) that is used to determine a tendency increasing or decreasing in reactor power. When calculating sequential values, a new value comes into the sum, the oldest value moves out, and the average value can be calculated as below [6, 7].

$$\overline{P}_{SM} = \overline{P}_{SM,pre} + \frac{1}{n}(P_M - P_{M-n}) \qquad (4)$$

where $\overline{P}_{SM}$ is the average value, $\overline{P}_{SM,pre}$ is the previous average, $P_M$ is the new sample, $P_{M-n}$ is the n-th old sample, n is window of the MA filter. The n-value will be changed during reactor operation depending on the fluctuation of power value and the current power level.

**B. Current to frequency converter**

Design of the I to F converter is based on the principle of charging and discharging of a capacitor through an integral circuit, as shown in Fig. 1. The converter circuit is calibrated with the input current of 300 μA corresponding to the output frequency of 50 kHz. Based on testing with the reactor power from 0.1% to 100% $P_{nominal}$, the coefficient Kwr was determined being 2.13 by formula (1). Accordingly, the value of the reactor power in the working range can be calculated using Eq. (5) when paired with the KNK-3 [1].

$$Pwr = 2{,}13 \times Fwr \times 10^{-3} \qquad (5)$$

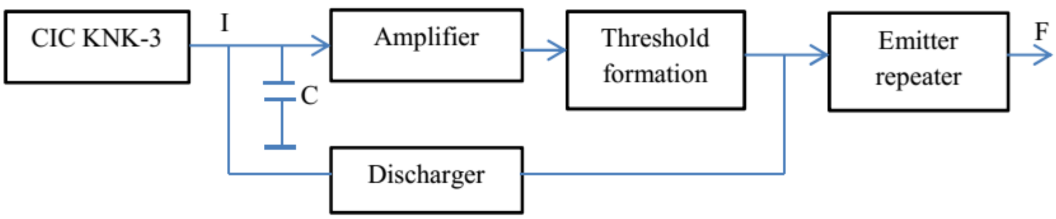


**Fig. 1.** Principle schema of the I to F converter

**C. Neutron flux measurement and control module**

The neutron flux measurement and control module for the working range (FPGA-WR) are illustrated in Fig. 2.

The FPGA-WR module is designed based on FPGA XC7A100T, on board clock frequency of 50 MHz, the output pulse from the I to F converter is sampled by a 32-bit counter, sampling time of 20 ms for calculating count per second (cps). This count is filtered by the MA filters for calculating the reactor power by Eq. (5) and the reactor period by Eq. (3). The coefficient of MA filter is changed automatically according to the pulse frequency values. The reactor power and period values are compared to the preset values for generating emergency signals to protect the reactor.

