COMPILATION OF LABORATORY PROTOCOLS FOR IAEA COMPENDIUM

Compendium on Research Reactors Utilization for Higher Education Programmes

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Content

1. MEASUREMENT OF THE THERMAL NEUTRON FLUX DENSITY IN THE REACTOR CORE 5
1.1. INTRODUCTION 5
1.1.1. Theoretical background 5
1.1.2. Experimental theory 6
1.2. PURPOSE 8
1.3. SAFETY MEASURES 8
1.4. INSTRUMENTS AND MATERIALS 9
1.5. EXPERIMENTAL PROCEDURES 9
1.6. MAIN PARAMETER(S) MEASURED 11
1.7. TYPICAL REAL DATA COLLECTED 11
1.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS 11
1.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS 12
1.10. RESULTS 12
1.11. CONCLUSIONS 13

2. MEASUREMENT OF THE FAST NEUTRON FLUX DENSITY IN THE REACTOR CORE CENTRE 14
2.1. INTRODUCTION 14
2.1.1. Theoretical background 14
2.2. PURPOSE 15
2.3. SAFETY MEASURES 15
2.4. INSTRUMENTS AND MATERIALS 15
2.5. EXPERIMENTAL PROCEDURES 15
2.6. MAIN PARAMETER(S) MEASURED 16
2.7. TYPICAL REAL DATA COLLECTED 16
2.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS 16
2.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS 17
2.10. RESULTS 17
2.11. CONCLUSIONS 18

3. DETERMINATION OF THE MATERIAL AND VOID FEEDBACK ON REACTIVITY 20
3.1. INTRODUCTION 20
3.2. PURPOSE 20
3.3. SAFETY MEASURES 20
3.4. INSTRUMENTS AND MATERIALS 20
3.5. EXPERIMENTAL PROCEDURES 20
3.6. MAIN PARAMETER(S) MEASURED 21
3.7. TYPICAL REAL DATA COLLECTED 21
3.8. PRE-KNOWLEDGE REQUIRED FROM STUDENTS 22
3.9. RESULTS 22
3.10. CONCLUSIONS 24

4. DETERMINATION OF THE REACTIVITY AND THE REACTOR PERIOD 25
4.1. INTRODUCTION 25
4.2. PURPOSE 27
4.3. SAFETY MEASURES 27
4.4. INSTRUMENTS AND MATERIALS 27
4.5. EXPERIMENTAL PROCEDURES 27
4.6. MAIN PARAMETER(S) MEASURED 28
4.7. TYPICAL REAL DATA COLLECTED 28
4.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS 28

4.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS **29**
4.10. RESULTS **29**
4.11. CONCLUSIONS **30**

5. MEASUREMENT OF THE BACKGROUND RADIATION AROUND THE OPERATING REACTOR **31**
5.1. INTRODUCTION **31**
5.2. PURPOSE **31**
5.3. SAFETY MEASURES **31**
5.4. INSTRUMENTS AND MATERIALS **31**
5.5. EXPERIMENTAL PROCEDURES **31**
5.6. MAIN PARAMETER(S) MEASURED **33**
5.7. TYPICAL REAL DATA COLLECTED **33**
5.8. PRE-KNOWLEDGE REQUIRED FROM STUDENTS **33**
5.9. RESULTS **33**
5.10. CONCLUSIONS **33**

6. CRITICAL EXPERIMENT **34**
6.1. INTRODUCTION **34**
6.1.1. Theoretical background 34
6.2. PURPOSE **35**
6.3. SAFETY MEASURES **35**
6.4. INSTRUMENTS AND MATERIALS **35**
6.5. EXPERIMENTAL PROCEDURES **36**
6.6. MAIN PARAMETER(S) MEASURED **36**
6.7. TYPICAL REAL DATA COLLECTED **36**
6.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS **37**
6.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS **37**
6.10. RESULTS **37**
6.11. CONCLUSIONS **38**

7. CONTROL ROD CALIBRATION AND DETERMINATION OF EXCESS REACTIVITY IN THE CORE **40**
7.1. INTRODUCTION **40**
7.1.1. Theoretical background 40
7.2. PURPOSE **42**
7.3. SAFETY MEASURES **42**
7.4. INSTRUMENTS AND MATERIALS **43**
7.5. EXPERIMENTAL PROCEDURES **43**
7.6. MAIN PARAMETER(S) MEASURED **43**
7.7. TYPICAL REAL DATA COLLECTED **44**
7.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS **44**
7.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS **44**
7.10. RESULTS **45**
7.11. CONCLUSIONS **46**

8. REACTIVITY VALUES OF FUEL ELEMENTS IN VARIOUS CORE POSITIONS **48**
8.1. INTRODUCTION **48**
8.1.1. Theoretical background 48
8.2. PURPOSE **48**
8.3. SAFETY MEASURES **48**
8.4. INSTRUMENTS AND MATERIALS **48**
8.5. EXPERIMENTAL PROCEDURES **49**
8.6. MAIN PARAMETER(S) MEASURED **49**
8.7. TYPICAL REAL DATA COLLECTED **49**
8.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS **50**

8.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS **50**
8.10. RESULTS **50**
8.11. CONCLUSIONS **51**

9. REACTOR POWER CALIBRATION AND TEMPERATURE COEFFICIENT OF THE REACTIVITY **52**
9.1. INTRODUCTION **52**
9.1.1. Theoretical background 52
9.2. PURPOSE **53**
9.3. SAFETY MEASURES **53**
9.4. INSTRUMENTS AND MATERIALS **53**
9.5. EXPERIMENTAL PROCEDURES **53**
9.6. MAIN PARAMETER(S) MEASURED **54**
9.7. TYPICAL REAL DATA COLLECTED **54**
9.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS **55**
9.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS **55**
9.10. RESULTS **55**
9.11. CONCLUSIONS **56**

10. NEUTRON FLUX DENSITY MEASUREMENT USING COMPENSATED IONIZATION CHAMBERS **57**
10.1. INTRODUCTION **57**
10.2. PURPOSE **60**
10.3. SAFETY MEASURES **60**
10.4. INSTRUMENTS AND MATERIALS **60**
10.5. EXPERIMENTAL PROCEDURES **61**
10.6. MAIN PARAMETER(S) MEASURED **61**
10.7. TYPICAL REAL DATA COLLECTED **61**
10.8. PRE-KNOWLEDGE REQUIRED FROM STUDENTS **62**
10.9. RESULTS **62**
10.10. CONCLUSIONS **63**

11. NEUTRON FLUX DENSITY MEASUREMENT WITH SELF-POWERED NEUTRON DETECTORS **64**
11.1. INTRODUCTION **64**
11.2. PURPOSE **67**
11.3. SAFETY MEASURES **67**
11.4. INSTRUMENTS AND MATERIALS **67**
11.5. EXPERIMENTAL PROCEDURES **67**
11.6. MAIN PARAMETER(S) MEASURED **67**
11.7. TYPICAL REAL DATA COLLECTED **68**
11.8. PRE-KNOWLEDGE REQUIRED FROM STUDENTS **68**
11.9. RESULTS **68**
11.10. CONCLUSIONS **68**

12. DEMONSTRATION OF A REACTOR PULSE WITH DIFFERENT REACTIVITY INSERTION **69**
12.1. INTRODUCTION **69**
12.2. PURPOSE **69**
12.3. SAFETY MEASURES **70**
12.4. INSTRUMENTS AND MATERIALS **70**
12.5. EXPERIMENTAL PROCEDURE **70**
12.6. MAIN PARAMETER MEASURED **71**
12.7. TYPICAL DATA COLLECTED **71**
12.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS **71**
12.9. PRE-KNNOWLEDGE REQUIRED FROM STUDENTS **71**

12.10. RESULTS..... 72

1. MEASUREMENT OF THE THERMAL NEUTRON FLUX DENSITY IN THE REACTOR CORE

1.1. INTRODUCTION

1.1.1. Theoretical background

The spatial distribution of the neutron flux density uses the diffusion approximated one group neutron balance equation:

$$-D\Delta\phi(\vec{r},t) + \Sigma_a(\vec{r}) \times \phi(\vec{r},t) - k_\infty \Sigma_a(\vec{r}) \times \phi(\vec{r},t) = -\frac{\partial n(\vec{r},t)}{\partial t} \quad (1.1)$$

Where D is the diffusion coefficient, Σ_a is the macroscopic absorption cross-section and k_∞ is the infinite multiplication factor of the system. The flux density f is taken as zero at the extrapolated boundary of the reactor core because of its boundary conditions. For the stationary case, Eq. (1.1) can be transformed into

$$\Delta\Phi(\vec{r}) + B_m^2 \times \Phi(\vec{r}) = 0 \quad (1.2)$$

$$B_m^2 = \frac{k_\infty - 1}{L^2} \quad (1.3)$$

Where L is the diffusion length, B_g^2 is the geometric buckling and B_m^2 is the material buckling of the reactor system.

The solution of Eq. (1.2) determines the geometrical buckling while the solution of Eq. (1.3) determines the material buckling. For a critical reactor core:

$$B_m^2 = B_g^2 \quad (1.4)$$

The infinite multiplication k_∞ is defined as the ratio of the number of neutrons in generation i to the number of neutrons in the preceding generation i-1. It is given as

$$k_\infty = \eta \cdot \varepsilon \cdot p \cdot f \quad (1.5)$$

Where $\eta = \nu \times \frac{\Sigma_f}{\Sigma_a}$ is the reproduction factor (>1), ε is the fast fission factor (>1), p represents the resonance escape probability (<1) and f is the thermal utilization factor (<1).

For a finite reactor system, the effective multiplication factor k_{eff} becomes

$$k_{eff} = k_\infty \cdot P_T \cdot P_F \quad (1.6)$$

Where P_T represents the non-leakage probability of the thermal neutrons and P_F is the non-leakage probability of the fast neutrons.

The subject probabilities are expressed by

$$P_T = \frac{1}{1 + L^2 B^2} \qquad P_F = e^{-B^2 \tau} \tag{1.7}$$

where τ is the Fermi age for thermal neutrons. For a bare cylindrical core Eq. (1.2) is solved by

$$\Phi(\vec{r}) = R(r) \times Z(z) \tag{1.8}$$

leading to

$$\begin{aligned} Z_m(z) &= A_m \cos\left(\frac{m \cdot z \cdot \pi}{H}\right) \\ R_n(r) &= A_n \times J_0\left(\frac{x_n r}{R}\right) \end{aligned} \tag{1.9}$$

where H represents an extrapolated height of the core, R shows an extrapolated radius of the core, J_0 is the Bessel function of first kind of order zero and x_n is nth zero solution of J_0. The stationary neutron flux density distribution is therefore

$$\Phi(\vec{r}) = \Phi_0 \times J_0\left(\frac{2.405 \cdot r}{R}\right)\cos\left(\frac{\pi \cdot z}{H}\right) \tag{1.10}$$

1.1.2. Experimental theory

The neutron flux density is measured by the gold foil activation method. The activation rate of a thin foil for mono-energetic neutrons is given by

$$C = N \cdot \sigma_a \cdot \phi \cdot d \tag{1.11}$$

Where C is the activation rate, N is the number of nuclei/cm^3, σ_a is the activation cross-section and d is the foil thickness. In the thermal region, most of the materials possess a 1/v activation cross-section. Therefore, the activation is proportional to the neutron density

$$C = \int N \cdot \sigma_a(v) \cdot \phi(v) dv = \int N \cdot \frac{k}{v} n(v) v dv = k \cdot n \tag{1.12}$$

Because of the radioactive decay, some corrections are required to implement:

$$\frac{dB(t)}{dt} = C \cdot F - \lambda \cdot B(t) \tag{1.13}$$

Where $B(t)$ is the number of radioactive nuclei and λ is the decay constant. The second term in Eq. (1.13) describes those nuclei already decaying during activation since the activity is given as $A(t) = \lambda B(t)$. The time dependent activity is obtained by integrating Eq. (1.13):

$$A = A_\infty\left(1 - e^{-\lambda t_1}\right) \qquad A_\infty = C \times F \tag{1.14}$$

Here t_1 represents the activation time. For the time between end of activation and start of the measurement, the radioactive decay needs also to be taken into consideration. Therefore,

$$A = A_\infty\left(1 - e^{-\lambda t_1}\right) \times e^{-\lambda t_2} = A_\infty \times T^{-1} \tag{1.15}$$

Here t_2 is the decay time. In order to perform only thermal activation, a cadmium (Cd) difference measurement is required, i.e. a Cd-covered foil is activated under the same conditions, as an uncovered foil and the difference between their activities are determined. The Cd absorbs all neutrons in the thermal region (i.e. below energy of 0.4 eV). Since this value of the Cd cut-off energy is not perfectly correct (see *FIG. **1.1***), a correction is required by the Cd-correction factor F_{Cd}.

$$C_{th} = C_t - F_{Cd} \cdot C_{Cd} \tag{1.16}$$

C_t represents the activity of the uncovered foil, F_{Cd} is the cadmium-correction factor and C_{Cd} shows the activity of the Cd-covered foil. The activity measurement can be performed using a $2\pi-\beta$ counter.

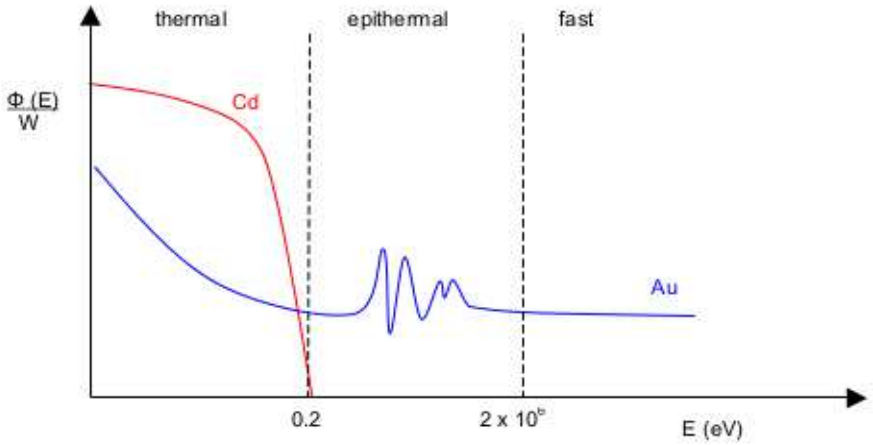


FIG. 1.1. Comparison of Neutron Cross-Section of Absorption of Gold and Cadmium.

Further, two more factors have to be taken into account. These are the self-absorption of the β radiation in the foil and the sensitivity of the counter. The self-absorption can be expressed as

$$S = \frac{\int_0^{\theta} e^{-dx} dx}{\theta} = \frac{1 - e^{\alpha\theta}}{\alpha\theta} \tag{1.17}$$

Where α is foil absorption coefficient for β rays for ^{197}Au and θ is foil thickness.

The sensitivity R of the counter is determined by measuring the same gold foil twice i.e. once deposited on the sample holder of the counter giving R_1 and the other covered on the top with a second sample holder giving R_2.

$$R = \frac{R_2}{R_1} \tag{1.18}$$

Therefore, the activity and activation are related through

$$A = \frac{R \cdot F \cdot S \cdot C}{T} \tag{1.19}$$

Where F is the area of the foil.

If a Maxwell-Boltzmann spectrum of thermal neutrons is assumed, then activation of a foil and the thermal neutron flux density for 1/v absorbers are connected through the following relation:

$$C = \frac{\sqrt{\pi}}{2} \cdot N \cdot \sigma_0 \cdot d \cdot \phi_{th} \sqrt{\frac{T_0}{T_N}}\, G \tag{1.20}$$

Where σ is cross-section at 2200 m/s (T_0 = 293.6 K), T_N is the neutron temperature and G is the neutron self-shielding factor. For thin foils, this approximates to

$$G = \frac{1}{2N \cdot \sigma_0 \cdot d + 1} = \frac{1}{2\mu \cdot \theta + 1} \tag{1.21}$$

and

$$N \cdot \sigma_0 \cdot d \cong \mu \cdot \theta \tag{1.22}$$

Where μ is the neutron mass absorption coefficient. If the Au-foil is very thin then the thermal neutron flux density is calculated as

$$\Phi = \frac{2T}{R \cdot F \cdot S \cdot \mu \cdot \theta \cdot G \cdot \sqrt{\pi}} \sqrt{\frac{T_N}{T_0}} \left(A_t - F_{Cd} \cdot A_{Cd} \right) \tag{1.23}$$

1.2. PURPOSE

The measurement of the thermal neutron flux density distribution in the core is of significant importance. The measured flux density values are applied for calibration of nuclear channels, assessment of absolute power, power distribution in the core, identification of hot spots and calculation of fuel burn up. It is also important to know the thermal flux density at irradiation sites from the reactor user's points of view. Since the flux density distribution is proportional to the power, one can determine the reactor power by measuring the relative flux density distribution in the reactor and the absolute flux density level in a certain reactor location.

The purpose of this experiment is to provide all participants an overview of the neutron flux distribution in the reactor core. The students will become familiar with neutron activation and get experienced in handling irradiated samples and short-lived radioisotopes. They will also learn how to deal with an appropriate counter system to measure the produced activity of radioactive isotopes.

1.3. SAFETY MEASURES

The standards regulations for control of radiation exposure in the laboratory for health and safety should be demonstrated to the experimenters. The general radiation protection safety measures for each student during this experiment are:

- Personal dosimeter for each course participant provided by the health physics group;
- Use of lab coat, gloves and tweezers for handling of irradiated foils;
- The irradiated foils should only be handled after a proper decay time;
- Spend as little time as possible in the control area;
- Avoid standing directly above the reactor core;

- Each course participant should wash his/her hands after the completion of the experiment and should check for possible contamination using a hand-foot monitor.

1.4. INSTRUMENTS AND MATERIALS

This experiment measures the flux density distribution and is performed in the reactor core. It needs the following equipment:

- TRIGA Mark II reactor with an operating power of 10 W for this experiment;
- An appropriate number of gold foils (m≈7.9 mg; r=0.25 cm) and cadmium foils (r=0.35 cm). The weight and size of the foils can be different for different types of the reactors; (Attention, cadmium is toxic and should be handled with care!)
- Aluminium sample holders (l=60 cm; r=0.3 cm) to attach the foils during activation;
- A 2π counter to measure the foil activity;
- A contamination monitor.

1.5. EXPERIMENTAL PROCEDURES

As foil material, Au foils are used with a weight of 40 mg/cm^2 and a diameter of 5 mm. The foils are initially cleaned with IPA and taped on specially designed sample holders, made of aluminium (see *FIG. **1.2***), and then secured by a cover plate clamp with countersunk screw for security. In order to obtain the same reactor power and irradiation time, two identical foils are activated, one foil covered with the Cd and the other Au foil uncovered. The Cd-covered Au foil is fixed at least 15 mm away from the uncovered foil to avoid flux depression. The sample holders, which contain two samples each, are inserted into the experimental holes of the reactor's grid plate. (see *FIG. **1.3***)

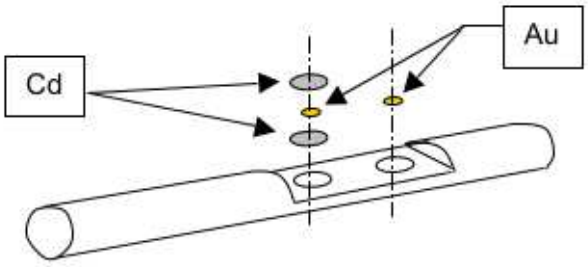


FIG. 1.2. Drawing of the Al sample holder.

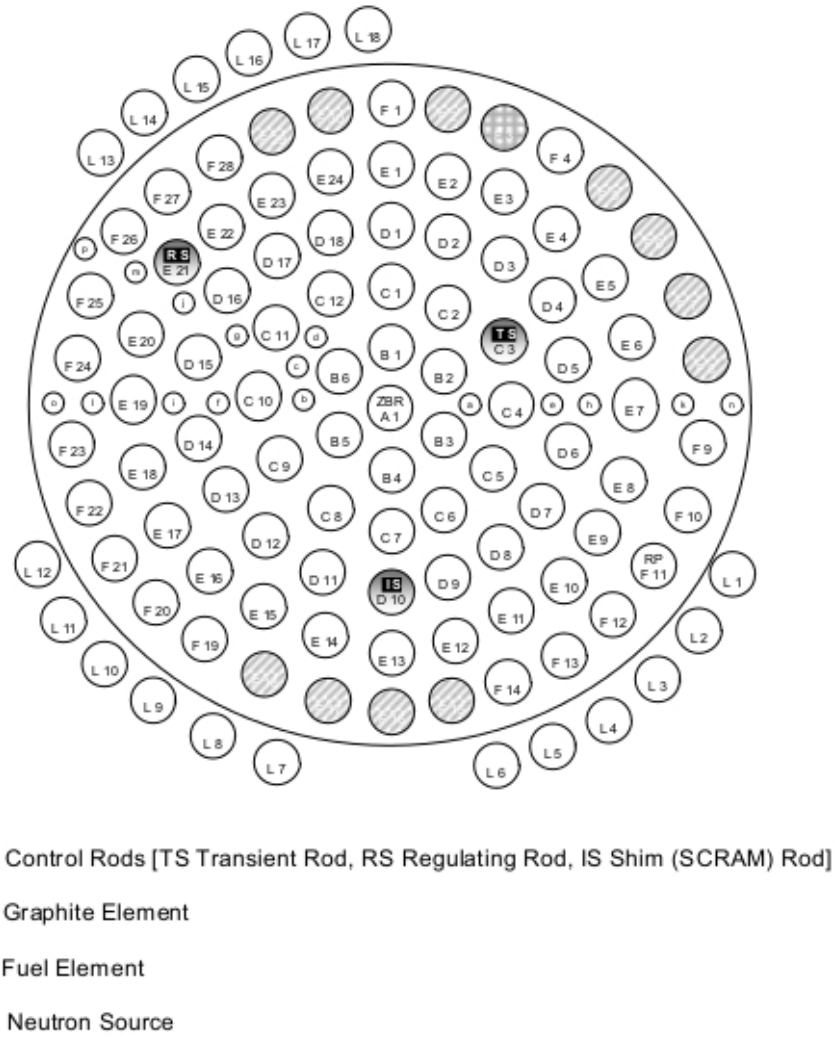


FIG. 1.3. Core map of the TRIGA reactor.

