11

Improving the Performance of the Power Monitoring Channel

M. Hashemi-Tilehnoee[1,*] and F. Javidkia[2]
[1]*Department of Engineering, Aliabad Katoul Branch, Islamic Azad University, Aliabad Katoul*
[2]*School of Mechanical Engineering, Shiraz University, Shiraz*
Iran

1. Introduction

In this chapter, different methods for monitoring and controlling power in nuclear reactors are reviewed. At first, some primary concepts like neutron flux and reactor power are introduced. Then, some new researches about improvements on power-monitoring channels, which are instrument channels important to reactor safety and control, are reviewed. Furthermore, some new research trends and developed design in relation with power monitoring channel are discussed. Power monitoring channels are employed widely in fuel management techniques, optimization of fuel arrangement and reduction in consumption and depletion of fuel in reactor core. Power reactors are equipped with neutron flux detectors, as well as a number of other sensors (e.g. thermocouples, pressure and flow sensors, ex-vessel accelerometers). The main purpose of in-core flux detectors is to measure the neutron flux distribution and reactor power. The detectors are used for flux mapping for in-core fuel management purposes, for control actions and for initiating reactor protection functions in the case of an abnormal event (IAEA, 2008). Thus, optimization on power monitoring channel will result in a better reactor control and increase the safety parameters of reactor during operation.

2. Neutron flux

It is convenient to consider the number of neutrons existing in one cubic centimeter at any one instant and the total distance they travel each second while in that cubic centimeter. The number of neutrons existing in a cm^3 of material at any instant is called neutron density and is represented by the symbol n with units of neutrons/cm^3. The total distance these neutrons can travel each second will be determined by their velocity.

A good way of defining neutron flux (ϕ) is to consider it to be the total path length covered by all neutrons in one cubic centimeter during one second. Mathematically, this is the equation below.

$$\phi = n\, v \tag{1}$$

* Corresponding Author

where: ϕ = neutron flux (neutrons cm^{-2} s^{-1}), n = neutron density (neutrons cm^{-3}), and v = neutron velocity (cm s^{-1}). The term neutron flux in some applications (for example, cross section measurement) is used as parallel beams of neutrons travelling in a single direction. The intensity of a neutron beam is the product of the neutron density times the average neutron velocity. The directional beam intensity is equal to the number of neutrons per unit area and time (neutrons cm^{-2} s^{-1}) falling on a surface perpendicular to the direction of the beam. One can think of the neutron flux in a reactor as being comprised of many neutron beams travelling in various directions. Then, the neutron flux becomes the scalar sum of these directional flux intensities. Macroscopic cross sections for neutron reactions with materials determine the probability of one neutron undergoing a specific reaction per centimeter of travel through that material. If one wants to determine how many reactions will actually occur, it is necessary to know how many neutrons are travelling through the material and how many centimeters they travel each second. Since the atoms in a reactor do not interact preferentially with neutrons from any particular direction, all of these directional beams contribute to the total rate of reaction. In reality, at a given point within a reactor, neutrons will be travelling in all directions (DOE, 1993).

3. Power monitoring in nuclear reactors

In order to ensure predictable temperatures and uniform depletion of the fuel installed in a reactor, numerous measures are taken to provide an even distribution of flux throughout the power producing section of the reactor. This shaping, or flattening, of the neutron flux is normally achieved through the use of reflectors that affect the flux profile across the core, or by the installation of poisons to suppress the neutron flux where desired. The last method, although effective at shaping the flux, is the least desirable since it reduces the neutron economy by absorbing the neutrons (DOE, 1993).

In recent years, power monitoring systems are under developing in research centers. Sakai et al. (Sakai et al., 2010) invented a power monitoring system for boiling water reactors (BWRs). In the BWR, the output power alternately falls and rises due to the generation and disappearance of voids, respectively, which may possibly generate power oscillation whereby the output power of the nuclear reactor oscillates and is amplified. The power monitoring system has a local power range monitor (LPRM) unit that has a plurality of local power channels to obtain local neutron distribution in a nuclear reactor core; an averaged power range monitor (APRM) unit that receives power output signals from the LPRM unit and obtains averaged output power signal of the reactor core as a whole; and an oscillation power range monitor (OPRM) unit that receives the power output signals from the LPRM unit and monitors power oscillation of the reactor core. The output signals from the LPRM unit to the APRM unit and the output signals from the LPRM unit to the OPRM unit are independent. A new flux mapping system (FMS) in Korea Electric Power Research Institute (KEPRI) was installed in Kori's unit 1 nuclear power plant. An in-core neutron FMS in a pressurized water reactor (PWR) yields information on the neutron flux distribution in the reactor core at selected core locations by means of movable detectors. The FMS having movable neutron detectors is equipped with detector cable drive units and path selectors located inside the reactor containment vessel. The drive units push and pull their detector cables, which run through guide tubes, and the path selectors route the detector cables into the predetermined guide tubes. Typically, 36–58 guide tubes (thimbles) are allocated in the reactor depending on the number of fuel assemblies. A control system of FMS is located at the main control room to control the detector drive system and measure the flux signal sensed by the detectors. The flux mapping data are used to verify the reactor core design parameters, and to determine the fission power distribution in the core. The new designed path selector for a guide the neutron detectors through the reactor core are shown in Figure 1.

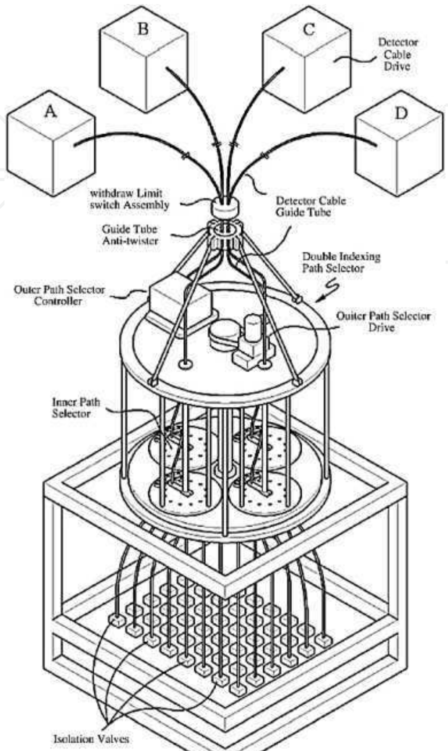


Fig. 1. The new designed path selector for KEPRI unit 1 reactor (Cho et al., 2006)

The path selector system is composed of four inner path selectors and an outer path selector. With the benefit of the double indexing path selector mechanism, the reliability of the detector drive system has been improved five times higher than that of a conventional system. Currently, the developed in-core flux mapping systems have been deployed at the Kori nuclear units 1–4.

4. Neutron flux monitoring and measurement in nuclear reactors

In thermal nuclear reactors, most of the power is generated through fission induced by slow neutrons. Therefore, nuclear sensors those are to be part of reactor control or safety systems are generally based on detectors that respond primarily to slow neutrons. In principle, many of detector types can be adapted for application to reactor measurements. However, the extreme conditions associated with reactor operation often lead to substantial design changes, and a category of slow neutron detectors designed specifically for this application has gradually evolved.

4.1 Neutron detectors and instruments

It is conventional to subdivide reactor instruments into two categories: in-core and out-of-core. In-core sensors are those that are located within narrow coolant channels in the reactor core and are used to provide detailed knowledge of the flux shape within the core. These sensors can be either fixed in one location or provided with a movable drive and must obviously be of rather small size (typically on the order of 10mm diameter). Out-of-core detectors are located some distance from the core and thus respond to properties of the neutron flux integrated over the entire core. The detectors may be placed either inside or outside the pressure vessel and normally will be located in a much less severe environment compared with in-core detectors. Size restrictions are also less of a factor in their design. The majority of neutron sensors for reactor use are of the gas-filled type. Their advantages in this application include the inherent gamma-ray discrimination properties found in any gas detector, their wide dynamic range and long-term stability, and their resistance to radiation damage. Detectors based on scintillation processes are less suitable because of the enhanced gamma-ray sensitivity of solid or liquid scintillators, and the radiation induced spurious events that occur in photomultiplier tubes. Semiconductor detectors are very sensitive to radiation damage and are never used in reactor environments.

4.2 In-core neutron detectors

There is often a need to place neutron sensors within the core of a nuclear reactor to provide information on the spatial variation of the neutron flux. Because of the small size (1-7 cm) of the channel in which these instruments must be located, emphasis is placed on compactness and miniaturization in their design. They may either be left in a fixed position or provided with a motorized drive to allow traverses through the reactor core. Miniaturized fission chambers can be tailored for in-core use over any of the power ranges likely to be encountered in reactor operation. Walls of the chamber are usually lined with highly enriched uranium to enhance the ionization current. These small ion chambers are typically made using stainless steel walls and electrodes, and operating voltage varies from about 50 to 300 V. Argon is a common choice for the chamber fill gas and is used at a pressure of several atmospheres. The elevated pressure ensures that the range of fission fragments within the gas does not exceed the small dimensions of the detector. The gradual burn up of neutron-sensitive material is a serious problem for the long term operation of in-core detectors. Although the change in current-voltage characteristics with increased neutron flux may be greater for in-core detectors than out of core detectors, a similar effect is observed in both the compensated and uncompensated ion chambers used in pressurized water reactors (Knoll, 2000).

4.3 Self-powered detectors

A unique type of neutron detector that is widely applied for in-core use is the self-powered detector (SPD). These devices incorporate a material chosen for its relatively high cross section for neutron capture leading to subsequent beta or gamma decay. In its simplest form, the detector operates on the basis of directly measuring the beta decay current following capture of the neutrons. This current should then be proportional to the rate at which neutrons are captured in the detector. Because the beta decay current is measured directly, no external bias voltage need be applied to the detector, hence the name self-powered. Another form of the self-powered detector makes use of the gamma rays emitted following neutron capture. Some fraction of these gamma rays will interact to form secondary electrons through the Compton, photoelectric, and pair production mechanisms. The current of the secondary electrons can then be used as the basic detector signal. Nonetheless, the self powered neutron detector (SPND) remains the most common term applied to this family of devices. Compared with other neutron sensors, self-powered detectors have the advantages of small size, low cost, and the relatively simple electronics required in conjunction with their use. Disadvantages stem from the low level of output current produced by the devices, a relatively severe sensitivity of the output current to changes in the neutron energy spectrum, and, for many types, a rather slow response time. Because the signal from a single neutron interaction is at best a single electron, pulse mode operation is impractical and self-powered detectors are always operated in current mode.

