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Excore Neutron Flux Monitoring Method for an Accelerator Driven Sub-critical System

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Abstract: In an accelerator driven sub-critical (ADS) system, power control in sub-critical reactor is achieved through the control of the beam current. Excore neutron flux monitoring in an ADS system, not only provides indication of reactor power, but also provides important inputs to reactor protection system during startup and power operation, and thus plays a very important role in the control and protection of ADS system. This paper presents the excore neutron flux monitoring method which uses three fission chambers (FCs) and three uncompensated ion chambers (UICs). With three operation modes, pulse mode, current mode, and mean square voltage mode, an FC can monitor reactor power over a wide range from the source range to the intermediate and power ranges. The proposed monitoring method increases the redundancy of independent monitoring channels, improves the reliability of the protection system, and provides more information on axial power distribution. Since these neutron detectors are sensitive to the neutron energy spectrum, we propose an effective calibration method to provide the exact value of neutron flux, *i.e.*, these neutron detectors are calibrated with a standardized neutron source, and then, a correction factor is added in the calibration by comparing the neutron energy spectrum of the neutron source with that in ADS system. Based on Geant4 simulation, the correction factors of 5 and 42 are extracted for FCs and UICs, respectively.

Key words: accelerator driven sub-critical system; fission chamber; uncompensated ion chamber
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1 Introduction

In an Accelerator Driven Sub-critical (ADS) system, a heavy metal spallation target located at the centre of a sub-critical core is bombarded by high-energy protons from an accelerator. The spallation neutrons from the target are used as an intense external neutron source to drive the sub-critical reactor. The power control of the sub-critical reactor is achieved through the control of the beam current from the proton accelerator. Excore neutron flux monitoring system which provides indication of reactor power provides important inputs to both accelerator's control system and reactor's control system during startup and power operation. On the other hand, excore neutron flux monitoring system provides power level signals and the rate of change of power signals to the reactor protection system, and provides information on axial power distribution to the control room. Therefore, excore neutron flux monitoring plays a very important role in the control and protection of ADS system.

In a commercial pressurized water reactor (PWR) at several hundred MW power, excore neutron monitoring system detects the leakage neutrons from the reactor core from a completely shutdown condition up to 200% of full power. Three ranges (the source, intermediate and power ranges) of instrumentation are used to monitor reactor power over a wide range of 12 decades. The source range instrumentation which covers the lowest 6 decades of power from 10^{-9} to 10^{-3} percent of the power consists of two independent channels, each channel utilizing a proportional counter.

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The intermediate range instrumentation which covers about 8-decades of power from 10^{-6} to 20 percent of the power, consists of two independent, compensated ion chambers. The power range instrumentation which covers the top 2 or 3-decades of the power from 1 to 120 percent of the power, consists of four independent, uncompensated ion chambers. In this paper we discuss excore neutron flux monitoring method for ADS system. In order to decrease the number of detectors, the proportional counters for the source range and the compensated ion chambers for the intermediate rang are replaced by three fission chambers. With three operation modes, pulse mode, current mode, and mean square voltage mode, a fission chamber can monitor reactor power over a wide range of over 1 decades from the source range to the intermediate and power ranges.

2 Excore neutron monitoring method

To investigate the monitoring method of excore neutrons, we consider the China Initiative Accelerator Driven Sub-critical (CIADS) system which should be able to demonstrate the ADS concept at 10 MW power level with a maximum incore neutron flux of 2×10^{14} n/cm^2/s. Extension of the monitoring method to ADS system with arbitrary power level is straightforward. Fig. 1 shows the excore neutron monitoring system for CIADS system. At 10 MW power level, the leakage neutron flux from the core covers a wide range of about 10 decades from shutdown to full power. Three fission chambers (FCs) are used as the source, intermediate and power range instrumentation and three uncompensated ion chambers (UIC) are used as the power range instrumentation to monitor reactor power.