The foil activation is carried out at low reactor power. The reactor is brought to a power of 10 W and the sample holders are lowered by string into the irradiation tubes at five radial positions: 'b', 'f', 'i', 'l' and 'o' (see *FIG.* ***1.3***); each sample is lowered such that it is positioned alongside the core axial centreline, i.e. the region corresponding to the greatest flux. The resultant flux profile could then be determined from the results and plotted at the different radial positions.

After an irradiation time of 10 minutes the sample holders are removed from the reactor core but still stored in the reactor tank for several minutes, because of the activation of the aluminium. (^{28}Al has a half life of $\tau_{1/2}$=2.246 min)

$$^{27}_{13}Al + n \rightarrow ^{28}_{13}Al \xrightarrow{\beta^-} ^{28}_{14}Si \tag{1.24}$$

The sample holders are taken to the laboratory, where the samples are removed from the sample holders and placed, in turn, into the β-detector. The activity of the ^{198}Au ($\tau_{1/2}$=2.69 d)

$$^{197}_{79}Au + n \rightarrow ^{198}_{79}Au \xrightarrow{\beta^-} ^{198}_{80}Hg \tag{1.25}$$

is measured by a 2π counter, the counter plateau is determined first and the background of the counter has also been taken into account.

1.6. MAIN PARAMETER(S) MEASURED

To determine the neutron flux density in the reactor core, the activity of both, the activated Au foil and the activated Au foil covered with Cd, are measured. Since the activated ^{198}Au isotope has a short half life ($\tau_{1/2}$=2.69 d), the time between end of irradiation and begin of measurement has also to be measured.
Therefore the main parameters measured for this experiment are:

- A_t = count rate of the activated Au foil
- A_{Cd} = count rate of the activated Au foil covered with Cd
- t_1 = period of activation
- t_2 = period between end of irradiation and begin of measurement

1.7. TYPICAL REAL DATA COLLECTED

The collected data for this experiment are shown on TABLE **1.1**. Each sample is measured twice with the 2π counter and an average taken (accounting for the difference in time between the two readings). In addition three background counts were also taken with no sample in the detector, and the average value subtracted from the sample reading.

TABLE 1.1. MEASURED DATA

Sample No.	A_t [cps]	A_{Cd} [cps]	t_1 [min]	t_2 [min]
1	314.2	-	10	49
1_{Cd}	-	102.6	10	53
2	235	-	10	56
2_{Cd}	-	100.8	10	59
3	222.8	-	10	62
3_{Cd}	-	87.77	10	64
4	135.3	-	10	66
4_{Cd}	-	52.71	10	69
5	113.4	-	10	72
5_{Cd}	-	43.75	10	74

1.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS

The following formula is used to calculate the thermal neutron flux density in the core:

$$\phi = \frac{2}{R\cdot F\cdot S\cdot \theta\cdot G\cdot \mu\cdot \sqrt{\pi}}\cdot \sqrt{\frac{T_N}{T_0}}\cdot \left(1-e^{-\lambda t_1}\right)^{-1}\cdot \left[A_t e^{\lambda t_2} - F_{Cd}\cdot A_{Cd} e^{\lambda t_2'}\right] \qquad (1.26)$$

with the following parameters:

- R: is the efficiency of the detector (R=0.32 for the 2π counter)
- $F = \pi\cdot r^2$: is the area of the foil (F=0.196 cm^2)
- $\theta = \frac{m}{F}$: θ=40 mg/cm^2 with m~ 7.9 mg
- $S = \frac{1-e^{\alpha\cdot\theta}}{\alpha\cdot\theta}$: is the self absorption factor of the β radiation in the foil (S=0.6968 with α=19.3 cm^{2}g^{-1})

- $G = \frac{1}{2\mu \cdot \theta + 1}$: is the neutron self shielding factor (G=0.9766)
- μ: is the neutron mass absorption coefficient (for Au μ=0.3 cm^2g^{-1})
- T_N/T_0: T_N is the neutron temperature (T_N/T_0= 1.07)
- $\lambda = \frac{\ln(2)}{\tau_{1/2}}$: is the decay constant (λ=1.783 × 10^{-4} min)
- t_1: is the measured time period of activation (t_1=10 min)
- t_2: is the measured time period from the end of irradiation to the start of measurement
- t_2': is the measured time period from the end of irradiation for Cd covered foil to the start of measurement
- A_t: is the measured count rate of the irradiated uncovered Au foil
- A_{Cd}: is the measured count rate of the irradiated Cd covered Au foil
- F_{Cd}: is the cadmium correction factor (F_{Cd}= 1.22)

1.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS

The students should have an understanding in

- nuclear and reactor physics: knowledge of nuclear fission, types of decay, cross section, nuclear reactions particularly neutron activation, multiplication factor, neutron flux density and its behavior.
- radiation physics and protection: the paticipants should have a profound understanding of radioactive dacay, dose rate and limits, activity, and all aspects regarding radiation protection.

1.10. RESULTS

Using Eq. (1.26), with the constant parameters provided in SECTION 1.8 and the measured values listed in TABLE **1.1**, the thermal neutron flux shown on TABLE **1.2** are obtained. *FIG. **1.4*** shows the thermal neutron flux along the distance from the centre of the core.

TABLE 1.2. THERMAL NEUTRON FLUX CALCULATIONS

Sample No.	Core Position*	Thermal Neutron Flux Φ [$cm^{-2}s^{-1}$]	Distance from Center of Core [cm]
1	b	2.44×10^8	5.30
2	f	1.45×10^8	10.60
3	i	1.50×10^8	13.50
4	l	9.19×10^7	18.50
5	o	7.78×10^7	21.30

*core position regarding *FIG. **1.3***

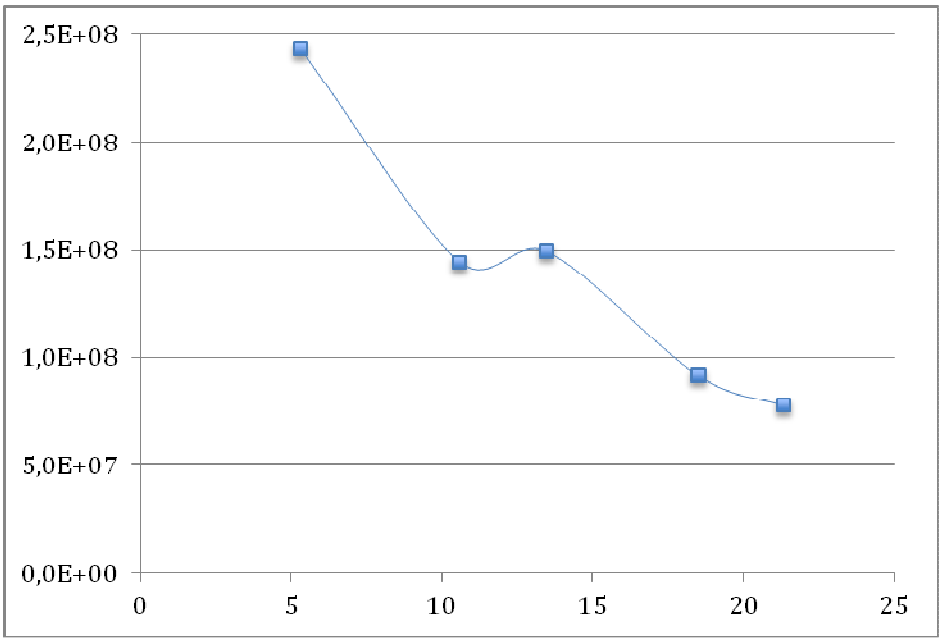


FIG. 1.4. Graph of the radial thermal flux distribution.

1.11. CONCLUSIONS

The thermal neutron flux of the TRIGA reactor in Vienna was measured by activation of Au foils. The diagram on *FIG. **1.4*** shows the radial distribution of the thermal neutron flux in the reactor core. It can be noticed that the highest neutron flux is situated at the core centre and decreases towards the reactor edge, which should roughly follow the shape of a Bessel function. This shape can be easily changed and modified by reflectors, control rods and fuel elements with variable burn up and enrichments.
Due to different distances between the sample holders and the fuel elements in the core, changes in the neutron flux distribution can occur. Since core position 'i' has a shorter distance to its surrounding fuel elements, a neutron flux peak at the core distance of 13.5 cm can be seen.

2. MEASUREMENT OF THE FAST NEUTRON FLUX DENSITY IN THE REACTOR CORE CENTRE

2.1. INTRODUCTION

2.1.1. Theoretical background

Neutrons emitted from the source in a reactor core or neutrons emitted after fission, are of higher energy and called fast neutrons. The strongest source for fast neutrons is the thermal reactor core. The energy distribution in a reactor core is divided in three different energy regions: (fast energy range, the fast flux Φ_f), epithermal energy range (the epithermal flux Φ_{ep}), and thermal energy range (thermal flux Φ_{th}), which describes the neutrons in thermal equilibrium. *FIG. **2.1*** shows the typical energy distribution of fission neutrons.

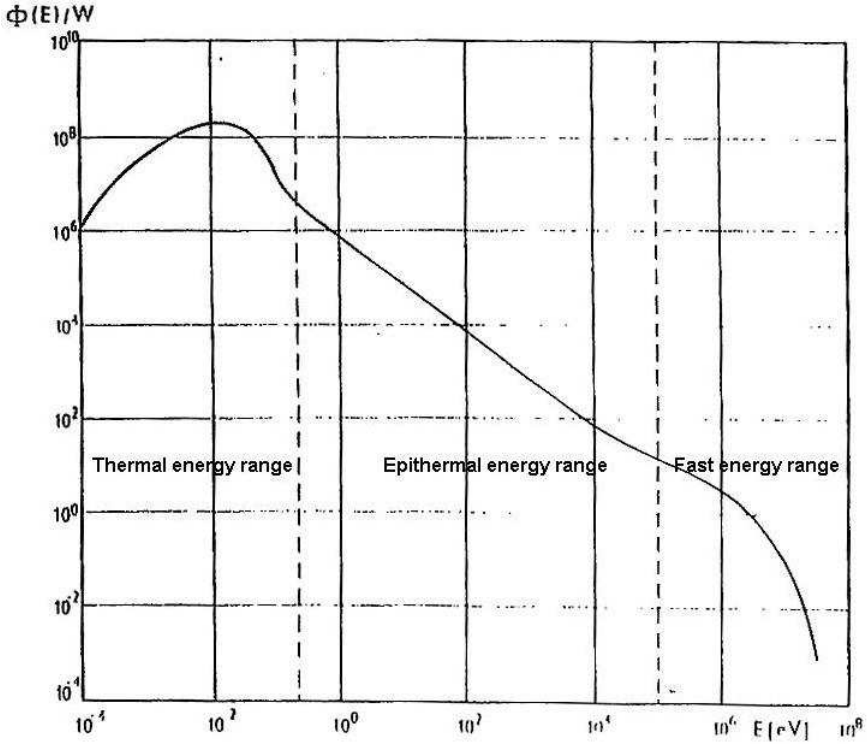


FIG. 2.1. Typical neutron energy spectrum in a reactor core.

$$\Phi(E)dE = \Phi_{th}\frac{E}{(kT)^2}e^{-\frac{E}{kT}}dE + \frac{\Phi_{ep}}{E}\Delta(E)dE + \Phi_f\frac{\sqrt{\pi}}{2}\frac{\sqrt{E}}{T_f}e^{-\frac{E}{T_f}}dE \qquad (2.1)$$

Where Φ_{th}, Φ_{ep} and Φ_f represent the flux density in thermal, epithermal and fast energy range ($E_{th} < 0{,}2$ eV, $0{,}2$ eV$< E_{ep} < 2$ MeV, $E_f > 2$ MeV) respectively. The fast and the thermal region are described through a Maxwell Boltzmann distribution, as the epithermal region is characterized through a 1/E distribution.

To measure the flux density of the fast neutrons, strong nuclear reactions with high threshold energies are used. This means that only neutrons above a certain energy limit contribute to the activation of the foil. Such reactions involve (n, p), (n, α), (n, f) and (n, 2n) reactions. The threshold energy limit depends on the foil materials and should be between 1–20 MeV. To select a certain material, the following characteristics have to be considered.

- The distribution of the cross-section as a function of the energy must be known;
- The reaction should generate radioactive nuclei which can be clearly identified and the activity can be absolutely determined;
- The nuclei should have a half-life longer than some minutes.

It is noticeable that a precise fast flux measurement is much more complicated than a thermal flux, because the cross-sections in that region are much smaller with larger uncertainties.

2.2. PURPOSE

This experiment provides all participants with the theoretical and practical differences between thermal and fast neutron flux measurements in the reactor core. The students will become familiar with neutron activation and gain experience in handling irradiated samples and short-lived radioisotopes. They will also learn how to use an appropriate counter system to measure the produced activity of the radioactive isotopes.

2.3. SAFETY MEASURES

The standards regulations for control of radiation exposure in the laboratory for health and safety should be demonstrated to the experimenters. The general radiation protection safety measures for each student during this experiment are:

- Personal dosimeter for each course participant provided by the health physics group;
- Use of lab coat, gloves and tweezers for handling irradiated foils;
- The irradiated foils should only be handled after a proper decay time;
- Spend as little time as possible in the control area;
- Avoid standing directly above the reactor core;
- Each course participant should wash his/her hands after the completion of the experiment and should check for possible contamination using a hand-foot monitor.

2.4. INSTRUMENTS AND MATERIALS

This experiment measures the fast neutron flux in the reactor core centre and is performed on the reactor platform. It needs the following equipment:

- TRIGA Mark II reactor with an operating power of 1 kW for this experiment;
- An appropriate number of iron foils (r=0.25 cm) and cadmium foils (r=0.35 cm). The weight and size of the foils can be different for different types of the reactors; (Attention, cadmium is toxic and should be handled with care!)
- Plexiglas sample holders and graphite cylinder to position the foils in the core centre;
- A 2π counter to measure the foil activity;
- A contamination monitor.

2.5. EXPERIMENTAL PROCEDURES

Iron as a sensor material is selected for this experiment. Its activity can be measured with a counter system. The effective absorption energy is high enough (6 MeV), and the half life (2.58 h) is ideal for a short irradiation time. After the irradiation of the foils, a decay time

between 30 and 60 minutes is recommended. The foil is placed with a Plexiglas sample holder in a graphite cylinder in the centre of the core.

In the experimental procedure, two iron foils are cleaned with IPA. Those foils are sandwiched between two pieces of cadmium, and installed into the sample holder. The water filled graphite cylinder is lowered into the core axial centreline and irradiated for a period of 10 minutes at a power of 1 kW. After irradiation the sample holder is left to cool for 2–3 hours. Afterwards the samples are extracted from the sample holder and the activity measured in the 2π□β-detector.

2.6. MAIN PARAMETER(S) MEASURED

To determine the fast neutron flux in the reactor core centre, the activity of the irradiated Fe foil is measured. Since the activated ^{56}Fe isotope has a short half life ($\tau_{1/2}$=2.58 h), the time between end of irradiation and begin of measurement has also to be measured. Therefore the main parameters measured for this experiment are:

- A = count rate of the activated Fe foil
- t_1 = period of activation
- t_2 = period between end of irradiation and begin of measurement

2.7. TYPICAL REAL DATA COLLECTED

The collected data for this experiment are shown on TABLE **2.1**. Each sample is measured twice with the 2π counter and an average taken (accounting for the difference in time between the two readings). In addition three background counts are also taken with no sample in the detector, and the average value subtracted from the sample reading.

TABLE 2.1. MEASURED DATA

Sample No.	A [cps]	t_1 [min]	t_2 [min]
1	3927.314	10	92.5
2	3678.572	10	105.0
Background	0.861	-	-

2.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS

In order to calculate the fast neutron flux, the following formula is used:

$$\Phi_{fast} = \frac{A \cdot M}{G \cdot N_A} \sigma^{-1} \left[\left(1 - e^{\frac{-\ln(2) \cdot t_1}{\tau_{1/2}}} \right) \cdot e^{\frac{-\ln(2) \cdot t_2}{\tau_{1/2}}} \right]^{-1} \qquad (2.2)$$

with the following parameters:

- A: is the measured count rate of the irradiated Fe foil
- M: is the atomic weight of Fe (55.845 amu)
- G: is the foil weight (G=3.858 × 10^{-2} g)
- N_A: is the Avogadro Number (N_A=6.02252 × 10^{23} mol^{-1})
- σ_{eff}: is the cross section in mb (1 mb=10^{-27} cm^2)

- t_1: is the measured time period of activation (t_1=10 min)
- t_2: is the measured time period from the end of irradiation to the start of measurement
- $\tau_{1/2}$: is the half life ($\tau_{1/2}$=2.58 h).

The values used for the cross section σ_{eff} are listed on TABLE **2.2**.

TABLE 2.2. CROSS SECTION VALUES FOR ^{56}Fe

σ_{eff} [mb][1]	σ_{eff} [mb][2]	σ_{eff} [mb][3]	σ_{eff} [mb][4]	σ_{eff} [mb][5]
1.07	118	12	15	13.11

2.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS

The students should have an understanding in

- nuclear and reactor physics: knowledge of nuclear fission, types of decay, cross section, nuclear reactions particularly neutron activation, multiplication factor, neutron flux density and its behavior.
- radiation physics and protection: the paticipants should have a profound understanding of radioactive dacay, dose rate and limits, activity, and all aspects regarding radiation protection.

2.10. RESULTS

Using Eq. (2.2) with the measured values in TABLE **2.1**, the fast neutron flux Φ_{fast} in the core centre, listed on TABLE **2.3** are obtained. The fast neutron flux is plotted as a function of various values for cross sections of ^{56}Fe, as shown on *FIG. **2.1***.

TABLE 2.3. FAST NEUTRON FLUX IN THE CORE CENTRE

	σ_{eff} [mb][(a)]	σ_{eff} [mb][(b)]	σ_{eff} [mb][(c)]	σ_{eff} [mb][(d)]	σ_{eff} [mb][(e)]
Φ_{fast} Sample 1 [$cm^{-2}s^{-1}$]	2.08×10^{11}	1.89×10^{9}	1.86×10^{10}	1.49×10^{10}	1.70×10^{10}
Φ_{fast} Sample 2 [$cm^{-2}s^{-1}$]	1.60×10^{11}	1.45×10^{9}	1.43×10^{10}	1.14×10^{10}	1.30×10^{10}

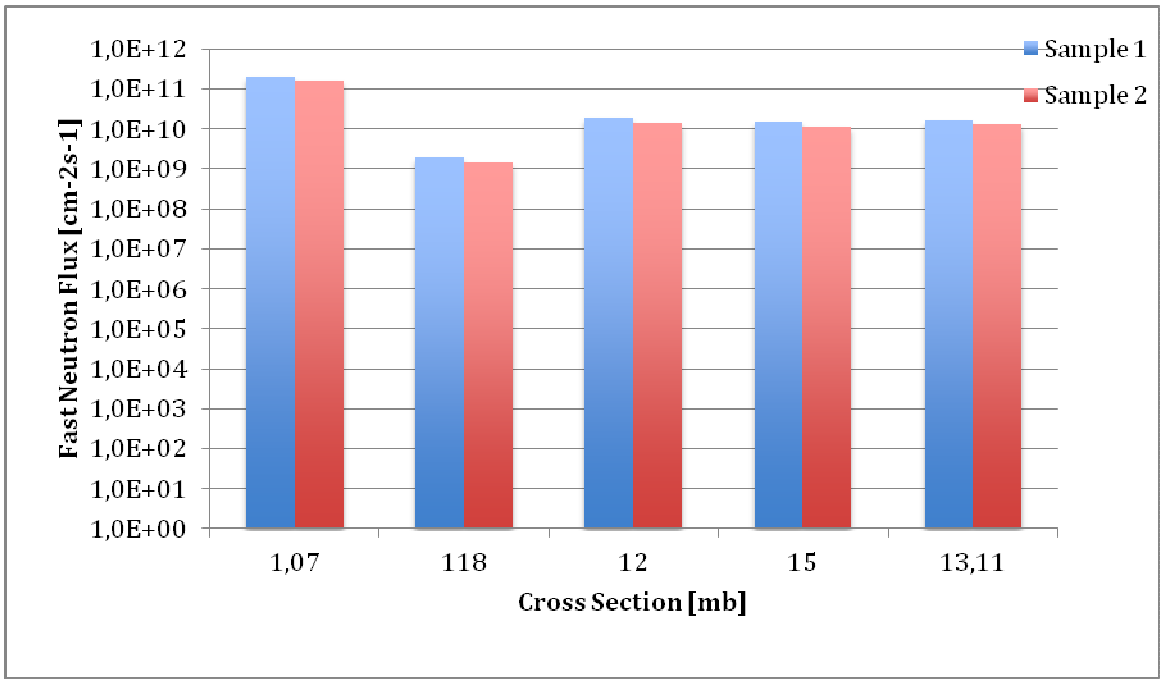


FIG. 2.2. Variation of fast neutron flux density as a function of various values of cross section for iron.

2.11. CONCLUSIONS

Unfortunately, the measurement of the fast neutron flux is far more difficult than of the thermal flux because the bulk material's capture cross-section in this region is much smaller and offers a high degree of uncertainty. On TABLE **2.4** some relevant foil materials for fast neutron flux density measurements are listed, together with their most important parameters. Iron is selected as sample material due to its relative abundance and favourable nuclear properties. But ^{56}Fe has a relatively short half-life of 2.58 hours, hence the β^- measurement must be swift in order to capture meaningful data.