Figure 2 shows a sketch of a typical SPD based on beta decay.

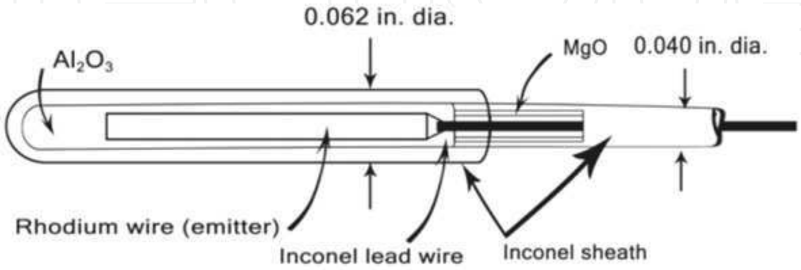


Fig. 2. Cross sectional view of a specific SPD design (Knoll, 2000)

The heart of the device is the emitter, which is made from a material chosen for its relatively high cross section for neutron capture leading to a beta-active radioisotope. Ideally, the remainder of the detector does not interact strongly with the neutrons, and construction materials are chosen from those with relatively low neutron cross sections.

4.4 Neutron instruments (NI) and detectors in pressurized water reactors

Detectors for the routine monitoring of reactor power in a PWR are located outside the reactor pressure vessel and are characterized by the following typical environmental conditions: neutron flux up to 10^{11} n cm^{-2} s^{-1}, gamma irradiation rates up to 10^6 R h^{-1}, and temperatures of approximately 100 °C. Out-of-core sensors are the usual basis of reactor control and safety channels in a PWR. In choosing specific detector types, consideration must be given to the expected neutron signal level compared with noise sources, the speed of response of the detector, and the ability to discriminate against gamma-induced signals.

Each of these criteria assumes different importance over various ranges of reactor power, and as a result multiple detector systems are usually provided, each designed to cover a specific subset of the power range (Knoll, 2000). Figure 3 illustrates a typical scheme for a PWR in which three sets of sensors with overlapping operating ranges are used to cover the entire power range of the reactor.

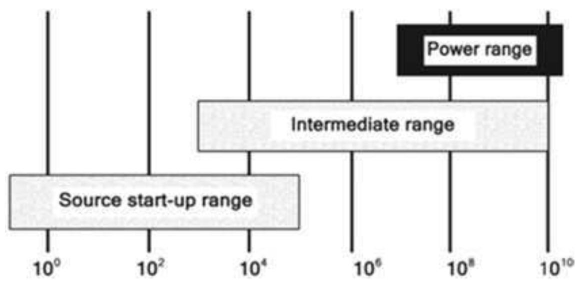


Fig. 3. Typical ranges covered by out-of-core neutron detectors in a PWR (Knoll, 2000)

The lowest range, usually called the source start-up range, is encountered first when bringing up reactor power from shut-down conditions. This range is characterized by conditions in which the gamma flux from the fission product inventory in the core may be large compared with the small neutron flux at these low power levels. Under these conditions, good discrimination against gamma rays is at a premium. Also, the expected neutron interaction rates will be relatively low in this range. Pulse mode operation of either fission chambers or BF_3 proportional counters is therefore possible, and the required gamma-ray discrimination can be accomplished by accepting only the much larger amplitude neutron pulses. As the power level is increased, an intermediate range is encountered in which pulse mode operation is no longer possible because of the excessive neutron interaction rate. In this region the gamma-ray-induced events are still significant compared with the neutron flux, and therefore simple current mode operation is not suitable. The MSV mode of operation can reduce the importance of the gamma-ray signal in this range, but a more common method used in PWRs is to employ direct gamma-ray compensation using a compensated ionization chamber (CIC). A third range of operation corresponds to the region near the full operating power of the reactor. The neutron flux here is usually so large that gamma-ray-induced currents in ion chambers are no longer significant, and simple uncompensated ion chambers are commonly used as the principal neutron sensor. Because these instruments are often part of the reactor safety system, there is a premium on simplicity that also favors uncompensated ion chamber construction.

4.5 Neutron instruments and detectors in boiling water reactors

The BWR NI system, like the PWR system, has three overlapping ranges as illustrated in Figure 4.

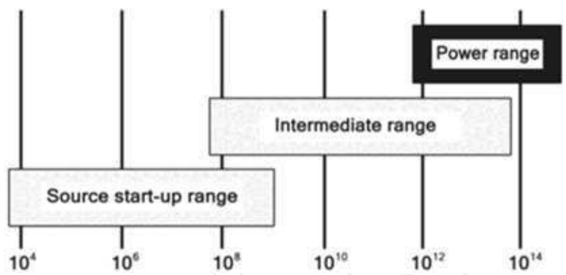


Fig. 4. Typical ranges covered by in-core neutron detectors in a BWR (Knoll, 2000)

The three systems are called source, intermediate, and power range monitors. Unlike the PWR, which uses out-of-core neutron detectors, the neutron detectors are all located in-core. There are also many more detectors used in the BWR NI system than in the PWR system.

The source range monitoring system typically consists of four in-core fission chambers operating in pulse mode. Pulse mode operation provides good discrimination against gamma rays, which is necessary when measuring a relatively low neutron flux in the presence of a high gamma flux. A typical intermediate range monitoring system has eight in-core fission chambers operating in the mean square voltage (MSV) mode. The MSV mode promotes the enhanced neutron to gamma response required to provide a proper measure of neutron flux in the presence of gamma rays for both control and safety requirements. The power range monitoring system typically consists of 144-164 fission ion chambers distributed throughout the core. The fission chambers operate in current mode and are called local power range monitors (LPRM). Current mode operation provides satisfactory neutron response at the high flux levels encountered between 2 and 150% full power. In a typical system, approximately 20 LPRMs are summed to provide input to one of the seven or eight average power range monitoring (APRM) systems. The APRM system provides input for both control and reactor protection systems. In-core flux detectors are used at high power levels (above 10% of full power) because they provide spatial information needed, at high power, to control xenon-induced flux tilts and to achieve the optimum flux distribution for maximum power output. The control system flux detectors are of two types. One type has an inconel emitter and is used for the zone control system. The other type has a vanadium emitter, and is used for the flux mapping system. For power mapping validation, channel temperature differentials are used with measured flows (instrumented channels) or predicted flows (other channels) to determine the estimated channel powers, which are then compared with the powers calculated from the flux mapping readings; this provides an ongoing validation of the accuracy of the flux mapping channel powers.

4.6 Neutron instruments and detectors in CANDU reactors

In CANDU reactors, three instrumentation systems are provided to measure reactor thermal neutron flux over the full power range of the reactor (Knoll, 2000). Start-up instrumentation covers the eight-decade range from 10^{-14} to 10^{-6} of full power; the ion chamber system extends from 10^{-7} to 1.5 of full power, and the in-core flux detector system provides accurate spatial measurement in the uppermost decade of power (10% to 120% of full power). The fuel channel temperature monitoring system is provided for channel flow verification and for power mapping validation. The self-powered in-core flux detectors are installed in flux detector assemblies to measure local flux in the regions associated with the liquid zone controllers. The flux mapping system uses vanadium detectors distributed throughout the core to provide point measurements of the flux. The fast, approximate estimate of reactor power is obtained by either taking the median ion chamber signal (at powers below 5% of full power) or the average of the in-core inconel flux detectors (above 15% of full power) or a mixture of both (5% to 15% of full power).

5. Several advanced power measuring and monitoring systems

The power range channels of nuclear reactors are linear, which cover only one decade, so they do not show any response during the startup and intermediate range of the reactor operation. So, there is no prior indication of the channels during startup and intermediate operating ranges in case of failure of the detectors or any other electronic fault in the channel. Some new reliable instrument channels for power measurement will be studied in this section.

5.1 A wide-range reactor power measuring channel

The power range channels of nuclear reactors are linear, which cover only one decade, so they do not show any response during the startup and intermediate range of the reactor operation. So, there is no prior indication of the channels during startup and intermediate operating ranges in case of failure of the detectors or any other electronic fault in the channel. A new reliable instrument channel for power measurement will be studied in this section. The device could be programmed to work in the logarithmic, linear, and log-linear modes during different operation time of the reactor life cycle. A new reliable nuclear channel has been developed for reactor power measurement, which can be programmed to work in the logarithmic mode during startup and intermediate range of operation, and as the reactor enters into the power range, the channel automatically switches to the linear mode of operation. The log-linear mode operation of the channel provides wide-range monitoring, which improves the self-monitoring capabilities and the availability of the reactor. The channel can be programmed for logarithmic, linear, or log-linear mode of operation. In the log-linear mode, the channel operates partially in log mode and automatically switches to linear mode at any preset point. The channel was tested at Pakistan Research Reactor-1 (PARR-1), and the results were found in very good agreement with the designed specifications. A wide range nuclear channel is designed to measure the reactor power in the full operating range from the startup region to 150% of full power. In the new channel, the status of the channels may be monitored before their actual operating range. The channel provides both logarithmic and linear mode of operation by automatic operating mode selection. The channel can be programmed for operation in any mode, log, linear or log-linear, in any range. In the log-linear mode, the logarithmic mode of operation is used for monitoring the operational status of the channel from reactor startup to little kilowatt reactor power where the mode of operation is automatically changed to linear mode for measurement of the reactor power. At the low power operation, the channel will provide monitoring of the proper functioning of the channel, which includes connection of the electronics with the chamber and functioning of the chamber, amplifier, high-voltage supply of the chamber, and auxiliary power supply of the channel. The channel has been developed using reliable components, and design has been verified under recommended reliability test procedures. The channel consists of different electronic circuits in modular form including programmable log-linear amplifier, isolation amplifier, alarm unit, fault monitor, high-voltage supply, dc-dc converter, and indicator. The channel is tested at PARR-1 from reactor startup to full reactor power. Before testing at the reactor, the channel was calibrated and tested in the lab by using a standard current source. The channel has been designed and developed for use in PARR-1 for reactor power measurement. The response of the channel was continuously compared with ^{16}N channel of PARR-1, and the test channel was calibrated according to the ^{16}N channel at 1 MW. After calibration, it was noticed that the test channel gave the same output as the ^{16}N channel. The channel response with Reactor Power is shown in Figure 5.