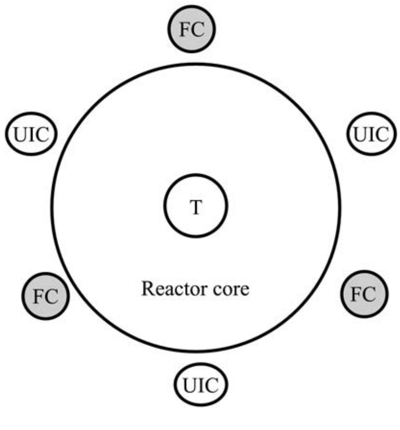

Fig. 1 Horizontal cross section of excore neutron detectors for ADS system
T: target, UIC: uncompensated ionization chambers, FC: fission chamber.

A fission chamber is a gas filled ionization chamber to which a fissile coating is added. When neutrons enter an FC, the neutron-induced fission generates two heavily charged ions. The two fission products are emitted in two opposite directions. The one fission product emitted out of the deposit generates a signal by ionizing the internal filling gas. Gamma radiation also produces ionization of the internal gas in the detector. But the signal generated from the fission is significantly greater than the signal from a gamma photon and the signal is also greater than the signal from a detector using boron as the neutron-sensitive element. Thus a fission chamber, in conjunction with good system design practices, is inherently better at rejecting gamma signals than other common neutron detectors. This rejection is essential for systems that are expected to measure a neutron flux accurately in the presence of a high gamma radiation.

Since an FC can be used as three ranges (source, intermediate and power ranges) of instrumentation, there are three advantages in the excore neutron monitoring method for CIADS system. The first advantage is that three independent channels are used for the source and intermediate range, in comparison with two independent channels in PWR. As shown in Fig. 1, three FCs are located 120 degrees apart outside the core. This location with three FCs provides more information in azimuthal power distribution during a reactor startup than the location with two detectors in PWRs. The second advantage is that the power range instrumentation consists of six independent channels, in comparison with four independent channels in PWRs. Since the excore neutron monitoring system is used for the reactor protection system (RPS), this method increases the redundancy of each protective function and thus increases the reliability of RPS. The third advantage is to provide more information on axial power distribution during power operations. To explain the third advantage, we show a vertical cutaway view of excore neutron detectors in Fig. 2. Each set of UIC consists of an upper detector and a lower detector, mounted inside the same instrument well. The outputs of both upper and lower detec-

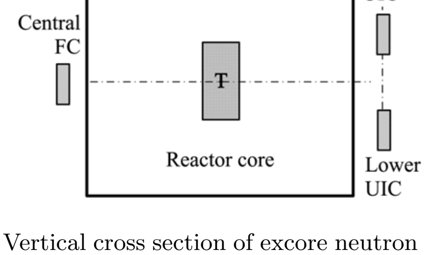

Fig. 2 Vertical cross section of excore neutron detectors for ADS system
T: target, UIC: uncompensated ionization chambers, FC: fission chamber.

tors are combined to produce a channel total power signal. On the other hand, each FC is located vertically at the central position. The nine detector outputs (three upper UICs, three lower UICs and three central FCs) are compared to each other to provide axial and azimuthal power distribution during power operations to the reactor operator.

3 Electronics of fission chamber

To explain why an FC can monitor neutron flux in a very wide range, in this section we discuss the electronics of FCs in three modes. The neutron-induced fission in an FC results in two highly charged fission fragments that create secondary ionization in the argonnitrogen fill gas. The output of the FC is fed to a pre-amplifier located in the containment. The preamplifier increases the signal to noise ratio of the detector outputs and transmits the signal through organic high-immunity cables to the electronic system in the RPS cabinets located in the control room.