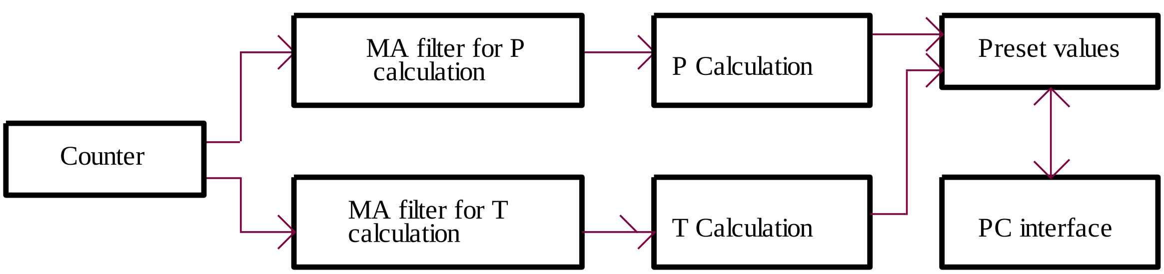


**Fig. 2**. Principle schema used for neutron flux measurement FPGA-WR

## III. EXPERIMENTAL RESULTS AND DISCUSSION

### A. Test of the FPGA-WR module with simulated signals

The PGT-17R module for simulation of the reactor power and period was produced by JSC SNIIP SYSTEMATOM Russia and has been used for checking working capability of the neutron flux measuring module BPM-107R of the ASUZ-14R complex. The PGT-17R allows generating pulse signals, that their frequencies are proportional to the change of the reactor power according to formula (2). The experimental schema to measure the reactor power and period using PGT-17R is shown in Fig. 3. The simulated period was chosen to be 20 s, the initial and final frequency values of simulating were 100 Hz and 50,000 Hz, respectively. The simulated pulse signals from the PGT-17R are simultaneously transferred to the BPM-107R and the FPGA-WR modules. The reactor power and period values are recorded by the computer via Terminal v1.9b software.

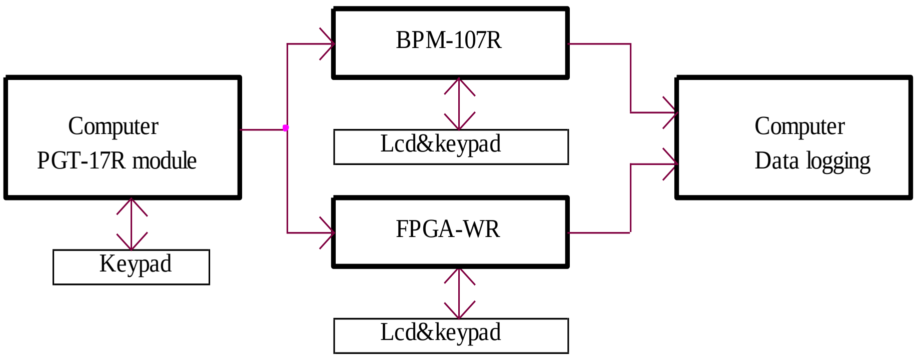


**Fig. 3.** Principle schema for measurement of the reactor power and period using PGT-17R module

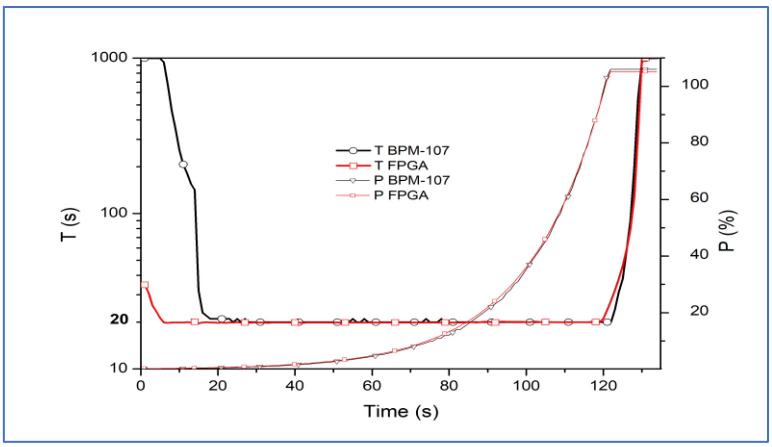


**Fig. 4.** Testing results of the BPM-107R and FPGA-WR modules using PGT-17R module with $F_{WR}$ from 100 Hz to 50,000 Hz at the period of 20 s

The obtained testing results in Fig. 4 show that the reactor power and period values, which were determined by the FPGA-WR and BPM-107R modules, are equivalent with the estimated discrepancy within 5%.