Fast neutrons are more likely to collide with atoms in materials that have a higher cross section, and this scattering would cause the fast neutrons to lose energy. Therefore the fast neutron flux falls exponentially with an increase in fast neutron capture cross-section (see **Error! Reference source not found.**).

TABLE 2.4. THE MOST IMPORTANT PARAMETERS OF SOME RELEVANT FOIL MATERIALS FOR FAST FLUX DENSITY MEASUREMENTS

Element	$\tau_{1/2}$	Reaction	n-energy [MeV]	σ_{eff} [mb]					Atomic weight M [amu]	Foil weight G [g]	Atoms per foil	Abundance
				*(a)	*(b)	*(c)	*(d)	*(e)				
Ti	3.43 d	$^{47}Ti(n, p)^{47}Sc$	3.1	16.3	51.5	48	50-60	40.87				0.073
	85 d	$^{46}Ti(n, p)^{46}Sc$	3.9	10.5	520	25	15	23.5	47.867	2.214×10^{-3}	2.786×10^{19}	0.08
	44 h	$^{48}Ti(n, p)^{48}Sc$	7.6	0.272	58	10	8	10.608				0.738
Mg	15 h	$^{24}Mg(n, p)^{24}Na$	6.8	1.53	200	45	35	44.62	24.305	8.541×10^{-2}	2.116×10^{21}	0.7899
Co	10.48 m	$^{59}Co(n,\square g)^{60}Co$	132 [eV]	20000	-	58	-	846	58.933	4.369×10^{-3}	4.465×10^{19}	1
	2.6 h	$^{59}Co(n,\square a)^{56}Mn$	6.8	0.156	28.2	3	3	2.8				1
Fe	314 d	$^{54}Fe(n, p)^{54}Mn$	3.1	82.5	320	150	120-150	185.6	55.845	3.858×10^{-2}	4.161×10^{20}	0.058
	2.58 h	$^{56}Fe(n, p)^{56}Mn$	6.0	1.07	118	12	15	13.11				0.9172

*(a) Gerhard Erdtman: Neutron Activation Tables

*(b) Friedrich Bensch, Carl M. Fleck: Neutronen-Physikalisches Praktikum I

*(c) International Atomic Energy Agency, Vienna 1987: Handbook on Nuclear Activation Data

*(d) Victoria McLane, Charles L. Dunford: Neutron Cross Sections

*(e) International Atomic Energy Agency

3. DETERMINATION OF THE MATERIAL AND VOID FEEDBACK ON REACTIVITY

3.1. INTRODUCTION

The operational safety of the reactor needs the information of reactivity effects on the core caused by small perturbations. In this experiment, the small perturbations are created in the Central Irradiation Channel of the TRIGA Mark II reactor core to investigate their effects on the core reactivity. Three different kinds of perturbations are considered in this measurement. A cylindrical void (air), heavy water (D_2O) and cadmium (Cd) samples are inserted into the core central position separately and their neutronics behaviour along the axial length of the core is analysed by moving the sample in axial steps.

3.2. PURPOSE

Very small disturbances within the reactor can cause major effects in reactivity, which for instance whilst critical or close to criticality, could lead to loss of control and ultimately fuel damage. The purpose of this experiment is to observe the correlation function and void coefficient, by inserting different materials into the core and observing their effects on reactivity. The students will become familiar with those effects on reactivity and hence on reactor power.

3.3. SAFETY MEASURES

This experiment is performed in the reactor hall and control room of the TRIGA reactor, and needs to follow the following radiation protection instructions:

- Personal dosimeter for each course participant provided by the health physics group;
- Use of lab coat and gloves when handling the samples;
- Spend as little time as possible in the control area;
- Avoid standing directly above the reactor core;
- Each course participant should wash his/her hands after the completion of the experiment and should check for possible contamination using a hand-foot monitor.

3.4. INSTRUMENTS AND MATERIALS

Only the reactor operator is regulating the position of all three control rods. This experiment requires the uses of the following equipment:

- TRIGA Mark II reactor with an operating power of 10 W for this experiment;
- An appropriate amount of heavy water (66.42 cm^3), cadmium (1.25 cm^3) and light water (66.42 cm^3) samples;
- A dosimeter and count rate monitor for measuring dose rate and activity control;
- Sample holders (polyethylene bottles) to keep the samples in the core central position for irradiation;
- A contamination monitor.

3.5. EXPERIMENTAL PROCEDURES

This experiment is performed in the core central position of the reactor. Three cylindrical samples are prepared for this perturbation study.

1. The first sample comprises a polyethylene bottle containing just air (i.e. a void), suitable sealed to prevent ingress of the coolant water;
2. The second sample consists of a tightly rolled sheet of cadmium placed in a polyethylene bottle and filled with reactor water;
3. For the third sample, the bottle is filled with heavy water (D_2O).

The void and heavy water sample have a typical volume of 66.42 cm^3 while the cadmium sample has a volume of 1.25 cm^3. All three samples are placed in a polyethylene bottle with negligible neutron absorption cross-section. This experiment is performed at low power of 10 W. First of all, the void sample is inserted into the core central position and moved vertically in 5 cm steps from bottom to top of the core. For each 5 cm step, the regulating rod position is adjusted to maintain a steady power of 10 W. Once the reactor power has stabilized, the rod position and therefore the reactivity effect is recorded. This experiment is carried out in auto-mode of the reactor control system to maintain the power at 10 W. The same procedure is repeated for heavy water and cadmium samples.

3.6. MAIN PARAMETER(S) MEASURED

To determine the effects of different materials on reactivity the main following parameters are measured:

- p_1 = height of the regulating rod for different positions of air sample;
- p_2 = height of the regulating rod for different positions of water sample;
- p_3 = height of the regulating rod for different positions of heavy water sample.

3.7. TYPICAL REAL DATA COLLECTED

The measured positions of the regulating rod are shown on TABLE **3.1** for the air sample, on TABLE **3.2** for the water sample, and on TABLE **3.3** for the heavy water sample.

TABLE 3.1. REGULATING ROD POSITION FOR AIR SAMPLE

Vertical position [cm]	Regulating rod height p_1 [mm]
0	198
5	198
10	202
15	206
20	210
25	201
30	193
35	191
40	194
45	198
50	203
55	202
60	199
65	198

TABLE 3.2. REGULATING ROD POSITION FOR WATER SAMPLE

Vertical position [cm]	Regulating rod height p_2 [mm]
0	198
5	199

10	199
15	201
20	206
25	224
30	260
35	296
40	320
45	320
50	294
55	264
60	231
65	204
70	200

TABLE 3.3. REGULATING ROD POSITION FOR HEAVY WATER SAMPLE

Vertical position [cm]	Regulating rod height p_3 [mm]
0	198
5	199
10	199
15	198
20	197
25	193
30	184
35	176
40	178
45	179
50	187
55	192
60	196
65	199
70	198

3.8. PRE-KNOWLEDGE REQUIRED FROM STUDENTS

The students should have an understanding in

- nuclear and reactor physics: knowledge of nuclear fission, types of decay, cross section, nuclear reactions, multiplication factor, void coefficient of reactivity, criticality, neutron flux density and its behavior.
- radiation physics and protection: the paticipants should have a profound understanding about radioactive dacay, dose rate and limits, activity, and all aspects regarding radiation protection.

3.9. RESULTS

The following curves on *FIG. **3.1** - FIG. **3.3*** shows the effects of air, Cd/water and heavy water on reactivity of the TRIGA Mark II reactor.

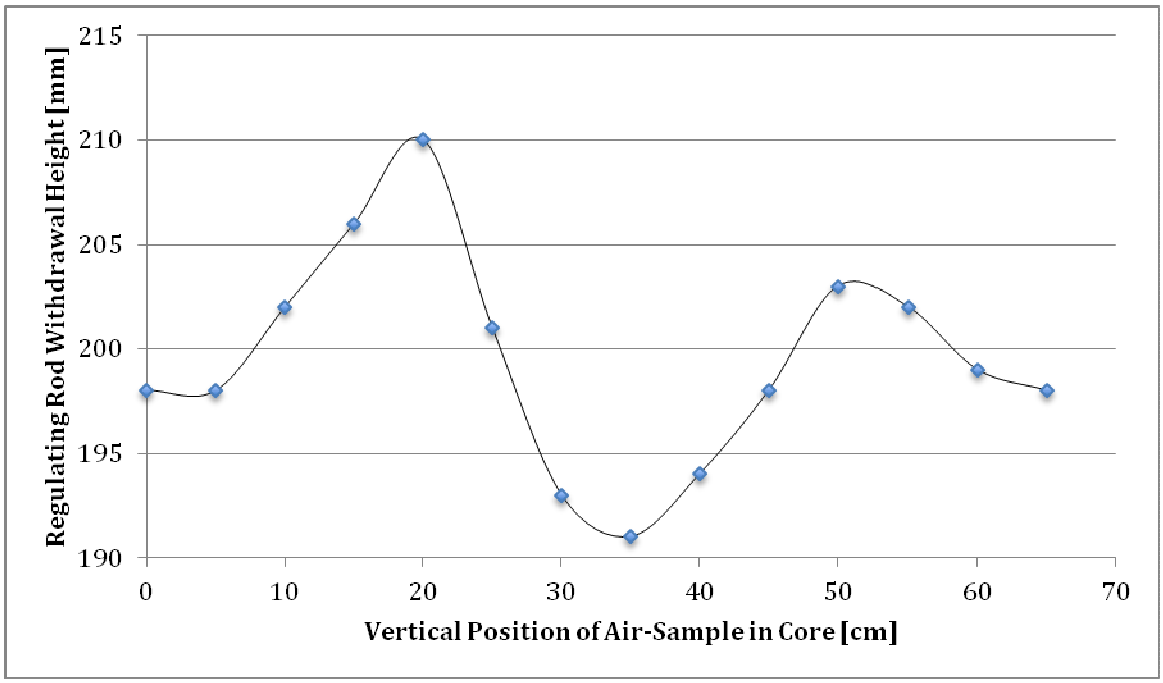


FIG. 3.1. Behaviour of reactivity along vertical position of air sample.

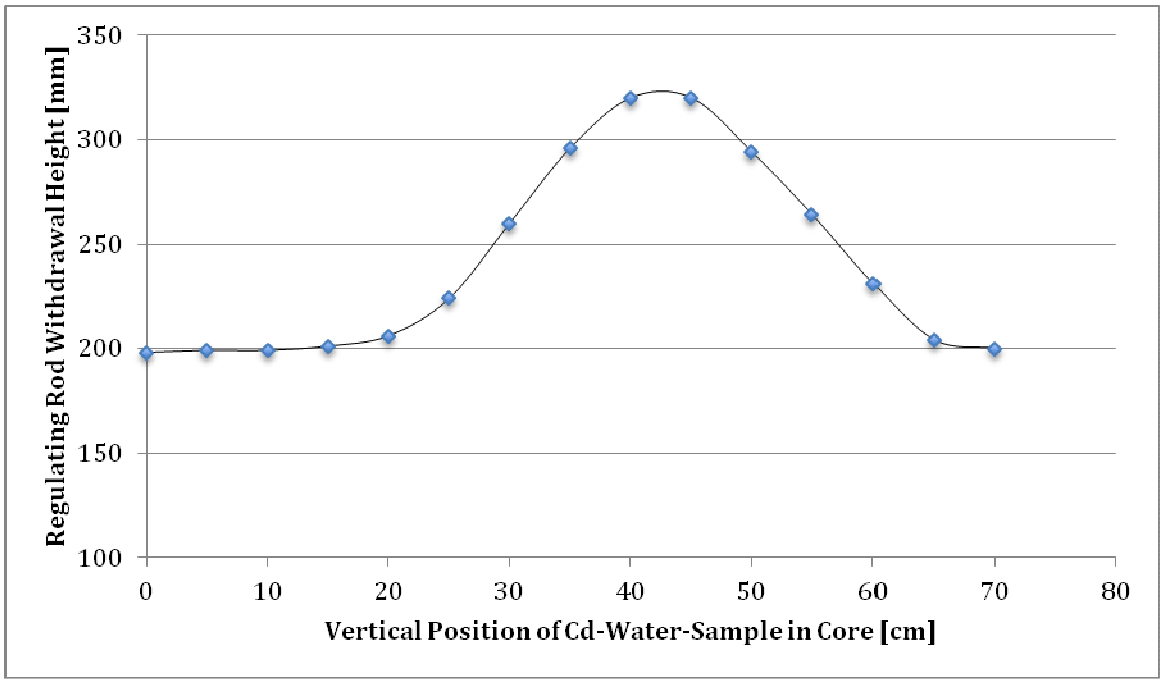


FIG. 3.2. Behaviour of reactivity along vertical position of cadmium-water sample.

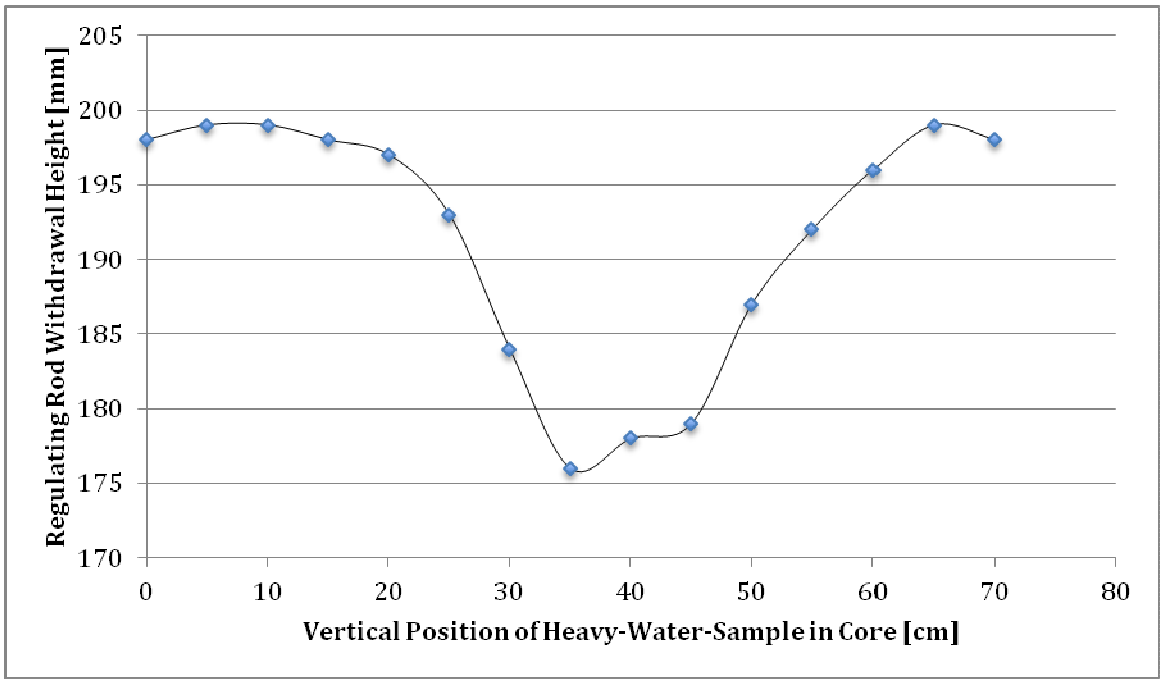


FIG. 3.3. Behaviour of reactivity along vertical position of heavy water sample.

3.10. CONCLUSIONS

The void effects from the bottom to the top of the core central position of the TRIGA Mark II research reactor show that the void coefficient increases from bottom to centre and then decreases from centre to top. At bottom and top ends of the reactor core the void coefficient is slightly negative while in the centre of the core central position it is positive. The reason of this trend is due to the fact that both ends of the core provide leakage routes to the neutrons, while no leakage route is available from the centre of the reactor core.

When the cadmium capsule is moved from bottom to top of the reactor core along the core central position, it shows a very strong reactivity effect due to its high absorption cross section for thermal neutron throughout the axial length of the core central position. It is interesting to note that cadmium introduces a stronger negative reactivity in the centre than at the bottom and top ends of the core. This behaviour is due to the dominant effect of neutron absorption over the leakage.

The heavy water introduces positive reactivity in the centre and negative reactivity in the bottom and top regions of the core. This is due to its relatively smaller absorption cross section than ordinary water along the axial length of the core central position of the TRIGA Mark II reactor core.

4. DETERMINATION OF THE REACTIVITY AND THE REACTOR PERIOD

4.1. INTRODUCTION

The reactor period is the time during which the neutron flux density and hence the reactor power is increased by factor of *e*. The change of the neutron flux density per unit time and unit volume follows:

$$\frac{dn}{dt} = \text{neutron production - leakage - absorption} \tag{4.1}$$

The effective multiplication factor k_{eff} is given as:

$$k_{eff} = k_{\infty} \cdot P \tag{4.2}$$

where k_{eff} defines the number of fission neutrons per absorbed thermal neutron with consideration of a possible leakage, while *P* is the non-leakage probability. These k_{eff} neutrons are now available for absorption or fission. Theoretically, the emission of one neutron is necessary to sustain the fission chain reaction and remaining neutrons (k_{eff} - 1) are available for a power increase or to compensate for any other loss due to experiments. With

$$\Sigma_a \cdot \Phi = \Sigma_a \cdot n \cdot v \tag{4.3}$$

and the dependent neutron flux density, the expression for neutron absorption per cm^3 and second is given by:

$$\frac{dn}{dt} = \left(k_{eff} - 1\right) \Sigma_a \cdot n \cdot v \tag{4.4}$$

Its solution becomes:

$$n = n_0 \cdot e^{(k_{eff} - 1)\Sigma_a \cdot v \cdot t} \tag{4.5}$$

and

$$n = n_0 \cdot e^{t/T} \tag{4.6}$$

where *T* is the reactor period and *t* is the chronological time. The reactor period from Eq.(4.6) is given as:

$$T = \frac{1}{\Sigma_a \cdot v \cdot \left(k_{eff} - 1\right)} = \frac{l}{k_{eff} - 1} \tag{4.7}$$

where *l* is the mean neutron lifetime of prompt neutrons in seconds. Eq. (4.7) shows that *T* is in the range of 0.1 s, if $l \approx 10^{-3}$ s and k_{eff}=1.01. A reactor with such a short period would be impossible to control. However a few fission products (i.e. Br, Rb, I) emit so-called delayed neutrons, which are extremely important for reactor control. These delayed neutrons can be subdivided into 6 groups as shown in TABLE **4.1**.

TABLE 4.1. THE DELAYED NEUTRON PRECURSOR GROUPS FOR THERMAL FISSION IN ^{235}U

Group	Average energy	Half-life	Fractional contribution

	[keV]	[s]	of each group β_i
1	250	55.72	2.1×10^{-4}
2	570	22.72	14.1×10^{-4}
3	412	6.22	12.6×10^{-4}
4	670	2.30	25.3×10^{-4}
5	400	0.61	7.4×10^{-4}
6	698	0.23	2.7×10^{-4}
			$\beta = 64.2$

The overall contribution of delayed neutrons to all fission neutrons is only 0.642 % for ^{235}U. The mean lifetime of the delayed neutrons is given as:

$$\bar{l} = \sum_i (\beta_i \cdot t_i) \tag{4.8}$$

where t is the mean lifetime of the mother nuclide of group i and β_i is the fraction of delayed neutrons of group i. Hence the mean lifetime of all neutrons is given as:

$$\bar{l} = \sum_i (\beta_i \cdot t_i) + l \tag{4.9}$$

The change of the neutron flux density depends on the production rate of the precursors, which emit these delayed neutrons. The change in concentration C_i of the precursor nuclei is as follows:

$$\frac{dC_i}{dt} = \Sigma_a \cdot \phi \cdot k_{eff} \cdot \beta_i - \lambda_i \cdot C_i . \tag{4.10}$$

Furthermore, the time dependent neutron flux density including delayed neutrons is coupled with C_i in Eq. (4.11):

$$\frac{dn}{dt} = \Sigma_a \cdot \phi \cdot k_{eff} \cdot (1 - \beta_i) + \sum_i (\lambda_i \cdot C_i) - \Sigma_a \cdot \phi \tag{4.11}$$

These equations are solved by using:

$$n = A \cdot e^{t/T} \tag{4.12}$$

and

$$C_i = B_i \cdot e^{t/T} \tag{4.13}$$

When using the expression, reactivity ρ leads to:

$$\rho = \frac{k_{eff} - 1}{k_{eff}} \tag{4.14}$$

The In-hour-equation connects reactivity and reactor period as follow:

$$\rho = \frac{l}{l+T} + \frac{T}{l+T} \cdot \sum_{i=1}^{6} \left(\frac{\beta_i}{1 + \lambda_i \cdot T} \right) \tag{4.15}$$

For simplification a weighted single group of delayed neutrons is sometimes used, in which case the mean lifetime is:

$$\tau = \frac{1}{\lambda} = \frac{1}{\beta}\sum_{i=1}^{6}\left(\frac{\beta_i}{\lambda_i}\right) \approx 13 \text{ s} \tag{4.16}$$

Then the following simplified equation is obtained:

$$\rho = \frac{l}{l+T} + \frac{T}{l+T} \cdot \frac{\beta}{1+\lambda \cdot T} \tag{4.17}$$

4.2. PURPOSE

This experiment examines the effects of the control rod on the reactor power. The change of reactivity and reactor period caused by control rod movements is observed. The participants will learn the meaning of control rods and their nexus to reactivity, and become familiar with the effects.

4.3. SAFETY MEASURES

This experiment is performed in the control room of the TRIGA reactor, and needs to follow the following radiation protection instructions:

- Personal dosimeter for each course participant provided by the health physics group;
- Spend as little time as possible in the control area;
- Each course participant should wash his/her hands after the completion of the experiment and should check for possible contamination using a hand-foot monitor.

4.4. INSTRUMENTS AND MATERIALS

Only the reactor operator is regulating the position of all three control rods. The following equipment are used:

- TRIGA Mark II reactor with an operating power of 10 W for this experiment;
- Several stop watches;
- A contamination monitor.