Fig. 3 shows the main components of electronics for an FC. An FC used in excore neutron measurement can cover over 10 decades of power. For example, the CFUL08, a commercial excore FC from PHOTONIS company in France, covers 10 decades of power when working in three modes. This FC has a cylindrical geometry with a diameter of 48 cm and a sensitive length of 21.1 cm. A fissionable material, uranium with 9%-enriched ^{235}U, is used for the coating onto the outer surface of the anode, which makes the FC sensitive to neutrons. The inter-electrode gap is filled with argon +4% nitrogen at 250 kPa. The neutron flux ranges are $1\sim10^6$ n/cm^2/s in the pulse mode, $8\times10^4\sim2\times10^9$ n/cm^2/s in Campbelling mode and $10^6\sim10^{10}$ n/cm^2/s in the current mode.

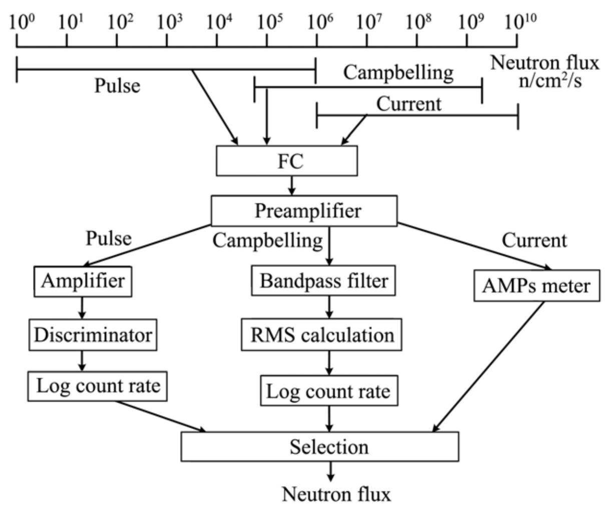

Fig. 3 Block diagram of electronics for fission chamber

3.1 Pulse mode and current mode

When an FC works in pulse mode, its electronics are similar to that of a boron-lined proportional counter or a proportional counter with boron trifluoride which is used as the source range instrumentation. As shown in Fig. 3, the preamplifier amplifies the neutron pulses from the FC and then output the signal to the amplifier and discriminator located in the in the RPS cabinets. The discriminator cuts off the low amplitude gamma-induced pulses and noise from the input signal. The log count rate amplifier converts the neutron pulses into a logarithmic signal. The output of the log count rate amplifier is routed to a local meter at the RPS cabinets.

When an FC works in current mode, its electronics are similar to that of an uncompensated ion chamber which is used as the power range instrumentation. An FC is sensitive to both gamma and neutrons; however, in the power range of operation the neutron flux level is many times greater than the gamma flux. Therefore, no gamma compensation is required in the power range. As shown in Fig. 3, the individual pulses from the preamplifier of an FC combine to give an average DC current that is linearly proportional to the detection rate and can be used to give an indication of the neutron flux.

3.2 Campbelling mode

The Cambelling mode (also called mean square voltage mode) is of great interest as it drastically diminishes the disturbance of gamma rays on neutron signal[1−3]. Based on Ref. [3], the general form of the second order Campbelling equation for a signal $\eta(t)$ consisting of general pulses $\psi(x,t)$ with a randomly distributed amplitude x is given by Eq. (1),

$$D^2(\eta)=s_0x_0^2\int_{-\infty}^{\infty}f^2(t)\mathrm{d}t \tag{1}$$

where D^2 is the mean square value of detector signal $\eta(t)$, s is the count rate (*i.e.* the mean neutron detection rate), x^2 is the mean square of amplitudes x which is proportional to the mean square charge per detection, and $f(t)$ is the pulse shape with $\psi(x,t)=x\cdot f(t)$. This means that the mean of the square of the current is proportional to the neutron flux. A significant advantage of the Campbell measuring mode over current mode is that it has an inherent ability to reject small pulses, such as those induced by gamma photons. Typically, the gamma contribution to the mean square of amplitudes is two decades less than the neutron contribution. In a real application the pulse shape $f(t)$ and the amplitude distribution x can be measured and considered as known information. The variance D^2 of the

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signal is determined from measurement, and then the count rate s of the signal can be estimated.