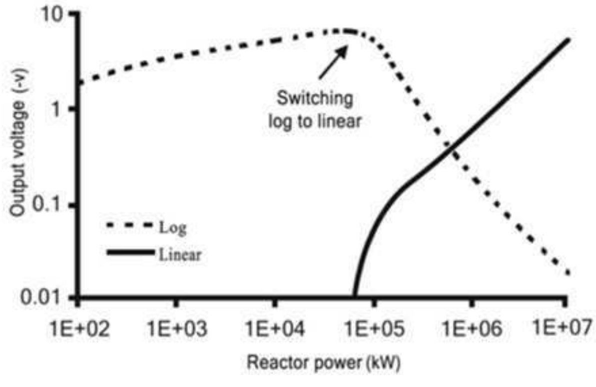


Fig. 5. Response of the channel at different reactor power at PARR-1 (Tahir Khaleeq et al. 2003)

The channel shows an excellent linearity. A very important check was the response of the test channel at the operating mode switching level, and it was found that the channel smoothly switched from log to linear operating mode. The designed channel has shown good performance throughout the operation and on applying different tests. The self-monitoring capabilities of the channel will improve the availability of the system.

5.2 A new developed monitoring channel using ^{16}N detector

^{16}N is one of the radioactive isotopes of nitrogen, which is produced in reactor coolant (water) emitting a Gamma ray with energy about 6 MeV and is detectable by out-core instruments. In this section, a ^{16}N instrument channel in relation to reactor power measurement will be studied. The reactor power and the rate of production of ^{16}N have a linear relation with good approximation. A research type of ^{16}N power monitoring channel subjected to use in Tehran Research Reactor (TRR). Tehran Research Reactor is a 5 MW pool-type reactor which use a 20% enriched MTR plate type fuel. When a reactor is operating, a fission neutron interacts with oxygen atom (^{16}O) present in the water around the reactor core, and convert the oxygen atom into radioactive isotope ^{16}N according to the following (n, p) reaction. Also another possible reaction is production of ^{19}O by $^{18}_{8}O\ (n,\gamma)\ ^{19}_{8}O$ reaction.

Of course, water has to be rich of $^{18}_{8}O$ for at least 22% to have a significant role in ^{19}O producing, but ^{18}O is exist naturally (0.2%)

$$^{1}_{0}n + ^{18}_{8}O \rightarrow ^{19}_{8}O + \gamma(2.8\ \mathrm{MeV}) \tag{2}$$

$$^{1}_{0}n + ^{16}_{8}O \rightarrow ^{1}_{1}P + ^{16}_{7}N^{*} \tag{3}$$

$^{16}N^{*}$ is produced and radiate gamma rays (6MeV) and β particles during its decay chain.

$$^{16}_{7}N^{*} \xrightarrow{7.2\ s} ^{16}_{8}O + ^{0}_{-1}\beta + \gamma(6.13\ \mathrm{MeV}) \tag{4}$$

In addition to $^{16}_{8}O$ (99.76%) and $^{18}_{8}O$ (0.2%), other isotope of oxygen is also exist naturally in water, including $^{17}_{8}O$ (0.04%). $^{17}_{7}N$ (0.037%) produced from $^{17}_{8}O$ by the (n, p) reaction which will decayed through beta emission.

$$^{17}_{8}O + ^{1}_{0}n \rightarrow ^{17}_{7}N + ^{1}_{1}H \tag{5}$$

$$^{17}_{7}N \xrightarrow{4.14\ s} ^{16}_{8}O + ^{0}_{-1}\beta + ^{1}_{0}n \tag{6}$$

Since activity ratio of ^{16}N to ^{17}N is 257/1, thus activity of ^{17}N does not count much and is negligible. Primary water containing this radioactive ^{16}N is passed through the hold-up tank (with capacity of 384.8 m^3, maximum amount of water that can pour to the hold-up tank is 172 m^3 and reactor core flow is 500 $m^3\ h^{-1}$), which is placed under the reactor core and water flow from core down to this tank by gravity force. The hold-up tank delays the water for about 20.7 min. During this period activity of the short lived ^{16}N ($T_{1/2}$ = 7.4 s) decays down to low level. The decay tank and the piping connection to the reactor pool are covered with heavy concrete shielding in order to attenuate the energy of gamma emitted by the ^{16}N nuclei. To investigate the amount of ^{16}N in Tehran Research Reactor by direct measurements of gamma radiation and examine the changes with reactor power, the existing detectors in the reactor control room used and experiment was performed. To assess gamma spectrum for the evaluation of ^{16}N in reactor pool a portable gamma spectroscopy system which includes a sodium-iodide detector is used. The sodium-iodide (NaI) detector which is installed at reactor outlet water side is used for counting Gamma rays due to decay of ^{16}N which depends directly on the amount of ^{16}N. Some advantages of the power measurement using ^{16}N system:

- Power measurement by ^{16}N system uses the gamma from decay of ^{16}N isotope only, so other gammas from impurities do not intervened the measurements.
- Since ^{16}N system installed far from the core, fission products and its gamma rays would not have any effects on the measurements.
- Energy dissipation of heat exchanged with surroundings would not intervene, because water temperature would not use in this system for reactor power measurements.

It is expected that the amount of ^{16}N which is produced in reactor water has linear relation with the reactor power. Comparison of theory and experience is shown in Figure 6.

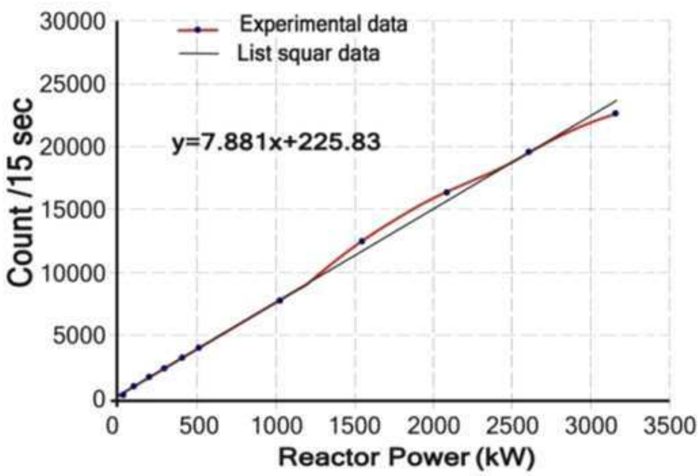


Fig. 6. Comparison of theory and experimental data from ^{16}N channel (Sadeghi, 2010)

Based on graph which resulted from experimental data and the straight line equation using least squire fit, it is appear that the experimental line deviated from what it expected; it means that the line is not completely straight. It seems this small deviation is due to the increasing water temperature around the core in higher power, density reduction and outlet water flow reduction which cause ^{16}O reduction and so ^{16}N. At the same time the amount of ^{16}N production decreases and thus decreasing gamma radiations, this will reduce the number of counting, but on the other hand, since the number of fast neutron production in reactor can increase according to reactor power and moderator density became less, the possibility of neutron interaction with water would increased. During past years, linearity of the curve as the experimental condition and the measurements were improved. Now that this linearity is achieved, by referring to the graph, it could conclude that ^{16}N system is suitable to measure the reactor power. Safety object of the new channel is evaluated by the radiation risk of ^{16}N, dose measurement performed in the area close to the hold-up tank for gamma and beta radiations. The dose received in these areas (except near the hold-up tank charcoal filter box which is shielded) are below the recommended dose limits for the radiation workers (0.05 Sv/year), therefore it can be seen that the radiation risk of ^{16}N is reduced due to design of the piping system and hold-up tank which is distanced from the core to overlap the decay time. Thus, ^{16}N decay through the piping and hold-up tank is reduced to a safe working level. It could be seen that ^{16}N system is able to measure the reactor power enough accurately to be used as a channel of information. For the pool type research reactor which has only one shut down system also could be used to increase the reactor safety (Sadeghi, 2010).

6. Power monitoring by some developed detectors and new methods

In this section, several neutron detectors and power monitoring systems are reviewed.

Application of a micro-pocket fission detector for in-core flux measurements is described in section 6.1. SIC neutron monitoring system is examined experimentally and theoretically. Development of an inconel self-powered neutron detector (SPND) for in-core power monitoring will be reviewed in section 6.3. Furthermore, a prototype cubic meter antineutrino detector which is used as a new device for measuring the thermal power as an out-core detection system, will be discussed. Finally, two passive approaches for power measurement are discussed.

6.1 Micro-pocket fission detectors (MPFD) for in-core neutron flux monitoring

There is a need for neutron radiation detectors capable of withstanding intense radiation fields, capable of performing ‘’in-core’’ reactor measurements, capable of pulse mode and current mode operation, capable of discriminating neutron signals from background gamma ray signals, and that are tiny enough to be inserted directly into a nuclear reactor without significantly perturbing the neutron flux. A device that has the above features is the subject of a Nuclear Engineering Research Initiative (NERI) research project, in which miniaturized fission chambers are being developed and deployed in the Kansas State University (K-State) TRIGA Mark-II research reactor (McGregor, 2005). The unique miniaturized neutron detectors are to be used for three specific purposes (1) as reactor power-level monitors, (2) power transient monitors, and (3) real-time monitoring of the thermal and fast neutron flux profiles in the core. The third application has the unique benefit of providing information that, with mathematical inversion techniques, can be used to infer the three-dimensional (3D) distribution of fission neutron production in the core. Micro-pocket fission detectors (MPFD) are capable of performing near-core and in-core reactor power measurements. The basic design utilizes neutron reactive material confined within a micro-sized gas pocket, thus forming a miniature fission chamber. The housing of the chamber is fabricated from inexpensive ceramic materials, the detectors can be placed throughout the core to enable the 3D mapping of the neutron flux profile in ‘’real-time’’. Initial tests have shown these devices to be radiation hard and potentially capable of operating in a neutron fluence exceeding 10^{19} cm^{-2} without noticeable degradation. Figure 7 shows a cutaway view of the basic detector concept. It consists of a small ceramic structure, within which is a miniature gas-filled pocket.