4.5. EXPERIMENTAL PROCEDURES

For this experiment, several stop watches are used to measure the time during which the reactor power increases by a factor of 1.5.
The experiment is conducted in the control room of the TRIGA reactor. Manual control mode is selected to prevent the automatic system from driving the regulating rod, and the control rod drive motor magnets are energized on both the shim rod (safety rod) and the regulating rod. Air is applied to the transient rod, and is fully withdrawn to a height of 499 mm. The exact position of the control rods may be taken from the rod position indicator in the control desk. The reactor is brought to a power of 10 W.

After obtaining a constant power of 10 W, the regulating rod is withdrawn from its starting position of 100 mm to the first height of interest. The reactivity addition caused by this

movement causes the reactor power to increase. The time, which takes the reactor power to rise by 50%, from

- 30 to 45 W,
- 40 to 60 W,
- 50 to 75 W and
- 60 to 90 W

is recorded. Therefore several stop watches are needed. After obtaining the time periods the reactor power is again stabilized at 10 W, and the experiment is repeated for several other heights of the regulating rod (see TABLE **4.2**). The reactor period *T* is obtained from the measured time periods, and from that the reactivity.

4.6. MAIN PARAMETER(S) MEASURED

To determine the effect of the control rod on the reactivity, the following main time periods are measured:

- t_1: time for reactor to reach 45 W from 30 W power;
- t_2: time for reactor to reach 60 W from 40 W power;
- t_3: time for reactor to reach 75 W from 50 W power;
- t_4: time for reactor to reach 90 W from 60 W power.

Each time period (t_1, t_1, t_1, and t_1) is measured for different positions of the regulating rod (distance from core ground):

- d_1: 160 mm above core ground;
- d_2: 170 mm above core ground;
- d_3: 180 mm above core ground;
- d_4: 190 mm above core ground;
- d_5: 200 mm above core ground;
- d_6: 210 mm above core ground.

4.7. TYPICAL REAL DATA COLLECTED

The measured time periods for different heights of the regulating rod are shown in the following TABLE **4.2**.

TABLE 4.2. MEASURED TIME FOR DIFFERENT REGULATING ROD POSITIONS

Rod position (distance) [mm]	Time Period [s]				Average [s]
	t_1	t_2	t_3	t_4	[illegible]
160	21.56	21.34	22.09	21.65	21.66
170	17.73	20.10	18.34	19.02	18.80
180	15.33	14.78	15.15	15.41	15.17
190	10.26	10.72	10.41	10.58	10.49
200	7.84	7.91	8.62	8.61	8.25
210	7.08	7.19	8.00	8.04	7.58

4.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS

The relation between two power levels describes the following equation:

$$P(t) = P(0) \cdot e^{t/T} \tag{4.18}$$

where T is the reactor period in seconds and t is the time elapsed between the two power levels (i.e. from 30 W to 45 W = 1.5 times power increase measured with a stop watch). With the power increase of a factor 1.5, Eq. (4.18) becomes

$$T = \frac{t}{\ln(1.5)} = 2.47 \cdot t \tag{4.19}$$

The reactivity $\rho = \rho(T)$ is calculated according to Eq.(4.15) and Eq.(4.17). For the mean lifetime of the neutrons l the value 10^{-3} s is used. The units of reactivity are measured in either ρ or Dollar (\$). ρ can be used as an absolute value according to Eq.(4.14) and Eq.(4.17), or as a percentage ($\rho_{abs} \times 100\%$). The transfer from percent (%) to dollar (\$) depends on the reactor type. For the TRIGA Mark-II the following value has to be used:

$$\rho(\%) = \rho(\$) \cdot 0.73 \tag{4.20}$$

For TRIGA reactors, instead of the absolute value $\beta = 0.64\%$ (= constant for ^{235}U), $\beta_{eff} = 0.73\%$ is used because the delayed neutrons seem to be available at a higher fraction due to the reactor design. This is because the delayed neutrons are produced with a slightly lower energy (≈ 200 keV) than fission neutrons (≈ 2 MeV) and are therefore thermalized faster. The delayed neutron fraction β_{eff} differs for different types of reactors and fuels.

4.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS

The students should have an understanding in

- nuclear and reactor physics: knowledge of nuclear fission, types of decay, cross section, nuclear reactions, multiplication factor, in-hour equation, core reactivity, criticality, neutron flux density and its behavior.
- radiation physics and protection: the paticipants should have a profound understanding about radioactive dacay, dose rates and limits, activity, and all aspects regarding radiation protection.

4.10. RESULTS

Using Eq.(4.19) and Eq.(4.17) with the measured values in TABLE **4.2**, the following reactivity values on TABLE **4.3** are obtained and graphically shown on *FIG. 4.1*.

TABLE 4.3. EFFECTS OF CONTROL ROD MOVEMENT ON REACTOR PERIOD AND REACTIVITY

Rod position [mm]	T [s]	Δρ [cents]
160	53.50	14.27
170	46.43	15.97
180	37.46	18.81
190	25.92	24.39
200	20.37	28.45
210	18.72	29.93

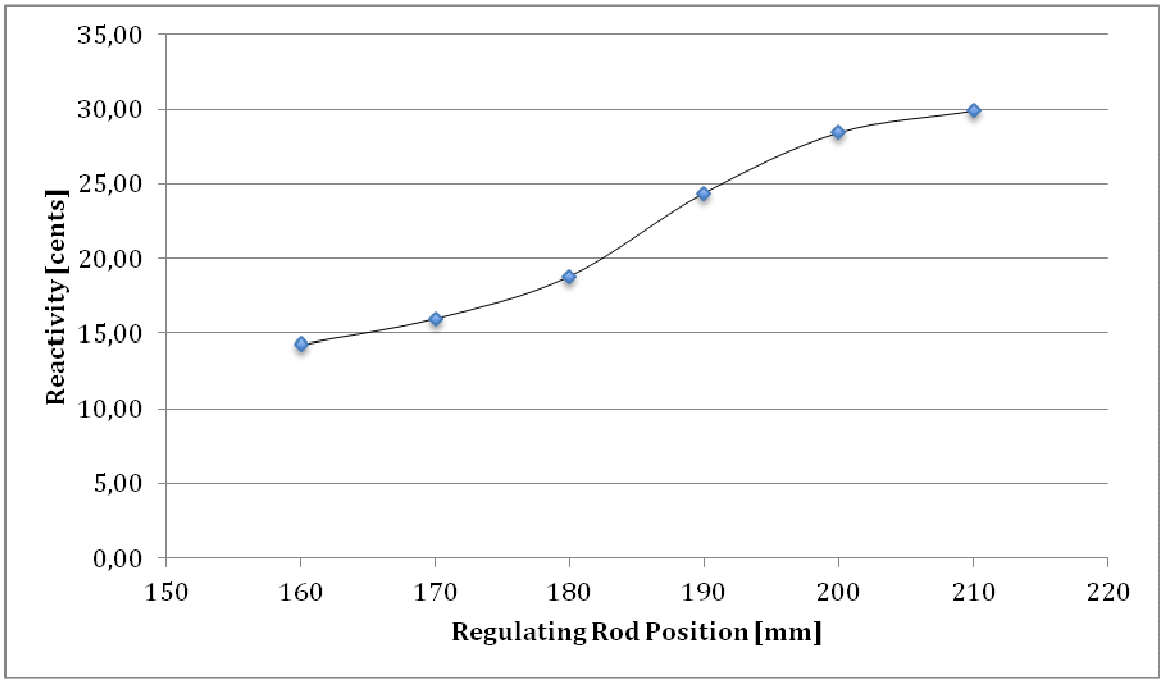


FIG. 4.1. Effects of control rod withdrawal on reactivity.

4.11. CONCLUSIONS

It can be seen on *FIG. **4.1*** that withdrawing the control rod from the core results in a decrease in reactor period, and hence an increase in reactivity. The removal of the highly absorbent boron carbide control rod from the region of neutron flux enables more fission to occur. But the effect on the reactivity is not linear. Since the axial neutron flux distribution has a shape of a cosine, the control rod has a greater effect near the core's axial centreline. Hence smaller movements near the core centreline have a greater effect on reactivity.

When reactor physics was developed after World War II the Dollar was the most important currency in the world. As prompt criticality is the most dangerous situation in a reactor, physicists decided to nominate the reactivity value, which puts the reactor into the prompt critical state as 1 Dollar.

5. MEASUREMENT OF THE BACKGROUND RADIATION AROUND THE OPERATING REACTOR

5.1. INTRODUCTION

Keeping in mind the safety of the radiation workers and experimenters around the reactor shield, e.g. at beam tubes, thermal column, etc., it is necessary to measure the background radiation levels around the operating reactor at various power levels. The TRIGA reactor in Vienna operates up to 250 kW and is equipped with four beam tubes and one thermal column. All beam tubes and the thermal columns are used for basic and applied academic research.

5.2. PURPOSE

This experiment measures the background level at different reactor positions. These measurements are focused on those positions where staff members or students remain for longer periods of time.

5.3. SAFETY MEASURES

To determine the background radiation level around the reactor hall, this experiment needs the following radiation protection measures for the safety of each participant:

- Personal dosimeter for each course participant provided by the health physics group;
- Spend as little time as possible in the control area;
- Each course participant should wash his/her hands after the completion of the experiment and should check for possible contamination using a hand-foot monitor.

5.4. INSTRUMENTS AND MATERIALS

The whole reactor hall is surveyed for this experiment. The following main instruments are required during this experiment:

- A portable gamma dose rate and count rate monitor;
- A portable neutron dose rate monitor;
- Radiation safety markers to highlight the selected area.

5.5. EXPERIMENTAL PROCEDURES

The reactor hall including the reactor platform is surveyed and the points of interests are selected and marked for measurements. Those points of interests around the reactor hall are marked on FIG. 5.2. In performing the measurements portable detectors are used, both a gamma dose rate monitor and a neutron dose rate monitor.

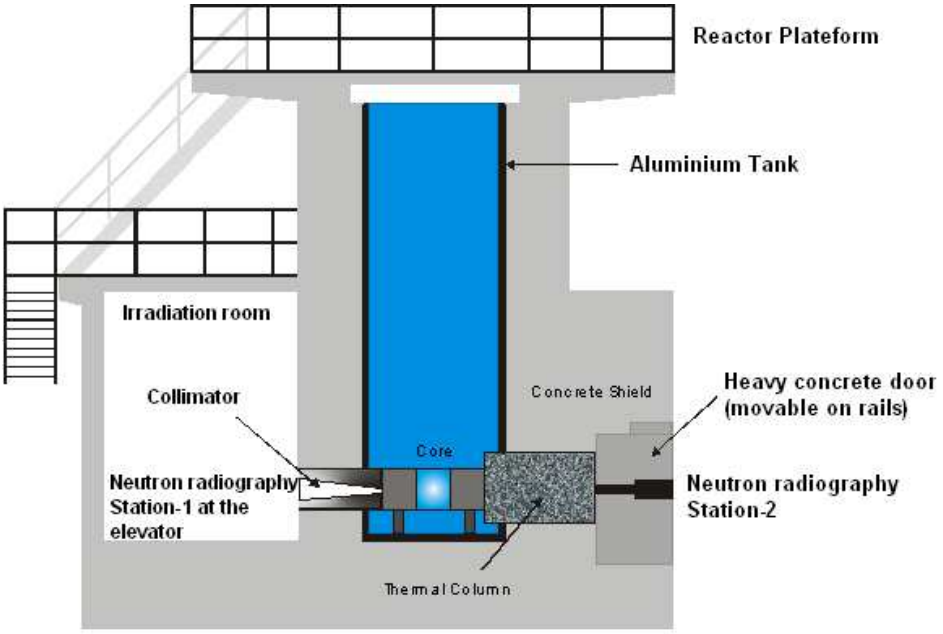


FIG. 5.1. *Vertical cross-section of the TRIGA Mark II reactor.*

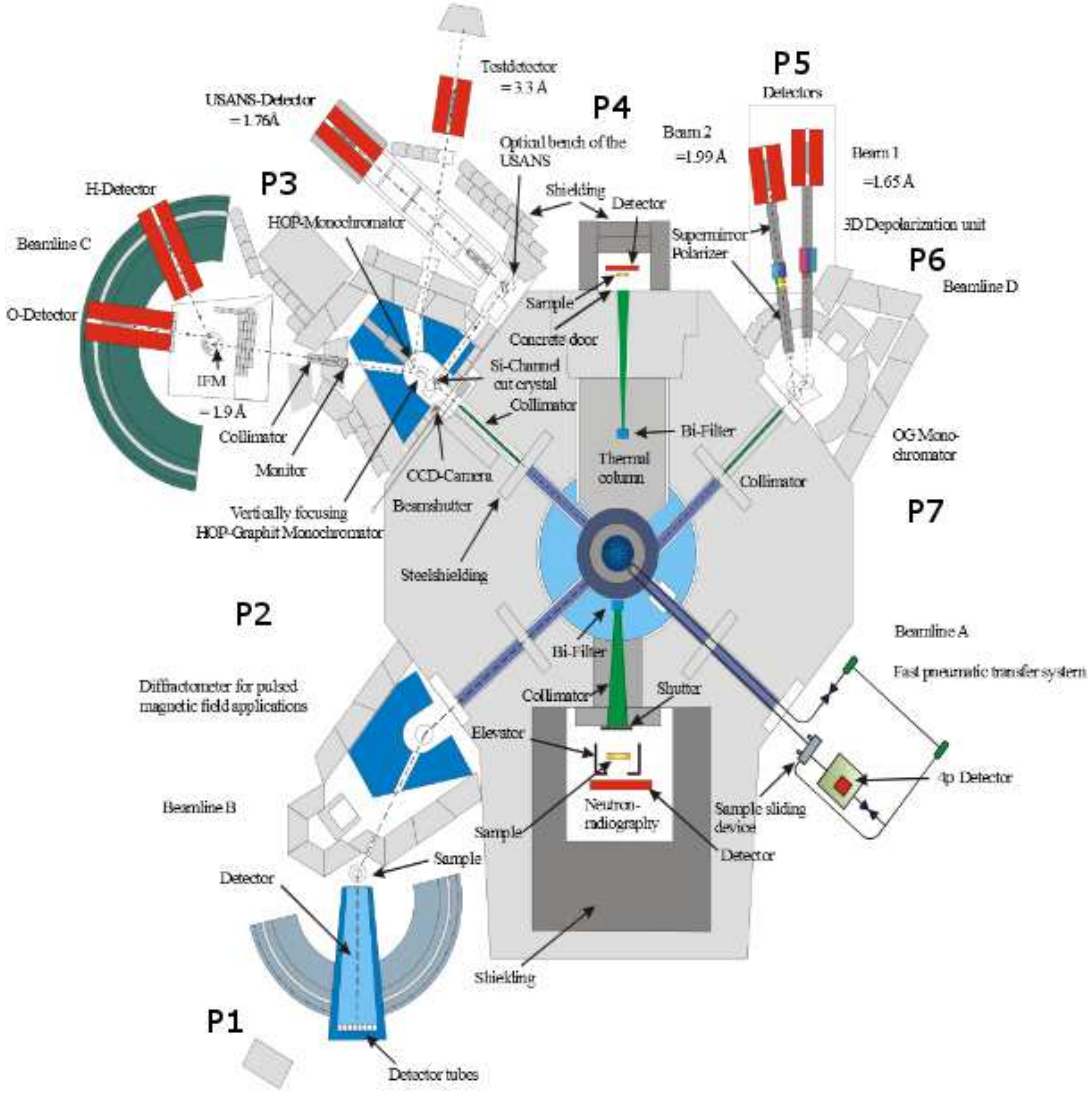


FIG. 5.2. *Horizontal cross-section of the TRIGA Mark II reactor.*

5.6. MAIN PARAMETER(S) MEASURED

For this experiment the following parameters are measured:

- gamma dose rate at different positions in the reactor hall;
- neutron dose rate at different positions in the reactor hall.

5.7. TYPICAL REAL DATA COLLECTED

TABLE **5**.**1** shows the neutron- and gamma dose rates measured at different positions in the reactor hall.

TABLE 5.1. DOSE RATE MEASUREMENTS ON WORKPLACES IN THE REACTOR HALL

Point of interest	Neutron dose rate [μSv/h]	Gamma dose rate [μSv/h]
P1	0.20	0.35
P2	0.10	0.33
P3	0.16	0.30
P4	0.44	0.40
P5	1.23	0.70
P6	0.25	0.23
P7	0.23	0.27

5.8. PRE-KNOWLEDGE REQUIRED FROM STUDENTS

The students should have an understanding in

- radiation physics and protection: the paticipants should have a profound understanding about radioactive dacay, dose rate and limits, activity, and all aspects regarding radiation protection.

5.9. RESULTS

The measured dose rates shown on TABLE **5**.**1** are all within the allowed radiation exposure for radiation workers.

5.10. CONCLUSIONS

The radiation dose rates in the reactor hall is higher than the natural background radiation dose, but within the exposure limits. A negative atmospheric pressure is maintained within the reactor hall by controlled, filtered ventilation system with detectors and alarms, for preventing any release of airborne radioactive particles of the controlled area.

The ‘As Low As Reasonably Practicable’ (ALARP) principle applies to all practices employing nuclear materials, and means that the radiation doses to any individual (workers, operators and the public) should be acceptably low, and reasonable (e.g. education and research at the TRIGA).

6. CRITICAL EXPERIMENT

6.1. INTRODUCTION

Safety is the top requirement in reactor operation. The criticality condition is an important safety design parameter in any type of reactor and the approach to criticality is the most fundamental experiment in nuclear reactor technology. When a reactor is initially loaded with fuel, the amount of fuel necessary for reactor criticality is usually not known very accurately. Therefore, the prediction of the critical mass by neutronics calculations based on reactor theory is necessary for the safe loading of fuel. The physical characteristics of a nuclear reactor as well as the validation of the calculation methods and the nuclear data employed may also be better understood by the comparison of the predicted and measured critical mass.

6.1.1. Theoretical background

Before the first start-up of any reactor a critical experiment needs to be performed. In this experiment, the reactor core is loaded with fuel elements in a number of steps. A neutron with a source strength of typically 10^6 neutrons per second (in case of TRIGA Mark II of Vienna) is installed at one position in the core. These source neutrons introduce fission in ^{235}U atoms in the surrounding fuel to produce fission neutrons. The relative number of neutrons near the core is measured with a fission chamber. It is well known that the ratio of the neutron number between two successive generations is called the multiplication factor k_∞ , defined as:

$$k_\infty = \frac{\text{Number of neutrons in generation}}{\text{Number of neutrons in generation i} \quad -1} \tag{6.1}$$

and

$$k_\infty = \eta \cdot \varepsilon \cdot p \cdot f \tag{6.2}$$

Where ε is the fast fission factor, p is the resonance escape probability, η is the reproduction factor and f is the thermal utilization factor.

The value of k_∞ indicates whether the chain reaction is building up or dying away. If $k_\infty=1$, then the chain reaction, and hence reactor power, is steady and the reactor is critical. If $k_\infty>1$ then the neutron population per successive generations is building and the reactor is super-critical with an increasing reactor power. Conversely, if $k_\infty<1$ the reactor is sub-critical and the chain reaction is dying away. In practice, the term reactivity ρ is used in preference to the multiplication factor, where:

$$\rho = \frac{k-1}{k} \tag{6.3}$$

The value for ρ indicates how far the reactor is from criticality:

- If $k=1$, $\rho=0$ → reactor is critical
- If $k>1$, $\rho>0$ → reactor is super-critical
- If $k<1$, $\rho<0$ → reactor is sub-critical

The ideal critical system keeps the number of neutrons constant. In case of a subcritical system, an increase of the neutron density is also observed because of the presence of the

neutron source. Suppose that the source emits Q neutrons per unit time. Then k^1Q neutrons are produced at the end of the first generation and k^2Q at the end of the second generation, and so forth. This is a geometrical sequence with a multiplication factor <1, and therefore, a finite sum:

$$Q + k \cdot Q + k^2 \cdot Q + ... = \frac{Q}{1-k} \tag{6.4}$$

After some time when the neutron distribution in the system becomes uniform, a stable neutron flux is measured which is considerably higher than the source strength. The neutron flux increases further when the multiplication factor approaches to 1. When k becomes equal to 1, the neutron flux would increase steadily if it is not limited by factors such as temperature effects (Doppler Effect) in the fuel.

6.2. PURPOSE

Critical experiments must be conducted before the start up of any reactor to confirm that the theoretical calculations correspond to the physical reactor build, thus ensuring that the assembly is safe. The participants will become familiar with criticality of the TRIGA Mark II reactor by following the changes, and measuring the average count rate of the reactor with different amount of mass (fuel elements) in the core.

6.3. SAFETY MEASURES

The standards regulations for control of radiation exposure in the laboratory for health and safety should be demonstrated to the experimenters. The general radiation protection safety measures for each student during this experiment are:

- Personal dosimeter for each course participant provided by the health physics group;
- Use of lab coat in the reactor hall;
- Spend as little time as possible in the control area;
- Avoid standing directly above the reactor core;
- Each course participant should wash his/her hands after the completion of the experiment and should check for possible contamination using a hand-foot monitor.

6.4. INSTRUMENTS AND MATERIALS

The critical experiment is performed directly in the reactor core and involves the removal and reloading of about 10 fuel elements. Mainly it uses the fuel handling tool and fuel storage rack. It needs the following equipment:

- TRIGA Mark II reactor with fuel storage rack in the reactor tank;
- Fuel handling tool for the operator (designed to handle TRIGA fuel elements);
- Corona detector;
- A contamination monitor.