As shown in Fig. 3, the Campbelling circuitry consists of a bandpass filter, a calculator of root-mean-squared (rms) voltage, and a logarithmic count rate amplifier. The bandpass filter is used to set the initial operating point of the Campbelling circuit. This is accomplished by setting the frequency of the ac voltage that will be passed to the remainder of the circuitry. From the bandpass filter, the signal is routed to the rms calculator where the signal becomes proportional to power. In accordance with Campbell's Theorem, the root mean square (rms) value of the preamplifiers's output voltage is proportional to the average pulse rate from the detector. The logarithmic amplifier is used to convert the output of the rms calculator to a logarithmic signal. A logarithmic signal is required to allow accurate resolution of the wide range of neutron flux indication. The Campbelling circuitry makes use of a condition called pulse pileup, *i.e.* the pulses are occurring at such a rate that they are piling on top of one another. This means that the neutron flux in the Campbelling mode should be more than a minimum value, such as 8×10^4 n/cm^2/s for the example FC CFUL08.

4 Calibration of neutron flux

The excore neutron flux monitoring system consists of several ion chambers to which a neutron-sensitive material is added. For example, an FC is coated with a fissionable material and an uncompensated ion chamber is coated with boron enriched with ^{10}B. These neutron-sensitive materials, either ^{10}B or a fissionable material, are sensitive to the neutron energy spectrum, *i.e.* the rates of neutron-induced reactions vary with neutron energy. When using these neutron detectors in ADS system, an absolute calibration is required to provide the exact value of neutron flux.

Calibration of these neutron detectors consists in establishing the relation between the measured indication and the neutron flux. The absolute calibration can be performed in a neutron environment where the neutron energy spectrum is similar to that in ADS system. However, it is difficult to find such a neutron environment. Thus, we suggest that these neutron detectors can be calibrated with a standardized ^{252}Cf fission source. Then, a correction factor is added in the calibration by comparing the neutron energy spectrum of ^{252}Cf fission source with that in ADS system. In this section, we will evaluate the detection efficiency of neutron detectors for the different neutron energy spectrums. Then we discuss how to extract the correction factor of the calibration

4.1 Geant4 based MC simulation

The detection efficiency of neutron detectors for the different neutron energy spectrums have been evaluated by performing the Monte Carlo (MC) simulation with the Geant4(GEometry ANd Tracking) toolkit[4, 5]. The Monte Carlo simulation based on the Geant4 toolkit has also been performed for neutron detectors (*e.g.* Ref. [6]). The Geant4 toolkit has been employed widely in basic physics research to simulate the propagation of particles and nuclei in extended media. The Geant4 toolkit provides a complete range of functionalities for simulating the passage of particles through matter: geometry, tracking, hits and physics models (the so-called physics lists). To simulate neutron-induced reactions with Geant4, we have used neutron high precision (G4HP) model with G4NDL based on ENDF-VII data. Neutron High Precision models are based on the ENDF/B-VII data format and deal with the detailed transport of neutrons from thermal energies up to 20 MeV. These models use the G4NDL neutron data library which was derived from the evaluated neutron data libraries Brond, CENDL, EEF, ENDF, FENDL, JEF, JENDL, and MENDL.