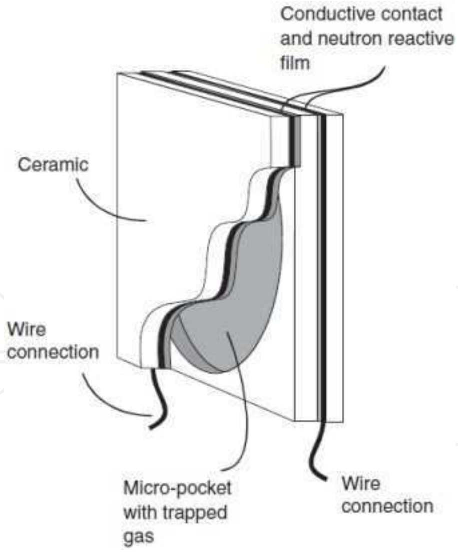


Fig. 7. Cut away view of MPFD (McGregor, 2005)

A conductive layer is deposited on opposing sides of the device, but not the perimeter. Neutron reactive material, such as ^{235}U, ^{232}Th, ^{10}B, or some material containing ^{6}Li, is applied over the conductive contact(s). Although both sides may be coated with neutron reactive material, only one side needs to be coated for the device to work. The ceramic pieces must be insulators and must not be composed of neutron-absorbing material. For instance, aluminum oxide or oxidized silicon may be used. Connecting wires must be sealed well so that no gas leaks out. Additionally, the ceramic pieces must be sealed with high temperature cement such that the seal integrity is secure within the hostile environment of a reactor core.

By in-core evaluation the device demonstrated excellent count-rate linearity with reactor power. Further, the small size and minute amount of uranium used permitted pulse mode operation without appreciable deadtime distortions or problems. MPFDs have, thus far, shown exceptional radiation hardness to neutrons, gamma rays, and charged-particle reaction products, while showing no performance degradation for devices exposed to neutron fluences exceeding 10^{19} cm^{-2}. Further, pulse mode operated devices have shown a linear relation to reactor power for neutron fluxes up to 10^{12} cm^{-2} s^{-1}, and smaller MPFDs are expected to operate in pulse mode in even higher neutron fluxes. The next generation of MPFDs will be composed of a triad of detectors on a single substrate, one with a ^{232}Th coating, one with a ^{235}U coating, and one with no coating. Such a triad permits monitoring of the fast neutron flux, the thermal neutron flux, and the gamma ray background, all at the same time. Further, the devices behave as point detectors, which greatly simplify data interpretation. Data from such a MPFD array can be converted into a power density map of the reactor core for real-time analysis. Mathematical models are under development that can relate the power density profiles in the reactor's fuel rods to the flux densities at the detector locations. Key to this formulation is the construction of an appropriate response function that gives the flux at any position in the core to the fast neutrons born at an arbitrary axial depth in any of the core fuel rods. Response functions have been derived and used to illustrate the analysis methods. Thus far, modeled results using predicted sensitivities of the MPFDs indicated that the power density in the fuel can be determined provided that appropriate boundary conditions regarding device placement are met. Good matching to power density profiles can be achieved with as few as five detector triads per detector string.

6.2 Experimental and computational evaluation of the response of a SiC neutron monitoring system in a thermal neutron field

Silicon carbide (SiC) is an interesting material for nuclear-reactor power monitor detectors. It has a wide band-gap, small volume and high break down electric field. In addition, SiC is chemically and neutronically inactive. Using SiC power monitors as in-core detectors provides the ability for high counting rate that may help to increase the safety margins of nuclear reactors. To observe the triton response in the SiC p-n diode, a detector with a 1.56 µm LiF converter (with 95% enriched ^{6}LiF) was used. ^{6}Li atoms in the LiF converter may absorb thermal neutrons and generate 2.05 MeV alpha and 2.73 MeV triton particles (^{6}Li(n,^{3}H)α reaction). An 8 µm Al layer was used to minimize damage in the SiC by blocking all alpha particles. However, most tritons have enough energy to pass through this layer and reach the 4.8 µm SiC active layer. The diameter of the LiF converter is 0.508 cm and the SiC diode area is 1.1 mm x 1.1 mm (diode is a square). The active area of the diode is approximately 0.965 mm^2. Upon irradiation in the thermal column (TC) facility, one can observe the triton peak in the recorded detector pulse-height spectra and the concomitant triton induced radiation damage on the detector. A schematic of the detector is shown in Figure 8.

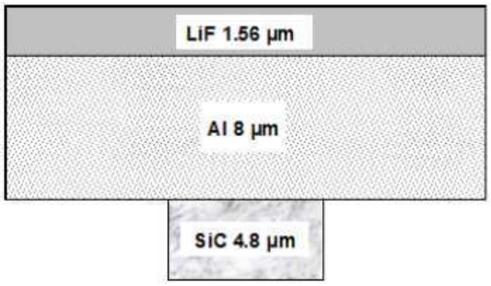


Fig. 8. Schematic of side view of SiC detector. The diameter of the LiF converter is 0.508 cm and the SiC diode active area is 0.965 mm^2 (diode is a square). Only the active region of SiC is shown (Blue and Miller, 2008).

The SiC detector package was connected to a pulse processing system consisting of a preamplifier (ORTEK 142 B) and a digital spectrum analyzer (Canberra DSA 2000). An oscilloscope (Hewlett Packard 54601B, 100 MHz) was used to study the shape of the signal from the amplifier. Bias voltage was provided by the DSA to the detector through the preamplifier. A power monitoring program was used to verify the reactor power that was displayed in the control room. In addition, the degradation of the SiC detectors in the TC's thermal neutron environment was evaluated in terms of dose and dose rate effects. After irradiating the detector at 455 kW, the count rate per kW decreased by a factor of 2 after 11 hr. The I-V characteristics recorded during pre-irradiation and post-irradiation, confirm degradation of the detectors. A theoretical model of the SiC schottky diode detectors was constructed based on MCNP and TRIM computer codes to study the damage induced by tritons for a given diode detector package configuration in the TC's thermal neutron environment. The predicted count rate was compared with the experimental results that were obtained in the TC irradiation field using a charge sensitive preamplifier. The experimental results are in agreement with the predicted response to within a factor of three. I-V measurements show some annealing effects occurring at room temperature. Maintaining the detectors at a higher temperature during irradiation may cause more annealing to occur, thus reducing degradation of the detector. Experiments are necessary to test the degradation of the detector at elevated temperatures, to determine if the effects of annealing are sufficiently great so that the detectors may be useful for neutron power monitoring at high count rates.

6.3 Development of an inconel self powered neutron detector for in-core reactor monitoring

An inconel600 self-powered neutron detector has been developed and tested for in-core neutron monitoring (Alex, 2007). The sensing material in a self-powered detector is an emitter from which electrons are emitted when exposed to radiation. These electrons penetrate the thin insulation around the emitter and reach the outer sheath without polarising voltage. Some electrons are emitted from the insulator and sheath also. The net flow of electrons from the emitter gives rise to a DC signal in an external circuit between the emitter and sheath, which is proportional to the incident neutron flux. Rh and V SPDs work on the basis of (n, β) reaction and are used for flux mapping while Co and Pt SPDs work on the basis of (n, γ − e) prompt reaction and are used for reactor control and safety. However, the build-up of the ^{60}Co and ^{61}Co gives rise to background signal in the cobalt detector thereby reducing the useful life. In the case of the platinum detector, the detector responds to both reactor neutrons via (n, γ, e) interaction and reactor gamma rays via (γ, e) interaction. Since the neutron sensitivity varies with irradiation as a result of burn up while the gamma sensitivity remains the same, the dynamic response of a mixed response detector varies with time. This mixed and time-dependent response of platinum SPD gives rise to anomalous behaviour in some situations. Development of SPDs with inconel emitters as alternative to Co and Pt prompt SPDs has been reported in literatures. The detector (Figure 9) consists of a 2 mm diameter × 21 cm long inconel 600 emitter wire surrounded by a high purity alumina ceramic tube (2.2 mm ID × 2.8 mm OD). The assembly is enclosed in a 3 mm ID × 3.5 mm OD inconel600 tube.

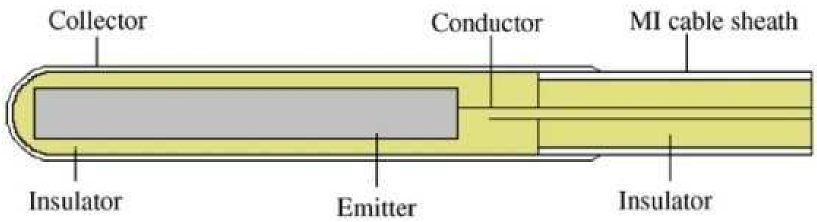


Fig. 9. Schematic diagram of self powered neutron detector

One end of the emitter is coupled to the conductor of a 2 mm diameter × 12 m long twin core mineral insulated (MI) cable while the detector sheath is laser welded to the MI cable sheath. The detector is integrally coupled to the MI cable and the cold end of the cable is sealed by a twin core ceramic-to-metal seal over which a Lemo connector is fitted.

The gamma sensitivity of the detectors was measured in pure gamma field using ^{60}Co source facility. The detectors were placed at a distance of 1m from the source for better source to detector geometry and 1m above the ground to minimize background from scattered rays. To estimate the gamma field at the detector location, a miniature gamma ion chamber (6 mm diameter and 25 mm long) was used. The calculated gamma sensitivity, 24.8 (fA R^{-1} h) was used to determine the gamma field at the self-powered neutron detector location. The three SPNDs (inconel600, cobalt, platinum) and were tested together with the miniature gamma chamber in a 200 kCi ^{60}Co source facility. The results showed that the gamma response of the inconel600 and Co detector was found to be similar. However, it was observed that unlike the platinum detector, which has positive response, the Co and inconel detectors showed negative response. The gamma sensitivity of the inconel600 detector is about 7.7 times lower than Pt detector. This low gamma response of the inconel600 detector improves the neutron to gamma ratio and makes it desirable for reactor safety and control applications. In addition to gamma sensitivity, the neutron sensitivity of SPNDs was tested in dry tube (55 mm diameter ×8.4 m long) in-core location of the Pool type reactor. The neutron sensitivity and the total sensitivity of the inconel600 detector were found to be lower than the Co detector. The total sensitivity of the inconel SPD is about 20–25% of the sensitivity of cobalt and about 35% of the sensitivity of platinum detectors of similar dimensions; however, it is proposed to improve the sensitivity by helically winding the detector with a short axial length. Finally by comparison, the performance of the inconel detector with cobalt and platinum detectors of similar dimensions, it is obvious that inconel SPD is a useful alternative to Co and Pt SPDs.