6.5. EXPERIMENTAL PROCEDURES

In this experiment, ten fuel elements (FE) are removed from the core and stored in the in-pool storage rack of the reactor tank. The neutron source stays in the core while all control rods are kept in fully withdrawn positions and the count rate of the corona detector is measured. The fuel elements are sequentially loaded from centre to the outside of the core to their respective core positions one by one. The count rate from the neutron detector is recorded after each insertion of one fuel element. When a certain number of fuel elements have been added, the reactor core achieves its critical state. This signal of criticality is seen by a steady increase of the count rate. The same experimental procedure is performed keeping all control rods completely in the core. In case of the TRIGA Mark II reactor of Vienna, keeping all three control rods fully inside the core, criticality is not reached after a complete core loading (presently 76 fuel elements). The number of fuel elements necessary for reactor criticality is determined by extrapolation of the criticality curve. In order to do this, the reciprocal count rate has to be compared with the number of fuel elements. Criticality is achieved when the reciprocal count rate approaches to zero.

6.6. MAIN PARAMETER(S) MEASURED

During the critical experiment the following average count rates are measured:

- S_U = average count rate when all control rods are driven fully up, at the beginning of the experiment
- S_D = average count rate when all control rods are driven fully down, at the beginning of the experiment
- S_{xxU} = average count rate when all control rods are driven fully up, with a certain amount of fuel elements in the core
- S_{xxD} = average count rate when all control rods are driven fully down, with a certain amount of fuel elements in the core

6.7. TYPICAL REAL DATA COLLECTED

The measured count rates are shown on TABLE **6.1** and TABLE **6.2**.

TABLE 6.1. MEASURED DATA AT THE BEGINNING OF THE EXPERIMENT

Amount of FE in core	S_D	S_U
66	1.987×10^3	4.383×10^3

TABLE 6.2. MEASURED COUNT RATES

Amount of FE in core	FE number	FE position	S_{xxD}	S_{xxU}
66	-	-	1.987×10^3	4.383×10^3
67	9902	B5	2.245×10^3	5.871×10^3
68	9970	B6	2.557×10^3	8.209×10^3
69	9911	C9	2.901×10^3	1.238×10^4
70	9912	C10	3.319×10^3	2.228×10^4
71	9926	D13	3.933×10^3	7.399×10^4
72	9927	D14	4.505×10^3	reactor ctritical
73	9947	E17	5.175×10^3	N/A

74	9948	E18	5.878×10^3	N/A
75	9956	F21	6.481×10^3	N/A
76	9957	F22	7.306×10^3	N/A

6.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS

The point of criticality can be estimated before it is reached, by plotting the graph of the reciprocal count rate against the number of fuel elements in the core and extrapolating to the x-axis. This enables a controlled (slow, steady) approach to the point of criticality.

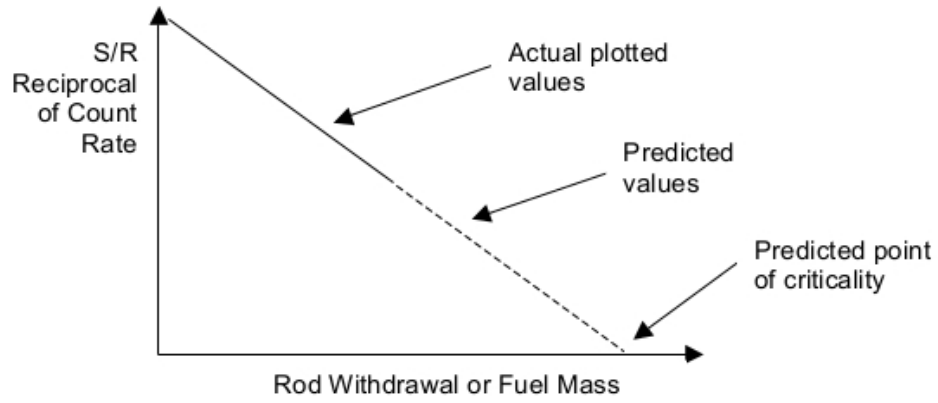


FIG. 6.1 Prediction of criticality of the reactor.

The reciprocal count rate can be determined by

$$\frac{S_D}{S_{xxD}} \quad \text{or} \quad \frac{S_U}{S_{xxU}}, \tag{6.5}$$

depending on the position of all three control rods, whether they are fully inserted or fully up.

6.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS

The students should have an understanding in

- reactor physics: nuclear fission, multiplication factor, criticality, and the neutron flux density should be familiar to all participants.
- radiation physics and protection: the paticipants should have a profound understanding about radioactive dacay, dose rates and limits, activity, and all aspects regarding radiation protection.
- theory of neutron detector.

6.10. RESULTS

By using the equation Eq.(6.5), with the measured data in TABLE **6.1** and in TABLE **6.2**, the reciprocal count rate listed on TABLE **6.3** are obtained, and plotted against the number of fuel elements loaded in the core (*FIG. **6.2***).

TABLE 6.3. RECIPROCAL COUNT RATE

Amount of FE in core	$\frac{S_D}{S_{xxD}}$	$\frac{S_U}{S_{xxU}}$
66	1	1
67	0.885	0.747
68	0.777	0.534
69	0.685	0.354
70	0.599	0.197
71	0.505	0.059
72	0.441	N/A
73	0.384	N/A
74	0.338	N/A
75	0.307	N/A
76	0.272	N/A

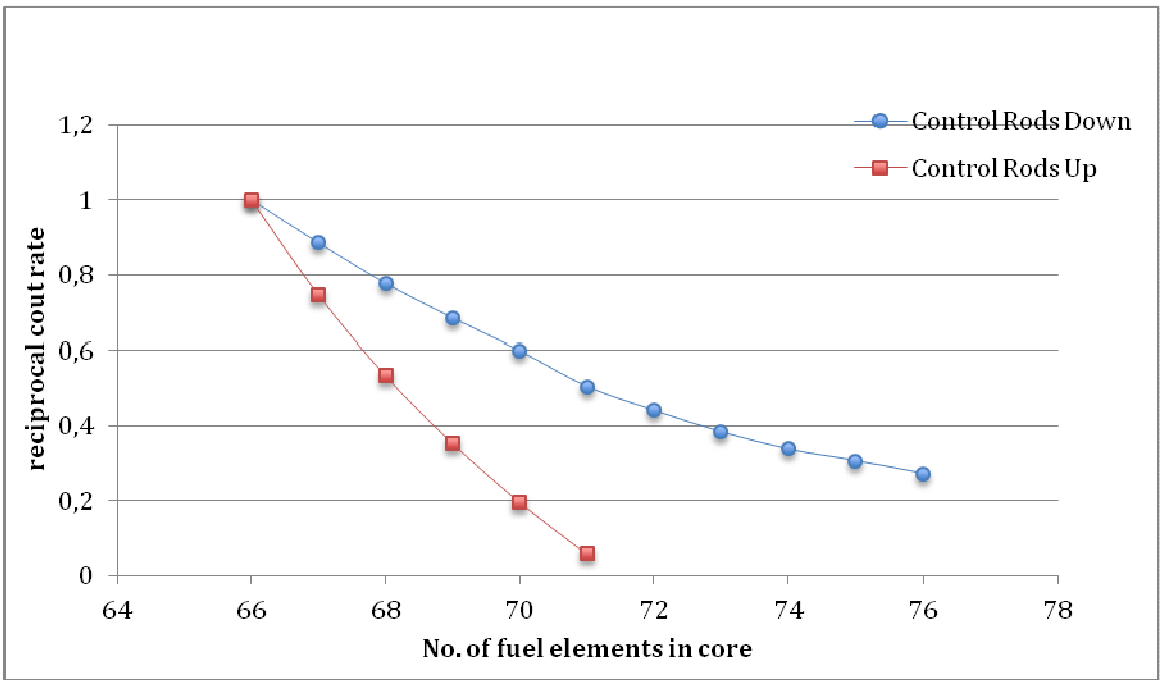


FIG. 6.2. Reciprocal count rate of different number of fuel elements, both with all control rods fully up and down.

The 'red' curve describes the reciprocal count rate with different amount of fuel elements loaded in the core, while all three control rods are fully withdrawn. On the other hand the 'blue' curve shows the reciprocal count rate in the core with all three control rods fully inserted.

6.11. CONCLUSIONS

With all control rods fully withdrawn and 71 fuel elements installed (from inner core rings to outer core rings), the reciprocal count rate approaches zero. As the 72th fuel element was being inserted, the count rate did not stabilize, hence the reactor was critical. Consequently, the number of counts in 30 seconds could not reach a steady rate. On the contrary, with all control rods fully inserted, the reactor was far from critical even with all 76 fuel elements installed in the core.

The positioning of the neutron detector with respect to the neutron source is crucial when estimating critical position – if the detector is situated too close or too far away from to the

source, the relationship between the reciprocal count rate and fuel loading becomes less linear and varies from the ideal curve on *FIG.* ***6.1***. In the TRIGA Mark II reactor core in Vienna, the neutron source is positioned at the top left location and the corona detector is almost directly opposite, bottom right.

7. CONTROL ROD CALIBRATION AND DETERMINATION OF EXCESS REACTIVITY IN THE CORE

7.1. INTRODUCTION

The reactivity of nuclear reactors changes during operation due to various causes. The operating reactor has always to be kept in critical condition (in this case its reactivity is zero) using control rods. The determination of excess reactivity and shutdown margin of a reactor is one of the strictest requirements for safe reactor operation. Thus, control rod calibration of a new built core is the most essential experiment to be performed immediately after the approach to criticality prior to the experiments.

7.1.1. Theoretical background

The reactivity effect of a control rod penetration into the reactor core depends on the:

- Neutron flux density at the point of insertion;
- Penetration length into the active core;
- Reactivity worth of control rods as a function local flux.

As a first approximation, the reactivity worth of a control rod is proportional to the square of the thermal neutron flux density at the point of insertion. Thus the efficiency of a given rod placed into the active core is high (reactivity worth) if the local flux is high. Generally, the thermal neutron flux density is highest in the centre and lowest at the boundaries of the reactor core. Therefore, rods designed for high efficiency are mainly set in the core centre.

It is notable that the control rods efficiency is less than it would be from the neutron flux value at the given place without the rod. The neutron absorber significantly reduces the neutron flux so that the rod is actually at a place of weaker than undisturbed flux. This flux distortion is obvious along the radius and the length of the control rod, as shown in *FIG. 7.**1*** and in *FIG. 7.**2***.

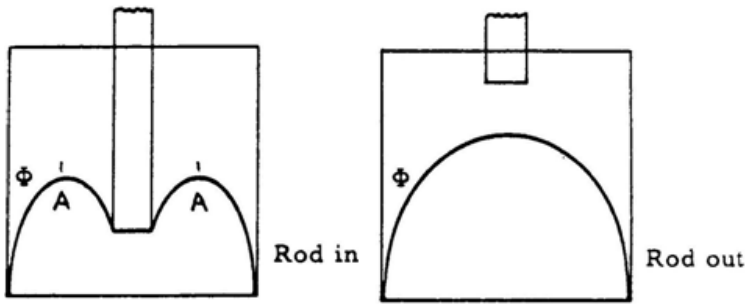


FIG. 7.1. Radial distribution of the neutron flux with and without a control rod.

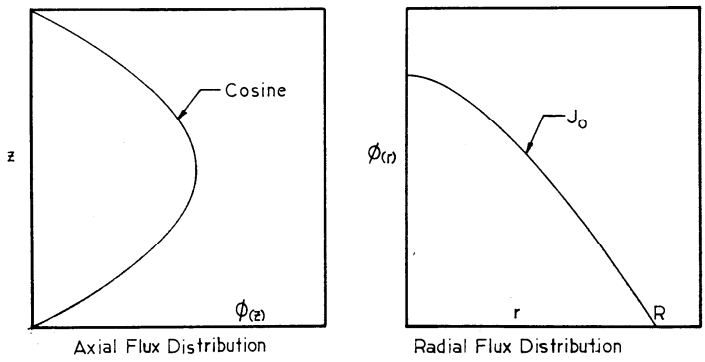


FIG. 7.2. Axial and vertical distribution of the undisturbed neutron flux.

The flux bias results in:

- Modified spatial distribution of the power density inside the active core, partly altering the performance of the given reactor, and partly affecting the temperature distribution in the core.
- Influence on individual rod worth values (flux-dependent) of the other rods, provided several ones are present in the core. For example, the sum of an individual rod worth values of two safety rods is less than the overall worth. This is the so-called rod interference.

For defining the reactivity worth of control rods as a function of penetration, two extreme positions of the rods are:

- Completely outside the core. In this position, the rod has zero effect on the reactivity.
- Completely inside the core. In this position, the rod has a maximum effect on the reactivity. This maximum efficiency is called the total rod reactivity value.

In case of uniform material distribution and symmetric flux distribution, the rod mid-height coincides with the active core mid-height.

There are several methods for determining the characteristic reactivity curve of a control rod. The selection of method depends on the reactor type, the individual rod values, the available instrument and time. The basic problem consists essentially of determining the reactivity for different control rod positions. The current methods are:

- determination of reactivity based on the reactor period measurement;
- rod calibration in a sub-critical system;
- inter calibration method;
- rod oscillation method;
- rod drop method.

The first and most frequently used method is the ‘rod calibration by the reactor period method’, as described below.

It is an absolute method (not relative) to obtain a “reactivity versus rod position” diagram. During this measurement, the reactor is in supercritical operation mode. With the rod in different withdrawn positions, the reactor is operating at low power and the reactor period is determined. The rod to be calibrated is its fully down position.

The calibration starts when the rod, which is to be calibrated, is slightly withdrawn at *P(0)* (power at time 0). After some time t, the reactor reaches the power *P(t)*. The relation between those two power levels describes the following equation:

$$P(t) = P(0) \cdot e^{t/T} \tag{7.1}$$

where T is the reactor period in seconds and t is the time elapsed between the two power levels (i.e. from 30 W to 45 W = 1.5 times power increase measured with a stop watch).

The In-hour equation relates the reactor period to the reactivity. Thus, the reactivity is determined either from a diagram or from available tables. Continuing the experiment in rod steps, the total length of the control rod is calibrated as shown in *FIG. 7.3*.

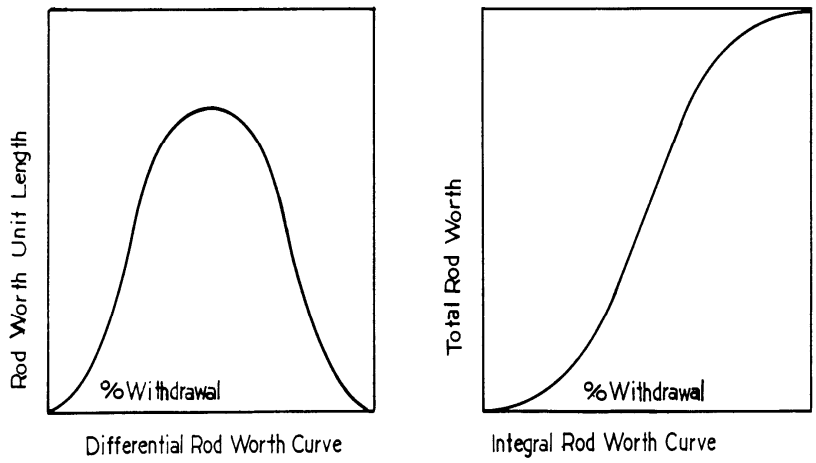


FIG. 7.3. Differential and integral control rod worth curves.

7.2. PURPOSE

This exercise examines the efficiency of control rods and measures the reactivity worth of the regulating rod. A visual representation of control rod worth may be obtained by plotting rod withdrawal against the excess reactivity. The resultant curve shows reactivity effect for different vertical positions of the calibrating rod. The participants will learn the meaning of calibration of the regulating rod, and become familiar with the effect of control rods on the reactivity of the core.

7.3. SAFETY MEASURES

The standards regulations for control of radiation exposure in the laboratory for health and safety should be demonstrated to the experimenters. The general radiation protection safety measures for each student during this experiment are:

- Personal dosimeter for each course participant provided by the health physics group;
- Spend as little time as possible in the control area;
- Each course participant should wash his/her hands after the completion of the experiment and should check for possible contamination using a hand-foot monitor.

7.4. INSTRUMENTS AND MATERIALS

The calibration of the control rod is performed in the control room of the TRIGA reactor. Only the reactor operator is regulating the position of all three control rods. This experiment mainly uses the following equipment:

- Several stop watches;
- A contamination monitor.

7.5. EXPERIMENTAL PROCEDURES

For this experiment, several stop watches are used to measure the time during which the reactor power is increases by a factor of 1.5. The experiment is conducted in the control room of the TRIGA reactor. Manual control mode is selected to prevent the automatic system from driving the regulating rod, and the control rod drive motor magnets are energized on both the shim rod (safety rod) and the regulating rod. Air is applied to the transient rod, and is fully withdrawn to a height of 499 mm. The exact position of the control rods may be taken from the rod position indicator in the control desk. To achieve a reactor power of 10 W, the shim rod is slowly withdrawn to a height of 200 mm, ensuring that the reactor period do not exceed 30 W. Care was necessary as the shim rod has a higher rod worth compared to the regulating rod.

The regulating rod was withdrawn continuously to 75 mm. The reactivity addition caused by this movement causes the reactor power to increase and the reactor to become supercritical. The time, which takes the reactor power to rise by 50%, from

- 30 to 45 W,
- 40 to 60 W,
- 50 to 75 W and
- 60 to 90 W

is recorded. Therefore several stop watches are needed. After that experiment the reactor power is reduced again to 10 W by inserting the shim rod, and the experiment is repeated for several other heights of the regulating rod (see TABLE **7.1**). The obtained reactivity values are applied to plot the integral and differential characteristic curve. Using the rod calibration curve of each control rod, the reactivity value $\Delta\rho$ is determined by the difference $\Delta\rho$ at the present rod position to the fully out position of the control rod. The total core excess reactivity is the sum of the individual $\Delta\rho$ of all rods.

7.6. MAIN PARAMETER(S) MEASURED

For the calibration of the control rod the following main time periods are measured:

- t_1: time for reactor to reach 45 W from 30 W power;
- t_2: time for reactor to reach 60 W from 40 W power;
- t_3: time for reactor to reach 75 W from 50 W power;
- t_4: time for reactor to reach 90 W from 60 W power.

Each time period (t_1, t_2, t_3, and t_4) is measured for different positions of the regulating rod (distance from core ground):

- d_1: 75 mm above core ground;

- d_2: 150 mm above core ground;
- d_3: 225 mm above core ground;
- d_4: 300 mm above core ground;
- d_5: 375 mm above core ground;
- d_6: 500 mm above core ground.

7.7. TYPICAL REAL DATA COLLECTED

The measured time periods for the rod calibration are shown in the following TABLE **7.1**.

TABLE 7.1. MEASURED TIME FOR REGULATING ROD CALIBRATION

Rod position (distance) [mm]	Time Period [s]				Average [s]
	t_1	t_2	t_3	t_4	
75	27.10 27.28	27.64 27.62	29.13 29.71	28.71 -	28.17
150	9.60 10.01	10.71 9.87	9.94 9.89	9.31 9.77	9.89
225	8.97 9.30	9.11 9.10	9.59 9.02	10.31 9.94	9.42
300	9.20 8.95	10.28 9.32	9.83 9.71	9.09 9.28	9.46
375	16.21 15.47	16.71 16.81	17.53 17.34	18.09 18.07	17.03
500	36.43 36.63	35.34 35.13	34.78 35.11	34.89 35.93	35.53

7.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS

For a power increase of a factor 1.5,

$$P(t) = 1.5 \cdot P(0) \tag{7.2}$$

the Eq. (7.1) becomes

$$1.5 \cdot P(0) = P(0) \cdot e^{t/T} \tag{7.3}$$

and the reactor period

$$T = \frac{t}{\ln(1.5)} = 2.47 \cdot t\,. \tag{7.4}$$

With the reactor period T, the reactivity value $\Delta\rho$ is determined from available tables.

7.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS

The students should have an understanding in

- nuclear and reactor physics: knowledge of nuclear fission, types of decay, cross section, nuclear reactions, multiplication factor, core reactivity, criticality, neutron flux density and its behavior.
- radiation physics and protection: the paticipants should have a profound understanding about radioactive dacay, dose rates and limits, activity, and all aspects regarding radiation protection.

7.10. RESULTS

By using Eq. (7.4) for the reactor period, with the measured data on TABLE **7.1**, the following reactivity values and total reactivity on TABLE **7.2** were obtained. The differential reactivity of the regulating rod is plotted on *FIG. 7.4* and the integral rod worth on *FIG. 7.5*.

TABLE 7.2. REACTOR PERIOD AND REACTIVITY VALUES OF REGULATING ROD

Rod position (distance) [mm]	T [s]	Δρ [$]	ρ [$]*
75	69.58	0.13	0.13
150	24.42	0.25	0.38
225	23.26	0.26	0.64
300	23.36	0.26	0.90
375	42.06	0.18	1.08
500	87.76	0.10	1.18

*Integrated reactivity in [$]

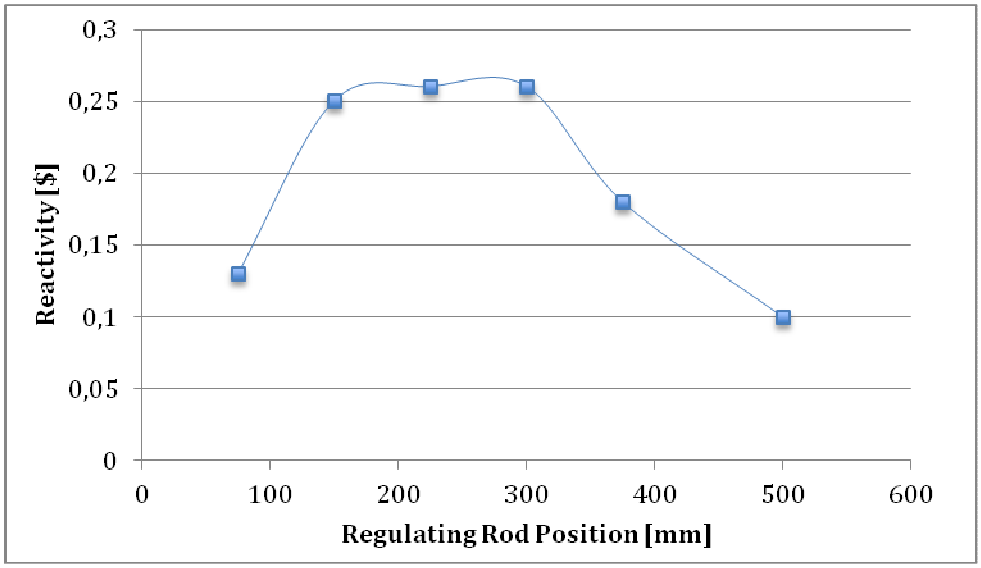


FIG. 7.4. Differential Characteristic Curve for the Regulating Rod.