### B. Test of the FPGA-WR module on the DNRR

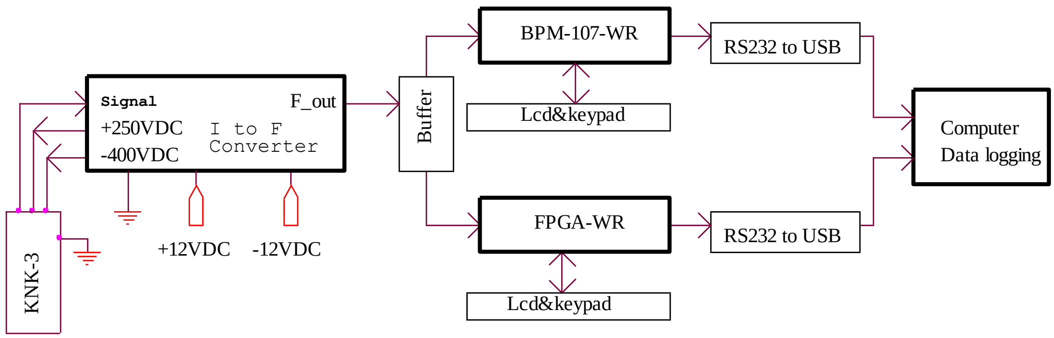


**Fig. 5.** Principle schema for testing the reactor power and period using KNK-3

The output pulse signals of the I to F converter is accessed to both the BPM-107R and FPGA-WR modules through a digital buffer. The values of the reactor power and period are logged by the computer through serial interfaces RS-232 to USB. In this experiment, the reactor was operated only up to 80% $P_{nominal}$ and the results are shown in Figs. 6 and 7.