6.4 Monitoring the thermal power of nuclear reactors with a prototype cubic meter antineutrino detector

A new power monitoring method applied to a pressurized water reactors designed by combustion engineering. The method estimate quickly and precisely a reactor's operational status and thermal power can be monitored over hour to month time scales, using the antineutrino rate as measured by a cubic meter scale detector. Antineutrino emission in nuclear reactors arises from the beta decay of neutron-rich fragments produced by heavy element fissions, and is thereby linked to the fissile isotope production and consumption processes of interest for reactor safeguards. On average, fission is followed by the production of approximately six antineutrinos. The antineutrinos emerge from the core isotropically, and effectively without attenuation. Over the few MeV energy range within which, reactor antineutrinos are typically detected, the average number of antineutrinos produced per fission is significantly different for the two major fissile elements, ^{235}U and ^{239}Pu. Hence, as the core evolves and the relative mass fractions and fission rates of these two elements change, the measured antineutrino flux in this energy range will also change. It is useful to express the relation between fuel isotopic and the antineutrino count rate explicitly in terms of the reactor thermal power, P_{th}. The thermal power is defined as

$$P_{th} = \sum_i N_i^f . E_i^f \tag{7}$$

where N_i^f is the number of fissions per unit time for isotope i, and E_i^f is the thermal energy released per fission for this isotope. The sum runs over all fissioning isotopes, with ^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu accounting for more than 99% of all fissions. The antineutrino emission rate $n_{\bar{\nu}}(t)$ can then be expressed in terms of the power fractions and the total thermal power as:

$$n_{\bar{\nu}}(t) = P_{th}(t) \sum_i \frac{f_i(t)}{E_i^f} \int \varphi_i\,(E_{\bar{\nu}})\, dE_{\bar{\nu}} \tag{8}$$

where the explicit time dependence of the fission fractions and, possibly, the thermal power are noted. $\varphi(E_{\bar{\nu}})$, is the energy dependent antineutrino number density per MeV and fission for the ith isotope. $\varphi(E_{\bar{\nu}})$ has been measured and tabulated. Equation 7 defines the burn-up effect. The fission rates $N_i^f(t)$ and power fractions $f_i(t)$ change by several tens of percent throughout a typical reactor cycle as ^{235}U is consumed and ^{239}Pu produced and consumed in the core. These changes directly affect the antineutrino emission rate $n_{\bar{\nu}}(t)$. Reactor antineutrinos are normally detected via the inverse beta decay process on quasi-free protons in hydrogenous scintillator. In this charged current interaction, the antineutrino $\bar{\nu}$ converts the proton into a neutron and a positron: $\bar{\nu} + p \rightarrow e^+ + n$. For this process, the cross section σ is small, with a numerical value of only $\sim 10^{-43}\text{cm}^2$. The small cross section can be compensated for with an intense source such as a nuclear reactor. For example, cubic meter scale hydrogenous scintillator detectors, containing $\sim 10^{28}$ target protons N_p, will register thousands of interactions per day at standoff distances of 10-50 meters from typical commercial nuclear reactors. In a measurement time T, the number of antineutrinos detected via the inverse beta decay process is:

$$N_{\bar{\nu}}(t) = \left(\frac{TN_p}{4\pi D^2}\right) P_{th}(t) \sum_i \frac{f_i(t)}{E_i^f} \int \sigma\, \varphi_i\, \epsilon\, dE_{\bar{\nu}} \tag{9}$$

In the above equation, σ is the energy dependent cross section for the inverse beta decay interaction, N_p is the number of target protons in the active volume of the detector, and D is the distance from the detector to the center of the reactor core. ϵ is the intrinsic detection efficiency, which may depend on both energy and time. The antineutrino energy density and the detection efficiency are folded with the cross section σ, integrated over all antineutrino energies, and summed over all isotopes i to yield the antineutrino detection rate. The SONGS1 detector consists of three subsystems; a central detector, a passive shield, and a muon veto system. Figure 10 shows a cut away diagram of the SONGS1 detector. Further information can be found in (Bowden, 2007) and (Bernstein et al., 2007).

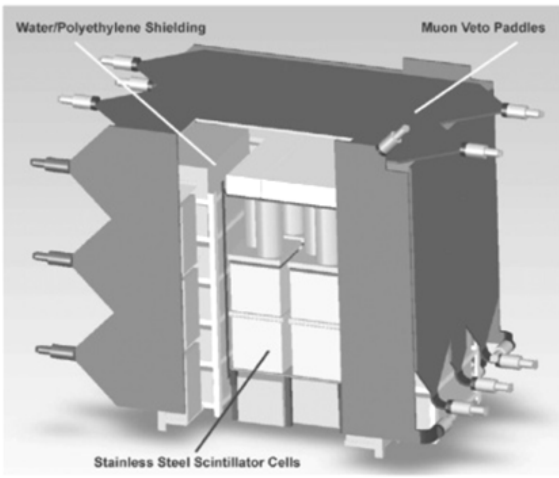


Fig. 10. A cut away diagram of the SONGS1 detector (showing the major subsystems).

This prototype that is operated at 25 meter standoff from a reactor core, can detect a prompt reactor shutdown within five hours, and monitor relative thermal power to 3.5% within 7 days. Monitoring of short-term power changes in this way may be useful in the context of International Atomic Energy Agency's (IAEA) Reactor Safeguards Regime, or other cooperative monitoring regimes.

6.5 Application of Cherenkov radiation and a designed detector for power monitoring

Cherenkov radiation is a process that could be used as an excess channel for power measurement to enhance redundancy and diversity of a reactor. This is especially easy to establish in a pool type research reactor (the TRR). A simple photo diode array is used in Tehran Research Reactor to measure and display power in parallel with the existing conventional detectors (Arkani and Gharib, 2009). Experimental measurements on this channel showed that a good linearity exists above 100 kW range. The system has been in use for more than a year and has shown reliability and precision. Nevertheless, the system is subject to further modifications, in particular for application to lower power ranges. TRR is originally equipped with four channels, namely, a fission chamber (FC), a compensated ionization chamber (CIC), and two uncompensated ionization chambers (UIC). However, in order to improve the power measuring system, two more channels have also been considered for implementation in recent years. One of these channels is based on 16O (n,p) ^{16}N reaction which is very attractive due to the short half life of ^{16}N (about 7 s). The other channel, at the center of our attention in this work, is based on measurement of Cherenkov radiation produced within and around the core. This channel has a fast response to power change and has been in operation since early 2007. It has been established that the movement of a fast charged particle in a transparent medium results in a characteristic radiation known as Cherenkov radiation. The bulk of radiation seen in and around a nuclear reactor core is mainly due to Beta and Gamma particles either from fission products or directly emanating from the fission process (prompt fission gamma rays). As it will be explained more thoroughly in the following section, Cherenkov radiation is produced through a number of ways when: (a) beta particles emitted by fission products travel with speeds greater than the speed of light in water and (b) indirect ionization by Gamma radiation produces electrons due to photo electric effect, Compton effect and pair production effect. Among these electrons, Compton electrons are the main contributors to Cherenkov radiation. It is established that Cherenkov light is produced by charged particles which pass through a transparent medium faster than the phase velocity of light in that medium. Considering the fact that speed of light in water is 220,000 km/s, the corresponding electron energy that is required to produce Cherenkov light is 0.26 MeV. This is the threshold energy for electrons that are energetic enough to produce Cherenkov light. It is the principal basis of Cherenkov light production in pool type research reactors in which the light is readily visible. For prompt Gamma rays, in general, it makes it possible to assume that Cherenkov light intensity is a linear function of reactor power. It is clear that neutron intensity, fission rate, power density, and total power itself are all inter-related by a linear relationship. In other words, Cherenkov light intensity is also directly proportional to the fission rate. This leads us to the fact that the measured Cherenkov light intensity at any point in a reactor is linearly proportional to the instantaneous power. As long as the measurement point is fixed, the total power could easily be derived from the light intensity with proper calibration. It should be noted here that, as mentioned before, Cherenkov light is also emitted by the electrons produced by the indirect ionization of fission products by Gamma rays, which are confined in fuel elements. For this reason, a linear relationship between reactor power and Cherenkov light intensity would only hold at the higher power range where fission power is dominant in comparison with residual power. Cherenkov light emanating from core is collected by a collimator right above the core and reflected by a mirror onto a sensitive part of the PDA. Figure 11 shows the integrated system at work, overlooking the core.

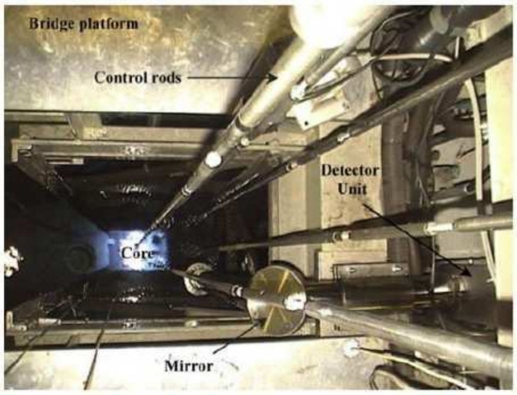


Fig. 11. Power measuring channel at work in TRR while receiving Cherenkov light (Arkani and Gharib, 2009).

An important factor to be checked is the system fidelity. This means that the response of the system must be the same when the reactor power is raised or lowered. There is a good fidelity within the linearity range by comparison of the Cherenkov system with the output of CIC power monitoring channel. Moreover, there has been no drift observed in the system in the long run as the system functioned properly for almost 2 years since it was installed. Finally, it is necessary to examine whether the reading from the Cherenkov detector is consistent with other channels. Finally, it is necessary to examine whether the reading from the Cherenkov detector is consistent with other channels. Figure 12 shows its good consistency with other conventional channels (only the fission chamber is shown for the sake of simplicity) within a typical shift operation.