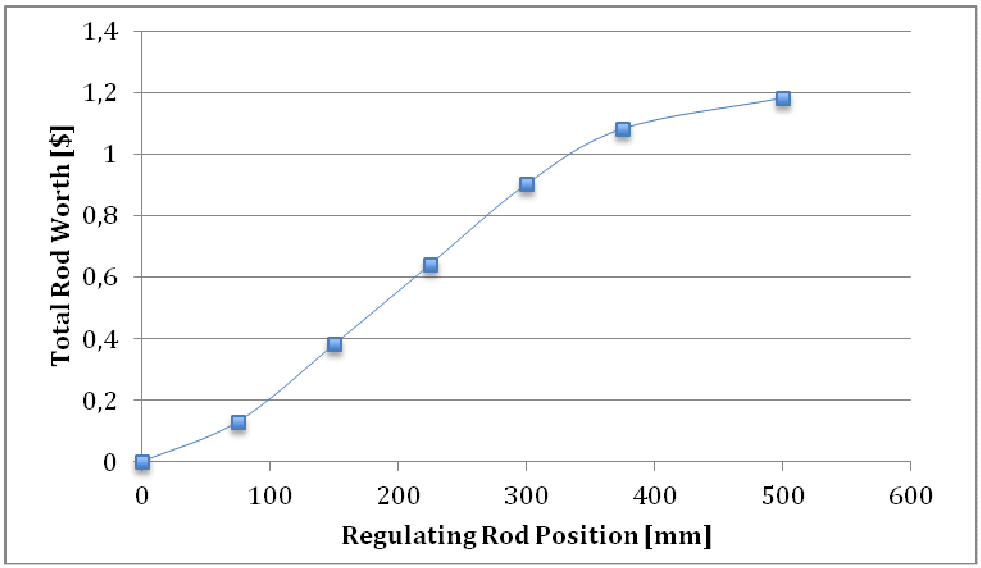


FIG. 7.5. Integrated Reactivity of the Regulating Rod.

7.11. CONCLUSIONS

This absolute method for determining the reactivity produces very accurate results and its advantage is that it enables a direct plot of reactivity against rod position. The drawback is that this experiment is performed in supercritical state and also takes a longer time to measure.

The results plotted in *FIG. 7.5* are typical for an integral rod worth curve, with the steepest gradient representing the greatest change in reactivity for height of withdrawal. The total reactivity worth of the regulating rod was found to be $1.18. The difference in reactivity per increment is shown on *FIG. 7.4*.

FIG. 7.6 shows pictorially the characteristics of core excess and shutdown margin. In a critical reactor, core excess is the amount of additional reactivity possessed by a control rod. Shutdown margin is the amount of negative reactivity that the control rod has. The overall reactivity of the TRIGA core in Vienna is approximately $2, with a total control rod worth of around $6 (regulating rod: ~$1.18; transient rod: ~$2; shim rod: ~$3). Hence the control rods provide considerable excess shutdown margin. Nuclear reactors must always be capable of shutting down even in the event of the most reactive rod being stuck fully out. Clearly, the TRIGA complies with this requirement as even with the shim rod fully out, the shutdown margin is $3 whilst the core reactivity is only $2.

Errors in measurement always account for some variation from an ideal set of results, e.g. the recorded timings with commercial stop watches performed by separate participants. Also the core power was read from digital instrumentation, though an analogue meter with a larger display may have been easier to read.

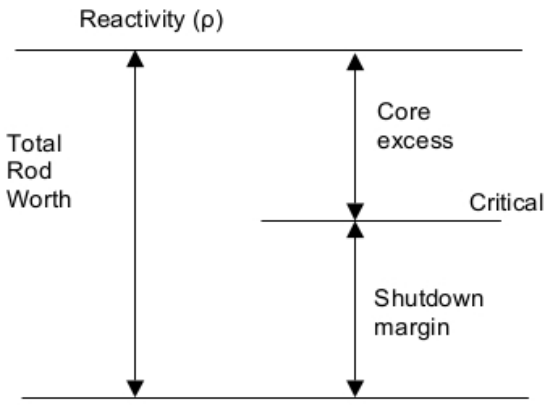


FIG. 7.6. Characteristics of core excess and shutdown margin.

8. REACTIVITY VALUES OF FUEL ELEMENTS IN VARIOUS CORE POSITIONS

8.1. INTRODUCTION

This experiment provides the reactivity values of fuel elements in various core positions. In the TRIGA core lattice, the reactivity worth of fuel elements for a well-thermalized, small, compact and uniform core is proportional to the square of the thermal neutron flux density, integrated over the entire fissile volume of the fuel elements. It also gives an idea of the distribution of neutron flux in the core.

8.1.1. Theoretical background

The neutron flux density distribution in a cylindrical core is calculated by solving the diffusion equation in cylindrical coordinates. The solution (flux) of the diffusion-approximated equation is:

$$\phi(r,z) = A\, J_0\left(\frac{2.405\, r}{R}\right)\cos\frac{\pi z}{H} \qquad (8.1)$$

Eq. (8.1) shows that the neutron flux density decreases both in the axial and radial directions towards the edges of the core. It is obvious that fuel elements in the different core positions have different reactivity values.

8.2. PURPOSE

This exercise examines the reactivity worth of fuel elements on different positions in the reactor core. A visual representation of reactivity worth may be obtained by plotting the horizontal distance of the fuel element to the core centre against the excess reactivity. The participants will learn the meaning of reactivity worth of fuel elements. They will become familiar with the effect of the horizontal distance of fuel elements on the reactivity of the core.

8.3. SAFETY MEASURES

The standards regulations for control of radiation exposure in the laboratory for health and safety should be demonstrated to the experimenters. The general radiation protection safety measures for each student during this experiment are:

- Personal dosimeter for each course participant provided by the health physics group;
- Spend as little time as possible in the control area;
- Avoid standing directly above the reactor core;
- Each course participant should wash his/her hands after the completion of the experiment and should check for possible contamination using a hand-foot monitor.

8.4. INSTRUMENTS AND MATERIALS

This experiment is performed in the control room of the TRIGA reactor. Only the reactor operator is regulating the position of all three control rods and handling the fuel elements. This experiment mainly uses the following equipment:

- TRIGA Mark II reactor with fuel storage rack in the reactor tank;

- Five TRIGA fuel elements from the core;
- Fuel handling tool for the operator (designed to handle TRIGA fuel elements);
- A contamination monitor.

8.5. EXPERIMENTAL PROCEDURES

The fuel and graphite elements are selected at different distances to the core centre. This experiment is performed at low power (10 W) and the regulating rod is used to measure the reactivity worth of the individual fuel or graphite element, as the control rod system is set into the automatic mode, which automatically compensates reactivity perturbations.

During operation, the fuel element to be measured is removed from the reactor core and stored into the fuel storage rack. Because of the decrease in fuel amount from the critical core, the reactor power is decreased. The automatic control system adjusts the regulating rod positions again to bring the reactor back to the same power level. The difference of two regulating rod positions (before and after withdrawal of a fuel element) is expressed in terms of reactivity values by using the regulating rod calibration curve. The experiment is repeated with fuel elements in different positions and also if possible with reflector elements.

The reshuffling the fuel elements back to the reactor core is performed slowly to avoid any sudden rise of power and a possible reactor shutdown due to the sudden reactivity increase. For the evaluation of this experiment, the distance of the fuel element from the centre of the core is required. The TRIGA Mark II core has a cylindrical core lattice in which fuel elements are arranged into five concentric rings. For near centre fuel elements the regulating rod reactivity value might be too low, so the shim rod has also to be removed. The changes in the shim rod reactivity value have to be added to the regulating rod reactivity value.

8.6. MAIN PARAMETER(S) MEASURED

For this experiment the following distances of the regulating rod are measured:

- d_1: height of regulating rod on 10 W power, before fuel element removal;
- d_2: height of regulating rod on 10 W power, after fuel element removal.

To obtain the differential reactivity worth of the fuel elements, the following parameters are taken from the graph on *FIG. 7.5* in CHAPTER 7:

- ρ_1: reactivity worth at regulating rod position d_1;
- ρ_2: reactivity worth at regulating rod position d_2.

8.7. TYPICAL REAL DATA COLLECTED

The measured distances (heights) of the regulating rod for different fuel elements at various locations in the core are shown in the following table (TABLE **8.1**).

TABLE 8.1. MEASURED HEIGHTS OF REGULATING ROD

Fuel element number	Core position of fuel element	Radial distance from core centre [cm]	Regulating rod position [mm]	
			d_1	d_2
2103	F21	20.33	95	206
2108	E17	16.33	97	262

2109	D13	12.15	96	356
9589	C9	8.18	98	372
10077	B5	4.2	100	497*

8.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS

To determine the reactivity values of the fuel elements, the characteristic curve of the integrated reactivity of the regulating rod (*FIG. 7.5*) from CHAPTER 7 is used.

8.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS

The students should have an understanding in

- reactor physics: fundamental nuclear physics and reactor physics, core reactivity, criticality and the neutron flux density should be familiar to all participants.
- radiation physics and protection: the paticipants should have a profound understanding about radioactive dacay, dose rates and limits, activity, and all aspects regarding radiation protection.

8.10. RESULTS

The reactivity values of all measured fuel elements are listed on TABLE **8.2** and plotted against the distance from the core centre on *FIG. **8.1***

TABLE 8.2. REACTIVITY VALUES OF FUEL ELEMENTS

Regulating rod position [mm]		Reactivity values [$]		
d_1	d_2	ρ_1	ρ_2	$\Delta\rho$
95	206	0.16	0.43	0.27
97	262	0.16	0.59	0.43
96	356	0.16	0.99	0.83
98	372	0.16	1.04	0.88
100	497*	0.17	1.5	1.33

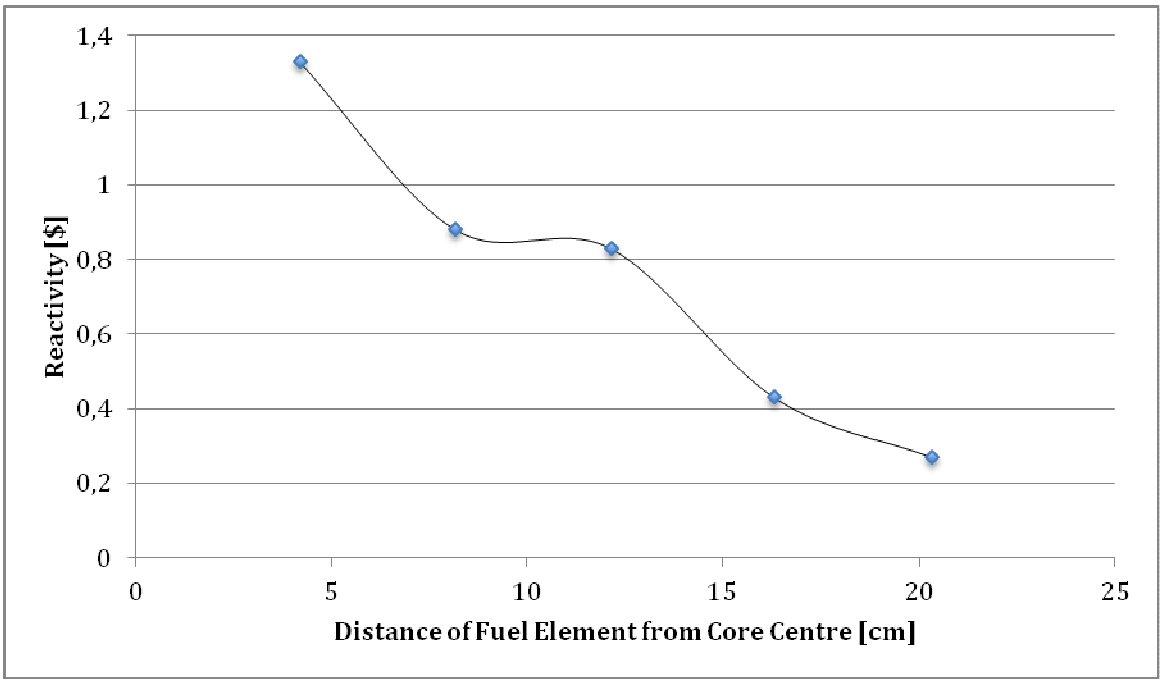


FIG. 8.1. Reactivity Value of Fuel at Different Distances form the Core Centre.

8.11. CONCLUSIONS

It can be seen on *FIG.* ***8.1***, that the reactivity worth of the fuel elements are very much dependent on their distance to the centre of the core. The closer the removed fuel element is to the centre of the core, the further the regulating rod has to be withdrawn from the core in order to maintain the preset power of 10 W, hence the greater is the reactivity worth of the fuel element.

9. REACTOR POWER CALIBRATION AND TEMPERATURE COEFFICIENT OF THE REACTIVITY

9.1. INTRODUCTION

The thermal power of a water-cooled reactor can be determined by measuring the temperature increase of the water during a given time interval. The electrical power of a nuclear power plant can be calculated by multiplying the thermal power by its thermal efficiency factor. The efficiency of a nuclear power plant is approximately 37% (European Pressurized Water Reactor).

9.1.1. Theoretical background

The temperature fluctuations during reactor operation are unavoidable and any temperature change influences the core reactivity. With the rise in temperature the mean energy of the thermal neutrons is increased. This influences the neutron absorption, as the neutron cross-section depends on the neutron energy (fuel temperature coefficient). Furthermore, the mean free path and the non-leakage factor depend on the density and therefore on the temperature (density temperature coefficient).

The nuclear temperature coefficient results mainly from the energy dependence of the absorption and scattering cross-sections. The kinetic energy of the neutrons is proportional to the absolute temperature T; therefore the cross-sections vary in the 1/v-region as:

$$\sigma_a = \sigma_{a0} \cdot \left(\frac{T_0}{T} \right)^{1/2} . \tag{9.1}$$

The same dependence can be shown for the scattering cross-sections. Therefore, all reactor parameters depending on these factors are also temperature dependent:

$$L^2 = \frac{1}{3 \cdot \Sigma_a(T) \cdot \Sigma_s(T)} \tag{9.2}$$

$$\tau = \tau_0 - \int_{E_{th_0}}^{E_{th}} \frac{1}{3 \cdot \xi \cdot \Sigma_S(T) \cdot \Sigma_a(T) \cdot \xi} \frac{dE}{E} \tag{9.3}$$

Therefore k_{eff} or reactivity ρ are temperature dependent:

$$\rho(T) = \frac{k_\infty - 1}{k_\infty} - \frac{B^2}{k_\infty} \cdot \left[\tau_0 - \frac{1}{3 \cdot \xi \cdot \Sigma_S \cdot \Sigma_t} \ln\left(\frac{T}{T_0} \right) + L_0^2 \cdot \left(\frac{T}{T_0} \right)^{0.6} \right] . \tag{9.4}$$

Additionally, the nuclear temperature coefficient has to be considered. The macroscopic cross-sections are proportional to the number of atoms per unit volume. L^2 and τ are proportional to the inverse density square:

$$L^2 + \tau = L_0^2 \cdot \left(d_0 / d \right)^2 + \tau_0 \cdot \left(d_0 / d \right)^2 \tag{9.5}$$

As long as the sum of all effects is negative, the temperature coefficient of the reactivity is also negative and the reactor is called self-stabilizing. In contrast, with a positive temperature coefficient, the reactor power increases with increasing temperature, which makes the reactor unstable. The TRIGA type reactors are inherently safe because of very large negative temperature coefficients of the reactivity, which comes mainly from the Zr-H moderator.

9.2. PURPOSE

The purpose of this experiment is to calibrate the reactor power by examine the water temperature increase. The participants will observe the temperature rise due to control rod movement and compare the results with those from the power calibration. The students will afterwards determine the temperature coefficient of reactivity.

9.3. SAFETY MEASURES

The standards regulations for control of radiation exposure in the laboratory for health and safety should be demonstrated to the experimenters. The general radiation protection safety measures for each student during this experiment are:

- Personal dosimeter for each course participant provided by the health physics group;
- Spend as little time as possible in the control area;
- Avoid standing directly above the reactor core;
- Each course participant should wash his/her hands after the completion of the experiment and should check for possible contamination using a hand-foot monitor.

9.4. INSTRUMENTS AND MATERIALS

This experiment is performed in the reactor hall and the control room of the TRIGA reactor. Only the reactor operator is regulating the position of all three control rods. The following equipment and instruments are needed:

- Pool water heaters;
- A pool temperature detector;
- A contamination monitor.

9.5. EXPERIMENTAL PROCEDURES

Before the start-up of the reactor 5 immersion heaters with a power of 20 kW heat the pool water electrically. This results in the rise of the water temperature (5.19°C ± 0.03) in a certain time interval (5 hours × 20 kW = 100 kWh), which is measured very accurately by thermometers. This procedure provides the calibration of the temperature increase in terms of thermal power. For example, if the reactor is operated at an unknown power (i.e. ~100 kW) with the cooling system shut down, the rise in water temperature is increased and can be compared to that power calibration with the heaters. Thus the thermal power of the operating reactor is calculated easily.

To measure the temperature coefficient of the reactivity it is necessary to operate the reactor first at a low power, i.e. 10 W. The position of all control rods and the temperature of the fuel element are noted. Then the reactor power is increased to 100 kW and the control rod positions and fuel element temperature are recorded again. At this higher power level the control rods are withdrawn from the core relative to the 10 W operation of the reactor. The

difference of reactivity $\Delta\rho$ is compensated by the increased fuel temperature ΔT. The $\Delta\rho$ can be found from the control rod calibration curve, and the ΔT can be read from the instrumentation.

9.6. MAIN PARAMETER(S) MEASURED

For this experiment the following parameters are measured:

- T_i: temperature of the core at different time points;
- positions of all three control rods (transient, shim, and regulating) at power level 10 W and 100 kW;
- average temperature of fuel elements (thermocouples) at power level 10 W and 100 kW;

The reactivity at power level 10 W and 100 kW are taken from the graph on *FIG. 7.5* in CHAPTER 7:

- Reactivity worth of regulating rod position at 10 W power;
- Reactivity worth of regulating rod position at 100 kW power.

9.7. TYPICAL REAL DATA COLLECTED

The measured temperature behaviour of the reactor core over time is shown on TABLE **9.1**. TABLE **9.2** shows the measured parameters at different power levels.

TABLE 9.1. REACTOR POOL TEMPERATURE OVER TIME AT 100 kW POWER

t [min]	T [°C]
0	27.8
5	28.5
10	29.2
15	29.7
20	30.1
25	30.8
30	31.2
35	31.6
40	32.1
45	32.5
50	33.1
55	33.5
60	34
65	34.3
70	34.9

TABLE 9.2, MEASUERED PARAMETERS OF VARIOUS POWER LEVEL

Reactor power	Rod position [mm]			Temperature [°C]	
	Transient	Shim	Regulating	Water	Fuel element
10 W	499	277	117	27.8	27.8
100 kW	499	277	356	28.3	104

9.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS

To determine the reactivity values at different power levels the characteristic curve of the integrated reactivity of the regulating rod (*FIG. 7.5*) from CHAPTER 7 is used.

The actual reactor power P (in kW) can be calculated as follows:

$$P = \frac{\Delta T}{5.19} \cdot 100 \tag{9.6}$$

where ΔT is the difference in temperature during one hour.

From $\Delta\rho$ and ΔT the temperature coefficient of reactivity $\frac{\Delta\rho}{\Delta T}$ is calculated.

9.9. PRE-KNOWLEDGE REQUIRED FROM STUDENTS

The students should have an understanding in

- reactor physics: fundamental nuclear physics and reactor physics, core reactivity, criticality and the neutron flux density should be familiar to all participants,
- radiation physics and protection: the paticipants should have a profound understanding about radioactive dacay, dose rates and limits, activity, and all aspects regarding radiation protection,

9.10. RESULTS

The temperature increase of the reactor pool at a power of 100 kW is plotted on *FIG. **9.1*** and ΔT is obtained with a value of 0.0979°C per minute. Therefore the actual reactor power is calculated with Eq.(9.6) and is found to be 113.18 kW.

From TABLE **9.3** the temperature coefficient can be calculated and is found to be -1.05 cents/°C.

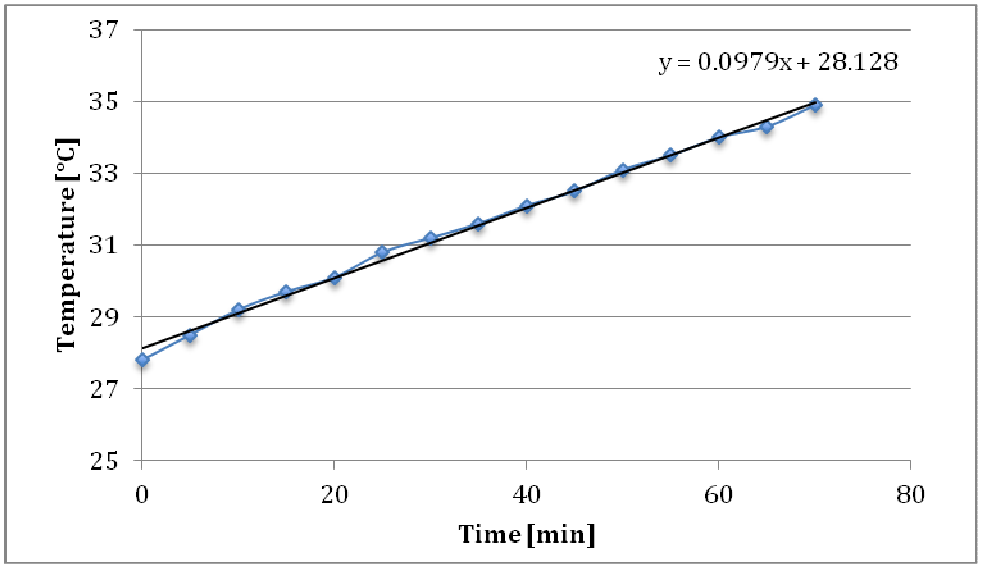


FIG. 9.1. Variation of reactor pool temperature in time, at 100 kW power.