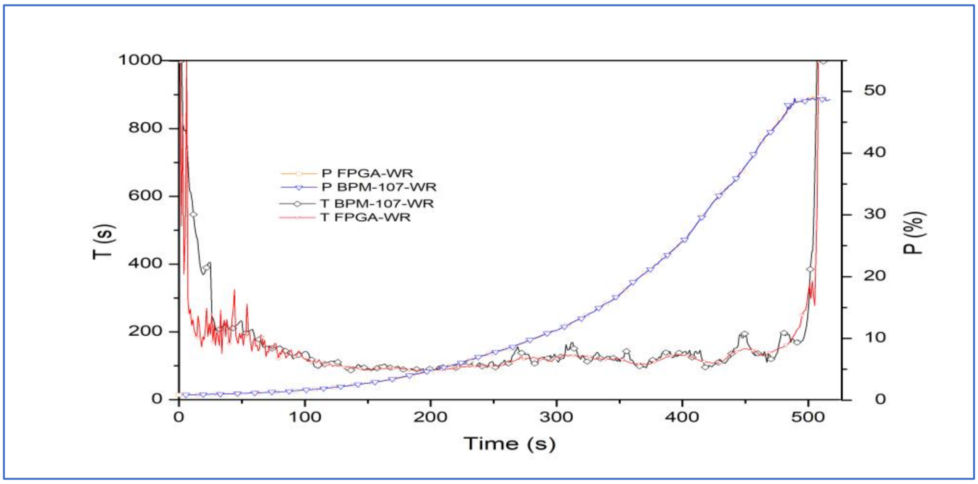


**Fig. 6.** Results of power and period measurements in the range from 0.5 to 50% $P_{nominal}$

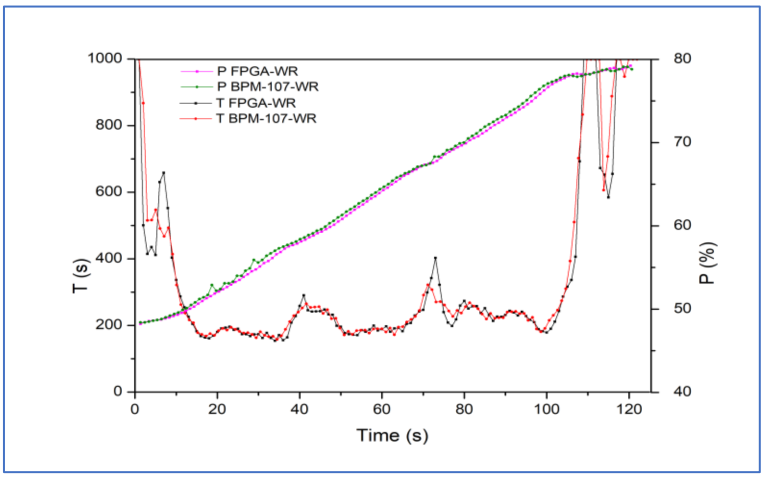


**Fig. 7.** Results of power and period measurements in the range from 50% to 80% $P_{nominal}$

The experimental results of the power and period on the reactor from 0.5% to 80% $P_{nominal}$ (Figs. 6 and 7) of FPGA-WR and BPM-107R are quite similar in full working range with the estimated discrepancy within 5%. The measured values of the reactor power and period with simulated signals of the PGT-17R module (as shown in Fig. 4) and by real neutron signals from the reactor (as shown in Figs. 6 and 7) indicate that the current design configuration of FPGA-WR has a function completely equivalent to that of the current BPM-107R module.

### C. Response time of emergency protection signals by reactor power and period

Fast response ability of a NFME for emergency conditions to stop chain reactions is one of important parameters of a CPS in nuclear reactor. The experimental schema to measure the time of generating the emergency protection signal by the reactor power and period is shown in Fig. 8.

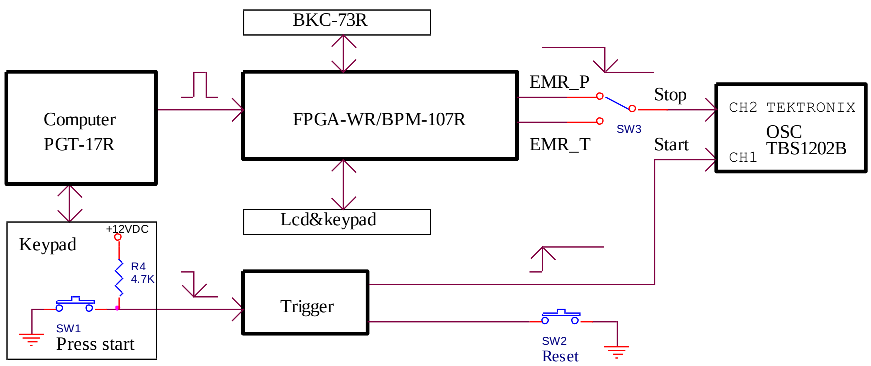


**Fig. 8.** Principle schema for testing response time of FPGA-WR and BPM-107R modules

The PGT-17R is also used for checking the time for generation of the actuation preset values by reactor power and period of the BPM-107R and FPGA-WR modules. The emergency set-points by the reactor power and period are shown in Tables I and II. The response time is determined from the moment of pressing the start key for starting timer until the control module determines that the reactor period value is less than 20 s or the power value exceeds the preset value by 10%. This response time is measured by the oscilloscope TBS1202B of TEKTRONIX. The measured values of reactor power and period are referred in [2].

**Table I.** Time for producing power emergency signals of BPM-107R and FPGA-WR modules

| **Reactor power level** (% $P_{nominal}$) | **Scram set-point** (% $P_{nominal}$) | **Response time of power emergency signal of BPM-107R** (s) | **Response time of power emergency signal of FPGA-WR** (s) |
|---|---|---|---|
| 5 | 8 | 0.08 ± 0.002 | 0.05 ± 0.002 |
| 10 | 15 | 0.07 ± 0.002 | 0.05 ± 0.002 |
| 30 | 40 | 0.06 ± 0.002 | 0.05 ± 0.002 |
| 50 | 60 | 0.05 ± 0.002 | 0.05 ± 0.002 |
| 70 | 80 | 0.05 ± 0.002 | 0.05 ± 0.002 |
| 90 | 100 | 0.07 ± 0.002 | 0.05 ± 0.002 |
| 100 | 110 | 0.09 ± 0.002 | 0.05 ± 0.002 |

The testing results in Table I show that due to operation in parallel processing of the FPGA-WR module, its response time of power emergency signal is smaller and quite stable in about 0.05 s with the sampling time of 20 ms, whereas, that of the BPM-107R module has a fluctuation in the range from 0.05 to 0.09 s because the sampling and processing cycles are performed sequentially by the microprocessor controller. The obtained testing results of the FPGA-WR module had a good agreement with the requirement of the ASUZ-14R complex that the response time of reactor power in the working range should be not more than 0.5 ± 0.02 s [2].