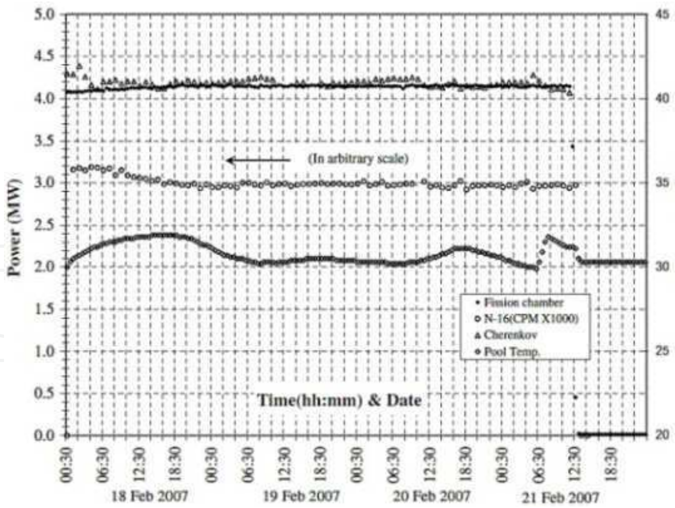


Fig. 12. Comparison of Cherenkov detector output with other regular channels within a typical operation shift of TRR (Arkani and Gharib, 2009).

It is observed that the steadiness and stability of the Cherenkov detector is as good as other existing channels. The ^{16}N counts and pool average temperature are also included as further confirmation of the general behavior of the reactor during the operation. Reasonable stability is observed in the hourly readings of all the channels. Based on statistics, the output value of the present PDA system is valid within ±1% at its nominal power. It is concluded that, at least for the case of research reactors, one can simply increase redundancy and diversity of medium-range reactors by employing the Cherenkov detector as an auxiliary tool for monitoring purposes. It is seen that such a system can provide a stable and reliable tool for the major part of power range, and it can assist in the reactor operation with additional safety interlocks to issue appropriate signals. The advantage of the present detector system over conventional ones is that it is far from the radiation source and thus easily accessible for maintenance and fine tuning. It contains no consumable materials to degrade in long term, and it is relatively inexpensive and simple. Nevertheless, a drawback of the Cherenkov system, which is also true about uncompensated ionization chambers, is its lack of linearity in the low power range.

6.6 A digital reactivity meter related to reactor power measuring process

Reactivity is a physical characteristic of the core (based on composition, geometry, temperature, pressure, and the ability of the core to produce fission neutrons) and may be either constant or changing with time. In reactor operation or experiments, signals indicating reactor power (or neutron flux) and reactor period are generally used for direct information on the state of the reactor. However, the most important time dependent parameter is reactivity and continuous information on its value from instant to instant should be highly useful. Since reactivity measurement is one of the challenges of monitoring, control and investigation of a nuclear reactor and is in relation with reactor power measuring. Thus, design and construction of a digital reactivity meter as a continuous monitoring of the reactivity will be reviewed in a research reactor. The device receives amplified output of the fission chamber, which is in *mA* range, as the input. Using amplifier circuits, this current is converted to voltage and then digitalized with a microcontroller to be sent to serial port of computer. The device itself consists of software, which is a MATLAB real time programming for the computation of reactivity by the solution of neutron kinetic equations. After data processing the reactivity is calculated and presented using LCD. Tehran research reactor is selected to test the reactivity meter device. The results of applying this reactivity meter in TRR are compared with the experimental data of control rod worth, void coefficient of reactivity and reactivity changes during approach to full power. Three experiments for system verification for TRR are; determination of control rod worth, void coefficient experiment, and measuring of reactivity during approach to full power (Khalafi and Mosavi, 2011). For investigating the results of reactivity meter, the reactor power and reactivity plots during the step-wise approach to full power of a particular run of TRR reactor are shown in Figure 13. In this experiment the reactor power was initially stable and critical at 100 kW and a positive reactivity insertion was introduced in the core by changes in control rods positions.

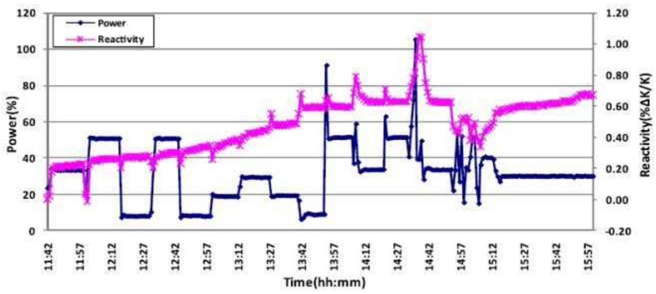


Fig. 13. Power and reactivity plots versus time (Khalafi and Mosavi, 2011).

The maximum relative error in three experiments is 13.3%. This error is caused by discrete signal that is transferred to the reactivity meter device. A great portion of the data is lost in the discrete signal and some others in the sampling process. As described in this section, the system of a digital reactivity meter developed on a PIC microcontroller and the personal computer is proved to function satisfactorily in the nuclear research reactor and the utilization of the plant instrument signals makes the system simple and economical. Besides, this device can be used to determine the positive reactivity worth of the fresh fuel and the reflector elements added to the core, effectively. According to the above experiments, the relative error of the digital reactivity meter can be reduced by increasing the sampling frequency of the device. Also by using digital signal processing (DSP) utilities, the rate and accuracy of the reactivity meter can be improved. Because derivative circuits are not used in this device, the error due to the noise that is observed in analog circuits decreases extremely.

7. Application of computational codes in simulation, modeling and development of the power monitoring tools

Some developed codes and simulators for improving the power monitoring will be reviewed in this section. For example, MCNP (monte-carlo n-particle transport code) is developed for neutron detector design, or modeling a fission chamber to optimize its performance

7.1 Computational tools to conduct experimental optimization

Research reactors need a handy computational tool to predict spatial flux changes and following power distribution due to experimental requirements. Therefore it is important to get accurate and precise information ahead of any modifications. To meet this demand, flux measurements were conducted in case that a typical flux trap inside the core to be allocated. In TRR, one of standard fuel boxes, in position D6 in core configuration of the year 1999, was taken out of the core and a water trap was formed in its place. With the aid of miniature neutron detector (MND) using standard procedure, thermal neutron flux is measured inside the water trap. To calculate the flux and power theoretically, two different computational approaches such as diffusion and Monte Carlo methods were chosen. Combination of cell calculation transport code, WIMS-D5, and three-dimensional core calculation diffusion code, such as CITATION, were used to calculate neutron flux inside the whole core either in two or five energy groups. However, MCNP-4, as a Monte Carlo code, was used to calculate neutron flux again inside the whole core as well as inside the trap (Khalafi and Gharib, 1999). Figure 14 shows axial thermal flux distribution along the D6 position by measurement and computation.

It is obvious from the figure; the both calculation codes are satisfactory and a good agreement exists between detector measurements and code computations. However, diffusion method is a rational choice especially for survey calculation where the Monte Carlo approach is more time demanding. For some consideration, in order to measure spectrum, a fixed point on the midplane along D6 axis was chosen. A variety of foils of different material was selected as measuring windows to determine differential fluxes at specified energy bins. Metal foils such as Ti, Se, Mg, Ni, Al, Co, Au, In, and Fe were selected as energy windows. These foils are sensitive to a part of neutron energy spectrum starting form high energies and ending to thermal energies. Induced activity of each foil is measured based on gamma spectroscopy using high purity Germanium (HPGe) detector. By providing raw counts to SAND-II computer code, neutron energy spectrum was calculated. The measured and calculated spectrum using neutron detector, MCNP and WIMS codes is shown in Figure 15.

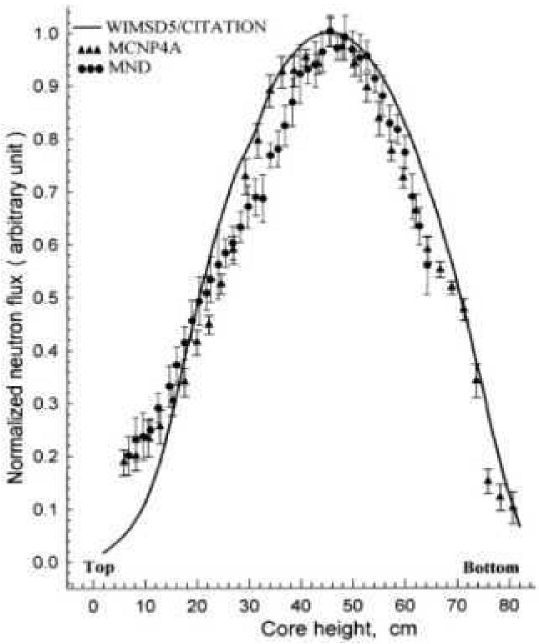


Fig. 14. Axial thermal neutron flux distribution in trap at D6 position (Khalafi and Gharib, 1999)

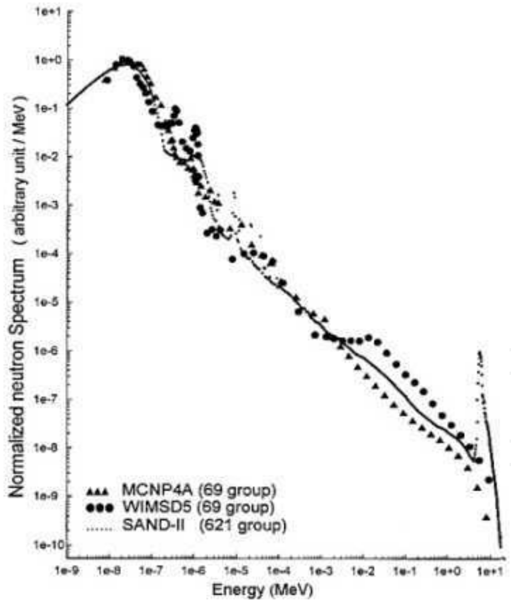


Fig. 15. Detector measured and MCNP and WIMS code calculated neutron spectrum(Khalafi and Gharib, 1999)

Spectrum calculations were also checked against measurements. Monte Carlo shows a better prediction while WIMS provides a fair result. It is notable that combination of WIMS/CITATION would be sufficient for neutron flux calculations while Monte Carlo technique should be reserved for the final stages of simulation. A good choice of computational tools would save time a lot in this respect and one is encouraged to perform a comprehensive simulation ahead of design and construction of irradiation facility.