TABLE 9.3, TEMPERATURE COEFFICIENT OF REACTIVITY

	Reactor Power	Fuel element temperature [°C]	Regulating rod position [mm]	Δρ [cents]
	10 W	27.8	117	-
	100 kW	104	356	-80
ΔT [°C]		76.2		

9.11. CONCLUSIONS

The fuel element's temperature is measured by thermocouples installed in the centre of the fuel element axis. This implies that the average fuel elements temperature is less than the measured temperature. In first approximation a linear temperature distribution is used (see *FIG. **9.1***), however, for more detailed calculations the actual temperature distribution is calculated.

From $\Delta\rho$ and ΔT the temperature coefficient of reactivity is found to be -1.05 cents/°C. Reactor plants having a negative temperature coefficient of reactivity are inherently safe systems, because an increase in temperature (for instance due to an increase in power) has the effect of reducing reactivity, i.e. trying to shutdown the reactor.

10. NEUTRON FLUX DENSITY MEASUREMENT USING COMPENSATED IONIZATION CHAMBERS

10.1. INTRODUCTION

The neutron flux density is the most important parameter which must be controlled during reactor operation. Its range varies from source level (shut down reactor) to 10^{13} n/cm^2/s or even more at full power operation. However there are four different types of radiations in a reactor core, i.e. prompt, delayed gamma radiation and prompt, delayed neutrons, with a large variety of energies.

In the shutdown situation condition, the delayed gamma radiation and some source neutrons are present. During the start-up procedure, the ratio between gammas and neutrons changes constantly while at full power level, the neutron signal dominating the gamma radiation. Therefore, very special detectors are required to measure the neutron induced signal at different power levels.

In shutdown and low power mode, the optimal detectors are fission chambers (FC); at medium power level, the compensated ionization chambers (CIC) are used; while at full power level, uncompensated ionization chambers (UIC) are applied. The UIC is similar to CIC except it does not include the compensating electrode – it is just a chamber with a boron-coated electrode. UIC are used at high reactor power level where the gamma induced signal is negligible compared to the neutron induced signal.

The CIC detection system reduces the gamma influence rate and selectively measures neutrons even in a high gamma background. A CIC is a differential chamber comprising three concentric electrodes, which define two chambers in the same housing. The first chamber is located between the external positive polarized electrode and the central electrode, which collects the signal. The surfaces of the electrodes are coated with a boron deposit and therefore are sensitive to neutrons and gamma rays. The schematic diagram of CIC is shown in *FIG. **10.1***.

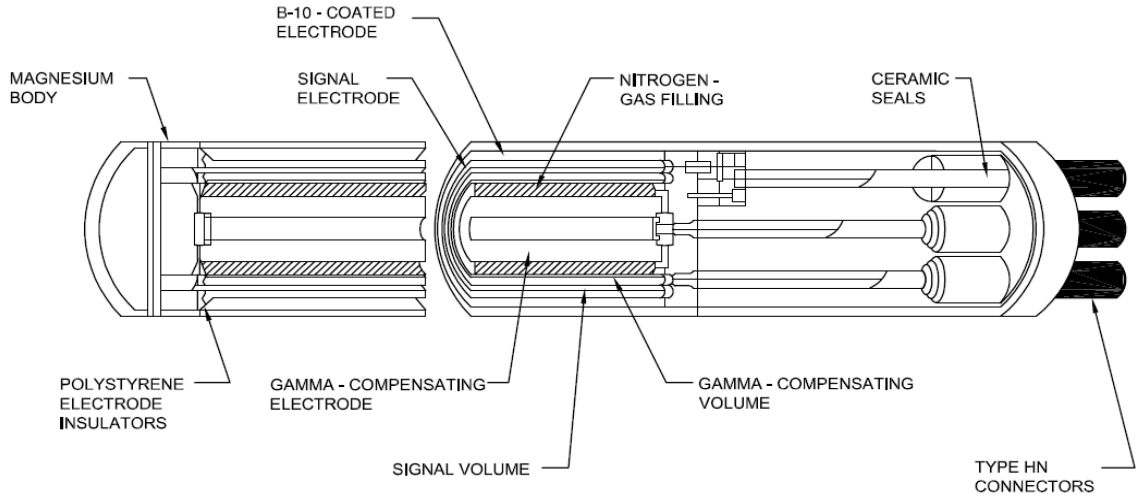


FIG. 10.1. Design of a compensated ionization chambers.

The most probable reaction is the $^{10}B(n, \alpha)^7Li$ reaction which converts the thermal neutrons into directly detectable alpha particles as shown in the following reactions:

$$^{10}_{5}B + ^{1}_{0}n \longrightarrow ^{7}_{3}Li^{*} + ^{4}_{2}He \quad , \text{Q} = 2.31 \text{ MeV} \tag{10.1}$$

$$^{10}_{5}B + ^{1}_{0}n \longrightarrow ^{7}_{3}Li \; + ^{4}_{2}He \quad , \text{Q} = 2.792 \text{ MeV} \tag{10.2}$$

The reaction product ^{7}Li may be left in its ground state or in its first excited state. When thermal neutrons are used to induce the reaction, about 94% of all reactions lead to the excited state; the remaining 6% of reaction products leads directly to the ground state. These nuclear reactions produce of about 2.31 and 2.792 MeV respectively. Alpha particles and ^{7}Li ionize gas and produce a charge pulse or an ionization current due to the neutron reaction in boron. Also gamma radiation from fission products produces electrons through photoelectric, Compton and pair production effect in the chamber.

The second chamber in the CIC is located between the internal negative polarized electrode and the signal electrode. The surface is not coated with a boron deposit and is therefore only sensitive to gamma rays. The current of these two chambers is subtracted from each other due to their opposite polarization and the resulting current is therefore mainly proportional to the neutron induced signal only, as shown below:

- Chamber one with boron coating: n+γ sensitive (+HV)
- Chamber two without boron: γ sensitive (-HV)

By subtracting the two signals, the gamma signal is compensated and only neutrons counts for the final current signal, which is graphically shown on *FIG.* ***10.2***.

$$n + \gamma - \gamma \longrightarrow n \tag{10.3}$$

A typical compensation plot of a CIC is shown on *FIG.* ***10.3***, and *FIG.* ***10.4*** shows a typical saturation curve. The main characteristics of a CIC are the following:

- It is mainly sensitive to neutrons;
- A typical neutron sensitivity is 10^{-14} A/n·v to 10^{-13} A/n·v, where n·v is the neutron flux density (neutrons/cm^2/s);
- The chamber has to be compensated for after installation in the reactor in the presence of a high gamma field but in the absence of neutrons (i.e. soon after reactor shut down with removed neutron source).

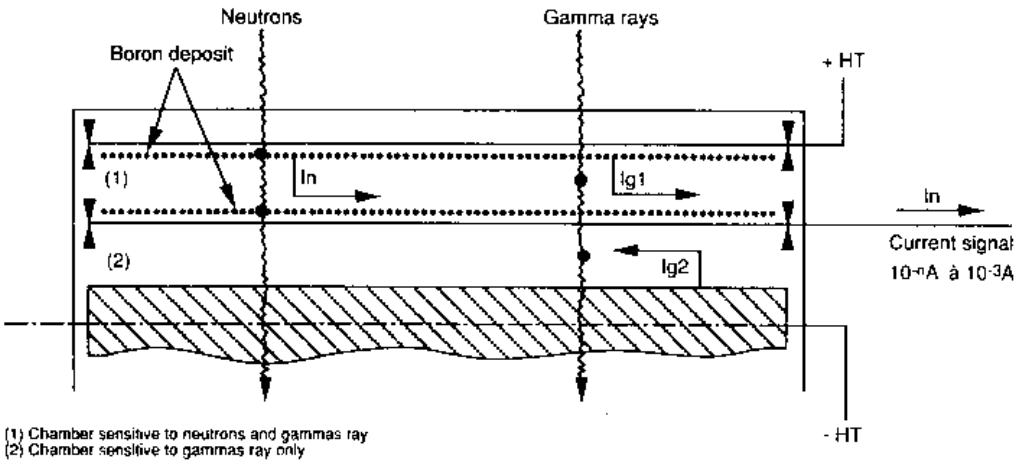


FIG. 10.2. Compensated ionization chamber mimic diagram.

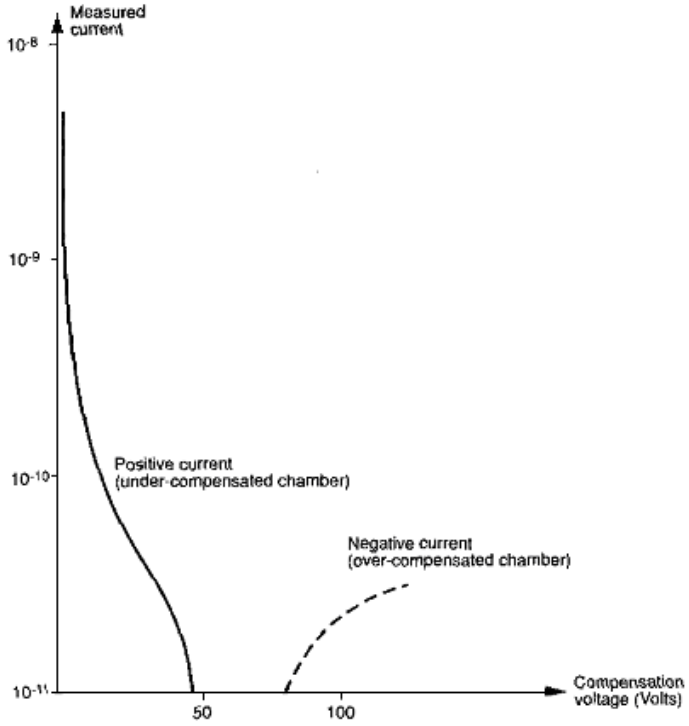


FIG. 10.3. Typical diagram of a compensation curve.

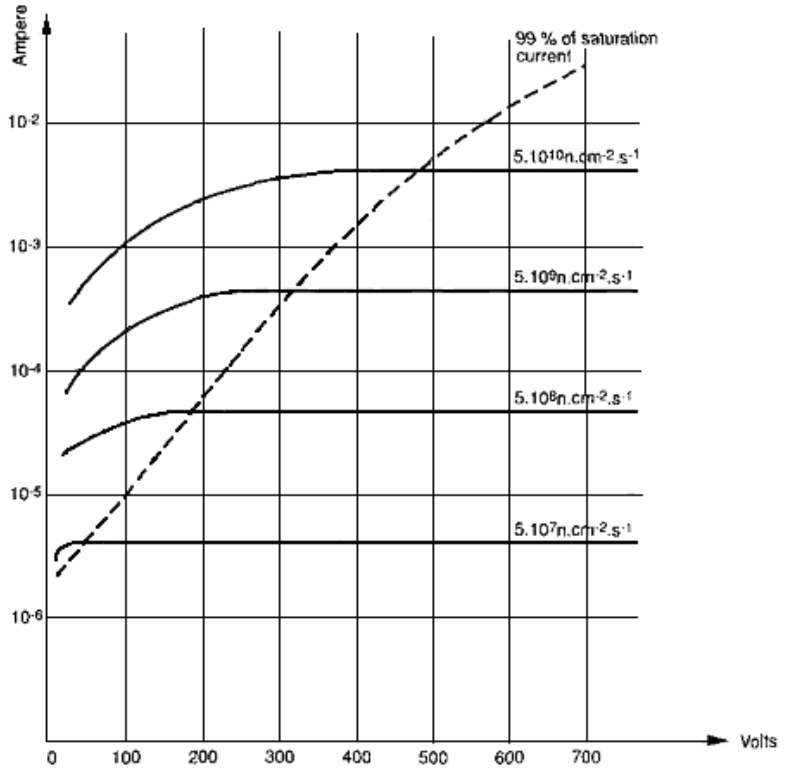


FIG. 10.4. Typical saturation curve diagram.

10.2. PURPOSE

This exercise and the next two, studies the properties and limitations of different types of detector used in a reactor. The experiment specially examines the effects of varying the applied high voltage to a compensated ionization chamber installed in the TRIGA, and determines its saturation voltage for the detectors. The participants will learn the meaning and use of the compensated ionization chamber installed in the TRIGA reactor.

10.3. SAFETY MEASURES

The standards regulations for control of radiation exposure in the laboratory for health and safety should be demonstrated to the experimenters. The general radiation protection safety measures for each student during this experiment are:

- Personal dosimeter for each course participant provided by the health physics group;
- Use of lab coat in the reactor hall;
- Spend as little time as possible in the control area;
- Avoid standing directly above the reactor core;
- Each course participant should wash his/her hands after the completion of the experiment and should check for possible contamination using a hand-foot monitor.

10.4. INSTRUMENTS AND MATERIALS

During this special practical exercise performed on the reactor platform, already installed reactor instrumentation is extensively used. The following equipment are used:

- TRIGA Mark II with an operating power of 10 W, 1 kW, 25 kW and 100 kW for this experiment;
- A compensated ionization chamber installed in the reactor;
- A contamination monitor.

10.5. EXPERIMENTAL PROCEDURES

For this experiment the TRIGA reactor is stabilized at four different power levels, firstly at 10 W. A positive high voltage is applied to the compensated ion chamber, which is increased stepwise. The resulting chamber current output is recorded until a plateau was reached. This procedure is then repeated for a reactor power of 1 kW, 25 kW and 100 kW.

Afterwards a second method is performed. At a fixed positive and negative high voltage (+800 V, -150 V) the reactor power is increased incrementally from 1 kW up to 250 kW and the chamber current is recorded.

10.6. MAIN PARAMETER(S) MEASURED

The following parameters are measured to obtain the saturation curve for different reactor power level:

- Current of different positive voltage and reactor power;
- Current of fixed voltage but different reactor power.

10.7. TYPICAL REAL DATA COLLECTED

During the experiments with the compensated ionization chamber the following currents listed in TABLE **10.1** and TABLE **10.2** were collected.

TABLE 10.1. CHAMBER CURRENT AS A FUNCTION OF + HV AND REACTOR POWER

+HV	Reactor power 10 W (current [A])	Reactor power 1 kW (current [A])	Reactor power 25 kW (current [A])	Reactor power 100 kW (current [A])
+4	1.80×10^{-8}	8.50×10^{-7}	7.80×10^{-6}	2.24×10^{-5}
+5	3.40×10^{-8}	1.30×10^{-6}	1.10×10^{-5}	3.22×10^{-5}
+6	4.18×10^{-8}	1.60×10^{-6}	1.40×10^{-5}	3.94×10^{-5}
+7	4.65×10^{-8}	1.80×10^{-6}	1.63×10^{-5}	4.57×10^{-5}
+8	4.80×10^{-8}	2.00×10^{-6}	1.82×10^{-5}	5.09×10^{-5}
+9	4.90×10^{-8}	2.20×10^{-6}	2.00×10^{-5}	5.61×10^{-5}
+10	4.88×10^{-8}	2.40×10^{-6}	2.16×10^{-5}	6.04×10^{-5}
+20	5.05×10^{-8}	3.70×10^{-6}	3.45×10^{-5}	9.62×10^{-5}
+30	5.10×10^{-8}	4.60×10^{-6}	4.27×10^{-5}	1.21×10^{-4}
+40	5.12×10^{-8}	5.20×10^{-6}	5.02×10^{-5}	1.42×10^{-4}
+50	5.13×10^{-8}	5.00×10^{-6}	5.57×10^{-5}	1.58×10^{-4}
+100	5.14×10^{-8}	5.30×10^{-6}	7.98×10^{-5}	2.23×10^{-4}
+150	5.20×10^{-8}	5.28×10^{-6}	9.60×10^{-5}	2.73×10^{-4}
+200	5.18×10^{-8}	5.25×10^{-6}	9.75×10^{-5}	3.14×10^{-4}
+300	5.19×10^{-8}	5.24×10^{-6}	9.75×10^{-5}	3.76×10^{-4}
+400	5.19×10^{-8}	5.30×10^{-6}	9.75×10^{-5}	3.76×10^{-4}
+500	5.26×10^{-8}	5.35×10^{-6}	9.75×10^{-5}	3.78×10^{-4}

TABLE 10.2. CHAMBER CURRENT AS A FUNCTION OF REACTOR POWER

Reactor power [kW]	Current [A]
1	4.40×10^{-6}
3	1.00×10^{-5}
10	3.70×10^{-5}
25	9.00×10^{-5}
100	3.20×10^{-4}
250	8.60×10^{-4}

10.8. PRE-KNOWLEDGE REQUIRED FROM STUDENTS

The students should have an understanding in

- nuclear and reactor physics: knowledge of nuclear fission, types of decay, cross section, nuclear reactions, core reactivity, criticality, neutron flux density and its behavior.
- Principle of different types of neutron detector;
- radiation physics and protection: the paticipants should have a profound understanding about radioactive dacay, dose rates and limits, activity, and all aspects regarding radiation protection.

10.9. RESULTS

The obtained currents listed on TABLE **10.1** are plotted on *FIG. **10.5***. The relation of detector current and reactor power is shown graphically on *FIG. **10.6***.

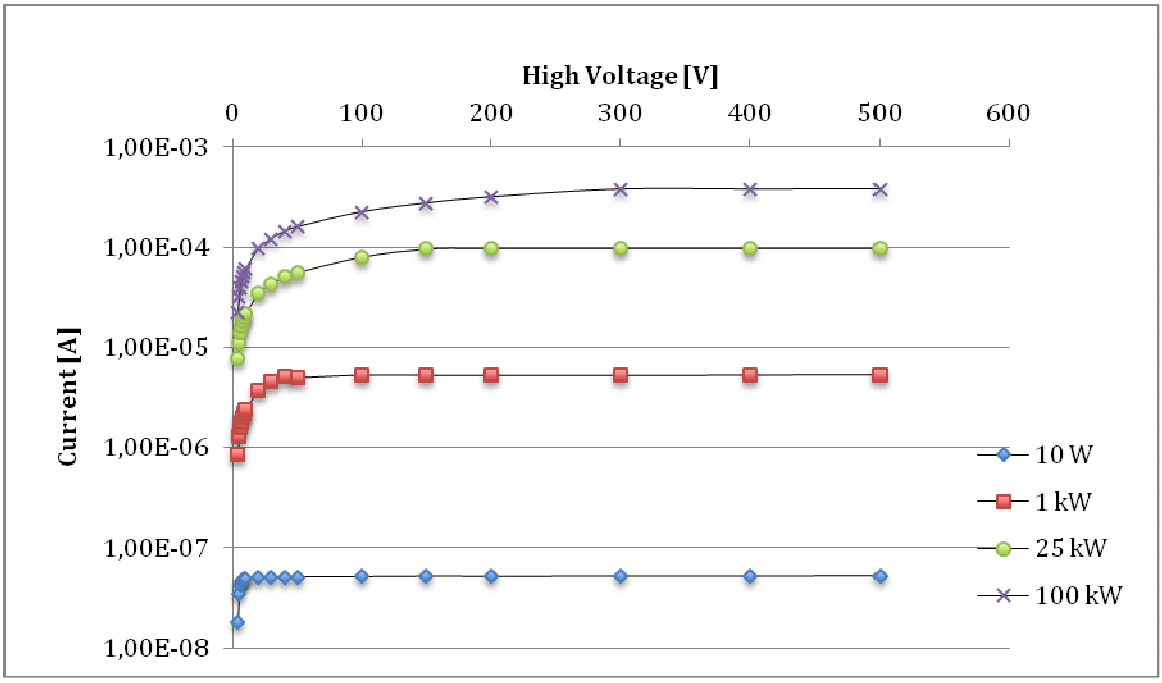


FIG. 10.5. Compensated ionization chamber response at different voltage.

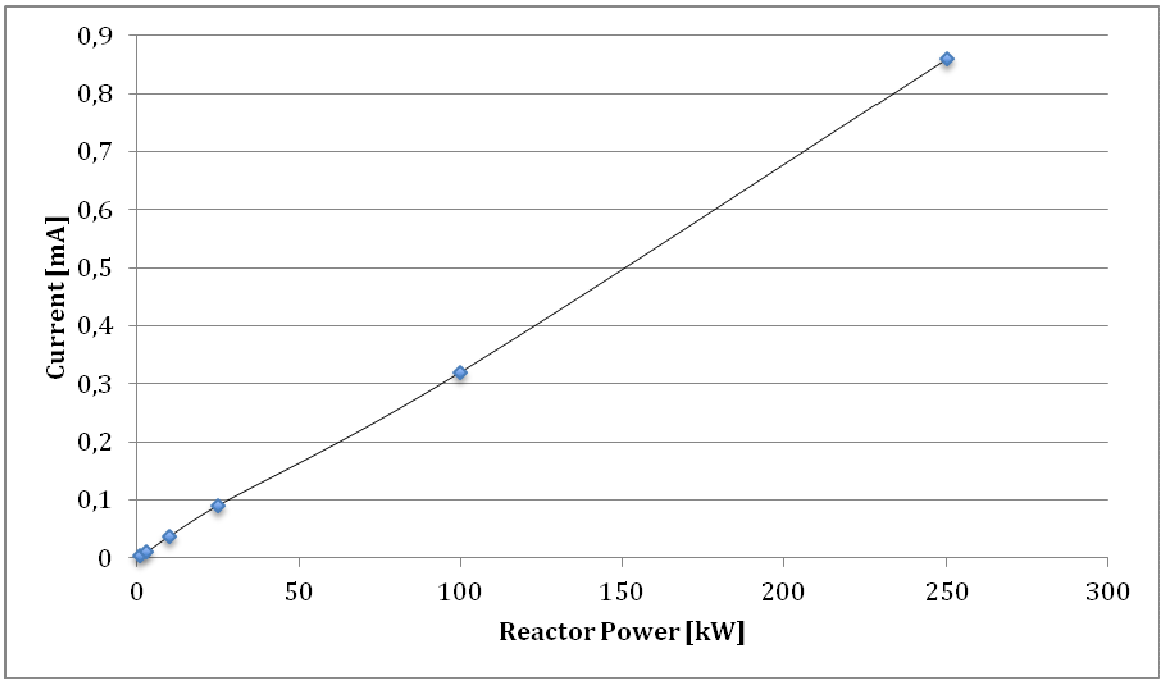


FIG. 10.6. The relation between detector current and reactor power.