Fig. 4 shows the model of the detector in the simulation, where a coating is added on the surface of a cylindrical anode. A neutron with energy E impinges on the coating in the detector with an incident angle θ. Eenergy E is sampled randomly from a known neutron energy spectrum. Two neutron energy spectra are compared, the one from ^{252}Cf fission source and the other one from the French Generation-IV sodium-cooled fast reactor[7]. Since the neutron energy spectrum in ADS system is a fast neutron spectrum, the neutron energy spectrum at the outer part of the lateral neutron shield (OLNS) in French fast reactor is used as the excore neutron energy spectrum in the simulation. In addition, the neutron energy spectrum from the FAst Spectrum Transmutation Experimental Facility (FASTEF) by the MYRRHA-FASTEF team[8] is also used to analyze the detection efficiency of fission

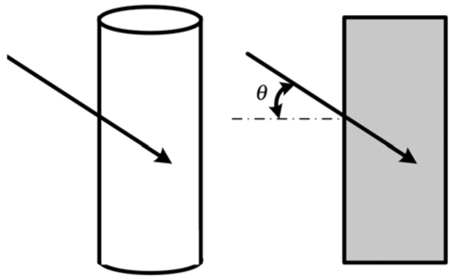

Fig. 4 Schematic view (left) and cutaway view (right) of a cylindrical detector.

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chamber. The FASTEF facility should be able to demonstrate the ADS concept at 10 MW power level. Since the sub-critical cores of both CIADS and FASTEF facilities are cooled with LBE, both facilities are fast neutron spectrum facilities.

When a neutron-induced fission occurs, two heavily charged particles (*i.e.* two fission fragments) are generated. Since the coating is very thin, one of the fission fragments will emit out of the coating to generate a signal by ionizing the internal filling gas. Thus, in the simulation the detector response is calculated by accumulating the number N_f of the neutron-induced fission. The detection efficiency of an FC is extracted by dividing N_f with the total number N of incident neutrons. On the other hand, in the neutron-induced interaction with ^{10}B, two charged particles (^{6}Li and α particles) are generated. Since the ^{10}B coating of an ionization chamber is very thin, one of the charged particles will emit out of the coating to generate a signal. The detection efficiency of an UIC is calculated by dividing N_{Li} with the total number N of incident neutrons, where N_{Li} is the number N_f of ^{6}Li particles. Normally, the numbers of ^{6}Li and α particles generated by the neutron-induced interaction with ^{10}B should be the same.

4.2 Detection efficiency of fission chamber

To evaluate the detection efficiency of an FC, we have taken the CFUL08 from PHOTONIS company as a sample detector. A fissionable material, uranium with 9%-enriched ^{235}U, is used for the coating onto the outer surface of the anode. The FC has a cylindrical geometry with a diameter of 48 cm and a sensitive length of 21 cm. Neutron impinges on the lateral surface of the cylinder with an incident angle θ. In the simulation, the detection efficiency, also known as sensitivity or response of a detector, is defined as the probability for a neutron to produce a charged particle coming out of the converter layer.

Fig. 5 shows the detection efficiency of FC with $\theta \leqslant 45°$ as a function of the thickness of a fissile coating, where the detection efficiencies to the neutrons from both ^{252}Cf fission source and the fast reactors are compared. The solid line in Fig. 5 represents the efficiencies to the fast neutrons at the outer part of the lateral neutron shield (OLNS) in French fast reactor, while the dotted line represents the efficiencies to the fission neutrons in Belgium's ADS facility. It is shown clearly that the detection efficiency of FC varies with the neutron energy spectrum. The detection efficiency to the neutrons from the French fast reactor is 5 folds higher than that the neutrons from the ^{252}Cf source If a current of 1 pA is measured by an FC for ^{252}Cf fission source with a flux of 10^7n/cm^2/s , this means that a current of 5 pA will be measured by the same FC forthe neutrons from the fast reactor with the same flux. Thus, a correction factor in the calibration is required to provide the exact value of neutron flux.