Time for producing emergency signals by period of a NFME is duration from the moment that the reactor period is smaller than the preset values until the period emergency signal EMR-T is generated [2]. The simulated frequency is proportional to the change of the reactor power according to formula (2) with the period of 10 s and 20 s, and the emergency preset value of period is 20 s, the response time by period emergency signals are presented in Table II.

**Table II.** Time for producing period emergency signals of BPM-107R and FPGA-WR modules

| **Power level** (% $P_{nominal}$) | **Frequency value** (Hz) | **Response time of period emergency signal of BPM-107R** (s) | | **Response time of period emergency signal of FPGA-WR** (s) | |
|---|---|---|---|---|---|
| | | T = 10 | T = 20 | T = 10 | T = 20 |
| 0.2 | 100 | 10.3 ± 0.1 | 27.0 ± 0.1 | 9.7 ± 0.1 | 23.8 ± 0.1 |
| 1 | 469 | 5.1 ± 0.1 | 9.3 ± 0.1 | 5.2 ± 0.1 | 9.4 ± 0.1 |
| 5 | 2,347 | 3.7 ± 0.1 | 7.2 ± 0.1 | 3.8 ± 0.1 | 5.4 ± 0.1 |
| 10 | 4,695 | 3.5 ± 0.1 | 6.9 ± 0.1 | 3.4 ± 0.1 | 4.4 ± 0.1 |
| 50 | 23,474 | 3.4 ± 0.1 | 6.7 ± 0.1 | 3.4 ± 0.1 | 5.4 ± 0.1 |
| 70 | 32,864 | 3.3 ± 0.1 | 6.7 ± 0.1 | 3.2 ± 0.1 | 4.9 ± 0.1 |

The experimental results in Table II show that the response time of period emergency signals of FPGA-WR module is mostly smaller than that of BPM-107R module. The response time of the emergency protection signal by period of the FPGA-WR is selected according to the input frequency range. The obtained results show that the response time by period of the FPGA-WR module is in a good agreement with the control system ASUZ-14R requirement.

## IV. CONCLUSION

The neutron flux measurement module FPGA-WR using the DSP and MA filtering techniques was developed, combined with the KNK-3 gamma-compensated ionization chamber and the I to F converter circuit, to measure the reactor power and period in the range from 0.1 to 120% $P_{nominal}$ of the DNRR.

Testing results of FPGA-WR module using the simulated signals from PGT-17R and the real neutron signals from the reactor were compared with those of the BPM-107R of the ASUZ-14R complex. The parameters of reactor power and period obtained from the two modules are similar. The response time by reactor power signals of the FPGA-WR module is stable at 0.05 s, meanwhile it fluctuates in the range from 0.05 to 0.09 s in case of the BPM-107R module. The response times by reactor period signals of the two modules are similar in the range from 9.7 to 3.2 s with the simulated period of 10 s and from 23.8 to 4.9 s with the simulated period of 20 s, and depending on the reactor power level in the range from 0.2 to 70 % $P_{nominal}$.

The obtained results allow to conclude that the FPGA-WR together with the neutron detector KNK-3 through the I to F converter can be performed as an independent neutron flux measurement channel for testing of reactor parameters, research and training purposes at the DNRR; and the current design configuration of the FPGA-WR can replace the BPM-107R module for operating the reactor in the working range.

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