10.10. CONCLUSIONS

FIG. ***10.5*** represents the saturation curves of the compensated ionization chamber for different power levels. An increase of the applied voltage above a certain value has little effect upon the current. This threshold depends on the reactors power operation. There is consequently a desire to operate at high voltages in order to guarantee a linear response from the detector, removing the uncertainty of the voltage causing fluctuations in the output.
A constant current for a reactor operation of 100 kW power starts approximately at 300 volts. Hence, setting the high voltage to 800 volts, all measured currents for different reactor power up to 250 kW indicates linearity, which is shown graphically on *FIG.* ***10.6***.

11. NEUTRON FLUX DENSITY MEASUREMENT WITH SELF-POWERED NEUTRON DETECTORS

11.1. INTRODUCTION

The principle of measuring neutron-induced electrons through (n, β) reaction was known and patented in early 1938. Neutron detectors operating without any external voltage were reported in 1961, but the practical application in neutron flux measurement techniques was first reported in 1964. During the following years a large number of papers on self-powered neutron detectors were published. As these detectors operated without any external voltage, they were called self-powered neutron detectors (SPND).
Most of the new emitter materials were developed between 1969 and 1974. Besides the most important emitter materials Co, Rh and V, other emitter materials such as Ag, Al, Au, ^{11}B, Ce, Er, Gd, Hf, Pd, Pt, Os, Ta, Ti, W and Yb were also investigated. Another part of the developmental work concentrated on new geometric detector forms where detector bundles with different emitter lengths were used to measure the axial power distribution in the core.

TABLE 11.1. SOME CHARACTERISTIC DATA OF β-EMITTER MATERIALS

Isotope	Natural abundance [%]	Neutron cross-section [b]	Half-life	Maximum β-decay [MeV]	Burn up per month at 10^{14} /cm^2/s [%]
^{27}Al	100	0.23	2.3 min	2.9	0.006
^{107}Ag	48.65	35	2.3 min	1.8	0.9
^{109}Ag	51.35	89	24 s	2.8	2.3
^{11}B	80.2	0.005	0.02 min	13.4	
^{51}V	99.76	4.8	3.76 min	2.5	0.12
^{103}Rh	100	150	4.4 min/42 s	2.5	3.9
^{59}Co	100	37	10^{-14} s	-	1.0
^{55}Mn	100	13.3	2.58 h	2.8	0.34
^{7}Li	92.6	0.037	0.85 s	13.0	
Natural Er	100	162			4.0
Natural Hf	100	102			3.2
Natural Pt	100	10			0.25

The most common detector form is a coaxial cylinder with an emitter in the centre, which is surrounded completely by an insulator. The outer sheath acts as case and collector as shown in *FIG. **11.1*** and *FIG. **11.2***. The detector geometry can be adapted for special applications; small plates, hooks or spirals are offered by the industry. TABLE **11.1** gives a summary of commonly used emitter materials for self-powered neutron detectors.

The emitter material must have an appropriate neutron absorption cross-section, which is usually a compromise between detector sensitivity and detector burn-up. The detector signal decreases with burn-up and the burn up rate depends mainly on the absorption cross-section of the emitter and local flux density values. Other emitter material selection criteria are the

energy of the radiation emitted after a neutron capture, the half-life of the produced nuclides, their daughter products and the melting point.

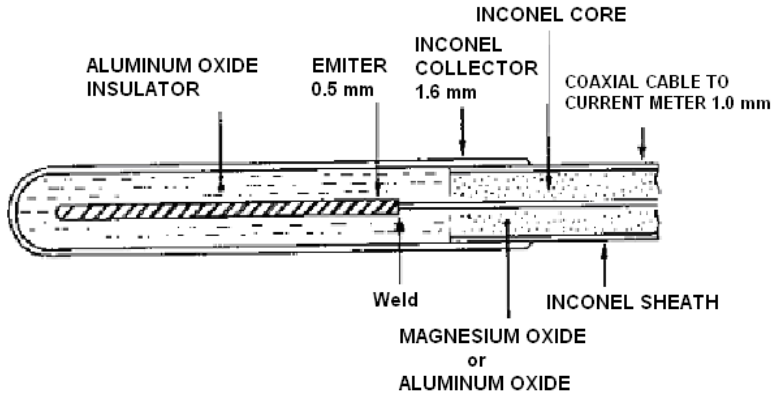


FIG. 11.1. Coaxial cylinder with an emitter surrounded by an insulator.

The insulator usually consists of a high-temperature resistant material (like Al_2O_3) and must have a thickness that allows the produced electrons ((n, γ) (γ, e)) to reach the collector. An optimization of the detector geometry, especially the emitter diameter and the insulator thickness, is necessary.

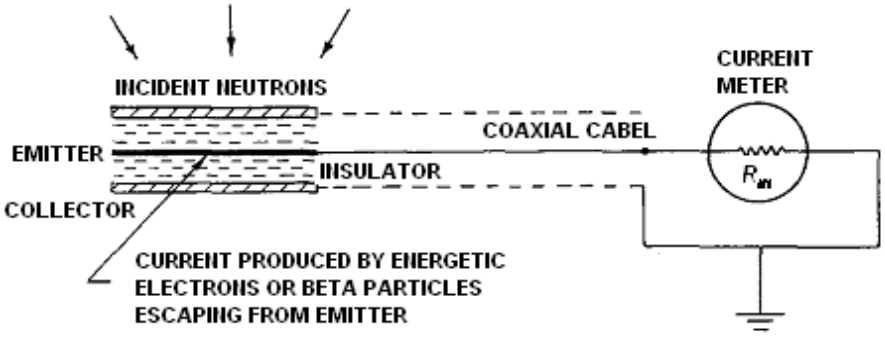


FIG. 11.2. Self-powered neutron detector.

The collector has to collect the emitted electrons and gives the detector its mechanical stability. It must be corrosion resistant and should not undergo dimensional changes at high temperatures. After a neutron capture in the emitter a ((n, γ) (γ, e)) reaction produces electrons. The constant loss of electrons from the emitter produces a current between emitter and collector, which can be measured directly by an ammeter. The current is proportional to the rate of absorption of neutrons in the emitter and thus proportional to the local neutron flux. A typical detection sensitivity is 10^{-21} A/n·v.

In some emitter materials the electrons are produced with a time delay according to the half-life of the nuclide. Therefore the detector gives a delayed response to a neutron flux variation and a constant signal is measured only after the saturation activity of the emitter material has been reached. Typical delayed self-powered neutron detectors are Rh and V detectors, which have a response time of several minutes. For this reason they cannot be used in the reactor protection system as shown in *FIG. **11.3***.

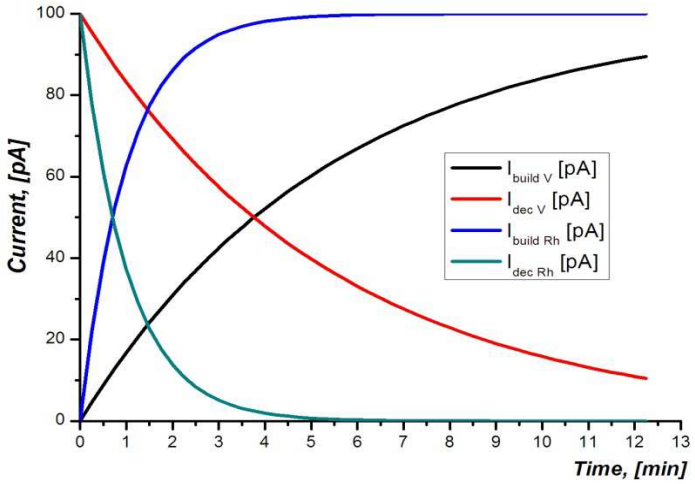


FIG. 11.3. SPND response to step change in neutron flux.

If the ((n, γ) (γ, e)) reactions take place immediately (i.e. 10^{-14} s for a Co detector), the detector responds promptly to any neutron flux change and can be used in the reactor protection system. As the prompt self-powered neutron detectors are very important for power reactor instrumentation, the signal composition will be analysed more closely.

The main part of the signal is prompt and is produced by the reaction $^{59}Co(n, \gamma)^{60*}Co$. In addition, heavier nuclides like ^{60}Co and ^{61}Co are produced in the emitter material. The gamma radiation emitted from this nuclear reaction produces electrons in the emitter by photo, Compton, pair-production processes or by internal conversions, which penetrates the insulator and are collected by the collector. Now the prompt signal $I_{pr,n}$ resulting from the direct nuclear reaction has to be distinguished from the delayed signal I_β from the produced heavier isotopes. These isotopes usually emit β radiation and delayed gamma radiation, both producing an unwanted delayed detector signal together with the prompt signal.

The other sources of radiation disturbances are neutron capture gamma rays from the reactor core and the prompt ($I_{pr,\gamma}$) and delayed ($I_{del,\gamma}$) gamma radiation from nuclear fission. These radiation components penetrate the detector and cable from outside and produce secondary electrons registered as detector noise. The polarity of this noise signal depends on the atomic numbers of collector, insulator and emitter. Therefore, the total detector signal is composed of three parts:

$$I_{tot} = I_{pr,n} + I_{pr,\gamma} + I_{del,\gamma} + I_\beta \qquad (11.1)$$

From the above mentioned components contained in $I_{pr,\gamma}$ one component depends on the fission rate. The component from neutron capture gamma rays depends on the location where the detector is installed in the reactor core. Therefore, $I_{pr,\gamma}$ depends on the reactor type and detector core position. For one special location of a new Co detector, $I_{pr,n}$ is about 83% and $I_{pr,\gamma}$ about 17% of the total signal. For other emitter materials these values depend on the atomic number of the emitter. The composition of the two different noise signals ($I_{del,\gamma}$ and I_β) can accidentally produce a total signal I_{tot}, which may be approximately constant over several years even if the true detector signal $I_{pr,n}$ decreases according to the detector burn-up. Thus the signal to noise ratio is decreased while the total signal is constant. The behaviour of I_{tot} depends very much on the local neutron flux and the above-mentioned effects of burn-up compensation by the creation of noise signals, and is predominant at a flux level of about 1×10^{14} $cm^{-2}s^{-1}$. From a safety point of view this effect is not acceptable; therefore, a periodic

calibration of the detector signal as well as an investigation of its composition is necessary. This can be done either during reactor operation with movable detector systems or during reactor shutdown periods when the prompt detector signal is zero.

11.2. PURPOSE

The purpose of this exercise is to provide students an insight of self-powered neutron detectors. The participants will learn the meaning and use of the SPND installed in the TRIGA reactor.

11.3. SAFETY MEASURES

The standards regulations for control of radiation exposure in the laboratory for health and safety should be demonstrated to the experimenters. The general radiation protection safety measures for each student during this experiment are:

- Personal dosimeter for each course participant provided by the health physics group;
- Use of lab coat in the reactor hall;
- Spend as little time as possible in the control area;
- Avoid standing directly above the reactor core;
- Each course participant should wash his/her hands after the completion of the experiment and should check for possible contamination using a hand-foot monitor.

11.4. INSTRUMENTS AND MATERIALS

During this special practical exercise performed on the reactor platform, already installed reactor instrumentation is extensively used. The following equipment are used:

- TRIGA Mark II with an operating power of 30 kW for this experiment;
- A vanadium self-powered neutron detector;
- A contamination monitor.

11.5. EXPERIMENTAL PROCEDURES

A SPND is installed in the centre of the reactor core and the reactor is started to a power level of 30 kW. According to the selected emitter material the current of the SPND will increase due to the activation of the emitter material. It takes about 10 half-lives until a constant signal is obtained. Then the reactor is shut down and the decrease of the signal is observed, which again depends on the half-life of the emitter material. From these values the emitter material can be identified.

11.6. MAIN PARAMETER(S) MEASURED

The following parameters are measured during this experiment:

- Current output during constant power operation: The detector current will increase as a function of time depending on the emitter type, the current has to be plotted as a function of activation time.
- Current output following a reactor scram: The current will decrease as a function of time depending on the emitter type, the current has to be plotted as a function of the

decay time. From this time plot the half life and therefore the emitter type can be determined (see also fig. 12.3.)

11.7. TYPICAL REAL DATA COLLECTED

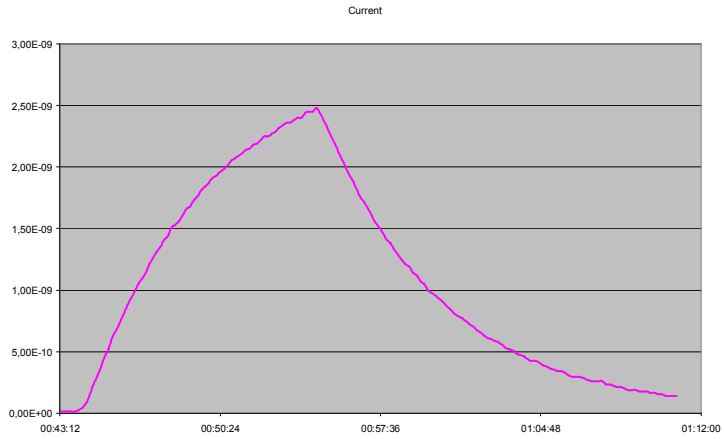


FIG. 12.4. SPND current as a function of time at constant power level and after reactor scram

11.8. PRE-KNOWLEDGE REQUIRED FROM STUDENTS

The students should have an understanding in

- nuclear and reactor physics: knowledge of nuclear fission, types of decay, cross section, nuclear reactions, core reactivity, criticality, neutron flux density and its behavior.
- Principle of different types of neutron detector;
- radiation physics and protection: the paticipants should have a profound understanding about radioactive dacay, dose rates and limits, activity, and all aspects regarding radiation protection.

11.9. RESULTS

It is possible to determine the unknown emitter material of an SPND detector by activation in an operating reactor core and by decay after a reactor scram.

11.10. CONCLUSIONS

SPN detectors are very small and cheap neutron detectors with rather low detector sensitivity. As all solid state detectors they can be inserted directly into the reactor core and can be used for online neutron flux monitoring. Due to their low detector sensitivity very sensitive ammeters have to be used for the current measurement.

12. DEMONSTRATION OF A REACTOR PULSE WITH DIFFERENT REACTIVITY INSERTION

12.1. INTRODUCTION

Due to the unique characteristics of the Zr-H moderator used in TRIGA reactors one can withdraw a control rod completely and suddenly from the critical core. Any other reactor would be severely damaged by this operation type which can be performed routinely at the TRIGA and results in a power pulse or power burst. The property of the Zr-H moderator making this possible is that the H-atoms form oscillations around its lattice position. The energy of these oscillations is quantized like in a three-dimensional harmonic oscillator (Einstein oscillator). Incident neutrons are moderated and are then in thermal equilibrium with these atoms. During a power burst the fuel temperature rises and, therefore, the population at higher oscillation energy levels of the moderator atoms increases, thus reducing the moderating effect of the neutrons. The neutrons may even be accelerated by the collisions with the H-atoms and the thermal neutron spectrum is hardened. Therefore, the number of fissions and the reactor power is decreased.

12.2. PURPOSE

The purpose of this experiment is to demonstrate the pulsing capability of TRIGA reactors and to explain in detail the negative temperature effect and the structure of the U-Zr-H fuel.

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The property of the Zr-H moderator making this possible is that the H-atoms form oscillations around its lattice position. The energy of these oscillations is quantized like in a three-dimensional harmonic oscillator (Einstein oscillator). Incident neutrons are moderated and are then in thermal equilibrium with these atoms. During a power burst the fuel temperature rises and, therefore, the population at higher oscillation energy levels of the moderator atoms increases, thus reducing the moderating effect of the neutrons. The neutrons may even be accelerated by the collisions with the H-atoms and the thermal neutron spectrum is hardened. Therefore, the number of fissions and the reactor power is decreased.

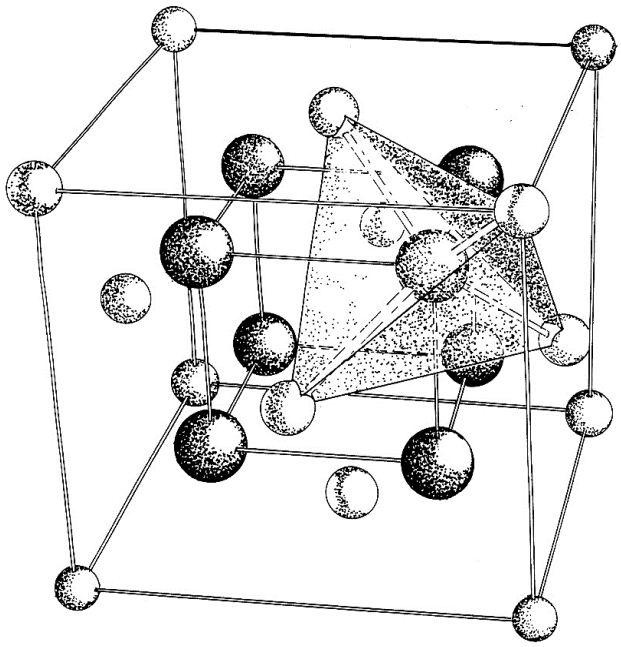


Fig. 12.1: Unit cell of ZrH2, 4 Zr atoms surround 1 H atom (black)

12.3. SAFETY MEASURES

The standards regulations for control of radiation exposure at top of the reactor tank for health and safety are required. The general radiation protection safety measures for each student during this experiment are:
Personal dosimeter for each course participant provided by the health physics group;

- Use of lab coat in the reactor hall;
- Spend as little time as possible in the control area;
- Avoid standing directly above the reactor core;
- Each course participant should wash his/her hands after the completion of the experiment and should check for possible contamination using a hand-foot monitor.

12.4. INSTRUMENTS AND MATERIALS

This experiment is uniquely controlled by reactor I&C system and through the inherent negative temperature effect of the U-Zr-H fuel material.

12.5. EXPERIMENTAL PROCEDURE

The reactor power is raised to a power level of less than 10 W with the shim rod and the regulating rod, the pulse rod stays in its lower position. After switching the control instrumentation to "pulse mode" the pulse rod (maximum reactivity value ~ 2 $!) is shot out of the critical core and the power pulse takes place. The power peak value can be varied by changing the shock absorber position which determines how far the pulse rod is fired out from the core and then determines the amount of reactivity inserted into the critical core. During and after the pulse the fuel element temperature and maximum power can be read from the instrumentation. The procedure may be repeated with different shock absorber positions. The whole procedure takes about 40 ms while the peak power increases to 250 MW and the peak neutron flux to 1 x1016 cm-2s-1.

12.6. MAIN PARAMETER MEASURED

The main parameters of the reactivity insertion is displayed at the control console which are:

- Peak pulse power
- Integrated energy released during the pulse
- Fuel temperature
- Time behaviour of the reactor power

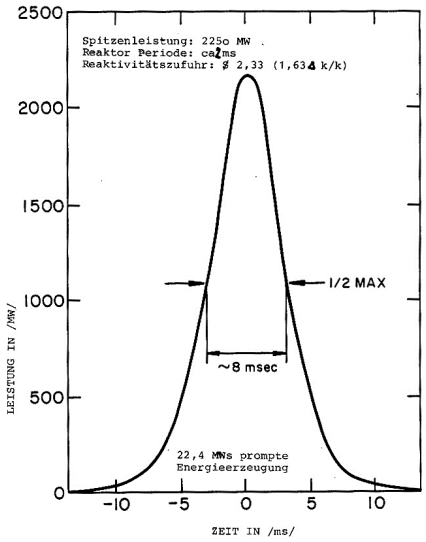


Fig. 13.2: Typical power transient (not from TRIGA Vienna)
Peak power: 2250 MW
Reactor period: 2 ms
Reactivity insertion: $ 2.33 (= 1.63 □k/k)

12.7. TYPICAL DATA COLLECTED

The only data and diagrams collected during the pulse are those which are stored in the digital reactor control console and will be transferred electronically to the students for further evaluation.

12.8. DATA ANALYSIS, ASSUMPTIONS AND EQUATIONS

If different reactivity insertions (1,1$, 1,2$, 1,5$, 1,8$, 2,0$) are demonstrated the impact of this step insertions on the data collected (see § 13.6) will be evaluated.

12.9. PRE-KNNOWLEDGE REQUIRED FROM STUDENTS

The students should have an understanding in

- nuclear and reactor physics: knowledge of nuclear fission, types of decay, cross section, nuclear reactions, core reactivity, criticality, neutron flux density and its behaviour.
- Principle of different types of neutron detector;
- radiation physics and protection: the participants should have a profound understanding about radioactive decay, dose rates and limits, activity, and all aspects regarding radiation protection.

12.10. RESULTS

The fuel temperature, the integrated power release and the peak power should be displayed as a function of the reactivity insertions.

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