7.2 Optimizing the performance of a neutron detector in the power monitoring channel of TRR

A fission chamber was utilized for neutron detection in TRR. It was a valuable instrument for in-core/out-core information and the core status monitoring during normal and transient operations. A general theoretical model is presented to calculate the current-voltage characteristics and associated sensitivity for a fission chamber. The chamber was used in the research nuclear reactor, TRR, and a flux-mapping experiment was performed. The experimental current measurement in certain locations of the reactor was compared with the theoretical model results. The characteristic curves were obtained as a function of fission rate, chamber geometry, and chamber gas pressure. An important part of the calculation was related to the operation of the fission chamber in the ionization zone and the applied voltages affecting two phenomena, recombination and avalanche. In developing the theoretical model, the MCNP code was used to compute the fission rate and the SRIM program for ion-pairs computations. In modeling the source for MCNP, the chamber was placed in a volume surrounded by standard air. Figure 16 illustrates the geometrical details of the MCNP simulation (Hashemi-Tilehnoee and Hadad, 2009).

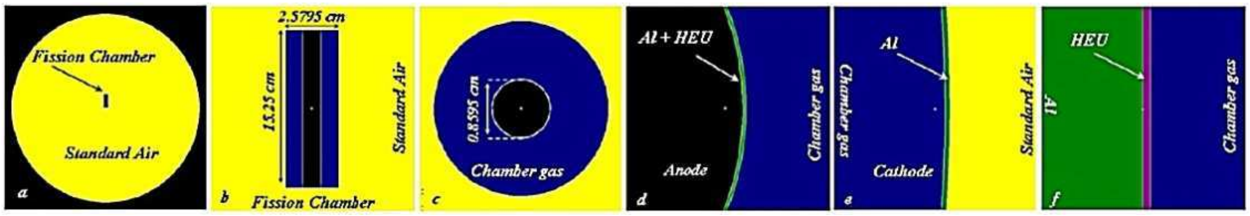


Fig. 16. The geometrical details of the MCNP simulation. (a) The chamber is placed in a volume surrounded by standard air, (b) the chamber geometrical details, width = 2.5795cm and length = 15.25 cm, (c) the cross section of the fission chamber, (d) the anode, (e) the cathode and (f) the fissile element coating.

The theoretical model together with the mentioned codes was used to evaluate the effects of different applicable variations on the chamber's parameters. An effective approach in decreasing the minimum voltage in the plateau zone, and retaining the chamber in the ionization zone, is to reduce the chamber gas pressure. However, by reducing the pressure, we decrease the gas density. This leads to the reduction of ion-pairs generation rate. Reduction of ion-pairs would affect the sensitivity. At high pressures, the plateau zone width would be extended. This extension needs a stronger electric field, which in turn causes the distortion of the electric field due to space charge effect. Thus, pressure is an important parameter in design considerations. Variations in the enrichment of the fissile element resulted in the enhancement of the fission rate and hence the sensitivity while retaining the applied voltage and plateau zone width. However, surface mass increase would require more applied voltage. Sensitivity of detection of the neutron flux would increase by decreasing the inter-electrode gap. In addition, it increases the width of the plateau zone. This extension optimizes the chamber performance and decreases the detection errors. Furthermore, by decreasing the inter-electrode gap, the fission chamber can be used in a low flux neutron surrounding for detection with high resolution. In contrast, by increasing the inter-electrode gap, the fission chamber can be used in a high flux nuclear reactor. Since the pressure variations have significant effects on the sensitivity, the detector components should be designed in accordance with the location, temperature, and neutron flux of the nuclear reactor core. Finally, applying the proper voltage not only enhances the sensitivity and readout, but also increases the longevity of the chamber.

In addition, the chamber is modeled by GEANT4 to evaluate its sensitivity to gamma ray, which exists as background. Figure 17 illustrates geometry of the modeled chamber in GEANT4. The unwanted noises from gamma ray in the core are dispensable, but in laboratory, this sensitivity must be accounted for the experiments as a disturbance signal.

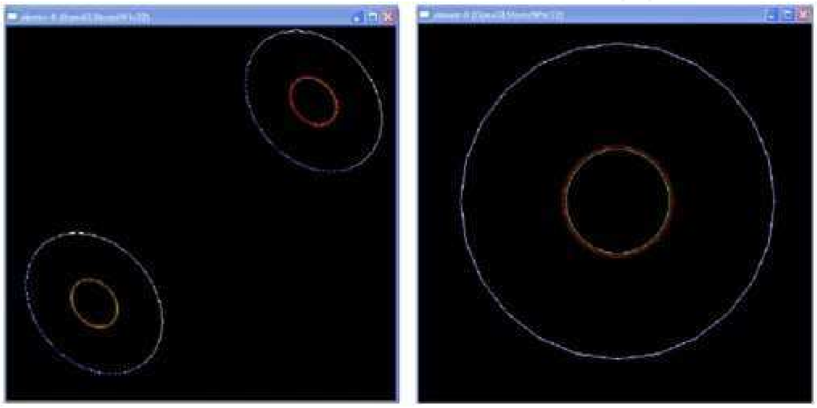


Fig. 17. Geometry of the modeled chamber in GEANT4

8. Thermal methods for power monitoring of nuclear reactor

Power monitoring using thermal power produced by reactor core is a method that is used in many reactors. To explain how the method is used for reactor power measurement, a research reactor is studied in this section. In IPR-R1, a TRIGA Mark I Research Reactor, the power is measured by four nuclear channels. The departure channel consists of a fission counter with a pulse amplifier that a logarithmic count rate circuit. The logarithmic channel consists of a compensated ion chamber, whose signal is the input to a logarithmic amplifier, which gives a logarithmic power indication from less than 0.1 W to full power. The linear channel consists of a compensated ion chamber, whose signal is the input to a sensitive amplifier and recorder with a range switch, which gives accurate power information from source level to full power on a linear recorder. The percent channel consists of an uncompensated ion chamber, whose signal is the input to a power level monitor circuit and meter, which is calibrated in percentage of full power. The ionization chamber neutron detector measures the flux of neutrons thermalized in the vicinity of the detector. In the present research, three new processes for reactor power measurement by thermal ways were developed as a result of the experiments. One method uses the temperature difference between an instrumented fuel element and the pool water below the reactor core. The other two methods consist in the steady-state energy balance of the primary and secondary reactor cooling loops. A stainless steel-clad fuel element is instrumented with three thermocouples along its centerline in order to evaluate the reactor thermal hydraulic performance. These processes make it possible on-line or off-line evaluation of the reactor power and the analysis of its behavior.

8.1 Power measuring channel by fuel and pool temperature

To evaluate the thermal hydraulic performance of the IPR-R1 reactor one instrumented fuel element was put in the core for the experiments. The instrumented fuel is identical to standard fuel elements but it is equipped with three chromel-alumel thermocouples, embedded in the zirconium pin centerline. The sensitive tips of the thermocouples are located one at the center of the fuel section and the other two 25.4 mm above, and 25.4 mm below the center. Figure 18 shows the diagram and design of the instrumented fuel element (Zacarias Mesquita and Cesar Rezende, 2010).

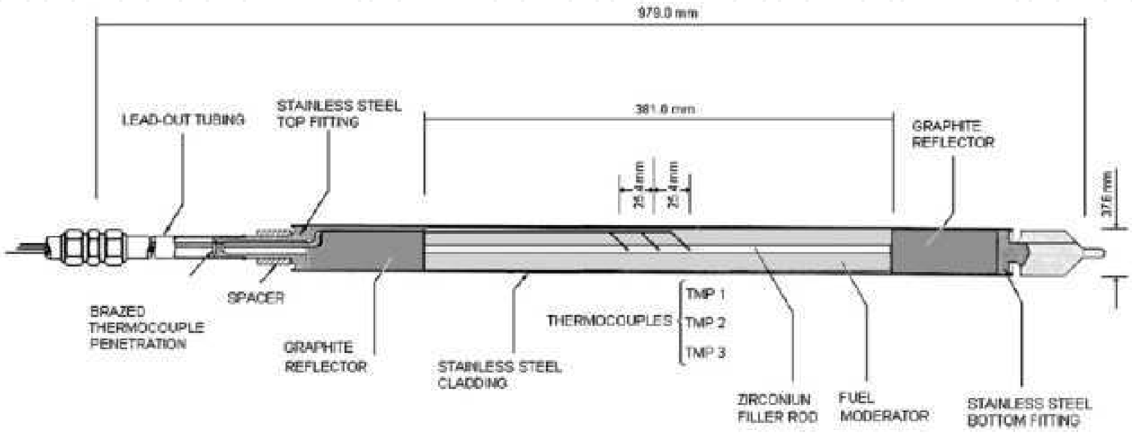


Fig. 18. Diagram of the instrumented fuel element (Zacarias Mesquita and Cesar Rezende, 2010)

The instrumented fuel element which is placed in proper thimble (B6 position) is obvious in Figure 19, a core upper view.

Fig. 19. Core upper view with the instrumented fuel element (Zacarias Mesquita and Cesar Rezende, 2010)

During the experiments it was observed that the temperature difference between fuel element and the pool water below the reactor core (primary loop inlet temperature) do not change for the same power value. Figure 20 compares the reactor power measuring results using the linear neutron channel and the temperature difference channel method (Zacarias Mesquita and Cesar Rezende, 2007).

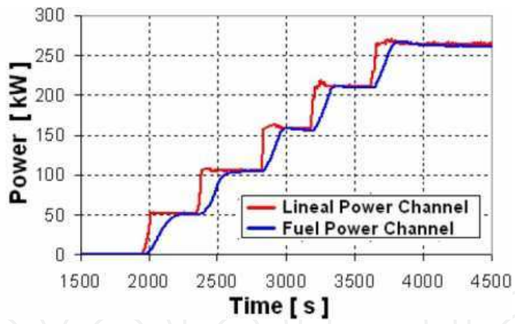


Fig. 20. Reactor power measured by neutron channel and by fuel element temperature (Zacarias Mesquita and Cesar Rezende, 2007).

There is a good agreement between the two results, although the temperature difference method presents a delay in its response, and it is useful for steady-state or very slow transient. It is notable that the thermal balance method presented in this report is now the standard methodology used for the IPR-R1 TRIGA Reactor power calibration. The heat balance and fuel temperature methods are accurate, but impractical methods for monitoring the instantaneous reactor power level, particularly during transients. For transients the power is monitored by the nuclear detectors, which are calibrated by the thermal balance method (Zacarias Mesquita and Cesar Rezende, 2007).