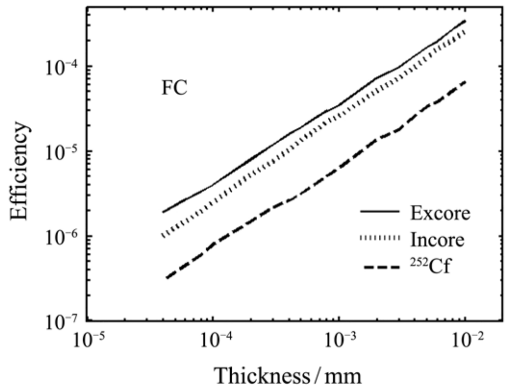

Fig. 5 The detection efficiency of FC as a function of the thickness of a fissile coating with $\theta \leqslant 45°$. Legend "excore" represents the fast neutrons at the outer part of the lateral neutron shield in French fast reactor, while legend "incore" represents the fission neutrons in Belgium's ADS facility.

4.3 Detection efficiency of uncompensated ionization chamber

To evaluate the detection efficiency of uncompensated ionization chamber (UIC) for neutron detection, we have taken the boron-lined ion chamber from LND company in Oceanside, New York, USA as a sample detector. Boron with 96%-enriched ^{10}B is used for the coating onto the outer surface of the anode. The coated thickness is 1 mg/cm^2 and the coated area is 219 cm^2. Fig. 6 shows the detection efficiency of UIC with $\theta \leqslant 45°$ as a function of the coated thickness, where the detection efficiencies to the neutrons from

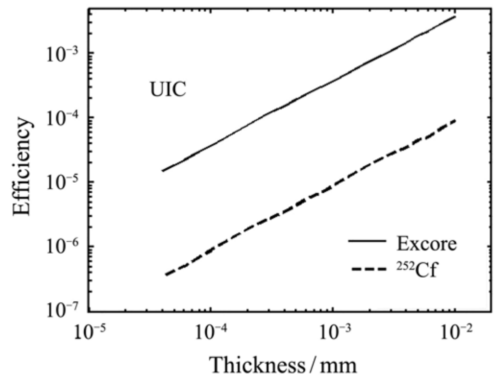

Fig. 6 The detection efficiency of UIC as a function of the thickness of a fissile coating with $\theta \leqslant 45°$ Legend "excore" represents the fast neutrons at the outer part of the lateral neutron shield in French fast reactor.

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both ^{252}Cf fission source and the French fast reactor are compared. It is shown that the detection efficiency to the neutrons from the fast reactor is 42 folds higher than that the neutrons from the ^{252}Cf source. For example, assume that a current of 1 pA is measured when the neutrons from the ^{252}Cf fission source with a flux of 10^7 n/cm^2/s impinge on the coating of an UIC. The results in Fig. 6 indicate that a current of 42 pA will be measured when the neutrons from the fast reactor with the same flux impinge on the coating.

4.4 Correction factor of calibration

Based on the results with Geant4 simulations, Table 1 lists the correction factors F with different incident angles which are calculated as follows

$$F = \frac{1}{N}\sum_{i=1}^{N}\frac{p_i}{q_i} \ , \tag{2}$$

where N is the number of simulated thicknesses and N =18 in our simulation. p_i and q_i are the detection efficiencies with the $i-$th coating thickness to the neutrons from the French fast reactor and ^{252}Cf source, respectively.

Table 1 Correction factor of calibration.

Detector	$\theta \leqslant 60°$	$\theta \leqslant 45°$	$\theta = 0°$	Calculation
FC	5.6	5.5	5.2	3.83
UIC	40.0	41.9	46.3	49.9

The correction factor of calibration can also be extracted from the reaction cross sections[9]. Letting n_i be the number of nuclei of the isotope i in the coating and σ_i be the reaction cross section of the isotope i, the correction factor F is given by

$$F = \int_0^{\infty}\sum_i n_i\sigma_i\phi_1(E)\mathrm{d}E \Big/ \int_0^{\infty}\sum_i n_i\sigma_i\phi_2(E)\mathrm{d}E \ , \tag{3}$$

where $\phi_1(E)$ and $\phi_2(E)$ are the neutron flux in each energy bin for neutron energy spectrums 1 and 2. In this paper, neutron energy spectrums 1 and 2 are the neutron energy spectrums from ^{252}Cf fission source and the French fast reactor, respectively. Table 1 lists the correction factors F extracted from the reaction cross sections. The correction factors with several angles, $\theta \leqslant 6°$, $\theta \leqslant 45°$ and $\theta = 0°$ are close to the calculated value from the reaction cross sections. Based on Geant4 simulations with $\theta \leqslant 45°$, the correction factors of 5 and 42 are extracted for FCs and UICs, respectively.