8.2 Power measuring channel by thermal balance

The new developed on-line monitoring method which is based on a temperature difference between an instrumented fuel element and the pool water below a research reactor in practice, as known power measuring by thermal balance is as following. The reactor core is cooled by natural convection of demineralized light water in the reactor pool. Heat is removed from the reactor pool and released into the atmosphere through the primary cooling loop, the secondary cooling loop and the cooling tower. Pool temperature depends on reactor power, as well as external temperature, because the latter affects heat dissipation in the cooling tower. The total power is determined by the thermal balance of cooling water flowing through the primary and secondary loops added to the calculated heat losses. These losses represent a very small fraction of the total power (about 1.5% of total). The inlet and outlet temperatures are measured by four platinum resistance thermometers (PT-100) positioned at the inlet and at the outlet pipes of the primary and secondary cooling loops. The flow rate in the primary loop is measured by an orifice plate and a differential pressure transmitter. The flow in the secondary loop is measured by a flow-meter. The pressure transmitter and the temperature measuring lines were calibrated and an adjusted equation was added to the data acquisition system. The steady-state is reached after some hours of reactor operation, so that the power dissipated in the cooling system added with the losses should be equal to the core power. The thermal power dissipated in the primary and secondary loops were given by:

$$q_{cool} = \dot{m}.c_P.\Delta T \tag{10}$$

where q_{cool} is the thermal power dissipated in each loop (kW), $\dot{m}$ is the flow rate of the coolant water in the loop (kg. s^{-1}), c_P is the specific heat of the coolant (kJ kg^{-1} °C^{-1}), and ΔT is the difference between the temperatures at loop the inlet and outlet (°C). Figure 21 shows the power evolution in the primary and secondary loops during one reactor operation.

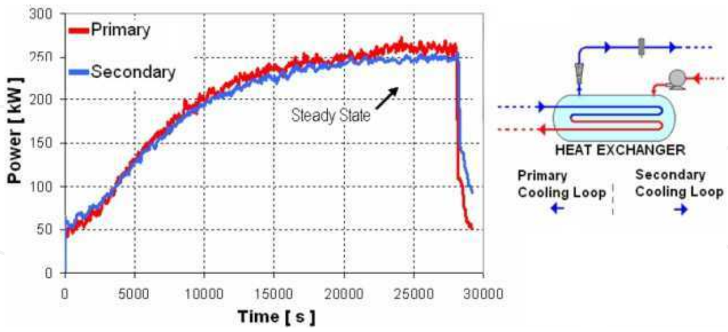


Fig. 21. Thermal power evolution in the cooling system (Zacarias Mesquita and Cesar Rezende, 2007).

9. Acknowledgement

The authors are thankful InTech publication for their scientific effort to develop the sciences. We also, thank Mrs. M. Mohammadi for her effort to preparing the manuscript.

10. Conclusion

Power monitoring channels play a major role in retaining a safe reliable operation of nuclear reactors and nuclear power plants. Accurate power monitoring using advanced developed channels could make nuclear reactors a more reliable energy source and change public mind about this major energy resource. Regarding harsh accidents such as Chernobyl, Three-Mile Island and the recent accidents in Fukushima nuclear power plants and their dangerous effects on the environment and human life, the importance of developing reactor safety system like power monitoring channels are more attended. New generations of nuclear power plants are much safer than their predecessors because of their new accurate safety systems and more reliable monitoring channels. They produce energy from nuclear fission and are the cleanest, safe and environment-friendly source of energy among many investigated power resources (Javidkia et al., 2011).

There is no doubt that nuclear power is the only feasible green and economic solution for today's increasing energy demand. Therefore, studying, researches and more investments on the power monitoring systems and channel in nuclear reactors will help to create an inexhaustible source of safe and clean energy.

11. References

IAEA, (2008). *On-line monitoring for improving performance of nuclear power plants part 2: process and component condition monitoring and diagnostics*, International atomic energy agency (IAEA), Vienna, ISSN 1995–7807; no. NP-T-1.2, STI/PUB/1323, ISBN 978–92–0–101208–1.

Department of energy, (1993). DOE fundamentals handbook, Nuclear physics and reactor theory, Vol.1 and 2, DOE-HDBK-1019/1-93 & 2-93.

Sakai, H.; Oda, N.; Miyazaki, T.; Sato, T. (2010). *Power monitoring system*, US 2010/0254504 A1.

Cho, B.; Shin, C.; Byun, S.; Park, J.; Park, S. (2006). *Development of an innovative neutron flux mapping system,* Nuclear Engineering and Design, Vol. 236, 1201–1209.

Knoll, G.E. (2000). *Radiation detection and measurement*, 3rd edition, ISBN 0-471-07338 , John Wiley & Son.

Tahir Khaleeq, M.; Zaka, I.; Qaiser, Hameed.; Nayyar, B. H.; Ahmad Ghumman, I.; Ali, A. (2003). *A New Wide-Range Reactor Power-Measuring Channel,* IEEE transactions on nuclear science, Vol. 50, No. 6.

Sadeghi, N. (2010). *Estimation of reactor power using N16 production rate and its radiation risk assessment in Tehran Research Reactor (TRR)*, Nuclear Engineering and Design, Vol. 240, 3607–3610.

McGregor, D. S.; F. Ohmes, M.; E. Ortiz, R.; Sabbir Ahmed, A.S.M.; Kenneth Shultis, J. (2005). Micro-pocket fission detectors (MPFD) for in-core neutron flux monitoring, Nuclear Instruments and Methods in Physics Research A, Vol. 554, 494–499.

Blue, T. E.; Miller, D.W. (2008). *Nuclear reactor power monitoring using silicon carbide semiconductor radiation detectors*, nuclear energy research initiative (NERI), Project Number DE-FG03-02SF22620.

Alex, M.; Ghodgaonkar, M.D. (2007). *Development of an inconel self powered neutron detector for in-core reactor monitoring*. Nuclear Instruments and Methods in Physics Research A, Vol. 574, 127–132.

Bowden, N.S.; Bernstein, A.; Allen, M.; Brennan, J.S.; Cunningham, M., Estrada, J.K.; Greaves, C.M.R.; Hagmann, C.; Lund, J.; Mengesha, W.; Weinbeck, T.D.; Winant, C.D. (2007). *Experimental results from an antineutrino detector for cooperative monitoring of nuclear reactors*, Nuclear Instruments and Methods in Physics Research A, Vol. 572, 985–998.

Bernstein, A.; Bowden, N. S.; Misner, A.; Palmer, T. (2007). *Monitoring the thermal power of nuclear reactors with a prototype cubic meter antineutrino detector*, Journal of Applied Physics, UCRL-JRNL-233165.

Arkani, M.; Gharib, M. (2009), *Reactor core power measurement using Cherenkov radiation and its application in Tehran Research Reactor*, Annals of Nuclear Energy, Vol. 36 , 896–900.

Khalafi, H.; Mosavi, S.H.; Mirvakili, S.M. (2011), *Design and construction of a digital real time reactivity meter for Tehran research reactor*, Progress in Nuclear Energy, Vol. 53 , 100-105.

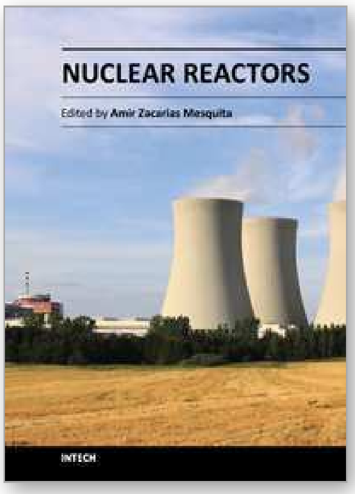
Khalafi, H.; Gharib, M. (1999). *Calculational tools to conduct experimental optimization in Tehran Research Reactor*, Annals of Nuclear Energy, Vol. 26 , 1601–1610.

Hashemi-Tilehnoee, M. ; Hadad, K. (2009). *Optimizing the performance of a neutron detector in the power monitoring channel of Tehran Research Reactor (TRR)*, Nuclear Engineering and Design, Vol. 239, 1260–1266.

Zacarias Mesquita, A.; Cesar Rezende, H. (2010). *Thermal methods for on-line power monitoring of the IPR-R1 TRIGA Reactor*, Progress in Nuclear Energy, Vol. 52, 268–272.

Zacarias Mesquita, A.; Cesar Rezende, H. (2007). *Power measure channels of the IPR-R1 TRIGA research nuclear reactor by thermal methods*, 19th International Congress of Mechanical Engineering, Proceedings of COBEM 2007, Brazil.

Javidkia, F.; Hashemi-Tilehnoee, M.; Zabihi ,V. (2011) *A Comparison between Fossil and Nuclear Power Plants Pollutions and Their Environmental Effects*, Journal of Energy and Power Engineering, Vol. 5, 811-820.



Nuclear Reactors
Edited by Prof. Amir Mesquita

ISBN 978-953-51-0018-8
Hard cover, 338 pages
Publisher InTech
Published online 10, February, 2012
Published in print edition February, 2012

This book presents a comprehensive review of studies in nuclear reactors technology from authors across the globe. Topics discussed in this compilation include: thermal hydraulic investigation of TRIGA type research reactor, materials testing reactor and high temperature gas-cooled reactor; the use of radiogenic lead recovered from ores as a coolant for fast reactors; decay heat in reactors and spent-fuel pools; present status of two-phase flow studies in reactor components; thermal aspects of conventional and alternative fuels in supercritical water?cooled reactor; two-phase flow coolant behavior in boiling water reactors under earthquake condition; simulation of nuclear reactors core; fuel life control in light-water reactors; methods for monitoring and controlling power in nuclear reactors; structural materials modeling for the next generation of nuclear reactors; application of the results of finite group theory in reactor physics; and the usability of vermiculite as a shield for nuclear reactor.

InTech Europe
University Campus STeP Ri
Slavka Krautzeka 83/A
51000 Rijeka, Croatia
Phone: +385 (51) 770 447
Fax: +385 (51) 686 166
www.intechopen.com

InTech China
Unit 405, Office Block, Hotel Equatorial Shanghai
No.65, Yan An Road (West), Shanghai, 200040, China
中国上海市延安西路65号上海国际贵都大饭店办公楼405单元
Phone: +86-21-62489820
Fax: +86-21-62489821