It is interesting to validate the extraction method of correction factors by comparing calculated values with experimentally measured values. Since the cross sections of neutron-induced reactions with either ^{235}U or ^{10}B vary dramatically with neutron energy, one can use different neutron sources with different energy spectra to test a given detector, such as a compensated ion chambers or an FC. For example, a bare and D^2 O-moderated ^{252}Cf source can be used as neutron sources with different energy spectra[10]. A calibrated bonner sphere spectrometer which consists of a set of polyethylene spheres and a ^{3}He ionization chamber can be used to characterize D^2 O-moderated ^{252}Cf source[10]. Then, a correction factor of calibration can be extracted experimentally by comparing the output currents of the detectors with two sources, *i.e.* a bare and a D^2 O-moderated ^{252}Cf source.

Finally, we summarize the calibration method as follows. First each neutron detector is calibrated with an intense calibration neutron source, such as a standardized ^{252}Cf fission source. Then, a correction factor F of calibration is extracted by comparing the neutron energy spectrum of neutron source with that in ADS system. Without any built ADS facility in the world, the neutron energy spectrum in an ADS system has to be estimated with simulation method. After building new ADS facility, the neutron energy spectrum can be measured with detectors. The final neutron flux in ADS system should be divided by F folds from the measured indication.

5 Conclusion

This paper has discussed the excore neutron flux monitoring method which uses three FCs and three UICs. With three operation modes, pulse mode, current mode, and mean square voltage mode, an FC can monitor reactor power over a wide range from the source range to the intermediate and power ranges. Thus, three, three, and six neutron detectors are used as the source, intermediate and power range instrumentation. The proposed monitoring method increases the redundancy of independent monitoring channels, improves the reliability of the protection system, and provides more information on axial power distribution. Since these neutron detectors are sensitive to the neutron energy spectrum, we have proposed an effective calibration method to provide the exact value of neutron flux, *i.e.*, these neutron detectors are calibrated with a standardized neutron source, and then, a correction factor is added in the calibration by comparing the neutron energy spectrum of the neutron source with that in ADS system. Based on Geant4 simulation, the correction factors of 5 and 42 are extracted for FCs and UICs, respectively.

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加速器驱动次临界系统中堆外中子注量率监测方法

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摘要： 在加速器驱动的次临界(ADS)系统中，次临界反应堆的功率控制是通过控制束流强度来实现。监测堆外中子注量率，不仅提供了反应堆功率指示，也为反应堆保护系统在启动和运行阶段提供了重要的监测信息，因此，堆外中子注量率的监测在ADS系统的控制与保护中起着非常重要的作用。采用3套裂变室和3套非补偿电离室来监测ADS堆外中子注量率。由于裂变室有脉冲、电流和均方电压3种操作模式，1套裂变室可以监测源量程、中间量程和功率量程等宽范围的反应堆功率。所以，使用的监测方法有3个优点，即：增加了监测通道的冗余度，提高了保护系统的可靠性，以及能提供更多的轴向功率分布信息。由于这些中子探测器对中子能谱很敏感，提出了一种有效的校准方法，即先用一个标准的中子源校准这些中子探测器，然后再将中子注量率除以一个修正因素。基于Geant4仿真结果显示，所提取的裂变室和非补偿电离室的修正因素分别为5和42。

关键词： 加速器驱动次临界系统；裂变室；非补偿电离室

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